Reserpine (Yohimban-16-carboxylic acid, 11,17-dimethoxy-18-[3,4,5-trimethoxybenzoyl)oxy]-, methyl ester)
Resorcinol (1,3-Benzenediol)
Saccharin and salts (1,2-Benzoisothiazolin-3-one, 1,1-dioxide, and salts)
Safrole (Benzene, 1,2-methylenedioxy-4-allyl-)
Selenious acid (Selenium dioxide)
Selenium and compounds, N.O.S.[3]
Selenium sulfide (Sulfur selenide)
Selenourea (Carbamimidoselenoic acid)
Silver and compounds, N.O.S.[3]
Silver cyanide
Sodium cyanide
Streptozotocin (D-Glucopyranose, 2-deoxy-2-(3-methyl-3-nitrosoureido)-)
Strontium sulfide
Strychnine and salts (Strychnidin-10-one, and salts)
1,2,4,5-Tetrachlorobenzene (Benzene, 1,2,4,5-tetrachloro-)
2,3,7,8-Tetrachlorodibenzo-p-dioxin (TCDD) (Dibenzo-p-dioxin, 2,3,7,8-tetrachloro-)
Tetrachloroethane, N.O.S.[3] (Ethane, tetrachloro-, N.O.S.[3])
1,1,1,2-Tetrachlorethane (Ethane, 1,1,1,2-tetrachloro-)
1,1,2,2-Tetrachlorethane (Ethane, 1,1,2,2-tetrachloro-)
Tetrachloroethane (Ethene, 1,1,2,2-tetrachloro-)
Tetrachloromethane (Carbon tetrachloride)
2,3,4,6,-Tetrachlorophenol (Phenol, 2,3,4,6-tetrachloro-)
Tetraethyldithiopyrophosphate (Dithiopyrophosphoric acid, tetraethyl-ester)
Tetraethyl lead (Plumbane, tetraethyl-)
Tetraethylpyrophosphate (Pyrophosphoric acide, tetraethyl ester)
Tetranitromethane (Methane, tetranitro-)
Thallium and compounds, N.O.S.[3]
Thallic oxide (Thallium (III) oxide)
Thallium (I) acetate (Acetic acid, thallium (I) salt)
Thallium (I) carbonate (Carbonic acid, dithallium (I) salt)
Thallium (I) chloride
Thallium (I) nitrate (Nitric acid, thallium (I) salt)
Thallium selenite
Thallium (I) sulfate (Sulfuric acid, thallium (I) salt)
Thioacetamide (Ethanethioamide)
Thiosemicarbazide (Hydrazinecarbothioamide)
Thiourea (Carbamide thio-)
Thiuram (Bis(dimethylthiocarbamoyl) disulfide)
Thorium and compounds, N.O.S.,[3] when producing thorium byproduct material
Toluene (Benzene, methyl-)
Toluenediamine (Diaminotoluene)
o-Toluidine hydrochloride (Benzenamine, 2-methyl-, hydrochloride)
Tolylene diisocyanate (Benzene, 1,3-diisocyanatomethyl-)
Toxaphene (Camphene, octachloro-)
Tribromomethane (Bromoform)
1,2,4-Trichlorobenzene (Benzene, 1,2,4-trichloro-)
1,1,1-Trichloroethane (Methyl chloroform)
1,1,2-Trichloroethane (Ethane, 1,1,2-trichloro-)
Trichloroethene (Trichloroethylene)
Trichloromethanethiol (Methanethiol, trichloro-)
Trichloromonofluoromethane (Methane, trichlorofluoro-)
2,4,5-Trichlorophenol (Phenol, 2,4,5-trichloro-)
2,4,6-Trichlorophenol (Phenol, 2,4,6-trichloro-)
2,4,5-Trichlorophenoxyacetic acid (2,4,5-T) (Acetic acid, 2,4,5-trichlorophenoxy-)
2,4,5-Trichlorophenoxypropionic acid (2,4,5-TP) (Silvex) (Propionoic acid, 2-(2,4,5-trichlorophenoxy)-)
Trichloropropane, N.O.S.[3] (Propane, trichloro-, N.O.S.[3])
1,2,3-Trichloropropane (Propane, 1,2,3-trichloro-)
O,O,O-Triethyl phosphorothioate (Phosphorothioic acid, O,O,O-triethyl ester)
sym-Trinitrobenzene (Benzene, 1,3,5-trinitro-)
Tris(1-azridinyl) phosphine sulfide (Phosphine sulfide, tris(1-aziridinyl-)
Tris(2,3-dibromopropyl) phosphate (1-Propanol, 2,3-dibromo-, phosphate)
Trypan blue (2,7-Naphthalenedisulfonic acid, 3,3′-[(3,3′-dimethyl (1,1′-biphenyl)- 4,4′-diyl)bis(azo)]bis(5-amino-4-hydroxy-, tetrasodium salt)
Uracil mustard (Uracil 5-[bis(2-chloroethyl)amino]-)
Uranium and compounds, N.O.S.[3]
Vanadic acid, ammonium salt (ammonium vanadate)
Vanadium pentoxide (Vanadium (V) oxide)
Vinyl chloride (Ethene, chloro-)
Zinc cyanide
Zinc phosphide

[50 FR 41862, Oct. 16, 1985, as amended at 52 FR 31611, Aug. 21, 1987; 52 FR 43562, Nov. 13, 1987; 53 FR 19248, May 27, 1988; 55 FR 45600, Oct. 30, 1990; 56 FR 23473, May 21, 1991; 58 FR 67661, Dec. 22, 1993; 59 FR 28229, June 1, 1994; 64 FR 17510, Apr. 12, 1999; 77 FR 35570, June 17, 2012; 81 FR 86909, Dec. 2, 2016]

PART 50—DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

GENERAL PROVISIONS

AUTHORITY: Atomic Energy Act of 1954, secs. 11, 101, 102, 103, 104, 105, 108, 122, 147, 149, 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.

SOURCE: 21 FR 355, Jan. 19, 1956, unless otherwise noted.

GENERAL PROVISIONS

§ 50.1 Basis, purpose, and procedures applicable.

The regulations in this part are promulgated by the Nuclear Regulatory Commission pursuant to the Atomic Energy Act of 1954, as amended (68 Stat. 919), and Title II of the Energy Reorganization Act of 1974 (88 Stat. 1242), to provide for the licensing of production and utilization facilities. This part also gives notice to all persons who knowingly provide to any licensee, applicant, contractor, or subcontractor, components, equipment, materials, or other goods or services, that relate to a licensee's or applicant's activities subject to this part, that they may be individually subject to NRC enforcement action for violation of § 50.5.

[63 FR 1897, Jan. 13, 1998]

§ 50.2 Definitions.

As used in this part,

Act means the Atomic Energy Act of 1954 (68 Stat. 919) including any amendments thereto.

Alternate ac source means an alternating current (ac) power source that is available to and located at or nearby a nuclear power plant and meets the following requirements:

(1) Is connectable to but not normally connected to the offsite or onsite emergency ac power systems;

(2) Has minimum potential for common mode failure with offsite power or the onsite emergency ac power sources;

(3) Is available in a timely manner after the onset of station blackout; and

(4) Has sufficient capacity and reliability for operation of all systems required for coping with station blackout and for the time required to bring and maintain the plant in safe shutdown (non-design basis accident).

Applicant means a person or an entity applying for a license, permit, or other form of Commission permission or approval under this part or part 52 of this chapter.

Atomic energy means all forms of energy released in the course of nuclear fission or nuclear transformation.

Atomic weapon means any device utilizing atomic energy, exclusive of the means for transporting or propelling the device (where such means is a separable and divisible part of the device), the prinicipal purpose of which is for use as, or for development of, a weapon, a weapon prototype, or a weapon test device.

Basic component means, for the purposes of § 50.55(e) of this chapter:

(1) When applied to nuclear power reactors, any plant structure, system, component, or part thereof necessary to assure

(i) The integrity of the reactor coolant pressure boundary,

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

(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), or § 100.11 of this chapter, as applicable.

(2) When applied to other types of facilities or portions of such facilities for which construction permits are issued under § 50.23, a component, structure, system or part thereof that is directly procured by the construction permit holder for the facility subject to the regulations of this part and in which a defect or failure to comply with any applicable regulation in this chapter, order, or license issued by the Commission could create a substantial safety hazard.

(3) In all cases, *basic component* includes safety related design, analysis, inspection, testing, fabrication, replacement parts, or consulting services that are associated with the component hardware, whether these services are performed by the component supplier or other supplier.

Byproduct material means—

(1) Any radioactive material (except special nuclear material) yielded in, or made radioactive by, exposure to the radiation incident to the process of producing or using special nuclear material;

(2)(i) Any discrete source of radium-226 that is produced, extracted, or converted after extraction, before, on, or after August 8, 2005, for use for a commercial, medical, or research activity; or

(ii) Any material that—

(A) Has been made radioactive by use of a particle accelerator; and

(B) Is produced, extracted, or converted after extraction, before, on, or after August 8, 2005, for use for a commercial, medical, or research activity; and

(3) Any discrete source of naturally occurring radioactive material, other than source material, that—

(i) The Commission, in consultation with the Administrator of the Environmental Protection Agency, the Secretary of Energy, the Secretary of Homeland Security, and the head of any other appropriate Federal agency, determines would pose a threat similar to the threat posed by a discrete source of radium-226 to the public health and safety or the common defense and security; and

(ii) Before, on, or after August 8, 2005, is extracted or converted after extraction for use in a commercial, medical, or research activity.

Certified fuel handler means, for a nuclear power reactor facility, a non-licensed operator who has qualified in accordance with a fuel handler training program approved by the Commission.

Commission means the Nuclear Regulatory Commission or its duly authorized representatives.

Committed dose equivalent means the dose equivalent to organs or tissues of reference that will be received from an intake of radioactive material by an individual during the 50-year period following the intake.

Committed effective dose equivalent is the sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to these organs or tissues.

Common defense and security means the common defense and security of the United States.

Construction or *constructing* means, for the purposes of § 50.55(e), the analysis, design, manufacture, fabrication, quality assurance, placement, erection, installation, modification, inspection, or testing of a facility or activity which is subject to the regulations in this part and consulting services related to the facility or activity that are safety related.

Controls when used with respect to nuclear reactors means apparatus and mechanisms, the manipulation of which directly affects the reactivity or power level of the reactor.

Controls when used with respect to any other facility means apparatus and mechanisms, the manipulation of which could affect the chemical, physical, metallurgical, or nuclear process of the facility in such a manner as to affect the protection of health and safety against radiation.

Cost of service regulation means the traditional system of rate regulation, or similar regulation, including ''price cap'' or ''incentive'' regulation, in which a rate regulatory authority generally allows an electric utility to charge its customers the reasonable and prudent costs of providing electricity services, including capital, operations, maintenance, fuel, decommissioning, and other costs required to provide such services.

Decommission means to remove a facility 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.

Deep-dose equivalent, which applies to external whole-body exposure, is the dose equivalent at a tissue depth of 1 cm (1000mg/cm^2).

Defect means, for the purposes of § 50.55(e) of this chapter:

(1) A deviation in a basic component delivered to a purchaser for use in a facility or activity subject to a construction permit under this part, if on the basis of an evaluation, the deviation could create a substantial safety hazard; or

(2) The installation, use, or operation of a basic component containing, a defect as defined in paragraph (1) of this definition; or

(3) A deviation in a portion of a facility subject to the construction permit of this part provided the deviation could, on the basis of an evaluation, create a substantial safety hazard.

Department and *Department of Energy* means the Department of Energy established by the Department of Energy Organization Act (Pub. L. 95–91, 91 Stat. 565, 42 U.S.C. 7101 *et seq.*), to the extent that the department, or its duly authorized representatives, exercises functions formerly vested in the Atomic Energy Commission, its Chairman, members, officers and components and transferred to the U.S. Energy Research and Development Administration and to the Administrator thereof pursuant to sections 104 (b), (c) and (d) of the Energy Reorganization Act of 1974 (Pub. L. 93–438, 88 Stat. 1233 at 1237, 42 U.S.C. 5814) and retransferred to the Secretary of Energy pursuant to section 301(a) of the Department of Energy Organization Act (Pub. L. 95–91, 91 Stat. 565 at 577–578, 42 U.S.C. 7151).

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 meet its functional goals.

Deviation means, for the purposes of §50.55(e) of this chapter, a departure from the technical or quality assurance requirements defined in procurement documents, safety analysis report, construction permit, or other documents provided for basic components installed in a facility subject to the regulations of this part.

Director means, for the purposes of §50.55(e) of this chapter, an individual, appointed or elected according to law, who is authorized to manage and direct the affairs of a corporation, partnership or other entity.

Discovery means, for the purposes of §50.55(e) of this chapter, the completion of the documentation first identifying the existence of a deviation or failure to comply potentially associated with a substantial safety hazard within the evaluation procedures discussed in §50.55(e)(1).

Electric utility means any entity that generates or distributes electricity and which 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.''

Evaluation means, for the purposes of §50.55(e) of this chapter, the process of determining whether a particular deviation could create a substantial safety hazard or determining whether a failure to comply is associated with a substantial safety hazard.

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 shall normally be prohibited. In any event, residents shall 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.

Federal Government funding for conversion means funds appropriated to the Department of Energy or to any other Federal Agency to pay directly to or to reimburse non-power reactor licensees for costs attendant to conversion.

Federal licensee means any NRC licensee, the obligations of which are guaranteed by and supported by the full faith and credit of the United States Government.

Fuel acceptable to the Commission means that the fuel replacing the existing HEU fuel in a specific non-power reactor (1) meets the operating requirements of the existing license or, through appropriate NRC safety review and approval, can be used in a manner which protects public health and safety and promotes the common defense and security; and (2) meets the Commission's policy of limiting, to the maximum extent possible, the use of HEU fuel in that reactor.

Government agency means any executive department, commission, independent establishment, corporation, wholly or partly owned by the United States of America which is an instrumentality of the United States, or any board, bureau, division, service, office, officer, authority, administration, or other establishment in the executive branch of the Government.

Highly enriched uranium (HEU) fuel means fuel in which the weight percent of U–235 in the uranium is 20% or greater. Target material, special instrumentation, or experimental devices using HEU are not included.

Historical site assessment means the identification of potential, likely, or known sources of radioactive material and radioactive contamination based on existing or derived information for the purpose of classifying a facility or site, or parts thereof, as impacted or non-impacted.

Impacted areas mean the areas with some reasonable potential for residual radioactivity in excess of natural background or fallout levels.

Incentive regulation means the system of rate regulation in which a rate regulatory authority establishes rates that an electric generator may charge its customers that are based on specified performance factors, in addition to cost-of-service factors.

License means a license, including a construction permit or operating license under this part, an early site permit, combined license or manufacturing license under part 52 of this chapter, or a renewed license issued by the Commission under this part, part 52, or part 54 of this chapter.

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

Low enriched uranium (LEU) fuel means fuel in which the weight percent of U–235 in the uranium is less than 20%.

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 in their behalf in the event of a serious accident. These guides do not specify a permissible population density or total population within this zone 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.

Major decommissioning activity means, for a nuclear power reactor facility, any activity that results in permanent removal of major radioactive components, permanently modifies the structure of the containment, or results in dismantling components for shipment containing greater than class C waste in accordance with § 61.55 of this chapter.

Major radioactive components means, for a nuclear power reactor facility, the reactor vessel and internals, steam generators, pressurizers, large bore reactor coolant system piping, and other large components that are radioactive to a comparable degree.

Non-bypassable charges mean those charges imposed over an established time period by a Government authority that affected persons or entities are required to pay to cover costs associated with the decommissioning of a nuclear power plant. Such charges include, but are not limited to, wire charges, stranded cost charges, transition charges, exit fees, other similar charges, or the securitized proceeds of a revenue stream.

Non-impacted areas mean the areas with no reasonable potential for residual radioactivity in excess of natural background or fallout levels.

Non-power reactor means a research or test reactor licensed under §§ 50.21(c) or 50.22 of this part for research and development.

Notification means the telephonic communication to the NRC Operations Center or written transmittal of information to the NRC Document Control Desk.

Nuclear reactor means an apparatus, other than an atomic weapon, designed or used to sustain nuclear fission in a self-supporting chain reaction.

Permanent cessation of operation(s) means, for a nuclear power reactor facility, a certification by a licensee to the NRC that it has permanently ceased or will permanently cease reactor operation(s), or a final legally effective order to permanently cease operation(s) has come into effect.

Permanent fuel removal means, for a nuclear power reactor facility, a certification by the licensee to the NRC that it has permanently removed all fuel assemblies from the reactor vessel.

Person means (1) any individual, corporation, partnership, firm, association, trust, estate, public or private institution, group, government agency other than the Commission or the Department, except that the Department shall be considered a person to the extent that its facilities are subject to the licensing and related regulatory authority of the Commission pursuant to section 202 of the Energy Reorganization Act of 1974, 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.

Price-cap regulation means the system of rate regulation in which a rate regulatory authority establishes rates that an electric generator may charge its customers that are based on a specified maximum price of electricity.

Procurement document means, for the purposes of § 50.55(e) of this chapter, a contract that defines the requirements which facilities or basic components must meet in order to be considered acceptable by the purchaser.

Produce, when used in relation to special nuclear material, means (1) to manufacture, make, produce, or refine special nuclear material; (2) to separate special nuclear material from other substances in which such material may be contained; or (3) to make or to produce new special nuclear material.

Production facility means:

(1) Any nuclear reactor designed or used primarily for the formation of plutonium or uranium–233; or

(2) Any facility designed or used for the separation of the isotopes of plutonium, except laboratory scale facilities designed or used for experimental or analytical purposes only; or

(3) Any facility designed or used for the processing of irradiated materials containing special nuclear material, except (i) laboratory scale facilities designed or used for experimental or analytical purposes, (ii) facilities in which the only special nuclear materials contained in the irradiated material to be processed are uranium enriched in the isotope U–235 and plutonium produced by the irradiation, if the material processed contains not more than 10^{-6} grams of plutonium per gram of U–235 and has fission product activity not in excess of 0.25 millicuries of fission products per gram of U–235, and (iii) facilities in which processing is conducted pursuant to a license issued under parts 30 and 70 of this chapter, or equivalent regulations of an Agreement State, for the receipt, possession, use, and transfer of irradiated special nuclear material, which authorizes the processing of the irradiated material on a batch basis for the separation of selected fission products and limits the process batch to not more than 100 grams of uranium enriched in the isotope 235 and not more than 15 grams of any other special nuclear material.

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

Reactor coolant pressure boundary means all those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, 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,

(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.

Research and development means (1) theoretical analysis, exploration, or experimentation; or (2) the extension of investigative findings and theories of a scientific or technical nature into practical application for experimental and demonstration purposes, including the experimental production and testing of models, devices, equipment, materials, and processes.

Responsible officer means, for the purposes of § 50.55(e) of this chapter, the president, vice-president, or other individual in the organization of a corporation, partnership, or other entity who is vested with executive authority over activities subject to this part.

Restricted Data means all data concerning (1) design, manufacture, or utilization of atomic weapons; (2) the production of special nuclear material; or (3) the use of special nuclear material in the production of energy, but shall not include data declassified or removed from the Restricted Data category pursuant to section 142 of the Act.

Safe shutdown (non-design basis accident (non-DBA)) for station blackout means bringing the plant to those shutdown conditions specified in plant technical specifications as Hot Standby or Hot Shutdown, as appropriate (plants have the option of maintaining the RCS at normal operating temperatures or at reduced temperatures).

Safety-related structures, systems and components 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 § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

Source material means source material as defined in subsection 11z. of the Act and in the regulations contained in part 40 of this chapter.

Source term refers to the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release.

Special nuclear material 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 act, 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.

Station blackout means the complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant (*i.e.*, loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency ac power system). Station blackout does not include the loss of available ac power to buses fed by station batteries through inverters or by alternate ac sources as defined in this section, nor does it assume a concurrent single failure or design basis accident. At single unit sites, any emergency ac power source(s) in excess of the number required to meet minimum redundancy requirements (*i.e.*, single failure) for safe shutdown (non-DBA) is assumed to be available and may be designated as an alternate power source(s) provided the applicable requirements are met. At multi-unit sites, where the combination of emergency ac power sources exceeds the minimum redundancy requirements for safe shutdown (non-DBA) of all units, the remaining emergency ac power sources may be used as alternate ac power sources provided they meet the applicable requirements. If these criteria are not met, station blackout must be assumed on all the units.

Substantial safety hazard means, for the purposes of § 50.55(e) of this chapter, 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 authorized by the construction permit issued under this part.

Testing facility means a nuclear reactor which is of a type described in § 50.21(c) of this part and for which an application has been filed for a license authorizing operation at:

(1) A thermal power level in excess of 10 megawatts; or

(2) A thermal power level in excess of 1 megawatt, if the reactor is to contain:

(i) A circulating loop through the core in which the applicant proposes to conduct fuel experiments; or

(ii) A liquid fuel loading; or

(iii) An experimental facility in the core in excess of 16 square inches in cross-section.

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).

Unique purpose means a project, program, or commercial activity which cannot reasonably be accomplished without the use of HEU fuel, and may include: (1) A specific experiment, program, or commercial activity (typically long-term) that significantly serves the U.S. national interest and cannot be accomplished without the use of HEU fuel; (2) Reactor physics or reactor development based explicitly on the use of HEU fuel; (3) Research projects based on neutron flux levels or spectra attainable only with HEU fuel; or (4) A reactor core of special design that could not perform its intended function without using HEU fuel.

United States, when used in a geographical sense, includes Puerto Rico and all territories and possessions of the United States.

Utilization facility means:

(1) Any nuclear reactor other than one designed or used primarily for the formation of plutonium or U–233; or

(2) An accelerator-driven subcritical operating assembly used for the irradiation of materials containing special nuclear material and described in the application assigned docket number 50–608.

NOTE: Pursuant to subsections 11v. and 11cc., respectively, of the Act, the Commission may from time to time add to, or otherwise alter, the foregoing definitions of production and utilization facility. It may also include as a facility an important component part especially designed for a facility, but has not at this time included any component parts in the definitions.

[21 FR 355, Jan. 19, 1956]

EDITORIAL NOTE: For FEDERAL REGISTER citations affecting § 50.2, see the List of CFR Sections Affected, which appears in the Finding Aids section of the printed volume and at *www.govinfo.gov*.

§ 50.3 Interpretations.

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

§ 50.4 Written communications.

(a) *General requirements*. All correspondence, reports, applications, and other written communications from the applicant or licensee to the Nuclear Regulatory Commission 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 must be submitted to the persons listed below (addresses for the NRC Regional Offices are listed in appendix D to part 20 of this chapter).

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

(2) *Applications for permits and licenses, and amendments to applications.* Applications for construction permits, applications for operating licenses and amendments to either type of application must be submitted as follows, except as otherwise specified in paragraphs (b)(3) through (b)(7) in this section.

(i) Applications for licenses for facilities described in § 50.21 (a) and (c) and amendments to these applications must be sent to the NRC's Document Control Desk, with a copy to the appropriate Regional Office. If the application or amendment is on paper, the submission to the Document Control Desk must be the signed original.

(ii) Applications for permits and licenses for facilities described in § 50.21(b) or § 50.22, and amendments to these applications must be sent 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 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 under § 50.30(a)(6) must be submitted to the NRC's Document Control Desk, with a copy to the appropriate Regional Office. If the communication is on paper, the submission to the Document Control Desk must be the signed original.

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

(i) Physical security plan under § 50.34;

(ii) Safeguards contingency plan under § 50.34;

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

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

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

(i) Emergency plan under § 50.34;

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

(iii) Emergency implementing procedures under appendix E.V of this part.

(6) *Updated FSAR.* An updated Final Safety Analysis Report (FSAR) or replacement pages, under § 50.71(e) 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 § 50.54(a)(3) or § 50.55(f)(3), or a change to a licensee's NRC-accepted quality assurance topical report under § 50.54(a)(3) or § 50.55(f)(3), must be submitted to the NRC's Document Control Desk, with a copy to the appropriate Regional Office, and a copy to the appropriate NRC Resident Inspector if one has been assigned to the site of the facility. If the communication is on paper, the submission to the Document Control Desk must be the signed original.

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

(8) *Certification of permanent cessation of operations.* The licensee's certification of permanent cessation of operations, under § 50.82(a)(1), must state the date on which operations have ceased or will cease, and must be submitted to the NRC's Document Control Desk. This submission must be under oath or affirmation.

(9) *Certification of permanent fuel removal.* The licensee's certification of permanent fuel removal, under § 50.82(a)(1), must state the date on which the fuel was removed from the reactor vessel and the disposition of the fuel, and must be submitted to the NRC's Document Control Desk. This submission must be under oath or affirmation.

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

(d) *Regulation governing submission.* Licensees and applicants 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.

(e) *Conflicting requirements.* The communications requirements contained in this section and §§ 50.12, 50.30, 50.36, 50.36a. 50.44, 50.49, 50.54, 50.55, 50.55a, 50.59, 50.62, 50.71, 50.73, 50.82, 50.90, and 50.91 supersede and replace all existing requirements in any license conditions or technical specifications in effect on January 5, 1987. Exceptions to these requirements must be approved by the Office of the Chief Information Officer, Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone

(301) 415–7233, e-mail *INFOCOLLECTS@nrc.gov*.

[68 FR 58808, Oct. 10, 2003, as amended at 74 FR 62682, Dec. 1, 2009; 80 FR 74979, Dec. 1, 2015]

§ 50.5 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 (a)(2) of this section may be subject to enforcement action in accordance with the procedures in 10 CFR part 2, subpart B.

(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.

[63 FR 1897, Jan. 13, 1998]

§ 50.7 Employee protection.

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

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

(i) Providing the Commission or his or her employer information about alleged violations of either of the statutes named in paragraph (a) introductory text 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) introductory text or under these requirements if the employee has identified the alleged illegality to the employer;

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

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

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

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

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

Atomic Energy Act of 1954, as amended.

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

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

(1) Denial, revocation, or suspension of the license.

(2) Imposition of a civil penalty on the licensee, applicant, or a contractor or subcontractor of the licensee or applicant.

(3) Other enforcement action.

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

(e)(1) Each licensee and each applicant for a license shall 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 not 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 U.S. Nuclear Regulatory Commission Regional Office listed in appendix D to part 20 of this chapter, 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 Energy Reorganization Act of 1974, as amended, may contain any provision which would prohibit, restrict, or otherwise discourage an employee from participating in protected activity as defined in paragraph (a)(1) of this section including, but not limited to, providing information to the NRC or to his or her employer on potential violations or other matters within NRC's regulatory responsibilities.

[58 FR 52410, Oct. 8, 1993, as amended at 60 FR 24551, May 9, 1995; 61 FR 6765, Feb. 22, 1996; 68 FR 58809, Oct. 10, 2003; 72 FR 63974, Nov. 14, 2007; 73 FR 30458, May 28, 2008; 79 FR 66603, Nov. 10, 2014; 83 FR 58465, Nov. 20, 2018]

§ 50.8 Information collection requirements: OMB approval.

(a) The Nuclear Regulatory Commission has submitted the information collection requirements contained in this part to the Office of Management and Budget (OMB) for approval as required by the Paperwork Reduction Act (44 U.S.C. 3501 *et seq.*). The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number. OMB has approved the information collection requirements contained in this part under control number 3150–0011.

(b) The approved information collection requirements contained in this part appear in §§ 50.12, 50.30, 50.33, 50.34, 50.34a, 50.35, 50.36, 50.36a, 50.36b, 50.44, 50.46, 50.47, 50.48, 50.49, 50.54, 50.55, 50.55a, 50.59, 50.60, 50.61, 50.61a, 50.62, 50.63, 50.64, 50.65, 50.66, 50.68, 50.69, 50.70, 50.71, 50.72, 50.74, 50.75, 50.80, 50.82, 50.90, 50.91, 50.120, 50.150, 50.155, and appendices A, B, E, G, H, I, J, K, M, N,O, Q, R, and S to this part.

(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:

(1) In § 50.73, NRC Form 366 is approved under control number 3150–0104.

(2) In § 50.78, IAEA Design Information Questionnaire forms are approved under control number 3150–0056.

(3) In § 50.78, DOC/NRC Forms AP–1, AP–A, and associated forms are approved under control numbers 0694–0135.

[49 FR 19627, May 9, 1984, as amended at 58 FR 68731, Dec. 29, 1993; 60 FR 65468, Dec. 19, 1995; 61 FR 65172, Dec. 11, 1996; 62 FR 52187, Oct. 6, 1997; 67 FR 67099, Nov. 4, 2002; 68 FR 19727, Apr. 22, 2003; 69 FR 68046, Nov. 22, 2004; 70 FR 61887, Oct. 27, 2005; 73 FR 78605, Dec. 23, 2008; 74 FR 28145, June 12, 2009; 75 FR 22, Jan. 4, 2010; 77 FR 39907, July 6, 2012; 83 FR 58465, Nov. 20, 2018; 84 FR 39718, Aug. 9, 2019; 85 FR 65662, Oct. 16, 2020]

§ 50.9 Completeness and accuracy of information.

(a) Information provided to the Commission by an applicant for a 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 shall be complete and accurate in all material respects.

(b) Each applicant or licensee shall 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 shall be provided to the Administrator of the appropriate Regional Office within two working days of identifying the information. This requirement is not applicable to information which is already required to be provided to the Commission by other reporting or updating requirements.

[52 FR 49372, Dec. 31, 1987]

REQUIREMENT OF LICENSE, EXCEPTIONS

§ 50.10 License required; limited work authorization.

(a) *Definitions*. As used in this section, *construction* means the activities in paragraph (a)(1) of this section, and does not mean the activities in paragraph (a)(2) of this section.

(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, or components (SSCs) of a facility, as defined in 10 CFR 50.2;

(ii) SSCs relied upon to mitigate accidents or transients or 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 10 CFR 50.48 and criterion 3 of 10 CFR part 50, appendix A; and

(vii) Onsite emergency facilities, that is, technical support and operations support centers, necessary to comply with 10 CFR 50.47 and 10 CFR part 50, appendix E.

(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 sewerage treatment facilities, and transmission lines;

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

(ix) Manufacture of a nuclear power reactor under a manufacturing license under subpart F of part 52 of this chapter to be installed at the proposed site and to be part of the proposed facility; or

(x) With respect to production or utilization facilities, other than testing facilities and nuclear power plants, required to be licensed under Section 104.a or Section 104.c of the Act, the erection of buildings which will be used for activities other than operation of a facility and which may also be used to house a facility (e.g., the construction of a college laboratory building with space for installation of a training reactor).

(b) *Requirement for license.* Except as provided in §50.11 of this chapter, no person within the United States shall transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, or use any production or utilization facility except as authorized by a license issued by the Commission.

(c) *Requirement for construction permit, early site permit authorizing limited work authorization activities, combined license, or limited work authorization.* No person may begin the construction of a production or utilization facility on a site on which the facility is to be operated until that person has been issued either a construction permit under this part, a combined license under part 52 of this chapter, an early site permit authorizing the activities under paragraph (d) of this section, or a limited work authorization under paragraph (d) of this section.

(d) *Request for limited work authorization.* (1) Any person to whom the Commission may otherwise issue either a license or permit under Sections 103, 104.b, or 185 of the Act for a facility of the type specified in §§50.21(b)(2), (b)(3), or 50.22 of this chapter, or a testing facility, may request a limited work authorization 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 paragraph (c) of this section.

(2) An application for a limited work authorization may be submitted as part of a complete application for a construction permit or combined license in accordance with 10 CFR 2.101(a)(1) through (a)(5), or as a partial application in accordance with 10 CFR 2.101(a)(9). An application for a limited work authorization must be submitted by an applicant for or holder of an early site permit as a complete application in accordance with 10 CFR 2.101(a)(1) through (a)(4).

(3) The application must include:

(i) A safety analysis report required by 10 CFR 50.34, 10 CFR 52.17 or 10 CFR 52.79 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, but limited to those portions of the facility that are within the scope of the limited work authorization. The safety analysis report must demonstrate that activities conducted under the limited work authorization 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 limited work authorization;

(ii) An environmental report in accordance with § 51.49 of this chapter; and

(iii) A plan for redress of activities performed under the limited work authorization, should limited work activities be terminated by the holder or the limited work authorization 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.

(e) *Issuance of limited work authorization.* (1) The Director of the Office of Nuclear Reactor Regulation may issue a limited work authorization only after:

(i) The NRC staff issues the final environmental impact statement for the limited work authorization in accordance with subpart A of part 51 of this chapter;

(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 Act, and the Commission's regulations applicable to the activities to be conducted under the limited work authorization, have been met. The applicant is technically qualified to engage in the activities authorized. Issuance of the limited work authorization 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 limited work authorization that would constitute good cause for withholding the authorization.

(2) Each limited work authorization will specify the activities that the holder is authorized to perform.

(f) *Effect of limited work authorization.* Any activities undertaken under a limited work authorization are entirely at the risk of the applicant and, except as to the matters determined under paragraph (e)(1) of this section, the issuance of the limited work authorization has no bearing on the issuance of a construction permit or combined license with respect to the requirements of the Act, and rules, regulations, or orders issued under the Act. The environmental impact statement for a construction permit or combined license application for which a limited work authorization was previously issued will not address, and the presiding officer will not consider, the sunk costs of the holder of limited work authorization in determining the proposed action (*i.e.*, issuance of the construction permit or combined license).

(g) *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 limited work authorization 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 limited work authorization, or upon effectiveness of the Commission's final decision denying the associated construction permit application or the underlying combined license application, as applicable.

[72 FR 57441, Oct. 9, 2007; 84 FR 65644, Nov. 29, 2019]

§ 50.11 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 Act, 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 Administration facilities of the types subject to licensing pursuant to section 202 of the Energy Reorganization Act of 1974 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 under a prime contract for:

(A) The performance of work for the Department 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 production or utilization facility in a United States owned vehicle or vessel; or

(ii) By a prime contractor or subcontractor of the Commission or the Department 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;

(2)(i) The construction or operation of a production or utilization facility for the Department at a United States government-owned or controlled site, including the transportation of the production or 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 production or utilization facility for the Department 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 production or utilization facility for the Department in a United States government-owned vehicle or vessel: *Provided,* That such activities are conducted by a prime contractor of the Department under a prime contract with the Department.

(ii) The construction or operation of a production or utilization facility by a prime contractor or subcontractor of the Commission or the Department 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.

(c) The transportation or possession of any production or utilization facility by a common or contract carrier or warehousemen in the regular course of carriage for another or storage incident thereto.

[40 FR 8788, Mar. 3, 1975, as amended at 65 FR 54950, Sept. 12, 2000]

§ 50.12 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—

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

(2) 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; or

(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; or

(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; or

(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; or

(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 satisfying paragraph (a)(2) of this section, the exemption may not be granted until the Executive Director for Operations has consulted with the Commission.

(b) Any person may request an exemption permitting the conduct of activities prior to the issuance of a construction permit prohibited by § 50.10. 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 effected 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 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 shall not be deemed to constitute a commitment to issue a construction permit. During the period of any exemption granted pursuant to this paragraph (b), any activities conducted shall be carried out in such a manner as will minimize or reduce their environmental impact.

[37 FR 5748, Mar. 21, 1972, as amended at 40 FR 8789, Mar. 3, 1975; 50 FR 50777, Dec. 12, 1985]

§ 50.13 Attacks and destructive acts by enemies of the United States; and defense activities.

An applicant for a license to construct and operate a production or utilization facility, or for an amendment to such license, is 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.

[32 FR 13445, Sept. 26, 1967]

CLASSIFICATION AND DESCRIPTION OF LICENSES

§ 50.20 Two classes of licenses.

Licenses will be issued to named persons applying to the Commission therefor, and will be either class 104 or class 103.

§ 50.21 Class 104 licenses; for medical therapy and research and development facilities.

A class 104 license will be issued, to an applicant who qualifies, for any one or more of the following: to transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, or use.

(a) A utilization facility for use in medical therapy; or

(b)(1) A production or utilization facility the construction or operation of which was licensed pursuant to subsection 104b of the Act prior to December 19, 1970;

(2) A production or utilization facility for industrial or commercial purposes constructed or operated under an arrangement with the Administration entered into under the Cooperative Power Reactor Demonstration Program, except as otherwise specifically required by applicable law; and

(3) A production or utilization facility for industrial or commercial purposes, when specifically authorized by law.

(c) A production or utilization facility, which is useful in the conduct of research and development activities of the types specified in section 31 of the Act, and which is not a facility of the type specified in paragraph (b) of this section or in § 50.22.

[21 FR 355, Jan. 19, 1956, as amended at 31 FR 15145, Dec. 2, 1966; 35 FR 19659, Dec. 29, 1970; 38 FR 11446, May 8, 1973; 43 FR 6924, Feb. 17, 1978]

§ 50.22 Class 103 licenses; for commercial and industrial facilities.

A class 103 license will be issued, to an applicant who qualifies, for any one or more of the following: To transfer or receive in interstate commerce, 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 production or utilization facility which is useful in the conduct of research and development activities of the types specified in section 31 of the Act, 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.

[38 FR 11446, May 8, 1973, as amended at 43 FR 6924, Feb. 17, 1978]

§ 50.23 Construction permits.

A construction permit for the construction of a production or utilization facility will be issued before the issuance of a license if the application is otherwise acceptable, and will be converted upon completion of the facility and Commission action, into a license as provided in § 50.56. However, if a combined license for a nuclear power reactor is issued under part 52 of this chapter, the construction permit and operating license are deemed to be combined in a single license. A construction permit for the alteration of a production or utilization facility will be issued before the issuance of an amendment of a license, if the application for amendment is otherwise acceptable, as provided in § 50.92.

[72 FR 49490, Aug. 28, 2007, as amended at 81 FR 86909, Dec. 2, 2016]

APPLICATIONS FOR LICENSES, CERTIFICATIONS, AND REGULATORY APPROVALS; FORM; CONTENTS; INELIGIBILITY OF CERTAIN APPLICANTS

§ 50.30 Filing of application; oath or affirmation.

(a) *Serving of applications.* (1) Each filing of an application for a standard design approval or license to construct and/or operate, or manufacture, a production or utilization facility (including an early site permit, combined license, and manufacturing license under part 52 of this chapter), and any amendments to the applications, must be submitted to the U.S. Nuclear Regulatory Commission in accordance with § 50.4 or § 52.3 of this chapter, as applicable.

(2) The applicant shall maintain the capability to generate additional copies of the general information and the safety analysis report, or part thereof or amendment thereto, for subsequent distribution in accordance with the written instructions of the Director, Office of Nuclear Reactor Regulation, or Director, Office of Nuclear Material Safety and Safeguards, as appropriate.

(3) Each applicant for a construction permit under this part, or an early site permit, combined license, or manufacturing license under part 52 of this chapter, shall, upon notification by the Atomic Safety and Licensing Board appointed to conduct the public hearing required by the Atomic Energy Act, 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 the Atomic Safety and Licensing Board. Any subsequent amendment to the application must be served on those served copies of the application and must be submitted to the U.S. Nuclear Regulatory Commission as specified in § 50.4 or § 52.3 of this chapter, as applicable.

(4) 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 shall certify that the updated copies of the application contain the current contents of the application submitted in accordance with the requirements of this part.

(5) 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 which is the subject of the application for public inspection and copying.

(6) 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 § 50.4 or § 52.3 of this chapter, as applicable, to enable the Director, Office of Nuclear Reactor Regulation, or the Director, Office of Nuclear

Material Safety and Safeguards, as appropriate, to determine whether the application is sufficiently complete to permit docketing.

(b) *Oath or affirmation.* Each application for a standard design approval 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) *Application for operating licenses.* The holder of a construction permit for a production or utilization facility shall, at the time of submission of the final safety analysis report, file an application for an operating license or an amendment to an application for a license to construct and operate a production or utilization facility for the issuance of an operating license, as appropriate. The application or amendment shall state the name of the applicant, the name, location and power level, if any, of the facility and the time when the facility is expected to be ready for operation, and may incorporate by reference any pertinent information submitted in accordance with §50.33 with the application for a construction permit.

(e) *Filing Fees.* Each application for a standard design approval or production or utilization facility license, including, whenever appropriate, a construction permit or early site permit, other than a license exempted from part 170 of this chapter, shall be accompanied by the fee prescribed in part 170 of this chapter. 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, combined license, or manufacturing license for a nuclear power reactor, testing facility, fuel reprocessing plant, or other production or utilization facility whose construction or operation may be determined by the Commission to have a significant impact in the environment, shall be accompanied by an Environmental Report required under subpart A of part 51 of this chapter.

[23 FR 3115, May 10, 1958, as amended at 33 FR 10924, Aug. 1, 1968; 34 FR 6307, Apr. 3, 1969; 35 FR 19660, Dec. 29, 1970; 37 FR 5749, Mar. 21, 1972; 51 FR 40307, Nov. 6, 1986; 64 FR 48951, Sept. 9, 1999; 68 FR 58809, Oct. 10, 2003; 72 FR 49490, Aug. 28, 2007; 73 FR 5721, Jan. 31, 2008; 84 FR 65644, Nov. 29, 2019]

§ 50.31 Combining applications.

An applicant may combine in one his several applications for different kinds of licenses under the regulations in this chapter.

§ 50.32 Elimination of repetition.

In his application, the applicant may incorporate by reference information contained in previous applications, statements or reports filed with the Commission: *Provided,* That such references are clear and specific.

§ 50.33 Contents of applications; general information.

Each application shall state:

(a) Name of applicant;

(b) Address of applicant;

(c) Description of business or occupation of applicant;

(d)(1) If applicant is an individual, state citizenship.

(2) If applicant is a partnership, state 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, state:

(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;

(iii) Whether it is owned, controlled, or dominated by an alien, a foreign corporation, or foreign government, and if so, give details.

(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 class 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) Except for an electric utility applicant for a license to operate a utilization facility of the type described in §50.21(b) or §50.22, information sufficient to demonstrate to the Commission the financial qualification of the applicant to carry out, in accordance with regulations in this chapter, the activities for which the permit or license is sought. As applicable, the following should be provided:

(1) If the application is for a construction permit, the applicant shall 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 shall submit estimates of the total construction costs of the facility and related fuel cycle costs, and shall indicate the source(s) of funds to cover these costs.

(2) If the application is for an operating license, the applicant shall 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 shall submit estimates for total annual operating costs for each of the first five years of operation of the facility. The applicant shall also indicate the source(s) of funds to cover these costs. An applicant seeking to renew or extend the term of an operating license for a power reactor need not submit the financial information that is required in an application for an initial license. Applicants to renew or extend the term of an operating license for a nonpower reactor shall include the financial information that is required in an application for an initial license.

(3) If the application is for a combined license under subpart C of part 52 of this chapter, the applicant shall submit the information described in paragraphs (f)(1) and (f)(2) of this section.

(4) Each application for a construction permit, operating license, or combined license submitted by a newly-formed entity organized for the primary purpose of constructing and/or operating a facility must also include information showing:

(i) The legal and financial relationships it has or proposes to have with its stockholders or owners;

(ii) The stockholders' or owners' financial ability to meet 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.

(5) 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.

(g) If the application is for an operating license or combined license for a nuclear power reactor, or if the application is for an early site permit and contains plans for coping with emergencies under §52.17(b)(2)(ii) of this chapter, the applicant shall submit radiological emergency response plans of State and local governmental entities in the United States that are wholly or partially within the plume exposure pathway emergency planning zone (EPZ),[4] as well as the plans of State governments wholly or partially within the ingestion pathway EPZ.[5] If the application is for an early site permit that, under 10 CFR 52.17(b)(2)(i), proposes major features of the emergency plans describing the EPZs, then the descriptions of the EPZs must meet the

[4] Emergency planning zones (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.

[5] If the State and local 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.

requirements of this paragraph. Generally, the plume exposure pathway EPZ for nuclear power reactors shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall 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 MW thermal. The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway.

(h) If the applicant, other than an applicant for a combined license, proposes to construct or alter a production or utilization facility, the application shall state the earliest and latest dates for completion of the construction or alteration.

(i) If the proposed activity is the generation and distribution of electric energy under a class 103 license, 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.

(j) If the application contains Restricted Data or other defense information, it shall be prepared in such manner that all Restricted Data and other defense information are separated from the unclassified information.

(k)(1) For an application for an operating license or combined license for a production or utilization facility, information in the form of a report, as described in § 50.75, indicating how reasonable assurance will be provided that funds will be available to decommission the facility.

(2) On or before July 26, 1990, each holder of an operating license for a production or utilization facility in effect on July 27, 1990, shall submit information in the form of a report as described in § 50.75 of this part, indicating how reasonable assurance will be provided that funds will be available to decommission the facility.

[21 FR 355, Jan. 19, 1956, as amended at 35 FR 19660, Dec. 29, 1970; 38 FR 3956, Feb. 9, 1973; 45 FR 55408, Aug. 19, 1980; 49 FR 35752, Sept. 12, 1984; 53 FR 24049, June 27, 1988; 69 FR 4448, Jan. 30, 2004; 72 FR 49490, Aug. 28, 2007]

§ 50.34 Contents of applications; technical information.

(a) *Preliminary safety analysis report.* Each application for a construction permit shall include a preliminary safety analysis report. The minimum information[5] to be included shall consist of the following:

(1) Stationary power reactor applicants for a construction permit who apply on or after January 10, 1997, shall comply with paragraph (a)(1)(ii) of this section. All other applicants for a construction permit shall comply with paragraph (a)(1)(i) of this section.

(i) A description and safety assessment of the site on which the facility is to be located, with appropriate attention to features affecting facility design. Special attention should be directed to the site evaluation factors identified in part 100 of this chapter. The assessment must contain an analysis and evaluation of the major structures, systems and components of the facility which bear significantly on the acceptability of the site under the site evaluation factors identified in part 100 of this chapter, assuming that the facility will be operated at the ultimate power level which is contemplated by the applicant. With respect to operation at the projected initial power

[5] The applicant may provide information required by this paragraph in the form of a discussion, with specific references, of similarities to and differences from, facilities of similar design for which applications have previously been filed with the Commission.

level, the applicant is required to submit information prescribed in paragraphs (a)(2) through (a)(8) of this section, as well as the information required by this paragraph, in support of the application for a construction permit, or a design approval.

(ii) A description and safety assessment of the site and a safety assessment of the facility. It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. The following power reactor design characteristics and proposed operation will be taken into consideration by the Commission:

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

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

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

(D) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant shall assume a fission product release[6] from the core into the containment assuming that the facility is operated at the ultimate power level contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable site characteristics, including site meteorology, to evaluate the offsite radiological consequences. Site characteristics must comply with part 100 of this chapter. The evaluation must determine that:

(*1*) An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem[7] 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 total effective dose equivalent (TEDE);

(E) With respect to operation at the projected initial power level, the applicant is required to submit information prescribed in paragraphs (a)(2) through (a)(8) of this section, as well as the information required by paragraph (a)(1)(i) of this section, in support of the application for a construction permit.

(2) A summary description and discussion of the facility, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

---

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

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

(3) The preliminary design of the facility including:

(i) The principal design criteria for the facility.[8] Appendix A, General Design Criteria for Nuclear Power Plants, establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants for construction permits in establishing principal design criteria for other types of nuclear power units;

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

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

(4) A preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling performance and the need for high point vents following postulated loss-of-coolant accidents must be performed in accordance with the requirements of § 50.46 and § 50.46a of this part for facilities for which construction permits may be issued after December 28, 1974.

(5) An identification and justification for the selection of those variables, conditions, or other items which are determined as the result of preliminary safety analysis and evaluation to be probable subjects of technical specifications for the facility, with special attention given to those items which may significantly influence the final design: *Provided, however,* That this requirement is not applicable to an application for a construction permit filed prior to January 16, 1969.

(6) A preliminary plan for the applicant's organization, training of personnel, and conduct of operations.

(7) A description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility. Appendix B, ''Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,'' sets forth the requirements for quality assurance programs for nuclear power plants and fuel reprocessing plants. The description of the quality assurance program for a nuclear power plant or a fuel reprocessing plant shall include a discussion of how the applicable requirements of appendix B will be satisfied.

(8) An identification of those structures, systems, or components of the facility, if any, which require research and development to confirm the adequacy of their design; and identification and description of the research and development program which will be conducted to resolve any safety questions associated with such structures, systems or components; and a schedule of the research and development program showing that such safety questions will be resolved at or before the latest date stated in the application for completion of construction of the facility.

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

(10) A discussion of the applicant's preliminary plans for coping with emergencies. Appendix E sets forth items which shall be included in these plans.

(11) On or after February 5, 1979, applicants who apply for construction permits for nuclear power plants to be built on multiunit sites shall identify potential hazards to the structures, systems and components important to safety of operating nuclear facilities from construction activities. A discussion shall also be included of any managerial and administrative controls

[8] General design criteria for chemical processing facilities are being developed.

[9] [Reserved]

that will be used during construction to assure the safety of the operating unit.

(12) On or after January 10, 1997, stationary power reactor applicants who apply for a construction permit, as partial conformance to General Design Criterion 2 of appendix A to this part, shall comply with the earthquake engineering criteria in appendix S to this part.

(13) On or after July 13, 2009, power reactor applicants who apply for a construction permit shall submit the information required by 10 CFR 50.150(b) as a part of their preliminary safety analysis report.

(b) *Final safety analysis report.* Each application for an operating license shall include a final safety analysis report. The final safety analysis report shall include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and shall include the following:

(1) 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 part 100 of this chapter.

(2) A description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

(i) For nuclear reactors, such items as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent.

(ii) For facilities other than nuclear reactors, such items as the chemical, physical, metallurgical, or nuclear process to be performed, instrumentation and control systems, ventilation and filter systems, electrical systems, auxiliary and emergency systems, and radioactive waste handling systems shall be discussed insofar as they are pertinent.

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

(4) A final analysis and evaluation of the design and performance of structures, systems, and components with the objective stated in paragraph (a)(4) of this section and taking into account any pertinent information developed since the submittal of the preliminary safety analysis report. Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §50.46 for facilities for which a license to operate may be issued after December 28, 1974.

(5) 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.

(6) The following information concerning facility operation:

(i) The applicant's organizational structure, allocations of responsibilities and authorities, and personnel qualifications requirements.

(ii) Managerial and administrative controls to be used to assure safe operation. Appendix B, ''Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,'' sets forth the requirements for such controls for nuclear power plants and fuel reprocessing plants. The information on the controls to be used for a nuclear power plant or a fuel reprocessing plant shall include a discussion of how the applicable requirements of appendix B will be satisfied.

(iii) Plans for preoperational testing and initial operations.

(iv) Plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components.

(v) Plans for coping with emergencies, which shall include the items specified in appendix E.

(vi) Proposed technical specifications prepared in accordance with the requirements of § 50.36.

(vii) On or after February 5, 1979, applicants who apply for operating licenses for nuclear power plants to be operated on multiunit sites shall include an evaluation of the potential hazards to the structures, systems, and components important to safety of operating units resulting from construction activities, as well as a description of the managerial and administrative controls to be used to provide assurance that the limiting conditions for operation are not exceeded as a result of construction activities at the multiunit sites.

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

(8) A description and plans for implementation of an operator requalification program. The operator requalification program must as a minimum, meet the requirements for those programs contained in § 55.59 of part 55 of this chapter.

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

(10) On or after January 10, 1997, stationary power reactor applicants who apply for an operating license, as partial conformance to General Design Criterion 2 of appendix A to this part, shall comply with the earthquake engineering criteria of appendix S to this part. However, for those operating license applicants and holders whose construction permit was issued before January 10, 1997, the earthquake engineering criteria in Section VI of appendix A to part 100 of this chapter continues to apply.

(11) On or after January 10, 1997, stationary power reactor applicants who apply for an operating license, shall provide a description and safety assessment of the site and of the facility as in § 50.34(a)(1)(ii). However, for either an operating license applicant or holder whose construction permit was issued before January 10, 1997, the reactor site criteria in part 100 of this chapter and the seismic and geologic siting criteria in appendix A to part 100 of this chapter continues to apply.

(12) On or after July 13, 2009, power reactor applicants who apply for an operating license which is subject to 10 CFR 50.150(a) shall submit the information required by 10 CFR 50.150(b) as a part of their final safety analysis report.

(c) *Physical security plan.* (1) Each applicant for an operating license for a production or utilization facility that will be subject to §§ 73.50 and 73.60 of this chapter must include a physical security plan.

(2) Each applicant for an operating license for a utilization facility that will be subject to the requirements of § 73.55 of this chapter must include a physical security plan, a training and qualification plan in accordance with the criteria set forth in appendix B to part 73 of this chapter, and a cyber security plan in accordance with the criteria set forth in § 73.54 of this chapter.

(3) The physical security plan must describe how the applicant will meet the requirements of part 73 of this chapter (and part 11 of this chapter, if applicable, including the identification and description of jobs as required by § 11.11(a) of this chapter, at the proposed facility). Security plans 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.

(d) *Safeguards contingency plan.* (1) Each application for a license to operate a production or utilization facility that will be subject to §§ 73.50 and 73.60 of this chapter must include a licensee safeguards contingency plan in accordance with the criteria set forth in section I of appendix C to part 73 of this chapter. The ''implementation procedures'' required per section I of appendix C to part 73 of this chapter do not have to be submitted to the Commission for approval.

(2) Each application for a license to operate a utilization facility that will be subject to § 73.55 of this chapter must include a licensee safeguards contingency plan in accordance with the criteria set forth in section II of appendix C to part 73 of this chapter. The ''implementing procedures'' required in section II of appendix C to part 73 of this chapter do not have to be submitted to the Commission for approval.

(e) *Protection against unauthorized disclosure.* Each applicant for an operating license for a production or utilization facility, who prepares a physical security plan, a safeguards contingency plan, a training and qualification plan, or a cyber security plan, shall protect the plans and other related Safeguards Information against unauthorized disclosure in accordance with the requirements of § 73.21 of this chapter.

(f) *Additional TMI-related requirements.* In addition to the requirements of paragraph (a) of this section, each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982, shall meet the requirements in paragraphs (f)(1) through (3) of this section. This regulation applies to the pending applications by Duke Power Company (Perkins Nuclear Station, Units 1, 2, and 3), Houston Lighting & Power Company (Allens Creek Nuclear Generating Station, Unit 1), Portland General Electric Company (Pebble Springs Nuclear Plant, Units 1 and 2), Public Service Company of Oklahoma (Black Fox Station, Units 1 and 2), Puget Sound Power & Light Company (Skagit/Hanford Nuclear Power Project, Units 1 and 2), and Offshore Power Systems (License to Manufacture Floating Nuclear Plants). The number of units that will be specified in the manufacturing license above, if issued, will be that number whose start of manufacture, as defined in the license application, can practically begin within a 10-year period commencing on the date of issuance of the manufacturing license, but in no event will that number be in excess of ten. The manufacturing license will require the plant design to be updated no later than 5 years after its approval. Paragraphs (f)(1)(xii), (2)(ix), and (3)(v) of this section, pertaining to hydrogen control measures, must be met by all applicants covered by this regulation. However, the Commission may decide to impose additional requirements and the issue of whether compliance with these provisions, together with 10 CFR 50.44 and criterion 50 of appendix A to 10 CFR part 50, is sufficient for issuance of that manufacturing license which may be considered in the manufacturing license proceeding. In addition, each applicant for a design certification, design approval, combined license, or manufacturing license under part 52 of this chapter shall demonstrate compliance with the technically relevant portions of the requirements in paragraphs (f)(1) through (3) of this section, except for paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v).

(1) To satisfy the following requirements, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of these studies are factored into the final design of the facility. For licensees identified in the introduction to paragraph (f) of this section, all studies must be completed no later than 2 years following the issuance of the construction permit or manufacturing license.[10] For all other applicants, the studies must be submitted as part of the final safety analysis report.

(i) Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. (II.B.8)

(ii) Perform an evaluation of the proposed auxiliary feedwater system (AFWS), to include (applicable to PWR's only) (II.E.1.1):

(A) A simplified AFWS reliability analysis using event-tree and fault-tree logic techniques.

(B) A design review of AFWS.

[10] Alphanumeric designations correspond to the related action plan items in NUREG 0718 and NUREG–0660, ''NRC Action Plan Developed as a Result of the TMI–2 Accident.'' They are provided herein for information only.

(C) An evaluation of AFWS flow design bases and criteria.

(iii) Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss-of-coolant accident with subsequent reactor coolant pump seal damage. (II.K.2.16 and II.K.3.25)

(iv) Perform an analysis of the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV). If this probability is a significant contributor to the probability of small-break LOCA's from all causes, provide a description and evaluation of the effect on small-break LOCA probability of an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened. (Applicable to PWR's only). (II.K.3.2)

(v) Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, ''high pressure core spray'' for ''high pressure coolant injection'' and ''HPCS'' for ''HPCI'') (Applicable to BWR's only). (II.K.3.13)

(vi) Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. (Applicable to BWR's only). (II.K.3.16)

(vii) Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system (ADS) design modifications that would eliminate the need for manual activation to ensure adequate core cooling. (Applicable to BWR's only). (II.K.3.18)

(viii) Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present. (Applicable to BWR's only). (II.K.3.21)

(ix) Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems, following a complete loss of offsite power to the plant for at least two (2) hours. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, ''high pressure core spray'' for ''high pressure coolant injection'' and ''HPCS'' for ''HPCI'') (Applicable to BWR's only). (II.K.3.24)

(x) Perform a study to ensure that the Automatic Depressurization System, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves. (Applicable to BWR's only). (II.K.3.28)

(xi) Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. (Applicable to BWR's only) (II.K.3.45)

(xii) Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f)(2)(ix) of this section. As a minimum include consideration of a hydrogen ignition and post-accident inerting system. The evaluation shall include:

(A) A comparison of costs and benefits of the alternative systems considered.

(B) For the selected system, analyses and test data to verify compliance with the requirements of (f)(2)(ix) of this section.

(C) For the selected system, preliminary design descriptions of equipment, function, and layout.

(2) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues.

(i) Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCA's. (Applicable to construction permit applicants only) (I.A.4.2.)

(ii) Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts. (Applicable to construction permit applicants only) (I.C.9)

(iii) Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts. (I.D.1)

(iv) Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. (I.D.2)

(v) Provide for automatic indication of the bypassed and operable status of safety systems. (I.D.3)

(vi) Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. (II.B.1)

(vii) Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term[11] radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. (II.B.2)

(viii) Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term[11] radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, radioiodines and cesiums, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. (II.B.3)

(ix) Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (f)(1)(xii) of this section is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that: (II.B.8)

(A) Uniformly distributed hydrogen concentrations in the containment do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

(B) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment

[11] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

integrity or loss of appropriate mitigating features.

(C) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction including the environmental conditions created by activation of the hydrogen control system.

(D) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.

(x) Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWR's, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transients without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. (II.D.1)

(xi) Provide direct indication of relief and safety valve position (open or closed) in the control room. (II.D.3)

(xii) Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room. (Applicable to PWR's only) (II.E.1.2)

(xiii) Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. (Applicable to PWR's only) (II.E.3.1)

(xiv) Provide containment isolation systems that: (II.E.4.2)

(A) Ensure all non-essential systems are isolated automatically by the containment isolation system,

(B) For each non-essential penetration (except instrument lines) have two isolation barriers in series,

(C) Do not result in reopening of the containment isolation valves on resetting of the isolation signal,

(D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation,

(E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

(xv) Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. (II.E.4.4)

(xvi) Establish a design criterion for the allowable number of actuation cycles of the emergency core cooling system and reactor protection system consistent with the expected occurrence rates of severe overcooling events (considering both anticipated transients and accidents). (Applicable to B&W designs only). (II.E.5.1)

(xvii) Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples. (II.F.1)

(xviii) Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWR's, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWR's and BWR's. (II.F.2)

(xix) Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage. (II.F.3)

(xx) Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) Level indicators are powered from vital buses;

(B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety and (C) electric power is provided from emergency power sources. (Applicable to PWR's only). (II.G.1)

(xxi) Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. (Applicable to BWR's only). (II.K.1.22)

(xxii) Perform a failure modes and effects analysis of the integrated control system (ICS) to include consideration of failures and effects of input and output signals to the ICS. (Applicable to B&W-designed plants only). (II.K.2.9)

(xxiii) Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip. (Applicable to B&W-designed plants only). (II.K.2.10)

(xxiv) Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. (Applicable to BWR's only). (II.K.3.23)

(xxv) Provide an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a nearsite Emergency Operations Facility. (III.A.1.2).

(xxvi) Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term [11] radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. (III.D.1.1)

(xxvii) Provide for monitoring of inplant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions. (III.D.3.3)

(xxviii) Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source term [11] release, and make necessary design provisions to preclude such problems. (III.D.3.4)

(3) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy paragraph (a)(1) of this section or to address the applicant's technical qualifications and management structure and competence.

(i) Provide administrative procedures for evaluating operating, design and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant. (I.C.5)

(ii) Ensure that the quality assurance (QA) list required by Criterion II, app. B, 10 CFR part 50 includes all structures, systems, and components important to safety. (I.F.1)

(iii) Establish a quality assurance (QA) program based on consideration of: (A) Ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (C) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (D) establishing criteria for determining QA programmatic requirements; (E) establishing qualification requirements for QA and QC personnel; (F) sizing the QA staff commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of ''as-built'' documentation; and (H) providing a QA role in design and analysis activities. (I.F.2)

(iv) Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system. (II.B.8)

(v) Provide preliminary design information at a level of detail consistent with that normally required at the construction permit stage of review sufficient to demonstrate that: (II.B.8)

(A)(*1*) Containment integrity will be maintained (*i.e.*, for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, subarticle NE–3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler Pressure Vessel Code, Section III, Division 2 subarticle CC–3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

(*2*) Subarticle NE–3220, Division 1, and subarticle CC–3720, Division 2, of section III of the July 1, 1980 ASME Boiler and Pressure Vessel Code, which are referenced in paragraphs (f)(3)(v)(A)(*1*) and (f)(3)(v)(B)(*1*) of this section, were approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the FEDERAL REGISTER. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. It is also available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852–2738.

(B)(*1*) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, subarticle NE–3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, subarticle CC–3720, Service Load Category, (*2*) The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting.

(vi) For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere. (II.E.4.1)

(vii) Provide a description of the management plan for design and construction activities, to include: (A) The organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the transition to operation; (E) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort. (II.J.3.1)

(g) *Combustible gas control.* All applicants for a reactor construction permit or operating license whose application is submitted after October 16, 2003, shall include the analyses, and the descriptions of the equipment and systems required by § 50.44 as a part of their application.

(h) *Conformance with the Standard Review Plan (SRP).* (1)(i) Applications for light water cooled nuclear power plant operating licenses docketed after May 17, 1982 shall include an evaluation of the facility against the Standard Review Plan (SRP) in effect on May 17, 1982 or the SRP revision in effect six months prior to the docket date of the application, whichever is later.

(ii) Applications for light-water-cooled nuclear power plant construction permits docketed after May 17, 1982, shall include an evaluation of the facility against the SRP in effect on May 17, 1982, or the SRP revision in effect six months before the docket date of the application, whichever is later.

(2) The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for a facility and those corresponding features, techniques, and measures given in the SRP acceptance criteria. Where such a difference exists, the evaluation shall discuss how the alternative proposed provides an acceptable method of complying with those rules or regulations of Commission, or portions thereof, that underlie the corresponding SRP acceptance criteria.

(3) The SRP was issued to establish criteria that the NRC staff intends to use in evaluating whether an applicant/licensee meets the Commission's regulations. The SRP is not a substitute for the regulations, and compliance is not a requirement. Applicants shall identify differences from the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the Commission's regulations.

(i) *Mitigation of beyond-design-basis events.* Each application for a power reactor operating license under this part must include the applicant's plans for implementing the requirements of § 50.155, including a schedule for achieving full compliance with these requirements and a description of the equipment upon which the strategies and guidelines required by § 50.155(b)(1) rely, including the planned locations of the equipment and how the equipment meets the requirements of § 50.155(c).

[33 FR 18612, Dec. 17, 1968]

EDITORIAL NOTE: For FEDERAL REGISTER citations affecting § 50.34, see the List of CFR Sections Affected, which appears in the Finding Aids section of the printed volume and at *www.govinfo.gov*.

§ 50.34a Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors.

(a) An application for a construction permit shall include a description of the preliminary design of equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, including expected operational occurrences. In the case of an application filed on or after January 2, 1971, the application shall also identify the design objectives, and the means to be employed, for keeping levels of radioactive material in effluents to unrestricted areas as low as is reasonably achievable. The term "as low as is reasonably achievable" as used in this part means as low as is reasonably achievable taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconomic considerations, and in relation to the use of atomic energy in the public interest. The guides set out in appendix I to this part provide numerical guidance on design objectives for light-water-cooled nuclear power reactors to meet the requirements that radioactive material in effluents released to unrestricted areas be kept as low as is reasonably achievable. These numerical guides for design objectives and limiting conditions for operation are not to be construed as radiation protection standards.

(b) Each application for a construction permit shall include:

(1) A description of the preliminary design of equipment to be installed under paragraph (a) of this section;

(2) An estimate of:

(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

(ii) The quantity 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.

(3) A general description of the provisions for packaging, storage, and shipment offsite of solid waste containing radioactive materials resulting from treatment of gaseous and liquid effluents and from other sources.

(c) Each application for an operating license shall include:

(1) A description of the equipment and procedures for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems, under paragraph (a) of this section; and

(2) A revised estimate of the information required in paragraph (b)(2) of this section if the expected releases and exposures differ significantly from the estimates submitted in the application for a construction permit.

(d) Each application for a combined license under part 52 of this chapter shall include:

(1) A description of the equipment and procedures for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems, under paragraph (a) of this section; and

(2) The information required in paragraph (b)(2) of this section.

(e) Each application for a design approval, a design certification, or a manufacturing license under part 52 of this chapter shall include:

(1) A description of the equipment for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems, under paragraph (a) of this section; and

(2) The information required in paragraph (b)(2) of this section.

[72 FR 49492, Aug. 28, 2007]

§ 50.35 Issuance of construction permits.[1]

(a) When an applicant has not supplied initially all of the technical information required to complete the application and support the issuance of a construction permit which approves all proposed design features, the Commission may issue a construction permit if the Commission finds that (1) the applicant has described the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design, 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 final safety analysis report; (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 that (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 in part 100 of this chapter, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

NOTE: When an applicant has supplied initially all of the technical information required to complete the application, including the final design of the facility, the findings required above will be appropriately modified to reflect that fact.

[1] The Commission may issue a provisional construction permit pursuant to the regulations in this part in effect on March 30, 1970, for any facility for which a notice of hearing on an application for a provisional construction permit has been published on or before that date.

(b) A construction permit will constitute an authorization to the applicant to proceed with construction but will 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 his option, may request such approvals in the construction permit or, from time to time, by amendment of his construction permit. 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.

(c) Any construction permit will be subject to the limitation that a license authorizing operation of the facility will not be issued by the Commission until (1) the applicant has submitted to the Commission, by amendment to the application, the complete final safety analysis report, portions of which may be submitted and evaluated from time to time, and (2) the Commission has found that the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the facility in accordance with the requirements of the license and the regulations in this chapter.

[27 FR 12915, Dec. 29, 1962, as amended at 31 FR 12780, Sept. 30, 1966; 35 FR 5318, Mar. 31, 1970; 35 FR 6644, Apr. 25, 1970; 35 FR 11461, July 7, 1970]

§ 50.36 Technical specifications.

(a)(1) Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section. A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications.

(a)(2) Each applicant for a design certification or manufacturing license under part 52 of this chapter shall include in its application proposed generic technical specifications in accordance with the requirements of this section for the portion of the plant that is within the scope of the design certification or manufacturing license application.

(b) Each license authorizing operation of a production or utilization facility of a type described in § 50.21 or § 50.22 will include technical specifications. The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to § 50.34. The Commission may include such additional technical specifications as the Commission finds appropriate.

(c) Technical specifications will include items in the following categories:

(1) *Safety limits, limiting safety system settings, and limiting control settings.* (i)(A) Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. Operation must not be resumed until authorized by the Commission. The licensee shall retain the record of the results of each review until the Commission terminates the license for the reactor, except for nuclear power reactors licensed under § 50.21(b) or § 50.22 of this part. For these reactors, the licensee shall notify the Commission as required by § 50.72 and submit a Licensee Event Report to the Commission as required by § 50.73. Licensees in these cases shall retain the records of the review for a period of three years following issuance of a Licensee Event Report.

(B) Safety limits for fuel reprocessing plants are those bounds within which the process variables must be maintained for adequate control of the operation and that must not be exceeded in order to protect the integrity of the physical system that is designed to guard against the uncontrolled release or radioactivity. If any safety limit for a fuel reprocessing plant is exceeded, corrective action must be taken as stated in the technical specification or the affected part of the process, or the entire process if required, must be shut down, unless this action would further reduce the margin of safety. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. If a portion of the process or the entire process has been shutdown, operation must not be resumed until authorized by the Commission. The licensee shall retain the record of the results of each review until the Commission terminates the license for the plant.

(ii)(A) Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. The licensee shall retain the record of the results of each review until the Commission terminates the license for the reactor except for nuclear power reactors licensed under § 50.21(b) or § 50.22 of this part. For these reactors, the licensee shall notify the Commission as required by § 50.72 and submit a Licensee Event Report to the Commission as required by § 50.73. Licensees in these cases shall retain the records of the review for a period of three years following issuance of a Licensee Event Report.

(B) Limiting control settings for fuel reprocessing plants are settings for automatic alarm or protective devices related to those variables having significant safety functions. Where a limiting control setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that protective action, either automatic or manual, will correct the abnormal situation before a safety limit is exceeded. If, during operation, the automatic alarm or protective devices do not function as required, the licensee shall take appropriate action to maintain the variables within the limiting control-setting values and to repair promptly the automatic devices or to shut down the affected part of the process and, if required, to shut down the entire process for repair of automatic devices. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. The licensee shall retain the record of the results of each review until the Commission terminates the license for the plant.

(2) *Limiting conditions for operation.* (i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met. When a limiting condition for operation of any process step in the system of a fuel reprocessing plant is not met, the licensee shall shut down that part of the operation or follow any remedial action permitted by the technical specifications until the condition can be met. In the case of a nuclear reactor not licensed under § 50.21(b) or § 50.22 of this part or fuel reprocessing plant, the licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. The licensee shall retain the record of the results of each review until the Commission terminates the license for the nuclear reactor or the fuel reprocessing plant. In the case of nuclear power reactors licensed under § 50.21(b) or § 50.22,

the licensee shall notify the Commission if required by § 50.72 and shall submit a Licensee Event Report to the Commission as required by § 50.73. In this case, licensees shall retain records associated with preparation of a Licensee Event Report for a period of three years following issuance of the report. For events which do not require a Licensee Event Report, the licensee shall retain each record as required by the technical specifications.

(ii) A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

(A) *Criterion 1.* Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

(B) *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(C) *Criterion 3.* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(D) *Criterion 4.* A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

(iii) A licensee is not required to propose to modify technical specifications that are included in any license issued before August 18, 1995, to satisfy the criteria in paragraph (c)(2)(ii) of this section.

