AUTHENTICATED
U.S. GOVERNMENT
INFORMATION
GPO

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, l,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]

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AUTHORITY: Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246 (42 U.S.C. 5841, 5842, 5846).

Section 50.7 also issued under Pub. L. 95–601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Section 50.10 also issued under secs. 101, 185, 68 Stat. 955 as amended (42 U.S.C. 2131, 2235), sec. 102, Pub. L. 91–190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.13, 50.54(dd), and 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and appendix Q also issued under sec. 102, Pub. L. 91–190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97–415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80–50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C 2237).

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).

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) 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.

By–product material means any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material.

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.

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.

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-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,[1] 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,

[1] The Department facilities identified in section 202 are:

(1) Demonstration Liquid Metal Fast Breeder reactors when operated as part of the power generation facilities of an electric utility system, or when operated in any other manner for the purpose of demonstrating the suitability for commercial application of such a reactor.

(2) Other demonstration nuclear reactors, except those in existence on January 19, 1975, when operated as part of the power generation facilities of an electric utility system, or when operated in any other manner for the purpose of demonstrating the suitability for commercial application of such a reactor.

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.

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.

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 deep-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 any nuclear reactor other than one designed or used primarily for the formation of plutonium or U–233.

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 additional FEDERAL REGISTER citations affecting §50.2, see the List of CFR Sections Affected in the Finding Aids section of this volume.

§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) *Address requirements.* The signed original of 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 addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555.

(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 Nuclear Regulatory Commission at the locations and in the quantities set forth below (addresses for the NRC Regional Offices are listed in appendix D of 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, 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: the signed original to the Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, one copy to the appropriate Regional Office, and one 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) of this section.

(i) Applications for licenses for facilities described in § 50.21 (a) and (c) and amendments to these applications: The signed original must be sent to the Nuclear Regulatory Commission. Document Control Desk, Washington, DC 20555 and one copy to the appropriate Regional Office.

(ii) Applications for permits and licenses for facilities described in § 50.21(b) or § 50.22, and amendments to these applications: the signed original and 37 copies must be sent to the Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, one copy to the appropriate Regional Office, and one copy to the appropriate NRC Resident Inspector, if one has been assigned to the site of the facility.

(3) *Acceptance review application.* Written communications required for an application for determination of suitability for docketing pursuant to § 50.30(a)(6) must be submitted as follows: the signed original and 13 copies to the Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555 and one copy to the appropriate Regional Office.

(4) *Security plan and related submittals.* Written communications, as defined in paragraphs (b)(4) (i) through (iv) of this section must be submitted as follows: The signed original and three copies to the Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, and two copies to the appropriate Regional Office;

(i) Physical security plan pursuant to § 50.34;

(ii) Safeguards contingency plan pursuant to § 50.34;

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

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

(5) *Emergency plan and related submittals.* Written communications as defined in paragraphs (b)(5)(i) through (iii) in this section, must be submitted as follows: the signed original to the Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, two copies to the appropriate Regional Office, and one copy to the appropriate NRC Resident Inspector if

one has been assigned to the site of the facility.

(i) Emergency plan pursuant to § 50.34;

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

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

(6) *Updated FSAR.* An updated Final Safety Analysis Report (FSAR) or replacement pages, pursuant to § 50.71(e) must be submitted as follows: the signed original and 10 copies to the Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, one copy to the appropriate Regional Office, and one copy to the appropriate NRC Resident Inspector if one has been assigned to the site of the facility.

(7) *Quality assurance related submittals.* (i) A change to the Safety Analysis Report quality assurance program description pursuant to § 50.54(a)(3) or § 50.55(f)(3), or a change to a licensee's NRC-accepted quality assurance topical report pursuant to § 50.54(a)(3) or § 50.55(f)(3), must be submitted as follows: the signed original to the Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, one copy to the appropriate Regional Office, and one copy to the appropriate NRC Resident Inspector if one has been assigned to the site of the facility.

(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 as follows: one signed original to the Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555.

(8) *Certification of permanent cessation of operations.* The licensee's certification of permanent cessation of operations, pursuant to § 50.82(a)(1), must state the date on which operations have ceased or will cease, and the signed and notarized original must be submitted to: The Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555–0001.

(9) *Certification of permanent fuel removal.* The licensee's certification of permanent fuel removal, pursuant to § 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 the signed and notarized original must be submitted to: The Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555–0001.

(c) *Form of communications.* All 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 may be granted for the submittal of micrographic, photographic, or electronic forms. Prior to making any submittal in other than paper form, the applicant or licensee must contact the Information and Records Management Branch, Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 415–7230, to obtain specifications, copy requirements, and prior approval.

(d) *Delivery of communications.* Written communications may be delivered to the Document Control Desk at 11555 Rockville Pike, Rockville, Maryland between the hours of 8:15 a.m. and 4:00 p.m. Eastern Time. If a submittal due date falls on Saturday, Sunday, or Federal holiday, the next Federal working day becomes the official due date.

(e) *Regulation governing submission.* Licensees and applicants submitting correspondence, reports, and other written communications pursuant to 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 submittal, the specific regulation or other basis, requiring submission.

(f) *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 Information and Records Management

Branch, Nuclear Regulatory Commission, Washington, DC 20555, Telephone (301) 415–7230.

[51 FR 40306, Nov. 6, 1986, as amended at 52 FR 31611, Aug. 21, 1987; 53 FR 6139, Mar. 1, 1988; 60 FR 24551, May 9, 1995; 61 FR 39298, July 29, 1996]

§ 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 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(c). 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 or by calling the NRC Information and Records Management Branch at (301) 415–7230.

(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]

§ 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 §§ 50.30, 50.33, 50.33a, 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, 50.62, 50.63, 50.64, 50.65, 50.66, 50.71, 50.72, 50.74, 50.75, 50.80, 50.82, 50.90, 50.91, 50.120, 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, Form N–71 is approved under control number 3150–0056.

[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]

§ 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.

(a) Except as provided in § 50.11, 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.

(b) No person shall begin the construction of a production or utilization facility on a site on which the facility is to be operated until a construction permit has been issued. As used in this paragraph, the term ''construction'' shall be deemed to include pouring the foundation for, or the installation of, any portion of the permanent facility on the site, but does not include:

(1) Site exploration, site excavation, preparation of the site for construction of the facility, including the driving of piles, and construction of roadways, railroad spurs, and transmission lines;

(2) Procurement or manufacture of components of the facility;

(3) Construction of non-nuclear facilities (such as turbogenerators and turbine buildings) and temporary buildings (such as construction equipment storage sheds) for use in connection with the construction of the facility; and

(4) With respect to production or utilization facilities, other than testing facilities, required to be licensed pursuant to section 104a or section 104c of the Act, the construction of buildings which will be used for activities other than operation of a facility and which may also be used to house a facility. (For example, the construction of a college laboratory building with space for installation of a training reactor is not affected by this paragraph. This paragraph does not apply to production or utilization facilities subject to paragraph (c) of this section.

(c) Notwithstanding the provisions of paragraph (b) of this section, and subject to paragraphs (d) and (e) of this section, no person shall effect commencement of construction of a production or utilization facility subject to the provisions of § 51.20(b) of this chapter on a site on which the facility is to be operated until a construction permit has been issued. As used in this paragraph, the term ''commencement of construction'' means any clearing of land, excavation or other substantial action that would adversely affect the environment of a site, but does not mean:

(1) Changes desirable for the temporary use of the land for public recreational uses, necessary borings to determine foundation conditions or other

preconstruction monitoring to establish background information related to the suitability of the site or to the protection of environmental values;

(2) Procurement or manufacture of components of the facility; and

(3) With respect to production or utilization facilities, other than testing facilities, required to be licensed pursuant to section 104a or section 104c of the Act, the construction of buildings which will be used for activities other than operation of a facility and which may also be used to house a facility. (For example, the construction of a college laboratory building with space for installation of a training reactor is not affected by this paragraph.)

(d)(1) Each person subject to the provisions of paragraph (c) of this section, who is, on March 21, 1972, conducting activities permitted pursuant to paragraph (b) of this section in effect prior to March 21, 1972, may furnish to the Commission within 30 days after March 21, 1972 or such later date as may be approved by the Commission upon good cause shown, a written statement of any reasons, with supporting factual submission, why, with reference to the factors stated in paragraph (d)(2) of this section, the activities should be continued, pending the issuance of a construction permit, notwithstanding the provisions of paragraph (c) of this section. If such written statement has been submitted, within the time specified, such activities may continue to be conducted pending Commission action pursuant to paragraph (d)(2) of this section.

(2) Upon submission of a statement of reasons pursuant to paragraph (d)(1) of this section the Commission may authorize the continued conduct of activities permitted by paragraph (b) of this section in effect prior to March 21, 1972, upon consideration and balancing of the following factors:

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

(ii) Whether redress of any adverse environmental impact from continuation of the activities can reasonably be effected should such redress be necessary;

(iii) Whether continuation of the activities would foreclose subsequent adoption of alternatives; and

(iv) The effect of delay in conducting such activities on the public interest, including the power needs to be served 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.

(3) Activities permitted to be continued pursuant to this paragraph (d) shall be conducted in such a manner as will minimize or reduce their environmental impact.

(e)(1) The Director of Nuclear Reactor Regulation may authorize an applicant for a construction permit for a utilization facility which is subject to § 51.20(b) of this chapter, and is of the type specified in § 50.21(b) (2) or (3) or § 50.22 or is a testing facility to conduct the following activities: (i) Preparation of the site for construction of the facility (including such activities as clearing, grading, construction of temporary access roads and borrow areas); (ii) installation of temporary construction support facilities (including such items as warehouse and shop facilities, utilities, concrete mixing plants, docking and unloading facilities, and construction support buildings); (iii) excavation for facility structures; (iv) construction of service facilities (including such facilities as roadways, paving, railroad spurs, fencing, exterior utility and lighting systems, transmission lines, and sanitary sewerage treatment facilities); and (v) the construction of structures, systems and components which do not prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. No such authorization shall be granted unless the staff has completed a final environmental impact statement on the issuance of the construction permit as required by subpart A of part 51 of this chapter.

(2) Such an authorization shall be granted only after the presiding officer in the proceeding on the construction permit application (i) has made all the findings required by §§ 51.104(b) and 51.105 of this chapter to be made prior to issuance of the construction permit

for the facility, and (ii) has determined that, based upon the available information and review to date, there is reasonable assurance that the proposed site is a suitable location for a reactor of the general size and type proposed from the standpoint of radiological health and safety considerations under the Act and rules and regulations promulgated by the Commission pursuant thereto.

(3)(i) The Director of Nuclear Reactor Regulation may authorize an applicant for a construction permit for a utilization facility which is subject to § 51.20(b) of this chapter, and is of the type specified in § 50.21(b) (2) or (3) or § 50.22 or is a testing facility to conduct, in addition to the activities described in paragraph (e)(1) of this section, the installation of structural foundations, including any necessary subsurface preparation, for structures, systems and components which prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

(ii) Such an authorization, which may be combined with the authorization described in paragraph (e)(1) of this section, or may be granted at a later time, shall be granted only after the presiding officer in the proceeding on the construction permit application has, in addition to making the findings and determinations required by paragraph (e)(2) of this section, determined that there are no unresolved safety issues relating to the additional activities that may be authorized pursuant to this paragraph that would constitute good cause for withholding authorization.

(4) Any activities undertaken pursuant to an authorization granted under this paragraph shall be entirely at the risk of the applicant and, except as to matters determined under paragraphs (e)(2) and (e)(3)(ii), the grant of the authorization shall have no bearing on the issuance of a construction permit with respect to the requirements of the Act, and rules, regulations, or orders promulgated pursuant thereto.

[21 FR 355, Jan. 19, 1956, as amended at 25 FR 8712, Sept. 9, 1960; 33 FR 2381, Jan. 31, 1968; 35 FR 11460, July 7, 1970; 37 FR 5748, Mar. 21, 1972; 39 FR 14508, Apr. 24, 1974; 39 FR 26279, July 18, 1974; 39 FR 33202, Sept. 16, 1974; 43 FR 6924, Feb. 17, 1978; 49 FR 9403, Mar. 12, 1984]

§ 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[2] 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

[2] The Department facilities identified in section 202 are:

(1) Demonstration Liquid Metal Fast Breeder reactors when operated as part of the power generation facilities of an electric utility system, or when operated in any other manner for the purpose of demonstrating the suitability for commercial application of such a reactor.

(2) Other demonstration nuclear reactors, except those in existence on January 19, 1975, when operated as part of the power generation facilities of an electric utility system, or when operated in any other manner for the purpose of demonstrating the suitability for commercial application of such a reactor.

(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]

§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 prior to the issuance of a license if the application is otherwise acceptable, and will be converted upon due completion of the facility and Commission action into a license as provided in §50.56 of this part. A construction permit for the alteration of a production or utilization facility will be issued prior to the issuance of an amendment of a license, if the application for amendment is otherwise acceptable, as provided in §50.91.

[21 FR 355, June 19, 1956, as amended at 35 FR 11461, July 17, 1970]

APPLICATIONS FOR LICENSES, FORM, CONTENTS, INELIGIBILITY OF CERTAIN APPLICANTS

§50.30 Filing of application for licenses; oath or affirmation.

(a) *Serving of applications.* (1) Each filing of an application for a license to construct and/or operate a production or utilization facility (including amendments to the applications) must be submitted to the U.S. Nuclear Regulatory Commission in accordance with §50.4.

(2) An additional 10 copies of the general information and 30 copies of the safety analysis report, or part thereof or amendment thereto, must be retained by the applicant for distribution in accordance with the written instructions of the Director, Office of Nuclear Reactor Regulation, or the Director, Office of Nuclear Material Safety and Safeguards, as appropriate.

(3) Each applicant shall, upon notification by the Atomic Safety and Licensing Board appointed to conduct the public hearing required by the Atomic Energy Act for the issuance of a construction permit, 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. In addition, at that time the applicant shall serve a copy of the updated application on the Atomic Safety and Licensing Appeal Panel. 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.

(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 establish a Local Public Document Room near the site of the proposed facility, for the use of the public, where a copy of the application, subsequent amendments, and other records pertinent to the facility will be available for public inspection and copying.

(6) The serving of copies required by this section must not occur until the application has been docketed pursuant to §2.101(a) of this chapter. Copies must be submitted to the Commission, as specified in §50.4, 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 license, including whenever appropriate a construction 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 production or utilization facility license, including, whenever appropriate, a construction 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 or operating license, except as provided in § 170.21 of this chapter.

(f) *Environmental report.* An application for a construction permit or an operating license for a nuclear power reactor, testing facility, fuel reprocessing plant, or such other production or utilization facility whose construction or operation may be determined by the Commission to have a significant impact on the environment shall be accompanied by any Environmental Report required pursuant to 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]

§ 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 application to renew or extend the term of an operating license must include the same financial information as is required in an application for an initial license.

(3) Each application for a construction permit or an operating license submitted by a newly-formed entity organized for the primary purpose of constructing 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) Its 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.

(4) The Commission may request an established entity or newly-formed entity to submit additional or more detailed information respecting its financial arrangements and status of funds if the Commission considers this information appropriate. This may include information regarding a licensee's ability to continue the conduct of the activities authorized by the license and to decommission the facility.

(g) If the application is for an operating license for a nuclear power reactor, 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)[3], as well as the plans of State governments wholly or partially within the ingestion pathway EPZ.[4] 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 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.

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[3] 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.

[4] 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.

(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 for a production or utilization facility, 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.

(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]

§ 50.33a Information requested by the Attorney General for antitrust review.

