This draft *Federal Register* notice contains the latest draft proposed rule language that the NRC staff has publicly released to support interactions with the Advisory Committee on Reactor Safeguards (ACRS). This version is based on reviews by NRC staff and consideration of stakeholder input. The NRC staff expects to adopt further changes in the draft proposed rule language.

This language has not been subject to complete NRC management or legal review, and its contents should not be interpreted as official agency positions. The NRC staff plans to continue working on the draft proposed rule language provided in this document.

Please note that blue text indicates conforming changes to existing rule language in parts other than Part 53.

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Flags, Organization and functions (Government Agencies), Seals and insignia.

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and reactors, Penalties, Reporting and recordkeeping requirements, Sex discrimination, Source material, Special nuclear material, Waste treatment and disposal.

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10 CFR Part 20

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10 CFR Part 21

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10 CFR Part 26

[Part 26 placeholder]

10 CFR Part 30

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10 CFR Part 40

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10 CFR Part 50

Administrative practice and procedure, Antitrust, Backfitting, Classified information, Criminal penalties, Education, Emergency planning, Fire prevention, Fire protection, Incorporation by reference, Intergovernmental relations, Nuclear power plants and reactors, Penalties, Radiation protection, Reactor siting criteria, Reporting and recordkeeping requirements, Whistleblowing.

10 CFR Part 51

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10 CFR Part 52

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10 CFR Part 53

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10 CFR Part 73

[Part 73 placeholder]

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10 CFR Part 95

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10 CFR Part 140

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10 CFR Part 170

Byproduct material, Import and export licenses, Intergovernmental relations, Non-payment penalties, Nuclear energy, Nuclear materials, Nuclear power plants and reactors, Source material, Special nuclear material.

10 CFR Part 171

Annual charges, Approvals, Byproduct material, Holders of certificates, Intergovernmental relations, Nonpayment penalties, Nuclear materials, Nuclear power plants and reactors, Registrations, Source material, Special nuclear material.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; and 5 U.S.C. 552 and 553, the NRC is adopting the following amendments to 10 CFR parts 1, 2, 10, 19, 20, 21, 25, 26, 30, 40, 50, 51, 52, 73, 74, 95, 140, 150, 170, and 171 and adding part 53:

PART 1 – STATEMENT OF ORGANIZATION AND GENERAL INFORMATION

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

Authority: Atomic Energy Act of 1954, secs. 23, 25, 29, 161, 191 (42 U.S.C. 2033, 2035, 2039, 2201, 2241); Energy Reorganization Act of 1974, secs. 201, 203, 204, 205, 209 (42 U.S.C. 5841, 5843, 5844, 5845, 5849); Administrative Procedure Act (5 U.S.C. 552, 553); Reorganization Plan No. 1 of 1980, 5 U.S.C. Appendix (Reorganization Plans).

2. Revise § 1.43(a)(2) to read as follows:

§ 1.43 Office of Nuclear Reactor Regulation.

* * * * *

(a) * * *

(2) Receipt, possession, and ownership of source, byproduct, and special nuclear material used or produced at facilities licensed under 10 CFR parts 50, 52, 53, and 54;

* * * * *

PART 2 – AGENCY RULES OF PRACTICE AND PROCEDURE

1. The authority citation for part 2 continues to read as follows:

Authority: Atomic Energy Act of 1954, secs. 29, 53, 62, 63, 81, 102, 103, 104, 105, 161, 181, 182, 183, 184, 186, 189, 191, 234 (42 U.S.C. 2039, 2073, 2092, 2093, 2111, 2132, 2133, 2134, 2135, 2201, 2231, 2232,2233, 2234, 2236, 2239, 2241, 2282); Energy Reorganization Act of 1974, secs. 201, 206 (42 U.S.C. 5841, 5846); Nuclear Waste Policy Act of 1982, secs. 114(f), 134, 135, 141 (42 U.S.C. 10134(f), 10154, 10155, 10161); Administrative Procedure Act (5 U.S.C. 552, 553, 554, 557, 558); National Environmental Policy Act of 1969 (42 U.S.C. 4332); 44 U.S.C. 3504 note. Section 2.205(j) also issued under 28 U.S.C. 2461 note.

2. Revise § 2.1, paragraph (e) to read as follows:

§ 2.1 Scope.

* * * * *

(e) Standard design approvals under parts 52 and 53 of this chapter.

3. Revise § 2.4 to read as follows:

§ 2.4 Definitions.

* * * * *

Contested proceeding means –

* * *

(2) A proceeding in which the NRC is imposing a civil penalty or other enforcement action, and the subject of the civil penalty or enforcement action is an applicant for or holder of a license or permit, or is or was an applicant for or holder of a license or permit, or is or was an applicant for a standard design certification under parts 52 or 53 of this chapter; and

* * * * *

Facility means production facility or a utilization facility as defined in §§ 50.2 and 53.020 of this chapter.

* * * * *

Subpart A – Procedure for Issuance, Amendment, Transfer, or Renewal of a License, and Standard Design Approval

4. Revise § 2.100 to read as follows:

§ 2.100 Scope of subpart.

This subpart prescribes the procedure for issuance of a license; amendment of a license at the request of the licensee; transfer and renewal of license; and issuance of a standard design approval under subpart E of part 52, §§ 53.1200 through 53.1221, or §§ 53.4800 through 53.4821 of this chapter.

5. Amend § 2.101 by revising paragraphs (a)(3)(i), (a)(5), (a)(9), (a-1), and (c)(1) to read as follows:

§ 2.101 Filing of application.

(a)* * *

(3)* * *

(i) Submit to the Director, Office of Nuclear Reactor Regulation, or Director, Office of Nuclear Material Safety and Safeguards, as appropriate, such additional copies as the regulations in part 50, part 53 and subpart A of part 51 of this chapter require;

* * *

(5) An applicant for a construction permit under part 50 or part 53 of this chapter or a combined license under part 52 or part 53 of this chapter for a production or utilization facility which is subject to § 51.20(b) of this chapter, and is of the type specified in § 50.21(b)(2) or (b)(3); or § 50.22; or part 53, as applicable, of this chapter, or is a testing facility, may submit the information required of applicants by parts 50, 52, or 53 of this chapter in two parts. One part shall be accompanied by the information required by § 50.30(f) of this chapter, § 52.80(b) of this chapter, § 53.1100(f), or § 53.4700(f) of this chapter, as applicable. The other part shall include any information required by § 50.34(a) and, if applicable, § 50.34a of this chapter; or §§ 52.79 and 52.80(a); or §§ 53.1109, 53.1306, 53.1309, and 53.1312; or §§ 53.1109, 53.1366,

53.1369, and 53.1372; or §§ 53.4709, 53.4906, 53.4909, and 53.4912; or §§ 53.4709, 53.4966, 53.4969, 53.4972, as applicable. One part may precede or follow other parts by no longer than 6 months. If it is determined that either of the parts as described above is incomplete and not acceptable for processing, the Director, Office of Nuclear Reactor Regulation, or Director, Office of Nuclear Material Safety and Safeguards, as appropriate, will inform the applicant of this determination and the respects in which the document is deficient. Such a determination of completeness will generally be made within a period of 30 days. Whichever part is filed first shall also include the fee required by §§ 50.30(e), 53.1100(e), or 53.4700(e) and § 170.21 of this chapter and the information required by §§ 50.33, 50.34(a)(1), and 52.79(a)(1); or §§ 53.1109, 53.1309, and 53.1369; or §§ 53.4709 53.4909, and 53.4969, as applicable, and § 50.37, 53.1115, or 53.4715, as applicable, of this chapter. The Director, Office of Nuclear Reactor Regulation, or Director, Office of Nuclear Material Safety and Safeguards, as appropriate, will accept for docketing an application for a construction permit under part 52 or part 53 of this chapter for a production or utilization facility which is subject to § 51.20(b) of this chapter, and is of the type specified in § 50.21(b)(2) or (b)(3), or § 50.22, or part 53, as applicable, of this chapter or is a testing facility where one part of the application as described above is complete and conforms to the requirements of part 50 of this chapter. The additional parts will be docketed upon a determination by the Director, Office of Nuclear Reactor Regulation, or Director, Office of Nuclear Material Safety and Safeguards, as appropriate, that it is complete.

* * *

(9) An applicant for a construction permit for a utilization facility which is subject to § 51.20(b) of this chapter and is of the type specified in § 50.21(b)(2) or (b)(3), or § 50.22, or part 53 of this chapter, an applicant for or holder of an early site permit under

part 52 or part 53 of this chapter, or an applicant for a combined license under part 52 or part 53 of this chapter, who seeks to conduct the activities authorized under §§ 50.10(d), 53.1130, or 53.4740 of this chapter may submit a complete application under paragraphs (a)(1) through (a)(4) of this section which includes the information required by §§ 50.10(d), 53.1130, or 53.4740 of this chapter. Alternatively, the applicant (other than an applicant for or holder of an early site permit) may submit its application in two parts:

(i) Part one must include the information required by §§ 50.33(a) through (f), 53.1109, or 53.4709 of this chapter, as applicable, and the information required by §§ 50.10(d)(2) and (d)(3), 53.1130(b)(2) and (b)(3), or 53.4740(b)(2) and (b)(3) of this chapter, as applicable.

* * *

(a-1) *Early consideration of site suitability issues*. An applicant for a construction permit under part 50 or part 53 of this chapter or a combined license under part 52 or part 53 of this chapter for a utilization facility which is subject to § 51.20(b) or part 53 of this chapter and is of the type specified in § 50.21(b)(2) or (b)(3), or § 50.22, or part 53 of this chapter, or is a testing facility, may request that the Commission conduct an early review and hearing and render an early partial decision in accordance with subpart F of this part on issues of site suitability within the purview of the applicable provisions of parts 50, 51, 52, 53 and 100 of this chapter.

(1) Construction permit. The applicant for the construction permit may submit the information required of applicants by the provisions of this chapter in three parts:

(i) Part one shall include or be accompanied by any information required by §§ 50.34(a)(1) and 50.30(f); § 53.1309(a) and § 53.1100(f); or § 53.4909(a) and § 53.4700(f) of this chapter, as applicable, which relates to the issue(s) of site suitability

for which an early review, hearing, and partial decision are sought, except that information with respect to operation of the facility at the projected initial power level need not be supplied, and shall include the information required by §§ 50.33(a) through (e), 53.1109(a) through (e), or 53.4709(a) through (e), as applicable; and §§ 50.37, 53.1115, or 53.4715 of this chapter, as applicable. The information submitted shall also include:

* * *

(B) A range of postulated facility design and operation parameters that is sufficient to enable the Commission to perform the requested review of site suitability issues under the applicable provisions of parts 50, 51, 53, and 100, and

* * *

(ii) Part two shall include or be accompanied by the remaining information required by §§ 50.30(f), 50.33, and 50.34(a)(1); or §§ 53.1312(a)(1) or 53.1419(a)(1), 53.1109, and 53.1309(a); or §§ 53.4912(a)(1) or 53.5019(a)(1)(i), 53.4709, and 53.4909(a) of this chapter, as applicable.

(iii) Part three shall include the remaining information required by §§ 50.34a and (in the case of a nuclear power reactor) 50.34(a), 53.1309, or 53.4909 of this chapter, as applicable.

* * *

(2) Combined license under 10 CFR part 52 or part 53. An applicant for a combined license under part 52 or part 53 of this chapter may submit the information required of applicants by the provisions of this chapter in three parts:

(i) Part one shall include or be accompanied by any information required by §§ 52.79(a)(1) and 50.30(f); §§ 53.1416(a) and 53.1100(f); or §§ 53.5016(a) and 53.4700(f) of this chapter, as applicable, which relates to the issue(s) of site suitability for

which an early review, hearing, and partial decision are sought, except that information with respect to operation of the facility at the projected initial power level need not be supplied, and shall include the information required by §§ 50.33(a) through (e); §§ 53.1109(a) through (e), or §§ 53.4709(a) through (e); and §§ 50.37, 53.1115, or 53.4715 of this chapter, as applicable. The information submitted shall also include:

* * *

(B) A range of postulated facility design and operation parameters that is sufficient to enable the Commission to perform the requested review of site suitability issues under the applicable provisions of parts 50, 51, 52, 53 and 100; and

* * *

(ii) Part two shall include or be accompanied by the remaining information required by §§ 50.30(f), 50.33, and 52.79(a)(1); §§ 53.1100(f), 53.1109, and 53.1416(a); §§ 53.4700(f), 53.4709, and 53.5016(a) of this chapter, as applicable.

(iii) Part three shall include the remaining information required by §§ 52.79 and 52.80; §§ 53.1416 and 53.1419; or §§ 53.5016 and 53.5019 of this chapter, as applicable.

* * * * *

6. Revise § 2.104, paragraph (a) and footnote 1 to read as follows:

§ 2.104 Notice of hearing.

(a) In the case of an application on which a hearing is required by the Act or this chapter, or in which the Commission finds that a hearing is required in the public interest, the Secretary will issue a notice of hearing to be published in the *Federal Register*. The notice must be published at least 15 days, and in the case of an application concerning a limited work authorization, construction permit, early site permit, or combined license for a facility of the type described in § 50.21(b) or 50.22, or subparts

H or R of part 53 of this chapter, as applicable, or a testing facility, at least 30 days, before the date set for hearing in the notice.[1] In addition, in the case of an application for a limited work authorization, construction permit, early site permit, or combined license for a facility of the type described in § 50.22 or subparts H or R of part 53 of this chapter, as applicable, or a testing facility, the notice must be issued as soon as practicable after the NRC has docketed the application. If the Commission decides, under § 2.101(a)(2), to determine the acceptability of the application based on its technical adequacy as well as completeness, the notice must be issued as soon as practicable after the application has been tendered.

* * * * *

[1] If the notice of hearing concerning an application for a limited work authorization, construction permit, early site permit, or combined license for a facility of the type described in §§ 50.21(b) or 50.22, or subparts H or R of part 53 of this chapter, as applicable, or a testing facility, does not specify the time and place of initial hearing, a subsequent notice will be published in the *Federal Register* which will provide at least 30-day notice of the time and place of that hearing. After this notice is given, the presiding officer may reschedule the commencement of the initial hearing for a later date or reconvene a recessed hearing without again providing at least 30-day notice.

7. Amend § 2.105 by revising the introductory text for paragraph (a) and revising paragraphs (a)(4), (a)(10), (a)(12), (a)(13) and (b)(3) as follows:

§ 2.105 Notice of proposed action.

(a) If a hearing is not required by the Act or this chapter, and if the Commission has not found that a hearing is in the public interest, it will, before acting thereon, publish in the *Federal Register*, as applicable, or on the NRC's Web site, http://www.nrc.gov, or both, at the Commission's discretion, either a notice of intended operation under

§§ 52.103(a), 53.1452(a), or 53.5052(a) of this chapter, as applicable, and a proposed finding that inspections, tests, analyses, and acceptance criteria for a combined license under subpart C of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 have been or will be met, or a notice of proposed action with respect to an application for:

* * *

(4) An amendment to an operating license, combined license, or manufacturing license for a facility licensed under §§ 50.21(b) or 50.22; §§ 53.1100 through 53.1480; or §§ 53.4700 through 53.5080 of this chapter, as applicable, or for a testing facility, as follows:

(i) If the Commission determines under §§ 50.58, subpart I of part 53, or subpart S of part 53 of this chapter that the amendment involves no significant hazards consideration, though it will provide notice of opportunity for a hearing pursuant to this section, it may make the amendment immediately effective and grant a hearing thereafter; or

(ii) If the Commission determines under §§ 50.58 and 50.91; § 53.1515; or § 53.6015 of this chapter, as applicable, that an emergency situation exists or that exigent circumstances exist and that the amendment involves no significant hazards consideration, it will provide notice of opportunity for a hearing pursuant to § 2.106 (if a hearing is requested, it will be held after issuance of the amendment);

* * *

(10) In the case of an application for an operating license for a facility of a type described in §§ 50.21(b) or ~~§~~ 50.22, or part 53 of this chapter or a testing facility, a notice of opportunity for hearing shall be issued as soon as practicable after the application has been docketed; or

* * *

(12) An amendment to an early site permit issued under subpart A of part 52, §§ 53.1140 through 53.1188, or §§ 53.4750 through 53.4798 of this chapter, as follows:

(i) If the early site permit does not provide authority to conduct the activities allowed under §§ 50.10(e)(1), 53.1130(b)(1), or 53.4740(b)(1) of this chapter, the amendment will involve no significant hazards consideration, and though the NRC will provide notice of opportunity for a hearing under this section, it may make the amendment immediately effective and grant a hearing thereafter; and

(ii) If the early site permit provides authority to conduct the activities allowed under §§ 50.10(e)(1), 53.1130(b)(1), or 53.4740(b)(1), and the Commission determines under §§ 50.58 and 50.91~~,~~; § 53.1515~~,~~; or § 53.6015 of this chapter that an emergency situation exists or that exigent circumstances exist and that the amendment involves no significant hazards consideration, it will provide notice of opportunity for a hearing under § 2.106 of this chapter (if a hearing is requested, which will be held after issuance of the amendment).

(13) A manufacturing license under subpart F of part 52 or subparts H or R of part 53 of this chapter.

(b) * * *

(3) For a notice of intended operation under §§ 52.103(a), 53.1452(a), or 53.5052(a) of this chapter, the following information:

(i) The identification of the NRC action as making the finding required under §§ 52.103(g), 53.1452(g), or 53.5052(g) of this chapter;

(ii) The manner in which the licensee notifications under 10 CFR 52.99(c), 53.1449(c), or 53.5049(c) which are required to be made available by 10 CFR 52.99(e)(2), 53.1449(e)(2), or 53.5049(e)(2) may be obtained and examined;

* * *

(iv) Any conditions, limitations, or restrictions to be placed on the license in connection with the finding under §§ 52.103(g), 53.1452(g), or 53.5052(g) of this chapter, and the expiration date or circumstances (if any) under which the conditions, limitations or restrictions will no longer apply.

* * * * *

8. Revise § 2.106, paragraphs (a)(2) and (3) and paragraph (b)(2) to read as follows:

§ 2.106 Notice of issuance.

(a) * * *

(2) An amendment of a license for a facility of the type described in §§ 50.21(b) or § 50.22, or part 53 of this chapter, as applicable, or a testing facility, whether or not a notice of proposed action has been previously published; and

(3) The finding under §§ 52.103(g), 53.1452(g), 53.5052(g) of this chapter.

(b) * * *

(2) In the case of a finding under §§ 52.103(g), 53.1452(g), or 53.5052(g) of this chapter:

* * * * *

9. Revise § 2.109, paragraphs (b), (c), and (d) to read as follows:

§ 2.109 Effect of timely renewal application.

* * * * *

(b) If the licensee of a nuclear power plant licensed under §§10 CFR 50.21(b) or 50.22, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter files a sufficient application for renewal of either an operating license or a combined license

at least 5 years before the expiration of the existing license, the existing license will not be deemed to have expired until the application has been finally determined.

(c) If the holder of an early site permit licensed under subpart A of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter, as applicable, files a sufficient application for renewal under §§ 52.29, 53.1173, or 53.4783 of this chapter, as applicable, at least 12 months before the expiration of the existing early site permit, the existing permit will not be deemed to have expired until the application has been finally determined.

(d) If the licensee of a manufacturing license under subpart F of part 52, or §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter files a sufficient application for renewal under §§ 52.177, 53.1295, or 53.4895 of this chapter at least 12 months before the expiration of the existing license, the existing license will not be deemed to have expired until the application has been finally determined.

* * * * *

10. Revise § 2.110, paragraphs (a)(1) and (b) to read as follows:

§ 2.110 Filing and administrative action on submittals for standard design approval or early review of site suitability issues.

(a)(1) A submittal for a standard design approval under subpart E of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter shall be subject to §§ 2.101(a) and 2.390 to the same extent as if it were an application for a permit or license.

* * *

(b) Upon initiation of review by the NRC staff of a submittal for an early review of site suitability issues under Appendix Q of part 50 of this chapter, or for a standard design approval under subpart E of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700

through 53.5080 of this chapter, the Director, Office of Nuclear Reactor Regulation, shall publish in the *Federal Register* a notice of receipt of the submittal, inviting comments from interested persons within 60 days of publication or other time as may be specified, for consideration by the NRC staff and ACRS in their review.

* * * * *

Subpart B – Procedure for Imposing Requirements by Order, or for Modification, Suspension, or Revocation of a License, or for Imposing Civil Penalties

11. Revise § 2.202, paragraph (e) to read as follows:

§ 2.202 Orders.

* * * * *

(e)(1) If the order involves the modification of a part 50 or part 53 license and is a backfit, the requirements of §§ 50.109, 53.1590, or 53.6090 of this chapter, as applicable, shall be followed, unless the licensee has consented to the action required.

(2) If the order involves the modification of combined license under subpart C of part 52, or subparts H or R of part 53 of this chapter, the requirements of §§ 52.98, 53.1443, or 53.5043 of this chapter, as applicable, shall be followed unless the licensee has consented to the action required.

(3) If the order involves a change to an early site permit under subpart A of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter, the requirements of §§ 52.39, 53.1188, or 53.4798 of this chapter, as applicable, must be followed, unless the applicant or licensee has consented to the action required.

(4) If the order involves a change to a standard design certification rule referenced by that plant's application, the requirements, if any, in the referenced design certification rule with respect to changes must be followed, or, in the absence of these requirements, the requirements of §§ 52.63, 53.1263, or 53.4863 of this chapter, as

applicable, must be followed, unless the applicant or licensee has consented to follow the action required.

(5) If the order involves a change to a standard design approval referenced by that plant's application, the requirements of §§ 52.145, 53.1221, or 53.4821 of this chapter, as applicable, must be followed unless the applicant or licensee has consented to follow the action required.

(6) If the order involves a modification of a manufacturing license under subpart F of part 52, the requirements of §§ 52.171, 53.1288, or 53.4888 of this chapter, as applicable, must be followed, unless the applicant or licensee has consented to the action required.

Subpart C – Rules of General Applicability: Hearing Requests, Petitions to Intervene, Availability of Documents, Selection of Specific Hearing Procedures, Presiding Officer Powers, and General Hearing Management for NRC Adjudicatory Hearings

12. Revise § 2.309, paragraphs (a), (f)(1)(i), (vi), and (vii), (g), (h)(2), (i)(2), and (j)(1) and (2) to read as follows:

§ 2.309 Hearing requests, petitions to intervene, requirements for standing, and contentions.

(a) *General requirements*. Any person whose interest may be affected by a proceeding and who desires to participate as a party must file a written request for hearing and a specification of the contentions which the person seeks to have litigated in the hearing. In a proceeding under §§ ~~10 CFR~~ 52.103, 53.1452, or 53.5052, as applicable, the Commission, acting as the presiding officer, will grant the request if it determines that the requestor has standing under the provisions of paragraph (d) of this section and has proposed at least one admissible contention that meets the

requirements of paragraph (f) of this section. For all other proceedings, except as provided in paragraph (e) of this section, the Commission, presiding officer, or the Atomic Safety and Licensing Board designated to rule on the request for hearing and/or petition for leave to intervene, will grant the request/petition if it determines that the requestor/petitioner has standing under the provisions of paragraph (d) of this section and has proposed at least one admissible contention that meets the requirements of paragraph (f) of this section. In ruling on the request for hearing/petition to intervene submitted by petitioners seeking to intervene in the proceeding on the HLW repository, the Commission, the presiding officer, or the Atomic Safety and Licensing Board shall also consider any failure of the petitioner to participate as a potential party in the pre-license application phase under subpart J of this part in addition to the factors in paragraph (d) of this section. If a request for hearing or petition to intervene is filed in response to any notice of hearing or opportunity for hearing, the applicant/licensee shall be deemed to be a party.

* * * * *

(f) * * *

(1) * * *

(i) Provide a specific statement of the issue of law or fact to be raised or controverted, provided further, that the issue of law or fact to be raised in a request for hearing under §§ ~~10 CFR~~ 52.103(b), 53.1452(b), or 53.5052(b), as applicable, must be directed at demonstrating that one or more of the acceptance criteria in the combined license have not been, or will not be met, and that the specific operational consequences of nonconformance would be contrary to providing reasonable assurance of adequate protection of the public health and safety;

* * *

(vi) In a proceeding other than one under §§ ~~10 CFR~~ 52.103, 53.1452, or 53.5052, provide sufficient information to show that a genuine dispute exists with the applicant/licensee on a material issue of law or fact. This information must include references to specific portions of the application (including the applicant's environmental report and safety report) that the petitioner disputes and the supporting reasons for each dispute, or, if the petitioner believes that the application fails to contain information on a relevant matter as required by law, the identification of each failure and the supporting reasons for the petitioner's belief; and

(vii) In a proceeding under §§ ~~10 CFR~~ 52.103(b), 53.1452(b), or 53.5052(b), as applicable, the information must be sufficient, and include supporting information showing, prima facie, that one or more of the acceptance criteria in the combined license have not been, or will not be met, and that the specific operational consequences of nonconformance would be contrary to providing reasonable assurance of adequate protection of the public health and safety. This information must include the specific portion of the report required by §§ ~~10 CFR~~ 52.99(c), 53.1449(c), or 53.5049(c), as applicable, which the requestor believes is inaccurate, incorrect, and/or incomplete (i.e., fails to contain the necessary information required by §§ 52.99(c), 53.1449(c), or 53.5049(c), as applicable). If the requestor identifies a specific portion of the §§ 52.99(c), 53.1449(c), or 53.5049(c), as applicable, report as incomplete and the requestor contends that the incomplete portion prevents the requestor from making the necessary *prima facie* showing, then the requestor must explain why this deficiency prevents the requestor from making the *prima facie* showing.

* * *

(g) *Selection of hearing procedures*. A request for hearing and/or petition for leave to intervene may, except in a proceeding under §§ ~~10 CFR~~ 52.103, 53.1452, or

53.5052, as applicable, also address the selection of hearing procedures, taking into account the provisions of § 2.310. If a request/petition relies upon § 2.310(d), the request/petition must demonstrate, by reference to the contention and the bases provided and the specific procedures in subpart G of this part, that resolution of the contention necessitates resolution of material issues of fact which may be best determined through the use of the identified procedures.

(h) * * *

(1) * * *

(2) If the proceeding pertains to a production or utilization facility (as defined in §§ 50.2 or 53.020 of this chapter) located within the boundaries of the State, local governmental body, or Federally-recognized Indian Tribe seeking to participate as a party, no further demonstration of standing is required. If the production or utilization facility is not located within the boundaries of the State, local governmental body, or Federally-recognized Indian Tribe seeking to participate as a party, the State, local governmental body, or Federally-recognized Indian Tribe also must demonstrate standing.

* * *

(i) * * *

(2) Except in a proceeding under §§ 52.103, 53.1452, or 53.5052 of this chapter, as applicable, the participant who filed the hearing request, intervention petition, or motion for leave to file new or amended contentions after the deadline may file a reply to any answer. The reply must be filed within 7 days after service of that answer.

* * *

(j) * * *

(1) In all proceedings other than a proceeding under §§ 52.103, 53.1452, or 53.5052 of this chapter, as applicable, the presiding officer shall issue a decision on each request for hearing or petition to intervene within 45 days of the conclusion of the initial pre-hearing conference or, if no pre-hearing conference is conducted, within 45 days after the filing of answers and replies under paragraph (i) of this section. With respect to a request to admit amended or new contentions, the presiding officer shall issue a decision on each such request within 45 days of the conclusion of any pre-hearing conference that may be conducted regarding the proposed amended or new contentions or, if no pre-hearing conference is conducted, within 45 days after the filing of answers and replies, if any. In the event the presiding officer cannot issue a decision within 45 days, the presiding officer shall issue a notice advising the Commission and the parties, and the notice shall include the expected date of when the decision will issue.

(2) The Commission, acting as the presiding officer, shall expeditiously grant or deny the request for hearing in a proceeding under §§ 52.103, 53.1452, or 53.5052 of this chapter, as applicable. The Commission's decision may not be the subject of any appeal under § 2.311.

13. Amend § 2.310 by revising paragraph (a), the introductory text for paragraph (h), and paragraphs (i) and (j) to read as follows:

§ 2.310 Selection of hearing procedures.

* * * * *

(a) Except as determined through the application of paragraphs (b) through (h) of this section, proceedings for the grant, renewal, licensee-initiated amendment, or termination of licenses or permits subject to parts 30, 32 through 36, 39, 40, 50, 52, 53,

54, 55, 61, 70 and 72 of this chapter may be conducted under the procedures of subpart L of this part.

* * * * *

(h) Except as determined through the application of paragraphs (b) through (g) of this section, proceedings for the grant, renewal, licensee-initiated amendment, or termination of licenses or permits subject to parts 30, 32 through 36, 39, 40, 50, 52, 53, 54, 55, 61, 70 and 72 of this chapter, and proceedings on an application for the direct or indirect transfer of control of an NRC license may be conducted under the procedures of subpart N of this part if—

* * *

(i) In design certification rulemaking proceedings under part 52 or part 53 of this chapter, any informal hearing held under §§ 52.51, 53.1242, or 53.4842 of this chapter, as applicable, must be conducted under the procedures of subpart O of this part.

(j) Proceedings on a Commission finding under §§ ~~10 CFR~~ 52.103(c) and (g); or 53.1452(c) and (g); or 53.5052(c) and (g), as applicable, shall be conducted in accordance with the procedures designated by the Commission in each proceeding.

* * * * *

14. Revise § 2.329, paragraph (a) as follows:

§ 2.329 Prehearing conference.

(a) Necessity for prehearing conference; timing. The Commission or the presiding officer may, and in the case of a proceeding on an application for a construction permit or an operating license for a facility of a type described in §§ 50.21(b) or 50.22, or part 53 of this chapter, or a testing facility, shall direct the parties or their counsel to appear at a specified time and place for a conference or conferences before trial. A prehearing conference in a proceeding involving a

construction permit or operating license for a facility of a type described in §§ 50.21(b) or 50.22 or part 53 of this chapter must be held within sixty (60) days after discovery has been completed or any other time specified by the Commission or the presiding officer.

* * * * *

15. Revise § 2.339, paragraph (d) to read as follows:

§ 2.339 Expedited decisionmaking procedure.

* * * * *

(d) The provisions of this section do not apply to an initial decision directing the issuance of a limited work authorization under §§ ~~10 CFR~~ 50.10, 53.1130, or 53.4740;~~,~~ an early site permit under subpart A of part 52, or §§ 53.1140 or 53.4750 of this chapter;~~,~~ a construction permit or construction authorization;~~,~~ a combined license under subpart C of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter;~~,~~ or a manufacturing license under subpart F of part 52, §§ 53.1270 or 53.4870.

16. Amend § 2.340 as follows:

a. Revise the introductory text for paragraph (b) and paragraphs (b)(1) and (2);

b. Revise paragraph (c);

c. Revise the introductory text for paragraph (d) and paragraphs (d)(1) and (2);

d. Revise paragraph (f);

e. Revise paragraph (i); and

f. Revise the introductory text for paragraph (j) and paragraph (j)(1)

§ 2.340 Initial decision in certain contested proceedings; immediate effectiveness of initial decisions; issuance of authorizations, permits and licenses.

* * * * *

(b) *Initial decision—combined license under 10 CFR part 52 or part 53*. (1) Matters in controversy; presiding officer consideration of matters not put in controversy by parties. In any initial decision in a contested proceeding on an application for a combined license under part 52 or part 53 of this chapter (including an amendment to or renewal of a combined license), the presiding officer shall make findings of fact and conclusions of law on the matters put into controversy by the parties and any matter designated by the Commission to be decided by the presiding officer. The presiding officer shall also make findings of fact and conclusions of law on any matter not put into controversy by the parties, but only to the extent that the presiding officer determines that a serious safety, environmental, or common defense and security matter exists, and the Commission approves of an examination of and decision on the matter upon its referral by the presiding officer under, inter alia, the provisions of §§ 2.323 and 2.341.

(2) Presiding officer initial decision and issuance of permit or license. (i) In a contested proceeding for the initial issuance or renewal of a combined license under part 52 or part 53 of this chapter, or the amendment of a combined license where the NRC has not made a determination of no significant hazards consideration, the Commission or the Director, Office of Nuclear Reactor Regulation, as appropriate, after making the requisite findings, shall issue, deny, or appropriately condition the permit or license in accordance with the presiding officer's initial decision once that decision becomes effective.

(ii) In a contested proceeding for the amendment of a combined license under part 52 or part 53 of this chapter where the NRC has made a determination of no significant hazards consideration, the Commission or the Director, Office of Nuclear Reactor Regulation, as appropriate (appropriate official), after making the requisite findings and complying with any applicable provisions of §§ 2.1202(a) or ~~§~~ 2.1403(a),

may issue the amendment before the presiding officer's initial decision becomes effective. Once the presiding officer's initial decision becomes effective, the appropriate official shall take action with respect to that amendment in accordance with the initial decision. If the presiding officer's initial decision becomes effective before the appropriate official issues the amendment, then the appropriate official, after making the requisite findings, shall issue, deny, or appropriately condition the amendment in accordance with the presiding officer's initial decision.

(c) *Initial decision on findings under* §§ ~~10 CFR~~*52.103*, *53.1452, or 53.5052 with respect to acceptance criteria in nuclear power reactor combined licenses*. In any initial decision under §§ 52.103(g), 53.1452(g), or 53.5052(g) of this chapter with respect to whether acceptance criteria have been or will be met, the presiding officer shall make findings of fact and conclusions of law on the matters put into controversy by the parties, and any matter designated by the Commission to be decided by the presiding officer. Matters not put into controversy by the parties, but identified by the presiding officer as matters requiring further examination, shall be referred to the Commission for its determination; the Commission may, in its discretion, treat any of these referred matters as a request for action under § 2.206 and process the matter in accordance with §§ 52.103(f), 53.1452(f), or 53.5052(f) of this chapter.

(d) *Initial decision—manufacturing license under 10 CFR part 52 or part 53*. (1) Matters in controversy; presiding officer consideration of matters not put in controversy by parties. In any initial decision in a contested proceeding on an application for a manufacturing license under subpart F of part 52, or subparts H or R of part 53 of this chapter (including an amendment to or renewal of a manufacturing license), the presiding officer shall make findings of fact and conclusions of law on the matters put into controversy by the parties and any matter designated by the Commission to be

decided by the presiding officer. The presiding officer also shall make findings of fact and conclusions of law on any matter not put into controversy by the parties, but only to the extent that the presiding officer determines that a serious safety, environmental, or common defense and security matter exists, and the Commission approves of an examination of and decision on the matter upon its referral by the presiding officer under, inter alia, the provisions of §§ 2.323 and 2.341.

(2) Presiding officer initial decision and issuance of permit or license. (i) In a contested proceeding for the initial issuance or renewal of a manufacturing license under subpart C of part 52, or subparts H or R of part 53of this chapter, or the amendment of a manufacturing license, the Commission or the Director, Office of Nuclear Reactor Regulation, as appropriate, after making the requisite findings, shall issue, deny, or appropriately condition the permit or license in accordance with the presiding officer's initial decision once that decision becomes effective.

(ii) In a contested proceeding for the initial issuance or renewal of a manufacturing license under subpart F of part 52, or subparts H or R of part 53of this chapter, or the amendment of a manufacturing license, the Commission or the Director, Office of Nuclear Reactor Regulation, as appropriate, may issue the license, permit, or license amendment in accordance with §§ 2.1202(a) or ~~§~~ 2.1403(a) before the presiding officer's initial decision becomes effective. If, however, the presiding officer's initial decision becomes effective before the license, permit, or license amendment is issued under §§ 2.1202 or ~~§~~ 2.1403, then the Commission or the Director, Office of Nuclear Reactor Regulation, as appropriate, shall issue, deny, or appropriately condition the license, permit, or license amendment in accordance with the presiding officer's initial decision.

* * * * *

(f) *Immediate effectiveness of certain presiding officer decisions*. A presiding officer's initial decision directing the issuance or amendment of a limited work authorization under §§ 50.10, 53.1130, or 53.4740 of this chapter; an early site permit under subpart A of part 52; §§ 53.1100 through 53.1480; or §§ 53.4700 through 53.5080 of this chapter; a construction permit or construction authorization under part 50 or part 53 of this chapter; an operating license under part 50 or part 53 of this chapter; a combined license under subpart C of part 52 or part 53 of this chapter; a manufacturing license under subpart F of part 52 or part 53 of this chapter; a renewed license under part 54 or part 53, or a license under part 72 of this chapter to store spent fuel in an independent spent fuel storage facility (ISFSI) or a monitored retrievable storage installation (MRS); an initial decision directing issuance of a license under part 61 of this chapter; or an initial decision under §§ 52.103(g), 53.1452(g), or 53.5052(g) of this chapter that acceptance criteria in a combined license have been met, is immediately effective upon issuance unless the presiding officer finds that good cause has been shown by a party why the initial decision should not become immediately effective.

* * * * *

(i) *Issuance of authorizations, permits, and licenses—production and utilization facilities*. The Commission or the Director, Office of Nuclear Reactor Regulation, as appropriate, shall issue a limited work authorization under §§ 50.10, 53.1130, or 53.4740 of this chapter; an early site permit under subpart A of part 52 or subparts H or R of part 53 of this chapter; a construction permit or construction authorization under part 50 or part 53 of this chapter; an operating license under part 50 or part 53 of this chapter; a combined license under subpart C of part 52 or part 53 of this chapter; or a

manufacturing license under subpart F of part 52 or part 53 of this chapter within 10 days from the date of issuance of the initial decision:

* * *

(j) *Issuance of finding on acceptance criteria under §§ ~~10 CFR~~ 52.103, 53.1452, or 53.5052*. The Commission or the Director, Office of Nuclear Reactor Regulation, as appropriate, shall make the finding under §§ ~~10 CFR~~ 52.103(g), 53.1452(g), or 53.5052(g) that acceptance criteria in a combined license are met within 10 days from the date of the presiding officer's initial decision:

(1) If the Commission or the Director is otherwise able to make the finding under §§ ~~10 CFR~~ 52.103(g), 53.1452(g), or 53.5052(g) that the prescribed acceptance criteria are met for those acceptance criteria not within the scope of the initial decision of the presiding officer;

* * *

* * * * *

17. Revise § 2.341, paragraph (a)(1) as follows:

§ 2.341 Review of decisions and actions of a presiding officer.

(a)(1) Review of decisions and actions of a presiding officer are treated under this section; provided, however, that no party may request further Commission review of a Commission determination to allow a period of interim operation under §§ 52.103(c), 53.1452(c), or 53.5052(c) of this chapter. This section does not apply to appeals under § 2.311 or to appeals in the high-level waste proceeding, which are governed by § 2.1015.

* * * * *

Subpart D – Additional Procedures Applicable to Proceedings for the Issuance of Licenses to Construct and/or Operate Nuclear Power Plants of Identical Design at Multiple Sites

18. Revise § 2.400 to read as follows:

§ 2.400 Scope of subpart.

This subpart describes procedures applicable to licensing proceedings which involve the consideration in hearings of a number of applications, filed by one or more applicants pursuant to appendix N of parts 50 or 52, or §§ 53.1470 or 53.5070 of this chapter, for licenses to construct and/or operate nuclear power reactors of identical design to be located at multiple sites.

19. Revise § 2.401, paragraph (a) to read as follows:

§ 2.401 Notice of hearing on construction permit or combined license applications pursuant to appendix N of 10 CFR parts 50 or 52, or part 53.

(a) In the case of applications pursuant to appendix N of part 50, or §§ 53.1470 or 53.5070 of this chapter for construction permits for nuclear power reactors of the type described in § 50.22 or part 53 of this chapter, or applications pursuant to appendix N of part 52, or §§ 53.1470 or 53.5070 of this chapter for combined licenses, the Secretary will issue notices of hearing pursuant to § 2.104.

* * * * *

20. Revise § 2.402, paragraph (a) as follows:

§ 2.402 Separate hearings on separate issues; consolidation of proceedings.

(a) In the case of applications under appendix N of part 50, or §§ 53.1470 or 53.5070 of this chapter for construction permits for nuclear power reactors of a type described in § ~~10 CFR~~ 50.22 or part 53, or applications pursuant to appendix N of part 52, or §§ 53.1470 or 53.5070 of this chapter for combined licenses, the Commission or

the presiding officer may order separate hearings on particular phases of the proceeding, such as matters related to the acceptability of the design of the reactor, in the context of the site parameters postulated for the design or environmental matters.

* * * * *

21. Revise § 2.403 as follows:

§ 2.403 Notice of proposed action on applications for operating licenses pursuant to appendix N of 10 CFR part 50.

In the case of applications pursuant to appendix N of part 50, or §§ 53.1470 or 53.5070 of this chapter for operating licenses for nuclear power reactors, if the Commission has not found that a hearing is in the public interest, the Commission or the Director, Office of Nuclear Reactor Regulation, as appropriate, will, prior to acting thereon, cause to be published in the *Federal Register*, pursuant to § 2.105, a notice of proposed action with respect to each application as soon as practicable after the applications have been docketed.

22. Revise § 2.404 as follows:

§ 2.404 Hearings on applications for operating licenses pursuant to appendix N of 10 CFR part 50.

If a request for a hearing and/or petition for leave to intervene is filed within the time prescribed in the notice of proposed action on an application for an operating license pursuant to appendix N of part 50, or §§ 53.1470 or 53.5070 of this chapter with respect to a specific reactor(s) at a specific site, and the Commission, the Chief Administrative Judge, or a presiding officer has issued a notice of hearing or other appropriate order, then the Commission, the Chief Administrative Judge, or the presiding officer may order separate hearings on particular phases of the proceeding and/or consolidate for hearing two or more proceedings in the manner described in § 2.402.

23. Revise the fourth sentence of § 2.405 to read as follows:

§ 2.405 Initial decisions in consolidated hearings.

* * * No construction permit, full-power operating license, or combined license under part 52 or part 53 of this chapter will be issued until an initial decision has been issued on all phases of the hearing and all issues under the Act and the National Environmental Policy Act of 1969 appropriate to the proceeding have been resolved.

24. Revise § 2.406 to read as follows:

§ 2.406 Finality of decisions on separate issues.

Notwithstanding any other provision of this chapter, in a proceeding conducted pursuant to this subpart and appendices N of parts 50 or 52, or §§ 53.1470 or 53.5070 of this chapter, no matter which has been reserved for consideration in one phase of the hearing shall be considered at another phase of the hearing except on the basis of significant new information that substantially affects the conclusion(s) reached at the other phase or other good cause.

Subpart E – Additional Procedures Applicable to Proceedings for the Issuance of Licenses to Manufacture Nuclear Power Reactors to be Operated at Sites Not Identified in the License Application and Related Licensing Proceedings

25. Revise § 2.500 to read as follows:

§ 2.500 Scope of subpart.

This subpart prescribes procedures applicable to licensing proceedings which involve the consideration in separate hearings of an application for a license to manufacture nuclear power reactors under subpart F of part 52~~,~~ or subparts H or R of part 53 of this chapter.

26. In § 2.501, revise the introductory text for paragraph (a) and revise paragraph (b)(1)(vii) and the introductory text for paragraph (b)(3) to read as follows:

§ 2.501 Notice of hearing on application under subpart F of 10 CFR part 52 for a license to manufacture nuclear power reactors.

(a) In the case of an application under subpart F of part 52, or subparts H or R of part 53 of this chapter for a license to manufacture nuclear power reactors of the type described in § 50.22 or part 53 of this chapter to be operated at sites not identified in the license application, the Secretary will issue a notice of hearing to be published in the *Federal Register* at least 30 days before the date set for hearing in the notice.[1] The notice shall be issued as soon as practicable after the application has been docketed. The notice will state:

* * * * *

Subpart F – Additional Procedures Applicable to Early Partial Decisions on Site Suitability Issues in Connection with an Application for a Construction Permit or Combined License to Construct Certain Utilization Facilities; and Advance Issuance of Limited Work Authorizations

27. Revise § 2.643, paragraph (b) to read as follows:

§ 2.643 Acceptance and docketing of application for limited work authorization.

* * * * *

(b) The Director will accept for docketing part one of an application for a construction permit for a utilization facility which is subject to § 51.20(b) of this chapter and is of the type specified in § 50.21(b)(2) or (3); or § 50.22; or part 53 of this chapter, or an application for a combined license where part one of the application as described in § 2.101(a)(9) is complete. Part one will not be considered complete unless it contains the information required by §§ 50.10(d)(3), 53.1130(a)(3), or 53.4740(a)(3) of this chapter. Upon assignment of a docket number, the procedures in § 2.101(a)(3) and (4)

relating to formal docketing and the submission and distribution of additional copies of the application must be followed.

* * * * *

28. Revise § 2.645, paragraph (a) to read as follows:

§ 2.645 Notice of hearing.

(a) The notice of hearing on part one of the application must set forth the matters of fact and law to be considered, as required by § 2.104, which will be modified to state that the hearing will relate only to the matters related to §§ 50.33(a) through (f), 53.1109, or 53.4709 of this chapter, and the limited work authorization.

* * * * *

29. Revise the third sentence of § 2.649 to read as follows:

§ 2.649 Partial decisions on limited work authorization.

* * * A limited work authorization may not be issued under §§ ~~10 CFR~~ 50.10(d), 53.1130(a), or 53.4740(a) without completion of the review for limited work authorizations required by subpart A of part 51 of this chapter. * * *

Subpart H – Rulemaking

30. Revise § 2.800, paragraphs (c) and (d) to read as follows:

§ 2.800 Scope and applicability.

* * * * *

(c) The procedures in §§ 2.802 through 2.803 apply to all petitions for rulemaking except for initial applications for standard design certification rulemaking under subpart B of part 52, subpart H of part 53, or subpart R of part 53 of this chapter, and subsequent petitions for amendment of an existing design certification rule filed by the original applicant for the design certification rule.

(d) The procedures in §§ 2.811 through 2.819, as supplemented by the provisions of subpart B of part 52, or subparts H or R of part, as applicable, apply to standard design certification rulemaking.

31. Revise § 2.801 to read as follows:

§ 2.801 Initiation of rulemaking.

Rulemaking may be initiated by the Commission at its own instance, on the recommendation of another agency of the United States, or on the petition of any other interested person, including an application for design certification under subpart B of part 52, or subparts H or R of part 53 of this chapter.

32. Revise § 2.813, paragraph (a) to read as follows:

§ 2.813 Written communications.

(a) General requirements. All correspondence, reports, and other written communications from the applicant to the Nuclear Regulatory Commission concerning the regulations in this subpart, and parts 50, 52, 53, and 100 of this chapter 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/e-submittals.html*; by e-mail to *MSHD.Resource@nrc.gov*; or by writing the Office of the Chief Information Officer, 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.

* * * * *

Subpart K – Hybrid Hearing Procedures for Expansion of Spent Nuclear Fuel Storage Capacity at Civilian Nuclear Power Reactors

33. Revise the first sentence of § 2.1103 to read as follows:

§ 2.1103 Scope of subpart K.

The provisions of this subpart, together with subpart C and applicable provisions of subparts G and L of this part, govern all adjudicatory proceedings on applications filed after January 7, 1983, for a license or license amendment under part 50 or part 53 of this chapter, to expand the spent fuel storage capacity at the site of a civilian nuclear power plant, through the use of high density fuel storage racks, fuel rod compaction, the transshipment of spent nuclear fuel to another civilian nuclear power reactor within the same utility system, the construction of additional spent nuclear fuel pool capacity or dry storage capacity, or by other means. * * *

Subpart L – Simplified Hearing Procedures for NRC Adjudications

34. Revise § 2.1202, paragraphs (a)(1), (a)(2), (a)(3), and (a)(6) to read as follows:

§ 2.1202 Authority and role of NRC staff.

(a) * * *

(1) An application to construct and/or operate a production or utilization facility (including an application for a limited work authorization under §§~~10 CFR~~ 50.12,

53.1130, or 53.4740, or an application for a combined license under subpart C of 10 CFR part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080;

(2) An application for an early site permit under subpart A of 10 CFR part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080;

(3) An application for a manufacturing license under subpart F of 10 CFR part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080;

* * *

(6) Production or utilization facility licensing actions that involve significant hazards considerations as defined in §§~~10 CFR~~ 50.92, 53.1520, or 53.6020.

* * * * *

Subpart M – Procedures for Hearings on License Transfer Applications

35. Revise § 2.1301, paragraph (b) to read as follows:

§ 2.1301 Public notice of receipt of a license transfer application.

* * * * *

(b) The Commission will also publish in the *Federal Register* a notice of receipt of an application for approval of a license transfer involving 10 CFR part 50, part 52, and part 53 licenses, major fuel cycle facility licenses issued under part 70, or part 72 licenses. This notice constitutes the notice required by § 2.105 with respect to all matters related to the application requiring NRC approval.

* * * * *

Subpart N – Expedited Proceedings with Oral Hearings

36. Revise § 2.1403, paragraph (a)(3) to read as follows:

§ 2.1403 Authority and role of the NRC staff.

(a) * * *

(3) Production or utilization facility licensing actions that involve significant hazards considerations as defined in §§~~10 CFR~~ 50.92, 53.1520, or 53.6020.

* * * * *

Subpart O – Legislative Hearings

37. Revise § 2.1500, paragraph (a) to read as follows:

§ 2.1500 Purpose and scope.

* * * * *

(a) Any design certification rulemaking hearings under subpart B of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter that the Commission may choose to conduct; and

* * * * *

38. Revise § 2.1502, paragraphs (a) and (b)(1) to read as follows:

§ 2.1502 Commission decision to hold legislative hearing.

(a) The Commission may, in its discretion, hold a legislative hearing in either a design certification rulemaking under §§ 52.51(b), 53.1242(b)(2), or 53.4842(b)(2) of this chapter, or a proceeding where a question has been certified to it under § 2.335(d).

(b) * * *

(1) Hearing in design certification rulemakings. If, at the time a proposed design certification rule is published in the *Federal Register* under §§ 52.51(a), 53.1242(b), or 53.4842(b) of this chapter, the Commission decides that a legislative hearing should be held, the information required by paragraph (c) of this section must be included in the *Federal Register* notice for the proposed design certification rule. If, following the submission of written public comments submitted on the proposed design certification rule which are submitted in accordance with §§ 52.51(a), 53.1242(b), or 53.4842(b) of this chapter, the Commission decides to conduct a legislative hearing, the Commission

shall publish a notice in the *Federal Register* and on the NRC Web site indicating its determination to conduct a legislative hearing. The notice 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.

* * * * *

PART 10 – CRITERIA AND PROCEDURES FOR DETERMINING ELIGIBILITY FOR ACCESS TO RESTRICTED DATA OR NATIONAL SECURITY INFORMATION OR AN EMPLOYMENT CLEARANCE

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

Authority: Atomic Energy Act of 1954, secs. 145, 161 (42 U.S.C. 2165, 2201); Energy Reorganization Act of 1974, sec. 201 (42 U.S.C. 5841); E.O. 10450, 18 FR 2489, 3 CFR, 1949-1953 Comp., p. 936, as amended; E.O. 10865, 25 FR 1583, 3 CFR, 1959-1963 Comp., p. 398, as amended; E.O. 12968, 60 FR 40245, 3 CFR, 1995 Comp., p. 391.

Subpart A – General Provisions

2. Revise § 10.1, paragraph (a)(3) to read as follows:

§ 10.1 Purpose.

(a) * * *

(3) The eligibility of individuals who are employed by or are applicants for employment with NRC licensees, certificate holders, holders of standard design approvals under part 52 or part 53 of this chapter, applicants for licenses, certificates, and NRC approvals, and others who may require access related to a license, certificate, or NRC approval, or other activities as the Commission may determine, for access to Restricted Data under the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974, or for access to national security information.

* * * * *

3. Revise § 10.2, paragraph (b) to read as follows:

§ 10.2 Scope.

* * * * *

(b) NRC licensees, certificate holders and holders of standard design approvals under part 52 or part 53 of this chapter, applicants for licenses, certificates, and standard design approvals under part 52 or part 53 of this chapter, and their employees (including consultants) and applicants for employment (including consulting);

* * * * *

PART 11 – Criteria and Procedures for Determining Eligibility for Access to or Control Over Special Nuclear Material

1. The authority citation for 10 CFR part 11 continues to read as follows:

Authority: Atomic Energy Act of 1954, secs. 161, 223 (42 U.S.C. 2201, 2273); Energy Reorganization Act of 1974, sec. 201 (42 U.S.C. 5841); 44 U.S.C. 3504 note. Section 11.15(e) also issued under 31 U.S.C. 9701; 42 U.S.C. 2214.

2. Revise § 11.7 to read as follows:

§ 11.7 Definitions.

Terms defined in parts 10, 25, 50, 53, 70, 72, 73, and 95 of this chapter have the same meaning when used in this part.

* * * * *

PART 19 – NOTICES, INSTRUCTIONS AND REPORTS TO WORKERS: INSPECTION AND INVESTIGATIONS

1. The authority citation for part 19 continues to read as follows:

Authority: Atomic Energy Act of 1954, secs. 53, 63, 81, 103, 104, 161, 223, 234, 1701 (42 U.S.C. 2073, 2093, 2111, 2133, 2134, 2201, 2273, 2282, 2297f); Energy Reorganization Act of 1974, secs. 201, 211, 401 (42 U.S.C. 5841, 5851, 5891); 44 U.S.C. 3504 note.

2. Revise § 19.2, paragraphs (a)(1) – (4) to read as follows:

§ 19.2 Scope.

(a) * * *

(1) All persons who receive, possess, use, or transfer material licensed by the NRC under the regulations in parts 30 through 36, 39, 40, 60, 61, 63, 70, or 72 of this chapter, including persons licensed to operate a production or utilization facility under parts 50, 52, or 53 of this chapter, persons licensed to possess power reactor spent fuel in an independent spent fuel storage installation (ISFSI) under part 72 of this chapter, and in accordance with ~~10 CFR~~ 76.60 to persons required to obtain a certificate of compliance or an approved compliance plan under part 76 of this chapter;

(2) All applicants for and holders of licenses (including construction permits and early site permits) under parts 50, 52, 53 and 54 of this chapter;

(3) All applicants for and holders of a standard design approval under subpart E of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter; and

(4) All applicants for a standard design certification under subpart B of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter, and those (former) applicants whose designs have been certified under that subpart.

* * * * *

3. Revise § 19.3 to read as follows:

§ 19.3 Definitions.

* * * * *

License means a license issued under the regulations in parts 30 through 36,39, 40, 60, 61, 63, 70, or 72 of this chapter, including licenses to manufacture, construct and/or operate a production or utilization facility under parts 50, 52, 53, or 54 of this chapter.

* * * * *

Regulated entities means any individual, person, organization, or corporation that is subject to the regulatory jurisdiction of the NRC, including (but not limited to) an applicant for or holder of a standard design approval under subpart E of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter or a standard design certification under subpart B of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter.

* * * * *

4. In § 19.11, revise the introductory text for paragraphs (a) and (b) and revise paragraph (e)(1) to read as follows:

§ 19.11 Posting of notices to workers.

(a) Each licensee (except for a holder of an early site permit under subpart A of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter, or a holder of a manufacturing license under subpart F of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter) shall post current copies of the following documents:

* * *

(b) Each applicant for and holder of a standard design approval under subpart E of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter;~~,~~ each applicant for an early site permit under subpart A of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter;~~,~~ each applicant for a standard design certification under subpart B of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter;~~,~~ and each applicant for and holder of a manufacturing license under subpart F of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter shall post:

* * * * *

(e)(1) Each licensee, each applicant for a specific license, each applicant for or holder of a standard design approval under subpart E of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter;~~,~~ each applicant for an early site permit under subpart A of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter;~~,~~ and each applicant for a standard design certification under subpart B of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter shall prominently post NRC Form 3, "Notice to Employees," dated August 1997. Later versions of NRC Form 3 that supersede the August 1997 version shall replace the previously posted version within 30 days of receiving the revised NRC Form 3 from the Commission.

* * * * *

5. Revise § 19.14, paragraph (a) to read as follows:

§ 19.14 Presence of representatives of licenses and regulated entities, and workers during inspections.

(a) Each licensee, applicant for a license, applicant for or holder of a standard design approval under subpart E of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter;~~,~~ applicant for an early site permit under subpart A of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter;~~,~~ and applicant for a standard design certification under subpart B of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter shall afford to the Commission at all reasonable times opportunity to inspect materials, activities, facilities, premises, and records under the regulations in this chapter.

* * * * *

6. Revise § 19.20 to read as follows:

§ 19.20 Employee protection.

Employment discrimination by a licensee, a holder of a certificate of compliance issued under part 76 of this chapter or regulated entity subject to the requirements in this part as delineated in § 19.2(a), or a contractor or subcontractor of a licensee, a holder of a certificate of compliance issued under part 76 of this chapter, or regulated entity subject to the requirements in this part as delineated in § 19.2(a), against an employee for engaging in protected activities under this part or parts 30, 40, 50, 52, 53, 54, 60, 61, 63, 70, 72, 76, or 150 of this chapter is prohibited.

PART 20 – STANDARDS FOR PROTECTION AGAINST RADIATION

1. The authority citation for part 20 continues to read as follows:

Authority: Atomic Energy Act of 1954, secs. 11, 53, 63, 65, 81, 103, 104, 161, 170H, 182, 186, 223, 234, 274, 1701 (42 U.S.C. 2014, 2073, 2093, 2095, 2111, 2133, 2134, 2201, 2210h, 2232, 2236, 2273, 2282, 2021, 2297f); Energy Reorganization Act of 1974, secs. 201, 202 (42 U.S.C. 5841, 5842); Low-Level Radioactive Waste Policy Amendments Act of 1985, sec. 2 (42 U.S.C. 2021b); 44 U.S.C. 3504 note.

2. Revise the first sentence in § 20.1002 to read as follows:

§ 20.1002 Scope.

The regulations in this part apply to persons licensed by the Commission to receive, possess, use, transfer, or dispose of byproduct, source, or special nuclear material or to operate a production or utilization facility under parts 30 through 36, 39, 40, 50, 52, 53, 60, 61, 63, 70, or 72 of this chapter, and in accordance with § ~~10 CFR~~ 76.60 to persons required to obtain a certificate of compliance or an approved compliance plan under part 76 of this chapter. * * *

3. Revise § 20.1003 to read as follows:

§ 20.1003 Definitions.

* * * * *

License means a license issued under the regulations in parts 30 through 36, 39, 40, 50, 53, 60, 61, 63, 70, or 72 of this chapter.

* * * * *

Subpart B – Radiation Protection Programs

4. Revise § 20.1101, paragraph (d) to read as follows:

§ 20.1101 Radiation protection programs.

* * * * *

(d) To implement the ALARA requirements of § 20.1101 (b), and notwithstanding the requirements in § 20.1301 of this part, a constraint on air emissions of radioactive material to the environment, excluding Radon-222 and its daughters, shall be established by licensees other than those subject to §§ 50.34a, 53.260(b), or 53.4730(a)(3) such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent in excess of 10 mrem (0.1 mSv) per year from these emissions. If a licensee subject to this requirement exceeds this dose constraint, the licensee shall report the exceedance as provided in § 20.2203 and promptly take appropriate corrective action to ensure against recurrence.

Subpart E – Radiological Criteria for License Termination

5. Revise § 20.1401, paragraphs (a) and (c) to read as follows:

§ 20.1401 General provisions and scope.

(a) The criteria in this subpart apply to the decommissioning of facilities licensed under parts 30, 40, 50, 52, 53, 60, 61, 63, 70, and 72 of this chapter, and release of part of a facility or site for unrestricted use in accordance with § 50.83, § 53.1080, or § 53.4680 of this chapter, as well as other facilities subject to the Commission's jurisdiction under the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974, as amended. For high-level and low-level waste disposal

facilities (10 CFR parts 60, 61, and 63), the criteria apply only to ancillary surface facilities that support radioactive waste disposal activities. The criteria do not apply to uranium and thorium recovery facilities already subject to appendix A to 10 CFR part 40 or the uranium solution extraction facilities.