(3) *Surveillance requirements.* Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

(4) *Design features.* Design features to be included are those features of the facility 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 (c) (1), (2), and (3) of this section.

(5) *Administrative controls.* Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in § 50.4.

(6) *Decommissioning.* This paragraph applies only to nuclear power reactor facilities that have submitted the certifications required by § 50.82(a)(1) and to non-power reactor facilities which are not authorized to operate. Technical specifications involving safety limits, limiting safety system settings, and limiting control system settings; limiting conditions for operation; surveillance requirements; design features; and administrative controls will be developed on a case-by-case basis.

(7) *Initial notification.* Reports made to the Commission by licensees in response to the requirements of this section must be made as follows:

(i) Licensees that have an installed Emergency Notification System shall make the initial notification to the NRC Operations Center in accordance with § 50.72 of this part.

(ii) All other licensees shall make the initial notification by telephone to the Administrator of the appropriate NRC Regional Office listed in appendix D, part 20, of this chapter.

(8) *Written Reports.* Licensees for nuclear power reactors licensed under § 50.21(b) and § 50.22 of this part shall submit written reports to the Commission in accordance with § 50.73 of this part for events described in paragraphs (c)(1) and (c)(2) of this section. For all licensees, the Commission may require Special Reports as appropriate.

(d)(1) This section shall not be deemed to modify the technical specifications included in any license issued prior to January 16, 1969. A license in which technical specifications have not been designated shall be deemed to include the entire safety analysis report as technical specifications.

(2) An applicant for a license authorizing operation of a production or utilization facility to whom a construction permit has been issued prior to January 16, 1969, may submit technical specifications in accordance with this section, or in accordance with the requirements of this part in effect prior to January 16, 1969.

(3) At the initiative of the Commission or the licensee, any license may be amended to include technical specifications of the scope and content which would be required if a new license were being issued.

(e) The provisions of this section apply to each nuclear reactor licensee whose authority to operate the reactor has been removed by license amendment, order, or regulation.

[33 FR 18612, Dec. 17, 1968, as amended at 48 FR 33860, July 26, 1983; 51 FR 40308, Nov. 6, 1986; 53 FR 19249, May 27, 1988; 60 FR 36959, July 19, 1995; 61 FR 39299, July 29, 1996; 72 FR 49493, Aug. 28, 2007; 73 FR 54932, Sept. 24, 2008; 84 FR 63568, Nov. 18, 2019]

§ 50.36a Technical specifications on effluents from nuclear power reactors.

(a) To keep releases of radioactive materials to unrestricted areas during normal conditions, including expected occurrences, as low as is reasonably achievable, each licensee of a nuclear power reactor and each applicant for a design certification or a manufacturing license will include technical specifications that, in addition to requiring compliance with applicable provisions of § 20.1301 of this chapter, require that:

(1) Operating procedures developed pursuant to § 50.34a(c) for the control of effluents be established and followed and that the radioactive waste system, pursuant to § 50.34a, be maintained and used. The licensee shall retain the operating procedures in effect as a record until the Commission terminates the license and shall retain each superseded revision of the procedures for 3 years from the date it was superseded.

(2) Each holder of an operating license, and each holder of a combined license after the Commission has made the finding under § 52.103(g) of this chapter, 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 § 50.4, and the time between submission of the reports must be no longer than 12 months. If quantities of radioactive materials released during the reporting period are significantly above design objectives, 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.

(b) In establishing and implementing the operating procedures described in paragraph (a) of this section, the licensee shall be guided by the following considerations: Experience with the design, construction, and operation of nuclear power reactors indicates that compliance with the technical specifications described in this section will keep average annual releases of radioactive material in effluents and their resultant committed effective dose equivalents at small percentages of the dose limits specified in § 20.1301 and in the license. At the same time, the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power even under unusual conditions which may temporarily result in releases higher than such small percentages, but still within the limits specified in § 20.1301 of this chapter and in the license. It is expected that in using this flexibility under unusual conditions, the licensee will exert its best efforts to keep levels of radioactive material in effluents as low as is reasonably achievable. The guides set out in appendix I, provide numerical guidance on limiting conditions for operation for light-water cooled nuclear power reactors to meet the requirement that radioactive materials in effluents released to unrestricted areas be kept as low as is reasonably achievable.

[61 FR 39299, July 29, 1996, as amended at 72 FR 49493, Aug. 28, 2007]

§ 50.36b Environmental conditions.

(a) Each construction permit under this part, each early site permit under part 52 of this chapter, and each combined license under part 52 of this chapter may include conditions to protect the environment during construction. These conditions are to be set out in an attachment to the permit or license, which is incorporated in and made a part of the permit or 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 production or utilization facility, including a combined license under part 52 of this chapter, and each license for a nuclear power reactor facility for which the certification of permanent cessation of operations required under § 50.82(a)(1) or § 52.110(a) of this chapter has been submitted, which is of a type described in § 50.21(b)(2) or (3) or § 50.22 or is a testing facility, may include conditions to protect the environment 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 pursuant to §§ 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.

[72 FR 49493, Aug. 28, 2007]

§ 50.37 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, construction permit, early site permit, or standard design approval, or before the Commission has adopted a final standard design certification rule under part 52 of this chapter, the applicant shall agree in writing that it will not permit any individual to have access to any facility to possess Restricted Data or classified National Security Information until the individual and/or facility has been approved for access under the provisions of 10 CFR parts 25 and/or 95. The agreement of the applicant becomes part of the license, or construction permit, or standard design approval.

[72 FR 49493, Aug. 28, 2007]

§ 50.38 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.

[21 FR 355, Jan. 16, 1956, as amended at 43 FR 6924, Feb. 17, 1978]

§ 50.39 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 part 2 of this chapter.

STANDARDS FOR LICENSES, CERTIFICATIONS, AND REGULATORY APPROVALS

§ 50.40 Common standards.

In determining that a construction permit or operating license in this part, or early site permit, combined license, or manufacturing license in part 52 of this chapter will be issued to an applicant, the Commission will be guided by the following considerations:

(a) 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 part 20 of this chapter, and that the health and safety of the public will not be endangered.

(b) 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 §50.21(b) or §50.22 or for an applicant for a manufacturing license.

(c) 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.

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

[72 FR 49493, Aug. 28, 2007]

§ 50.41 Additional standards for class 104 licenses.

In determining that a class 104 license will be issued to an applicant, the Commission will, in addition to applying the standards set forth in §50.40 be guided by the following considerations:

(a) The Commission will permit the widest amount of effective medical therapy possible with the amount of special nuclear material available for such purposes.

(b) The Commission will permit the conduct of widespread and diverse research and development.

(c) [Reserved]

[21 FR 355, Jan. 19, 1956, as amended at 35 FR 19660, Dec. 29, 1970; 73 FR 44620, July 31, 2008]

§ 50.42 Additional standard for class 103 licenses.

In determining whether a class 103 license will be issued to an applicant, the Commission will, in addition to applying the standards set forth in §50.40, consider whether the proposed activities will serve a useful purpose proportionate to the quantities of special nuclear material or source material to be utilized.

[73 FR 44620, July 31, 2008]

§ 50.43 Additional standards and provisions affecting class 103 licenses and certifications for commercial power.

In addition to applying the standards set forth in §§50.40 and 50.42, paragraphs (a) through (e) of this section apply in the case of a class 103 license for a facility for the generation of commercial power. For a design certification under part 52 of this chapter, only paragraph (e) of this section applies.

(a) The NRC will:

(1) 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;

(2) 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

(3) Publish notice of the application once each week for 4 consecutive weeks in the FEDERAL REGISTER. No license will be issued by the NRC prior to the giving of these notices and until 4 weeks after the last notice is published in the FEDERAL REGISTER.

(b) 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 other than high cost power areas; and, fourth, to all other applicants.

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

(d) 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 or operating license under this part, or a combined license under part 52 of this chapter for a utilization facility for the primary purpose of producing electric energy for disposition for ultimate public consumption.

(e) 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:

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

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

(iii) 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

(2) 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.

[21 FR 355, Jan. 19, 1956, as amended at 35 FR 19660, Dec. 29, 1970; 63 FR 50480, Sept. 22, 1998; 72 FR 49494, Aug. 28, 2007; 82 FR 52825, Nov. 15, 2017]

§ 50.44 Combustible gas control for nuclear power reactors.

(a) *Definitions*—(1) *Inerted atmosphere* means a containment atmosphere with less than 4 percent oxygen by volume.

(2) *Mixed atmosphere* means that the concentration of combustible gases in any part of the containment is below a level that supports combustion or detonation that could cause loss of containment integrity.

(b) *Requirements for currently-licensed reactors*. Each boiling or pressurized water nuclear power reactor with an operating license on October 16, 2003, except for those facilities for which the certifications required under § 50.82(a)(1) have been submitted, must comply with the following requirements, as applicable:

(1) *Mixed atmosphere*. All containments must have a capability for ensuring a mixed atmosphere.

(2) *Combustible gas control*. (i) All boiling water reactors with Mark I or Mark II type containments must have an inerted atmosphere.

(ii) All boiling water reactors with Mark III type containments and all pressurized water reactors with ice condenser containments must have the capability for controlling combustible gas generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume) so that there is no loss of containment structural integrity.

(3) *Equipment survivability*. All boiling water reactors with Mark III containments and all pressurized water reactors with ice condenser containments that do not rely upon an inerted atmosphere inside containment to control combustible gases must be able to establish and maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. Environmental conditions caused by local detonations of hydrogen must also be included, unless such detonations can be shown unlikely to occur. The amount of hydrogen to be considered must be equivalent to that generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume).

(4) *Monitoring*. (i) Equipment must be provided for monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. Equipment for monitoring oxygen must be functional, reliable, and capable of continuously measuring the concentration of oxygen in the containment atmosphere following a significant beyond design-basis accident for combustible gas control and accident management, including emergency planning.

(ii) Equipment must be provided for monitoring hydrogen in the containment. Equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design-basis accident for accident management, including emergency planning.

(5) *Analyses*. Each holder of an operating license for a boiling water reactor with a Mark III type of containment or for a pressurized water reactor with an ice condenser type of containment, shall perform an analysis that:

(i) Provides an evaluation of the consequences of large amounts of hydrogen generated after the start of an accident (hydrogen resulting from the metal-water reaction of up to and including 75 percent of the fuel cladding surrounding the active fuel region, excluding the cladding surrounding the plenum volume) and include consideration of hydrogen control measures as appropriate;

(ii) Includes the period of recovery from the degraded condition;

(iii) Uses accident scenarios that are accepted by the NRC staff. These scenarios must be accompanied by sufficient supporting justification to show that they describe the behavior of the reactor system during and following an accident resulting in a degraded core.

(iv) Supports the design of the hydrogen control system selected to meet the requirements of this section; and,

(v) Demonstrates, for those reactors that do not rely upon an inerted atmosphere to comply with paragraph (b)(2)(ii) of this section, that:

(A) Containment structural integrity is maintained. Containment structural integrity must be demonstrated by use of an analytical technique that is accepted by the NRC staff in accordance with § 50.90. This demonstration must include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. This method could include the use of actual material properties with suitable margins to account for uncertainties in modeling, in material properties, in construction tolerances, and so on; and

(B) Systems and components necessary to establish and maintain safe shutdown and to maintain containment integrity will be capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen, including local detonations, unless such detonations can be shown unlikely to occur.

(c) *Requirements for future water-cooled reactor applicants and licensees.*[2] The requirements in this paragraph

[2] The requirements of this paragraph apply only to water-cooled reactor designs with characteristics (e.g., type and quantity of cladding materials) such that the potential for production of combustible gases is comparable to light water reactor designs licensed as of October 16, 2003.

apply to all water-cooled reactor construction permits or operating licenses under this part, and to all water-cooled reactor design approvals, design certifications, combined licenses or manufacturing licenses under part 52 of this chapter, any of which are issued after October 16, 2003.

(1) *Mixed atmosphere.* All containments must have a capability for ensuring a mixed atmosphere during design-basis and significant beyond design-basis accidents.

(2) *Combustible gas control.* All containments must have an inerted atmosphere, or must limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume) and maintain containment structural integrity and appropriate accident mitigating features.

(3) *Equipment Survivability.* Containments that do not rely upon an inerted atmosphere to control combustible gases must be able to establish and maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. Environmental conditions caused by local detonations of hydrogen must also be included, unless such detonations can be shown unlikely to occur. The amount of hydrogen to be considered must be equivalent to that generated from a fuel clad-coolant reaction involving 100 percent of the fuel cladding surrounding the active fuel region.

(4) *Monitoring.* (i) Equipment must be provided for monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. Equipment for monitoring oxygen must be functional, reliable, and capable of continuously measuring the concentration of oxygen in the containment atmosphere following a significant beyond design-basis accident for combustible gas control and accident management, including emergency planning.

(ii) Equipment must be provided for monitoring hydrogen in the containment. Equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design-basis accident for accident management, including emergency planning.

(5) *Structural analysis.* An applicant must perform an analysis that demonstrates containment structural integrity. This demonstration must use an analytical technique that is accepted by the NRC and include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The analysis must address an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by hydrogen burning. Systems necessary to ensure containment integrity must also be demonstrated to perform their function under these conditions.

(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 to all design approvals, design certifications, combined licenses, or manufacturing licenses under part 52 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:

(1) Information addressing whether accidents involving combustible gases are technically relevant for their design, and

(2) If accidents involving combustible gases are found to be technically relevant, information (including a design-specific probabilistic risk assessment) demonstrating that the safety impacts of combustible gases during design-basis and significant beyond design-basis accidents have been addressed to ensure adequate protection of public

health and safety and common defense and security.

[68 FR 54141, Sept. 16, 2003]

§ 50.45 Standards for construction permits, operating licenses, and combined licenses.

(a) An applicant for an operating license or an amendment of an operating license who proposes to construct or alter a production or utilization facility will be initially granted a construction permit if the application is in conformity with and acceptable under the criteria of §§ 50.31 through 50.38, and the standards of §§ 50.40 through 50.43, as applicable.

(b) A holder of a combined license who proposes, after the Commission makes the finding under § 52.103(g) of this chapter, to alter the licensed facility will be initially granted a construction permit if the application is in conformity with and acceptable under the criteria of §§ 50.30 through 50.33, § 50.34(f), §§ 50.34a through 50.38, the standards of §§ 50.40 through 50.43, as applicable, and §§ 52.79 and 52.80 of this chapter.

[72 FR 49494, Aug. 28, 2007]

§ 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) have been submitted.

(ii) Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K ECCS Evaluation Models.

(2) The Director of Nuclear Reactor Regulation may impose restrictions on reactor operation if it is found that the evaluations of ECCS cooling performance submitted are not consistent with paragraphs (a)(1) (i) and (ii) of this section.

(3)(i) Each applicant for or holder of an operating license or construction permit issued under this part, applicant for a standard design certification under part 52 of this chapter (including an applicant after the Commission has adopted a final design certification regulation), or an applicant for or holder of a standard design approval, a combined license or a manufacturing license issued under part 52 of this chapter, 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 or § 52.3 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. 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 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.

(b)(1) *Peak cladding temperature.* The calculated maximum fuel element cladding temperature shall not exceed 2200 °F.

(2) *Maximum cladding oxidation.* The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening.

(3) *Maximum hydrogen generation.* The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) *Coolable geometry*. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) *Long-term cooling*. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

(c) As used in this section:

(1) Loss-of-coolant accidents (LOCA's) are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system.

(2) An evaluation model is the calculational framework for evaluating the behavior of the reactor system during a postulated loss-of-coolant accident (LOCA). It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.

(d) The requirements of this section are in addition to any other requirements applicable to ECCS set forth in this part. The criteria set forth in paragraph (b), with cooling performance calculated in accordance with an acceptable evaluation model, are in implementation of the general requirements with respect to ECCS cooling performance design set forth in this part, including in particular Criterion 35 of appendix A.

[39 FR 1002, Jan. 4, 1974, as amended at 53 FR 36004, Sept. 16, 1988; 57 FR 39358, Aug. 31, 1992; 61 FR 39299, July 29, 1996; 62 FR 59276, Nov. 3, 1997; 72 FR 49494, Aug. 28, 2007]

§ 50.46a Acceptance criteria for reactor coolant system venting systems.

Each nuclear power reactor must be provided with high point vents for the reactor coolant system, for the reactor vessel head, and for other systems required to maintain adequate core cooling if the accumulation of noncondensible gases would cause the loss of function of these systems. High point vents are not required for the tubes in U-tube steam generators. Acceptable venting systems must meet the following criteria:

(a) The high point vents must be remotely operated from the control room.

(b) The design of the vents and associated controls, instruments and power sources must conform to appendix A and appendix B of this part.

(c) The vent system must be designed to ensure that:

(1) The vents will perform their safety functions; and

(2) There would not be inadvertent or irreversible actuation of a vent.

[68 FR 54142, Sept. 16, 2003]

§ 50.47 Emergency plans.

(a)(1)(i) Except as provided in paragraph (d) of this section, no initial operating license for a nuclear power reactor 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 nuclear power reactor operating license.

(ii) No initial combined license under part 52 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.

(iii) If an application for an early site permit under subpart A of part 52 of this chapter includes complete and integrated emergency plans under 10 CFR 52.17(b)(2)(ii), 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.

(iv) If an application for an early site permit proposes major features of the emergency plans under 10 CFR 52.17(b)(2)(i), 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 10 CFR 50.47 and 10 CFR part 50, appendix E, within the scope of emergency preparedness matters addressed in the major features.

(2) The NRC will base its finding on a review of the Federal Emergency Management Agency (FEMA) findings and determinations as to whether State and local emergency plans are adequate and whether there is reasonable assurance that they can be implemented, and on the NRC assessment as to whether the applicant's onsite emergency plans are adequate and whether there is reasonable assurance that they can be implemented. A FEMA finding will primarily be based on a review of the plans. Any other information already available to FEMA may be considered in assessing whether there is reasonable assurance that the plans can be implemented. In any NRC licensing proceeding, a FEMA finding will constitute a rebuttable presumption on questions of adequacy and implementation capability.

(b) The onsite and, except as provided in paragraph (d) of this section, offsite emergency response plans for nuclear power reactors must meet the following standards:

(1) Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the Emergency Planning Zones have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.

(2) On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available and the interfaces among various onsite response activities and offsite support and response activities are specified.

(3) Arrangements for requesting and effectively using assistance resources have been made, arrangements to accommodate State and local staff at the licensee's Emergency Operations Facility have been made, and other organizations capable of augmenting the planned response have been identified.

(4) A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.

(5) Procedures have been established for notification, by the licensee, of State and local response organizations and for notification of emergency personnel by all organizations; the content of initial and followup messages to response organizations and the public has been established; and means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established.

(6) Provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.

(7) Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors), the principal points of contact with the news media for dissemination of information during an emergency (including the physical location or locations) are established in advance, and procedures for coordinated dissemination of information to the public are established.

(8) Adequate emergency facilities and equipment to support the emergency response are provided and maintained.

(9) Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

(10) A range of protective actions has been developed for the plume exposure pathway EPZ for emergency workers and the public. In developing this range of actions, consideration has been given to evacuation, sheltering, and, as a supplement to these, the prophylactic use of potassium iodide (KI), as appropriate. Evacuation time estimates have been developed by applicants and licensees. Licensees shall update the evacuation time estimates on a periodic basis. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.

(11) Means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.

(12) Arrangements are made for medical services for contaminated injured individuals.

(13) General plans for recovery and reentry are developed.

(14) Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.

(15) Radiological emergency response training is provided to those who may be called on to assist in an emergency.

(16) Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.

(c)(1) Failure to meet the applicable standards set forth in paragraph (b) of this section may result in the Commission declining to issue an operating license; however, the applicant will have an opportunity to demonstrate to the satisfaction of the Commission that deficiencies in the plans are not significant for the plant in question, that adequate interim compensating actions have been or will be taken promptly, or that there are other compelling reasons to permit plant operations. Where an applicant for an operating license asserts that its inability to demonstrate compliance with the requirements of paragraph (b) of this section results wholly or substantially from the decision of state and/or local governments not to participate further in emergency planning, an operating license may be issued if the applicant demonstrates to the Commission's satisfaction that:

(i) The applicant's inability to comply with the requirements of paragraph (b) of this section is wholly or substantially the result of the non-participation of state and/or local governments.

(ii) The applicant has made a sustained, good faith effort to secure and retain the participation of the pertinent state and/or local governmental authorities, including the furnishing of copies of its emergency plan.

(iii) The applicant's emergency plan provides reasonable assurance that public health and safety is not endangered by operation of the facility concerned. To make that finding, the applicant must demonstrate that, as outlined below, adequate protective measures can and will be taken in the event of an emergency. A utility plan will be evaluated against the same planning standards applicable to a state or local plan, as listed in paragraph (b) of this section, with due allowance made both for—

(A) Those elements for which state and/or local non-participation makes compliance infeasible and

(B) The utility's measures designed to compensate for any deficiencies resulting from state and/or local non-participation.

In making its determination on the adequacy of a utility plan, the NRC will recognize the reality that in an actual emergency, state and local government officials will exercise their best efforts to protect the health and safety of the public. The NRC will determine the adequacy of that expected response, in combination with the utility's compensating measures, on a case-by-case basis, subject to the following guidance. In addressing the circumstance where applicant's inability to comply with the requirements of paragraph (b) of this section is wholly or substantially the result of non-participation of state and/or local governments, it may be presumed that in the event of an actual radiological emergency state and local officials would generally follow the utility plan. However, this presumption may be rebutted by, for example, a good faith and timely proffer of an adequate and feasible state and/or local radiological emergency plan that would in fact be relied upon in a radiological emergency.

(2) Generally, the plume exposure pathway EPZ for nuclear power plants shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to 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 nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway.

(d) Notwithstanding the requirements of paragraphs (a) and (b) of this section, and except as specified by this paragraph, no NRC or FEMA review, findings, or determinations concerning the state of offsite emergency preparedness or the adequacy of and capability to implement State and local or utility offsite emergency plans are required prior to issuance of an operating license authorizing only fuel loading or low power testing and training (up to 5 percent of the rated thermal power). Insofar as emergency planning and preparedness requirements are concerned, a license authorizing fuel loading and/or low power testing and training may be issued after a finding is made by the NRC that the state of onsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. The NRC will base this finding on its assessment of the applicant's onsite emergency plans against the pertinent standards in paragraph (b) of this section and appendix E. Review of applicant's emergency plans will include the following standards with offsite aspects:

(1) Arrangements for requesting and effectively using offsite assistance on site have been made, arrangements to accommodate State and local staff at the licensee's Emergency Operations Facility have been made, and other organizations capable of augmenting the planned onsite response have been identified.

(2) Procedures have been established for licensee communications with State and local response organizations, including initial notification of the declaration of emergency and periodic provision of plant and response status reports.

(3) Provisions exist for prompt communications among principal response organizations to offsite emergency personnel who would be responding onsite.

(4) Adequate emergency facilities and equipment to support the emergency response onsite are provided and maintained.

(5) Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use onsite.

(6) Arrangements are made for medical services for contaminated and injured onsite individuals.

(7) Radiological emergency response training has been made available to those offsite who may be called to assist in an emergency onsite.

(e) Notwithstanding the requirements of paragraph (b) of this section and the provisions of § 52.103 of this chapter, a holder of a combined license under part 52 of this chapter may not load fuel or operate except as provided in accordance with appendix E to part 50 and § 50.54(gg).

[45 FR 55409, Aug. 8, 1980, as amended at 47 FR 30235, July 13, 1982; 47 FR 40537, Sept. 15, 1982; 49 FR 27736, July 6, 1984; 50 FR 19324, May 8, 1985; 52 FR 42085, Nov. 3, 1987; 53 FR 36959, Sept. 23, 1988; 56 FR 64976, Dec. 13, 1991; 61 FR 30132, June 14, 1996; 66 FR 5440, Jan. 19, 2001; 72 FR 49495, Aug. 28, 2007; 76 FR 72595, Nov. 23, 2011; 78 FR 34248, June 7, 2013]

§ 50.48 Fire protection.

(a)(1) Each holder of an operating license issued under this part or a combined license issued under part 52 of this chapter must have a fire protection plan that satisfies Criterion 3 of appendix A to this part. This fire protection plan must:

(i) Describe the overall fire protection program for the facility;

(ii) Identify the various positions within the licensee's organization that are responsible for the program;

(iii) State the authorities that are delegated to each of these positions to implement those responsibilities; and

(iv) Outline the plans for fire protection, fire detection and suppression capability, and limitation of fire damage.

(2) The plan must also describe specific features necessary to implement the program described in paragraph (a)(1) of this section such as—

(i) Administrative controls and personnel requirements for fire prevention and manual fire suppression activities;

(ii) Automatic and manually operated fire detection and suppression systems; and

(iii) The means to limit fire damage to structures, systems, or components important to safety so that the capability to shut down the plant safely is ensured.

(3) The licensee shall retain the fire protection plan and each change to the plan as a record until the Commission terminates the reactor license. The licensee shall retain each superseded revision of the procedures for 3 years from the date it was superseded.

(4) Each applicant for a design approval, design certification, or manufacturing license under part 52 of this chapter must have a description and analysis of the fire protection design features for the standard plant necessary to demonstrate compliance with Criterion 3 of appendix A to this part.

(b) Appendix R to this part establishes fire protection features required to satisfy Criterion 3 of appendix A to this part with respect to certain generic issues for nuclear power plants licensed to operate before January 1, 1979.

(1) Except for the requirements of Sections III.G, III.J, and III.O, the provisions of Appendix R to this part do not apply to nuclear power plants licensed to operate before January 1, 1979, to the extent that—

(i) Fire protection features proposed or implemented by the licensee have been accepted by the NRC staff as satisfying the provisions of Appendix A to Branch Technical Position (BTP) APCSB 9.5–1 reflected in NRC fire protection safety evaluation reports issued before the effective date of February 19, 1981; or

(ii) Fire protection features were accepted by the NRC staff in comprehensive fire protection safety evaluation reports issued before Appendix A to Branch Technical Position (BTP) APCSB 9.5–1 was published in August 1976.

(2) With respect to all other fire protection features covered by Appendix R, all nuclear power plants licensed to operate before January 1, 1979, must satisfy the applicable requirements of Appendix R to this part, including specifically the requirements of Sections III.G, III.J, and III.O.

(c) *National Fire Protection Association Standard NFPA 805*—(1) *Approval of incorporation by reference.* National Fire Protection Association (NFPA) Standard 805, ''Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition'' (NFPA 805), which is referenced in this section, was approved for incorporation by reference by the Director of the Federal Register pursuant to 5 U.S.C. 552(a) and 1 CFR part 51. Copies of NFPA 805 may be purchased from the NFPA Customer Service Department, 1 Batterymarch Park, P.O. Box 9101, Quincy, MA 02269–9101 and in PDF format through the NFPA Online Catalog (*www.nfpa.org*) or by calling 1–800–344–3555 or (617) 770–3000. Copies are also available for inspection at the NRC Library, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852–2738, and at the NRC Public Document Room, Building One White Flint North, Room O1-F15, 11555 Rockville Pike, Rockville, Maryland 20852–2738. Copies are also available at the National Archives and Records Administration (NARA). For information on the availability of this material at NARA, call (202) 741–6030, or go to: *http://www.archives.gov/federal_register/code_of_federal_regulations/ibr_locations.html.*

(2) *Exceptions, modifications, and supplementation of NFPA 805.* As used in this section, references to NFPA 805 are to the 2001 Edition, with the following exceptions, modifications, and supplementation:

(i) *Life Safety Goal, Objectives, and Criteria.* The Life Safety Goal, Objectives, and Criteria of Chapter 1 are not endorsed.

(ii) *Plant Damage/Business Interruption Goal, Objectives, and Criteria.* The Plant Damage/Business Interruption Goal, Objectives, and Criteria of Chapter 1 are not endorsed.

(iii) *Use of feed-and-bleed.* In demonstrating compliance with the performance criteria of Sections 1.5.1(b) and (c), a high-pressure charging/injection pump coupled with the pressurizer power-operated relief valves (PORVs) as the sole fire-protected safe shutdown path for maintaining reactor coolant inventory, pressure control, and decay heat removal capability (*i.e.*, feed-and-bleed) for pressurized-water reactors (PWRs) is not permitted.

(iv) *Uncertainty analysis.* An uncertainty analysis performed in accordance with

Section 2.7.3.5 is not required to support deterministic approach calculations.

(v) *Existing cables.* In lieu of installing cables meeting flame propagation tests as required by Section 3.3.5.3, a flame-retardant coating may be applied to the electric cables, or an automatic fixed fire suppression system may be installed to provide an equivalent level of protection. In addition, the italicized exception to Section 3.3.5.3 is not endorsed.

(vi) *Water supply and distribution.* The italicized exception to Section 3.6.4 is not endorsed. Licensees who wish to use the exception to Section 3.6.4 must submit a request for a license amendment in accordance with paragraph (c)(2)(vii) of this section.

(vii) *Performance-based methods.* Notwithstanding the prohibition in Section 3.1 against the use of performance-based methods, the fire protection program elements and minimum design requirements of Chapter 3 may be subject to the performance-based methods permitted elsewhere in the standard. Licensees who wish to use performance-based methods for these fire protection program elements and minimum design requirements shall submit a request in the form of an application for license amendment under §50.90. The Director of the Office of Nuclear Reactor Regulation, or a designee of the Director, may approve the application if the Director or designee determines that the performance-based approach;

(A) Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;

(B) Maintains safety margins; and

(C) Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

(3) *Compliance with NFPA 805.* (i) A licensee may maintain a fire protection program that complies with NFPA 805 as an alternative to complying with paragraph (b) of this section for plants licensed to operate before January 1, 1979, or the fire protection license conditions for plants licensed to operate after January 1, 1979. The licensee shall submit a request to comply with NFPA 805 in the form of an application for license amendment under §50.90. The application must identify any orders and license conditions that must be revised or superseded, and contain any necessary revisions to the plant's technical specifications and the bases thereof. The Director of the Office of Nuclear Reactor Regulation, or a designee of the Director, may approve the application if the Director or designee determines that the licensee has identified orders, license conditions, and the technical specifications that must be revised or superseded, and that any

necessary revisions are adequate. Any approval by the Director or the designee must be in the form of a license amendment approving the use of NFPA 805 together with any necessary revisions to the technical specifications.

(ii) The licensee shall complete its implementation of the methodology in Chapter 2 of NFPA 805 (including all required evaluations and analyses) and, upon completion, modify the fire protection plan required by paragraph (a) of this section to reflect the licensee's decision to comply with NFPA 805, before changing its fire protection program or nuclear power plant as permitted by NFPA 805.

(4) *Risk-informed or performance-based alternatives to compliance with NFPA 805.* A licensee may submit a request to use risk-informed or performance-based alternatives to compliance with NFPA 805. The request must be in the form of an application for license amendment under § 50.90 of this chapter. The Director of the Office of Nuclear Reactor Regulation, or designee of the Director, may approve the application if the Director or designee determines that the proposed alternatives:

(i) Satisfy the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;

(ii) Maintain safety margins; and

(iii) Maintain fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

(d)–(e) [Reserved]

(f) Licensees that have submitted the certifications required under § 50.82(a)(1) shall maintain a fire protection program to address the potential for fires that could cause the release or spread of radioactive materials (*i.e.*, that could result in a radiological hazard). A fire protection program that complies with NFPA 805 shall be deemed to be acceptable for complying with the requirements of this paragraph.

(1) The objectives of the 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 shall assess the fire protection program on a regular basis. The licensee shall revise the plan as appropriate throughout the various stages of facility decommissioning.

(3) The licensee may make changes to the fire protection program without NRC approval if these changes do not reduce the effectiveness of fire protection for facilities, systems, and equipment that could result in a radiological hazard, taking into account the decommissioning plant conditions and activities.

[65 FR 38190, June 20, 2000, as amended at 69 FR 33550, June 16, 2004; 72 FR 49495, Aug. 28, 2007]

§ 50.49 Environmental qualification of electric equipment important to safety for nuclear power plants.

(a) Each holder of or an applicant for an operating license issued under this part, or a combined license or manufacturing license issued under part 52 of this chapter, other than a nuclear power plant for which the certifications required under § 50.82(a)(1) or § 52.110(a)(1) of this chapter have been submitted, shall establish a program for qualifying the electric equipment defined in paragraph (b) of this section. For a manufacturing license, only electric equipment defined in paragraph (b) which is within the scope of the manufactured reactor must be included in the program.

(b) Electric equipment important to safety covered by this section is:

(1) Safety-related electric equipment.[3]

(i) This equipment is that relied upon to remain functional during and following design basis events to ensure—

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

[3] Safety-related electric equipment is referred to as "Class 1E" equipment in IEEE 323–1974. Copies of this standard may be obtained from the Institute of Electrical and Electronics Engineers, Inc., 345 East 47th Street, New York, NY 10017.

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

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

(ii) Design basis events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed to ensure functions (b)(1)(i) (A) through (C) of this section.

(2) Nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified in subparagraphs (b)(1)(i)(A) through (C) of this section by the safety-related equipment.

(3) Certain post-accident monitoring equipment.[4]

(c) Requirements for (1) dynamic and seismic qualification of electric equipment important to safety, (2) protection of electric equipment important to safety against other natural phenomena and external events, and (3) environmental qualification of electric equipment important to safety located in a mild environment are not included within the scope of this section. A mild environment is an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences.

(d) The applicant or licensee shall prepare a list of electric equipment important to safety covered by this section. In addition, the applicant or licensee shall include the information in paragraphs (d)(1), (2), and (3) of this section for this electric equipment important to safety in a qualification file. The applicant or licensee shall keep the list and information in the file current and retain the file in auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for future use to permit verification that each item of electric equipment is important to safely meet the requirements of paragraph (j) of this section.

(1) The performance specifications under conditions existing during and following design basis accidents.

(2) The voltage, frequency, load, and other electrical characteristics for which the performance specified in accordance with paragraph (d)(1) of this section can be ensured.

(3) The environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence at the location where the equipment must perform as specified in accordance with paragraphs (d) (1) and (2) of this section.

(e) The electric equipment qualification program must include and be based on the following:

(1) *Temperature and pressure.* The time-dependent temperature and pressure at the location of the electric equipment important to safety must be established for the most severe design basis accident during or following which this equipment is required to remain functional.

(2) *Humidity.* Humidity during design basis accidents must be considered.

(3) *Chemical effects.* The composition of chemicals used must be at least as severe as that resulting from the most limiting mode of plant operation (e.g., containment spray, emergency core cooling, or recirculation from containment sump). If the composition of the chemical spray can be affected by equipment malfunctions, the most severe chemical spray environment that results from a single failure in the spray system must be assumed.

(4) *Radiation.* The radiation environment must be based on the type of radiation, the total dose expected during normal operation over the installed life

[4] Specific guidance concerning the types of variables to be monitored is provided in Revision 2 of Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Copies of the Regulatory Guide may be purchased through the U.S. Government Publishing Office by calling 202–512–1800 or by writing to the U.S. Government Publishing Office, P.O. Box 37082, Washington, DC 20013–7082.

of the equipment, and the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional, including the radiation resulting from recirculating fluids for equipment located near the recirculating lines and including dose-rate effects.

(5) *Aging*. Equipment qualified by test must be preconditioned by natural or artificial (accelerated) aging to its end-of-installed life condition. Consideration must be given to all significant types of degradation which can have an effect on the functional capability of the equipment. If preconditioning to an end-of-installed life condition is not practicable, the equipment may be preconditioned to a shorter designated life. The equipment must be replaced or refurbished at the end of this designated life unless ongoing qualification demonstrates that the item has additional life.

(6) *Submergence* (if subject to being submerged).

(7) *Synergistic effects*. Synergistic effects must be considered when these effects are believed to have a significant effect on equipment performance.

(8) *Margins*. Margins must be applied to account for unquantified uncertainty, such as the effects of production variations and inaccuracies in test instruments. These margins are in addition to any conservatisms applied during the derivation of local environmental conditions of the equipment unless these conservatisms can be quantified and shown to contain appropriate margins.

(f) Each item of electric equipment important to safety must be qualified by one of the following methods:

(1) Testing an identical item of equipment under identical conditions or under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.

(2) Testing a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable.

(3) Experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.

(4) Analysis in combination with partial type test data that supports the analytical assumptions and conclusions.

(g) Each holder of an operating license issued prior to February 22, 1983, shall, by May 20, 1983, identify the electric equipment important to safety within the scope of this section already qualified and submit a schedule for either the qualification to the provisions of this section or for the replacement of the remaining electric equipment important to safety within the scope of this section. This schedule must establish a goal of final environmental qualification of the electric equipment within the scope of this section by the end of the second refueling outage after March 31, 1982 or by March 31, 1985, whichever is earlier. The Director of the Office of Nuclear Reactor Regulation may grant requests for extensions of this deadline to a date no later than November 30, 1985, for specific pieces of equipment if these requests are filed on a timely basis and demonstrate good cause for the extension, such as procurement lead time, test complications, and installation problems. In exceptional cases, the Commission itself may consider and grant extensions beyond November 30, 1985, for completion of environmental qualification.

The schedule in this paragraph supersedes the June 30, 1982, deadline, or any other previously imposed date, for environmental qualification of electric equipment contained in certain nuclear power operating licenses.

(h) Each license shall notify the Commission as specified in § 50.4 of any significant equipment qualification problem that may require extension of the completion date provided in accordance with paragraph (g) of this section within 60 days of its discovery.

(i) Applicants for operating licenses granted after February 22, 1983, but prior to November 30, 1985, shall perform an analysis to ensure that the plant can be safely operated pending completion of equipment qualification required by this section. This analysis must be submitted, as specified in § 50.4, for consideration prior to the granting of an operating license and must include, where appropriate, consideration of:

(1) Accomplishing the safety function by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified.

(2) The validity of partial test data in support of the original qualification.

(3) Limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.

(4) Completion of the safety function prior to exposure to the accident environment resulting from a design basis event and ensuring that the subsequent failure of the equipment does not degrade any safety function or mislead the operator.

(5) No significant degradation of any safety function or misleading information to the operator as a result of failure of equipment under the accident environment resulting from a design basis event.

(j) A record of the qualification, including documentation in paragraph (d) of this section, must be maintained in an auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for future use to permit verification that each item of electric equipment important to safety covered by this section:

(1) Is qualified for its application; and

(2) Meets its specified performance requirements when it is subjected to the conditions predicted to be present when it must perform its safety function up to the end of its qualified life.

(k) Applicants for and holders of operating licenses are not required to requalify electric equipment important to safety in accordance with the provisions of this section if the Commission has previously required qualification of that equipment in accordance with ''Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors,'' November 1979 (DOR Guidelines), or NUREG–0588 (For Comment version), ''Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment.''

(l) Replacement equipment must be qualified in accordance with the provisions of this section unless there are sound reasons to the contrary.

[48 FR 2733, Jan. 21, 1983, as amended at 49 FR 45576, Nov. 19, 1984; 51 FR 40308, Nov. 6, 1986; 51 FR 43709, Dec. 3, 1986; 52 FR 31611, Aug. 21, 1987; 53 FR 19250, May 27, 1988; 61 FR 39300, July 29, 1996; 61 FR 65173, Dec. 11, 1996; 62 FR 47271, Sept. 8, 1997; 64 FR 72001, Dec. 23, 1999; 66 FR 64738, Dec. 14, 2001; 72 FR 49495, Aug. 28, 2007; 80 FR 45843, Aug. 3, 2015]

ISSUANCE, LIMITATIONS, AND CONDITIONS OF LICENSES AND CONSTRUCTION PERMITS

§ 50.50 Issuance of licenses and construction permits.

Upon determination that an application for a license meets the standards and requirements of the act and regulations, and that notifications, if any, to other agencies or bodies have been duly made, the Commission will issue a license, or if appropriate a construction permit, in such form and containing such conditions and limitations including technical specifications, as it deems appropriate and necessary.

§ 50.51 Continuation of license.

(a) Each license will be issued for a fixed period of time to be specified in the license but in no case to exceed 40 years from date of issuance. Where the operation of a facility is involved, the Commission will issue the license for the term requested by the applicant or for the estimated useful life of the facility if the Commission determines that the estimated useful life is less than the term requested. Where construction of a facility is involved, the Commission may specify in the construction permit the period for which the license will be issued if approved pursuant to § 50.56. Licenses may be renewed by the Commission upon the expiration of the period. Renewal of operating licenses for nuclear power plants is governed by 10 CFR part 54. Application for termination of license is to be made pursuant to § 50.82.

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

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

(2) Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the provisions of the specific 10 CFR part 50 license for the facility.

[56 FR 64976, Dec. 13, 1991, as amended at 61 FR 39300, July 29, 1996]

§ 50.52 Combining licenses.

The Commission may combine in a single license the activities of an applicant which would otherwise be licensed severally.

§ 50.53 Jurisdictional limitations.

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

[21 FR 355, Jan. 19, 1956, as amended at 43 FR 6924, Feb. 17, 1978]

§ 50.54 Conditions of licenses.

The following paragraphs of this section, with the exception of paragraphs (r) and (gg), and the applicable requirements of 10 CFR 50.55a, are conditions in every nuclear power reactor operating license issued under this part. The following paragraphs with the exception of paragraph (r), (s), and (u) of this section are conditions in every combined license issued under part 52 of this chapter, provided, however, that paragraphs (i) introductory text, (i)(1), (j), (k), (l), (m), (n), (w), (x), (y), (z), and (hh) of this section are only applicable after the Commission makes the finding under § 52.103(g) of this chapter.

(a)(1) Each nuclear power plant or fuel reprocessing plant licensee subject to the quality assurance criteria in appendix B of this part shall implement, under § 50.34(b)(6)(ii) or § 52.79 of this chapter, the quality assurance program described or referenced in the safety analysis report, including changes to that report. However, a holder of a combined license under part 52 of this chapter shall implement the quality assurance program described or referenced in the safety analysis report applicable to operation 30 days prior to the scheduled date for the initial loading of fuel.

(2) Each licensee described in paragraph (a)(1) of this section shall, by June 10, 1983, submit to the appropriate NRC Regional Office shown in appendix D of part 20 of this chapter the current description of the quality assurance program it is implementing for inclusion in the Safety Analysis Report, unless there are no changes to the description previously accepted by NRC. This submittal must identify changes made to the quality assurance program description since the description was submitted to NRC. (Should a licensee need additional time beyond June 10, 1983 to submit its current quality assurance program description to NRC, it shall notify the appropriate NRC Regional Office in writing, explain why additional time is needed, and provide a schedule for NRC approval showing when its current quality assurance program description will be submitted.)

(3) Each licensee described in paragraph (a)(1) of this section 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 § 50.71(e). 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:

(i) The use of a QA standard approved by the NRC which is more recent than the QA standard in the licensee's current QA program at the time of the change;

(ii) 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;

(iii) The use of generic organizational position titles that clearly denote the position function, supplemented as necessary by descriptive text, rather than specific titles;

(iv) The use of generic organizational charts to indicate functional relationships, authorities, and responsibilities, or, alternately, the use of descriptive text;

(v) 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

(vi) 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.

(4) 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:

(i) 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 § 50.4.

(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 appendix B 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 change 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.

(b) No right to the special nuclear material shall be conferred by the license except as may be defined by the license.

(c) Neither the license, 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 act and give its consent in writing.

(d) The license shall 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 act in a state of war or national emergency declared by Congress.

(e) The license shall be subject to revocation, suspension, modification, or amendment for cause as provided in the act and regulations, in accordance with the procedures provided by the act and regulations.

(f) The licensee shall at any time before expiration of the license, upon request of the Commission, submit, as specified in § 50.4, 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.

(g) The issuance or existence of the license shall not be deemed to waive, or relieve the licensee from compliance with, the antitrust laws, as specified in subsection 105a of the Act. In the event that the licensee should be found by a court of competent jurisdiction to have violated any provision of such antitrust laws in the conduct of the licensed activity, the Commission may suspend or revoke the license or take such other action with respect to it as shall be deemed necessary.

(h) The license shall be subject to the provisions of the Act now or hereafter in effect 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.

(i) Except as provided in § 55.13 of this chapter, the licensee may not permit the manipulation of the controls of any facility by anyone who is not a licensed operator or senior operator as provided in part 55 of this chapter.

(i–1) Within 3 months after either the issuance of an operating license or the date that the Commission makes the finding under § 52.103(g) of this chapter for a combined license, as applicable, the licensee shall have in effect an operator requalification program. The operator requalification program must, as a minimum, meet the requirements of § 55.59(c) of this chapter. Notwithstanding the provisions of § 50.59, the licensee may not, except as specifically authorized by the Commission decrease the scope of an approved operator requalification program.

(j) Apparatus and mechanisms other than controls, the operation of which may affect the reactivity or power level of a reactor shall be manipulated only with the knowledge and consent of an operator or senior operator licensed pursuant to part 55 of this chapter present at the controls.

(k) An operator or senior operator licensed pursuant to part 55 of this chapter shall be present at the controls at all times during the operation of the facility.

(l) The licensee shall designate individuals to be responsible for directing the licensed activities of licensed operators. These individuals shall be licensed as senior operators pursuant to part 55 of this chapter.

(m)(1) A senior operator licensed pursuant to part 55 of this chapter shall be present at the facility or readily available on call at all times during its operation, and shall be present at the facility during initial start-up and approach to power, recovery from an unplanned or unscheduled shut-down or significant reduction in power, and refueling, or as otherwise prescribed in the facility license.

(2) Notwithstanding any other provisions of this section, by January 1, 1984, licensees of nuclear power units shall meet the following requirements:

(i) Each licensee shall meet the minimum licensed operator staffing requirements in the following table:

MINIMUM REQUIREMENTS [1] PER SHIFT FOR ON-SITE STAFFING OF NUCLEAR POWER UNITS BY OPERATORS AND SENIOR OPERATORS LICENSED UNDER 10 CFR PART 55

Number of nuclear power units operating [2]	Position	One unit	Two units		Three units	
		One control room	One control room	Two control rooms	Two control rooms	Three control rooms
None	Senior Operator	1	1	1	1	1
	Operator	1	2	2	3	3
One	Senior Operator	2	2	2	2	2
	Operator	2	3	3	4	4
Two	Senior Operator	....	2	3	[3] 3	3
	Operator	....	3	4	[3] 5	5
Three	Senior Operator	....	....	....	3	4
	Operator	....	....	....	5	6

[1] Temporary deviations from the numbers required by this table shall be in accordance with criteria established in the unit's technical specifications.

[2] For the purpose of this table, a nuclear power unit is considered to be operating when it is in a mode other than cold shutdown or refueling as defined by the unit's technical specifications.

[3] The number of required licensed personnel when the operating nuclear power units are controlled from a common control room are two senior operators and four operators.

(ii) Each licensee shall have at its site a person holding a senior operator license for all fueled units at the site who is assigned responsibility for overall plant operation at all times there is fuel in any unit. If a single senior operator does not hold a senior operator license on all fueled units at the site, then the licensee must have at the site two or more senior operators, who in combination are licensed as senior operators on all fueled units.

(iii) When a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times.

(iv) Each licensee shall have present, during alteration of the core of a nuclear power unit (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person.

(3) Licensees who cannot meet the January 1, 1984 deadline must submit by October 1, 1983 a request for an extension to the Director of the Office of Nuclear Regulation and demonstrate good cause for the request.

(n) The licensee shall not, except as authorized pursuant to a construction permit, make any alteration in the facility constituting a change from the technical specifications previously incorporated in a license or construction permit pursuant to §50.36 of this part.

(o) Primary reactor containments for water cooled power reactors, other than facilities for which the certifications required under §§50.82(a)(1) or 52.110(a)(1) of this chapter have been submitted, shall be subject to the requirements set forth in appendix J to this part.

(p)(1) 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 cyber security plan prepared under §50.34(c) or §52.79(a), 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 prepared under §50.34(d) or §52.79(a), 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 §50.90.

(2) The licensee may make changes to the plans referenced in paragraph (p)(1) 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 §50.4 or §52.3 of this chapter, 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:

(i) All safeguards capabilities specified in the safeguards contingency plan available and functional;

(ii) Detailed procedures developed according to appendix C to part 73 of this chapter available at the licensee's site; and

(iii) All appropriate personnel trained to respond to safeguards incidents as outlined in the plan and specified in the detailed procedures.

(3) 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:

(i) At intervals not to exceed 12 months; or

(ii) 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.

(4) 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.

(q) *Emergency plans*—(1) Definitions for the purpose of this section:

(i) *Change* means an action that results in modification or addition to, or removal from, the licensee's emergency plan. All such changes are subject to the provisions of this section except where the applicable regulations establish specific criteria for accomplishing a particular change.

(ii) *Emergency plan* means the document(s), prepared and maintained by the licensee, that identify and describe the licensee's methods for maintaining emergency preparedness and responding to emergencies. An emergency plan includes the plan as originally approved by the NRC and all subsequent changes made by the licensee with, and without, prior NRC review and approval under paragraph (q) of this section.

(iii) *Emergency planning function* means a capability or resource necessary to prepare for and respond to a radiological emergency, as set forth in the elements of section IV. of appendix E to this part and, for nuclear power reactor licensees, the planning standards of § 50.47(b).

(iv) *Reduction in effectiveness* means 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.

(2) A holder of a license under this part, or a combined license under part 52 of this chapter after the Commission makes the finding under § 52.103(g) of this chapter, shall follow and maintain the effectiveness of an emergency plan that meets the requirements in appendix E to this part and, for nuclear power reactor licensees, the planning standards of § 50.47(b).

(3) 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 meet the requirements in appendix E to this part and, for nuclear power reactor licensees, the planning standards of § 50.47(b).

(4) The changes to a licensee's emergency plan that reduce the effectiveness of the plan as defined in paragraph (q)(1)(iv) of this section may not be implemented without prior approval by the NRC. A licensee desiring to make such a change after February 21, 2012 shall submit an application for an amendment to its license. In addition to the filing requirements of §§ 50.90 and 50.91, 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 meet the requirements in appendix E to this part and, for nuclear power reactor licensees, the planning standards of § 50.47(b).

(5) 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 § 50.4, a report of each such change made after February 21, 2012, including a summary of its analysis, within 30 days after the change is put in effect.

(6) The nuclear power reactor licensee shall retain the emergency plan and each change for which prior NRC approval was obtained pursuant to paragraph (q)(4) of this section as a

record until the Commission terminates the license for the nuclear power reactor.

(r) [Reserved]

(s)(1) [Reserved]

(2)(i) [Reserved]

(ii) If after April 1, 1981, 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 reactor shall be shut down until such deficiencies are remedied or whether other enforcement action is appropriate. In determining whether a shutdown 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 there are other compelling reasons for continued operation.

(3) 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.

(t)(1) 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:

(i) At intervals not to exceed 12 months or,

(ii) 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.

(2) 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.

(u) [Reserved]

(v) Each licensee subject to the requirements of Part 73 of this chapter shall ensure that Safeguards Information is protected against unauthorized disclosure in accordance with the requirements in §73.21 and the requirements in §73.22 or §73.23 of this chapter, as applicable.

(w) Each power reactor licensee under this part for a production or utilization facility of the type described in §50.21(b) or §50.22 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 reactor, to stabilize and decontaminate the reactor and the reactor station site at which the reactor experiencing the accident is located, provided that:

(1) The insurance required by paragraph (w) of this section must have a minimum coverage limit for each reactor station site of either $1.06 billion 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 (w)(4) of this section, any proceeds must be payable first for stabilization of the reactor and next for decontamination of the reactor and the reactor station site. If a licensee's coverage falls below the required minimum, the licensee shall 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.

(2)(i) With respect to policies issued or annually renewed on or after April 2, 1991, 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 reactor, to ensure that the reactor 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 (w)(4) 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 reactor or on the reactor station site such that there is a present danger of release off site in amounts that would pose a threat to the public health and safety.

(ii) The stabilization and decontamination requirements set forth in paragraph (w)(4) of this section must apply uniformly to all insurance policies required under paragraph (w) of this section.

(3) 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.

(4)(i) In the event of an accident at the licensee's reactor, whenever the estimated costs of stabilizing the licensed reactor and of decontaminating the reactor and the reactor station site exceed $100 million, the proceeds of the insurance required by paragraph (w) of this section must be dedicated to and used, first, to ensure that the licensed reactor 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 reactor and the reactor station 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 (w)(4)(iii) and (w)(4)(iv) of this section, as the Director may find necessary to protect the public health and safety. Actions needed to bring the reactor to and maintain the reactor in a safe and stable condition may include one or more of the following, as appropriate:

(A) Shutdown of the reactor;

(B) Establishment and maintenance of long-term cooling with stable decay heat removal;

(C) Maintenance of sub-criticality;

(D) Control of radioactive releases; and

(E) 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.

(ii) The licensee shall inform the Director of the Office of Nuclear Reactor Regulation in writing when the reactor 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 reactor is in this condition, or at such earlier time as the licensee may elect or the Director may for good cause direct, the licensee shall 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 § 50.82 for authority to decommission the reactor and to surrender the license voluntarily. Cleanup operations may include one or more of the following, as appropriate:

(A) Processing any contaminated water generated by the accident and by decontamination operations to remove radioactive materials;

(B) Decontamination of surfaces inside the auxiliary and fuel-handling buildings and the reactor building to levels consistent with the Commission's occupational exposure limits in 10 CFR part 20, and decontamination or disposal of equipment;

(C) Decontamination or removal and disposal of internal parts and damaged fuel from the reactor vessel; and

(D) Cleanup of the reactor coolant system.

(iii) 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 § 50.82 for authority to decommission the reactor and to surrender the license voluntarily. The Director shall approve or disapprove, 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 1 year, at which time it may be renewed. Each subsequent renewal order, if imposed, may be effective for not more than 6 months.

(iv) Of the balance of the proceeds of the required insurance not already expended to place the reactor in a safe and stable condition pursuant to paragraph (w)(2)(i) 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 shall 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.

(x) A licensee 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.

(y) Licensee action permitted by paragraph (x) of this section shall be approved, as a minimum, by a licensed senior operator, or, at a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted, by either a licensed senior operator or a certified fuel handler, prior to taking the action.

(z) Each licensee with a utilization facility licensed pursuant to sections 103 or 104b. of the Act shall immediately notify the NRC Operations Center of the occurrence of any event specified in § 50.72 of this part.

(aa) The license shall be subject to all conditions deemed 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).)

(bb) For nuclear power reactors licensed by the NRC, the licensee shall, within 2 years following permanent cessation of operation of the reactor or 5 years before expiration of the reactor operating license, whichever occurs first, submit written notification to the Commission for its review and preliminary approval of the program by which the licensee intends to manage and provide funding for the management of all irradiated fuel at the reactor following permanent cessation of operation of the reactor until title to the irradiated fuel and possession of the fuel is transferred to the Secretary of Energy for its ultimate disposal in a repository. Licensees of nuclear power reactors that have permanently ceased operation by April 4, 1994 are required to submit such written notification by April 4, 1996. Final Commission review will be undertaken as part of any proceeding for continued licensing under part 50 or part 72 of this chapter. The licensee must demonstrate to NRC that the elected actions will be consistent with NRC requirements for licensed possession of irradiated nuclear fuel and that the actions will be implemented on a timely basis. Where implementation of such actions requires NRC authorizations, the licensee shall verify in the notification that submittals for such actions have been or will be made to NRC and shall identify them. A copy of the notification shall be retained by the licensee as a record until expiration of the reactor operating license. The licensee shall notify the NRC of any significant changes in the proposed waste management program as described in the initial notification.

(cc)(1) Each licensee 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 that term is defined in 11 U.S.C. 101(14)) controlling the licensee or listing the license or licensee as property of the estate; or

(iii) An affiliate (as that term is defined in 11 U.S.C. 101(2)) 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.

(dd) A licensee may take reasonable action that departs from a license condition or a technical specification (contained in a license issued under 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 meet 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 paragraph (x) of this section, which remains in effect unless otherwise directed by the Commission during a national security emergency.

(ee)(1) Each license issued under this part authorizing the possession of byproduct and special nuclear material produced in the operation of the licensed reactor includes, whether stated in the license or not, the authorization to receive back that same material, in the same or altered form or combined with byproduct or special nuclear material produced in the operation of another reactor of the same licensee located at that site, from a licensee of the Commission or an Agreement State, or from a non-licensed entity authorized to possess the material.

(2) The authorizations in this subsection are subject to the same limitations and requirements applicable to the original possession of the material.

(3) This paragraph does not authorize the receipt of any material recovered from the reprocessing of irradiated fuel.

(ff) For licensees of nuclear power plants that have implemented the earthquake engineering criteria in appendix S to this part, plant shutdown is required as provided in paragraph IV(a)(3) of appendix S to this part. Prior to resuming operations, the licensee shall demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public and the licensing basis is maintained.

(gg)(1) Notwithstanding 10 CFR 52.103, if, following the conduct of the exercise required by paragraph IV.f.2.a of appendix E to part 50 of this chapter, FEMA identifies one or more deficiencies in the state of offsite emergency preparedness, the holder of a combined license under 10 CFR part 52 may operate at up to 5 percent of rated thermal power only if the Commission finds that the state of onsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. The NRC will base this finding on its assessment of the applicant's onsite emergency plans against the pertinent standards in §50.47 and appendix E to this part. Review of the applicant's emergency plans will include the following standards with offsite aspects:

(i) Arrangements for requesting and effectively using offsite assistance onsite have been made, arrangements to accommodate State and local staff at the licensee's Emergency Operations Facility have been made, and other organizations capable of augmenting the planned onsite response have been identified.

(ii) Procedures have been established for licensee communications with State and local response organizations, including initial notification of the declaration of emergency and periodic provision of plant and response status reports.

(iii) Provisions exist for prompt communications among principal response organizations to offsite emergency personnel who would be responding onsite.

(iv) Adequate emergency facilities and equipment to support the emergency response onsite are provided and maintained.

(v) Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use onsite.

(vi) Arrangements are made for medical services for contaminated and injured onsite individuals.

(vii) Radiological emergency response training has been made available to those offsite who may be called to assist in an emergency onsite.

(2) The condition in this paragraph, regarding operation at up to 5 percent power, ceases to apply 30 days after FEMA informs the NRC that the offsite deficiencies have been corrected, unless the NRC notifies the combined license holder before the expiration of the 30-day period that the Commission finds under paragraphs (s)(2) and (3) of this section 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.

(hh) (1) Each licensee shall develop, implement and maintain 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.

(2) Paragraph (hh)(1) of this section does not apply to a licensee that has submitted the certifications required under §50.82(a)(1) or §52.110(a) of this chapter.

(ii) [Reserved]

(jj) Structures, systems, and components subject to the codes and standards in 10 CFR 50.55a must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

[21 FR 355, Jan. 19, 1956]

EDITORIAL NOTE: For FEDERAL REGISTER citations affecting §50.54, see the List of CFR Sections Affected, which appears in the Finding Aids section of the printed volume and at *www.govinfo.gov.*

§ 50.55 Conditions of construction permits, early site permits, combined licenses, and manufacturing licenses.

Each construction permit for a utilization facility is subject to the following terms and conditions and the applicable requirements of § 50.55a; each construction permit for a production facility is subject to the following terms and conditions with the exception of paragraph (i); each early site permit is subject to the terms and conditions in paragraph (f) of this section; each manufacturing license is subject to the terms and conditions in paragraphs (e), (f), and (i) of this section and the applicable requirements of § 50.55a; and each combined license is subject to the terms and conditions in paragraphs (e), (f), and (i) of this section and the applicable requirements of § 50.55a until the date that the Commission makes the finding under § 52.103(g) of this chapter:

(a) The construction permit shall state the earliest and latest dates for completion of the construction or modification.

(b) If the proposed construction or modification of the facility is not completed by the latest completion date, the construction permit shall expire and all rights are 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 attributable 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.

(c) Except as modified by this section and § 50.55a, the construction permit shall be subject to the same conditions to which a license is subject.

(d) At or about the time of completion of the construction or modification of the facility, the applicant will file any additional information needed to bring the original application for license up to date, and will file an application for an operating license or an amendment to an application for a license to construct and operate the facility for the issuance of an operating license, as appropriate, as specified in § 50.30(d) of this part.

(e)(1) *Definitions.* For purposes of this paragraph, the definitions in § 21.3 of this chapter apply.

(2) *Posting requirements.* (i) Each individual, partnership, corporation, dedicating entity, or other entity subject to the regulations in this part shall post current copies of the regulations in this part; Section 206 of the Energy Reorganization Act of 1974 (ERA); and procedures adopted under the regulations in this part. These documents must be posted in a conspicuous position on any premises within the United States where the activities subject to this part are conducted.

(ii) If posting of the regulations in this part or the procedures adopted under the regulations in this part is not practicable, the licensee or firm subject to the regulations in this part 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 the regulation, procedures, and reports may be examined.

(3) *Procedures.* Each individual, corporation, partnership, or other entity holding a facility construction permit subject to this part, combined license (until the Commission makes the finding under 10 CFR 52.103(g)), and manufacturing license under 10 CFR part 52 must adopt appropriate procedures to—

(i) 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 (e)(3)(ii) 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.

(ii) 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 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 (e)(4)(v) 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.

(iii) Ensure that a director or responsible officer of the holder of a facility construction permit subject to this part, combined license (until the Commission makes the finding under 10 CFR 52.103(g)), and manufacturing license under 10 CFR part 52 is informed as soon as practicable, and, in all cases, within the 5 working days after completion of the evaluation described in paragraph (e)(3)(i) or (e)(3)(ii) of this section, if the construction or manufacture of a facility or activity, or a basic component supplied for such facility or activity—

(A) Fails to comply with the AEA, as amended, or any applicable regulation, order, or license of the Commission, relating to a substantial safety hazard;

(B) Contains a defect; or

(C) Undergoes any significant breakdown in any portion of the quality assurance program conducted under the requirements of appendix B to 10 CFR part 50 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.

(4) *Notification.* (i) The holder of a facility construction permit subject to this part, combined license (until the Commission makes the finding under 10 CFR 52.103(g)), and manufacturing license who obtains information reasonably indicating that the facility fails to comply with the AEA, as amended, 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 or responsible officer or designated person as discussed in paragraph (e)(4)(v) of this section.

(ii) The holder of a facility construction permit subject to this part, combined license, or manufacturing license, who 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 or responsible officer or designated person as discussed in paragraph (e)(4)(v) of this section.

(iii) The holder of a facility construction permit subject to this part, combined license, or manufacturing license, who obtains information reasonably indicating that the quality assurance program has undergone any significant breakdown discussed in paragraph (e)(3)(iii)(C) of this section must notify the Commission of the breakdown in the quality assurance program through a director or responsible officer or designated person as discussed in paragraph (e)(4)(v) of this section.

(iv) A dedicating entity is responsible for identifying and evaluating deviations and reporting defects and failures to comply associated with substantial safety hazards for dedicated items; and maintaining auditable records for the dedication process.

(v) 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 part 21 of this chapter, and satisfies the responsibility of individual directors or responsible officers of holders of construction permits issued under § 50.23, holders of combined licenses (until the Commission makes the finding under § 52.103 of this chapter), and holders of manufacturing licenses 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.

(5) *Notification—timing and where sent.* The notification required by paragraph (e)(4) of this section must consist of—

(i) Initial notification by facsimile, which is the preferred method of notification, to the NRC Operations Center at (301) 816–5151 or by telephone at (301) 816–5100 within 2 days following receipt of information by the director or responsible corporate officer under paragraph (e)(3)(iii) 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 (e)(3)(ii) of this section.

(ii) Written notification submitted to the Document Control Desk, U.S. Nuclear Regulatory Commission, by an appropriate method listed in § 50.4, with a copy to the appropriate Regional Administrator at the address specified in appendix D to part 20 of this chapter and a copy to the appropriate NRC resident inspector within 30 days following receipt of information by the director or responsible corporate officer under paragraph (e)(3)(iii) of this section, on the identification of a defect or failure to comply.

(6) *Content of notification.* The written notification required by paragraph (e)(5)(ii) of this section must clearly indicate that the written notification is being submitted under § 50.55(e) and include the following information, to the extent known.

(i) Name and address of the individual or individuals informing the Commission.

(ii) 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.

(iii) Identification of the firm constructing or manufacturing the facility or supplying the basic component which fails to comply or contains a defect.

(iv) 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.

(v) The date on which the information of a defect or failure to comply was obtained.

(vi) In the case of a basic component which contains a defect or fails to comply, the number and location of all the basic components in use at the facility subject to the regulations in this part.

(vii) In the case of a completed reactor manufactured under part 52 of this chapter, the entities to which the reactor was supplied.

(viii) 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.

(ix) 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.

(7) *Procurement documents.* Each individual, corporation, partnership, dedicating entity, or other entity subject to the regulations in this part shall ensure that each procurement document for a facility, or a basic component specifies or is issued by the entity subject to the regulations, when applicable, that the provisions of 10 CFR part 21 or 10 CFR 50.55(e) applies, as applicable.

(8) *Coordination with 10 CFR part 21.* The requirements of § 50.55(e) are satisfied when the defect or failure to comply associated with a substantial safety hazard has been previously reported under part 21 of this chapter, under § 73.71 of this chapter, or under §§ 50.55(e) or 50.73. For holders of construction permits issued before October 29, 1991, evaluation, reporting and recordkeeping requirements of § 50.55(e) may be met by complying with the comparable requirements of part 21 of this chapter.

(9) *Records retention.* The holder of a construction permit, combined license, and manufacturing license must prepare and maintain records necessary to

accomplish the purposes of this section, specifically—

(i) Retain procurement documents, which define the requirements that facilities or basic components must meet in order to be considered acceptable, for the lifetime of the facility or basic component.

(ii) Retain records of evaluations of all deviations and failures to comply under paragraph (e)(3)(i) of this section for the longest of:

(A) Ten (10) years from the date of the evaluation;

(B) Five (5) years from the date that an early site permit is referenced in an application for a combined license; or

(C) Five (5) years from the date of delivery of a manufactured reactor.

(iii) Retain records of all interim reports to the Commission made under paragraph (e)(3)(ii) of this section, or notifications to the Commission made under paragraph (e)(4) of this section for the minimum time periods stated in paragraph (e)(9)(ii) of this section;

(iv) Suppliers of basic components must retain records of:

(A) All notifications sent to affected licensees or purchasers under paragraph (e)(4)(iv) of this section for a minimum of ten (10) years following the date of the notification;

(B) The facilities or other purchasers to whom basic components or associated services were supplied for a minimum of fifteen (15) years from the delivery of the basic component or associated services.

(v) Maintaining records in accordance with this section satisfies the recordkeeping obligations under part 21 of this chapter of the entities, including directors or responsible officers thereof, subject to this section.

(f)(1) Each nuclear power plant or fuel reprocessing plant construction permit holder subject to the quality assurance criteria in appendix B of this part shall implement, pursuant to §50.34(a)(7) of this part, the quality assurance program described or referenced in the Safety Analysis Report, including changes to that report.

(2) Each construction permit holder described in paragraph (f)(1) of this section shall, by June 10, 1983, submit to the appropriate NRC Regional Office shown in appendix D of part 20 of this chapter the current description of the quality assurance program it is implementing for inclusion in the Safety Analysis Report, unless there are no changes to the description previously accepted by NRC. This submittal must identify changes made to the quality assurance program description since the description was submitted to NRC. (Should a permit holder need additional time beyond June 10, 1983 to submit its current quality assurance program description to NRC, it shall notify the appropriate NRC Regional Office in writing, explain why additional time is needed, and provide a schedule for NRC approval showing when its current quality assurance program description will be submitted.)

(3) After March 11, 1983, each construction permit holder described in paragraph (f)(1) of this section may make a change to a previously accepted quality assurance program description included or referenced in the Safety Analysis Report, 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 do 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 §50.4. Changes made to NRC-accepted quality assurance topical report descriptions must be submitted as specified in §50.4.

(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 appendix B 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.

(4) Each holder of an early site permit or a manufacturing license under part 52 of this chapter shall implement the quality assurance program described or referenced in the safety analysis report, including changes to that report. Each holder of a combined license shall implement the quality assurance program for design and construction described or referenced in the safety analysis report, including changes to that report, provided, however, that the holder of a combined license is not subject to the terms and conditions in this paragraph after the Commission makes the finding under § 52.103(g) of this chapter.

(i) Each holder described in paragraph (f)(4) of this section may make a change to a previously accepted quality assurance program description included or referenced in the safety analysis report, if 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:

(A) Changes to the safety analysis report must be submitted for review as specified in § 50.4. Changes made to NRC-accepted quality assurance topical report descriptions must be submitted as specified in § 50.4.

(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 appendix B 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 changes must be maintained as a facility record for three (3) 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.

(ii) [Reserved]

(g)–(h) [Reserved]

(i) Structures, systems, and components subject to the codes and standards in 10 CFR 50.55a must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

[21 FR 355, Jan. 19, 1956, as amended at 32 FR 4055, Mar. 15, 1967; 35 FR 11461, July 17, 1970; 35 FR 19661, Dec. 29, 1970; 36 FR 11424, June 12, 1971; 37 FR 6460, Mar. 30, 1972; 38 FR 1272, Jan. 11, 1973; 41 FR 16446, Apr. 19, 1976; 42 FR 43385, Aug. 29, 1977; 48 FR 1029, Jan. 10, 1983; 51 FR 40309, Nov. 6, 1986; 56 FR 36091, July 31, 1991; 59 FR 14087, Mar. 25, 1994; 68 FR 58809, Oct. 10, 2003; 72 FR 49497, Aug. 28, 2007; 78 FR 34248, June 7, 2013; 79 FR 65798, Nov. 5, 2014]

§ 50.55a Codes and standards.

(a) *Documents approved for incorporation by reference.* The standards listed in this paragraph (a) have been approved for incorporation by reference by the Director of the Federal Register pursuant to 5 U.S.C. 552(a) and 1 CFR part 51. The standards are available for inspection, by appointment, at the NRC Technical Library, which is located at Two White Flint North, 11545 Rockville Pike, Rockville, Maryland

20852; telephone: 301–415–7000; email: *Library.Resource@nrc.gov;* or at the National Archives and Records Administration (NARA). For information on the availability of this material at NARA, email *fr.inspection@nara.gov* or go to *www.archives.gov/federal-register/cfr/ibr-locations.html*.

(1) *American Society of Mechanical Engineers (ASME)*, Three Park Avenue, New York, NY 10016; telephone: 1–800–843–2763; *https://www.asme.org/Codes/*.

(i) *ASME Boiler and Pressure Vessel Code, Section III*. The editions and addenda for Section III of the ASME Boiler and Pressure Vessel Code (excluding Nonmandatory Appendices) are listed in this paragraph (a)(1)(i), but limited by those provisions identified in paragraph (b)(1) of this section.

(A) ''Rules for Construction of Nuclear Vessels:''

(*1*) 1963 Edition,
(*2*) Summer 1964 Addenda,
(*3*) Winter 1964 Addenda,
(*4*) 1965 Edition,
(*5*) 1965 Summer Addenda,
(*6*) 1965 Winter Addenda,
(*7*) 1966 Summer Addenda,
(*8*) 1966 Winter Addenda,
(*9*) 1967 Summer Addenda,
(*10*) 1967 Winter Addenda,
(*11*) 1968 Edition,
(*12*) 1968 Summer Addenda,
(*13*)1968 Winter Addenda,
(*14*) 1969 Summer Addenda,
(*15*) 1969 Winter Addenda,
(*16*) 1970 Summer Addenda, and
(*17*) 1970 Winter Addenda.