(a)(1) An applicant for a construction permit for a nuclear power reactor shall submit the information requested by the Attorney General as described in appendix L to this part, if the application is for a class 103 permit and if the applicant has electrical generating capacity exceeding 1400 MW(e).

(2) An applicant for a construction permit for a nuclear power reactor shall submit the information requested by the Attorney General as described in paragraph 9 of section II of appendix L if the applicant has electrical generating capacity exceeding 200 MW(e) but no more than 1400 MW(e). Upon request of the Commission, the applicant shall furnish the other information described in appendix L.

(3) An applicant for a construction permit for a nuclear power reactor is not required to submit the information described in appendix L unless specifically requested by the Commission to provide the information, if the applicant has electrical generating capacity of 200 MW(e) or less.

(4) The information described in paragraphs (a) (1) and (2) of this section shall be submitted as a separate document prior to any other part of the license application as provided in paragraph (b) and in accordance with § 2.101 of this chapter.

(b) Except as provided in paragraph (d), any person who applies for a class 103 construction permit for a nuclear power reactor on or after July 28, 1975 shall submit the document titled ''Information Requested by the Attorney General for Antitrust Review'' at least nine (9) months but not more than thirty-six months prior to the date of submittal of any part of the application for a class 103 construction permit.

(c) [Reserved]

(d) Any person who applies for a class 103 construction permit for a nuclear power reactor pursuant to the provisions of § 2.101(a–1) and subpart F of part 2 of this chapter shall submit the document title ''Information Requested by the Attorney General for Antitrust Review'' at least nine (9) months but not more than thirty-six months prior to the filing of part two or part three of the application, whichever part is filed first, as specified in § 2.101(a–1) of this chapter.

(e) Any person who applies for a class 103 construction permit for a fuel reprocessing plant shall submit the information requested by the Attorney General for antitrust review, as a separate document, as soon as possible and in accordance with § 2.101 of this chapter.

[39 FR 34395, Sept. 25, 1974, as amended at 42 FR 22887, May 5, 1977; 42 FR 25721, May 19, 1977; 43 FR 49775, Oct. 25, 1978; 44 FR 60716, Oct. 22, 1979; 57 FR 18391, Apr. 30, 1992]

§ 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:

[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.

(1) Stationary power reactor applicants for a construction permit pursuant to this part, or a design certification or combined license pursuant to part 52 of this chapter 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 pursuant to this part or a design certification or combined license pursuant to part 52 of this chapter, 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 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).

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[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,

Continued

(*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 this paragraph (a)(1)(i), in support of the application for a construction permit, or a design approval.

(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.

(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 (i) the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and (ii) 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 following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of § 50.46 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

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.

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

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 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 pursuant to this part, or a design certification or combined license pursuant to part 52 of this chapter, 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.

(b) *Final safety analysis report.* Each application for a license to operate a facility 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 or 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 pursuant to this part, or a design certification or combined license pursuant to part 52 of this chapter, 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 prior to 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 pursuant to this part, or a combined license pursuant to part 52 of this chapter, shall provide a description and safety assessment of the site and of the facility as in § 50.34(a)(1)(ii) of this part. However, for either an operating license applicant or holder whose construction permit was issued prior to 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.

(c) Each application for a license to operate a production or utilization facility must include a physical security plan. The plan must describe how the applicant will meet the requirements of part 73 (and part 11 of this chapter, if applicable, including the identification and description of jobs as required by § 11.11(a), at the proposed facility). The plan must list tests, inspections, audits, and other means to be used to demonstrate compliance with the requirements of 10 CFR parts 11 and 73, if applicable.

(d) *Safeguards contingency plan.* Each application for a license to operate a production or utilization facility that will be subject to §§ 73.50, 73.55, or § 73.60 of this chapter must include a licensee

safeguards contingency plan in accordance with the criteria set forth in appendix C to 10 CFR part 73. The safeguards contingency plan shall include plans for dealing with threats, thefts, and radiological sabotage, as defined in part 73 of this chapter, relating to the special nuclear material and nuclear facilities licensed under this chapter and in the applicant's possession and control. Each application for such a license shall include the first four categories of information contained in the applicant's safeguards contingency plan. (The first four categories of information as set forth in appendix C to 10 CFR part 73 are Background, Generic Planning Base, Licensee Planning Base, and Responsibility Matrix. The fifth category of information, Procedures, does not have to be submitted for approval.)[9]

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

(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 rule applies only 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, if issued, will be that number whose start of manufacture, as defined in the license application, can practically begin within a ten-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 five 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 rule. 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 the manufacturing license may be considered in the manufacturing license proceeding.

(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 such studies are factored into the final design of the facility. All studies shall be completed no later than two years following issuance of the construction permit or manufacturing license.[10]

(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.

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

[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 TID 14844 source term 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 TID–14844 source term 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, iodines 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 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) TID 14844 source term 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 a TID 14844 source term 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, Subsubarticle 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 Subsubarticle 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, Subsubarticle 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, Subsubarticle 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) *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, manufacturing licenses, and preliminary or final design approvals for standard plants 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 prior to 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.

[33 FR 18612, Dec. 17, 1968]

EDITORIAL NOTE: For additional FEDERAL REGISTER citations affecting §50.34, see the List of CFR Sections Affected in the Finding Aids section of this volume.

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

(a) An application for a permit to construct a nuclear power reactor 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 utilization 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 permit to construct a nuclear power reactor shall include:

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

(2) An estimate of:

(i) The quantity of each of the principal radio-nuclides 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 radio-nuclides 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 a license to operate a nuclear power reactor 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, pursuant to paragraph (a) of this section; an (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.

[35 FR 18387, Dec. 3, 1970, as amended at 40 FR 58847, Dec. 19, 1975; 61 FR 65172, Dec. 11, 1996]

§ 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.

(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) 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.

[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) 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 of the 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]

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

(a) In order 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 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 licensee 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]

§ 50.36b Environmental conditions.

Each license authorizing operation of a production or utilization facility, and each license for a nuclear power reactor facility for which the certification of permanent cessation of operations required under § 50.82(a)(1) 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 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 and 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.

[61 FR 39299, July 29, 1996]

§ 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 or construction permit, the applicant shall agree in writing that it will not permit any individual to have access to or any facility to possess Restricted Data or classified National Security Information until the individual and/or facility has been approved for such access under the provisions of 10 CFR parts 25 and/or 95. The agreement of the applicant in this regard shall be deemed part of the license or construction permit, whether so stated therein or not.

[62 FR 17690, Apr. 11, 1997]

§ 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 AND CONSTRUCTION PERMITS

§50.40 Common standards.

In determining that a license will be issued to an applicant, the Commission will be guided by the following considerations:

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

(b) The applicant 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.

(c) The issuance of a 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 part 51 have been satisfied.

[21 FR 355, Jan. 19, 1956, as amended at 36 FR 12731, July 7, 1971; 49 FR 9404, Mar. 12, 1984; 49 FR 35753, Sept. 12, 1984]

§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) An application for a class 104 operating license as to which a person who intervened or sought by timely written notice to the Commission to intervene in the construction permit proceeding for the facility to obtain a determination of antitrust considerations or to advance a jurisdictional basis for such determination has requested an antitrust review under section 105 of the Act within 25 days after the date of publication in the FEDERAL REGISTER of notice of filing of the application for an operating license or December 19, 1970, whichever is later, is also subject to the provisions of §50.42(b).

[21 FR 355, Jan. 19, 1956, as amended at 35 FR 19660, Dec. 29, 1970]

§50.42 Additional standards 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, be guided by the following considerations:

(a) The proposed activities will serve a useful purpose proportionate to the quantities of special nuclear material or source material to be utilized.

(b) Due account will be taken of the advice provided by the Attorney General, pursuant to subsection 105c of the Act, and to such evidence as may be provided during any proceedings in connection with the antitrust aspects of the application. For this purpose, the Commission will promptly transmit to the Attorney General a copy of the license application, and request such advice as the Attorney General determines to be appropriate in regard

to the finding to be made by the Commission as to whether the proposed license would create or maintain a situation inconsistent with the antitrust laws, as specified in subsection 105a of the Act: *Provided,* That this requirement will not apply with respect to the types of class 103 licenses which the Commission, with the approval of the Attorney General, may determine would not significantly affect the applicant's activities under the antitrust laws: *And provided further,* That this requirement will not apply to an application for a license to operate a production or utilization facility for which a class 103 construction permit was issued unless the Commission, after consultation with the Attorney General, determines such review is advisable on the ground that significant changes in the licensee's activities or proposed activities have occurred subsequent to the previous review by the Attorney General and the Commission. Upon receipt of the Attorney General's advice, the Commission will cause such advice to be published in the FEDERAL REGISTER. After consideration of the antitrust aspects of the application, the Commission, if it finds that the license to be issued or continued, would create or maintain a situation inconsistent with the antitrust laws as specified in subsection 105a of the Act, will consider, in determining whether a license should be issued or continued, such other factors as the Commission in its judgment deems necessary to protect the public interest, including the need for power in the affected area.[1]

[21 FR 355, Jan. 19, 1956, as amended at 35 FR 11461, July 17, 1970; 35 FR 19660, Dec. 29, 1970]

[1] As permitted by subsection 105c(8) of the Act, with respect to proceedings in which an application for a construction permit was filed prior to Dec. 19, 1970, and proceedings in which a written request for antitrust review of an application for an operating license to be issued under section 104b has been made by a person who intervened or sought by timely written notice to the Atomic Energy Commission to intervene in the construction permit proceeding for the facility to obtain a determination of antitrust considerations or to advance a jurisdictional basis for such determination within 25 days after the date of publication in the FEDERAL REGISTER of notice of filing of the application for an operating license or Dec. 19, 1970, whichever is later, the Commission may issue a construction permit or operating license in advance of consideration of, and findings with respect to the antitrust aspects of the application, provided that the permit or license so issued contains the condition specified in § 50.55b.

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

In addition to applying the standards set forth in §§ 50.40 and 50.42, in the case of a class 103 license for a facility for the generation of commercial power:

(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 herein 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 license for the construction and operation of a utilization facility for the primary purpose of producing electric energy for disposition for ultimate public consumption.

[21 FR 355, Jan. 19, 1956, as amended at 35 FR 19660, Dec. 29, 1970; 63 FR 50480, Sept. 22, 1998]

§50.44 Standards for combustible gas control system in light-water-cooled power reactors.

(a) Each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets withincylindrical zircaloy or ZIRLO cladding, must, as provided in paragraphs (b) through (d) of this section, include means for control of hydrogen gas that may be generated, following a postulated loss-of-coolant accident (LOCA) by—

(1) Metal-water reaction involving the fuel cladding and the reactor coolant,

(2) Radiolytic decomposition of the reactor coolant, and

(3) Corrosion of metals.

This section does not apply to a nuclear power reactor facility for which the certifications required under §50.82(a)(1) have been submitted.

(b) Each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with the capability for—

(1) Measuring the hydrogen concentration in the containment,

(2) Insuring a mixed atmosphere in the containment, and

(3) Controlling combustible gas concentrations in the containment following a postulated LOCA.

(c)(1) For each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy or ZIRLO cladding, it must be shown that during the time period following a postulated LOCA, but prior to effective operation of the combustible gas control system, either:

(i) An uncontrolled hydrogen-oxygen recombination would not take place in the containment; or

(ii) The plant could withstand the consequences of uncontrolled hydrogen-oxygen recombination without loss of safety function.

(2) If the conditions set out in paragraph (c)(1) of this section cannot be shown, the containment shall be provided with an inerted or an oxygen deficient atmosphere in order to provide protection against hydrogen burning and explosions during the time period specified in paragraph (c)(1) of this section.

(3) Notwithstanding paragraphs (c)(1) and (c)(2) of this section:

(i) Effective May 4, 1982 or 6 months after initial criticality, whichever is later, an inerted atmosphere shall be provided for each boiling light-water nuclear power reactor with a Mark I or Mark II type containment; and

(ii) By the end of the first scheduled outage beginning after July 5, 1982 and of sufficient duration to permit required modifications, each light-water nuclear power reactor that relies upon a purge/repressurization system as the primary means for controlling combustible gases following a LOCA shall be provided with either an internal recombiner or the capability to install an external recombiner following the start of an accident. The internal or external recombiners must meet the combustible gas control requirements in paragraph (d) of this section. The containment penetrations used for external recombiners must either be:

(A) Dedicated to that service only, conform to the requirements of Criteria 54 and 56 of appendix A of this part, be designed against postulated single failures for containment isolation purposes, and be sized to satisfy the flow requirements of the external recombiners, or

(B) Of a combined design for use by either external recombiners or purge/repressurization systems and other systems, conform to the requirements of criteria 54 and 56 of appendix A of this part, be designed against postulated single failures both for containment isolation purposes and for operation of the external recombiners or purge/repressurization systems, and be sized to satisfy the flow requirements of the external recombiners or purge repressurization systems.

(iii) To provide improved operational capability to maintain adequate core

cooling following an accident, by the end of the first scheduled outage beginning after July 1, 1982 and of sufficient duration to permit required modifications, each light-water nuclear power reactor shall 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, however, for the tubes in U-tube steam generators.) The high point vents must be remotely operated from the control room. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents and associated controls, instruments and power sources must conform to the requirements of appendix A and appendix B of this part. In particular, the vent system shall be designed to ensure a low probability that (A) the vents will not perform their safety functions and (B) there would be inadvertent or irreversible actuation of a vent. Furthermore, the use of these vents during and following an accident must not aggravate the challenge to the containment or the course of the accident.

(iv)(A) Each licensee with a boiling light-water nuclear power reactor with a Mark III type of containment and each licensee with a pressurized light-water nuclear power reactor with an ice condenser type of containment issued a construction permit before March 28, 1979, shall provide its nuclear power reactor with a hydrogen control system justified by a suitable program of experiment and analysis. The hydrogen control system must be capable of handling without loss of containment structural integrity an amount of hydrogen equivalent to that generated from a metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume).

(B) Containment structural integrity must be demonstrated by use of an analytical technique that is accepted by the NRC staff. 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. Another method could include a showing that the following specific criteria of the ASME Boiler and Pressure Vessel Code are met:

(*1*) That steel containments meet the requirements of the ASME Boiler and Pressure Vessel Code (Edition and Addenda as incorporated by reference in § 50.55a(b)(1) of this part), specifically in Section III, Division 1, Subsubarticle NE–3220, Service Level C Limits, considering pressure and dead load alone (evaluation of instability is not required); and

(*2*) That concrete containments meet the requirements of the ASME Boiler and Pressure vessel Code, Section III, Division 2, Subsubarticle CC–3720, Factored Load Category, considering pressure and dead load alone.

(C) Subsubarticle NE–3220, Divisin 1, and Subsubarticle CC–3720, Division 2, of Section III of the ASME Boiler and Pressure Vessel Code, referenced in paragraphs (c)(3)(iv)(B)(*1*) and (c)(3)(iv)(B)(*2*) of this section, have been 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 Street, New York, N.Y. 10017. It is also available for inspection at the Nuclear Regulatory Commission's Public Document Room, 2120 L Street, NW., Washington, DC.

(D) If the hydrogen control system relies on post-accident inerting, the containment structure must be capable of withstanding the increased pressure:

(*1*) During the accident, where it is acceptable to show that it does not exceed Service Level C Limits or the Factored load Category (as described in paragraph (c)(3)(iv)(B) of this section; and

(*2*) Following inadvertent full inerting during normal plant operations, where it is acceptable to show that it does not exceed either the Service Level A Limits of Subsubarticle NE–3220 (for a steel containment) or the Service Load Category of Subsubarticle CC–3720 (for a concrete containment.)