* * * * *

(c) After a site has been decommissioned and the license terminated in accordance with the criteria in this subpart, or after part of a facility or site has been released for unrestricted use in accordance with § 50.83, § 53.1080, or § 53.4680 of this chapter and in accordance with the criteria in this subpart, the Commission will require additional cleanup only, if based on new information, it determines that the criteria of this subpart were not met and residual radioactivity remaining at the site could result in significant threat to public health and safety.

* * * * *

6. Revise § 20.1403, paragraph (d) to read as follows:

§ 20.1403 Criteria for license termination under restricted conditions.

* * * * *

(d) The licensee has submitted a decommissioning plan or License Termination Plan (LTP) to the Commission indicating the licensee's intent to decommission in accordance with §§ 30.36(d), 40.42(d), 50.82 (a) and (b), 53.1000 through 53.1080, 53.4600 through 53.4680, 70.38(d), or 72.54 of this chapter, and specifying that the licensee intends to decommission by restricting use of the site. The licensee shall document in the LTP or decommissioning plan how the advice of individuals and institutions in the community who may be affected by the decommissioning has been sought and incorporated, as appropriate, following analysis of that advice.

* * * * *

7. Revise § 20.1404, paragraph (a)(4) to read as follows:

§ 20.1404 Alternate criteria for license termination.

(a) * * *

(4) Has submitted a decommissioning plan or License Termination Plan (LTP) to the Commission indicating the licensee's intent to decommission in accordance with §§ 30.36(d), 40.42(d), 50.82 (a) and (b), 53.1000 through 53.1080, 53.4600 through 53.4680, 70.38(d), or 72.54 of this chapter, and specifying that the licensee proposes to decommission by use of alternate criteria. The licensee shall document in the decommissioning plan or LTP how the advice of individuals and institutions in the community who may be affected by the decommissioning has been sought and addressed, as appropriate, following analysis of that advice. In seeking such advice, the licensee shall provide for:

* * * * *

8. Revise § 20.1406, paragraphs (a) and (b) to read as follows:

§ 20.1406 Minimization of contamination.

(a) Applicants for licenses, other than early site permits and manufacturing licenses under part 52 or part 53 of this chapter and renewals, whose applications are submitted after August 20, 1997, shall describe in the application how facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

(b) Applicants for standard design certifications, standard design approvals, and manufacturing licenses under part 52 or part 53 of this chapter, whose applications are submitted after August 20, 1997, shall describe in the application how facility design will minimize, to the extent practicable, contamination of the facility and the

environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

(c) Licensees shall, to the extent practical, conduct operations to minimize the introduction of residual radioactivity into the site, including the subsurface, in accordance with the existing radiation protection requirements in subpart B and radiological criteria for license termination in subpart E of this part.

Subpart F – Surveys and Monitoring

9. Revise § 20.1501, paragraph (b) to read as follows:

§ 20.1501 General.

* * * * *

(b) Notwithstanding § 20.2103(a) of this part, records from surveys describing the location and amount of subsurface residual radioactivity identified at the site must be kept with records important for decommissioning, and such records must be retained in accordance with §§ 30.35(g), 40.36(f), 50.75(g), 53.1000 through 53.1080, 53.4600 through 53.4680, 70.25(g), or 72.30(d), as applicable.

* * * * *

Subpart J – Precautionary Procedures

10. Revise § 20.1905, paragraph (g) to read as follows:

§ 20.1905 Exemptions to labeling requirements.

* * * * *

(g) Containers holding licensed material (other than sealed sources that are either specifically or generally licensed) at a facility licensed under ~~P~~parts 50, 52, or 53 of this chapter, not including non-power reactors, that are within an area posted under the requirements in § 20.1902 if the containers are:

* * *

Subpart K – Waste Disposal

11. Revise § 20.2004, paragraph (b)(1) to read as follows:

§ 20.2004 Treatment or disposal by incineration.

* * * * *

(b)(1) Waste oils (petroleum derived or synthetic oils used principally as lubricants, coolants, hydraulic or insulating fluids, or metalworking oils) that have been radioactively contaminated in the course of the operation or maintenance of a nuclear power reactor licensed under part 50 or part 53 of this chapter may be incinerated on the site where generated provided that the total radioactive effluents from the facility, including the effluents from such incineration, conform to the requirements of appendix I to part 50, or §§ 53.260(b) or 53.4730(a)(3) of this chapter and the effluent release limits contained in applicable license conditions other than effluent limits specifically related to incineration of waste oil. The licensee shall report any changes or additions to the information supplied under §§ 50.34, 50.34a, 53.1100 through 53.1480, or 53.4700 through 53.5080 of this chapter associated with this incineration pursuant to § 50.71, 53.1620, or 53.6320 of this chapter, as appropriate. The licensee shall also follow the procedures of §§ 50.59, 53.1565, or 53.6065 of this chapter with respect to such changes to the facility or procedures.

* * *

Subpart M – Reports

12. Revise § 20.2201, paragraphs (a)(2)(i), (b)(2)(i), and (c) to read as follows:

§ 20.2201 Reports of theft or loss of licensed material.

(a) * * *

(2) * * *

(i) Licensees having an installed Emergency Notification System shall make the reports to the NRC Operations Center in accordance with §§ 50.72, 53.1630, or 53.6330 of this chapter, and

* * *

(b) * * *

(2) * * *

(i) For holders of an operating license for a nuclear power plant, the events included in paragraph (b) of this section must be reported in accordance with the procedures described in §§ 50.73(b), (c), (d), (e), and (g), 53.1640(b), (c), (d), (e), and (g), or 53.6340(b), (c), (d), (e), and (g) of this chapter and must include the information required in paragraph (b)(1) of this section, and

* * *

(c) A duplicate report is not required under paragraph (b) of this section if the licensee is also required to submit a report pursuant to §§ 30.55(c), 37.57, 37.81, 40.64(c), 50.72, 50.73, 53.1630, 53.1640, 53.6330, 53.6340, 70.52, 73.27(b), 73.67(e)(3)(vii), 73.67(g)(3)(iii), 73.71, or 150.19(c) of this chapter.

* * * * *

13. Revise § 20.2202, paragraph (d)(1) to read as follows:

§ 20.2202 Notification of incidents.

* * * * *

(d) * * *

(1) Licensees having an installed Emergency Notification System shall make the reports required by paragraphs (a) and (b) of this section to the NRC Operations Center in accordance with §§ 50.72, 53.1630, or 53.6330; and

* * * * *

14. Revise § 20.2203, paragraph (c) to read as follows:

§ 20.2203 Reports of exposures, radiation levels, and concentrations of radioactive material exceeding the constraints or limits.

* * * * *

(c) For holders of an operating license or a combined license for a nuclear power plant, the occurrences included in paragraph (a) of this section must be reported in accordance with the procedures described in §§ 50.73(b), (c), (d), (e), and (g), 53.1640(b), (c), (d), (e), and (g), or 53.6340(b), (c), (d), (e), and (g) of this chapter, and must include the information required by paragraph (b) of this section. Occurrences reported in accordance with §§ 50.73, 53.1640, or 53.6340 of this chapter need not be reported by a duplicate report under paragraph (a) of this section.

* * * * *

15. Revise § 20.2206, paragraph (a)(1) to read as follows:

§ 20.2206 Reports of individual monitoring.

(a) * * *

(1) Operate a nuclear reactor designed to produce electrical or heat energy pursuant to §§ 50.21(b), ~~or §~~ 50.22, or 53.010 of this chapter or a testing facility as defined in § 50.2 of this chapter; or

* * * * *

PART 21 – REPORTING OF DEFECTS AND NONCOMPLIANCE

1. The authority citation for part 21 continues to read as follows:

Authority: Atomic Energy Act of 1954, secs. 53, 63, 81, 103, 104, 161, 223, 234, 1701 (42 U.S.C. 2073, 2093, 2111, 2133, 2134, 2201, 2273, 2282, 2297f); Energy Reorganization Act of 1974, secs. 201, 206 (42 U.S.C. 5841, 5846); Nuclear Waste Policy Act of 1982, secs. 135, 141 (42 U.S.C. 10155, 10161); 44 U.S.C. 3504 note.

2. Revise § 21.2, paragraphs (a)(2)-(4) and paragraphs (b) and (c) to read as follows:

§ 21.2 Scope.

(a) * * *

(2) Each individual, corporation, partnership, or other entity doing business within the United States, and each director and responsible officer of such an organization, that constructs a production or utilization facility licensed for manufacture, construction, or operation under parts 50, 52, or 53 of this chapter, an ISFSI for the storage of spent fuel licensed under part 72 of this chapter, an MRS for the storage of spent fuel or high-level radioactive waste under part 72 of this chapter, or a geologic repository for the disposal of high-level radioactive waste under part 60 or 63 of this chapter; or supplies basic components for a facility or activity licensed, other than for export, under parts 30, 40, 50, 52, 53, 60, 61, 63, 70, 71, or part 72 of this chapter;

(3) Each individual, corporation, partnership, or other entity doing business within the United States, and each director and responsible officer of such an organization, applying for a design certification rule under part 52 or part 53 of this chapter; or supplying basic components with respect to that design certification, and each individual, corporation, partnership, or other entity doing business within the United States, and each director and responsible officer of such an organization, whose application for design certification has been granted under part 52 or part 53 of this chapter, or who has supplied or is supplying basic components with respect to that design certification;

(4) Each individual, corporation, partnership, or other entity doing business within the United States, and each director and responsible officer of such an organization, applying for or holding a standard design approval under part 52 or part 53 of this

chapter; or supplying basic components with respect to a standard design approval under part 52 or part 53 of this chapter;

(b) For persons licensed to construct a facility under either a construction permit issued under §§ 50.23, 53.1333, or 53.4933 of this chapter or a combined license under part 52 or part 53 of this chapter (for the period of construction until the date that the Commission makes the finding under §§ 52.103(g), 53.1452(g), or 53.5052(g) of this chapter), or to manufacture a facility under part 52 or part 53 of this chapter, evaluation of potential defects and failures to comply and reporting of defects and failures to comply under §§ 50.55(e), 53.605, or 53.4105 of this chapter satisfies each person's evaluation, notification, and reporting obligation to report defects and failures to comply under this part and the responsibility of individual directors and responsible officers of these licensees to report defects under Section 206 of the Energy Reorganization Act of 1974.

(c) For persons licensed to operate a nuclear power plant under part 50, part 52, or part 53 of this chapter, evaluation of potential defects and appropriate reporting of defects under §§ 50.72, 50.73, 53.1630, 53.1640, 53.6330, 53.6340, or 73.71 of this chapter, satisfies each person's evaluation, notification, and reporting obligation to report defects under this part, and the responsibility of individual directors and responsible officers of these licensees to report defects under Section 206 of the Energy Reorganization Act of 1974.

* * * * *

3. Revise § 21.3 to read as follows:

§ 21.3 Definitions.

* * * * *

Basic component. (1)(i) When applied to nuclear power plants licensed under 10 CFR part 50, part 52, or part 53 of this chapter, basic component means a structure,

system, or component, or part thereof that affects its safety function necessary to assure:

* * *

(C) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in §§ 50.34(a)(1), 50.67(b)(2), 53.210, 53.4730(a)(1)(vi), or 100.11 of this chapter, as applicable.

(ii) Basic components are items designed and manufactured under a quality assurance program complying with appendix B to part 50, §§ 53.1800 through 53.1890, or §§ 53.6600 through 53.6690 of this chapter, or commercial grade items which have successfully completed the dedication process.

(2) When applied to standard design certifications under subpart B of part 52, or subparts H or R of part 53 of this chapter and standard design approvals under part 52 or part 53 of this chapter, basic component means the design or procurement information approved or to be approved within the scope of the design certification or approval for a structure, system, or component, or part thereof, that affects its safety function necessary to assure:

* * *

(iii) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in §§ 50.34(a)(1), 50.67(b)(2), 53.210, 53.4730(a)(1)(vi), or 100.11 of this chapter, as applicable.

* * *

(4) In all cases, basic component includes safety-related design, analysis, inspection, testing, fabrication, replacement of parts, or consulting services that are

associated with the component hardware, design certification, design approval, or information in support of an early site permit application under part 52 or part 53 of this chapter, whether these services are performed by the component supplier or others.

Commercial grade item. (1) When applied to nuclear power plants licensed pursuant to 10 CFR ~~P~~part 50 or 10 CFR part 53, commercial grade item means a structure, system, or component, or part thereof that affects its safety function, that was not designed and manufactured as a basic component. Commercial grade items do not include items where the design and manufacturing process require in-process inspections and verifications to ensure that defects or failures to comply are identified and corrected (i.e., one or more critical characteristics of the item cannot be verified).

* * * * *

Critical characteristics. When applied to nuclear power plants licensed pursuant to 10 CFR ~~P~~part 50 or 10 CFR part 53, critical characteristics are those important design, material, and performance characteristics of a commercial grade item that, once verified, will provide reasonable assurance that the item will perform its intended safety function.

Dedicating entity. When applied to nuclear power plants licensed pursuant to 10 CFR ~~P~~part 50 or 10 CFR part 53, dedicating entity means the organization that performs the dedication process. Dedication may be performed by the manufacturer of the item, a third-party dedicating entity, or the licensee itself. The dedicating entity, pursuant to § 21.21(c) of this part, is responsible for identifying and evaluating deviations, reporting defects and failures to comply for the dedicated item, and maintaining auditable records of the dedication process.

Dedication. (1) When applied to nuclear power plants licensed pursuant to 10 CFR ~~P~~part 30, 40, 50, 53, 60, dedication is an acceptance process undertaken to

provide reasonable assurance that a commercial grade item to be used as a basic component will perform its intended safety function and, in this respect, is deemed equivalent to an item designed and manufactured under a 10 CFR ~~P~~part 50, appendix B, or subparts K or U of part 53, quality assurance program. This assurance is achieved by identifying the critical characteristics of the item and verifying their acceptability by inspections, tests, or analyses performed by the purchaser or third-party dedicating entity after delivery, supplemented as necessary by one or more of the following: commercial grade surveys; product inspections or witness at holdpoints at the manufacturer's facility, and analysis of historical records for acceptable performance. In all cases, the dedication process must be conducted in accordance with the applicable provisions of 10 CFR ~~P~~part 50, appendix B, or subparts K or U of part 53. The process is considered complete when the item is designated for use as a basic component.

* * *

Defect means: * * *

(3) A deviation in a portion of a facility subject to the early site permit, standard design certification, standard design approval, construction permit, combined license or manufacturing licensing requirements of part 50, part 52, or part 53 of this chapter, provided the deviation could, on the basis of an evaluation, create a substantial safety hazard and the portion of the facility containing the deviation has been offered to the purchaser for acceptance;

(4) A condition or circumstance involving a basic component that could contribute to the exceeding of a safety limit, as defined in the technical specifications of a license for operation issued under part 50, part 52, or part 53 of this chapter; or

* * * * *

Substantial safety hazard means a loss of safety function to the extent that there is a major reduction in the degree of protection provided to public health and safety for any facility or activity licensed or otherwise approved or regulated by the NRC, other than for export, under parts 30, 40, 50, 52, 53, 60, 61, 63, 70, 71, or 72 of this chapter.

* * * * *

4. Revise § 21.21, paragraphs (a)(3), (a)(3)(i), and (d)(1)(i)-(ii) to read as follows:

§ 21.21 Notification of failure to comply or existence of a defect and its evaluation.

(a) * * *

(3) Ensure that a director or responsible officer subject to the regulations of this part is informed as soon as practicable, and, in all cases, within the 5 working days after completion of the evaluation described in paragraphs (a)(1) or (a)(2) of this section if the manufacture, construction, or operation of a facility or activity, a basic component supplied for such facility or activity, or the design certification or design approval under part 52 or part 53 of this chapter –

(i) Fails to comply with the Atomic Energy Act of 1954, as amended, or any applicable rule, regulation, order, or license of the Commission or standard design approval under part 52 or part 53 of this chapter, relating to a substantial safety hazard, or

* * * * *

(d)(1) * * *

(i) The manufacture, construction or operation of a facility or an activity within the United States that is subject to the licensing requirements under parts 30, 40, 50, 52, 53,

60, 61, 63, 70, 71, or 72 of this chapter and that is within his or her organization's responsibility; or

(ii) A basic component that is within his or her organization's responsibility and is supplied for a facility or an activity within the United States that is subject to the licensing, design certification, or approval requirements under parts 30, 40, 50, 52, 53, 60, 61, 63, 70, 71, or 72 of this chapter.

* * * * *

5. Revise § 21.51, paragraphs (a)(4)-(5) to read as follows:

§ 21.51 Maintenance and inspection of records.

(a) * * *

(4) Applicants for standard design certification under subpart B of part 52, §§ 53.1230 through 53.1263, or §§ 53.4830 through 53.4863 of this chapter and others providing a design which is the subject of a design certification, during and following Commission adoption of a final design certification rule for that design, shall retain any notifications sent to purchasers and affected licensees for a minimum of 5 years after the date of the notification, and retain a record of the purchasers for 15 years after delivery of design which is the subject of the design certification rule or service associated with the design.

(5) Applicants for or holders of a standard design approval under subpart E of part 52, §§ 53.1200 through 53.1221, or §§ 53.4800 through 53.4821 of this chapter and others providing a design which is the subject of a design approval retain any notifications sent to purchasers and affected licensees for a minimum of 5 years after the date of the notification, and retain a record of the purchasers for 15 years after delivery of the design which is the subject of the design approval or service associated with the design.

* * * * *

6. Revise § 21.61, paragraph (b) to read as follows:

§ 21.61 Failure to notify.

* * * * *

(b) Any NRC licensee or applicant for a license (including an applicant for, or holder of, a permit), applicant for a design certification under part 52 or part 53 of this chapter during the pendency of its application, applicant for a design certification after Commission adoption of a final design certification rule for that design, or applicant for or holder of a standard design approval under part 52 or part 53 of this chapter subject to the regulations in this part who fails to provide the notice required by § 21.21, or otherwise fails to comply with the applicable requirements of this part shall be subject to a civil penalty as provided by Section 234 of the Atomic Energy Act of 1954, as amended.

* * * * *

PART 25 – DEFINITIONS

1. The authority citation for part 25 continues to read as follows:

Authority: Atomic Energy Act of 1954, secs. 145, 161, 223, 234 (42 U.S.C. 2165, 2201, 2273, 2282); Energy Reorganization Act of 1974, sec. 201 (42 U.S.C. 5841); 44 U.S.C. 3504 note; E.O. 10865, 25 FR 1583, as amended, 3 CFR, 1959-1963 Comp., p. 398; E.O. 12829, 58 FR 3479, 3 CFR, 1993 Comp., p. 570; E.O. 13526, 75 FR 707, 3 CFR, 2009 Comp., p. 298; E.O. 12968, 60 FR 40245, 3 CFR, 1995 Comp., p. 391.

Section 25.17(f) and Appendix A also issued under 31 U.S.C. 9701; 42 U.S.C. 2214.

2. Revise § 25.5 to read as follows:

§ 25.5 Definitions.

* * * * *

License means a license issued pursuant to 10 CFR parts 50, 52, 53, 60, 63, 70, or 72.

* * * * *

3. Revise § 25.17, paragraph (a) to read as follows:

§ 25.17 Approval for processing applicants for access authorization.

(a) Access authorizations must be requested for licensee employees or other persons (e.g., 10 CFR part 2, subpart I) who need access to classified information in connection with activities under 10 CFR parts 50, 52, 53, 54, 60, 63, 70, 72, or 76.

* * * * *

4. Revise § 25.35, paragraph (a) to read as follows:

§ 25.35 Classified visits.

(a) The number of classified visits must be held to a minimum. The licensee, certificate holder, applicant for a standard design certification under part 52 or part 53 of this chapter (including an applicant after the Commission has adopted a final standard design certification rule under part 52 or part 53 of this chapter), or other facility, or an applicant for or holder of a standard design approval under part 52 or part 53 of this chapter shall determine that the visit is necessary and that the purpose of the visit cannot be achieved without access to, or disclosure of, classified information. All classified visits require advance notification to, and approval of, the organization to be visited. In urgent cases, visit information may be furnished by telephone and confirmed in writing.

* * * * *

PART 26—FITNESS FOR DUTY PROGRAMS.

[Part 26 placeholder.]

PART 30 – RULES OF GENERAL APPLICABILITY TO DOMESTIC LICENSING OF BYPRODUCT MATERIAL

1. The authority citation for part 30 continues to read as follows:

Authority: Atomic Energy Act of 1954, secs. 11, 81, 161, 181, 182, 183, 184, 186, 187, 223, 234, 274 (42 U.S.C. 2014, 2111, 2201, 2231, 2232, 2233, 2234, 2236, 2237, 2273, 2282, 2021); Energy Reorganization Act of 1974, secs. 201, 202, 206, 211 (42 U.S.C. 5841, 5842, 5846, 5851); 44 U.S.C. 3504 note.

2. Revise § 30.4, paragraph *Utilization facility* to read as follows:

§ 30.4 Definitions.

* * * * *

Utilization facility means a utilization facility as defined in the regulations contained in part 50 or part 53 of this chapter;

* * * * *

3. Revise § 30.50, paragraph (c)(3) to read as follows:

§ 30.50 Reporting requirements.

* * * * *

(c) * * *

(3) The provisions of § 30.50 do not apply to licensees subject to the notification requirements in §§ 50.72, 53.1630, or 53.6330. They do apply to those part 50 and part 53 licensees possessing material licensed under part 30, who are not subject to the notification requirements in §§ 50.72, 53.1630, or 53.6330.

PART 40 – DOMESTIC LICENSING OF SOURCE MATERIAL

1. The authority citation for part 40 continues to read as follows:

Authority: Atomic Energy Act of 1954, secs. 62, 63, 64, 65, 69, 81, 83, 84, 122, 161, 181, 182, 183, 184, 186, 187, 193, 223, 234, 274, 275 (42 U.S.C. 2092, 2093, 2094, 2095, 2099, 2111, 2113, 2114, 2152, 2201, 2231, 2232, 2233, 2234, 2236, 2237, 2243, 2273, 2282, 2021, 2022); Energy Reorganization Act of 1974, secs. 201, 202, 206, 211 (42 U.S.C. 5841, 5842, 5846, 5851); Uranium Mill Tailings Radiation Control Act of 1978, sec. 104 (42 U.S.C. 7914); 44 U.S.C. 3504 note.

2. Revise § 40.60, paragraph (c)(3) to read as follows:

§ 40.60 Reporting requirements.

* * * * *

(c) * * *

(3) The provisions of § 40.60 do not apply to licensees subject to the notification requirements in §§ 50.72, 53.1630, or 53.6330. They do apply to those part 50 and part 53 licensees possessing material licensed under part 40 who are not subject to the notification requirements in §§ 50.72, 53.1630, or 53.6330.

PART 50 – DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for part 50 continues to read as follows:

Authority: Atomic Energy Act of 1954, secs. 62, 63, 64, 65, 69, 81, 83, 84, 122, 161, 181, 182, 183, 184, 186, 187, 193, 223, 234, 274, 275 (42 U.S.C. 2092, 2093, 2094, 2095, 2099, 2111, 2113, 2114, 2152, 2201, 2231, 2232, 2233, 2234, 2236, 2237, 2243, 2273, 2282, 2021, 2022); Energy Reorganization Act of 1974, secs. 201, 202, 206, 211 (42 U.S.C. 5841, 5842, 5846, 5851); Uranium Mill Tailings Radiation Control Act of 1978, sec. 104 (42 U.S.C. 7914); 44 U.S.C. 3504 note.

2. In § 50.44, revise the introductory paragraphs (c) and (d) to read as follows:

§ 50.44 Combustible gas control for nuclear power reactors.

* * * * *

(c) *Requirements for future water-cooled reactor applicants and licensees.*[2] The requirements in this paragraph apply to all water-cooled reactor construction permits or operating licenses under this part or Framework B of Part 53 of this chapter, and to all

water-cooled reactor design approvals, design certifications, combined licenses or manufacturing licenses under part 52 or Framework B of Part 53 of this chapter, any of which are issued after October 16, 2003.

* * * * *

(d) *Requirements for future non water-cooled reactor applicants and licensees and certain water-cooled reactor applicants and licensees.* The requirements in this paragraph apply to all construction permits and operating licenses under this part and Framework B of Part 53 of this chapter, and to all design approvals, design certifications, combined licenses, or manufacturing licenses under part 52 and Framework B of Part 53 of this chapter, for non water-cooled reactors and water-cooled reactors that do not fall within the description in paragraph (c), footnote 1 of this section, any of which are issued after October 16, 2003. Applications subject to this paragraph must include:

* * * * *

3. In § 50.46, revise paragraphs (a)(1)(i) and (a)(3)(i) through (iii) to read as follows:

§ 50.46 Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors

(a)(1)(i) Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-

coolant accidents are calculated. Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II Required Documentation, sets forth the documentation requirements for each evaluation model. This section does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) or § 53.4670(a) have been submitted.

* * * * *

(3)(i) Each applicant for or holder of an operating license or construction permit issued under this part or Framework B under part 53 of this chapter subject to this section, applicant for a standard design certification under part 52 or Framework B under part 53 of this chapter (including an applicant after the Commission has adopted a final design certification regulation) subject to this section, or an applicant for or holder of a standard design approval, a combined license, or a manufacturing license issued under part 52 or Framework B under part 53 of this chapter subject to this section, shall estimate the effect of any change to or error in an acceptable evaluation model or in the application of such a model to determine if the change or error is significant. For this purpose, a significant change or error is one which results in a calculated peak fuel cladding temperature different by more than 50 °F from the temperature calculated for the limiting transient using the last acceptable model, or is a cumulation of changes and

errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50 °F.

(ii) For each change to or error discovered in an acceptable evaluation model or in the application of such a model that affects the temperature calculation, the applicant or holder of a construction permit, operating license, combined license, or manufacturing license shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually as specified in § 50.4, § 52.3, or § 53.040 of this chapter, as applicable. If the change or error is significant, the applicant or licensee shall provide this report within 30 days and include with the report a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with § 50.46 requirements. This schedule may be developed using an integrated scheduling system previously approved for the facility by the NRC. For those facilities not using an NRC approved integrated scheduling system, a schedule will be established by the NRC staff within 60 days of receipt of the proposed schedule. Any change or error correction that results in a calculated ECCS performance that does not conform to the criteria set forth in paragraph (b) of this section is a reportable event as described in §§ 50.55(e), 50.72, and 50.73. For applicants or holders of a construction permit, operating license, combined license, or manufacturing license under Framework B of part 53, the reporting requirements of §§ 53.4105, 53.6330, and 53.6340 of this chapter apply in lieu of the reporting requirements of §§ 50.55(e), 50.72, and 50.73. The affected applicant or licensee shall propose immediate steps to demonstrate compliance or bring plant design or operation into compliance with § 50.46 requirements.

(iii) For each change to or error discovered in an acceptable evaluation model or in the application of such a model that affects the temperature calculation, the applicant or holder of a standard design approval or the applicant for a standard design

certification (including an applicant after the Commission has adopted a final design certification rule) shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission and to any applicant or licensee referencing the design approval or design certification at least annually as specified in § 52.3 or § 53.040 of this chapter. If the change or error is significant, the applicant or holder of the design approval or the applicant for the design certification shall provide this report within 30 days and include with the report a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with § 50.46 requirements. The affected applicant or holder shall propose immediate steps to demonstrate compliance or bring plant design into compliance with § 50.46 requirements.

* * * * *

5. Amend § 50.47 by:

a. Revising paragraphs (a)(1)(ii); and

b. Adding paragraphs (a)(v) and (a)(vi);

The revisions to read as follows:

§ 50.47 Emergency plans.

(a)(1) * * *

(ii) No initial combined license under part 52 or part 53 of this chapter will be issued unless a finding is made by the NRC that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. No finding under this section is necessary for issuance of a renewed combined license.

* * *

(v) If an application for an early site permit under part 53 of this chapter includes complete and integrated emergency plans under §§ 53.855 or 53.4320, no early site permit

will be issued unless a finding is made by the NRC that the emergency plans provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

(vi) If an application for an early site permit proposes major features of the emergency plans under part 53, no early site permit will be issued unless a finding is made by the NRC that the major features are acceptable in accordance with the applicable standards of § 50.47 or 53.855, and 10 CFR part 50, appendix E, within the scope of emergency preparedness matters addressed in the major features.

5. Amend § 50.55a by:

a. Revising paragraphs (b)(1), (b)(2)(xxi)(B)(*3*), (b)(3)(iii), and (b)(4);

b. Revising introductory text of paragraph (c);

c. Revising introductory text of paragraph (d) and (d)(1);

d. Revising introductory text of paragraph (e) and (e)(1);

e. Revising introductory text of paragraph (f) and (f)(3) and (f)(3)(iii)(B), (f)(3)(iv)(B), and (f)(4)(i);

f. Revising introductory text of paragraph (g) and paragraphs (g)(2)(ii), (g)(3)(ii), and (g)(4)(i) and (v);

g. Revising paragraph introductory text of paragraph (d) and (d)(1);

The revisions to read as follows:

§ 50.55a Codes and standards.

* * * * *

(b) * * *

(1) *Conditions on ASME BPV Code Section III*. Each manufacturing license, standard design approval, and design certification under 10 CFR part 52 and boiling or pressurized water-cooled commercial nuclear plants under 10 CFR part 53 Framework B

is subject to the following conditions. As used in this section, references to Section III refer to Section III of the ASME BPV Code and include the 1963 Edition through 1973 Winter Addenda and the 1974 Edition (Division 1) through the 2017 Edition (Division 1), subject to the following conditions:

* * * * *

(2) * * *

(xxi) * * *

(B) * * *

(*3*) The provisions of IWB–2500(g) and Table IWB–2500–1 Notes 6 and 7 for examination of Examination Category B–D Item Numbers B3.90 and B3.100 shall not be used to eliminate the preservice or inservice volumetric examination of plants with a Combined Operating License pursuant to 10 CFR part 52 or boiling or pressurized water-cooled commercial nuclear plants under Framework B of 10 CFR part 53, or a plant that receives its operating license after October 22, 2015.

* * * * *

(3) * * *

(iii) *OM condition: New reactors*. In addition to complying with the provisions in the ASME OM Code with the conditions specified in paragraph (b)(3) of this section, holders of operating licenses for nuclear power reactors that received construction permits under this part on or after the date 12 months after August 17, 2017, and holders of combined licenses issued under 10 CFR part 52, whose initial fuel loading occurs on or after the date 12 months after August 17, 2017, and holders of operating licenses or combined licenses for boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53, shall also comply with the following conditions, as applicable:

* * * * *

(4) *Conditions on Design, Fabrication, and Materials Code Cases*. Each manufacturing license, standard design approval, and design certification application under part 52 or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 of this chapter is subject to the following conditions. Licensees may apply the ASME BPV Code Cases listed in NRC Regulatory Guide 1.84, as incorporated by reference in paragraph (a)(3)(i) of this section, without prior NRC approval, subject to the following conditions:

* * * * *

(c) *Reactor coolant pressure boundary*. Systems and components of boiling and pressurized water-cooled nuclear power reactors must satisfy the requirements of the ASME BPV Code as specified in this paragraph. Each manufacturing license, standard design approval, and design certification application under part 52 or Framework B of part 53 of this chapter and each combined license under part 52 or boiling or pressurized water-cooled commercial nuclear plants Framework B of part 53 for a utilization facility is subject to the following conditions:

* * * * *

(d) *Quality Group B components*. Systems and components of boiling and pressurized water-cooled nuclear power reactors must satisfy the requirements of the ASME BPV Code as specified in this paragraph. Each manufacturing license, standard design approval, and design certification application under part 52 or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 of this chapter, and each combined license under part 52 or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 for a utilization facility is subject to the following conditions:

(1) *Standards requirement for Quality Group B components*. For a nuclear power plant whose application for a construction permit under this part, or a combined license or manufacturing license under part 52 of this chapter, docketed after May 14, 1984, or for an application for a standard design approval or a standard design certification docketed after May 14, 1984, components classified Quality Group B [7] must satisfy the requirements for Class 2 Components in Section III of the ASME BPV Code. For a commercial nuclear plant whose application for a construction permit, combined license, manufacturing license, standard design approval, or standard design certification, uses a design for a boiling or pressurized water cooled commercial nuclear plant under 10 CFR part 53 Framework B, components classified Quality Group B [7] must satisfy the requirements for Class 2 Components in Section III of the ASME BPV Code.

* * * * *

(e) *Quality Group C components*. Systems and components of boiling and pressurized water-cooled nuclear power reactors must satisfy the requirements of the ASME BPV Code as specified in this paragraph. Each manufacturing license, standard design approval, and design certification application under part 52 or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 of this chapter and each combined license under part 52 or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 for a utilization facility is subject to the following conditions.

(1) *Standards requirement for Quality Group C components.* For a nuclear power plant whose application for a construction permit under this part or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53, or a combined license or manufacturing license under part 52 or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 of this chapter, docketed after

May 14, 1984, or for an application for a standard design approval or a standard design certification docketed after May 14, 1984, components classified Quality Group C [7] must satisfy the requirements for Class 3 components in Section III of the ASME BPV Code.

* * * * *

(f) *Preservice and inservice testing requirements*. Systems and components of boiling and pressurized water-cooled nuclear power reactors must satisfy the requirements for preservice and inservice testing (referred to in this paragraph (f) collectively as inservice testing) of the ASME BPV Code and ASME OM Code as specified in this paragraph (f). Each operating license for a boiling or pressurized water-cooled nuclear facility is subject to the following conditions. Each combined license for a boiling or pressurized water-cooled nuclear facility is subject to the following conditions, but the conditions in paragraphs (f)(4) through (6) of this section must be met only after the Commission makes the finding under § 52.103(g) or § 53.5052(g) of this chapter. Requirements for inservice inspection of Class 1, Class 2, Class 3, Class MC, and Class CC components (including their supports) are located in paragraph (g) of this section.

* * * * *

(3) *Design and accessibility requirements for performing inservice testing in plants with CPs issued after 1974*. For a boiling or pressurized water-cooled nuclear power facility whose construction permit under this part or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 of this chapter, or design approval, design certification, combined license, or manufacturing license under part 52 or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 of this chapter was issued on or after July 1, 1974:

* * * * *

(iii) * * *

(B) *Class 1 pumps and valves: Second provision*. In facilities whose construction permit under this part or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 of this chapter, or design certification, design approval, combined license, or manufacturing license under part 52 or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 ~~of this chapter~~, issued on or after November 22, 1999, pumps and valves that are classified as ASME BPV Code Class 1 must be designed and provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in editions and addenda of the ASME OM Code (or the optional ASME OM Code Cases listed in NRC Regulatory Guide 1.192, as incorporated by reference in paragraph (a)(3)(iii) of this section), incorporated by reference in paragraph (a)(1)(iv) of this section at the time the construction permit, combined license, manufacturing license, design certification, or design approval is issued.

(iv) * * *

(B) *Class 2 and 3 pumps and valves: Second provision*. In facilities whose construction permit under this part or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 of this chapter, or design certification, design approval, combined license, or manufacturing license under part 52 or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 ~~of this chapter~~, issued on or after November 22, 1999, pumps and valves that are classified as ASME BPV Code Class 2 and 3 must be designed and provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in editions and addenda of the ASME OM Code (or the optional ASME OM Code Cases listed in NRC Regulatory Guide 1.192, as incorporated by reference in paragraph (a)(3)(iii) of this section), incorporated by reference in

paragraph (a)(1)(iv) of this section at the time the construction permit, combined license, or design certification is issued.

* * * * *

(4) * * *

(i) *Applicable IST Code: Initial 120-month interval*. Inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during the initial 120-month interval must comply with the requirements in the latest edition and addenda of the ASME OM Code incorporated by reference in paragraph (a)(1)(iv) of this section on the date 18 months before the date of issuance of the operating license under this part or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 of this chapter, or 18 months before the date scheduled for initial loading of fuel under a combined license under part 52 or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 of this chapter (or the optional ASME OM Code Cases listed in NRC Regulatory Guide 1.192, as incorporated by reference in paragraph (a)(3)(iii) of this section, subject to the conditions listed in paragraph (b) of this section).

* * * * *

(g) *Preservice and inservice inspection requirements*. Systems and components of boiling and pressurized water-cooled nuclear power reactors must satisfy the requirements of the ASME BPV Code as specified in this paragraph. Each operating license for a boiling or pressurized water-cooled nuclear facility is subject to the following conditions. Each combined license for a boiling or pressurized water-cooled nuclear facility is subject to the following conditions, but the conditions in paragraphs (g)(4) through (6) of this section must be met only after the Commission makes the finding under § 52.103(g) or § 53.5052(g) of this chapter. Requirements for inservice testing of

Class 1, Class 2, and Class 3 pumps and valves are located in paragraph (f) of this section.

* * * * *

(2) * * *

(ii) *Accessibility requirements for plants with CPs issued after 1974*. For a boiling or pressurized water-cooled nuclear power facility, whose construction permit under this part or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 of this chapter, or design certification, design approval, combined license, or manufacturing license under part 52 or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 ~~of this chapter~~, was issued on or after July 1, 1974, components that are classified as ASME BPV Code Class 1, Class 2, and Class 3 and supports for components that are classified as ASME BPV Code Class 1, Class 2, and Class 3 must be designed and provided with the access necessary to perform the required preservice and inservice examinations set forth in editions and addenda of Section III or Section XI of the ASME BPV Code incorporated by reference in paragraph (a)(1) of this section (or the optional ASME BPV Code Cases listed in NRC Regulatory Guide 1.147, as incorporated by reference in paragraph (a)(3)(ii) of this section) applied to the construction of the particular component.

* * * * *

(3) * * *

(ii) *Preservice examination requirements for plants with CPs issued after 1974*. For a boiling or pressurized water-cooled nuclear power facility, whose construction permit under this part or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 of this chapter, or design certification, design approval, combined license, or manufacturing license under part 52 or boiling or pressurized

water-cooled commercial nuclear plants under Framework B of part 53 of this chapter, was issued on or after July 1, 1974, components that are classified as ASME BPV Code Class 1, Class 2, and Class 3 and supports for components that are classified as ASME BPV Code Class 1, Class 2, and Class 3 must satisfy the preservice examination requirements set forth in the editions and addenda of Section III or Section XI of the ASME BPV Code incorporated by reference in paragraph (a)(1) of this section (or the optional ASME BPV Code Cases listed in NRC Regulatory Guide 1.147, as incorporated by reference in paragraph (a)(3)(ii) of this section) applied to the construction of the particular component.

* * * * *

(4) * * *

(i) *Applicable ISI Code: Initial 120-month interval*. Inservice examination of components and system pressure tests conducted during the initial 120-month inspection interval must comply with the requirements in the latest edition and addenda of the ASME Code incorporated by reference in paragraph (a) of this section on the date 18 months before the date of issuance of the operating license under this part or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 of this chapter, or 18 months before the date scheduled for initial loading of fuel under a combined license under part 52 or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 ~~of this chapter~~ (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147, when using ASME BPV Code, Section XI, or NRC Regulatory Guide 1.192, when using the ASME OM Code, as incorporated by reference in paragraphs (a)(3)(ii) and (iii) of this section, respectively), subject to the conditions listed in paragraph (b) of this section. Licensees may, at any time in their 120-month ISI interval, elect to use the Appendix VIII in the latest edition and addenda of the

ASME BPV Code incorporated by reference in paragraph (a) of this section, subject to any applicable conditions listed in paragraph (b) of this section. Licensees using this option must also use the same edition and addenda of Appendix I, Subarticle I–3200, as Appendix VIII, including any applicable conditions listed in paragraph (b) of this section.

* * * * *

(v) *Applicable ISI Code: Metal and concrete containments*. For a boiling or pressurized water-cooled nuclear power facility whose construction permit under this part or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 of this chapter, or combined license under part 52 or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 was issued after January 1, 1956, the following are required:

* * * * *

(h) * * *

(3) *Safety systems*. Applications filed on or after May 13, 1999, for construction permits and operating licenses under this part or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 of this chapter, and for design approvals, design certifications, and combined licenses under part 52 or boiling or pressurized water-cooled commercial nuclear plants under Framework B of part 53 of this chapter, must satisfy the requirements for safety systems in IEEE Std. 603–1991 and the correction sheet dated January 30, 1995.

* * * * *

6. In § 50.60, revise paragraphs (a) and (b) to read as follows:

§ 50.60 Acceptance criteria for fracture prevention measures for light-water nuclear power reactors for normal operation.

(a) Except as provided in paragraph (b) of this section, all light-water nuclear power reactors, other than reactor facilities for which the certifications required under § 50.82(a)(1) or § 53.4670(a) of this chapter have been submitted, must satisfy the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in appendices G and H to this part.

(b) Proposed alternatives to the described requirements in Appendices G and H of this part or portions thereof may be used when an exemption is granted by the Commission under § 50.12 or § 53.080 of this chapter.

7. In § 50.61, revise paragraph (b)(1) to read as follows:

§ 50.61 Fracture toughness requirements for protection against pressurized thermal shock events.

* * * * *

(b) *Requirements*. (1) For each pressurized-water nuclear power reactor for which an operating license has been issued under this part or Framework B of part 53 of this chapter, or a combined license issued under Part 52 or Framework B of part 53 of this chapter, other than a nuclear power reactor facility for which the certification required under § 50.82(a)(1) or § 53.4670(a) has been submitted, the licensee shall have projected values of RT_{PTS} or RT_{MAX-X}, accepted by the NRC, for each reactor vessel beltline material. For pressurized-water nuclear power reactors for which a construction permit was issued under this part before February 3, 2010 and whose reactor vessel was designed and fabricated to the 1998 Edition or earlier of the ASME Code, the projected values must be in accordance with this section or § 50.61a. For pressurized-water nuclear power reactors for which a construction permit is issued under this part or Framework B of part 53 of this chapter after February 3, 2010 and whose reactor vessel is designed and fabricated to an ASME Code after the 1998 Edition, or for

which a combined license is issued under Part 52 or Framework B of part 53 of this chapter, the projected values must be in accordance with this section. When determining compliance with this section, the assessment of RT_{PTS} must use the calculation procedures described in paragraph (c)(1) of this section and perform the evaluations described in paragraphs (c)(2) and (c)(3) of this section. The assessment must specify the bases for the projected value of RT_{PTS} for each vessel beltline material, including the assumptions regarding core loading patterns, and must specify the copper and nickel contents and the fluence value used in the calculation for each beltline material. This assessment must be updated whenever there is a significant [2] change in projected values of RT_{PTS}, or upon request for a change in the expiration date for operation of the facility.

* * * * *

8. In § 50.62, revise paragraphs (a), (b), (c)(4) and (6), and (d) to read as follows:

§ 50.62 Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.

(a) *Applicability*. The requirements of this section apply to all commercial light-water-cooled nuclear power plants licensed under this part 50 or part 52 of this chapter, or Framework B of part 53, other than nuclear power reactor facilities for which the certifications required under §§ 50.82(a)(1), 52.110(a), or § 53.4670(a) have been submitted.

(b) *Definition*. For purposes of this section, *Anticipated Transients Without Scram* (ATWS) means an anticipated operational occurrence as defined in appendix A of this part or as defined in § 53.028 for commercial nuclear plants licensed under

Framework B of part 53, followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion 20 of appendix A of this part.

(c) * * *

(4) Each boiling-water reactor must have a standby liquid control system (SLCS) with the capability of injecting into the reactor pressure vessel a borated water solution at such a flow rate, level of boron concentration and boron-10 isotope enrichment, and accounting for reactor pressure vessel volume, that the resulting reactivity control is at least equivalent to that resulting from injection of 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor pressure vessel for a given core design. The SLCS and its injection location must be designed to perform its function in a reliable manner. The SLCS initiation must be automatic and must be designed to perform its function in a reliable manner for plants granted a construction permit under this part or Framework B of part 53 of this chapter after July 26, 1984, and for plants granted a construction permit prior to July 26, 1984, that have already been designed and built to include this feature.

* * * * *

(6) Information sufficient to demonstrate to the Commission the adequacy of items in paragraphs (c)(1) through (c)(5) of this section shall be submitted to the Commission as specified in § 50.4 or § 53.040.

(d) *Implementation*. For each light-water-cooled nuclear power plant operating license issued before September 27, 2007, by 180 days after the issuance of the QA guidance for non-safety-related components, each licensee shall develop and submit to the Commission, as specified in § 50.4 or § 53.040, a proposed schedule for meeting the requirements of paragraphs (c)(1) through (c)(5) of this section. Each shall include an

explanation of the schedule along with a justification if the schedule calls for final implementation later than the second refueling outage after July 26, 1984, or the date of issuance of a license authorizing operation above 5 percent of full power. A final schedule shall then be mutually agreed upon by the Commission and licensee. For each light-water-cooled nuclear power plant operating license application under this part, or operating license or combined license application for a light-water cooled commercial nuclear plant under Framework B of part 53 of this chapter, submitted after September 27, 2007, the applicant shall submit information in its Final Safety Analysis Report demonstrating how it will comply with paragraphs (c)(1) through (c)(5) of this section.

9. In § 50.63, revise paragraphs (a)(1) and (c)(2) to read as follows:

§ 50.63 Loss of all alternating current power.

(a) *Requirements*. (1) Each light-water-cooled nuclear power plant licensed to operate under this part or § 53.4960 of Framework B of 10 CFR part 53, each light-water-cooled nuclear power plant licensed under subpart C of 10 CFR part 52 after the Commission makes the finding under § 52.103(g) of this chapter, each light-water-cooled nuclear power plant licensed under § 53.5010 of Framework B of 10 CFR part 53 after the Commission makes the finding under § 53.5052(g) of this chapter, and each design for a light-water-cooled nuclear power plant approved under a standard design approval, standard design certification, and manufacturing license under part 52 or under Framework B of part 53 of this chapter must be able to withstand for a specified duration and recover from a station blackout as defined in § 50.2. The specified station blackout duration shall be based on the following factors:

* * * * *

(c) * * *

(2) *Alternate ac source*: The alternate ac power source(s), as defined in § 50.2, will constitute acceptable capability to withstand station blackout provided an analysis is performed which demonstrates that the plant has this capability from onset of the station blackout until the alternate ac source(s) and required shutdown equipment are started and lined up to operate. The time required for startup and alignment of the alternate ac power source(s) and this equipment shall be demonstrated by test. Alternate ac source(s) serving a multiple unit site where onsite emergency ac sources are not shared between units must have, as a minimum, the capacity and capability for coping with a station blackout in any of the units. At sites, subject to this section, where onsite emergency ac sources are shared between units, the alternate ac source(s) must have the capacity and capability as required to ensure that all units can be brought to and maintained in safe shutdown (non-DBA) as defined in § 50.2. If the alternate ac source(s) meets the above requirements and can be demonstrated by test to be available to power the shutdown buses within 10 minutes of the onset of station blackout, then no coping analysis is required.

* * * * *

10. In Appendix A to part 50, revise the first paragraph in the Introduction section and the last paragraph of Criterion 19 to read as follows:

Appendix A to Part 50—General Design Criteria for Nuclear Power Plants

Introduction

Under the provisions of § 50.34 and § 53.4909, an application for a construction permit must include the principal design criteria for a proposed facility. Under the provisions of 10 CFR 52.47 and 10 CFR 53.4839, 10 CFR 52.79 and 10 CFR 53.5016, 10 CFR 52.137 and 10 CFR 53.4809, and 10 CFR 52.157 and 10 CFR 53.4879, an application

for a design certification, combined license, design approval, or manufacturing license, respectively, must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

* * * * *

Criteria

* * * * *

Criterion 19—Control room. * * *

Applicants for and holders of construction permits and operating licenses under this part or water-cooled commercial nuclear reactors under Framework B of part 53 of this chapter who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 or water-cooled commercial nuclear reactors under Framework B of part 53 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 or water-cooled commercial nuclear reactors under Framework B of part 53 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under § 50.67, shall satisfy the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.

11. In Appendix G to part 50, revise section IV.A.1.c to read as follows:

Appendix G to Part 50—Fracture Toughness Requirements

* * * * *

IV. Fracture Toughness Requirements

A. * * *

1. * * *

c. The analysis for satisfying the requirements of section IV.A.1 of this appendix must be submitted, as specified in § 50.4 or § 53.040 of this chapter, for review and approval on an individual case basis at least three years prior to the date when the predicted Charpy upper-shelf energy will no longer satisfy the requirements of section IV.A.1 of this appendix, or on a schedule approved by the Director, Office of Nuclear Reactor Regulation.

* * * * *

12. In appendix H to part 50, revise sections III.B.3 and IV.A to read as follows:

Appendix H to Part 50—Reactor Vessel Material Surveillance Program Requirements

* * * * *

III. Surveillance Program Criteria

* * * * *

B. * * *

3. A proposed withdrawal schedule must be submitted with a technical justification as specified in § 50.4 or § 53.040 of this chapter. The proposed schedule must be approved prior to implementation.

* * * * *

IV. Report of Test Results

A. Each capsule withdrawal and the test results must be the subject of a summary technical report to be submitted, as specified in § 50.4 or § 53.040 of this chapter, within eighteen months of the date of capsule withdrawal, unless an extension is granted by the Director, Office of Nuclear Reactor Regulation.

* * * * *

13. In appendix J to part 50, under Option A revise the Introduction paragraph and section II.A; under Option B revise the Introduction paragraph, section IV. Second paragraph and section V.A. and V.B.1 through 3 to read as follows:

Appendix J to Part 50—Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors

* * * * *

Option A—Prescriptive Requirements

* * * * *

I. Introduction

One of the conditions of all operating licenses under this part, operating licenses for water-cooled commercial nuclear reactors under Framework B of part 53 of this chapter, combined licenses under part 52 of this chapter for water-cooled power reactors, and combined licenses for water-cooled commercial nuclear reactors under Framework B of part 53, as specified in § 50.54(o) or § 53.4410 of this chapter, is that primary reactor containments shall satisfy the containment leakage test requirements set forth in this appendix. These test requirements provide for preoperational and periodic verification by tests of the leak-tight integrity of the primary reactor containment, and systems and components which penetrate containment of water-cooled power reactors, and establish the acceptance criteria for these tests. The purposes of the tests are to assure that (a) leakage through the primary reactor containment and systems and components

penetrating primary containment shall not exceed allowable leakage rate values as specified in the technical specifications or associated bases; and (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment. These test requirements may also be used for guidance in establishing appropriate containment leakage test requirements in technical specifications or associated bases for other types of nuclear power reactors.

II. Explanation of Terms

A. "Primary reactor containment" means the structure or vessel that encloses the components of the reactor coolant pressure boundary, as defined in § 50.2 or § 53.028, and serves as an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.

* * * * *

Option B—Performance-Based Requirements

* * * * *

I. Introduction

One of the conditions required of all operating licenses and combined licenses for light water–cooled power reactors as specified in § 50.54(o) or § 53.4410 of this chapter is that primary reactor containments satisfy the leakage-rate test requirements in either Option A or B of this appendix. These test requirements ensure that (a) leakage through these containments or systems and components penetrating these containments does not exceed allowable leakage rates specified in the technical specifications; and (b) integrity of the containment structure is maintained during its service life. Option B of this

appendix identifies the performance-based requirements and criteria for preoperational and subsequent periodic leakage-rate testing.[3]

* * * * *

IV. Recordkeeping

* * * * *

If the test results exceed the performance criteria (La) as defined in the plant Technical Specifications, those exceedances must be assessed for Emergency Notification System reporting under § 50.72 (b)(2)(i) or § 53.6330(b)(2)(i) of this chapter, and for a Licensee Event Report under § 50.73 (a)(2)(ii) or § 53.6340(a)(2)(ii) of this chapter.

* * * * *

V. Application

A. Applicability

The requirements in either or both Option B, III.A for Type A tests, and Option B, III.B for Type B and C tests, may be adopted on a voluntary basis by an operating nuclear power reactor licensee as specified in § 50.54 or § 53.4410 of this chapter in substitution of the requirements for those tests contained in Option A of this appendix. If the requirements for tests in Option B, III.A or Option B, III.B are implemented, the recordkeeping requirements in Option B, IV for these tests must be substituted for the reporting requirements of these tests contained in Option A of this appendix.

B. Implementation

1. Specific exemptions to Option A of this appendix that have been formally approved by the AEC or NRC, according to 10 CFR 50.12 or 10 CFR 53.080, are still applicable to Option B of this appendix if necessary, unless specifically revoked by the NRC.

2. A licensee or applicant for an operating license under this part or under Framework B of part 53 or a combined license under part 52 or under Framework B of part 53 of this

chapter may adopt Option B, or parts thereof, as specified in Section V.A of this appendix, by submitting its implementation plan and request for revision to technical specifications (see paragraph B.3 of this section) to the Director, Office of Nuclear Reactor Regulation.

3. The regulatory guide or other implementation document used by a licensee or applicant for an operating license under this part or under Framework B of part 53, or a combined license under part 52 or under Framework B of part 53 of this chapter to develop a performance-based leakage-testing program must be included, by general reference, in the plant technical specifications. The submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide.

* * * * *

14. In appendix S to part 50, revise the paragraph under “General Information,” and paragraph (a) under the Introduction section, and revise the following definitions under section III: Combined license, Design approval, Design certification, Manufacturing license, Structures, systems, and components required to withstand the effects of the safe-shutdown earthquake ground motion or surface deformation.

The revisions to read as follows:

Appendix S to Part 50—Earthquake Engineering Criteria for Nuclear Power Plants

General Information

This appendix applies to applicants for a construction permit or operating license under part 50, ~~or~~ a design certification, combined license, design approval, or manufacturing license under part 52 of this chapter, on or after January 10, 1997, or a construction permit, operating license, design certification, combined license, design

approval, or manufacturing license under Framework B of part 53 of this chapter except for those using the alternative seismic design criteria of § 53.4733. However, for either an operating license applicant or holder whose construction permit was issued before January 10, 1997, the earthquake engineering criteria in Section VI of appendix A to 10 CFR part 100 continue to apply. Paragraphs IV.a.1.i, IV.a.1.ii, IV.4.b, and IV.4.c of this appendix apply to applicants for an early site permit under part 52.

I. *Introduction*

(a) Each applicant for a construction permit, operating license, design certification, combined license, design approval, or manufacturing license is required by §§ 50.34(a)(12), 50.34(b)(10), or 10 CFR 52.47, 52.79, 52.137, or 52.157, or 53.4709, 53.4839, 53.4879, 53.4909, 53.4969, or 53.5016 and General Design Criterion 2 of appendix A to this part, to design nuclear power plant structures, systems, and components important to safety to withstand the effects of natural phenomena, such as earthquakes, without loss of capability to perform their safety functions. Also, as specified in §§ 50.54(ff) or 53.4215, nuclear power plants that have implemented the earthquake engineering criteria described herein must shut down if the criteria in paragraph IV(a)(3) of this appendix are exceeded.

* * * * *

III. *Definitions*

As used in these criteria:

Combined license means a combined construction permit and operating license with conditions for a nuclear power facility issued under subpart C of part 52 or Framework B of part 53 of this chapter.

Design Approval means an NRC staff approval, issued under subpart E of part 52 or Framework B of part 53 of this chapter, of a final standard design for a nuclear power reactor of the type described in 10 CFR 50.22.

Design Certification means a Commission approval, issued under subpart B of part 52 or Framework B of part 53 of this chapter, of a standard design for a nuclear power facility.

Manufacturing license means a license, issued under subpart F of part 52 or Framework B of part 53 of this chapter, authorizing the manufacture of nuclear power reactors but not their installation into facilities located at the sites on which the facilities are to be operated.

* * * * *

Structures, systems, and components required to withstand the effects of the safe-shutdown earthquake ground motion or surface deformation are those necessary to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe-shutdown condition; or

(3) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of § 50.34(a)(1) or § 53.4730(a)(1)(vi).

* * * * *

PART 51 – ENVIRONMENTAL PROTECTION REGULATIONS FOR DOMESTIC LICENSING AND RELATED REGULATORY FUNCTIONS

1. The authority citation for part 51 continues to read as follows:

Authority: Atomic Energy Act of 1954, secs. 161, 193 (42 U.S.C. 2201, 2243); Energy Reorganization Act of 1974, secs. 201, 202 (42 U.S.C. 5841, 5842); National Environmental Policy Act of 1969 (42 U.S.C. 4332, 4334, 4335); Nuclear Waste Policy Act of 1982, secs. 144(f), 121, 135, 141, 148 (42 U.S.C. 10134(f), 10141, 10155, 10161, 10168); 44 U.S.C. 3504 note.

Subpart A – National Environmental Policy Act – Regulations Implementing Section 102(2)

2. Revise § 51.20, paragraphs (b)(1)-(2) to read as follows:

§ 51.20 Criteria for and identification of licensing and regulatory actions requiring environmental impact statements.

* * * * *

(b) * * *

(1) Issuance of a limited work authorization or a permit to construct a nuclear power reactor, testing facility, or fuel reprocessing plant under part 50 of this chapter, issuance of an early site permit under part 52 of this chapter, or issuance of a construction permit or early site permit under part 53 of this chapter.

(2) Issuance or renewal of a full power or design capacity license to operate a nuclear power reactor, testing facility, or fuel reprocessing plant under parts 50 or 53 of this chapter, or a combined license under parts 52 or 53 of this chapter.

* * * * *

3. Revise § 51.22, paragraphs (c)(3), (c)(9), (c)(12), (c)(17), and (c)(22)-(23) to read as follows:

§ 51.22 Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review.

* * * * *

(c) * * *

(3) Amendments to parts 20, 30, 31, 32, 33, 34, 35, 37, 39, 40, 50, 51, 52, 53, 54, 60, 61, 63, 70, 71, 72, 73, 74, 81, and 100 of this chapter which relate to—

* * *

(9) Issuance of an amendment to a permit or license for a reactor under parts 50, 52, or 53 of this chapter that changes a requirement or issuance of an exemption from a requirement, with respect to installation or use of a facility component located within the restricted area, as defined in part 20 of this chapter; or the issuance of an amendment to a permit or license for a reactor under parts 50, 52, or 53 of this chapter that changes an inspection or a surveillance requirement; provided that:

* * *

(12) Issuance of an amendment to a license under parts 50, 52, 53, 60, 61, 63, 70, 72, or 75 of this chapter relating solely to safeguards matters (i.e., protection against sabotage or loss or diversion of special nuclear material) or issuance of an approval of a safeguards plan submitted under parts 50, 52, 53, 70, 72, and 73 of this chapter, provided that the amendment or approval does not involve any significant construction impacts. These amendments and approvals are confined to—

* * *

(17) Issuance of an amendment to a permit or license under parts 30, 40, 50, 52, 53, or part 70 of this chapter which deletes any limiting condition of operation or monitoring requirement based on or applicable to any matter subject to the provisions of the Federal Water Pollution Control Act.

* * *

(22) Issuance of a standard design approval under parts 52 or 53 of this chapter.

(23) The Commission finding for a combined license under §§ 52.103(g), 53.1452(g), or 53.5052(g) of this chapter.

* * * * *

4. Revise § 51.26, paragraph (d) to read as follows:

§ 51.26 Requirement to publish notice of intent and conduct scoping process.

* * * * *

(d) Whenever the appropriate NRC staff director determines that a supplement to an environmental impact statement will be prepared by the NRC, a notice of intent will be prepared as provided in § 51.27, and will be published in the *Federal Register* as provided in § 51.116. The NRC staff need not conduct a scoping process (see §§ 51.27, 51.28, and 51.29), provided, however, that if scoping is conducted, then the scoping must be directed at matters to be addressed in the supplement. If scoping is conducted in a proceeding for a combined license referencing an early site permit under parts 52 or 53, then the scoping must be directed at matters to be addressed in the supplement as described in § 51.92(e).