(B) ''Rules for Construction of Nuclear Power Plant Components:''

(*1*) 1971 Edition,
(*2*) 1971 Summer Addenda,
(*3*) 1971 Winter Addenda,
(*4*) 1972 Summer Addenda,
(*5*) 1972 Winter Addenda,
(*6*) 1973 Summer Addenda, and
(*7*) 1973 Winter Addenda.

(C) ''Division 1 Rules for Construction of Nuclear Power Plant Components:''

(*1*) 1974 Edition,
(*2*) 1974 Summer Addenda,
(*3*) 1974 Winter Addenda,
(*4*) 1975 Summer Addenda,
(*5*) 1975 Winter Addenda,
(*6*) 1976 Summer Addenda, and
(*7*) 1976 Winter Addenda;

(D) ''Rules for Construction of Nuclear Power Plant Components—Division 1'';

(*1*) 1977 Edition,
(*2*) 1977 Summer Addenda,
(*3*) 1977 Winter Addenda,
(*4*) 1978 Summer Addenda,
(*5*) 1978 Winter Addenda,
(*6*) 1979 Summer Addenda,
(*7*) 1979 Winter Addenda,
(*8*) 1980 Edition,
(*9*) 1980 Summer Addenda,
(*10*) 1980 Winter Addenda,
(*11*) 1981 Summer Addenda,
(*12*) 1981 Winter Addenda,
(*13*) 1982 Summer Addenda,
(*14*) 1982 Winter Addenda,
(*15*) 1983 Edition,
(*16*) 1983 Summer Addenda,
(*17*) 1983 Winter Addenda,
(*18*) 1984 Summer Addenda,
(*19*) 1984 Winter Addenda,
(*20*) 1985 Summer Addenda,
(*21*) 1985 Winter Addenda,
(*22*) 1986 Edition,
(*23*) 1986 Addenda,
(*24*) 1987 Addenda,
(*25*) 1988 Addenda,
(*26*) 1989 Edition,
(*27*) 1989 Addenda,
(*28*) 1990 Addenda,
(*29*) 1991 Addenda,
(*30*) 1992 Edition,
(*31*) 1992 Addenda,
(*32*) 1993 Addenda,
(*33*) 1994 Addenda,
(*34*) 1995 Edition,
(*35*) 1995 Addenda,
(*36*) 1996 Addenda, and
(*37*) 1997 Addenda.

(E) ''Rules for Construction of Nuclear Facility Components—Division 1:''

(*1*) 1998 Edition,
(*2*) 1998 Addenda,
(*3*) 1999 Addenda,
(*4*) 2000 Addenda,
(*5*) 2001 Edition,
(*6*) 2001 Addenda,
(*7*) 2002 Addenda,
(*8*) 2003 Addenda,
(*9*) 2004 Edition,
(*10*) 2005 Addenda,
(*11*) 2006 Addenda,
(*12*) 2007 Edition,
(*13*) 2008 Addenda,
(*14*) 2009b Addenda (including Subsection NCA; and Division 1 subsections NB through NH and Appendices),

(*15*) 2010 Edition (including Subsection NCA; and Division 1 subsections NB through NH and Appendices),

(*16*) 2011a Addenda (including Subsection NCA; and Division 1 subsections NB through NH and Appendices),

(*17*) 2013 Edition (including Subsection NCA; and Division 1 subsections NB through NH and Appendices),

(*18*) 2015 Edition (including Subsection NCA; and Division 1 subsections NB through NH and Appendices);

(*19*) 2017 Edition (including Subsection NCA; and Division 1 subsections NB through NG and Appendices); and

(*20*) 2019 Edition (including Subsection NCA; and Division 1 subsections NB through NG and Appendices).

(ii) *ASME Boiler and Pressure Vessel Code, Section XI.* The editions and addenda for Section XI of the ASME Boiler and Pressure Vessel Code are listed in this paragraph (a)(1)(ii), but limited by those provisions identified in paragraph (b)(2) of this section.

(A) [Reserved]

(B) ''Rules for Inservice Inspection of Nuclear Power Plant Components:''

(*1*) 1974 Edition;

(*2*) 1974 Summer Addenda;

(*3*) 1974 Winter Addenda; and

(*4*) 1975 Summer Addenda.

(C) ''Rules for Inservice Inspection of Nuclear Power Plant Components—Division 1:''

(*1*)–(*32*) [Reserved]

(*33*) 1995 Edition;

(*34*) 1995 Addenda;

(*35*) 1996 Addenda;

(*36*) 1997 Addenda;

(*37*)–(*40*) [Reserved]

(*41*) 2001 Edition;

(*42*) 2001 Addenda;

(*43*) 2002 Addenda;

(*44*) 2003 Addenda;

(*45*) 2004 Edition;

(*46*) 2005 Addenda;

(*47*) 2006 Addenda;

(*48*) 2007 Edition;

(*49*) 2008 Addenda;

(*50*) 2009b Addenda;

(*51*) 2010 Edition;

(*52*) 2011a Addenda;

(*53*) 2013 Edition;

(*54*) 2015 Edition;

(*55*) 2017 Edition; and

(*56*) 2019 Edition.

(iii) *ASME Code Cases: Nuclear Components*—(A) *ASME BPV Code Case N–513–3 Mandatory Appendix I.* ASME BPV Code Case N–513–3, ''Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping Section XI, Division 1,'' Mandatory Appendix I, ''Relations for F_m, F_b, and F for Through-Wall Flaws'' (Approval Date: January 26, 2009). ASME BPV Code Case N–513–3 Mandatory Appendix I is referenced in paragraph (b)(2)(xxxiv)(B) of this section.

(B) *ASME BPV Code Case N–722–1.* ASME BPV Code Case N–722–1, ''Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials, Section XI, Division 1'' (Approval Date: January 26, 2009), with the conditions in paragraph (g)(6)(ii)(E) of this section.

(C) *ASME BPV Code Case N–729–6.* ASME BPV Code Case N–729–6, ''Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1'' (Approval Date: March 3, 2016), with the conditions in paragraph (g)(6)(ii)(D) of this section.

(D) *ASME BPV Code Case N–770–5.* ASME BPV Code Case N–770–5, ''Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities Section XI, Division 1'' (Approval Date: November 7, 2016), with the conditions in paragraph (g)(6)(ii)(F) of this section.

(E) [Reserved]

(F) *ASME BPV Code Case N–852.* ASME BPV Code Case N–852, ''Application of the ASME NPT Stamp, Section III, Division 1; Section III, Division 2; Section III, Division 3; Section III, Division 5'' (Approval Date: February 9, 2015). ASME BPV Code Case N–852 is referenced in paragraph (b)(1)(ix) of this section.

(G) [Reserved]

(H) *ASME OM Code Case OMN–28*. ASME OM Case OMN–28, ''Alternative Valve Position Verification Approach to Satisfy ISTC–3700 for Valves Not Susceptible to Stem-Disk Separation.'' Issued March 4, 2021. OMN–28 is referenced in paragraph (b)(3)(xi) of this section.

(iv) *ASME Operation and Maintenance Code*. The editions and addenda for the ASME Operation and Maintenance of Nuclear Power Plants are listed in this paragraph (a)(1)(iv), but limited by those provisions identified in paragraph (b)(3) of this section.

(A) ''Code for Operation and Maintenance of Nuclear Power Plants:''

(*1*) 1995 Edition;
(*2*) 1996 Addenda;
(*3*) 1997 Addenda;
(*4*) 1998 Edition;
(*5*) 1999 Addenda;
(*6*) 2000 Addenda;
(*7*) 2001 Edition;
(*8*) 2002 Addenda;
(*9*) 2003 Addenda;
(*10*) 2004 Edition;
(*11*) 2005 Addenda; and
(*12*) 2006 Addenda.

(B) ''Operation and Maintenance of Nuclear Power Plants, Division 1: Section IST Rules for Inservice Testing of Light-Water Reactor Power Plants:''

(*1*) 2009 Edition.
(*2*) [Reserved]

(C) Operation and Maintenance of Nuclear Power Plants:

(*1*) 2012 Edition, ''Division 1: OM Code: Section IST'';
(*2*) 2017 Edition; and
(*3*) 2020 Edition.

(v) *ASME Quality Assurance Requirements*. (A) ASME NQA–1, ''Quality Assurance Program Requirements for Nuclear Facilities:''

(*1*) NQA–1—1983 Edition;
(*2*) NQA–1a—1983 Addenda;
(*3*) NQA–1b—1984 Addenda;
(*4*) NQA–1c—1985 Addenda;
(*5*) NQA–1—1986 Edition;
(*6*) NQA–1a—1986 Addenda;
(*7*) NQA–1b—1987 Addenda;
(*8*) NQA–1c—1988 Addenda;
(*9*) NQA–1—1989 Edition;
(*10*) NQA–1a—1989 Addenda;
(*11*) NQA–1b—1991 Addenda; and
(*12*) NQA–1c—1992 Addenda.

(B) ASME NQA–1, ''Quality Assurance Requirements for Nuclear Facility Applications:''

(*1*) NQA–1—1994 Edition;
(*2*) NQA–1—2008 Edition;
(*3*) NQA–1a—2009;
(*4*) NQA–1b—2011 Addenda;
(*5*) NQA–1—2012; and
(*6*) NQA–1—2015.

(2) *Institute of Electrical and Electronics Engineers (IEEE) Service Center*, 445 Hoes Lane, Piscataway, NJ 08855; telephone: 1–800–678–4333; *http://ieeexplore.ieee.org*.

(i) *IEEE standard 279–1968*. (IEEE Std 279–1968), ''Proposed IEEE Criteria for Nuclear Power Plant Protection Systems'' (Approval Date: August 30, 1968), referenced in paragraph (h)(2) of this section. (Copies of this document may be purchased from IHS Global, 15 Inverness Way East, Englewood, CO 80112; *https://global.ihs.com.*)

(ii) *IEEE standard 279–1971*. (IEEE Std 279–1971), ''Criteria for Protection Systems for Nuclear Power Generating Stations'' (Approval Date: June 3, 1971), referenced in paragraph (h)(2) of this section.

(iii) *IEEE standard 603–1991*. (IEEE Std 603–1991), ''Standard Criteria for Safety Systems for Nuclear Power Generating Stations'' (Approval Date: June 27, 1991), referenced in paragraphs (h)(2) and (h)(3) of this section. All other standards that are referenced in IEEE Std 603–1991 are not approved for incorporation by reference.

(iv) *IEEE standard 603–1991, correction sheet*. (IEEE Std 603–1991 correction sheet), ''Standard Criteria for Safety Systems for Nuclear Power Generating Stations, Correction Sheet, Issued January 30, 1995,'' referenced in paragraphs (h)(2) and (h)(3) of this section. (This correction sheet is available from IEEE at *http://standards.ieee.org/findstds/errata/*).

(3) *U.S. Nuclear Regulatory Commission (NRC) Public Document Room*, 11555 Rockville Pike, Rockville, Maryland 20852; telephone: 1–800–397–4209; email: *pdr.resource@nrc.gov; https://www.nrc.gov/reading-rm/doc-collections/reg-guides/*. The use of code cases listed in the NRC regulatory guides in paragraphs (a)(3)(i) through (iii) of this section is acceptable with the specified conditions in those guides when implementing the editions and addenda of the ASME BPV Code and ASME OM

Code incorporated by reference in paragraph (a)(1) of this section. The NRC report in paragraph (a)(3)(iv) of this section is acceptable as specified in the conditions when implementing code cases listed in the NRC regulatory guides in paragraphs (a)(3)(i) through (iii) of this section.

(i) *NRC Regulatory Guide 1.84, Revision 39*. NRC Regulatory Guide 1.84, Revision 39, ''Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,'' issued December 2021, with the requirements in paragraph (b)(4) of this section.

(ii) *NRC Regulatory Guide 1.147, Revision 20*. NRC Regulatory Guide 1.147, Revision 20, ''Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1,'' issued December 2021, which lists ASME Code Cases that the NRC has approved in accordance with the requirements in paragraph (b)(5) of this section.

(iii) *NRC Regulatory Guide 1.192, Revision 4*. NRC Regulatory Guide 1.192, Revision 3, ''Operation and Maintenance Code Case Acceptability, ASME OM Code,'' issued December 2021, which lists ASME Code Cases that the NRC has approved in accordance with the requirements in paragraph (b)(6) of this section.

(iv) *NUREG–2228*. NUREG–2228, ''Weld Residual Stress Finite Element Analysis Validation: Part II—Proposed Validation Procedure,'' Published July 2020 (including Errata September 22, 2021), which is referenced in RG 1.147, Revision 20.

(4) Electric Power Research Institute, Materials Reliability Program, 3420 Hillview Avenue, Palo Alto, CA 94304–1338; telephone: 1–650–855–2000; *http://www.epri.com*.

(i) ''Materials Reliability Program: Topical Report for Primary Water Stress Corrosion Cracking Mitigation by Surface Stress Improvement (MRP–335, Revision 3–A)'', EPRI approval date: November 2016.

(ii) [Reserved]

(b) *Use and conditions on the use of standards*. Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME BPV Code and the ASME OM Code as specified in this paragraph (b). Each combined license for a utilization facility is subject to the following conditions.

(1) *Conditions on ASME BPV Code Section III*. Each manufacturing license, standard design approval, and design certification under 10 CFR part 52 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 latest edition and addenda incorporated by reference in paragraph (a)(1)(i) of this section, subject to the following conditions:

(i) *Section III condition: Section III materials*. When applying the 1992 Edition of Section III, applicants or licensees must apply the 1992 Edition with the 1992 Addenda of Section II of the ASME Boiler and Pressure Vessel Code.

(ii) *Section III condition: Weld leg dimensions*. When applying the 1989 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1) of this section, applicants and licensees may not apply the Section III provisions identified in table 1 to this paragraph (b)(1)(ii) for welds with leg size less than 1.09 t_n:

TABLE 1 TO PARAGRAPH (b)(1)(ii)—PROHIBITED CODE PROVISIONS

Editions and addenda	Code provision
1989 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(i) of this section.	Subparagraph NB–3683.4(c)(1); Subparagraph NB–3683.4(c)(2).
1989 Addenda through 2003 Addenda	Footnote 11 to Figure NC–3673.2(b)–1; Note 11 to Figure ND–3673.2(b)–1.
2004 Edition through 2010 Edition	Footnote 13 to Figure NC–3673.2(b)–1; Note 13 to Figure ND–3673.2(b)–1.
2011 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(i) of this section.	Footnote 11 to Table NC–3673.2(b)–1; Note 11 to Table ND–3673.2(b)–1.

(iii) *Section III condition: Seismic design of piping.* Applicants or licensees may use Subarticles NB–3200, NB–3600, NC–3600, and ND–3600 for seismic design of piping, up to and including the 1993 Addenda, subject to the condition specified in paragraph (b)(1)(ii) of this section. Applicants or licensees may not use these subarticles for seismic design of piping in the 1994 Addenda through the 2005 Addenda incorporated by reference in paragraph (a)(1) of this section, except that Subarticle NB–3200 in the 2004 Edition through the 2017 Edition may be used by applicants and licensees, subject to the condition in paragraph (b)(1)(iii)(A) of this section. Applicants or licensees may use Subarticles NB–3600, NC–3600, and ND–3600 for the seismic design of piping in the 2006 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(i) of this section, subject to the conditions of this paragraph (b)(1)(iii) corresponding to those subarticles.

(A) *Seismic design of piping: First provision.* When applying Note (1) of Figure NB–3222–1 for Level B service limits, the calculation of Pb stresses must include reversing dynamic loads (including inertia earthquake effects) if evaluation of these loads is required by NB–3223(b).

(B) *Seismic design of piping: Second provision.* For Class 1 piping, the material and Do/t requirements of NB–3656(b) must be met for all Service Limits when the Service Limits include reversing dynamic loads, and the alternative rules for reversing dynamic loads are used.

(iv) *Section III condition: Quality assurance.* When applying editions and addenda later than the 1989 Edition of Section III, an applicant or licensee may use the requirements of NQA–1, ''Quality Assurance Requirements for Nuclear Facility Applications,'' that is both incorporated by reference in paragraph (a)(1)(v) of this section and specified in either NCA–4000 or NCA–7000 of that Edition and Addenda of Section III, provided that the administrative, quality, and technical provisions contained in that Edition and Addenda of Section III are used in conjunction with the applicant's or licensee's appendix B to this part quality assurance program; and that the applicant's or licensee's Section III activities comply with those commitments contained in the applicant's or licensee's quality assurance program description. Where NQA–1 and Section III do not address the commitments contained in the applicant's or licensee's appendix B quality assurance program description, those licensee commitments must be applied to Section III activities.

(v) *Section III condition: Independence of inspection.* Applicants or licensees may not apply the exception in NCA–4134.10(a) of Section III, 1995 Edition through 2009b Addenda of the 2007 Edition, from paragraph 3.1 of Supplement 10S–1 of NQA–1–1994 Edition.

(vi) *Section III condition: Subsection NH.* The provisions in Subsection NH, ''Class 1 Components in Elevated Temperature Service,'' 1995 Addenda through all editions and addenda up to and including the 2013 Edition incorporated by reference in paragraph (a)(1) of this section, may only be used for the design and construction of Type 316 stainless steel pressurizer heater sleeves where service conditions do not cause the components to reach temperatures exceeding 900 °F.

(vii) *Section III condition: Capacity certification and demonstration of function of incompressible-fluid pressure-relief valves.* When applying the 2006 Addenda through all editions and addenda up to and including the latest edition and addenda incorporated by reference in paragraph (a)(1)(i) of this section, applicants and licensees may use paragraph NB–7742, except that paragraph NB–7742(a)(2) may not be used. For a valve design of a single size to be certified over a range of set pressures, the demonstration of function tests under paragraph NB–7742 must be conducted as prescribed in NB–7732.2 on two valves covering the minimum set pressure for the design and the maximum set pressure that can be accommodated at the demonstration facility selected for the test.

(viii) *Section III condition: Use of ASME certification marks.* When applying editions and addenda earlier than the 2011 Addenda to the 2010 Edition, licensees may use either the ASME BPV Code Symbol Stamps or the ASME Certification Marks with the appropriate certification designators and class designators as specified in the 2013 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1) of this section.

(ix) *Section III Condition: NPT Code Symbol Stamps.* Licensees may use the NPT Code Symbol Stamp with the letters arranged horizontally as specified in ASME BPV Code Case N–852 for the service life of a component that had the NPT Code Symbol Stamp applied during the time period from January 1, 2005, through December 31, 2015.

(x) *Section III Condition: Visual examination of bolts, studs and nuts.* Applicants or licensees applying the provisions of NB–2582, NC–2582, ND–2582, NE–2582, NF–2582, NG–2582 in the 2017 Edition of Section III through the latest edition and addenda incorporated by reference in paragraph (a)(1)(i) of this section, must apply paragraphs (b)(1)(x)(A) and (B) of this section.

(A) *Visual examination of bolts, studs, and nuts: First provision.* When applying the provisions of NB–2582, NC–2582, ND–2582, NE–2582, NF–2582, NG–2582 in the 2017 Edition of Section III through the latest edition and addenda incorporated by reference in paragraph (a)(1)(i) of this section, the visual examinations are required to be performed in accordance with procedures qualified to NB–5100, NC–5100, ND–5100, NE–5100, NF–5100, NG–5100 and performed by personnel qualified in accordance with NB–5500, NC–5500, ND–5500, NE–5500, NF–5500, and NG–5500.

(B) *Visual examination of bolts, studs, and nuts: Second provision.* When applying the provisions of NB–2582, NC–2582, ND–2582, NE–2582, NF–2582, and NG–2582 in the 2017 Edition of Section III through the latest edition and addenda incorporated by reference in paragraph (a)(1)(i) of this section, bolts, studs, and nuts must be visually examined for discontinuities including cracks, bursts, seams, folds, thread lap, voids, and tool marks.

(xi) *Section III condition: Mandatory Appendix XXVI.* When applying the 2015 and 2017 Editions of Section III, Mandatory Appendix XXVI, ''Rules for Construction of Class 3 Buried Polyethylene Pressure Piping,'' applicants or licensees must meet the following conditions:

(A) *Mandatory Appendix XXVI: First provision.* When performing fusing procedure qualification testing in accordance with XXVI–2300 and XXVI–4330 the following essential variables must be used for the performance qualification tests of butt fusion joints:

(*1*) Joint Type: A change in the type of joint from that qualified, except that a square butt joint qualifies as a mitered joint.

(*2*) Pipe Surface Alignment: A change in the pipe outside diameter (O.D.) surface misalignment of more than 10 percent of the wall thickness of the thinner member to be fused.

(*3*) PE Material: Each lot of polyethylene source material to be used in production (XXVI–2310(c)).

(*4*) Wall Thickness: Each thickness to be fused in production (XXVI–2310(c)).

(*5*) Diameter: Each diameter to be fused in production (XXVI–2310(c)).

(*6*) Cross-sectional Area: Each combination of thickness and diameter (XXVI–2310(c)).

(*7*) Position: Maximum machine carriage slope when greater than 20 degrees from horizontal (XXVI–4321(c)).

(*8*) Heater Surface Temperature: A change in the heater surface temperature to a value beyond the range tested (XXVI–2321).

(*9*) Ambient Temperature: A change in ambient temperature to less than 50 °F (10 °C) or greater than 125 °F (52 °C) (XXVI–4412(b)).

(*10*) Interfacial Pressure: A change in interfacial pressure to a value beyond the range tested (XXVI–2321).

(*11*) Decrease in Melt Bead Width: A decrease in melt bead size from that qualified.

(*12*) Increase in Heater Removal Time: An increase in heater plate removal time from that qualified.

(*13*) Decrease in Cool-down Time: A decrease in the cooling time at pressure from that qualified.

(*14*) Fusing Machine Carriage Model: A change in the fusing machine carriage model from that tested (XXVI–2310(d)).

(B) *Mandatory Appendix XXVI: Second provision.* When performing procedure qualification for high speed tensile impact testing of butt fusion joints in accordance with XXVI–2300 or XXVI–4330, breaks in the specimen that are away from the fusion zone must be retested. When performing fusing operator qualification bend tests of butt fusion joints in accordance with XXVI–4342, guided side bend testing must be used for all thicknesses greater than 1.25 inches.

(C) *Mandatory Appendix XXVI: Third provision*. When performing fusing procedure qualification tests in accordance with 2017 Edition of BPV Code Section III XXVI–2300 and XXVI–4330, the following essential variables must be used for the testing of electrofusion joints:

(*1*) Joint Design: A change in the design of an electrofusion joint.

(*2*) Fit-up Gap: An increase in the maximum radial fit-up gap qualified.

(*3*) Pipe PE Material: A change in the PE designation or cell classification of the pipe from that tested (XXVI–2322(a)).

(*4*) Fitting PE Material: A change in the manufacturing facility or production lot from that tested (XXVI–2322(b)).

(*5*) Pipe Wall Thickness: Each thickness to be fused in production (XXVI–2310(c)).

(*6*) Fitting Manufacturer: A change in fitting manufacturer.

(*7*) Pipe Diameter: Each diameter to be fused in production (XXVI–2310(c)).

(*8*) Cool-down Time: A decrease in the cool time at pressure from that qualified.

(*9*) Fusion Voltage: A change in fusion voltage.

(*10*) Nominal Fusion Time: A change in the nominal fusion time.

(*11*) Material Temperature Range: A change in material fusing temperature beyond the range qualified.

(*12*) Power Supply: A change in the make or model of electrofusion control box (XXVI–2310(f)).

(*13*) Power Cord: A change in power cord material, length, or diameter that reduces current at the coil to below the minimum qualified.

(*14*) Processor: A change in the manufacturer or model number of the processor. (XXVI–2310(f)).

(*15*) Saddle Clamp: A change in the type of saddle clamp.

(*16*) Scraping Device: A change from a clean peeling scraping tool to any other type of tool.

(xii) *Section III condition: Certifying Engineer*. When applying the 2017 and later editions of ASME BPV Code Section III, the NRC does not permit applicants and licensees to use a Certifying Engineer who is not a Registered Professional Engineer qualified in accordance with paragraph XXIII–1222 for Code-related activities that are applicable to U.S. nuclear facilities regulated by the NRC. The use of paragraph XXIII–1223 is prohibited.

(xiii) *Section III Condition: Preservice Inspection of Steam Generator Tubes*. Applicants or licensees applying the provisions of NB–5283 and NB–5360 in the 2019 Edition of Section III, must apply paragraphs (b)(1)(xiii)(A) and (B) of this section.

(A) *Preservice Inspection of Steam Generator Tubes: First provision*. When applying the provisions of NB–5283 in the 2019 Edition of Section III, a full-length preservice examination of 100 percent of the steam generator tubing in each newly installed steam generator must be performed prior to plant startup.

(B) *Preservice Inspection of Steam Generator Tubes: Second provision*. When applying the provisions of NB–5360 in the 2019 Edition of Section III, flaws revealed during preservice examination of steam generator tubing performed in accordance with paragraph (b)(1)(xiii)(A) of this section must be evaluated using the criteria in the design specifications.

(2) *Conditions on ASME BPV Code, Section XI*. As used in this section, references to Section XI refer to Section XI, Division 1, of the ASME BPV Code, and include the 1970 Edition through the 1976 Winter Addenda and the 1977 Edition through the latest edition incorporated by reference in paragraph (a)(1)(ii) of this section, subject to the following conditions:

(i) [Reserved]

(ii) *Section XI condition: Pressure-retaining welds in ASME Code Class 1 piping (applies to Table IWB–2500 and IWB–2500–1 and Category B–J)*. If the facility's application for a construction permit was docketed prior to July 1, 1978, the extent of examination for Code Class 1 pipe welds may be determined by the requirements of Table IWB–2500 and Table IWB–2600 Category B–J of Section XI of the ASME BPV Code in the 1974 Edition and Addenda through the Summer 1975 Addenda or other requirements the NRC may adopt.

(iii)–(vii) [Reserved]

(viii) *Section XI condition: Concrete containment examinations.* Applicants or licensees applying Subsection IWL, 2001 Edition through the 2004 Edition, up to and including the 2006 Addenda, must apply paragraphs (b)(2)(viii)(E) through (G) of this section. Applicants or licensees applying Subsection IWL, 2007 Edition up to and including the 2008 Addenda must apply paragraph (b)(2)(viii)(E) of this section. Applicants or licensees applying Subsection IWL, 2007 Edition with the 2009 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, must apply paragraphs (b)(2)(viii)(H) and (I) of this section.

(A)–(D) [Reserved]

(E) *Concrete containment examinations: Fifth provision.* For Class CC applications, the applicant or licensee must evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or the result in degradation to such inaccessible areas. For each inaccessible area identified, the applicant or licensee must provide the following in the ISI Summary Report required by IWA–6000:

(*1*) A description of the type and estimated extent of degradation, and the conditions that led to the degradation;

(*2*) An evaluation of each area, and the result of the evaluation; and

(*3*) A description of necessary corrective actions.

(F) *Concrete containment examinations: Sixth provision.* Personnel that examine containment concrete surfaces and tendon hardware, wires, or strands must meet the qualification provisions in IWA–2300. The ''owner-defined'' personnel qualification provisions in IWL–2310(d) are not approved for use.

(G) *Concrete containment examinations: Seventh provision.* Corrosion protection material must be restored following concrete containment post-tensioning system repair and replacement activities in accordance with the quality assurance program requirements specified in IWA–1400.

(H) *Concrete containment examinations: Eighth provision.* For each inaccessible area of concrete identified for evaluation under IWL–2512(a), or identified as susceptible to deterioration under IWL–2512(b), the licensee must provide the applicable information specified in paragraphs (b)(2)(viii)(E)(*1*), (*2*), and (*3*) of this section in the ISI Summary Report required by IWA–6000.

(I) *Concrete containment examinations: Ninth provision.* During the period of extended operation of a renewed license under part 54 of this chapter, the licensee must perform the technical evaluation under IWL–2512(b) of inaccessible below-grade concrete surfaces exposed to foundation soil, backfill, or groundwater at periodic intervals not to exceed 5 years. In addition, the licensee must examine representative samples of the exposed portions of the below-grade concrete, when such below-grade concrete is excavated for any reason.

(ix) *Section XI condition: Metal containment examinations.* Applicants or licensees applying Subsection IWE, 2001 Edition up to and including the 2003 Addenda, must satisfy the requirements of paragraphs (b)(2)(ix)(A) and (B), (F) through (I), and (K) of this section. Applicants or licensees applying Subsection IWE, 2004 Edition, up to and including the 2005 Addenda, must satisfy the requirements of paragraphs (b)(2)(ix)(A) and (B), (F) through (H), and (K) of this section. Applicants or licensees applying Subsection IWE, 2004 Edition with the 2006 Addenda, must satisfy the requirements of paragraphs (b)(2)(ix)(A)(*2*) and (b)(2)(ix)(B) and (K) of this section. Applicants or licensees applying Subsection IWE, 2007 Edition through the 2015 Edition, must satisfy the requirements of paragraphs (b)(2)(ix)(A)(*2*) and (b)(2)(ix)(B), (J), and (K) of this section. Applicants or licensees applying Subsection IWE, 2017 Edition, through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section must satisfy the requirements of paragraphs (b)(2)(ix)(A)(*2*) and (b)(2)(ix)(B) and (J) of this section.

(A) *Metal containment examinations: First provision.* For Class MC applications, the following apply to inaccessible areas.

(*1*) The applicant or licensee must evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or could result in degradation to such inaccessible areas.

(*2*) For each inaccessible area identified for evaluation, the applicant or licensee must provide the following in the ISI Summary Report as required by IWA–6000:

(*i*) A description of the type and estimated extent of degradation, and the conditions that led to the degradation;

(*ii*) An evaluation of each area, and the result of the evaluation; and

(*iii*) A description of necessary corrective actions.

(B) *Metal containment examinations: Second provision.* When performing remotely the visual examinations required by Subsection IWE, the maximum direct examination distance specified in Table IWA–2210–1 (2001 Edition through 2004 Edition) or Table IWA–2211–1 (2005 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1) of this section) may be extended and the minimum illumination requirements specified may be decreased provided that the conditions or indications for which the visual examination is performed can be detected at the chosen distance and illumination.

(C)–(E) [Reserved]

(F) *Metal containment examinations: Sixth provision.* VT–1 and VT–3 examinations must be conducted in accordance with IWA–2200. Personnel conducting examinations in accordance with the VT–1 or VT–3 examination method must be qualified in accordance with IWA–2300. The ''owner-defined'' personnel qualification provisions in IWE–2330(a) for personnel that conduct VT–1 and VT–3 examinations are not approved for use.

(G) *Metal containment examinations: Seventh provision.* The VT–3 examination method must be used to conduct the examinations in Items E1.12 and E1.20 of Table IWE–2500–1, and the VT–1 examination method must be used to conduct the examination in Item E4.11 of Table IWE–2500–1. An examination of the pressure-retaining bolted connections in Item E1.11 of Table IWE–2500–1 using the VT–3 examination method must be conducted once each interval. The ''owner-defined'' visual examination provisions in IWE–2310(a) are not approved for use for VT–1 and VT–3 examinations.

(H) *Metal containment examinations: Eighth provision.* Containment bolted connections that are disassembled during the scheduled performance of the examinations in Item E1.11 of Table IWE–2500–1 must be examined using the VT–3 examination method. Flaws or degradation identified during the performance of a VT–3 examination must be examined in accordance with the VT–1 examination method. The criteria in the material specification or IWB–3517.1 must be used to evaluate containment bolting flaws or degradation. As an alternative to performing VT–3 examinations of containment bolted connections that are disassembled during the scheduled performance of Item E1.11, VT–3 examinations of containment bolted connections may be conducted whenever containment bolted connections are disassembled for any reason.

(I) *Metal containment examinations: Ninth provision.* The ultrasonic examination acceptance standard specified in IWE–3511.3 for Class MC pressure-retaining components must also be applied to metallic liners of Class CC pressure-retaining components.

(J) *Metal containment examinations: Tenth provision.* In general, a repair/replacement activity such as replacing a large containment penetration, cutting a large construction opening in the containment pressure boundary to replace steam generators, reactor vessel heads, pressurizers, or other major equipment; or other similar modification is considered a major containment modification. When applying IWE–5000 to Class MC pressure-retaining components, any major containment modification or repair/replacement must be followed by a Type A test to provide assurance of both containment structural integrity and leak-tight integrity prior to returning to service, in accordance with appendix J to this part, Option A or Option B, on which the applicant's or licensee's Containment Leak-Rate Testing Program is based. When applying IWE–5000, if a Type A, B, or C Test is performed, the test pressure and acceptance standard for the test must be in accordance with appendix J to this part.

(K) *Metal Containment Examinations: Eleventh provision*. A general visual examination of containment leak chase channel moisture barriers must be performed once each interval, in accordance with the completion percentages in Table IWE 2411–1 of the 2017 Edition. Examination shall include the moisture barrier materials (caulking, gaskets, coatings, etc.) that prevent water from accessing the embedded containment liner within the leak chase channel system. Caps of stub tubes extending to or above the concrete floor interface may be inspected, provided the configuration of the cap functions as a moisture barrier as described previously. Leak chase channel system closures need not be disassembled for performance of examinations if the moisture barrier material is clearly visible without disassembly, or coatings are intact. The closures are acceptable if no damage or degradation exists that would allow intrusion of moisture against inaccessible surfaces of the metal containment shell or liner within the leak chase channel system. Examinations that identify flaws or relevant conditions shall be extended in accordance with paragraph IWE 2430 of the 2017 Edition.

(x) *Section XI condition: Quality assurance*. When applying the editions and addenda later than the 1989 Edition of ASME BPV Code, Section XI, licensees may use any edition or addenda of NQA–1, ''Quality Assurance Requirements for Nuclear Facility Applications,'' that is both incorporated by reference in paragraph (a)(1)(v) of this section and specified in Table IWA 1600–1 of that edition and addenda of Section XI, provided that the licensee uses its appendix B to this part quality assurance program in conjunction with Section XI requirements and the commitments contained in the licensee's quality assurance program description. Where NQA–1 and Section XI do not address the commitments contained in the licensee's appendix B quality assurance program description, those licensee commitments must be applied to Section XI activities.

(xi) [Reserved]

(xii) *Section XI condition: Underwater welding*. The provisions in IWA–4660, ''Underwater Welding,'' of Section XI, 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, are approved for use on irradiated material with the following conditions:

(A) *Underwater welding: First provision*. Licensees must obtain NRC approval in accordance with paragraph (z) of this section regarding the welding technique to be used prior to performing welding on ferritic material exposed to fast neutron fluence greater than 1×10^{17} n/cm^2 (E > 1 MeV).

(B) *Underwater welding: Second provision*. Licensees must obtain NRC approval in accordance with paragraph (z) of this section regarding the welding technique to be used prior to performing welding on austenitic material other than P–No. 8 material exposed to thermal neutron fluence greater than 1×10^{17} n/cm^2 (E < 0.5 eV). Licensees must obtain NRC approval in accordance with paragraph (z) regarding the welding technique to be used prior to performing welding on P–No. 8 austenitic material exposed to thermal neutron fluence greater than 1×10^{17} n/cm^2 (E < 0.5 eV) and measured or calculated helium concentration of the material greater than 0.1 atomic parts per million.

(xiii) [Reserved]

(xiv) *Section XI condition: Appendix VIII personnel qualification*. All personnel qualified for performing ultrasonic examinations in accordance with Appendix VIII must receive 8 hours of annual hands-on training on specimens that contain cracks. Licensees applying the 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section may use the annual practice requirements in VII–4240 of Appendix VII of Section XI in place of the 8 hours of annual hands-on training provided that the supplemental practice is performed on material or welds that contain cracks, or by analyzing prerecorded data from material or welds that contain cracks. In either case, training must be completed no earlier than 6 months prior to performing ultrasonic examinations at a licensee's facility.

(xv) *Section XI condition: Appendix VIII specimen set and qualification requirements.* Licensees using Appendix VIII in the 2001 Edition of the ASME Boiler and Pressure Vessel Code may elect to comply with all of the provisions in paragraphs (b)(2)(xv)(A) through (M) of this section, except for paragraph (b)(2)(xv)(F) of this section, which may be used at the licensee's option. Licensees using editions and addenda after 2001 Edition through the 2006 Addenda must use the 2001 Edition of Appendix VIII and may elect to comply with all of the provisions in paragraphs (b)(2)(xv)(A) through (M) of this section, except for paragraph (b)(2)(xv)(F) of this section, which may be used at the licensee's option.

(A) *Specimen set and qualification: First provision.* When applying Supplements 2, 3, and 10 to Appendix VIII, the following examination coverage criteria requirements must be used:

(*1*) Piping must be examined in two axial directions, and when examination in the circumferential direction is required, the circumferential examination must be performed in two directions, provided access is available. Dissimilar metal welds must be examined axially and circumferentially.

(*2*) Where examination from both sides is not possible, full coverage credit may be claimed from a single side for ferritic welds. Where examination from both sides is not possible on austenitic welds or dissimilar metal welds, full coverage credit from a single side may be claimed only after completing a successful single-sided Appendix VIII demonstration using flaws on the opposite side of the weld. Dissimilar metal weld qualifications must be demonstrated from the austenitic side of the weld, and the qualification may be expanded for austenitic welds with no austenitic sides using a separate add-on performance demonstration. Dissimilar metal welds may be examined from either side of the weld.

(B) *Specimen set and qualification: Second provision.* The following conditions must be used in addition to the requirements of Supplement 4 to Appendix VIII:

(*1*) Paragraph 3.1, Detection acceptance criteria—Personnel are qualified for detection if the results of the performance demonstration satisfy the detection requirements of ASME Section XI, Appendix VIII, Table VIII–S4–1, and no flaw greater than 0.25 inch through-wall dimension is missed.

(*2*) Paragraph 1.1(c), Detection test matrix—Flaws smaller than the 50 percent of allowable flaw size, as defined in IWB–3500, need not be included as detection flaws. For procedures applied from the inside surface, use the minimum thickness specified in the scope of the procedure to calculate a/t. For procedures applied from the outside surface, the actual thickness of the test specimen is to be used to calculate a/t.

(C) *Specimen set and qualification: Third provision.* When applying Supplement 4 to Appendix VIII, the following conditions must be used:

(*1*) A depth sizing requirement of 0.15 inch RMS must be used in lieu of the requirements in Subparagraphs 3.2(a) and 3.2(c), and a length sizing requirement of 0.75 inch RMS must be used in lieu of the requirement in Subparagraph 3.2(b).

(*2*) In lieu of the location acceptance criteria requirements of Subparagraph 2.1(b), a flaw will be considered detected when reported within 1.0 inch or 10 percent of the metal path to the flaw, whichever is greater, of its true location in the X and Y directions.

(*3*) In lieu of the flaw type requirements of Subparagraph 1.1(e)(1), a minimum of 70 percent of the flaws in the detection and sizing tests must be cracks. Notches, if used, must be limited by the following:

(*i*) Notches must be limited to the case where examinations are performed from the clad surface.

(*ii*) Notches must be semielliptical with a tip width of less than or equal to 0.010 inches.

(*iii*) Notches must be perpendicular to the surface within ±2 degrees.

(*4*) In lieu of the detection test matrix requirements in paragraphs 1.1(e)(2) and 1.1(e)(3), personnel demonstration test sets must contain a representative distribution of flaw orientations, sizes, and locations.

(D) *Specimen set and qualification: Fourth provision.* The following conditions must be used in addition to the requirements of Supplement 6 to Appendix VIII:

(*1*) Paragraph 3.1, Detection Acceptance Criteria—Personnel are qualified for detection if:

(*i*) No surface connected flaw greater than 0.25 inch through-wall has been missed.

(*ii*) No embedded flaw greater than 0.50 inch through-wall has been missed.

(*2*) Paragraph 3.1, Detection Acceptance Criteria—For procedure qualification, all flaws within the scope of the procedure are detected.

(*3*) Paragraph 1.1(b) for detection and sizing test flaws and locations—Flaws smaller than the 50 percent of allowable flaw size, as defined in IWB–3500, need not be included as detection flaws. Flaws that are less than the allowable flaw size, as defined in IWB–3500, may be used as detection and sizing flaws.

(*4*) Notches are not permitted.

(E) *Specimen set and qualification: Fifth provision.* When applying Supplement 6 to Appendix VIII, the following conditions must be used:

(*1*) A depth sizing requirement of 0.25 inch RMS must be used in lieu of the requirements of subparagraphs 3.2(a), 3.2(c)(2), and 3.2(c)(3).

(*2*) In lieu of the location acceptance criteria requirements in Subparagraph 2.1(b), a flaw will be considered detected when reported within 1.0 inch or 10 percent of the metal path to the flaw, whichever is greater, of its true location in the X and Y directions.

(*3*) In lieu of the length sizing criteria requirements of Subparagraph 3.2(b), a length sizing acceptance criteria of 0.75 inch RMS must be used.

(*4*) In lieu of the detection specimen requirements in Subparagraph 1.1(e)(1), a minimum of 55 percent of the flaws must be cracks. The remaining flaws may be cracks or fabrication type flaws, such as slag and lack of fusion. The use of notches is not allowed.

(*5*) In lieu of paragraphs 1.1(e)(2) and 1.1(e)(3) detection test matrix, personnel demonstration test sets must contain a representative distribution of flaw orientations, sizes, and locations.

(F) *Specimen set and qualification: Sixth provision.* The following conditions may be used for personnel qualification for combined Supplement 4 to Appendix VIII and Supplement 6 to Appendix VIII qualification. Licensees choosing to apply this combined qualification must apply all of the provisions of Supplements 4 and 6 including the following conditions:

(*1*) For detection and sizing, the total number of flaws must be at least 10. A minimum of 5 flaws must be from Supplement 4, and a minimum of 50 percent of the flaws must be from Supplement 6. At least 50 percent of the flaws in any sizing must be cracks. Notches are not acceptable for Supplement 6.

(*2*) Examination personnel are qualified for detection and length sizing when the results of any combined performance demonstration satisfy the acceptance criteria of Supplement 4 to Appendix VIII.

(*3*) Examination personnel are qualified for depth sizing when Supplement 4 to Appendix VIII and Supplement 6 to Appendix VIII flaws are sized within the respective acceptance criteria of those supplements.

(G) *Specimen set and qualification: Seventh provision.* When applying Supplement 4 to Appendix VIII, Supplement 6 to Appendix VIII, or combined Supplement 4 and Supplement 6 qualification, the following additional conditions must be used, and examination coverage must include:

(*1*) The clad-to-base-metal-interface, including a minimum of 15 percent T (measured from the clad-to-base-metal-interface), must be examined from four orthogonal directions using procedures and personnel qualified in accordance with Supplement 4 to Appendix VIII.

(*2*) If the clad-to-base-metal-interface procedure demonstrates detectability of flaws with a tilt angle relative to the weld centerline of at least 45 degrees, the remainder of the examination volume is considered fully examined if coverage is obtained in one parallel and one perpendicular direction. This must be accomplished using a procedure and personnel qualified for single-side examination in accordance with Supplement 6. Subsequent examinations of this volume may be performed using examination techniques qualified for a tilt angle of at least 10 degrees.

(*3*) The examination volume not addressed by paragraph (b)(2)(xv)(G)(*1*) of this section is considered fully examined if coverage is obtained in one parallel and one perpendicular direction, using a procedure and personnel qualified for single sided examination when the conditions in paragraph (b)(2)(xv)(G)(*2*) are met.

(H) *Specimen set and qualification: Eighth provision.* When applying Supplement 5 to Appendix VIII, at least 50 percent of the flaws in the demonstration test set must be cracks and the maximum misorientation must be demonstrated with cracks. Flaws in nozzles with bore diameters equal to or less than 4 inches may be notches.

(I) *Specimen set and qualification: Ninth provision.* When applying Supplement 5, Paragraph (a), to Appendix VIII, the number of false calls allowed must be D/10, with a maximum of 3, where D is the diameter of the nozzle.

(J) [Reserved]

(K) *Specimen set and qualification: Eleventh provision.* When performing nozzle-to-vessel weld examinations, the following conditions must be used when the requirements contained in Supplement 7 to Appendix VIII are applied for nozzle-to-vessel welds in conjunction with Supplement 4 to Appendix VIII, Supplement 6 to Appendix VIII, or combined Supplement 4 and Supplement 6 qualification.

(*1*) For examination of nozzle-to-vessel welds conducted from the bore, the following conditions are required to qualify the procedures, equipment, and personnel:

(*i*) For detection, a minimum of four flaws in one or more full-scale nozzle mock-ups must be added to the test set. The specimens must comply with Supplement 6, paragraph 1.1, to Appendix VIII, except for flaw locations specified in Table VIII S6–1. Flaws may be notches, fabrication flaws, or cracks. Seventy-five (75) percent of the flaws must be cracks or fabrication flaws. Flaw locations and orientations must be selected from the choices shown in paragraph (b)(2)(xv)(K)(*4*) of this section, Table VIII–S7–1—Modified, with the exception that flaws in the outer eighty-five (85) percent of the weld need not be perpendicular to the weld. There may be no more than two flaws from each category, and at least one subsurface flaw must be included.

(*ii*) For length sizing, a minimum of four flaws as in paragraph (b)(2)(xv)(K)(*1*)(*i*) of this section must be included in the test set. The length sizing results must be added to the results of combined Supplement 4 to Appendix VIII and Supplement 6 to Appendix VIII. The combined results must meet the acceptance standards contained in paragraph (b)(2)(xv)(E)(*3*) of this section.

(*iii*) For depth sizing, a minimum of four flaws as in paragraph (b)(2)(xv)(K)(*1*)(*i*) of this section must be included in the test set. Their depths must be distributed over the ranges of Supplement 4, Paragraph 1.1, to Appendix VIII, for the inner 15 percent of the wall thickness and Supplement 6, Paragraph 1.1, to Appendix VIII, for the remainder of the wall thickness. The depth sizing results must be combined with the sizing results from Supplement 4 to Appendix VIII for the inner 15 percent and to Supplement 6 to Appendix VIII for the remainder of the wall thickness. The combined results must meet the depth sizing acceptance criteria contained in paragraphs (b)(2)(xv)(C)(*1*), (b)(2)(xv)(E)(*1*), and (b)(2)(xv)(F)(*3*) of this section.

(*2*) For examination of reactor pressure vessel nozzle-to-vessel welds conducted from the inside of the vessel, the following conditions are required:

(*i*) The clad-to-base-metal-interface and the adjacent examination volume to a minimum depth of 15 percent T (measured from the clad-to-base-metal-interface) must be examined from four orthogonal directions using a procedure and personnel qualified in accordance with Supplement 4 to Appendix VIII as conditioned by paragraphs (b)(2)(xv)(B) and (C) of this section.

(*ii*) When the examination volume defined in paragraph (b)(2)(xv)(K)(*2*)(*i*) of this section cannot be effectively examined in all four directions, the examination must be augmented by examination from the nozzle bore using a procedure and personnel qualified in accordance with paragraph (b)(2)(xv)(K)(*1*) of this section.

(*iii*) The remainder of the examination volume not covered by paragraph (b)(2)(xv)(K)(*2*)(*ii*) of this section or a combination of paragraphs (b)(2)(xv)(K)(*2*)(*i*) and (*ii*) of this section, must be examined from the nozzle bore using a procedure and personnel qualified in accordance with paragraph (b)(2)(xv)(K)(*1*) of this section, or from the vessel shell using a procedure and personnel qualified for single sided examination in accordance with Supplement 6 to Appendix VIII, as conditioned by paragraphs (b)(2)(xv)(D) through (G) of this section.

(*3*) For examination of reactor pressure vessel nozzle-to-shell welds conducted from the outside of the vessel, the following conditions are required:

(*i*) The clad-to-base-metal-interface and the adjacent metal to a depth of 15 percent T (measured from the clad-to-base-metal-interface) must be examined from one radial and two opposing circumferential directions using a procedure and personnel qualified in accordance with Supplement 4 to Appendix VIII, as conditioned by paragraphs (b)(2)(xv)(B) and (C) of this section, for examinations performed in the radial direction, and Supplement 5 to Appendix VIII, as conditioned by paragraph (b)(2)(xv)(J) of this section, for examinations performed in the circumferential direction.

(*ii*) The examination volume not addressed by paragraph (b)(2)(xv)(K)(*3*)(*i*) of this section must be examined in a minimum of one radial direction using a procedure and personnel qualified for single sided examination in accordance with Supplement 6 to Appendix VIII, as conditioned by paragraphs (b)(2)(xv)(D) through (G) of this section.

(*4*) Table VIII–S7–1, ''Flaw Locations and Orientations,'' Supplement 7 to Appendix VIII, is conditioned as follows:

TABLE 2 TO PARAGRAPH (B)(2)(XV)(K)(*4*)—TABLE VIII: S7–1—MODIFIED

[Flaw locations and orientations]

	Parallel to weld	Perpendicular to weld
Inner 15 percent	X	X
Outside Diameter Surface	X	
Subsurface	X	....

(L) *Specimen set and qualification: Twelfth provision*. As a condition to the requirements of Supplement 8, Subparagraph 1.1(c), to Appendix VIII, notches may be located within one diameter of each end of the bolt or stud.

(M) *Specimen set and qualification: Thirteenth provision*. When implementing Supplement 12 to Appendix VIII, only the provisions related to the coordinated implementation of Supplement 3 to Supplement 2 performance demonstrations are to be applied.

(xvi) *Section XI condition: Appendix VIII single side ferritic vessel and piping and stainless steel piping examinations*. When applying editions and addenda prior to the 2007 Edition of Section XI, the following conditions apply.

(A) *Ferritic and stainless steel piping examinations: First provision*. Examinations performed from one side of a ferritic vessel weld must be conducted with equipment, procedures, and personnel that have demonstrated proficiency with single side examinations. To demonstrate equivalency to two sided examinations, the demonstration must be performed to the requirements of Appendix VIII, as conditioned by this paragraph and paragraphs (b)(2)(xv)(B) through (G) of this section, on specimens containing flaws with non-optimum sound energy reflecting characteristics or flaws similar to those in the vessel being examined.

(B) *Ferritic and stainless steel piping examinations: Second provision*. Examinations performed from one side of a ferritic or stainless steel pipe weld must be conducted with equipment, procedures, and personnel that have demonstrated proficiency with single side examinations. To demonstrate equivalency to two sided examinations, the demonstration must be performed to the requirements of Appendix VIII, as conditioned by this paragraph and paragraph (b)(2)(xv)(A) of this section.

(xvii) [Reserved]

(xviii) *Section XI condition: NDE personnel certification*—(A) *NDE personnel certification: First provision*. Level I and

II nondestructive examination personnel must be recertified on a 3-year interval in lieu of the 5-year interval specified in IWA–2314(a) and IWA–2314(b) of the 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section.

(B) *NDE personnel certification: Second provision.* When applying editions and addenda prior to the 2007 Edition of Section XI, paragraph IWA–2316 may only be used to qualify personnel that observe leakage during system leakage and hydrostatic tests conducted in accordance with IWA 5211(a) and (b).

(C) *NDE personnel certification: Third provision.* When applying editions and addenda prior to the 2005 Addenda of Section XI, licensee's qualifying visual examination personnel for VT–3 visual examination under paragraph IWA–2317 of Section XI must demonstrate the proficiency of the training by administering an initial qualification examination and administering subsequent examinations on a 3-year interval.

(D) *NDE personnel certification: Fourth provision.* The use of Appendix VII, Table VII–4110–1 and Appendix VIII, Subarticle VIII–2200 of the 2011 Addenda through the latest edition incorporated by reference in paragraph (a)(1)(ii) of this section is prohibited. When using ASME BPV Code, Section XI editions and addenda later than the 2010 Edition, licensees and applicants must use the prerequisites for ultrasonic examination personnel certifications in Appendix VII, Table VII–4110–1 and Appendix VIII, Subarticle VIII–2200 in the 2010 Edition.

(*1*) As an alternative to Note (c) in Table VII–4110–1 of ASME BPV Code, Section XI, 2010 Edition, the 250 hours of Level I experience time may be reduced to 175 hours, if the experience time includes a minimum of 125 hours of field experience and 50 hours of laboratory practice beyond the requirements of for training in accordance with Appendix VII Subarticle 4220, provided those practice hours are dedicated to the Level I or Level II skill areas as described in ANSI/ASNT CP–189.

(*2*) As an alternative to Note (d) in Table VII–4110–1 of ASME BPV Code, Section XI, 2010 Edition, the 800 hours of Level II experience time may be reduced to 720 hours, if the experience time includes a minimum of 400 hours of field experience and a minimum of 320 hours of laboratory practice. The practice must be dedicated to scanning specimens containing flaws in materials representative of those in actual power plant components. Additionally, for Level II Certification, the candidate must pass a Mandatory Appendix VIII, Supplement 2 performance demonstration for detection and length sizing.

(xix) *Section XI condition: Substitution of alternative methods.* The provisions for substituting alternative examination methods, a combination of methods, or newly developed techniques in the 1997 Addenda of IWA–2240 must be applied when using the 2001 Edition through the 2004 Edition of Section XI of the ASME BPV Code. The provisions in IWA–4520(c), 2001 Edition through the 2004 Edition, allowing the substitution of alternative methods, a combination of methods, or newly developed techniques for the methods specified in the Construction Code, are not approved for use. The provisions in IWA–4520(b)(2) and IWA–4521 of the 2008 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, allowing the substitution of ultrasonic examination for radiographic examination specified in the Construction Code, are not approved for use.

(xx) *Section XI condition: System leakage tests*—(A) *System leakage tests: First provision.* When performing system leakage tests in accordance with IWA–5213(a), 2001 Edition through 2002 Addenda, the licensee must maintain a 10-minute hold time after test pressure has been reached for Class 2 and Class 3 components that are not in use during normal operating conditions. No hold time is required for the remaining Class 2 and Class 3 components provided that the system has been in operation for at least 4 hours for insulated components or 10 minutes for uninsulated components.

(B) *System leakage tests: Second provision.* The nondestructive examination method and acceptance criteria of the 1992 Edition or later of Section III shall be met when performing system leakage tests (in lieu of a hydrostatic test) in accordance with IWA–4520 after repair and replacement activities performed by welding or brazing on a pressure retaining boundary using the 2003 Addenda through the latest edition and addenda of Section XI incorporated by reference in paragraph (a)(1)(ii) of this section. The nondestructive examination and pressure testing may be performed using procedures and personnel meeting the requirements of the licensee's/applicant's current ISI code of record.

(C) *System leakage tests: Third provision.* The use of the provisions for an alternative BWR pressure test at reduced pressure to satisfy IWA–4540 requirements as described in IWB–5210(c) of Section XI, 2017 Edition and IWA–5213(b)(2) and IWB–5221(d) of Section XI, 2017 Edition through the latest edition incorporated by reference in paragraph (a)(1)(ii) of this section may be used subject to the following conditions:

(*1*) The use of nuclear heat to conduct the BWR Class 1 system leakage test is prohibited (*i.e.*, the reactor must be in a non-critical state), except during refueling outages in which the ASME Section XI Category B–P pressure test has already been performed, or at the end of mid-cycle maintenance outages fourteen (14) days or less in duration.

(*2*) In lieu of the test condition holding time of IWA–5213(b)(2), after pressurization to test conditions, and before the visual examinations commence, the holding time shall be 1 hour for non-insulated components.

(xxi) *Section XI condition: Table IWB–2500–1 examination requirements.* (A) [Reserved]

(B) *Table IWB–2500–1 examination.* Use of the provisions of IWB–2500(f) and (g) and Table IWB–2500–1 Notes 6 and 7 of Section XI, 2017 Edition through the latest edition incorporated by reference in paragraph (a)(1)(ii) of this section, for examination of Examination Category B–D Item Numbers B3.90 and B3.100 shall be subject to the following conditions:

(*1*) A plant-specific evaluation demonstrating the criteria of IWB–2500(f) are met must be maintained in accordance with IWA–1400(l).

(*2*) The use of the provisions of IWB–2500(f) and Table IWB–2500–1 Note 6 for examination of Examination Category B–D Item Numbers B3.90 is prohibited for plants with renewed licenses in accordance with 10 CFR part 54.

(*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 a plant that receives its operating license after October 22, 2015.

(xxii) *Section XI condition: Surface examination.* The use of the provision in IWA–2220, ''Surface Examination,'' of Section XI, 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, that allows use of an ultrasonic examination method is prohibited.

(xxiii) *Section XI condition: Evaluation of thermally cut surfaces.* The use of the provisions for eliminating mechanical processing of thermally cut surfaces in IWA–4461.4.2 of Section XI, 2001 Edition through the 2009 Addenda, is prohibited.

(xxiv) *Section XI condition: Incorporation of the performance demonstration initiative and addition of ultrasonic examination criteria.* The use of Appendix VIII and the supplements to Appendix VIII and Article I–3000 of Section XI of the ASME BPV Code, 2002 Addenda through the 2006 Addenda, is prohibited.

(xxv) *Section XV Condition: Mitigation of defects by modification.* Use of the provisions of IWA–4340 must be subject to the following conditions:

(A) *Mitigation of defects by modification: First provision.* The use of the provisions for mitigation of defects by modification in IWA–4340 of Section XI 2001 Edition through the 2010 Addenda, is prohibited.

(B) *Mitigation of defects by modification: Second provision.* The provisions for mitigation of defects by modification in IWA–4340 of Section XI, 2011 Edition through the latest edition incorporated by reference in paragraph (a)(1)(ii) of this section, may be used subject to the following conditions:

(*1*) The use of the provisions in IWA 4340 to mitigate crack-like defects or those associated with flow accelerated corrosion are prohibited.

(*2*) The design of a modification that mitigates a defect must incorporate a loss of material rate either 2 times the actual measured corrosion rate, which must be established based on wall thickness measurements conducted at least twice, in that pipe location or another location with similar corrosion conditions, similar flow characteristics, and the same piping configuration (*e.g.*, straight run of pipe, elbow, tee) as the encapsulated area, or 4 times the estimated maximum corrosion rate for the piping system.

(*3*) The licensee must perform a wall thickness examination in the vicinity of the modification and relevant pipe base metal at half its expected life or, if the modification has an expected life greater than 19 years, once per interval starting with the interval subsequent to the mitigation, and the results must be used to confirm corrosion rates, determine the next inspection date, and confirm the design inputs.

(*i*) For buried pipe locations where the loss of material has occurred due to internal corrosion, the wall thickness examinations may be conducted at a different location in the same system as long as: Wall thickness measurements were conducted at the different location at the same time as installation of the modification; the flow rate is the same or higher at the different location; the piping configuration is the same (*e.g.*, straight run of pipe, elbow, tee); and if pitting occurred at the modification location, but not the different location, wall loss values must be multiplied by four (instead of two) times the actual measured corrosion rate. Where wall loss values are greater than that assumed during the design of the modification, the structural integrity of the modification must be reanalyzed. Additionally, if the extent of degradation is different (*i.e.*, percent wall loss plus or minus 25 percent) or the corrosion mechanism (*e.g.*, general, pitting) is not the same at the different location as at the modification location, the modification must be examined at half its expected life or 10 years, whichever is sooner.

(*ii*) For buried pipe locations where loss of material has occurred due to external corrosion, the modification must be examined at half its expected life or 10 years, whichever is sooner. Alternatively, when the modification has been recoated prior to return to service, the modification may be examined at half its expected life or during the subsequent 10-year inspection interval after installation, whichever is sooner.

(xxvi) *Section XI condition: Pressure Testing of Class 1, 2, and 3 Mechanical Joints.* Mechanical joints in Class 1, 2, and 3 piping and components greater than NPS–1 that are disassembled and reassembled during the performance of a Section XI repair/replacement activity requiring documentation on a Form NIS–2 must be verified to be leak tight. The verification must be performed to the standards of the licensee's appendix B to this part quality assurance program.

(xxvii) *Section XI condition: Removal of insulation.* When performing visual examination in accordance with IWA–5242 of Section XI of the ASME BPV Code, 2003 Addenda through the 2006 Addenda, or IWA–5241 of the 2007 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, insulation must be removed from 17–4 PH or 410 stainless steel studs or bolts aged at a temperature below 1100 °F or having a Rockwell Method C hardness value above 30, and from A–286 stainless steel studs or bolts preloaded to 100,000 pounds per square inch or higher.

(xxviii) *Section XI condition: Analysis of flaws.* Licensees using ASME BPV Code, Section XI, Appendix A, must use the following conditions when implementing Equation (2) in A–4300(b)(1):

For $R < 0$, ΔK_I depends on the crack depth (a), and the flow stress (σ_f). The flow stress is defined by $\sigma_f = 1/2(\sigma_{ys} + \sigma_{ult})$, where σ_{ys} is the yield strength and σ_{ult} is the ultimate tensilestrength in units ksi (MPa) and (a) is in units in. (mm). For $-2 \le R \le 0$ and $K_{max} - K_{min} \le 0.8 \times 1.12\ \sigma_f\sqrt{(\pi a)}$, $S = 1$ and $\Delta K_I = K_{max}$. For $R < -2$ and $K_{max} - K_{min} \le 0.8 \times 1.12\ \sigma_f\sqrt{(\pi a)}$, $S = 1$ and $\Delta K_I = (1 - R)\ K_{max}/3$. For $R < 0$ and $K_{max} - K_{min} > 0.8 \times 1.12\sigma_f\sqrt{(\pi a)}$, $S = 1$ and $\Delta K_I = K_{max} - K_{min}$.

(xxix) *Section XI condition: Nonmandatory Appendix R.* (A) Nonmandatory Appendix R, ''Risk-Informed Inspection Requirements for Piping Supplement 1—Risk-Informed Selection Process—Method A,'' of Section XI, 2005 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, may not be implemented without prior NRC authorization of the proposed alternative in accordance with paragraph (z) of this section.

(B) Nonmandatory Appendix R, ''Risk-Informed Inspection Requirements for Piping, Supplement 2—Risk-Informed Selection Process—Method B'' of Section XI, 2005 Addenda through the 2015 Edition, may not be implemented without prior NRC authorization of the proposed alternative in accordance with paragraph (z) of this section.

(C) Nonmandatory Appendix R, ''Risk-Informed Inspection Requirements for Piping, Supplement 2—Risk-Informed Selection Process—Method B'' of Section XI, 2017 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, may be implemented without prior NRC authorization of the proposed alternative in accordance with paragraph (z) of this section.

(xxx) [Reserved]

(xxxi) *Section XI condition: Mechanical clamping devices.* When installing a mechanical clamping device on an ASME BPV Code class piping system, Appendix W of Section XI shall be treated as a mandatory appendix and all of the provisions of Appendix W shall be met for the mechanical clamping device being installed. Additionally, use of IWA–4131.1(c) of the 2010 Edition of Section XI and IWA–4131.1(d) of the 2011 Addenda of the 2010 Edition and later versions of Section XI is prohibited on small item Class 1 piping and portions of a piping system that form the containment boundary.

(xxxii) *Section XI condition: Summary report submittal.* When using ASME BPV Code, Section XI, 2010 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, Summary Reports and Owner's Activity Reports described in IWA–6230 must be submitted to the NRC. Preservice inspection reports for examinations prior to commercial service must be submitted prior to the date of placement of the unit into commercial service. For preservice and inservice examinations performed following placement of the unit into commercial service, reports must be submitted within 120 calendar days of the completion of each refueling outage.

(xxxiii) *Section XI condition: Risk-Informed allowable pressure.* The use of Paragraph G–2216 in Appendix G in the 2011 Addenda and later editions and addenda of the ASME BPV Code, Section XI is prohibited.

(xxxiv) *Section XI condition: Nonmandatory Appendix U.* When using Nonmandatory Appendix U of the ASME BPV Code, Section XI, 2013 Edition through the latest edition incorporated by reference in paragraph (a)(1)(ii) of this section, the following conditions apply:

(A) The repair or replacement activities temporarily deferred under the provisions of Nonmandatory Appendix U must be performed during the next scheduled refueling outage.

(B) In lieu of the appendix referenced in paragraph U–S1–4.2.1(c) of Appendix U of the 2013 and the 2015 Editions the mandatory appendix in ASME BPV Code Case N–513–3 must be used.

(xxxv) *Section XI condition: Use of RT_{T0} in the K_{Ia} and K_{Ic} equations.*

(A) When using the 2013 Edition of the ASME BPV Code, Section XI, Appendix A, paragraph A–4200, if T_0 is available, then RT_{T0} may be used in place of RT_{NDT} for applications using the K_{Ic} equation and the associated K_{Ic} curve, but not for applications using the K_{Ia} equation and the associated K_{Ia} curve.

(B) When using the 2015 Edition of the ASME BPV Code, Section XI, Appendix A, paragraph A–4200 subparagraph (c) RT_{KIa} shall be defined as $RT_{KIa} = T_0 + 90.267 \exp(-0.003406T_0)$ for U.S. Customary Units.

(xxxvi) *Section XI condition: Fracture toughness of irradiated materials.* When using the 2013 Edition through the latest edition incorporated by reference in paragraph (a)(1)(ii) of this section of the ASME BPV Code, Section XI, Appendix A paragraph A–4400, the licensee shall obtain NRC approval under paragraph (z) of this section before using irradiated T_0 and the associated RT_{T0} in establishing fracture toughness of irradiated materials.

(xxxviii) *Section XI condition: ASME Code Section XI Appendix III Supplement 2*. Licensees applying the provisions of ASME Code Section XI Appendix III Supplement 2, ''Welds in Cast Austenitic Materials,'' are subject to the following conditions:

(A) *ASME Code Section XI Appendix III Supplement 2: First provision.*: First provision. In lieu of Paragraph (c)(1)(–c)(–2), licensees shall use a search unit with a center frequency of 500 kHz with a tolerance of ± 20 percent.

(B) *ASME Code Section XI Appendix III Supplement 2: Second provision.*: Second provision. In lieu of Paragraph (c)(1)(–d), the search unit shall produce angles including, but not limited to, 30 to 55 degrees with a maximum increment of 5 degrees.

(xxxix) *Section XI condition: Defect Removal*. The use of the provisions for removal of defects by welding or brazing in IWA–4421(c)(1) and IWA–4421(c)(2) of Section XI, 2017 Edition through the latest edition incorporated by reference in paragraph (a)(1)(ii) of this section may be used subject to the following conditions:

(A) *Defect removal requirements: First provision*. The provisions of subparagraph IWA 4421(c)(1) shall not be used to contain or isolate a defective area without removal of the defect.

(B) *Defect removal requirements: Second provision*. The provisions of subparagraph IWA–4421(c)(2) shall not be used for crack-like defects.

(xl) [Reserved]

(xli) *Section XI condition: Preservice Volumetric and Surface Examinations Acceptance*. The use of the provisions for accepting flaws by analytical evaluation during preservice inspection in IWB–3112(a)(3) and IWC–3112(a)(3) of Section XI, 2013 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section is prohibited.

(xlii) *Section XI condition: Steam Generator Nozzle-to-Component welds and Reactor Vessel Nozzle-to-Component welds*. Licensees applying the provisions of Table IWB–2500–1, Examination Category B–F, Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles, Item B5.11 (NPS 4 or Larger Nozzle-to-Component Butt Welds) of the 2013 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section and Item B5.71 (NPS 4 or Larger Nozzle-to-Component Butt Welds) of the 2011a Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section must also meet the following conditions:

(A) Ultrasonic examination procedures, equipment, and personnel shall be qualified by performance demonstration in accordance with Mandatory Appendix VIII.

(B) When applying the examination requirements of Figure IWB–2500–8, the volumetric examination volume shall be extended to include 100 percent of the weld volume, except as provided in paragraph (b)(2)(xlii)(B)(*1*) of this section:

(*1*) If the examination volume that can be obtained by performance demonstration qualified procedures is less than 100 percent of the weld volume, the licensee may ultrasonically examine the qualified volume and perform a flaw evaluation of the largest hypothetical crack that could exist in the volume not qualified for ultrasonic examination, subject to prior NRC authorization in accordance with paragraph (z) of this section.

(*2*) [Reserved]

(xliii) *Section XI condition: Section XI Condition: Regulatory Submittal Requirements*. Licensees shall submit for NRC review and approval the following analyses:

(A) The analytical evaluation determining the effects of an out-of-limit condition on the structural integrity of the Reactor Coolant System, as described in IWB–3720(a);

(B) Determination of T_0 and RT_{T0}, as described in Nonmandatory Appendix A, A–4200(c); and

(C) Determination of T_0 and RT_{T0}, as described in Nonmandatory Appendix G, G–2110(c).

(3) *Conditions on ASME OM Code*. As used in this section, references to the ASME OM Code are to the ASME OM

Code, Subsections ISTA, ISTB, ISTC, ISTD, ISTE, and ISTF; Mandatory Appendices I, II, III, IV, and V; and Nonmandatory Appendices A through H and J through M, in the editions and addenda of the ASME OM Code incorporated by reference in paragraph (a)(1)(iv) of this section. Mandatory appendices must be used if required by the OM Code; nonmandatory appendices are approved for use by the NRC but need not be used. When implementing the ASME OM Code, conditions are applicable only as specified in the following paragraphs:

(i) *OM condition: Quality assurance.* When applying editions and addenda of the ASME OM Code, the requirements of ASME Standard NQA–1, ''Quality Assurance Requirements for Nuclear Facility Applications,'' 1994 Edition, 2008 Edition, and 2009–1a Addenda, are acceptable as permitted by either ISTA 1.4 of the 1995 Edition through 1997 Addenda or ISTA–1500 of the 1998 Edition through the latest edition and addenda of the ASME OM Code incorporated by reference in paragraph (a)(1)(iv) of this section, provided the licensee uses its appendix B to this part quality assurance program in conjunction with the ASME OM Code requirements and the commitments contained in the licensee's quality assurance program description. Where NQA–1 and the ASME OM Code do not address the commitments contained in the licensee's appendix B quality assurance program description, the commitments must be applied to ASME OM Code activities.

(ii) *OM condition: Motor-Operated Valve (MOV) testing.* Licensees must comply with the provisions for testing MOVs in ASME OM Code, ISTC 4.2, 1995 Edition with the 1996 and 1997 Addenda, or ISTC–3500, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(iv) of this section, and must establish a program to ensure that MOVs continue to be capable of performing their design basis safety functions. Licensees implementing ASME OM Code, Mandatory Appendix III, ''Preservice and Inservice Testing of Active Electric Motor Operated Valve Assemblies in Light-Water Reactor Power Plants,'' of the 2009 Edition, through the latest edition and addenda of the ASME OM Code incorporated by reference in paragraph (a)(1)(iv) of this section shall comply with the following conditions:

(A) *MOV diagnostic test interval.* Licensees shall evaluate the adequacy of the diagnostic test intervals established for MOVs within the scope of ASME OM Code, Appendix III, not later than 5 years or three refueling outages (whichever is longer) from initial implementation of ASME OM Code, Appendix III.