(*3*) Modest deviations from the criteria in paragraph (c)(3)(iv)(D) of this section will be considered by the Commission if good cause is shown.

(E) If the hydrogen control system relies on post-accident inerting, the systems and components required to establish and maintain safe shutdown and containment integrity must be designed and qualified for the environment caused by such inerting. Furthermore, inadvertent full inerting during normal plant operations must not adversely affect systems and components needed for safe operation of the plant.

(v)(A) Each licensee with a boiling light-water nuclear power reactor with a Mark III type of containment and each licensee with a pressurized light-water nuclear power reactor with an ice condenser type of containment issued a construction permit before March 28, 1979, for a reactor that does not rely upon an inerted atmosphere to control hydrogen inside the containment, shall provide its nuclear power reactor with systems and components necessary to establish and maintain safe shutdown and to maintain containment integrity. These systems and components must be 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.

(B) The amount of hydrogen to be considered is equivalent to that generated from a metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume).

(vi)(A) Each applicant for or holder of an operating license for a boiling light-water nuclear power reactor with a Mark III type of containment or for a pressurized light-water nuclear power reactor with an ice condenser type of containment issued a construction permit before March 28, 1979, shall submit an analysis to the Commission as specified in § 50.4.

(B) The analysis required by paragraph (c)(3)(vi)(A) of this section must:

(*1*) Provide 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% 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;

(*2*) Include the period of recovery from the degraded condition;

(*3*) Use 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.

(*4*) Support the design of the hydrogen control system selected under paragraph (c)(3)(iv) of this section; and,

(*5*) Show that for those reactors described in paragraph (c)(3)(iv) of this section that do not rely upon an inerted atmosphere to control hydrogen inside the containment:

(*i*) The containment structural integrity as described in paragraph (c)(3)(iv) of this section will be maintained; and

(*ii*) 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 the effect of local detonations, unless such detonations can be shown unlikely to occur.

(vii)(A) By June 25, 1985, each applicant for or holder of an operating license subject to the requirements of paragraphs (c)(3) (iv), (v) and (vi) of this section shall develop and submit to the Commission a proposed schedule for meeting these requirements. The schedule may be developed using integrated scheduling systems previously approved for the facility by the NRC.

(B) For each applicant for an operating license as of Febuary 25, 1985, the schedule shall provide for compliance with the requirements of paragraph (c)(3)(iv)(A) of this section prior to operation of the reactor in excess of 5 percent power. Completed final analyses are not necessary for a staff determination that a plant is safe to operate at full power provided that prior to such operation an applicant has provided a preliminary analysis which the staff has determined provides a satisfactory basis for a decision to support interim operation at full power until the final analysis has been completed. However, the record in this rulemaking shows that such preliminary analyses are not necessary for a staff determination that a plant is safe to operate at full power if the staff has determined for similar plants, referenced in this notice of rulemaking, that similar systems provide a satisfactory basis for a decision to support operation at full power until the preliminary analyses have been completed.

(C) For those holders of operating licenses containing license conditions on Hydrogen Control Measures covered by this section, the schedule shall be consistent with those license conditions, or approved amendments thereto.

(D) For those facilities not having an NRC approved integrated scheduling system, a final schedule for meeting the requirements of paragraphs (c)(3)(iv), (v) and (vi) of this section shall be established by the NRC staff within 90 days of receipt of a proposed schedule from the licensee or applicant, taking into account the current status of efforts at the facility to comply with the requirements; analyses that may be provided by applicants or licensees regarding the impacts of these requirements on other scheduled plant modifications, including any NRC-mandated safety modifications, and their relative importance to safety; and the Commission's objective that these requirements be complied with without undue delay.

(d)(1) For facilities that are in compliance with § 50.46(b), the amount of hydrogen contributed by core metal-water reaction (percentage of fuel cladding that reacts with water), as a result of degradation, but not total failure, of emergency core cooling functioning shall be assumed either to be five times the total amount of hydrogen calculated in demonstrating compliance with § 50.46(b)(3), or to be the amount that would result from reaction of all the metal in the outside surfaces of the cladding cylinders surrounding the fuel (excluding the cladding surrounding the plenum volume) to a depth of 0.00023 inch (0.0058 mm), whichever amount is greater. A time period of 2 minutes shall be used as the interval after the postulated LOCA over which the metal-water reaction occurs.

(2) For facilities as to which no evaluation of compliance in accordance with § 50.46(b) has been submitted and evaluated, the amounts of hydrogen so contributed shall be assumed to be that amount resulting from the reaction of 5 percent of the mass of metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume.

(e) For facilities whose notice of hearing on the application for a construction permit was published on or after November 5, 1970, purging and/or repressurization shall not be the primary means for controlling combustible gases following a LOCA. However, the capability for controlled purging shall be provided. For these facilities, the primary means for controlling combustible gases following a LOCA shall consist of a combustible gas control system, such as recombiners, that does not result in a significant release from containment.

(f) For facilities with respect to which the notice of hearing on the application for a construction permit was published between December 22, 1968, and November 5, 1970, if the incremental radiation dose from purging (and repressurization if a repressurization system is provided) occurring at all points beyond the exclusion area boundary after a postulated LOCA calculated in accordance with § 100.11(a)(2) of this chapter is less than 2.5 rem to the whole body and less than 30 rem to the thyroid, and if the combined radiation dose at the low population zone outer boundary from purging and the postulated LOCA calculated in accordance with § 100.11(a)(2) of this chapter is

less than 25 rem to the whole body and less than 300 rem to the thyroid, only a purging system is necessary, provided that the purging system and any filtration system associated with it are designed to conform with the general requirements of Criteria 41, 42, and 43 of appendix A to this part. Otherwise the facility shall be provided with another type of combustible gas control system (a repressurization system is acceptable) designed to conform with the general requirements of Criteria 41, 42, and 43 of appendix A to this part. If a purge system is used as part of the repressurization system, the purge system shall be designed to conform with the general requirements of Criteria 41, 42, and 43 of appendix A to this part. The containment shall not be repressurized beyond 50 percent of the containment design pressure.

(g) For facilities with respect to which the notice of hearing on the application for a construction permit was published on or before December 22, 1968, if the combined radiation dose at the low population zone outer boundary from purging (and repressurization if a repressurization system is provided) and the postulated LOCA calculated in accordance with §100.11(a)(2) of this chapter is less than 25 rem to the whole body and less than 300 rem to the thyroid, only a purging system is necessary, provided that the purging system and any filtration system associated with it are designed to conform with the general requirements of Criteria 41, 42, and 43 of appendix A to this part. Otherwise, the facility shall be provided with another type of combustible gas control system (a repressurization system is acceptable) designed to conform with the general requirements of Criteria 41, 42, and 43 of appendix A to this part. If a purge system is used as part of the repressurization system, it shall be designed to conform with the general requirements of Criteria 41, 42, and 43 of appendix A to this part. The containment shall not be repressurized beyond 50 percent of the containment design pressure.

(h) As used in this section: (1) Degradation, but not total failure, of emergency core cooling functioning means that the performance of the emergency core cooling system is postulated, for purposes of design of the combustible gas control system, not to meet the acceptance criteria in §50.46 and that there could be localized clad melting and metal-water reaction to the extent postulated in paragraph (d) of this section. The degree of performance degradation is not postulated to be sufficient to cause core meltdown.

(2) A combustible gas control system is a system that operates after a LOCA to maintain the concentrations of combustible gases within the containment, such as hydrogen, below flammability limits. Combustible gas control systems are of two types: (i) Systems that allow controlled release from containment, through filters if necessary, such as purging systems and repressurization systems, and (ii) systems that do not result in a significant release from containment such as recombiners.

(3) A purging system is a system for the controlled release of the containment atmosphere to the environment through filters if needed.

(4) A repressurization system is a system used to dilute the concentration of combustible gas within containment by adding inert gas or air to the containment. Dilution of the combustible gas results in a delay in time until a flammable concentration is reached and permits fission product decay. Operation is limited to a containment repressurization to 50 percent of the containment design pressure. A purging system is normally part of the repressurization system.

[43 FR 50163, Oct. 27, 1978, as amended at 46 FR 58486, Dec. 2, 1981; 50 FR 3504, Jan. 25, 1985; 50 FR 5567, Feb. 11, 1985; 51 FR 40308, Nov. 6, 1986; 53 FR 43420, Oct. 27, 1988; 57 FR 39358, Aug. 31, 1992, 61 FR 39299, July 29, 1996]

§50.45 Standards for construction permits.

An applicant for a license or an amendment of a 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.

§ 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 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 licensee 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. If the change or error is signficant, 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.

(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]

§ 50.47 Emergency plans.

(a)(1) 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.

(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 near-site 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 have been developed for the plume exposure pathway EPZ for emergency workers and the public. 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 sectionresults wholly or substantially from the decision of stateand/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 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 emergencyplans 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 near-site 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.

[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]

§ 50.48 Fire protection.

(a) Each operating nuclear power plant must have a fire protection plan that satisfies Criterion 3 of appendix A of this part. This fire protection plan must describe the overall fire protection program for the facility, identify the various positions within the licensee's organization that are responsible for the program, state the authorities that are delegated to each of these positions to implement those responsibilities, and outline the plans for fire protection, fire detection and suppression capability, and limitation of fire damage. The plan must also describe specific features necessary to implement the program described above, such as administrative controls and personnel requirements for fire prevention and manual fire suppression activities, automatic and manually operated fire detection and suppression systems, and the means to limit fire damage to structures, systems, or components important to safety so that the capability to safely shut down the plant is ensured.[3] The licensee shall retain the

[3] Basic fire protection guidance for nuclear power plants is contained in two NRC documents:

• Branch Technical Position Auxiliary Power Conversion System Branch BTP APCSB 9.5–1, ''Guidelines for Fire Protection for Nuclear Power Plants,'' for new plants docketed after July 1, 1976, dated May 1976.

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

(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 prior to January 1, 1979. Except for the requirements of sections III.G, III.J, and III.O, the provisions of appendix R to this part shall not be applicable to nuclear power plants licensed to operate prior to January 1, 1979, to the extent that 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[4]reflected in staff fire protection safety evaluation reports issued prior to the effective date of this rule, or to the extent that fire protection features were accepted by the 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. With respect to all other fire protection features covered by appendix R, all nuclear power plants licensed to operate prior to January 1, 1979 shall satisfy the applicable requirements of appendix R to this part, including specifically the requirements of sections III.G, III.J, and III.O.

(c) All fire protection modifications required to satisfy the provisions of appendix R to this part or directly affected by such requirements shall be completed on the following schedule:

(1) Those fire protection features that involve revisions of administrative controls, manpower changes, and training, shall be implemented within 30 days after the effective date of this section and appendix R to this part.

(2) Those fire protection features that involve installation of modifications that do not require prior NRC approval or plant shutdown shall be implemented within 9 months after the effective date of this section and appendix R to this part.

(3) Those fire protection features, except for those requiring prior NRC approval by paragraph (c)(5) of this section, that involve installation of modifications that do require plant shutdown, the need for which is justified in the plans and schedules required by the provisions of paragraph (c)(5) of this section, shall be implemented before startup after the earliest of the following events commencing 180 days or more after the effective date of this section and appendix R to this part:

(i) The first refueling outage;

(ii) Another planned outage that lasts for at least 60 days; or

(iii) An unplanned outage that lasts for at least 120 days.

(4) Those fire protection features that require prior NRC approval by paragraph (c)(5) of this section, shall be implemented within the following schedule: Dedicated shutdown systems—30 months after NRC approval; modifications requiring plant shutdown—before startup after the earliest of the events given in paragraph (c)(3) commencing 180 days after NRC approval; modifications not requiring plant shutdown—6 months after NRC approval.

(5) Licensees shall make any modifications necessary to comply with

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• Appendix A to BTP APCSB 9.5–1, ''Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976,'' for plants that were operating or under various stages of design or construction before July 1, 1976, dated August 23, 1976.

Also see Note 4.

[4]Clarification and guidance with respect to permissible alternatives to satisfy Appendix A to BTP APCSB 9.5–1 has been provided in four other NRC documents.

''Supplementary Guidance on Information Needed for Fire Protection Evaluation,'' dated October 21, 1976.

''Sample Technical Specification,'' dated May 12, 1977.

''Nuclear Plant Fire Protection Functional Responsibilities, Administrative Control and Quality Assurance,'' dated June 14, 1977.

''Manpower Requirements for Operating Reactors,'' dated May 11, 1978.

A Fire Protection Safety Evaluation Report that has been issued for each operating plant states how these guidelines were applied to each facility and identifies open fire protection issues that will be resolved when the facility satisfies the appropriate requirements of appendix R to this part.

these requirements in accordance with the above schedule without prior review and approval by NRC except for modifications required by section III.G.3 of appendix R to this part. Licensees shall submit plans and schedules for meeting the provisions of paragraphs (c)(2), (c)(3), and (c)(4) within 30 days after the effective date of this section and appendix R to this part. Licensees shall submit design descriptions of modifications needed to satisfy section III.G.3 of appendix R to this part within 30 days after the the effective date of this section and appendix R to this part.

(6) In the event that a request for exemption from a requirement to comply with one or more of the provisions of appendix R filed within 30 days of the effective date of this rule is based on an assertion by the licensee that such required modifications would not enhance fire protection safety in the facility or that such modifications may be detrimental to overall facility safety, the schedule requirements of paragraph (c) shall be tolled until final Commission action on the exemption request upon a determination by the Director of Nuclear Reactor Regulation that the licensee has provided a sound technical basis for such assertion that warrants further staff review of the request.

(d) Fire protection features accepted by the NRC staff in Fire Protection Safety Evaluation Reports referred to in paragraph (b) of this section and supplements to such reports, other than features covered by paragraph (c), shall be completed as soon as practicable but no later than the completion date currently specified in license conditions or technical specifications for such facility, or the date determined by paragraphs (d)(1) through (d)(4) of this section, whichever is sooner, unless the Director of Nuclear Reactor Regulation determines, upon a showing by the licensee, that there is good cause for extending such date and that the public health and safety is not adversely affected by such extension. Extensions of such date shall not exceed the dates determined by paragraphs (c)(1) through (c)(4) of this section.

(1) Those fire protection features that involve revisions of administrative controls, manpower changes, and training shall be implemented within 4 months after the date of the NRC staff Fire Protection Evaluation Report accepting or requiring such features.

(2) Those fire protection features involving installation of modifications not requiring prior approval or plant shutdown shall be implemented within 12 months after the date of the NRC staff Fire Protection Safety Evaluation Report accepting or requiring such features.

(3) Those fire protection features, including alternative shutdown capability, involving installation of modifications requiring plant shutdown shall be implemented before the start-up after the earliest of the following events commencing 9 months or more after the date of the NRC staff Fire Protection Safety Evaluation Report accepting or requiring such features:

(i) The first refueling outage;

(ii) Another planned outage that lasts for at least 60 days; or

(iii) An unplanned outage that lasts for at least 120 days.

(4) Those fire protection features involving dedicated shutdown capability requiring new buildings and systems shall be implemented within 30 months of NRC approval. Other modifications requiring NRC approval prior to installation shall be implemented within 6 months after NRC approval.

(e) Nuclear power plants licensed to operate after January 1, 1979, shall complete all fire protection modifications needed to satisfy Criterion 3 of appendix A to this part in accordance with the provisions of their licenses.

(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 which could cause the release or spread of radioactive materials (i.e., which could result in a radiological hazard).