5. In § 51.30, revise the introductory text for paragraph (a) and revise paragraphs (d) and (e) to read as follows:

§ 51.30 Environmental assessment.

(a) An environmental assessment for proposed actions, other than those for a standard design certification under 10 CFR parts 52 or 53, or a manufacturing license under parts 52 or 53, shall identify the proposed action and include:

* * * * *

(d) An environmental assessment for a standard design certification under subpart B of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter must identify the proposed action, and will be limited to the consideration of the costs and benefits of severe accident mitigation design alternatives and the bases for not incorporating severe accident mitigation design alternatives in the design

certification. An environmental assessment for an amendment to a design certification will be limited to the consideration of whether the design change which is the subject of the proposed amendment renders a severe accident mitigation design alternative previously rejected in the earlier environmental assessment to become cost beneficial, or results in the identification of new severe accident mitigation design alternatives, in which case the costs and benefits of new severe accident mitigation design alternatives and the bases for not incorporating new severe accident mitigation design alternatives in the design certification must be addressed.

(e) An environmental assessment for a manufacturing license under subpart F of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter must identify the proposed action and will be limited to the consideration of the costs and benefits of severe accident mitigation design alternatives and the bases for not incorporating severe accident mitigation design alternatives in the manufacturing license. An environmental assessment for an amendment to a manufacturing license will be limited to consideration of whether the design change which is the subject of the proposed amendment either renders a severe accident mitigation design alternative previously rejected in an environmental assessment to become cost beneficial, or results in the identification of new severe accident mitigation design alternatives, in which case the costs and benefits of new severe accident mitigation design alternatives and the bases for not incorporating new severe accident mitigation design alternatives in the manufacturing license must be addressed. In either case, the environmental assessment will not address the environmental impacts associated with manufacturing the reactor under the manufacturing license.

6. Revise § 51.31, paragraph (a) to read as follows:

§ 51.31 Determinations based on environmental assessment.

(a) *General*. Upon completion of an environmental assessment for proposed actions other than those involving a standard design certification or a manufacturing license under parts 52 or 53 of this chapter, the appropriate NRC staff director will determine whether to prepare an environmental impact statement or a finding of no significant impact on the proposed action. As provided in § 51.33, a determination to prepare a draft finding of no significant impact may be made.

* * * * *

7. Revise § 51.32, paragraphs (b)(1) and (3) to read as follows:

§ 51.32 Finding of no significant impact.

* * * * *

(b) * * *

(1) A standard design certification under subpart B of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter;

* * *

(3) A manufacturing license under subpart F of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter; or

* * *

8. In § 51.49, revise the introductory text for paragraph (c) to read as follows:

§ 51.49 Environmental report-limited work authorization.

* * * * *

(c) *Limited work authorization submitted as part of an early site permit application*. Each applicant for an early site permit under subpart A of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter requesting a limited work authorization shall submit with its application the environmental report

required by § 51.50(b). Each environmental report must contain the following information:

* * * * *

9. Revise § 51.50, paragraphs (a), (b)(4), and the introductory text for paragraph (c) to read as follows:

§ 51.50 Environmental report – construction permit, early site permit, or combined license stage.

(a) *Construction permit stage*. Each applicant for a permit to construct a production or utilization facility covered by § 51.20 shall submit with its application a separate document, entitled "Applicant's Environmental Report—Construction Permit Stage," which shall contain the information specified in §§ 51.45, 51.51, and 51.52. Each environmental report shall identify procedures for reporting and keeping records of environmental data, and any conditions and monitoring requirements for protecting the non-aquatic environment, proposed for possible inclusion in the license as environmental conditions in accordance with §§ 50.36b, 53.1112, or 53.4712 of this chapter. As stated in § 51.23, no discussion of the environmental impacts of the continued storage of spent fuel is required in this report.

(b) * * *

(4) Each environmental report must identify the procedures for reporting and keeping records of environmental data, and any conditions and monitoring requirements for protecting the non-aquatic environment, proposed for possible inclusion in the license as environmental conditions in accordance with §§ 50.36b, 53.1112, or 53.4712 of this chapter.

(c) *Combined license stage*. Each applicant for a combined license shall submit with its application a separate document, entitled "Applicant's Environmental Report—

Combined License Stage." Each environmental report shall contain the information specified in §§ 51.45, 51.51, and 51.52, as modified in this paragraph. For other than light-water-cooled nuclear power reactors, the environmental report shall contain the basis for evaluating the contribution of the environmental effects of fuel cycle activities for the nuclear power reactor. Each environmental report shall identify procedures for reporting and keeping records of environmental data, and any conditions and monitoring requirements for protecting the non-aquatic environment, proposed for possible inclusion in the license as environmental conditions in accordance with §§ 50.36b, 53.1112, or 53.4712 of this chapter. The combined license environmental report may reference information contained in a final environmental document previously prepared by the NRC staff. As stated in § 51.23, no discussion of the environmental impacts of the continued storage of spent fuel is required in this report.

* * *

10. Revise § 51.53, paragraph (d) to read as follows:

§ 51.53 Postconstruction environmental reports.

* * * * *

(d) *Postoperating license stage*. Each applicant for a license amendment authorizing decommissioning activities for a production or utilization facility either for unrestricted use or based on continuing use restrictions applicable to the site; and each applicant for a license amendment approving a license termination plan or decommissioning plan under § 50.82, § 53.1080, or § 53.4680 of this chapter either for unrestricted use or based on continuing use restrictions applicable to the site; and each applicant for a license or license amendment to store spent fuel at a nuclear power reactor after expiration of the operating license for the nuclear power reactor shall submit with its application a separate document, entitled "Supplement to Applicant's

Environmental Report—Post Operating License Stage," which will update "Applicant's Environmental Report—Operating License Stage," as appropriate, to reflect any new information or significant environmental change associated with the applicant's proposed decommissioning activities or with the applicant's proposed activities with respect to the planned storage of spent fuel. As stated in § 51.23, no discussion of the environmental impacts of the continued storage of spent fuel is required in this report. The "Supplement to Applicant's Environmental Report—Post Operating License Stage" may incorporate by reference any information contained in "Applicant's Environmental Report—Construction Permit Stage."

11. Revise § 51.54, paragraph (a) to read as follows:

§ 51.54 Environmental report – manufacturing license.

(a) Each applicant for a manufacturing license under subpart F of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter shall submit with its application a separate document entitled, "Applicant's Environmental Report—Manufacturing License." The environmental report must address 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. The environmental report need not address the environmental impacts associated with manufacturing the reactor under the manufacturing license, the benefits and impacts of utilizing the reactor in a nuclear power plant, or an evaluation of alternative energy sources.

* * * * *

12. Revise § 51.55, paragraph (a) to read as follows:

§ 51.55 Environmental report – standard design certification.

(a) Each applicant for a standard design certification under subpart B of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter shall submit with its application a separate document entitled, "Applicant's Environmental Report—Standard Design Certification." The environmental report must address the costs and benefits of severe accident mitigation design alternatives, and the bases for not incorporating severe accident mitigation design alternatives in the design to be certified.

* * * * *

13. Revise § 51.58, paragraph (b) to read as follows:

§ 51.58 Environmental report – number of copies; distribution.

* * * * *

(b) Each applicant for a license to manufacture a nuclear power reactor, or for an amendment to a license to manufacture, seeking approval of the final design of the nuclear power reactor under subpart F of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080 of this chapter, shall submit to the Commission an environmental report or any supplement to an environmental report in the manner specified in §§ 52.3 or 53.040 of this chapter. The applicant shall maintain the capability to generate additional copies of the environmental report or any supplement to the environmental report for subsequent distribution to parties and Boards in the NRC proceeding; Federal, State, and local officials; and any affected Indian Tribes, in accordance with written instructions issued by the Director of the Office of Nuclear Reactor Regulation.

14. In § 51.77, revise the introductory text for paragraph (a) to read as follows:

§ 51.77 Distribution of draft environmental impact statement.

(a) In addition to the distribution authorized by § 51.74, a copy of a draft environmental statement for a licensing action for a production or utilization facility, except an action authorizing issuance, amendment or renewal of a license to manufacture a nuclear power reactor pursuant to 10 CFR part 52, appendix M, or §§ 53.1270, or 53.4870 will also be distributed to:

* * * * *

15. Revise § 51.92, paragraph (b) to read as follows:

§ 51.92 Supplement to the final environmental impact statement.

* * * * *

(b) In a proceeding for a combined license application under 10 CFR parts 52 or 53 referencing an early site permit under parts 52 or 53, the NRC staff shall prepare a supplement to the final environmental impact statement for the referenced early site permit in accordance with paragraph (e) of this section.

* * * * *

16. In § 51.95, revise the introductory text for paragraph (c) to read as follows:

§ 51.95 Postconstruction environmental impact statements.

* * * * *

(c) Operating license renewal stage. In connection with the renewal of an operating license or combined license for a nuclear power plant under 10 CFR parts 52, 53 or 54 of this chapter, the Commission shall prepare an environmental impact statement, which is a supplement to the Commission's NUREG–1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants" (June 2013), which is available in the NRC's Public Document Room, 11555 Rockville Pike, Rockville, Maryland 20852.

* * * * *

17. Revise § 51.101, paragraph (a)(2) to read as follows:

§ 51.101 Limitations on actions.

(a) * * *

(2) Any action concerning the proposal taken by an applicant which would (i) have an adverse environmental impact, or (ii) limit the choice of reasonable alternatives may be grounds for denial of the license. In the case of an application covered by §§ 30.32(f), 40.31(f), 50.10(c), 53.1130, 53.4740, 70.21(f), or §§ 72.16 and 72.34 of this chapter, the provisions of this paragraph will be applied in accordance with §§ 30.33(a)(5), 40.32(e), 50.10 (c), 53.1130, 53.4740, , 70.23(a)(7), or § 72.40(b) of this chapter, as appropriate.

* * * * *

18. Revise § 51.103, paragraph (a)(6) to read as follows:

§ 51.103 Record of decision – general.

(a) * * *

(6) In a construction permit or a combined license proceeding where a limited work authorization under §§ 10 CFR parts 50.10, 53.1130, or 53.4740 was issued, the Commission's decision on the construction permit or combined license application will not address or consider the sunk costs associated with the limited work authorization in determining the proposed action.

* * * * *

19. Revise § 51.105, paragraph (c)(1) to read as follows:

§ 51.105 Public hearings in proceedings for issuance of construction permits or early site permits; limited work authorizations.

* * * * *

(c)(1) In addition to complying with the applicable provisions of § 51.104, in any proceeding for the issuance of a construction permit for a nuclear power plant or an early site permit under parts 52 or 53 of this chapter, where the applicant requests a limited work authorization under §§ 50.10(d), 53.1130, or 53.4740 of this chapter, the presiding officer shall—

* * *

20. In § 51.107, revise the introductory text for paragraphs (a) and (b), and revise paragraph (d)(1), to read as follows:

§ 51.107 Public hearings in proceedings for issuance of combined licenses; limited work authorizations.

(a) In addition to complying with the applicable requirements of § 51.104, in a proceeding for the issuance of a combined license for a nuclear power reactor under parts 52 or 53 of this chapter, the presiding officer will:

* * *

(b) If a combined license application references an early site permit, then the presiding officer in the combined license hearing shall not admit any contention proffered by any party on environmental issues which have been accorded finality under §§ 52.39, 53.1188, or 53.4798 of this chapter, unless the contention:

* * * * *

(d)(1) In any proceeding for the issuance of a combined license where the applicant requests a limited work authorization under §§ 50.10(d), 53.1130(a), or 53.4740(a) of this chapter, the presiding officer, in addition to complying with any applicable provision of § 51.104, shall:

* * *

21. Revise § 51.108 to read as follows:

§ 51.108 Public hearings on Commission findings that inspections, tests, analyses, and acceptance criteria of combined licenses are met.

In any public hearing requested under §§~~10 CFR~~ 52.103(b), 53.1452(b), or 53.5052(b), the Commission will not admit any contentions on environmental issues, the adequacy of the environmental impact statement for the combined license issued under subpart C of part 52, §§ 53.1100 through 53.1480, or §§ 53.4700 through 53.5080, or the adequacy of any other environmental impact statement or environmental assessment referenced in the combined license application. The Commission will not make any environmental findings in connection with the finding under §§~~10 CFR~~ 52.103(g), 53.1452(g), or 53.5052(g).

PART 53—RISK-INFORMED, TECHNOLOGY-INCLUSIVE REGULATORY FRAMEWORK FOR COMMERCIAL NUCLEAR PLANTS

1. The authority citation for part 53 to read as follows:

Authority: Atomic Energy Act of 1954, secs. 11, 101, 103, 108, 122, 147, 161, 181, 182, 183, 184, 185, 186, 187, 189, 223, 234 (42 U.S.C. 2014, 2131, 2132, 2133, 2134, 2135, 2138, 2152, 2167, 2169, 2201, 2231, 2232, 2233, 2234, 2235, 2236, 2237, 2239, 2273, 2282); Energy Reorganization Act of 1974, secs. 201, 202, 206, 211 (42 U.S.C. 5841, 5842, 5846, 5851); Nuclear Waste Policy Act of 1982, sec. 306 (42 U.S.C. 10226); National Environmental Policy Act of 1969 (42 U.S.C. 4332); 44 U.S.C. 3504 note; Sec. 109, Pub. L. 96-295, 94 Stat. 783; Pub. L. 115-439, 132 Stat. 5571.

2. The title to part 53 to read as set out above.

3. Add part 53 to 10 CFR Chapter 1, to read as follows:

Sec.

53.000 Purpose.

53.010 Frameworks.

Subpart A—General Provisions

53.015 Scope.

53.020 Definitions.
53.024 Definitions specific to Framework A.
53.028 Definitions specific to Framework B.
53.030 Reserved.
53.040 Written communications.
53.050 Deliberate misconduct.
53.060 Employee protection.
53.070 Completeness and accuracy of information.
53.080 Specific exemptions.
53.090 Common standards
53.100 Jurisdictional limits.
53.110 Attacks and destructive acts.
53.115 Rights related to special nuclear material.
53.117 License suspension and rights of recapture.
53.120 Information collection requirements: OMB approval.

Subpart B—Technology-Inclusive Safety Requirements

Sec.
53.200 Safety objectives.
53.210 Safety criteria for design-basis accidents.
53.220 Safety criteria for licensing-basis events other than design-basis accidents.
53.230 Safety functions.
53.240 Licensing-basis events.
53.250 Defense in depth.
53.260 Normal operations.
53.270 Protection of plant workers.

Subpart C—Design and Analysis Requirements

Sec.
53.400 Design features for licensing-basis events.
53.410 Functional design criteria for design-basis accidents.

53.420 Functional design criteria for licensing-basis events other than design-basis accidents.

53.425 Design features and functional design criteria for normal operations.

53.430 Design features and functional design criteria for protection of plant workers.

53.440 Design requirements.

53.450 Analysis requirements.

53.460 Safety categorization and special treatment.

53.470 Maintaining analytical safety margins used to justify operational flexibilities.

53.480 Earthquake engineering.

Subpart D—Siting Requirements

Sec.

53.500 General siting.

53.510 External hazards.

53.520 Site characteristics.

53.530 Population-related considerations.

53.540 Siting interfaces.

Subpart E—Construction and Manufacturing Requirements

Sec.

53.600 Construction and manufacturing – scope and purpose.

53.605 Reporting of defects and noncompliance.

53.610 Construction.

53.620 Manufacturing.

Subpart F—Requirements for Operation

Sec.

53.700 Operational objectives.

53.710 Maintaining capabilities and availability of structures, systems, and components.

53.715 Maintenance, repair, and inspection programs.

53.720 Response to seismic events.

53.725 General staffing, training, personnel qualifications, and human factors requirements.

53.726 Communications.

53.727 Information collection requirements.

53.728 Completeness and accuracy of information.

53.730 Defining, fulfilling, and maintaining the role of personnel in ensuring safe operations.

53.735 General exemptions.

53.740 Facility licensee requirements – General

53.745 Operator license requirements.

53.750 Violations.

53.755 Criminal penalties.

53.760 Operator licensing.

53.765 Medical requirements.

53.770 Incapacitation because of disability or illness.

53.775 Applications for operators and senior operators.

53.780 Training, examination, and proficiency program.

53.785 Conditions of operator and senior operator licenses.

53.790 Issuance, modification, and revocation of operator and senior operator licenses.

53.795 Expiration and renewal of operator and senior operator licenses.

53.800 Facility licensees that comply with §§ 53.800 through 53.820.

53.805 Facility licensee requirements related to generally licensed reactor operators.

53.810 Generally licensed reactor operators.

53.815 Generally licensed reactor operator training, examination, and proficiency programs.

53.820 Expiration.

53.830 Training and qualification of commercial nuclear plant personnel.

53.845 Programs.

53.850 Radiation protection.

53.855 Emergency preparedness.

53.860 Security programs.

53.865 Quality assurance.

53.870 Integrity assessment programs.

53.875 Fire protection.

53.880 Inservice inspection/inservice testing.

53.890 Facility safety program.

53.910 Procedures and guidelines.

Subpart G¬—Decommissioning Requirements

Sec.

53.1000 Scope and purpose.

53.1010 Financial assurance for decommissioning.

53.1020 Cost estimates for required decommissioning funds.

53.1030 Annual adjustments.

53.1040 Methods for providing financial assurance for decommissioning funds.

53.1045 Financial management of decommissioning funds.

53.1050 NRC oversight.

53.1060 Reporting and recordkeeping requirements.

53.1070 Termination of license.

53.1075 Program requirements during decommissioning.

53.1080 Release of part of a commercial nuclear plant or site for unrestricted use.

Subpart H—Licenses, Certifications and Approvals

Sec.

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53.6040 Updating licensing basis information and determining the need for NRC approval.

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§ 53.000 Purpose.

This part provides optional frameworks for the issuance, amendment, renewal, and termination of licenses, permits, certifications, and approvals for commercial nuclear plants licensed under Section 103 of the Atomic Energy Act of 1954, as amended (AEA) (68 Stat. 919), and Title II of the Energy Reorganization Act of 1974, as amended (ERA)

(88 Stat. 1242). Also, this part 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 U.S. Nuclear Regulatory Commission (NRC) enforcement action for violation of the provisions in § 53.050.

§ 53.010 Frameworks.

This part provides two optional frameworks, Framework A and Framework B. The two frameworks are distinct. A license issued under Framework A is subject to the requirements under Subpart A and Subparts B through K of this part. A license issued under Framework B is subject to the requirements under Subpart A and Subparts N through U of this part. Applicants and licensees subject to the rules in this part must only use the subparts applicable to one framework, except where stated. Consequently, an applicant for a license, certificate, or permit may not reference or rely on a license, certificate, or permit issued under another framework.

Subpart A—General Provisions

§ 53.015 Scope.

Subpart A provides general provisions applicable to all applicants and licensees subject to the rules of this part.

§ 53.020 Definitions.

For the purpose of this part:

Applicant means a person applying for a license, permit, or other form of Commission permission or approval under this part.

Certified fuel handler means, for a commercial nuclear plant, either:

(1) A non-licensed operator who has qualified in accordance with a fuel handler training program approved by the Commission; or

(2) A non-licensed operator who demonstrates compliance with the following criteria:

(i) Has qualified in accordance with a fuel handler training program that demonstrates compliance with the same requirements as training programs for non-licensed operators required by § 53.830, and

(ii) Is responsible for decisions on:

(A) Safe conduct of decommissioning activities,

(B) Safe handling and storage of spent fuel, and

(C) Appropriate response to plant emergencies.

Combined license means a combined construction permit and operating license with conditions for a commercial nuclear plant issued under this part.

Commercial nuclear plant means a facility consisting of one or more commercial nuclear reactors and associated co-located support facilities, including the collection of buildings, radionuclide sources, and structures, systems, and components for which a license(s) is being sought under this part, that is used for producing power for commercial electric power or other commercial purposes. For the purposes of requirements in this part that reference requirements in 10 CFR part 50, a commercial nuclear plant is equivalent to a nuclear power plant.

Commercial nuclear reactor means an apparatus, other than an atomic weapon, designed or used to sustain nuclear fission. For the purposes of requirements in this part

that reference requirements in 10 CFR part 50, a commercial nuclear reactor is equivalent to a nuclear reactor as defined in 10 CFR 50.2.

Commission means the NRC or its duly authorized representatives.

Consensus code or standard means any technical standard that is:

(1) Developed or adopted by a voluntary consensus standard body under procedures that assure that persons having interests within the scope of the standard that are affected by the provisions of the standard have reached substantial agreement on its adoption;

(2) Formulated in a manner that afforded an opportunity for diverse views to be considered; and

(3) Designated by the standards body as a consensus code or standard.

Custom combined license means a combined license that does not reference a standard design certification or design certification.

Decommission or decommissioning means to remove a plant or site safely from service and reduce residual radioactivity to a level that permits:

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

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

Defense in depth means inclusion of two or more independent and redundant layers of defense in the design of a facility and its operating procedures to compensate for uncertainties such that no single layer of defense, no matter how robust, is exclusively relied upon. Defense in depth includes, but is not limited to, the use of access controls, physical barriers, redundant and diverse safety functions, and emergency response measures.

Design features means the active and passive SSCs and inherent characteristics of those SSCs that contribute to limiting the total effective dose equivalent to individual members of the public during normal operations and prevent or mitigate the consequences of event sequences.

Electric utility means any entity that generates or distributes electricity and that recovers the cost of this electricity, either directly or indirectly, through rates established by the entity itself or by a separate regulatory authority. Investor-owned utilities, including generation or distribution subsidiaries, public utility districts, municipalities, rural electric cooperatives, and State and Federal agencies, including associations of any of the foregoing, are included within the meaning of "electric utility."

Event sequence means a postulated initiating event defined for a set of initial plant conditions followed by system, safety function, and operator successes or failures, and terminating in a specified end state depending on the system, safety function, and operator successes and failures (e.g., prevention of release of radioactive material or release in one of the reactor-specific release categories). An event sequence may include many unique variations of events that are similar in terms of results or end states.

Exclusion area means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to the facility as to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety. Residence within the exclusion area must normally be prohibited. In any event, residents must be subject to ready removal in case

of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazards to the public health and safety will result.

Fission product release means the amount and composition of radioactive material released to the environment, after accounting for any retention of radionuclides provided by reactor design features.

Fuel means special nuclear material (SNM) or source material, discrete elements that physically contain SNM or source material, and homogeneous mixtures that contain SNM or source material, intended to or used to create power in a commercial nuclear plant.

License means a limited work authorization (LWA), construction permit, operating license, early site permit, combined license, or manufacturing license under this part, or a renewed license issued by the Commission under this part.

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

Licensing basis information means the information contained in regulations, orders, licenses, certifications, or approvals issued by the NRC for a commercial nuclear plant licensed under this part and that information submitted to the NRC by an applicant or licensee in a Safety Analysis Report, program description, or other licensing-related document required under this part.

Low population zone means the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken on their behalf in the event of a serious accident. A permissible population density or total population within this zone is not included in this definition because the situation may

vary from case to case. Whether a specific number of people can, for example, be evacuated from a specific area or instructed to take shelter on a timely basis, will depend on many factors such as location, number and size of highways, scope and extent of advance planning, and actual distribution of residents within the area.

Manufactured reactor means the essential portions of a nuclear reactor that are manufactured under a manufacturing license and subsequently transported and incorporated into a commercial nuclear plant under a combined license.

Manufactured reactor module means a manufactured reactor loaded with fuel prior to transport to a licensed location for installation and commercial operation.

Manufacturing license means a license issued under this part that authorizes the manufacture of a manufactured reactor or manufactured reactor module but not their construction, installation, or operation.

Person means: (1) any individual, corporation, partnership, firm, association, trust, estate, public or private institution, group, government agency other than the Commission, any State or any political subdivision of, or any political entity within a State, any foreign government or nation or any political subdivision of any such government or nation, or other entity; and (2) any legal successor, representative, agent, or agency of the foregoing.

Population center distance means the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents.

Probabilistic risk assessment (PRA) means a quantitative assessment of the risk associated with plant operation and maintenance that is measured in terms of event sequence occurrence frequencies and consequences.

Programmatic controls means administrative procedures that govern human action in implementing programs and operating, monitoring, and maintaining SSCs and

equipment of a commercial nuclear plant. Programmatic controls are specified in an application for a requested activity of the Commission.

Prototype plant means a nuclear reactor that is used to test design features. A 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.

Quality assurance means all those planned and systematic actions necessary to ensure that an SSC will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system which provide a means to control the quality of the material, structure, component, or system to predetermined requirements.

Safety function means a purpose served by a design feature, human action, or programmatic control to prevent or mitigate unplanned events and thereby demonstrate compliance with requirements in part 53 for limiting risks to public health and safety. Safety functions can be performed by any combination of the elements listed above and can be specified at the plant level or at the level of a particular barrier or system. The approach to identifying and addressing safety functions in Frameworks A and B are as follows:

(1) Within Framework A, the primary safety function is stated to be limiting the release of radioactive materials. Additional safety functions supporting the retention of radioactive materials, such as controlling reactivity, heat generation, heat removal, and chemical interactions, are determined for each reactor design by analyzing a spectrum of unplanned events.

(2) Within Framework B, multiple plant-level safety functions are assumed to apply to all reactor designs based on established requirements and historical practices. These fundamental safety functions include the control of reactivity, removal of heat, and limiting the release of radioactive materials. The protection of a specific barrier or system that contributes to meeting plant-level safety criteria may also be referred to as a safety function.

Site characteristics means the actual physical, environmental, and demographic features of a site. Site characteristics are specified in an early site permit or in a Preliminary or Final Safety Analysis Report for an LWA, a construction permit, or a combined license, as applicable.

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.

Special nuclear material (SNM) means: (1) plutonium, uranium-233, uranium enriched in the isotope-233 or in the isotope-235, and any other material which the Commission, pursuant to the provisions of Section 51 of the AEA, determines to be special nuclear material, but does not include source material; or (2) any material artificially enriched by any of the foregoing, but does not include source material.

Standard design means a design which is sufficiently detailed and complete to support certification or approval in accordance with subpart H or subpart R of this part, and which is usable under Framework A or Framework B of this part, as appropriate, 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 H or subpart R of this part, of a final standard design for a

commercial nuclear plant. 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 H or subpart R of this part, of a final standard design for a nuclear power facility. This design may be referred to as a certified standard design.

Total effective dose equivalent (TEDE) means the sum of the effective dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).

Utilization facility means any commercial nuclear reactor other than one designed or used primarily for the formation of plutonium or uranium–233.

§ 53.024 Definitions Specific to Framework A.

For the purpose of Framework A of this part:

Anticipated event sequence means event sequences expected to occur one or more times during the life of a commercial nuclear plant. Anticipated event sequences take into account the expected response of all SSCs within the plant, regardless of safety classification.

Construction means the activities in paragraph (1) below and does not mean the activities in paragraph (2) below.

(1) Activities constituting construction are those activities credited or relied upon for demonstrating compliance with the safety criteria defined in subpart B of this part which are conducted on-site to build the commercial nuclear plant, including the driving of piles; subsurface preparation; placement of backfill, concrete, or permanent retaining walls within an excavation; installation of foundations; or in-place assembly, erection, fabrication, or testing, which are for:

(i) Safety-related (SR) and non-safety-related but safety-significant (NSRSS) structures, systems, or components (SSCs) of a facility;

(ii) SSCs necessary to comply with 10 CFR part 73; or

(iii) Onsite emergency facilities necessary to comply with § 53.855.

(2) Construction does not include:

(i) Changes for temporary use of the land for public recreational purposes;

(ii) Site exploration, including necessary borings to determine foundation conditions or other preconstruction monitoring to establish background information related to the suitability of the site, the environmental impacts of construction or operation, or the protection of environmental values;

(iii) Preparation of a site for construction of a facility, including clearing of the site, grading, installation of drainage, erosion and other environmental mitigation measures, and construction of temporary roads and borrow areas;

(iv) Erection of fences and other access control measures;

(v) Excavation;

(vi) Erection of support buildings (such as construction equipment storage sheds, warehouse and shop facilities, utilities, concrete mixing plants, docking and unloading facilities, and office buildings) for use in connection with the construction of the facility;

(vii) Building of service facilities (such as paved roads, parking lots, railroad spurs, exterior utility and lighting systems, potable water systems, sanitary sewage treatment facilities, and transmission lines);

(viii) Procurement or fabrication of components or portions of the proposed facility occurring at locations other than the final, in-place location at the facility; or

(ix) Manufacture of a nuclear power reactor under a manufacturing license under subpart H of this part to be installed at the proposed site and to be part of the proposed facility.

Design-basis accidents (DBAs) means postulated event sequences that are used to set functional design criteria and performance objectives for the design of safety-related SSCs through deterministic analyses. DBAs are a type of licensing-basis event (LBE) and are based on the capabilities and reliabilities of safety-related SSCs needed to mitigate and prevent event sequences, respectively.

Design-basis external hazard level means the level of severity or intensity of an external hazard for which the safety-related SSCs are designed to withstand with no adverse impact on their capability to perform their safety functions.

Functional design criteria means metrics for the performance of SSCs. For safety-related SSCs, these criteria define performance metrics necessary to demonstrate compliance with safety criteria in § 53.210. For NSRSS SSCs, these criteria define performance metrics necessary to demonstrate compliance with the safety criteria in § 53.220.

Licensing-basis events (LBEs) means a collection of event sequences considered in the design and licensing of the commercial nuclear plant. LBEs are unplanned events and include anticipated event sequences, unlikely event sequences, very unlikely event sequences, and DBAs.

Non-Safety-Related but Safety-Significant (NSRSS) SSCs means those SSCs which are not safety related but are relied on to achieve adequate defense in depth or perform risk-significant functions and warrant special treatment.

Non-Safety-Significant (NSS) SSCs means those SSCs that are not safety related, or NSRSS, are not relied on to achieve adequate defense in depth or to perform risk-significant functions, and do not warrant special treatment.

Safety criteria means performance-based metrics that establish a level of safety provided in requirements in §§ 53.210 and 53.220.

Safety-related (SR) SSCs means those SSCs that are relied upon to demonstrate compliance with the safety criteria in § 53.210 and warrant special treatment.

Special treatment means those requirements, such as quality assurance and programmatic controls, that ensure that safety-related and NSRSS SSCs will provide defense in depth or perform risk-significant functions. The requirements also ensure that the SSCs will perform under the service conditions and with the reliability assumed in the analysis performed in accordance with § 53.450 to demonstrate compliance with the safety criteria in §§ 53.210 and 53.220.

Unlikely event sequences means event sequences that are not expected to occur in the life of a commercial nuclear plant and are less likely than anticipated event sequences, but are infrequent rather than rare. Unlikely event sequences take into account the expected response of all SSCs within the plant regardless of safety classification.

Very unlikely event sequences means event sequences that are not expected to occur in the life of a commercial nuclear plant, are less likely than an unlikely event sequence, and are rare. Very unlikely event sequences take into account the expected response of all SSCs within the plant regardless of safety classification.

§ 53.028 Definitions Specific to Framework B.

For the purpose of Framework B of this part:

Anticipated operational occurrences (AOOs) means those conditions of normal operation which are expected to occur one or more times during the life of the commercial nuclear reactor.

Construction means the activities in paragraph (1) below, and does not mean the activities in paragraph (2) below.

(1) Activities constituting construction are the driving of piles; subsurface preparation; placement of backfill, concrete, or permanent retaining walls within an excavation; installation of foundations; or in-place assembly, erection, fabrication, or testing, which are for:

(i) Safety-related structures, systems, and components (SSCs), as defined in § 53.028;

(ii) SSCs that are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures;

(iii) SSCs whose failure could prevent safety-related SSCs from fulfilling their safety-related function;

(iv) SSCs whose failure could cause a reactor scram or actuation of a safety-related system;

(v) SSCs necessary to comply with 10 CFR part 73;

(vi) SSCs necessary to comply with § 53.4350; or

(vii) Onsite emergency facilities necessary to comply with § 53.4320.

(2) Construction does not include:

(i) Changes for temporary use of the land for public recreational purposes;

(ii) Site exploration, including necessary borings to determine foundation conditions or other preconstruction monitoring to establish background information

related to the suitability of the site, the environmental impacts of construction or operation, or the protection of environmental values;

(iii) Preparation of a site for construction of a facility, including clearing of the site, grading, installation of drainage, erosion and other environmental mitigation measures, and construction of temporary roads and borrow areas;

(iv) Erection of fences and other access control measures;

(v) Excavation;

(vi) Erection of support buildings (such as construction equipment storage sheds, warehouse and shop facilities, utilities, concrete mixing plants, docking and unloading facilities, and office buildings) for use in connection with the construction of the facility;

(vii) Building of service facilities (such as paved roads, parking lots, railroad spurs, exterior utility and lighting systems, potable water systems, sanitary sewage treatment facilities, and transmission lines);

(viii) Procurement or fabrication of components or portions of the proposed facility occurring at locations other than the final, in-place location at the facility; or

(ix) Manufacture of a manufactured reactor under a manufacturing license to be installed at the proposed site and to be part of the proposed facility.

Design bases means that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be: (1) restraints derived from generally accepted "state of the art" practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must demonstrate compliance with its functional goals.

Functional containment means a barrier, or a set of barriers taken together, that effectively limits the physical transport of radioactive material to the environment.

Reactor coolant pressure boundary means, for a light-water reactor, all those pressure-containing components, such as pressure vessels, piping, pumps, and valves, which are:

(1) Part of the reactor coolant system; or

(2) Connected to the reactor coolant system, up to and including any and all of the following:

(i) The outermost containment isolation valve in system piping which penetrates primary reactor containment;

(ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment; and

(iii) The reactor coolant system safety and relief valves.

For nuclear power reactors of the direct cycle boiling water type, the reactor coolant system extends to and includes the outermost containment isolation valve in the main steam and feedwater piping.

Safety-related (SR) SSCs for light-water commercial nuclear reactors means those structures, systems, and components that are relied upon to remain functional during and following design-basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 53.4730(a)(1)(vi).

For non-light-water commercial nuclear reactors, *safety-related structures, systems, and components* means those structures, systems, and components that are relied on to remain functional during and following design-basis events to assure:

(1) The capability to perform safety functions determined in accordance with § 53.4730(a)(5)(ii) and (36), including cooling to maintain the integrity of systems and barriers credited in the safety analyses such that these SSCs function as credited;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 53.4730(a)(1)(vi).

Severe nuclear accident means those events that progress beyond the design-basis accidents in which substantial damage is done to the reactor core or to any other structure, vessel, or retention system that contains a significant inventory of radiological material, whether or not there are serious offsite consequences.

§ 53.030 Reserved.

§ 53.040 Written communications.

(a) *General requirements.* All correspondence, reports, applications, and other written communications from the applicant or licensee to the NRC concerning the regulations in this part or individual license conditions 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 8:15 a.m. and 4 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/e-submittals.html;* by e-mail to *MSHD.Resource@nrc.gov;* or by writing the Office of the Chief Information Officer, 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 below (addresses for the NRC Regional Offices are listed in appendix D to 10 CFR part 20).

(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 licenses, permits, and design approvals issued pursuant to this part, must be submitted as follows, except as otherwise specified in paragraphs (b)(2) through (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 this part.

(2) *Applications for permits and licenses, and amendments to applications.* Applications for licenses, permits, and design approvals 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 facility) or the place of manufacture of a reactor licensed under this part, except as otherwise specified in paragraphs (b)(3) through (9) 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 (v) 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. Submissions should include the following as appropriate:

(i) Physical security plan;

(ii) Safeguards contingency plan;

(iii) Cybersecurity plan;

(iv) Change to security plan, guard training and qualification plan, safeguards contingency plan, or cybersecurity plan made without prior Commission approval under §§ 53.1565 or 53.6065; and

(v) Application for amendment of physical security plan, guard training and qualification plan, safeguards contingency plan, or cybersecurity plan under §§ 53.1510 or 53.6010.

(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. Submissions should include the following as appropriate:

(i) Emergency plan;

(ii) Change to an emergency plan under §§ 53.1565 or 53.6065; and

(iii) Emergency implementing procedures under §§ 53.855 or 53.4320.

(6) *Updated Final Safety Analysis Report.* An Updated Final Safety Analysis Report (UFSAR) or replacement pages under §§ 53.1545 or 53.6045 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. 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 §§ 53.1565 or 53.6065, or a change to a licensee's NRC-accepted quality assurance topical report under §§ 53.1565 or 53.6065, 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 non-licensees (i.e., architect/engineers, nuclear steam supply system (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 subpart G or subpart Q of this part, 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 subpart G or subpart Q of this part, 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 demonstrate compliance with 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.* Licensees, applicants, 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.

§ 53.050 Deliberate misconduct.

(a) Any licensee, applicant for a license, employee of a licensee or applicant; or any contractor (including a supplier or consultant), subcontractor, employee of a contractor or subcontractor of any licensee or applicant for a license, who knowingly provides to any licensee, applicant, contractor, or subcontractor, any components, equipment, materials, or other goods or services that relate to a licensee's or applicant's activities in this part, may not:

(1) Engage in deliberate misconduct that causes or would have caused, if not detected, a licensee 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; or

(2) Deliberately submit to the NRC, a licensee, an applicant, or a licensee'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.

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

(c) For the purposes of paragraph (a)(1) of this section, deliberate misconduct by a person means an intentional act or omission that the person knows:

(1) Would cause a licensee 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; or

(2) Constitutes a violation of a requirement, procedure, instruction, contract, purchase order, or policy of a licensee, applicant, contractor, or subcontractor.

§ 53.060 Employee protection.

(a) Discrimination by a holder or applicant for an NRC license or design approval, or a contractor or subcontractor of a holder or applicant for an NRC license or 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 ERA and in general are related to the administration or enforcement of a requirement imposed under the AEA or the ERA.

(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 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 paragraph (a) of this section or under these requirements if the employee has identified the alleged illegality to the employer;

(iii) Requesting the NRC 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 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 ERA or the AEA.

(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, 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 holder or applicant for an NRC license or design approval, or a contractor or subcontractor of a holder or applicant for an NRC license or design approval, 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 holder or applicant for an NRC license or design approval, or a contractor or subcontractor of a holder or applicant for a Commission license, permit, or design approval; or

(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 holder or applicant for a license or design approval, must prominently post the revision of NRC Form 3, "Notice to Employees," referenced in 10 CFR 19.11(e)(1). 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 no later than 30 days after an application is docketed and remain posted while the application is pending before the Commission, during the term of the license, and for 30 days following license termination.

(2) Copies of NRC Form 3 may be obtained by writing to the Regional Administrator of the appropriate NRC Regional Office listed in appendix D to 10 CFR part 20, via email to *Forms.Resource@nrc.gov*, or by visiting the NRC's online library at *http://www.nrc.gov/reading-rm/doc-collections/forms/*.

(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 pursuant to Section 211 of the ERA, 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) 10 CFR part 19 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.

§ 53.070 Completeness and accuracy of information.

(a) Information provided to the Commission by a holder of a license, permit, design certification, or standard design approval under this part or an applicant for a license, permit, design certification, or 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 applicant or the licensee must be complete and accurate in all material respects.

(b) Each applicant or licensee must notify the Commission of information identified by the applicant or licensee as having for the regulated activity a significant implication for public health and safety or common defense and security. An applicant or licensee violates this paragraph only if the applicant or licensee fails to notify the Commission of information that the applicant or licensee has identified as having a significant implication for public health and safety or common defense and security. Notification must 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.

§ 53.080 Specific exemptions.

(a) 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, which are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security.

(b) The Commission will not consider granting an exemption unless special circumstances are present. Special circumstances are present whenever:

(i) Application of the regulation in the particular circumstances conflicts with other rules or requirements of the Commission;

(ii) Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule;

(iii) Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated;

(iv) The exemption would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption;

(v) The exemption would provide only temporary relief from the applicable regulation and the licensee or applicant has made good faith efforts to comply with the regulation; or

(vi) There is present any other material circumstance not considered when the regulation was adopted for which it would be in the public interest to grant an exemption.

If such condition is relied on exclusively for demonstrating compliance with paragraph (b) of this section, the exemption may not be granted until the Executive Director for Operations has consulted with the Commission.

(c) Any person may request an exemption permitting the conduct of construction activities prior to the issuance of a construction permit. The Commission may grant such an exemption upon considering and balancing the following factors:

(1) Whether conduct of the proposed activities will give rise to a significant adverse impact on the environment and the nature and extent of such impact, if any;

(2) Whether redress of any adverse environment impact from conduct of the proposed activities can reasonably be effective should such redress be necessary;

(3) Whether conduct of the proposed activities would foreclose subsequent adoption of alternatives; and

(4) The effect of delay in conducting such activities on the public interest, including whether the power needs to be used by the proposed facility, the availability of alternative sources, if any, to meet those needs on a timely basis and delay costs to the applicant and to consumers.

Issuance of such an exemption must not be deemed to constitute a commitment to issue a construction permit. During the period of any exemption granted pursuant to paragraph (c) of this section, any activities conducted must be carried out in such a manner as will minimize or reduce their environmental impact.

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, must be governed by the exemption requirements of those parts.

§ 53.090 Standards for review.

(a) *Common standards*. In determining that a construction permit, operating license, early site permit, combined license, or manufacturing license in this part will be issued to an applicant, the Commission will be guided by the following considerations:

(1) Except for an early site permit or manufacturing license, the processes to be performed, the operating procedures, the facility and equipment, the use of the facility, and other technical specifications, or the proposals, in regard to any of the foregoing, collectively provide reasonable assurance that the applicant will comply with the regulations in this chapter, including the regulations in 10 CFR part 20, and that the health and safety of the public will not be endangered.

(2) The applicant for a construction permit, operating license, combined license, or manufacturing license is technically and financially qualified to engage in the proposed activities in accordance with the regulations in this chapter. However, no consideration of financial qualification is necessary for an electric utility applicant for an operating license for a utilization facility of the type described in paragraph (d) of this section or for an applicant for a manufacturing license.

(3) The issuance of a construction permit, operating license, early site permit, combined license, or manufacturing license to the applicant will not, in the opinion of the Commission, be inimical to the common defense and security or to the health and safety of the public.

(4) Any applicable requirements of subpart A of 10 CFR part 51 have been satisfied.

(b) *Additional standards for licenses*. In determining whether a license will be issued to an applicant, the Commission will, in addition to applying the standards set forth in paragraph (a) of this section, consider whether the proposed activities will serve

a useful purpose proportionate to the quantities of special nuclear material or source material to be utilized.

(c) *Additional standards and provisions affecting licenses and certifications for commercial power*. In addition to applying the standards set forth in paragraphs (a) and (b) of this section, paragraphs (c)(1) through (c)(5) of this section apply in the case of a license for a facility for the generation of commercial power. For a design certification under this part, only paragraph (c)(5) of this section applies.

(1) The NRC will:

(i) Give notice in writing of each application to the regulatory agency or State as may have jurisdiction over the rates and services incident to the proposed activity;

(ii) Publish notice of the application in trade or news publications as it deems appropriate to give reasonable notice to municipalities, private utilities, public bodies, and cooperatives which might have a potential interest in the utilization or production facility; and

(iii) Publish notice of the application once each week for four consecutive weeks in the *Federal Register*. No license will be issued by the NRC prior to the giving of these notices and until four weeks after the last notice is published in the *Federal Register*.

(2) If there are conflicting applications for a limited opportunity for such license, the Commission will give preferred consideration in the following order: first, to applications submitted by public or cooperative bodies for facilities to be located in high cost power areas in the United States; second, to applications submitted by others for facilities to be located in such areas; third, to applications submitted by public or cooperative bodies for facilities to be located in areas other than high cost power areas; and, fourth, to all other applicants.

(3) The licensee who transmits electric energy in interstate commerce, or sells it at wholesale in interstate commerce, must be subject to the regulatory provisions of the Federal Power Act.

(4) Nothing shall preclude any government agency, now or hereafter authorized by law to engage in the production, marketing, or distribution of electric energy, if otherwise qualified, from obtaining a construction permit, operating license, or combined license under this part for a utilization facility for the primary purpose of producing electric energy for disposition for ultimate public consumption.

(5) Applications for a design certification, combined license, manufacturing license, operating license, or standard design approval that propose nuclear reactor designs which differ 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, will be approved only if:

(i)(A) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;

(B) Interdependent effects among the safety features of the design are acceptable, as demonstrated by analysis, appropriate test programs, experience, or a combination thereof; and

(C) Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions; or

(ii) There has been acceptable testing of a prototype plant over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences,

including equilibrium core conditions. If a prototype plant is used to comply with the testing requirements, then the NRC may impose additional requirements on siting, safety features, or operational conditions for the prototype plant to protect the public and the plant staff from the possible consequences of accidents during the testing period.

(d) *Licenses for commercial nuclear plants*. A license will be issued, to an applicant who qualifies, for any one or more of the following: to transfer or receive in interstate commerce, or manufacture, produce, transfer, acquire, possess, or use a production or utilization facility for industrial or commercial purposes. *Provided, however*, that in the case of a utilization facility which is useful in the conduct of research and development activities of the types specified in Section 31 of the AEA, such facility is deemed to be for industrial or commercial purposes if the facility is to be used so that more than 50 percent of the annual cost of owning and operating the facility is devoted to the production of materials, products, or energy for sale or commercial distribution, or to the sale of services, other than research and development or education or training.

§ 53.100 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.

§ 53.110 Attacks and destructive acts.

Licensees, applicants for licenses, permits, certifications, and design approvals, and applicants for an amendment to any license, permit, certification, or design approval under this part are not 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.

§ 53.115 Rights related to special nuclear material.

(a) No right to the special nuclear material must be conferred by a license issued under this part except as may be defined by the license.

(b) Neither a license issued under this part, nor any right thereunder, nor any right to utilize or produce special nuclear material shall be transferred, assigned, or disposed of in any manner, either voluntarily or involuntarily, directly or indirectly, through transfer of control of the license to any person, unless the Commission shall, after securing full information, find that the transfer is in accordance with the provisions of the AEA (68 Stat. 919) and give its consent in writing.

§ 53.117 License suspension and rights of recapture.

Any license issued under this part must be subject to suspension and to the rights of recapture of the material or control of the facility reserved to the Commission under Section 108 of the AEA in a state of war or national emergency declared by Congress.

§ 53.120 Information collection requirements: OMB approval.

(a) The NRC 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 [TBD].

(b) The approved information collection requirements contained in this part appear in § 53.XX.

(c) This part contains information collection requirements in addition to those approved under the control number specified in paragraph (a) of this section. These information collection requirements and the control numbers under which they are approved are as follows:

[To be added.]

Subpart B — Technology-Inclusive Safety Requirements

§ 53.200 Safety objectives.

Each commercial nuclear plant must be designed, constructed, operated, and decommissioned to limit the possibility of an immediate threat to the public health and safety. In addition, additional measures must be taken for each commercial nuclear plant as may be appropriate when considering potential risks to public health and safety. These safety objectives must be carried out by meeting the safety criteria and other requirements identified in this subpart.

§ 53.210 Safety criteria for design-basis accidents.

Design features and programmatic controls must be provided for each commercial nuclear plant such that identification and analyses of design-basis accidents in accordance with § 53.240 demonstrate the following:

(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 (250 mSv) total effective dose equivalent; 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 (250 mSv) total effective dose equivalent.[1]

[1] The use of 25 rem total effective dose equivalent 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.

§ 53.220 Safety criteria for licensing-basis events other than design-basis accidents.

Design features and programmatic controls must be provided for each commercial nuclear plant such that identification and analyses of LBEs other than design-basis accidents in accordance with § 53.240 demonstrate the following:

(a) Ensure plant structures, systems and components (SSCs), personnel, and programs provide the necessary capabilities and maintain the necessary reliability to address LBEs other than design-basis accidents in accordance with §§ 53.240 and 53.450(e), and provide measures for defense-in-depth in accordance with § 53.250; and

(b) Maintain overall cumulative plant risk from LBEs other than design-basis accidents analyzed in accordance with § 53.450(e) such that the calculated risk to an average individual in the vicinity of the commercial nuclear plant of prompt fatalities remains below five in 10 million years, and the calculated risk to the population in the area near a commercial nuclear plant of cancer fatalities remains below two in one million years.

§ 53.230 Safety functions.

(a) The primary safety function is limiting the release of radioactive materials from the facility and must be maintained during normal operation and for LBEs over the life of the plant.

(b) Additional safety functions needed to support the retention of radioactive materials during LBEs—such as controlling reactivity, heat generation, heat removal, and chemical interactions— must be identified for each commercial nuclear plant.

(c) The primary and additional safety functions are required to satisfy the safety criteria defined in §§ 53.210 and 53.220, or more restrictive alternative criteria adopted under § 53.470, and must be fulfilled by the design features, human actions, and programmatic controls specified throughout Framework A of this part.

§ 53.240 Licensing-basis events.

(a) LBEs must be identified for each commercial nuclear plant and analyzed in accordance with § 53.450 to demonstrate that the safety requirements in this subpart have been satisfied.

(b) LBEs ranging from anticipated event sequences to very unlikely event sequences must collectively address combinations of malfunctions of plant SSCs, human errors, facility hazards, and the effects of external hazards.

(c) The analysis of LBEs must:

(1) include analysis of one or more design-basis accidents in accordance with § 53.450(f);

(2) confirm the adequacy of design features and programmatic controls needed to satisfy the safety criteria defined in §§ 53.210 and 53.220, or more restrictive alternative criteria adopted under § 53.470, and

(3) establish related functional requirements for plant SSCs, personnel, and programs.

§ 53.250 Defense in depth.

(a) Measures must be taken for each commercial nuclear plant to ensure appropriate defense in depth is provided to compensate for uncertainties in the analysis of the safety criteria such that there is reasonable assurance that the safety criteria in this subpart are met over the life of the plant.

(b) The uncertainties that must be addressed under paragraph (a) of this section include those related to the state of knowledge and modeling capabilities, the ability of barriers to limit the release of radioactive materials from the facility during LBEs other than design-basis accidents, the reliability and performance of plant SSCs and personnel, and the effectiveness of programmatic controls.

(c) The safety analysis may not rely upon a single engineered design feature, human action, or programmatic control, no matter how robust, to address the range of LBEs other than design-basis accidents.

§ 53.260 Normal operations.

(a) Licensees under Framework A of this part must ensure that normal plant operations do not result in public doses or dose rates in unrestricted areas that exceed the limits provided in Subpart D to 10 CFR part 20.

(b) A combination of design features and programmatic controls must be established such that the estimated total effective dose equivalent to individual members of the public resulting from normal plant operation is as low as is reasonably achievable in accordance with 10 CFR part 20.

§ 53.270 Protection of plant workers.

(a) Licensees under Framework A of this part must ensure that radiological dose to plant workers does not exceed the occupational dose limits provided in subpart C to 10 CFR part 20.

(b) A combination of design features and programmatic controls must, to the extent practical, be based upon sound radiation protection principles to achieve occupational doses that are as low as is reasonably achievable in accordance with 10 CFR part 20.

Subpart C — Design and Analysis Requirements

§ 53.400 Design features for licensing-basis events.

(a) Design features must be provided for each commercial nuclear plant such that, when combined with corresponding human actions and programmatic controls, the plant will satisfy the safety criteria defined in §§ 53.210 and 53.220, or more restrictive alternative criteria adopted under § 53.470.

(b) Design features must ensure that the safety functions identified in § 53.230 are fulfilled during LBEs.

§ 53.410 Functional design criteria for design-basis accidents.

(a) Functional design criteria must be defined for each design feature required by § 53.400 and relied upon to demonstrate compliance with the safety criteria defined in § 53.210.

(b) Corresponding human actions and programmatic controls must be identified and implemented in accordance with this and other subparts to achieve and maintain the reliability and capability of SSCs relied upon to satisfy the defined functional design criteria and the safety criteria required in § 53.210, and to maintain consistency with analyses required by § 53.450(f).

§ **53.415 Protection against external hazards.**

Safety-related SSCs must be protected against or must be designed to withstand the effects of natural phenomena (e.g., earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches) and human-related hazards (e.g., dams, transportation routes, military and industrial facilities) considering an event severity up to the design basis external hazard levels as determined under § 53.510 without losing the capability to perform the safety functions stated in § 53.230. Specific requirements for earthquake engineering are included in § 53.480.

§ 53.420 Functional design criteria for licensing-basis events other than design-basis accidents.

(a) Functional design criteria must be defined for each design feature required by § 53.400 and relied upon to:

(1) demonstrate compliance with the safety criteria in § 53.220 or more restrictive alternative criteria adopted under § 53.470; and

(2) demonstrate compliance with the evaluation criteria in § 53.450(e) or more restrictive alternative criteria adopted under § 53.470.

(b) Corresponding human actions and programmatic controls must be identified and implemented in accordance with this and other subparts to achieve and maintain the reliability and capability of SSCs relied upon to:

(1) satisfy the safety criteria in § 53.220 or more restrictive alternative criteria adopted under § 53.470; and

(2) satisfy the evaluation criteria in § 53.450(e) or more restrictive alternate criteria adopted under § 53.470.

§ 53.425 Design features and functional design criteria for normal operations.

(a) Design features must be provided for each commercial nuclear plant such that, when combined with corresponding programmatic controls, the requirements in § 53.260(a) can be met.

(b) Functional design criteria must be defined for each design feature relied upon to demonstrate compliance with § 53.260(a).

(c) Corresponding programmatic controls, including monitoring programs, must be identified and implemented to confirm that the criteria in § 53.260(a) are not exceeded.

(d) Functional design criteria must be defined for design features to ensure that plant SSCs and corresponding programmatic controls, including monitoring programs,

ensure that liquid, gaseous, and solid wastes are controlled such that the public doses are kept as low as reasonably achievable in accordance with § 53.260(b). A guide for meeting as low as is reasonably achievable in § 53.260(b) is that the estimated annual dose to the maximally exposed member of the public does not exceed 10 mrem total effective dose equivalent. The design objective of maintaining doses below 10 mrem/year should not be construed as a radiation protection standard.

§ 53.430 Design features and functional design criteria for protection of plant workers.

(a) Design features must be provided for each commercial nuclear plant such that, when combined with corresponding programmatic controls, the requirements in § 53.270(a) can be met.

(b) Functional design criteria must be defined for each design feature relied upon to demonstrate compliance with § 53.270(a).

(c) Corresponding programmatic controls, including monitoring programs, must be identified and implemented to confirm that the worker protection criteria in § 53.270(a) are not exceeded.

(d) Functional design criteria must be defined for design features to ensure that plant SSCs and corresponding programmatic controls, including monitoring programs, satisfy § 53.270(b).

§ 53.440 Design requirements.

(a) Analysis, appropriate test programs, prototype testing, operating experience, or a combination thereof must demonstrate that each design feature required by § 53.400 meets the defined functional design criteria required by §§ 53.410 and 53.420.

This demonstration must consider interdependent effects throughout the commercial nuclear plant and the range of conditions under which the design features required by § 53.400 must function throughout the plant's lifetime.

(b) The design features required by § 53.400 must, wherever applicable, be designed using generally accepted consensus codes and standards that have been endorsed or otherwise found acceptable by the NRC.

(c) The materials used for safety-related (SR) and NSRSS SSCs must be qualified for their service conditions over the plant lifetime.

(d) Possible degradation mechanisms related to aging, fatigue, chemical interactions, operating temperatures, effects of irradiation, and other environmental factors that may affect the performance of safety-related and NSRSS SSCs must be evaluated and used to inform the design and the development of integrity assessment programs under § 53.870.

(e)(1) SR and NSRSS SSCs must be designed and located to minimize, consistent with other safety requirements in this part, the probability and effect of fires and explosions.

(2) Noncombustible and fire-resistant materials must be used wherever practical throughout the facility, particularly in locations with SR and NSRSS SSCs.

(3) Fire detection and fire suppression systems of appropriate capacity and capability must be provided and designed to minimize the adverse effects of fires on SR and NSRSS SSCs.

(4) Fire suppression systems must be designed to ensure that their rupture or inadvertent operation does not significantly impair the ability of SR and NSRSS SSCs to perform their safety functions to satisfy § 53.230.

(f) Safety and security must be considered together in the design process such that, where possible, security issues are effectively resolved through design and engineered security features.

(g) The reactor system and waste stores for each commercial nuclear plant must be capable of achieving and maintaining a subcritical condition during normal operations and following any LBE identified in accordance with § 53.240.

(h) Each commercial nuclear plant must have a capability to provide long-term cooling of the reactor fuel and waste stores following normal operations or any LBE identified in accordance with § 53.240.

(i) The design, analysis, staffing, and programmatic controls for each commercial nuclear plant must consider the number of reactors, waste stores, and other significant inventories of radioactive materials and the associated operating configurations, common systems, system interfaces, and system interactions.

(j)(1) Design features must be provided and related functional design criteria defined such that, with limited use of operator actions, one or more physical barriers are maintained to limit the release of radionuclides from reactor systems, waste stores, or other significant inventories of radioactive materials assuming the impact of a large, commercial aircraft.

(2) The functional design criteria for those design features provided to address the requirements in paragraph (j)(1) of this section must be based on an assessment of the impact of a large, commercial aircraft used for long distance flights in the United States, with aviation fuel loading typically used in such flights, and an impact speed and angle of impact considering the ability of both experienced and inexperienced pilots to control large, commercial aircraft at low altitude representative of a nuclear power plant's low profile.[1]

(k) Design features and related functional design criteria must be defined such that analyses demonstrate a low risk of permanent injury to the public due to the health effects of the chemical hazards of licensed material.

(l) Measures must be taken during the design of commercial nuclear plants to minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste in accordance with § 20.1406 of this chapter.

(m)(1) Each commercial nuclear plant must include criticality monitoring capabilities meeting the requirements of either 10 CFR 70.24 of this chapter or paragraph (m)(2) of this section.

(2) In lieu of maintaining a monitoring system capable of detecting criticality as described in 10 CFR 70.24, criticality accident requirements may be satisfied by:

(i) Demonstrating the sub-criticality of special nuclear material, except when it is inside the reactor and the reactor is being operated, by maintaining k-effective below 0.95 at a 95 percent probability, 95 percent confidence level, under conditions that maximize reactivity for the applicable storage and handling configurations, and

(ii) Providing radiation monitors for fuel storage and associated handling areas when fuel is present to detect excessive radiation levels and to support initiating appropriate safety actions.

(3) While a spent fuel transportation package approved under Part 71 of this chapter or spent fuel storage cask approved under Part 72 of this chapter is in the special nuclear material handing or storage area, the requirements in Part 71 or 72 of this chapter, as applicable, and the requirements of the Certificate of Compliance for that package or cask, are the applicable requirements for the fuel within that package or cask.

(n)(1) The design of each commercial nuclear plant must reflect state-of-the-art human factors principles for safe and reliable performance in all settings that human activities are expected for performing or supporting the continued availability of plant safety or emergency response functions.

(2) The design must provide for the capabilities described in § 53.730(b) to ensure the plant staff are able to monitor plant conditions and respond to events.

(3) The means by which the design and human actions together will achieve the safety requirements of Subpart B must be evaluated and used to inform the design and the development of the concept of operations required by § 53.730(c).

(4) A functional requirements analysis and function allocation must be used to ensure that plant design features address how safety functions and functional safety criteria are satisfied, and how the safety functions will be assigned to appropriate combinations of human action, automation, active safety features, passive safety features, or inherent safety characteristics.

[1] Changes to the detailed parameters on aircraft impact characteristics set forth in guidance shall be approved by the Commission.

§ 53.450 Analysis requirements.

(a) *Requirement to have a probabilistic risk assessment.* A probabilistic risk assessment (PRA) of each commercial nuclear plant must be performed to identify potential failures, susceptibility to internal and external hazards, and other contributing factors to event sequences that might challenge the safety functions identified in § 53.230 and to support demonstrating that each commercial nuclear plant meets the safety criteria of § 53.220, or more restrictive alternative criteria adopted under § 53.470.