(B) *MOV testing impact on risk.* Licensees shall ensure that the potential increase in core damage frequency and large early release frequency associated with the extension is acceptably small when extending exercise test intervals for high risk MOVs beyond a quarterly frequency.

(C) *MOV risk categorization.* When applying Appendix III to the ASME OM Code, licensees shall categorize MOVs according to their safety significance using the methodology described in ASME OM Code Case OMN–3, ''Requirements for Safety Significance Categorization of Components Using Risk Insights for Inservice Testing of LWR Power Plants,'' subject to the conditions applicable to OMN–3 which are set forth in Regulatory Guide 1.192, or using an MOV risk ranking methodology accepted by the NRC on a plant-specific or industry-wide basis in accordance with the conditions in the applicable safety evaluation.

(D) *MOV stroke time.* When applying Paragraph III–3600, ''MOV Exercising Requirements,'' of Appendix III to the ASME OM Code, licensees shall verify that the stroke time of MOVs specified in plant technical specifications satisfies the assumptions in the plant's safety analyses.

(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 August 17, 2018, and holders of combined licenses issued under 10 CFR part 52, whose initial fuel loading occurs on or after August 17, 2018, must also comply with the following conditions, as applicable:

(A) *Power-operated valves.* Licensees must periodically verify the capability of power-operated valves to perform their design-basis safety functions.

(B) *Check valves.* Licensees must perform bi-directional testing of check valves within the IST program where practicable.

(C) *Flow-induced vibration.* Licensees must monitor flow-induced vibration from hydrodynamic loads and acoustic resonance during preservice testing or inservice testing to identify potential adverse flow effects on components within the scope of the IST program.

(D) *High risk non-safety systems.* Licensees must assess the operational readiness of pumps, valves, and dynamic restraints within the scope of the Regulatory Treatment of Non-Safety Systems for applicable reactor designs.

(iv) *OM condition: Check valves (Appendix II).* Appendix II of the ASME OM Code, 2003 Addenda through the 2012 Edition, is acceptable for use with the following requirements. Trending and evaluation must support the determination that the valve or group of valves is capable of performing its intended function(s) over the entire interval. At least one of the Appendix II condition monitoring activities for a valve group must be performed on each valve of the group at approximate equal intervals not to exceed the maximum interval shown in the following table:

TABLE 3 TO PARAGRAPH (b)(3)(iv)—MAXIMUM INTERVALS FOR USE WHEN APPLYING INTERVAL EXTENSIONS

Group size	Maximum interval between activities of member valves in the groups (years)	Maximum interval between activities of each valve in the group (years)
≥4	4.5	16
3	4.5	12
2	6	12
1	Not applicable	10

(v) *OM condition: Snubbers ISTD.* Article IWF–5000, ''Inservice Inspection Requirements for Snubbers,'' of the ASME BPV Code, Section XI, must be used when performing inservice inspection examinations and tests of snubbers at nuclear power plants, except as conditioned in paragraphs (b)(3)(v)(A) and (B) of this section.

(A) *Snubbers: First provision.* Licensees may use Subsection ISTD, ''Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Power Plants,'' ASME OM Code, 1995 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(iv) of this section, in place of the requirements for snubbers in the editions and addenda up to the 2005 Addenda of the ASME BPV Code, Section XI, IWF–5200(a) and (b) and IWF–5300(a) and (b), by making appropriate changes to their technical specifications or licensee-controlled documents. Preservice and inservice examinations must be performed using the VT–3 visual examination method described in IWA–2213.

(B) *Snubbers: Second provision.* Licensees must comply with the provisions for examining and testing snubbers in Subsection ISTD of the ASME OM Code and make appropriate changes to their technical specifications or licensee-controlled documents when using the 2006 Addenda and later editions and addenda of Section XI of the ASME BPV Code.

(vi) *OM condition: Exercise interval for manual valves.* Manual valves must be exercised on a 2-year interval rather than the 5-year interval specified in paragraph ISTC–3540 of the 1999 through the 2005 Addenda of the ASME OM Code, provided that adverse conditions do not require more frequent testing.

(vii) [Reserved]

(viii) *OM condition: Subsection ISTE.* Licensees may not implement the risk-informed approach for inservice testing (IST) of pumps and valves specified in Subsection ISTE, ''Risk-Informed Inservice Testing of Components in

Light-Water Reactor Nuclear Power Plants,'' in the ASME OM Code, 2009 Edition through the 2017 Edition, without first obtaining NRC authorization to use Subsection ISTE as an alternative to the applicable IST requirements in the ASME OM Code, pursuant to paragraph (z) of this section.

(ix) *OM condition: Subsection ISTF*. Licensees applying Subsection ISTF, 2012 Edition must satisfy the requirements of Mandatory Appendix V, ''Pump Periodic Verification Test Program,'' of the ASME OM Code in that edition.

(x) [Reserved]

(xi) *OM condition: Valve Position Indication*. When implementing paragraph ISTC–3700, ''Position Verification Testing,'' in the ASME OM Code, 2012 Edition through the latest edition of the ASME OM Code incorporated by reference in paragraph (a)(1)(iv) of this section, licensees must verify that valve operation is accurately indicated by supplementing valve position indicating lights with other indications, such as flow meters or other suitable instrumentation to provide assurance of proper obturator position for valves with remote position indication within the scope of Subsection ISTC including its mandatory appendices and their verification methods and frequencies. For valves not susceptible to stem-disk separation, licensees may implement ASME OM Code Case OMN–28, ''Alternative Valve Position Verification Approach to Satisfy ISTC–3700 for Valves Not Susceptible to Stem-Disk Separation,'' which is incorporated by reference in paragraph (a)(1)(iii)(H) of this section. Where plant conditions make it impractical to perform the initial ISTC–3700 test as supplemented by paragraph (b)(3)(xi) of this section by the date 2 years following the previously performed ISTC–3700 test, a licensee may justify an extension of this initial supplemental valve position verification provided the ISTC–3700 test as supplemented by paragraph (b)(3)(xi) of this section is performed at the next available opportunity and no later than the next plant shutdown. This one-time extension of the ISTC–3700 test schedule as supplemented by paragraph (b)(3)(xi) of this section is acceptable provided the licensee has available for NRC review documented justification based on information obtained over the previous 5 years of the structural integrity of the stem-disk connection for the applicable valves. The licensee's justification could be based on, for example, verification of the valve stem-disk connection through an appropriate weak link analysis, appropriate disk motion confirmed during diagnostic testing, or allowance and cessation of flow through the valves. The licensee's justification must provide reasonable assurance that the remote indicating lights accurately reveal the position of the valve obturator until the next ISTC–3700 test as supplemented by paragraph (b)(3)(xi) of this section is performed.

(4) *Conditions on Design, Fabrication, and Materials Code Cases*. Each manufacturing license, standard design approval, and design certification application under part 52 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:

(i) *Design, Fabrication, and Materials Code Case condition: Applying Code Cases*. When an applicant or licensee initially applies a listed Code Case, the applicant or licensee must apply the most recent version of that Code Case incorporated by reference in paragraph (a) of this section.

(ii) *Design, Fabrication, and Materials Code Case condition: Applying different revisions of Code Cases*. If an applicant or licensee has previously applied a Code Case and a later version of the Code Case is incorporated by reference in paragraph (a) of this section, the applicant or licensee may continue to apply the previous version of the Code Case as authorized or may apply the later version of the Code Case, including any NRC-specified conditions placed on its use, until it updates its Code of Record for the component being constructed.

(iii) *Design, Fabrication, and Materials Code Case condition: Applying annulled Code Cases*. Application of an annulled Code Case is prohibited unless an applicant or licensee applied the listed Code Case prior to it being listed as annulled in Regulatory Guide 1.84. If an applicant or licensee has applied a listed Code Case that is later listed as annulled in Regulatory Guide 1.84, the applicant or licensee may continue to apply the Code Case until it updates its Code of Record for the component being constructed.

(5) *Conditions on inservice inspection Code Cases*. Licensees may apply the ASME BPV Code Cases listed in NRC Regulatory Guide 1.147, as incorporated by reference in paragraph (a)(3)(ii) of this section, without prior NRC approval, subject to the following:

(i) *ISI Code Case condition: Applying Code Cases*. When a licensee initially applies a listed Code Case, the licensee must apply the most recent version of that Code Case incorporated by reference in paragraph (a) of this section.

(ii) *ISI Code Case condition: Applying different revisions of Code Cases*. If a licensee has previously applied a Code Case and a later version of the Code Case is incorporated by reference in paragraph (a) of this section, the licensee may continue to apply, to the end of the current 120-month interval, the previous version of the Code Case, as authorized, or may apply the later version of the Code Case, including any NRC-specified conditions placed on its use. Licensees who choose to continue use of the Code Case during subsequent 120-month ISI program intervals will be required to implement the latest version incorporated by reference into this section as listed in Tables 1 and 2 of NRC Regulatory Guide 1.147, as incorporated by reference in paragraph (a)(3)(ii) of this section.

(iii) *ISI Code Case condition: Applying annulled Code Cases*. Application of an annulled Code Case is prohibited unless a licensee previously applied the listed Code Case prior to it being listed as annulled in NRC Regulatory Guide 1.147. If a licensee has applied a listed Code Case that is later listed as annulled in NRC Regulatory Guide 1.147, the licensee may continue to apply the Code Case to the end of the current 120-month interval.

(6) *Conditions on ASME OM Code Cases*. Licensees may apply the ASME OM Code Cases listed in NRC Regulatory Guide 1.192, as incorporated by reference in paragraph (a)(3)(iii) of this section, without prior NRC approval, subject to the following:

(i) *OM Code Case condition: Applying Code Cases*. When a licensee initially applies a listed Code Case, the licensee must apply the most recent version of that Code Case incorporated by reference in paragraph (a) of this section.

(ii) *OM Code Case condition: Applying different revisions of Code Cases*. If a licensee has previously applied a Code Case and a later version of the Code Case is incorporated by reference in paragraph (a) of this section, the licensee may continue to apply, to the end of the current 120-month interval, the previous version of the Code Case, as authorized, or may apply the later version of the Code Case, including any NRC-specified conditions placed on its use. Licensees who choose to continue use of the Code Case during subsequent 120-month ISI program intervals will be required to implement the latest version incorporated by reference into this section as listed in Tables 1 and 2 of NRC Regulatory Guide 1.192, as incorporated by reference in paragraph (a)(3)(iii) of this section.

(iii) *OM Code Case condition: Applying annulled Code Cases*. Application of an annulled Code Case is prohibited unless a licensee previously applied the listed Code Case prior to it being listed as annulled in NRC Regulatory Guide 1.192. If a licensee has applied a listed Code Case that is later listed as annulled in NRC Regulatory Guide 1.192, the licensee may continue to apply the Code Case to the end of the current 120-month interval.

(c) *Reactor coolant pressure boundary*. Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet 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 of this chapter and each combined license for a utilization facility is subject to the following conditions:

(1) *Standards requirement for reactor coolant pressure boundary components*. Components that are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III[14] of the ASME

BPV Code, except as provided in paragraphs (c)(2) through (4) of this section.

(2) *Exceptions to reactor coolant pressure boundary standards requirement.* Components that are connected to the reactor coolant system and are part of the reactor coolant pressure boundary as defined in § 50.2 need not meet the requirements of paragraph (c)(1) of this section, provided that:

(i) *Exceptions: Shutdown and cooling capability.* In the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system; or

(ii) *Exceptions: Isolation capability.* The component is or can be isolated from the reactor coolant system by two valves in series (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation and, assuming the other valve is open, its closure time must be such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.

(3) *Applicable Code and Code Cases and conditions on their use.* The Code edition, addenda, and optional ASME Code Cases to be applied to components of the reactor coolant pressure boundary must be determined by the provisions of paragraph NCA–1140, Subsection NCA of Section III of the ASME BPV Code, subject to the following conditions:

(i) *Reactor coolant pressure boundary condition: Code edition and addenda.* The edition and addenda applied to a component must be those that are incorporated by reference in paragraph (a)(1)(i) of this section;

(ii) *Reactor coolant pressure boundary condition: Earliest edition and addenda for pressure vessel.* The ASME Code provisions applied to the pressure vessel may be dated no earlier than the summer 1972 Addenda of the 1971 Edition;

(iii) *Reactor coolant pressure boundary condition: Earliest edition and addenda for piping, pumps, and valves.* The ASME Code provisions applied to piping, pumps, and valves may be dated no earlier than the Winter 1972 Addenda of the 1971 Edition; and

(iv) *Reactor coolant pressure boundary condition: Use of Code Cases.* The optional Code Cases applied to a component must be those listed in NRC Regulatory Guide 1.84 that is incorporated by reference in paragraph (a)(3)(i) of this section.

(4) *Standards requirement for components in older plants.* For a nuclear power plant whose construction permit was issued prior to May 14, 1984, the applicable Code edition and addenda for a component of the reactor coolant pressure boundary continue to be that Code edition and addenda that were required by Commission regulations for such a component at the time of issuance of the construction permit.

(d) *Quality Group B components.* Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet 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 of this chapter, and each combined license 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 meet the requirements for Class 2 Components in Section III of the ASME BPV Code.

(2) *Quality Group B: Applicable Code and Code Cases and conditions on their use.* The Code edition, addenda, and optional ASME Code Cases to be applied to the systems and components identified in paragraph (d)(1) of this section must be determined by the rules of paragraph NCA–1140, Subsection NCA of Section III of the ASME BPV Code, subject to the following conditions:

(i) *Quality Group B condition: Code edition and addenda.* The edition and addenda must be those that are incorporated by reference in paragraph (a)(1)(i) of this section;

(ii) *Quality Group B condition: Earliest edition and addenda for components.* The ASME Code provisions applied to the systems and components may be dated no earlier than the 1980 Edition; and

(iii) *Quality Group B condition: Use of Code Cases.* The optional Code Cases must be those listed in NRC Regulatory Guide 1.84 that is incorporated by reference in paragraph (a)(3)(i) of this section.

(e) *Quality Group C components.* Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet 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 of this chapter and each combined license 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 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 C[7] must meet the requirements for Class 3 components in Section III of the ASME BPV Code.

(2) *Quality Group C applicable Code and Code Cases and conditions on their use.* The Code edition, addenda, and optional ASME Code Cases to be applied to the systems and components identified in paragraph (e)(1) of this section must be determined by the rules of paragraph NCA–1140, subsection NCA of Section III of the ASME BPV Code, subject to the following conditions:

(i) *Quality Group C condition: Code edition and addenda.* The edition and addenda must be those incorporated by reference in paragraph (a)(1)(i) of this section;

(ii) *Quality Group C condition: Earliest edition and addenda for components.* The ASME Code provisions applied to the systems and components may be dated no earlier than the 1980 Edition; and

(iii) *Quality Group C condition: Use of Code Cases.* The optional Code Cases must be those listed in NRC Regulatory Guide 1.84 that is incorporated by reference in paragraph (a)(3)(i) of this section.

(f) *Preservice and inservice testing requirements.* Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet 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) 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.

(1) *Inservice testing requirements for older plants (pre-1971 CPs).* For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued prior to January 1, 1971, pumps and valves must meet the test requirements of paragraphs (f)(4) and (5) of this section to the extent practical. Pumps and valves that are part of the reactor coolant pressure boundary must meet the requirements applicable to components that are classified as ASME Code Class 1. Other pumps and valves that perform a function to shut down the reactor or maintain the reactor in a safe shutdown condition, mitigate the consequences of an accident, or provide overpressure protection for safety-related systems (in meeting the requirements of the 1986 Edition, or later, of the BPV or OM Code) must meet the test requirements applicable to components that are classified as ASME Code Class 2 or Class 3.

(2) *Design and accessibility requirements for performing inservice testing in plants with CPs issued between 1971 and 1974.* For a boiling or pressurized water-cooled nuclear power facility whose

construction permit was issued on or after January 1, 1971, but before July 1, 1974, pumps and valves that are classified as ASME BPV Code Class 1 and Class 2 must be designed and provided with access to enable the performance of inservice tests for operational readiness set forth in editions and addenda of Section XI of the ASME BPV Code incorporated by reference in paragraph (a)(1)(ii) of this section (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147 or NRC Regulatory Guide 1.192, as incorporated by reference in paragraphs (a)(3)(ii) and (iii) of this section, respectively) in effect 6 months before the date of issuance of the construction permit. The pumps and valves may meet the inservice test requirements set forth in subsequent editions of this Code and addenda that are incorporated by reference in paragraph (a)(1)(ii) of this section (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147 or NRC Regulatory Guide 1.192, as incorporated by reference in paragraphs (a)(3)(ii) and (iii) of this section, respectively), subject to the applicable conditions listed therein.

(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 design approval, design certification, combined license, or manufacturing license under part 52 of this chapter was issued on or after July 1, 1974:

(i)–(ii) [Reserved]

(iii) *IST design and accessibility requirements: Class 1 pumps and valves.* (A) *Class 1 pumps and valves: First provision.* In facilities whose construction permit was issued before 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 the editions and addenda of Section XI of the ASME BPV Code incorporated by reference in paragraph (a)(1)(ii) of this section (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147 or NRC Regulatory Guide 1.192, as incorporated by reference in paragraphs (a)(3)(ii) and (iii) of this section, respectively) applied to the construction of the particular pump or valve or the summer 1973 Addenda, whichever is later.

(B) *Class 1 pumps and valves: Second provision.* In facilities whose construction permit under this part, or design certification, design approval, combined license, or manufacturing license under part 52 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) *IST design and accessibility requirements: Class 2 and 3 pumps and valves.* (A) *Class 2 and 3 pumps and valves: First provision.* In facilities whose construction permit was issued before November 22, 1999, pumps and valves that are classified as ASME BPV Code Class 2 and Class 3 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in the editions and addenda of Section XI of the ASME BPV Code incorporated by reference in paragraph (a)(1)(ii) 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 pump or valve or the Summer 1973 Addenda, whichever is later.

(B) *Class 2 and 3 pumps and valves: Second provision.* In facilities whose construction permit under this part, or design certification, design approval, combined license, or manufacturing license under part 52 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.

(v) *IST design and accessibility requirements: Meeting later IST requirements.* All pumps and valves may meet the test requirements set forth in subsequent editions of codes and addenda or portions thereof that are incorporated by reference in paragraph (a) of this section, subject to the conditions listed in paragraph (b) of this section.

(4) *Inservice testing standards requirement for operating plants.* Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, pumps and valves that are within the scope of the ASME OM Code must meet the inservice test requirements (except design and access provisions) set forth in the ASME OM Code and addenda that become effective subsequent to editions and addenda specified in paragraphs (f)(2) and (3) of this section and that are incorporated by reference in paragraph (a)(1)(iv) of this section, to the extent practical within the limitations of design, geometry, and materials of construction of the components. The inservice test requirements for pumps and valves that are within the scope of the ASME OM Code but are not classified as ASME BPV Code Class 1, Class 2, or Class 3 may be satisfied as an augmented IST program. This use of an augmented IST program is acceptable without prior NRC approval provided the basis for deviations from the ASME OM Code, as incorporated by reference in this section, demonstrates an acceptable level of quality and safety, or that implementing the Code provisions would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, where documented and available for NRC review. When using the 2006 Addenda or later of the ASME BPV Code, Section XI, the inservice examination, testing, and service life monitoring requirements for dynamic restraints (snubbers) must meet the requirements set forth in the applicable ASME OM Code as specified in paragraph (b)(3)(v)(B) of this section. When using the 2005 Addenda or earlier edition or addenda of the ASME BPV Code, Section XI, the inservice examination, testing, and service life monitoring requirements for dynamic restraints (snubbers) must meet the requirements set forth in either the applicable ASME OM Code or ASME BPV Code, Section XI as specified in paragraph (b)(3)(v) of this section.

(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 18 months before the date scheduled for initial loading of fuel under a combined license under part 52 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).

(ii) *Applicable IST Code: Successive 120-month intervals.* Inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during successive 120-month intervals must comply with the requirements of the latest edition and addenda of the ASME OM Code incorporated by reference in paragraph (a)(1)(iv) of this section 18 months before the start of the 120-month interval (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147 or NRC Regulatory Guide 1.192 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.

(iii) [Reserved]

(iv) *Applicable IST Code: Use of later Code editions and addenda.* Inservice tests of pumps and valves may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in paragraph (a)(1)(iv) of this section, subject to the conditions listed in paragraph (b) of this section, and subject to NRC approval. Portions of editions or addenda may be used, provided that all related requirements of the respective editions or addenda are met.

(5) *Requirements for updating IST programs*—(i) *IST program update: Applicable IST Code editions and addenda.* The inservice test program for a boiling or pressurized water-cooled nuclear power facility must be revised by the licensee, as necessary, to meet the requirements of paragraph (f)(4) of this section.

(ii) *IST program update: Conflicting IST Code requirements with technical specifications.* If a revised inservice test program for a facility conflicts with the technical specifications for the facility, the licensee must apply to the Commission for amendment of the technical specifications to conform the technical specifications to the revised program. The licensee must submit this application, as specified in § 50.4, at least 6 months before the start of the period during which the provisions become applicable, as determined by paragraph (f)(4) of this section.

(iii) *IST program update: Notification of impractical IST Code requirements.* If the licensee has determined that conformance with certain Code requirements is impractical for its facility, the licensee must notify the Commission and submit, as specified in § 50.4, information to support the determination.

(iv) *IST program update: Schedule for completing impracticality determinations.* Where a pump or valve test requirement by the Code or addenda is determined to be impractical by the licensee and is not included in the revised inservice test program (as permitted by paragraph (f)(4) of this section), the basis for this determination must be submitted for NRC review and approval not later than 12 months after the expiration of the initial 120-month interval of operation from the start of facility commercial operation and each subsequent 120-month interval of operation during which the test is determined to be impractical.

(6) *Actions by the Commission for evaluating impractical and augmented IST Code requirements*—(i) *Impractical IST requirements: Granting of relief.* The Commission will evaluate determinations under paragraph (f)(5) of this section that code requirements are impractical. The Commission may grant relief and may impose such alternative requirements as it determines are authorized by law, will not endanger life or property or the common defense and security, and are otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

(ii) *Augmented IST requirements.* The Commission may require the licensee to follow an augmented inservice test program for pumps and valves for which the Commission deems that added assurance of operational readiness is necessary.

(7) *Inservice testing reporting requirements.* Inservice Testing Program Test and Examination Plans (IST Plans) for pumps, valves, and dynamic restraints (snubbers) prepared to meet the requirements of the ASME OM Code must be submitted to the NRC as specified in § 50.4. IST Plans must be submitted within 90 days of their implementation for the applicable 120-month IST Program interval. Electronic submission is preferred.

(g) *Preservice and inservice inspection requirements.* Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet 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) 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.

(1) *Inservice inspection requirements for older plants (pre-1971 CPs).* For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued before January 1, 1971, components (including supports) must meet the requirements of paragraphs (g)(4) and (g)(5) of this section to the extent practical. Components that are part of the reactor coolant pressure boundary and their supports must meet the requirements applicable to components that are classified as ASME Code Class 1. Other safety-related pressure vessels, piping, pumps and valves, and their supports must meet the requirements applicable to components that are classified as ASME Code Class 2 or Class 3.

(2) *Accessibility requirements*—(i) *Accessibility requirements for plants with CPs issued between 1971 and 1974.* For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, components that are classified as ASME BPV Code Class 1 and Class 2 and supports for components that are classified as ASME BPV Code Class 1 and Class 2 must be designed and be 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) in effect 6 months before the date of issuance of the construction permit.

(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 design certification, design approval, combined license, or manufacturing license under part 52 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.

(iii) *Accessibility requirements: Meeting later Code requirements.* All components (including supports) may meet the requirements set forth in subsequent editions of codes and addenda or portions thereof that are incorporated by reference in paragraph (a) of this section, subject to the conditions listed therein.

(3) *Preservice examination requirements*—(i) *Preservice examination requirements for plants with CPs issued between 1971 and 1974.* For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, components that are classified as ASME BPV Code Class 1 and Class 2 and supports for components that are classified as ASME BPV Code Class 1 and Class 2 must meet the preservice examination requirements 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) in effect 6 months before the date of issuance of the construction permit.

(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 design certification, design approval, combined license, or manufacturing license under part 52 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 meet 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.

(iii)–(iv) [Reserved]

(v) *Preservice examination requirements: Meeting later Code requirements.* All components (including supports) may meet the requirements set forth in subsequent editions of codes and addenda or portions thereof that are incorporated by reference in paragraph (a) of this section, subject to the conditions listed therein.

(4) *Inservice inspection standards requirement for operating plants.* Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) that are classified as ASME Code Class 1, Class 2, and Class 3 must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions and addenda of the ASME BPV Code that become effective subsequent to editions specified in paragraphs (g)(2) and (3) of this section and that are incorporated by reference in paragraph (a)(1)(ii) or (iv) of this section for snubber examination and testing of this section, to the extent practical within the limitations of design, geometry, and materials of construction of the components. Components that are classified as Class MC pressure retaining components and their integral attachments, and components that are classified as Class CC pressure retaining components and their integral attachments, must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of the ASME BPV Code and addenda that are incorporated by reference in paragraph (a)(1)(ii) of this section subject to the condition listed in paragraph (b)(2)(vi) of this section and the conditions listed in paragraphs (b)(2)(viii) and (ix) of this section, to the extent practical within the limitation of design, geometry, and materials of construction of the components. When using the 2006 Addenda or later of the ASME BPV Code, Section XI, the inservice examination, testing, and service life monitoring requirements for dynamic restraints (snubbers) must meet the requirements set forth in the applicable ASME OM Code as specified in paragraph (b)(3)(v)(B) of this section. When using the 2005 Addenda or earlier edition or addenda of the ASME BPV Code, Section XI, the inservice examination, testing, and service life monitoring requirements for dynamic restraints (snubbers) must meet the requirements set forth in either the applicable ASME OM Code or ASME BPV Code, Section XI as specified in paragraph (b)(3)(v) of this section.

(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 18 months before the date scheduled for initial loading of fuel under a combined license under part 52 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.

(ii) *Applicable ISI Code: Successive 120-month intervals.* Inservice examination of components and system pressure tests conducted during successive 120-

month inspection intervals must comply with the requirements of the latest edition and addenda of the ASME Code incorporated by reference in paragraph (a) of this section 18 months before the start of the 120-month inspection interval (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), subject to the conditions listed in paragraph (b) of this section. However, a licensee whose inservice inspection interval commences during the 12 through 18-month period after June 3, 2020, may delay the update of their Appendix VIII program by up to 18 months after June 3, 2020. Alternatively, 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.

(iii) *Applicable ISI Code: Optional surface examination requirement.* When applying editions and addenda prior to the 2003 Addenda of Section XI of the ASME BPV Code, licensees may, but are not required to, perform the surface examinations of high-pressure safety injection systems specified in Table IWB–2500–1, Examination Category B–J, Item Numbers B9.20, B9.21, and B9.22.

(iv) *Applicable ISI Code: Use of subsequent Code editions and addenda.* Inservice examination of components and system pressure tests may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in paragraph (a) of this section, subject to the conditions listed in paragraph (b) of this section, and subject to Commission approval. Portions of editions or addenda may be used, provided that all related requirements of the respective editions or addenda are met.

(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 combined license under part 52 of this chapter was issued after January 1, 1956, the following are required:

(A) *Metal and concrete containments: First provision.* Metal containment pressure retaining components and their integral attachments must meet the inservice inspection, repair, and replacement requirements applicable to components that are classified as ASME Code Class MC;

(B) *Metal and concrete containments: Second provision.* Metallic shell and penetration liners that are pressure retaining components and their integral attachments in concrete containments must meet the inservice inspection, repair, and replacement requirements applicable to components that are classified as ASME Code Class MC; and

(C) *Metal and concrete containments: Third provision.* Concrete containment pressure retaining components and their integral attachments, and the post-tensioning systems of concrete containments, must meet the inservice inspections, repair, and replacement requirements applicable to components that are classified as ASME Code Class CC.

(5) *Requirements for updating ISI programs*—(i) *ISI program update: Applicable ISI Code editions and addenda.* The inservice inspection program for a boiling or pressurized water-cooled nuclear power facility must be revised by the licensee, as necessary, to meet the requirements of paragraph (g)(4) of this section.

(ii) *ISI program update: Conflicting ISI Code requirements with technical specifications.* If a revised inservice inspection program for a facility conflicts with the technical specifications for the facility, the licensee must apply to the Commission for amendment of the technical specifications to conform the technical specifications to the revised program. The licensee must submit this application, as specified in § 50.4, at least six months before the start of the period during which the provisions become applicable, as determined by paragraph (g)(4) of this section.

(iii) *ISI program update: Notification of impractical ISI Code requirements.* If the licensee has determined that conformance with a Code requirement is impractical for its facility the licensee must notify the NRC and submit, as specified in § 50.4, information to support the determinations. Determinations of impracticality in accordance with this section must be based on the demonstrated limitations experienced when attempting to comply with the Code requirements during the inservice inspection interval for which the request is being submitted. Requests for relief made in accordance with this section must be submitted to the NRC no later than 12 months after the expiration of the initial or subsequent 120-month inspection interval for which relief is sought.

(iv) *ISI program update: Schedule for completing impracticality determinations.* Where the licensee determines that an examination required by Code edition or addenda is impractical, the basis for this determination must be submitted for NRC review and approval not later than 12 months after the expiration of the initial or subsequent 120-month inspection interval for which relief is sought.

(6) *Actions by the Commission for evaluating impractical and augmented ISI Code requirements*—(i) *Impractical ISI requirements: Granting of relief.* The Commission will evaluate determinations under paragraph (g)(5) of this section that code requirements are impractical. The Commission may grant such relief and may impose such alternative requirements as it determines are authorized by law, will not endanger life or property or the common defense and security, and are otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

(ii) *Augmented ISI program.* The Commission may require the licensee to follow an augmented inservice inspection program for systems and components for which the Commission deems that added assurance of structural reliability is necessary.

(A) [Reserved]

(B) *Augmented ISI requirements: Submitting containment ISI programs.* Licensees do not have to submit to the NRC for approval of their containment inservice inspection programs that were developed to satisfy the requirements of Subsection IWE and Subsection IWL with specified conditions. The program elements and the required documentation must be maintained on site for audit.

(C) [Reserved]

(D) *Augmented ISI requirements: Reactor vessel head inspections*—(*1*) *Implementation.* Holders of operating licenses or combined licenses for pressurized-water reactors as of or after June 3, 2020 shall implement the requirements of ASME BPV Code Case N–729–6 instead of ASME BPV Code Case N–729–4, subject to the conditions specified in paragraphs (g)(6)(ii)(D)(*2*) through (*8*) of this section, by no later than one year after June 3, 2020. All previous NRC-approved alternatives from the requirements of paragraph (g)(6)(ii)(D) of this section remain valid.

(*2*) *Appendix I use.* If Appendix I is used, Section I–3000 must be implemented to define an alternative examination area or volume.

(*3*) *Bare metal visual frequency.* Instead of Note 4 of ASME BPV Code Case N–729–6, the following shall be implemented. If effective degradation years (EDY) < 8 and if no flaws are found that are attributed to primary water stress corrosion cracking:

(*i*) A bare metal visual examination is not required during refueling outages when a volumetric or surface examination is performed; and

(*ii*) If a wetted surface examination has been performed of all of the partial penetration welds during the previous non-visual examination, the reexamination frequency may be extended to every third refueling outage or 5 calendar years, whichever is less, provided an IWA–2212 VT–2 visual examination of the head is performed under the insulation through multiple access points in outages that the VE is not completed. This IWA–2212 VT–2 visual examination may be performed with the reactor vessel depressurized.

(*4*) *Surface exam acceptance criteria.* In addition to the requirements of paragraph 3132.1(b) of ASME BPV Code Case N–729–6, a component whose surface examination detects rounded indications greater than allowed in paragraph NB–5352 in size on the partial-penetration or associated fillet weld shall be classified as having an unacceptable indication and corrected in accordance with the provisions of paragraph 3132.2 of ASME BPV Code Case N–729–6.

(*5*) *Peening.* In lieu of inspection requirements of Table 1, Items B4.50 and B4.60, and all other requirements in ASME BPV Code Case N–729–6 pertaining to peening, in order for a RPV upper head with nozzles and associated J-groove welds mitigated by peening to obtain examination relief from the requirements of Table 1 for unmitigated heads, peening must meet the performance criteria, qualification, and examination requirements stated in MRP–335, Revision 3–A, with the exception that a plant-specific alternative request is not required and NRC condition 5.4 of MRP–335, Revision 3–A does not apply.

(*6*) *Baseline Examinations.* In lieu of the requirements for Note 7(c) the baseline volumetric and surface examination for plants with a RPV Head with less than 8 EDY shall be performed by 2.25 reinspection years (RIY) after initial startup not to exceed 8 years.

(*7*) *Sister Plants.* Note 10 of ASME BPV Code Case N–729–6 shall not be implemented without prior NRC approval.

(*8*) *Volumetric Leak Path.* In lieu of paragraph 3200(b) requirement for a surface examination of the partial penetration weld, a volumetric leak path assessment of the nozzle may be performed in accordance with Note 6 of Table 1 of N–729–6.

(E) *Augmented ISI requirements: Reactor coolant pressure boundary visual inspections* [10]—(*1*) All licensees of pressurized water reactors must augment their inservice inspection program by implementing ASME Code Case N–722–1, subject to the conditions specified in paragraphs (g)(6)(ii)(E)(*2*) through (*4*) of this section. The inspection requirements of ASME Code Case N–722–1 do not apply to components with pressure retaining welds fabricated with Alloy 600/82/182 materials that have been mitigated by weld overlay or stress improvement.

(*2*) If a visual examination determines that leakage is occurring from a specific item listed in Table 1 of ASME Code Case N–722–1 that is not exempted by the ASME Code, Section XI, IWB–1220(b)(1), additional actions must be performed to characterize the location, orientation, and length of a crack or cracks in Alloy 600 nozzle wrought material and location, orientation, and length of a crack or cracks in Alloy 82/182 butt welds. Alternatively, licensees may replace the Alloy 600/82/182 materials in all the components under the item number of the leaking component.

(*3*) If the actions in paragraph (g)(6)(ii)(E)(*2*) of this section determine that a flaw is circumferentially oriented and potentially a result of primary water stress corrosion cracking, licensees must perform non-visual NDE inspections of components that fall under that ASME Code Case N–722–1 item number. The number of components inspected must equal or exceed the number of components found to be leaking under that item number. If circumferential cracking is identified in the sample, non-visual NDE must be performed in the remaining components under that item number.

(*4*) If ultrasonic examinations of butt welds are used to meet the NDE requirements in paragraphs (g)(6)(ii)(E)(*2*) or (*3*) of this section, they must be performed using the appropriate supplement of Section XI, Appendix VIII, of the ASME BPV Code.

(F) *Augmented ISI requirements: Examination requirements for Class 1 piping and nozzle dissimilar-metal butt welds*—(*1*) *Implementation.* Holders of operating licenses or combined licenses for pressurized-water reactors as of or after June 3, 2020, shall implement the requirements of ASME BPV Code Case N–770–5 instead of ASME BPV Code Case N–770–2, subject to the conditions specified in paragraphs (g)(6)(ii)(F)(*2*) through (*16*) of this section, by no later than one year after June 3, 2020. All NRC authorized alternatives from previous versions of paragraph (g)(6)(ii)(F) of this section remain applicable.

(*2*) *Categorization.* (*i*) Welds that have been mitigated by the Mechanical Stress Improvement Process (MSIP™) may be categorized as Inspection Items D or E, as appropriate, provided the criteria in Appendix I of the code case have been met.

(*ii*) In order to be categorized as peened welds, in lieu of inspection category L requirements and examinations, welds must meet the performance criteria, qualification and examination requirements as stated by MRP–335, Revision 3–A, with the exception that no plant-specific alternative is required.

(*iii*) Other mitigated welds shall be identified as the appropriate inspection item of the NRC authorized alternative or NRC-approved code case for the mitigation type in Regulatory Guide 1.147.

(*iv*) All other butt welds that rely on Alloy 82/182 for structural integrity shall be categorized as Inspection Items A–1, A–2, B–1 or B–2, as appropriate.

(*v*) Paragraph -1100(e) of ASME BPV Code Case N–770–5 shall not be used to exempt welds that rely on Alloy 82/182 for structural integrity from any requirement of this section.

(*3*) [Reserved]

(*4*) *Examination coverage.* When implementing Paragraph -2500(a) of ASME BPV Code Case N–770–5, essentially 100 percent of the required volumetric examination coverage shall be obtained, including greater than 90 percent of the volumetric examination coverage for circumferential flaws. Licensees are prohibited from using Paragraphs -2500(c) and -2500(d) of ASME BPV Code Case N–770–5 to meet examination requirements.

(*5*) *Inlay/onlay inspection frequency.* All hot-leg operating temperature welds in Inspection Items G, H, J, and K shall be inspected each inspection interval. A 25 percent sample of Inspection Items G, H, J, and K cold-leg operating temperature welds shall be inspected whenever the core barrel is removed (unless it has already been inspected within the past 10 years) or within 20 years, whichever is less.

(*6*) *Reporting requirements.* The licensee will promptly notify the NRC regarding any volumetric examination of a mitigated weld that detects growth of existing flaws in the required examination volume that exceed the previous IWB–3600 flaw evaluations, new flaws, or any indication in the weld overlay or excavate and weld repair material characterized as stress corrosion cracking. Additionally, the licensee will submit to the NRC a report summarizing the evaluation, along with inputs, methodologies, assumptions, and causes of the new flaw or flaw growth within 30 days following plant startup.

(*7*) *Defining ''t''.* For Inspection Items G, H, J, and K, when applying the acceptance standards of ASME BPV Code, Section XI, IWB–3514, for planar flaws contained within the inlay or onlay, the thickness ''t'' in IWB–3514 is the thickness of the inlay or onlay. For planar flaws in the balance of the dissimilar metal weld examination volume, the thickness ''t'' in IWB–3514 is the combined thickness of the inlay or onlay and the dissimilar metal weld.

(*8*) *Optimized weld overlay examination.* Initial inservice examination of Inspection Item C–2 welds shall be performed between the third refueling outage and no later than 10 years after application of the overlay.

(*9*) *Deferrals.* (*i*) The initial inservice volumetric examination of optimized weld overlays, Inspection Item C–2, shall not be deferred.

(*ii*) Volumetric inspection of peened dissimilar metal butt welds shall not be deferred.

(*iii*) For Inspection Item M–2, N–1 and N–2 welds, the second required inservice volumetric examination shall not be deferred.

(*10*) *Examination technique.* Note 14(b) of Table 1 and Note (b) of Figure 5(a) of ASME BPV Code Case N–770–5 may only be implemented if the requirements of Note 14(a) of Table 1 of ASME BPV Code Case N–770–5 cannot be met.

(*11*) [Reserved]

(*12*) *Stress improvement inspection coverage.* Under Paragraph I.5.1, for cast stainless steel items, the required examination volume shall be examined by Appendix VIII procedures to the maximum extent practical including 100 percent of the susceptible material volume.

(*13*) *Encoded ultrasonic examination.* Ultrasonic examinations of non-mitigated or cracked mitigated dissimilar metal butt welds in the reactor coolant pressure boundary must be performed in accordance with the requirements of

Table 1 for Inspection Item A–1, A–2, B–1, B–2, E, F–2, J, K, N–1, N–2 and O. Essentially 100 percent of the required inspection volume shall be examined using an encoded method.

(*14*) *Excavate and weld repair cold leg.* For cold leg temperature M–2, N–1 and N–2 welds, initial volumetric inspection after application of an excavate and weld repair (EWR) shall be performed during the second refueling outage.

(*15*) *Cracked excavate and weld repair.* In lieu of the examination requirements for cracked welds with 360 excavate and weld repairs, Inspection Item N–1 of Table 1, welds shall be examined during the first or second refueling outage following EWR. Examination volumes that show no indication of crack growth or new cracking shall be examined once each inspection interval thereafter.

(*16*) *Partial arc excavate and weld repair.* Inspection Item O cannot be used without NRC review and approval.

(h) *Protection and safety systems.* Protection systems of nuclear power reactors of all types must meet the requirements specified in this paragraph. Each combined license for a utilization facility is subject to the following conditions.

(1) [Reserved]

(2) *Protection systems.* For nuclear power plants with construction permits issued after January 1, 1971, but before May 13, 1999, protection systems must meet the requirements in IEEE Std 279–1968, ''Proposed IEEE Criteria for Nuclear Power Plant Protection Systems,'' or the requirements in IEEE Std 279–1971, ''Criteria for Protection Systems for Nuclear Power Generating Stations,'' or the requirements in IEEE Std 603–1991, ''Criteria for Safety Systems for Nuclear Power Generating Stations,'' and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, protection systems must be consistent with their licensing basis or may meet the requirements of IEEE Std. 603–1991 and the correction sheet dated January 30, 1995.

(3) *Safety systems.* Applications filed on or after May 13, 1999, for construction permits and operating licenses under this part, and for design approvals, design certifications, and combined licenses under part 52 of this chapter, must meet the requirements for safety systems in IEEE Std. 603–1991 and the correction sheet dated January 30, 1995.

(i)–(y) [Reserved]

(z) *Alternatives to codes and standards requirements.* Alternatives to the requirements of paragraphs (b) through (h) of this section or portions thereof may be used when authorized by the Director, Office of Nuclear Reactor Regulation. A proposed alternative must be submitted and authorized prior to implementation. The applicant or licensee must demonstrate that:

(1) *Acceptable level of quality and safety.* The proposed alternative would provide an acceptable level of quality and safety; or

(2) *Hardship without a compensating increase in quality and safety.* Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Footnotes to § 50.55a:

[1] USAS and ASME Code addenda issued prior to the winter 1977 Addenda are considered to be ''in effect'' or ''effective'' 6 months after their date of issuance and after they are incorporated by reference in paragraph (a) of this section. Addenda to the ASME Code issued after the summer 1977 Addenda are considered to be ''in effect'' or ''effective'' after the date of publication of the addenda and after they are incorporated by reference in paragraph (a) of this section.

[2–3] [Reserved]

[4] For ASME Code editions and addenda issued prior to the winter 1977 Addenda, the Code edition and addenda applicable to the component is governed by the order or contract date for the component, not the contract date for the nuclear energy system. For the winter 1977 Addenda and subsequent editions and addenda the method for determining the applicable Code editions and addenda is contained in Paragraph NCA 1140 of Section III of the ASME Code.

[5–6] [Reserved]

[7] Guidance for quality group classifications of components that are to be included in the safety analysis reports pursuant to § 50.34(a) and § 50.34(b) may be found in Regulatory Guide 1.26, ''Quality Group Classifications and Standards for Water-, Steam-, and Radiological-Waste-Containing Components of Nuclear Power Plants,'' and in Section 3.2.2 of NUREG–0800, ''Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants.''

[8–9] [Reserved]

[10] For inspections to be conducted once per interval, the inspections must be performed in accordance with the schedule in Section XI, paragraph IWB–2400, except for plants with inservice inspection programs based on a Section XI edition or addenda prior to the 1994 Addenda. For plants with inservice inspection programs based on a Section XI edition or addenda prior to the 1994 Addenda, the inspection must be performed in accordance with the schedule in Section XI, paragraph IWB–2400, of the 1994 Addenda.

[79 FR 65798, Nov. 5, 2014, as amended at 79 FR 66603, Nov. 10, 2014; 79 FR 73462, Dec. 11, 2014; 82 FR 52825, Dec. 15, 2017; 83 FR 2354, Jan. 17, 2018; 83 FR 2526, Jan. 18, 2018; 84 FR 65644, Nov. 29, 2019; 85 FR 14756, Mar. 16, 2020; 85 FR 26576, May 4, 2020; 85 FR 34088, June 3, 2020; 85 FR 65662, Oct. 16, 2020; 87 FR 11949, Mar. 3, 2022; 87 FR 65148, Oct. 27, 2022; 87 FR 73633, Dec. 1, 2022]

§ 50.56 Conversion of construction permit to license; or amendment of license.

Upon completion of the construction or alteration of a facility, in compliance with the terms and conditions of the construction permit and subject to any necessary testing of the facility for health or safety purposes, the Commission will, in the absence of good cause shown to the contrary, issue a license of the class for which the construction permit was issued or an appropriate amendment of the license, as the case may be.

[21 FR 355, Jan. 19, 1956, as amended at 35 FR 11461, July 17, 1970; 75 FR 73944, Nov. 30, 2010]

§ 50.57 Issuance of operating license. [1]

(a)Pursuant to § 50.56, an operating license may be issued by the Commission, up to the full term authorized by § 50.51, upon finding that:

(1) Construction of the facility has been substantially completed, in conformity with the construction permit and the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

(2) The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

(3) There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations in this chapter; and

(4) The applicant is technically and financially qualified to engage in the activities authorized by the operating license in accordance with the regulations in this chapter. However, no finding of financial qualification is necessary for an electric utility applicant for an operating license for a utilization facility of the type described in § 50.21(b) or § 50.22.

(5) The applicable provisions of part 140 of this chapter have been satisfied; and

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

(b) Each operating license will 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.

(c) An applicant may, in a case where a hearing is held in connection with a pending proceeding under this section make a motion in writing, under this paragraph (c), for an operating license authorizing low-power testing (operation at not more than 1 percent of full power for the purpose of testing the facility), and further operations short of full power operation. Action on such a motion by the presiding officer shall be taken with due regard to the rights of the parties to the proceedings, including the right of any party to be heard to the extent that his contentions are relevant to the activity to be authorized. Before taking any action on such a motion that any party opposes, the presiding officer shall make findings on the matters specified in paragraph (a) of this section as to which there is a

[1] The Commission may issue a provisional operating license pursuant to the regulations in this part in effect on March 30, 1970, for any facility for which a notice of hearing on an application for a provisional operating license or a notice of proposed issuance of a provisional operating license has been published on or before that date.

controversy, in the form of an initial decision with respect to the contested activity sought to be authorized. The Director of Nuclear Reactor Regulation will make findings on all other matters specified in paragraph (a) of this section. If no party opposes the motion, the presiding officer will issue an order in accordance with § 2.319(p) authorizing the Director of Nuclear Reactor Regulation to make appropriate findings on the matters specified in paragraph (a) of this section and to issue a license for the requested operation.

[35 FR 5318, Mar. 31, 1970, as amended at 35 FR 6644, Apr. 25, 1970; 37 FR 11873, June 15, 1972; 37 FR 15142, July 28, 1972; 49 FR 35753, Sept. 12, 1984; 51 FR 7765, Mar. 6, 1986; 69 FR 2275, Jan. 14, 2004]

§ 50.58 Hearings and report of the Advisory Committee on Reactor Safeguards.

(a) Each application for a construction permit or an operating license for a facility which is of a type described in § 50.21(b) or § 50.22, or for a testing facility, shall be referred to the Advisory Committee on Reactor Safeguards for a review and report. An application for an amendment to such a construction permit or operating license may be referred to the Advisory Committee on Reactor Safeguards for review and report. Any report shall be made part of the record of the application and available to the public, except to the extent that security classification prevents disclosure.

(b)(1) The Commission will hold a hearing after at least 30-days' notice and publication once in the FEDERAL REGISTER on each application for a construction permit for a production or utilization facility which is of a type described in § 50.21(b) or § 50.22, or for a testing facility.

(2) When a construction permit has been issued for such a facility following the holding of a public hearing, and an application is made for an operating license or for an amendment to a construction permit or operating license, the Commission may hold a hearing after at least 30-days' notice and publication once in the FEDERAL REGISTER, or, in the absence of a request therefor by any person whose interest may be affected, may issue an operating license or an amendment to a construction permit or operating license without a hearing, upon 30-days' notice and publication once in the FEDERAL REGISTER of its intent to do so.

(3) If the Commission finds, in an emergency situation, as defined in § 50.91, that no significant hazards consideration is presented by an application for an amendment to an operating license, it may dispense with public notice and comment and may issue the amendment. If the Commission finds that exigent circumstances exist, as described in § 50.91, it may reduce the period provided for public notice and comment.

(4) Both in an emergency situation and in the case of exigent circumstances, the Commission will provide 30 days notice of opportunity for a hearing, though this notice may be published after issuance of the amendment if the Commission determines that no significant hazards consideration is involved.

(5) The Commission will use the standards in § 50.92 to determine whether a significant hazards consideration is presented by an amendment to an operating license for a facility of the type described in § 50.21(b) or § 50.22, or which is a testing facility, and may make the amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

(6) No petition or other request for review of or hearing on the staff's significant hazards consideration determination will be entertained by the Commission. The staff's determination is final, subject only to the Commission's discretion, on its own initiative, to review the determination.

[27 FR 12186, Dec. 8, 1962, as amended at 35 FR 11461, July 17, 1970; 39 FR 10555, Mar. 21, 1974; 51 FR 7765, Mar. 6, 1986]

§ 50.59 Changes, tests, and experiments.

(a) Definitions for the purposes of this section:

(1) *Change* means a modification or addition to, or removal from, the facility or procedures that affects a design 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 FSAR (as updated) used in establishing the design bases or in the safety analyses* means:

(i) Changing any of the elements of the method described in the FSAR (as updated) 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 final safety analysis report (as updated)* means:

(i) The structures, systems, and components (SSC) that are described in the final safety analysis report (FSAR) (as updated),

(ii) The design and performance requirements for such SSCs described in the FSAR (as updated), and

(iii) The evaluations or methods of evaluation included in the FSAR (as updated) for such SSCs which demonstrate that their intended function(s) will be accomplished.

(4) *Final Safety Analysis Report (as updated)* means the Final Safety Analysis Report (or Final Hazards Summary Report) submitted in accordance with § 50.34, as amended and supplemented, and as updated per the requirements of § 50.71(e) or § 50.71(f), as applicable.

(5) *Procedures as described in the final safety analysis report (as updated)* means those procedures that contain information described in the FSAR (as updated) such as how structures, systems, and components are operated and controlled (including assumed operator actions and response times).

(6) *Tests or experiments not described in the final safety analysis report (as updated)* means any activity where any structure, system, or component is utilized or controlled in a manner which is either:

(i) Outside the reference bounds of the design bases as described in the final safety analysis report (as updated) or

(ii) Inconsistent with the analyses or descriptions in the final safety analysis report (as updated).

(b) This section applies to each holder of an operating license issued under this part or a combined license issued under part 52 of this chapter, including the holder of a license authorizing operation of a nuclear power reactor that has submitted the certification of permanent cessation of operations required under § 50.82(a)(1) or § 50.110 or a reactor licensee whose license has been amended to allow possession of nuclear fuel but not operation of the facility.

(c)(1) A licensee may make changes in the facility as described in the final safety analysis report (as updated), make changes in the procedures as described in the final safety analysis report (as updated), and conduct tests or experiments not described in the final safety analysis report (as updated) without obtaining a license amendment pursuant to § 50.90 only if:

(i) A change to the technical specifications incorporated in the license is not required, and

(ii) The change, test, or experiment does not meet any of the criteria in paragraph (c)(2) of this section.

(2) A licensee shall obtain a license amendment pursuant to § 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would:

(i) Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the final safety analysis report (as updated);

(ii) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the final safety analysis report (as updated);

(iii) Result in more than a minimal increase in the consequences of an accident previously evaluated in the final safety analysis report (as updated);

(iv) Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the final safety analysis report (as updated);

(v) Create a possibility for an accident of a different type than any previously evaluated in the final safety analysis report (as updated);

(vi) Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the final safety analysis report (as updated);

(vii) Result in a design basis limit for a fission product barrier as described in the FSAR (as updated) being exceeded or altered; or

(viii) Result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses.

(3) In implementing this paragraph, the FSAR (as updated) is considered to include FSAR changes resulting from evaluations performed pursuant to this section and analyses performed pursuant to § 50.90 since submittal of the last update of the final safety analysis report pursuant to § 50.71 of this part.

(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.

(d)(1) The licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments made pursuant to paragraph (c) of this section. These records must include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license amendment pursuant to paragraph (c)(2) of this section.

(2) The licensee shall submit, as specified in § 50.4 or § 52.3 of this chapter, as applicable, a report containing a brief description of any changes, tests, and experiments, 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 52.103(g).

(3) The records of changes in the facility must be maintained until the termination of an operating license issued under this part, a combined license issued under part 52 of this chapter, or the termination of a license issued under 10 CFR part 54, whichever is later. Records of changes in procedures and records of tests and experiments must be maintained for a period of 5 years.

[64 FR 53613, Oct. 4, 1999, as amended at 66 FR 64738, Dec. 14, 2001; 72 FR 49500, Aug. 28, 2007]

§ 50.60 Acceptance criteria for fracture prevention measures for lightwater 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) have been submitted, must meet 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.

[48 FR 24009, May 27, 1983, as amended at 50 FR 50777, Dec. 12, 1985; 61 FR 39300, July 29, 1996]

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

(a) *Definitions*. For the purposes of this section:

(1) *ASME Code* means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division I, ''Rules for the Construction of Nuclear Power Plant Components,'' edition and addenda and any limitations and modifications thereof as specified in § 50.55a.

(2) *Pressurized Thermal Shock Event* means an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel.

(3) *Reactor Vessel Beltline* means the region of the reactor vessel (shell material including welds, heat affected

zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

(4) RT_{NDT} means the reference temperature for a reactor vessel material, under any conditions. For the reactor vessel beltline materials, RT_{NDT} must account for the effects of neutron radiation.

(5) $RT_{NDT(U)}$ means the reference temperature for a reactor vessel material in the pre-service or unirradiated condition, evaluated according to the procedures in the ASME Code, Paragraph NB–2331 or other methods approved by the Director, Office of Nuclear Reactor Regulation.

(6) *EOL Fluence* means the best-estimate neutron fluence projected for a specific vessel beltline material at the clad-base-metal interface on the inside surface of the vessel at the location where the material receives the highest fluence on the expiration date of the operating license.

(7) RT_{PTS} means the reference temperature, RT_{NDT}, evaluated for the EOL Fluence for each of the vessel beltline materials, using the procedures of paragraph (c) of this section.

(8) *PTS Screening Criterion* means the value of RT_{PTS} for the vessel beltline material above which the plant cannot continue to operate without justification.

(b) *Requirements.* (1) For each pressurized water nuclear power reactor for which an operating license has been issued under this part or a combined license issued under Part 52 of this chapter, other than a nuclear power reactor facility for which the certification required under § 50.82(a)(1) 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 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, 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) 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.

(2) The pressurized thermal shock (PTS) screening criterion is 270 °F for plates, forgings, and axial weld materials, and 300 °F for circumferential weld materials. For the purpose of comparison with this criterion, the value of RT_{PTS} for the reactor vessel must be evaluated according to the procedures of paragraph (c) of this section, for each weld and plate, or forging, in the reactor vessel beltline. RT_{PTS} must be determined for each vessel beltline material using the EOL fluence for that material.

(3) For each pressurized water nuclear power reactor for which the value of RT_{PTS} for any material in the beltline is projected to exceed the PTS screening criterion using the EOL fluence, the licensee shall implement those flux reduction programs that are reasonably practicable to avoid exceeding the PTS screening criterion set

[2] Changes to RT_{PTS} values are considered significant if either the previous value or the current value, or both values, exceed the screening criterion before the expiration of the operating license or the combined license under Part 52 of this chapter, including any renewed term, if applicable for the plant.

forth in paragraph (b)(2) of this section. The schedule for implementation of flux reduction measures may take into account the schedule for submittal and anticipated approval by the Director, Office of Nuclear Reactor Regulation, of detailed plant-specific analyses, submitted to demonstrate acceptable risk with RT_{PTS} above the screening limit due to plant modifications, new information or new analysis techniques.

(4) For each pressurized water nuclear power reactor for which the analysis required by paragraph (b)(3) of this section indicates that no reasonably practicable flux reduction program will prevent RT_{PTS} from exceeding the PTS screening criterion using the EOL fluence, the licensee shall submit a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criterion is allowed. In the analysis, the licensee may determine the properties of the reactor vessel materials based on available information, research results, and plant surveillance data, and may use probabilistic fracture mechanics techniques. This analysis must be submitted at least three years before RT_{PTS} is projected to exceed the PTS screening criterion.

(5) After consideration of the licensee's analyses, including effects of proposed corrective actions, if any, submitted in accordance with paragraphs (b)(3) and (b)(4) of this section, the Director, Office of Nuclear Reactor Regulation, may, on a case-by-case basis, approve operation of the facility with RT_{PTS} in excess of the PTS screening criterion. The Director, Office of Nuclear Reactor Regulation, will consider factors significantly affecting the potential for failure of the reactor vessel in reaching a decision.

(6) If the Director, Office of Nuclear Reactor Regulation, concludes, pursuant to paragraph (b)(5) of this section, that operation of the facility with RT_{PTS} in excess of the PTS screening criterion cannot be approved on the basis of the licensee's analyses submitted in accordance with paragraphs (b)(3) and (b)(4) of this section, the licensee shall request and receive approval by the Director, Office of Nuclear Reactor Regulation, prior to any operation beyond the criterion. The request must be based upon modifications to equipment, systems, and operation of the facility in addition to those previously proposed in the submitted analyses that would reduce the potential for failure of the reactor vessel due to PTS events, or upon further analyses based upon new information or improved methodology.

(7) If the limiting RT_{PTS} value of the plant is projected to exceed the screening criteria in paragraph (b)(2), or the criteria in paragraphs (b)(3) through (b)(6) of this section cannot be satisfied, the reactor vessel beltline may be given a thermal annealing treatment to recover the fracture toughness of the material, subject to the requirements of § 50.66. The reactor vessel may continue to be operated only for that service period within which the predicted fracture toughness of the vessel beltline materials satisfy the requirements of paragraphs (b)(2) through (b)(6) of this section, with RT_{PTS} accounting for the effects of annealing and subsequent irradiation.

(c) *Calculation of RT_{PTS}.* RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence for the material. RT_{PTS} must be evaluated using the same procedures used to calculate RT_{NDT}, as indicated in paragraph (c)(1) of this section, and as provided in paragraphs (c)(2) and (c)(3) of this section.

(1) Equation 1 must be used to calculate values of RT_{NDT} for each weld and plate, or forging, in the reactor vessel beltline.

Equation 1: $RT_{NDT} = RT_{NDT(U)} + M + \Delta RT_{NDT}$

(i) If a measured value of $RT_{NDT(U)}$ is not available, a generic mean value for the class[3] of material may be used if there are sufficient test results to establish a mean and a standard deviation for the class.

---

[3] The class of material for estimating $RT_{NDT(U)}$ is generally determined for welds by the type of welding flux (Linde 80, or other), and for base metal by the material specification.

(ii) For generic values of weld metal, the following generic mean values must be used unless justification for different values is provided: 0 °F for welds made with Linde 80 flux, and −56 °F for welds made with Linde 0091, 1092 and 124 and ARCOS B–5 weld fluxes.

(iii) *M* means the margin to be added to account for uncertainties in the values of $RT_{NDT(U)}$, copper and nickel contents, fluence and the calculational procedures. M is evaluated from Equation 2.

Equation 2: $$M = 2\sqrt{\sigma_U^2 + \sigma_\Delta^2}$$

(A) In Equation 2, σ_U is the standard deviation for $RT_{NDT(U)}$. If a measured value of $RT_{NDT(U)}$ is used, then σ_U is determined from the precision of the test method. If a measured value of $RT_{NDT(U)}$ is not available and a generic mean value for that class of materials is used, then σ_U is the standard deviation obtained from the set of data used to establish the mean. If a generic mean value given in paragraph (c)(1)(i)(B) of this section for welds is used, then σ_U is 17 °F.

(B) In Equation 2, σ_Δ is the standard deviation for ΔRT_{NDT}. The value of σ_Δ to be used is 28 °F for welds and 17 °F for base metal; the value of σ_Δ need not exceed one-half of ΔRT_{NDT}.

(iv) ΔRT_{NDT} is the mean value of the transition temperature shift, or change in RT_{NDT}, due to irradiation, and must be calculated using Equation 3.

Equation 3: $\Delta RT_{NDT} = (CF)f^{(0.28-0.10 \log f)}$

(A) *CF* (°F) is the chemistry factor, which is a function of copper and nickel content. CF is given in table 1 for welds and in table 2 for base metal (plates and forgings). Linear interpolation is permitted. In tables 1 and 2, ''Wt − % copper'' and ''Wt − % nickel'' are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging. For a weld, the best estimate values will normally be the mean of the measured values for a weld deposit made using the same weld wire heat number as the critical vessel weld. If these values are not available, the upper limiting values given in the material specifications to which the vessel material was fabricated may be used. If not available, conservative estimates (mean plus one standard deviation) based on generic data[4] may be used if justification is provided. If none of these alternatives are available, 0.35% copper and 1.0% nickel must be assumed.

(B) *f* is the best estimate neutron fluence, in units of 10^{19} n/cm^2 (E greater than 1 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence for the period of service in question. As specified in this paragraph, the EOL fluence for the vessel beltline material is used in calculating KRT_{PTS}.

(v) Equation 4 must be used for determining RT_{PTS} using equation 3 with EOL fluence values for determining ΔRT_{PTS}.

Equation 4: $RT_{PTS} = RT_{NDT(U)} + M + \Delta RT_{PTS}$

(2) To verify that RT_{NDT} for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and any related surveillance program[5] results.

(i) Results from the plant-specific surveillance program must be integrated into the RT_{NDT} estimate if the plant-specific surveillance data has been deemed credible as judged by the following criteria:

(A) The materials in the surveillance capsules must be those which are the controlling materials with regard to radiation embrittlement.

(B) Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions must

[4] Data from reactor vessels fabricated to the same material specification in the same shop as the vessel in question and in the same time period is an example of ''generic data.''

[5] Surveillance program results means any data that demonstrates the embrittlement trends for the limiting beltline material, including but not limited to data from test reactors or from surveillance programs at other plants with or without surveillance program integrated per 10 CFR part 50, appendix H.

be small enough to permit the determination of the 30-foot-pound temperature unambiguously.

(C) Where there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values must be less than 28 °F for welds and 17 °F for base metal. Even if the range in the capsule fluences is large (two or more orders of magnitude), the scatter may not exceed twice those values.

(D) The irradiation temperature of the Charpy specimens in the capsule must equal the vessel wall temperature at the cladding/base metal interface within ±25 °F.

(E) The surveillance data for the correlation monitor material in the capsule, if present, must fall within the scatter band of the data base for the material.

(ii)(A) Surveillance data deemed credible according to the criteria of paragraph (c)(2)(i) of this section must be used to determine a material-specific value of CF for use in Equation 3. A material-specific value of CF is determined from Equation 5.

$$\text{Equation 5:}\quad CF = \frac{\sum_{i=1}^{n}\left[A_i \times f_i^{\left(0.28-0.10 \log f_i\right)}\right]}{\sum_{i=1}^{n}\left[f_i^{\left(0.56-0.20 \log f_i\right)}\right]}$$

(B) In Equation 5, ''n'' is the number of surveillance data points, ''A_i'' is the measured value of ΔRT_{NDT} and ''f_i'' is the fluence for each surveillance data point. If there is clear evidence that the copper and nickel content of the surveillance weld differs from the vessel weld, *i.e.*, differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld, the measured values of ΔRT_{NDT} must be adjusted for differences in copper and nickel content by multiplying them by the ratio of the chemistry factor for the vessel material to that for the surveillance weld.

(iii) For cases in which the results from a credible plant-specific surveillance program are used, the value of σ_Δ to be used is 14 °F for welds and 8.5 °F for base metal; the value of σ_Δ need not exceed one-half of ΔRT_{NDT}.

(iv) The use of results from the plant-specific surveillance program may result in an RT_{NDT} that is higher or lower than those determined in paragraph (c)(1).

(3) Any information that is believed to improve the accuracy of the RT_{PTS} value significantly must be reported to the Director, Office of Nuclear Reactor Regulation. Any value of RT_{PTS} that has been modified using the procedures of paragraph (c)(2) of this section is subject to the approval of the Director, Office of Nuclear Reactor Regulation, when used as provided in this section.

TABLE 1—CHEMISTRY FACTOR FOR WELD METALS, °F

Copper, wt-%	Nickel, wt-%						
	0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	21	26	27	27	27	27	27
0.03	22	35	41	41	41	41	41
0.04	24	43	54	54	54	54	54
0.05	26	49	67	68	68	68	68
0.06	29	52	77	82	82	82	82
0.07	32	55	85	95	95	95	95
0.08	36	58	90	106	108	108	108
0.09	40	61	94	115	122	122	122
0.10	44	65	97	122	133	135	135
0.11	49	68	101	130	144	148	148
0.12	52	72	103	135	153	161	161
0.13	58	76	106	139	162	172	176
0.14	61	79	109	142	168	182	188
0.15	66	84	112	146	175	191	200
0.16	70	88	115	149	178	199	211
0.17	75	92	119	151	184	207	221
0.18	79	95	122	154	187	214	230
0.19	83	100	126	157	191	220	238
0.20	88	104	129	160	194	223	245
0.21	92	108	133	164	197	229	252
0.22	97	112	137	167	200	232	257
0.23	101	117	140	169	203	236	263
0.24	105	121	144	173	206	239	268
0.25	110	126	148	176	209	243	272
0.26	113	130	151	180	212	246	276
0.27	119	134	155	184	216	249	280
0.28	122	138	160	187	218	251	284
0.29	128	142	164	191	222	254	287
0.30	131	146	167	194	225	257	290
0.31	136	151	172	198	228	260	293

TABLE 1—CHEMISTRY FACTOR FOR WELD METALS, °F—Continued

Copper, wt-%	Nickel, wt-%						
	0	0.20	0.40	0.60	0.80	1.00	1.20
0.32	140	155	175	202	231	263	296
0.33	144	160	180	205	234	266	299
0.34	149	164	184	209	238	269	302
0.35	153	168	187	212	241	272	305
0.36	158	172	191	216	245	275	308
0.37	162	177	196	220	248	278	311
0.38	166	182	200	223	250	281	314
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

TABLE 2—CHEMISTRY FACTOR FOR BASE METALS, °F

Copper, wt-%	Nickel, wt-%						
	0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	20	20	20	20	20	20	20
0.03	20	20	20	20	20	20	20
0.04	22	26	26	26	26	26	26
0.05	25	31	31	31	31	31	31
0.06	28	37	37	37	37	37	37
0.07	31	43	44	44	44	44	44
0.08	34	48	51	51	51	51	51
0.09	37	53	58	58	58	58	58
0.10	41	58	65	65	67	67	67
0.11	45	62	72	74	77	77	77
0.12	49	67	79	83	86	86	86
0.13	53	71	85	91	96	96	96
0.14	57	75	91	100	105	106	106
0.15	61	80	99	110	115	117	117
0.16	65	84	104	118	123	125	125
0.17	69	88	110	127	132	135	135
0.18	73	92	115	134	141	144	144
0.19	78	97	120	142	150	154	154
0.20	82	102	125	149	159	164	165
0.21	86	107	129	155	167	172	174
0.22	91	112	134	161	176	181	184
0.23	95	117	138	167	184	190	194
0.24	100	121	143	172	191	199	204
0.25	104	126	148	176	199	208	214
0.26	109	130	151	180	205	216	221
0.27	114	134	155	184	211	225	230
0.28	119	138	160	187	216	233	239
0.29	124	142	164	191	221	241	248
0.30	129	146	167	194	225	249	257
0.31	134	151	172	198	228	255	266
0.32	139	155	175	202	231	260	274
0.33	144	160	180	205	234	264	282
0.34	149	164	184	209	238	268	290
0.35	153	168	187	212	241	272	298
0.36	158	173	191	216	245	275	303
0.37	162	177	196	220	248	278	308
0.38	166	182	200	223	250	281	313
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

[60 FR 65468, Dec. 19, 1995, as amended at 61 FR 39300, July 29, 1996; 72 FR 49500, Aug. 28, 2007; 73 FR 5722, Jan. 31, 2008; 75 FR 23, Jan. 4, 2010; 84 FR 65644, Nov. 29, 2019]

§ 50.61a Alternate fracture toughness requirements for protection against pressurized thermal shock events.

(a) *Definitions.* Terms in this section have the same meaning as those presented in 10 CFR 50.61(a), with the exception of the term ''ASME Code.''

(1) *ASME Code* means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division I, ''Rules for the Construction of Nuclear Power Plant Components,'' and Section XI, Division I, ''Rules for Inservice Inspection of Nuclear Power Plant Components,'' edition and addenda and any limitations and modifications thereof as specified in § 50.55a.

(2) RT_{MAX-AW} means the material property which characterizes the reactor vessel's resistance to fracture initiating from flaws found along axial weld fusion lines. RT_{MAX-AW} is determined under the provisions of paragraph (f) of this section and has units of °F.

(3) RT_{MAX-PL} means the material property which characterizes the reactor vessel's resistance to fracture initiating from flaws found in plates in regions that are not associated with welds found in plates. RT_{MAX-PL} is determined under the provisions of paragraph (f) of this section and has units of °F.

(4) RT_{MAX-FO} means the material property which characterizes the reactor vessel's resistance to fracture initiating from flaws in forgings that are not associated with welds found in forgings. RT_{MAX-FO} is determined under the provisions of paragraph (f) of this section and has units of °F.

(5) RT_{MAX-CW} means the material property which characterizes the reactor vessel's resistance to fracture initiating from flaws found along the circumferential weld fusion lines. RT_{MAX-CW} is determined under the provisions of paragraph (f) of this section and has units of °F.

(6) RT_{MAX-X} means any or all of the material properties RT_{MAX-AW}, RT_{MAX-PL}, RT_{MAX-FO}, RT_{MAX-CW}, or sum of RT_{MAX-AW} and RT_{MAX-PL}, for a particular reactor vessel.

(7) φt means fast neutron fluence for neutrons with energies greater than 1.0 MeV. φt is utilized under the provisions of paragraph (g) of this section and has units of n/cm^2.

(8) φ means average neutron flux for neutrons with energies greater than 1.0 MeV. φ is utilized under the provisions of paragraph (g) of this section and has units of n/cm^2/sec.

(9) ΔT_{30} means the shift in the Charpy V-notch transition temperature at the 30 ft-lb energy level produced by irradiation. The ΔT_{30} value is utilized under the provisions of paragraph (g) of this section and has units of °F.

(10) *Surveillance data* means any data that demonstrates the embrittlement trends for the beltline materials, including, but not limited to, surveillance programs at other plants with or without a surveillance program integrated under 10 CFR part 50, appendix H.

(11) T_C means cold leg temperature under normal full power operating conditions, as a time-weighted average from the start of full power operation through the end of licensed operation. T_C has units of °F.

(12) *CRP* means the copper rich precipitate term in the embrittlement model from this section. The CRP term is defined in paragraph (g) of this section.

(13) *MD* means the matrix damage term in the embrittlement model for this section. The MD term is defined in paragraph (g) of this section.

(b) *Applicability*. The requirements of this section apply to each holder of an operating license for a pressurized water nuclear power reactor whose construction permit was issued before February 3, 2010 and whose reactor vessel was designed and fabricated to the ASME Boiler and Pressure Vessel Code, 1998 Edition or earlier. The requirements of this section may be implemented as an alternative to the requirements of 10 CFR 50.61.