(1) The objectives of the fire protection program are to—

(i) Reasonably prevent such fires from occurring;

(ii) Rapidly detect, control, and extinguish those fires which do occur and

which 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 fire protection program must be assessed by the licensee on a regular basis and revised 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 which could result in a radiological hazard, taking into account the decommissioning plant conditions and activities.

[45 FR 76610, Nov. 19, 1980, as amended at 53 FR 19250, May 27, 1988; 61 FR 39300, July 29, 1996]

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

(a) Each holder of or an applicant for a license for a nuclear power plant, other than a nuclear power plant for which the certifications required under §50.82(a)(1) have been submitted, shall establish a program for qualifying the electric equipment defined in paragraph (b) of this section.

(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;

(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) 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 (i) through (iii) of paragraph (b)(1) 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

[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.

[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 Printing Office by calling 202–275–2060 or by writing to the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013–7082.

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 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]

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.

Whether stated therein or not, the following shall be deemed conditions in every license issued:

(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, pursuant to § 50.34(b)(6)(ii) of this part, the quality assurance program described or referenced in the Safety Analysis Report, including changes to that report.

(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) After March 11, 1983, 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, 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 the NRC in accordance with the requirements of § 50.71(e). Changes to the quality assurance program description that do reduce the commitments must be submitted to NRC and receive NRC approval prior to implementation, as follows:

(i) Changes made to the Safety Analysis Report must be submitted, 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.

(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 three months after issuance of an operating license, the licensee shall have in effect an operator requalification program which 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) 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 effecting the actions and decisions contained in the Responsibility Matrix of the safeguards contingency plan. The licensee may make no change which would decrease the effectiveness of a security plan, or guard training and qualification plan, prepared pursuant to § 50.34(c) 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 pursuant to § 50.34(d) 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 an amendment to the licensee's license pursuant to § 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 three years from the date of the change, and shall submit, as specified in § 50.4, a report containing a description of each change within two 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 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 maintanence of its safeguards contingency plan. To this end, the licensee shall provide for a review at least every 12 months of the safeguards contingency plan by individuals independent of both security program management and personnel who have direct responsibility for implementation of the security program. 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 two years.

(q) A licensee authorized to possess and operate a nuclear power reactor shall follow and maintain in effect emergency plans which meet the standards in § 50.47(b) and the requirements in appendix E of this part. A licensee authorized to possess and/or operate a research reactor or a fuel facility shall follow and maintain in effect emergency plans which meet the requirements in appendix E to this part. The licensee shall retain the emergency plan and each change that decreases the effectiveness of the plan as a record until the Commission terminates the license for the nuclear power reactor. The nuclear power reactor licensee may make changes to these plans without Commission approval only if the changes do not decrease the effectiveness of the plans and the plans, as changed, continue to meet the standards of § 50.47(b) and the requirements of appendix E to this part. The research reactor and/or the fuel facility licensee may make changes to these plans without Commission approval only if these changes do not decrease the effectiveness of the plans and the plans, as changed, continue to meet the requirements of appendix E to this part. This nuclear power reactor, research reactor, or fuel facility licensee shall retain a record of each change to the emergency plan made without prior Commission approval for a period of three years from the date of the change. Proposed changes that decrease the effectiveness of the approved emergency plans may not be implemented without application to and approval by the Commission. The licensee shall submit, as specified in § 50.4, a report of each proposed change for approval. If a change is made without approval, the licensee shall submit, as specified in § 50.4, a report of each change within 30 days after the change is made.

(r) Each licensee who is authorized to possess and/or operate a research or test reactor facility with an authorized power level greater than or equal to 2 MW thermal, under a licensee of the type specified in § 50.21(c), shall submit emergency plans complying with 10 CFR part 50, appendix E, to the Director of the Office of Nuclear Reactor Regulation for approval by September 7, 1982. Each licensee who is authorized to possess and/or operate a research or test reactor facility with an authorized power level less than 2 MW thermal, under a license of the type specified in § 50.21(c), shall submit emergency plans complying with 10 CFR part 50, appendix E, to the Director of the Office of Nuclear Reactor Regulation for approval by November 3, 1982.

(s)(1) Each licensee who is authorized to possess and/or operate a nuclear power reactor shall submit to NRC within 60 days of the effective date of this amendment the radiological emergency response plans of State and local governmental entities in the United States that are wholly or partially within a plume exposure pathway EPZ, as well as the plans of State governments wholly or partially within an ingestion pathway EPZ.[1,2] Ten (10) copies of the above plans shall be forwarded to the Director of Nuclear Reactor Regulation with 3 copies to the Administrator of the appropriate NRC regional office. 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 for 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 EPZ shall focus on such actions as are

[1] 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.

[2] 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.

appropriate to protect the food ingestion pathway.

(2)(i) For operating power reactors, the licensee, State, and local emergency response plans shall be implemented by April 1, 1981, except as provided in section IV.D.3 of appendix E to this part.

(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 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) A nuclear power reactor licensee shall provide for the development, revision, implementation, and maintenance of its emergency preparedness program. To this end, the licensee shall provide for a review of its emergency preparedness program at least every 12 months by persons who have no direct responsibility for implementation of the emergency preparedness program. The review shall 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, shall be documented, reported to the licensee's corporate and plant management, and retained for a period of five years. The part of the review involving the evaluation for adequacy of interface with State and local governments shall be available to the appropriate State and local governments.

(u) Within 60 days after the effective date of this amendment, each nuclear power reactor licensee shall submit to the NRC plans for coping with emergencies that meet standards in § 50.47(b) and the requirements of appendix E to this part.

(v) Each licensee subject to the requirements of part 73 of this chapter shall ensure that physical security, safeguards contingency and guard qualification and training plans and other related Safeguards Information are protected against unauthorized disclosure in accordance with the requirements of § 73.21 of this chapter, as appropriate.

(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.

[21 FR 355, Jan. 19, 1956]

EDITORIAL NOTE: For additional FEDERAL REGISTER citations affecting § 50.54, see the List of CFR Sections Affected in the Finding Aids section of this volume.

§50.55 Conditions of construction permits.

Each construction permit shall be subject to the following terms and conditions:

(a) The 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 permit shall expire and all rights thereunder shall be forfeited: *Provided, however,* That 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) Each individual, corporation, partnership, or other entity holding a facility construction permit subject to this part 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)(1)(ii) of this section, in all cases within 60 days of discovery, in order 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)(7) 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 is informed as soon as practicable, and, in all cases, within the 5 working days after completion of the evaluation described in paragraph (e)(1)(i) or (e)(1)(ii) of this section, if the construction of a facility or activity, or a basic component supplied for such facility or activity—

(A) Fails to comply with the Atomic Energy Act of 1954, as amended, or any applicable rule, 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 pursuant to the requirements of appendix B to 10 CFR part 50 which could have produced a defect in a basic component. Such 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 or installation.

(2) The holder of a facility construction permit subject to this part who obtains information reasonably indicating that the facility fails to comply with the Atomic Energy Act of 1954, as amended, or any applicable rule, 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)(7) of this section.

(3) The holder of a facility construction permit subject to this part who obtains information reasonably indicating the existence of any defect found in construction or any defect found in the final design of a facility as approved and released for construction must notify the Commission of the defect through a director or responsible officer or designated person as discussed in paragraph (e)(7) of this section.

(4) The holder of a facility construction permit subject to this part who obtains information reasonably indicating that the quality assurance program has undergone any significant breakdown discussed in paragraph (e)(1)(ii)(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)(7) of this section.

(5) The notification requirements of paragraphs (e)(2), (e)(3), and (e)(4) of this section 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 or construction permit. 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 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 of this chapter to report defects, and failures to comply associated with substantial safety hazards under section 206 of the Energy Reorganization Act of 1974.

(6) The notification required by paragraphs (e)(2), (e)(3), and (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 two days following receipt of information by the director or responsible corporate officer under paragraph (e)(1)(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)(1)(ii).

(ii) Written notification submitted to the Document Control Desk, U.S. Nuclear Regulatory Commission, Washington, DC 20555, 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)(1)(iii) of this section, on the identification of a defect or failure to comply.

(7) 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.

(8) The written notification required by paragraph (e)(6)(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 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 such defect or failure to comply.

(v) The date on which the information of such 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 components in use at the facility subject to the regulations in this part.

(vii) 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.

(viii) 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.

(9) The holder of a construction permit 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 basic component.

(ii) Retain evaluations of all deviations and failures to comply for a minimum of five years.

(iii) Maintaining records in accordance with this section satisfies the construction permit holders recordkeeping obligations under part 21 of this chapter. The recordkeeping obligations of responsible officers and directors under part 21 of this chapter are met by recordkeeping in accordance with this section.

(10) The requirements of this § 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 or under § 73.71 of this chapter under § 50.55(e) or § 50.73 of this part. For holders of construction permits issued prior to 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.

(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.

[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]

§ 50.55a Codes and standards.

Each operating license for a boiling or pressurized water-cooled nuclear power facility is subject to the conditions in paragraphs (f) and (g) of this section and each construction permit for a utilization facility is subject to the following conditions in addition to those specified in § 50.55.

(a)(1) Structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

(2) Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME Boiler and Pressure Vessel Code specified in paragraphs (b), (c), (d), (e), (f), and (g) of this section. Protection systems of nuclear power reactors of all types must meet the requirements specified in paragraph (h) of this section.

(3) Proposed alternatives to the requirements of paragraphs (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant shall demonstrate that:

(i) The proposed alternatives would provide an acceptable level of quality and safety, or

(ii) 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.

(b) The ASME Boiler and Pressure Vessel Code, which is referenced in the following paragraphs, was 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 St., New York, NY 10017. It is also available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852–2738.

(1) As used in this section, references to section III of the ASME Boiler and Pressure Vessel Code refer to section III, Division 1, and include addenda through the 1988 Addenda and editions through the 1989 Edition.

(2) As used in this section, references to Section XI of the ASME Boiler and Pressure Vessel Code refer to Class 1, Class 2, and Class 3 components of Section XI, Division 1, and include addenda through the 1988 Addenda and editions through the 1989 Edition, and Class MC and Class CC components of Section XI, Division 1, 1992 Edition with the 1992 Addenda, subject to the following limitations and modifications:

(i) *Limitations on specific editions and addenda.* When applying the 1974 Edition, only the addenda through the Summer 1975 Addenda may be used. When applying the 1977 Edition, all of the addenda through the Summer 1978 Addenda must also be used. Addenda and editions subsequent to the Summer 1978 Addenda, that are incorporated by reference in paragraph (b)(2) of this section are not affected by these limitations.

(ii) *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 Code in the 1974 Edition and addenda through the Summer 1975 Addenda or other requirements the Commission may adopt.

(iii) *Steam generator tubing (modifies Article IWB–2000).* If the technical specifications of a nuclear power plant include surveillance requirements for steam generators different than those in Article IWB–2000, the inservice inspection program for steam generator tubing is governed by the requirements in the technical specifications.

(iv) *Pressure-retaining welds in ASME Code Class 2 piping (applies to Tables IWC–2520 or IWC–2520–1, Category C–F).* (A) Appropriate Code Class 2 pipe welds in Residual Heat Removal Systems, Emergency Core Cooling Systems, and Containment Heat Removal Systems, must be examined. When applying editions and addenda up to the 1983 Edition through the Summer 1983 Addenda of section XI of the ASME Code, the extent of examination for these systems must be determined by the requirements of paragraph IWC–1220, Table IWC–2520 Category C–F and C–G, and paragraph IWC–2411 in the 1974 Edition and Addenda through the Summer 1975 Addenda.

(B) For a nuclear power plant whose application for a construction permit was docketed prior to July 1, 1978, when applying editions and addenda up to the 1983 Edition through the Summer 1983 Addenda of section XI of the ASME Code, the extent of examination for Code Class 2 pipe welds may be determined by the requirements of paragraph IWC–1220, Table IWC–2520 Category C–F and C–G and paragraph IWC–2411 in the 1974 Edition and Addenda through the Summer 1975 Addenda of Section XI of the ASME Code or other requirements the Commission may adopt.

(v) Evaluation procedures and acceptance criteria for austenitic piping (applies to IWB–3640). When applying the Winter 1983 Addenda and Winter 1984 Addenda, the rules of paragraph IWB–3640 may be used for all applications permitted in that paragraph, except those associated with submerged arc welds (SAW) or shielded metal arc welds (SMAW). For SAW or SMAW, use paragraph IWB–3640, as modified by the Winter 1985 Addenda.

(vi) *Effective edition and addenda of Subsection IWE and Subsection IWL, Section XI.* The 1992 Edition with the 1992 Addenda of Subsection IWE and Subsection IWL shall be used by licensees when performing containment examinations as modified and supplemented by the requirements in § 50.55a(b)(2)(ix) and § 50.55a(b)(2)(x).

(vii) *Inservice testing of containment isolation valves.* When using subsection IWV in the 1988 Addenda or the 1989 Edition of section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, leakage rates for Category A containment isolation valves that do not provide a reactor coolant system pressure isolation function must be analyzed in accordance with paragraph 4.2.2.3(e) of part 10, and corrective actions for these valves must be made in accordance with paragraph 4.2.2.3(f) of part 10 of ASME/ANSI OMa–1988 Addenda to ASME/ANSI OM–1987.

(viii) *Section XI References to OM Part 4, OM Part 6 and OM Part 10 (Table IWA–1600–1).* When using Table IWA–1600–1, ''Referenced Standards and Specifications'' in the section XI, Division 1, 1987 Addenda, 1988 Addenda, or 1989 Edition, the specified ''Revision Date or Indicator'' for ASME/ANSI OM Part 4, ASME/ANSI part 6, and ASME/ANSI part 10 shall be the OMa–1988 Addenda to the OM–1987 Edition.

(ix) *Examination of concrete containments.* (A) Grease caps that are accessible must be visually examined to detect grease leakage or grease cap deformations. Grease caps must be removed for this examination when there is evidence of grease cap deformation that indicates deterioration of anchorage hardware.

(B) When evaluation of consecutive surveillances of prestressing forces for the same tendon or tendons in a group indicates a trend of prestress loss such that the tendon force(s) would be less than the minimum design prestress requirements before the next inspection interval, an evaluation shall be performed and reported in the Engineering

Evaluation Report as prescribed in IWL–3300.

(C) When the elongation corresponding to a specific load (adjusted for effective wires or strands) during retensioning of tendons differs by more than 10 percent from that recorded during the last measurement, an evaluation must be performed to determine whether the difference is related to wire failures or slip of wires in anchorages. A difference of more than 10 percent must be identified in the ISI Summary Report required by IWA–6000.

(D) The licensee shall report the following conditions, if they occur, in the ISI Summary Report required by IWA–6000:

(*1*) The sampled sheathing filler grease contains chemically combined water exceeding 10 percent by weight or the presence of free water;

(*2*) The absolute difference between the amount removed and the amount replaced exceeds 10 percent of the tendon net duct volume.

(*3*) Grease leakage is detected during general visual examination of the containment surface.

(E) For Class CC applications, the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, the licensee shall 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.

(x) *Examination of metal containments and the liners of concrete containments.* (A) For Class MC applications, the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, the licensee shall 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.

(B) When performing remotely the visual examinations required by Subsection IWE, the maximum direct examination distance specified in Table IWA–2210–1 may be extended and the minimum illumination requirements specified in Table IWA–2210–1 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) The examinations specified in Examination Category E–B, Pressure Retaining Welds, and Examination Category E–F, Pressure Retaining Dissimilar Metal Welds, are optional.

(D) Section 50.55a(b)(2)(x)(D) may be used as an alternative to the requirements of IWE–2430.