(b) *Specific uses of analyses*. The PRA in combination with other generally accepted approaches for systematically evaluating engineered systems must be used:

(1) In informing the selection of the LBEs, as described in § 53.240, which must be considered in the design to determine compliance with the safety criteria in Subpart B of this part.

(2) For informing the classification of SSCs according to their safety significance in accordance with § 53.460 and for identifying the environmental conditions under which the SSCs and operating staff must perform their safety functions.

(3) In evaluating the adequacy of defense-in-depth measures required in accordance with § 53.250.

(4) To identify and assess all plant operating states where there is the potential for the uncontrolled release of radioactive material to the environment.

(5) To identify and assess events that challenge plant control and safety systems whose failure could lead to the uncontrolled release of radioactive material to the environment. These include internal events, such as human errors and equipment failures, and external events identified in accordance with Subpart D of this part.

(c) *Maintenance and upgrade of analyses*. The PRA must be maintained and upgraded in conformance with generally accepted methods, standards, and practices that have been endorsed or otherwise found acceptable by the NRC.

(d) *Qualification of analytical codes*. The analytical codes used in modeling plant behavior in analyses of LBEs (including but not limited to thermodynamics, reactor physics, fuel performance, and mechanistic source term codes) must be qualified for the range of conditions for which they are to be used.

(e) *Analyses of licensing-basis events other than design-basis accidents*. (1) Analyses must be performed for LBEs other than design-basis accidents. These LBEs

must be identified using insights from a PRA in combination with other generally accepted approaches for systematically evaluating engineered systems to identify and analyze equipment failures and human errors.

(2) The analysis of LBEs other than design-basis accidents must include definition of evaluation criteria for each event or specific categories of LBEs to determine the acceptability of the plant response to the challenges posed by internal and external hazards to provide an appropriate level of safety.

(3) The analyses of LBEs other than design-basis accidents must address event sequences from initiation to a defined end state and be used in combination with other engineering analyses to demonstrate that the functional design criteria required by § 53.420 provide sufficient barriers to the unplanned release of radionuclides to satisfy the evaluation criteria defined for each LBE other than design-basis accidents, to satisfy the safety criteria of § 53.220, and provide defense in depth as required by § 53.250.

(4) The methodology used to identify, categorize, and analyze LBEs must include a means to identify event sequences deemed significant for controlling the risks posed to public health and safety.

(f) *Analysis of design-basis accidents*. (1) The analysis of LBEs required by § 53.240 must include analysis of design-basis accidents that address possible challenges to the safety functions identified in accordance with § 53.230. The events selected as design-basis accidents must be those that, if not terminated, have the potential for exceeding the safety criteria in § 53.210.

(2) The design-basis accidents selected must be analyzed using deterministic methods that address event sequences from initiation to a safe stable end state and assume only the safety-related SSCs identified in accordance with § 53.460 and human

actions addressed by the requirements of Subpart F are available to perform the safety functions identified in accordance with § 53.230.

(3) The analysis must conservatively demonstrate compliance with the safety criteria in § 53.210.

(g) *Other required analyses*. Analyses must be performed to assess:

(1) *Fire protection.* fire protection measures to demonstrate, through inclusion of fires in the analysis of LBEs or by separate analyses, that a fire or explosion in any plant area would not:

(i) prevent equipment from fulfilling the safety functions identified in accordance with § 53.230, or

(ii) challenge the safety criteria in §§ 53.210 and 53.220

(2) *Aircraft impact.* measures provided to protect against aircraft impacts as required by § 53.440(j)

(3) *Effluents.* measures taken to satisfy the requirements of § 53.425 by estimating:

(i) The quantity of each of the principal radionuclides expected to be released annually to unrestricted areas in liquid effluents produced during normal reactor operations and the dose to the maximally exposed member of the public in unrestricted areas.

(ii) The quantities of each of the principal radionuclides of the gases, halides, and particulates expected to be released annually to unrestricted areas in gaseous effluents produced during normal reactor operations and the dose to the maximally exposed member of the public in unrestricted areas.

(iii) The annual external radiation dose to unrestricted areas and the maximally exposed member of the public in unrestricted areas due to contained radiation sources from the planned licensed operations during normal reactor operations.

§ **53.460 Safety categorization and special treatment.**

(a) SSCs must be classified according to their safety significance. The categories must include “Safety-Related” (SR), “Non-Safety-Related but Safety-Significant” (NSRSS), and “Non-Safety-Significant” (NSS), as defined in subpart A of this part.

(b) For SR and NSRSS SSCs, the conditions under which they must perform their safety function in § 53.230 must be identified. Special treatment must be established in accordance with this and other Subparts to provide confidence that the SSCs will perform under the service conditions and with the reliability consistent with the analysis performed in accordance with § 53.450 to demonstrate meeting the safety criteria in §§ 53.210 and 53.220, or more restrictive alternative criteria adopted under § 53.470.

(1) The special treatments for SR SSCs must include meeting the applicable quality assurance requirements from Subpart K of this part.

(2) The special treatments for NSRSS SSCs may include meeting selected quality assurance requirements from Subpart K of this part when such treatment is needed to address performance requirements, equipment reliability, or uncertainties.

(c) Human actions needed to prevent or mitigate LBEs must be identified, be able to be performed reliably under the postulated environmental conditions, and be addressed by programs established in accordance with Subpart F of this part to provide confidence that those actions will be performed as assumed in the analysis performed in

accordance with § 53.450 to demonstrate meeting the criteria in §§ 53.210, 53.220, and 53.450(e), or more restrictive alternative criteria adopted under § 53.470.

§ 53.470 Maintaining analytical safety margins used to justify operational flexibilities.

Where an applicant or licensee so chooses, alternative criteria more restrictive than those defined in §§ 53.220 and 53.450(e) may be adopted to support operational flexibilities. In such cases, applicants and licensees must ensure that the functional design criteria of § 53.420, the analysis requirements of § 53.450(e), and identification of special treatment of SSCs and human actions under § 53.460 reflect and support the use of alternative criteria to justify operational flexibilities. Licensees must ensure that measures taken to provide the analytical margins supporting operational flexibilities are incorporated into design features and programmatic controls and are maintained within programs required in other Subparts.

§ 53.480 Earthquake engineering.

(a) SSCs classified as safety related or NSRSS must be able to withstand the effects of earthquakes, commensurate with safety significance, without loss of capability to perform their role in fulfilling the safety functions required by § 53.230.

(b) For the purpose of this section:

Design Basis Ground Motions (DBGMs) are the vibratory ground motions for which certain SSCs must be designed to remain functional.

Operating basis earthquake (OBE) ground motion is the vibratory ground motion for which those features of the commercial nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to

remain functional. The OBE ground motion is used in § 53.720, “Response to seismic events.”

Response spectrum is a plot of the maximum responses (acceleration, velocity, or displacement) of idealized single-degree-of-freedom oscillators as a function of the natural frequencies of the oscillators for a given damping value. The response spectrum is calculated for a specified vibratory motion input at the oscillators' supports.

Surface deformation is the distortion of geologic strata on or near the ground surface that occurs because of tectonic forces that result from earthquakes.

(c) *Vibratory Ground Motion*

(1) *Design Basis Ground Motions.*

(i) The DBGMs must be derived from the Site Ground Motion Response Spectra (GMRS) developed in accordance with § 53.510(c), by taking into consideration the functional design criteria of SSCs in accordance with §§ 53.410 and 53.420. The horizontal component of the DBGM(s) in the free-field at the foundation level of the structures must be an appropriate response spectrum that is determined based on the risk-significance of SSCs and their safety functions. In view of the limited data available on vibratory ground motion of strong earthquakes, it is acceptable that the design response spectra be smoothed spectra.

(ii) The commercial nuclear power plant must be designed so that, if the DBGMs occur, the following SSCs must remain functional and within applicable stress, strain, and deformation limits:

(A) SSCs for which functional design criteria are established in accordance with §§ 53.410 or 53.420; and

(B) SSCs classified as safety related or NSRSS commensurate with safety significance in accordance with § 53.460.

(iii) In addition to seismic loads, applicable concurrent normal operating, functional, and accident-induced loads must be taken into account in the design of the safety-related SSCs and, commensurate with safety significance, NSRSS SSCs.

(iv) The design of the commercial nuclear power plant must take into account the possible effects of the DBGM on the facility foundations by ground disruption, such as fissuring, lateral spreads, differential settlement, liquefaction, and landsliding.

(v) The SSCs fulfilling the safety functions required by § 53.230 must be demonstrated through design, testing, or qualification methods to be able to fulfill those safety functions during and after the vibratory ground motion associated with the DBGMs.

(vi) The evaluation of SSCs required by this section to show they are able to function during and following earthquake ground motion must take into account soil-structure interaction effects and the expected duration of vibratory motion. It is permissible to design for strain limits in excess of yield strain in some of these safety-related SSCs during the DBGMs and under the postulated concurrent loads, provided the necessary safety functions are maintained.

(2) *Operating Basis Earthquake Ground Motion.*

(i) The OBE Ground Motion must be characterized by response spectra. The value of the OBE Ground Motion must be set to one-third or less of the DBGMs response spectra.

(3) Reserved

(4) *Required Seismic Instrumentation.* Suitable instrumentation must be provided so that the seismic response of commercial nuclear power plant SR or NSRSS SSCs can be evaluated promptly after an earthquake.

(d) *Surface Deformation.*

(1) The potential for surface deformation must be taken into account in the design of the commercial nuclear power plant by providing reasonable assurance that in the event of deformation, the following SSCs will remain functional:

(i) SSCs for which functional design criteria are established in accordance with §§ 53.410 and 53.420; and

(ii) SSCs classified as safety related or NSRSS in accordance with § 53.460.

(2) In addition to surface deformation induced loads, the design of SSCs must take into account, commensurate with safety significance, seismic loads and applicable concurrent functional and accident-induced loads.

(3) The design provisions for surface deformation must be based on its postulated occurrence in any direction and azimuth and under any part of the commercial nuclear power plant, unless evidence indicates this assumption is not appropriate, and must take into account the estimated rate at which the surface deformation may occur.

(e) *Seismically Induced Floods and Water Waves and Other Design Conditions.* Seismically induced floods and water waves from either locally or distantly generated seismic activity and other design conditions determined pursuant to Subpart D must be taken into account in the design of the commercial nuclear power plant so as to prevent undue risk to the health and safety of the public.

(f) *Analysis.* The analyses required by § 53.450 must address seismic hazards and related SSC responses in determining that the safety criteria defined in § 53.220 will be met.

(g) Functional design criteria, human actions, and programmatic controls needed to address seismic events must be identified and implemented in accordance with this and other subparts to achieve and maintain the performance of SSCs relied upon to

satisfy the safety criteria in § 53.220 and to maintain consistency with analyses required by § 53.450 when accounting for the site-specific frequencies and magnitudes of earthquakes for a commercial nuclear plant.

Subpart D—Siting Requirements

§ 53.500 General siting and siting assessment.

(a) The siting of each commercial nuclear plant must be supported by assessments of proposed sites such that the design, including design features and programmatic controls corresponding to the site characteristics, satisfies the safety criteria defined in §§ 53.210 and 53.220 or more restrictive alternative criteria adopted under § 53.470. The siting assessment must ensure that site characteristics that might contribute to the initiation, progression, or consequences of LBEs analyzed in accordance with §§ 53.450 and 53.480 are identified and mitigated by design features or programmatic controls. The siting assessment must take into consideration the potential adverse impacts that a commercial nuclear plant may have on nearby populations as a result of normal operations or LBEs.

(b) Activities performed to identify site characteristics or otherwise needed to determine site-specific contributors to functional design criteria or analysis assumptions under Subpart C must satisfy the applicable special treatment requirements of § 53.460, including, where applicable, the quality assurance requirements from Subpart K of this part.

§ 53.510 External hazards.

(a) *General external hazard requirements.* The design basis external hazard level for the relevant external hazards for a site must be identified and characterized based on

site-specific assessments of natural and manmade hazards with the potential to adversely affect plant functions. The external hazard frequencies and magnitudes determined from the site-specific assessments must take into account uncertainties and variabilities in data, models, and methods relied on to characterize the external hazards.

(b) *Definitions.* For the purpose of this section, the following terms mean:

Geological Siting Factors are geological and seismic factors that may affect the design and operation of the proposed commercial nuclear plant.

Ground Motion Response Spectra (GMRS) are the site-specific GMRS resulting from the geologic investigations and evaluations of the site vicinity and region and used to determine DBGMs for SSCs under § 53.480.

Probabilistic Seismic Hazard Analysis (PSHA) is an analytical methodology that incorporates uncertainty into estimates of an annual frequency of exceedance for a certain ground motion parameter (e.g., peak ground acceleration, peak ground velocity, response spectral values) at a site.

(c) *Geological Investigations*. The GMRS for the site must be determined based on the results of investigations of the geological, seismological, and engineering characteristics of the site and its environs and must be characterized by both horizontal and vertical free-field GMRS at the free ground surface. The size of the region to be investigated and the type of data pertinent to the investigations must be determined based on the nature of the region surrounding the site. Data on vibratory ground motion, earthquake recurrence rates, fault geometry and slip rates, and site subsurface material properties must be obtained by reviewing pertinent literature and carrying out field investigations. Uncertainties are inherent in the parameters and models used to estimate the GMRS for the site. The site assessment must reflect these uncertainties through an appropriate analysis, such as PSHA.

(d) *Geologic and Seismic Siting Factors*. The geologic and seismic siting factors considered for design under §§ 53.415 and 53.480 must include, but are not limited to, determination of the potential for surface tectonic and nontectonic deformations, the size and character of seismically induced floods and water waves that could affect a site from either locally or distantly generated seismic activity, soil and rock stability, liquefaction potential, and natural and artificial slope stability.

§ 53.520 Site characteristics.

Site characteristics that might contribute to the initiation, progression, or consequences of LBEs analyzed in accordance with § 53.450 must be identified, assessed, and considered in the design and analyses required by Subpart C of this part.

§ 53.530 Population-related considerations.

Every site must have an exclusion area, low population zone, and a population center distance as defined in § 53.020.

(a) The offsite radiological consequences estimated by the analyses required by § 53.450(f) must be used to confirm that:

(1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following onset of the postulated fission product release would not receive a radiation dose in excess of 25 rem (250 mSv) 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 (250 mSv) TEDE.

(b) The population center distance must be at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. The boundary of the population center shall be determined upon consideration of population distribution. Political boundaries are not controlling in the calculation of population center distance.

(c) Reactor sites should be located away from very densely populated centers. Areas of low population density are, generally, preferred. However, in determining the acceptability of a particular site located away from a very densely populated center but not in an area of low population density, consideration will be given to safety, environmental, economic, or other factors, which may result in the site being found acceptable.

§ 53.540 Siting interfaces.

Site characteristics must be addressed by the design features, programmatic controls, and supporting analyses used to demonstrate that the safety criteria in §§ 53.210 and 53.220 are met for each commercial nuclear plant. Site characteristics must be such that adequate emergency plans and security plans can be developed and maintained. Changes to site characteristics over the lifetime of a commercial nuclear plant must be considered in the assessments performed under the facility safety program (FSP) required by § 53.890.

Subpart E—Construction and Manufacturing Requirements

§ 53.600 Construction and manufacturing - scope and purpose.

This subpart applies to those construction and manufacturing activities authorized by a CP, a COL, an ML, or an LWA issued under Framework A of this part.

§ 53.605 Reporting of defects and noncompliance.

Each construction permit and manufacturing license issued under Framework A of this part is subject to the terms and conditions in this section, and each combined license issued under Framework A of this part is subject to the terms and conditions in this section until the date that the Commission makes the finding under § 53.1452(g) of this chapter.

(a) *Definitions*. The definitions in 10 CFR 21.3 apply to this section.

(b) *Posting requirements*.

(1) Each individual, partnership, corporation, dedicating entity, or other entity subject to the regulations in this part shall post current copies of this section and the regulations in part 21 of this chapter; Section 206 of the ERA; and procedures adopted under the regulations. These documents must be posted in a conspicuous position on any premises within the United States where the activities subject to the license are conducted.

(2) If posting of these regulations or the procedures adopted under them is not practical, the licensee may, in addition to posting Section 206 of the ERA, post a notice which describes the regulations/procedures, including the name of the individual to whom reports may be made, and states where they may be examined.

(c) *Procedures*. The holder of a construction permit, a combined license, or a manufacturing license subject to this section must adopt appropriate procedures to –

(1) Evaluate deviations and failures to comply to identify defects and failures to comply associated with substantial safety hazards as soon as practicable, and, except as provided in paragraph (c)(2) of this section, in all cases within 60 days of discovery, to

identify a reportable defect or failure to comply that could create a substantial safety hazard, were it to remain uncorrected.

(2) Ensure that if an evaluation of an identified deviation or failure to comply potentially associated with a substantial safety hazard cannot be completed within 60 days from the discovery of the deviation or failure to comply, an interim report is prepared and submitted to the Commission through a director or responsible officer or designated person as discussed in paragraph (d)(5) of this section. The interim report should describe the deviation or failure to comply that is being evaluated and should also state when the evaluation will be completed. This interim report must be submitted in writing within 60 days of discovery of the deviation or failure to comply.

(3) Ensure that a director or responsible officer of the holder of a construction permit, combined license, or manufacturing license subject to this section is informed as soon as practicable, and, in all cases, within the 5 working days after completion of the evaluation described in paragraph (c)(1) or (c)(2) of this section, if the construction or manufacture of a facility or activity, or a basic component supplied for such a facility or activity –

(i) Fails to comply with the AEA or any applicable regulation, order, or license of the Commission relating to a substantial safety hazard;

(ii) Contains a defect; or

(iii) Underwent any significant breakdown in any portion of the quality assurance program conducted under the requirements of Subpart K to this part which could have produced a defect in a basic component. These breakdowns in the quality assurance program are reportable whether or not the breakdown actually resulted in a defect in a design approved and released for construction, installation or manufacture.

(d)(1) The holder of a construction permit, combined license, or manufacturing license subject to this section that obtains information reasonably indicating that the facility or manufactured reactors fail to comply with the AEA or any applicable regulation, order, or license of the Commission relating to a substantial safety hazard must notify the Commission of the failure to comply through a director, responsible officer, or designated person as discussed in paragraph (d)(5) of this section.

(2) The holder of a construction permit, combined license, or manufacturing license subject to this section that obtains information reasonably indicating the existence of any defect found in the construction or manufacture, or any defect found in the final design of a facility as approved and released for construction or manufacture, must notify the Commission of the defect through a director, responsible officer, or designated person as discussed in paragraph (d)(5) of this section.

(3) The holder of a construction permit, combined license, or manufacturing license subject to this part, who obtains information reasonably indicating that the quality assurance program has undergone any significant breakdown discussed in paragraph (c)(3)(iii) of this section must notify the Commission of the breakdown in the quality assurance program through a director, responsible officer, or designated person as discussed in paragraph (d)(5) of this section.

(4) When acting as a dedicating entity, the holder of a construction permit, combined license, or manufacturing license subject to this section is responsible for identifying and evaluating deviations; reporting defects and failures to comply associated with substantial safety hazards for dedicated items; and maintaining auditable records for the dedication process.

(5) The notification requirements of this paragraph apply to all defects and failures to comply associated with a substantial safety hazard regardless of whether

extensive evaluation, redesign, or repair is required to conform to the criteria and bases stated in the Safety Analysis Report, construction permit, combined license, or manufacturing license. Evaluation of potential defects and failures to comply and reporting of defects and failures to comply under this section satisfies the construction permit holder's, combined license holder's, and manufacturing license holder's evaluation and notification obligations under 10 CFR part 21, and satisfies the responsibility of individual directors or responsible officers or holders of a construction permit, combined license, or manufacturing license subject to this section to report defects, and failures to comply associated with substantial safety hazards under section 206 of the ERA. The director or responsible officer may authorize an individual to provide the notification required by this section, provided that this must not relieve the director or responsible officer of his or her responsibility under this section.

(e) *Notification – timing and where sent*. The notification required by paragraph (d) of this section must consist of –

(1) Initial notification by telephone, facsimile, or e-mail identified in Appendix A to part 73 of this chapter to the NRC Operations Center within 2 days following receipt of information by the director or responsible corporate officer under paragraph (c)(3) of this section, on the identification of a defect or a failure to comply. Verification that the facsimile has been received should be made by calling the NRC Operations Center. This paragraph does not apply to interim reports described in paragraph (c)(2) of this section.

(2) Written notification submitted to the Document Control Desk, NRC, by an appropriate method listed in § 53.040, with a copy to the appropriate Regional Administrator at the address specified in appendix D to 10 CFR part 20 and a copy to the appropriate NRC resident inspector, if applicable, within 30 days following receipt of

information by the director or responsible corporate officer under paragraph (c)(3) of this section, on the identification of a defect or failure to comply.

(f) *Content of notification*. The written notification required by paragraph (e)(2) of this section must clearly indicate that the written notification is being submitted under this section and include the following information, to the extent known.

(1) Name and address of the individual or individuals informing the Commission.

(2) Identification of the facility, the activity, or the basic component supplied for the facility or the activity within the United States which contains a defect or fails to comply.

(3) Identification of the firm constructing or manufacturing the facility or supplying the basic component which fails to comply or contains a defect.

(4) Nature of the defect or failure to comply and the safety hazard which is created or could be created by the defect or failure to comply.

(5) The date on which the information of a defect or failure to comply was obtained.

(6) In the case of a basic component which contains a defect or failure to comply, the number and location of these components in use at the facility subject to the regulations in this part.

(7) In the case of a completed reactor manufactured under Framework A of this part, the entities to which the reactor was supplied.

(8) The corrective action which has been, is being, or will be taken; the name of the individual or organization responsible for the action; and the length of time that has been or will be taken to complete the action.

(9) Any advice related to the defect or failure to comply about the facility, activity, or basic component that has been, is being, or will be given to other entities.

(g) *Procurement documents*. Each holder of a construction permit, combined license, or manufacturing license subject to this section shall ensure that each procurement document for a facility or a basic component specifies the provisions of 10 CFR part 21 or this section that apply, as applicable.

(h) *Coordination with 10 CFR part 21*. The requirements of this section are satisfied when the defect or failure to comply associated with a substantial safety hazard has been previously reported under 10 CFR part 21, under 10 CFR 73.71, under this section or under § 53.1640.

(i) *Records retention*. The holder of a construction permit, combined license, or manufacturing license subject to this section must prepare and maintain records necessary to accomplish the purposes of this section, specifically –

(1) Retain procurement documents, which define the requirements that facilities or basic components must satisfy in order to be considered acceptable, for the lifetime of the facility or basic component.

(2) Retain records of evaluations of all deviations and failures to comply for the longest of:

(i) Ten (10) years from the date of the evaluation;

(ii) Five (5) years from the date that an early site permit is referenced in an application for a combined license; or

(iii) Five (5) years from the date of delivery of a manufactured reactor.

(3) Retain records of all interim reports to the Commission made under paragraph (c)(2) of this section, or notifications to the Commission made under paragraph (d) of this section for the minimum time periods stated in paragraph (i)(2) of this section;

(4) Suppliers of basic components must retain records of:

(i) All notifications sent to affected licensees or purchasers under paragraph (d)(4) of this section for a minimum of ten (10) years following the date of the notification;

(ii) The facilities or other purchasers to whom the basic components or associated services were supplied for a minimum of fifteen (15) years from the delivery of the basic component or associated services.

(5) Maintaining reports in accordance with this section satisfies the recordkeeping obligations under 10 CFR part 21 of the entities, including directors or responsible officers thereof, subject to this section.

§ 53.610 Construction.

(a) *Management and control*. Licensees must ensure that the following plans, programs, and organizational units are developed and implemented to manage and control the construction activities:

(1) Programs to ensure that the construction of a commercial nuclear plant supports the eventual compliance with the design and analysis requirements in subpart C of this part.

(2) An organization, headed by qualified personnel, responsible for managing, controlling, and evaluating the adequacy of the construction activities.

(3) Procedures describing the qualifications for personnel in key positions in the licensee’s management and control organization and the organizational responsibilities, authority, and interfaces with other parts of the licensee’s organization.

(4) Procedures to evaluate the applicability of other national and international construction experience to the planned and ongoing construction activities and to ensure the applicable experience will be provided to those constructing the plant.

(5) A fitness-for-duty (FFD) program, under 10 CFR part 26.

(6)(i) A Quality Assurance (QA) Program meeting the requirements of Subpart K to this part as required by 53.460(b).

(ii) Appropriate programmatic controls to provide special treatment for NSRSS SSCs.

(7) A radiation protection program, in accordance with 10 CFR part 20, that includes measures for monitoring the dose to individuals working with radioactive materials brought onto the site, as applicable.

(8) An information security program in accordance with 10 CFR 73.21, 73.22, and 73.23, as applicable.

(9) A cybersecurity program established in accordance with 10 CFR 73.54 or 73.110, as applicable.

(b) *Construction activities.* No person may begin the construction of a commercial nuclear plant on a site on which the facility is to be operated under Framework A of this part until that person has been issued either a construction permit or combined license, an early site permit authorizing activities under § 53.1130, or an LWA under Framework A of this part.

(1) Licensees must satisfy the following requirements:

(i) As appropriate, considering the types and quantities of radioactive materials being brought onto the site:

(A) The licensee must maintain and follow a special nuclear material (SNM) material control and accounting (MC&A) program, a measurement control program, and other material control procedures that include corresponding record management requirements as required by the provisions of 10 CFR 70.32. Prior to initial receipt of SNM onsite, the licensee shall implement a SNM MC&A Program in accordance with 10 CFR part 74.

(B) Procedures must be in place to receive, possess, use, and store source, byproduct, and SNM in accordance with applicable portions of 10 CFR parts 30, 40, and 70.

(C) A plant staff training program associated with the receipt of radioactive material must be approved and implemented prior to initial receipt of byproduct, source or SNM (excluding exempt quantities as described in 10 CFR 30.18).

(ii) For construction of a commercial nuclear plant involving multiple reactor units, plans and procedures must be in place to prevent or mitigate potential hazards to the SSCs of operating units resulting from construction activities, including the managerial and administrative controls to be used to provide assurance that the limiting conditions for operation of the operating units are not exceeded as a result of construction activities.

(iii) Procedures must be in place prior to the start of construction activities that describe how construction will be controlled so as not to impact other features important to the design, such as dewatering, slope stability, backfill, compaction, and seepage.

(iv) For LWA holders, a plan must be developed for redress of activities performed under the LWA should one of the following situations arise:

(A) LWA work activities are terminated by the holder of the LWA;

(B) The LWA is revoked by the NRC; or

(C) The Commission denies the associated construction permit or combined license application.

(2) *Onsite fresh fuel*.

(i) Onsite fresh fuel must be protected and stored in compliance with 10 CFR 73.67.

(ii) Before initial fuel load into the reactor, a cybersecurity program that meets the requirements of 10 CFR 73.54 or 73.110, a physical security program that meets the requirements of 10 CFR 73.55 or 73.100, and an access authorization program that meets the requirements of 10 CFR 73.56 or 73.120 must be established, as applicable.

(iii) Holders of an OL or a COL after the Commission makes the finding under § 53.1452 must implement fire protection measures for work and storage areas (including adjacent fire areas that could affect the work or storage area) before initial receipt of byproduct, source, or non-fuel SNM (excluding exempt quantities as described in 10 CFR 30.18). The fire protection measures for areas associated with new fuel (including all fuel handling, fuel storage, and adjacent fire areas that could affect the new fuel) must be implemented before receipt of fuel. Prior to the receipt of fuel, a formal letter of agreement must be in place with the local fire department specifying the nature of arrangements in support of the fire protection program.

(c) *Inspection and acceptance*

(1) The licensee must have a process for accepting individual or groups of SSCs upon completion of construction and protecting them from damage or tampering as other construction activities continue.

(2) The post construction acceptance process must address the inspections, tests, analyses, and acceptance criteria (ITAAC) specified in the combined license under § 53.1440 or the equivalent verifications needed to support the issuance of an operating license under § 53.1387.

§ 53.620 Manufacturing.

(a) *Management and control*. Holders of manufacturing licenses must ensure that the following plans, programs, and organizational units are developed and implemented to manage and control the manufacturing activities within the scope of the ML:

(1) Programs to ensure that the manufacturing of a manufactured reactor, portions of a manufactured reactor, or a manufactured reactor module complies with the design and analysis requirements in subpart C of this part. The entity with design authority for the manufactured reactor or manufactured reactor module covered by the manufacturing license must be identified in the license.

(2) An organizational and management structure responsible for managing, controlling, and evaluating the adequacy of the reactor design and manufacturing activities.

(3) Procedures describing the qualifications for personnel in key positions in the licensee's management and control organization and the organizational responsibilities, authority, and interfaces with other parts of the licensee's organization.

(4) A program to evaluate the applicability of other national and international design and manufacturing experience to the planned and ongoing manufacturing activities.

(5) An FFD program, in accordance with 10 CFR part 26.

(6)(i) A QA program meeting the requirements of Subpart K of this part, to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the manufactured reactor or manufactured reactor module.

(ii) Appropriate programmatic controls to provide special treatment measures for NSRSS SSCs.

(7) A radiation protection program, in accordance with 10 CFR part 20, that includes measures for monitoring the dose to individuals if the manufacturing activities include working with radioactive materials.

(8) An information security program in accordance with 10 CFR 73.21, 73.22 and 73.23, as applicable.

(b) *Manufacturing activities*. Holders of manufacturing licenses must satisfy the following requirements:

(1) The manufacturing process must be conducted within facilities for which the manufacturing license holder has the authority to establish controls on any activity that might affect manufacturing. The licensee must establish access controls to the portions of each facility involved in the manufacturing processes governed by the ML.

(2) Manufacturing processes must be performed in accordance with the ML and the referenced codes and standards that have been endorsed or otherwise found acceptable by the NRC.

(3) A post-manufacturing inspection and acceptance process must be established and implemented before transporting a manufactured reactor or portions of a manufactured reactor for installation at a commercial nuclear plant and prior to and following the loading of fresh fuel into a manufactured reactor module. The process must consider the results of inspections, tests, and analyses that have been performed and the acceptance criteria that are necessary and sufficient to conclude that manufacturing activities have been completed in accordance with the ML.

(c) *Control of radioactive materials*. As appropriate considering the types and quantities of radioactive materials being brought into the manufacturing facility:

(1) Procedures must be in place to receive, transfer, possess, and use source, byproduct, and SNM in accordance with the applicable portions of 10 CFR parts 30, 40 and 70.

(2) A fire protection program must be established and implemented before the initial receipt of byproduct, source, or non-fuel SNM (excluding exempt quantities as described in 10 CFR 30.18). The fire protection measures for areas associated with fueling a manufactured reactor module (including all fuel handling, fuel storage and adjacent areas where a fire could affect the fresh fuel) must be implemented before receipt of fresh fuel at the manufacturer's facility. Prior to the receipt of fuel at the manufacturer's facility, a formal letter of agreement must be in place with the local fire department specifying the nature of arrangements in support of the fire protection program.

(3) An emergency plan appropriate for responding to the facility-specific hazards of an accidental release of radioactive material and to limit the health effects of the associated chemical hazards of licensed material must be approved and implemented prior to the receipt of byproduct, source, or SNM (excluding exempt quantities as described in 10 CFR 30.18).

(4) A plant staff training program associated with the receipt of radioactive material must be approved and implemented before initial receipt of byproduct, source, or special nuclear material (excluding exempt quantities as described in 10 CFR 30.18).

(5) Procedures shall be in place to describe how the manufacturing facility design and manufacturing process will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste. Manufacturing licensees shall, to the extent practical, conduct operations to minimize the introduction of residual

radioactivity into the facility site, including the subsurface, under § 20.1406 of this chapter.

(d) *Fuel loading*

(1) (i) A manufacturing license may include authorizing the loading of fuel into a manufactured reactor module only if the module is configured during its loading and storage to provide at least two independent mechanisms each of which is sufficient to prevent criticality assuming maximum reactivity of the fissile material would be attained from possible fuel configurations, neutron moderation, and neutron reflection from the module and surrounding materials. The Commission has determined that any such manufactured reactor module in which these mechanisms have been installed is not a utilization facility as defined in section 11cc. of the AEA or § 53.020 until it is installed in its final place of use and the Commission has found that both the ITAAC in the ML are met under § 53.620(f) and the ITAAC in the COL that authorized reactor construction are met under § 53.1452(g); and

(ii) The Commission has determined that, upon a Commission finding with respect to a particular module that the ITAAC are met in accordance with a COL and § 53.1452(g) that the manufactured reactor module is a utilization facility and all COL provisions and regulations applicable to the type of commercial nuclear plant for which the Commission has made the finding apply to that manufactured reactor module.

(2) If the ML authorizes fuel loading into a manufactured reactor module at the manufacturing facility, the following must be in place prior to the receipt of SNM:

(i) Radiation monitoring instrumentation and alarms.

(ii) Measures to prevent criticality accidents in accordance with §§ 70.61 and 70.64 of this chapter and to detect potential criticality accidents in accordance with § 53.440(m).

(iii) Procedures, equipment, and personnel qualified to handle fresh fuel, load it into the reactor, monitor the reactivity, and secure the fuel and reactor assembly for shipment.

(iv) A physical security program for the storage of fresh fuel in accordance with 10 CFR 73.67.

(v) An MC&A program in accordance with 10 CFR part 74.

(3) The storage, movement, and loading of fresh fuel into the manufactured reactor module within the manufacturing facility must comply with the requirements of §§ 70.61, 70.62 and 70.64 of this chapter.

(4) The loading or unloading of fresh fuel into or from a manufactured reactor module and any changes to the configuration of reactivity-related systems for the manufactured reactor module must be performed by a certified fuel handler meeting the requirements in subpart F.

(e) *Transportation.*

(1) A holder of a manufacturing license may not transport or allow to be removed from the places of manufacture the manufactured reactor or major portions thereof as defined in the ML except to the site of a licensee with a combined license. The combined license must authorize the construction of a commercial nuclear plant using the manufactured reactor(s).

(2) A holder of a manufacturing license shall include, in any contract governing the transport of a manufactured reactor or major portions thereof as defined in the ML from the places 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.

(3) Procedures governing the preparation of the manufactured reactor or major portions thereof as defined in the ML for transport and the conduct of the transport must be documented and approved prior to transport. The procedures must implement the protective measures and restrictions described in the ML to protect the reactor from potential conditions that would adversely affect the safe operation of a commercial nuclear plant.

(4) The packaging and shipping of any fueled manufactured reactor module must be done in compliance with 10 CFR parts 71 and 73.

(f) *Acceptance and installation at the site*.

(1) Installation at the site must follow the regulations in § 53.610.

(2) Upon arrival at the site, the manufactured reactor, portions of a manufactured reactor, or a manufactured reactor module may not be installed in its final place of use unless the COL holder performs inspections, using approved procedures, and verifies it is in acceptable condition in compliance with the ML. These inspections must confirm that all interface requirements between the manufactured reactor, portions of a manufactured reactor, or a manufactured reactor module and the remaining portions of the commercial nuclear power plant are met.

Subpart F—Requirements for Operation

§ 53.700 Operational objectives.

(a) Each holder of an operating license or combined license under Framework A of this part must develop, implement, and maintain controls for plant SSCs, responsibilities of plant personnel, and plant programs during the operating life of each commercial nuclear plant such that the requirements defined in subpart B are satisfied.

(b) Each holder of an operating license or combined license under Framework A of this part must maintain the capabilities, availability, and reliability of plant SSCs to ensure that the safety functions identified in § 53.230 will be performed if called upon during LBEs.

(c) Each holder of an operating license or combined license under Framework A of this part must ensure that plant personnel have adequate knowledge and skills to perform their assigned duties that support the performance of the safety functions identified in § 53.230.

(d) Each holder of an operating license or combined license under Framework A of this part must implement plant programs sufficient to ensure that the safety functions identified in § 53.230 will be performed if called upon during normal operations and LBEs.

§ 53.710 Maintaining capabilities and availability of structures, systems, and components.

Controls must be provided for each commercial nuclear plant such that the capabilities, availability, and reliability of plant SSCs, when combined with corresponding programmatic controls and human actions, provide that the safety criteria defined in §§ 53.210 and 53.220 will be met.

(a) Technical specifications must be developed, implemented, and maintained that define conditions or limitations on plant operations that are necessary to ensure that SR SSCs fulfill the safety functions identified in § 53.230 and support meeting the safety criteria of § 53.210. The technical specifications must describe the following requirements:

(1) Limits on the inventory of radioactive materials within the reactor system and supporting systems with the potential, individually or collectively, to cause a release exceeding the safety criteria in § 53.210 as a result of a design-basis accident analyzed in accordance with § 53.450(f).

(2) Operating limits for the facility that if exceeded could lead to a failure to perform a required safety function necessary to demonstrate compliance with the safety criteria in § 53.210.

(3) For each SSC classified as SR in accordance with § 53.460, technical specifications must define:

(i) *Limiting conditions for operation*. Limiting conditions for operation are the lowest functional capability or performance levels of SR SSCs required to ensure that the design-basis accidents analyzed in accordance with § 53.450(f) would not give rise to an immediate threat to the public health and safety as represented by the safety criteria of § 53.210. When a limiting condition for operation is not met, the licensee must shut down the plant or follow any remedial action permitted by the technical specifications until the condition can be met.

(ii) *Surveillance requirements.* Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained and that the limiting conditions for operation will be met.

(4) Design elements to be included are those elements of the plant such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (a)(1)-(3) of this section.

(5) Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the plant in a safe manner. Each licensee must submit any reports to the Commission pursuant to approved technical specifications as specified in § 53.040.

(b) Controls on plant operations, including availability controls, must be developed and implemented to ensure that the configurations and special treatments for NSRSS SSCs provide the capabilities, availability, and reliability required to demonstrate compliance with the criteria of §§ 53.220 and 53.450(e). The controls must:

(1)(i) Identify who within the commercial nuclear plant has authority to make configuration changes;

(ii) Establish processes to make configuration changes to NSRSS SSCs; and

(iii) Establish processes to ensure that all departments of the commercial nuclear plant affected by the configuration changes are formally notified and approve of the change.

(2) Describe how the special treatments for each NSRSS SSC will be established and maintained over the operating life of the commercial nuclear plant.

§ 53.715 Maintenance, repair, and inspection programs.

(a) A program to control maintenance activities and monitor the performance or condition of SR and NSRSS SSCs must be developed, implemented, and maintained to ensure that the safety criteria defined in §§ 53.210 and 53.220 will be met.

(b) Whenever a licensee determines through activities related to maintenance, repair, and inspection of SSCs, the activities under § 53.710, or otherwise that the performance or condition of an NSRSS SSC does not demonstrate compliance with

established special treatments or performance goals related to capabilities, availability, or reliability, the licensee must take appropriate corrective action.

(c) Performance and condition monitoring activities and associated goals and preventive maintenance activities must be evaluated at least every 24 months. The evaluations must take into account, where practical, industry-wide operating experience. Adjustments must be made where necessary to ensure that the objective of preventing failures of SSCs through maintenance is appropriately balanced against the objective of minimizing unavailability of SSCs due to monitoring or preventive maintenance.

(d) Before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee must assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of the assessment may be limited to SSCs that a risk-informed evaluation process determines are necessary to ensure that the criteria defined in §§ 53.210, 53.220, and 53.450(e) will be met.

§ 53.720 Response to seismic events.

If vibratory ground motion exceeding that of the OBE Ground Motion or significant plant damage due to vibratory ground motion occurs, the licensee must shut down the commercial nuclear plant. If SSCs necessary for the safe shutdown of the commercial nuclear plant are not available after the occurrence of this vibratory ground motion, the licensee must consult with the Commission and must propose a plan for the timely, safe shutdown of the commercial nuclear power plant. Prior to resuming operations, the licensee must demonstrate to the Commission that those features necessary for continued operation without undue risk to the health and safety of the

public or necessary to maintain the licensing basis of the commercial nuclear plant were either not functionally damaged or have been repaired.

§ 53.725 General staffing, training, personnel qualifications, and human factors requirements.

(a) *Purpose and applicability*. The regulations in §§ 53.725 through 53.830 address areas related to staffing, training, personnel qualifications, and human factors engineering for applicants for or holders of operating licenses or combined licenses under Frameworks A and B of this part. These regulations are organized as follows:

(1) Sections 53.725 through 53.755 address general requirements for staffing, training, personnel qualifications, and human factors engineering. The regulations within these sections are applicable to all applicants for or holders of operating licenses or combined licenses under Frameworks A and B of this part, except where specifically stated otherwise.

(2) Sections 53.760 through 53.795 address operator and senior operator licensing requirements. The regulations within these sections are applicable to those applicants for or holders of operating licenses or combined licenses under this part that do not demonstrate compliance with the criteria provided under § 53.800 and have not yet certified the permanent cessation of operations and permanent removal of fuel from the reactor vessel as described under §§ 53.1070 or 53.4670, as applicable.

(3) Sections 53.800 through 53.820 address generally licensed reactor operator requirements. The regulations within these sections are in lieu of §§ 53.760 through 53.795 for those applicants for or holders of operating licenses or combined licenses under this part that demonstrate compliance with the criteria provided under § 53.800 and have not yet certified the permanent cessation of operations and permanent

removal of fuel from the reactor vessel as described under §§ 53.1070 or 53.4670, as applicable.

(4) Section 53.830 provides general personnel training requirements. The regulations within this section are applicable to all applicants for or holders of operating licenses or combined licenses under this part.

(b) *Definitions*. When used in §§ 53.725 through 53.830, *applicant* refers to an applicant for an operator or senior operator license; *licensee* refers to the holder of an operator, senior operator, or generally licensed reactor operator license; and *facility licensee* refers to the licensee for the commercial nuclear plant where the applicant would be licensed or the licensee is licensed. As also used in §§ 53.725 through 53.830:

Automation means a device or system that accomplishes (partially or fully) a function or task.

Auxiliary operator means any individual who operates components of a commercial nuclear plant but does not manipulate controls or direct the manipulation of controls of the plant and is not required to be licensed under the provisions of this part.

Controls when used with respect to a nuclear reactor means apparatus and mechanisms, the manipulation of which directly affects the reactivity or power level of the reactor.

Generally licensed reactor operator means any individual licensed under the provisions of § 53.810 to manipulate controls of a self-reliant mitigation facility and to direct the licensed activities of generally licensed reactor operators.

Load following means a commercial nuclear plant automatically changing its output to match expected demand in response to externally originated instructions or signals.

Operator means any individual licensed under the provisions of §§ 53.760 through 53.795 to manipulate controls of a commercial nuclear plant.

Performance testing means testing conducted to verify a simulation facility's performance as compared to actual or predicted reference plant performance.

Reference plant means the specific commercial nuclear plant on which a simulation facility's configuration, system control arrangement, and design data are based. The reference plant may or may not be constructed.

Self-reliant mitigation facility means a commercial nuclear plant design that demonstrates compliance with the operating and technical characteristics defined under § 53.800.

Senior operator means any individual licensed under the provisions of §§ 53.760 through 53.795 to manipulate controls of a commercial nuclear plant and to direct the licensed activities of operators.

Simulation facility means an interface designed to provide a realistic imitation of the operation of a commercial nuclear plant used for the administration of examinations, for training, and/or to demonstrate compliance with experience requirements for applicants or licensees. A simulation facility may rely, in whole or part, upon the physical utilization of the reference plant itself.

Systems approach to training means a training program that includes the following five elements:

(1) Systematic analysis of the jobs to be performed.

(2) Learning objectives derived from the analysis which describe desired performance after training.

(3) Training design and implementation based on the learning objectives.

(4) Evaluation of trainee mastery of the objectives during training.

(5) Evaluation and revision of the training based on the performance of trained personnel in the job setting.

§ 53.726 Communications.

An applicant or licensee or facility licensee must submit any communication or report concerning the regulations contained within §§ 53.725 through 53.830 and must submit any application filed under these regulations to the Commission.

§ 53.727 Information collection requirements.

(a) The NRC has submitted the information collection requirements contained within §§ 53.725 through 53.830 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 within §§ 53.725 through 53.830 under control number XXXX-XXXX.

(b) The approved information collection requirements appear in §§ 53.765, 53.770, 53.775, 53.780, and 53.795.

(c) Sections 53.725 through 53.830 contain information collection requirements in addition to those approved under the control number specified in paragraph (a) of this section. These information collection requirements and the control numbers under which they are approved are as follows:

(1) In §§ 53.765, 53.770, 53,780, and 53,795, NRC Form 396 is approved under control number XXXX-XXXX.

(2) In §§ 53.775 and 53.795, NRC Form 398 is approved under control number XXXX-XXXX.

(3) In § 53.780, clearance is approved under control number XXXX-XXXX.

§ 53.728 Completeness and accuracy of information.

Information provided to the Commission by an applicant for an operator or senior operator license or by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee must be complete and accurate in all material respects.

§ 53.730 Defining, fulfilling, and maintaining the role of personnel in ensuring safe operations.

Each applicant for or holder of an operating license or combined license for a commercial nuclear plant under this part must comply with the following:

(a) *Human factors engineering design requirements*. The plant design must reflect state-of-the-art human factors engineering principles for safe and reliable performance in all locations that human activities are expected for performing or supporting the continued availability of plant safety or emergency response functions.

(b) *Human system interface design requirements.* The plant design must provide for the following to support operating personnel in monitoring plant conditions and responding to plant events:

(1) features for displaying to operating personnel a minimum set of parameters that define the safety status of the plant and are capable of displaying both the full range of important plant parameters and data trends on demand, as well as indicating when process limits are being approached or exceeded;

(2) automatic indication of the bypassed and operable status of safety systems;

(3) direct indication of SSC status that relates to the ability of the SSC to perform its safety function, such as relief and safety valve position (i.e., open or closed) for barriers important to fulfilling safety functions of with such devices, and ultimate heat sink and cooling system status and availability;

(4) instrumentation to measure, record, and display key plant parameters related to the performance of SSCs and the integrity of barriers important to fulfilling safety functions to support operators in monitoring plant conditions and responding to plant events. Examples include temperatures and pressures within important systems or structures, core or fuel system conditions (including possible damage states), temperatures and levels associated with cooling functions, combustible gas concentrations, radiation levels in systems and within structures, and radioactive effluent releases;

(5) leakage control and detection in the design of systems that pass through barriers important to fulfilling safety functions for the release of radionuclides. An example is an SSC that penetrates a containment structure that might contain radioactive materials that could contribute to the source term during an accident;

(6) monitoring of in-plant radiation and airborne radioactivity as appropriate for a broad range of normal operating and accident conditions; and

(7) for applicants or holders of an operating license or combined license subject to the provisions of §§ 53.800 through 53.820, the plant design must also provide the generally licensed reactor operators with the capability to do the following:

(i) receive plant operating data, including reactor parameters and information needed for the evaluation of emergency conditions.

(ii) immediately initiate a reactor shutdown from their location.

(iii) promptly dispatch operations and maintenance personnel.

(iv) immediately implement responsibilities under the facility emergency plan, as applicable.

(c) *Concept of operations.* A concept of operations that is of sufficient scope and detail to address the following must be provided:

(1) plant goals;

(2) the roles and responsibilities of operating personnel and automation (or any combination thereof) that are responsible for completing plant functions;

(3) staffing, qualifications, and training;

(4) the management of normal operations;

(5) the management of off-normal conditions and emergencies;

(6) the management of maintenance and modifications; and

(7) the management of tests, inspections, and surveillances.

(d) *Functional requirements analysis and function allocation*. A functional requirements analysis and a function allocation must be provided that are sufficient to demonstrate compliance with the following:

(1) the functional requirements analysis must address how safety functions and functional safety criteria are satisfied, and

(2) the function allocation must describe how the safety functions will be assigned to human action, automation, active safety features, passive safety features, and/or inherent safety characteristics.

(e) *Programmatic requirements*. A program, during construction and during operation, as applicable, for evaluating and applying operating experience must be developed, implemented, and maintained.

(f) *Staffing plan.* A staffing plan must be developed and comply with the following:

(1) The staffing plan must include a description of how engineering expertise will be available to the on-shift operating personnel during all plant conditions, to assist if they encounter a situation not covered by procedures or training. Engineering expertise includes familiarity with the operation of the plant for which the expertise is provided and one of the following:

(i) a bachelor's degree in engineering, engineering technology, or physical science from an institution accredited by a U.S. Government recognized accrediting body or equivalent; or

(ii) a Professional Engineer's license from a U.S. State or territory.

(2) Applicants for or holders of operating licenses or combined licenses subject to the provisions of §§ 53.760 through 53.795 must include within their staffing plans a description of how the proposed numbers, positions, and qualifications of operators and senior operators across all modes of plant operations will be sufficient to ensure that plant safety functions will be maintained. This description must be supported by human factors engineering analyses and assessments.

(3) Applicants for or holders of operating licenses or combined licenses subject to the provisions of §§ 53.800 through 53.820 must include within their staffing plans a description of how generally licensed reactor operator staffing that is both sufficient to continually monitor the operations of fueled reactors and to provide for a continuity of responsibility for facility operations at all times during the operating phase will be maintained.

(4) Applicants for or holders of operating licenses or combined licenses under this part must include within their staffing plans a description of how the numbers, positions, and responsibilities of personnel contained within those plans will adequately support all necessary functions within areas such as plant operations, equipment

surveillance and maintenance, radiological protection, chemistry control, fire brigades, engineering, security, and emergency response.

(5) The staffing plan must be approved by the NRC as part of its approval of the operating license or combined license for the plant. The approved staffing plan is subject to the requirements of §§ 53.1565 or 53.6065, as applicable.

(g) *Training, examination, and proficiency programs*. Develop, implement, and maintain programs that comply with the following requirements. These programs must be approved by the NRC as part of its approval of the operating license or combined license for the plant:

(1) For those applicants for or holders of operating licenses or combined licenses under this part subject to the provisions of §§ 53.760 through 53.795:

(i) The operator licensing initial training program required under § 53.780(a);

(ii) The operator licensing initial examination program required under § 53.780(b);

(iii) The operator licensing requalification program required under § 53.780(c); and

(iv) The operator proficiency program required under § 53.780(g).

(2) For those applicants for or holders of operating licenses or combined licenses under this part subject to the provisions of §§ 53.800 through 53.820, the generally licensed reactor operator training, examination, and proficiency programs required under § 53.815.

(3) The operator licensing requalification programs required under §§ 53.780(c) or 53.815(b) must be implemented upon commencing the administration of initial examinations under the operator licensing examination program required under §§ 53.780(b) or 53.815(b), respectively.

§ 53.735 General exemptions.

The regulations in §§ 53.725 through 53.820 do not require a license for an individual who –

(a) Under the direction and in the presence of an operator or senior operator or a generally licensed reactor operator, as appropriate, manipulates the controls of a commercial nuclear plant as a part of the individual's training in a facility licensee's training program as approved by the Commission to qualify for an operator or senior operator license or a generally licensed reactor operator license, as appropriate, under these regulations; or

(b) Under the direction and in the presence of a senior operator or generally licensed reactor operator, as appropriate, manipulates the controls of a commercial nuclear plant to load or unload the fuel into, out of, or within the reactor vessel while the reactor is not operating.

§ 53.740 Facility licensee requirements – General.

(a) Facility licensees must demonstrate compliance with the requirements of either §§ 53.760 through 53.795 or §§ 53.800 through 53.820, as appropriate.

(b) The facility licensee must maintain the staffing complement described under its approved facility staffing plan until such time as the permanent cessation of operations and permanent removal of fuel from the reactor vessel has been certified as described under §§ 53.1070 or 53.4670, as applicable. The approved staffing plan is subject to the requirements of §§ 53.1565 or 53.6065, as applicable.

(c) Except as provided under § 53.735, the facility licensee may not permit the manipulation of the controls of a commercial nuclear plant by anyone who is not an operator or senior operator or generally licensed reactor operator, as appropriate.

(d) Facility licensees subject to the requirements of §§ 53.760 through 53.795 and that have not yet certified the permanent cessation of operations and permanent removal of fuel from the reactor vessel as described under §§ 53.1070 or 53.4670, as applicable, must designate senior operators to be responsible for supervising the licensed activities of operators.

(e) Apparatus and mechanisms other than controls, the operation of which may affect the reactivity or power level of a reactor, must be manipulated only while plant conditions are being monitored by an individual who is an operator or senior operator or a generally licensed reactor operator, as appropriate.

(f)(1)Load following is permitted if at least one of the following is immediately capable of refusing demands when they could challenge the safe operation of the plant or when precluded by the plant equipment conditions:

(i) the actuation of an automatic protection system that utilizes setpoints more conservative than those otherwise credited for the purposes of reactor protection; or

(ii) an automated control system; or

(iii) an operator or senior operator or a generally licensed reactor operator, as appropriate.

(2) The provisions of paragraph (e) of this section do not apply during load following operations.

(g)(1) Facility licensees subject to the requirements of §§ 53.760 through 53.795 must have present during alteration of the core (including fuel loading or transfer) an individual holding a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the facility licensee must not assign other duties to this person.

(2) Facility licensees subject to the requirements of §§ 53.800 through 53.820 must have present during alteration of the core (including fuel loading or transfer) an individual holding a generally licensed reactor operator license to directly supervise the activity and, during this time, the facility licensee must not assign other duties to this person.

(3) The provisions of paragraphs (g)(1) and (g)(2) of this section do not apply to core alterations performed as part of refueling operations while a facility that is capable of online refueling is operating at power.

(h) Facility licensees may take reasonable action that departs from a license condition or a technical specification (contained in a license issued under this part) in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with license conditions and technical specifications that can provide adequate or equivalent protection is immediately apparent. Such facility licensee action must be approved, as a minimum, by a senior operator or a generally licensed reactor operator, as applicable, or, after certifying the permanent cessation of operations and permanent removal of fuel from the reactor vessel as described under §§ 53.1070 or 53.4670, as applicable, by a certified fuel handler, senior operator, or generally licensed reactor operator, as applicable, prior to taking the action.

§ 53.745 Operator license requirements.

A person must be authorized by a license issued by the Commission to perform the function of an operator, senior operator, or generally licensed reactor operator as defined in this part.

§ 53.750 Violations.

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

(1) The AEA;

(2) Title II of the ERA; or

(3) A regulation or order issued pursuant to those Acts.

(b) The Commission may obtain a court order for the payment of a civil penalty imposed under section 234 of the AEA:

(1) For violations of –

(i) Sections 53, 57, 62, 63, 81, 82, 101, 103, 104, 107, or 109 of the AEA;

(ii) Section 206 of the ERA;

(iii) Any rule, regulation, or order issued pursuant to 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 AEA.

§ 53.755 Criminal penalties.

(a) Section 223 of the AEA 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 AEA. For purposes of section 223, all the regulations in §§ 53.725 through 53.830 are issued under one or more of sections 161b, 161i, or 161o, except for the sections listed in paragraph (b) of this section.

(b) The regulations in §§ 53.725 through 53.830 that are not issued under sections 161b, 161i, or 161o for the purposes of section 223 are as follows:

§§ 53.725(b), 53.726, 53.727, 53.735, 53.750, 53.755, 53.760, 53.775, 53.780(f), 53.790, 53.795, 53.800, 53.810(e), 53.815(f) and 53.820.

§ 53.760 Operator licensing.

(a) *Applicability*. Sections 53.760 through 53.795 address operator licensing requirements. The regulations within these sections are applicable to all applicants for, or holders of, operating licenses or combined licenses for commercial nuclear plants licensed under this part except for those subject to §§ 53.800 through 53.820 and those that have certified the permanent cessation of operations and permanent removal of fuel from the reactor vessel as described under §§ 53.1070 or 53.4670, as applicable.

(b) Reserved.

§ 53.765 Medical requirements.

(a) An applicant for an operator or senior operator license must have a medical examination by a physician. An operator or senior operator must have a medical examination by a physician every two years.

(b) To certify the medical fitness of an applicant for an operator or senior operator license, an authorized representative of the facility licensee must complete and sign NRC Form 396, “Certification of Medical Examination by Facility Licensee,” which can be obtained by writing the Office of the Chief Information Officer, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, by calling (301) 415-5877, or by visiting the NRC’s website at https://www.nrc.gov and selecting forms from the index found on the home page, or by other means provided by the NRC.

(1) Form NRC 396 must certify that a physician has conducted the medical examination of the applicant as required in paragraph (a) of this section.

(2) When the medical certification requests a conditional license based on medical evidence, the medical evidence must be submitted on NRC Form 396 to the Commission to enable the Commission to make a determination in accordance with § 53.775(b).

(c) The facility licensee must document and maintain the results of medical qualifications data, test results, and each operator's or senior operator's medical history for the current license period and provide the documentation to the Commission upon request. The facility licensee must retain this documentation while an individual performs the functions of an operator or senior operator.

§ 53.770 Incapacitation because of disability or illness.

If, during the term of the operator or senior operator license, the licensee develops a permanent physical or mental condition that causes the licensee to fail to demonstrate compliance with the requirements of § 53.775(b)(1)(i), the facility licensee must notify the Commission within 30 days of learning of the diagnosis. For conditions for which a conditional license (as described in § 53.775(b)) is requested, the facility licensee must provide medical certification on Form NRC 396 to the Commission (as described in § 53.765(b)).

§ 53.775 Applications for operators and senior operators

(a) *How to apply*. (1) The applicant for an operator or senior operator license must:

(i) Complete NRC Form 398, "Personal Qualification Statement-- Licensee," which can be obtained by writing the Office of the Chief Information Officer, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, by calling (301) 415-5877, or by

visiting the NRC's website at https://www.nrc.gov and selecting forms from the index found on the home page, or by other means provided by the NRC;

(ii) File an original of NRC Form 398, or an equivalent electronic submittal, together with the information required in paragraphs (a)(1)(iii) and (a)(1)(iv) of this section, with the appropriate Regional Administrator.

(iii) Provide evidence that the applicant, as a trainee, has successfully demonstrated competence in manipulating the controls of either the facility for which a license is sought or a simulation facility that demonstrates compliance with the requirements of § 53.780(e). For operators applying for a senior operator license, certification that the operator has successfully operated the controls of the facility as an operator will be accepted; and

(iv) Provide certification by the facility licensee of medical condition and general health on Form NRC 396, to comply with § 53.765.

(2) The Commission may at any time after the application has been filed, and before the license has expired, require further information under oath or affirmation to enable it to determine whether to grant or deny the application or whether to revoke, modify, or suspend the license.

(3) An applicant whose application has been denied because of a medical condition or general health may submit a further medical report at any time as a supplement to the application.

(4) Each application and statement must contain complete and accurate disclosure as to all matters required to be disclosed. The applicant must sign statements required by paragraphs (a)(1)(i) and (a)(1)(ii) of this section.

(b) *Disposition of an initial application.* (1) The Commission will approve an initial application if it finds that the following criteria are met:

(i) *Health*. The applicant's medical condition and general health will not adversely affect the performance of assigned operator or senior operator job duties or cause operational errors endangering public health and safety. The Commission will base its finding upon the certification by the facility licensee as detailed in § 53.765(b).

(ii) *Examination.* The applicant has passed the requisite examination in accordance with § 53.780(b). The examination determines whether the applicant for an operator's or senior operator's license has learned to operate a facility competently and safely, and additionally, in the case of a senior operator, whether the applicant has learned to supervise the licensed activities of operators competently and safely.

(2) *Conditional license*. If an applicant's general medical condition does not demonstrate compliance with the minimum standards under § 53.775(b)(1)(i), the Commission may approve the application and include conditions in the license to accommodate the medical condition. The Commission will consider the recommendations and supporting evidence of the facility licensee and of the examining physician (provided on Form NRC 396) in arriving at its decision.