(c) *Request for approval.* Before the implementation of this section, each licensee shall submit a request for approval in the form of an application for a license amendment in accordance with § 50.90 together with the documentation required by paragraphs (c)(1), (c)(2), and (c)(3) of this section for review and approval by the Director of the Office of Nuclear Reactor Regulation (Director). The application must be submitted for review and approval by the Director at least three years before the limiting RT_{PTS} value calculated under 10 CFR 50.61 is projected to exceed the PTS screening criteria in 10 CFR 50.61 for plants licensed under this part.

(1) Each licensee shall have projected values of RT_{MAX-X} for each reactor vessel beltline material for the EOL fluence of the material. The assessment of RT_{MAX-X} values must use the calculation procedures given in paragraphs (f) and (g) of this section. The assessment must specify the bases for the projected value of RT_{MAX-X} for each reactor vessel beltline material, including the assumptions regarding future plant operation (e.g., core loading patterns, projected capacity factors); the copper (Cu), phosphorus (P), manganese (Mn), and nickel (Ni) contents; the reactor cold leg temperature (T_C); and the neutron flux and fluence values used in the calculation for each beltline material. Assessments performed under paragraphs (f)(6) and (f)(7) of this section, shall be submitted by the licensee to the Director in its license amendment application to utilize § 50.61a.

(2) Each licensee shall perform an examination and an assessment of flaws in the reactor vessel beltline as required by paragraph (e) of this section. The licensee shall verify that the requirements of paragraphs (e), (e)(1), (e)(2), and (e)(3) of this section have been met. The licensee must submit to the Director, in its application to use § 50.61a, the adjustments made to the volumetric test data to account for NDE-related uncertainties as described in paragraph (e)(1) of this section, all information required by paragraph (e)(1)(iii) of this section, and, if applicable, analyses performed under paragraphs (e)(4), (e)(5) and (e)(6) of this section.

(3) Each licensee shall compare the projected RT_{MAX-X} values for plates, forgings, axial welds, and circumferential welds to the PTS screening criteria in Table 1 of this section, for the purpose of evaluating a reactor vessel's susceptibility to fracture due to a PTS event. If any of the projected RT_{MAX-X} values are greater than the PTS screening criteria in Table 1 of this section, then the licensee may propose the compensatory actions or plant-specific

analyses as required in paragraphs (d)(3) through (d)(7) of this section, as applicable, to justify operation beyond the PTS screening criteria in Table 1 of this section.

(d) *Subsequent requirements.* Licensees who have been approved to use 10 CFR 50.61a under the requirements of paragraph (c) of this section shall comply with the requirements of this paragraph.

(1) Whenever there is a significant change in projected values of RT_{MAX-X}, so that the previous value, the current value, or both values, exceed the screening criteria before the expiration of the plant operating license; or upon the licensee's request for a change in the expiration date for operation of the facility; a re-assessment of RT_{MAX-X} values documented consistent with the requirements of paragraph (c)(1) and (c)(3) of this section must be submitted in the form of a license amendment for review and approval by the Director. If the surveillance data used to perform the re-assessment of RT_{MAX-X} values meet the requirements of paragraph (f)(6)(v) of this section, the licensee shall submit the data and the results of the analysis of the data to the Director for review and approval within one year after the capsule is withdrawn from the vessel. If the surveillance data meet the requirements of paragraph (f)(6)(vi) of this section, the licensee shall submit the data, the results of the analysis of the data, and proposed ΔT_{30} and RT_{MAX-X} values considering the surveillance data in the form of a license amendment to the Director for review and approval within two years after the capsule is withdrawn from the vessel. If the Director does not approve the assessment of RT_{MAX-X} values, then the licensee shall perform the actions required in paragraphs (d)(3) through (d)(7) of this section, as necessary, before operation beyond the PTS screening criteria in Table 1 of this section.

(2) The licensee shall verify that the requirements of paragraphs (e), (e)(1), (e)(2), and (e)(3) of this section have been met. The licensee must submit, within 120 days after completing a volumetric examination of reactor vessel beltline materials as required by ASME Code, Section XI, the adjustments made to the volumetric test data to account for NDE-related uncertainties as described in paragraph (e)(1) of this section and all information required by paragraph (e)(1)(iii) of this section in the form of a license amendment for review and approval by the Director. If a licensee is required to implement paragraphs (e)(4), (e)(5), and (e)(6) of this section, the information required in these paragraphs must be submitted in the form of a license amendment for review and approval by the Director within one year after completing a volumetric examination of reactor vessel materials as required by ASME Code, Section XI.

(3) If the value of RT_{MAX-X} is projected to exceed the PTS screening criteria, then the licensee shall implement those flux reduction programs that are reasonably practicable to avoid exceeding the PTS screening criteria. The schedule for implementation of flux reduction measures may take into account the schedule for review and anticipated approval by the Director of detailed plant-specific analyses which demonstrate acceptable risk with RT_{MAX-X} values above the PTS screening criteria due to plant modifications, new information, or new analysis techniques.

(4) If the analysis required by paragraph (d)(3) of this section indicates that no reasonably practicable flux reduction program will prevent the RT_{MAX-X} value for one or more reactor vessel beltline materials from exceeding the PTS screening criteria, then the licensee shall perform a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent the potential for an unacceptably high probability of failure of the reactor vessel as a result of postulated PTS events. In the analysis, the licensee may determine the properties of the reactor vessel materials based on available information, research results and plant surveillance data, and may use probabilistic fracture mechanics techniques. This analysis and the description of the modifications must be submitted to the Director in the form of a license amendment at least three years before RT_{MAX-X} is projected to exceed the PTS screening criteria.

(5) After consideration of the licensee's analyses, including effects of proposed corrective actions, if any, submitted under paragraphs (d)(3) and (d)(4) of this section, the Director may, on a case-by-case basis, approve operation of the facility with RT_{MAX-X} values in excess of the PTS screening criteria. The Director will consider factors significantly affecting the potential for failure of the reactor vessel in reaching a decision. The Director shall impose the modifications to equipment, systems and operations described to meet paragraph (d)(4) of this section.

(6) If the Director concludes, under paragraph (d)(5) of this section, that operation of the facility with RT_{MAX-X} values in excess of the PTS screening criteria cannot be approved on the basis of the licensee's analyses submitted under paragraphs (d)(3) and (d)(4) of this section, then the licensee shall request a license amendment, and receive approval by the Director, before any operation beyond the PTS screening criteria. The request must be based on modifications to equipment, systems, and operation of the facility in addition to those previously proposed in the submitted analyses that would reduce the potential for failure of the reactor vessel due to PTS events, or on further analyses based on new information or improved methodology. The licensee must show that the proposed alternatives provide reasonable assurance of adequate protection of the public health and safety.

(7) If the limiting RT_{MAX-X} value of the facility is projected to exceed the PTS screening criteria and the requirements of paragraphs (d)(3) through (d)(6) of this section cannot be satisfied, the reactor vessel beltline may be given a thermal annealing treatment under the requirements of § 50.66 to recover the fracture toughness of the material. The reactor vessel may be used only for that service period within which the predicted fracture toughness of the reactor vessel beltline materials satisfy the requirements of paragraphs (d)(1) through (d)(6) of this section, with RT_{MAX-X} values accounting for the effects of annealing and subsequent irradiation.

(e) *Examination and flaw assessment requirements.* The volumetric examination results evaluated under paragraphs (e)(1), (e)(2), and (e)(3) of this section must be acquired using procedures, equipment and personnel that have been qualified under the ASME Code, Section XI, Appendix VIII, Supplement 4 and Supplement 6, as specified in 10 CFR 50.55a(b)(2)(xv).

(1) The licensee shall verify that the flaw density and size distributions within the volume described in ASME Code, Section XI,[1] Figures IWB–2500–1 and IWB–2500–2 and limited to a depth from the clad-to-base metal interface of 1-inch or 10 percent of the vessel thickness, whichever is greater, do not exceed the limits in Tables 2 and 3 of this section based on the test results from the volumetric examination. The values in Tables 2 and 3 represent actual flaw sizes. Test results from the volumetric examination may be adjusted to account for the effects of NDE-related uncertainties. The methodology to account for NDE-related uncertainties must be based on statistical data from the qualification tests and any other tests that measure the difference between the actual flaw size and the NDE detected flaw size. Licensees who adjust their test data to account for NDE-related uncertainties to verify conformance with the values in Tables 2 and 3 shall prepare and submit the methodology used to estimate the NDE uncertainty, the statistical data used to adjust the test data and an explanation of how the data was analyzed for review and approval by the Director in accordance with paragraphs (c)(2) and (d)(2) of this section. The verification of the flaw density and size distributions shall be performed line-by-line for Tables 2 and 3. If the flaw density and size distribution exceeds the limitations specified in Tables 2 and 3 of this section, the licensee shall perform the analyses required by paragraph (e)(4) of this section. If analyses are required in accordance with paragraph (e)(4) of this section, the licensee must address the effects on through-

[1] For forgings susceptible to underclad cracking the determination of the flaw density for that forging from the licensee's inspection shall exclude those indications identified as underclad cracks.

wall crack frequency (TWCF) in accordance with paragraph (e)(5) of this section and must prepare and submit a neutron fluence map in accordance with the requirements of paragraph (e)(6) of this section.

(i) The licensee shall determine the allowable number of weld flaws in the reactor vessel beltline by multiplying the values in Table 2 of this section by the total length of the reactor vessel beltline welds that were volumetrically inspected and dividing by 1000 inches of weld length.

(ii) The licensee shall determine the allowable number of plate or forging flaws in their reactor vessel beltline by multiplying the values in Table 3 of this section by the total surface area of the reactor vessel beltline plates or forgings that were volumetrically inspected and dividing by 1000 square inches.

(iii) For each flaw detected in the inspection volume described in paragraph (e)(1) with a through-wall extent equal to or greater than 0.075 inches, the licensee shall document the dimensions of the flaw, including through-wall extent and length, whether the flaw is axial or circumferential in orientation and its location within the reactor vessel, including its azimuthal and axial positions and its depth embedded from the clad-to-base metal interface.

(2) The licensee shall identify, as part of the examination required by paragraph (c)(2) of this section and any subsequent ASME Code, Section XI ultrasonic examination of the beltline welds, any flaws within the inspection volume described in paragraph (e)(1) of this section that are equal to or greater than 0.075 inches in through-wall depth, axially-oriented, and located at the clad-to-base metal interface. The licensee shall verify that these flaws do not open to the vessel inside surface using surface or visual examination technique capable of detecting and characterizing service induced cracking of the reactor vessel cladding.

(3) The licensee shall verify, as part of the examination required by paragraph (c)(2) of this section and any subsequent ASME Code, Section XI ultrasonic examination of the beltline welds, that all flaws between the clad-to-base metal interface and three-eights of the reactor vessel thickness from the interior surface are within the allowable values in ASME Code, Section XI, Table IWB–3510–1.

(4) The licensee shall perform analyses to demonstrate that the reactor vessel will have a TWCF of less than 1×10^{-6} per reactor year if the ASME Code, Section XI volumetric examination required by paragraph (c)(2) or (d)(2) of this section indicates any of the following:

(i) The flaw density and size in the inspection volume described in paragraph (e)(1) exceed the limits in Tables 2 or 3 of this section;

(ii) There are axial flaws that penetrate through the clad into the low alloy steel reactor vessel shell, at a depth equal to or greater than 0.075 inches in through-wall extent from the clad-to-base metal interface; or

(iii) Any flaws between the clad-to-base metal interface and three-eighths[2] of the vessel thickness exceed the size allowable in ASME Code, Section XI, Table IWB–3510–1.

(5) The analyses required by paragraph (e)(4) of this section must address the effects on TWCF of the known sizes and locations of all flaws detected by the ASME Code, Section XI, Appendix VIII, Supplement 4 and Supplement 6 ultrasonic examination out to three-eights of the vessel thickness from the inner surface, and may also take into account other reactor vessel-specific information, including fracture toughness information.

(6) For all flaw assessments performed in accordance with paragraph (e)(4) of this section, the licensee shall prepare and submit a neutron fluence map, projected to the date of license expiration, for the reactor vessel beltline clad-to-base metal interface and indexed in a manner that allows the determination of the neutron fluence at the location of the detected flaws.

(f) *Calculation of RT_{MAX-X} values.* Each licensee shall calculate RT_{MAX-X} values for each reactor vessel beltline material using φt. The neutron flux (φ[t]),

[2] Because flaws greater than three-eights of the vessel wall thickness from the inside surface do not contribute to TWCF, flaws greater than three-eights of the vessel wall thickness from the inside surface need not be analyzed for their contribution to PTS.

must be calculated using a methodology that has been benchmarked to experimental measurements and with quantified uncertainties and possible biases.[3]

(1) The values of RT_{MAX-AW}, RT_{MAX-PL}, RT_{MAX-FO}, and RT_{MAX-CW} must be determined using Equations 1 through 4 of this section. When calculating RT_{MAX-AW} using Equation 1, RT_{MAX-AW} is the maximum value of ($RT_{NDT(U)}$ + ΔT_{30}) for the weld and for the adjoining plates. When calculating RT_{MAX-CW} using Equation 4, RT_{MAX-CW} is the maximum value of ($RT_{NDT(U)}$ + ΔT_{30}) for the circumferential weld and for the adjoining plates or forgings.

(2) The values of ΔT_{30} must be determined using Equations 5, 6 and 7 of this section, unless the conditions specified in paragraph (f)(6)(v) of this section are not met, for each axial weld, plate, forging, and circumferential weld. The ΔT_{30} value for each axial weld calculated as specified by Equation 1 of this section must be calculated for the maximum fluence ($\varphi t_{AXIAL-WELD}$) occurring along a particular axial weld at the clad-to-base metal interface. The ΔT_{30} value for each plate calculated as specified by Equation 1 of this section must also be calculated using the same value of $\varphi t_{AXIAL-WELD}$ used for the axial weld. The ΔT_{30} values in Equation 1 shall be calculated for the weld itself and each adjoining plate. The ΔT_{30} value for each plate or forging calculated as specified by Equations 2 and 3 of this section must be calculated for the maximum fluence (φt_{MAX}) occurring at the clad-to-base metal interface over the entire area of each plate or forging. In Equation 4, the fluence ($\varphi t_{WELD-CIRC}$) value used for calculating the plate, forging, and circumferential weld ΔT_{30} value is the maximum fluence occurring for each material along the circumferential weld at the clad-to-base metal interface. The ΔT_{30} values in Equation 4 shall be calculated for the circumferential weld and for the adjoining plates or forgings. If the conditions specified in paragraph (f)(6)(v) of this section are not met, licensees must propose ΔT_{30} and RT_{MAX-X} values in accordance with paragraph (f)(6)(vi) of this section.

(3) The values of Cu, Mn, P, and Ni in Equations 6 and 7 of this section must represent the best estimate values for the material. For a plate or forging, the best estimate value is normally the mean of the measured values for that plate or forging. For a weld, the best estimate value is normally the mean of the measured values for a weld deposit made using the same weld wire heat number as the critical vessel weld. If these values are not available, either the upper limiting values given in the material specifications to which the vessel material was fabricated, or conservative estimates (*i.e.*, mean plus one standard deviation) based on generic data[4] as shown in Table 4 of this section for P and Mn, must be used.

(4) The values of $RT_{NDT(U)}$ must be evaluated according to the procedures in the ASME Code, Section III, paragraph NB–2331. If any other method is used for this evaluation, the licensee shall submit the proposed method for review and approval by the Director along with the calculation of RT_{MAX-X} values required in paragraph (c)(1) of this section.

(i) If a measured value of $RT_{NDT(U)}$ is not available, a generic mean value of $RT_{NDT(U)}$ for the class[5] of material must be used if there are sufficient test results to establish a mean.

(ii) The following generic mean values of $RT_{NDT(U)}$ must be used unless justification for different values is provided: 0 °F for welds made with Linde 80 weld flux; and −56 °F for welds made with Linde 0091, 1092, and 124 and ARCOS B–5 weld fluxes.

(5) The value of T_C in Equation 6 of this section must represent the time-weighted average of the reactor cold leg temperature under normal operating full power conditions from the beginning of full power operation through the end of licensed operation.

[3] Regulatory Guide 1.190 dated March 2001, establishes acceptable methods for determining neutron flux.

[4] Data from reactor vessels fabricated to the same material specification in the same shop as the vessel in question and in the same time is an example of ''generic data.''

[5] The class of material for estimating $RT_{NDT(U)}$ must be determined by the type of welding flux (Linde 80, or other) for welds or by the material specification for base metal.

(6) The licensee shall verify that an appropriate RT_{MAX-X} value has been calculated for each reactor vessel beltline material by considering plant-specific information that could affect the use of the model (*i.e.*, Equations 5, 6 and 7) of this section for the determination of a material's ΔT_{30} value.

(i) The licensee shall evaluate the results from a plant-specific or integrated surveillance program if the surveillance data satisfy the criteria described in paragraphs (f)(6)(i)(A) and (f)(6)(i)(B) of this section:

(A) The surveillance material must be a heat-specific match for one or more of the materials for which RT_{MAX-X} is being calculated. The 30-foot-pound transition temperature must be determined as specified by the requirements of 10 CFR part 50, Appendix H.

(B) If three or more surveillance data points measured at three or more different neutron fluences exist for a specific material, the licensee shall determine if the surveillance data show a significantly different trend than the embrittlement model predicts. This must be achieved by evaluating the surveillance data for consistency with the embrittlement model by following the procedures specified by paragraphs (f)(6)(ii), (f)(6)(iii), and (f)(6)(iv) of this section. If fewer than three surveillance data points exist for a specific material, then the embrittlement model must be used without performing the consistency check.

(ii) The licensee shall estimate the mean deviation from the embrittlement model for the specific data set (*i.e.*, a group of surveillance data points representative of a given material). The mean deviation from the embrittlement model for a given data set must be calculated using Equations 8 and 9 of this section. The mean deviation for the data set must be compared to the maximum heat-average residual given in Table 5 or derived using Equation 10 of this section. The maximum heat-average residual is based on the material group into which the surveillance material falls and the number of surveillance data points. For surveillance data sets with greater than 8 data points, the maximum credible heat-average residual must be calculated using Equation 10 of this section. The value of σ used in Equation 10 of this section must be obtained from Table 5 of this section.

(iii) The licensee shall estimate the slope of the embrittlement model residuals (estimated using Equation 8) plotted as a function of the base 10 logarithm of neutron fluence for the specific data set. The licensee shall estimate the T-statistic for this slope (T_{SURV}) using Equation 11 and compare this value to the maximum permissible T-statistic (T_{MAX}) in Table 6. For surveillance data sets with greater than 15 data points, the T_{MAX} value must be calculated using Student's T distribution with a significance level (α) of 1 percent for a one-tailed test.

(iv) The licensee shall estimate the two largest positive deviations (*i.e.*, outliers) from the embrittlement model for the specific data set using Equations 8 and 12. The licensee shall compare the largest normalized residual (r^*) to the appropriate allowable value from the third column in Table 7 and the second largest normalized residual to the appropriate allowable value from the second column in Table 7.

(v) The ΔT_{30} value must be determined using Equations 5, 6, and 7 of this section if all three of the following criteria are satisfied:

(A) The mean deviation from the embrittlement model for the data set is equal to or less than the value in Table 5 or the value derived using Equation 10 of this section;

(B) The T-statistic for the slope (T_{SURV}) estimated using Equation 11 is equal to or less than the Maximum permissible T-statistic (T_{MAX}) in Table 6; and

(C) The largest normalized residual value is equal to or less than the appropriate allowable value from the third column in Table 7 and the second largest normalized residual value is equal to or less than the appropriate allowable value from the second column in Table 7. If any of these criteria is not satisfied, the licensee must propose ΔT_{30} and RT_{MAX-X} values in accordance with paragraph (f)(6)(vi) of this section.

(vi) If any of the criteria described in paragraph (f)(6)(v) of this section are not satisfied, the licensee shall review the data base for that heat in detail, including all parameters used in Equations 5, 6, and 7 of this section and the data used to determine the baseline Charpy V-notch curve for the material in an unirradiated condition. The licensee shall submit an evaluation of the surveillance data to the NRC and shall propose ΔT_{30} and RT_{MAX-X} values, considering their plant-specific surveillance data, to be used for evaluation relative to the acceptance criteria of this rule. These evaluations must be submitted for review and approval by the Director in the form of a license amendment in accordance with the requirements of paragraphs (c)(1) and (d)(1) of this section.

(7) The licensee shall report any information that significantly influences the RT_{MAX-X} value to the Director in accordance with the requirements of paragraphs (c)(1) and (d)(1) of this section.

(g) *Equations and variables used in this section.*

$$\text{Equation 1: } RT_{MAX-AW} = MAX\left\{\left[RT_{NDT(U)-plate} + \Delta T_{30-plate}\right], \left[RT_{NDT(U)-axial\ weld} + \Delta T_{30-axial\ weld}\right]\right\}$$

$$\text{Equation 2: } RT_{MAX-PL} = RT_{NDT(U)-plate} + \Delta T_{30-plate}$$

$$\text{Equation 3: } RT_{MAX-FO} = RT_{NDT(U)-forging} + \Delta T_{30-forging}$$

$$\text{Equation 4: } RT_{MAX-CW} = MAX\left\{\left[RT_{NDT(U)-plate} + \Delta T_{30-plate}\right], \left[RT_{NDT(U)-circweld} + \Delta T_{30-circweld}\right], \left[RT_{NDT(U)-forging} + \Delta T_{30-forging}\right]\right\}$$

$$\text{Equation 5: } \Delta T_{30} = MD + CRP$$

$$\text{Equation 6: } MD = A \times (1 - 0.001718 \times T_C) \times \left(1 + 6.13 \times P \times Mn^{2.471}\right) \times \varphi t_e^{0.5}$$

$$\text{Equation 7: } CRP = B \times \left(1 + 3.77 \times Ni^{1.191}\right) \times f(Cu_e, P) \times g(Cu_e, Ni, \varphi t_e)$$

Where:

P [wt-&%] = phosphorus content
Mn [wt-%] = manganese content
Ni [wt-%] = nickel content
Cu [wt-%] = copper content
A = 1.140×10^{-7} for forgings
A = 1.561×10^{-7} for plates
A = 1.417×10^{-7} for welds
B = 102.3 for forgings
B = 102.5 for plates in non-Combustion Engineering manufactured vessels
B = 135.2 for plates in Combustion Engineering vessels
B = 155.0 for welds

$\varphi t_e = \varphi t$ for $\varphi \geq 4.39 \times 10^{10}$ n/cm²/sec
$\varphi t_e = \varphi t \times (4.39 \times 10^{10}/\varphi)^{0.2595}$ for $\varphi < 4.39 \times 10^{10}$ n/cm²/sec

Where:

φ [n/cm²/sec] = average neutron flux
t [sec] = time that the reactor has been in full power operation

φt [n/cm²] = $\varphi \times t$

$f(Cu_e,P) = 0$ for Cu ≤ 0.072
$f(Cu_e,P) = [Cu_e - 0.072]^{0.668}$ for Cu >0.072 and P ≤ 0.008
$f(Cu_e,P) = [Cu_e - 0.072 + 1.359 \times (P - 0.008)]^{0.668}$ for Cu >0.072 and P >0.008

Where:

$Cu_e = 0$ for Cu ≤ 0.072
Cu_e = MIN (Cu, maximum Cu_e) for Cu >0.072
maximum Cu_e = 0.243 for Linde 80 welds
maximum Cu_e = 0.301 for all other materials

$g(Cu_e,Ni,\varphi t_e) = 0.5 + (0.5 \times \tanh\{[\log_{10}(\varphi t_e) + (1.1390 \times Cu_e) - (0.448 \times Ni) - 18.120]/0.629\}$

Equation 8: Residual (r) = measured ΔT_{30} − predicted ΔT_{30} (by Equations 5, 6 and 7)

$$\text{Equation 9: Mean deviation for a data set of n data points} = (1/n) \times \sum_{i=1}^{n} r_i$$

Equation 10: Maximum credible heat-average residual = $2.33\sigma/n^{0.5}$

Where:

n = number of surveillance data points (sample size) in the specific data set

σ = standard deviation of the residuals about the model for a relevant material group given in Table 5.

$$\text{Equation 11: } T_{SURV} = \frac{m}{se(m)}$$

Where:

m is the slope of a plot of all of the r values (estimated using Equation 8) versus the base 10 logarithm of the neutron fluence for each r value. The slope shall be estimated using the method of least squares.

se(m) is the least squares estimate of the standard-error associated with the estimated slope value *m*.

$$\text{Equation 12: } r^* = \frac{r}{\sigma}$$

Where:

r is defined using Equation 8 and σ is given in Table 5

TABLE 1—PTS SCREENING CRITERIA

Product form and RT_{MAX-X} values	RT_{MAX-X} limits [°F] for different vessel wall thicknesses [6] (T_{WALL})		
	T_{WALL} ≤9.5 in.	9.5 in. <T_{WALL} ≤10.5 in.	10.5 in. <T_{WALL} ≤11.5 in.
Axial Weld—RT_{MAX-AW}	269	230	222
Plate—RT_{MAX-PL}	356	305	293
Forging without underclad cracks—RT_{MAX-FO} [7]	356	305	293
Axial Weld and Plate—RT_{MAX-AW} + RT_{MAX-PL}	538	476	445
Circumferential Weld—RT_{MAX-CW} [8]	312	277	269
Forging with underclad cracks—RT_{MAX-FO} [9]	246	241	239

[6] Wall thickness is the beltline wall thickness including the clad thickness.

[7] Forgings without underclad cracks apply to forgings for which no underclad cracks have been detected and that were fabricated in accordance with Regulatory Guide 1.43.

[8] RT_{PTS} limits contribute 1×10^{-8} per reactor year to the reactor vessel TWCF.

[9] Forgings with underclad cracks apply to forgings that have detected underclad cracking or were not fabricated in accordance with Regulatory Guide 1.43.

TABLE 2—ALLOWABLE NUMBER OF FLAWS IN WELDS

Through-wall extent, TWE [in.]		Maximum number of flaws per 1,000-inches of weld length in the inspection volume that are greater than or equal to TWE_{MIN} and less than TWE_{MAX}
TWE_{MIN}	TWE_{MAX}	
0	0.075	No Limit.
0.075	0.475	166.70.
0.125	0.475	90.80.
0.175	0.475	22.82.
0.225	0.475	8.66.
0.275	0.475	4.01.
0.325	0.475	3.01.

TABLE 2—ALLOWABLE NUMBER OF FLAWS IN WELDS—Continued

Through-wall extent, TWE [in.]		Maximum number of flaws per 1,000-inches of weld length in the inspection volume that are greater than or equal to TWE_{MIN} and less than TWE_{MAX}
TWE_{MIN}	TWE_{MAX}	
0.375	0.475	1.49.
0.425	0.475	1.00.
0.475	Infinite	0.00.

TABLE 3—ALLOWABLE NUMBER OF FLAWS IN PLATES AND FORGINGS

Through-wall extent, TWE [in.]		Maximum number of flaws per 1,000 square-inches of inside surface area in the inspection volume that are greater than or equal to TWE_{MIN} and less than TWE_{MAX}. This flaw density does not include underclad cracks in forgings
TWE_{MIN}	TWE_{MAX}	
0	0.075	No Limit.
0.075	0.375	8.05.
0.125	0.375	3.15.
0.175	0.375	0.85.
0.225	0.375	0.29.
0.275	0.375	0.08.
0.325	0.375	0.01.
0.375	Infinite	0.00.

TABLE 4—CONSERVATIVE ESTIMATES FOR CHEMICAL ELEMENT WEIGHT PERCENTAGES

Materials	P	Mn
Plates	0.014	1.45
Forgings	0.016	1.11
Welds	0.019	1.63

TABLE 5—MAXIMUM HEAT-AVERAGE RESIDUAL [°F] FOR RELEVANT MATERIAL GROUPS BY NUMBER OF AVAILABLE DATA POINTS (SIGNIFICANCE LEVEL = 1%)

Material group	σ [°F]	Number of available data points					
		3	4	5	6	7	8
Welds, for Cu >0.072	26.4	35.5	30.8	27.5	25.1	23.2	21.7
Plates, for Cu >0.072	21.2	28.5	24.7	22.1	20.2	18.7	17.5
Forgings, for Cu >0.072	19.6	26.4	22.8	20.4	18.6	17.3	16.1
Weld, Plate or Forging, for Cu ≤0.072	18.6	25.0	21.7	19.4	17.7	16.4	15.3

TABLE 6—T_{MAX} VALUES FOR THE SLOPE DEVIATION TEST (SIGNIFICANCE LEVEL = 1%)

Number of available data points (n)	T_{MAX}
3	31.82
4	6.96
5	4.54
6	3.75
7	3.36
8	3.14
9	3.00
10	2.90
11	2.82
12	2.76
13	2.72
14	2.68
15	2.65

TABLE 7—THRESHOLD VALUES FOR THE OUTLIER DEVIATION TEST (SIGNIFICANCE LEVEL = 1%)

Number of available data points (n)	Second largest allowable normalized residual value (r*)	Largest allowable normalized residual value (r*)
3	1.55	2.71
4	1.73	2.81
5	1.84	2.88
6	1.93	2.93
7	2.00	2.98
8	2.05	3.02
9	2.11	3.06
10	2.16	3.09
11	2.19	3.12
12	2.23	3.14
13	2.26	3.17
14	2.29	3.19
15	2.32	3.21

[75 FR 23, Jan. 4, 2010, as amended at 75 FR 5495, Feb. 3, 2010; 75 FR 10411, Mar. 8, 2010; 75 FR 72653, Nov. 26, 2010]

§ 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, other than nuclear power reactor facilities for which the certifications required under § 50.82(a)(1) have been submitted.

(b) *Definition.* For purposes of this section, *Anticipated Transient Without Scram* (ATWS) means an anticipated operational occurrence as defined in appendix A of this part 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) *Requirements.* (1) Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system.

(2) Each pressurized water reactor manufactured by Combustion Engineering or by Babcock and Wilcox must have a diverse scram system from the sensor output to interruption of power to the control rods. This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system (from sensor output to interruption of power to the control rods).

(3) Each boiling water reactor must have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI system must have redundant scram air header exhaust valves. The ARI must be designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

(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 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.

(5) Each boiling water reactor must have equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner.

(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.

(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, 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 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.

[49 FR 26044, June 26, 1984; 49 FR 27736, July 6, 1984, as amended at 51 FR 40310, Nov. 6, 1986; 54 FR 13362, Apr. 3, 1989; 61 FR 39301, July 29, 1996; 72 FR 49500, Aug. 28, 2007]

§ 50.63 Loss of all alternating current power.

(a) *Requirements.* (1) Each light-water-cooled nuclear power plant licensed to operate under this part, 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, 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 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:

(i) The redundancy of the onsite emergency ac power sources;

(ii) The reliability of the onsite emergency ac power sources;

(iii) The expected frequency of loss of offsite power; and

(iv) The probable time needed to restore offsite power.

(2) The reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Licensees are expected to have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review.

(b) *Limitation of scope.* Paragraph (c) of this section does not apply to those plants licensed to operate prior to *July 21, 1988*, if the capability to withstand station blackout was specifically addressed in the operating license proceeding and was explicitly approved by the NRC.

(c) *Implementation*—(1) *Information Submittal.* For each light-water-cooled nuclear power plant operating license application submitted after September 27, 2007, the applicant shall submit the information defined below in its final safety analysis report.

(i) A proposed station blackout duration to be used in determining compliance with paragraph (a) of this section, including a justification for the selection based on the four factors identified in paragraph (a) of this section;

(ii) A description of the procedures that will be implemented for station blackout events for the duration determined in paragraph (c)(1)(i) of this section and for recovery therefrom; and

(iii) A list of modifications to equipment and associated procedures, if any, necessary to meet the requirements of paragraph (a) of this section, for the specified station blackout duration determined in paragraph (c)(1)(i) of this section, and a proposed schedule for implementing the stated modifications.

(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 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.

(3) *Regulatory Assessment:* After consideration of the information submitted in accordance with paragraph (c)(1) of this section, the Director, Office of Nuclear Reactor Regulation, will notify the licensee of the Director's conclusions regarding the adequacy of the proposed specified station blackout duration, the proposed equipment modifications and procedures, and the proposed schedule for implementing the procedures and modifications for compliance with paragraph (a) this section.

[53 FR 23215, June 21, 1988, as amended at 63 FR 50480, Sept. 22, 1998; 72 FR 49501, Aug. 28, 2007; 86 FR 43402, Aug. 9, 2021]

§ 50.64 Limitations on the use of highly enriched uranium (HEU) in domestic non-power reactors.

(a) *Applicability.* The requirements of this section apply to all non-power reactors.

(b) *Requirements.* (1) The Commission will not issue a construction permit after March 27, 1986 for a non-power reactor where the applicant proposes to use highly enriched uranium (HEU) fuel, unless the applicant demonstrates that the proposed reactor will have a unique purpose as defined in § 50.2.

(2) Unless the Commission has determined, based on a request submitted in accordance with paragraph (c)(1) of this section, that the non-power reactor has a unique purpose, each licensee authorized to possess and use HEU fuel in connection with the reactor's operation shall:

(i) Not initiate acquisition of additional HEU fuel, if low enriched uranium (LEU) fuel acceptable to the Commission for that reactor is available when it proposes that acquisition; and

(ii) Replace all HEU fuel in its possession with available LEU fuel acceptable to the Commission for that reactor, in accordance with a schedule determined pursuant to paragraph (c)(2) of this section.

(3) If not required by paragraphs (b) (1) and (2) of this section to use LEU fuel, the applicant or licensee must use HEU fuel of enrichment as close to 20% as is available and acceptable to the Commisson.

(c) *Implementation.* (1) Any request by a licensee for a determination that a non-power reactor has a unique purpose as defined in § 50.2 should be submitted with supporting documentation to the Director of the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, by September 29, 1986.

(2) (i) By March 27, 1987 and at 12-month intervals thereafter, each licensee of a non-power reactor authorized to possess and use HEU fuel shall

develop and submit to the Director of the Office of Nuclear Reactor Regulation a written proposal for meeting the requirements of paragraph (b) (2) or (3) of this section. The licensee shall include in the proposal a certification that Federal Government funding for conversion is available through the Department of Energy (DOE) or other appropriate Federal Agency. The licensee shall also include in the proposal a schedule for conversion, based upon availability of replacement fuel acceptable to the Commisson for that reactor and upon consideration of other factors such as the availability of shipping casks, implementation of arrangements for the available financial support, and reactor usage.

(ii) If Federal Government funding for conversion cannot be certified, the proposal's contents may be limited to a statement of this fact. If a statement of non-availability of Federal Government funding for conversion is submitted by a licensee, then it shall be required to resubmit a proposal for meeting the requirements of paragraph (b) (2) or (3) of this section at 12-month intervals.

(iii) The proposal shall include, to the extent required to effect the conversion, all necessary changes in the license, facility, or procedures. Supporting safety analyses should be provided so as to meet the schedule established for conversion. As long as Federal Government funding for conversion is not available, the resubmittal may be a reiteration of the original proposal. The Director of the Office of Nuclear Reactor Regulation shall review the proposal and confirm the status of Federal Government funding for conversion and, if a schedule for conversion has been submitted by the licensee, will then determine a final schedule.

(3) After review of the safety analysis required by paragraph (c)(2), the Director of the Office of Nuclear Reactor Regulation will issue an appropriate enforcement order directing both the conversion and, to the extent consistent with protection of the public health and safety, any necessary changes to the license, facility, or procedures.

[51 FR 6519, Feb. 25, 1986]

§ 50.65 Requirements for monitoring the effectiveness of maintenance at nuclear power plants.

The requirements of this section are applicable during all conditions of plant operation, including normal shutdown operations.

(a)(1) Each holder of an operating license for a nuclear power plant under this part and each holder of a combined license under part 52 of this chapter after the Commission makes the finding under § 52.103(g) of this chapter, shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components, as defined in paragraph (b) of this section, are capable of fulfilling their intended functions. These goals shall be established commensurate with safety and, where practical, take into account industry-wide operating experience. When the performance or condition of a structure, system, or component does not meet established goals, appropriate corrective action shall be taken. For a nuclear power plant for which the licensee has submitted the certifications specified in § 50.82(a)(1) or 52.110(a)(1) of this chapter, as applicable, this section shall only apply to the extent that the licensee shall monitor the performance or condition of all structures, systems, or components associated with the storage, control, and maintenance of spent fuel in a safe condition, in a manner sufficient to provide reasonable assurance that these structures, systems, and components are capable of fulfilling their intended functions.

(2) Monitoring as specified in paragraph (a)(1) of this section is not required where it has been demonstrated that the performance or condition of a structure, system, or component is being effectively controlled through the performance of appropriate preventive maintenance, such that the structure, system, or component remains capable of performing its intended function.

(3) Performance and condition monitoring activities and associated goals and preventive maintenance activities shall be evaluated at least every refueling cycle provided the interval between

evaluations does not exceed 24 months. The evaluations shall take into account, where practical, industry-wide operating experience. Adjustments shall be made where necessary to ensure that the objective of preventing failures of structures, systems, and components through maintenance is appropriately balanced against the objective of minimizing unavailability of structures, systems, and components due to monitoring or preventive maintenance.

(4) Before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of the assessment may be limited to structures, systems, and components that a risk-informed evaluation process has shown to be significant to public health and safety.

(b) The scope of the monitoring program specified in paragraph (a)(1) of this section shall include safety related and nonsafety related structures, systems, and components, as follows:

(1) Safety-related structures, systems and components that are relied upon to remain functional during and following design basis events to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in § 50.34(a)(1), § 50.67(b)(2), or § 100.11 of this chapter, as applicable.

(2) Nonsafety related structures, systems, or components:

(i) That are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures (EOPs); or

(ii) Whose failure could prevent safety-related structures, systems, and components from fulfilling their safety-related function; or

(iii) Whose failure could cause a reactor scram or actuation of a safety-related system.

(c) The requirements of this section shall be implemented by each licensee no later than July 10, 1996.

[56 FR 31324, July 10, 1991, as amended at 58 FR 33996, June 23, 1993; 61 FR 39301, July 29, 1996; 61 FR 65173, Dec. 11, 1996; 62 FR 47271, Sept. 8, 1997; 62 FR 59276, Nov. 3, 1997; 64 FR 38557, July 19, 1999; 64 FR 72001, Dec. 23, 1999; 72 FR 49501, Aug. 28, 2007]

EFFECTIVE DATE NOTE: See 64 FR 38551, July 19, 1999, for effectiveness of § 50.65 (a)(3) and (a)(4).

§ 50.66 Requirements for thermal annealing of the reactor pressure vessel.

(a) For those light water nuclear power reactors where neutron radiation has reduced the fracture toughness of the reactor vessel materials, a thermal annealing may be applied to the reactor vessel to recover the fracture toughness of the material. The use of a thermal annealing treatment is subject to the requirements in this section. A report describing the licensee's plan for conducting the thermal annealing must be submitted in accordance with § 50.4 at least three years prior to the date at which the limiting fracture toughness criteria in § 50.61 or appendix G to part 50 would be exceeded. Within three years of the submittal of the Thermal Annealing Report and at least thirty days prior to the start of the thermal annealing, the NRC will review the Thermal Annealing Report and make available the results of its evaluation at the NRC Web site, *http://www.nrc.gov*. The licensee may begin the thermal anneal after:

(1) Submitting the Thermal Annealing Report required by paragraph (b) of this section;

(2) The NRC makes available the results of its evaluation of the Thermal Annealing Report at the NRC Web site, *http://www.nrc.gov;* and

(3) The requirements of paragraph (f)(1) of this section have been satisfied.

(b) *Thermal Annealing Report.* The Thermal Annealing Report must include: a Thermal Annealing Operating Plan; a Requalification Inspection and Test Program; a Fracture Toughness Recovery and Reembrittlement Trend

Assurance Program; and an Identification of Changes Requiring a License Amendment.

(1) *Thermal Annealing Operating Plan.* The thermal annealing operating plan must include:

(i) A detailed description of the pressure vessel and all structures and components that are expected to experience significant thermal or stress effects during the thermal annealing operation;

(ii) An evaluation of the effects of mechanical and thermal stresses and temperatures on the vessel, containment, biological shield, attached piping and appurtenances, and adjacent equipment and components to demonstrate that operability of the reactor will not be detrimentally affected. This evaluation must include:

(A) Detailed thermal and structural analyses to establish the time and temperature profile of the annealing operation. These analyses must include heatup and cooldown rates, and must demonstrate that localized temperatures, thermal stress gradients, and subsequent residual stresses will not result in unacceptable dimensional changes or distortions in the vessel, attached piping and appurtenances, and that the thermal annealing cycle will not result in unacceptable degradation of the fatigue life of these components.

(B) The effects of localized high temperatures on degradation of the concrete adjacent to the vessel and changes in thermal and mechanical properties, if any, of the reactor vessel insulation, and on detrimental effects, if any, on containment and the biological shield. If the design temperature limitations for the adjacent concrete structure are to be exceeded during the thermal annealing operation, an acceptable maximum temperature for the concrete must be established for the annealing operation using appropriate test data.

(iii) The methods, including heat source, instrumentation and procedures proposed for performing the thermal annealing. This shall include any special precautions necessary to minimize occupational exposure, in accordance with the As Low As Reasonably Achievable (ALARA) principle and the provisions of § 20.1206.

(iv) The proposed thermal annealing operating parameters, including bounding conditions for temperatures and times, and heatup and cooldown schedules.

(A) The thermal annealing time and temperature parameters selected must be based on projecting sufficient recovery of fracture toughness, using the procedures of paragraph (e) of this section, to satisfy the requirements of § 50.60 and § 50.61 for the proposed period of operation addressed in the application.

(B) The time and temperature parameters evaluated as part of the thermal annealing operating plan, and supported by the evaluation results of paragraph (b)(1)(ii) of this section, represent the bounding times and temperatures for the thermal annealing operation. If these bounding conditions for times and temperatures are violated during the thermal annealing operation, then the annealing operation is considered not in accordance with the Thermal Annealing Operating Plan, as required by paragraph (c)(1) of this section, and the licensee must comply with paragraph (c)(2) of this section.

(2) *Requalification Inspection and Test Program.* The inspection and test program to requalify the annealed reactor vessel must include the detailed monitoring, inspections, and tests proposed to demonstrate that the limitations on temperatures, times and temperature profiles, and stresses evaluated for the proposed thermal annealing conditions of paragraph (b)(1)(iv) of this section have not been exceeded, and to determine the thermal annealing time and temperature to be used in quantifying the fracture toughness recovery. The requalification inspection and test program must demonstrate that the thermal annealing operation has not degraded the reactor vessel, attached piping or appurtenances, or the adjacent concrete structures to a degree that could affect the safe operation of the reactor.

(3) *Fracture Toughness Recovery and Reembrittlement Trend Assurance Program.* The percent recovery of RT_{NDT} and Charpy upper-shelf energy due to the thermal annealing treatment must be determined based on the time and

temperature of the actual vessel thermal anneal. The recovery of RT_{NDT} and Charpy upper-shelf energy provide the basis for establishing the post-anneal RT_{NDT} and Charpy upper-shelf energy for each vessel material. Changes in the RT_{NDT} and Charpy upper-shelf energy with subsequent plant operation must be determined using the post-anneal values of these parameters in conjunction with the projected reembrittlement trend determined in accordance with paragraph (b)(3)(ii) of this section. Recovery and reembrittlement evaluations shall include:

(i) *Recovery Evaluations.* (A) The percent recovery of both RT_{NDT} and Charpy upper-shelf energy must be determined by one of the procedures described in paragraph (e) of this section, using the proposed lower bound thermal annealing time and temperature conditions described in the operating plan.

(B) If the percent recovery is determined from testing surveillance specimens or from testing materials removed from the reactor vessel, then it shall be demonstrated that the proposed thermal annealing parameters used in the test program are equal to or bounded by those used in the vessel annealing operation.

(C) If generic computational methods are used, appropriate justification must be submitted as a part of the application.

(ii) *Reembrittlement Evaluations.* (A) The projected post-anneal reembrittlement of RT_{NDT} must be calculated using the procedures in § 50.61(c), or must be determined using the same basis as that used for the pre-anneal operating period. The projected change due to post-anneal reembrittlement for Charpy upper-shelf energy must be determined using the same basis as that used for the pre-anneal operating period.

(B) The post-anneal reembrittlement trend of both RT_{NDT} and Charpy upper-shelf energy must be estimated, and must be monitored using a surveillance program defined in the Thermal Annealing Report and which conforms to the intent of appendix H of this part, ''Reactor Vessel Material Surveillance Program Requirements.''

(4) *Identification of Changes Requiring a License Amendment.* Any changes to the facility as described in the final safety analysis report (as updated) which requires a license amendment pursuant to § 50.59(c)(2) of this part, and any changes to the Technical Specifications, which are necessary to either conduct the thermal annealing or to operate the nuclear power reactor following the annealing must be identified. The section shall demonstrate that the Commission's requirements continue to be complied with, and that there is reasonable assurance of adequate protection to the public health and safety following the changes.

(c) *Completion or Termination of Thermal Annealing.* (1) If the thermal annealing was completed in accordance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program, the licensee shall so confirm in writing to the Director, Office of Nuclear Reactor Regulation. The licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met.

(2) If the thermal annealing was completed but the annealing was not performed in accordance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program, the licensee shall submit a summary of lack of compliance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program and a justification for subsequent operation to the Director, Office of Nuclear Reactor Regulation. Any changes to the facility as described in the final safety analysis report (as updated) which are attributable to the noncompliances and which require a license amendment pursuant to § 50.59(c)(2) and any changes to the Technical Specifications shall also be identified.

(i) If no changes requiring a license amendment pursuant to § 50.59(c)(2) or changes to Technical Specifications are identified, the licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met.

(ii) If any changes requiring a license amendment pursuant to § 50.59(c)(2) or changes to the Technical Specifications are identified, the licensee may not restart its reactor until approval is obtained from the Director, Office of Nuclear Reactor Regulation and the requirements of paragraph (f)(2) of this section have been met.

(3) If the thermal annealing was terminated prior to completion, the licensee shall immediately notify the NRC of the premature termination of the thermal anneal.

(i) If the partial annealing was otherwise performed in accordance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program, and the licensee does not elect to take credit for any recovery, the licensee need not submit the Thermal Annealing Results Report required by paragraph (d) of this section but instead shall confirm in writing to the Director, Office of Nuclear Reactor Regulation that the partial annealing was otherwise performed in accordance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program. The licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met.

(ii) If the partial annealing was otherwise performed in accordance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program, and the licensee elects to take full or partial credit for the partial annealing, the licensee shall confirm in writing to the Director, Office of Nuclear Reactor Regulation that the partial annealing was otherwise performed in compliance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program. The licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met.

(iii) If the partial annealing was not performed in accordance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program, the licensee shall submit a summary of lack of compliance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program and a justification for subsequent operation to the Director, Office of Nuclear Reactor Regulation. Any changes to the facility as described in the final safety analysis report (as updated) which are attributable to the noncompliances and which require a license amendment pursuant to § 50.59(c)(2) and any changes to the technical specifications which are required as a result of the noncompliances, shall also be identified.

(A) If no changes requiring a license amendment pursuant to § 50.59(c)(2) or changes to Technical Specifications are identified, the licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met.

(B) If any changes requiring a license amendment pursuant to § 50.59(c)(2) or changes to Technical Specifications are identified, the licensee may not restart its reactor until approval is obtained from the Director, Office of Nuclear Reactor Regulation and the requirements of paragraph (f)(2) of this section have been met.

(d) *Thermal Annealing Results Report.* Every licensee that either completes a thermal annealing, or that terminates an annealing but elects to take full or partial credit for the annealing, shall provide the following information within three months of completing the thermal anneal, unless an extension is authorized by the Director, Office of Nuclear Reactor Regulation:

(1) The time and temperature profiles of the actual thermal annealing;

(2) The post-anneal RT_{NDT} and Charpy upper-shelf energy values of the reactor vessel materials for use in subsequent reactor operation;

(3) The projected post-anneal reembrittlement trends for both RT_{NDT} and Charpy upper-shelf energy; and

(4) The projected values of RT_{PTS} and Charpy upper-shelf energy at the end of the proposed period of operation addressed in the Thermal Annealing Report.

(e) *Procedures for Determining the Recovery of Fracture Toughness.* The procedures of this paragraph must be used to determine the percent recovery of ΔRT_{NDT}, R_t, and percent recovery of

Charpy upper-shelf energy, R_u. In all cases, R_t and R_u may not exceed 100.

(1) For those reactors with surveillance programs which have developed credible surveillance data as defined in § 50.61, percent recovery due to thermal annealing (R_t and R_u) must be evaluated by testing surveillance specimens that have been withdrawn from the surveillance program and that have been annealed under the same time and temperature conditions as those given the beltline material.

(2) Alternatively, the percent recovery due to thermal annealing (R_t and R_u) may be determined from the results of a verification test program employing materials removed from the beltline region of the reactor vessel[6] and that have been annealed under the same time and temperature conditions as those given the beltline material.

(3) Generic computational methods may be used to determine recovery if adequate justification is provided.

(f) *Public information and participation.* (1) Upon receipt of a Thermal Annealing Report, and a minimum of 30 days before the licensee starts thermal annealing, the Commission shall:

(i) Notify and solicit comments from local and State governments in the vicinity of the site where the thermal annealing will take place and any Indian Nation or other indigenous people that have treaty or statutory rights that could be affected by the thermal annealing,

(ii) Publish a notice of a public meeting in the FEDERAL REGISTER and in a forum, such as local newspapers, which is readily accessible to individuals in the vicinity of the site, to solicit comments from the public, and

(iii) Hold a public meeting on the licensee's Thermal Annealing Report.

(2) Within 15 days after the NRC's receipt of the licensee submissions required by paragraphs (c)(1), (c)(2) and (c)(3)(i) through (iii) of this section, the NRC staff shall make available at the NRC Web site, *http://www.nrc.gov*, a summary of its inspection of the licensee's thermal annealing, and the Commission shall hold a public meeting:

(i) For the licensee to explain to NRC and the public the results of the reactor pressure vessel annealing,

(ii) for the NRC to discuss its inspection of the reactor vessel annealing, and

(iii) for the NRC to receive public comments on the annealing.

(3) Within 45 days of NRC's receipt of the licensee submissions required by paragraphs (c)(1), (c)(2) and (c)(3)(i) through (iii) of this section, the NRC staff shall complete full documentation of its inspection of the licensee's annealing process and make available this documentation at the NRC Web site, *http://www.nrc.gov*.

[60 FR 65472, Dec. 19, 1995, as amended at 64 FR 48952, Sept. 9, 1999; 64 FR 53613, Oct. 4, 1999]

EFFECTIVE DATE NOTE: See 64 FR 53582, Oct. 4, 1999, for effectiveness of § 50.66 (b) introductory text, paragraphs (b)(4), (c)(2), and (c)(3)(iii).

§ 50.67 Accident source term.

(a) *Applicability.* The requirements of this section apply to all holders of operating licenses issued prior to January 10, 1997, and holders of renewed licenses under part 54 of this chapter whose initial operating license was issued prior to January 10, 1997, who seek to revise the current accident source term used in their design basis radiological analyses.

(b) *Requirements.* (1) A licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under § 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents[1] previously analyzed in the safety analysis report.

[6] For those cases where materials are removed from the beltline of the pressure vessel, the stress limits of the applicable portions of the ASME Code Section III must be satisfied, including consideration of fatigue and corrosion, regardless of the Code of record for the vessel design.

[1] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of

(2) The NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:

(i) 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 0.25 Sv (25 rem)[2] total effective dose equivalent (TEDE).

(ii) 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 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

[64 FR 72001, Dec. 23, 1999]

§ 50.68 Criticality accident requirements.

(a) Each holder of a construction permit or operating license for a nuclear power reactor issued under this part or a combined license for a nuclear power reactor issued under part 52 of this chapter, shall comply with either 10 CFR 70.24 of this chapter or the requirements in paragraph (b) of this section.

(b) Each licensee shall comply with the following requirements in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24:

(1) Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

(2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

(3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used.

(4) If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

(5) The quantity of SNM, other than nuclear fuel stored onsite, is less than the quantity necessary for a critical mass.

(6) Radiation monitors are provided in storage and associated handling

---

the core with subsequent release of appreciable quantities of fission products.

[2]2 The use of 0.25 Sv (25 rem) TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, this 0.25 Sv (25 rem) TEDE value has been stated in this section as a reference value, which can be used in the evaluation of proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation.

areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions.

(7) The maximum nominal U–235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight.

(8) The FSAR is amended no later than the next update which § 50.71(e) of this part requires, indicating that the licensee has chosen to comply with § 50.68(b).

(c) 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 spent fuel pool:

(1) The requirements in § 50.68(b) do not apply to the fuel located within that package or cask; and

(2) 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, apply to the fuel within that package or cask.

[63 FR 63130, Nov. 12, 1998, as amended at 71 FR 66652, Nov. 16, 2006]

§ 50.69 Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors.

(a) *Definitions.*

Risk-Informed Safety Class (RISC)–1 structures, systems, and components (SSCs) means safety-related SSCs that perform safety significant functions.

Risk-Informed Safety Class (RISC)–2 structures, systems and components (SSCs) means nonsafety-related SSCs that perform safety significant functions.

Risk-Informed Safety Class (RISC)–3 structures, systems and components (SSCs) means safety-related SSCs that perform low safety significant functions.

Risk-Informed Safety Class (RISC)–4 structures, systems and components (SSCs) means nonsafety-related SSCs that perform low safety significant functions.

Safety significant function means a function whose degradation or loss could result in a significant adverse effect on defense-in-depth, safety margin, or risk.

(b) *Applicability and scope of risk-informed treatment of SSCs and submittal/approval process.* (1) A holder of a license to operate a light water reactor (LWR) nuclear power plant under this part; a holder of a renewed LWR license under part 54 of this chapter; an applicant for a construction permit or operating license under this part; or an applicant for a design approval, a combined license, or manufacturing license under part 52 of this chapter; may voluntarily comply with the requirements in this section as an alternative to compliance with the following requirements for RISC–3 and RISC–4 SSCs:

(i) 10 CFR part 21.

(ii) The portion of 10 CFR 50.46a(b) that imposes requirements to conform to Appendix B to 10 CFR part 50.

(iii) 10 CFR 50.49.

(iv) 10 CFR 50.55(e).

(v) The inservice testing requirements in 10 CFR 50.55a(f); the inservice inspection, and repair and replacement (with the exception of fracture toughness), requirements for ASME Class 2 and Class 3 SSCs in 10 CFR 50.55a(g); and the electrical component quality and qualification requirements in Section 4.3 and 4.4 of IEEE 279, and Sections 5.3 and 5.4 of IEEE 603–1991, as incorporated by reference in 10 CFR 50.55a(h).

(vi) 10 CFR 50.65, except for paragraph (a)(4).

(vii) 10 CFR 50.72.

(viii) 10 CFR 50.73.

(ix) Appendix B to 10 CFR part 50.

(x) The Type B and Type C leakage testing requirements in both Options A and B of Appendix J to 10 CFR part 50, for penetrations and valves meeting the following criteria:

(A) Containment penetrations that are either 1-inch nominal size or less, or continuously pressurized.

(B) Containment isolation valves that meet one or more of the following criteria:

(*1*) The valve is required to be open under accident conditions to prevent or mitigate core damage events;

(*2*) The valve is normally closed and in a physically closed, water-filled system;

(*3*) The valve is in a physically closed system whose piping pressure rating exceeds the containment design pressure rating and is not connected to the reactor coolant pressure boundary; or

(*4*) The valve is 1-inch nominal size or less.

(xi) Appendix A to part 100, Sections VI(a)(1) and VI(a)(2), to the extent that these regulations require qualification testing and specific engineering methods to demonstrate that SSCs are designed to withstand the Safe Shutdown Earthquake and Operating Basis Earthquake.

(2) A licensee voluntarily choosing to implement this section shall submit an application for license amendment under § 50.90 that contains the following information:

(i) A description of the process for categorization of RISC–1, RISC–2, RISC–3 and RISC–4 SSCs.

(ii) A description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant-specific probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs.

(iii) Results of the PRA review process conducted to meet § 50.69(c)(1)(i).

(iv) A description of, and basis for acceptability of, the evaluations to be conducted to satisfy § 50.69(c)(1)(iv). The evaluations must include the effects of common cause interaction susceptibility, and the potential impacts from known degradation mechanisms for both active and passive functions, and address internally and externally initiated events and plant operating modes (e.g., full power and shutdown conditions).

(3) The Commission will approve a licensee's implementation of this section if it determines that the process for categorization of RISC–1, RISC–2, RISC–3, and RISC–4 SSCs satisfies the requirements of § 50.69(c) by issuing a license amendment approving the licensee's use of this section.

(4) An applicant choosing to implement this section shall include the information in § 50.69(b)(2) as part of application. The Commission will approve an applicant's implementation of this section if it determines that the process for categorization of RISC–1, RISC–2, RISC–3, and RISC–4 SSCs satisfies the requirements of § 50.69(c).

(c) *SSC Categorization Process.* (1) SSCs must be categorized as RISC–1, RISC–2, RISC–3, or RISC–4 SSCs using a categorization process that determines if an SSC performs one or more safety significant functions and identifies those functions. The process must:

(i) Consider results and insights from the plant-specific PRA. This PRA must at a minimum model severe accident scenarios resulting from internal initiating events occurring at full power operation. The PRA must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC.

(ii) Determine SSC functional importance using an integrated, systematic process for addressing initiating events (internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include design bases functions and functions credited for mitigation and prevention of severe accidents. All aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience.

(iii) Maintain defense-in-depth.

(iv) Include evaluations that provide reasonable confidence that for SSCs categorized as RISC–3, sufficient safety margins are maintained and that any potential increases in core damage frequency (CDF) and large early release frequency (LERF) resulting from changes in treatment permitted by implementation of §§ 50.69(b)(1) and (d)(2) are small.

(v) Be performed for entire systems and structures, not for selected components within a system or structure.

(2) The SSCs must be categorized by an Integrated Decision-Making Panel (IDP) staffed with expert, plant-knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering, and system engineering.

(d) *Alternative treatment requirements*—(1) *RISC–1 and RISC 2 SSCs*. The licensee or applicant shall ensure that RISC–1 and RISC–2 SSCs perform their functions consistent with the categorization process assumptions by evaluating treatment being applied to these SSCs to ensure that it supports the key assumptions in the categorization process that relate to their assumed performance.

(2) *RISC–3 SSCs*. The licensee or applicant shall ensure, with reasonable confidence, that RISC–3 SSCs remain capable of performing their safety-related functions under design basis conditions, including seismic conditions and environmental conditions and effects throughout their service life. The treatment of RISC–3 SSCs must be consistent with the categorization process. Inspection and testing, and corrective action shall be provided for RISC–3 SSCs.

(i) Inspection and testing. Periodic inspection and testing activities must be conducted to determine that RISC–3 SSCs will remain capable of performing their safety-related functions under design basis conditions; and

(ii) Corrective action. Conditions that would prevent a RISC–3 SSC from performing its safety-related functions under design basis conditions must be corrected in a timely manner. For significant conditions adverse to quality, measures must be taken to provide reasonable confidence that the cause of the condition is determined and corrective action taken to preclude repetition.

(e) *Feedback and process adjustment*—(1) *RISC–1, RISC–2, RISC–3 and RISC–4 SSCs*. The licensee shall review changes to the plant, operational practices, applicable plant and industry operational experience, and, as appropriate, update the PRA and SSC categorization and treatment processes. The licensee shall perform this review in a timely manner but no longer than once every two refueling outages.

(2) *RISC–1 and RISC–2 SSCs*. The licensee shall monitor the performance of RISC–1 and RISC–2 SSCs. The licensee shall make adjustments as necessary to either the categorization or treatment processes so that the categorization process and results are maintained valid.

(3) *RISC–3 SSCs*. The licensee shall consider data collected in § 50.69(d)(2)(i) for RISC–3 SSCs to determine if there are any adverse changes in performance such that the SSC unreliability values approach or exceed the values used in the evaluations conducted to satisfy § 50.69(c)(1)(iv). The licensee shall make adjustments as necessary to the categorization or treatment processes so that the categorization process and results are maintained valid.

(f) *Program documentation, change control and records*. (1) The licensee or applicant shall document the basis for its categorization of any SSC under paragraph (c) of this section before removing any requirements under § 50.69(b)(1) for those SSCs.

(2) Following implementation of this section, licensees and applicants shall update their final safety analysis report (FSAR) to reflect which systems have been categorized, in accordance with § 50.71(e).

(3) When a licensee first implements this section for a SSC, changes to the FSAR for the implementation of the changes in accordance with § 50.69(d) need not include a supporting § 50.59 evaluation of the changes directly related to implementation. Thereafter, changes to the programs and procedures for implementation of § 50.69(d), as described in the FSAR, may be made if the requirements of this section and § 50.59 continue to be met.

(4) When a licensee first implements this section for a SSC, changes to the quality assurance plan for the implementation of the changes in accordance with § 50.69(d) need not include a supporting § 50.54(a) review of the changes directly related to implementation. Thereafter, changes to the programs and procedures for implementation of § 50.69(d), as described in the quality assurance plan may be made if the requirements of this section and § 50.54(a) continue to be met.

(g) *Reporting*. The licensee shall submit a licensee event report under § 50.73(b) for any event or condition that prevented, or would have prevented, a RISC–1 or RISC–2 SSC from performing a safety significant function.

[69 FR 68047, Nov. 22, 2004]

INSPECTIONS, RECORDS, REPORTS, NOTIFICATIONS

§ 50.70 Inspections.

(a) Each applicant for or holder of a license, including a construction permit or an early site permit, shall permit inspection, by duly authorized representatives of the Commission, of his 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, including Section 105 of the Act, and the Energy Reorganization Act of 1974, as amended.

(b)(1) Each licensee and each holder of a construction permit shall, 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 shall be furnished by each licensee and each holder of a construction permit. The office shall be convenient to and have full access to the facility and shall provide the inspector both visual and acoustic privacy.

(2) For a site with a single power reactor or fuel facility licensed under part 50 or part 52 of this chapter, or a facility issued a manufacturing license under part 52, the space provided shall be adequate to accommodate a full-time inspector, a part-time secretary and transient NRC personnel and will 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 containing multiple power reactor units or fuel facilities, additional space may be requested to accommodate additional full-time inspector(s). The office space that is provided shall 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) The licensee or construction permit holder shall 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.

(4) The licensee or construction permit holder (nuclear power reactor only) shall ensure that the arrival and presence of an NRC inspector, who has been properly authorized facility access as described in paragraph (b)(3) 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.

[21 FR 355, Jan. 19, 1956; 44 FR 47919, Aug. 16, 1979, as amended at 52 FR 31612, Aug. 21, 1987; 53 FR 42942, Oct. 25, 1988; 72 FR 49501, Aug. 28, 2007; 73 FR 5722, Jan. 31, 2008; 75 FR 73944, Nov. 30, 2010; 84 FR 65644, Nov. 29, 2019]

§ 50.71 Maintenance of records, making of reports.

(a) Each licensee, including each holder of a construction permit or early site permit, shall 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, including Section 105 of the Act, and the Energy Reorganization Act of 1974, as amended. Reports must be submitted in accordance with § 50.4 or 10 CFR 52.3, as applicable.

(b) With respect to any production or utilization facility of a type described in § 50.21(b) or 50.22, or a testing facility, each licensee and each holder of a construction permit shall submit its annual financial report, including the certified financial statements, to the Commission, as specified in § 50.4, 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 the Federal Energy

Regulatory Commission, need not submit the annual financial report or the certified financial statement under this paragraph.

(c) Records that are required by the regulations in this part or part 52 of this chapter, 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 maintained under this part or part 52 of this chapter may be the original or a reproduced copy or 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 shall maintain adequate safeguards against tampering with, and loss of records.

(2) If there is a conflict between the Commission's regulations in 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 this part for such records shall apply unless the Commission, pursuant to §50.12 of this part, has granted a specific exemption from the record retention requirements specified in the regulations in this part.

(e) Each person licensed to operate a nuclear power reactor under the provisions of §50.21 or §50.22, and each applicant for a combined license under part 52 of this chapter, shall update periodically, as provided in paragraphs (e) (3) and (4) of this section, the final safety analysis report (FSAR) originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the applicant or licensee or prepared by the applicant or licensee pursuant to Commission requirement since the submittal of the original FSAR, or as appropriate, the last update to the FSAR under this section. The submittal shall include the effects[1] of all changes made in the facility or procedures as described in the FSAR; all 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 §50.59(c)(2) or, in the case of a license that references a certified design, in accordance with §52.98(c) of this chapter; and all analyses of new safety issues performed by or on behalf of the applicant or licensee at Commission request. The updated information shall be appropriately located within the update to the FSAR.

(1) The licensee shall submit revisions containing updated information to the Commission, as specified in §50.4, on a replacement-page basis that is accompanied by a list which identifies the current pages of the FSAR following page replacement.

(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 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 §50.59 but not previously submitted to the Commission.

(3)(i) A revision of the original FSAR containing those original pages that

[1] Effects of changes includes appropriate revisions of descriptions in the FSAR such that the FSAR (as updated) is complete and accurate.

are still applicable plus new replacement pages shall be filed within 24 months of either July 22, 1980, or the date of issuance of the operating license, whichever is later, and shall bring the FSAR up to date as of a maximum of 6 months prior to the date of filing the revision.

(ii) [Reserved]

(iii) During the period from the docketing of an application for a combined license under subpart C of part 52 of this chapter until the Commission makes the finding under § 52.103(g) of this chapter, the update to the FSAR must be submitted annually.

(4) Subsequent revisions must be filed annually or 6 months after each refueling outage provided the interval between successive updates does not exceed 24 months. The revisions must reflect all changes up to a maximum of 6 months prior to the date of filling. For nuclear power reactor facilities that have submitted the certifications required by § 50.82(a)(1), subsequent revisions must be filed every 24 months.

(5) Each replacement page shall include both a change indicator for the area changed, e.g., a bold line vertically drawn in the margin adjacent to the portion actually changed, and a page change identification (date of change or change number or both).

(6) The updated FSAR shall be retained by the licensee until the Commission terminates their license.

(f) Each person licensed to manufacture a nuclear power reactor under subpart F of 10 CFR part 52 shall update the FSAR originally submitted as part of the application to reflect any modification to the design that is approved by the Commission under § 52.171 of this chapter, and any new analyses of the design performed by or on behalf of the licensee at the NRC's request. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the licensee or prepared by the licensee with respect to the modification approved under § 52.171 of this chapter or the analyses requested by the Commission under § 52.171 of this chapter. The updated information shall be appropriately located within the update to the FSAR.

(g) The provisions of this section apply to nuclear power reactor licensees that have submitted the certification of permanent cessation of operations required under §§ 50.82(a)(1)(i) or 52.110(a)(1) of this chapter. The provisions of paragraphs (a), (c), and (d) of this section also apply to non-power reactor licensees that are no longer authorized to operate.

(h)(1) No later than the scheduled date for initial loading of fuel, each holder of a combined license under subpart C of 10 CFR part 52 shall develop a level 1 and a level 2 probabilistic risk assessment (PRA). The PRA must cover those initiating events and modes for which NRC-endorsed consensus standards on PRA exist one year prior to the scheduled date for initial loading of fuel.

(2) Each holder of a combined license shall maintain and upgrade the PRA required by paragraph (h)(1) of this section. The upgraded PRA must cover initiating events and modes of operation contained in NRC-endorsed consensus standards on PRA in effect one year prior to each required upgrade. The PRA must be upgraded every four years until the permanent cessation of operations under § 52.110(a) of this chapter.

(3) Each holder of a combined license shall, no later than the date on which the licensee submits an application for a renewed license, upgrade the PRA required by paragraph (h)(1) of this section to cover all modes and all initiating events.

[33 FR 9704, July 4, 1968, as amended at 41 FR 18303, May 3, 1976; 45 FR 30615, May 9, 1980; 51 FR 40310, Nov. 6, 1986; 53 FR 19250, May 27, 1988; 57 FR 39358, Aug. 31, 1992; 61 FR 39301, July 29, 1996; 64 FR 53614, Oct. 4, 1999; 71 FR 29246, May 22, 2006; 72 FR 49501, Aug. 28, 2007; 86 FR 43402, Aug. 9, 2021]

EFFECTIVE DATE NOTE: See 64 FR 53582, Oct. 4, 1999, for effectiveness of § 50.71(e) introductory text.

§ 50.72 Immediate notification requirements for operating nuclear power reactors.

(a) *General requirements.*[1] (1) Each nuclear power reactor licensee licensed under §§ 50.21(b) or 50.22 holding an operating license under this part or a combined license under part 52 of this chapter after the Commission makes the finding under § 52.103(g), shall 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;[2] 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 shall 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 shall 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 shall activate the Emergency Response Data System (ERDS)[3] as soon as possible but not later than one hour after declaring an Emergency Class of alert, site area emergency, or general emergency. The ERDS 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 shall 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 shall 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 § 50.54(x) of this part.

(2) *Four-hour reports.* If not reported under paragraphs (a) or (b)(1) of this section, the licensee shall 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 nuclear plant shutdown required by the plant's Technical Specifications.

(ii)–(iii) [Reserved]

(iv)(A) Any event that results or should have resulted in emergency core cooling system (ECCS) discharge into the reactor coolant system as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

(B) 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.

(v)–(x) [Reserved]

(xi) 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 shall notify the NRC as soon as practical and in all cases within eight hours of the occurrence of any of the following:

[1] Other requirements for immediate notification of the NRC by licensed operating nuclear power reactors are contained elsewhere in this chapter, in particular §§ 20.1906, 20.2202, 50.36, 72.74, 72.75, and 73.71.

[2] These Emergency Classes are addressed in Appendix E of this part.

[3] Requirements for ERDS are addressed in Appendix E, Section VI.

(i) [Reserved]

(ii) Any event or condition that results in:

(A) The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or

(B) The nuclear power plant being in an unanalyzed condition that significantly degrades plant safety.

(iii) [Reserved]

(iv)(A) Any event or condition that results in valid actuation of any of the systems listed in paragraph (b)(3)(iv)(B) of this section, except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

(B) The systems to which the requirements of paragraph (b)(3)(iv)(A) of this section apply are:

(*1*) Reactor protection system (RPS) including: Reactor scram and reactor trip.[4]

(*2*) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs).

(*3*) Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: High-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems.

(*4*) ECCS for boiling water reactors (BWRs) including: High-pressure and low-pressure core spray systems; high-pressure coolant injection system; low pressure injection function of the residual heat removal system.

(*5*) BWR reactor core isolation cooling system; isolation condenser system; and feedwater coolant injection system.

(*6*) PWR auxiliary or emergency feedwater system.

(*7*) Containment heat removal and depressurization systems, including containment spray and fan cooler systems.

(*8*) Emergency ac electrical power systems, including: Emergency diesel generators (EDGs); hydroelectric facilities used in lieu of EDGs at the Oconee Station; and BWR dedicated Division 3 EDGs.