(*1*) If the examinations reveal flaws or areas of degradation exceeding the acceptance standards of Table IWE–3410–1, an evaluation shall be performed to determine whether additional component examinations are required. For each flaw or area of degradation identified which exceeds acceptance standards, the licensee shall provide the following in the ISI Summary Report required by IWA–6000:

(*i*) A description of each flaw or area, including the extent of degradation, and the conditions that led to the degradation;

(*ii*) The acceptability of each flaw or area, and the need for additional examinations to verify that similar degradation does not exist in similar components, and;

(*iii*) A description of necessary corrective actions.

(*2*) The number and type of additional examinations to ensure detection of similar degradation in similar components.

(E) A general visual examination as required by Subsection IWE shall be performed once each period.

(c) *Reactor coolant pressure boundary.* (1) Components which are part of the reactor coolant pressure boundary

must meet the requirements for Class 1 components in Section III[4,5] of the ASME Boiler and Pressure Vessel Code, except as provided in paragraphs (c)(2), (c)(3), and (c)(4) of this section.

(2) Components which 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:*

(i) 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) 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) The Code Edition, Addenda, and optional Code Cases[6] 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 Boiler and Pressure Vessel Code, but (i) the edition and addenda applied to a component must be those which are incorporated by reference in paragraph (b)(1) of this section, (ii) the ASME Code provisions applied to the pressure vessel may be dated no earlier than the Summer 1972 Addenda of the 1971 edition, (iii) 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) ASME Code Cases[thnsp;6] must have been determined suitable for use by the NRC.

(4) 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 component at the time of issuance of the construction permit.

(d) *Quality Group B components.* (1) For a nuclear power plant whose application for a construction permit is docketed after May 14, 1984 components classified Quality Group B[9] must meet the requirements for Class 2 Components in Section III of the ASME Boiler and Pressure Vessel Code.

(2) The Code Edition, Addenda, and optional Code Cases[6] 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 Boiler Vessel and Pressure Code, but (i) the edition and addenda must be those which are incorporated by reference in paragraph (b)(1) of this section, (ii) the ASME Code provisions applied to the systems and components may be dated no earlier than the 1980 Edition, and (iii) the ASME Code Cases[6] must have been determined suitable for use by the NRC.

(e) *Quality Group C components.* (1) For a nuclear power plant whose application for a construction permit is docketed after May 14, 1984 components classified Quality Group C[9] must meet the requirements for Class 3 components in Section III of the ASME Boiler and Pressure Vessel Code.

(2) The Code Edition, Addenda, and optional Code Cases[6] 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 Boiler and Pressure Vessel Code, but (i) the edition and addenda must be those which are incorporated by reference in paragraph (b)(1) of this section, (ii) the ASME Code provisions applied to the systems and components may be dated no earlier than the 1980 Edition, and (iii) the ASME Code Cases[6] must have been determined suitable for use by the NRC.

(f) *Inservice testing requirements.* (1) For a boiling or pressurized water-cooled nuclear power facility whose

See footnotes at end of section.

construction permit was issued prior to January 1, 1971, pumps and valves must meet the test requirement of paragraphs (f) (4) and (5) of this section to the extent practical. Pumps and valves which are part of the reactor coolant pressure boundary must meet the requirements applicable to components which are classified as ASME Code Class 1. Other safety-related pumps and valves must meet the requirements applicable to components which are classified as ASME Code Class 2 or Class 3.

(2) 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 which are classified as ASME Code Class 1 and Class 2 must be designed and be provided with access to enable the performance of inservice tests for operational readiness set forth in editions of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda[6] in effect 6 months prior to 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 which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed therein.

(3) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after July 1, 1974:

(i)–(ii) [Reserved]

(iii) Pumps and valves which are classified as ASME Code Class 1 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 section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda[6] applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

(iv) Pumps and valves which are classified as ASME 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 section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda[6] applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

(v) All pumps and valves may meet the test requirements set forth in subsequent editions of codes and addenda or portions thereof which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed in paragraph (b) of this section.

(4) Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) which 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 of the ASME Boiler and Pressure Vessel Code and Addenda that become effective subsequent to editions specified in paragraphs (g)(2) and (g)(3) of this section and that are incorporated by reference in paragraph (b) of this section, to the extent practical within the limitations of design, geometry and materials of construction of the components.

(i) 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 Code incorporated by reference in paragraph (b) of this section on the date 12 months prior to the date of issuance of the operating license, subject to the limitations and modifications listed in paragraph (b) of this section.

(ii) 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 Code incorporated by reference in paragraph (b) of this section 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed in paragraph (b) of this section.

(iii) [Reserved]

(iv) Inservice tests of pumps and valves may meet the requirements set

forth in subsequent editions and addenda that are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications 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.

(5)(i) 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) If a revised inservice test program for a facility conflicts with the technical specification for the facility, the licensee shall apply to the Commission for amendment of the technical specifications to conform the technical specification to the revised program. The licensee shall 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) If the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit, as specified in § 50.4, information to support the determination.

(iv) 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 demonstrated to the satisfaction of the Commission not later than 12 months after the expiration of the initial 120-month period of operation from start of facility commercial operation and each subsequent 120-month period of operation during which the test is determined to be impractical.

(6)(i) 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 is authorized by law and will not endanger life or property or the common defense and security and is 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) 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.

(g) *Inservice inspection requirements.* Requirements for inservice testing of Class 1, Class 2, and Class 3 pumps and valves are located in § 50.55a(f).

(1) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued prior to January 1, 1971, components (including supports) must meet the requirements of paragraphs (g) (4) and (5) of this section to the extent practical. Components which are part of the reactor coolant pressure boundary and their supports must meet the requirements applicable to components which are classified as ASME Code Class 1. Other safety-related pressure vessels, piping, pumps and valves must meet the requirements applicable to components which are classified as ASME Code Class 2 or Class 3.

(2) 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 (including supports) which are classified as ASME Code Class 1 and Class 2 must be designed and be provided with access to enable the performance of inservice examination of such components (including supports) and must meet the preservice examination requirements set forth in editions of section XI of the ASME Boiler and Pressure Vessel Code and Addenda[6] in effect six months prior to the date of issuance of the construction permit. The components (including supports) may meet the requirements set forth in subsequent editions of this code and addenda which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed in paragraph (b) of this section.

(3) For a boiling or pressurized water-cooled nuclear power facility whose

construction permit was issued on or after July 1, 1974:

(i) Components which are classified as ASME Code Class 1 must be designed and be provided with access to enable the performance of inservice examination of such components and must meet the preservice examination requirements set forth in section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda[6] applied to the construction of the particular component.

(ii) Components which are classified as ASME Code Class 2 and Class 3 and supports for components which are classified as ASME Code Class 1, Class 2, and Class 3 must be designed and be provided with access to enable the performance of inservice examination of such components and must meet the preservice examination requirements set forth in section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda[6] applied to the construction of the particular component.

(iii)–(iv) [Reserved]

(v) All components (including supports) may meet the requirements set forth in subsequent editions of codes and addenda or portions thereof which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed therein.

(4) Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) which 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 of the ASME Boiler and Pressure Vessel Code and Addenda that become effective subsequent to editions specified in paragraphs (g)(2) and (g)(3) of this section and that are incorporated by reference in paragraph (b) of this section, to the extent practical within the limitations of design, geometry and materials of construction of the components. Components which are classified as Class MC pressure retaining components and their integral attachments, and components which 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 Boiler and Pressure Vessel Code and Addenda that are incorporated by reference in paragraph (b) of this section, subject to the limitation listed in paragraph (b)(2)(vi) and the modifications listed in paragraphs (b)(2)(ix) and (b)(2)(x) of this section, to the extent practical within the limitations of design, geometry and materials of construction of the components.

(i) Inservice examinations 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 Code incorporated by reference in paragraph (b) of this section on the date 12 months prior to the date of issuance of the operating license, subject to the limitations and modifications listed in paragraph (b) of this section.

(ii) 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 Code incorporated by reference in paragraph (b) of this section 12 months prior to the start of the 120-month inspection interval, subject to the limitations and modifications listed in paragraph (b) of this section.

(iii) [Reserved]

(iv) 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 (b) of this section, subject to the limitations and modifications 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) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued after January 1, 1956:

(A) Metal containment pressure retaining components and their integral attachments must meet the inservice

inspection, repair, and replacement requirements applicable to components which are classified as ASME Code Class MC;

(B) Metallic shell and penetration liners which are pressure retaining components and their integral attachments in concrete containments must meet the inservice inspection, repair, and replacement requirements applicable to components which are classified as ASME Code Class MC; and

(C) Concrete containment pressure retaining components and their integral attachments, and the post-tensioning systems of concrete containments must meet the inservice inspection and repair requirements applicable to components which are classified as ASME Code Class CC.

(5)(i) 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) If a revised inservice inspection program for a facility conflicts with the technical specification for the facility, the licensee shall apply to the Commission for amendment of the technical specifications to conform the technical specification to the revised program. The licensee shall 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) If the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit, as specified in § 50.4, information to support the determinations.

(iv) Where an examination requirement by the code or addenda is determined to be impractical by the licensee and is not included in the revised inservice inspection program as permitted by paragraph (g)(4) of this section, the basis for this determination must be demonstrated to the satisfaction of the Commission not later than 12 months after the expiration of the initial 120-month period of operation from start of facility commercial operation and each subsequent 120-month period of operation during which the examination is determined to be impractical.

(6)(i) 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 is authorized by law and will not endanger life or property or the common defense and security and is 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) 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) Augmented examination of reactor vessel.

(*1*) All previously granted reliefs under § 50.55a to licensees for the extent of volumetric examination of reactor vessel shell welds specified in Item B1.10 of Examination Category B–A, ''Pressure Retaining Welds in Reactor Vessel,'' in Table IWB–2500–1 of subsection IWB in applicable edition and addenda of section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, during the inservice inspection interval in effect on September 8, 1992 are hereby revoked, subject to the specific modification in § 50.55a(g)(6)(ii)(A)*(3)(iv)* for licensees that defer the augmented examination in accordance with § 50.55a(g)(6)(ii)(A)*(3)*.

(*2*) All licensees shall augment their reactor vessel examination by implementing once, as part of the inservice inspection interval in effect on September 8, 1992, the examination requirements for reactor vessel shell welds specified in Item B1.10 of Examination Category B–A, ''Pressure Retaining Welds in Reactor Vessel,'' in Table IWB–2500–1 of subsection IWB of the 1989 Edition of section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, subject to the conditions specified in § 50.55a(g)(6)(ii)(A) (*3*) and

See footnotes at end of section.

(*4*). The augmented examination, when not deferred in accordance with the provisions of § 50.55a(g)(6)(ii)(A)(*3*), shall be performed in accordance with the related procedures specified in the section XI edition and addenda applicable to the inservice inspection interval in effect on September 8, 1992, and may be used as a substitute for the reactor vessel shell weld examination scheduled for implementation during the inservice inspection interval in effect on September 8, 1992. For the purpose of this augmented examination, ''essentially 100% as used in Table IWB–2500–1 means more than 90 percent of the examination volume of each weld, where the reduction in coverage is due to interference by another component, or part geometry.

(*3*) Licensees with fewer than 40 months remaining in the inservice inspection interval in effect on September 8, 1992 may defer the augmented reactor vessel examination specified in § 50.55a(g)(6)(ii)(A)(*2*) to the first period of the next inspection interval under the following conditions:

(*i*) The deferred augmented examination may not be used as a substitute for the reactor vessel shell weld examination scheduled for implementation during the inservice inspection interval in effect on September 8, 1992.

(*ii*) The deferred augmented examination may be used as a substitute for the reactor vessel shell weld examination normally scheduled for the inspection interval in which the deferred examination is performed.

(*iii*) If the deferred augmented examination is used as a substitute for the normally scheduled reactor vessel shell weld examination, subsequent reactor vessel shell weld examinations must be performed during the first period of successive inspection intervals.

(*iv*) Licensees that defer the augmented examination, as permitted herein, may retain all previously granted reliefs that otherwise would be revoked by § 50.55a(g)(6)(ii)(A)(*1*) for the inservice inspection interval in effect on September 8, 1992.

(*v*) Licensees with fewer than 40 months remaining in the inservice inspection interval in effect on September 8, 1992 may extend that interval in accordance with the provisions of section XI (1989 Edition) IWA–2430(d) for the purpose of implementing the augmented examination during that interval.

(*vi*) The deferred augmented examination shall be performed in accordance with the related procedures specified in the section XI edition and addenda applicable to the inspection interval in which the augmented examination is performed.

(*4*) The requirement for augmented examination of the reactor vessel may be satisfied by an examination of essentially 100 percent of the reactor vessel shell welds specified in § 50.55a(g)(6)(ii)(A)(*2*) that has been completed, or is scheduled for implementation with a written commitment, or is required by § 50.55a(g)(4)(i), during the inservice inspection interval in effect on September 8, 1992.

(*5*) Licensees that make a determination that they are unable to completely satisfy the requirements for the augmented reactor vessel shell weld examination specified in § 50.55a(g)(6)(ii)(A) shall submit information to the Commission to support the determination and shall propose an alternative to the examination requirements that would provide an acceptable level of quality and safety. The licensee may use the proposed alternative when authorized by the Director of the Office of Nuclear Reactor Regulation.

(B) *Expedited examination of containment.* (*1*) Licensees of all operating nuclear power plants shall implement the inservice examinations specified for the first period of the first inspection interval in Subsection IWE of the 1992 Edition with the 1992 Addenda in conjunction with the modifications specified in § 50.55a (b)(2)(ix) by September 9, 2001. The examination performed during the first period of the first inspection interval shall serve the same purpose for operating plants as the preservice examination specified for plants not yet in operation.

(*2*) Licensees of all operating nuclear power plants shall implement the inservice examinations which correspond to the number of years of operation which are specified in Subsection IWL

of the 1992 Edition with the 1992 Addenda in conjunction with the modifications specified in § 50.55a (b)(2)(ix) by September 9, 2001. The first examination performed shall serve the same purpose for operating plants as the preservice examination specified for plants not yet in operation.

(*3*) The expedited examination for Class MC components may be used to satisfy the requirements of routinely scheduled examinations of Subsection IWE subject to IWA–2430(d) when the expedited examination occurs during the first containment inspection interval.

(*4*) The requirement for the expedited examination of the containment post-tensioning system may be satisfied by the post-tensioning system examinations performed after September 9, 1996 as a result of licensee post-tensioning system programs accepted by the NRC prior to September 9, 1996.

(*5*) Licensees do not have to submit to the NRC staff for approval of their containment inservice inspection program which was developed to satisfy the requirements of Subsection IWE and Subsection IWL with specified modifications and a limitation. The program elements and the required documentation shall be maintained on site for audit.

(h) *Protection systems.* For construction permits issued after January 1, 1971, protection systems must meet the requirements set forth in editions or revisions of the Institute of Electrical and Electronics Engineers Standard, ''Criteria for Protection Systems for Nuclear Power Generating Stations,'' (IEEE–279) in effect[7] on the formal docket date[8] of the application for a construction permit. Protection systems may meet the requirements set forth in subsequent editions or revisions of IEEE–279 which become effective.

Footnotes to § 50.55a:

[1–3] [Reserved]

[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.

[4] 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 (b) 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 (b) of this section.

[5] 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.