(c) *Re-applications.*

(1) An applicant whose application for a license has been denied because of failure to pass the examination may file a new application. The application must be submitted on Form NRC 398 and include a statement signed by an authorized representative of the facility licensee by whom the applicant will be employed that states in detail the extent of the applicant's additional training and remediation since the denial and certifies that the applicant is ready for re-examination.

(2) An applicant who has passed a portion of the examination and failed another may request in a new application on Form NRC 398 to be excused from re-examination

on the portions of the examination that the applicant has passed. The Commission may in its discretion grant the request if it determines that sufficient justification is presented.

§ 53.780 Training, examination, and proficiency program.

(a) *Operator licensing initial training program*. (1) A program that is based upon a systems approach to training, as defined by § 53.725(b), must be utilized for the training of applicants for operator and senior operator licenses. The program must ensure that applicants at the facility will possess the knowledge, skills, and abilities necessary to protect the public health and maintain those plant safety functions specific to the facility design. The program must be approved by the Commission prior to its use for training applicants, as described under § 53.730(g). The approved operator licensing initial training program is subject to the requirements of §§ 53.1565 or 53.6065, as applicable.

(2) *Records*. The operator licensing initial training program documentation must include the following:

(i) The facility licensee must maintain records documenting the initial operator licensing training administered and completed by each applicant. The facility licensee must retain these records during the period in which any trainees subsequently remain licensed as operators or senior operators at the facility.

(ii) Each record required by this part must be legible throughout the retention period specified by each Commission regulation. The record may be the original, a reproduced copy, or an electronic copy provided that the copy is authenticated by authorized personnel.

(b) *Operator licensing initial examination program*. (1) The facility licensee must establish and implement an examination program for testing a representative sample of the knowledge, skills, and abilities needed to safely perform operator and senior

operator duties, to include both the examination methods and criteria to be used to assess passing performance. The program must provide for valid and reliable examinations and be approved by the Commission prior to its use for examining applicants, as described under § 53.730(g). The approved operator licensing initial examination program is subject to the requirements of §§ 53.1565 or 53.6065, as applicable.

(2) The facility licensee must submit prepared examinations to the Commission for review and approval in advance of their administration.

(3) The Commission will either administer an approved examination or allow the facility licensee to administer the examination. The facility licensee must ensure that sufficient advance notification is provided to the Commission to either administer the examination or allow for a representative of the Commission to be afforded the opportunity to be present when the facility licensee administers the examination.

(4) Graded examination documentation for each applicant must be promptly provided to the Commission for review in making operator licensing decisions.

(5) *Records.* The operator licensing initial examination program documentation must include the following:

(i) The facility licensee must maintain records documenting the participation of each operator and senior operator applicant in the initial examination. The records must contain copies of examinations administered, the answers given by the applicant, documentation of the grading of examinations, and documentation of any additional training administered in areas in which an applicant exhibited deficiencies. The facility licensee must retain these records during the period in which the associated operators or senior operators remain licensed at the facility.

(ii) Each record required by this part must be legible throughout the retention period specified by each Commission regulation. The record may be the original, a reproduced copy, or an electronic copy provided that the copy is authenticated by authorized personnel.

(c) *Operator licensing requalification program.* (1) A program based upon a systems approach to training, as defined by § 53.725(b), must be utilized for the continuing training of operators and senior operators.

(i) The program must ensure that operators and senior operators at the facility maintain the knowledge, skills, and abilities necessary to protect the public health and maintain those plant safety functions specific to the facility design. The program must be conducted for a continuous period not to exceed 24 months in duration.

(ii) The program must be approved by the Commission prior to its use for continuing training, as described under § 53.730(g). The approved operator licensing requalification program is subject to the requirements of §§ 53.1565 or 53.6065, as applicable.

(2) The following requirements apply to operator licensing requalification programs:

(i) The facility licensee must propose a requalification examination program for testing, for each requalification period, a sample of the topics included under the systems approach to training, to include both the examination methods and criteria to be used to assess passing performance. The program must provide for valid and reliable examinations and be approved by the Commission prior to its use for examining operators and senior operators, as described under § 53.730(g). The approved requalification examination program is subject to the requirements of §§ 53.1565 or 53.6065, as applicable.

(ii) The following requirements apply to the requalification examination program:

(A) The facility licensee must make prepared requalification examinations available to the Commission for review.

(B) The facility licensee must ensure that a representative of the Commission is afforded the opportunity to be present during requalification examination administration.

(C) The facility licensee must ensure that each operator and senior operator is administered a complete requalification examination on a periodicity not to exceed 24 months. Additionally, the facility licensee must ensure that any licensed operator or senior licensed operator who either demonstrates unsatisfactory performance on, or fails to complete, the biennial requalification examination is removed from the performance of licensed operator and senior licensed operator duties until such time that any necessary remedial training has been completed and a retake examination has been passed.

(D) The facility licensee must promptly provide a summary of examination results for each operator and senior operator following the completion of the requalification examination.

(3) *Records.* The operator licensing requalification program documentation must include the following:

(i) The facility licensee must maintain records documenting the participation of each operator and senior operator in the requalification program. The records must contain copies of examinations administered, the answers given by the operator or senior operator, documentation of the grading of examinations, and documentation of any additional training administered in areas in which an operator or senior operator exhibited deficiencies. The facility licensee must retain these records until the operator's or senior operator's license is renewed.

(ii) Each record required by this part must be legible throughout the retention period specified by each Commission regulation. The record may be the original, a reproduced copy, or an electronic copy provided that the copy is authenticated by authorized personnel.

(d) *Examination integrity*. Applicants, operators and senior operators, and facility licensees must not engage in any activity that compromises the integrity of any application or examination required by §§ 53.760 through 53.795. The integrity of an examination is considered compromised if any activity, regardless of intent, affected, or, but for detection, could have affected the equitable and consistent administration of the examination. This includes activities related to the preparation and certification of applications and all activities related to the preparation, administration, and grading of examinations required by §§ 53.760 through 53.795.

(e) *Simulation facilities.* (1) This section addresses the use of a simulation facility for the administration of examinations, for training, or to demonstrate compliance with experience requirements for applicants for operator and senior operator licenses.

(2) Simulation facilities used for training purposes, for demonstrating compliance with experience requirements, or for the conduct of examinations under § 53.780(b) and (c) must demonstrate compliance with the following criteria as they relate to the facility licensee's reference plant:

(i) The simulation facility must be of sufficient scope and fidelity for individuals to acquire and demonstrate the necessary knowledge, skills, and abilities to safely perform operator and senior operator duties.

(ii) The simulation facility must utilize models relating to nuclear, thermal-hydraulic, and other applicable design-specific characteristics that either replicate the most recent fuel load in the reference commercial nuclear plant or, prior to initial fuel

load, replicate the intended initial fuel load for the reference commercial nuclear plant, with the exception of those portions of the simulation facility that utilize the reference plant itself.

(iii) Simulation facility fidelity must be demonstrated so that significant control manipulations are completed without procedural exceptions, simulator performance exceptions, or deviation from the approved training scenario sequence.

(3) *Continued assurance of simulator fidelity*. Facility licensees that maintain a simulation facility that has been approved by the Commission for training purposes, demonstrating compliance with experience requirements, or the conduct of examinations under § 53.780(b) and (c) for the facility licensee's reference plant must:

(i) Conduct performance testing throughout the life of the simulation facility in a manner sufficient to ensure that paragraph (e)(2) of this section is met;

(ii) Retain the results of performance testing for four years after the completion of each performance test or until superseded by updated test results;

(iii) Promptly correct modeling and hardware discrepancies and discrepancies identified from scenario validation and from performance testing or provide justification as to why the presence of such discrepancies will not adversely affect the criteria of paragraph (e)(2) of this section;

(iv) Make the results of any uncorrected performance test failures that may exist at the time of the initial license examination or requalification examination available for NRC review, prior to or concurrent with preparations for each initial license examination or requalification examination; and

(v) Maintain the provisions for license application and examination integrity consistent with § 53.780(d).

(4) A simulation facility must demonstrate compliance with the requirements of paragraphs (e)(2) and (e)(3) of this section for the Commission to accept the simulation facility for conducting initial examinations as described in § 53.780(b), requalification training as described in § 53.780(c), or performing control manipulations that affect reactivity to establish eligibility for an operator or senior operator license as described in § 53.775(a).

(f) *Waiver of examination requirement.* On application, the Commission may waive any or all of the requirements for an examination if it finds that the applicant has demonstrated the required knowledge, skills, and abilities to safely operate the plant, and is capable of continuing to do so. The Commission may make such a finding based on demonstration of the following:

(1) operating experience at a comparable facility;

(2) proof of the applicant's past competent and safe performance; and

(3) proof of the applicant's current qualifications.

(g) *Proficiency.* The facility licensee must develop, implement, and maintain a proficiency program to ensure that operators and senior operators will actively perform the functions of an operator or senior operator, respectively, as needed to maintain proficiency with on-shift duties and familiarity with plant status. This program must include those steps that will be taken to re-establish proficiency when it cannot be maintained. This program must be approved by the Commission as part of its approval of the operating license or combined license for the plant. The approved proficiency program is subject to the requirements of §§ 53.1565 or 53.6065, as applicable.

§ 53.785 Conditions of operator and senior operator licenses.

Each operator and senior operator license contains and is subject to the following conditions whether stated in the license or not:

(a) Neither the license nor any right under the license may be assigned or otherwise transferred.

(b) The license is limited to the facility for which it is issued.

(c) The license is limited to those controls of the facility(ies) specified in the license.

(d) The license is subject to, and the licensee must observe, all applicable rules, regulations, and orders of the Commission.

(e) The licensee must maintain or re-establish proficiency in accordance with the facility licensee's Commission-approved proficiency program required under § 53.780(g).

(f) The licensee must be subject to the facility's Commission-approved operator licensing requalification and requalification examination programs required under § 53.780(c).

(g) The licensee must have a biennial medical examination as described by § 53.765.

(h) The licensee must notify the Commission within 30 days about a conviction for a felony.

(i) The licensee must not consume or ingest alcoholic beverages within the protected area of commercial nuclear plants. The licensee must not use, possess, or sell any illegal drugs. The licensee must not perform activities authorized by a license issued under this part while under the influence of alcohol or any prescription, over-the-counter, or illegal substance that could adversely affect his or her ability to safely and competently perform his or her licensed duties. For the purpose of this paragraph, with respect to alcoholic beverages and drugs, the term "under the influence" means the

licensee exceeded, as evidenced by a confirmed test result, the lower of the cutoff levels for drugs or alcohol contained in 10 CFR part 26, or as established by the facility licensee. The term "under the influence" also means the licensee could be mentally or physically impaired as a result of substance use including prescription and over-the-counter drugs, as determined under the provisions, policies, and procedures established by the facility licensee for its FFD program, in such a manner as to adversely affect his or her ability to safely and competently perform licensed duties.

(j) Each licensee must participate in the drug and alcohol testing programs as required under 10 CFR part 26.

(k) The licensee must comply with any other conditions that the Commission may impose to protect health or to minimize danger to life or property.

§ 53.790 Issuance, modification, and revocation of operator and senior operator licenses.

(a) *Issuance of operator and senior operator licenses.* If the Commission determines that an applicant for an operator license or a senior operator license demonstrates compliance with the requirements of the AEA and its regulations, it will issue a license in the form and containing any conditions and limitations it considers appropriate and necessary.

(b) *Modification and revocation of operator and senior operator licenses.* (1) The terms and conditions of all operator and senior operator licenses are subject to amendment, revision, or modification by reason of rules, regulations, or orders issued in accordance with the AEA or any amendments thereto.

(2) Any license may be revoked, suspended, or modified, in whole or in part:

(i) For any material false statement in the application or in any statement of fact required under section 182 of the AEA;

(ii) Because of conditions revealed by the application or statement of fact or any report, record, inspection, or other means that would warrant the Commission to refuse to grant a license on an original application;

(iii) For willful violation of, or failure to observe, any of the terms and conditions of the AEA or the license, or of any rule, regulation, or order of the Commission;

(iv) For any conduct determined by the Commission to be a hazard to safe operation of the facility; or

(v) For the sale, use, or possession of illegal drugs, or refusal to participate in the facility drug and alcohol testing program, or a confirmed positive test for drugs, drug metabolites, or alcohol in violation of the conditions and cutoff levels established by § 53.785(i) or the consumption of alcoholic beverages within the protected area of commercial nuclear plants, or a determination of unfitness for scheduled work as a result of the consumption of alcoholic beverages.

§ 53.795 Expiration and renewal of operator and senior operator licenses.

(a) *Expiration*. (1) Each operator license and senior operator license expires six years after the date of issuance, upon termination of employment with the facility licensee, or upon determination by the facility licensee that the licensed individual no longer needs to maintain a license.

(2) If a licensee files an application for renewal or an upgrade of an existing license on Form NRC 398 at least 30 days before the expiration of the existing license, it does not expire until disposition of the application for renewal or for an upgraded license

has been finally determined by the Commission. Filing by mail will be deemed to be complete at the time the application is postmarked

(b) *Renewal*. (1) The applicant for renewal of an operator license or senior operator license must:

(i) Complete and sign Form NRC 398 and include the number of the license for which renewal is sought.

(ii) File an original of NRC Form 398 as specified in § 53.726.

(iii) Provide written evidence of the applicant's experience under the existing license and the approximate number of hours that the licensee has operated the facility.

(iv) Provide a statement by an authorized representative of the facility licensee that during the effective term of the current license the applicant has satisfactorily completed the requalification program for the facility for which operator or senior operator license renewal is sought.

(v) Provide evidence that the applicant has discharged the license responsibilities competently and safely. The Commission may accept as evidence of the applicant's having met this requirement a certificate of an authorized representative of the facility licensee or holder of an authorization by which the licensee has been employed.

(vi) Provide certification by the facility licensee of medical condition and general health on Form NRC 396, to comply with § 53.765.

(2) The license will be renewed if the Commission finds that:

(i) The medical condition and the general health of the licensee continue to be such as not to cause operational errors that endanger public health and safety. The Commission will base this finding upon the certification by the facility licensee as described in § 53.765(b).

(ii) The licensee –

(A) Is capable of continuing to competently and safely assume licensed duties;

(B) Has successfully completed a requalification program that has been approved by the Commission as required by § 53.780(c); and

(C) Has passed the requalification examinations as required by § 53.780(c).

(iii) There is a continued need for an operator to operate or for a senior operator to supervise operators at the facility designated in the application.

(iv) The past performance of the licensee has been satisfactory to the Commission. In making its finding, the Commission will include in its evaluation information such as notices of violations or letters of reprimand in the licensee's docket.

§ 53.800 Facility licensees that comply with §§ 53.800 through 53.820.

(a) A commercial nuclear plant is a self-reliant mitigation facility if the NRC determined as part of its approval of the operating license or combined license for that plant that its design demonstrates compliance with the criteria of either paragraph (a)(1) or (a)(2) of this section, as applicable. A self-reliant mitigation facility is of a class, based upon the similarity of operating and technical characteristics of the plants in the class, that its licensee must comply with the requirements of §§ 53.800 through 53.820 in lieu of those in §§ 53.760 through 53.795.

(1) The criteria applicable under Framework A of this part are that:

(i) The safety criteria of §§ 53.210 and 53.220 and, if applicable, any alternative criteria used in accordance with § 53.470 must be met without reliance on human actions for event mitigation;

(ii) The safety functions of § 53.230 must be achieved without reliance on human actions for credited event mitigation;

(iii) The requirements associated with defense in depth, as described under § 53.250, must be met without reliance on human actions for event mitigation for the purposes of credited defense in depth;

(iv) The analysis of LBEs in accordance with § 53.450(e) must demonstrate that the evaluation criteria for each event sequence will be met and the analysis of design-basis accidents in accordance with § 53.450(f) must demonstrate that the evaluation criteria will be met, respectively, without reliance on human actions for credited event mitigation; and

(v) The plant response to LBEs must not credibly rely on human actions to assure the performance of SSCs. Compliance with this criterion may be achieved through the use of SSCs that function through inherent characteristics or have engineered protections against human failures.

(2) The criteria applicable under Framework B of this part are that:

(i) The plant design must provide for layered defense in depth without dependence upon any single barrier or reliance upon credited human action, and either:

(ii) For those commercial nuclear plants that do not demonstrate compliance with the criteria in § 53.4730(a)(34)(ii):

(A) The safety assessment performed under § 53.4730(a)(1)(vi) must demonstrate that evaluation requirements will be met without reliance on credited human action;

(B) The Probabilistic Risk Analysis performed under § 53.4730(a)(34)(i) must demonstrate that the evaluation criteria for each event sequence will be met without reliance on human actions for credited event mitigation;

(C) The Functional Requirements Analysis and Function Allocation performed under § 53.730(d) must demonstrate that functions required for safety are not reliant upon credited human action; and

(D) The plant response to LBEs must not credibly rely on human actions to assure the performance of SSCs. Compliance with this criterion may be achieved through the use of SSCs that function through inherent characteristics or have engineered protections against human failures.

(iii) For those commercial nuclear plants performing Alternative Evaluations for Risk Insights, the qualification for Alternative Evaluations for Risk Insights in § 53.4730(a)(34)(ii) must be demonstrated to be met.

(b) Reserved.

§ 53.805 Facility licensee requirements related to generally licensed reactor operators

(a) Licensees for self-reliant mitigation facilities that have not yet certified the permanent cessation of operations and permanent removal of fuel from the reactor vessel as described under §§ 53.1070 or 53.4670, as applicable, must demonstrate compliance with the following requirements:

(1) Ensure that, in addition to performing those items identified by the facility-specific systems approach to training conducted under § 53.815, generally licensed reactor operators are qualified to safely and competently:

(i) perform administrative tasks, including compliance with technical specifications, and perform operability determinations;

(ii) implement maintenance and configuration controls;

(iii) comply with radioactive release limitations;

(iv) understand plant operating data, including reactor parameters, and evaluate emergency conditions;

(v) initiate a reactor shutdown from necessary locations;

(vi) dispatch and direct operations and maintenance personnel;

(vii) implement any applicable responsibilities under the facility emergency plan; and

(viii) make required notifications to local, State, and Federal authorities.

(2) Develop, implement, and maintain facility technical specifications that provide the necessary administrative controls to ensure the implementation of these requirements.

(3) Develop, implement, and maintain the generally licensed reactor operator training, examination, and proficiency programs required under § 53.815.

(4) Ensure that generally licensed reactor operators are subject to the facility's generally licensed reactor operator training, examination, and proficiency programs required under § 53.815. Ensure that generally licensed reactor operators are subject to and comply with the applicable programmatic requirements for plant personnel required under 10 CFR parts 26 and 73. An individual that is not in compliance with any of these programs is not qualified to be in a position that may involve the manipulation of the controls of the commercial nuclear plant.

(5) Report annually to the NRC the identity of all generally licensed reactor operators at the commercial nuclear plant, including all additions and deletions since the previous report.

(b) Reserved.

§ 53.810 Generally licensed reactor operators.

(a) A general license to manipulate the controls of a self-reliant mitigation facility and to direct the licensed activities of generally licensed reactor operators is hereby issued to any individual employed in a position that may involve the manipulation of the controls of that self-reliant mitigation facility and who observes the restrictions of this section.

(b) A generally licensed reactor operator must observe the operating procedures and other conditions specified in the license authorizing operation of the facility.

(c) The general license is limited to the facility or facilities at which the operator is employed.

(d) The Commission will suspend the general license on an individual basis for violations of any provision of the AEA or any rule or regulation issued thereunder whenever the Commission deems such action desirable, including:

(1) For willful violation of, or failure to observe, any of the terms and conditions of the AEA or the general license, or of any rule, regulation, or order of the Commission;

(2) For any conduct determined by the Commission to be a hazard to safe operation of the facility; or

(3) For the sale, use, or possession of illegal drugs, or refusal to participate in the facility drug and alcohol testing program, or a confirmed positive test for drugs, drug metabolites, or alcohol in violation of the conditions and cutoff levels established by § 53.810(f) or the consumption of alcoholic beverages within the protected area of commercial nuclear plants, or a determination of unfitness for scheduled work as a result of the consumption of alcoholic beverages.

(e) The Commission may require information from a generally licensed reactor operator to determine whether a general license should be revoked or suspended with respect to that operator.

(f) The generally licensed reactor operator must not consume or ingest alcoholic beverages within the protected area of commercial nuclear plants. The generally licensed reactor operator must not use, possess, or sell any illegal drugs. The generally licensed reactor operator must not perform activities requiring a general license while under the influence of alcohol or any prescription, over-the-counter, or illegal substance that could adversely affect his or her ability to safely and competently perform these activities. For the purpose of this paragraph, with respect to alcoholic beverages and drugs, the term "under the influence" means the generally licensed reactor operator exceeded, as evidenced by a confirmed test result, the lower of the cutoff levels for drugs or alcohol contained in 10 CFR part 26 of this chapter, or as established by the facility licensee. The term "under the influence" also means the generally licensed reactor operator could be mentally or physically impaired as a result of substance use including prescription and over-the-counter drugs, as determined under the provisions, policies, and procedures established by the facility licensee for its FFD program, in such a manner as to adversely affect his or her ability to safely and competently perform generally licensed reactor operator duties.

(g) The generally licensed reactor operator must notify the Commission within 30 days about a conviction for a felony.

§ 53.815 Generally licensed reactor operator training, examination, and proficiency programs.

(a) Applicability. The requirements of this section apply to each licensee of a self-reliant mitigation facility that has not yet certified the permanent cessation of operations and permanent removal of fuel from the reactor vessel as described under §§ 53.1070 or 53.4670, as applicable.

(b) Requirements. (1) The licensee must develop, implement, and maintain training and examination programs that demonstrate compliance with the requirements of paragraphs (b)(2) through (b)(3) of this section.

(2) The training program must provide for both the initial and continuing training of generally licensed reactor operators and be derived from a systems approach to training as defined in this part.

(3)(i) The training program must incorporate the instructional requirements necessary to provide qualified generally licensed reactor operators to operate and maintain the facility in a safe manner in all modes of operation. The training program must comply with the facility license, including all technical specifications and applicable regulations. The facility licensee must periodically evaluate and revise the training program as appropriate to reflect industry experience and relevant changes, including changes to the facility, procedures, regulations, and quality assurance requirements. Facility licensee management must periodically review the training program for effectiveness.

(ii) The training program must ensure that generally licensed reactor operators have and maintain the necessary knowledge, skills, and abilities.

(iii) The training program must include the generally licensed reactor operator manipulating the controls of either the facility or a simulation facility that demonstrates compliance with the requirements of § 53.815(e).

(iv) The training program must include an initial examination program for testing a representative sample of the knowledge, skills, and abilities needed to safely perform generally licensed reactor operator duties, to include both the examination methods and criteria to be used to assess passing performance. The facility licensee must provide the

opportunity for a representative of the Commission to be present during initial examination administration.

(v) The training program must include a requalification examination program for testing a sample of the topics included under the systems approach to training, to include the examination methods and criteria to be used to assess passing performance. The requalification examination program must specify an appropriate periodicity for administering a complete requalification examination to each generally licensed reactor operator, and the facility licensee must provide the opportunity for a representative of the Commission to be present during requalification examination administration.

(A) The facility licensee shall ensure that any generally licensed reactor operator who either demonstrates unsatisfactory performance on, or fails to complete, the requalification examination is removed from the performance of generally licensed reactor operator duties until such time that any necessary remedial training has been completed and a retake examination has been passed.

(vi) The training program must be approved by the Commission prior to its use, as described under § 53.730(g). The examination program must provide for valid and reliable examinations and must be approved by the Commission prior to their use, as described under § 53.730(g). The approved programs are subject to the requirements of §§ 53.1565 or 53.6065, as applicable.

(c) *Records.* The following is required regarding the documentation of the generally licensed reactor operator training and examination programs:

(1) Sufficient records must be maintained by the facility licensee to maintain the integrity of the programs and kept available for NRC inspection to verify the adequacy of the programs.

(2) The facility licensee must maintain records documenting the participation of each generally licensed reactor operator in the training and examination programs. The records must contain copies of examinations administered, the answers given by the generally licensed reactor operator, documentation of the grading of examinations, and documentation of any additional training administered in areas in which a generally licensed reactor operator exhibited deficiencies. The facility licensee must retain these records while the associated generally licensed reactor operators remain employed at the facility.

(3) Each record required by this part must be legible throughout the retention period. The record may be the original, a reproduced copy, or an electronic copy provided that the copy is authenticated by authorized personnel.

(d) *Examination integrity.* Generally licensed reactor operators and facility licensees must not engage in any activity that compromises the integrity of any examination conducted under the generally licensed reactor operator training and examination programs. The integrity of an examination is considered compromised if any activity, regardless of intent, affected, or, but for detection, could have affected the equitable and consistent administration of the examination. This includes all activities related to the preparation, administration, and grading of examinations.

(e) *Simulation facilities.* (1) Simulation facilities used for training purposes, for maintaining proficiency, or for the conduct of examinations must demonstrate compliance with the following criteria as they relate to the facility licensee’s reference plant:

(i) The simulation facility must be of sufficient scope and fidelity for individuals to acquire and demonstrate the necessary knowledge, skills, and abilities to safely perform generally licensed reactor operator duties.

(ii) The simulation facility must utilize models relating to nuclear, thermal-hydraulic, and other applicable design-specific characteristics that either replicate the most recent fuel load in the reference commercial nuclear plant or, prior to initial fuel load, replicate the intended initial fuel load for the reference commercial nuclear plant, with the exception of those portions of the simulation facility that utilize the reference plant itself. (iii) Simulator fidelity must be demonstrated so that significant control manipulations are completed without procedural exceptions, simulator performance exceptions, or deviation from the approved training scenario sequence.

(2) *Continued assurance of simulator fidelity*. Facility licensees that maintain a simulation facility for training purposes, for maintaining proficiency, or for the conduct of examinations must:

(i) Conduct performance testing throughout the life of the simulation facility in a manner sufficient to ensure that paragraph (e)(1) of this section is met;

(ii) Retain the results of performance testing for four years after the completion of each performance test or until superseded by updated test results;

(iii) Promptly correct modeling and hardware discrepancies and discrepancies identified from scenario validation and from performance testing or provide justification for why the presence of such discrepancies will not adversely affect the criteria of paragraph (e)(1) of this section;

(iv) Make the results of any uncorrected performance test failures that may exist at the time of an inspection available for NRC review; and

(v) Maintain the provisions for examination integrity consistent with § 53.815(d).

(f) *Waiver of examination requirement*. The facility licensee may waive any or all of the requirements for an examination in accordance with the facility licensee's

Commission-approved generally licensed reactor operator training and examination programs.

(g) *Proficiency.* The facility licensee must develop, implement, and maintain a proficiency program to ensure that generally licensed reactor operators will maintain proficiency regarding position functions and familiarity with plant status. This program must include those steps that will be taken in order to re-establish proficiency when it cannot be maintained.

§ 53.820 Expiration.

The general license expires upon a generally licensed reactor operator no longer being employed in a position that may involve the manipulation of the controls of the self-reliant mitigation facility.

§ 53.830 Training and qualification of commercial nuclear plant personnel.

(a) This section addresses personnel training requirements. The regulations within this section are applicable to all applicants for or holders of operating licenses or combined licenses under this part.

(b) Prior to fuel load, each holder of an operating or combined license under this part must, with sufficient time to provide trained and qualified personnel to operate the facility, establish, implement, and maintain a training program that demonstrates compliance with the requirements of paragraphs (c) and (d) of this section.

(c) The training program must be derived from a systems approach to training as defined in this part and must provide, at a minimum, for the training and qualification of the following categories of commercial nuclear plant personnel:

(1) supervisors (e.g., shift supervisors);

(2) technicians (e.g., maintenance, chemistry, and radiological); and

(3) other appropriate operating personnel (e.g., auxiliary operators, certified fuel handlers, and individuals who provide engineering expertise to on-shift operating personnel).

(d) The training program must incorporate the instructional requirements necessary to provide qualified personnel to operate components of a commercial nuclear plant and maintain the facility in a safe manner in all modes of operation. The training program must be developed to be in compliance with the facility license, including all technical specifications and applicable regulations.

(1) The training program must be periodically evaluated and revised as appropriate to reflect industry experience and relevant changes, including changes to the facility, procedures, regulations, and quality assurance requirements. The training program must be periodically reviewed by facility licensee management for effectiveness.

(2) Sufficient records must be maintained by the facility licensee to maintain program integrity and kept available for NRC inspection to verify the adequacy of the training program.

§ 53.845 Programs.

Programs must be provided for each commercial nuclear plant such that, when combined with associated design features and human actions, the plant will demonstrate compliance with the safety criteria defined in §§ 53.210 and 53.220. Programs must also support continued assurance that the safety functions identified in § 53.230 are maintained during normal operations and LBEs. The required plant programs must include but are not necessarily limited to the programs described in the following

sections of this subpart. Licensees may combine, separate, and otherwise organize programs and related documents as appropriate for the technologies and organizations associated with the commercial nuclear plant.

§ 53.850 Radiation protection.

(a) Each holder of an operating license or combined license under this Framework A of this part must develop, implement, and maintain a Radiation Protection Program for operations that is commensurate with the scope and extent of licensed activities under this part and includes measures for limiting and monitoring radioactive plant effluents and limiting and monitoring the dose to individuals working with radioactive materials in accordance with 10 CFR part 20.

(b) Each licensee under Framework A of this part must develop, implement, and maintain a program for the control of radioactive effluents and for keeping the doses to members of the public from radioactive effluents as low as is reasonably achievable. The program must be contained in an Offsite Dose Calculations Manual (ODCM), must be implemented by procedures, and must include remedial actions to be taken whenever the program limits are exceeded. The ODCM must:

(1) Contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

(2) Contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by § 53.1645.

(c) Each licensee under Framework A of this part must develop, implement, and maintain a Process Control Program that identifies the administrative and operational controls for solid radioactive waste processing, process parameters, and surveillance requirements sufficient to ensure compliance with the requirements of 10 CFR part 20, 10 CFR part 61, and 10 CFR part 71.

§ 53.855 Emergency preparedness.

(a) Each holder of an operating license or combined license under this Framework A of this part must develop, implement, and maintain an emergency response plan that ensures that adequate protective measures can and will be taken in the event of a radiological emergency.

(b) The emergency response plan must contain information needed to demonstrate compliance with the requirements in appendix E to 10 CFR part 50 and the applicable standards in 10 CFR 50.47.

§ 53.860 Security program.

(a) *Physical Protection Program.* Each holder of an operating license or combined license under this Framework A of this part must develop, implement, and maintain a physical protection program demonstrating compliance with the following requirements:

(1) The licensee must implement security requirements for the protection of special nuclear material based on the type, enrichment, and quantity in accordance with 10 CFR part 73, as applicable, and implement security requirements for the protection of Category 1 and Category 2 quantities of radioactive material in accordance with 10 CFR part 37, as applicable; and

(2) The licensee must demonstrate compliance with the provisions set forth in either §§ 73.55 or 73.100, unless the licensee demonstrates compliance with the following criterion:

(i) The radiological consequences from a design-basis threat initiated event involving the loss of engineered systems for decay heat removal and possible breaches in physical structures surrounding the reactor, spent fuel, and other inventories of radioactive materials result in offsite doses below the values in § 53.210.

(ii) The applicant must perform a site-specific analysis, including identification of target sets, to demonstrate that the criterion in § 53.860(a)(2)(i) is met. The analysis must assume that licensee mitigation and recovery actions, including any operator action, are unavailable or ineffective. The licensee must maintain the analysis until the permanent cessation of operations and permanent removal of fuel from the reactor vessel as described under § 53.1070.

(b) *Fitness for Duty*. Each holder of an operating license or combined license under Framework A of this part must develop, implement, and maintain an FFD program that demonstrates compliance with the requirements in 10 CFR part 26.

(c) *Access Authorization*. Each holder of an operating license or combined license under this Framework A of this part must develop, implement, and maintain an access authorization program that demonstrates compliance with the requirements in § 73.120 if the criterion in § 53.860(a)(2)(i) is met, or the requirements in § 73.56 if the criterion is not met.

(d) *Cybersecurity*. Each holder of an operating license or combined license under Framework A of this part must develop, implement, and maintain a cybersecurity program that demonstrates compliance with the requirements in §§ 73.54 or 73.110.

(e) *Information Security*. Each holder of an operating license or combined license under Framework A of this part must develop, implement, and maintain an information protection system that demonstrates compliance with the requirements of §§ 73.21, 73.22, and 73.23, as applicable.

§ 53.865 Quality assurance.

Each holder of an operating license or combined license under Framework A of this part must develop, implement, and maintain a quality assurance program (QAP) in accordance with subpart K of this part. A written QAP manual must be developed and used to guide the conduct of the program in accordance with generally accepted consensus codes and standards that have been endorsed or otherwise found acceptable by the NRC.

§ 53.870 Integrity assessment programs.

Each holder of an operating license or combined license under Framework A of this part must develop, implement, and maintain an integrity assessment program to monitor, evaluate, and manage:

(a) The effects of plant aging on SR and NSRSS SSCs. The program may refer to surveillances, tests, and inspections conducted for specific SSCs in accordance with other requirements in this part or conducted in accordance with applicable consensus codes and standards endorsed or otherwise found acceptable by the NRC;

(b) Cyclic or transient load limits to ensure that SR and NSRSS SSCs are maintained within the applicable design limits; and

(c) Degradation mechanisms related to chemical interactions, operating temperatures, effects of irradiation, and other environmental factors to ensure that the

capabilities, availability, and reliability of SR and NSRSS SSCs demonstrate compliance with the functional design criteria of §§ 53.410 and 53.420.

§ 53.875 Fire protection.

(a)(1) Each holder of an operating license or combined license under Framework A of this part must have a fire protection plan that describes the overall fire protection program for the facility, identifies the various positions within the licensee's organization that are responsible for the program, states the authorities that are delegated to each of these positions to implement those responsibilities, and outlines the plans for fire protection, fire detection and suppression capability, and limitation of fire damage.

(2) The fire protection plan must also describe specific features necessary to implement the program described in paragraph (a)(1) of this section such as: administrative controls and personnel requirements for fire prevention and manual fire suppression activities; automatic and manually operated fire detection and suppression systems; and the means to limit fire damage to SR and NSRSS SSCs so that the capability to demonstrate compliance with the requirements of § 53.210 is ensured.

(b)(1) Each holder of an operating license or combined license under Framework A of this part must develop a performance-based or deterministic fire protection program that demonstrates compliance with the safety criteria outlined in §§ 53.210 and 53.220, related safety functions outlined in § 53.230, and defense in depth as outlined in § 53.250 with specific fire protection measures related to fire prevention, fire detection, and fire suppression.

(2) The fire protection program must comply with the following:

(i) SR and NSRSS SSCs must be designed, located, and maintained to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

(ii) Noncombustible and fire-resistant materials must be used wherever practical throughout the facility, particularly in locations with SR and NSRSS SSCs.

(iii) Fire detection and fire suppression systems of appropriate capacity and capability must be provided, and designed and maintained to minimize the adverse effects of fires on SR and NSRSS SSCs.

(iv) Fire suppression systems must be designed and maintained to ensure that their rupture or inadvertent operation does not significantly impair the ability of SR and NSRSS SSCs to perform their safety functions to satisfy § 53.230.

§ 53.880 Inservice inspection and inservice testing.

(a) Each holder of an operating license or combined license under Framework A of this part must develop, implement, and maintain a program for inservice inspection (ISI) and inservice testing (IST) prior to receiving an operating license or combined license. The ISI/IST program must include all inspections and tests required by the codes and standards used in the design and be supplemented by risk insights that identify the most important SSCs to plant safety. The types of testing and inspections and their frequency should be informed by risk insights to maintain the reliability and performance of SSCs consistent with the associated design and analyses activities involving those SSCs. Risk insights must also be used to determine when to conduct the inspections and tests (e.g., full power, shutdown, refueling) to minimize risk to the plant workers and the public. The ISI/IST program must be documented in a written manual and managed by qualified personnel reporting to the Plant Manager.

(b) Prior to plant operation, baseline inspections and testing must be performed using the same techniques as will be used for future inspections and testing. The results of these inspections and testing must be used as benchmarks for evaluating the results of future inspections and testing. Sufficient room and support must be provided to accommodate the personnel, ISI/IST equipment, and shielding necessary to perform the inspections and testing. Acceptance criteria for determining whether corrective action is needed must be developed (or taken from the codes and standards used in the design) for evaluating the results of the inspections and testing. The results of the inspections and testing must be provided to the Plant Manager who is responsible for determining what, if any, corrective action is needed and when it should be taken. The ISI/IST results and corrective actions must be documented and the documentation retained for the life of the plant.

§ 53.890 Facility safety program.

(a) Each holder of an operating license or combined license under Framework A of this part must develop, implement, and maintain an FSP that includes a risk-informed, performance-based process to identify new or revised internal or external hazards to the facility and performance issues related to plant SSCs, programmatic controls, and human actions; assess changes in the risks posed to the public from the commercial nuclear plant; and, when appropriate, consider measures to mitigate or eliminate the resulting risks.

(b)(1) Each licensee must implement risk reduction measures as may be appropriate when considering potential risks to public health and safety, technology changes, economic costs, operating experience, new or revised hazard assessments, or other factors included in the FSP plan required by paragraph (c) of this section.

(2) Each licensee must develop, implement, and maintain a process to routinely assess potential changes in contributors to plant risks and identify when to assess potential risk reduction measures related to internal and external events, identified hazards, or other specific contributors to the overall cumulative risk from unplanned events. The process must, at a minimum, assess potential risk reduction measures when:

(i) the estimated frequency-weighted cumulative dose to nearby populations increases by 5 person-rem due to a new or revised hazard or could be decreased by 5 person-rem by a risk reduction measure; and

(ii) there is a significant reduction in or possibility to significantly increase the calculated margins between the frequency and consequences of LBEs and the evaluation criteria in § 53.450(e) or more restrictive alternate criteria adopted under § 53.470.

(3) Possible risk reduction measures for commercial nuclear plants whose licenses refer to certified designs or manufacturing licenses must also follow the change control and reporting provisions of subpart I related to changes to standardized designs. Licensees need not pursue risk reduction measures under this section if the only cost-effective measures would require a license amendment or exemption under subpart I due to references to a certified design or manufacturing license.

(c)(1) Each licensee subject to this section must adopt and implement an FSP under paragraph (b) of this section by using a written FSP plan that, at a minimum, describes the facility or facilities covered by the plan; facility environs that influence the assessments; and how the FSP addresses the role of and operating experience with

SSCs, personnel, and programmatic controls supporting the safety functions required by § 53.230.

(2) Each FSP plan must also describe:

(i) the methods used to identify and analyze current, new, or novel technologies that will mitigate or eliminate internal or external hazards and resulting risks from the release of radioactive materials;

(ii) the licensee's overall safety philosophy and intended safety culture to be practiced by its management, employees, and contractors;

(iii) the required participants in the FSP, which will include managers, employees, and contractors that directly support facility operations; maintain, inspect, or change plant SSCs or programmatic controls; or assess potential risk reduction measures;

(iv) the FSP-related training program; and

(v) periodic reviews of the effectiveness of the FSP and its implementation.

(d) The NRC will review the FSP plan as part of the applications for an operating license or combined license under subpart H. Approval of an FSP plan under Framework A of this part does not constitute approval of the specific actions the licensee will implement under its FSP plan pursuant to this section and must not be construed as establishing an NRC standard regarding those specific actions.

(e) Updates and revisions to the FSP plan must be provided biennially or more frequently in accordance with § 53.1560 and the licensee must obtain NRC approval of a proposed change in accordance with § 53.1565 if its implementation requires an exemption from the requirements in this section.

§ 53.910 Procedures and guidelines.

(a) Each holder of an operating license or combined license under this Framework A of this part must have a program for developing, implementing, and maintaining an integrated set of procedures, guidelines, and related supporting activities to support normal operations and respond to possible unplanned events.

(b) The program required by paragraph (a) of this section must include but is not limited to development, implementation, maintenance, and supporting activities of procedures and guidelines for the following:

(1) Plant operations;

(2) Maintenance activities under § 53.715;

(3) Program requirements under this subpart;

(4) Emergency operating procedures, if developed to address the role of human actions in responding to LBEs;

(5) Accident management guidelines, if developed to address the role of human actions in responding to LBEs;

(6) Procedures for each area in which licensed special nuclear material is handled, used, or stored to protect personnel upon the sounding of a criticality alarm required by § 53.440(m); and

(7) Procedures that describe how the licensee will address the following areas if the licensee is notified of a potential aircraft threat:

(i) Verification of the authenticity of threat notifications;

(ii) Maintenance of continuous communication with threat notification sources;

(iii) Contacting all onsite personnel and applicable offsite response organizations;

(iv) Onsite actions necessary to enhance the capability of the facility to mitigate the consequences of an aircraft impact;

(v) Measures to reduce visual discrimination of the site relative to its surroundings or individual buildings within the protected area;

(vi) Dispersal of equipment and personnel, as well as rapid entry into site protected areas for essential onsite personnel and offsite responders who are necessary to mitigate the event; and

(vii) Recall of site personnel.

Subpart G—Decommissioning Requirements

§ 53.1000 Scope and purpose.

This subpart defines the requirements related to decommissioning for applicants for, or holders of, an operating license or combined license under Framework A. The requirements related to maintaining financial assurance for decommissioning are in §§ 53.1010 through 53.1060. The requirements for transitioning from operations to decommissioning and for the release of property and termination of the license are in §§ 53.1070 through 53.1080.

§ 53.1010 Financial assurance for decommissioning.

(a) This section establishes requirements for indicating to the NRC how an applicant for or holder of an operating license or combined license under Framework A of this part will provide reasonable assurance that funds will be available for the decommissioning process. Reasonable assurance consists of a series of steps as provided in paragraph (b) of this section and §§ 53.1020, 53.1030 and 53.1040. Funding for the decommissioning of commercial nuclear plants may also be subject to the regulation of Federal or State government agencies (e.g., Federal Energy Regulatory

Commission (FERC) and State Public Utility Commissions (PUC)) that have jurisdiction over rate regulation. The requirements of this subpart, in particular § 53.1020, are in addition to, and not a substitution for, other requirements, and are not intended to be used by themselves or by other agencies to establish rates.

(b) Each applicant for an operating license or combined license under Framework A of this part must prepare a plan and an associated decommissioning report that ensures and documents that adequate funding will be available to decommission the facility. Each holder of an operating license or combined license must implement and maintain the plan.

(1)(i) Before the Commission issues an operating license under Framework A of this part, the applicant must update the decommissioning report to certify that it has provided financial assurance for decommissioning in the amount proposed in the application and approved by the NRC in accordance with § 53.1020.

(ii) No later than 30 days after the Commission issues the notice of intended operation under § 53.1452 for a combined license under Framework A of this part, the licensee must update the decommissioning report to certify that it has provided financial assurance for decommissioning in the amount proposed in the application and approved by the NRC in accordance with § 53.1020.

(2) The amount of financial assurance for decommissioning to be provided must be based on a site-specific cost estimate for decommissioning the facility in accordance with § 53.1020.

(3) The amount of financial assurance for decommissioning to be provided must be adjusted annually using a rate at least equal to that stated in § 53.1030.

(4) The amount of financial assurance for decommissioning to be provided must be covered by one or more of the methods described in § 53.1040 as acceptable to the

NRC. A copy of the financial instrument obtained to satisfy the requirements of § 53.1040 must be submitted to the NRC as part of the application for an operating license or combined license under Framework A of this part.

§ 53.1020 Cost estimates for decommissioning.

Cost estimates for decommissioning must be site-specific. Site-specific decommissioning cost estimates (DCEs) must account for the engineering, labor, equipment, transportation, disposal, and related charges needed to support termination of the license. They must include the costs for decontaminating structures, systems, and components and the site environs; removal of contaminated components and materials from the plant and the site environs; disposal of removed components and materials in appropriate facilities; and any other activities supporting the release of the property and termination of the license. They must also address the approach to annual adjustments required by § 53.1030. Finally, site-specific DCEs must include plans for adjusting levels of funds assured for decommissioning to demonstrate that a reasonable level of assurance will be provided that funds will be available when needed to cover the cost of decommissioning.

§ 53.1030 Annual adjustments to cost estimates for decommissioning.

Each holder of an operating license or combined license under Framework A of this part must annually adjust the cost estimate for decommissioning to account for escalation in labor, energy, and waste burial costs. Licensees may elect to use either a site-specific adjustment factor, approved as part of the plan and associated decommissioning report required by § 53.1010, in paragraph (a) of this section or the generic adjustment factor in paragraph (b) of this section.

(a) A site-specific adjustment factor must address the estimated contributions and escalation of costs for the following aspects of decommissioning:

(1) labor, materials, and services;

(2) energy and waste transportation; and

(3) radioactive waste burial or other disposition.

(b) A generic adjustment factor must be at least equal to 0.65 L + 0.13 E + 0.22 B, where L and E are escalation factors for labor and energy, respectively, and are to be taken from regional data of U.S. Department of Labor Bureau of Labor Statistics and B is an escalation factor for waste burial and is to be taken from NRC report NUREG-1307, "Report on Waste Burial Charges."

§ 53.1040 Methods for providing financial assurance for decommissioning.

Financial assurance for decommissioning is to be provided by the following methods.

(a) *Prepayment*. Prepayment is the deposit made preceding the start of operation or the transfer of a license under § 53.1570 into an account segregated from licensee assets and outside the administrative control of the licensee and its subsidiaries or affiliates of cash or liquid assets such that the amount of funds would be sufficient to pay decommissioning costs. Prepayment may be in the form of a trust, escrow account, or Government fund with payment by certificate of deposit, deposit of government or other securities, or other method acceptable to the NRC. This trust, escrow account, Government fund, or other type of agreement shall be established in writing and maintained at all times in the United States with an entity that is an appropriate State or Federal government agency, or an entity whose operations in which the prepayment deposit is managed are regulated and examined by a Federal or State agency. A

licensee that has prepaid funds based on a site-specific cost estimate under § 53.1020 may take credit for projected earnings on the prepaid decommissioning trust funds, using up to a 2 percent annual real rate of return through the time of termination of the license. A licensee may use a credit of greater than 2 percent if the licensee's rate-setting authority has specifically authorized a higher rate. Actual earnings on existing funds may be used to calculate future fund needs.

(b) *External sinking fund*. An external sinking fund is a fund established and maintained by setting funds aside periodically in an account segregated from licensee assets and outside the administrative control of the licensee and its subsidiaries or affiliates in which the total amount of funds would be sufficient to pay decommissioning costs. An external sinking fund may be in the form of a trust, escrow account, or Government fund, with payment by certificate of deposit, deposit of Government or other securities, or other method acceptable to the NRC. This trust, escrow account, Government fund, or other type of agreement shall be established in writing and maintained at all times in the United States with an entity that is an appropriate State or Federal government agency, or an entity whose operations in which the external sinking fund is managed are regulated and examined by a Federal or State agency. A licensee that has collected funds based on a site-specific cost estimate under § 53.1020 may take credit for projected earnings on the external sinking funds using up to a 2 percent annual real rate of return from the time of future funds' collection through the time of termination of the license. A licensee may use a credit of greater than 2 percent if the licensee's rate-setting authority has specifically authorized a higher rate. Actual earnings on existing funds may be used to calculate future fund needs. A licensee whose rates for decommissioning costs cover only a portion of these costs may make use of this method only for the portion of these costs that are collected in one of the manners described in

this paragraph. This method may be used as the exclusive mechanism relied upon for providing financial assurance for decommissioning in the following circumstances:

(1) By a licensee that recovers, either directly or indirectly, the estimated total cost of decommissioning through rates established by "cost of service" or similar ratemaking regulation. Public utility districts, municipalities, rural electric cooperatives, and State and Federal agencies, including associations of any of the foregoing, that establish their own rates and are able to recover their cost of service allocable to decommissioning, are deemed to satisfy this condition.

(2) By a licensee whose source of revenues for its external sinking fund is a "non-bypassable charge," the total amount of which will provide funds estimated to be needed for decommissioning pursuant to §§ 53.1020, 53.1060, or 53.1575.

(c) *A surety method, insurance, or other guarantee method*.

(1) These methods guarantee that decommissioning costs will be paid. A surety method may be in the form of a surety bond, or letter of credit. Any surety method or insurance used to provide financial assurance for decommissioning must contain the following conditions:

(i) The surety method or insurance must be open-ended, or, if written for a specified term, such as 5 years, must be renewed automatically, unless 90 days or more prior to the renewal day the issuer notifies the NRC, the beneficiary, and the licensee of its intention not to renew. The surety or insurance must also provide that the full-face amount be paid to the beneficiary automatically prior to the expiration without proof of forfeiture if the licensee fails to provide a replacement acceptable to the NRC within 30 days after receipt of notification of cancellation.

(ii) The surety or insurance must be payable to a trust established for decommissioning costs. The trustee and trust must be acceptable to the NRC. An

acceptable trustee includes an appropriate State or Federal government agency or an entity that has the authority to act as a trustee and whose trust operations are regulated and examined by a Federal or State agency.

(2) A parent company guarantee of funds for decommissioning costs based on a financial test may be used if the guarantee and test are as contained in appendix A to 10 CFR part 30.

(3) For commercial companies that issue bonds, a guarantee of funds by the applicant or licensee for decommissioning costs based on a financial test may be used if the guarantee and test are as contained in appendix C to 10 CFR part 30. For commercial companies that do not issue bonds, a guarantee of funds by the applicant or licensee for decommissioning costs may be used if the guarantee and test are as contained in appendix D to 10 CFR part 30. A guarantee by the applicant or licensee may not be used in any situation in which the applicant or licensee has a parent company holding majority control of voting stock of the company.

(d) For a Federal licensee, a statement of intent containing a cost estimate for decommissioning and indicating that funds for decommissioning will be obtained when necessary.

(e) Contractual obligation(s) on the part of a licensee's customer(s), the total amount of which over the duration of the contract(s) will provide the licensee's total share of uncollected funds estimated to be needed for decommissioning pursuant to §§ 53.1020, 53.1060, or 53.1575. To be acceptable to the NRC as a method of decommissioning funding assurance, the terms of the contract(s) shall include provisions that the buyer(s) of electricity or other products will pay for the decommissioning obligations specified in the contract(s), notwithstanding the operational status either of the licensed plant to which the contract(s) pertains or force majeure provisions. All

proceeds from the contract(s) for decommissioning funding will be deposited to the external sinking fund. The NRC reserves the right to evaluate the terms of any contract(s) and the financial qualifications of the contracting entity or entities offered as assurance for decommissioning funding.

(f) Any other mechanism, or combination of mechanisms, that provides, as determined by the NRC upon its evaluation of the specific circumstances of each licensee submittal, assurance of decommissioning funding equivalent to that provided by the mechanisms specified in paragraphs (a) through (e) of this section. Licensees who do not have sources of funding described in paragraph (b) of this section may use an external sinking fund in combination with a guarantee mechanism, as specified in paragraph (c) of this section, provided that the total amount of funds estimated to be necessary for decommissioning is assured.

§ 53.1045 Limitations on the use of decommissioning trust funds.

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

(i) The withdrawals are for expenses for decommissioning activities consistent with the definition of decommissioning in § 53.020;

(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 amount determined in accordance with § 53.1020 may be used for decommissioning planning. For licensees that have submitted the

certifications required under § 53.1575 and commencing 90 days after the NRC has received the post-shutdown decommissioning activities report (PSDAR) required by § 53.1060, an additional 20 percent may be used. An updated site-specific DCE must be submitted to the NRC prior to the licensee using any funding in excess of these amounts.

(b) Licensees that are not "electric utilities" as defined in § 53.020 that use prepayment or an external sinking fund to provide financial assurance shall provide in the terms of the arrangements governing the trust, escrow account, or Government fund, used to segregate and manage the funds that—

(1) The trustee, manager, investment advisor, or other person directing investment of the funds:

(i) Is prohibited from investing the funds in securities or other obligations of the licensee or any other owner or operator of any commercial nuclear plant or their affiliates, subsidiaries, successors or assigns, or in a mutual fund in which at least 50 percent of the fund is invested in the securities of a licensee or parent company whose subsidiary is an owner or operator of a foreign or domestic commercial nuclear plant. However, the funds may be invested in securities tied to market indices or other non-nuclear sector collective, commingled, or mutual funds, provided that this paragraph (b)(1)(i) shall not operate in such a way as to require the sale or transfer either in whole or in part, or other disposition of any such prohibited investment that was made before the publication date of this rule, and provided further that no more than 10 percent of trust assets may be indirectly invested in securities of any entity owning or operating one or more commercial nuclear plants.

(ii) Is obligated at all times to adhere to a standard of care set forth in the trust, which either shall be the standard of care, whether in investing or otherwise, required by

State or Federal law or one or more State or Federal regulatory agencies with jurisdiction over the trust funds, or, in the absence of any such standard of care, whether in investing or otherwise, that a prudent investor would use in the same circumstances. The term "prudent investor," shall have the same meaning as set forth in FERC's "Regulations Governing Nuclear Plant Decommissioning Trust Funds" at 18 CFR 35.32(a)(3), or any successor regulation.

(2) The licensee, its affiliates, and its subsidiaries are prohibited from being engaged as investment manager for the funds or from giving day-to-day management direction of the funds' investments or direction on individual investments by the funds, except in the case of passive fund management of trust funds where management is limited to investments tracking market indices.

(3) The trust, escrow account, Government fund, or other account used to segregate and manage the funds may not be amended in any material respect without written notification to the Director, Office of Nuclear Reactor Regulation, or Director, Office of Nuclear Material Safety and Safeguards, as applicable, at least 30 working days before the proposed effective date of the amendment. The licensee shall provide the text of the proposed amendment and a statement of the reason for the proposed amendment. The trust, escrow account, Government fund, or other account may not be amended if the person responsible for managing the trust, escrow account, Government fund, or other account receives written notice of objection from the Director, Office of Nuclear Reactor Regulation, or Director, Office of Nuclear Material Safety and Safeguards, as applicable, within the notice period.

(4) Except for withdrawals being made under paragraph (a) of this section or for payments of ordinary administrative costs (including taxes) and other incidental expenses of the fund (including legal, accounting, actuarial, and trustee expenses) in

connection with the operation of the fund, no disbursement or payment may be made from the trust, escrow account, Government fund, or other account used to segregate and manage the funds until written notice of the intention to make a disbursement or payment has been given to the Director, Office of Nuclear Reactor Regulation, or Director, Office of Nuclear Material Safety and Safeguards, as applicable, at least 30 working days before the date of the intended disbursement or payment. The disbursement or payment from the trust, escrow account, Government fund or other account may be made following the 30-working day notice period if the person responsible for managing the trust, escrow account, Government fund, or other account does not receive written notice of objection from the Director, Office of Nuclear Reactor Regulation, or Director, Office of Nuclear Material Safety and Safeguards, as applicable, within the notice period. Disbursements or payments from the trust, escrow account, Government fund, or other account used to segregate and manage the funds, other than for payment of ordinary administrative costs (including taxes) and other incidental expenses of the fund (including legal, accounting, actuarial, and trustee expenses) in connection with the operation of the fund, are restricted to decommissioning expenses or transfer to another financial assurance method acceptable under § 53.1040 until final decommissioning has been completed. After decommissioning has begun and withdrawals from the decommissioning fund are made under paragraph (a) of this section, no further notification need be made to the NRC.

(c) Licensees that are "electric utilities" under § 53.020 that use prepayment or an external sinking fund to provide financial assurance shall include a provision in the terms of the trust, escrow account, Government fund, or other account used to segregate and manage funds that except for withdrawals being made under paragraph (a) of this section or for payments of ordinary administrative costs (including taxes) and

other incidental expenses of the fund (including legal, accounting, actuarial, and trustee expenses) in connection with the operation of the fund, no disbursement or payment may be made from the trust, escrow account, Government fund, or other account used to segregate and manage the funds until written notice of the intention to make a disbursement or payment has been given the Director, Office of Nuclear Reactor Regulation, or Director, Office of Nuclear Material Safety and Safeguards, as applicable, at least 30 working days before the date of the intended disbursement or payment. The disbursement or payment from the trust, escrow account, Government fund or other account may be made following the 30-working day notice period if the person responsible for managing the trust, escrow account, Government fund, or other account does not receive written notice of objection from the Director, Office of Nuclear Reactor Regulation, or Director, Office of Nuclear Material Safety and Safeguards, as applicable, within the notice period. Disbursements or payments from the trust, escrow account, Government fund, or other account used to segregate and manage the funds, other than for payment of ordinary administrative costs (including taxes) and other incidental expenses of the fund (including legal, accounting, actuarial, and trustee expenses) in connection with the operation of the fund, are restricted to decommissioning expenses or transfer to another financial assurance method acceptable under § 53.1040 until final decommissioning has been completed. After decommissioning has begun and withdrawals from the decommissioning fund are made under paragraph (a) of this section, no further notification need be made to the NRC.

(d) A licensee that is not an "electric utility" under § 53.020 and using a surety method, insurance, or other guarantee method to provide financial assurance shall provide that the trust established for decommissioning costs to which the surety or

insurance is payable contains in its terms the requirements in § 53.1045(b)(1), (2), (3), and (4).

§ 53.1050 NRC oversight.

The NRC reserves the right to take the following steps in order to ensure a licensee's adequate accumulation of decommissioning funds: review, as needed, the rate of accumulation of decommissioning funds; and, either independently or in cooperation with FERC and the licensee's State PUC, take additional actions as appropriate on a case-by-case basis, including modification of a licensee's schedule for the accumulation of decommissioning funds.

§ 53.1060 Reporting and recordkeeping requirements.

(a) Each holder of an operating license under Framework A of this part or holder of a combined license under Framework A of this part after the date that the Commission has made the finding under § 53.1452(g) must report, at least once every 2 years, by March 31, on the status of its certification of decommissioning funding for each reactor or part of a reactor that it owns. The information in this report must include, at a minimum, the amount of decommissioning funds estimated to be required pursuant to §§ 53.1020 and 53.1030; the amount of decommissioning funds accumulated to the end of the calendar year preceding the date of the report; a schedule of the annual amounts remaining to be collected; the assumptions used regarding rates of escalation in decommissioning costs, rates of earnings on decommissioning funds, and rates of other factors used in funding projections; any contracts upon which the licensee is relying pursuant to § 53.1040(e); any modifications occurring to a licensee's current method of providing financial assurance since the last submitted report; and any material changes

to trust agreements. If any of the preceding items is not applicable, the licensee should so state in its report. Any licensee for a plant that is within 5 years of the projected end of its operation, or where conditions have changed such that it will close within 5 years (before the end of its licensed life), or that has already closed (before the end of its licensed life), or that is involved in a merger or an acquisition shall submit this report annually.

(b) Each holder of a combined license under Framework A of this part shall, 2 years before and 1-year before the scheduled date for initial loading of fuel, submit a report to the NRC containing a certification updating the DCEs and a copy of the financial instrument to be used to satisfy § 53.1040. No later than 30 days after the Commission publishes notice in the *Federal Register* under § 53.1452(a), the licensee shall submit a report containing a certification that financial assurance for decommissioning is being provided in an amount specified in the licensee's most recent updated certification, including a copy of the financial instrument obtained to satisfy § 53.1040.

(c) Each licensee shall keep records of information important to the safe and effective decommissioning of the facility in an identified location until the license is terminated by the Commission. If records of relevant information are kept for other purposes, reference to these records and their locations may be used. Information the Commission considers important to decommissioning consists of—

(1) Records of spills or other unusual occurrences involving the spread of contamination in and around the facility, equipment, or site. These records may be limited to instances when significant contamination remains after any cleanup procedures or when there is reasonable likelihood that contaminants may have spread to inaccessible areas as in the case of possible seepage into porous materials such as

concrete. These records must include any known information on identification of involved nuclides, quantities, forms, and concentrations.