(v) Any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to:

(A) Shut down the reactor and maintain it in a safe shutdown condition;

(B) Remove residual heat;

(C) Control the release of radioactive material; or

(D) Mitigate the consequences of an accident.

(vi) Events covered in paragraph (b)(3)(v) of this section 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 paragraph (b)(3)(v) of this section if redundant equipment in the same system was operable and available to perform the required safety function.

(vii)–(xi) [Reserved]

(xii) Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment.

(xiii) 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 telephone notifications made under paragraphs (a) and (b) of this section, in addition to making the required initial notification, each licensee, shall 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

[4] Actuation of the RPS when the reactor is critical is reportable under paragraph (b)(2)(iv)(B) of this section.

of response or protective measures taken, and (iii) information related to plant behavior that is not understood.

(3) Maintain an open, continuous communication channel with the NRC Operations Center upon request by the NRC.

[48 FR 39046, Aug. 29, 1983; 48 FR 40882, Sept. 12, 1983; 55 FR 29194, July 18, 1990, as amended at 56 FR 944, Jan. 10, 1991; 56 FR 23473, May 21, 1991; 56 FR 40184, Aug. 13, 1991; 57 FR 41381, Sept. 10, 1992; 58 FR 67661, Dec. 22, 1993; 59 FR 14087, Mar. 25, 1994; 65 FR 63786, Oct. 25, 2000; 72 FR 49502, Aug. 28, 2007; 85 FR 65662, Oct. 16, 2020; 87 FR 20697, Apr. 8, 2022]

§ 50.73 Licensee event report system.

(a) *Reportable events.* (1) The holder of an operating license under this part or a combined license under part 52 of this chapter (after the Commission has made the finding under § 52.103(g) of this chapter) for a nuclear power plant (licensee) shall submit a Licensee Event Report (LER) for any event of the type described in this paragraph within 60 days after the discovery of the event. In the case of an invalid actuation reported under § 50.73(a)(2)(iv), other than actuation of the reactor protection system (RPS) 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 shall 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 shall report:

(i)(A) The completion of any 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 discovery of the event.

(C) Any deviation from the plant's Technical Specifications authorized pursuant to § 50.54(x) of this part.

(ii) Any event or condition that resulted in:

(A) The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or

(B) The nuclear power plant being in an unanalyzed condition that significantly degraded plant safety.

(iii) Any natural phenomenon or other external condition that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant.

(iv)(A) Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section, except when:

(*1*) The actuation resulted from and was part of a pre-planned sequence during testing or reactor operation; or

(*2*) The actuation was invalid and;

(*i*) Occurred while the system was properly removed from service; or

(*ii*) Occurred after the safety function had been already completed.

(B) The systems to which the requirements of paragraph (a)(2)(iv)(A) of this section apply are:

(*1*) Reactor protection system (RPS) including: reactor scram or reactor trip.

(*2*) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs).

(*3*) Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: high-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems.

(*4*) ECCS for boiling water reactors (BWRs) including: high-pressure and low-pressure core spray systems; high-pressure coolant injection system; low pressure injection function of the residual heat removal system.

(*5*) BWR reactor core isolation cooling system; isolation condenser system; and feedwater coolant injection system.

(*6*) PWR auxiliary or emergency feedwater system.

(*7*) Containment heat removal and depressurization systems, including containment spray and fan cooler systems.

(*8*) Emergency ac electrical power systems, including: emergency diesel generators (EDGs); hydroelectric facilities used in lieu of EDGs at the Oconee Station; and BWR dedicated Division 3 EDGs.

(*9*) Emergency service water systems that do not normally run and that serve as ultimate heat sinks.

(v) Any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to:

(A) Shut down the reactor and maintain it in a safe shutdown condition;

(B) Remove residual heat;

(C) Control the release of radioactive material; or

(D) Mitigate the consequences of an accident.

(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, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to paragraph (a)(2)(v) of this section if redundant equipment in the same system was operable and available to perform the required safety function.

(vii) Any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a single system designed to:

(A) Shut down the reactor and maintain it in a safe shutdown condition;

(B) Remove residual heat;

(C) Control the release of radioactive material; or

(D) Mitigate the consequences of an accident.

(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 exceeded 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)(A) Any event or condition that as a result of a single cause could have prevented the fulfillment of a safety function for two or more trains or channels in different systems that are needed to:

(*1*) Shut down the reactor and maintain it in a safe shutdown condition;

(*2*) Remove residual heat;

(*3*) Control the release of radioactive material; or

(*4*) Mitigate the consequences of an accident.

(B) Events covered in paragraph (a)(2)(ix)(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)(ix)(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.

(x) Any event that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases.

(b) *Contents*. The Licensee Event Report 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 power 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 times of 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) The Energy Industry Identification System component function identifier and system name of each component or system referred to in the LER.

(*1*) The Energy Industry Identification System is defined in: IEEE Std 803–1983 (May 16, 1983) Recommended Practice for Unique Identification in Power Plants and Related Facilities—Principles and Definitions.

(*2*) IEEE Std 803–1983 has been approved for incorporation by reference by the Director of the Federal Register in accordance with 5 U.S.C. 552(a) and 1 CFR part 51.

(*3*) A notice of any changes made to the material incorporated by reference will be published in the FEDERAL REGISTER. Copies may be obtained from the Institute of Electrical and Electronics Engineers, 445 Hoes Lane, P.O. Box 1331, Piscataway, NJ 08855–1331. IEEE Std 803–1983 is available for inspection at the NRC's Technical Library, which is located in the Two White Flint North Building, 11545 Rockville Pike, Rockville, Maryland 20852–2738; or at the National Archives and Records Administration (NARA). For information on the availability of this material at NARA, call 202–741–6030, or go to: *http://www.archives.gov/federal_register/code_of_federal_regulations/ibr_locations.html.*

(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 train of a safety system inoperable, an estimate of the elapsed time from the discovery of the failure until the train 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 shall 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 shutdown, the availability of systems or components that are needed to shutdown 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 telephone number 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 shall submit, as specified in § 50.4, the requested information as a supplement to the initial LER.

(d) *Submission of reports.* Licensee Event Reports must be prepared on Form NRC 366 and submitted to the U.S. Nuclear Regulatory Commission, as specified in § 50.4.

(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 Occurrences" as defined in individual plant Technical Specifications.

[48 FR 33858, July 26, 1983, as amended at 49 FR 47824, Dec. 7, 1984; 51 FR 40310, Nov. 6, 1986; 56 FR 23473, May 21, 1991; 56 FR 61352, Dec. 3, 1991; 57 FR 41381, Sept. 10, 1992; 58 FR 67661, Dec. 22, 1993; 59 FR 50689, Oct. 5, 1994; 63 FR 50480, Sept. 22, 1998; 65 FR 63787, Oct. 25, 2000; 69 FR 18803, Apr. 9, 2004; 72 FR 49502, Aug. 28, 2007]

§ 50.74 Notification of change in operator or senior operator status.

Each licensee shall notify the appropriate NRC contact, as described in § 55.5 of this chapter, within 30 days of the following in regard to a licensed operator or senior operator:

(a) Permanent reassignment from the position for which the licensee has certified the need for a licensed operator or senior operator under § 55.31(a)(3) of this chapter;

(b) Termination of any operator or senior operator;

(c) Permanent disability or illness as described in § 55.25 of this chapter.

[52 FR 9469, Mar. 25, 1987, as amended at 60 FR 13616, Mar. 14, 1995; 68 FR 58809, Oct. 10, 2003; 86 FR 67842, Nov. 30, 2021]

§ 50.75 Reporting and recordkeeping for decommissioning planning.

(a) This section establishes requirements for indicating to NRC how a licensee will provide reasonable assurance that funds will be available for the decommissioning process. For power reactor licensees (except a holder of a manufacturing license under part 52 of this chapter), reasonable assurance consists of a series of steps as provided in paragraphs (b), (c), (e), and (f) of this section. Funding for the decommissioning of power reactors 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) that have jurisdiction over rate regulation. The requirements of this section, in particular paragraph (c) of this section, are in addition to, and not substitution for, other requirements, and are not intended to be used by themselves or by other agencies to establish rates.

(b) Each power reactor applicant for or holder of an operating license, and each applicant for a combined license under subpart C of 10 CFR part 52 for a production or utilization facility of the type and power level specified in paragraph (c) of this section shall submit a decommissioning report, as required by § 50.33(k).

(1) For an applicant for or holder of an operating license under part 50, the report must contain a certification that financial assurance for decommissioning will be (for a license applicant), or has been (for a license holder), provided in an amount which may be more, but not less, than the amount stated in the table in paragraph (c)(1) of this section adjusted using a rate at least equal to that stated in paragraph (c)(2) of this section. For an applicant for a combined license under subpart C of 10 CFR part 52, the report must contain a certification that financial assurance for decommissioning will be provided no later than 30 days after the Commission publishes notice in the FEDERAL REGISTER under § 52.103(a) in an amount which may be more, but not less, than the amount stated in the table in paragraph (c)(1) of this section, adjusted using a rate at least equal to that stated in paragraph (c)(2) of this section.

(2) The amount to be provided must be adjusted annually using a rate at

least equal to that stated in paragraph (c)(2) of this section.

(3) The amount must be covered by one or more of the methods described in paragraph (e) of this section as acceptable to the NRC.

(4) The amount stated in the applicant's or licensee's certification may be based on a cost estimate for decommissioning the facility. As part of the certification, a copy of the financial instrument obtained to satisfy the requirements of paragraph (e) of this section must be submitted to NRC; *provided, however,* that an applicant for or holder of a combined license need not obtain such financial instrument or submit a copy to the Commission except as provided in paragraph (e)(3) of this section.

(c) Table of minimum amounts (January 1986 dollars) required to demonstrate reasonable assurance of funds for decommissioning by reactor type and power level, P (in MWt); adjustment factor.[1]

	Millions
(1)(i) For a PWR:	
greater than or equal to 3400 MWt	$105
between 1200 MWt and 3400 MWt (For a PWR of less than 1200 MWt, use P = 1200 MWt)	$(75 + 0.0088P)
(ii) For a BWR:	
greater than or equal to 3400 MWt	$135
between 1200 MWt and 3400 MWt (For a BWR of less than 1200 MWt, use P = 1200 MWt)	$(104 + 0.009P)

(2) An adjustment factor at least equal to 0.65 L + 0.13 E + 0.22 B is to be used 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."

(d)(1) Each non-power reactor applicant for or holder of an operating license for a production or utilization facility shall submit a decommissioning report as required by § 50.33(k) of this part.

(2) The report must:

(i) Contain a cost estimate for decommissioning the facility;

(ii) Indicate which method or methods described in paragraph (e) of this section as acceptable to the NRC will be used to provide funds for decommissioning; and

(iii) Provide a description of the means of adjusting the cost estimate and associated funding level periodically over the life of the facility.

(e)(1) Financial assurance is to be provided by the following methods.

(i) *Prepayment.* Prepayment is the deposit made preceding the start of operation or the transfer of a license under § 50.80 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 at the time permanent termination of operations is expected. 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 estimate under § 50.75(b)(1) of this section may take credit for projected earnings on the prepaid decommissioning trust funds, using up to a 2 percent annual real rate of return from the time of future funds' collection through the projected decommissioning period, provided that the site-specific estimate is based on a

[1] Amounts are based on activities related to the definition of "Decommission" in § 50.2 of this part and do not include the cost of removal and disposal of spent fuel or of nonradioactive structures and materials beyond that necessary to terminate the license.

period of safe storage that is specifically described in the estimate. This includes the periods of safe storage, final dismantlement, and license termination. A licensee that has prepaid funds based on the formulas in § 50.75(c) of this section may take credit for projected earnings on the prepaid decommissioning funds using up to a 2 percent annual real rate of return up to the time of permanent termination of operations. A licensee may use a credit of greater than 2 percent if the licensee's rate-setting authority has specifically authorized a higher rate. However, licensees certifying only to the formula amounts (*i.e.*, not a site-specific estimate) can take a pro-rata credit during the immediate dismantlement period (*i.e.*, recognizing both cash expenditures and earnings the first 7 years after shutdown). Actual earnings on existing funds may be used to calculate future fund needs.

(ii) *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 at the time permanent termination of operations is expected. 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 linking fund is managed are regulated and examined by a Federal or State agency. A licensee that has collected funds based on a site-specific estimate under § 50.75(b)(1) of this section 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 decommissioning period, provided that the site-specific estimate is based on a period of safe storage that is specifically described in the estimate. This includes the periods of safe storage, final dismantlement, and license termination. A licensee that has collected funds based on the formulas in § 50.75(c) of this section may take credit for collected earnings on the decommissioning funds using up to a 2 percent annual real rate of return up to the time of permanent termination of operations. A licensee may use a credit of greater than 2 percent if the licensee's rate-setting authority has specifically authorized a higher rate. However, licensees certifying only to the formula amounts (*i.e.*, not a site-specific estimate) can take a pro-rata credit during the dismantlement period (*i.e.*, recognizing both cash expenditures and earnings the first 7 years after shutdown). 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, (e)(1)(ii). This method may be used as the exclusive mechanism relied upon for providing financial assurance for decommissioning in the following circumstances:

(A) 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 assumed to meet this condition.

(B) 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 §§ 50.75(c), 50.75(f), or 50.82 of this part.

(iii) A surety method, insurance, or other guarantee method:

(A) 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:

(*1*) 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.

(*2*) 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.

(B) 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.

(C) 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. For non-profit entities, such as colleges, universities, and non-profit hospitals, a guarantee of funds by the applicant or licensee may be used if the guarantee and test are as contained in appendix E 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.

(iv) For a power reactor licensee that is a Federal licensee, or for a non-power reactor licensee that is a Federal, State, or local government licensee, a statement of intent containing a cost estimate for decommissioning, and indicating that funds for decommissioning will be obtained when necessary.

(v) 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 §§ 50.75(c), 50.75(f), or § 50.82. To be acceptable to the NRC as a method of decommissioning funding assurance, the terms of the contract(s) shall include provisions that the electricity buyer(s) will pay for the decommissioning obligations specified in the contract(s), notwithstanding the operational status either of the licensed power reactor 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.

(vi) 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 (e)(1)(i) through (v) of this section. Licensees who do not have sources of funding described in paragraph (e)(1)(ii) of this section may use an external sinking fund in combination with a guarantee mechanism, as specified in paragraph (e)(1)(iii) of this section, provided that the total amount of funds estimated to be necessary for decommissioning is assured.

(2) 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 the 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.

(3) Each holder of a combined license under subpart C of 10 CFR part 52 shall, 2 years before and 1 year before the scheduled date for initial loading of fuel, consistent with the schedule required by § 52.99(a), submit a report to the NRC containing a certification updating the information described under paragraph (b)(1) of this section, including a copy of the financial instrument to be used. No later than 30 days after the Commission publishes notice in the FEDERAL REGISTER under 10 CFR 52.103(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 the requirements of paragraph (e) of this section.

(f)(1) Each power reactor licensee shall report, on a calendar-year basis, to the NRC by March 31, 1999, and at least once every 2 years thereafter on the status of its decommissioning funding for each reactor or part of a reactor that it owns. However, each holder of a combined license under part 52 of this chapter need not begin reporting until the date that the Commission has made the finding under § 52.103(g) of this chapter. The information in this report must include, at a minimum, the amount of decommissioning funds estimated to be required pursuant to 10 CFR 50.75(b) and (c); 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 paragraph (e)(1)(v) of this section; 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.

(2) Each power reactor licensee shall report, on a calendar-year basis, to the NRC by March 31, 1999, and at least once every 2 years thereafter on the status of its 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 10 CFR 50.75(b) and (c); 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 paragraph (e)(1)(v) of this section; 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.

(3) Each power reactor licensee shall at or about 5 years prior to the projected end of operations submit a preliminary decommissioning cost estimate which includes an up-to-date assessment of the major factors that could affect the cost to decommission.

(4) Each non-power reactor licensee shall at or about 2 years prior to the projected end of operations submit a preliminary decommissioning plan containing a cost estimate for decommissioning and an up-to-date assessment of the major factors that could affect planning for decommissioning. Factors to be considered in submitting this preliminary plan information include—

(i) The decommissioning alternative anticipated to be used. The requirements of § 50.82(b)(4)(i) must be considered at this time;

(ii) Major technical actions necessary to carry out decommissioning safely;

(iii) The current situation with regard to disposal of high-level and low-level radioactive waste;

(iv) Residual radioactivity criteria;

(v) Other site specific factors which could affect decommissioning planning and cost.

(5) If necessary, the cost estimate, for power and non-power reactors, shall also 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.

(g) 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, 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 (g)(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 § 50.83, 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.

(h)(1) Licensees that are not ''electric utilities'' as defined in § 50.2 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—

(i) The trustee, manager, investment advisor, or other person directing investment of the funds:

(A) Is prohibited from investing the funds in securities or other obligations of the licensee or any other owner or operator of any nuclear power reactor 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 nuclear power 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 subsection 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 nuclear power plants.

(B) 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 the Federal Energy Regulatory Commission's "Regulations Governing Nuclear Plant Decommissioning Trust Funds" at 18 CFR 35.32(a)(3), or any successor regulation.

(ii) 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.

(iii) 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; and

(iv) Except for withdrawals being made under §50.82(a)(8) 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 paragraph (e) of this section until final decommissioning has been completed. After decommissioning has begun and withdrawals from the decommissioning fund are made under § 50.82(a)(8), no further notification need be made to the NRC.

(2) Licensees that are ''electric utilities'' under § 50.2 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 § 50.82(a)(8) 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 paragraph (e) of this section until final decommissioning has been completed. After decommissioning has begun and withdrawals from the decommissioning fund are made under § 50.82(a)(8), no further notification need be made to the NRC.

(3) A licensee that is not an ''electric utility'' under § 50.2 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 paragraphs (h)(1)(i), (ii), (iii), and (iv) of this section.

(4) Unless otherwise determined by the Commission with regard to a specific application, the Commission has determined that any amendment to the license of a utilization facility that does no more than delete specific license conditions relating to the terms and conditions of decommissioning trust agreements involves ''no significant hazards consideration.''

(5) The provisions of paragraphs (h)(1) through (h)(3) of this section do not apply to any licensee that as of December 24, 2003, has existing license conditions relating to decommissioning trust agreements, so long as the licensee does not elect to amend those license conditions. If a licensee with existing license conditions relating to decommissioning trust agreements elects to amend those conditions, the license amendment shall be in accordance with the provisions of paragraph (h) of this section.

[53 FR 24049, June 27, 1988, as amended at 58 FR 68731, Dec. 29, 1993; 59 FR 1618, Jan. 12, 1994; 61 FR 39301, July 29, 1996; 63 FR 50480, Sept. 22, 1998; 63 FR 57236, Oct. 27, 1998; 68 FR 19727, Apr. 22, 2003; 67 FR 78350, Dec. 24, 2002; 68 FR 12571, Mar. 17, 2003; 68 FR 65388, Nov. 20, 2003; 72 FR 49502, Aug. 28, 2007; 73 FR 5722, Jan. 31, 2008; 77 FR 35571, June 17, 2012; 83 FR 30288, June 28, 2018; 84 FR 65644, Nov. 29, 2019; 87 FR 68031, Nov. 14, 2022]

§ 50.76 Licensee's change of status; financial qualifications.

An electric utility licensee holding an operating license (including a renewed license) for a nuclear power reactor, no later than seventy-five (75) days prior to ceasing to be an electric utility in any manner not involving a license transfer under § 50.80, shall provide the NRC with the financial qualifications information that would be required for obtaining an initial operating license as specified in § 50.33(f)(2). The financial qualifications information must address the first full five years of operation after the date the licensee ceases to be an electric utility.

[69 FR 4448, Jan. 30, 2004]

US/IAEA SAFEGUARDS AGREEMENT

§ 50.78 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 shall submit facility information, as described in § 75.10 of this chapter, on IAEA Design Information Questionnaire forms and site information on DOC/NRC Form AP–A and associated forms;

(b) As required by the Additional Protocol, shall submit location information described in § 75.11 of this chapter on DOC/NRC Form AP–1 and associated forms; and

(c) Shall permit verification thereof by the International Atomic Energy Agency (IAEA) and take other action as necessary to implement the US/IAEA Safeguards Agreement, as described in part 75 of this chapter.

[73 FR 78605, Dec. 23, 2008, as amended at 85 FR 65663, Oct. 16, 2020]

TRANSFERS OF LICENSES—CREDITORS' RIGHTS—SURRENDER OF LICENSES

§ 50.80 Transfer of licenses.

(a) No license for a production or utilization facility (including, but not limited to, permits under this part and part 52 of this chapter, and licenses under parts 50 and 52 of this chapter), 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) For a construction permit or operating license under this part, as much of the information described in §§ 50.33 and 50.34 of this part 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) For an early site permit under part 52 of this chapter, as much of the information described in §§ 52.16 and 52.17 of this chapter with respect to the identity and technical qualifications of the proposed transferee as would be required by those sections if the application were for an initial license.

(iii) For a combined license under part 52 of this chapter, as much of the information described in §§ 52.77 and 52.79 of this chapter 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.

(iv) For a manufacturing license under part 52 of this chapter, as much of the information described in §§ 52.156 and 52.157 of this chapter with respect to the identity and technical qualifications of the proposed transferee as would be required by those sections if the application were for an initial license.

(2) The application shall include also 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 pursuant to § 50.37. 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.

[26 FR 9546, Oct. 10, 1961, as amended at 35 FR 19661, Dec. 29, 1970; 38 FR 3956, Feb. 9, 1973; 65 FR 44660, July 19, 2000; 70 FR 61888, Oct. 27, 2005; 72 FR 49503, Aug. 28, 2007]

§ 50.81 Creditor regulations.

(a) Pursuant to section 184 of the Act, the Commission consents, without individual application, to the creation of any mortgage, pledge, or other lien upon any production or utilization 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 Atomic Energy Act of 1954, as amended, and regulations issued by the Commission pursuant to said Act; 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 § 50.80(b). The Commission will act upon such application pursuant to § 50.80 (c).

(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 this chapter, any construction permit under this part, and any early site permit under part 52 of this chapter, 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 subpart A of part 52 of this chapter, and a reactor manufactured under a manufacturing license under subpart F of part 52 of this chapter.

[26 FR 9546, Oct. 10, 1961, as amended at 32 FR 2562, Feb. 7, 1967; 72 FR 49504, Aug. 28, 2007]

§ 50.82 Termination of license.

For power reactor licensees who, before the effective date of this rule, either submitted a decommissioning plan for approval or possess an approved decommissioning plan, the plan is considered to be the PSDAR submittal required under paragraph (a)(4) of this section and the provisions of this section apply accordingly. For power reactor licensees whose decommissioning plan approval activities have been relegated to notice of opportunity for a hearing under subpart G of 10 CFR part 2, the public meeting convened and 90-day delay of major decommissioning activities required in paragraphs (a)(4)(ii) and (a)(5) of this section shall not apply, and any orders arising from proceedings under subpart G of 10 CFR part 2 shall continue and remain in effect absent any orders from the Commission.

(a) For power reactor licensees—

(1)(i) When a licensee has determined to permanently cease operations the licensee shall, within 30 days, submit a written certification to the NRC, consistent with the requirements of § 50.4(b)(8);

(ii) 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 § 50.4(b)(9) and;

(iii) For licensees whose licenses have been permanently modified to allow possession but not operation of the facility, before the effective date of this rule, the certifications required in paragraphs (a)(1) (i)–(ii) of this section shall be deemed to have been submitted.

(2) Upon docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel, or when a final legally effective order to permanently cease operations has come into effect, the 10 CFR part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel.

(3) 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.

(4)(i) Prior to or within 2 years following permanent cessation of operations, the licensee shall submit a post-shutdown decommissioning activities report (PSDAR) to the NRC, and a copy to the affected State(s). The 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.

(ii) The NRC shall notice receipt of the PSDAR and make the PSDAR available for public comment. The NRC shall also schedule a public meeting in the vicinity of the licensee's facility upon receipt of the PSDAR. The NRC shall publish a 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.

(5) Licensees shall not perform any major decommissioning activities, as defined in § 50.2, 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 § 50.82(a)(1), have been submitted.

(6) Licensees shall not perform any decommissioning activities, as defined in § 50.2, that—

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

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

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

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

(8)(i) Decommissioning trust funds may be used by licensees if—

(A) The withdrawals are for expenses for legitimate decommissioning activities consistent with the definition of decommissioning in § 50.2;

(B) 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;

(C) 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.

(ii) Initially, 3 percent of the generic amount specified in § 50.75 may be used for decommissioning planning. For licensees that have submitted the certifications required under § 50.82(a)(1) and commencing 90 days after the NRC has received the PSDAR, an additional 20 percent may be used. A site-specific decommissioning cost estimate must be submitted to the NRC prior to the licensee using any funding in excess of these amounts.

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

(iv) 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.

(v) After submitting its site-specific DCE required by paragraph (a)(4)(i) 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:

(A) 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;

(B) 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;

(C) Any modifications occurring to a licensee's current method of providing financial assurance since the last submitted report; and

(D) Any material changes to trust agreements or financial assurance contracts.

(vi) 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.

(vii) After submitting its site-specific DCE required by paragraph (a)(4)(i) 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:

(A) The amount of funds accumulated to cover the cost of managing the irradiated fuel;

(B) 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

(C) If the funds accumulated do not cover the projected cost, a plan to obtain additional funds to cover the cost.

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

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

(ii) The license termination plan must include—

(A) A site characterization;

(B) Identification of remaining dismantlement activities;

(C) Plans for site remediation;

(D) Detailed plans for the final radiation survey;

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

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

(G) 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.

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

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

(10) 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.

(11) 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, 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.

(b) For non-power reactor licensees—

(1) A licensee that permanently ceases operations must make application for license termination within 2 years following permanent cessation of operations, and in no case later than 1 year prior to expiration of the operating license. Each application for termination of a license must be accompanied or preceded by a proposed decommissioning plan. The contents of the decommissioning plan are specified in paragraph (b)(4) of this section.

(2) For decommissioning plans in which the major dismantlement activities are delayed by first placing the facility in storage, planning for these delayed activities may be less detailed. Updated detailed plans must be submitted and approved prior to the start of these activities.

(3) For decommissioning plans that delay completion of decommissioning by including a period of storage or surveillance, the licensee shall provide that—

(i) Funds needed to complete decommissioning be placed into an account segregated from the licensee's assets and outside the licensee's administrative control during the storage or surveillance period, or a surety method or fund statement of intent be maintained in accordance with the criteria of § 50.75(e); and

(ii) Means be included for adjusting cost estimates and associated funding levels over the storage or surveillance period.

(4) The proposed decommissioning plan must include—

(i) The choice of the alternative for decommissioning with a description of activities involved. An alternative is acceptable if it provides for completion of decommissioning without significant delay. Consideration will be given to an alternative which provides for delayed completion of decommissioning only when necessary to protect the public health and safety. Factors to be considered in evaluating an alternative which provides for delayed completion of decommissioning include unavailability of waste disposal capacity and other site-specific factors affecting the licensee's capability to carry out decommissioning, including the presence of other nuclear facilities at the site.

(ii) A description of the controls and limits on procedures and equipment to protect occupational and public health and safety;

(iii) A description of the planned final radiation survey;

(iv) An updated cost estimate for the chosen alternative for decommissioning, comparison of that estimate with present funds set aside for decommissioning, and plan for assuring the availability of adequate funds for completion of decommissioning; and

(v) A description of technical specifications, quality assurance provisions and physical security plan provisions in place during decommissioning.

(5) If the decommissioning plan demonstrates that the decommissioning will be performed in accordance with the regulations in this chapter and will not be inimical to the common defense and security or to the health and safety of the public, and after notice to interested persons, the Commission will approve, by amendment, the plan subject to such conditions and limitations as it deems appropriate and necessary. The approved decommissioning plan will be a supplement to the Safety Analysis report or equivalent.

(6) The Commission will terminate the license if it determines that—

(i) The decommissioning has been performed in accordance with the approved decommissioning plan, and

(ii) The terminal radiation survey and associated documentation demonstrate that the facility and site are suitable for release in accordance with the criteria for decommissioning in 10 CFR part 20, subpart E.

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

[61 FR 39301, July 29, 1996, as amended at 62 FR 39091, July 21, 1997; 68 FR 19727, Apr. 22, 2003; 76 FR 35571, June 17, 2011; 79 FR 66603, Nov. 10, 2014]

§ 50.83 Release of part of a power reactor facility or site for unrestricted use.

(a) Prior written NRC approval is required to release part of a facility or site for unrestricted use at any time before receiving approval of a license termination plan. Section 50.75 specifies recordkeeping requirements associated with partial release. Nuclear power reactor licensees seeking NRC 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 10 CFR Part 100 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 facility or site to be released; and

(3) Perform surveys adequate to demonstrate compliance with the radiological criteria for unrestricted use specified in 10 CFR 20.1402 for impacted areas.

(b) For release of non-impacted areas, the licensee may submit a written request for NRC 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 facility or site to be released;

(3) The schedule for release of the property;

(4) The results of the evaluations performed in accordance with § 50.59; 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 an approval request from the licensee for 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) Upon 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 10 CFR 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) Upon 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 in the vicinity of the licensee's facility for the purpose of obtaining public comments on the proposed release of part of the facility or site. The NRC shall publish a document in the FEDERAL REGISTER and in a forum, such as local newspapers, which is readily accessible to individuals in the vicinity of the site, announcing the date, time, and location of the meeting, along with a brief description of the purpose of the meeting.

[68 FR 19727, Apr. 22, 2003]

AMENDMENT OF LICENSE OR CONSTRUCTION PERMIT AT REQUEST OF HOLDER

§ 50.90 Application for amendment of license, construction permit, or early site permit.

Whenever a holder of a license, including a construction permit and operating license under this part, and an early site permit, combined license, and manufacturing license under part 52 of this chapter, desires to amend the license or permit, application for an amendment must be filed with the Commission, as specified in §§ 50.4 or 52.3 of this chapter, as applicable, fully describing the changes desired, and following as far as applicable, the form prescribed for original applications.

[72 FR 49504, Aug. 28, 2007]

§ 50.91 Notice for public comment; State consultation.

The Commission will use the following procedures for an application requesting an amendment to an operating license under this part or a combined license under part 52 of this chapter for a facility licensed under §§ 50.21(b) or 50.22, or for a testing facility, except for amendments subject to hearings governed by 10 CFR part 2, subpart L. For amendments subject to 10 CFR part 2, subpart L, the following procedures will apply only to the extent specifically referenced in § 2.309(b) of this chapter, except that notice of opportunity for hearing must be published in the FEDERAL REGISTER at least 30 days before the requested amendment is issued by the Commission:

(a) *Notice for public comment.* (1) At the time a licensee requests an amendment, it must provide to the Commission, in accordance with the distribution requirements specified in § 50.4, its analysis about the issue of no significant hazards consideration using the standards in § 50.92.

(2)(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 § 50.92, (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.

(3) 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 on no significant hazards consideration, unless it receives a request for a hearing on that amendment request.

(4) 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.

(5) 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 nuclear power plant, 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.

(6) 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)(2) 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;

(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)(2) 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.

(7) 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. (The Commission will make available to the licensee the name of the appropriate State official designated to receive such amendments.)

(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 available to the State official designated to consult with it about its proposed determination the names of the Project Manager or other NRC personnel it designated to consult with the State. 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 telephone 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 telephone 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 nuclear power plants.

[51 FR 7765, Mar. 6, 1986, as amended at 51 FR 40310, Nov. 6, 1986; 61 FR 39303, July 29, 1996; 69 FR 2276, Jan. 14, 2004; 72 FR 49504, Aug. 28, 2007]

§ 50.92 Issuance of amendment.

(a) In determining whether an amendment to a license, construction permit, or early site permit will be issued to the applicant, the Commission will be guided by the considerations which govern the issuance of initial licenses, construction permits, or early site permits 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 nuclear power reactor manufactured under part 52 of this chapter before its installation at a site, or a combined license before the date that the Commission makes the finding under § 52.103(g) of this chapter, 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 nuclear power plant).

(c) The Commission may make a final determination, under the procedures in § 50.91, that a proposed amendment to an operating license or a combined license for a facility or reactor licensed under §§ 50.21(b) or 50.22, or for a testing facility involves no significant hazards consideration, if operation of the facility 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 accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

[51 FR 7767, Mar. 6, 1986, as amended at 72 FR 49504, Aug. 28, 2007]

Revocation, Suspension, Modification, Amendment of Licenses and Construction Permits, Emergency Operations by the Commission

§ 50.100 Revocation, suspension, modification of licenses, permits, and approvals for cause.

A license, permit, or standard design approval under parts 50 or 52 of this chapter 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, permit, or approval on an original application (other than those relating to §§ 50.51, 50.42(a), and 50.43(b)); or for failure to manufacture a reactor, or construct or operate a facility in accordance with the terms of the permit or license, provided, however, that failure to make timely completion of the proposed construction or alteration of a facility under a construction permit under part 50 of this chapter or a combined license under part 52 of this chapter shall be governed by the provisions of § 50.55(b); or for violation of, or failure to observe, any of the terms and provisions of the act, regulations, license, permit, approval, or order of the Commission.

[72 FR 49504, Aug. 28, 2007]

§ 50.101 Retaking possession of special nuclear material.

Upon revocation of a license, the Commission may immediately cause the retaking of possession of all special nuclear material held by the licensee.

[21 FR 355, Jan. 19, 1956, as amended at 40 FR 8790, Mar. 3, 1975]

§ 50.102 Commission order for operation after revocation.

Whenever the Commission finds that the public convenience and necessity, or the Department finds that the production program of the Department requires continued operation of a production or utilization facility, the license for which has been revoked, the Commission may, after consultation with the appropriate federal or state regulatory agency having jurisdiction, order that possession be taken of such facility and that it be operated for a period of time as, in the judgment of the Commission, the public convenience and necessity or the production program of the Department may require, or until a license for operation of the facility shall become effective. Just compensation shall be paid for the use of the facility.

[40 FR 8790, Mar. 3, 1975]

§ 50.103 Suspension and operation in war or national emergency.

(a) Whenever Congress declares that a state of war or national emergency exists, the Commission, if it finds it necessary to the common defense and security, may,

(1) Suspend any license it has issued.

(2) Cause the recapture of special nuclear material.

(3) Order the operation of any licensed facility.

(4) Order entry into any plant or facility in order to recapture special nuclear material or to operate the facility.

(b) Just compensation shall be paid for any damages caused by recapture of special nuclear material or by operation of any facility, pursuant to this section.

[21 FR 355, Jan. 19, 1956, as amended at 35 FR 11416, July 17, 1970; 40 FR 8790, Mar. 3, 1975]

BACKFITTING

§ 50.109 Backfitting.

(a)(1) Backfitting is defined as 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:

(i) The date of issuance of the construction permit for the facility for facilities having construction permits issued after October 21, 1985;

(ii) Six (6) months before the date of docketing of the operating license application for the facility for facilities having construction permits issued before October 21, 1985;

(iii) The date of issuance of the operating license for the facility for facilities having operating licenses;

(iv) The date of issuance of the design approval under subpart E of part 52 of this chapter;

(v) The date of issuance of a manufacturing license under subpart F of part 52 of this chapter;

(vi) The date of issuance of the first construction permit issued for a duplicate design under appendix N of this part; or

(vii) The date of issuance of a combined license under subpart C of part 52 of this chapter, provided that if the combined license references an early site permit, the provisions in § 52.39 of this chapter apply with respect to the site characteristics, design parameters, and terms and conditions specified in the early site permit. If the combined license references a standard design certification rule under subpart B of 10 CFR part 52, the provisions in § 52.63 of this chapter apply with respect to the design matters resolved in the standard design certification rule, provided however, that if any specific backfitting limitations are included in a referenced design certification rule, those limitations shall govern. If the combined license references a standard design approval under subpart E of 10 CFR part 52, the provisions in § 52.145 of this chapter apply with respect to the design matters resolved in the standard design approval. If the combined license uses a reactor manufactured under a manufacturing license under subpart F of 10 CFR part 52, the provisions of § 52.171 of this chapter apply with respect to matters resolved in the manufacturing license proceeding.

(2) Except as provided in paragraph (a)(4) of this section, the Commission shall require a systematic and documented analysis pursuant to paragraph (c) 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 (c) 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) Paragraph (a)(3) of this section shall not apply to backfits imposed prior to October 21, 1985.

(c) 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.

(d) No licensing action will be withheld during the pendency of backfit analyses required by the Commission's rules.

(e) 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 FR 20610, June 6, 1988, as amended at 54 FR 15398, Apr. 18, 1989; 72 FR 49504, Aug. 28, 2007]

ENFORCEMENT

§ 50.110 Violations.

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

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

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

(3) A regulation or order issued 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 Atomic Energy Act:

(1) For violations of—

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

(ii) Section 206 of the Energy Reorganization Act;

(iii) Any 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 Atomic Energy Act of 1954, as amended.

[57 FR 55075, Nov. 24, 1992]

§ 50.111 Criminal penalties.

(a) Section 223 of the Atomic Energy Act of 1954, as amended, provides for criminal sanctions for willful violation of, attempted violation of, or conspiracy to violate, any regulation issued under sections 161b, 161i, or 161o of the Act. For purposes of section 223, all the regulations in part 50 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 10 CFR part 50 that are not issued under sections 161b, 161i, or 161o for the purposes of section 223 are as follows: §§ 50.1, 50.2, 50.3, 50.4, 50.8, 50.11, 50.12, 50.13, 50.20, 50.21, 50.22, 50.23, 50.30, 50.31, 50.32, 50.33, 50.34a, 50.35, 50.36b, 50.37, 50.38, 50.39, 50.40, 50.41, 50.42, 50.43, 50.45, 50.50, 50.51, 50.52, 50.53, 50.56, 50.57, 50.58, 50.81, 50.90, 50.91, 50.92, 50.100, 50.101, 50.102, 50.103, 50.109, 50.110, 50.111.

[57 FR 55075, Nov. 24, 1992, as amended at 61 FR 39303, July 29, 1996]

ADDITIONAL STANDARDS FOR LICENSES, CERTIFICATIONS, AND REGULATORY APPROVALS

§ 50.120 Training and qualification of nuclear power plant personnel.

(a) *Applicability*. The requirements of this section apply to each applicant for and each holder of an operating license issued under this part and each holder of a combined license issued under part 52 of this chapter for a nuclear power plant of the type specified in § 50.21(b) or § 50.22.

(b) *Requirements*. (1)(i) Each nuclear power plant operating license applicant, by 18 months prior to fuel load, and each holder of an operating license shall establish, implement, and maintain a training program that meets the requirements of paragraphs (b)(2) and (b)(3) of this section.

(ii) Each holder of a combined license shall establish, implement, and maintain the training program that meets the requirements of paragraphs (b)(2) and (b)(3) of this section, as described in the final safety analysis report no later than 18 months before the scheduled date for initial loading of fuel.

(2) The training program must be derived from a systems approach to training as defined in 10 CFR 55.4, and must provide for the training and qualification of the following categories of nuclear power plant personnel:

(i) Non-licensed operator.

(ii) Shift supervisor.

(iii) Shift technical advisor.

(iv) Instrument and control technician.

(v) Electrical maintenance personnel.

(vi) Mechanical maintenance personnel.

(vii) Radiological protection technician.

(viii) Chemistry technician.

(ix) Engineering support personnel.

(3) The training program must incorporate the instructional requirements necessary to provide qualified personnel to operate 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. The training program must be periodically evaluated and revised as appropriate to reflect industry experience as well as changes to the facility, procedures, regulations, and quality assurance requirements. The training program must be periodically reviewed by licensee management for effectiveness. Sufficient records must be maintained by the licensee to maintain program integrity and kept available for NRC inspection to verify the adequacy of the program.

[72 FR 49505, Aug. 28, 2007]

§ 50.150 Aircraft impact assessment.

(a) *Assessment requirements.* (1) *Assessment.* Each applicant listed in paragraph (a)(3) shall perform a design-specific assessment of the effects on the facility of the impact of a large, commercial aircraft. Using realistic analyses, the applicant shall identify and incorporate into the design those design features and functional capabilities to show that, with reduced use of operator actions:

(i) The reactor core remains cooled, or the containment remains intact; and

(ii) spent fuel cooling or spent fuel pool integrity is maintained.

(2) *Aircraft impact characteristics.*[1] The assessment must be based on the beyond-design-basis 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 the low altitude representative of a nuclear power plant's low profile.

(3) *Applicability.* The requirements of paragraphs (a)(1) and (a)(2) of this section apply to applicants for:

(i) Construction permits for nuclear power reactors issued under this part after July 13, 2009;

(ii) Operating licenses for nuclear power reactors issued under this part for which a construction permit was issued after July 13, 2009;

(iii)(A) Standard design certifications issued under part 52 of this chapter after July 13, 2009;

(B) Renewal of standard design certifications in effect on July 13, 2009 which have not been amended to comply with the requirements of this section by the time of application for renewal;

(iv) Standard design approvals issued under part 52 of this chapter after July 13, 2009;

(v) Combined licenses issued under part 52 of this chapter that:

(A) Do not reference a standard design certification, standard design approval, or manufactured reactor; or

(B) Reference a standard design certification issued before July 13, 2009 which has not been amended to address the requirements of this section; and

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

(vi) Manufacturing licenses issued under part 52 of this chapter that:

(A) Do not reference a standard design certification or standard design approval; or

(B) Reference a standard design certification issued before July 13, 2009 which has not been amended to address the requirements of this section.

(b) *Content of application.* For applicants identified in paragraph (a)(3) of this section, the preliminary or final safety analysis report, as applicable, must include a description of:

(1) The design features and functional capabilities identified in paragraph (a)(1) of this section; and

(2) How the design features and functional capabilities identified in paragraph (a)(1) of this section meet the assessment requirements in paragraph (a)(1) of this section.

(c) *Control of changes.* (1) For construction permits which are subject to paragraph (a) of this section, if the permit holder changes the information required by 10 CFR 50.34(a)(13) to be included in the preliminary safety analysis report, then the permit holder shall consider the effect of the changed feature or capability on the original assessment required by 10 CFR 50.150(a) and amend the information required by 10 CFR 50.34(a)(13) to be included in the preliminary safety analysis report to describe how the modified design features and functional capabilities continue to meet the assessment requirements in paragraph (a)(1) of this section.

(2) For operating licenses which are subject to paragraph (a) of this section, if the licensee changes the information required by 10 CFR 50.34(b)(12) to be included in the final safety analysis report, then the licensee shall consider the effect of the changed feature or capability on the original assessment required by 10 CFR 50.150(a) and amend the information required by 10 CFR 50.34(b)(12) to be included in the final safety analysis report to describe how the modified design features and functional capabilities continue to meet the assessment requirements in paragraph (a)(1) of this section.

(3) For standard design certifications which are subject to paragraph (a) of this section, generic changes to the information required by 10 CFR 52.47(a)(28) to be included in the final safety analysis report are governed by the applicable requirements of 10 CFR 52.63.

(4)(i) For combined licenses which are subject to paragraph (a) of this section, if the licensee changes the information required by 10 CFR 52.79(a)(47) to be included in the final safety analysis report, then the licensee shall consider the effect of the changed feature or capability on the original assessment required by 10 CFR 50.150(a) and amend the information required by 10 CFR 52.79(a)(47) to be included in the final safety analysis report to describe how the modified design features and functional capabilities continue to meet the assessment requirements in paragraph (a)(1) of this section.

(ii) For combined licenses which are not subject to paragraph (a) of this section but reference a standard design certification which is subject to paragraph (a) of this section, proposed departures from the information required by 10 CFR 52.47(a)(28) to be included in the final safety analysis report for the referenced standard design certification are governed by the change control requirements in the applicable design certification rule.

(iii) For combined licenses which are not subject to paragraph (a) of this section but reference a manufactured reactor which is subject to paragraph (a) of this section, proposed departures from the information required by 10 CFR 52.157(f)(32) to be included in the final safety analysis report for the manufacturing license are governed by the applicable requirements in 10 CFR 52.171(b)(2).

(5)(i) For manufacturing licenses which are subject to paragraph (a) of this section, generic changes to the information required by 10 CFR 52.157(f)(32) to be included in the final safety analysis report are governed by the applicable requirements of 10 CFR 52.171.

(ii) For manufacturing licenses which are not subject to paragraph (a) of this section but reference a standard design certification which is subject to paragraph (a) of this section, proposed departures from the information required by 10 CFR 52.47(a)(28) to be included in the final safety analysis report for the referenced standard design certification are governed by the change control requirements in the applicable design certification rule.

[74 FR 28146, June 12, 2009]

§ 50.155 Mitigation of beyond-design-basis events.

(a) *Applicability*. (1) Each holder of an operating license for a nuclear power reactor under this part and each holder of a combined license under part 52 of this chapter for which the Commission has made the finding under § 52.103(g) of this chapter shall comply with the requirements of this section until submittal of the license holder's certifications described in § 50.82(a)(1) or § 52.110(a) of this chapter.

(2)(i) Once the certifications described in § 50.82(a)(1) or § 52.110(a) of this chapter have been submitted by a licensee subject to the requirements of this section, that licensee need only comply with the requirements of paragraphs (b) through (d) and (f) of this section associated with spent fuel pool cooling capabilities.

(ii) Holders of operating licenses or combined licenses for which the certifications described in § 50.82(a)(1) or § 52.110(a) of this chapter have been submitted need not meet the requirements of this section except for the requirements of paragraph (b)(2) of this section associated with spent fuel pool cooling capabilities once the decay heat of the fuel in the spent fuel pool can be removed solely by heating and boiling of water within the spent fuel pool and the boil-off period provides sufficient time for the licensee to obtain off-site resources to sustain the spent fuel pool cooling function indefinitely, as demonstrated by an analysis performed and retained by the licensee.

(iii) The holder of the license for Millstone Power Station, Unit 1, is not subject to the requirements of this section.

(iv) Holders of operating licenses or combined licenses for which the certifications described in § 50.82(a)(1) or § 52.110(a) of this chapter have been submitted need not meet the requirements of this section once all irradiated fuel has been permanently removed from the spent fuel pool(s).

(b) *Strategies and guidelines*. Each applicant or licensee shall develop, implement, and maintain:

(1) Mitigation strategies for beyond-design-basis external events—Strategies and guidelines to mitigate beyond-design-basis external events from natural phenomena that are developed assuming a loss of all ac power concurrent with either a loss of normal access to the ultimate heat sink or, for passive reactor designs, a loss of normal access to the normal heat sink. These strategies and guidelines must be capable of being implemented site-wide and must include the following:

(i) Maintaining or restoring core cooling, containment, and spent fuel pool cooling capabilities; and

(ii) The acquisition and use of offsite assistance and resources to support the functions required by paragraph (b)(1)(i) of this section indefinitely, or until sufficient site functional capabilities can be maintained without the need for the mitigation strategies.

(2) Extensive damage mitigation guidelines—Strategies and guidelines to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant impacted by the event, due to explosions or fire, to include strategies and guidelines in the following areas:

(i) Firefighting;

(ii) Operations to mitigate fuel damage; and

(iii) Actions to minimize radiological release.

(c) *Equipment*. (1) The equipment relied on for the mitigation strategies and guidelines required by paragraph (b)(1) of this section must have sufficient capacity and capability to perform the functions required by paragraph (b)(1) of this section.

(2) The equipment relied on for the mitigation strategies and guidelines required by paragraph (b)(1) of this section must be reasonably protected from the effects of natural phenomena that are equivalent in magnitude to the phenomena assumed for developing the design basis of the facility.

(d) *Training requirements*. Each licensee shall provide for the training of personnel that perform activities in accordance with the capabilities required by paragraphs (b)(1) and (2) of this section.

(e) *Spent fuel pool monitoring.* In order to support effective prioritization of event mitigation and recovery actions, each licensee shall provide reliable means to remotely monitor wide-range water level for each spent fuel pool at its site until 5 years have elapsed since all of the fuel within that spent fuel pool was last used in a reactor vessel for power generation. This provision does not apply to General Electric Mark III upper containment pools.

(f) *Documentation of changes.* (1) A licensee may make changes in the implementation of the requirements in this section without NRC approval, provided that before implementing each such change, the licensee demonstrates that the provisions of this section continue to be met and maintains documentation of changes until the requirements of this section no longer apply.

(2) Changes in the implementation of requirements in this section subject to change control processes in addition to paragraph (f) of this section must be processed via their respective change control processes, unless the changes being evaluated impact only the implementation of the requirements of this section.

(g) *Implementation.* Each holder of an operating license for a nuclear power reactor under this part on September 9, 2019, and each holder of a combined license under part 52 of this chapter for which the Commission made the finding specified in 10 CFR 52.103(g) as of September 9, 2019, shall continue to comply with the provisions of paragraph (b)(2) of this section, and shall comply with all other provisions of this section no later than September 9, 2022, for licensees that received NRC Order EA–13–109 or September 9, 2021, for all other applicable licensees.

(h) *Withdrawal of orders and removal of license conditions.* (1) On September 9, 2022, Order EA–12–049, ''Order Modifying Licenses With Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events,'' and Order EA–12–051, ''Order Modifying Licenses With Regard to Reliable Spent Fuel Pool Instrumentation,'' are withdrawn for each licensee or construction permit holder that was issued those Orders.

(2) On September 9, 2019, Enrico Fermi Nuclear Plant Unit 3, License No. NPF–95, license conditions 2.D(12)(h), ''Reliable Spent Fuel Pool/ Buffer Pool Level Instrumentation,'' 2.D(12)(i), ''Emergency Planning Actions,'' and 2.D(12)(g), ''Mitigation Strategies for Beyond-Design-Basis External Events,'' except for 2.D(12)(g)1, are deemed removed from that license.

(3) On September 9, 2019, William States Lee III Nuclear Station, Unit 1, License No. NPF–101, license conditions 2.D(12)(d)11 regarding reliable spent fuel pool instrumentation, 2.D(12)(g), ''Emergency Planning Actions,'' and 2.D(12)(j), ''Mitigation Strategies for Beyond-Design-Basis External Events,'' except for 2.D(12)(j)1, and William States Lee III Nuclear Station, Unit 2, License No. NPF–102, license conditions 2.D(12)(d)11 regarding reliable spent fuel pool instrumentation, 2.D(12)(g), ''Emergency Planning Actions,'' and 2.D(12)(j), ''Mitigation Strategies for Beyond-Design-Basis External Events,'' except for 2.D(12)(j)1, are deemed removed from those licenses.

(4) On September 9, 2019, North Anna Unit 3, License No. NPF–103, license conditions 2.D(12)(g), ''Reliable Spent Fuel Pool/Buffer Pool Level Instrumentation,'' 2.D(12)(h), ''Emergency Planning Actions,'' and 2.D(12)(f), ''Mitigation Strategies for Beyond-Design-Basis External Events,'' except for 2.D(12)(f)1, are deemed removed from the license.

(5) On September 9, 2019, Turkey Point, Unit 6, License No. NPF–104, license conditions 2.D(12)(e)11 regarding reliable spent fuel pool instrumentation, 2.D(12)(g), ''Emergency Planning Actions,'' and 2.D(12)(h), ''Mitigation Strategies for Beyond-Design-Basis External Events,'' except for 2.D(12)(h)1, and Turkey Point, Unit 7, License No. NPF–105, license conditions 2.D(12)(e)11 regarding reliable spent fuel pool instrumentation, 2.D(12)(g), ''Emergency Planning Actions,'' and 2.D(12)(h), ''Mitigation Strategies for Beyond-Design-Basis External Events,'' except for

2.D(12)(h)1, are deemed removed from those licenses.

[84 FR 39718, Aug. 9, 2019]

APPENDIX A TO PART 50—GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

Table of Contents

INTRODUCTION

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

These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

The development of these General Design Criteria is not yet complete. For example, some of the definitions need further amplification. Also, some of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined. Their omission does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements. These matters include:

(1) Consideration of the need to design against single failures of passive components in fluid systems important to safety. (See Definition of Single Failure.)

(2) Consideration of redundancy and diversity requirements for fluid systems important to safety. A ''system'' could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem, and the required interconnection and independence of the subsystems have not yet been developed or defined. (See Criteria 34, 35, 38, 41, and 44.)

(3) Consideration of the type, size, and orientation of possible breaks in components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss-of-coolant accidents. (See Definition of Loss of Coolant Accidents.)

(4) Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of protection systems and reactivity control systems. (See Criteria 22, 24, 26, and 29.)

It is expected that the criteria will be augmented and changed from time to time as important new requirements for these and other features are developed.

There will be some water-cooled nuclear power plants for which the General Design Criteria are not sufficient and for which additional criteria must be identified and satisfied in the interest of public safety. In particular, it is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design. Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

DEFINITIONS AND EXPLANATIONS

Nuclear power unit. A nuclear power unit means a nuclear power reactor and associated equipment necessary for electric power generation and includes those structures, systems, and components required to provide reasonable assurance the facility can be operated without undue risk to the health and safety of the public.

Loss of coolant accidents. Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.[1]

Single failure. A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions.[2]

Anticipated operational occurrences. Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.

CRITERIA

I. Overall Requirements

Criterion 1—Quality standards and records. Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall

[1] Further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development.

[2] Single failures of passive components in electric systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.

be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Criterion 2—Design bases for protection against natural phenomena. Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Criterion 3—Fire protection. Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Criterion 4—Environmental and dynamic effects design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

Criterion 5—Sharing of structures, systems, and components. Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

II. Protection by Multiple Fission Product Barriers

Criterion 10—Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Criterion 11—Reactor inherent protection. The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Criterion 12—Suppression of reactor power oscillations. The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Criterion 13—Instrumentation and control. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Criterion 14—Reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Criterion 15—Reactor coolant system design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Criterion 16—Containment design. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Criterion 17—Electric power systems. An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

Criterion 18—Inspection and testing of electric power systems. Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Criterion 19—Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 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 meet 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.

III. Protection and Reactivity Control Systems

Criterion 20—Protection system functions. The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Criterion 21—Protection system reliability and testability. The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Criterion 22—Protection system independence. The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Criterion 23—Protection system failure modes. The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

Criterion 24—Separation of protection and control systems. The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Criterion 25—Protection system requirements for reactivity control malfunctions. The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Criterion 26—Reactivity control system redundancy and capability. Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Criterion 27—Combined reactivity control systems capability. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Criterion 28—Reactivity limits. The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Criterion 29—Protection against anticipated operational occurrences. The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

IV. Fluid Systems

Criterion 30—Quality of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical.

Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Criterion 31—Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

Criterion 32—Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Criterion 33—Reactor coolant makeup. A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Criterion 34—Residual heat removal. A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 35—Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 36—Inspection of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Criterion 37—Testing of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 38—Containment heat removal. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 39—Inspection of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

Criterion 40—Testing of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 41—Containment atmosphere cleanup. Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

Criterion 42—Inspection of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

Criterion 43—Testing of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Criterion 44—Cooling water. A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 45—Inspection of cooling water system. The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

Criterion 46—Testing of cooling water system. The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

V. Reactor Containment

Criterion 50—Containment design basis. The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by §50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Criterion 51—Fracture prevention of containment pressure boundary. The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.

Criterion 52—Capability for containment leakage rate testing. The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Criterion 53—Provisions for containment testing and inspection. The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

Criterion 54—Piping systems penetrating containment. Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Criterion 55—Reactor coolant pressure boundary penetrating containment. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Criterion 56—Primary containment isolation. Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Criterion 57—Closed system isolation valves. Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

VI. Fuel and Radioactivity Control

Criterion 60—Control of releases of radioactive materials to the environment. The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Criterion 61—Fuel storage and handling and radioactivity control. The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Criterion 62—Prevention of criticality in fuel storage and handling. Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Criterion 63—Monitoring fuel and waste storage. Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Criterion 64—Monitoring radioactivity releases. Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

[36 FR 3256, Feb. 20, 1971, as amended at 36 FR 12733, July 7, 1971; 41 FR 6258, Feb. 12, 1976; 43 FR 50163, Oct. 27, 1978; 51 FR 12505, Apr. 11, 1986; 52 FR 41294, Oct. 27, 1987; 64 FR 72002, Dec. 23, 1999; 72 FR 49505, Aug. 28, 2007]

Appendix B to Part 50—Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

Introduction. Every applicant for a construction permit is required by the provisions of § 50.34 to include in its preliminary safety analysis report a description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility. Every applicant for an operating license is required to include, in its final safety analysis report, information pertaining to the managerial and administrative controls to be used to assure safe operation. Every applicant for a combined license under part 52 of this chapter is required by the provisions of § 52.79 of this chapter to include in its final safety analysis report a description of the quality assurance applied to the design, and to be applied to the fabrication, construction, and testing of the structures, systems, and components of the facility and to the managerial and administrative controls to be used to assure safe operation. For applications submitted after September 27, 2007, every applicant for an early site permit under part 52 of this chapter is required by the provisions of § 52.17 of this chapter to include in its site safety analysis report a description of the quality assurance program applied to site activities related to the design, fabrication, construction, and testing of the structures, systems, and components of a facility or facilities that may be constructed on the site. Every applicant for a design approval or design certification under part 52 of this chapter is required by the provisions of 10 CFR 52.137 and 52.47, respectively, to include in its final safety analysis report a description of the quality assurance program applied to the design of the structures, systems, and components of the facility. Every applicant for a manufacturing license under part 52 of this chapter is required by the provisions of 10 CFR 52.157 to include in its final safety analysis report a description of the quality assurance program applied to the design, and to be applied to the manufacture of, the structures, systems, and components of the reactor. Nuclear power plants and fuel reprocessing plants include structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. This appendix establishes quality assurance requirements for the design, manufacture, construction, and operation of those structures, systems, and components. The pertinent requirements of this appendix 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.

As used in this appendix, ''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.

I. Organization

The applicant[1] shall be responsible for the establishment and execution of 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 appendix are being performed, shall have direct access to the levels of management necessary to perform this function.

II. 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 appendix. This program shall be documented by written policies, procedures, or instructions and shall be carried out throughout 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, to an extent consistent with their importance to safety. 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

[1] While the term ''applicant'' is used in these criteria, the requirements are, of course, applicable after such a person has received a license to construct and operate a nuclear power plant or a fuel reprocessing plant 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.

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.

III. DESIGN CONTROL

Measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in §50.2 and as specified in the license application, for those structures, systems, and components to which this appendix 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.

Measures shall be established for the identification and control of design interfaces and for 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.

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. 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 inservice inspection, maintenance, and repair; and delineation of acceptance criteria for inspections and tests.

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.

IV. PROCUREMENT DOCUMENT CONTROL

Measures shall be established to assure that applicable regulatory requirements, design bases, 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 appendix.

V. 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.

VI. 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.

VII. 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 nuclear power plant or fuel reprocessing plant site prior to installation or use of such material and equipment. This documentary evidence shall be retained at the nuclear power plant or fuel reprocessing plant site 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.

VIII. 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.

IX. 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.

X. 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.

XI. 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 nuclear power plant or fuel reprocessing plant 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.

XII. 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 specified periods to maintain accuracy within necessary limits.

XIII. 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.

XIV. 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 nuclear power plant or fuel reprocessing plant. 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 nuclear power plant or fuel reprocessing plant, such as by tagging valves and switches, to prevent inadvertent operation.

XV. 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.

XVI. 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.

XVII. 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.

XVIII. 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.

[35 FR 10499, June 27, 1970, as amended at 36 FR 18301, Sept. 11, 1971; 40 FR 3210, Jan. 20, 1975; 72 FR 49505, Aug. 28, 2007; 84 FR 63568, Nov. 18, 2019]

APPENDIX C TO PART 50—A GUIDE FOR THE FINANCIAL DATA AND RELATED INFORMATION REQUIRED TO ESTABLISH FINANCIAL QUALIFICATIONS FOR CONSTRUCTION PERMITS AND COMBINED LICENSES

GENERAL INFORMATION

This appendix is intended to appraise applicants for construction permits and combined licenses for production or utilization facilities of the types described in § 50.21(b) or § 50.22, or testing facilities, of the general kinds of financial data and other related information that will demonstrate the financial qualification of the applicant to carry out the activities for which the permit or license is sought. The kind and depth of information described in this guide is not intended to be a rigid and absolute requirement. In some instances, additional pertinent material may be needed. In any case, the applicant should include information other than that specified, if the information is pertinent to establishing the applicant's financial ability to carry out the activities for which the permit or license is sought.

It is important to observe also that both § 50.33(f) and this appendix distinguish between applicants which are established organizations and those which are newly-formed entities organized primarily for the purpose of engaging in the activity for which the permit is sought. Those in the former category will normally have a history of operating experience and be able to submit financial statements reflecting the financial results of past operations. With respect, however, to the applicant which is a newly formed company established primarily for the purpose of carrying out the licensed activity, with little or no prior operating history, somewhat more detailed data and supporting documentation will generally be necessary. For this reason, the appendix describes separately the scope of information to be included in applications by each of these two classes of applicants.

In determining an applicant's financial qualification, the Commission will require the minimum amount of information necessary for that purpose. No special forms are prescribed for submitting the information. In many cases, the financial information usually contained in current annual financial reports, including summary data of prior years, will be sufficient for the Commission's needs. The Commission reserves the right, however, to require additional financial information at the construction permit stage, particularly in cases in which the proposed power generating facility will be commonly owned by two or more existing companies or in which financing depends upon long-term arrangements for sharing of the power from the facility by two or more electrical generating companies.

Applicants are encouraged to consult with the Commission with respect to any questions they may have relating to the requirements of the Commission's regulations or the information set forth in this appendix.

I. Applicants Which Are Established Organizations

A. Applications for Construction Permits or Combined Licenses

1. *Estimate of construction costs.* For electric utilities, each applicant's estimate of the total cost of the proposed facility should be broken down as follows and be accompanied by a statement describing the bases from which the estimate is derived:

(a) Total nuclear production plant costs	$..........
(b) Transmission, distribution, and general plant costs	$..........
(c) Nuclear fuel inventory cost for first core [1]	$..........
Total estimated cost	$..........

[1] Section 2.790 of 10 CFR part 2 and §9.5 of 10 CFR part 9 indicate the circumstances under which information submitted by applicants may be withheld from public disclosure.

If the fuel is to be acquired by lease or other arrangement than purchase, the application should so state. The items to be included in these categories should be the same as those defined in the applicable electric plant and nuclear fuel inventory accounts prescribed by the Federal Energy Regulatory Commission or an explanation given as to any departure therefrom.

Since the composition of construction cost estimates for production and utilization facilities other than nuclear power reactors will vary according to the type of facility, no particular format is suggested for submitting such estimates. The estimate should, however, be itemized by categories of cost in sufficient detail to permit an evaluation of its reasonableness.

2. *Source of construction funds.* The application should include a brief statement of the applicant's general financial plan for financing the cost of the facility, identifying the source or sources upon which the applicant relies for the necessary construction funds, e.g., internal sources such as undistributed earnings and depreciation accruals, or external sources such as borrowings.

3. *Applicant's financial statements.* The application should also include the applicant's latest published annual financial report, together with any current interim financial statements that are pertinent. If an annual financial report is not published, the balance sheet and operating statement covering the latest complete accounting year together with all pertinent notes thereto and certification by a public accountant should be furnished.

II. Applicants Which Are Newly Formed Entities

A. Applications for Construction Permits or Combined Licenses

1. *Estimate of construction costs.* The information that will normally be required of applicants which are newly formed entities will not differ in scope from that required of established organizations. Accordingly, applicants should submit estimates as described above for established organizations.

2. *Source of construction funds.* The application should specifically identify the source or sources upon which the applicant relies for the funds necessary to pay the cost of constructing the facility, and the amount to be obtained from each. With respect to each source, the application should describe in detail the applicant's legal and financial relationships with its stockholders, corporate affiliates, or others (such as financial institutions) upon which the applicant is relying for financial assistance. If the sources of funds relied upon include parent companies or other corporate affiliates, information to support the financial capability of each such company or affiliate to meet its commitments to the applicant should be set forth in the application. This information should be of the same kind and scope as would be required if the parent companies or affiliates were in fact the applicant. Ordinarily, it will be necessary that copies of agreements or contracts among the companies be submitted.

As noted earlier in this appendix, an applicant which is a newly formed entity will normally not be in a position to submit the usual types of balance sheets and income statements reflecting the results of prior operations. The applicant should, however, include in its application a statement of its assets, liabilities, and capital structure as of the date of the application.

III. Annual Financial Statement

Each holder of a construction permit for a production or utilization facility of a type described in §50.21(b) or §50.22 or a testing facility, and each holder of a combined license issued under part 52 of this chapter, is required by §50.71(b) to file its annual financial report with the Commission at the time of issuance. This requirement does not apply to licensees or holders of construction permits for medical and research reactors.

IV. ADDITIONAL INFORMATION

The Commission may, from time to time, request the applicant, whether an established organization or newly formed entity, to submit additional or more detailed information respecting its financial arrangements and status of funds if such information is deemed necessary to enable the Commission to determine an applicant's financial qualifications for the license.

[49 FR 35753, Sept. 12, 1984, as amended at 50 FR 18853, May 3, 1985; 72 FR 49506, Aug. 28, 2007; 81 FR 86909, Dec. 2, 2016]

APPENDIX D TO PART 50 [RESERVED]

APPENDIX E TO PART 50—EMERGENCY PLANNING AND PREPAREDNESS FOR PRODUCTION AND UTILIZATION FACILITIES

Table of Contents

I. INTRODUCTION

1. Each applicant for a construction permit is required by §50.34(a) to include in the preliminary safety analysis report a discussion of preliminary plans for coping with emergencies. Each applicant for an operating license is required by §50.34(b) to include in the final safety analysis report plans for coping with emergencies. Each applicant for a combined license under subpart C of part 52 of this chapter is required by §52.79 of this chapter to include in the application plans for coping with emergencies. Each applicant for an early site permit under subpart A of part 52 of this chapter may submit plans for coping with emergencies under §52.17 of this chapter.

2. This appendix establishes minimum requirements for emergency plans for use in attaining an acceptable state of emergency preparedness. These plans shall be described generally in the preliminary safety analysis report for a construction permit and submitted as part of the final safety analysis report for an operating license. These plans, or major features thereof, may be submitted as part of the site safety analysis report for an early site permit.

3. The potential radiological hazards to the public associated with the operation of research and test reactors and fuel facilities licensed under 10 CFR parts 50 and 70 involve considerations different than those associated with nuclear power reactors. Consequently, the size of Emergency Planning Zones[1] (EPZs) for facilities other than power reactors and the degree to which compliance with the requirements of this section and sections II, III, IV, and V of this appendix as necessary will be determined on a case-by-case basis.[2]

4. Notwithstanding the above paragraphs, in the case of an operating license authorizing only fuel loading and/or low power operations up to 5 percent of rated power, no NRC or FEMA review, findings, or determinations concerning the state of offsite emergency preparedness or the adequacy of and the capability to implement State and local offsite emergency plans, as defined in this Appendix, are required prior to the issuance of such a license.

5. Each applicant for a combined license or early site permit under part 52 of this chapter whose application is docketed before December 23, 2011 may defer compliance with any change to emergency preparedness regulations under the final rule issued November 23, 2011. If that applicant chooses to defer compliance, it shall subsequently request to amend the combined license or early site permit to comply with those changes no later than December 31, 2013. An applicant that does not receive a combined license or early site permit before December 31, 2013, shall revise its combined license or early site permit application to comply with those changes no later than December 31, 2013. Notwithstanding any Commission finding under 10 CFR 52.103(g) regarding the combined license holder's facility, the combined license holder may not operate the facility until the NRC has approved the license

[1] EPZs for power reactors 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. The size of the EPZs for a nuclear power plant shall be determined in relation to 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 nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. Generally, the plume exposure pathway EPZ for nuclear power plants with an authorized power level greater than 250 MW thermal shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius.

[2] Regulatory Guide 2.6 will be used as guidance for the acceptability of research and test reactor emergency response plans.

amendment demonstrating compliance with the final rule.

6. The Tennessee Valley Authority Watts Bar Nuclear Plant, Unit 2, holding a construction permit under the provisions of part 50 of this chapter, shall meet the requirements of the final rule issued November 23, 2011 as applicable to operating nuclear power reactor licensees.

II. The Preliminary Safety Analysis Report

The Preliminary Safety Analysis Report shall contain 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 in the case of nuclear power reactors as well as the means by which the standards of §50.47(b) will be met.

As a minimum, the following items shall be described:

A. 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.

B. 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.

C. Protective measures to be taken within the site boundary and within each EPZ 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.

D. 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.

E. Provisions to be made for emergency treatment at offsite facilities of individuals injured as a result of licensed activities.

F. 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.

G. 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 shall 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.

H. 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 onsite technical support center and the emergency operations facility in assessing information, recommending protective action, and disseminating information to the public.

III. The Final Safety Analysis Report; Site Safety Analysis Report

The final safety analysis report or the site safety analysis report for an early site permit that includes complete and integrated emergency plans under §52.17(b)(2)(ii) of this chapter shall contain the plans for coping with emergencies. The plans shall be an expression of the overall concept of operation; they shall describe the essential elements of advance planning that have been considered and the provisions that have been made to cope with emergency situations. The plans shall incorporate information about the emergency response roles of supporting organizations and offsite agencies. That information shall be sufficient to provide assurance of coordination among the supporting groups and with the licensee. The site safety analysis report for an early site permit which proposes major features must address the relevant provisions of 10 CFR 50.47 and 10 CFR part 50, appendix E, within the scope of emergency preparedness matters addressed in the major features.

The plans submitted must include a description of the elements set out in Section IV for the emergency planning zones (EPZs) to an extent sufficient to demonstrate that the plans provide reasonable assurance that adequate protective measures can and will be taken in the event of an emergency.

IV. Content of Emergency Plans

1. The applicant's emergency plans shall contain, but not necessarily be limited to, information needed to demonstrate compliance with the elements set forth below, *i.e.*, organization for coping with radiological emergencies, assessment actions, activation of emergency organization, notification procedures, emergency facilities and equipment, training, maintaining emergency preparedness, recovery, and onsite protective actions

during hostile action. In addition, the emergency response plans submitted by an applicant for a nuclear power reactor operating license under this part, or for an early site permit (as applicable) or combined license under 10 CFR part 52, shall contain information needed to demonstrate compliance with the standards described in §50.47(b), and they will be evaluated against those standards.

2. This nuclear power reactor license applicant shall also provide an analysis of the time required to evacuate various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations, using the most recent U.S. Census Bureau data as of the date the applicant submits its application to the NRC.

3. Nuclear power reactor licensees shall use NRC approved evacuation time estimates (ETEs) and updates to the ETEs in the formulation of protective action recommendations and shall provide the ETEs and ETE updates to State and local governmental authorities for use in developing offsite protective action strategies.

4. Within 365 days of the later of the date of the availability of the most recent decennial census data from the U.S. Census Bureau or December 23, 2011, nuclear power reactor licensees shall develop an ETE analysis using this decennial data and submit it under §50.4 to the NRC. These licensees shall submit this ETE analysis to the NRC at least 180 days before using it to form protective action recommendations and providing it to State and local governmental authorities for use in developing offsite protective action strategies.