[6] ASME Code cases that have been determined suitable for use by the Commission staff are listed in NRC Regulatory Guide 1.84, ''Design and Code Case Acceptability—ASME Section III Division 1,'' NRC Regulatory Guide 1.85, ''Materials Code Case Acceptability—ASME Section III Division 1,'' and NRC Regulatory Guide 1.147, ''Inservice Inspection Code Case Acceptability—ASME Section XI Division 1.'' The use of other Code cases may be authorized by the Director of the Office of Nuclear Reactor Regulation upon request pursuant to § 50.55a(a)(3).

[7] For purposes of this regulation the proposed IEEE 279 became ''in effect'' on August 30, 1968, and the revised issue IEEE 279–1971 became ''in effect'' on June 3, 1971. Copies may be obtained from the Institute of Electrical and Electronics Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. Copies are available for inspection at the Commission's Technical Library, 11545 Rockville Pike, Two White Flint North, Rockville, Maryland 20852–2738.

[8] Where an application for a construction permit is submitted in four parts pursuant to the provisions of § 2.101(a–1) and subpart F of part 2 of this chapter, ''the formal docket date of the application for a construction permit'' for purposes of this section is the date of docketing of the information required by § 2.101(a–1), (2), or (3), whichever is later.

[9] Guidance for quality group classifications of components which 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.''

[36 FR 11424, June 12, 1971]

EDITORIAL NOTE: For FEDERAL REGISTER citations affecting § 50.55a, see the List of CFR Sections Affected in the Finding Aids section of this volume.

§ 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]

§ 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, pursuant to 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 proceeding, including the right of any party to be heard to the extent that his contentions are relevant to the activity to be authorized. Prior to taking any action on such a motion which 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 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 pursuant to § 2.730(e) of this chapter, authorizing the Director of Nuclear Reactor Regulation to make appropriate findings on the matters specified in

[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.

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]

§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)(1) The holder of a license authorizing operation of a production or utilization facility may (i) make changes in the facility as described in the safety analysis report, (ii) make changes in the procedures as described in the safety analysis report, and (iii) conduct tests or experiments not described in the safety analysis report, without prior Commission approval, unless the proposed change, test or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question.

(2) A proposed change, test, or experiment shall be deemed to involve an

unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

(b)(1) The licensee shall maintain records of changes in the facility and of changes in procedures made pursuant to this section, to the extent that these changes constitute changes in the facility as described in the safety analysis report or to the extent that they constitute changes in procedures as described in the safety analysis report. The licensee shall also maintain records of tests and experiments carried out pursuant to paragraph (a) of this section. These records must include a written safety evaluation which provides the bases for the determination that the change, test, or experiment does not involve an unreviewed safety question.

(2) The licensee shall submit, as specified in § 50.4, a report containing a brief description of any changes, tests, and experiments, including a summary of the safety evaluation of each. The report may be submitted annually or along with the FSAR updates as required by § 50.71(e), or at such shorter intervals as may be specified in the license.

(3) The records of changes in the facility shall be maintained until the date of termination of the license, and records of changes in procedures and records of tests and experiments shall be maintained for a period of five years.

(c) The holder of a license authorizing operation of a production or utilization facility who desires (1) a change in technical specifications or (2) to make a change in the facility or the procedures described in the safety analysis report or to conduct tests or experiments not described in the safety analysis report, which involve an unreviewed safety question or a change in technical specifications, shall submit an application for amendment of his license pursuant to § 50.90.

(d) The provisions of this section apply to each nuclear power reactor licensee that has submitted the certification of permanent cessation of operations required under § 50.82(a)(1)(i).

(e) The provisions of paragraphs (a) through (c) of this section apply to each non-power reactor licensee whose license no longer authorizes operation of the reactor.

[39 FR 10555, Mar. 21, 1974, as amended at 41 FR 16446, Apr. 19, 1976; 41 FR 18302, May 3, 1976; 42 FR 20139, Apr. 18, 1977; 51 FR 40309, Nov. 6, 1986; 57 FR 39358, Aug. 31, 1992; 61 FR 39300, July 29, 1996]

§ 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, other than a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted, the licensee shall have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material. The assessment of RT_{PTS} must use the calculation procedures given in paragraph (c)(1) of this section, except as provided 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 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.

[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 prior to the expiration of the operating license, including any renewed term, if applicable for the plant.

(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.

(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

[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.

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 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

[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.

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.

$$\text{Equation 5:} \quad CF = \frac{\sum_{i=1}^{n}\left[A_i \times f_i^{(0.28-0.10\log f_i)}\right]}{\sum_{i=1}^{n}\left[f_i^{(0.56-0.20\log f_i)}\right]}$$

(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
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]

§ 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.* 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.

[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]

§ 50.63 Loss of all alternating current power.

(a) *Requirements.* (1) Each light-water-cooled nuclear power plant licensed to operate 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 licensed to operate on or before *July 21, 1988,* the licensee shall submit the information defined below to the Director of the Office of Nuclear Reactor Regulation by *April 17, 1989.* For each light-water-cooled nuclear power plant licensed to operate after the effective date of this amendment, the licensee shall submit the information defined below to the Director by 270 days after the date of license issuance.

(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.

(4) *Implementation Schedule:* For each light-water-cooled nuclear power plant licensed to operate on or before June 21, 1988, the licensee shall, within 30 days of the notification provided in accordance with paragraph (c)(3) of this section, submit to the Director of the Office of Nuclear Reactor Regulation a schedule commitment for implementing any equipment and associated procedure modifications necessary to meet the requirements of paragraph (a) of this section. This submittal must include an explanation of the schedule and a justification if the schedule does not provide for completion of the modifications within two years of the notification provided in accordance with paragraph (c)(3) of this section. A final schedule for implementing modifications necessary to comply with the requirements of paragraph (a) of this section will be established by the NRC staff in consultation and coordination with the affected licensee.

[53 FR 23215, June 21, 1988, as amended at 63 FR 50480, Sept. 22, 1998]

§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.

(a)(1) Each holder of a license to operate a nuclear power plant under §§ 50.21(b) or 50.22 shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that such structures, systems, and components, as defined in paragraph (b), are capable of fulfilling their intended functions. Such 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), this section only shall 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 such 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 be conducted taking 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. In performing monitoring and preventive maintenance activities, an assessment of the total plant equipment that is out of service should be taken into account to determine the overall effect on performance of safety functions.

(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) 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]

§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 place the results of its evaluation in its Public Document Room. The licensee may begin the thermal anneal after:

(1) Submitting the Thermal Annealing Report required by paragraph (b) of this section;

(2) The NRC places the results of its evaluation of the Thermal Annealing Report in the Public Document Room; 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 Identification of Unreviewed Safety Questions and Technical Specification Changes.

(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 Unreviewed Safety Questions and Technical Specification Changes.* Any changes to the facility as described in the updated final safety analysis report constituting unreviewed safety questions, and any changes to the technical specifications, which are necessary to either conduct the thermal annealing or 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 updated final safety analysis report which are attributable to the noncompliances and constitute unreviewed safety questions, and any changes to the technical specifications which are required as a result of the noncompliances, shall also be identified.

(i) If no unreviewed safety questions 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 unreviewed safety questions 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.

(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 relevant portions of 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 updated final safety analysis report which are attributable to the noncompliances and constitute unreviewed safety questions, and any changes to the technical specifications which are required as a result of the noncompliances, shall also be identified.

(A) If no unreviewed safety questions 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 unreviewed safety questions 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)–(iii) of this section, the NRC staff shall place in the NRC Public Document Room 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)–(iii) of this section, the NRC staff shall complete full documentation of its inspection of the licensee's annealing process and place this documentation in the NRC Public Document Room.

[60 FR 65472, Dec. 19, 1995]

§ 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

[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.

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 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).

[63 FR 63130, Nov. 12, 1998]

INSPECTIONS, RECORDS, REPORTS, NOTIFICATIONS

§ 50.70 Inspections.

(a) Each licensee and each holder of a construction 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 as may be necessary to effectuate the purposes of the Act, including section 105 of the Act.

(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 pursuant to part 50, 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 on site 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 meaures 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]

§50.71 Maintenance of records, making of reports.

(a) Each licensee and each holder of a construction 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 rules, regulations, and orders of the Commission in effectuating the purposes of the Act, including section 105 of the Act. Reports must be submitted in accordance with §50.4.

(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.

(c) Records that are required by the regulations in this part, by license condition, or by technical specifications, must be retained for the period specified by the appropriate regulation, license condition, or technical specification. If a retention period is not otherwise specified, these records must be retained until the Commission terminates the facility license.

(d)(1) Records which must be maintained pursuant to this part may be the original or a reproduced copy or microform if such 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, 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 pursuant to the provisions of §50.21 or §50.22 of this part 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 operating license, to assure that the information included in the FSAR contains the latest material developed. 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 pursuant to Commission requirement since the submission of the original FSAR or, as appropriate, the last updated FSAR. The updated FSAR shall be revised to include the effects of: all changes made in the facility or procedures as described in the FSAR; all safety evaluations performed by the licensee either in support of requested license amendments or in support of conclusions that changes did not involve an unreviewed safety question; and all analyses of new safety issues performed by or on behalf of the licensee at Commission request. The updated information shall be appropriately located within 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 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) Not less than 15 days before § 50.71(e) becomes effective, the Director of the Office of Nuclear Reactor Regulation shall notify by letter the licensees of those nuclear power plants initially subject to the NRC's systematic evaluation program that they need not comply with the provisions of this section while the program is being conducted at their plant. The Director of the Office of Nuclear Reactor Regulation will notify by letter the licensee of each nuclear power plant being evaluated when the systematic evaluation program has been completed. Within 24 months after receipt of this notification, the licensee shall file a complete FSAR which is up to date as of a maximum of 6 months prior to the date of filing the revision.

(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) 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). 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.

[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]

§ 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 of this part 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) Of those non-Emergency events specified in paragraph (b) of this section.

(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 Operations Center.[3]

(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)[5] 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

[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, and 73.71.

[2] These Emergency Classes are addressed in appendix E of this part.

[3] Commercial telephone number of the NRC Operations Center is (301) 816–5100.

[4] [Reserved]

[5] Requirements for ERDS are addressed in appendix E, Section VI.

has the capability to transmit the exercise data.

(5) When making a report under paragraph (a)(3) of this section, the licensee shall identify:

(i) The Emergency Class declared; or

(ii) Either paragraph (b)(1), ''One-Hour Report,'' or paragraph (b)(2), ''Four-Hour Report,'' 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 of the following:

(i)(A) The initiation of any nuclear plant shutdown required by the plant's Technical Specifications.

(B) Any deviation from the plant's Technical Specifications authorized pursuant to § 50.54(x) of this part.

(ii) Any event or condition during operation that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or results in the nuclear power plant being:

(A) In an unanalyzed condition that significantly compromises plant safety;

(B) In a condition that is outside the design basis of the plant; or

(C) In a condition not covered by the plant's operating and emergency procedures.

(iii) Any natural phenomenon or other external condition that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the plant.

(iv) 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.

(v) Any event that results in a major loss of emergency assessment capability, offsite response capability, or communications capability (e.g., significant portion of control room indication, Emergency Notification System, or offsite notification system).

(vi) Any event that poses an actual threat to the safety of the nuclear power plant or significantly hampers 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.

(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) Any event, found while the reactor is shut down, that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant, including its principal safety barriers, being seriously degraded or being in an unanalyzed condition that significantly compromises plant safety.

(ii) Any event or condition that results in a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS), except when:

(A) The actuation results from and is part of a pre-planned sequence during testing or reactor operation;

(B) The actuation is invalid and:

(*1*) Occurs while the system is properly removed from service;

(*2*) Occurs after the safety function has been already completed; or

(*3*) Involves only the following specific ESFs or their equivalent systems:

(*i*) Reactor water clean-up system;

(*ii*) Control room emergency ventilation system;

(*iii*) Reactor building ventilation system;

(*iv*) Fuel building ventilation system; or

(*v*) Auxiliary building ventilation system.

(iii) Any event or condition that alone 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.

(iv)(A) Any airborne radioactive release that, when averaged over a time period of 1 hour, results in concentrations in an unrestricted area that exceed 20 times the applicable concentration specified in appendix B to part 20, table 2, column 1.

(B) Any liquid effluent release that, when averaged over a time of 1 hour, exceeds 20 times the applicable concentration 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. (Immediate notifications made under this paragraph also satisfy the requirements of § 20.2202 of this chapter.)

(v) Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment.

(vi) 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.

(vii) Any instance of:

(A) A defect in any spent fuel storage cask structure, system, or component which is important to safety; or

(B) A significant reduction in the effectiveness of any spent fuel storage cask confinement system during use of the storage cask under a general license issued under § 72.210 of this chapter.

A followup written report is required by § 72.216(b) of this chapter including a description of the means employed to repair any defects or damage and prevent recurrence, using instructions in § 72.4, within 30 days of the report submitted in paragraph (a). A copy of the written report must be sent to the administrator of the appropriate Nuclear Regulatory Commission regional office shown in appendix D to part 20 of this chapter.

(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 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]

§ 50.73 Licensee event report system.

(a) *Reportable events.* (1) The holder of an operating license for a nuclear power plant (licensee) shall submit a Licensee Event Report (LER) for any event of the type described in this paragraph within 30 days after the discovery of the event. Unless otherwise specified in this section, the licensee shall report an event 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; or

(B) Any operation or condition prohibited by the plant's Technical Specifications; or

(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 the condition of the nuclear power plant, including its principal

safety barriers, being seriously degraded, or that resulted in the nuclear power plant being:

(A) In an unanalyzed condition that significantly compromised plant safety;

(B) In a condition that was outside the design basis of the plant; or

(C) In a condition not covered by the plant's operating and emergency procedures.

(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) Any event or condition that resulted in a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS), except when:

(A) The actuation resulted from and was part of a pre-planned sequence during testing or reactor operation;

(B) The actuation was invalid and:

(*1*) Occurred while the system was properly removed from service;

(*2*) Occurred after the safety function had been already completed; or

(*3*) Involved only the following specific ESFs or their equivalent systems:

(*i*) Reactor water clean-up system;

(*ii*) Control room emergency ventilation system;

(*iii*) Reactor building ventilation system;

(*iv*) Fuel building ventilation system; or

(*v*) Auxiliary building ventilation system.

(v) Any event or condition that alone 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 this paragraph 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) Reports submitted to the Commission in accordance with paragraph (a)(2)(viii) of this section also meet the effluent release reporting requirements of § 20.2203(a)(3) of this chapter.

(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 Practices for Unique Identification 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.

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, 345 East 47th Street, New York, NY 10017. IEEE Std 803–1983 is available for inspection at the NRC's Technical Library, which is located at 11545 Rockville Pike, Rockville, Maryland 20852–2738; and at the Office of the Federal Register, 800 North Capitol Street, NW., suite 700, Washington, DC.

(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)*(1)* Operator actions that affected the course of the event, including operator errors, procedural deficiencies, or both, that contributed to the event.

(2) For each personnel error, the licensee shall discuss:

(i) Whether the error was a cognitive error (e.g., failure to recognize the actual plant condition, failure to realize which systems should be functioning, failure to recognize the true nature of the event) or a procedural error;

(ii) Whether the error was contrary to an approved procedure, was a direct result of an error in an approved procedure, or was associated with an activity or task that was not covered by an approved procedure;

(iii) Any unusual characteristics of the work location (e.g., heat, noise) that directly contributed to the error; and

(iv) The type of personnel involved (i.e., contractor personnel, licensed operator, nonlicensed operator, other licensee personnel).

(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 the availability of other systems or components that could have performed the same function as the components and systems that failed during the event.