(2) As-built drawings and modifications of structures and equipment in restricted areas where radioactive materials are used and/or stored and of locations of possible inaccessible contamination such as buried pipes which may be subject to contamination. If required drawings are referenced, each relevant document need not be indexed individually. If drawings are not available, the licensee shall substitute appropriate records of available information concerning these areas and locations.

(3) Records of the cost estimate performed for the decommissioning funding plan or of the amount certified for decommissioning, and records of the funding method used for assuring funds if either a funding plan or certification is used.

(4) Records of:

(i) The licensed site area, as originally licensed and any revisions, which must include a site map and any acquisition or use of property outside the originally licensed site area for the purpose of receiving, possessing, or using licensed materials;

(ii) The licensed activities carried out on the acquired or used property; and

(iii) The release and final disposition of any property recorded in paragraph (c)(4)(i) of this section, the historical site assessment performed for the release, radiation surveys performed to support release of the property, submittals to the NRC made in accordance with § 53.1575, and the methods employed to ensure that the property met the radiological criteria of 10 CFR part 20, subpart E, at the time the property was released.

(d) Each holder of an operating license or combined license under Framework A of this part shall at or about 5 years prior to the projected end of operations submit a

preliminary DCE which includes an up-to-date assessment of the major factors that could affect the cost to decommission.

(e) Prior to or within 2 years following permanent cessation of operations, the licensee shall submit a PSDAR to the NRC, and a copy to the affected State(s). The PSDAR must contain a description of the planned decommissioning activities along with a schedule for their accomplishment, 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, and a site-specific DCE, including the projected cost of managing irradiated fuel.

(f) 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.

(g) After submitting its site-specific DCE required by paragraph (e) of this section, and until the licensee has completed its final radiation survey and demonstrated that residual radioactivity has been reduced to a level that permits termination of its license, the licensee must annually submit to the NRC, by March 31, a financial assurance status report. The report must include the following information, current through the end of the previous calendar year:

(1) The amount spent on decommissioning, both cumulative and over the previous calendar year, the remaining balance of any decommissioning funds, and the amount provided by other financial assurance methods being relied upon;

(2) An estimate of the costs to complete decommissioning, reflecting any difference between actual and estimated costs for work performed during the year, and the decommissioning criteria upon which the estimate is based;

(3) Any modifications occurring to a licensee's current method of providing financial assurance since the last submitted report; and

(4) Any material changes to trust agreements or financial assurance contracts.

(5) If the sum of the balance of any remaining decommissioning funds, plus earnings on such funds calculated at not greater than a 2 percent real rate of return, together with the amount provided by other financial assurance methods being relied upon, does not cover the estimated cost to complete the decommissioning, the financial assurance status report must include additional financial assurance to cover the estimated cost of completion.

(h) After submitting its site-specific DCE required by paragraph (e) of this section, the licensee must annually submit to the NRC, by March 31, a report on the status of its funding for managing irradiated fuel. The report must include the following information, current through the end of the previous calendar year:

(1) The amount of funds accumulated to cover the cost of managing the irradiated fuel;

(2) The projected cost of managing irradiated fuel until title to the fuel and possession of the fuel is transferred to the Secretary of Energy; and

(3) If the funds accumulated do not cover the projected cost, a plan to obtain additional funds to cover the cost.

§ 53.1070 Termination of license.

For each holder of an operating license or combined license under Framework A of this part—

(a)(1) When the 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 § 53.040(b)(8);

(2) When appropriate to support decommissioning activities and the eventual permanent removal of fuel from the reactor vessel, the licensee must develop defueled technical specifications by reviewing the operational technical specifications and determining which specifications no longer apply during decommissioning and which ones should remain applicable. The licensee must make the appropriate submittals to the NRC in accordance with § 53.1510 to request changes to the technical specifications; and

(3)(i) 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 § 53.040(b)(9); and

(ii) The licensee may establish and maintain staffing consisting of certified fuel handlers, as defined under § 53.020, and other non-licensed personnel with appropriate qualifications, and in sufficient numbers, to ensure support for facility operations and radiological control activities, as required by the facility defueled technical specifications. These personnel must be subject to the training requirements of §§ 53.830.

(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 license issued under Framework A of this part 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) Prior to or within 2 years following permanent cessation of operations, the licensee shall submit a PSDAR and site-specific DCE in accordance with § 53.1060(e).

(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 readily accessible to individuals in the vicinity of the licensee’s facility upon receipt of the PSDAR. The NRC shall publish a notice 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 § 53.020, 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 paragraph (a) of this section, have been submitted.

(f) Licensees shall not perform any decommissioning activities, as defined in § 53.020, 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 § 53.1540 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 increase the decommissioning cost by more than 20 percent from the previously provided DCE.

(h) Licensees may use decommissioning trust funds consistent with the limitations of § 53.1045(a). Licensees must report on the status of decommissioning trust funds consistent with the requirements of § 53.1060.

(i) Licensees must submit an application for termination of license in accordance with § 53.1575. 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 Final Safety Analysis Report (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, pursuant to § 51.53, 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 readily accessible to individuals in the vicinity of the licensee’s facility upon receipt of the license termination plan. The NRC shall publish a notice 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.

(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 such conditions and limitations 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 10 CFR part 20, subpart E.

§ 53.1075 Program requirements during decommissioning.

(a) Licensees that have submitted the certifications required under § 53.1575 must maintain a decommissioning fire protection program to address the potential for fires that could cause the release or spread of radioactive materials.

(1) The objectives of the decommissioning fire protection program are to

(i) Reasonably prevent these fires from occurring;

(ii) Rapidly detect, control, and extinguish those fires that do occur and that could result in a radiological hazard; and

(iii) Ensure that the risk of fire induced radiological hazards to the public, environment, and plant personnel is minimized.

(2) The licensee must assess the decommissioning fire protection program on a regular basis. The licensee must revise the decommissioning fire protection program documentation as appropriate throughout the various stages of facility decommissioning.

(3) The licensee may make changes to the decommissioning fire protection program without NRC approval if these changes do not reduce the effectiveness of fire protection for structures, systems, and components that could result in a radiological hazard, taking into account the decommissioning plant conditions and activities.

(b) Reserved.

§ 53.1080 Release of part of a commercial nuclear plant or site for unrestricted use.

(a) Prior written NRC approval is required to release part of a commercial nuclear plant or site for unrestricted use at any time before receiving approval of a license termination plan. Section 53.1060 specifies recordkeeping requirements associated with partial release. Holders of an operating license or combined license under Framework A of this part seeking NRC review and approval shall—

(1) Evaluate the effect of releasing the property to ensure that—

(i) The dose to individual members of the public does not exceed the limits and standards of 10 CFR part 20, subpart D;

(ii) There is no reduction in the effectiveness of emergency planning or physical security;

(iii) Effluent releases remain within license conditions;

(iv) The environmental monitoring program and offsite dose calculation manual are revised to account for the changes;

(v) The siting criteria of subpart D of this part continue to be met; and

(vi) All other applicable statutory and regulatory requirements continue to be met.

(2) Perform a historical site assessment of the part of the commercial nuclear plant or site to be released; and

(3) Perform surveys adequate to demonstrate compliance with the radiological criteria for unrestricted use specified in § 20.1402 for impacted areas.

(b) For release of non-impacted areas, the licensee may submit a written request for NRC review and approval of the release if a license amendment is not otherwise required. The request submittal must include—

(1) The results of the evaluations performed in accordance with paragraphs (a)(1) and (a)(2) of this section;

(2) A description of the part of the commercial nuclear plant or site to be released;

(3) The schedule for release of the property;

(4) The results of the evaluations performed in accordance with § 53.1540; and

(5) A discussion that provides the reasons for concluding that the environmental impacts associated with the licensee's proposed release of the property will be bounded by appropriate previously issued environmental impact statements.

(c) After receiving a request from the licensee for NRC approval of the release of a non-impacted area, the NRC shall—

(1) Determine whether the licensee has adequately evaluated the effect of releasing the property as required by paragraph (a)(1) of this section;

(2) Determine whether the licensee's classification of any release areas as non-impacted is adequately justified; and

(3) If determining that the licensee's submittal is adequate, inform the licensee in writing that the release is approved.

(d) For release of impacted areas, the licensee shall submit an application for amendment of its license for the release of the property. The application must include—

(1) The information specified in paragraphs (b)(1) through (b)(3) of this section;

(2) The methods used for and results obtained from the radiation surveys required to demonstrate compliance with the radiological criteria for unrestricted use specified in § 20.1402; and

(3) A supplement to the environmental report, under § 51.53, describing any new information or significant environmental change associated with the licensee's proposed release of the property.

(e) After receiving a license amendment application from the licensee for the release of an impacted area, the NRC shall—

(1) Determine whether the licensee has adequately evaluated the effect of releasing the property as required by paragraph (a)(1) of this section;

(2) Determine whether the licensee's classification of any release areas as non-impacted is adequately justified;

(3) Determine whether the licensee's radiation survey for an impacted area is adequate; and

(4) If determining that the licensee's submittal is adequate, approve the licensee's amendment application.

(f) The NRC shall notice receipt of the release approval request or license amendment application and make the approval request or license amendment application available for public comment. Before acting on an approval request or license amendment application submitted in accordance with this section, the NRC shall conduct a public meeting readily accessible to individuals in the vicinity of the licensee's facility for the purpose of obtaining public comments on the proposed release of part of the commercial nuclear plant or site. 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.

Subpart H—Licenses, Certifications, and Approvals

§ 53.1100 Filing of application for licenses, certifications or approvals; oath or affirmation.

(a) *Serving of applications*.

(1) Each filing of an application for a standard design approval, standard design certification, or license under Framework A of this part, and any amendments to the applications, must be submitted to the NRC under § 53.040, as applicable.

(2) Each applicant for a construction permit, early site permit, combined license, or manufacturing license under Framework A of this part must, upon notification by the presiding officer designated to conduct the public hearing required by the AEA, update the application and serve the updated copies of the application or parts of it, eliminating all superseded information, together with an index of the updated application, as directed by presiding officer. Any subsequent amendment to the application must be served on those served copies of the application and must be submitted to the NRC as specified in § 53.040, as applicable.

(3) The applicant must make a copy of the updated application available at the public hearing for the use of any other parties to the proceeding, and must certify that the updated copies of the application contain the current contents of the application submitted in accordance with the requirements under Framework A of this part.

(4) At the time of filing an application, the Commission will make available at the NRC Web site, http://www.nrc.gov, a copy of the application, subsequent amendments, and other records pertinent to the matter that is the subject of the application for public inspection and copying.

(5) The serving of copies required by this section must not occur until the application has been docketed under § 2.101(a) of this chapter. Copies must be submitted to the Commission, as specified in § 53.040, as applicable, to enable the Director, Office of Nuclear Reactor Regulation to determine whether the application is sufficiently complete to permit docketing.

(b) *Oath or affirmation*. Each application for a standard design approval, standard

design certification, or license, including, whenever appropriate, a construction permit or early site permit, or amendment of it, and each amendment of each application must be executed in a signed original by the applicant or duly authorized officer thereof under oath or affirmation.

(c) [Reserved]

(d) [Reserved]

(e) *Filing fees*. Each application for a standard design approval, standard design certification, or commercial nuclear plant license under Framework A this part, including, whenever appropriate, a construction permit, combined license, operating license, manufacturing license, or early site permit, other than a license exempted from 10 CFR part 170, must be accompanied by the fee prescribed in 10 CFR part 170. No fee will be required to accompany an application for renewal, amendment, or termination of a construction permit, operating license, combined license, or manufacturing license, except as provided in § 170.21 of this chapter.

(f) *Environmental report*. An application for a construction permit, operating license, early site permit, design certification, combined license, or manufacturing license for a commercial nuclear plant must be accompanied by an environmental report required under subpart A of 10 CFR part 51.

§ 53.1101 Requirement for license.

Except as provided in § 53.1120, no person within the United States shall transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, or use any utilization facility except as authorized by a license issued by the Commission.

§ 53.1103 Combining applications and licenses.

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

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

§ 53.1106 Elimination of repetition.

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.

§ 53.1109 Contents of applications; general information.

Each application must include, unless otherwise indicated in this subpart:

(a) Name of applicant;

(b) Address of applicant;

(c) Description of business or occupation of applicant;

(d)(1) If applicant is an individual, the citizenship of the applicant;

(2) If applicant is a partnership, the name, citizenship and address of each partner and the principal location where the partnership does business;

(3) If applicant is a corporation or an unincorporated association, the following information:

(i) The State where it is incorporated or organized and the principal location where it does business;

(ii) The names, addresses and citizenship of its directors and of its principal

officers; and

(iii) Whether it is owned, controlled, or dominated by an alien, a foreign corporation, or foreign government, and if so, give details; or

(4) If the applicant is acting as agent or representative of another person in filing the application, identify the principal and furnish information required under this paragraph with respect to such principal;

(e) The type of license applied for, the use to which the facility will be put, the period of time for which the license is sought, and a list of other licenses, except operator's licenses, issued or applied for in connection with the proposed facility;

(f) [Reserved]

(g)(1) Except as provided in paragraph (g)(2) of this section, if the application is for an operating license or combined license for a commercial nuclear plant, or if the application is for an early site permit for a commercial nuclear plant and contains plans for coping with emergencies under § 53.1146(b)(2)(ii), radiological emergency response plans of State, local, and participating Tribal governmental entities in the United States that are wholly or partially within the plume exposure pathway emergency planning zone (EPZ),[1] and, for applicants choosing to comply with § 50.47 of this chapter and appendix E to 10 CFR part 50, the plans of State governments wholly or partially within the ingestion pathway EPZ.[2] Except as provided in paragraph (g)(2) of this section, if the application is for an early site permit that, under § 53.1146(b)(2)(i), proposes major features of the emergency plans describing the EPZs, then the descriptions of the EPZs must demonstrate compliance with the requirements of this paragraph. Generally, for applicants choosing to follow § 50.47 and appendix E to 10 CFR part 50, the plume exposure pathway EPZ for a commercial nuclear plant must consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ must consist of an area about 50

miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular commercial nuclear plant must be determined in relation to the local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The size of the EPZs also may be determined on a case-by-case basis for gas-cooled reactors and for reactors with an authorized power level less than 250 megawatt (MW) thermal. The plans for the ingestion pathway must focus on such actions as are appropriate to protect the food ingestion pathway.

(2) [*To be added when the EP for SMR and ONT final rule is published*.]

(h) [Reserved]

(i) A list of the names and addresses of such regulatory agencies as may have jurisdiction over the rates and services incident to the proposed activity, and a list of trade and news publications which circulate in the area where the proposed activity will be conducted and which are considered appropriate to give reasonable notice of the application to those municipalities, private utilities, public bodies, and cooperatives, which might have a potential interest in the facility; and

(j) If the application contains Restricted Data or other defense information, confirmation that all Restricted Data and other defense information are separated from the unclassified information.

(k) [Reserved]

* * * * *

[1] EPZs are discussed in NUREG–0396, EPA 520/1–78–016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light-Water Nuclear Power Plants," December 1978.

[2] If the State, local, and participating Tribal emergency response plans have been previously provided to the NRC for inclusion in the facility docket, the applicant need only provide the appropriate

reference to meet this requirement.

§ 53.1112 Environmental conditions.

(a) Each construction permit, early site permit, and combined license under Framework A of this part may include conditions to address environmental issues during construction. These conditions are to be set out in an attachment to the license, which is incorporated in and made a part of the license. These conditions will be derived from information contained in the environmental report submitted pursuant to § 51.50 of this chapter, as analyzed and evaluated in the NRC record of decision, and will identify the obligations of the licensee in the environmental area, including, as appropriate, requirements for reporting and keeping records of environmental data, and any conditions and monitoring requirement for the protection of the nonaquatic environment.

(b) Each license authorizing operation of a commercial nuclear plant, including a combined license, under Framework A of this part, and each license for a commercial nuclear plant for which the certification of permanent cessation of operations required under § 53.1070(a)(1) has been submitted may include conditions to address environmental issues during operation and decommissioning. These conditions are to be set out in an attachment to the license which is incorporated in and made a part of the license. These conditions will be derived from information contained in the environmental report or the supplement to the environmental report submitted under §§ 51.50 and 51.53 of this chapter as analyzed and evaluated in the NRC record of decision, and will identify the obligations of the licensee in the environmental area, including, as appropriate, requirements for reporting and keeping records of environmental data and any conditions and monitoring requirement for the protection of the nonaquatic environment.

§ 53.1115 Agreement limiting access to classified information.

As part of its application and in any event before the receipt of Restricted Data or classified National Security Information or the issuance of a license or standard design approval under Framework A of this part, or before the Commission has adopted a final standard design certification rule under Framework A of this part, the applicant must agree in writing that it will not permit any individual to have access to any facility or 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. The agreement of the applicant becomes part of the license or standard design approval.

§ 53.1118 Ineligibility of certain applicants.

Any person who is a citizen, national, or agent of a foreign country, or any corporation, or other entity which the Commission knows or has reason to believe is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government, shall be ineligible to apply for and obtain a license.

§ 53.1120 Exceptions and exemptions from licensing requirements.

Nothing in this part shall be deemed to require a license for:

(a) The manufacture, production, or acquisition by the Department of Defense of any utilization facility authorized pursuant to section 91 of the AEA, or the use of such facility by the Department of Defense or by a person under contract with and for the account of the Department of Defense;

(b) Except to the extent that the Department of Energy facilities of the types subject to licensing pursuant to section 202 of the ERA are involved:

(1)(i) The processing, fabrication or refining of special nuclear material or the separation of special nuclear material, or the separation of special nuclear material from other substances by a prime contractor of the Department of Energy under a prime contract for:

(A) The performance of work for the Department of Energy at a United States government-owned or controlled site;

(B) Research in, or development, manufacture, storage, testing or transportation of, atomic weapons or components thereof; or

(C) The use or operation of a utilization facility in a United States owned vehicle or vessel; or

(ii) The processing, fabrication or refining of special nuclear material of the separation of special nuclear material, or the separation of special nuclear material from other substances by a prime contractor or subcontractor of the Commission or the Department of Energy under a prime contract or subcontract when the Commission determines that the exemption of the prime contractor or subcontractor is authorized by law; and that, under the terms of the contract or subcontract, there is adequate assurance that the work thereunder can be accomplished without undue risk to the public health and safety; or

(2)(i) The construction or operation of a utilization facility for the Department of Energy at a United States government-owned or controlled site, including the transportation of the utilization facility to or from such site and the performance of contract services during temporary interruptions of such transportation; or the construction or operation of a utilization facility for the Department of Energy in the

performance of research in, or development, manufacture, storage, testing, or transportation of, atomic weapons or components thereof; or the use or operation of a utilization facility for the Department of Energy in a United States government-owned vehicle or vessel: provided, that such activities are conducted by a prime contractor of the Department of Energy under a prime contract with the Department of Energy; or

(ii) The construction or operation of a utilization facility by a prime contractor or subcontractor of the Commission or the Department of Energy under his prime contract or subcontract when the Commission determines that the exemption of the prime contractor or subcontractor is authorized by law; and that, under the terms of the contract or subcontract, there is adequate assurance that the work thereunder can be accomplished without undue risk to the public health and safety; or

(c) The transportation or possession of any utilization facility by a common or contract carrier or warehousemen in the regular course of carriage for another or storage incident thereto.

§ 53.1121 Public inspection of applications.

Applications and documents submitted to the Commission in connection with applications may be made available for public inspection in accordance with the provisions of the regulations contained in 10 CFR part 2.

§ 53.1124 Relationship between sections.

(a) *Limited work authorization.* An application for an LWA under Framework A of this part may be submitted as part of an application for an early site permit, construction permit, or combined license under Framework A of this part as required in § 53.1130(a)(2).

(b) *Early site permit*. (1) A holder of an early site permit may request an LWA.

(2) An application for a construction permit or combined license under Framework A of this part may, but need not, reference an early site permit.

(c) *Standard design approval.* An application for a standard design approval under Framework A of this part may, but need not, reference an operating license or custom combined license under Framework A of this part that is essentially the same as the information supporting the standard design for which approval is being requested.

(d) *Standard design certification.* An application for a standard design certification under Framework A of this part may, but need not, reference an operating license or custom combined license under Framework A of this part that is essentially the same as the standard design for which certification is being requested.

(e) *Manufacturing license.* (1) A manufactured reactor or manufactured reactor module manufactured under a manufacturing license (ML) issued under Framework A of this part may only be transported to and installed at a site for which a combined license (COL) under Framework A of this part has been issued. Manufactured reactor modules licensed for factory installation of fuel can only be shipped to sites for which an appropriate license, including for the possession of special nuclear material, has been issued.

(2) A manufacturing license applicant under Framework A of this part may reference a standard design certification or a standard design approval under Framework A of this part in its application.

(3) If licensed under Framework A of this part for factory installation of fuel, a license for receipt, possession, handling, and storage of special nuclear material under 10 CFR part 70, “Domestic licensing of special nuclear material,” must be obtained prior to receipt of the fuel at the manufacturer’s facility.

(f) *Construction permit.* An application for a construction permit may, but need not, reference a standard design certification or standard design approval issued under Framework A of this part, respectively, and may also reference an early site permit issued under Framework A of this part. In the absence of a demonstration that an entity other than the one originally sponsoring a standard design certification is qualified to supply a design, the Commission will entertain an application for a construction permit that references a standard design certification issued under Framework A of this part only if the entity that sponsored the certification supplies the design for the applicant's use.

(g) *Operating license.* (1) An application for an operating license under Framework A of this part may, but need not, reference an early site permit, standard design certification, or standard design approval issued under Framework A of this part. In the absence of a demonstration that an entity other than the one originally sponsoring a standard design certification is qualified to supply a design, the Commission will entertain an application for an operating license that references a standard design certification issued under Framework A of this part only if the entity that sponsored the certification supplies the design for the applicant's use.

(2) The holder of a construction permit must, at the time of submission of the FSAR, file an application for an operating license.

(h) *Combined licenses.* An application for a combined license under Framework A of this part may, but need not, reference an early site permit, standard design certification, standard design approval, or manufacturing license issued under Framework A of this part. In the absence of a demonstration that an entity other than the one originally sponsoring and obtaining a standard 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 Framework A of this part only if the entity that sponsored the certification supplies the design for the applicant's use.

§ 53.1130 Limited work authorizations.

(a) *Request for limited work authorization*. (1) Any person to whom the Commission may otherwise issue either a license or permit related to a commercial nuclear plant may request an LWA allowing that person to perform the driving of piles, subsurface preparation, placement of backfill, concrete, or permanent retaining walls within an excavation, installation of the foundation, including placement of concrete, any of which are for an SSC of the facility for which either a construction permit or combined license is otherwise required under § 53.610 of this part.

(2) An application for an LWA may be submitted as part of a complete application for a construction permit or combined license in accordance with § 2.101(a)(1) through (a)(5) of this chapter, or as a partial application in accordance with § 2.101(a)(9) of this chapter. An application for an LWA by the holder of an early site permit must be submitted as a complete application in accordance with § 2.101(a)(1) through (a)(4) of this chapter.

(3) The application must include:

(i) A Safety Analysis Report required by §§ 53.1309, or 53.1416 of this chapter, as applicable, a description of the activities requested to be performed, and the design and construction information otherwise required by the Commission's rules and regulations to be submitted for a construction permit or combined license under Framework A of this part but limited to those portions of the facility that are within the scope of the LWA;

(A) The Safety Analysis Report must demonstrate that activities conducted under

the LWA will be conducted in compliance with the technically relevant Commission requirements in 10 CFR chapter I applicable to the design of those portions of the facility within the scope of the LWA;

(B) [Reserved]

(ii) An environmental report in accordance with § 51.49 of this chapter; and

(iii) A plan for redress of activities performed under the LWA, should limited work activities be terminated by the holder or the LWA be revoked by the NRC or upon effectiveness of the Commission's final decision denying the associated construction permit or combined license application, as applicable.

(b) *Issuance of limited work authorization*. (1) The Director of the Office of Nuclear Reactor Regulation may issue an LWA only after:

(i) The NRC staff issues the final environmental impact statement for the LWA in accordance with subpart A of 10 CFR part 51;

(ii) The presiding officer makes the finding in §§ 51.105(c) or 51.107(d) of this chapter, as applicable;

(iii) The Director determines that the applicable standards and requirements of the AEA, and the Commission's regulations applicable to the activities to be conducted under the LWA, have been met. The applicant is technically qualified to engage in the activities authorized. Issuance of the LWA will provide reasonable assurance of adequate protection to public health and safety and will not be inimical to the common defense and security; and

(iv) The presiding officer finds that there are no unresolved safety issues relating to the activities to be conducted under the LWA that would constitute good cause for withholding the authorization.

(2) Each LWA will specify the activities that the holder is authorized to perform.

(c) *Effect of limited work authorization*. Any activities undertaken under an LWA are entirely at the risk of the applicant and, except as to the matters determined under paragraph (d) of this section, the issuance of the LWA has no bearing on the issuance of a construction permit or combined license with respect to the requirements of the AEA and rules, regulations, or orders issued under the AEA. The environmental impact statement for a construction permit or combined license application for which an LWA was previously issued will not address, and the presiding officer will not consider, the sunk costs of the holder of the LWA in determining the proposed action (i.e., issuance of the construction permit or combined license).

(d) *Implementation of redress plan*. If construction is terminated by the holder, the underlying application is withdrawn by the applicant or denied by the NRC, or the LWA is revoked by the NRC, then the holder must begin implementation of the redress plan in a reasonable time. The holder must also complete the redress of the site no later than 18 months after termination of construction, revocation of the LWA, or upon effectiveness of the Commission's final decision denying the associated construction permit application or the associated combined license application, as applicable.

§ 53.1140 Early site permits.

Sections 53.1143 through 53.1188 set out the requirements and procedures applicable to Commission issuance of an early site permit for approval of a site for a commercial nuclear plant, which may consist of one or more reactor modules separate from the filing of an application for a construction permit or combined license for the facility.

§ 53.1143 Filing of applications.

Any person who may apply for a construction permit or for a combined license under Framework A of this part, may file an application for an early site permit with the Director, Office of Nuclear Reactor Regulation. 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.

§ 53.1144 Contents of applications for early site permits; general information.

The application must contain all of the information required by § 53.1109(a) through (d) and (j).

§ 53.1146 Contents of applications for early site permits; technical information.

(a) The application must contain:

(1) A Site Safety Analysis Report that must 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, including intakes and outflows, where appropriate, 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 external hazards and site characteristics required by Framework A of this

part;

(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 assessment of the site on which a facility is to be located. The assessment must address the requirements of § 53.500;

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

(xi) A description of the quality assurance program required by subpart K applied to 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.

(2) A complete environmental report as required by § 51.50(b) of this chapter.

(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 § 53.855, such as the exact size and configuration of the EPZs, for review and approval by the NRC, in consultation with the Federal Emergency Management Agency (FEMA), as applicable, 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 FEMA, as applicable, in accordance with the applicable standards of § 53.855. To the extent approval of emergency plans is sought, the application must contain the information required by § 53.1109(g).

(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 must 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 AEA, 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 ITAAC.

(4) Under paragraphs (b)(1) and (b)(2)(i) of this section, the Site Safety Analysis Report must include, where appropriate, a description of contacts and arrangements made with Federal, State, participating Tribal, and local governmental agencies with emergency planning responsibilities. The Site Safety Analysis Report must contain any certifications that have been obtained. If these certifications, where appropriate, 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 must make good faith efforts, where appropriate, 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 an LWA under § 53.1130 be issued in conjunction with the early site permit. The application must include the information otherwise required by § 53.1130.

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

§ 53.1149 Review of applications.

(a) *Standards for review of applications.* Applications filed under Framework A of this part will be reviewed according to the applicable standards set out in Framework A of this part. In addition, the Commission must prepare an environmental impact statement during review of the application, in accordance with the applicable provisions of 10 CFR part 51. The Commission must determine, after consultation with FEMA, as applicable, whether the information required of the applicant by § 53.1146(b)(1) shows that there is not 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 § 53.1146(b)(2)(i) are acceptable in accordance with the applicable standards of § 53.855, and whether any emergency plans submitted by the applicant under § 53.1146(b)(2)(ii) provide reasonable assurance that adequate protective measures can and will be taken in the

event of a radiological emergency.

(b) *Administrative review of applications; hearings*. An early site permit application 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 must 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 Framework A of this part are governed by the procedures contained in subparts C, G, L, and N of 10 CFR part 2, as applicable.

§ 53.1155 Referral to the Advisory Committee on Reactor Safeguards.

The Commission must refer a copy of the application for an early site permit to the Advisory Committee on Reactor Safeguards (ACRS). The ACRS must report on those portions of the application which concern safety.

§ 53.1158 Issuance of early site permit.

(a) After conducting a hearing under § 53.1149(b) and receiving the report to be submitted by the ACRS under § 53.1155, 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 demonstrates compliance with the applicable standards and requirements of the AEA 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 AEA and the Commission's regulations;

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

(5) The proposed ITAAC, 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 AEA, 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 § 53.1146(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 must 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 must specify those § 53.1130(b) activities requested under § 53.1146(c) that the permit holder is authorized to perform.

§ 53.1161 Extent of activities permitted.

If the activities authorized by § 53.1158(c) are performed and the site is not referenced in an application for a construction permit or a combined license issued under Framework A 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 must redress the site under the terms of the site redress plan required by § 53.1146(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 must carry out the redress plan to the greatest extent possible consistent with the alternate use.

§ 53.1164 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.

§ 53.1167 Limited work authorization after issuance of early site permit.

A holder of an early site permit may request an LWA under § 53.1146(c).

§ 53.1170 Transfer of early site permit.

An application to transfer an early site permit will be processed under § 53.1570.

§ 53.1173 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 § 2.309 of this chapter. If a hearing is granted, notice of the hearing will be published in accordance with § 2.309 of this chapter.

(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 § 53.1164(b).

(d) The Commission must refer a copy of the application for renewal to the ACRS. The ACRS must report on those portions of the application which concern safety

and must apply the criteria set forth in § 53.1176.

§ 53.1176 Criteria for renewal.

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

(1) The site complies with the AEA, 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:

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

(ii) Are 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) Will provide 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 § 53.1176(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.

§ 53.1179 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.

§ 53.1182 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 must inform the Director, Office of Nuclear Reactor Regulation (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, under the provisions of § 53.1188. If the permit holder informs the Director that the holder no longer intends to use the site for a commercial nuclear power plant, the Director may terminate the permit.

§ 53.1188 Finality of early site permit determinations.

(a) *Commission finality*.

(1) While an early site permit is in effect under §§ 53.1164 or 53.1179, 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 § 53.1452(g), 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 must 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 commercial nuclear power plant, the Commission must 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 § 53.1565 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 commercial nuclear power plant, the Commission must treat as resolved changes that are equivalent to those that could be made under § 53.1565 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 must update the emergency preparedness information that was provided under § 53.1146(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 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 must 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 demonstrate compliance with 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 must 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 demonstrate compliance with the requirements of §§ 53.1510 and 53.1520.

§ 53.1200 Standard design approvals.

Sections 53.1203 through 53.1221 set out procedures for the filing, NRC staff review, and referral to the ACRS of standard designs, or major portions thereof, for a commercial nuclear plant under Framework A of this part.

§ 53.1203 Filing of applications.

Any person may submit a proposed standard design for a commercial nuclear

plant to the NRC staff for its review. The submittal may consist of either the final design for the entire facility or the final design for major portions thereof.

§ 53.1206 Contents of applications for standard design approvals; general information.

The application must contain all of the information required by § 53.1109(a) through (c) and (j).

§ 53.1209 Contents of applications for standard design approvals; technical information.

(a) *Major portion of a standard design.* 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. If an applicant seeks approval of a major portion of the design, the scope of the application for which approval is sought must include all functional design criteria necessary to demonstrate compliance with the safety criteria in §§ 53.210, 53.220 and 53.450(e), as applicable, for the major portion of the standard design for which NRC staff approval is sought. Such applicants must identify conditions related to interfaces with systems outside the scope of the major portion of the standard design for which NRC staff approval is sought, and functional or physical boundary conditions between the major portion of the standard design for which NRC staff approval is sought and the remainder of the standard design. These conditions must be demonstrated when the standard design approval is incorporated into a subsequent construction permit, design certification, manufacturing license, or

combined license application.

(b) *Final Safety Analysis Report.* The application must contain an FSAR that describes the facility 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, for which the applicant seeks design approval, and must include the following information:

(1) *Site Parameters*. The site parameters postulated for the design in accordance with Framework A of this part, including the design basis external hazard levels for the relevant external hazards, and an analysis and evaluation of the design in terms of those site parameters.

(2) *Design information.* Except as specified in this paragraph, an application for a standard design approval for a commercial nuclear plant must include the design information equivalent to that required for a standard design certification as provided in § 53.1239(a)(2) through (27) for those portions of a commercial nuclear plant included in the standard design approval.

§ 53.1210 Contents of applications for standard design approvals; other application content

(a) In addition to the FSAR, the application must also include the following:

(1) *Availability Controls (if not included in the FSAR)*. A description of the controls on plant operations, including availability controls, to provide reasonable confidence that the configurations and special treatments for NSRSS SSCs provide the capabilities and reliabilities required to demonstrate compliance with the safety criteria of § 53.220.

(2) *Safeguards Information*. A description of the program to protect Safeguards Information against unauthorized disclosure in accordance with the requirements in §§ 73.21 and 73.22 of this chapter, as applicable.

(b) If there are SSCs of the plant which required research and development to confirm the adequacy of their design, provide a report in the application which documents the resolution of any safety questions associated with such SSCs.

(c) A description of how the performance of each design feature has been demonstrated capable of fulfilling functional design criteria considering interdependent effects through either analysis, appropriate test programs, prototype testing, operating experience, or a combination thereof, in accordance with § 53.440(a).

§ 53.1212 Standards for review of applications.

Applications filed under this Framework A of this part will be reviewed for compliance with the standards set out in 10 CFR parts 20, 53, and 73.

§ 53.1215 Referral to the Advisory Committee on Reactor Safeguards.

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

§ 53.1218 Staff approval of design.

(a) Upon completion of its review of a submittal under §§ 53.1200 through 53.1221 and receipt of a report by the ACRS under § 53.1215, the NRC staff must 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.

(b) A standard design approval issued under this section 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, an operating license, a combined license, or a manufacturing license under Framework A of this part that references the design approval and is docketed before the date of expiration of the design approval.

§ 53.1221 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 standard design certification or individual facility license application under Framework A of this part that incorporates by reference a standard design approved under Framework A of this part 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 10 CFR part 2.

(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 § 53.1580 and must be approved by the Executive Director for Operations or authorized designee before issuance of the request.

(d) The Commission will require, before granting a construction permit, combined license, operating license, or manufacturing license that references a standard design approval, that engineering documents, such as analyses, drawings, procurement

specifications, or construction and installation specifications, be completed and available for audit if the more detailed information is necessary for the Commission to verify the information in the application and make its safety determination, including the determination that the application is consistent with the design approval information. This information may be acquired by appropriate arrangements with the design approval applicant.

§ 53.1230 Standard design certifications.

Sections 53.1230 through 53.1263 set forth the requirements and procedures applicable to the Commission's issuance of rules granting standard design certifications for commercial nuclear plants under Framework A of this part separate from the filing of an application for a construction permit or combined license for such a facility.

§ 53.1233 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 § 53.040 and §§ 2.811 through 2.819 of this chapter.

§ 53.1236 Contents of applications for standard design certifications; general information.

The application must contain all of the information required by § 53.1109(a) through (c) and (j).

§ 53.1239 Contents of applications for standard design certifications; 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. The Commission will require, before design certification, that engineering documents, such as analyses, drawings, procurement specifications, or construction and installation specifications, be completed and available for audit if the more detailed information is necessary for the Commission to verify the information in the application and make its safety determination.

(a) *Final Safety Analysis Report.* The application must contain an FSAR that describes the facility and the limits on its operation, and presents a safety analysis of the structures, systems, and components (SSCs), and must include the following information:

(1) *Site Parameters*. The site parameters postulated for the design in accordance with Framework A of this part, including the design basis external hazard levels for the relevant external hazards, and an analysis and evaluation of the design in terms of those site parameters.

(2*)*(i) *General Plant Description*. A general description of the commercial nuclear plant including reactor type, the intended use of the reactor, nuclear design (e.g., neutron spectrum, reactor control, multi-module reactor control), overall layout of the

plant including significant plant features and SSCs, maximum power level and the nature and inventory of radioactive materials.

(ii) *Safety functions*. A description of the primary and additional safety functions required under § 53.230 and a summary of how each safety function is satisfied.

(3) *Design Features and functional design criteria – licensing-basis events*. (i) A description of the design features required by § 53.400 and the functional design criteria required by §§ 53.410 and 53.420 that, when combined with corresponding human actions and programmatic controls, demonstrate that the plant will demonstrate compliance with the safety criteria defined in §§ 53.210 and 53.220, or more restrictive alternative criteria adopted under § 53.470, during LBEs.

(ii) A description of how design features demonstrate compliance with the requirements of § 53.440(a) – (i) and (k) – (m).

(4) *Design Features and Functional Design Criteria – Normal Operations*. A description of the design features and functional design criteria required by § 53.425 to demonstrate compliance with § 53.260 during normal operations.

(5) *Design Features and Functional Design Criteria – aircraft impact*. A description of the design features and functional design criteria required to demonstrate compliance with the requirements of § 53.440(j) for addressing the impact of a large, commercial aircraft.

(6) *Earthquake engineering*. The information necessary to demonstrate that the commercial nuclear plant complies with the earthquake engineering criteria in § 53.480.

(7) *Programmatic Controls and Interfaces*. (i) A description of the corresponding programmatic controls and interfaces necessary to achieve and maintain the reliability and capability of SSCs relied upon to demonstrate compliance with the functional design criteria required by §§ 53.410 and 53.420 and the safety criteria in §§ 53.210 and

53.220, or more restrictive alternative criteria adopted under § 53.470, and necessary to maintain consistency with analyses required by § 53.450.

(ii) For an application for a multi-module commercial nuclear plant, the programmatic controls and interfaces must also be described for different modular configurations, as required by § 53.440(i), including any restrictions that will be necessary during the construction and startup of any given module to ensure the safe operation of the overall commercial nuclear plant to be licensed under this part.

(8) *Programmatic Controls for Normal Operations*. A description of the corresponding programmatic controls, including monitoring programs, necessary to demonstrate that that the criteria defined in § 53.260 are satisfied during normal operations.

(9) *Design Features and Functional Design Criteria for the Protection of Plant Workers*. A description of the design features and functional design criteria required by § 53.430 to demonstrate compliance with § 53.270.

(10) *Programmatic Controls for Protection of Plant Workers*. A description of the corresponding programmatic controls, including monitoring programs, necessary to demonstrate that the worker protection criteria in § 53.270 are satisfied.

(11) *Codes and Standards*. A description of generally accepted consensus codes and standards used to design the design features, as required by § 53.440(b).

(12) *Materials*. A description of the materials used for safety-related (SR) and NSRSS SSCs and a description of the qualification of these materials for their service conditions over the plant lifetime, as required by with § 53.440(c).

(13) *Integrity Assessment Program*. A description of a design integrity assessment program that addresses the elements described in § 53.440(d).

(14) *Safety and Security*. Confirmation that safety and security were considered

together in the design process, as required by § 53.440(f).

(15) *Criticality*. Information demonstrating how the applicant will comply with requirements for criticality accidents in § 53.440(m).

(16) For an application for standard design certification of a multi-module commercial nuclear plant, the possible operating configurations of the reactor modules, including common systems, interface requirements, and system interactions, as required by § 53.440(i).

(17)(i) The classification of SSCs according to their safety significance in accordance with § 53.460(a).

(ii) For SR and NSRSS SSCs, the conditions under which they must perform the safety functions required by § 53.230, including environmental conditions.

(18) *Probabilistic Risk Assessment*. A description of the probabilistic risk assessment (PRA) required by § 53.450(a), and its results.

(19) *Analyses*. A description of the analyses performed to demonstrate compliance with the requirements in § 53.450(b) through 53.450(g), that includes the following information:

(i) A description of the analysis of LBEs and its results, as described in § 53.240. This analysis description must:

(A) Address the elements in § 53.450(e) and 53.450(f); and

(B) In accordance with § 53.460(c):

(*1*) Describe any human actions that are necessary to prevent or mitigate LBEs;

(*2*) Describe how those human actions are capable of being reliably performed under the postulated environmental conditions present; and

(*3*) Describe how those human actions would be addressed by programs established in accordance with subpart F of this part.

(ii)(A) A description of how SSCs needed to ensure the safety criteria defined in § 53.210 are designed to withstand the effects of external hazards as required by § 53.510.

(B) The information necessary to demonstrate that the commercial nuclear plant complies with the earthquake engineering criteria in § 53.480.

(iii) A description of the defense-in-depth measures required by § 53.250.

(iv) A description of all plant operating states where there is the potential for the uncontrolled release of radioactive material to the environment, as required by § 53.450(b)(4).

(v) A description of the events that challenge plant control and safety systems whose failure could lead to an undesirable end state and/or radioactive material release, as required by § 53.450(b)(5).

(vi) A description of the analytical codes used in modeling plant behavior in analyses of LBEs and how these codes are qualified for the range of conditions for which they were used, as required by § 53.450(d).

(vii) If not described in addressing paragraph (5) of this section, the results of other analyses required by § 53.450(g).

(20) *Special Treatments*. A description of special treatments established as required by § 53.460.

(21) *Analytical Margins*. A description of any alternative criteria adopted to demonstrate analytical margins supporting operational flexibilities, if applicable, as required by § 53.470.

(22) *Quality Assurance.* A description of the quality assurance program applied to the design of the structures, systems and components of the commercial nuclear plant, as required by § 53.460(b). The description of the quality assurance program for a

commercial nuclear plant must include a discussion of how the applicable requirements of subpart K of this part were satisfied.

(23) *Design Features and Controls to Address the Minimization of Contamination.* The information required by § 20.1406 of this chapter.

(24) *Interface Requirements*. (i) A description analysis, and evaluation of the interfaces between the standard design and the balance of the commercial nuclear power plant that may impact the ability of the plant to demonstrate compliance with the functional design criteria, performance objectives or the safety criteria required in §§ 53.210 or 53.220, or more restrictive alternative criteria adopted under § 53.470.

(ii) Confirmation that interface requirements are verifiable through inspections, testing, or analysis. These requirements must be sufficiently detailed to allow for completion of the final safety analysis by license applicants that reference the certified design under this subpart. The method to be used for verification of interface requirements must be included as part of the proposed ITAAC required by § 53.1241.

(iii) 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)(24)(i) of this section.

(25) *Technical Qualifications*. A description of the technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter.

(26) *Technical Specifications*. Proposed technical specifications prepared in accordance with the requirements of § 53.710(a) for those areas addressed by the design certification.

(27) *Role of personnel.* Information to address the following for the role of

personnel in ensuring safe operations:

(i) A description of how the human factors engineering design requirements of § 53.440(n)(1) are addressed;

(ii) A description of how the human system interface design requirements of § 53.440(n)(2) are addressed;

(iii) A concept of operations that is of sufficient scope and detail to address the requirements of § 53.440(n)(3);

(iv) A functional requirements analysis and function allocation that is of sufficient scope and detail to address the requirements of § 53.440(n)(4).

§ 53.1241 Contents of applications for standard design certifications; other application content.

(a) In addition to the FSAR, the application must also include the following:

(1) *Environmental report*. An environmental report as required by § 51.55 of this chapter.

(2) *Availability Controls* (if not included in the FSAR). A description of the controls on plant operations, including availability controls, to provide reasonable confidence that the configurations and special treatments for NSRSS SSCs provide the capabilities and reliabilities required to demonstrate compliance with the safety criteria of § 53.220, or more restrictive alternative criteria adopted under § 53.470.

(3) *Inspections, tests, analyses, and acceptance criteria.* The proposed ITAAC 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 AEA, and the Commission's

rules and regulations.

(4) *Safeguards information*. A description of the program to protect Safeguards Information against unauthorized disclosure in accordance with the requirements in §§ 73.21 and 73.22 of this chapter, as applicable.

(b) If there are SSCs of the plant which required research and development to confirm the adequacy of their design, provide a report in the application which documents the resolution of any safety questions associated with such SSCs.

(c) A description of how the performance of each design feature has been demonstrated capable of fulfilling functional design criteria considering interdependent effects through either analysis, appropriate test programs, prototype testing, operating experience, or a combination thereof, in accordance with § 53.440(a).

§ 53.1242 Review of applications.

(a*) Standards for review of applications.* Applications filed under Framework A of this part will be reviewed for compliance with the standards set out in 10 CFR parts 20, 51, 53, and 73.

(b) *Administrative review of applications; hearings.* (1) 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 must initiate the rulemaking after an application has been filed under § 53.1233 and must 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, the Commission decides that a legislative hearing should be held, the information required

by § 2.1502(c) of this chapter must be included in the *Federal Register* document for the proposed design certification.

(2) 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 10 CFR part 2. The Commission must publish a document in the *Federal Register* of its decision to hold a legislative hearing. The document must contain the information specified in § 2.1502(c) of this chapter and specify whether the Commission or a presiding officer will conduct the legislative hearing.

(3) Notwithstanding anything in § 2.390 of this chapter 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 must be published in chapter I of this title.

(c) *Reference to an issued operating license or combined license*. In those cases where a design certification application is preceded by the issuance of an operating license or custom combined license for a commercial nuclear plant that is essentially the same as the standard design for which certification is being requested, the NRC review will follow the processes for referencing a standard design approval in § 53.1221, to the extent practicable.

§ 53.1245 Referral to the Advisory Committee on Reactor Safeguards.

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

§ 53.1248 Issuance of standard design certification.

(a) After conducting a rulemaking proceeding under § 53.1242 on an application for a standard design certification and receiving the report to be submitted by the ACRS under § 53.1245, 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 demonstrates compliance with the applicable standards and requirements of the AEA 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 AEA and the Commission's regulations;

(4) The applicant is technically qualified;

(5) The proposed ITAAC 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 AEA, 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 must not permit any individual to have access to any facility or 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.

§ 53.1251 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 effective date of the rule.

(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 under Framework A of this part 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 § 53.1452 before operation begins under a combined license that references the design certification.

(c) An applicant for a construction permit, operating license, combined license, or manufacturing license under Framework A of this part may, at its own risk, reference in its application a design for which a design certification application has been docketed but not granted.

§ 53.1254 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 engineering documents, such as analyses, drawings, procurement specifications, or construction and installation specifications, be completed and available for audit if the more detailed information is necessary for the Commission to verify the information in the application and 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, under § 53.1251.

(c) The Commission must refer a copy of the application for renewal to the ACRS. The ACRS must report on those portions of the application which concern safety and must apply the criteria set forth in § 53.1257.

§ 53.1257 Criteria for renewal.

(a) The Commission must 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 AEA 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 must grant the amendment request if it determines that the amendment will comply with the AEA 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.

§ 53.1260 Duration of renewal.

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

§ 53.1263 Finality of standard design certifications.

(a)(1) While a standard design certification rule is in effect under § 53.1251 or

§ 53.1260, 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 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 § 53.1263(a)(1), 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 § 53.1263(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) 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 § 53.1248, 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 § 53.080 are present. In addition to the factors listed in § 53.080, the Commission must consider whether the special circumstances which § 53.080 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 § 2.335 of this chapter, in making the findings required for issuance of a combined license, construction permit, operating license, or manufacturing license, or for any hearing under § 53.1452, the Commission must treat as resolved those matters resolved in connection with the issuance or renewal of a design certification rule.

(b) An applicant 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 § 53.080. In addition to the factors listed in § 53.080, the Commission must consider whether the special circumstances that § 53.080 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.

(c) The Commission will require, before granting a construction permit, combined license, operating license, or manufacturing license that references a design certification rule, that engineering documents, such as analyses, drawings, procurement specifications, or construction and installation specifications, be completed and available for audit if the more detailed information is necessary for the Commission to verify the information in the application and make its safety determination, 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.

§ 53.1270 Manufacturing licenses.

Sections 53.1270 through 53.1295 set out the requirements and procedures applicable to Commission issuance of a license under Framework A of this part authorizing manufacture of manufactured reactors or manufactured reactor modules to be installed at sites not identified in the manufacturing license application.

§ 53.1273 Filing of applications.

(a) Any person, except one excluded by § 53.1118, may file an application for a manufacturing license under this section with the Director, Office of Nuclear Reactor Regulation.

(b) Applicants for manufactured reactor modules for which fuel is to be installed at the manufacturer’s facility and the manufactured reactor modules are to be

transported to a licensed site must also possess, apply for, or reference licenses and certifications required by 10 CFR parts 70 and 71.

§ 53.1276 Contents of applications for manufacturing licenses; general information.

Each application for a manufacturing license must include the information contained in § 53.1109(a) through (e), and (j).

§ 53.1279 Contents of applications for manufacturing licenses; technical information.

(a) *Final Safety Analysis Report-siting and design.* The application must include an FSAR 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 ensuring 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 manufacturing and installation specifications by an applicant who seeks to use the manufactured reactor or manufactured reactor module, and permit the preparation of acceptance and inspection requirements by the NRC. The application must include the following information:

(1) *Site Parameters.* The site parameters postulated for the design in accordance with Framework A of this part, including the design basis external hazard levels for the relevant external hazards, and an analysis and evaluation of the design in terms of those site parameters.

(2) *Design information.* Except as specified in this paragraph, the design information equivalent to that required for a standard design certification as defined in

§ 53.1239(a)(2) through (27) for those portions of a commercial nuclear plant included in the manufactured reactor or manufactured reactor module.

(3) *Quality assurance program.* A description of the quality assurance program, as required by § 53.620(a)(6), applied to the design, fabrication, and testing of the structures, systems, and components of the manufactured reactor or manufactured reactor module;

(4) *Conceptual designs*. Representative conceptual designs for one or more commercial nuclear plants using the manufactured reactor or manufactured reactor module;

(5) *Operating configurations*. If multiple manufactured reactors or manufactured reactor modules may be installed at a commercial nuclear plant, a description of the possible operating configurations, including common systems, interface requirements, and system interactions. The final safety analysis must also account for differences among the possible configurations, including any restrictions that will be necessary during the construction and startup of a given manufactured reactor or manufactured reactor module to ensure the safe operation of any commercial nuclear reactor already operating;

(6) *Interface requirements.* (i) The interface requirements between the manufactured reactor or manufactured reactor module and the remaining portions of the commercial nuclear power plant or connections to other facilities outside of the commercial nuclear plant.

(ii) Confirmation that interface requirements are verifiable through inspections, testing, or analysis. These requirements must be sufficiently detailed to allow for completion of the final safety analysis by license applicants that reference the manufactured reactor or manufactured reactor module manufactured under this subpart.

Applicants for a COL under § 53.1410 will need to verify the interface requirements at the installation site. The method to be used for verification of interface requirements must be included as part of the proposed ITAAC required by § 53.1282.

(iii) The FSAR must identify potential pathways for radionuclides produced within the manufactured reactor or manufactured reactor module to enter interfacing systems to support development of radiation monitoring programs required under subpart F.

(b) *Final Safety Analysis Report - Manufacturing information*. The FSAR must include the following information related to the manufacturing processes, organization, controls, and inspections:

(1) A description, including references to generally accepted consensus codes and standards, of the processes that will be used to procure, fabricate, and assemble components that make up the manufactured reactor or manufactured reactor module. The description should clearly define which activities are proposed to be within the scope of the manufacturing license and those, such as the making of a component to be procured from a separate company for installation in the manufactured reactor module, that are not considered to be within the scope of the manufacturing license.

(2) A description of the organizational and management structure singularly responsible for direction of design and manufacture of the manufactured reactor or manufactured reactor modules. The information should include a description of the management plan, technical qualifications, and controls in place to demonstrate compliance with the requirements of § 53.620, including those for any facility performing an activity within the scope of the manufacturing license.

(3) A description of the inspections and tests to be performed as part of the manufacturing process, including the inspection of procured components, inspection and testing of fabrication processes such as the molding, welding, or coating of components,

and inspections and testing of the assembled manufactured reactor, portions of the manufactured reactor, or the manufactured reactor module. Where applicable, the description should identify where the inspections and tests are used to close ITAAC from a standard design certification issued under § 53.1248 referenced in the application for a manufacturing license.

(c) *Deployment of the completed manufactured reactor or manufactured reactor module*. The application must include the following information related to the deployment of a manufactured reactor or manufactured reactor module:

(1) Procedures governing the preparation of the manufactured reactor, portions of the manufactured reactor, or manufactured reactor module for shipping to the site where it is to be operated; the conduct of shipping; and verifying the condition of the shipped items upon receipt at the site;

(2) Details of the interaction of the design, manufacture, and installation of a manufactured reactor or manufactured reactor module within the applicant's organization and the manner by which the applicant will ensure close integration between the designer, contractors, and any facility in which the manufactured reactor or manufactured reactor module is to be installed;

(3) A description of the measures used for the control of interfaces, including the consideration of key site parameters, between the holder of the manufacturing license and the holder of the combined license for the commercial nuclear plant at which the manufactured reactor or manufactured reactor module is to be installed;

(d) *Special considerations for factory fueling.* In addition to the above paragraphs, an application for a manufacturing license for a manufactured reactor module must include the following information related to the fueling operations and the required independent mechanisms to prevent inadvertent criticality and to otherwise

ensure the safety of workers and the public during the manufacture, storage, and transport of each manufactured reactor module:

(1) A description of the safety program and integrated safety analysis required by subpart H of 10 CFR part 70. The description must include the procedures to be used for receipt, storage and loading of the fuel into the manufactured reactor module. The description must either be in the form of a reference to the applicable part 70 application and license, if issued, or provided in the Safety Analysis Report supporting the manufacturing license if one application is used for both the manufacturing license and part 70 license.

(i) The application must specifically address the measures taken for fuel loading, in-factory inspections and non-nuclear testing, including at least two independent mechanisms each of which is sufficient to prevent inadvertent criticality, and an analysis of the safety and security of the manufactured reactor module within the factory, during possible periods of storage, and during transportation to the licensed site. The storage and transport of a fueled manufactured reactor module must comply with applicable regulations in § 53.620(d) and 10 CFR parts 70, 71, and 73.

(ii) The application must specifically address the functional design criteria and design features included in the manufactured reactor module, or physical or programmatic controls implemented during manufacturing, storage, or transport to prevent inadvertent criticality during various conditions, including when subject to potential hazards and human errors.

(2) A description of the procedures governing the transfer of authorities and responsibilities for the manufactured reactor module from the holder of the manufacturing license to the holder of the COL for the installation site.

(3) A description of the controls needed to demonstrate compliance with the

requirements of § 53.620 to address the receipt, storage, and loading of special nuclear material into a manufactured reactor module, including:

(i) The FFD program, in accordance with § 53.620(a)(5) and 10 CFR part 26.

(ii) A Radiation Protection Program in accordance with § 53.620(a)(7).

(iii) An information security program in accordance with § 53.620(a)(8).

(iv) A physical security program in accordance with § 53.620(c)(5).

(v) A fire protection program in accordance with § 53.620(c)(2).

(vi) An emergency plan in accordance with § 53.620(c)(3).

(vii) A description of the plant staff training program in accordance with § 53.620(d).

§ 53.1282 Contents of applications for manufacturing licenses; other application content.

(a)(1) *Inspections, tests, analyses, and acceptance criteria*. The application must contain proposed inspections, tests, and analyses that the combined license holder must 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 or reactor module has been manufactured in conformity with the manufacturing license; the provisions of the AEA, and the Commission's rules and regulations; and

(ii) The manufactured reactor or manufactured reactor module 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, a subsequent combined license application may include a notification that a required inspection, test, or analysis in the design certification ITAAC has been successfully completed during manufacture and that the corresponding acceptance criterion has been met. The *Federal Register* notification required by § 53.1422 must indicate that the application includes this notification.

(b) *Environmental report*. (1) The application must contain an environmental report as required by § 51.54 of this chapter.

(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 manufactured reactor or manufactured reactor module as used in a commercial nuclear plant. Nonetheless, an application for a manufacturing license that references a standard design certification but includes the installation of fuel at the factory must discuss severe accident mitigation design alternatives for the reactor module while at the factory and must also discuss severe accident mitigation alternatives for the factory itself.

(c) *Safeguards information*. The application must contain a description of the program to protect safeguards information against unauthorized disclosure in accordance with the requirements in §§ 73.21 and 73.22 of this chapter, as applicable.

(d) *Performance demonstration.* A description of how the performance of each design feature has been demonstrated capable of fulfilling functional design criteria considering interdependent effects through either analysis, appropriate test programs, prototype testing, operating experience, or a combination thereof, in accordance with

§ 53.440(a).

§ 53.1285 Review of applications.

(a) *Standards for review of applications.* Applications for manufacturing licenses under Framework A of this part will be reviewed according to the applicable standards set out in this subpart as well as applicable standards in 10 CFR parts 20, 25, 26, 51, 53, 70, 71, 73, and 75.

(b) *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, howeve*r, 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 module 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, E, G, L, and N.

§ 53.1286 Referral to the Advisory Committee on Reactor Safeguards.

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

§ 53.1287 Issuance of manufacturing license.

(a) After completing any hearing under § 53.1285(b), 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 AEA and the Commission's regulations have been met;

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

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

(4) The applicant is technically qualified to design and manufacture the proposed manufactured reactor or manufactured reactor modules;

(5) The proposed ITAAC are necessary and sufficient, within the scope of the manufacturing license, to provide reasonable assurance that the manufactured reactor or manufactured reactor module has been manufactured and will be operated in conformity with the license, the provisions of the AEA, 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 10 CFR part 51 have been made.

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

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

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

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

(4) The interface requirements to be met by the site-specific elements of the facility, such as the energy conversions systems and ultimate heat sink, not within the scope of the manufactured reactor or manufactured reactor modules.

§ 53.1288 Finality of manufacturing licenses.

(a)(1) Notwithstanding any provision in § 53.1590, during the term of a manufacturing license issued under Framework A of this part the Commission may not modify, rescind, or impose new requirements on the design of the manufactured reactor or manufactured reactor module, or the requirements for the manufacture of the manufactured reactor or manufactured reactor module, unless the Commission determines that a modification is necessary to bring the design of the reactor or reactor module 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 reactor or manufactured reactor module that is imposed by the Commission under paragraph (a)(1) of this section will be applied to all manufactured reactors or reactor modules manufactured under the license, including those that have already been transported and sited, except those manufactured reactors or manufactured reactor modules to which the modification has been rendered technically irrelevant by action taken under § 53.1530 or paragraph (b) of

this section.

(3) In making the findings required under Framework A of this part for issuance of a combined license, in any hearing under § 53.1452, or in any enforcement hearing other than one initiated by the Commission under paragraph (a)(1) of this section, for which a manufactured reactor or manufactured reactor module manufactured under this subpart is referenced or used, the Commission must 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 or manufactured reactor module, 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 manufactured reactor or manufactured reactor module to be manufactured.

(b) An applicant who references or uses a manufactured reactor or manufactured reactor module manufactured under a manufacturing license under Framework A of this part may include in the application a request for a departure from the design characteristics, site parameters, terms and conditions, or approved design of the manufactured reactor module. The Commission may grant a request only if it determines that the departure will comply with the requirements of 10 CFR 53.080, 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 combined license hearing.

§ 53.1291 Duration of manufacturing licenses.

A manufacturing license issued under Framework A of this part is valid for not

less than 5, nor more than 15 years from the date of issuance. Upon expiration of the manufacturing license, the manufacture of any uncompleted manufactured reactors or manufactured reactor modules must cease unless a timely application for renewal has been docketed with the NRC.

§ 53.1293 Transfer of manufacturing licenses.