5. During the years between decennial censuses, nuclear power reactor licensees shall estimate EPZ permanent resident population changes once a year, but no later than 365 days from the date of the previous estimate, using the most recent U.S. Census Bureau annual resident population estimate and State/local government population data, if available. These licensees shall maintain these estimates so that they are available for NRC inspection during the period between decennial censuses and shall submit these estimates to the NRC with any updated ETE analysis.

6. If at any time during the decennial period, the EPZ permanent resident population increases such that it causes the longest ETE value for the 2-mile zone or 5-mile zone, including all affected Emergency Response Planning Areas, or for the entire 10-mile EPZ to increase by 25 percent or 30 minutes, whichever is less, from the nuclear power reactor licensee's currently NRC approved or updated ETE, the licensee shall update the ETE analysis to reflect the impact of that population increase. The licensee shall submit the updated ETE analysis to the NRC under §50.4 no later than 365 days after the licensee's determination that the criteria for updating the ETE have been met and at least 180 days before using it to form protective action recommendations and providing it to State and local governmental authorities for use in developing offsite protective action strategies.

7. After an applicant for a combined license under part 52 of this chapter receives its license, the licensee shall conduct at least one review of any changes in the population of its EPZ at least 365 days prior to its scheduled fuel load. The licensee shall estimate EPZ permanent resident population changes using the most recent U.S. Census Bureau annual resident population estimate and State/local government population data, if available. If the EPZ permanent resident population increases such that it causes the longest ETE value for the 2-mile zone or 5-mile zone, including all affected Emergency Response Planning Areas, or for the entire 10-mile EPZ, to increase by 25 percent or 30 minutes, whichever is less, from the licensee's currently approved ETE, the licensee shall update the ETE analysis to reflect the impact of that population increase. The licensee shall submit the updated ETE analysis to the NRC for review under §50.4 of this chapter no later than 365 days before the licensee's scheduled fuel load.

A. Organization

The organization for coping with radiological emergencies shall be described, including definition of authorities, responsibilities, and duties of individuals assigned to the licensee's emergency organization and the means for notification of such individuals in the event of an emergency. Specifically, the following shall be included:

1. A description of the normal plant operating organization.

2. A description of the onsite emergency response organization (ERO) with a detailed discussion of:

a. Authorities, responsibilities, and duties of the individual(s) who will take charge during an emergency;

b. Plant staff emergency assignments;

c. Authorities, responsibilities, and duties of an onsite emergency coordinator who shall be in charge of the exchange of information with offsite authorities responsible for coordinating and implementing offsite emergency measures.

3. A description, by position and function to be performed, of the licensee's headquarters personnel who will be sent to the plant site to augment the onsite emergency organization.

4. Identification, by position and function to be performed, of persons within the licensee organization who will be responsible for making offsite dose projections, and a description of how these projections will be made and the results transmitted to State

and local authorities, NRC, and other appropriate governmental entities.

5. Identification, by position and function to be performed, of other employees of the licensee with special qualifications for coping with emergency conditions that may arise. Other persons with special qualifications, such as consultants, who are not employees of the licensee and who may be called upon for assistance for emergencies shall also be identified. The special qualifications of these persons shall be described.

6. A description of the local offsite services to be provided in support of the licensee's emergency organization.

7. By June 23, 2014, identification of, and a description of the assistance expected from, appropriate State, local, and Federal agencies with responsibilities for coping with emergencies, including hostile action at the site. For purposes of this appendix, ''hostile action'' is defined as an act directed toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force.

8. Identification of the State and/or local officials responsible for planning for, ordering, and controlling appropriate protective actions, including evacuations when necessary.

9. By December 24, 2012, for nuclear power reactor licensees, a detailed analysis demonstrating that on-shift personnel assigned emergency plan implementation functions are not assigned responsibilities that would prevent the timely performance of their assigned functions as specified in the emergency plan.

B. Assessment Actions

1. The means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials shall be described, including emergency action levels that are to be used as criteria for determining the need for notification and participation of local and State agencies, the Commission, and other Federal agencies, and the emergency action levels that are to be used for determining when and what type of protective measures should be considered within and outside the site boundary to protect health and safety. The emergency action levels shall be based on in-plant conditions and instrumentation in addition to onsite and offsite monitoring. By June 20, 2012, for nuclear power reactor licensees, these action levels must include hostile action that may adversely affect the nuclear power plant. The initial emergency action levels shall be discussed and agreed on by the applicant or licensee and state and local governmental authorities, and approved by the NRC. Thereafter, emergency action levels shall be reviewed with the State and local governmental authorities on an annual basis.

2. A licensee desiring to change its entire emergency action level scheme shall submit an application for an amendment to its license and receive NRC approval before implementing the change. Licensees shall follow the change process in §50.54(q) for all other emergency action level changes.

C. Activation of Emergency Organization

1. The entire spectrum of emergency conditions that involve the alerting or activating of progressively larger segments of the total emergency organization shall be described. The communication steps to be taken to alert or activate emergency personnel under each class of emergency shall be described. Emergency action levels (based not only on onsite and offsite radiation monitoring information but also on readings from a number of sensors that indicate a potential emergency, such as the pressure in containment and the response of the Emergency Core Cooling System) for notification of offsite agencies shall be described. The existence, but not the details, of a message authentication scheme shall be noted for such agencies. The emergency classes defined shall include: (1) Notification of unusual events, (2) alert, (3) site area emergency, and (4) general emergency. These classes are further discussed in NUREG–0654/FEMA–REP–1.

2. By June 20, 2012, nuclear power reactor licensees shall establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and shall promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. Licensees shall not construe these criteria as a grace period to attempt to restore plant conditions to avoid declaring an emergency action due to an emergency action level that has been exceeded. Licensees shall not construe these criteria as preventing implementation of response actions deemed by the licensee to be necessary to protect public health and safety provided that any delay in declaration does not deny the State and local authorities the opportunity to implement measures necessary to protect the public health and safety.

D. Notification Procedures

1. Administrative and physical means for notifying local, State, and Federal officials and agencies and agreements reached with these officials and agencies for the prompt notification of the public and for public evacuation or other protective measures,

should they become necessary, shall be described. This description shall include identification of the appropriate officials, by title and agency, of the State and local government agencies within the EPZs.

2. Provisions shall be described for yearly dissemination to the public within the plume exposure pathway EPZ of basic emergency planning information, such as the methods and times required for public notification and the protective actions planned if an accident occurs, general information as to the nature and effects of radiation, and a listing of local broadcast stations that will be used for dissemination of information during an emergency. Signs or other measures shall also be used to disseminate to any transient population within the plume exposure pathway EPZ appropriate information that would be helpful if an accident occurs.

3. A licensee shall have the capability to notify responsible State and local governmental agencies within 15 minutes after declaring an emergency. The licensee shall demonstrate that the appropriate governmental authorities have the capability to make a public alerting and notification decision promptly on being informed by the licensee of an emergency condition. Prior to initial operation greater than 5 percent of rated thermal power of the first reactor at a site, each nuclear power reactor licensee shall demonstrate that administrative and physical means have been established for alerting and providing prompt instructions to the public within the plume exposure pathway EPZ. The design objective of the prompt public alert and notification system shall be to have the capability to essentially complete the initial alerting and initiate notification of the public within the plume exposure pathway EPZ within about 15 minutes. The use of this alerting and notification capability will range from immediate alerting and notification of the public (within 15 minutes of the time that State and local officials are notified that a situation exists requiring urgent action) to the more likely events where there is substantial time available for the appropriate governmental authorities to make a judgment whether or not to activate the public alert and notification system. The alerting and notification capability shall additionally include administrative and physical means for a backup method of public alerting and notification capable of being used in the event the primary method of alerting and notification is unavailable during an emergency to alert or notify all or portions of the plume exposure pathway EPZ population. The backup method shall have the capability to alert and notify the public within the plume exposure pathway EPZ, but does not need to meet the 15-minute design objective for the primary prompt public alert and notification system. When there is a decision to activate the alert and notification system, the appropriate governmental authorities will determine whether to activate the entire alert and notification system simultaneously or in a graduated or staged manner. The responsibility for activating such a public alert and notification system shall remain with the appropriate governmental authorities.

E. Emergency Facilities and Equipment

Adequate provisions shall be made and described for emergency facilities and equipment, including:

1. Equipment at the site for personnel monitoring;

2. Equipment for determining the magnitude of and for continuously assessing the impact of the release of radioactive materials to the environment;

3. Facilities and supplies at the site for decontamination of onsite individuals;

4. Facilities and medical supplies at the site for appropriate emergency first aid treatment;

5. Arrangements for medical service providers qualified to handle radiological emergencies onsite;

6. Arrangements for transportation of contaminated injured individuals from the site to specifically identified treatment facilities outside the site boundary;

7. Arrangements for treatment of individuals injured in support of licensed activities on the site at treatment facilities outside the site boundary;

8.a. (i) A licensee onsite technical support center and an emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency;

(ii) For nuclear power reactor licensees, a licensee onsite operational support center;

b. For a nuclear power reactor licensee's emergency operations facility required by paragraph 8.a of this section, either a facility located between 10 miles and 25 miles of the nuclear power reactor site(s), or a primary facility located less than 10 miles from the nuclear power reactor site(s) and a backup facility located between 10 miles and 25 miles of the nuclear power reactor site(s). An emergency operations facility may serve more than one nuclear power reactor site. A licensee desiring to locate an emergency operations facility more than 25 miles from a nuclear power reactor site shall request prior Commission approval by submitting an application for an amendment to its license. For an emergency operations facility located more than 25 miles from a nuclear power reactor site, provisions must be made for locating NRC and offsite responders closer to the nuclear power reactor site so that NRC and offsite responders can interact face-to-face with emergency response personnel entering and leaving the nuclear power reactor site. Provisions for locating NRC and offsite responders closer to a nuclear power reactor site that is more than 25 miles from the emergency operations facility must include the following:

(1) Space for members of an NRC site team and Federal, State, and local responders;

(2) Additional space for conducting briefings with emergency response personnel;

(3) Communication with other licensee and offsite emergency response facilities;

(4) Access to plant data and radiological information; and

(5) Access to copying equipment and office supplies;

c. By June 20, 2012, for a nuclear power reactor licensee's emergency operations facility required by paragraph 8.a of this section, a facility having the following capabilities:

(1) The capability for obtaining and displaying plant data and radiological information for each reactor at a nuclear power reactor site and for each nuclear power reactor site that the facility serves;

(2) The capability to analyze plant technical information and provide technical briefings on event conditions and prognosis to licensee and offsite response organizations for each reactor at a nuclear power reactor site and for each nuclear power reactor site that the facility serves; and

(3) The capability to support response to events occurring simultaneously at more than one nuclear power reactor site if the emergency operations facility serves more than one site; and

d. For nuclear power reactor licensees, an alternative facility (or facilities) that would be accessible even if the site is under threat of or experiencing hostile action, to function as a staging area for augmentation of emergency response staff and collectively having the following characteristics: the capability for communication with the emergency operations facility, control room, and plant security; the capability to perform offsite notifications; and the capability for engineering assessment activities, including damage control team planning and preparation, for use when onsite emergency facilities cannot be safely accessed during hostile action. The requirements in this paragraph 8.d must be implemented no later than December 23, 2014, with the exception of the capability for staging emergency response organization personnel at the alternative facility (or facilities) and the capability for communications with the emergency operations facility, control room, and plant security, which must be implemented no later than June 20, 2012.

e. A licensee shall not be subject to the requirements of paragraph 8.b of this section for an existing emergency operations facility approved as of December 23, 2011;

9. At least one onsite and one offsite communications system; each system shall have a backup power source. All communication plans shall have arrangements for emergencies, including titles and alternates for those in charge at both ends of the communication links and the primary and backup means of communication. Where consistent with the function of the governmental agency, these arrangements will include:

a. Provision for communications with contiguous State/local governments within the plume exposure pathway EPZ. Such communications shall be tested monthly.

b. Provision for communications with Federal emergency response organizations. Such communications systems shall be tested annually.

c. Provision for communications among the nuclear power reactor control room, the onsite technical support center, and the emergency operations facility; and among the nuclear facility, the principal State and local emergency operations centers, and the field assessment teams. Such communications systems shall be tested annually.

d. Provisions for communications by the licensee with NRC Headquarters and the appropriate NRC Regional Office Operations Center from the nuclear power reactor control room, the onsite technical support center, and the emergency operations facility. Such communications shall be tested monthly.

F. Training

1. The program to provide for: (a) The training of employees and exercising, by periodic drills, of emergency plans to ensure that employees of the licensee are familiar with their specific emergency response duties, and (b) The participation in the training and drills by other persons whose assistance may be needed in the event of a radiological emergency shall be described. This shall include a description of specialized initial training and periodic retraining programs to be provided to each of the following categories of emergency personnel:

i. Directors and/or coordinators of the plant emergency organization;

ii. Personnel responsible for accident assessment, including control room shift personnel;

iii. Radiological monitoring teams;

iv. Fire control teams (fire brigades);

v. Repair and damage control teams;

vi. First aid and rescue teams;

vii. Medical support personnel;

viii. Licensee's headquarters support personnel;

ix. Security personnel.

In addition, a radiological orientation training program shall be made available to local services personnel; e.g., local emergency services/Civil Defense, local law enforcement personnel, local news media persons.

2. The plan shall describe provisions for the conduct of emergency preparedness exercises as follows: Exercises shall test the adequacy

of timing and content of implementing procedures and methods, test emergency equipment and communications networks, test the public alert and notification system, and ensure that emergency organization personnel are familiar with their duties.[3]

a. A full participation[4] exercise which tests as much of the licensee, State, and local emergency plans as is reasonably achievable without mandatory public participation shall be conducted for each site at which a power reactor is located. Nuclear power reactor licensees shall submit exercise scenarios under §50.4 at least 60 days before use in a full participation exercise required by this paragraph 2.a.

(i) For an operating license issued under this part, this exercise must be conducted within 2 years before the issuance of the first operating license for full power (one authorizing operation above 5 percent of rated thermal power) of the first reactor and shall include participation by each State and local government within the plume exposure pathway EPZ and each state within the ingestion exposure pathway EPZ. If the full participation exercise is conducted more than 1 year prior to issuance of an operating licensee for full power, an exercise which tests the licensee's onsite emergency plans must be conducted within one year before issuance of an operating license for full power. This exercise need not have State or local government participation.

(ii) For a combined license issued under part 52 of this chapter, this exercise must be conducted within two years of the scheduled date for initial loading of fuel. If the first full participation exercise is conducted more than one year before the scheduled date for initial loading of fuel, an exercise which tests the licensee's onsite emergency plans must be conducted within one year before the scheduled date for initial loading of fuel. This exercise need not have State or local government participation. If FEMA identifies one or more deficiencies in the state of offsite emergency preparedness as the result of the first full participation exercise, or if the Commission 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, the provisions of §50.54(gg) apply.

(iii) For a combined license issued under part 52 of this chapter, if the applicant currently has an operating reactor at the site, an exercise, either full or partial participation,[5] shall be conducted for each subsequent reactor constructed on the site. This exercise may be incorporated in the exercise requirements of Sections IV.F.2.b. and c. in this appendix. If FEMA identifies one or more deficiencies in the state of offsite emergency preparedness as the result of this exercise for the new reactor, or if the Commission 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, the provisions of §50.54(gg) apply.

b. Each licensee at each site shall conduct a subsequent exercise of its onsite emergency plan every 2 years. Nuclear power reactor licensees shall submit exercise scenarios under §50.4 at least 60 days before use in an exercise required by this paragraph 2.b. The exercise may be included in the full participation biennial exercise required by paragraph 2.c. of this section. In addition, the licensee shall take actions necessary to ensure that adequate emergency response capabilities are maintained during the interval between biennial exercises by conducting drills, including at least one drill involving a combination of some of the principal functional areas of the licensee's onsite emergency response capabilities. The principal functional areas of emergency response include activities such as management and coordination of emergency response, accident assessment, event classification, notification of offsite authorities, assessment of the onsite and offsite impact of radiological releases, protective action recommendation development, protective action decision making, plant system repair and mitigative action implementation. During these drills, activation of all of the licensee's emergency response facilities (Technical Support Center (TSC), Operations Support Center (OSC), and the Emergency Operations Facility (EOF)) would not be necessary, licensees would have

[3] Use of site specific simulators or computers is acceptable for any exercise.

[4] Full participation when used in conjunction with emergency preparedness exercises for a particular site means appropriate offsite local and State authorities and licensee personnel physically and actively take part in testing their integrated capability to adequately assess and respond to an accident at a commercial nuclear power plant. Full participation includes testing major observable portions of the onsite and offsite emergency plans and mobilization of State, local and licensee personnel and other resources in sufficient numbers to verify the capability to respond to the accident scenario.

[5] Partial participation when used in conjunction with emergency preparedness exercises for a particular site means appropriate offsite authorities shall actively take part in the exercise sufficient to test direction and control functions; *i.e.*, (a) protective action decision making related to emergency action levels, and (b) communication capabilities among affected State and local authorities and the licensee.

the opportunity to consider accident management strategies, supervised instruction would be permitted, operating staff in all participating facilities would have the opportunity to resolve problems (success paths) rather than have controllers intervene, and the drills may focus on the onsite exercise training objectives.

c. Offsite plans for each site shall be exercised biennially with full participation by each offsite authority having a role under the radiological response plan. Where the offsite authority has a role under a radiological response plan for more than one site, it shall fully participate in one exercise every two years and shall, at least, partially participate in other offsite plan exercises in this period. If two different licensees each have licensed facilities located either on the same site or on adjacent, contiguous sites, and share most of the elements defining co-located licensees,[6] then each licensee shall:

(1) Conduct an exercise biennially of its onsite emergency plan;

(2) Participate quadrennially in an offsite biennial full or partial participation exercise;

(3) Conduct emergency preparedness activities and interactions in the years between its participation in the offsite full or partial participation exercise with offsite authorities, to test and maintain interface among the affected State and local authorities and the licensee. Co-located licensees shall also participate in emergency preparedness activities and interaction with offsite authorities for the period between exercises;

(4) Conduct a hostile action exercise of its onsite emergency plan in each exercise cycle; and

(5) Participate in an offsite biennial full or partial participation hostile action exercise in alternating exercise cycles.

d. Each State with responsibility for nuclear power reactor emergency preparedness should fully participate in the ingestion pathway portion of exercises at least once every exercise cycle. In States with more than one nuclear power reactor plume exposure pathway EPZ, the State should rotate this participation from site to site. Each State with responsibility for nuclear power reactor emergency preparedness should fully participate in a hostile action exercise at least once every cycle and should fully participate in one hostile action exercise by December 31, 2015. States with more than one nuclear power reactor plume exposure pathway EPZ should rotate this participation from site to site.

e. Licensees shall enable any State or local government located within the plume exposure pathway EPZ to participate in the licensee's drills when requested by such State or local government.

f. Remedial exercises will be required if the emergency plan is not satisfactorily tested during the biennial exercise, such that NRC, in consultation with FEMA, cannot (1) find reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency or (2) determine that the Emergency Response Organization (ERO) has maintained key skills specific to emergency response. The extent of State and local participation in remedial exercises must be sufficient to show that appropriate corrective measures have been taken regarding the elements of the plan not properly tested in the previous exercises.

g. All exercises, drills, and training that provide performance opportunities to develop, maintain, or demonstrate key skills must provide for formal critiques in order to identify weak or deficient areas that need correction. Any weaknesses or deficiencies that are identified in a critique of exercises, drills, or training must be corrected.

h. The participation of State and local governments in an emergency exercise is not required to the extent that the applicant has identified those governments as refusing to participate further in emergency planning activities, pursuant to §50.47(c)(1). In such cases, an exercise shall be held with the applicant or licensee and such governmental entities as elect to participate in the emergency planning process.

i. Licensees shall use drill and exercise scenarios that provide reasonable assurance that anticipatory responses will not result from preconditioning of participants. Such scenarios for nuclear power reactor licensees must include a wide spectrum of radiological releases and events, including hostile action. Exercise and drill scenarios as appropriate must emphasize coordination among onsite and offsite response organizations.

j. (i) The exercises conducted under paragraph 2 of this section by nuclear power reactor licensees must provide the opportunity for the ERO to demonstrate proficiency in the key skills necessary to implement the principal functional areas of emergency response identified in paragraph 2.b of this section.

(ii) Each exercise must provide the opportunity for the ERO to demonstrate key skills specific to emergency response duties in the

[6] Co-located licensees are two different licensees whose licensed facilities are located either on the same site or on adjacent, contiguous sites, and that share most of the following emergency planning and siting elements:

a. Plume exposure and ingestion emergency planning zones;

b. Offsite governmental authorities;

c. Offsite emergency response organizations;

d. Public notification system; and/or

e. Emergency facilities.

control room, TSC, OSC, EOF, and joint information center.

(iii) In each 8-calendar-year exercise cycle, nuclear power reactor licensees shall vary the content of scenarios during exercises conducted under paragraph 2 of this section to provide the opportunity for the ERO to demonstrate proficiency in the key skills necessary to respond to the following scenario elements:

(1) Hostile action directed at the plant site;

(2) No radiological release or an unplanned minimal radiological release that does not require public protective actions;

(3) An initial classification of, or rapid escalation to, a Site Area Emergency or General Emergency;

(4) Implementation of strategies, procedures, and guidance under § 50.155(b)(2); and

(5) Integration of offsite resources with onsite response.

(iv) The licensee shall maintain a record of exercises conducted during each 8-year exercise cycle that documents the content of scenarios used to comply with the requirements of section IV.F.2.j of this appendix.

(v) Each licensee shall conduct a hostile action exercise for each of its sites no later than December 31, 2015.

(vi) The first 8-year exercise cycle for a site will begin in the calendar year in which the first hostile action exercise is conducted. For a site licensed under 10 CFR part 52, the first 8-year exercise cycle begins in the calendar year of the initial exercise required by section IV.F.2.a of this appendix.

G. Maintaining Emergency Preparedness

Provisions to be employed to ensure that the emergency plan, its implementing procedures, and emergency equipment and supplies are maintained up to date shall be described.

H. Recovery

Criteria to be used to determine when, following an accident, reentry of the facility would be appropriate or when operation could be resumed shall be described.

I. Onsite Protective Actions During Hostile Action

By June 20, 2012, for nuclear power reactor licensees, a range of protective actions to protect onsite personnel during hostile action must be developed to ensure the continued ability of the licensee to safely shut down the reactor and perform the functions of the licensee's emergency plan.

V. IMPLEMENTING PROCEDURES

No less than 180 days before the scheduled issuance of an operating license for a nuclear power reactor or a license to possess nuclear material, or the scheduled date for initial loading of fuel for a combined license under part 52 of this chapter, the applicant's or licensee's detailed implementing procedures for its emergency plan shall be submitted to the Commission as specified in § 50.4.

VI. EMERGENCY RESPONSE DATA SYSTEM

1. The Emergency Response Data System (ERDS) is a direct near real-time electronic data link between the licensee's onsite computer system and the NRC Operations Center that provides for the automated transmission of a limited data set of selected parameters. The ERDS supplements the existing voice transmission over the Emergency Notification System (ENS) by providing the NRC Operations Center with timely and accurate updates of a limited set of parameters from the licensee's installed onsite computer system in the event of an emergency. When selected plant data are not available on the licensee's onsite computer system, retrofitting of data points is not required. The licensee shall test the ERDS periodically to verify system availability and operability. The frequency of ERDS testing will be quarterly unless otherwise set by NRC based on demonstrated system performance.

2. Except for Big Rock Point and all nuclear power facilities that are shut down permanently or indefinitely, onsite hardware shall be provided at each unit by the licensee to interface with the NRC receiving system. Software, which will be made available by the NRC, will assemble the data to be transmitted and transmit data from each unit via an output port on the appropriate data system. The hardware and software must have the following characteristics:

a. Data points, if resident in the in-plant computer systems, must be transmitted for four selected types of plant conditions: Reactor core and coolant system conditions; reactor containment conditions; radioactivity release rates; and plant meteorological tower data. A separate data feed is required for each reactor unit. While it is recognized that ERDS is not a safety system, it is conceivable that a licensee's ERDS interface could communicate with a safety system. In this case, appropriate isolation devices would be required at these interfaces.[7] The data points, identified in the following parameters will be transmitted:

(i) For pressurized water reactors (PWRs), the selected plant parameters are: (1) Primary coolant system: pressure, temperatures (hot leg, cold leg, and core exit thermocouples), subcooling margin, pressurizer level, reactor coolant charging/makeup flow, reactor vessel level, reactor coolant flow, and reactor power; (2) Secondary coolant system: Steam generator levels and pressures, main feedwater flows, and auxiliary and emergency feedwater flows; (3) Safety

[7] See 10 CFR 50.55a(h) Protection Systems.

injection: High- and low-pressure safety injection flows, safety injection flows (Westinghouse), and borated water storage tank level; (4) Containment: pressure, temperatures, hydrogen concentration, and sump levels; (5) Radiation monitoring system: Reactor coolant radioactivity, containment radiation level, condenser air removal radiation level, effluent radiation monitors, and process radiation monitor levels; and (6) Meteorological data: wind speed, wind direction, and atmospheric stability.

(ii) For boiling water reactors (BWRs), the selected parameters are: (1) Reactor coolant system: Reactor pressure, reactor vessel level, feedwater flow, and reactor power; (2) Safety injection: Reactor core isolation cooling flow, high-pressure coolant injection/high-pressure core spray flow, core spray flow, low-pressure coolant injection flow, and condensate storage tank level; (3) Containment: drywell pressure, drywell temperatures, drywell sump levels, hydrogen and oxygen concentrations, suppression pool temperature, and suppression pool level; (4) Radiation monitoring system: Reactor coolant radioactivity level, primary containment radiation level, condenser off-gas radiation level, effluent radiation monitor, and process radiation levels; and (5) Meteorological data: Wind speed, wind direction, and atmospheric stability.

b. The system must be capable of transmitting all available ERDS parameters at time intervals of not less than 15 seconds or more than 60 seconds. Exceptions to this requirement will be considered on a case by case basis.

c. All link control and data transmission must be established in a format compatible with the NRC receiving system[8] as configured at the time of licensee implementation.

3. Maintaining Emergency Response Data System:

a. Any hardware and software changes that affect the transmitted data points identified in the ERDS Data Point Library[9] (site specific data base residing on the ERDS computer) must be submitted to the NRC within 30 days after the changes are completed.

b. Hardware and software changes, with the exception of data point modifications, that could affect the transmission format and computer communication protocol to the ERDS must be provided to the NRC as soon as practicable and at least 30 days prior to the modification.

[8] Guidance is provided in NUREG–1394, Revision 1.

[9] See NUREG–1394, Revision 1, appendix C, Data Point Library.

c. In the event of a failure of NRC-supplied equipment, a replacement will be furnished by the NRC for licensee installation.

[45 FR 55410, Aug. 19, 1980; 46 FR 28839, May 29, 1981, as amended at 46 FR 63032, Dec. 30, 1981; 47 FR 30236, July 13, 1982; 47 FR 57671, Dec. 28, 1982; 49 FR 27736, July 6, 1984; 51 FR 40310, Nov. 6, 1986; 52 FR 16829, May 6, 1987; 52 FR 42086, Nov. 3, 1987; 56 FR 40185, Aug. 13, 1991; 59 FR 14090, Mar. 25, 1994; 61 FR 30132, June 14, 1996; 72 FR 49506, Aug. 28, 2007; 73 FR 42674, July 23, 2008; 76 FR 72596, Nov. 23, 2011; 78 FR 34248, June 7, 2013; 80 FR 74980, Dec. 1, 2015; 84 FR 39719, Aug. 9, 2019; 86 FR 43402, Aug. 9, 2021; 86 FR 67842, Nov. 30, 2021]

EDITORIAL NOTE: At 72 FR 49506, Aug. 28, 2007, Appendix E to part 50 was amended by redesignating footnotes 6, 7, 8, 9, 10, 11 as 7, 8, 9, 10, 11, 12; however, the amendment could not be incorporated due to inaccurate amendatory instruction.

APPENDIX F TO PART 50—POLICY RELATING TO THE SITING OF FUEL REPROCESSING PLANTS AND RELATED WASTE MANAGEMENT FACILITIES

1. Public health and safety considerations relating to licensed fuel reprocessing plants do not require that such facilities be located on land owned and controlled by the Federal Government. Such plants, including the facilities for the temporary storage of high-level radioactive wastes, may be located on privately owned property.

2. A fuel reprocessing plant's inventory of high-level liquid radioactive wastes will be limited to that produced in the prior 5 years. (For the purpose of this statement of policy, ''high-level liquid radioactive wastes'' means those aqueous wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuels.) High-level liquid radioactive wastes shall be converted to a dry solid as required to comply with this inventory limitation, and placed in a sealed container prior to transfer to a Federal repository in a shipping cask meeting the requirements of 10 CFR part 71. The dry solid shall be chemically, thermally, and radiolytically stable to the extent that the equilibrium pressure in the sealed container will not exceed the safe operating pressure for that container during the period from canning through a minimum of 90 days after receipt (transfer of physical custody) at the Federal repository. All of these high-level radioactive wastes shall be transferred to a Federal repository no later than 10 years following separation of fission products from the irradiated fuel. Upon receipt, the Federal repository will assume permanent custody of these radioactive waste materials although

industry will pay the Federal Government a charge which together with interest on unexpended balances will be designed to defray all costs of disposal and perpetual surveillance. The Department of Energy will take title to the radioactive waste material upon transfer to a Federal repository. Before retirement of the reprocessing plant from operational status and before termination of licensing pursuant to §50.82, transfer of all such wastes to a Federal repository shall be completed. Federal repositories, which will be limited in number, will be designated later by the Commission.

3. Disposal of high-level radioactive fission product waste material will not be permitted on any land other than that owned and controlled by the Federal Government.

4. A design objective for fuel reprocessing plants shall be to facilitate decontamination and removal of all significant radioactive wastes at the time the facility is permanently decommissioned. Criteria for the extent of decontamination to be required upon decommissioning and license termination will be developed in consultation with competent groups. Opportunity will be afforded for public comment before such criteria are made effective.

5. Applicants proposing to operate fuel reprocessing plants, in submitting information concerning financial qualifications as required by §50.33(f), shall include information enabling the Commission to determine whether the applicant is financially qualified, among other things, to provide for the removal and disposal of radioactive wastes, during operation and upon decommissioning of the facility, in accordance with the Commission's regulations, including the requirements set out in this appendix.

6. With respect to fuel reprocessing plants already licensed, the licenses will be appropriately conditioned to carry out the purposes of the policy stated above with respect to high-level radioactive fission product wastes generated after installation of new equipment for interim storage of liquid wastes, or after installation of equipment required for solidification without interim liquid storage. In either case, such equipment shall be installed at the earliest practicable date, taking into account the time required for design, procurement and installation thereof. With respect to such plants, the application of the policy stated in this appendix to existing wastes and to wastes generated prior to the installation of such equipment, will be the subject of a further rulemaking proceeding.

[35 FR 17533, Nov. 14, 1970, as amended at 36 FR 5411, Mar. 23, 1971; 42 FR 20139, Apr. 18, 1977; 45 FR 14201, Mar. 5, 1980; 70 FR 3599, Jan. 26, 2005]

APPENDIX G TO PART 50—FRACTURE TOUGHNESS REQUIREMENTS

I. INTRODUCTION AND SCOPE

This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

The ASME Code forms the basis for the requirements of this appendix. "ASME Code" means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. If no section is specified, the reference is to Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components." "Section XI" means Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components." If no edition or addenda are specified, the ASME Code edition and addenda and any limitations and modifications thereof, which are specified in §50.55a, are applicable.

The sections, editions and addenda of the ASME Boiler and Pressure Vessel Code specified in §50.55a have been approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the FEDERAL REGISTER. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, NY 10017, and are available for inspection at the NRC Library, 11545 Rockville Pike, Two White Flint North, Rockville, MD 20852–2738.

The requirements of this appendix apply to the following materials:

A. Carbon and low-alloy ferritic steel plate, forgings, castings, and pipe with specified minimum yield strengths not over 50,000 psi (345 MPa), and to those with specified minimum yield strengths greater than 50,000 psi (345 MPa) but not over 90,000 psi (621 MPa) if qualified by using methods equivalent to those described in paragraph G–2110 of appendix G of section XI of the latest edition and addenda of the ASME Code incorporated by reference into §50.55a(b)(2).

B. Welds and weld heat-affected zones in the materials specified in paragraph I.A. of this appendix.

C. Materials for bolting and other types of fasteners with specified minimum yield strengths not over 130,000 psi (896 MPa).

NOTE: The adequacy of the fracture toughness of other ferritic materials not covered in this section must be demonstrated to the Director, Office of Nuclear Reactor Regulation, on an individual case basis.

II. DEFINITIONS

A. *Ferritic material* means carbon and low-alloy steels, higher alloy steels including all stainless alloys of the 4xx series, and maraging and precipitation hardening steels with a predominantly body-centered cubic crystal structure.

B. *System hydrostatic tests* means all preoperational system leakage and hydrostatic pressure tests and all system leakage and hydrostatic pressure tests performed during the service life of the pressure boundary in compliance with the ASME Code, Section XI.

C. *Specified minimum yield strength* means the minimum yield strength (in the unirradiated condition) of a material specified in the construction code under which the component is built under §50.55a.

D. RT_{NDT} means the reference temperature of the material, for all conditions.

(i) For the pre-service or unirradiated condition, RT_{NDT} is evaluated according to the procedures in the ASME Code, Paragraph NB–2331.

(ii) For the reactor vessel beltline materials, RT_{NDT} must account for the effects of neutron radiation.

E. ΔRT_{NDT} means the transition temperature shift, or change in RT_{NDT}, due to neutron radiation effects, which is evaluated as the difference in the 30 ft-lb (41 J) index temperatures from the average Charpy curves measured before and after irradiation.

F. *Beltline* or *Beltline region of reactor vessel* means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

III. FRACTURE TOUGHNESS TESTS

A. To demonstrate compliance with the fracture toughness requirements of section IV of this appendix, ferritic materials must be tested in accordance with the ASME Code and, for the beltline materials, the test requirements of appendix H of this part. For a reactor vessel that was constructed to an ASME code earlier than the Summer 1972 Addenda of the 1971 Edition (under §50.55a), the fracture toughness data and data analysis must be supplemented in a manner approved by the Director, Office of Nuclear Reactor Regulation, to demonstrate equivalence with the fracture toughness requirements of this appendix.

B. Test methods for supplemental fracture toughness tests described in paragraph IV.A.1.b of this appendix must be submitted to and approved by the Director, Office of Nuclear Reactor Regulation, prior to testing.

C. All fracture toughness test programs conducted in accordance with paragraphs III.A and III.B must comply with ASME Code requirements for calibration of test equipment, qualification of test personnel, and retention of records of these functions and of the test data.

IV. FRACTURE TOUGHNESS REQUIREMENTS

A. The pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials must meet the requirements of the ASME Code, supplemented by the additional requirements set forth below, for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences. Reactor vessels may continue to be operated only for that service period within which the requirements of this section are satisfied. For the reactor vessel beltline materials, including welds, plates and forgings, the values of RT_{NDT} and Charpy upper-shelf energy must account for the effects of neutron radiation, including the results of the surveillance program of appendix H of this part. The effects of neutron radiation must consider the radiation conditions (*i.e.*, the fluence) at the deepest point on the crack front of the flaw assumed in the analysis.

1. Reactor Vessel Charpy Upper-Shelf Energy Requirements

a. Reactor vessel beltline materials must have Charpy upper-shelf energy[1] in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 75 ft-lb (102 J) initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68 J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. This analysis must use the latest edition and addenda of the ASME Code incorporated by reference into §50.55a(b)(2) at the time the analysis is submitted.

[1] Defined in ASTME 185–79 and –82 which are incorporated by reference in appendix H to part 50.

b. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation may be obtained from results of supplemental fracture toughness tests for use in the analysis specified in section IV.A.1.a.

c. The analysis for satisfying the requirements of section IV.A.1 of this appendix must be submitted, as specified in §50.4, 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.

2. Pressure-Temperature Limits and Minimum Temperature Requirements

a. Pressure-temperature limits and minimum temperature requirements for the reactor vessel are given in table 1, and are defined by the operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operational occurrences), the vessel pressure, whether or not fuel is in the vessel, and whether the core is critical. In table 1, the vessel pressure is defined as a percentage of the preservice system hydrostatic test pressure. The appropriate requirements on both the pressure-temperature limits and the minimum permissible temperature must be met for all conditions.

b. The pressure-temperature limits identified as ''ASME Appendix G limits'' in table 1 require that the limits must be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G of Section XI of the ASME Code.

c. The minimum temperature requirements given in table 1 pertain to the controlling material, which is either the material in the closure flange or the material in the beltline region with the highest reference temperature. As specified in table 1, the minimum temperature requirements and the controlling material depend on the operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operational occurrences), the vessel pressure, whether fuel is in the vessel, and whether the core is critical. The metal temperature of the controlling material, in the region of the controlling material which has the least favorable combination of stress and temperature must exceed the appropriate minimum temperature requirement for the condition and pressure of the vessel specified in table 1.

d. Pressure tests and leak tests of the reactor vessel that are required by Section XI of the ASME Code must be completed before the core is critical.

B. If the procedures of section IV.A. of this appendix do not indicate the existence of an equivalent safety margin, the reactor vessel beltline may be given a thermal annealing treatment to recover the fracture toughness of the material, subject to the requirements of §50.66. The reactor vessel may continue to be operated only for that service period within which the predicted fracture toughness of the beltline region materials satisfies the requirements of section IV.A. of this appendix using the values of RT_{NDT} and Charpy upper-shelf energy that include the effects of annealing and subsequent irradiation.

TABLE 1—PRESSURE AND TEMPERATURE REQUIREMENTS FOR THE REACTOR PRESSURE VESSEL

Operating condition	Vessel pressure[1]	Requirements for pressure-temperature limits	Minimum temperature requirements
1. Hydrostatic pressure and leak tests (core is not critical):			
1.a Fuel in the vessel	≤20%	ASME Appendix G Limits	([2])
1.b Fuel in the vessel	>20%	ASME Appendix G Limits	([2]) + 90 °F ([6])
1.c No fuel in the vessel (Preservice Hydrotest Only).	ALL	(Not Applicable)	([3]) + 60 °F
2. Normal operation (incl. heat-up and cool-down), including anticipated operational occurrences:			
2.a Core not critical	≤20%	ASME Appendix G Limits	([2])
2.b Core not critical	>20%	ASME Appendix G Limits	([2]) + 120 °F ([6])
2.c Core critical	≤20%	ASME Appendix G Limits + 40 °F	Larger of [([4])] or [([2]) + 40 °F]
2.d Core critical	>20%	ASME Appendix G Limits + 40 °F	Larger of [([4])] or [([2]) + 160 °F]
2.e Core critical for BWR ([5])	≤20%	ASME Appendix G Limits + 40 °F	([2]) + 60 °F

[1] Percent of the preservice system hydrostatic test pressure.
[2] The highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload.
[3] The highest reference temperature of the vessel.
[4] The minimum permissible temperature for the inservice system hydrostatic pressure test.
[5] For boiling water reactors (BWR) with water level within the normal range for power operation.
[6] Lower temperatures are permissible if they can be justified by showing that the margins of safety of the controlling region are equivalent to those required for the beltline when it is controlling.

[60 FR 65474, Dec. 19, 1995, as amended at 73 FR 5723, Jan. 31, 2008; 78 FR 34248, June 7, 2013; 78 FR 75450, Dec. 12, 2013; 84 FR 65644, Nov. 29, 2019]

APPENDIX H TO PART 50—REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM REQUIREMENTS

I. Introduction
II. Definitions
III. Surveillance Program Criteria
IV. Report of Test Results

I. INTRODUCTION

The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. These data will be used as described in section IV of appendix G to part 50.

ASTM E 185–73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels"; ASTM E 185–79, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels"; and ASTM E 185–82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels"; which are referenced in the following paragraphs, have been approved for incorporation by reference by the Director of the Federal Register. Copies of ASTM E 185–73, –79, and –82, may be purchased from the American Society for Testing and Materials, 1916 Race Street, Philadelphia, PA 19103 and are available for inspection at the NRC Library, 11545 Rockville Pike, Two White Flint North, Rockville, MD 20852–2738.

II. DEFINITIONS

All terms used in this appendix have the same meaning as in appendix G.

III. SURVEILLANCE PROGRAM CRITERIA

A. No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence at the end of the design life of the vessel will not exceed 10^{17} n/cm^2 (E >1 MeV).

B. Reactor vessels that do not meet the conditions of paragraph III.A of this appendix must have their beltline materials monitored by a surveillance program complying with ASTM E 185, as modified by this appendix.

1. The design of the surveillance program and the withdrawal schedule must meet the requirements of the edition of the ASTM E 185 that is current on the issue date of the ASME code to which the reactor vessel was purchased; for reactor vessels purchased after 1982, the design of the surveillance program and the withdrawal schedule must meet the requirements of ASTM E 185–82. For reactor vessels purchased in or before 1982, later editions of ASTM E 185 may be used, but including only those editions through 1982. For each capsule withdrawal, the test procedures and reporting requirements must meet the requirements of the ASTM E 185 to the extent practicable for the configuration of the specimens in the capsule. If any of the optional provisions in paragraphs III.B.4(a) through (d) of this section are implemented in lieu of ASTM E 185, the number of specimens included or tested in the surveillance program shall be adjusted as specified in paragraphs III.B.4(a) through (d) of this section.

2. Surveillance specimen capsules must be located near the inside vessel wall in the beltline region so that the specimen irradiation history duplicates, to the extent practicable within the physical constraints of the system, the neutron spectrum, temperature history, and maximum neutron fluence experienced by the reactor vessel inner surface. If the capsule holders are attached to the vessel wall or to the vessel cladding, construction and inservice inspection of the attachments and attachment welds must be done according to the requirements for permanent structural attachments to reactor vessels given in Sections III and XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The design and location of the capsule holders must permit insertion of replacement capsules. Accelerated irradiation capsules may be used in addition to the required number of surveillance capsules.

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

4. Optional provisions. As used in this section, references to ASTM E 185 include the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased through the 1982 edition.

(a) *First Provision:* Heat-Affected Zone Specimens—The inclusion or testing of weld heat-affected zone Charpy impact specimens within the surveillance program as specified in ASTM E 185 is optional.

(b) *Second Provision:* Tension Specimens—If this provision is implemented, the minimum number of tension specimens to be included and tested in the surveillance program shall be as specified in paragraphs III.B.4(b)(i) and (ii) of this section.

(i) Unirradiated Tension Specimens—Two tension specimens from each base and weld material required by ASTM E 185 shall be tested, with one specimen tested at room temperature and the other specimen tested at the service temperature; and

(ii) Irradiated Tension Specimens—Two tension specimens from each base and weld material required by ASTM E 185 shall be included in each surveillance capsule and tested, with one specimen tested at room temperature and the other specimen tested at the service temperature.

(c) *Third Provision:* Correlation Monitor Materials—The testing of correlation monitor material specimens within the surveillance program as specified in ASTM E 185 is optional.

(d) *Fourth Provision:* Thermal Monitor—The inclusion or examination of thermal monitors within the surveillance program as specified in ASTM E 185 is optional.

C. Requirements for an Integrated Surveillance Program.

1. In an integrated surveillance program, the representative materials chosen for surveillance for a reactor are irradiated in one or more other reactors that have similar design and operating features. Integrated surveillance programs must be approved by the Director, Office of Nuclear Reactor Regulation, on a case-by-case basis. Criteria for approval include the following:

a. The reactor in which the materials will be irradiated and the reactor for which the materials are being irradiated must have sufficiently similar design and operating features to permit accurate comparisons of the predicted amount of radiation damage.

b. Each reactor must have an adequate dosimetry program.

c. There must be adequate arrangement for data sharing between plants.

d. There must be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

e. There must be substantial advantages to be gained, such as reduced power outages or reduced personnel exposure to radiation, as a direct result of not requiring surveillance capsules in all reactors in the set.

2. No reduction in the requirements for number of materials to be irradiated, specimen types, or number of specimens per reactor is permitted.

3. After (the effective date of this section), no reduction in the amount of testing is permitted unless previously authorized by the Director, Office of Nuclear Reactor Regulation.

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, within eighteen months of the date of capsule withdrawal, unless an extension is granted by the Director, Office of Nuclear Reactor Regulation.

B. The report must include the data required by ASTM E 185, as specified in paragraph III.B.1 of this appendix, and the results of all fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions.

C. If a change in the Technical Specifications is required, either in the pressure-temperature limits or in the operating procedures required to meet the limits, the expected date for submittal of the revised Technical Specifications must be provided with the report.

[60 FR 65476, Dec. 19, 1995, as amended at 68 FR 75390, Dec. 31, 2003; 73 FR 5723, Jan. 31, 2008; 84 FR 65644, Nov. 29, 2019; 85 FR 62207, Oct. 2, 2020]

APPENDIX I TO PART 50—NUMERICAL GUIDES FOR DESIGN OBJECTIVES AND LIMITING CONDITIONS FOR OPERATION TO MEET THE CRITERION ''AS LOW AS IS REASONABLY ACHIEVABLE'' FOR RADIOACTIVE MATERIAL IN LIGHT-WATER-COOLED NUCLEAR POWER REACTOR EFFLUENTS

SECTION I. *Introduction.* Section 50.34a provides that an application for a construction permit shall include a description of the preliminary design of equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal conditions, including expected occurrences. In the case of an application filed on or after January 2, 1971, the application must also identify the design objectives, and the means to be employed, for keeping levels of radioactive material in effluents to unrestricted areas as low as practicable. Sections 52.47, 52.79, 52.137, and 52.157 of this chapter provide that applications for design certification, combined license, design approval, or manufacturing license, respectively, shall include a description of the equipment and procedures for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems.

Section 50.36a contains provisions designed to assure that releases of radioactive material from nuclear power reactors to unrestricted areas during normal conditions, including expected occurrences, are kept as low as practicable.

SECTION II. *Guides on design objectives for light-water-cooled nuclear power reactors licensed under 10 CFR part 50 or part 52 of this chapter*. The guides on design objectives set forth in this section may be used by an applicant for a construction permit as guidance in meeting the requirements of § 50.34a(a), or by an applicant for a combined license under part 52 of this chapter as guidance in meeting the requirements of § 50.34a(d), or by an applicant for a design approval, a design certification, or a manufacturing license as guidance in meeting the requirements of § 50.34a(e). The applicant shall provide reasonable assurance that the following design objectives will be met.

A. The calculated annual total quantity of all radioactive material above background[1] to be released from each light-water-cooled nuclear power reactor to unrestricted areas will not result in an estimated annual dose or dose commitment from liquid effluents for any individual in an unrestricted area from all pathways of exposure in excess of 3 millirems to the total body or 10 millirems to any organ.

B.1. The calculated annual total quantity of all radioactive material above background to be released from each light-water-cooled nuclear power reactor to the atmosphere will not result in an estimated annual air dose from gaseous effluents at any location near ground level which could be occupied by individuals in unrestricted areas in excess of 10 millirads for gamma radiation or 20 millirads for beta radiation.

2. Notwithstanding the guidance of paragraph B.1:

(a) The Commission may specify, as guidance on design objectives, a lower quantity of radioactive material above background to be released to the atmosphere if it appears that the use of the design objectives in paragraph B.1 is likely to result in an estimated annual external dose from gaseous effluents to any individual in an unrestricted area in excess of 5 millirems to the total body; and

(b) Design objectives based upon a higher quantity of radioactive material above background to be released to the atmosphere than the quantity specified in paragraph B.1 will be deemed to meet the requirements for keeping levels of radioactive material in gaseous effluents as low as is reasonably achievable if the applicant provides reasonable assurance that the proposed higher quantity will not result in an estimated annual external dose from gaseous effluents to any individual in unrestricted areas in excess of 5 millirems to the total body or 15 millirems to the skin.

C. The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form above background to be released from each light-water-cooled nuclear power reactor in effluents to the atmosphere will not result in an estimated annual dose or dose commitment from such radioactive iodine and radioactive material in particulate form for any individual in an unrestricted area from all pathways of exposure in excess of 15 millirems to any organ.

D. In addition to the provisions of paragraphs A, B, and C above, the applicant shall include in the radwaste system all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. As an interim measure and until establishment and adoption of better values (or other appropriate criteria), the values $1000 per total body man-rem and $1000 per man-thyroid-rem (or such lesser values as may be demonstrated to be suitable in a particular case) shall be used in this cost-benefit analysis. The requirements of this paragraph D need not be complied with by persons who have filed applications for construction permits which were docketed on or after January 2, 1971, and prior to June 4, 1976, if the radwaste systems and equipment described in the preliminary or final safety analysis report and amendments thereto satisfy the Guides on Design Objectives for Light-Water-Cooled Nuclear Power Reactors proposed in the Concluding Statement of Position of the Regulatory Staff in Docket-RM-50-2 dated February 20, 1974, pp. 25–30, reproduced in the annex to this appendix I.

SECTION III. *Implementation*. A.1. Conformity with the guides on design objectives of Section II shall be demonstrated by calculational procedures based upon models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated, all uncertainties being considered together. Account shall be taken of the cumulative effect of all sources and pathways within the plant contributing to the particular type of effluent being considered. For determination of design objectives in accordance with the guides of Section II, the estimations of exposure shall be made with respect to such potential land and water usage and food pathways as could actually exist during the term of plant operation: *Provided*, That, if the requirements of paragraph B of Section III are fulfilled, the applicant shall be deemed to have complied with the requirements of paragraph C of Section II with respect to radioactive iodine if estimations of exposure

[1] Here and elsewhere in this appendix background means radioactive materials in the environment and in the effluents from light-water-cooled power reactors not generated in, or attributable to, the reactors of which specific account is required in determining design objectives.

are made on the basis of such food pathways and individual receptors as actually exist at the time the plant is licensed.

2. The characteristics attributed to a hypothetical receptor for the purpose of estimating internal dose commitment shall take into account reasonable deviations of individual habits from the average. The applicant may take account of any real phenomenon or factors actually affecting the estimate of radiation exposure, including the characteristics of the plant, modes of discharge of radioactive materials, physical processes tending to attenuate the quantity of radioactive material to which an individual would be exposed, and the effects of averaging exposures over times during which determining factors may fluctuate.

B. If the applicant determines design objectives with respect to radioactive iodine on the basis of existing conditions and if potential changes in land and water usage and food pathways could result in exposures in excess of the guideline values of paragraph C of Section II, the applicant shall provide reasonable assurance that a monitoring and surveillance program will be performed to determine:

1. The quantities of radioactive iodine actually released to the atmosphere and deposited relative to those estimated in the determination of design objectives;

2. Whether changes in land and water usage and food pathways which would result in individual exposures greater than originally estimated have occurred; and

3. The content of radioactive iodine and foods involved in the changes, if and when they occur.

SECTION IV. *Guides on technical specifications for limiting conditions for operation for light-water-cooled nuclear power reactors licensed under 10 CFR part 50 or part 52 of this chapter*. The guides on limiting conditions for operation for light-water-cooled nuclear power reactors set forth below may be used by an applicant for an operating license under this part or a design certification or combined license under part 52 of this chapter, or a licensee who has submitted a certification of permanent cessation of operations under §50.82(a)(1) or §52.110 of this chapter as guidance in developing technical specifications under §50.36a(a) to keep levels of radioactive materials in effluents to unrestricted areas as low as is reasonably achievable.

Section 50.36a(b) provides that licensees shall be guided by certain considerations in establishing and implementing operating procedures specified in technical specifications that take into account the need for operating flexibility and at the same time assure that the licensee will exert his best effort to keep levels of radioactive material in effluents as low as is reasonably achievable. The guidance set forth below provides additional and more specific guidance to licensees in this respect.

Through the use of the guides set forth in this section it is expected that the annual release of radioactive material in effluents from light-water-cooled nuclear power reactors can generally be maintained within the levels set forth as numerical guides for design objectives in Section II.

At the same time, the licensee is permitted the flexibility of operations, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power even under unusual conditions which may temporarily result in releases higher than numerical guides for design objectives but still within levels that assure that the average population exposure is equivalent to small fractions of doses from natural background radiation. It is expected that in using this operational flexibility under unusual conditions, the licensee will exert his best efforts to keep levels of radioactive material in effluents within the numerical guides for design objectives.

A. If the quantity of radioactive material actually released in effluents to unrestricted areas from a light-water-cooled nuclear power reactor during any calendar quarter is such that the resulting radiation exposure, calculated on the same basis as the respective design objective exposure, would exceed one-half the design objective annual exposure derived pursuant to Sections II and III, the licensee shall: [2]

1. Make an investigation to identify the causes for such release rates;

2. Define and initiate a program of corrective action; and

3. Report these actions as specified in §50.4, within 30 days from the end of the quarter during which the release occurred.

B. The licensee shall establish an appropriate surveillance and monitoring program to:

1. Provide data on quantities of radioactive material released in liquid and gaseous effluents to assure that the provisions of paragraph A of this section are met;

2. Provide data on measurable levels of radiation and radioactive materials in the environment to evaluate the relationship between quantities of radioactive material released in effluents and resultant radiation

[2] Section 50.36a(a)(2) requires the licensee to submit certain reports to the Commission with regard to the quantities of the principal radionuclides released to unrestricted areas. It also provides that, on the basis of such reports and any additional information the Commission may obtain from the licensee and others, the Commission may from time to time require the license to take such action as the Commission deems appropriate.

doses to individuals from principal pathways of exposure; and

3. Identify changes in the use of unrestricted areas (e.g., for agricultural purposes) to permit modifications in monitoring programs for evaluating doses to individuals from principal pathways of exposure.

C. If the data developed in the surveillance and monitoring program described in paragraph B of Section III or from other monitoring programs show that the relationship between the quantities of radioactive material released in liquid and gaseous effluents and the dose to individuals in unrestricted areas is significantly different from that assumed in the calculations used to determine design objectives pursuant to Sections II and III, the Commission may modify the quantities in the technical specifications defining the limiting conditions in a license to operate a light-water-cooled nuclear power reactor or a license whose holder has submitted a certification of permanent cessation of operations under §50.82(a)(1).

SECTION V. *Effective dates.* A. The guides for limiting conditions for operation set forth in this appendix shall be applicable in any case in which an application was filed on or after January 2, 1971, for a construction permit for a light-water-cooled nuclear power reactor under this part, or a design certification, a combined license, or a manufacturing license for a light-water-cooled nuclear power reactor under part 52 of this chapter.

B. For each light-water-cooled nuclear power reactor constructed pursuant to a permit for which application was filed prior to January 2, 1971, the holder of the permit or a license, authorizing operation of the reactor shall, within a period of twelve months from June 4, 1975, file with the Commission:

1. Such information as is necessary to evaluate the means employed for keeping levels of radioactivity in effluents to unrestricted areas as low as is reasonably achievable, including all such information as is required by §50.34a (b) and (c) not already contained in his application; and

2. Plans and proposed technical specifications developed for the purpose of keeping releases of radioactive materials to unrestricted areas during normal reactor operations, including expected operational occurrences, as low as is reasonably achievable.

CONCLUDING STATEMENT OF POSITION OF THE REGULATORY STAFF (DOCKET-RM-50-2)

GUIDES ON DESIGN OBJECTIVES FOR LIGHT-WATER-COOLED NUCLEAR POWER REACTORS

A. For radioactive material above background[1] in liquid effluents to be released to unrestricted areas:

1. The calculated annual total quantity of all radioactive material from all light-water-cooled nuclear power reactors at a site should not result in an annual dose or dose commitment to the total body or to any organ of an individual in an unrestricted area from all pathways of exposure in excess of 5 millirems; and

2. The calculated annual total quantity of radioactive material, except tritium and dissolved gases, should not exceed 5 curies for each light-water-cooled reactor at a site.

3. Notwithstanding the guidance in paragraph A.2, for a particular site, if an applicant for a permit to construct a light-water-cooled nuclear power reactor has proposed baseline in-plant control measures[2] to reduce the possible sources of radioactive material in liquid effluent releases and the calculated quantity exceeds the quantity set forth in paragraph A.2, the requirements for design objectives for radioactive material in liquid effluents may be deemed to have been met provided:

a. The applicant submits, as specified in §50.4, an evaluation of the potential for effects from long-term buildup on the environment in the vicinity of the site of radioactive material, with a radioactive half-life greater than one year, to be released; and

b. The provisions of paragraph A.1 are met.

B. For radioactive material above background in gaseous effluents the annual total quantity of radioactive material to be released to the atmosphere by all light-water-cooled nuclear power reactors at a site:

1. The calculated annual air dose due to gamma radiation at any location near ground level which could be occupied by individuals at or beyond the boundary of the site should not exceed 10 millirads; and

[1] "Background," means the quantity of radioactive material in the effluent from light-water-cooled nuclear power reactors at a site that did not originate in the reactors.

[2] Such measures may include treatment of clear liquid waste streams (normally tritiated, nonaerated, low conductivity equipment drains and pump seal leakoff), dirty liquid waste streams (normally nontritiated, aerated, high conductivity building sumps, floor and sample station drains), steam generator blowdown streams, chemical waste streams, low purity and high purity liquid streams (resin regenerate and laboratory wastes), as appropriate for the type of reactor.

2. The calculated annual air dose due to beta radiation at any location near ground level which could be occupied by individuals at or beyond the boundary of the site should not exceed 20 millirads.

3. Notwithstanding the guidance in paragraphs B.1 and B.2, for a particular site:

a. The Commission may specify, as guidance on design objectives, a lower quantity of radioactive material above background in gaseous effluents to be released to the atmosphere if it appears that the use of the design objectives described in paragraphs B.1 and B.2 is likely to result in an annual dose to an individual in an unrestricted area in excess of 5 millirems to the total body or 15 millirems to the skin; or

b. Design objectives based on a higher quantity of radioactive material above background in gaseous effluents to be released to the atmosphere than the quantity specified in paragraphs B.1 and B.2 may be deemed to meet the requirements for keeping levels of radioactive material in gaseous effluents as low as practicable if the applicant provides reasonable assurance that the proposed higher quantity will not result in annual doses to an individual in an unrestricted area in excess of 5 millirems to the total body or 15 millirems to the skin.

C. For radioactive iodine and radioactive material in particulate form above background released to the atmosphere:

1. The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form from all light-water-cooled nuclear power reactors at a site should not result in an annual dose or dose commitment to any organ of an individual in an unrestricted area from all pathways of exposure in excess of 15 millirems. In determining the dose or dose commitment the portion thereof due to intake of radioactive material via the food pathways may be evaluated at the locations where the food pathways actually exist; and

2. The calculated annual total quantity of iodine-131 in gaseous effluents should not exceed 1 curie for each light-water-cooled nuclear power reactor at a site.

3. Notwithstanding the guidance in paragraphs C.1 and C.2 for a particular site, if an applicant for a permit to construct a light-water-cooled nuclear power reactor has proposed baseline in-plant control measures[3] to reduce the possible sources of radioactive iodine releases, and the calculated annual quantities taking into account such control measures exceed the design objective quantities set forth in paragraphs C.1 and C.2, the requirements for design objectives for radioactive iodine and radioactive material in particulate form in gaseous effluents may be deemed to have been met provided the calculated annual total quantity of all radioactive iodine and radioactive material in particulate form that may be released in gaseous effluents does not exceed four times the quantity calculated pursuant to paragraph C.1.

[3] Such in-plant control measures may include treatment of steam generator blowdown tank exhaust, clean steam supplies for turbine gland seals, condenser vacuum systems, containment purging exhaust and ventilation exhaust systems and special design features to reduce contaminated steam and liquid leakage from valves and other sources such as sumps and tanks, as appropriate for the type of reactor.

[40 FR 19442, May 5, 1975, as amended at 40 FR 40818, Sept. 4, 1975; 40 FR 58847, Dec. 19, 1975; 41 FR 16447, Apr. 19, 1976; 42 FR 20139, Apr. 18, 1977; 51 FR 40311, Nov. 6, 1986; 61 FR 39303, July 29, 1996; 72 FR 49507, Aug. 28, 2007]

APPENDIX J TO PART 50—PRIMARY REACTOR CONTAINMENT LEAKAGE TESTING FOR WATER-COOLED POWER REACTORS

This appendix includes two options, A and B, either of which can be chosen for meeting the requirements of this appendix.

OPTION A—PRESCRIPTIVE REQUIREMENTS

Table of Contents

I. INTRODUCTION

One of the conditions of all operating licenses under this part and combined licenses under part 52 of this chapter for water-cooled power reactors as specified in §50.54(o) is that primary reactor containments shall meet 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, and serves as an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.

B. ''Containment isolation valve'' means any valve which is relied upon to perform a containment isolation function.

C. ''Reactor containment leakage test program'' includes the performance of Type A, Type B, and Type C tests, described in II.F, II.G, and II.H, respectively.

D. ''Leakage rate'' for test purposes is that leakage which occurs in a unit of time, stated as a percentage of weight of the original content of containment air at the leakage rate test pressure that escapes to the outside atmosphere during a 24-hour test period.

E. ''Overall integrated leakage rate'' means that leakage rate which obtains from a summation of leakage through all potential leakage paths including containment welds, valves, fittings, and components which penetrate containment.

F. ''Type A Tests'' means tests intended to measure the primary reactor containment overall integrated leakage rate (1) after the containment has been completed and is ready for operation, and (2) at periodic intervals thereafter.

G. ''Type B Tests'' means tests intended to detect local leaks and to measure leakage across each pressure-containing or leakage-limiting boundary for the following primary reactor containment penetrations:

1. Containment penetrations whose design incorporates resilient seals, gaskets, or sealant componds, piping penetrations fitted with expansion bellows, and electrical penetrations fitted with flexible metal seal assemblies.

2. Air lock door seals, including door operating mechanism penetrations which are part of the containment pressure boundary.

3. Doors with resilient seals or gaskets except for seal-welded doors.

4. Components other than those listed in II.G.1, II.G.2, or II.G.3 which must meet the acceptance criteria in III.B.3.

H. ''Type C Tests'' means tests intended to measure containment isolation valve leakage rates. The containment isolation valves included are those that:

1. Provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation, such as purge and ventilation, vacuum relief, and instrument valves;

2. Are required to close automatically upon receipt of a containment isolation signal in response to controls intended to effect containment isolation;

3. Are required to operate intermittently under postaccident conditions; and

4. Are in main steam and feedwater piping and other systems which penetrate containment of direct-cycle boiling water power reactors.

I. Pa (p.s.i.g.) means the calculated peak containment internal pressure related to the design basis accident and specified either in the technical specification or associated bases.

J. Pt (p.s.i.g.) means the containment vessel reduced test pressure selected to measure the integrated leakage rate during periodic Type A tests.

K. La (percent/24 hours) means the maximum allowable leakage rate at pressure Pa as specified for preoperational tests in the technical specifications or associated bases, and as specified for periodic tests in the operating license or combined license, including the technical specifications in any referenced design certification or manufactured reactor used at the facility.

L. Ld (percent/24 hours) means the design leakage rate at pressure, Pa, as specified in the technical specifications or associated bases.

M. Lt (percent/24 hours) means the maximum allowable leakage rate at pressure Pt derived from the preoperational test data as specified in III.A.4.(a)(iii).

N. Lam, Ltm (percent/24 hours) means the total measured containment leakage rates at pressure Pa and Pt, respectively, obtained from testing the containment with components and systems in the state as close as practical to that which would exist under design basis accident conditions (e.g., vented, drained, flooded or pressurized).

O. ''Acceptance criteria'' means the standard against which test results are to be compared for establishing the functional acceptability of the containment as a leakage limiting boundary.

III. LEAKAGE TESTING REQUIREMENTS

A program consisting of a schedule for conducting Type A, B, and C tests shall be developed for leak testing the primary reactor containment and related systems and components penetrating primary containment pressure boundary.

Upon completion of construction of the primary reactor containment, including installation of all portions of mechanical, fluid, electrical, and instrumentation systems penetrating the primary reactor containment pressure boundary, and prior to any reactor operating period, preoperational and periodic leakage rate tests, as applicable, shall be conducted in accordance with the following:

A. *Type A test*—1. *Pretest requirements.* (a) Containment inspection in accordance with V. A. shall be performed as a prerequisite to the performance of Type A tests. During the period between the initiation of the containment inspection and the performance of the Type A test, no repairs or adjustments shall be made so that the containment can be tested in as close to the ''as is'' condition as practical. During the period between the completion of one Type A test and the initiation of the containment inspection for the subsequent Type A test, repairs or adjustments shall be made to components whose leakage exceeds that specified in the technical specification as soon as practical after identification. If during a Type A test, including the supplemental test specified in III.A.3.(b), potentially excessive leakage paths are identified which will interfere with satisfactory completion of the test, or which result in the Type A test not meeting the acceptance criteria III.A.4.(b) or III.A.5.(b), the Type A test shall be terminated and the leakage through such paths shall be measured using local leakage testing methods. Repairs and/or adjustments to equipment shall be made and Type A test performed. The corrective action taken and the change in leakage rate determined from the tests and overall integrated leakage determined from local leak and Type A tests shall be included in the summary report required by V.B.

(b) Closure of containment isolation valves for the Type A test shall be accomplished by normal operation and without any preliminary exercising or adjustments (e.g., no tightening of valve after closure by valve motor). Repairs of maloperating or leaking valves shall be made as necessary. Information on any valve closure malfunction or valve leakage that require corrective action before the test, shall be included in the summary report required by V.B.

(c) The containment test conditions shall stabilize for a period of about 4 hours prior to the start of a leakage rate test.

(d) Those portions of the fluid systems that are part of the reactor coolant pressure boundary and are open directly to the containment atmosphere under post-accident conditions and become an extension of the boundary of the containment shall be opened or vented to the containment atmosphere prior to and during the test. Portions of closed systems inside containment that penetrate containment and rupture as a result of a loss of coolant accident shall be vented to the containment atmosphere. All vented systems shall be drained of water or other fluids to the extent necessary to assure exposure of the system containment isolation valves to containment air test pressure and to assure they will be subjected to the post accident differential pressure. Systems that are required to maintain the plant in a safe condition during the test shall be operable in their normal mode, and need not be vented. Systems that are normally filled with water and operating under post-accident conditions, such as the containment heat removal system, need not be vented. However, the containment isolation valves in the systems defined in III.A.1.(d) shall be tested in accordance with III.C. The measured leakage rate from these tests shall be included in the summary report required by V.B.

2. *Conduct of tests.* Preoperational leakage rate tests at either reduced or at peak pressure, shall be conducted at the intervals specified in III.D.

3. *Test Methods.* (a) All Type A tests shall be conducted in accordance with the provisions of the American National Standards N45.4–1972, ''Leakage Rate Testing of Containment Structures for Nuclear Reactors,'' March 16, 1972. In addition to the Total time and Point-to-Point methods described in that standard, the Mass Point Method, when used with a test duration of at least 24 hours, is an acceptable method to use to calculate leakage rates. A typical description of the Mass Point method can be found in the American National Standard ANSI/ANS 56.8–1987, ''Containment System Leakage Testing Requirements,'' January 20, 1987. Incorporation of ANSI N45.4–1972 by reference was approved by the Director of the Federal Register. Copies of this standard, as well as ANSI/ANS–56.8–1987, ''Containment System Leakage Testing Requirements'' (dated January 20, 1987) may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, IL 60525. A copy of each of these standards is available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852–2738.

(b) The accuracy of any Type A test shall be verified by a supplemental test. An acceptable method is described in Appendix C of ANSI N45.4–1972. The supplemental test method selected shall be conducted for sufficient duration to establish accurately the change in leakage rate between the Type A and supplemental test. Results from this supplemental test are acceptable provided the difference between the supplemental test data and the Type A test data is within 0.25 La (or 0.25 Lt). If results are not within 0.25 La (or 0.25 Lt), the reason shall be determined, corrective action taken, and a successful supplemental test performed.

(c) Test leakage rates shall be calculated using absolute values corrected for instrument error.

4. *Preoperational leakage rate tests.* (a) *Test pressure*—(1) *Reduced pressure tests.* (i) An initial test shall be performed at a pressure Pt, not less than 0.50 Pa to measure a leakage rate Ltm.

(ii) A second test shall be performed at pressure Pa to measure a leakage rate Lam.

(iii) The leakage characteristics yielded by measurements Ltm and Lam shall establish the maximum allowable test leakage rate Lt of not more than La (Ltm/Lam). In the event Ltm/Lam is greater than 0.7, Lt shall be specified as equal to La (Pt/Pa).[1]

(2) *Peak pressure tests.* A test shall be performed at pressure Pa to measure the leakage rate Lam.

(b) *Acceptance criteria*—(1) *Reduced pressure tests.* The leakage rate Ltm shall be less than 0.75 Lt.

(2) *Peak pressure tests.* The leakage rate Lam shall be less than 0.75 La and not greater than Ld.

5. *Periodic leakage rate tests*—(a) *Test pressure.* (1) Reduced pressure tests shall be conducted at Pt;

(2) Peak pressure tests shall be conducted at Pa.

(b) *Acceptance criteria*—(1) *Reduced pressure tests.* The leakage rate Ltm shall be less than 0.75 Lt. If local leakage measurements are taken to effect repairs in order to meet the acceptance criteria, these measurements shall be taken at a test pressure Pt.

(2) *Peak pressure tests.* The leakage rate Lam shall be less than 0.75 La. If local leakage measurements are taken to effect repairs in order to meet the acceptance criteria, these measurements shall be taken at a test pressure Pa.

6. *Additional requirements.* (a) If any periodic Type A test fails to meet the applicable acceptance criteria in III.A.5.(b), the test schedule applicable to subsequent Type A tests will be reviewed and approved by the Commission.

(b) If two consecutive periodic Type A tests fail to meet the applicable acceptance criteria in III.A.5(b), notwithstanding the periodic retest schedule of III.D., a Type A test shall be performed at each plant shutdown for refueling or approximately every 18 months, whichever occurs first, until two consecutive Type A tests meet the acceptance criteria in III.A.5(b), after which time the retest schedule specified in III.D. may be resumed.

B. *Type B tests*—1. *Test methods.* Acceptable means of performing preoperation and periodic Type B tests include:

(a) Examination by halide leak-detection method (or by other equivalent test methods such as mass spectrometer) of a test chamber, pressurized with air, nitrogen, or pneumatic fluid specified in the technical specifications or associated bases and constructed as part of individual containment penetrations.

(b) Measurement of the rate of pressure loss of the test chamber of the containment penetration pressurized with air, nitrogen, or pneumatic fluid specified in the technical specifications or associated bases.

(c) Leakage surveillance by means of a permanently installed system with provisions for continuous or intermittent pressurization of individual or groups of containment penetrations and measurement of rate of pressure loss of air, nitrogen, or pneumatic fluid specified in the technical specification or associated bases through the leak paths.

2. *Test pressure.* All preoperational and periodic Type B tests shall be performed by local pneumatic pressurization of the containment penetrations, either individually or in groups, at a pressure not less than Pa.

3. *Acceptance criteria.* (See also Type C tests.) (a) The combined leakage rate of all penetrations and valves subject to Type B and C tests shall be less than 0.60 La, with the exception of the valves specified in III.C.3.