(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 within 30 days of discovery of a reportable event or situation 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) *Exemptions.* Upon written request from a licensee including adequate justification or at the initiation of the NRC staff, the NRC Executive Director for Operations may, by a letter to the licensee, grant exemptions to the reporting requirements under this section.

(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]

§50.74 Notification of change in operator or senior operator status.

Each licensee shall notify the Commission in accordance with §50.4 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]

§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, 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, by other agencies to establish rates.

(b) Each power reactor applicant for or holder of an operating license 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) of this part.

(1) 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.

(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 use 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.

(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 into an account segregated from licensee assets and outside the licensee's administrative control of cash or liquid assets such that the amount of funds would be sufficient to pay decommissioning costs at the time termination of operation is expected. Prepayment may be in the form of a trust, escrow account, Government fund, certificate of deposit, deposit of Government securities or other payment acceptable to the NRC. A licensee 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. This includes the periods of safe storage, final dismantlement, and license termination, if the licensee's rate-setting authority does not authorize the use of another rate. However, 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 licensee's administrative control in which the total amount of funds would be sufficient to pay decommissioning costs at the time termination of operation is expected. An external sinking fund may be in the form of a trust, escrow account, Government fund, certificate of deposit, deposit of Government securities, or other payment acceptable to the NRC. A licensee 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. This includes the periods of safe storage, final dismantlement, and license termination, if the licensee's rate-setting authority does not authorize the use of another rate. However, actual earnings on existing funds may be used

[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.

to calculate future fund needs. A licensee, whose rates for decommissioning costs cover only a portion of such costs, may make use of this method only for that portion of such 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, letter of credit, or line 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(ies) 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.

(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. 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 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. 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 has already closed (before the end of its licensed life), or for plants involved in mergers or acquisitions shall submit this report annually.

(2) 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.

(3) 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.

(4) 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.

[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]

US/IAEA SAFEGUARDS AGREEMENT

§50.78 Installation information and verification.

Each holder of a construction permit shall, if requested by the Commission, submit installation information on Form N–71, permit verification thereof by the International Atomic Energy Agency, and take such other action as may be necessary to implement the US/IAEA Safeguards Agreement, in the manner set forth in §§75.6 and 75.11 through 75.14 of this chapter.

[49 FR 19627, May 9, 1984]

TRANSFERS OF LICENSES—CREDITORS' RIGHTS—SURRENDER OF LICENSES

§50.80 Transfer of licenses.

(a) No license for a production or utilization facility, 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 shall give its consent in writing.

(b) An application for transfer of a license shall include 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, and, if the license to be issued is a class 103 license, the information required by §50.33a. 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. 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 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]

§ 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 or construction permit which may be issued by the Commission with regard to the 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.

[26 FR 9546, Oct. 10, 1961, as amended at 32 FR 2562, Feb. 7, 1967]

§ 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 report must include a description of the planned decommissioning activities along with a schedule for their accomplishment, an estimate of expected costs, and a discussion that provides the reasons for concluding that the environmental impacts associated with site-specific decommissioning activities will be bounded by appropriate previously issued environmental impact statements.

(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.

(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.

(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 terminal radiation survey and associated documentation demonstrates that the facility and site are suitable for release in accordance with 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]

AMENDMENT OF LICENSE OR CONSTRUCTION PERMIT AT REQUEST OF HOLDER

§50.90 Application for amendment of license or construction permit.

Whenever a holder of a license or construction permit desires to amend the license or permit, application for an amendment must be filed with the Commission, as specified in §50.4, fully describing the changes desired, and following as far as applicable, the form prescribed for original applications.

[51 FR 40310, Nov. 6, 1986]

§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 for a facility licensed under § 50.21(b) or § 50.22 or for a testing facility, except for amendments subject to hearings governed by §§ 2.1201–2.1263 of this chapter. For amendments subject to §§ 2.1201–2.1263 of this chapter, the following procedures will apply only to the extent specifically referenced in § 2.1205 (c) and (d) of this chapter:

(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 upon issuance, even if adverse public comments have been received and even if an interested person meeting the provisions for intervention called for in § 2.714 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 called for in § 2.714 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]

§ 50.92 Issuance of amendment.

(a) In determining whether an amendment to a license or construction permit will be issued to the applicant, the Commission will be guided by the considerations which govern the issuance of initial licenses or construction 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. If the amendment involves a significant hazards consideration, the Commission will give notice of its proposed action (1) pursuant to § 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, pursuant to the procedures in § 50.91, that a proposed amendment to an operating license for a facility 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]

REVOCATION, SUSPENSION, MODIFICATION, AMENDMENT OF LICENSES AND CONSTRUCTION PERMITS, EMERGENCY OPERATIONS BY THE COMMISSION

§ 50.100 Revocation, suspension, modification of licenses and construction permits for cause.

A license or construction permit may be revoked, suspended, or modified, in whole or in part, for any material false statement in the application for license or in the supplemental or other statement of fact required of the applicant; or because of conditions revealed by the application for license or statement of fact or any report, record, inspection, or other means, which would warrant the Commission to refuse to grant a license on an original application (other than those relating to §§ 50.51, 50.42(a), and 50.43(b) of this part); or for failure to construct or operate a facility in accordance with the terms of the construction permit or license, provided that failure to make timely completion of the proposed construction or alteration of a facility under a construction permit 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, or order of the Commission.

§ 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 rules or the imposition of a regulatory staff position interpreting the Commission rules 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; or

(ii) Six months before the date of docketing of the operating license application for the facility for facilities having construction permits issued before October 21, 1985; or

(iii) The date of issuance of the operating license for the facility for facilities having operating licenses; or

(iv) The date of issuance of the design approval under appendix M, N, or O of part 52.

(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]

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]

§ 50.120 Training and qualification of nuclear power plant personnel.

(a) *Applicability*. The requirements of this section apply to each applicant for (applicant) and each holder of an operating license (licensee) for a nuclear power plant of the type specified in § 50.21(b) or § 50.22.

(b) *Requirements*. (1) Each nuclear power plant applicant, by November 22, 1993 or 18 months prior to fuel load, whichever is later, and each nuclear power plant licensee, by November 22, 1993 shall establish, implement, and maintain a training program derived from a systems approach to training as defined in 10 CFR 55.4. The training program 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.

(2) 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 so as 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.

[58 FR 21912, Apr. 26, 1993; 58 FR 39092, July 21, 1993]

APPENDIX A TO PART 50—GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

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INTRODUCTION

Pursuant to the provisions of §50.34, an application for a construction permit 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 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

[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.

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.

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.

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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. 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, 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 therefor. 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 (a) assuring that an appropriate quality assurance program is established and effectively executed and (b) verifying, such as by checking, auditing, and inspection, 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. Such persons and organizations performing quality assurance functions shall report to a management level such that this 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

[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. These criteria will also be used for guidance in evaluating the adequacy of quality assurance programs in use by holders of construction permits and operating licenses.

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 this 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 such levels of management as may be 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 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 3210D, Jan. 20, 1975]

APPENDIX C TO PART 50—A GUIDE FOR THE FINANCIAL DATA AND RELATED INFORMATION REQUIRED TO ESTABLISH FINANCIAL QUALIFICATIONS FOR FACILITY CONSTRUCTION PERMITS

GENERAL INFORMATION

This appendix is intended to apprise applicants for licenses to construct 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 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 such information is pertinent to establishing the applicant's financial ability to construct the proposed facility.

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

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

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 orcontracts 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 is required by §50.71(b) to file its annual financial report with the Commission at the time of issuance thereof. 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 Commiasion 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]

APPENDIX D TO PART 50 [RESERVED]

APPENDIX E TO PART 50—EMERGENCY PLANNING AND PREPAREDNESS FOR PRODUCTION AND UTILIZATION FACILITIES

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I. Introduction
II. The Preliminary Safety Analysis Report
III. The Final Safety Analysis Report
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V. Implementing Procedures
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I. INTRODUCTION

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.

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 and submitted as part of the final safety analysis report.

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

[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

Continued

reactors and the degree to which compliance with the requirements of this section and sections II, III, IV, and V as necessary will be determined on a case-by-case basis.[2]

Notwithstanding the above paragraphs, in the case of an operating license authorizing only fuel loading and/or low power operations up to 5% 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.

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 of the near-site emergency operations facility in assessing information, recommending protective action, and disseminating information to the public.

III. The Final Safety Analysis Report

The Final Safety Analysis Report 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 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

The applicant's emergency plans shall contain, but not necessarily be limited to, information needed to demonstrate compliance

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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.

with the elements set forth below, i.e., organization for coping with radiation emergencies, assessment action, activation of emergency organization, notification procedures, emergency facilities and equipment, training, maintaining emergency preparedness, and recovery. In addition, the emergency response plans submitted by an applicant for a nuclear power reactor operating license shall contain information needed to demonstrate compliance with the standards described in §50.47(b), and they will be evaluated against those standards. The nuclear power reactor operating license applicant shall also provide an analysis of the time required to evacuate and for taking other protective actions for various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations.

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 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 on 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. Identification of, and assistance expected from, appropriate State, local, and Federal agencies with responsibilities for coping with emergencies.

8. Identification of the State and/or local officials responsible for planning for, ordering, and controlling appropriate protective actions, including evacuations when necessary.

B. Assessment Actions

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 inplant conditions and instrumentation in addition to onsite and offsite monitoring. These emergency action levels shall be discussed and agreed on by the applicant and State and local governmental authorities and approved by NRC. They shall also be reviewed with the State and local governmental authorities on an annual basis.

C. Activation of Emergency Organization

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.

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.[1]

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 State/local officials have the capability to make a public notification decision promptly on being informed by the licensee of an emergency condition. By February 1, 1982, 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 four-month period in 10 CFR 50.54(s)(2) for the correction of emergency plan deficiencies shall not apply to the initial installation of this public notification system that is required by February 1, 1982. The four-month period will apply to correction of deficiencies identified during the initial installation and testing of the prompt public notification systems as well as those deficiencies discovered thereafter. The design objective of the prompt public notification system shall be to have the capability to essentially complete the initial notification of the public within the plume exposure pathway EPZ within about 15 minutes. The use of this notification capability will range from immediate 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 State and local governmental officials to make a judgment whether or not to activate the public notification system. Where there is a decision to activate the notification system, the State and local officials will determine whether to activate the entire notification system simultaneously or in a graduated or staged manner. The responsibility for activating such a public 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 the services of physicians and other medical personnel qualified to handle radiation emergencies on-site;

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 licensee onsite technical support center and a licensee near-site emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency;

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 near-site 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.

[1] See footnote 1 to section I.

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 near-site 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 radiation 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 radiation 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 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. This exercise shall be conducted within two years before the issuance of the first operating license for full power (one authorizing operation above 5% of rated 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 one year prior to issuance of an operating licensee for full power, an exercise which tests the licensee's onsite emergency plans shall be conducted within one year before issuance of an operating license for full power. This exercise need not have State or local government participation.

b. Each licensee at each site shall conduct an exercise of its onsite emergency plan every 2 years. 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, protective action decision-making, and plant system repair and corrective actions. 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 the opportunity to consider accident management strategies, supervised instruction would be permitted, operating staff would have the opportunity to resolve problems (success paths) rather than have controllers intervene, and the drills could focus on onsite training objectives.

c. Offsite plans for each site shall be exercised biennially with full participation by each offsite authority having a role under the 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

[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.

least, partially participate[5] in other offsite plan exercises in this period.

d. A State should fully participate in the ingestion pathway portion of exercises at least once every six years. In States with more than one site, the State 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 find reasonable assurance that adequate protective measures can be taken in the event of a radiological emergency. 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 training, including exercises, shall provide for formal critiques in order to identify weak or deficient areas that need correction. Any weaknesses or deficiencies that are identified shall 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 10 CFR 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.

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.

[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.

V. Implementing Procedures

No less than 180 days prior to the scheduled issuance of an operating license for a nuclear power reactor or a license to possess nuclear material the applicant's detailed implementing procedures for its emergency plan shall be submitted to the Commission as specified in §50.4. Licensees who are authorized to operate a nuclear power facility shall submit any changes to the emergency plan or procedures to the Commission, as specified in §50.4, within 30 days of such changes.

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.[6] 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

[6] See 10 CFR 50.55a(h) Protection Systems.

(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 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[7] 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[8] (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.

c. In the event of a failure of the NRC supplied onsite modem, a replacement unit will be furnished by the NRC for licensee installation.

4. Implementing the Emergency Response Data System Program:

a. Each licensee shall develop and submit an ERDS implementation program plan to the NRC by October 28, 1991. To ensure compatibility with the guidance provided for the ERDS, the ERDS implementation program plan,[9] must include, but not be limited to, information on the licensee's computer system configuration (i.e., hardware and software), interface, and procedures.

b. Licensees must comply with appendix E to part 50, section V.

c. Licensees that have submitted the required information under the voluntary ERDS implementation program will not be required to resubmit this information. The licensee shall meet the implementation schedule of appendix E to part 50, section VI.4d.

d. Each licensee shall complete implementation of the ERDS by February 13, 1993, or before initial escalation to full power, whichever comes later. Licensees with currently operational ERDS interfaces approved under the voluntary ERDS implementation program[10] will not be required to submit another implementation plan and will be considered to have met the requirements for ERDS under appendix E to part 50, section VI.1 and 2 of this part.

[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]

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

[7] Guidance is provided in NUREG–1394, Revision 1.

[8] See NUREG–1394, Revision 1, appendix C, Data Point Library.

[9] See NUREG–1394, Revision 1, section 3.

[10] See NUREG–1394, Revision 1.

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]

APPENDIX G TO PART 50—FRACTURE TOUGHNESS REQUIREMENTS

I. Introduction and scope.
II. Definitions.
III. Fracture toughness tests.
IV. 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 analyses 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.

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 3, 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 3, 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 3 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 3 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 3, 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

[1] Defined in ASTM E 185–79 and –82 which are incorporated by reference in appendix H to part 50.

TABLE 1—PRESSURE AND TEMPERATURE REQUIREMENTS FOR THE REACTOR PRESSURE VESSEL—Continued

Operating condition	Vessel pressure [1]	Requirements for pressure-temperature limits	Minimum temperature requirements
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]

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, –79, and –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 ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased. 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 ASTM E 185–82 to the extent practicable for the configuration of the specimens in the capsule.

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.

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 one year 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]

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 permit to construct a nuclear power reactor 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.

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.

SEC. II. *Guides on design objectives for light-water-cooled nuclear power reactors licensed under 10 CFR part 50.* The guides on design objectives set forth in this section may be used by an applicant for a permit to construct a light-water-cooled nuclear power reactor as guidance in meeting the requirements of §50.34a(a). 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]

[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

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.

SEC. 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 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

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which specific account is required in determining design objectives.

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.

SEC. IV. *Guides on technical specifications for limiting conditions for operation for light-water-cooled nuclear power reactors licensed under 10 CFR part 50.* The guides on limiting conditions for operation for light-water-cooled nuclear power reactors set forth below may be used by an applicant for a license to operate a light-water-cooled nuclear power reactor or a licensee who has submitted a certification of permanent cessation of operations under §50.82(a)(1) 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 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).

SEC. V. *Effective dates.* A. The guides for limiting conditions for operation set forth in

[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.

this appendix shall be applicable in any case in which an application was filed on or after January 2, 1971, for a permit to construct a light-water-cooled nuclear power reactor.

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

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

[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.

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.

[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]

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

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I. INTRODUCTION

One of the conditions of all operating licenses 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 such 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(v), 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

[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.

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.

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,

[1] Such inservice inspections are required by § 50.55a.

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.

(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

[2] Such inservice inspections are required by § 50.55a.