A manufacturing license may be transferred under § 53.1570.

§ 53.1295 Renewal of manufacturing licenses.

(a)(1) Not less than 12 months, nor more than five years before the expiration of the manufacturing license, or any later renewal period, the holder of the manufacturing license issued under Framework A of this part 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.

(2) The filing of an application for a renewed license must be in accordance with subpart A of 10 CFR part 2 and § 53.1100.

(3) A manufacturing license issued under Framework A of this part, 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 under § 53.1291, the holder of a manufacturing license may not begin manufacture of a manufactured reactor or manufactured reactor modules less than 6 months before the expiration of the license.

(4) Any person whose interest may be affected by renewal of the license may request a hearing on the application for renewal. The request for a hearing must comply

with § 2.309 of this chapter. If a hearing is granted, notice of the hearing will be published in accordance with § 2.104 of this chapter.

(5) The Commission must refer a copy of the application for renewal to the ACRS. The ACRS must report on those portions of the application which concern safety and must apply the criteria set forth in § 53.1285.

(b) The Commission may grant the renewal if the Commission determines:

(1) The manufacturing license complies with the AEA and the Commission's regulations and orders applicable and in effect at the time the manufacturing license 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 manufacturing license 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.

(c) 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 must be subject to the requirements of § 53.1288.

§ 53.1300 Construction permits.

Sections 53.1300 through 53.1348 set out the requirements and procedures

applicable to Commission issuance of a construction permit for commercial nuclear plants. A construction permit for the construction of a commercial nuclear plant under Framework A of this part will be issued before the issuance of an operating license if the application is otherwise acceptable and will be converted upon completion of the facility and Commission action, into an operating license as provided under §§ 53.1360 through 53.1405.

§ 53.1306 Contents of applications for construction permits; general information.

An application for a construction permit must include the information required by § 53.1109 and the following information:

(a) Except for an application submitted by an electric utility applicant, information sufficient to demonstrate to the Commission the financial qualification of the applicant to carry out, under the regulations in this chapter, the activities for which the permit is sought. As applicable, the following should be provided:

(1) The information that demonstrates that the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover estimated construction costs and related fuel cycle costs, including estimates of the total construction costs and related fuel cycle costs of the facility and must indicate the source(s) of funds to cover these costs;

(2) Each application for a construction permit submitted by a newly-formed entity organized for the primary purpose of constructing and operating a facility must also include information showing:

(i) The legal and financial relationships the entity has or proposes to have with its stockholders or owners;

(ii) The stockholders' or owners' financial ability to demonstrate compliance with

any contractual obligation to the entity which they have incurred or proposed to incur; and

(iii) Any other information considered necessary by the Commission to enable it to determine the applicant's financial qualification; and

(3) The Commission may request an established entity or newly-formed entity to submit additional or more detailed information respecting its financial arrangements and status of funds if the Commission considers this information appropriate. This may include information regarding an applicant’s ability to continue the conduct of the activities authorized by the construction permit and to decommission the facility.

(b) If the applicant proposes to construct or alter a facility, the application must state the earliest and latest dates for completion of the construction or alteration.

§ 53.1309 Contents of applications for construction permits; technical information.

The application must contain a Preliminary Safety Analysis Report (PSAR) that describes the facility and the limits on its operation, and presents a preliminary safety analysis of the structures, systems, and components of the facility as a whole. The PSAR must include the following information, at a level of detail sufficient to enable the Commission to reach a conclusion on safety matters that must be resolved by the Commission before issuance of a construction permit:

(a) (1) *Site information.* An application for a construction permit for a commercial nuclear reactor must include the site information equivalent to that required for an early site permit in § 53.1146(a)(1)(iv) through (x).

(2) *Design information*. Except as specified in this paragraph, an application for a construction permit for a commercial nuclear plant must include the design information equivalent to that required for a standard design certification as defined in

§ 53.1239(a)(2) through (27).

(i) *Quality assurance program.* A description of the quality assurance program, as required by § 53.610(a)(6), applied to the design, fabrication, construction and testing of the structures, systems, and components of the facility.

(ii) *Preliminary design information.* The information provided in the application may include some aspects of the design that are not fully developed, and the information is therefore preliminary. The completed design, including any changes during construction, must be described in the FSAR required in § 53.1369 that supports an application for an operating license.

(iii) *Planned research or testing*. Descriptions of how design features and related functional design criteria will fulfil the safety criteria in subpart B, or more restrictive alternative criteria adopted under § 53.470, and how that has been or will be demonstrated through either analysis, appropriate test programs, experience, or a combination thereof. Where any design feature has not been fully developed or demonstrated to fulfill the functional design criteria at the time of an application for a construction permit, the applicant must provide a plan for future analysis, research and development, test programs, gathering of experience, or a combination thereof to provide reasonable confidence that the required demonstration will be available for an application for an operating license.

(iv) *Programmatic controls*. Descriptions of the programmatic controls may include those to be provided in the FSAR or other licensing basis documents because they are necessary to achieve and maintain the reliability and capability of SSCs relied upon to demonstrate compliance with the established safety criteria and functional design criteria required in subpart B, and to maintain consistency with analyses required by § 53.450.

(3) *Technical qualifications*. A description of the technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter.

(4) *Emergency preparedness*. A preliminary description of the plans for coping with emergencies with sufficient information to ensure the compatibility of proposed emergency plans for both onsite areas and the EPZs, with facility design features, site layout, and site location with respect to such considerations as access routes, surrounding population distributions, land use, and local jurisdictional boundaries for the EPZs as well as the means by which the standards of § 53.855 will be met. As a minimum, the following items must be described:

(i) Onsite and offsite organizations for coping with emergencies and the means for notification, in the event of an emergency, of persons assigned to the emergency organizations.

(ii) Contacts and arrangements made and documented with local, State, and Federal governmental agencies with responsibility for coping with emergencies, including identification of the principal agencies.

(iii) Protective measures to be to protect health and safety in the event of an accident; procedures by which these measures are to be carried out (e.g., in the case of an evacuation, who authorizes the evacuation, how the public is to be notified and instructed, how the evacuation is to be carried out); and the expected response of offsite agencies in the event of an emergency.

(iv) Features of the facility to be provided for onsite emergency first aid and decontamination and for emergency transportation of onsite individuals to offsite treatment facilities.

(v) Provisions to be made for emergency treatment at offsite facilities of

individuals injured as a result of licensed activities.

(vi) Provisions for a training program for employees of the licensee, including those who are assigned specific authority and responsibility in the event of an emergency, and for other persons who are not employees of the licensee but whose assistance may be needed in the event of a radiological emergency.

(vii) A preliminary analysis that projects the time and means to be employed in the notification of State and local governments and the public in the event of an emergency. A nuclear power plant applicant with an EPZ extending beyond the site boundary must perform a preliminary analysis of the time required to evacuate various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations, noting major impediments to the evacuation or taking of protective actions.

(viii) A preliminary analysis reflecting the need to include facilities, systems, and methods for identifying the degree of seriousness and potential scope of radiological consequences of emergency situations within and outside the site boundary, including capabilities for dose projection using real-time meteorological information and for dispatch of radiological monitoring teams within the EPZs; and a preliminary analysis reflecting the role of the emergency response facility(ies) in assessing information, recommending protective action, and disseminating information to the public.

(5) *Physical security*. A report that provides a preliminary description of how the site characteristics support the development of adequate security plans and measures consistent with the requirements in § 53.540.

(b) *Safeguards information.* A description of the program to protect Safeguards Information against unauthorized disclosure in accordance with the requirements in §§ 73.21 and 73.22 of this chapter, as applicable.

§ 53.1312 Contents of applications for construction permits; other application content.

(a) In addition to the PSAR, the application must include the following:

(1) An environmental report either under § 51.50(a) of this chapter if an LWA under § 53.1130 is not requested in conjunction with the construction permit application, or under §§ 51.49 and 51.50(a) of this chapter if an LWA is requested in conjunction with the construction permit application; or

(2) If the applicant wishes to request that an LWA under § 53.1130 be issued before issuance of the construction permit, the information otherwise required by § 53.1130, in accordance with either § 2.101(a)(1) through (a)(5), or § 2.101(a)(9) of this chapter.

(b) If the construction permit application references an early site permit, standard design approval, or standard design certification issued under Framework A of this part, then the following requirements apply:

(1) The PSAR need not contain information or analyses submitted to the Commission in connection with the referenced NRC approval, permit, or certification, provided, however, that the PSAR incorporates the material by reference and confirms that the site and design of the facility falls within parameter values postulated in the referenced NRC approval, permit, or certification.

(2) The PSAR must provide a means to demonstrate that all terms and conditions that have been included in the referenced NRC approval, permit, or certification will be satisfied by the date of issuance of the operating license, as appropriate. If the PSAR does not demonstrate that each site characteristic falls within the corresponding postulated site parameter and each design characteristic of the facility

falls within the corresponding postulated design parameter, the application must justify a departure, variance, or exemption from the referenced NRC approval, license, or certification in regard to that particular site or design characteristic in compliance with the requirements of Framework A of this part.

(3) If a referenced early site permit approves complete and integrated emergency plans, or major features of emergency plans, then the PSAR must include any new or additional information that updates and corrects the information that was provided under § 53.1146(b)(2) and discuss whether the new or additional information materially changes the bases for compliance with the applicable requirements.

§ 53.1315 Review of applications.

(a) *Standards for review of applications.* Applications filed under Framework A of this part will be reviewed according to the standards set out in 10 CFR parts 20, 51, 53, 73, and 140.

(b) *Administrative review of applications; hearings*. A proceeding on a construction permit application 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). All hearings on construction permit applications are governed by the procedures contained in 10 CFR part 2.

§ 53.1318 Finality of referenced NRC approvals, permits, and certifications.

If the application for a construction permit under this part references an early site permit, standard design approval, or standard design certification, the scope and nature

of matters resolved for the application are governed by the relevant provisions addressing finality, including §§ 53.1188, 53.1221, and 53.1263.

§ 53.1324 Referral to the Advisory Committee on Reactor Safeguards.

The Commission must refer a copy of the application to the ACRS. The ACRS must report on those portions of the application that concern safety and must apply the standards referenced in § 53.1315, in accordance with the finality provisions in § 53.1318.

§ 53.1327 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 § 53.1130, the applicant may not perform those activities without obtaining the separate authorization required by § 53.1130. Authorization may be granted only after the presiding officer in the proceeding on the application has made the findings and determination required by § 53.1130(b)(1)(ii) and (iv), and the Director of the Office of Nuclear Reactor Regulation makes the determination required by § 53.1130(b)(1)(iii).

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

§ 53.1330 Exemptions, departures, and variances.

(a) Applicants for a construction permit under this subpart, or any amendment to a construction permit, may include in the application a request for an exemption from

one or more of the Commission's regulations. The Commission may grant a request if it determines that the exemption complies with § 53.080.

(b) An applicant for a construction permit who has filed an application referencing an NRC approval, permit, or certification issued under Framework A of this part may include in the application a request for exemptions, departures, or variances related to the subject referenced NRC approval, permit, or certification. In determining whether to grant the departure, variance, or exemption, the Commission must apply the same technically relevant criteria as were applicable to the application for the original or renewed approval, license, or certification.

§ 53.1333 Issuance of construction permits.

(a) After conducting a hearing in accordance with § 53.1315 and receiving the report submitted by the ACRS, the Commission may issue a construction permit only if the Commission finds that:

(1) The applicant has described the proposed design of the facility and has identified the major features or components incorporated therein for the protection of the health and safety of the public;

(2) Such further technical or design information as may be required to complete the safety analysis, and which can reasonably be left for later consideration, will be supplied in the FSAR;

(3) Safety features or components, if any, which require research and development have been described by the applicant and the applicant has identified, and there will be conducted, a research and development program reasonably designed to resolve any safety questions associated with such features or components; and

(4) On the basis of the foregoing, there is reasonable assurance that,;

(i) Such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility; and

(ii) Taking into consideration the site criteria contained subpart D, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

(b) A construction permit must contain the terms and conditions for the permit, as the Commission deems necessary and appropriate. The Commission may, in its discretion, incorporate in any construction permit provisions requiring the applicant to furnish periodic reports of the progress and results of research and development programs designed to resolve safety questions.

§ 53.1336 Finality of construction permits.

Notwithstanding any provision in § 53.1590, a construction permit constitutes an authorization to proceed with construction but does not constitute Commission approval of the safety of any design feature or specification unless the applicant specifically requests such approval and such approval is incorporated in the permit. The applicant, at its option, may request such approvals in the construction permit or by amendment to the construction permit. If approved by the NRC and included in the permit, the NRC will consider modifications to the approved design features or specifications in accordance with § 53.1590.

§ 53.1342 Duration of construction permit.

(a) A construction permit will state the earliest and latest dates for completion of construction or alteration of the facility, not to exceed 40 years from date of issuance.

(b) If the proposed construction or alteration of the facility is not completed by the latest completion date, the construction permit must expire, and all rights forfeited. However, upon good cause shown, the Commission will extend the completion date for a reasonable period of time. The Commission will recognize, among other things, developmental problems attributed to the experimental nature of the facility or fire, flood explosion, strike, sabotage, domestic violence, enemy action an act of the elements, and other acts beyond the control of the permit holder, as a basis for extending the completion date.

§ 53.1345 Transfer of construction permits.

A construction permit may be transferred under § 53.1570.

§ 53.1348 Termination of construction permits.

When a permit holder has determined to permanently cease construction, the holder must, within 30 days, submit a written certification to the NRC.

§ 53.1360 Operating licenses.

Sections 53.1360 through 53.1405 set out the requirements and procedures applicable to Commission issuance of an operating license for a nuclear power facility.

§ 53.1366 Contents of applications for operating licenses; general information.

An application for an operating license must include the information required by § 53.1109 and the following information:

(a) Except for an electric utility applicant, information sufficient to demonstrate to

the Commission the financial qualification of the applicant to carry out, in accordance with the regulations in this chapter, the activities for which the license is sought. As applicable, the following should be provided:

(1) The applicant must submit information that demonstrates the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover estimated operation costs for the period of the license. The applicant must submit estimates for total annual operating costs for each of the first five years of operation of the facility. The applicant must also indicate the source(s) of funds to cover these costs.

(2) Each application for an operating license submitted by a newly-formed entity organized for the primary purpose of operating the facility must also include information showing:

(i) The legal and financial relationships the entity has or proposes to have with its stockholders or owners;

(ii) The stockholders' or owners' financial ability to demonstrate compliance with any contractual obligation to the entity which they have incurred or proposed to incur; and

(iii) Any other information considered necessary by the Commission to enable it to determine the applicant's financial qualification.

(3) The Commission may request an established entity or newly-formed entity to submit additional or more detailed information respecting its financial arrangements and status of funds if the Commission considers this information appropriate. This may include information regarding a licensee’s ability to continue the conduct of the activities authorized by the license and to decommission the facility.

(b) The application must include information in the form of a report, as described in subpart G, indicating how reasonable assurance will be provided that funds will be

available to decommission the facility.

§ 53.1369 Contents of applications for operating licenses; technical information.

Final Safety Analysis Report. The application must contain an FSAR that describes the facility and the limits on its operation, and presents a safety analysis of the structures, systems, and components of the facility as a whole. The FSAR must include the following information, at a level of detail sufficient to enable the Commission to reach a final conclusion on all safety matters that must be resolved by the Commission before issuance of an operating license. The application must include the following information:

(a) *Site information*. An application for an operating license for a commercial nuclear reactor must include the site information equivalent to that required for an early site permit in § 53.1146(a)(1)(iv) through (x), including all current information, such as the results of environmental and meteorological monitoring programs, which has been developed since issuance of the construction permit, relating to site evaluation factors identified in Framework A of this part.

(b) *Design information*. Except as specified in this paragraph, an application for an operating license for a commercial nuclear plant must include the final design information equivalent to that required for a standard design certification as defined in § 53.1239(a)(2) through (27).

(1) The completed design, including any changes during construction, must be described.

(2) Where any design feature had not been fully developed or demonstrated at the time of application for the construction permit, the applicant must provide the analysis, research and development, test programs, gathering of experience, or a combination thereof to provide the required demonstration to fulfill the functional design

criteria.

(c) *Technical qualifications*. A description of the technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter.

(d) *Integrity assessment program*. A description of an Integrity Assessment Program that addresses the elements described in § 53.870.

(e) *Safeguards information.* A description of the program to protect Safeguards Information against unauthorized disclosure in accordance with the requirements in §§ 73.21 and 73.22 of this chapter, as applicable.

(f) *Emergency response facility or facilities*. Description of location and capabilities to be established for command and control, support, and coordination of onsite and offsite, as applicable, functions during reactor accident conditions.

(g) *Role of personnel.* (1) A description of the completed assessments related to the role of personnel in ensuring safe operations considering the analyses required by § 53.730. These assessments must include the following:

(i) Human Factors Engineering Design Requirements of § 53.730(a);

(ii) Human System Interface Design Requirements of § 53.730(b);

(iii) Concept of Operations of § 53.730(c);

(iv) Functional Requirements Analysis and Function Allocation of § 53.730(d);

(2) A description of the programs to be used for the following as required by § 53.730(e):

(i) Evaluating and applying operating experience; and

(ii) Developing and maintaining plant procedures.

(3) A staffing plan and supporting analyses as required by § 53.730(f).

(h) *Training, examination, and proficiency programs.* (1) The training,

examination, and proficiency programs required by § 53.730(g);

(2) A description of the training programs required by § 53.830.

(i) *Emergency plan*. Emergency plans complying with the requirements of § 53.855.

(1) Include all emergency plan certifications, as applicable, that have been obtained from the State, local, and participating Tribal governmental agencies with emergency planning responsibilities that are wholly or partially within the EPZ plume exposure pathway. These certifications must state 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) These agencies are committed to executing their responsibilities under the plans in the event of an emergency.

(2) 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.

(3) If complete and integrated emergency plans were approved as part of an early site permit, or submitted, reviewed, and approved as part of the construction permit application, new certifications that demonstrate compliance with the requirements of paragraph (i)(1) of this section are not required.

(j) *Organization.* A description of the applicant's organizational structure, allocations or responsibilities and authorities, and personnel qualifications requirements for operation.

(k) *Maintenance program*. A description of a maintenance program that

demonstrates compliance with the requirements in § 53.715.

(l) *Quality assurance*. A description of the quality assurance program that demonstrates compliance with the requirements of § 53.865.

(m) *Radiation protection program*. A radiation protection program description that demonstrates compliance with the requirements of § 53.850.

(n) *Security program*. A physical security plan that describes how the applicant will demonstrate compliance with the requirements of § 53.860 (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.

(o) *Safeguards contingency plan*. A safeguards contingency plan in accordance with the criteria set forth in appendix C to 10 CFR part 73. The safeguards contingency plan must include plans for dealing with threats, thefts, and radiological sabotage, as defined in 10 CFR part 73, 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 must include the information contained in the applicant's safeguards contingency plan. (Implementing procedures required for this plan need not be submitted for approval.)[4]

(p) *Security training and qualification*. A training and qualification plan that describes how the applicant will demonstrate compliance with the criteria set forth in § 73.100 of this chapter or appendix B to 10 CFR part 73.

(q) *Cybersecurity plan*. A cybersecurity plan in accordance with the criteria set forth in § 73.54 or 73.110 of this chapter.

(r) *Security, safeguards and cybersecurity plan implementation*. A description of

the implementation of the physical security plan, safeguards contingency plan, security training and qualification plan, and cybersecurity plan. Each applicant who prepares a physical security program, a safeguards contingency plan, a security training and qualification plan, or a cybersecurity plan must protect the plans and other related Safeguards Information against unauthorized disclosure in accordance with the requirements of §§ 73.21 and 73.22 of this chapter.

(s) *Fire protection program*. A fire protection program description that demonstrates compliance with the requirements of § 53.875.

(t) *Inservice inspection/inservice testing program*. An ISI/IST program description that demonstrates compliance with the requirements of § 53.880.

(u) Reserved

(v) *Facility safety program.* An FSP plan that demonstrates compliance with the requirements of § 53.890.

(w) *General employee training*. A description of the training program required to demonstrate compliance with § 53.830 and its implementation.

(x) *Fitness-for-duty program*. A description of the FFD program required by 10 CFR part 26 and its implementation.

(y) A description and evaluation of the results of the applicant's programs, including research and development, if any, to demonstrate that any safety questions identified at the construction permit stage have been resolved.

(z) A description of how the performance of each safety design feature has been demonstrated capable of fulfilling functional design criteria considering interdependent effects through either analysis, appropriate test programs, prototype testing, operating experience, or a combination thereof, in accordance with § 53.440(a).

(aa) *Technical specifications*. Proposed technical specifications prepared in

accordance with the requirements of § 53.710(a).

* * * * *

[4] A physical security plan that contains all the information required in both 10 CFR 73.55 or 10 CFR 73.100 and appendix C to 10 CFR part 73 demonstrates compliance with the requirement for a contingency plan.

§ 53.1372 Contents of applications for operating licenses; other application content.

In addition to the FSAR, the application must also include the following:

(a) *Environmental report*. An environmental report in accordance with § 51.53(b) of this chapter.

(b) *Availability controls (if not included in the FSAR*). A description of the controls on plant operations, including availability controls, to provide reasonable confidence of safe operation and that the configurations and special treatments for NSRSS SSCs provide the capabilities and reliabilities required to demonstrate compliance with the safety criteria of § 53.220, or more restrictive alternative criteria adopted under § 53.470, if not addressed by Technical Specifications under § 53.1369(aa).

§ 53.1375 Review of applications.

(a) *Standards for review of applications.* Applications filed under this Framework A of this part will be reviewed according to the standards set out in 10 CFR parts 20, 26, 51, 53, 73, and 140. Upon receipt of an application, the NRC will:

(1) Give notice in writing to the regulatory agency or State as may have jurisdiction over the rates and services incident to the proposed activity;

(2) Publish notice of the application in trade or news publications as appropriate

to give reasonable notice to municipalities, private utilities, public bodies and cooperatives which might have a potential interest in the facility; and

(3) Publish notice of the application once each week for four consecutive weeks in the *Federal Register*.

(b) *Administrative review of applications; hearings*. A proceeding on an operating 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). All hearings on operating licenses are governed by the procedures contained in 10 CFR part 2.

§ 53.1381 Referral to the Advisory Committee on Reactor Safeguards.

The Commission must refer a copy of the application to the ACRS. The ACRS must report on those portions of the application that concern safety and must apply the standards referenced in § 53.1375.

§ 53.1384 Exemptions, departures, and variances.

(a) Applicants for an operating license under this subpart, or any amendment to an operating license, may include in the application a request for an exemption from one or more of the Commission's regulations. The Commission may grant an exemption request if it determines that the exemption complies with § 53.080.

(b) An applicant for an operating license who has filed an application referencing a NRC approval, permit, license, or certification issued under Framework A of this part may include in the application a request for, departures, variances, or exemptions related to the subject referenced NRC approval, permit, license, or certification. In

determining whether to grant the departure, variance, or exemption, the Commission must apply the same technically relevant criteria as were applicable to the application for the original or renewed approval, license, or certification.

§ 53.1387 Issuance of operating licenses.

(a)(1) After receiving the report submitted by the ACRS, the Commission may issue an operating license if the Commission finds that:

(i) Construction of the facility has been substantially completed in conformity with the construction permit and the application as amended, the provisions of the AEA, and the rules and regulations of the Commission;

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

(iii) The facility will operate in conformity with the application as amended, the provisions of the AEA, and the rules and regulations of the Commission;

(iv) There is reasonable assurance that:

(A) the activities authorized by the operating license can be conducted without endangering the health and safety of the public; and

(B) such activities will be conducted in compliance with the regulations in this chapter.

(v) The applicant is technically and financially qualified to engage in the activities authorized, however, no finding of financial qualification is necessary for an electric utility applicant for an operating license;

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

(vii) The applicable provisions of 10 CFR part 140 have been satisfied; and

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

(2) [Reserved]

(b) Fuel loading may not begin until the operating license is issued.

(c) The operating license may include appropriate provisions with respect to any uncompleted items of construction and such limitations or conditions as are required to assure that operation during the period of the completion of such items will not endanger public health and safety.

(d) The Commission will issue an operating license in such form and containing such conditions and limitations, including technical specifications, as it deems necessary and appropriate.

§ 53.1390 Finality of operating licenses.

After issuance of an operating license, the Commission may not modify, add, or delete any term or condition of the operating license, except in accordance with the provisions of § 53.1590.

§ 53.1396 Duration of operating license.

The Commission will issue an operating license under Framework A of this part for the term requested by the applicant, not to exceed 40 years from the date of issuance, or for the estimated useful life of the facility if the Commission determines that the estimated useful life is less than the term requested.

§ 53.1399 Transfer of an operating license.

An operating license may be transferred in accordance with § 53.1570.

§ 53.1402 Application for renewal.

The filing of an application for a renewed license must be in accordance with § 53.1595.

§ 53.1405 Continuation of an operating license.

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

(a) 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

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

§ 53.1410 Combined licenses.

Sections 53.1410 through 53.1461 set out the requirements and procedures applicable to Commission issuance of combined licenses for commercial nuclear plants under Framework A of this part.

§ 53.1413 Contents of applications for combined licenses; general information.

An application for a combined license must include the information required by § 53.1109 and the following information:

(a) Except for an electric utility applicant, the application must include information sufficient to demonstrate to the Commission the financial qualification of the applicant to carry out, in accordance with the regulations in this chapter, the activities for which the permit or license is sought. As applicable, the following should be provided:

(1) The applicant must submit information that demonstrates that the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover estimated construction costs and related fuel cycle costs. The applicant must submit estimates of the total construction costs of the facility and related fuel cycle costs and must indicate the source(s) of funds to cover these costs.

(2) The applicant must submit information that demonstrates the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover estimated operation costs for the period of the license. The applicant must submit estimates for total annual operating costs for each of the first five years of operation of the facility. The applicant must also indicate the source(s) of funds to cover these costs.

(3) Each application for a combined license submitted by a newly-formed entity organized for the primary purpose of constructing and operating a facility must also include information showing:

(i) The legal and financial relationships the entity has or proposes to have with its stockholders or owners; and

(ii) The stockholders' or owners' financial ability to demonstrate compliance with any contractual obligation to the entity which they have incurred or proposed to incur; and

(iii) Any other information considered necessary by the Commission to enable it to determine the applicant's financial qualification.

(4) The Commission may request an established entity or newly-formed entity to

submit additional or more detailed information respecting its financial arrangements and status of funds if the Commission considers this information appropriate. This may include information regarding a licensee's ability to continue the conduct of the activities authorized by the license and to decommission the facility.

(b) The application must include information in the form of a report, as described in subpart G of this part, indicating how reasonable assurance will be provided that funds will be available to decommission the facility.

§ 53.1416 Contents of applications for combined licenses; technical information.

(a) *Final Safety Analysis Report.* The application must contain an FSAR that describes the facility and the limits on its operation and presents a safety analysis of the structures, systems, and components of the facility as a whole. The Commission will require, before issuance of a combined license, that engineering documents, such as analyses, drawings, procurement specifications, or construction and installation specifications, be completed and available for audit if the more detailed information is necessary for the Commission to verify the information in the application and make its safety determination. The FSAR must include the following information, at a level of detail 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) *Site information*. An application for a combined license for a commercial nuclear reactor must include the site information equivalent to that required for an early site permit in § 53.1146(a)(1)(iv) through (x).

(2) *Design information*. An application for a combined license for a commercial nuclear plant must include the design information equivalent to that required for a

standard design certification as defined in § 53.1239(a)(2) through (27).

(3) *Technical qualifications*. A description of the technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter.

(4) *Integrity assessment program*. A description of an Integrity Assessment Program that addresses the elements described in § 53.870.

(5) *Safeguards information*. A description of the program to protect Safeguards Information against unauthorized disclosure in accordance with the requirements in §§ 73.21 and 73.22 of this chapter, as applicable.

(6) *Emergency response facility or facilities*. Description of the locations and capabilities to be established for command and control, support, and coordination of onsite and offsite, as applicable, functions during reactor accident conditions.

(7) *Role of personnel*. (i) A description of the completed assessments related to the role of personnel in ensuring safe operations considering the analyses required by § 53.730. These assessments must include:

(A) Human Factors Engineering Design Requirements of § 53.730(a);

(B) Human System Interface Design Requirements of § 53.730(b);

(C) Concept of Operations of § 53.730(c); and

(D) Functional Requirements Analysis and Function Allocation of § 53.730(d);

(ii) A description of the programs to be used for the following as required by § 53.730(e):

(A) Evaluating and applying operating experience; and

(B) Developing and maintaining plant procedures.

(iii) A staffing plan and supporting analyses as required by § 53.730(f).

(8) *Training, examination, and proficiency programs.* (i) The training,

examination, and proficiency programs required by § 53.730(g); and

(ii) A description of the training programs required by § 53.830.

(9) *Emergency plan*. Emergency plans complying with the requirements of § 53.855.

(i) Include, as applicable, all emergency plan certifications that have been obtained from the State, local and participating Tribal 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.

(10) *Organization.* A description of the applicant's organizational structure, allocations of responsibilities and authorities, and personnel qualifications requirements for operation.

(11) *Maintenance program*. A description of a maintenance program that demonstrates compliance with the requirements in § 53.715.

(12) *Quality assurance*. A description of the quality assurance program that demonstrates compliance with the requirements of § 53.865.

(13) *Radiation protection program*. A radiation protection program description that demonstrates compliance with the requirements of § 53.850.

(14) *Security program*. A physical security plan that describes how the applicant will demonstrate compliance with the requirements of § 53.860 (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.

(15) *Safeguards contingency plan*. A safeguards contingency plan in accordance with the criteria set forth in appendix C to 10 CFR part 73. The safeguards contingency plan must include plans for dealing with threats, thefts, and radiological sabotage, as defined in 10 CFR part 73, 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 must include the information contained in the applicant's safeguards contingency plan.[5] (Implementing procedures required for this plan need not be submitted for approval.)

(16) *Security training and qualification*. A training and qualification plan that describes how the applicant will demonstrate compliance with the criteria set forth in § 73.100 of this chapter or appendix B to 10 CFR part 73.

(17) *Cybersecurity plan*. A cybersecurity plan in accordance with the criteria set forth in 10 CFR 73.54 or 73.110.

(18) *Security plan implementation*. A description of the implementation of the safeguards contingency plan, training and qualification plan, and cybersecurity plan. Each applicant who prepares a physical security plan, a safeguards contingency plan, a training and qualification plan, or a cybersecurity plan, must protect the plans and other related Safeguards Information against unauthorized disclosure in accordance with the requirements of §§ 73.21 and 73.22 of this chapter.

(19) *Fire protection program*. A fire protection program description that demonstrates compliance with the requirements of § 53.875.

(20) *Inservice inspection/inservice testing program*. An ISI/IST program description that demonstrates compliance with the requirements of § 53.880.

(21) Reserved.

(22) *Facility safety program*. An FSP plan that demonstrates compliance with the requirements of § 53.890.

(23) *General employee training*. A description of the training program required to demonstrate compliance with § 53.830 and its implementation.

(24) *Fitness-for-duty program*. A description of the FFD program required by 10 CFR part 26 and its implementation.

(25) *Technical specifications*. Proposed technical specifications prepared in accordance with the requirements of § 53.710(a).

(b) If there are SSCs of the plant for which research and development is necessary to confirm the adequacy of their design, a report which documents the resolution of any safety questions associated with such SSCs.

(c) A description of how the performance of each safety design feature has been demonstrated capable of fulfilling functional design criteria considering interdependent effects through either analysis, appropriate test programs, prototype testing, operating experience, or a combination thereof, in accordance with § 53.440(a).

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

(1) The FSAR need not contain information or analyses submitted to the Commission in connection with the early site permit provided that the FSAR either include or incorporate by reference the early site permit Site Safety Analysis Report and

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 FSAR does not demonstrate that design of the facility falls within the site characteristics and design parameters, the application must include a request for a variance that complies with the requirements of §§ 53.1188(d) and 53.1437.

(3) The FSAR must demonstrate that all terms and conditions that have been included in the early site permit 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 FSAR must include any new or additional information that updates and corrects the information that was provided under § 53.1146(b)(2), 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 change in an emergency plan that results in reducing the licensee's capability to perform an emergency planning function in the event of a radiological emergency.

(5) If complete and integrated emergency plans are approved as part of the early site permit, new certifications meeting the requirements of paragraph (a)(9)(i) of this section are not required.

(e) If the combined license application references a standard design approval,

then the following requirements apply:

(1) The FSAR need not contain information or analyses submitted to the Commission in connection with the design approval, provided, however, that the FSAR must either include or incorporate by reference the standard design approval FSAR 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 FSAR must demonstrate that all terms and conditions that have been included in the design approval will be satisfied by the date of issuance of the combined license.

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

(1) The FSAR need not contain information or analyses submitted to the Commission in connection with the standard design certification, provided, however, that the FSAR must either include or incorporate by reference the standard design certification FSAR 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 standard design certification. In addition, the plant specific PRA information must use the PRA information for the standard design certification and must be updated to account for site-specific design information and any design changes or departures.

(2) The FSAR must demonstrate that the interface requirements established for the design under § 53.1239(a)(24) have been met.

(3) The FSAR must demonstrate that all requirements and restrictions set forth in the referenced standard design certification rule must be satisfied by the date of issuance of the combined license. Any requirements and restrictions set forth in the referenced standard 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.

(g) If the combined license application references the use of one or more manufactured reactors licensed under § 53.1270, then the following requirements apply:

(1) The FSAR need not contain information or analyses submitted to the Commission in connection with the manufacturing license, provided, however, that the FSAR must either include or incorporate by reference the manufacturing license FSAR 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 FSAR must demonstrate that the interface requirements established for the design have been met.

(3) The FSAR must demonstrate that all terms and conditions that have been included in the manufacturing license 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.

(h) Each applicant for a combined license under this part must protect

Safeguards Information against unauthorized disclosure in accordance with the requirements in §§ 73.21 and 73.22 of this chapter, as applicable.

* * * * *

[5] A physical security plan that contains all the information required in both 10 CFR 73.55 or 10 CFR 73.100 and appendix C to 10 CFR part 73 demonstrates compliance with the requirement for a contingency plan.

§ 53.1419 Contents of applications for combined licenses; other application content.

(a) In addition to the FSAR, the application must also include the following:

(1) *Environmental report.* (i) An environmental report either in accordance with § 51.50(c) of this chapter if an LWA under § 53.1130 is not requested in conjunction with the combined license application, or in accordance with §§ 51.49 and 51.50(c) of this chapter if an LWA is requested in conjunction with the combined license application; or

(ii) If the applicant wishes to request that an LWA under § 53.1130 be issued before issuance of the combined license, the information otherwise required by § 53.1130, in accordance with either § 2.101(a)(1) through (a)(4), or 2.101(a)(9) of this chapter;

(2) *Availability controls (if not included in the FSAR).* A description of the controls on plant operations, including availability controls, to provide reasonable confidence of safe operation and that the configurations and special treatments for NSRSS SSCs provide the capabilities and reliabilities required to demonstrate compliance with the safety criteria of § 53.220, or more restrictive alternative criteria adopted under § 53.470, if not addressed by Technical Specifications under § 53.1416(a)(25); and

(3) *ITAAC.* The proposed inspections, tests, and analyses, including those

applicable to emergency planning, that the licensee must 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 AEA, and the Commission's rules and regulations.

(i) 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.

(ii) 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 standard design certification.

(iii) If the application references a manufacturing license, the ITAAC contained in the manufacturing license must apply to those portions of the facility design which are approved in the manufacturing license.

(iv) If the application references an early site permit with ITAAC, a standard design certification, a manufacturing license, or combination thereof, 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 § 53.1422 of this chapter must indicate that the application includes this notification.

§ 53.1422 Review of applications.

(a) *Standards for review of applications.* Applications filed under Framework A of this part will be reviewed according to the standards set out in 10 CFR parts 20, 51, 53, 73, and 140.

(b) *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.

§ 53.1425 Finality of referenced NRC approvals.

If the application for a combined license under Framework A of this part references an early site permit, standard design certification rule, standard design approval, or manufacturing license, issued under this part, the scope and nature of matters resolved for the application and any combined license issued are governed by the relevant provisions addressing finality, including §§ 53.1188, 53.1263, 53.1221, and 53.1288.

§ 53.1431 Referral to the Advisory Committee on Reactor Safeguards.

The Commission must refer a copy of the application to the ACRS. The ACRS must report on those portions of the application that concern safety and must apply the standards referenced in § 53.1422, in accordance with the finality provisions in § 53.1425.

§ 53.1434 Authorization to conduct limited work authorization activities.

(a) If the application for a combined license under Framework A of this part does

not reference an early site permit which authorizes the holder to perform the activities under § 53.1130(b), the applicant may not perform those activities without obtaining the separate authorization required by § 53.1130(a). Authorization may be granted only after the presiding officer in the proceeding on the application has made the findings and determination required by § 53.1130(b)(1)(ii) and (b)(1)(iv), and the Director of the Office of Nuclear Reactor Regulation makes the determination required by § 53.1130(b)(1)(iii).

(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 must implement the approved site redress plan.

§ 53.1437 Exemptions, departures, and variances.

(a) Applicants for a combined license, 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 standard design certification rule, the Commission may grant the request if it determines that the exemption complies with any exemption provisions of the referenced standard design certification rule, or with § 53.1263 if there are no applicable exemption provisions in the referenced standard design certification rule.

(2) For all other requests for exemptions, the Commission may grant a request if it determines that the exemption complies with § 53.080.

(b) An applicant for a combined license who has filed an application referencing an early site permit issued under § 53.1158 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 must apply the same technically relevant criteria as were applicable to the application for the original or renewed site permit. Once a 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 § 53.1270 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 § 53.080, 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) of this section 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.

§ 53.1440 Issuance of combined licenses.

(a)(1) After conducting a hearing in accordance with § 53.1422(b) 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 AEA 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 AEA, and the Commission's

regulations;

(iv) The applicant is technically and financially qualified to engage in the activities authorized; however, no finding of financial qualification is necessary for an electric utility applicant for a combined license;

(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 10 CFR part 51 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 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 § 53.1452(g) with respect to those acceptance criteria are unnecessary.

(b) The Commission must identify within the combined license the inspections, tests, and analyses, including those applicable to emergency planning, that the licensee must 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 AEA, and the Commission's rules and regulations.

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

§ 53.1443 Finality of combined licenses.

(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

ITAAC contained in the license that are not derived from a referenced standard design certification or manufacturing license, except in accordance with the provisions of §§ 53.1452 or 53.1590.

(b) If the combined license does not reference a standard design certification or a reactor manufactured under § 53.1270, then a licensee may make changes in the facility as described in the FSAR (as updated), make changes in the procedures as described in the FSAR (as updated), and conduct tests or experiments not described in the FSAR (as updated) under the applicable change processes in subpart I.

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

(1) Changes to or departures from information within the scope of the referenced standard 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 standard design certification rule are subject to the applicable change processes in subpart I of this part, unless they also involve changes to or noncompliance with information within the scope of the referenced standard design certification rule. In these cases, the applicable provisions of this section and the standard design certification rule apply.

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

(1) Changes to or departures from information within the scope of the manufactured reactor's design are subject to the change processes in § 53.1288; and

(2) Changes that are not within the scope of the manufactured reactor's design are subject to the applicable change processes in subpart I.

(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 § 53.1515.

(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, and 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.

§ 53.1449 Inspection during construction.

(a) *Licensee schedule for inspections, tests, or analyses*. The licensee must submit to the NRC, no later than 1 year after issuance of the combined license or at the start of construction as defined at § 53.020, whichever is later, its schedule for completing the inspections, tests, or analyses in the ITAAC. The licensee must 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 must 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) *Licensee and applicant conduct of activities subject to ITAAC*. 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 preoperational activities, even though the NRC may not have found that any one of the prescribed acceptance criteria are met.

(c) *Licensee notifications*. (1) ITAAC closure notification. The licensee must notify the NRC that prescribed inspections, tests, and analyses have been performed and that the prescribed acceptance criteria are met. The notification must contain sufficient information to demonstrate that the prescribed inspections, test, and analyses have been performed and that the prescribed acceptance criteria are met.

(2) *ITAAC post-closure notifications*. Following the licensee's ITAAC closure notifications under paragraph (c)(1) of this section until the Commission makes the finding under 53.1452(g), the licensee must notify the NRC, in a timely manner, of new information that materially alters the basis for determining that either inspections, tests, and analyses were performed as required, or that acceptance criteria are met. The notification must contain sufficient information to demonstrate that, notwithstanding the new information, the prescribed inspections, tests, and analyses have been performed as required, and the prescribed acceptance criteria are met.

(3) *Uncompleted ITAAC notification*. 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 must notify the NRC that the prescribed inspections, tests, and 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, and 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 are met.

(4) *All ITAAC complete notification*. The licensee must notify the NRC that all ITAAC are complete.

(d) *Licensee determination of noncompliance with ITAAC*. (1) In the event that an activity is subject to a ITAAC derived from a referenced standard design certification and the licensee has not demonstrated that the prescribed acceptance criteria are 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 subpart I.

(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 prescribed acceptance criteria are met, the licensee may take corrective actions to successfully complete that ITAAC or request a license amendment under subpart I.

(e) *NRC inspection, publication of notices, and availability of licensee notifications*. The NRC must 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 53.1452, the NRC must publish notices in the *Federal Register* of the NRC staff's determination of the successful completion of inspections, tests, and analyses.

(2) The NRC must make publicly available the licensee notifications under paragraph (c) of this section. The NRC must, no later than the date of publication of the notice of intended operation required by 53.1452(a), make publicly available those licensee notifications under paragraph (c) of this section that have been submitted to the NRC at least seven days before that notice.

§ 53.1452 Operation under a combined license.

(a) The licensee must notify the NRC of its scheduled date for initial loading of fuel no later than 270 days before the scheduled date and must 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 Framework A of this part, the Commission must 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 must not be granted for those ITAAC which the Commission found were met under § 53.1440(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, must determine whether to grant or deny the request for hearing in accordance with the applicable requirements of § 2.309 of this chapter. If the Commission grants the request, the Commission, acting as the presiding officer, must 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 must allow operation during an interim period under the combined license.

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

(e) The Commission must, 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 § 2.206 of this chapter. The petitioner must 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 must 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 must 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 § 53.1440(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 FSAR (as updated) must comply with the requirements in § 53.1443(e) or (f), as applicable.

§ 53.1455 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 § 53.1452(g) or allowing operation during an interim period under the combined license under § 53.1452(c).

§ 53.1456 Transfer of a combined license.

A combined license may be transferred under § 53.1570.

§ 53.1458 Application for renewal.

The filing of an application for a renewed license must be in accordance with § 53.1595.

§ 53.1461 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 facility, until the Commission notifies the licensee in writing that the license is terminated. During this period of continued effectiveness, the licensee must:

(a) 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

(b) 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.

§ 53.1470 Standardization of commercial nuclear power plant designs: licenses to construct and operate nuclear power reactors of identical design at multiple sites.

(a) Except as otherwise specified in this section, the provisions of this section apply to construction permit, operating license, and combined license applications under Framework A of this part.

(b) Each application for a construction permit, operating license, or combined license submitted pursuant to this section must be submitted as specified in §§ 53.1300, 53.1360, or 53.1410 and § 2.101 of this chapter. Each application should state that the applicant wishes to have the application considered under this section and should list each of the applications to be treated together under this section.

(c) Each application must include the information required by the applicable sections of this subpart, *provided however*, that the application must identify the common design, and, if applicable, reference a standard design certification or standard design approval under Framework A of this part, or the use of a reactor manufactured under Framework A of this part. The FSAR for each application must either incorporate by reference or include the final safety analysis of the common design, including, if applicable, the FSAR for the referenced standard design certification or the manufactured reactor.

(d) Each application submitted pursuant to this section must contain an environmental report as required by §§ 53.1312, 53.1372, or, 53.1419, as applicable, 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.

(e) Upon a determination that each application is acceptable for docketing under § 2.101 of this chapter, each application will be docketed and a notice of docketing for each application will be published in the *Federal Register*, in accordance with § 2.104 of this chapter, *provided, however*, that the notice must state that the application will be processed under the provisions of this section and subpart D of 10 CFR part 2. At the discretion of the Commission, a single notice of docketing for multiple applications may be published in the *Federal Register.*

(f) The NRC must prepare an environmental assessment or draft and final environmental impact statements for each of the applications under 10 CFR part 51. Scoping under §§ 51.28 and 51.29 of this chapter for each of the 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 assessment or environmental impact statement for each of the applications must incorporate by reference the standard design certification environmental assessment. If the applications do not reference a standard design certification, then the NRC must prepare environmental assessments or 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 assessment or environmental impact statement prepared for each application. Scoping under §§ 51.28 and 51.29 of this

chapter for the supplemental environmental impact statement may be conducted simultaneously and may be part of the scoping for each of the applications.

(g) The ACRS must report on each of the applications as required by the applicable sections of this subpart. Each report must be limited to those safety matters for each application which are not relevant to the common design. In addition, the ACRS must 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 Framework A of this part.

(h) The Commission must 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 10 CFR part 2 relating to applications for construction permits, operating licenses, and combined licenses. The presiding officer must issue a partial initial decision on the common design.

(i) 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 section and subpart D of 10 CFR part 2.

(j) As used in this section, the design of a nuclear power reactor included in a single referenced Safety Analysis Report means the design of those structures, systems, and components important to radiological health and safety and the common defense and security.

§ 53.1480 Limited combined license supporting testing of manufactured reactor modules.

[Under development]

Subpart I—Maintaining and Revising Licensing Basis Information

§ 53.1500 Licensing basis information.

This subpart provides the requirements for each holder of a license for a commercial nuclear plant licensed under Framework A of this part to maintain licensing basis information as defined in § 53.020; evaluate changes to site characteristics, plant design features, and programmatic controls to determine needed approvals and revisions; and submit appropriate updates to the NRC.

§ 53.1502 Specific terms and conditions of licenses.

(a) Each license issued under Framework A of this part is subject to the provisions of the AEA and to all rules, regulations, and orders of the Commission. The terms and conditions of the license shall be subject to amendment, revision, or modification, by reason of amendments of the Act or by reason of rules, regulations, and orders issued in accordance with the terms of the Act.

(b) Each license issued under Framework A of this part must be subject to all conditions imposed as a matter of law by sections 401(a)(2) and 401(d) of the Federal Water Pollution Control Act, as amended (33 U.S.C.A. 1341(a)(2) and (d)).

(c) A holder of an operating license or combined license under Framework A of this part may take reasonable action that departs from a license condition or a technical specification (included in a license issued under Framework A of this part) in a national security emergency:

(1) When this action is immediately needed to implement national security objectives as designated by the national command authority through the Commission, and

(2) No action consistent with license conditions and technical specifications that can satisfy national security objectives is immediately apparent.

A national security emergency is established by a law enacted by the Congress or by an order or directive issued by the President pursuant to statutes or the Constitution of the United States. The authority under this paragraph must be exercised in accordance with law, including section 57e. of the Act, and is in addition to the authority granted under § 53.740(h), which remains in effect unless otherwise directed by the Commission during a national security emergency.

(d)(1) If the NRC finds that the state of emergency preparedness does not provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency (including findings based on requirements of appendix E, section IV.D.3) and if the deficiencies (including deficiencies based on requirements of appendix E, section IV.D.3) are not corrected within four months of that finding, the Commission will determine whether the facility shall be shut down or cease operations until such deficiencies are remedied or whether other enforcement action is appropriate. In determining whether a shutdown, cessation of operations, or other enforcement action is appropriate, the Commission shall take into account, among other factors, whether the licensee can demonstrate to the Commission’s satisfaction that the deficiencies in the plan are not significant for the plant in question, or that adequate interim compensating actions have been or will be taken promptly, or that that there are other compelling reasons for continued operation.

(2) If the planning standards for radiological emergency preparedness apply to offsite emergency response plans, then the NRC will base its finding on a review of the FEMA findings and determinations as to whether State and local emergency plans are adequate and capable of being implemented, and on the NRC assessment as to

whether the licensee's emergency plans are adequate and capable of being implemented. Nothing in this paragraph shall be construed as limiting the authority of the Commission to take action under any other regulation or authority of the Commission or at any time other than that specified in this paragraph.

§ 53.1505 Changes to licensing basis information requiring NRC approval.

(a) Sections 53.1510 through 53.1520 provide the process for a licensee to request and the NRC to issue amendments to licenses, including any conditions contained therein, technical specifications or other attachments to a license, and any orders issued by the NRC modifying a license. Sections 53.1525 and 53.1530 govern proposed changes to a commercial nuclear plant referencing a certified design or manufacturing license.

(b) A licensee may propose changing licensing basis information established by NRC regulations by requesting an exemption in accordance with § 53.080.

§ 53.1510 Application for amendment of license.

Whenever a holder of a license under Framework A of this part desires to amend the license, an application for an amendment must be filed with the Commission, as specified in § 53.040, that fully describes the changes desired, and following as far as applicable, the form prescribed for original applications. Applications for amendments involving changes to plant SSCs, programmatic controls, or the role of plant personnel must include an assessment of the changes in relation to the safety requirements in subpart B of this part and the analyses requirements of § 53.450, an analysis of whether the amendment involves no significant hazards consideration using the standards in § 53.1520, and a consideration of environmental factors.

§ 53.1515 Public notices; state consultation.

The Commission will use the following procedures for an application requesting an amendment to an OL or COL issued under Framework A of this part.

(a) *Public notices.*

(1)(i) The Commission may publish in the *Federal Register* under § 2.105 an individual notice of proposed action for an amendment for which it makes a proposed determination that no significant hazards consideration is involved, or, at least once every 30 days, publish a periodic *Federal Register* notice of proposed actions, which identifies each amendment issued and each amendment proposed to be issued since the last such periodic notice, or it may publish both such notices.

(ii) For each amendment proposed to be issued, the notice will (A) contain the staff's proposed determination, under the standards in § 53.1520, (B) provide a brief description of the amendment and of the facility involved, (C) solicit public comments on the proposed determination, and (D) provide for a 30-day comment period.

(iii) The comment period will begin on the day after the date of the publication of the first notice, and, normally, the amendment will not be granted until after this comment period expires.

(2) The Commission may inform the public about the final disposition of an amendment request for which it has made a proposed determination of no significant hazards consideration either by issuing an individual notice of issuance under § 2.106 of this chapter or by publishing such a notice in its periodic system of *Federal Register* notices. In either event, it will not make and will not publish a final determination of no significant hazards consideration, unless it receives a request for a hearing on that amendment request.

(3) Where the Commission makes a final determination that no significant hazards consideration is involved and that the amendment should be issued, the amendment will be effective on issuance, even if adverse public comments have been received and even if an interested person meeting the provisions for intervention called for in § 2.309 of this chapter has filed a request for a hearing. The Commission need hold any required hearing only after it issues an amendment, unless it determines that a significant hazards consideration is involved, in which case the Commission will provide an opportunity for a prior hearing.

(4) Where the Commission finds that an emergency situation exists, in that failure to act in a timely way would result in derating or shutdown of a commercial nuclear reactor, or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, it may issue a license amendment involving no significant hazards consideration without prior notice and opportunity for a hearing or for public comment. In such a situation, the Commission will not publish a notice of proposed determination on no significant hazards consideration, but will publish a notice of issuance under § 2.106 of this chapter, providing for opportunity for a hearing and for public comment after issuance. The Commission expects its licensees to apply for license amendments in timely fashion. It will decline to dispense with notice and comment on the determination of no significant hazards consideration if it determines that the licensee has abused the emergency provision by failing to make timely application for the amendment and thus itself creating the emergency. Whenever an emergency situation exists, a licensee requesting an amendment must explain why this emergency situation occurred and why it could not avoid this situation, and the Commission will assess the licensee's reasons for failing to file an application sufficiently in advance of that event.

(5) Where the Commission finds that exigent circumstances exist, in that a licensee and the Commission must act quickly and that time does not permit the Commission to publish a *Federal Register* notice allowing 30 days for prior public comment, and it also determines that the amendment involves no significant hazards considerations, it:

(i)(A) Will either issue a *Federal Register* notice providing notice of an opportunity for hearing and allowing at least two weeks from the date of the notice for prior public comment; or

(B) Will use local media to provide reasonable notice to the public in the area surrounding a licensee's facility of the licensee's amendment and of its proposed determination as described in paragraph (a)(1) of this section, consulting with the licensee on the proposed media release and on the geographical area of its coverage;

(ii) Will provide for a reasonable opportunity for the public to comment, using its best efforts to make available to the public whatever means of communication it can for the public to respond quickly, and, in the case of telephone comments, have these comments recorded or transcribed, as necessary and appropriate;

(iii) When it has issued a local media release, may inform the licensee of the public's comments, as necessary and appropriate;

(iv) Will publish a notice of issuance under § 2.106;

(v) Will provide a hearing after issuance, if one has been requested by a person who satisfies the provisions for intervention specified in § 2.309 of this chapter; and

(vi) Will require the licensee to explain the exigency and why the licensee cannot avoid it and use its normal public notice and comment procedures in paragraph (a)(1) of this section if it determines that the licensee has failed to use its best efforts to make a

timely application for the amendment in order to create the exigency and to take advantage of this procedure.

(6) Where the Commission finds that significant hazards considerations are involved, it will issue a *Federal Register* notice providing an opportunity for a prior hearing even in an emergency situation, unless it finds an imminent danger to the health or safety of the public, in which case it will issue an appropriate order or rule under 10 CFR part 2.

(b) *State consultation*.

(1) At the time a licensee requests an amendment, it must notify the State in which its facility is located of its request by providing that State with a copy of its application and its reasoned analysis about no significant hazards considerations and indicate on the application that it has done so.

(2) The Commission will advise the State of its proposed determination about no significant hazards consideration normally by sending it a copy of the *Federal Register* notice.

(3) The Commission will make the names of the Project Manager or other NRC personnel it designated to consult with the State available to the State official designated to consult about its proposed determination. The Commission will consider any comments of that State official. If it does not hear from the State in a timely manner, it will consider that the State has no interest in its determination; nonetheless, to ensure that the State is aware of the application, before it issues the amendment, it will make a good faith effort to communicate directly with that official. (Inability to consult with a responsible State official following good faith attempts will not prevent the Commission from making effective a license amendment involving no significant hazards consideration.)

(4) The Commission will make a good faith attempt to consult with the State before it issues a license amendment involving no significant hazards consideration. If, however, it does not have time to use its normal consultation procedures because of an emergency situation, it will attempt to communicate directly with the appropriate State official. (Inability to consult with a responsible State official following good faith attempts will not prevent the Commission from making effective a license amendment involving no significant hazards consideration, if the Commission deems it necessary in an emergency situation.)

(5) After the Commission issues the requested amendment, it will send a copy of its determination to the State.

(c) *Caveats about State consultation.*

(1) The State consultation procedures in paragraph (b) of this section do not give the State a right:

(i) To veto the Commission's proposed or final determination;

(ii) To a hearing on the determination before the amendment becomes effective; or

(iii) To insist upon a postponement of the determination or upon issuance of the amendment.

(2) These procedures do not alter present provisions of law that reserve to the Commission exclusive responsibility for setting and enforcing radiological health and safety requirements for commercial nuclear power plants.

§ 53.1520 Issuance of amendment.

(a) In determining whether an amendment to a license will be issued to the applicant, the Commission will be guided by the considerations which govern the

issuance of initial licenses to the extent applicable and appropriate. If the application involves the material alteration of a licensed facility, a construction permit will be issued before the issuance of the amendment to the license, provided however, that if the application involves a material alteration to a manufactured reactor under Framework A of this part before its installation at a site, or a combined license before the date that the Commission makes the finding under § 53.1452(g), no application for a construction permit is required. If the amendment involves a significant hazards consideration, the Commission will give notice of its proposed action:

(1) Under § 2.105 of this chapter before acting thereon; and

(2) As soon as practicable after the application has been docketed.

(b) The Commission will be particularly sensitive to a license amendment request that involves irreversible consequences (such as one that permits a significant increase in the amount of effluents or radiation emitted by a commercial nuclear power plant).

(c) The Commission may make a final determination, under the procedures in § 53.1515, that a proposed amendment to an operating license or a combined license for a commercial nuclear plant under Framework A of this part involves no significant hazards consideration, if operation of the plant in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of an accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

§ 53.1525 Revising certification information within a design certification rule.

(a) A holder of an operating license or combined license who references a design certification rule issued under Framework A of this part must request an exemption if proposing to change 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 § 53.080 and that the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the departure.

(b) The request for an exemption must be included with any associated license amendment request, which must be requested and processed in accordance with §§ 53.1510, 53.1515, and 53.1520.

(c) Licensees must evaluate changes to the design as described in the FSAR not involving changes to the certification information using the criteria in § 53.1550.

§ 53.1530 Revising design information within a manufacturing license.

(a) The holder of a manufacturing license may not make changes to the design of the manufactured reactor or manufactured reactor module authorized to be manufactured without obtaining an amendment pursuant to § 53.1510 and as applicable, 53.1520.

(b) The holder of a combined license under Framework A of this part who references or uses a manufactured reactor under Framework A of this part must request approval for any proposed departure from the design characteristics, site parameters, terms and conditions, or approved design of the manufactured reactor. The application for such departures must be submitted and processed in accordance with §§ 53.1510, 53.1515, and 53.1520. In those cases where a manufacturing license references a design certification rule, the amendment application from the holder of the combined

license must also request an exemption from the design certification rule in accordance with § 53.1525 if one or more elements of the certification information are adversely affected by the proposed change. The holder of the combined licensee must evaluate changes to the commercial nuclear plant as described in the FSAR but outside of the scope of the referenced manufacturing license using the criteria in § 53.1550.

§ 53.1535 Amendments during construction.

(a) The holder of a construction permit or LWA under Framework A of this part may request an amendment to the construction permit or LWA in order to gain Commission approval of the safety of selected design features or specifications, including proposed departures from a design certification rule or manufacturing license. Amendments to construction permits or LWAs under Framework A of this part must be requested and processed in accordance with §§ 53.1510 and 53.1520.

(b) The holder of a combined license under Framework A of this part for which the NRC has not yet made a finding in accordance with § 53.1452(g) must request amendments required by §§ 53.1525 or 53.1550 no later than 45 days from the date the licensee begins the construction of the SSCs to implement the change or departure requiring NRC approval. The licensee proceeds with such changes at its own risk recognizing that there is a possibility that the amendment will not be granted.

§ 53.1540 Updating licensing basis information and determining the need for NRC approval.

(a) Sections 53.1545 through 53.1565 provide the process for a holder of an OL or COL to modify licensing basis information and to evaluate potential changes to its facilities, procedures, programs, and organizations to determine if NRC approval is

required. These sections also apply to the holder of a license authorizing operation of a nuclear power reactor that has submitted the certification of permanent cessation of operations required under § 53.1070(a) or a reactor licensee whose license has been amended to allow possession of nuclear fuel but not operation of the facility.

(b) Definitions for the purposes of §§ 53.1545 through 53.1565:

(1) Change means a modification or addition to, or removal from, the commercial nuclear plant or procedures that affects a safety function, method of performing or controlling the function, or an evaluation that demonstrates that intended functions will be accomplished.

(2) Departure from a method of evaluation described in the Updated Final Safety Analysis Report (UFSAR) used in establishing the functional design criteria for SR SSCs or in the safety analyses means:

(i) Changing any of the elements of the method described in the UFSAR unless the results of the analysis are conservative or essentially the same; or

(ii) Changing from a method described in the FSAR to another method unless that method has been approved by NRC for the intended application.

(3) Facility as described in the UFSAR means:

(i) The structures, systems, and components (SSC) that are described in the UFSAR,

(ii) The design and performance requirements for such SSCs described in the UFSAR, and

(iii) The evaluations or methods of evaluation included in the UFSAR for such SSCs which demonstrate that their intended function(s) will be accomplished.