(b) Leakage measurements obtained through component leakage surveillance systems (e.g., continuous pressurization of individual containment components) that maintains a pressure not less than Pa at individual test chambers of containment penetrations during normal reactor operation, are acceptable in lieu of Type B tests.

C. *Type C tests*—1. *Test method.* Type C tests shall be performed by local pressurization. The pressure shall be applied in the same direction as that when the value would be required to perform its safety function, unless it can be determined that the results from the tests for a pressure applied in a different direction will provide equivalent or more conservative results. The test methods in III.B.1 may be substituted where appropriate. Each valve to be tested shall be closed by normal operation and without any preliminary exercising or adjustments (e.g., no tightening of valve after closure by valve motor).

2. *Test pressure.* (a) Valves, unless pressurized with fluid (e.g., water, nitrogen) from a seal system, shall be pressurized with air or nitrogen at a pressure of Pa.

[1] Such inservice inspections are required by § 50.55a.

(b) Valves, which are sealed with fluid from a seal system shall be pressurized with that fluid to a pressure not less than 1.10 Pa.

3. *Acceptance criterion.* The combined leakage rate for all penetrations and valves subject to Type B and C tests shall be less than 0.60 La. Leakage from containment isolation valves that are sealed with fluid from a seal system may be excluded when determining the combined leakage rate: *Provided,* That;

(a) Such valves have been demonstrated to have fluid leakage rates that do not exceed those specified in the technical specifications or associated bases, and

(b) The installed isolation valve seal-water system fluid inventory is sufficient to assure the sealing function for at least 30 days at a pressure of 1.10 Pa.

D. *Periodic retest schedule*—1. *Type A test.* (a) After the preoperational leakage rate tests, a set of three Type A tests shall be performed, at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year plant inservice inspections.[2]

(b) Permissible periods for testing. The performance of Type A tests shall be limited to periods when the plant facility is nonoperational and secured in the shutdown condition under the administrative control and in accordance with the safety procedures defined in the license.

2. *Type B tests.* (a) Type B tests, except tests for air locks, shall be performed during reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than 2 years. If opened following a Type A or B test, containment penetrations subject to Type B testing shall be Type B tested prior to returning the reactor to an operating mode requiring containment integrity. For primary reactor containment penetrations employing a continuous leakage monitoring system, Type B tests, except for tests of air locks, may, notwithstanding the test schedule specified under III.D.1., be performed every other reactor shutdown for refueling but in no case at intervals greater than 3 years.

(b)(i) Air locks shall be tested prior to initial fuel loading and at 6-month intervals thereafter at an internal pressure not less than P_a.

(ii) Air locks opened during periods when containment integrity is not required by the plant's Technical Specifications shall be tested at the end of such periods at not less than P_a.

(iii) Air locks opened during periods when containment integrity is required by the plant's Technical Specifications shall be tested within 3 days after being opened. For air lock doors opened more frequently than once every 3 days, the air lock shall be tested at least once every 3 days during the period of frequent openings. For air lock doors having testable seals, testing the seals fulfills the 3-day test requirements. In the event that the testing for this 3-day interval cannot be at P_a, the test pressure shall be as stated in the Technical Specifications. Air lock door seal testing shall not be substituted for the 6-month test of the entire air lock at not less than P_a.

(iv) The acceptance criteria for air lock testing shall be stated in the Technical Specifications.

3. *Type C tests.* Type C tests shall be performed during each reactor shutdown for refueling but in no case at intervals greater than 2 years.

IV. Special Testing Requirements

A. *Containment modification.* Any major modification, replacement of a component which is part of the primary reactor containment boundary, or resealing a seal-welded door, performed after the preoperational leakage rate test shall be followed by either a Type A, Type B, or Type C test, as applicable for the area affected by the modification. The measured leakage from this test shall be included in the summary report required by V.B. The acceptance criteria of III.A.5.(b), III.B.3., or III.C.3., as appropriate, shall be met. Minor modifications, replacements, or resealing of seal-welded doors, performed directly prior to the conduct of a scheduled Type A test do not require a separate test.

B. *Multiple leakage barrier or subatmospheric containments.* The primary reactor containment barrier of a multiple barrier or subatmospheric containment shall be subjected to Type A tests to verify that its leakage rate meets the requirements of this appendix. Other structures of multiple barrier or subatmospheric containments (e.g., secondary containments for boiling water reactors and shield buildings for pressurized water reactors that enclose the entire primary reactor containment or portions thereof) shall be subject to individual tests in accordance with the procedures specified in the technical specifications, or associated bases.

V. Inspection and Reporting of Tests

A. *Containment inspection.* A general inspection of the accessible interior and exterior surfaces of the containment structures and components shall be performed prior to any Type A test to uncover any evidence of structural deterioration which may affect either the containment structural integrity or leak-tightness. If there is evidence of structural deterioration, Type A tests shall not be performed until corrective action is taken in

[2] Such inservice inspections are required by §50.55a.

accordance with repair procedures, non destructive examinations, and tests as specified in the applicable code specified in §50.55a at the commencement of repair work. Such structural deterioration and corrective actions taken shall be included in the summary report required by V.B.

B. *Recordkeeping of test results.* 1. The preoperational and periodic tests must be documented in a readily available summary report that will be made available for inspection, upon request, at the nuclear power plant. The summary report shall include a schematic arrangement of the leakage rate measurement system, the instrumentation used, the supplemental test method, and the test program selected as applicable to the preoperational test, and all the subsequent periodic tests. The report shall contain an analysis and interpretation of the leakage rate test data for the Type A test results to the extent necessary to demonstrate the acceptability of the containment's leakage rate in meeting acceptance criteria.

2. For each periodic test, leakage test results from Type A, B, and C tests shall be included in the summary report. The summary report shall contain an analysis and interpretation of the Type A test results and a summary analysis of periodic Type B and Type C tests that were performed since the last type A test. Leakage test results from type A, B, and C tests that failed to meet the acceptance criteria of III.A.5(b), III.B.3, and III.C.3, respectively, shall be included in a separate accompanying summary report that includes an analysis and interpretation of the test data, the least squares fit analysis of the test data, the instrumentation error analysis, and the structural conditions of the containment or components, if any, which contributed to the failure in meeting the acceptance criteria. Results and analyses of the supplemental verification test employed to demonstrate the validity of the leakage rate test measurements shall also be included.

OPTION B—PERFORMANCE-BASED REQUIREMENTS

Table of Contents

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) is that primary reactor containments meet 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]

II. DEFINITIONS

Performance criteria means the performance standards against which test results are to be compared for establishing the acceptability of the containment system as a leakage-limiting boundary.

Containment system means the principal barrier, after the reactor coolant pressure boundary, to prevent the release of quantities of radioactive material that would have a significant radiological effect on the health of the public.

Overall integrated leakage rate means the total leakage rate through all tested leakage paths, including containment welds, valves, fittings, and components that penetrate the containment system.

La (percent/24 hours) means the maximum allowable leakage rate at pressure Pa as specified in the Technical Specifications.

Pa (p.s.i.g) means the calculated peak containment internal pressure related to the design basis loss-of-coolant accident as specified in the Technical Specifications.

III. PERFORMANCE-BASED LEAKAGE-TEST REQUIREMENTS

A. Type A Test

Type A tests to measure the containment system overall integrated leakage rate must be conducted under conditions representing design basis loss-of-coolant accident containment peak pressure. A Type A test must be conducted (1) after the containment system has been completed and is ready for operation and (2) at a periodic interval based on the historical performance of the overall containment system as a barrier to fission product releases to reduce the risk from reactor accidents. A general visual inspection of the accessible interior and exterior surfaces of the containment system for structural deterioration which may affect the containment leak-tight integrity must be

[3] Specific guidance concerning a performance-based leakage-test program, acceptable leakage-rate test methods, procedures, and analyses that may be used to implement these requirements and criteria are provided in Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program.''

conducted prior to each test, and at a periodic interval between tests based on the performance of the containment system. The leakage rate must not exceed the allowable leakage rate (La) with margin, as specified in the Technical Specifications. The test results must be compared with previous results to examine the performance history of the overall containment system to limit leakage.

B. Type B and C Tests

Type B pneumatic tests to detect and measure local leakage rates across pressure retaining, leakage-limiting boundaries, and Type C pneumatic tests to measure containment isolation valve leakage rates, must be conducted (1) prior to initial criticality, and (2) periodically thereafter at intervals based on the safety significance and historical performance of each boundary and isolation valve to ensure the integrity of the overall containment system as a barrier to fission product release to reduce the risk from reactor accidents. The performance-based testing program must contain a performance criterion for Type B and C tests, consideration of leakage-rate limits and factors that are indicative of or affect performance, when establishing test intervals, evaluations of performance of containment system components, and comparison to previous test results to examine the performance history of the overall containment system to limit leakage. The tests must demonstrate that the sum of the leakage rates at accident pressure of Type B tests, and pathway leakage rates from Type C tests, is less than the performance criterion (La) with margin, as specified in the Technical Specification.

IV. RECORDKEEPING

The results of the preoperational and periodic Type A, B, and C tests must be documented to show that performance criteria for leakage have been met. The comparison to previous results of the performance of the overall containment system and of individual components within it must be documented to show that the test intervals established for the containment system and components within it are adequate. These records must be available for inspection at plant sites.

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), and for a Licensee Event Report under §50.73 (a)(2)(ii).

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 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, 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 a combined license under part 52 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 a combined license under part 52 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.

4. The detailed licensee programs for conducting testing under Option B must be available at the plant site for NRC inspection.

[38 FR 4386, Feb. 14, 1973; 38 FR 5997, Mar. 6, 1973, as amended at 41 FR 16447, Apr. 19, 1976; 45 FR 62789, Sept. 22, 1980; 51 FR 40311, Nov. 6, 1986; 53 FR 45891, Nov. 15, 1988; 57 FR 61786, Dec. 29, 1992; 59 FR 50689, Oct. 5, 1994; 60 FR 13616, Mar. 14, 1995; 60 FR 49504, Sept. 26, 1995; 72 FR 49508, Aug. 28, 2007; 73 FR 5723, Jan. 31, 2008; 84 FR 63568, Nov. 18, 2019; 84 FR 65644, Nov. 29, 2019]

APPENDIX K TO PART 50—ECCS EVALUATION MODELS

I. Required and Acceptable Features of Evaluation Models.

II. Required Documentation.

I. REQUIRED AND ACCEPTABLE FEATURES OF THE EVALUATION MODELS

A. *Sources of heat during the LOCA.* For the heat sources listed in paragraphs I.A.1 to 4 of this appendix it must be assumed that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error), with the maximum peaking factor allowed by the technical specifications. An assumed power level lower than the level specified in this paragraph (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error. A range of power distribution shapes and peaking factors representing power distributions that may occur over the core lifetime must be studied. The selected combination of power distribution shape and peaking factor should be the one that results in the most severe calculated consequences for the spectrum of postulated breaks and single failures that are analyzed.

1. *The Initial Stored Energy in the Fuel.* The steady-state temperature distribution and stored energy in the fuel before the hypothetical accident shall be calculated for the burn-up that yields the highest calculated cladding temperature (or, optionally, the highest calculated stored energy.) To accomplish this, the thermal conductivity of the UO_2 shall be evaluated as a function of burn-up and temperature, taking into consideration differences in initial density, and the thermal conductance of the gap between the UO_2 and the cladding shall be evaluated as a function of the burn-up, taking into consideration fuel densification and expansion, the composition and pressure of the gases within the fuel rod, the initial cold gap dimension with its tolerances, and cladding creep.

2. *Fission Heat.* Fission heat shall be calculated using reactivity and reactor kinetics. Shutdown reactivities resulting from temperatures and voids shall be given their minimum plausible values, including allowance for uncertainties, for the range of power distribution shapes and peaking factors indicated to be studied above. Rod trip and insertion may be assumed if they are calculated to occur.

3. *Decay of Actinides.* The heat from the radioactive decay of actinides, including neptunium and plutonium generated during operation, as well as isotopes of uranium, shall be calculated in accordance with fuel cycle calculations and known radioactive properties. The actinide decay heat chosen shall be that appropriate for the time in the fuel cycle that yields the highest calculated fuel temperature during the LOCA.

4. *Fission Product Decay.* The heat generation rates from radioactive decay of fission products shall be assumed to be equal to 1.2 times the values for infinite operating time in the ANS Standard (Proposed American Nuclear Society Standards—''Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors.'' Approved by Subcommittee ANS–5, ANS Standards Committee, October 1971). This standard has been approved for incorporation by reference by the Director of the Federal Register. A copy of the standard is available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852–2738. The fraction of the locally generated gamma energy that is deposited in the fuel (including the cladding) may be different from 1.0; the value used shall be justified by a suitable calculation.

5. *Metal—Water Reaction Rate.* The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation (Baker, L., Just, L.C., ''Studies of Metal Water Reactions at High Temperatures, III. Experimental and Theoretical Studies of the Zirconium-Water Reaction,'' ANL–6548, page 7, May 1962). This publication has been approved for incorporation by reference by the Director of the Federal Register. A copy of the publication is available for inspection at the NRC Library, 11545 Rockville Pike, Two White Flint North, Rockville, Maryland 20852–2738. The reaction shall be assumed not to be steam limited. For rods whose cladding is calculated to rupture during the LOCA, the inside of the cladding shall be assumed to react after the rupture. The calculation of the reaction rate on the inside of the cladding shall also follow the Baker-Just equation, starting at the time when the cladding is calculated to rupture, and extending around the cladding inner circumference and axially no less that 1.5 inches each way from the location of the rupture, with the reaction assumed not to be steam limited.

6. *Reactor Internals Heat Transfer.* Heat transfer from piping, vessel walls, and non-fuel internal hardware shall be taken into account.

7. *Pressurized Water Reactor Primary-to-Secondary Heat Transfer.* Heat transferred between primary and secondary systems through heat exchangers (steam generators) shall be taken into account. (Not applicable to Boiling Water Reactors.)

B. Swelling and Rupture of the Cladding and Fuel Rod Thermal Parameters

Each evaluation model shall include a provision for predicting cladding swelling and rupture from consideration of the axial temperature distribution of the cladding and from the difference in pressure between the inside and outside of the cladding, both as functions of time. To be acceptable the swelling and rupture calculations shall be based on applicable data in such a way that the degree of swelling and incidence of rupture are not underestimated. The degree of swelling and rupture shall be taken into account in calculations of gap conductance, cladding oxidation and embrittlement, and hydrogen generation.

The calculations of fuel and cladding temperatures as a function of time shall use values for gap conductance and other thermal parameters as functions of temperature and other applicable time-dependent variables. The gap conductance shall be varied in accordance with changes in gap dimensions and any other applicable variables.

C. Blowdown Phenomena

1. *Break Characteristics and Flow.* a. In analyses of hypothetical loss-of-coolant accidents, a spectrum of possible pipe breaks shall be considered. This spectrum shall include instantaneous double-ended breaks ranging in cross-sectional area up to and including that of the largest pipe in the primary coolant system. The analysis shall also include the effects of longitudinal splits in the largest pipes, with the split area equal to the cross-sectional area of the pipe.

b. *Discharge Model.* For all times after the discharging fluid has been calculated to be two-phase in composition, the discharge rate shall be calculated by use of the Moody model (F.J. Moody, ''Maximum Flow Rate of a Single Component, Two-Phase Mixture,'' Journal of Heat Transfer, Trans American Society of Mechanical Engineers, 87, No. 1, February, 1965). This publication has been approved for incorporation by reference by the Director of the Federal Register. A copy of this publication is available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852–2738. The calculation shall be conducted with at least three values of a discharge coefficient applied to the postulated break area, these values spanning the range from 0.6 to 1.0. If the results indicate that the maximum clad temperature for the hypothetical accident is to be found at an even lower value of the discharge coefficient, the range of discharge coefficients shall be extended until the maximum clad temperatures calculated by this variation has been achieved.

c. *End of Blowdown.* (Applies Only to Pressurized Water Reactors.) For postulated cold leg breaks, all emergency cooling water injected into the inlet lines or the reactor vessel during the bypass period shall in the calculations be subtracted from the reactor vessel calculated inventory. This may be executed in the calculation during the bypass period, or as an alternative the amount of emergency core cooling water calculated to be injected during the bypass period may be subtracted later in the calculation from the water remaining in the inlet lines, downcomer, and reactor vessel lower plenum after the bypass period. This bypassing shall end in the calculation at a time designated as the ''end of bypass,'' after which the expulsion or entrainment mechanisms responsible for the bypassing are calculated not to be effective. The end-of-bypass definition used in the calculation shall be justified by a suitable combination of analysis and experimental data. Acceptable methods for defining ''end of bypass'' include, but are not limited to, the following: (1) Prediction of the blowdown calculation of downward flow in the downcomer for the remainder of the blowdown period; (2) Prediction of a threshold for droplet entrainment in the upward velocity, using local fluid conditions and a conservative critical Weber number.

d. *Noding Near the Break and the ECCS Injection Points.* The noding in the vicinity of and including the broken or split sections of pipe and the points of ECCS injection shall be chosen to permit a reliable analysis of the thermodynamic history in these regions during blowdown.

2. *Frictional Pressure Drops.* The frictional losses in pipes and other components including the reactor core shall be calculated using models that include realistic variation of friction factor with Reynolds number, and realistic two-phase friction multipliers that have been adequately verified by comparison with experimental data, or models that prove at least equally conservative with respect to maximum clad temperature calculated during the hypothetical accident. The modified Baroczy correlation (Baroczy, C. J., ''A Systematic Correlation for Two-Phase Pressure Drop,'' *Chem. Enging. Prog. Symp. Series,* No. 64, Vol. 62, 1965) or a combination of the Thom correlation (Thom, J.R.S., ''Prediction of Pressure Drop During Forced Circulation Boiling of Water,'' *Int. J. of Heat & Mass Transfer,* 7, 709–724, 1964) for pressures equal to or greater than 250 psia and the Martinelli-Nelson correlation (Martinelli, R. C. Nelson, D.B., ''Prediction of Pressure Drop During Forced Circulation Boiling of Water,'' *Transactions of ASME,* 695–702, 1948) for pressures lower than 250 psia is acceptable as a basis for calculating realistic two-phase friction multipliers.

3. *Momentum Equation.* The following effects shall be taken into account in the conservation of momentum equation: (1) temporal change of momentum, (2) momentum convection, (3) area change momentum flux, (4) momentum change due to compressibility, (5) pressure loss resulting from wall friction, (6) pressure loss resulting from area change, and (7) gravitational acceleration. Any omission of one or more of these terms under stated circumstances shall be justified by comparative analyses or by experimental data.

4. *Critical Heat Flux.* a. Correlations developed from appropriate steady-state and transient-state experimental data are acceptable for use in predicting the critical heat flux (CHF) during LOCA transients. The computer programs in which these correlations are used shall contain suitable checks to assure that the physical parameters are within

the range of parameters specified for use of the correlations by their respective authors.

b. Steady-state CHF correlations acceptable for use in LOCA transients include, but are not limited to, the following:

(1) *W 3.* L. S. Tong, ''Prediction of Departure from Nucleate Boiling for an Axially Non-uniform Heat Flux Distribution,'' *Journal of Nuclear Energy*, Vol. 21, 241–248, 1967.

(2) *B&W–2.* J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, L. J. Stanek, ''Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water,'' *Two-Phase Flow and Heat Transfer in Rod Bundles,* ASME, New York, 1969.

(3) *Hench-Levy.* J. M. Healzer, J. E. Hench, E. Janssen, S. Levy, ''Design Basis for Critical Heat Flux Condition in Boiling Water Reactors,'' APED–5186, GE Company Private report, July 1966.

(4) *Macbeth.* R. V. Macbeth, ''An Appraisal of Forced Convection Burnout Data,'' *Proceedings of the Institute of Mechanical Engineers*, 1965–1966.

(5) *Barnett.* P. G. Barnett, ''A Correlation of Burnout Data for Uniformly Heated Annuli and Its Uses for Predicting Burnout in Uniformly Heated Rod Bundles,'' AEEW-R 463, 1966.

(6) *Hughes.* E. D. Hughes, ''A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia,'' IN–1412, Idaho Nuclear Corporation, July 1970.

c. Correlations of appropriate transient CHF data may be accepted for use in LOCA transient analyses if comparisons between the data and the correlations are provided to demonstrate that the correlations predict values of CHF which allow for uncertainty in the experimental data throughout the range of parameters for which the correlations are to be used. Where appropriate, the comparisons shall use statistical uncertainty analysis of the data to demonstrate the conservatism of the transient correlation.

d. Transient CHF correlations acceptable for use in LOCA transients include, but are not limited to, the following:

(1) *GE transient CHF.* B. C. Slifer, J. E. Hench, ''Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors,'' NEDO–10329, General Electric Company, Equation C–32, April 1971.

e. After CHF is first predicted at an axial fuel rod location during blowdown, the calculation shall not use nucleate boiling heat transfer correlations at that location subsequently during the blowdown even if the calculated local fluid and surface conditions would apparently justify the reestablishment of nucleate boiling. Heat transfer assumptions characteristic of return to nucleate boiling (rewetting) shall be permitted when justified by the calculated local fluid and surface conditions during the reflood portion of a LOCA.

5. *Post-CHF Heat Transfer Correlations.* a. Correlations of heat transfer from the fuel cladding to the surrounding fluid in the post-CHF regimes of transition and film boiling shall be compared to applicable steady-state and transient-state data using statistical correlation and uncertainty analyses. Such comparison shall demonstrate that the correlations predict values of heat transfer coefficient equal to or less than the mean value of the applicable experimental heat transfer data throughout the range of parameters for which the correlations are to be used. The comparisons shall quantify the relation of the correlations to the statistical uncertainty of the applicable data.

b. The Groeneveld flow film boiling correlation (equation 5.7 of D.C. Groeneveld, ''An Investigation of Heat Transfer in the Liquid Deficient Regime,'' AECL–3281, revised December 1969) and the Westinghouse correlation of steady-state transition boiling (''Proprietary Redirect/Rebuttal Testimony of Westinghouse Electric Corporation,'' USNRC Docket RM–50–1, page 25–1, October 26, 1972) are acceptable for use in the post-CHF boiling regimes. In addition, the transition boiling correlation of McDonough, Milich, and King (J.B. McDonough, W. Milich, E.C. King, ''An Experimental Study of Partial Film Boiling Region with Water at Elevated Pressures in a Round Vertical Tube,'' Chemical Engineering Progress Symposium Series, Vol. 57, No. 32, pages 197–208, (1961) is suitable for use between nucleate and film boiling. Use of all these correlations is restricted as follows:

(1) The Groeneveld correlation shall not be used in the region near its low-pressure singularity,

(2) The first term (nucleate) of the Westinghouse correlation and the entire McDonough, Milich, and King correlation shall not be used during the blowdown after the temperature difference between the clad and the saturated fluid first exceeds 300 °F,

(3) Transition boiling heat transfer shall not be reapplied for the remainder of the LOCA blowdown, even if the clad superheat returns below 300 °F, except for the reflood portion of the LOCA when justified by the calculated local fluid and surface conditions.

c. Evaluation models approved after October 17, 1988, which make use of the Dougall-Rohsenow flow film boiling correlation (R.S. Dougall and W.M. Rohsenow, ''Film Boiling on the Inside of Vertical Tubes with Upward Flow of Fluid at Low Qualities,'' MIT Report Number 9079 26, Cambridge, Massachusetts, September 1963) may not use this correlation under conditions where nonconservative predictions of heat transfer result. Evaluation models that make use of the Dougall-Rohsenow correlation and were approved prior to October 17, 1988, continue to be acceptable until a change is made to, or an error is corrected in, the evaluation model that results in a significant reduction in the overall conservatism in the evaluation model. At that time continued use of the Dougall-Rohsenow correlation under conditions where nonconservative predictions of heat transfer result will no longer be acceptable. For this purpose, a significant reduction in the overall conservatism in the evaluation model would be a reduction in the calculated peak fuel cladding temperature of at least 50 °F from that which would have been calculated on October 17, 1988, due either to individual changes or error corrections or the net effect of an accumulation of changes or error corrections.

6. *Pump Modeling*. The characteristics of rotating primary system pumps (axial flow, turbine, or centrifugal) shall be derived from a dynamic model that includes momentum transfer between the fluid and the rotating member, with variable pump speed as a function of time. The pump model resistance used for analysis should be justified. The pump model for the two-phase region shall be verified by applicable two-phase pump performance data. For BWR's after saturation is calculated at the pump suction, the pump head may be assumed to vary linearly with quality, going to zero for one percent quality at the pump suction, so long as the analysis shows that core flow stops before the quality at pump suction reaches one percent.

7. *Core Flow Distribution During Blowdown*. (Applies only to pressurized water reactors.)

a. The flow rate through the hot region of the core during blowdown shall be calculated as a function of time. For the purpose of these calculations the hot region chosen shall not be greater than the size of one fuel assembly. Calculations of average flow and flow in the hot region shall take into account cross flow between regions and any flow blockage calculated to occur during blowdown as a result of cladding swelling or rupture. The calculated flow shall be smoothed to eliminate any calculated rapid oscillations (period less than 0.1 seconds).

b. A method shall be specified for determining the enthalpy to be used as input data to the hot channel heatup analysis from quantities calculated in the blowdown analysis, consistent with the flow distribution calculations.

D. Post-Blowdown Phenomena; Heat Removal by the ECCS

1. *Single Failure Criterion*. An analysis of possible failure modes of ECCS equipment and of their effects on ECCS performance must be made. In carrying out the accident evaluation the combination of ECCS subsystems assumed to be operative shall be those available after the most damaging single failure of ECCS equipment has taken place.

2. *Containment Pressure*. The containment pressure used for evaluating cooling effectiveness during reflood and spray cooling shall not exceed a pressure calculated conservatively for this purpose. The calculation shall include the effects of operation of all installed pressure-reducing systems and processes.

3. *Calculation of Reflood Rate for Pressurized Water Reactors*. The refilling of the reactor vessel and the time and rate of reflooding of the core shall be calculated by an acceptable model that takes into consideration the thermal and hydraulic characteristics of the core and of the reactor system. The primary system coolant pumps shall be assumed to have locked impellers if this assumption leads to the maximum calculated cladding temperature; otherwise the pump rotor shall be assumed to be running free. The ratio of the total fluid flow at the core exit plane to the total liquid flow at the core inlet plane (carryover fraction) shall be used to determine the core exit flow and shall be determined in accordance with applicable experimental data (for example, ''PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report,'' Westinghouse Report WCAP–7665, April 1971; ''PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report,'' Westinghouse Report WCAP–7435, January 1970; ''PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Group II Test Report,'' Westinghouse Report WCAP–7544, September 1970; ''PWR FLECHT Final Report Supplement,'' Westinghouse Report WCAP–7931, October 1972).

The effects on reflooding rate of the compressed gas in the accumulator which is discharged following accumulator water discharge shall also be taken into account.

4. *Steam Interaction with Emergency Core Cooling Water in Pressurized Water Reactors*. The thermal-hydraulic interaction between steam and all emergency core cooling water shall be taken into account in calculating the core reflooding rate. During refill and reflood, the calculated steam flow in unbroken reactor coolant pipes shall be taken to be zero during the time that accumulators are discharging water into those pipes unless experimental evidence is available regarding the realistic thermal-hydraulic interaction between the steam and the liquid. In this case, the experimental data may be used to support an alternate assumption.

5. *Refill and Reflood Heat Transfer for Pressurized Water Reactors*. a. For reflood rates of one inch per second or higher, reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores including FLECHT results (''PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report,'' Westinghouse Report WCAP–7665, April 1971). The use of a correlation derived from FLECHT data shall be demonstrated to be conservative for the

transient to which it is applied; presently available FLECHT heat transfer correlations (''PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report,'' Westinghouse Report WCAP–7544, September 1970; ''PWR FLECHT Final Report Supplement,'' Westinghouse Report WCAP–7931, October 1972) are not acceptable. Westinghouse Report WCAP–7665 has been approved for incorporation by reference by the Director of the Federal Register. A copy of this report is available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852–2738. New correlations or modifications to the FLECHT heat transfer correlations are acceptable only after they are demonstrated to be conservative, by comparison with FLECHT data, for a range of parameters consistent with the transient to which they are applied.

b. During refill and during reflood when reflood rates are less than one inch per second, heat transfer calculations shall be based on the assumption that cooling is only by steam, and shall take into account any flow blockage calculated to occur as a result of cladding swelling or rupture as such blockage might affect both local steam flow and heat transfer.

6. *Convective Heat Transfer Coefficients for Boiling Water Reactor Fuel Rods Under Spray Cooling*. Following the blowdown period, convective heat transfer shall be calculated using coefficients based on appropriate experimental data. For reactors with jet pumps and having fuel rods in a 7×7 fuel assembly array, the following convective coefficients are acceptable:

a. During the period following lower plenum flashing but prior to the core spray reaching rated flow, a convective heat transfer coefficient of zero shall be applied to all fuel rods.

b. During the period after core spray reaches rated flow but prior to reflooding, convective heat transfer coefficients of 3.0, 3.5, 1.5, and 1.5 Btu-hr^{-1}-ft^{-2} °F^{-1} shall be applied to the fuel rods in the outer corners, outer row, next to outer row, and to those remaining in the interior, respectively, of the assembly.

c. After the two-phase reflooding fluid reaches the level under consideration, a convective heat transfer coefficient of 25 Btu-hr^{-1}-ft^{-2} °F^{-1} shall be applied to all fuel rods.

7. *The Boiling Water Reactor Channel Box Under Spray Cooling*. Following the blowdown period, heat transfer from, and wetting of, the channel box shall be based on appropriate experimental data. For reactors with jet pumps and fuel rods in a 7×7 fuel assembly array, the following heat transfer coefficients and wetting time correlation are acceptable.

a. During the period after lower plenum flashing, but prior to core spray reaching rated flow, a convective coefficient of zero shall be applied to the fuel assembly channel box.

b. During the period after core spray reaches rated flow, but prior to wetting of the channel, a convective heat transfer coefficient of 5 Btu-hr^{-1}-ft^{-2}- °F^{-1} shall be applied to both sides of the channel box.

c. Wetting of the channel box shall be assumed to occur 60 seconds after the time determined using the correlation based on the Yamanouchi analysis (''Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors,'' General Electric Company Report NEDO–10329, April 1971). This report was approved for incorporation by reference by the Director of the Federal Register. A copy of the report is available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852–2738.

II. Required Documentation

1. a. A description of each evaluation model shall be furnished. The description shall be sufficiently complete to permit technical review of the analytical approach including the equations used, their approximations in difference form, the assumptions made, and the values of all parameters or the procedure for their selection, as for example, in accordance with a specified physical law or empirical correlation.

b. A complete listing of each computer program, in the same form as used in the evaluation model, must be furnished to the Nuclear Regulatory Commission upon request.

2. For each computer program, solution convergence shall be demonstrated by studies of system modeling or noding and calculational time steps.

3. Appropriate sensitivity studies shall be performed for each evaluation model, to evaluate the effect on the calculated results of variations in noding, phenomena assumed in the calculation to predominate, including pump operation or locking, and values of parameters over their applicable ranges. For items to which results are shown to be sensitive, the choices made shall be justified.

4. To the extent practicable, predictions of the evaluation model, or portions thereof, shall be compared with applicable experimental information.

5. General Standards for Acceptability—Elements of evaluation models reviewed will include technical adequacy of the calculational methods, including: For models covered by §50.46(a)(1)(ii), compliance with required features of section I of this appendix K; and, for models covered by §50.46(a)(1)(i), assurance of a high level of

probability that the performance criteria of §50.46(b) would not be exceeded.

[39 FR 1003, Jan. 4, 1974, as amended at 51 FR 40311, Nov. 6, 1986; 53 FR 36005, Sept. 16, 1988; 57 FR 61786, Dec. 29, 1992; 59 FR 50689, Oct. 5, 1994; 60 FR 24552, May 9, 1995; 65 FR 34921, June 1, 2000]

APPENDIXES L–M TO PART 50 [RESERVED]

APPENDIX N TO PART 50—STANDARDIZATION OF NUCLEAR POWER PLANT DESIGNS: PERMITS TO CONSTRUCT AND LICENSES TO OPERATE NUCLEAR POWER REACTORS OF IDENTICAL DESIGN AT MULTIPLE SITES

Section 101 of the Atomic Energy Act of 1954, as amended, and §50.10 of this part require a Commission license to transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, use, import or export any production or utilization facility. The regulations in this part require the issuance of a construction permit by the Commission before commencement of construction of a production or utilization facility, except as provided in §50.10(e), and the issuance of an operating license before operation of the facility.

The Commission's regulations in part 2 of this chapter specifically provide for the holding of hearings on particular issues separately from other issues involved in hearings in licensing proceedings (§2.761a, appendix A, section I(c)), and for the consolidation of adjudicatory proceedings and of the presentations of parties in adjudicatory proceedings such as licensing proceedings (§§2.715a, 2.716).

This appendix sets out the particular requirements and provisions applicable to situations in which applications are filed by one or more applicants for licenses to construct and operate nuclear power reactors of essentially the same design to be located at different sites.[1]

1. Except as otherwise specified in this appendix or as the context otherwise indicates, the provisions of this part applicable to construction permits and operating licenses, including the requirement in §50.58 for review of the application by the Advisory Committee on Reactor Safeguards and the holding of public hearings, apply to construction permits and operating licenses subject to this appendix N.

2. Applications for construction permits submitted pursuant to this appendix must include the information required by §§50.33, 50.34(a) and 50.34a(a) and (b) and be submitted as specified in §50.4. The applicant shall also submit the information required by §51.50 of this chapter.

3. Applications for operating licenses submitted pursuant to this appendix N shall include the information required by §§50.33, 50.34(b) and (c), and 50.34a(c). The applicant shall also submit the information required by §51.53 of this chapter. For the technical information required by §§50.34(b)(2) through (5) and 50.34a(c), reference may be made to a single final safety analysis of the design.

[40 FR 2977, Jan. 17, 1975, as amended at 49 FR 9405, Mar. 12, 1984; 51 FR 40311, Nov. 6, 1986; 70 FR 61888, Oct. 27, 2005]

APPENDIXES O–P TO PART 50 [RESERVED]

APPENDIX Q TO PART 50—PRE-APPLICATION EARLY REVIEW OF SITE SUITABILITY ISSUES

This appendix sets out procedures for the filing, Staff review, and referral to the Advisory Committee on Reactor Safeguards of requests for early review of one or more site suitability issues relating to the construction and operation of certain utilization facilities separately from and prior to the submittal of applications for construction permits for the facilities. The appendix also sets out procedures for the preparation and issuance of Staff Site Reports and for their incorporation by reference in applications for the construction and operation of certain utilization facilities. The utilization facilities are those which are subject to §51.20(b) of this chapter and are of the type specified in §50.21(b) (2) or (3) or §50.22 or are testing facilities. This appendix does not apply to proceedings conducted pursuant to subpart F of part 2 of this chapter.

1. Any person may submit information regarding one or more site suitability issues to the Commission's Staff for its review separately from and prior to an application for a construction permit for a facility. Such a submittal shall be accompanied by any fee required by part 170 of this chapter and shall consist of the portion of the information required of applicants for construction permits by §§50.33(a)–(c) and (e), and, insofar as it relates to the issue(s) of site suitability for which early review is sought, by §§50.34(a)(1) and 50.30(f), except that information with respect to operation of the facility at the projected initial power level need not be supplied.

2. The submittal for early review of site suitability issue(s) must be made in the same manner and in the same number of copies as

[1] If the design for the power reactor(s) proposed in a particular application is not identical to the others, that application may not be processed under this appendix and subpart D of part 2 of this chapter.

provided in §§50.4 and 50.30 for license applications. The submittal must include sufficient information concerning a range of postulated facility design and operation parameters to enable the Staff to perform the requested review of site suitability issues. The submittal must contain suggested conclusions on the issues of site suitability submitted for review and must be accompanied by a statement of the bases or the reasons for those conclusions. The submittal must also list, to the extent possible, any long-range objectives for ultimate development of the site, state whether any site selection process was used in preparing the submittal, describe any site selection process used, and explain what consideration, if any, was given to alternative sites.

3. The Staff shall publish a notice of docketing of the submittal in the FEDERAL REGISTER, and shall send a copy of the notice of docketing to the Governor or other appropriate official of the State in which the site is located. This notice shall identify the location of the site, briefly describe the site suitability issue(s) under review, and invite comments from Federal, State, and local agencies and interested persons within 120 days of publication or such other time as may be specified, for consideration by the staff in connection with the initiation or outcome of the review and, if appropriate by the ACRS, in connection with the outcome of their review. The person requesting review shall serve a copy of the submittal on the Governor or other appropriate official of the State in which the site is located, and on the chief executive of the municipality in which the site is located or, if the site is not located in a municipality, on the chief executive of the county. The portion of the submittal containing information required of applicants for construction permits by §§50.33(a)–(c) and (e) and 50.34(a)(1) will be referred to the Advisory Committee on Reactor Safeguards (ACRS) for a review and report. There will be no referral to the ACRS unless early review of the site safety issues under §50.34(a)(1) is requested.

4. Upon completion of review by the NRC staff and, if appropriate by the ACRS, of a submittal under this appendix, the NRC staff shall prepare a Staff Site Report which shall identify the location of the site, state the site suitability issues reviewed, explain the nature and scope of the review, state the conclusions of the staff regarding the issues reviewed and state the reasons for those conclusions. Upon issuance of an NRC Staff Site Report, the NRC staff shall publish a notice of the availability of the report in the FEDERAL REGISTER and shall make the report available at the NRC Web site, *http://www.nrc.gov*. The NRC staff shall also send a copy of the report to the Governor or other appropriate official of the State in which the site is located, and to the chief executive of the municipality in which the site is located or, if the site is not located in a municipality, to the chief executive of the county.

5. Any Staff Site Report prepared and issued in accordance with this appendix may be incorporated by reference, as appropriate, in 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 of this chapter or is a testing facility. The conclusions of the Staff Site Report will be reexamined by the staff where five years or more have elapsed between the issuance of the Staff Site Report and its incorporation by reference in a construction permit application.

6. Issuance of a Staff Site Report shall not constitute a commitment to issue a permit or license, to permit on-site work under §50.10(e), or in any way affect the authority of the Commission, Atomic Safety and Licensing Appeal Panel, Atomic Safety and Licensing Board, and other presiding officers in any proceeding under subpart F and/or G of part 2 of this chapter.

7. The staff will not conduct more than one review of site suitability issues with regard to a particular site prior to the full construction permit review required by subpart A of part 51 of this chapter. The staff may decline to prepare and issue a Staff Site Report in response to a submittal under this appendix where it appears that, (a) in cases where no review of the relative merits of the submitted site and alternative sites under subpart A of part 51 of this chapter is requested, there is a reasonable likelihood that further Staff review would identify one or more preferable alternative sites and the Staff review of one or more site suitability issues would lead to an irreversible and irretrievable commitment of resources prior to the submittal of the analysis of alternative sites in the Environmental Report that would prejudice the later review and decision on alternative sites under subpart F and/or G of part 2 and subpart A of part 51 of this chapter; or (b) in cases where, in the judgment of the Staff, early review of any site suitability issue or issues would not be in the public interest, considering (1) the degree of likelihood that any early findings on those issues would retain their validity in later reviews, (2) the objections, if any, of cognizant state or local government agencies to the conduct of an early review on those issues, and (3) the possible effect on the public interest of having an early, if not necessarily conclusive, resolution of those issues.

[42 FR 22887, May 5, 1977, as amended at 49 FR 9405, Mar. 12, 1984; 51 FR 40311, Nov. 6, 1986; 53 FR 43420, Oct. 27, 1988; 64 FR 48952, Sept. 9, 1999]

APPENDIX R TO PART 50—FIRE PROTECTION PROGRAM FOR NUCLEAR POWER FACILITIES OPERATING PRIOR TO JANUARY 1, 1979

I. INTRODUCTION AND SCOPE

This appendix applies to licensed nuclear power electric generating stations that were operating prior to January 1, 1979, except to the extent set forth in §50.48(b) of this part. With respect to certain generic issues for such facilities it sets forth fire protection features required to satisfy Criterion 3 of appendix A to this part.

Criterion 3 of appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."

When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boiloff.

The phrases "important to safety," or "safety-related," will be used throughout this appendix R as applying to all safety functions. The phrase "safe shutdown" will be used throughout this appendix as applying to both hot and cold shutdown functions.

Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under postfire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents. Three levels of fire damage limits are established according to the safety functions of the structure, system, or component:

Safety function	Fire damage limits
Hot Shutdown	One train of equipment necessary to achieve hot shutdown from either the control room or emergency control station(s) must be maintained free of fire damage by a single fire, including an exposure fire. [1]
Cold Shutdown	Both trains of equipment necessary to achieve cold shutdown may be damaged by a single fire, including an exposure fire, but damage must be limited so that at least one train can be repaired or made operable within 72 hours using onsite capability.
Design Basis Accidents.	Both trains of equipment necessary for mitigation of consequences following design basis accidents may be damaged by a single exposure fire.

[1] *Exposure Fire.* An exposure fire is a fire in a given area that involves either in situ or transient combustibles and is external to any structures, systems, or components located in or adjacent to that same area. The effects of such fire (e.g., smoke, heat, or ignition) can adversely affect those structures, systems, or components important to safety. Thus, a fire involving one train of safe shutdown equipment may constitute an exposure fire for the redundant train located in the same area, and a fire involving combustibles other than either redundant train may constitute an exposure fire to both redundant trains located in the same area.

The most stringent fire damage limit shall apply for those systems that fall into more than one category. Redundant systems used to mitigate the consequences of other design basis accidents but not necessary for safe shutdown may be lost to a single exposure fire. However, protection shall be provided so that a fire within only one such system will not damage the redundant system.

II. GENERAL REQUIREMENTS

A. *Fire protection program.* A fire protection program shall be established at each nuclear power plant. The program shall establish the fire protection policy for the protection of structures, systems, and components important to safety at each plant and the procedures, equipment, and personnel required to implement the program at the plant site.

The fire protection program shall be under the direction of an individual who has been delegated authority commensurate with the responsibilities of the position and who has available staff personnel knowledgeable in both fire protection and nuclear safety.

The fire protection program shall extend the concept of defense-in-depth to fire protection in fire areas important to safety, with the following objectives:

- To prevent fires from starting;
- To detect rapidly, control, and extinguish promptly those fires that do occur;
- To provide protection for structures, systems, and components important to safety so that a fire that is not promptly extinguished by the fire suppression activities will not prevent the safe shutdown of the plant.

B. *Fire hazards analysis.* A fire hazards analysis shall be performed by qualified fire protection and reactor systems engineers to (1) consider potential in situ and transient fire hazards; (2) determine the consequences of fire in any location in the plant on the ability to safely shut down the reactor or on the ability to minimize and control the release of radioactivity to the environment; and (3) specify measures for fire prevention, fire detection, fire suppression, and fire containment and alternative shutdown capability as required for each fire area containing structures, systems, and components important to safety in accordance with NRC guidelines and regulations.

C. *Fire prevention features.* Fire protection features shall meet the following general requirements for all fire areas that contain or present a fire hazard to structures, systems, or components important to safety.

1. In situ fire hazards shall be identified and suitable protection provided.

2. Transient fire hazards associated with normal operation, maintenance, repair, or modification activities shall be identified and eliminated where possible. Those transient fire hazards that can not be eliminated shall be controlled and suitable protection provided.

3. Fire detection systems, portable extinguishers, and standpipe and hose stations shall be installed.

4. Fire barriers or automatic suppression systems or both shall be installed as necessary to protect redundant systems or components necessary for safe shutdown.

5. A site fire brigade shall be established, trained, and equipped and shall be on site at all times.

6. Fire detection and suppression systems shall be designed, installed, maintained, and tested by personnel properly qualified by experience and training in fire protection systems.

7. Surveillance procedures shall be established to ensure that fire barriers are in place and that fire suppression systems and components are operable.

D. *Alternative or dedicated shutdown capability.* In areas where the fire protection features cannot ensure safe shutdown capability in the event of a fire in that area, alternative or dedicated safe shutdown capability shall be provided.

III. SPECIFIC REQUIREMENTS

A. *Water supplies for fire suppression systems.* Two separate water supplies shall be provided to furnish necessary water volume and pressure to the fire main loop.

Each supply shall consist of a storage tank, pump, piping, and appropriate isolation and control valves. Two separate redundant suctions in one or more intake structures from a large body of water (river, lake, etc.) will satisfy the requirement for two separated water storage tanks. These supplies shall be separated so that a failure of one supply will not result in a failure of the other supply.

Each supply of the fire water distribution system shall be capable of providing for a period of 2 hours the maximum expected water demands as determined by the fire hazards analysis for safety-related areas or other areas that present a fire exposure hazard to safety-related areas.

When storage tanks are used for combined service-water/fire-water uses the minimum volume for fire uses shall be ensured by means of dedicated tanks or by some physical means such as a vertical standpipe for other water service. Administrative controls, including locks for tank outlet valves, are unacceptable as the only means to ensure minimum water volume.

Other water systems used as one of the two fire water supplies shall be permanently connected to the fire main system and shall be capable of automatic alignment to the fire main system. Pumps, controls, and power supplies in these systems shall satisfy the requirements for the main fire pumps. The use of other water systems for fire protection shall not be incompatible with their functions required for safe plant shutdown. Failure of the other system shall not degrade the fire main system.

B. *Sectional isolation valves.* Sectional isolation valves such as post indicator valves or key operated valves shall be installed in the fire main loop to permit isolation of portions of the fire main loop for maintenance or repair without interrupting the entire water supply.

C. *Hydrant isolation valves.* Valves shall be installed to permit isolation of outside hydrants from the fire main for maintenance or repair without interrupting the water supply to automatic or manual fire suppression systems in any area containing or presenting a fire hazard to safety-related or safe shutdown equipment.

D. *Manual fire suppression.* Standpipe and hose systems shall be installed so that at least one effective hose stream will be able to reach any location that contains or presents an exposure fire hazard to structures, systems, or components important to safety.

Access to permit effective functioning of the fire brigade shall be provided to all areas that contain or present an exposure fire hazard to structures, systems, or components important to safety.

Standpipe and hose stations shall be inside PWR containments and BWR containments that are not inerted. Standpipe and hose stations inside containment may be connected to a high quality water supply of sufficient quantity and pressure other than the fire main loop if plant-specific features prevent extending the fire main supply inside containment. For BWR drywells, standpipe and hose stations shall be placed outside the dry well with adequate lengths of hose to reach any location inside the dry well with an effective hose stream.

E. *Hydrostatic hose tests.* Fire hose shall be hydrostatically tested at a pressure of 150 psi or 50 psi above maximum fire main operating pressure, whichever is greater. Hose stored in outside hose houses shall be tested annually. Interior standpipe hose shall be tested every three years.

F. *Automatic fire detection.* Automatic fire detection systems shall be installed in all areas of the plant that contain or present an exposure fire hazard to safe shutdown or safety-related systems or components. These fire detection systems shall be capable of operating with or without offsite power.

G. *Fire protection of safe shutdown capability*. 1. Fire protection features shall be provided for structures, systems, and components important to safe shutdown. These features shall be capable of limiting fire damage so that:

a. One train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or emergency control station(s) is free of fire damage; and

b. Systems necessary to achieve and maintain cold shutdown from either the control room or emergency control station(s) can be repaired within 72 hours.

2. Except as provided for in paragraph G.3 of this section, where cables or equipment, including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided:

a. Separation of cables and equipment and associated non-safety circuits of redundant trains by a fire barrier having a 3-hour rating. Structural steel forming a part of or supporting such fire barriers shall be protected to provide fire resistance equivalent to that required of the barrier;

b. Separation of cables and equipment and associated non-safety circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustible or fire hazards. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area; or

c. Enclosure of cable and equipment and associated non-safety circuits of one redundant train in a fire barrier having a 1-hour rating, In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area;

Inside noninerted containments one of the fire protection means specified above or one of the following fire protection means shall be provided:

d. Separation of cables and equipment and associated non-safety circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards;

e. Installation of fire detectors and an automatic fire suppression system in the fire area; or

f. Separation of cables and equipment and associated non-safety circuits of redundant trains by a noncombustible radiant energy shield.

3. Alternative or dedicated shutdown capability and its associated circuits,[1] independent of cables, systems or components in the area, room, zone under consideration should be provided:

a. Where the protection of systems whose function is required for hot shutdown does not satisfy the requirement of paragraph G.2 of this section; or

b. Where redundant trains of systems required for hot shutdown located in the same fire area may be subject to damage from fire suppression activities or from the rupture or inadvertent operation of fire suppression systems.

In addition, fire detection and a fixed fire suppression system shall be installed in the area, room, or zone under consideration.

H. *Fire brigade*. A site fire brigade trained and equipped for fire fighting shall be established to ensure adequate manual fire fighting capability for all areas of the plant containing structures, systems, or components important to safety. The fire brigade shall be at least five members on each shift. The brigade leader and at least two brigade members shall have sufficient training in or knowledge of plant safety-related systems to understand the effects of fire and fire suppressants on safe shutdown capability. The qualification of fire brigade members shall include an annual physical examination to determine their ability to perform strenuous fire fighting activities. The shift supervisor shall not be a member of the fire brigade. The brigade leader shall be competent to assess the potential safety consequences of a fire and advise control room personnel. Such competence by the brigade leader may be evidenced by possession of an operator's license or equivalent knowledge of plant safety-related systems.

The minimum equipment provided for the brigade shall consist of personal protective equipment such as turnout coats, boots, gloves, hard hats, emergency communications equipment, portable lights, portable ventilation equipment, and portable extinguishers. Self-contained breathing apparatus using full-face positive-pressure masks approved by NIOSH (National Institute for Occupational Safety and Health—approval formerly given by the U.S. Bureau of Mines) shall be provided for fire brigade, damage control, and control room personnel. At least 10 masks shall be available for fire brigade personnel. Control room personnel may be

[1] Alternative shutdown capability is provided by rerouting, relocating, or modifying existing systems; dedicated shutdown capability is provided by installing new structures and systems for the function of post-fire shutdown.

furnished breathing air by a manifold system piped from a storage reservoir if practical. Service or rated operating life shall be a minimum of one-half hour for the self-contained units.

At least a 1-hour supply of breathing air in extra bottles shall be located on the plant site for each unit of self-contained breathing appratus. In addition, an onsite 6-hour supply of reserve air shall be provided and arranged to permit quick and complete replenishment of exhausted air supply bottles as they are returned. If compressors are used as a source of breathing air, only units approved for breathing air shall be used and the compressors shall be operable assuming a loss of offsite power. Special care must be taken to locate the compressor in areas free of dust and contaminants.

I. *Fire brigade training*. The fire brigade training program shall ensure that the capability to fight potential fires is established and maintained. The program shall consist of an initial classroom instruction program followed by periodic classroom instruction, fire fighting practice, and fire drills:

1. *Instruction*

a. The initial classroom instruction shall include:

(1) Indoctrination of the plant fire fighting plan with specific identification of each individual's responsibilities.

(2) Identification of the type and location of fire hazards and associated types of fires that could occur in the plant.

(3) The toxic and corrosive characteristics of expected products of combustion.

(4) Identification of the location of fire fighting equipment for each fire area and familiarization with the layout of the plant, including access and egress routes to each area.

(5) The proper use of available fire fighting equipment and the correct method of fighting each type of fire. The types of fires covered should include fires in energized electrical equipment, fires in cables and cable trays, hydrogen fires, fires involving flammable and combustible liquids or hazardous process chemicals, fires resulting from construction or modifications (welding), and record file fires.

(6) The proper use of communication, lighting, ventilation, and emergency breathing equipment.

(7) The proper method for fighting fires inside buildings and confined spaces.

(8) The direction and coordination of the fire fighting activities (fire brigade leaders only).

(9) Detailed review of fire fighting strategies and procedures.

(10) Review of the latest plant modifications and corresponding changes in fire fighting plans.

NOTE: Items (9) and (10) may be deleted from the training of no more than two of the non-operations personnel who may be assigned to the fire brigade.

b. The instruction shall be provided by qualified individuals who are knowledgeable, experienced, and suitably trained in fighting the types of fires that could occur in the plant and in using the types of equipment available in the nuclear power plant.

c. Instruction shall be provided to all fire brigade members and fire brigade leaders.

d. Regular planned meetings shall be held at least every 3 months for all brigade members to review changes in the fire protection program and other subjects as necessary.

e. Periodic refresher training sessions shall be held to repeat the classroom instruction program for all brigade members over a two-year period. These sessions may be concurrent with the regular planned meetings.

2. *Practice*

Practice sessions shall be held for each shift fire brigade on the proper method of fighting the various types of fires that could occur in a nuclear power plant. These sessions shall provide brigade members with experience in actual fire extinguishment and the use of emergency breathing apparatus under strenuous conditions encountered in fire fighting. These practice sessions shall be provided at least once per year for each fire brigade member.

3. *Drills*

a. Fire brigade drills shall be performed in the plant so that the fire brigade can practice as a team.

b. Drills shall be performed at regular intervals not to exceed 3 months for each shift fire brigade. Each fire brigade member should participate in each drill, but must participate in at least two drills per year.

A sufficient number of these drills, but not less than one for each shift fire brigade per year, shall be unannounced to determine the fire fighting readiness of the plant fire brigade, brigade leader, and fire protection systems and equipment. Persons planning and authorizing an unannounced drill shall ensure that the responding shift fire brigade members are not aware that a drill is being planned until it is begun. Unannounced drills shall not be scheduled closer than four weeks.

At least one drill per year shall be performed on a ''back shift'' for each shift fire brigade.

c. The drills shall be preplanned to establish the training objectives of the drill and shall be critiqued to determine how well the training objectives have been met. Unannounced drills shall be planned and critiqued by members of the management staff responsible for plant safety and fire protection. Performance deficiencies of a fire brigade or of individual fire brigade members shall be remedied by scheduling additional training for the brigade or members. Unsatisfactory drill performance shall be followed by a repeat drill within 30 days.

d. At 3-year intervals, a randomly selected unannounced drill must be critiqued by qualified individuals independent of the licensee's staff. A copy of the written report from these individuals must be available for NRC review and shall be retained as a record as specified in section III.I.4 of this appendix.

e. Drills shall as a minimum include the following:

(1) Assessment of fire alarm effectiveness, time required to notify and assemble fire brigade, and selection, placement and use of equipment, and fire fighting strategies.

(2) Assessment of each brigade member's knowledge of his or her role in the fire fighting strategy for the area assumed to contain the fire. Assessment of the brigade member's conformance with established plant fire fighting procedures and use of fire fighting equipment, including self-contained emergency breathing apparatus, communication equipment, and ventilation equipment, to the extent practicable.

(3) The simulated use of fire fighting equipment required to cope with the situation and type of fire selected for the drill. The area and type of fire chosen for the drill should differ from those used in the previous drill so that brigade members are trained in fighting fires in various plant areas. The situation selected should simulate the size and arrangement of a fire that could reasonably occur in the area selected, allowing for fire development due to the time required to respond, to obtain equipment, and organize for the fire, assuming loss of automatic suppression capability.

(4) Assessment of brigade leader's direction of the fire fighting effort as to thoroughness, accuracy, and effectiveness.

4. *Records*

Individual records of training provided to each fire brigade member, including drill critiques, shall be maintained for at least 3 years to ensure that each member receives training in all parts of the training program. These records of training shall be available for NRC review. Retraining or broadened training for fire fighting within buildings shall be scheduled for all those brigade members whose performance records show deficiencies.

J. *Emergency lighting.* Emergency lighting units with at least an 8-hour battery power supply shall be provided in all areas needed for operation of safe shutdown equipment and in access and egress routes thereto.

K. *Administrative controls.* Administrative controls shall be established to minimize fire hazards in areas containing structures, systems, and components important to safety. These controls shall establish procedures to:

1. Govern the handling and limitation of the use of ordinary combustible materials, combustible and flammable gases and liquids, high efficiency particulate air and charcoal filters, dry ion exchange resins, or other combustible supplies in safety-related areas.

2. Prohibit the storage of combustibles in safety-related areas or establish designated storage areas with appropriate fire protection.

3. Govern the handling of and limit transient fire loads such as combustible and flammable liquids, wood and plastic products, or other combustible materials in buildings containing safety-related systems or equipment during all phases of operating, and especially during maintenance, modification, or refueling operations.

4. Designate the onsite staff member responsible for the inplant fire protection review of proposed work activities to identify potential transient fire hazards and specify required additional fire protection in the work activity procedure.

5. Govern the use of ignition sources by use of a flame permit system to control welding, flame cutting, brazing, or soldering operations. A separate permit shall be issued for each area where work is to be done. If work continues over more than one shift, the permit shall be valid for not more than 24 hours when the plant is operating or for the duration of a particular job during plant shutdown.

6. Control the removal from the area of all waste, debris, scrap, oil spills, or other combustibles resulting from the work activity immediately following completion of the activity, or at the end of each work shift, whichever comes first.

7. Maintain the periodic housekeeping inspections to ensure continued compliance with these administrative controls.

8. Control the use of specific combustibles in safety-related areas. All wood used in safety-related areas during maintenance, modification, or refueling operations (such as lay-down blocks or scaffolding) shall be treated with a flame retardant. Equipment or supplies (such as new fuel) shipped in untreated combustible packing containers may be unpacked in safety-related areas if required for valid operating reasons. However, all combustible materials shall be removed from the area immediately following the unpacking. Such transient combustible material, unless stored in approved containers, shall not be left unattended during lunch breaks, shift changes, or other similar periods. Loose combustible packing material such as wood or paper excelsior, or polyethylene sheeting shall be placed in metal containers with tight-fitting self-closing metal covers.

9. Control actions to be taken by an individual discovering a fire, for example, notification of control room, attempt to extinguish fire, and actuation of local fire suppression systems.

10. Control actions to be taken by the control room operator to determine the need for brigade assistance upon report of a fire or receipt of alarm on control room annunciator panel, for example, announcing location of fire over PA system, sounding fire alarms, and notifying the shift supervisor and the fire brigade leader of the type, size, and location of the fire.

11. Control actions to be taken by the fire brigade after notification by the control room operator of a fire, for example, assembling in a designated location, receiving directions from the fire brigade leader, and discharging specific fire fighting responsibilities including selection and transportation of fire fighting equipment to fire location, selection of protective equipment, operating instructions for use of fire suppression systems, and use of preplanned strategies for fighting fires in specific areas.

12. Define the strategies for fighting fires in all safety-related areas and areas presenting a hazard to safety-related equipment. These strategies shall designate:

a. Fire hazards in each area covered by the specific prefire plans.

b. Fire extinguishants best suited for controlling the fires associated with the fire hazards in that area and the nearest location of these extinguishants.

c. Most favorable direction from which to attack a fire in each area in view of the ventilation direction, access hallways, stairs, and doors that are most likely to be free of fire, and the best station or elevation for fighting the fire. All access and egress routes that involve locked doors should be specifically identified in the procedure with the appropriate precautions and methods for access specified.

d. Plant systems that should be managed to reduce the damage potential during a local fire and the location of local and remote controls for such management (e.g., any hydraulic or electrical systems in the zone covered by the specific fire fighting procedure that could increase the hazards in the area because of overpressurization or electrical hazards).

e. Vital heat-sensitive system components that need to be kept cool while fighting a local fire. Particularly hazardous combustibles that need cooling should be designated.

f. Organization of fire fighting brigades and the assignment of special duties according to job title so that all fire fighting functions are covered by any complete shift personnel complement. These duties include command control of the brigade, transporting fire suppression and support equipment to the fire scenes, applying the extinguishant to the fire, communication with the control room, and coordination with outside fire departments.

g. Potential radiological and toxic hazards in fire zones.

h. Ventilation system operation that ensures desired plant air distribution when the ventilation flow is modified for fire containment or smoke clearing operations.

i. Operations requiring control room and shift engineer coordination or authorization.

j. Instructions for plant operators and general plant personnel during fire.

L. *Alternative and dedicated shutdown capability*. 1. Alternative or dedicated shutdown capability provided for a specific fire area shall be able to (a) achieve and maintain subcritical reactivity conditions in the reactor; (b) maintain reactor coolant inventory; (c) achieve and maintain hot standby[2] conditions for a PWR (hot shutdown[2] for a BWR); (d) achieve cold shutdown conditions within 72 hours; and (e) maintain cold shutdown conditions thereafter. During the postfire shutdown, the reactor coolant system process variables shall be maintained within those predicted for a loss of normal a.c. power, and the fission product boundary integrity shall not be affected; i.e., there shall be no fuel clad damage, rupture of any primary coolant boundary, or rupture of the containment boundary.

2. The performance goals for the shutdown functions shall be:

a. The reactivity control function shall be capable of achieving and maintaining cold shutdown reactivity conditions.

b. The reactor coolant makeup function shall be capable of maintaining the reactor coolant level above the top of the core for BWRs and be within the level indication in the pressurizer for PWRs.

c. The reactor heat removal function shall be capable of achieving and maintaining decay heat removal.

d. The process monitoring function shall be capable of providing direct readings of the process variables necessary to perform and control the above functions.

e. The supporting functions shall be capable of providing the process cooling, lubrication, etc., necessary to permit the operation of the equipment used for safe shutdown functions.

3. The shutdown capability for specific fire areas may be unique for each such area, or it may be one unique combination of systems for all such areas. In either case, the alternative shutdown capability shall be independent of the specific fire area(s) and shall accommodate postfire conditions where offsite power is available and where offsite power is not available for 72 hours. Procedures shall be in effect to implement this capability.

4. If the capability to achieve and maintain cold shutdown will not be available because of fire damage, the equipment and systems

[2] As defined in the Standard Technical Specifications.

comprising the means to achieve and maintain the hot standby or hot shutdown condition shall be capable of maintaining such conditions until cold shutdown can be achieved. If such equipment and systems will not be capable of being powered by both onsite and offsite electric power systems because of fire damage, an independent onsite power system shall be provided. The number of operating shift personnel, exclusive of fire brigade members, required to operate such equipment and systems shall be on site at all times.

5. Equipment and systems comprising the means to achieve and maintain cold shutdown conditions shall not be damaged by fire; or the fire damage to such equipment and systems shall be limited so that the systems can be made operable and cold shutdown can be achieved within 72 hours. Materials for such repairs shall be readily available on site and procedures shall be in effect to implement such repairs. If such equipment and systems used prior to 72 hours after the fire will not be capable of being powered by both onsite and offsite electric power systems because of fire damage, an independent onsite power system shall be provided. Equipment and systems used after 72 hours may be powered by offsite power only.

6. Shutdown systems installed to ensure postfire shutdown capability need not be designed to meet seismic Category I criteria, single failure criteria, or other design basis accident criteria, except where required for other reasons, e.g., because of interface with or impact on existing safety systems, or because of adverse valve actions due to fire damage.

7. The safe shutdown equipment and systems for each fire area shall be known to be isolated from associated non-safety circuits in the fire area so that hot shorts, open circuits, or shorts to ground in the associated circuits will not prevent operation of the safe shutdown equipment. The separation and barriers between trays and conduits containing associated circuits of one safe shutdown division and trays and conduits containing associated circuits or safe shutdown cables from the redundant division, or the isolation of these associated circuits from the safe shutdown equipment, shall be such that a postulated fire involving associated circuits will not prevent safe shutdown.[3]

M. *Fire barrier cable penetration seal qualification.* Penetration seal designs must be qualified by tests that are comparable to tests used to rate fire barriers. The acceptance criteria for the test must include the following:

1. The cable fire barrier penetration seal has withstood the fire endurance test without passage of flame or ignition of cables on the unexposed side for a period of time equivalent to the fire resistance rating required of the barrier;

2. The temperature levels recorded for the unexposed side are analyzed and demonstrate that the maximum temperature is sufficiently below the cable insulation ignition temperature; and

3. The fire barrier penetration seal remains intact and does not allow projection of water beyond the unexposed surface during the hose stream test.

N. *Fire doors.* Fire doors shall be self-closing or provided with closing mechanisms and shall be inspected semiannually to verify that automatic hold-open, release, and closing mechanisms and latches are operable.

One of the following measures shall be provided to ensure they will protect the opening as required in case of fire:

1. Fire doors shall be kept closed and electrically supervised at a continuously manned location;

2. Fire doors shall be locked closed and inspected weekly to verify that the doors are in the closed position;

3. Fire doors shall be provided with automatic hold-open and release mechanisms and inspected daily to verify that doorways are free of obstructions; or

4. Fire doors shall be kept closed and inspected daily to verify that they are in the closed position.

The fire brigade leader shall have ready access to keys for any locked fire doors.

Areas protected by automatic total flooding gas suppression systems shall have electrically supervised self-closing fire doors or shall satisfy option 1 above.

O. *Oil collection system for reactor coolant pump.* The reactor coolant pump shall be equipped with an oil collection system if the containment is not inerted during normal operation. The oil collection system shall be so designed, engineered, and installed that failure will not lead to fire during normal or design basis accident conditions and that there is reasonable assurance that the system will withstand the Safe Shutdown Earthquake.[4]

Such collection systems shall be capable of collecting lube oil from all potential pressurized and unpressurized leakage sites in the reactor coolant pump lube oil systems. Leakage shall be collected and drained to a vented closed container that can hold the entire

[3] An acceptable method of complying with this alternative would be to meet Regulatory Guide 1.75 position 4 related to associated circuits and IEEE Std 384–1974 (Section 4.5) where trays from redundant safety divisions are so protected that postulated fires affect trays from only one safety division.

[4] See Regulatory Guide 1.29—''Seismic Design Classification'' paragraph C.2.

lube oil system inventory. A flame arrester is required in the vent if the flash point characteristics of the oil present the hazard of fire flashback. Leakage points to be protected shall include lift pump and piping, overflow lines, lube oil cooler, oil fill and drain lines and plugs, flanged connections on oil lines, and lube oil reservoirs where such features exist on the reactor coolant pumps. The drain line shall be large enough to accommodate the largest potential oil leak.

[45 FR 76611, Nov. 19, 1980; 46 FR 44735, Sept. 8, 1981, as amended at 53 FR 19251, May 27, 1988; 65 FR 38191, June 20, 2000; 77 FR 39907, July 6, 2012; 85 FR 65663, Oct. 16, 2020]

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. 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, 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), 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.

(b) These criteria implement General Design Criterion 2 insofar as it requires structures, systems, and components important to safety to withstand the effects of earthquakes.

II. Scope

The evaluations described in this appendix are within the scope of investigations permitted by § 50.10(c)(1).

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 of this chapter.

Design Approval means an NRC staff approval, issued under subpart E of part 52 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 of this chapter, of a standard design for a nuclear power facility.

Manufacturing license means a license, issued under subpart F of part 52 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.

Operating basis earthquake ground motion (OBE) is the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional. The operating basis earthquake ground motion is only associated with plant shutdown and inspection unless specifically selected by the applicant as a design input.

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.

Safe-shutdown earthquake ground motion (SSE) is the vibratory ground motion for which certain structures, systems, and components must be designed to remain functional.

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).

Surface deformation is distortion of geologic strata at or near the ground surface by the processes of folding or faulting as a result of various earth forces. Tectonic surface deformation is associated with earthquake processes.

IV. APPLICATION TO ENGINEERING DESIGN

The following are pursuant to the seismic and geologic design basis requirements of §100.23 of this chapter:

(a) *Vibratory Ground Motion.* (1) *Safe Shutdown Earthquake Ground Motion.* (i) The Safe Shutdown Earthquake Ground Motion must be characterized by free-field ground motion response spectra at the free ground surface. In view of the limited data available on vibratory ground motions of strong earthquakes, it usually will be appropriate that the design response spectra be smoothed spectra. The horizontal component of the Safe Shutdown Earthquake Ground Motion in the free-field at the foundation level of the structures must be an appropriate response spectrum with a peak ground acceleration of at least 0.1g.

(ii) The nuclear power plant must be designed so that, if the Safe Shutdown Earthquake Ground Motion occurs, certain structures, systems, and components will remain functional and within applicable stress, strain, and deformation limits. In addition to seismic loads, applicable concurrent normal operating, functional, and accident-induced loads must be taken into account in the design of these safety-related structures, systems, and components. The design of the nuclear power plant must also take into account the possible effects of the Safe Shutdown Earthquake Ground Motion on the facility foundations by ground disruption, such as fissuring, lateral spreads, differential settlement, liquefaction, and landsliding, as required in §100.23 of this chapter.

(iii) The required safety functions of structures, systems, and components must be assured during and after the vibratory ground motion associated with the Safe Shutdown Earthquake Ground Motion through design, testing, or qualification methods.

(iv) The evaluation 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 structures, systems, and components during the Safe Shutdown Earthquake Ground Motion and under the postulated concurrent loads, provided the necessary safety functions are maintained.

(2) *Operating Basis Earthquake Ground Motion.* (i) The Operating Basis Earthquake Ground Motion must be characterized by response spectra. The value of the Operating Basis Earthquake Ground Motion must be set to one of the following choices:

(A) One-third or less of the Safe Shutdown Earthquake Ground Motion design response spectra. The requirements associated with this Operating Basis Earthquake Ground Motion in Paragraph (a)(2)(i)(B)(*I*) can be satisfied without the applicant performing explicit response or design analyses, or

(B) A value greater than one-third of the Safe Shutdown Earthquake Ground Motion design response spectra. Analysis and design must be performed to demonstrate that the requirements associated with this Operating Basis Earthquake Ground Motion in Paragraph (a)(2)(i)(B)(*I*) are satisfied. The design must take into account soil-structure interaction effects and the duration of vibratory ground motion.

(*I*) When subjected to the effects of the Operating Basis Earthquake Ground Motion in combination with normal operating loads, all structures, systems, and components of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public must remain functional and within applicable stress, strain, and deformation limits.

(3) *Required Plant Shutdown.* If vibratory ground motion exceeding that of the Operating Basis Earthquake Ground Motion or if significant plant damage occurs, the licensee must shut down the nuclear power plant. If systems, structures, or components necessary for the safe shutdown of the nuclear power plant are not available after the occurrence of the Operating Basis Earthquake Ground Motion, the licensee must consult with the Commission and must propose a plan for the timely, safe shutdown of the nuclear power plant. Prior to resuming operations, the licensee must demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public and the licensing basis is maintained.

(4) *Required Seismic Instrumentation.* Suitable instrumentation must be provided so that the seismic response of nuclear power plant features important to safety can be evaluated promptly after an earthquake.

(b) *Surface Deformation.* The potential for surface deformation must be taken into account in the design of the nuclear power plant by providing reasonable assurance that in the event of deformation, certain structures, systems, and components will remain functional. In addition to surface deformation induced loads, the design of safety features must take into account seismic loads and applicable concurrent functional and accident-induced loads. The design provisions for surface deformation must be based on its postulated occurrence in any direction and azimuth and under any part of the 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.

(c) *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 §100.23 of this chapter must be taken into account in the design of the nuclear power plant so as to prevent undue risk to the health and safety of the public.

[61 FR 65173, Dec. 11, 1996, as amended at 72 FR 49508, Aug. 28, 2007]