(4) *Final Safety Analysis Report (as updated)* means the FSAR submitted in accordance with §§ 53.1369 or 53.1416, as amended and supplemented, and as updated per the requirements in § 53.1545, as applicable.

(5) *Procedures as described in the Final Safety Analysis Report (as updated)* means those procedures that contain information described in the UFSAR such as how structures, systems, and components are operated and controlled (including assumed operator actions and response times).

§ 53.1545 Updating Final Safety Analysis Reports.

(a) Each holder of an operating license or combined license under Framework B of this part for which the Commission has made the finding under § 53.1452(g) must update the FSAR originally submitted as part of the application for the license every 24 months or more frequently to assure that the information included in the report contains the latest information developed. The submittal must include the effects on the content of the FSAR of:

(1) changes made to the facility or procedures as described in the FSAR;

(2) safety analyses and evaluations performed by the licensee either in support of approved license amendments or in support of conclusions that changes did not require a license amendment in accordance with § 53.1550;

(3) updates to the probabilistic risk assessments required under § 53.450;

(4) the cumulative effects of the changes to the facility or procedures on the margins to the safety criteria in §§ 53.210, 53.220, 53.450(e), and 53.470 since the last FSAR update.

(5) analyses of new safety issues performed by or on behalf of the licensee at Commission request.

(b)(1) The licensee must submit revisions containing updated information to the Commission, as specified in § 53.040, identifying the location of revised or new information.

(2) The submittal must include:

(i) a certification by a duly authorized officer of the licensee that either the information accurately presents changes made since the previous submittal, necessary to reflect information and analyses submitted to the Commission or prepared pursuant to Commission requirement, or that no such changes were made; and

(ii) an identification of changes made under the provisions of § 53.1550 but not previously submitted to the Commission.

(c) Each applicant for or holder of a combined license under Framework A of this part for which the Commission has not made the finding under § 53.1452(g) must submit an update to the FSAR annually by providing the information required in (a)(1) through (a)(5) of this section and meeting the requirements of paragraph (b) of this section. Combined license applicants who have requested the NRC to suspend its review of the combined license application and combined license holders who have informed the NRC that they do not plan to pursue construction need not submit an annual update of the FSAR. If a combined license applicant requests that the NRC resume its review, or a combined license holder notifies the NRC that the combined license holder plans to commence or resume construction, then the combined license applicant or holder shall submit to NRC an update to its FSAR within 90 days of the request or notification, as applicable, and annually thereafter.

(d) The updated FSAR must be retained by the licensee until the Commission terminates its license.

(f) Each holder of a manufacturing license under Framework A of this part must submit an update of the FSAR reflecting any modification to the design that is directed or approved by the Commission under §§ 53.1288 or 53.1530, and any new analyses of the design requested by the Commission under § 53.1580.

§ 53.1550 Evaluating changes to facility as described in Final Safety Analysis Reports.

(a) A licensee may make changes in the facility as described in the UFSAR and make changes in the procedures as described in the UFSAR without obtaining a license amendment pursuant to § 53.1510 only if:

(1) A change to the technical specifications incorporated in the license is not required and

(2) The change meets all of the following criteria:

(i) Does not result in an increase to the frequency or consequences of an event sequence such that an event sequence not previously identified as risk significant becomes risk significant by the analyses performed in accordance with § 53.450(e).

(ii) Does not result in an increase to the frequency or consequences of an event sequence such that an event sequence identified as risk significant in accordance with § 53.450(e) exceeds the LBE evaluation criteria required to be established in accordance with § 53.450(e).

(iii) Does not result in an increase to the frequency or consequences of one or more event sequences such that the margin between the calculated cumulative risks

posed by the commercial nuclear plant and the safety criteria of § 53.220 decreases by 10 percent or more.

(iv) Does not involve a departure from a method of evaluation described in the UFSAR used in assessing LBEs in accordance with § 53.450 unless the results of the analysis under § 53.450 are conservative or essentially the same, the revised method of evaluation has been previously approved by the NRC for the intended application, or the revised method of evaluation can be used in accordance with an NRC endorsed consensus code or standard.

(v) Does not result in the escalation in the safety classification of an SSC from non-safety-related (NSR) to NSRSS or from NSRSS to SR

(vi) Does not result in more than a minimal decrease in defense-in-depth

(vii) For commercial nuclear plants licensed under Framework A of this part for which alternative evaluation criteria are adopted in accordance with § 53.470, does not result in a change to the frequency or consequences of event sequences such that the calculated margins between the results for event sequences evaluated in accordance with § 53.450(e) and the alternative evaluation criteria decreases by 25 percent or more.

(viii) Does not result in the identification of a new design-basis accident in accordance with § 53.450(f).

(ix) Does not result in a decrease by 10 percent or more in the margin between the consequence of any design-basis accident and the safety criteria in § 53.210.

(ixi) Does not prevent meeting the design requirements in § 53.440(j) to limit the release of radionuclides from reactor systems, waste stores, or other significant inventories of radioactive materials assuming the impact of a large, commercial aircraft.

(3) In implementing this paragraph, the UFSAR is considered to include FSAR changes since submittal of the last update of the FSAR pursuant to § 53.1545.

(4) The provisions in this section do not apply to changes to the facility or procedures when the applicable regulations establish more specific criteria for accomplishing such changes.

(b)(1) A licensee who references a design certification rule may make departures from the standard design, without prior Commission approval, unless the proposed departure involves a change to the design as described in the rule certifying the design, in which case the requirements of § 53.1525 are applicable.

(2) The licensee must maintain records of all departures from the certified design of the facility and these records must be maintained and available for audit until the date of termination of the license. The licensee must identify the location and nature of departures from licensing basis information within supporting documents for a certified design within the updates to the Safety Analysis Report required by § 53.1545.

(3) Licensees for which the NRC has docketed the certifications required under § 53.1070 are not required to retain records of departures from the design of the facility associated with structures, systems, and components that have been permanently removed from service using an NRC-approved change process.

(c)(1) The licensee must maintain records of changes in the facility and procedures made pursuant to paragraph (a) of this section. These records must include a written evaluation which provides the bases for the determination that the change does not require a license amendment pursuant to paragraph (a)(2) of this section.

(2) The licensee must submit, as specified in § 53.040, a report containing a brief description of any departures and changes, including a summary of the evaluation of each. A report must be submitted at intervals not to exceed 24 months. For combined licenses, the report must be submitted at intervals not to exceed 6 months during the

period from the date of application for a combined license to the date the Commission makes its findings under 10 CFR 53.1452(g).

(3) The records of changes in the facility must be maintained until the termination of an operating license or combined license issued under Framework A of this part, or the termination of a renewed license issued under § 53.1595, whichever is later. Records of changes in procedures must be maintained for a period of 5 years.

§ 53.1560 Updating program documents included in licensing basis information.

(a) Each holder under Framework A of this part of an operating license or combined license for which the Commission has made the finding under § 53.1452(g) must biennially or more frequently update the program documents submitted as part of an application to obtain or maintain the license to assure that the information included in the documents contains the latest information developed. The submittals shall include the effects on the content of the program documents of:

(1) changes made in the facility, procedures, licensee's organization, or site environs;

(2) safety analyses and evaluations performed by the applicant or licensee either in support of approved license amendments or in support of conclusions that changes did not require a license amendment in accordance with § 53.1550;

(3) analyses of new safety issues performed by or on behalf of the licensee at Commission request; and

(4) changes to the programs as a result of operating experience, corrective actions, or other reasons deemed appropriate to ensure the programs serve their underlying purpose to support the requirements in subpart B of this part or other NRC regulations.

(b)(1) The licensee shall submit revisions containing updated information to the Commission, as specified in § 53.040, identifying the location of revised or new information.

(2) The submittal shall include (i) a certification by a duly authorized officer of the licensee that either the information accurately presents changes made since the previous submittals, necessary to reflect information and analyses submitted to the Commission or prepared pursuant to Commission requirement, or that no such changes were made; and (ii) an identification of changes made under the provisions of § 53.1550 but not previously submitted to the Commission.

(c) The updated program documents shall be retained by the licensee until the Commission terminates their license.

§ 53.1565 Evaluating changes to programs included in licensing basis information.

(a) A licensee may make changes to the facility, procedures, or organizations or address changes to site environs as described in the program documents included in licensing basis information without obtaining prior NRC approval only if:

(1) A change to the technical specifications incorporated in the license is not required,

(2) An exemption from an NRC regulation is not required,

(3) The change conforms to program-specific requirements included in regulations in Framework A of this part, technical specifications, or the NRC-approved program document included and reviewed as part of a license application under subpart H or an amendment under this subpart.

(b) In implementing this paragraph, the program documents (as updated) include changes since submittal of the last updates of the program documents pursuant to § 53.1560.

(c) The provisions in this section do not apply to changes to the program documents when the applicable regulations establish more specific criteria for accomplishing such changes.

(d) To make changes to the facility, procedures, or organizations or to address changes to site environs as described in the program documents included in licensing basis information for individual programs, the following requirements must be satisfied:

(1) *Quality assurance program—operation*. (i) Each holder under Framework A of this part of an operating license or combined license, after the Commission makes the finding under § 53.1452(g), may make a change to a previously accepted quality assurance program description included or referenced in the Safety Analysis Report without prior NRC approval, provided the change does not reduce the commitments in the program description as accepted by the NRC. Changes to the quality assurance program description that do not reduce the commitments must be submitted to the NRC in accordance with the requirements of § 53.1545. In addition to quality assurance program changes involving administrative improvements and clarifications, spelling corrections, punctuation, or editorial items, the following changes are not considered to be reductions in commitment:

(A) The use of a QA standard approved by the NRC which is more recent than the QA standard in the licensee's QA program at the time of the change;

(B) The use of a quality assurance alternative or exception approved by an NRC safety evaluation, provided that the bases of the NRC approval are applicable to the licensee's facility;

(C) The use of generic organizational position titles that clearly denote the position function, supplemented as necessary by descriptive text, rather than specific titles;

(D) The use of generic organizational charts to indicate functional relationships, authorities, and responsibilities, or, alternately, the use of descriptive text;

(E) The elimination of quality assurance program information that duplicates language in quality assurance regulatory guides and quality assurance standards to which the licensee is committed; and

(F)(i) Organizational revisions that ensure that persons and organizations performing quality assurance functions continue to have the requisite authority and organizational freedom, including sufficient independence from cost and schedule when opposed to safety considerations.

(ii) Changes to the quality assurance program description that do reduce the commitments must be submitted to the NRC and receive NRC approval prior to implementation, as follows:

(A) Changes made to the quality assurance program description as presented in the Safety Analysis Report or in a topical report must be submitted as specified in § 53.040.

(B) The submittal of a change to the Safety Analysis Report quality assurance program description must include all pages affected by that change and must be accompanied by a forwarding letter identifying the change, the reason for the change, and the basis for concluding that the revised program incorporating the change continues to satisfy the criteria of subpart K of this part and the Safety Analysis Report quality assurance program description commitments previously accepted by the NRC

(the letter need not provide the basis for changes that correct spelling, punctuation, or editorial items).

(C) A copy of the forwarding letter identifying the change must be maintained as a facility record for three years.

(D) Changes to the quality assurance program description included or referenced in the Safety Analysis Report shall be regarded as accepted by the Commission upon receipt of a letter to this effect from the appropriate reviewing office of the Commission or 60 days after submittal to the Commission, whichever occurs first.

(2) *Quality assurance program—siting, construction, and manufacturing*. Each holder of an LWA, an early site permit, a construction permit, a manufacturing license, or a combined license, before the Commission makes the finding under § 53.1452(g) of this chapter, under Framework A of this part may make a change to a previously accepted quality assurance program description included or referenced in the Safety Analysis Report without prior NRC approval, provided the change does not reduce the commitments in the program description previously accepted by the NRC. Changes to the quality assurance program description that do not reduce the commitments must be submitted to NRC within 90 days. Changes to the quality assurance program description that reduce the commitments must be submitted to NRC and receive NRC approval before implementation, as follows:

(i) Changes to the Safety Analysis Report must be submitted for review as specified in § 53.040. Changes made to NRC-accepted quality assurance topical report descriptions must be submitted as specified in § 53.040.

(ii) The submittal of a change to the Safety Analysis Report quality assurance program description must include all pages affected by that change and must be accompanied by a forwarding letter identifying the change, the reason for the change,

and the basis for concluding that the revised program incorporating the change continues to satisfy the criteria of subpart K of this part and the Safety Analysis Report quality assurance program description commitments previously accepted by the NRC (the letter need not provide the basis for changes that correct spelling, punctuation, or editorial items).

(iii) A copy of the forwarding letter identifying the changes must be maintained as a facility record for three years.

(iv) Changes to the quality assurance program description included or referenced in the Safety Analysis Report shall be regarded as accepted by the Commission upon receipt of a letter to this effect from the appropriate reviewing office of the Commission or 60 days after submittal to the Commission, whichever occurs first.

(3) *Emergency preparedness program*.

(i)(A) The licensee shall provide for the development, revision, implementation, and maintenance of its emergency preparedness program. The licensee shall ensure that all program elements are reviewed by persons who have no direct responsibility for the implementation of the emergency preparedness program either:

(*1*) At intervals not to exceed 12 months or,

(*2*) As necessary, based on an assessment by the licensee against performance indicators, and as soon as reasonably practicable after a change occurs in personnel, procedures, equipment, or facilities that potentially could adversely affect emergency preparedness, but no longer than 12 months after the change. In any case, all elements of the emergency preparedness program must be reviewed at least once every 24 months.

(B).The review must include an evaluation for adequacy of interfaces with State and local governments and of licensee drills, exercises, capabilities, and procedures.

The results of the review, along with recommendations for improvements, must be documented, reported to the licensee's corporate and plant management, and retained for a period of 5 years. The part of the review involving the evaluation for adequacy of interface with State and local governments must be available to the appropriate State and local governments.

(ii) The licensee may make changes to its emergency plan without NRC approval only if the licensee performs and retains an analysis demonstrating that the changes do not reduce the effectiveness of the plan and the plan, as changed, continues to satisfy the requirements in § 53.855. A change reduces the effectiveness of the plan if it results in reducing the licensee's capability to perform an emergency planning function required by § 53.855 in the event of a radiological emergency.

(iii) The licensee shall retain a record of each change to the emergency plan made without prior NRC approval for a period of three years from the date of the change and shall submit, as specified in § 53.040, a report of each such change, including a summary of its analysis, within 30 days after the change is put in effect.

(iv) The changes to a licensee's emergency plan that reduce the effectiveness of the plan may not be implemented without prior approval by the NRC. A licensee desiring to make such a change must submit an application for an amendment to its license. In addition to the filing requirements of §§ 53.1510, 53.1515, and 53.1520, the request must include all emergency plan pages affected by that change and must be accompanied by a forwarding letter identifying the change, the reason for the change, and the basis for concluding that the licensee's emergency plan, as revised, will continue to satisfy the requirements of § 53.855.

(v) The nuclear power reactor licensee must retain the emergency plan and each change for which NRC approval was obtained, pursuant to paragraph (d)(3)(iv) of this

section, as a record until the Commission terminates the license for the nuclear power reactor.

(4) *Security programs.*

(i) The licensee shall prepare and maintain safeguards contingency plan procedures in accordance with appendix C of part 73 of this chapter for affecting the actions and decisions contained in the Responsibility Matrix of the safeguards contingency plan. The licensee may not make a change which would decrease the effectiveness of a physical security plan, or guard training and qualification plan, or cybersecurity plan submitted under subpart H or part 73 of this chapter, or of the first four categories of information (Background, Generic Planning Base, Licensee Planning Base, Responsibility Matrix) contained in a licensee safeguards contingency plan submitted under subpart H or part 73 of this chapter, as applicable, without prior approval of the Commission. A licensee desiring to make such a change shall submit an application for amendment to the licensee's license under §§ 53.1510, 53.1515, and 53.1520.

(ii) The licensee may make changes to the plans referenced in paragraph (4)(i) of this section, without prior Commission approval if the changes do not decrease the safeguards effectiveness of the plan. The licensee shall maintain records of changes to the plans made without prior Commission approval for a period of 3 years from the date of the change, and shall submit, as specified in § 53.040, a report containing a description of each change within 2 months after the change is made. Prior to the safeguards contingency plan being put into effect, the licensee shall have:

(A) All safeguards capabilities specified in the safeguards contingency plan available and functional;

(B) Detailed procedures developed according to appendix C to part 73 of this chapter available at the licensee's site; and

(C) All appropriate personnel trained to respond to safeguards incidents as outlined in the plan and specified in the detailed procedures.

(iii) The licensee shall provide for the development, revision, implementation, and maintenance of its safeguards contingency plan. The licensee shall ensure that all program elements are reviewed by individuals independent of both security program management and personnel who have direct responsibility for implementation of the security program either:

(A) At intervals not to exceed 12 months; or

(B) As necessary, based on an assessment by the licensee against performance indicators, and as soon as reasonably practicable after a change occurs in personnel, procedures, equipment, or facilities that potentially could adversely affect security, but no longer than 12 months after the change. In any case, all elements of the safeguards contingency plan must be reviewed at least once every 24 months.

(iv) The review must include a review and audit of safeguards contingency procedures and practices, an audit of the security system testing and maintenance program, and a test of the safeguards systems along with commitments established for response by local law enforcement authorities. The results of the review and audit, along with recommendations for improvements, must be documented, reported to the licensee’s corporate and plant management, and kept available at the plant for inspection for a period of 3 years.

§ 53.1570 Transfer of licenses.

(a) No commercial nuclear plant license issued under Framework A of this part, or any right thereunder, shall be transferred, assigned, or in any manner disposed of, either voluntarily or involuntarily, directly or indirectly, through transfer of control of the license to any person, unless the Commission gives its consent in writing.

(b)(1) An application for transfer of a license shall include:

(i) As much of the information described in §§ 53.1109, 53.1306, 53.1366, and 53.1413 with respect to the identity and technical and financial qualifications of the proposed transferee as would be required by those sections if the application were for an initial license. The Commission may require additional information such as data respecting proposed safeguards against hazards from radioactive materials and the applicant's qualifications to protect against such hazards.

(ii) A statement of the purposes for which the transfer of the license is requested, the nature of the transaction necessitating or making desirable the transfer of the license, and an agreement to limit access to Restricted Data or Classified National Security Information pursuant to § 53.1115. The Commission may require any person who submits an application for license pursuant to the provisions of this section to file a written consent from the existing licensee or a certified copy of an order or judgment of a court of competent jurisdiction attesting to the person's right (subject to the licensing requirements of the Act and these regulations) to possession of the facility or site involved.

(c) After appropriate notice to interested persons, including the existing licensee, and observance of such procedures as may be required by the Act or regulations or orders of the Commission, the Commission will approve an application for the transfer of a license, if the Commission determines:

(1) That the proposed transferee is qualified to be the holder of the license; and

(2) That transfer of the license is otherwise consistent with applicable provisions of law, regulations, and orders issued by the Commission pursuant thereto.

§ 53.1575 Termination of license.

(a) When the holder of an operating license or combined license under Framework A of this part has determined to permanently cease operations the licensee shall, within 30 days, submit a written certification to the NRC, consistent with the requirements of§ 53.1070.

(b) Once fuel has been permanently removed from the reactor system, the licensee shall submit a written certification to the NRC that meets the requirements of § 53.1070.

(c)(1) Upon docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor system, or when a final legally effective order to permanently cease operations has come into effect, the license no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor system.

(2) Activities associated with decommissioning will be carried out in accordance with the requirements and procedures in subpart G.

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

(i) The remaining dismantlement has been performed in accordance with the approved license termination plan required in subpart G, and

(ii) 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 10 CFR part 20, subpart E.

(d) A holder of a construction permit or combined license under Framework A of this part may request the termination of the license as well as licenses issued by the NRC under parts 30, 40, 70 of this chapter prior to plant operations. Such requests may support an immediate NRC approval of the site for unrestricted use.

§ 53.1580 Information requests.

Any licensee under Framework A of this part shall at any time before expiration of the license, upon request of the Commission, submit, as specified in § 53.040 written statements, signed under oath or affirmation, to enable the Commission to determine whether or not the license should be modified, suspended, or revoked. Except for information sought to verify licensee compliance with the current licensing basis for that facility, the NRC must prepare the reason or reasons for each information request prior to 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 such justification provided for an evaluation performed by the NRC staff must be approved by the Executive Director for Operations or his or her designee prior to issuance of the request.

§ 53.1585 Revocation, suspension, modification of licenses and approvals for cause.

A license or standard design approval issued under Framework A of this part may be revoked, suspended, or modified, in whole or in part, for any material false statement in the application or in the supplemental or other statement of fact required of the applicant; or because of conditions revealed by the application or statement of fact of any report, record, inspection, or other means which would warrant the Commission to

refuse to grant a license or approval on an original application; or for failure to manufacture a reactor, or construct or operate a facility in accordance with the terms of the license, provided, however, that failure to make timely completion of the proposed construction or alteration of a facility under a construction permit under Framework A of this part shall be governed by the provisions of § 53.1342(b); or for violation of, or failure to observe, any of the terms and provisions of the act, regulations, license, approval, or order of the Commission.

§ 53.1590 Backfitting.

(a)(1) Backfitting means the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission's regulations or the imposition of a regulatory staff position interpreting the Commission's regulations that is either new or different from a previously applicable staff position after the date of the commercial nuclear plant license issued under Framework A of this part.

(2) Except as provided in paragraph (a)(4) of this section, the Commission shall require a systematic and documented analysis pursuant to paragraph (b) of this section for backfits which it seeks to impose.

(3) Except as provided in paragraph (a)(4) of this section, the Commission shall require the backfitting of a facility only when it determines, based on the analysis described in paragraph (b) of this section, that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection.

(4) The provisions of paragraphs (a)(2) and (a)(3) of this section are inapplicable and, therefore, backfit analysis is not required and the standards in paragraph (a)(3) of this section do not apply where the Commission or staff, as appropriate, finds and declares, with appropriated documented evaluation for its finding, either:

(i) That a modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee; or

(ii) That regulatory action is necessary to ensure that the facility provides adequate protection to the health and safety of the public and is in accord with the common defense and security; or

(iii) That the regulatory action involves defining or redefining what level of protection to the public health and safety or common defense and security should be regarded as adequate.

(5) The Commission shall always require the backfitting of a facility if it determines that such regulatory action is necessary to ensure that the facility provides adequate protection to the health and safety of the public and is in accord with the common defense and security.

(6) The documented evaluation required by paragraph (a)(4) of this section shall include a statement of the objectives of and reasons for the modification and the basis for invoking the exception. If immediately effective regulatory action is required, then the documented evaluation may follow rather than precede the regulatory action.

(7) If there are two or more ways to achieve compliance with a license or the rules or orders of the Commission, or with written licensee commitments, or there are two or more ways to reach a level of protection which is adequate, then ordinarily the applicant or licensee is free to choose the way which best suits its purposes. However,

should it be necessary or appropriate for the Commission to prescribe a specific way to comply with its requirements or to achieve adequate protection, then cost may be a factor in selecting the way, provided that the objective of compliance or adequate protection is met.

(b) In reaching the determination required by paragraph (a)(3) of this section, the Commission will consider how the backfit should be scheduled in light of other ongoing regulatory activities at the facility and, in addition, will consider information available concerning any of the following factors as may be appropriate and any other information relevant and material to the proposed backfit:

(1) Statement of the specific objectives that the proposed backfit is designed to achieve;

(2) General description of the activity that would be required by the licensee or applicant in order to complete the backfit;

(3) Potential change in the risk to the public from the accidental off-site release of radioactive material;

(4) Potential impact on radiological exposure of facility employees;

(5) Installation and continuing costs associated with the backfit, including the cost of facility downtime or the cost of construction delay

(6) The potential safety impact of changes in plant or operational complexity, including the relationship to proposed and existing regulatory requirements;

(7) The estimated resource burden on the NRC associated with the proposed backfit and the availability of such resources;

(8) The potential impact of differences in facility type, design or age on the relevancy and practicality of the proposed backfit;

(9) Whether the proposed backfit is interim or final and, if interim, the justification for imposing the proposed backfit on an interim basis.

(c) No licensing action will be withheld during the pendency of backfit analyses required by the Commission's rules.

(d) The Executive Director for Operations shall be responsible for implementation of this section, and all analyses required by this section shall be approved by the Executive Director for Operations or his designee.

§ 53.1595 Renewal.

Licenses may be renewed by the Commission upon expiration of the period of the license.

Subpart J—Reporting and Other Administrative Requirements

§ 53.1600 General information.

Each applicant and licensee under Framework A of this part must ensure that NRC inspectors have unfettered access to sites and facilities licensed or proposed to be licensed in § 53.1610, must maintain records and make reports to the NRC in accordance with requirements in §§ 53.1620 through 53.1650, must satisfy financial qualification and reporting requirements in §§ 53.1670 through 53.1700, and must obtain and maintain required financial protections in case of an accident in §§ 53.1720 and 53.1730.

§ 53.1610 Unfettered access for inspections.

(a) Each applicant for or holder of a manufacturing license, operating license, combined license, construction permit or an early site permit, must permit inspection by

duly authorized representatives of the Commission, of its records, premises, activities, and of licensed materials in possession or use, related to the license or construction permit or early site permit as may be necessary to effectuate the purposes of the Act, as amended, and the ERA.

(b)(1) Each holder of a manufacturing license, operating license, combined license, or construction permit must, upon request by the Director, Office of Nuclear Reactor Regulation, provide rent-free office space for the exclusive use of the Commission inspection personnel. Heat, air conditioning, light, electrical outlets, and janitorial services must be furnished by each licensee and each holder of a construction permit. The office must be convenient to and have full access to the facility and shall provide the inspectors both visual and acoustic privacy.

(2) For a site or facility with an assigned resident inspector, the space provided must be adequate to accommodate a full-time inspector, a part-time secretary, and transient NRC personnel and must be generally commensurate with other office facilities at the site. A space of 250 square feet either within the site's office complex or in an office trailer or other onsite space is suggested as a guide. For sites or facilities assigned multiple resident inspectors, additional space may be requested. The office space that is provided must be subject to the approval of the Director, Office of Nuclear Reactor Regulation. All furniture, supplies, and communication equipment will be furnished by the Commission.

(3) For a site or facility without an assigned resident inspector, temporary space to accommodate periodic or special inspections must be provided. The office space must be generally commensurate with other office accommodations at the site.

(4) The licensee or permit holder must afford any NRC resident inspector assigned to that site, or other NRC inspectors identified by the Regional Administrator as

likely to inspect the facility, immediate unfettered access, equivalent to access provided regular plant employees, following proper identification and compliance with applicable access control measures for security, radiological protection, and personal safety.
(5) The licensee or permit holder must ensure that the arrival and presence of an NRC inspector, who has been properly authorized facility access as described in paragraph (b)(4) of this section, is not announced or otherwise communicated by its employees or contractors to other persons at the facility unless specifically requested by the NRC inspector.

§ 53.1620 Maintenance of records, making of reports.

(a) Each holder of a manufacturing license, operating license, combined license, construction permit or early site permit, must maintain all records and make all reports, in connection with the activity, as may be required by the conditions of the license or permit or by the regulations, and orders of the Commission in effectuating the purposes of the Act and the ERA. Reports must be submitted in accordance with § 53.040.

(b) Reserved

(c) Records that are required by the regulations in Framework A of this part, by license condition, or by technical specifications must be retained for the period specified by the appropriate regulation, license condition, or technical specification. If a retention period is not otherwise specified, these records must be retained until the Commission terminates the facility license or, in the case of an early site permit, until the permit expires.

(d)(1) Records which must be retained under Framework A of this part may be the original or a reproduced copy or a microform if the reproduced copy or microform is duly authenticated by authorized personnel and the microform is capable of producing a

clear and legible copy after storage for the period specified by Commission regulations. The record may also be stored in electronic media with the capability of producing legible, accurate, and complete records during the required retention period. Records such as letters, drawings, and specifications, must include all pertinent information such as stamps, initials, and signatures. The licensee must maintain adequate safeguards against tampering with, and loss of records.

(2) If there is a conflict between the Commission's regulations in Framework A of this part, license condition, or technical specification, or other written Commission approval or authorization pertaining to the retention period for the same type of record, the retention period specified in the regulations in Framework A of this part for such records shall apply unless the Commission, pursuant to § 53.080 of this part, has granted a specific exemption from the record retention requirements in the regulations in Framework A of this part.

(e) Each licensee shall notify the Commission as specified in § 53.040 of this chapter, of successfully completing power ascension testing or startup testing as applicable within 30 calendar days of completing the testing.

§ 53.1630 Immediate notification requirements for operating commercial nuclear plants.

(a) *General requirements.**: (1) Each holder of an operating license under Framework A of this part or a combined license under Framework A of this part after the Commission makes the finding under § 53.1452(g), must notify the NRC Operations Center via the Emergency Notification System of:

(i) The declaration of any of the Emergency Classes specified in the licensee's approved Emergency Plan; or

(ii) Those non-emergency events specified in paragraph (b) of this section that occurred within three years of the date of discovery.

(2) If the Emergency Notification System is inoperative, the licensee must make the required notifications via commercial telephone service, other dedicated telephone system, or any other method which will ensure that a report is made as soon as practical to the NRC Headquarters Operations Center at the numbers specified in appendix A to part 73 of this chapter.

(3) The licensee must notify the NRC immediately after notification of the appropriate State or local agencies and not later than one hour after the time the licensee declares one of the Emergency Classes.

(4) The licensee must activate the data links with the NRC as specified in their emergency plans after declaring an Emergency Class for events of actual or potential substantial degradation of plant safety or security, probable risk to site personnel life or, site equipment damage caused by hostile action. The data links may also be activated by the licensee during emergency drills or exercises if the licensee's computer system has the capability to transmit the exercise data.

(5) When making a report under paragraph (a)(1) of this section, the licensee must identify:

(i) The Emergency Class declared; or

(ii) Paragraph (b)(1), "One-hour reports," paragraph (b)(2), "Four-hour reports," or paragraph (b)(3), "Eight-hour reports," as the paragraph of this section requiring notification of the non-emergency event.

(b) *Non-emergency events* – (1) *One-hour reports*. If not reported as a declaration of an Emergency Class under paragraph (a) of this section, the licensee must notify the NRC as soon as practical and in all cases within one hour of the

occurrence of any deviation from the plant's Technical Specifications authorized pursuant to § 53.740(h) of this part.

(2) *Four-hour reports*. If not reported under paragraphs (a) or (b)(1) of this section, the licensee must notify the NRC as soon as practical, and in all cases, within four hours of the occurrence of any of the following:

(i) The initiation of any commercial nuclear plant shutdown required by the plant's Technical Specifications.

(ii) Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

(iii) Any event or condition that results in an unplanned actuation of a safety-related standby cooling system or the unplanned sole reliance on a safety-related standby cooling system for those systems that are in constant operation.

(iv) Any event or condition that results in an unplanned movement of, change of state in, or chemical interaction involving a significant amount of radioactive material within the commercial nuclear plant.

(v) Any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactively contaminated materials.

(3) *Eight-hour reports*. If not reported under paragraphs (a), (b)(1) or (b)(2) of this section, the licensee must notify the NRC as soon as practical and in all cases within eight hours of the occurrence of any of the following:

(i) Any event or condition that results in:

(A) The condition of the commercial nuclear plant, including its principal safety barriers, being seriously degraded; or

(B) The commercial nuclear plant being in a condition not analyzed under § 53.450 that significantly degrades plant safety.

(ii) Any event or condition that results in valid actuation of a safety-related system, except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

(iii) Any event or condition that at the time of discovery could have prevented the fulfilment of the safety functions defined in § 53.230. Events covered may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to this paragraph if other equipment was operable and available to perform the required safety function.

(iv) Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment.

(v) Any event that results in a major loss of emergency assessment capability, offsite response capability, or offsite communications capability (e.g., significant portion of control room indication, Emergency Notification System, or offsite notification system).

(c) *Followup Notification*: With respect to the notifications made under paragraphs (a) and (b) of this section, in addition to making the required initial notification, each licensee, must during the course of the event:

(1) Immediately Report: (i) any further degradation in the level of safety of the plant or other worsening plant conditions, including those that require the declaration of any of the Emergency Classes, if such a declaration has not been previously made, or

(ii) any change from one Emergency Class to another, or

(iii) a termination of the Emergency Class.

(2) Immediately Report: (i) the results of ensuing evaluations or assessments of plant conditions,

(ii) the effectiveness of response or protective measures taken, and

(iii) important information related to plant behavior that is not understood.

(3) Maintain an open, continuous communication channel with the NRC Operation Center upon request by the NRC.

*Other requirements for immediate notification of the NRC by licensed operating commercial nuclear plants are contained elsewhere in this chapter, in particular §§ 20.1906, 20.2202, 72.216, 73.71, and 73.77.

§ 53.1640 Licensee event report system.

(a) *Reportable events*. (1) Each commercial nuclear plant licensee holding an operating license under Framework A of this part or a combined license under Framework A of this part after the Commission makes the finding under § 53.1452(g), must submit a Licensee Event Report (LER) for any event of the type described in this paragraph within 60 days after discovery of the event. In the case of an invalid actuation reported under § 53.1640(a)(2), other than automatic reactor shutdown when the reactor is critical, the licensee may, at its option, provide a telephone notification to the NRC Operations Center within 60 days after discovery of the event instead of submitting a written LER. Unless otherwise specified in this section, the licensee must report an event if it occurred within 3 years of the date of discovery regardless of the plant mode or power level, and regardless of the significance of the structure, system, or component that initiated the event.

(2) The licensee must report:

(i)(A) The completion of any commercial nuclear plant shutdown required by the plant's Technical Specifications.

(B) Any operation or condition which was prohibited by the plant's Technical Specifications except when:

(1) The Technical Specification is administrative in nature;

(2) The event consisted solely of a case of a late surveillance test where the oversight was corrected, the test was performed, and the equipment was found to be capable of performing its specified safety functions; or

(3) The Technical Specification was revised prior to discovery of the event such that the operation or condition was no longer prohibited at the time of the event.

(C) Any deviation from the plant's Technical Specifications authorized pursuant to § 53.740(h).

(ii) Any event or condition that resulted in:

(A) The condition of the commercial nuclear plant, including its principal safety barriers, being seriously degraded; or

(B) The commercial nuclear plant being in a condition not analyzed under § 53.450 that significantly degrades plant safety.

(iii) Any natural phenomena or other external condition that posed an actual threat to the safety of the commercial nuclear plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the commercial nuclear plant.

(iv) Any event or condition that resulted in inadvertent operation of any SSC classified as SR for an identified safety function under § 53.460 or the unplanned sole reliance on a SR system for those systems that are in constant operation, except when:

(A) The actuation resulted from and was part of a pre-planned sequence during testing; or

(B) The actuation was invalid and;

(*1*) Occurred while the system was properly removed from service; or

(*2*) Occurred after the safety function had been already completed.

(v) Any event or condition that could have prevented the fulfillment of the safety functions listed in § 53.230.

(vi) Events covered in paragraph (a)(2)(v) of this section may include one or more procedural errors, equipment failures, and/or discovery of design, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to paragraph (a)(2)(v) of this section if any other equipment was operable and available to perform the required safety function.

(vii)(A) Any event or condition that as a result of a single cause could have prevented the fulfillment of any of the safety functions listed in § 53.230.

(B) Events covered in paragraph (a)(2)(viii)(A) of this section may include cases of procedural error, equipment failure, and/or discovery of a design, analysis, fabrication, construction, and/or procedural inadequacy. However, licensees are not required to report an event pursuant to paragraph (a)(2)(viii)(A) of this section if the event results from:

(*1*) A shared dependency among trains or channels that is a natural or expected consequence of the approved plant design; or

(*2*) Normal and expected wear or degradation.

(viii)(A) Any airborne radioactive release that, when averaged over a time period of 1-hour, resulted in airborne radionuclide concentrations in an unrestricted area that

exceeds 20 times the applicable concentration limits specified in appendix B to part 20, table 2, column 1.

(B) Any liquid effluent release that, when averaged over a time period of 1-hour, exceeds 20 times the applicable concentrations specified in appendix B to part 20, table 2, column 2, at the point of entry into the receiving waters (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases.

(ix) Any event that posed an actual threat to the safety of the commercial nuclear plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the plant, including fires, toxic gas releases, or radioactive releases.

(b) *Contents*. The LER shall contain:

(1) A brief abstract describing the major occurrences during the event, including all component or system failures that contributed to the event and significant corrective action taken or planned to prevent recurrence.

(2)(i) A clear, specific narrative description of what occurred so that knowledgeable readers conversant with the design of commercial nuclear plants, but not familiar with the details of a particular plant, can understand the complete event.

(ii) The narrative description must include the following specific information as appropriate for the particular event:

(A) Plant operating conditions before the event.

(B) Status of structures, components, or systems that were inoperable at the start of the event and that contributed to the event.

(C) Dates and approximate time of the occurrences.

(D) The cause of each component or system failure or personnel error, if known.

(E) The failure mode, mechanism, and effect of each failed component, if known.

[(F) Reserved]

(G) For failures of components with multiple functions, include a list of systems or secondary functions that were also affected.

(H) For failure that rendered a component or system classified as SR or NSRSS inoperable, an estimate of the elapsed time from the discovery of the failure until the component or system was returned to service.

(I) The method of discovery of each component or system failure or procedural error.

(J) For each human performance related root cause, the licensee must discuss the cause(s) and circumstances.

(K) Automatically and manually initiated safety system responses.

(L) The manufacturer and model number (or other identification) of each component that failed during the event.

(3) An assessment of the safety consequences and implications of the event. This assessment must include:

(i) The availability of systems or components that could have performed the same function as the components and systems that failed during the event, and

(ii) For events that occurred when the reactor was shut down, the availability of systems or components that are needed to shut down the reactor and maintain safe shutdown conditions, remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident.

(4) A description of any corrective actions planned as a result of the event, including those to reduce the probability of similar events occurring in the future.

(5) Reference to any previous similar events at the same plant that are known to the licensee.

(6) The name and contact information of a person within the licensee's organization who is knowledgeable about the event and can provide additional information concerning the event and the plant's characteristics.

(c) *Supplemental Information*: The Commission may require the licensee to submit specific additional information beyond that required by paragraph (b) of this section if the Commission finds that supplemental material is necessary for complete understanding of an unusually complex or significant event. These requests for supplemental information will be made in writing and the licensee must submit, as specified in § 53.040, the requested information as a supplement to the initial LER.

(d) *Submission of Reports*: LERs must be prepared on Form NRC 366 and submitted to the NRC, as specified in § 53.040.

(e) *Report Legibility*: The reports and copies that licensees are required to submit to the Commission under the provisions of this section must be of sufficient quality to permit legible reproduction and micrographic processing.

(f) [Reserved]

(g) *Reportable Occurrences*: The requirements contained in this section replace all existing requirements for licensees to report "Reportable Occurrenc

§ 53.1645 Effluent reports.

Each holder of an operating license, and each holder of a combined license after the Commission has made the finding under § 53.1452(g), shall submit a report to the Commission annually that specifies the quantity of each of the principal radionuclides released to unrestricted areas in liquid and in gaseous effluents during the previous 12 months, including any other information as may be required by the Commission to estimate maximum potential annual radiation doses to the public resulting from effluent

releases. The report must be submitted as specified in § 53.040, and the time between submission of the reports must be no longer than 12 months. If the total effective dose equivalent to the maximally exposed individual members of the public in unrestricted areas during the reporting period are significantly above established design objectives or 10 mrem/year total effective dose equivalent, the report must cover this specifically. On the basis of these reports and any additional information the Commission may obtain from the licensee or others, the Commission may require the licensee to take action as the Commission deems appropriate.

§ 53.1650 Facility information and verification.

(a) In response to a written request by the Commission, each applicant for a construction permit or license and each recipient of a construction permit or a license must submit facility information, as described in § 75.10 of this chapter, on International Atomic Energy Agency (IAEA) Design Information Questionnaire forms and site information on DOC/NRC Form AP-A and associated forms;

(b) As required by the Additional Protocol, must submit location information described in § 75.11 of this chapter on DOC/NRC Form AP-1 and associated forms; and (c) Must permit verification thereof by the IAEA and take other action as necessary to implement the US/IAEA Safeguards Agreement, as described in Part 75 of this chapter.

§ 53.1660 Financial requirements.

Sections 53.1670 through 53.1700 set out the requirements and procedures related to financial qualifications and related reporting requirements.

§ 53.1670 Financial qualifications.

Except for an electric utility applicant for a license to operate a commercial nuclear plant, an applicant for a construction permit, operating license, or combined license under this part must possess or have reasonable assurance of obtaining the funds necessary for the activities for which the permit or license is sought.

§ 53.1680 Annual financial reports.

With respect to any commercial nuclear plant of a type described in § 53.020, each licensee and each holder of a construction permit must submit its annual financial report, including the certified financial statements, to the Commission, as specified in § 53.040, upon issuance of the report. However, licensees and holders of a construction permit who submit a Form 10-Q with the Securities and Exchange Commission or a Form 1 with FERC, need not submit the annual financial report or the certified financial statement under this paragraph.

§ 53.1690 Licensee's change of status; financial qualifications.

(a) An electric utility licensee holding an operating license or combined license (including a renewed license) for a commercial nuclear plant, no later than seventy-five (75) days prior to ceasing to be an electric utility in any manner not involving a license transfer under §§ 53.1399 or 53.1456 must provide the NRC with the financial qualifications information that would be required for obtaining an initial operating license or combined license under Framework A of this part. The financial qualifications information must address the first full five years of operation after the date the licensee ceases to be an electric utility.

(b)(1) Any holder of a license issued under this part shall notify the appropriate NRC Regional Administrator, in writing, immediately following the filing of a voluntary or

involuntary petition for bankruptcy under any chapter of title 11 (Bankruptcy) of the United States Code by or against:

(i) The licensee;

(ii) An entity (as 11 U.S.C. 101(14) defines that term) controlling the licensee or listing the license or licensee as property of the estate; or

(iii) An affiliate (as 11 U.S.C. 101(2) defines that term) of the licensee.

(2) This notification must indicate:

(i) The bankruptcy court in which the petition for bankruptcy was filed; and

(ii) The date of the filing of the petition.

§ 53.1700 Creditor regulations.

(a) Pursuant to section 184 of the AEA, the Commission consents, without individual application, to the creation of any mortgage, pledge, or other lien upon any facility not owned by the United States which is the subject of a license or upon any leasehold or other interest in such facility; *provided*:

(1) That the rights of any creditor so secured may be exercised only in compliance with and subject to the same requirements and restrictions as would apply to the licensee pursuant to the provisions of the license, the AEA, and regulations issued by the Commission pursuant to the AEA; and

(2) That no creditor so secured may take possession of the facility pursuant to the provisions of this section prior to either the issuance of a license from the Commission authorizing such possession or the transfer of the license.

(b) Any creditor so secured may apply for transfer of the license covering such facility by filing an application for transfer of the license pursuant to § 53.1570. The Commission will act upon such application pursuant to Subpart I of this part.

(c) Nothing contained in this regulation shall be deemed to affect the means of acquiring, or the priority of, any tax lien or other lien provided by law.

(d) As used in this section: (1) "*License*" includes any license under Framework A of this part, which may be issued by the Commission with regard to a facility;

(2) "*Creditor*" includes, without implied limitation, the trustee under any mortgage, pledge or lien on a facility made to secure any creditor, any trustee or receiver of the facility appointed by a court of competent jurisdiction in any action brought for the benefit of any creditor secured by such mortgage, pledge or lien, any purchaser of such facility at the sale thereof upon foreclosure of such mortgage, pledge, or lien or upon exercise of any power of sale contained therein, or any assignee of any such purchaser.

(3) "*Facility*" includes but is not limited to, a site which is the subject of an early site permit under Framework A of this part, and a reactor manufactured under a manufacturing license under Framework A of this part.

§ 53.1710 Financial protection.

Sections 53.1720 and 53.1730 set out the requirements and procedures related to licensees obtaining and maintaining insurance to cover stabilization and decontamination activities in the event of an accident and financial protection in accordance with Part 140, "Financial Protection Requirements and Indemnity Agreements," of this chapter.

§ 53.1720 Insurance required to stabilize and decontaminate plant following an accident.

Each commercial nuclear plant licensee under Framework A of this part shall take reasonable steps to obtain insurance available at reasonable costs and on

reasonable terms from private sources or to demonstrate to the satisfaction of the NRC that it possesses an equivalent amount of protection covering the licensee's obligation, in the event of an accident at the licensee's commercial nuclear reactor, to stabilize and decontaminate the plant and the plant site at which such an accident may occur, provided that:

(a) The insurance required by this section must have a minimum coverage limit for each commercial nuclear plant site of $1.06 billion, an amount based on plant-specific estimates of costs to stabilize and decontaminate a plant, or whatever amount of insurance is generally available from private sources, whichever is less. The required insurance must clearly state that, as and to the extent provided in paragraph (d)(1) of this section, any proceeds must be payable first for stabilization of the plant and next for decontamination of the plant and the plant site. If a licensee's coverage falls below the required minimum, the licensee must within 60 days take all reasonable steps to restore its coverage to the required minimum. The required insurance may, at the option of the licensee, be included within policies that also provide coverage for other risks, including, but not limited to, the risk of direct physical damage.

(b)(1) With respect to policies issued or annually renewed, the proceeds of such required insurance must be dedicated, as and to the extent provided in this paragraph, to reimbursement or payment on behalf of the insured of reasonable expenses incurred or estimated to be incurred by the licensee in taking action to fulfill the licensee's obligation, in the event of an accident at the licensee's plant, to ensure that the plant is in, or is returned to, and maintained in, a safe and stable condition and that radioactive contamination is removed or controlled such that personnel exposures are consistent with the occupational exposure limits in 10 CFR part 20. These actions must be consistent with any other obligation the licensee may have under this chapter and must

be subject to paragraph (d) of this section. As used in this section, an "accident" means an event that involves the release of radioactive material from its intended place of confinement within the commercial nuclear plant such that there is a present danger of release off site in amounts that would pose a threat to the public health and safety.

(2) The stabilization and decontamination requirements set forth in paragraph (d) of this section must apply uniformly to all insurance policies required under this section.

(c) The licensee shall report to the NRC on April 1 of each year the current levels of this insurance or financial security it maintains and the sources of this insurance or financial security.

(d)(1) In the event of an accident at the licensee's plant, whenever the estimated costs of stabilizing the licensed plant and of decontaminating the plant and the plant site exceed one tenth of the minimum insurance under paragraph (a) of this section, the proceeds of the insurance required by this section must be dedicated to and used, first, to ensure that the licensed plant is in, or is returned to, and can be maintained in, a safe and stable condition so as to prevent any significant risk to the public health and safety and, second, to decontaminate the plant and the plant site in accordance with the licensee's cleanup plan as approved by order of the Director of the Office of Nuclear Reactor Regulation. This priority on insurance proceeds must remain in effect for 60 days or, upon order of the Director, for such longer periods, in increments not to exceed 60 days except as provided for activities under the cleanup plan required in paragraphs (d)(3) and (d)(4) of this section, as the Director may find necessary to protect the public health and safety. Actions needed to bring the plant to and maintain the plant in a safe and stable condition may include one or more of the following, as appropriate:

(i) Shutdown of the reactor(s) and other processes at the plant;

(ii) Establishment and maintenance of long-term cooling with stable decay heat removal;

(iii) Maintenance of sub-criticality;

(iv) Control of radioactive releases; and

(v) Securing of structures, systems, or components to minimize radiation exposure to onsite personnel or to the offsite public or to facilitate later decontamination or both.

(2) The licensee must inform the Director of the Office of Nuclear Reactor Regulation in writing when the plant is and can be maintained in a safe and stable condition so as to prevent any significant risk to the public health and safety. Within 30 days after the licensee informs the Director that the plant is in this condition, or at such earlier time as the licensee may elect or the Director may for good cause direct, the licensee must prepare and submit a cleanup plan for the Director's approval. The cleanup plan must identify and contain an estimate of the cost of each cleanup operation that will be required to decontaminate the reactor sufficiently to permit the licensee either to resume operation of the reactor or to apply to the Commission under Subpart G for authority to decommission the reactor and to surrender the license voluntarily. Cleanup operations may include one or more of the following, as appropriate:

(i) Processing any contaminated materials generated by the accident and by decontamination operations to remove radioactive materials;

(ii) Decontamination of surfaces inside the plant buildings to levels consistent with the Commission's occupational exposure limits in 10 CFR part 20, and decontamination or disposal of equipment;

(iii) Decontamination or removal and disposal of internal parts, damaged fuel from the reactor coolant or fuel systems, or related process or waste systems; and

(iv) Cleanup of the reactor coolant or fuel systems or related process or waste systems.

(3) Following review of the licensee's cleanup plan, the Director will order the licensee to complete all operations that the Director finds are necessary to decontaminate the reactor sufficiently to permit the licensee either to resume operation of the reactor or to apply to the Commission under Subpart G for authority to decommission the reactor and to surrender the license voluntarily. The Director approves or disapproves, in whole or in part for stated reasons, the licensee's estimate of cleanup costs for such operations. Such order may not be effective for more than one year, at which time it may be renewed. Each subsequent renewal order, if imposed, may be effective for not more than 6 months.

(4) Of the balance of the proceeds of the required insurance not already expended to place the plant in a safe and stable condition pursuant to paragraph (b)(1) of this section, an amount sufficient to cover the expenses of completion of those decontamination operations that are the subject of the Director's order must be dedicated to such use, provided that, upon certification to the Director of the amounts expended previously and from time to time for stabilization and decontamination and upon further certification to the Director as to the sufficiency of the dedicated amount remaining, policies of insurance may provide for payment to the licensee or other loss payees of amounts not so dedicated, and the licensee may proceed to use in parallel (and not in preference thereto) any insurance proceeds not so dedicated for other purposes.

§ 53.1730 Financial protection requirements.

Commercial nuclear plant licensees must satisfy the applicable provisions of Part 140, “Financial Protection Requirements and Indemnity Agreements,” of this chapter.

Subpart K—Quality Assurance Criteria for Commercial Nuclear Plants

§ 53.1800 General provisions.

(a) Commercial nuclear plants and manufactured reactors include structures, systems and components that prevent or mitigate the consequences of LBEs, including design-basis accidents, as described in § 53.240, that could cause undue risk to the health and safety of the public. This subpart establishes quality assurance requirements for the design, manufacture, construction and operation of those structures, systems and components classified as safety related under Framework A of this part. The pertinent requirements of this subpart apply to all activities affecting the safety-related functions of those structures, systems and components; these activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying.

(b) As used in this subpart, “quality assurance” comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system which provide a means to control the quality of the material, structure, component, or system to predetermined requirements.

§ 53.1805 Organization.

The applicant[1] must establish and execute the quality assurance program. The applicant may delegate to others, such as contractors, agents, or consultants, the work of establishing and executing the quality assurance program, or any part thereof, but

shall retain responsibility for the quality assurance program. The authority and duties of persons and organizations performing activities affecting the safety-related functions of structures, systems, and components shall be clearly established and delineated in writing. These activities include both the performing functions of attaining quality objectives and the quality assurance functions. The quality assurance functions are those of (1) assuring that an appropriate quality assurance program is established and effectively executed; and (2) verifying, such as by checking, auditing, and inspecting, that activities affecting the safety-related functions have been correctly performed. The persons and organizations performing quality assurance functions shall have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions. The persons and organizations performing quality assurance functions shall report to a management level so that the required authority and organizational freedom, including sufficient independence from cost and schedule when opposed to safety considerations, are provided. Because of the many variables involved, such as the number of personnel, the type of activity being performed, and the location or locations where activities are performed, the organizational structure for executing the quality assurance program may take various forms, provided that the persons and organizations assigned the quality assurance functions have the required authority and organizational freedom. Irrespective of the organizational structure, the individual(s) assigned the responsibility for assuring effective execution of any portion of the quality assurance program at any location where activities subject to this subpart are being performed, shall have direct access to the levels of management necessary to perform this function.

[1] While the term “applicant” is used in these criteria, the requirements are applicable after such a person has received a license to construct and operate a commercial nuclear power plant or manufacturing facility or has received an early site permit, design approval, design certification or manufacturing license, as

applicable. These criteria will also be used for guidance in evaluating the adequacy of quality assurance programs in use by holders of construction permits, operating licenses, early site permits, design approvals, combined licenses and manufacturing licenses.

§ 53.1810 Quality assurance program.

The applicant shall establish at the earliest practicable time, consistent with the schedule for accomplishing the activities, a quality assurance program which complies with the requirements of this Subpart. The program shall be documented by written policies, procedures, or instructions and shall be carried out throughout the plant life in accordance with those policies, procedures, or instructions. The applicant shall identify the structures, systems, and components to be covered by the quality assurance program and the major organizations participating in the program, together with the designated functions of these organizations. The quality assurance program shall provide control over activities affecting the quality of the identified structures, systems, and components. Activities affecting quality shall be accomplished under suitably controlled conditions. Controlled conditions include the use of appropriate equipment; suitable environmental conditions for accomplishing the activity, such as adequate cleanness; and assurance that all prerequisites for the given activity have been satisfied. The program shall take into account the need for special controls, processes, test equipment, tools, and skills to attain the required quality, and the need for verification of quality by inspection and test. The program shall provide for indoctrination and training of personnel performing activities affecting quality as necessary to assure that suitable proficiency is achieved and maintained. The applicant shall regularly review the status and adequacy of the quality assurance program. Management of other organizations participating in the quality assurance program shall regularly review the status and adequacy of that part of the quality assurance program which they are executing.

§ 53.1815 Design control.

(a) Measures shall be established to assure that applicable regulatory requirements and the functional design criteria, as specified in the license application, for those structures, systems, and components to which this subpart applies are correctly translated into specifications, drawings, procedures and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems, and components.

(b) Measures shall be established for the identification and control of design interfaces and for the coordination among participating design organizations. These measures shall include the establishment of procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces.

(c) The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

(d) The verifying or checking process shall be performed by individuals or groups other than those who performed the original design but who may be from the same organization. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, it shall include suitable qualifications testing of a prototype unit under the most adverse design conditions.

Design control measures shall be applied to items such as the following: reactor physics, stress, thermal hydraulic, and accident analyses; compatibility of materials; accessibility for ISI, maintenance, and repair; and delineation of acceptance criteria for inspections and tests.

(e) Design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design and be approved by the organization that performed the original design unless the applicant designates another responsible organization.

§ 53.1820 Procurement document control.

Measures shall be established to assure that applicable regulatory requirements, functional design criteria, and other requirements which are necessary to assure adequate quality are suitably included or referenced in the documents for procurement of material, equipment, and services, whether purchased by the applicant or by its contractors or subcontractors. To the extent necessary, procurement documents shall require contractors or subcontractors to provide a quality assurance program consistent with the pertinent provisions of this subpart.

§ 53.1825 Instructions, procedures, and drawings.

Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

§ 53.1830 Document control.

Measures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality. These measures shall assure that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel and are distributed to and used at the location where the prescribed activity is performed. Changes to documents shall be reviewed and approved by the same organizations that performed the original review and approval unless the applicant designates another responsible organization.

§ 53.1835 Control of purchased material, equipment, and services.

Measures shall be established to assure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents. These measures shall include provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by the contractor or subcontractor, inspection at the contractor or subcontractor source, and examination of products upon delivery. Documentary evidence that material and equipment conform to the procurement requirements shall be available at the commercial nuclear plant site or manufacturing facility prior to installation or use of such material and equipment. This documentary evidence shall be retained at the commercial nuclear power plant site or manufacturing facility and shall be sufficient to identify the specific requirements, such as codes, standards, or specifications, met by the purchased material and equipment. The effectiveness of the control of quality by contractors and subcontractors shall be assessed by the applicant or designee at intervals consistent

with the importance, complexity, and quantity of the product or services.

§ 53.1840 Identification and control of materials, parts, and components.

Measures shall be established for the identification and control of materials, parts, and components, including partially fabricated assemblies. These measures shall assure that identification of the item is maintained by heat number, part number, serial number, or other appropriate means, either on the item or on records traceable to the item, as required throughout fabrication, erection, installation, and use of the item. These identification and control measures shall be designed to prevent the use of incorrect or defective material, parts, and components.

§ 53.1845 Control of special processes.

Measures shall be established to assure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

§ 53.1850 Inspection.

A program for inspection of activities affecting quality shall be established and executed by or for the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. Such inspection shall be performed by individuals other than those who performed the activity being inspected. Examinations, measurements, or tests of material or products processed shall be performed for each work operation where necessary to assure quality. If inspection of processed material or products is impossible or disadvantageous,

indirect control by monitoring processing methods, equipment, and personnel shall be provided. Both inspection and process monitoring shall be provided when control is inadequate without both. If mandatory inspection hold points, which require witnessing or inspecting by the applicant's designated representative and beyond which work shall not proceed without the consent of its designated representative are required, the specific hold points shall be indicated in appropriate documents.

§ 53.1855 Test control.

A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. The test program shall include, as appropriate, proof tests prior to installation, preoperational tests, and operational tests during commercial nuclear power plant and manufacturing facility operation, of structures, systems, and components. Test procedures shall include provisions for assuring that all prerequisites for the given test have been met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions. Test results shall be documented and evaluated to assure that test requirements have been satisfied.

§ 53.1860 Control of measuring and test equipment.

Measures shall be established to assure that tools, gages, instruments, and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated, and adjusted at specific periods to maintain accuracy within necessary limits.

§ 53.1865 Handling, storage and shipping.

Measures shall be established to control the handling, storage, shipping, cleaning and preservation of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration. When necessary for particular products, special protective environments, such as inert gas atmosphere, specific moisture content levels, and temperature levels, shall be specified and provided.

§ 53.1870 Inspection, test, and operating status.

Measures shall be established to indicate, by the use of markings such as stamps, tags, labels, routing cards, or other suitable means, the status of inspections and tests performed upon individual items of the commercial nuclear power plant or manufactured reactor module. These measures shall provide for the identification of items which have satisfactorily passed required inspections and tests, where necessary to preclude inadvertent bypassing of such inspections and tests. Measures shall also be established for indicating the operating status of structures, systems, and components of the commercial nuclear power plant or manufactured reactor module, such as by tagging valves and switches, to prevent inadvertent operation.

§ 53.1875 Nonconforming materials, parts, or components.

Measures shall be established to control materials, parts, or components which do not conform to requirements in order to prevent their inadvertent use or installation. These measures shall include, as appropriate, procedures for identification, documentation, segregation, disposition, and notification to affected organizations. Nonconforming items shall be reviewed and accepted, rejected, repaired, or reworked in accordance with documented procedures.

§ 53.1880 Corrective action.

Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management.

§ 53.1885 Quality assurance records.

Sufficient records shall be maintained to furnish evidence of activities affecting quality. The records shall include at least the following: Operating logs and the results of reviews, inspections, tests, audits, monitoring of work performance, and materials analyses. The records shall also include closely-related data such as qualifications of personnel, procedures, and equipment. Inspection and test records shall, as a minimum, identify the inspector or data recorder, the type of observation, the results, the acceptability, and the action taken in connection with any deficiencies noted. Records shall be identifiable and retrievable. Consistent with applicable regulatory requirements, the applicant shall establish requirements concerning record retention, such as duration, location, and assigned responsibility.

§ 53.1890 Audits.

A comprehensive system of planned and periodic audits shall be carried out to verify compliance with all aspects of the quality assurance program and to determine the

effectiveness of the program. The audits shall be performed in accordance with the written procedures or check lists by appropriately trained personnel not having direct responsibilities in the areas being audited. Audit results shall be documented and reviewed by management having responsibility in the area audited. Followup action, including reaudit of deficient areas, shall be taken where indicated.