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 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

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I. Introduction.
II. Definitions.
III. Performance-based leakage-test requirements.

A. Type A test.
B. Type B and C tests.
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I. INTRODUCTION

One of the conditions required of all operating 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 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.

[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.''

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)(1)(ii) and §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 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 below) to the Director of the Office of Nuclear Reactor Regulation.

3. The regulatory guide or other implementation document used by a licensee, or applicant for an operating license, 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]

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 1 to 4 below it shall 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 such uncertainties as instrumentation error), with the maximum peaking factor allowed by the technical specifications. A range of power distribution shapes and peaking factors representing power distributions that may occur over the core lifetime shall be studied and the one selected should be that which results in the most severe calculated consequences, for the spectrum of postulated breaks and single failures 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 nonfuel 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 $Btu\text{-}hr^{-1}\text{-}ft^{-2}\ {}^{\circ}F^{-1}$ 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]

Appendix L to Part 50—Information Requested by the Attorney General for Antitrust Review Facility License Applications

Introduction. The information in this appendix is that requested by the Attorney General in connection with his review, pursuant to section 105c of the Atomic Energy Act of 1954, as amended, of certain license applications for nuclear power plants. The applicant shall submit the information as a separate document titled, ''Information Requested by the Attorney General for Antitrust Review.'' Twenty (20) copies shall be submitted prior to any other part of the facility license application as provided in §50.33a and in accordance with §2.101 of this chapter.

I. Definitions

1. ''Applicant'' means the entity applying for authority to construct or operate subject unit and each corporate parent, subsidiary and affiliate. Where application is made by two or more electric utilities not under common ownership or control, each utility, subject to the applicable exclusions contained in §50.33a, should set forth separate responses to each item herein.

2. ''Subject unit'' means the nuclear generating unit or units for which application for construction or operation is being made.

3. ''Electric utility'' or ''system'' means any entity owning, controlling or operating facilities for the generation or transmission or distribution of electric power.

4. ''Coordination'' means any arrangement between two or more systems for generation and transmission planning, or operation of two or more interconnected electric utilities not under common ownership or control, including but not limited to arrangements for sharing operating and installed reserves, arrangements for joint or staggered construction of generating facilities, economy energy transactions, capacity transactions based on load diversities, thermalhydro generation pooling, common maintenance arrangements, and joint use of transmission facilities or wheeling.

5. ''Coordinating power and energy'' means energy transmitted in accordance with an arrangement for coordination including but not limited to emergency power, economy energy, deficiency power and associated energy, and maintenance power and energy.

6. Except where specifically mentioned otherwise, the term ''reserve generating capacity'' or ''reserves'' shall refer to installed reserves in contrast to spinning or operating reserves.

II. Required Information

1. State separately for hydroelectric and thermal generating resources applicant's most recent peak load and dependable capacity for the same time period. State applicant's dependable capacity at time of system peak for each of the next 10 years for which information is available. Identify each new unit or resource. For hydroelectric generating capacity, indicate the number of kilowatt hours of use associated with each kilowatt of capacity during the ''adverse water year'' upon which dependable capacity is based. Indicate average annual kilowatt hour loads per kilowatt, associated with each system peak shown (exclusive of interchange arrangements).

2. State applicant's estimated annual load growth for each of the next 20 years or for the period applicant utilizes in system planning. Indicate growth both in kilowatt requirements and kilowatt hour requirements.

3. State estimated annual load growth in kilowatts and kilowatt hours of companies or pools upon which the economic justification of the subject unit is based for each of the next 20 years or for the period applicant utilized in system planning. Identify each company or pool member.

4. For the year the subject unit would first come on line, state estimated annual load growth in kilowatts and kilowatt hours of any coordinating group or pool of which the applicant is a member (other than the coordinating group or pool referred to in the applicant's response to item 3) which has generating and/or transmission planning functions. Identify each company or pool member whose loads are indicated in the response thereto.

5. State applicant's minimum installed reserve criterion (as a percentage of load)[1] for the period when the subject unit will first come on line. If the applicant shares reserves with other systems, identify the other systems and provide minimum installed reserve criterion (as a percentage of loan)[1] by contracting parties or pool for the period when the proposed unit will first come on line.

6. Describe methods used as a basis to establish, or as a guide in establishing the criteria for applicant's and/or applicant's pool's minimum amount of installed reserves (e.g., (a) single largest unit down, (b) probability methods such as loss of load one day in 20 years, loss of capacity once in 5 years, (c) other methods and/or (d) judgment. List contingencies other than risk of forced outage that enter into the determination).

7. Indicate whether applicant's system interconnections are credited explicitly or implicitly in establishing applicant's installed reserves.

8. List rights to receive emergency power and obligations to deliver emergency power, rights or obligations to receive or deliver deficiency power or unit power, or other coordinating arrangements, by reference to applicant's Federal Power Commission (FPC) rate schedules (i.e., ABC Power & Light Co., FPC Rate Schedule No. 15 including supplements 1–5),[2] and also by reference to applicant's state commission filings. Where documents are not on file with the FPC, supply copies, or where not reduced to writing, describe arrangements. Identify for each such arrangement the participating parties other than applicant. Provide one line electrical and geographic diagrams of coordinating groups or power pools (with generation or transmission planning functions) of which applicant's generation and transmission facilities constitute a part.

9. List, and provide the mailing address for non-affiliated electric utility systems with peak loads smaller than applicant's which serve either at wholesale or at retail adjacent to areas served by the applicant.

Provide a geographic one-line diagram of applicant's generating and transmission facilities (including subtransmission) indicating the location of adjacent systems and as to such systems indicate (if available) their load, their annual load growth, their generating capacity, their largest thermal generating unit size, and their minimum reserve criteria.

10. List separately those systems in Item 9 which purchase from applicant (a) all bulk power supply and (b) systems which purchase partial bulk power supply requirements. Where information is available to applicant, identify those Item 9 systems purchasing part or all of their bulk power supply requirements from suppliers other than applicant.

11. State as to all power generated and sold by applicant the most recent average cost of bulk power supply experienced by applicant (a) at site of generating facilities, (b) at the delivery points from the primary transmission (backbone) system, (c) at delivery

[1] Indicate whether loads other than peak loads are considered.

[2] List separately and identify certificates of concurrence.

points from the secondary transmission system, and (d) at delivery points from the distribution system, in terms of dollars per kilowatt per year, in mills per kilowatt hour, and in both the kilowatt costs and kilowatt hour costs divided by the kilowatt hours. If wholesale sales are made at varying voltages, indicate average costs at each voltage.

12. State (a) for generating facilities and (b) for transmission subdivided by voltage classes, the most recent estimated cost of applicant's bulk power supply expansion program of which the subject unit is a part, in terms of dollars per kilowatt per year, in mills per kilowatt hour and in both the kilowatt costs and kilowatt hour costs divided by the kilowatt hours. Also state separately the most recently estimated cost of the subject unit(s).

13. List and describe all requests for, or indications of interest in, interconnection and/or coordination and purchases or sales of coordinating power and energy from adjacent utilities listed in item 9 since 1960 and state applicant's response thereto. List and describe all requests for, or indications of interest in, supply of full or partial requirements of bulk power for the same period and state applicant's response thereto.

14. List (a) agreements to which applicant is a party (reproducing relevant paragraphs) and (b) State laws (supply citations only) which restrict or preclude coordination by, with, between, or among any electric utilities or systems identified in applicant's response to items 8 and 9. List (a) agreements to which the applicant is a party (reproducing relevant paragraphs) and (b) State laws (supply citations only) which restrict or preclude substitution of service or establishment of service of full or partial bulk power supply requirements by an electric utility other than applicant to systems identified in items 8 and 9. Where the contract provision appears in contracts or rate schedules on file with a Federal agency, identify each in the same form as in previous responses. Where the contract has not been filed with a Federal agency, a copy should be supplied unless it has been supplied pursuant to another item hereto. Where it is not in writing, it should be described.

15. State, at point of delivery, average future costs of power purchased from applicant to adjacent systems identified in applicant's response to item 9 in terms of dollars/month/kw for capacity, mills/kw for energy and mills/kwh for both power and energy at purchaser's present load factor (a) at present load, (b) at 50 percent increase over present load, (c) at 100 percent increase over present load, and (d) at 200 percent increase over present load. (All costs should be determined under present rate schedules.) Where sales are made under contracts or rate schedules on file with a Federal agency and not included in the response to item 9, identify each in the same form as in previous responses. Where the contract has not been filed with a Federal agency, a copy should be supplied.

16. State whether applicant has prepared, caused to be prepared, or received engineering studies for generation and transmission expansion programs which include loads of each system in item 9.

17. List adjacent systems to which applicant has offered to sponsor or to conduct system surveys in contemplation of an offer by applicant to purchase, merge or consolidate with said adjacent system, subsequent to January 1, 1960.

18. List applicant's offers or proposals to purchase, merge or consolidate with electric utilities, subsequent to January 1, 1960.

19. List all acquisitions of or mergers or consolidations with electric utilities by applicant, subsequent to January 1, 1960, including:

(a) The name and principal place of business of the system prior to the acquisition, merger or consolidation;

(b) The date the acquisition, merger or consolidation was consummated;

(c) Gross annual revenue and most recent peak load, dependable capacity and the largest thermal generating unit of the system, prior to the dates of consummation.

20. State applicant's six (or fewer if there are not six) lowest industrial or large commercial rates for firm electric power supply in terms of cost for power and energy in mills per kilowatt hour (and separately, the demand and energy components) and indicate the portion of the charge attributed to bulk power supply. State the rates or rate blocks applicant utilizes for its six (or fewer if there are not six) promotional services such as electric space heating, electric hot water heating, and the like, in terms of mills per kilowatt hour for power and energy and indicate the portion of the rate or rate blocks attributed to bulk power supply.

[38 FR 3956, Feb. 9, 1973, as amended at 39 FR 34395, Sept. 25, 1974; 43 FR 49775, Oct. 25, 1978; 44 FR 60716, Oct. 22, 1979]

APPENDIX M TO PART 50—STANDARDIZATION OF DESIGN; MANUFACTURE OF NUCLEAR POWER REACTORS; CONSTRUCTION AND OPERATION OF NUCLEAR POWER REACTORS MANUFACTURED PURSUANT TO COMMISSION LICENSE

Section 101 of the Atomic Energy Act of 1954, as amended, and §50.10 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 the part require the

issuance of a construction permit by the Commission before commencement of construction of a production or utilization facility, and the issuance of an operating license before operation of the facility. The provisions of this part relating to the facility licensing process are, in general, predicated on the assumption that the facility will be assembled and constructed on the site at which it is to be operated. In those circumstances, both facility design and site-related issues can be considered in the initial, construction permit stage of the licensing process.

However, under the Atomic Energy Act, a license may be sought and issued authorizing the manufacture of facilities but not their construction and installation at the sites on which the facilities are to be operated. Prior to the ''commencement of construction'', as defined in §50.10(c), of a facility (manufactured pursuant to such a Commission license) on the site at which it is to operate—that is preparation of the site and installation of the facility—a construction permit that, among other things, reflects approval of the site on which the facility is to be operated, must be issued by the Commission. This appendix sets out the particular requirements and provisions applicable to such situations where nuclear power reactors to be manufactured pursuant to a Commission license and subsequently installed at the site pursuant to a Commission construction permit, are of the type described in §50.22. It thus codifies one approach to the standardization of nuclear power reactors.

1. Except as otherwise specified in this appendix or as the context otherwise indicates, the provisions in this part applicable to construction permits, including the requirement in §50.58 for review of the application by the Advisory Committee on Reactor Safeguards and the holding of a public hearing, apply in context, with respect to matters of radiological health and safety, environmental protection, and the common defense and security, to licenses pursuant to this appendix M to manufacture nuclear power reactors (manufacturing licenses) to be operated at sites not identified in the license application.

2. An application for a manufacturing license pursuant to this appendix M must be submitted, as specified in §50.4, and meet all the requirements of §§50.34(a) (1)–(9) and 50.34a (a) and (b), except that the preliminary safety analysis report shall be designated as a ''design report'' and any required information or analyses relating to site matters shall be predicated on postulated site parameters which must be specified in the application. The application must also include information pertaining to design features of the proposed reactor(s) that affect plans for coping with emergencies in the operation of the reactor(s).

3. An applicant for a manufacturing license pursuant to this appendix M shall submit with his application an environmental report as required of applicants for construction permits in accordance with subpart A of part 51 of this chapter, provided, however, that such report shall be directed at the manufacture of the reactor(s) at the manufacturing site; and, in general terms, at the construction and operation of the reactor(s) at an hypothetical site or sites having characteristics that fall within the postulated site parameters. The related draft and final environmental impact statement prepared by the Commission's regulatory staff will be similarly directed.

4. (a) Sections 50.10 (b) and (c), 50.12(b), 50.23, 50.30(d), 50.34(a)(10), 50.34a(c), 50.35 (a) and (c), 50.40(a), 50.45, 50.55(d), 50.56, and appendix J do not apply to manufacturing licenses. Appendices E and H apply to manufacturing licenses only to the extent that the requirements of these appendixes involve facility design features.

(b) The financial information submitted pursuant to §50.33(f) and appendix C shall be directed at a demonstration of the financial qualifications of the applicant for the manufacturing license to carry out the manufacturing activity for which the license is sought.

5. The Commission may issue a license to manufacture one or more nuclear power reactors to be operated at sites not identified in the license application if the Commission finds that:

(a) The applicant has described the proposed design of and the site parameters postulated for the reactor(s), 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.

(b) Such further technical or design information as may be required to complete the design report and which can reasonably be left for later consideration, will be supplied in a supplement to the design report.

(c) 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

(d) On the basis of the foregoing, there is reasonable assurance that (i) such safety questions will be satisfactorily resolved before any of the proposed nuclear power reactor(s) are removed from the manufacturing site and (ii) taking into consideration the site criteria contained in part 100 of this chapter, the proposed reactor(s) can be constructed and operated at sites having characteristics that fall within the site parameters

postulated for the design of the reactor(s) without undue risk to the health and safety of the public.

(e) The applicant is technically and financially qualified to design and manufacture the proposed nuclear power reactor(s).

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

(g) On the basis of the evaluations and analyses of the environmental effects of the proposed action required by subpart A of part 51 of this chapter and paragraph 3 of this appendix, the action called for is the issuance of the license.

NOTE: When an applicant has supplied initially all of the technical information required to complete the application, including the final design of the reactor(s), the findings required for the issuance of the license will be appropriately modified to reflect that fact.

6. Each manufacturing license issued pursuant to this appendix will specify the number of nuclear power reactors authorized to be manufactured and the latest date for the completion of the manufacture of all such reactors. Upon good cause shown, the Commission will extend such completion date for a reasonable period of time.

7. The holder of a manufacturing license issued pursuant to this appendix M shall submit to the Commission the final design of the nuclear power reactor(s) covered by the license as soon as such design has been completed. Such submittal shall be in the form of an application for amendment of the manufacturing license.

8. The prohibition in §50.10(c) against commencement of construction of a production or utilization facility prior to issuance of a construction permit applies to the transport of a nuclear power reactor(s) manufactured pursuant to a license issued pursuant to this appendix from the manufacturing facility to the site at which the reactor(s) will be installed and operated. In addition, such nuclear power reactor(s) shall not be removed from the manufacturing site until the final design of the reactor(s) has been approved by the Commission in accordance with paragraph 7.

9. An application for a permit to construct a nuclear power reactor(s) which is the subject of an application for a manufacturing license pursuant to this appendix M need not contain such information or analyses as have previously been submitted to the Commission in connection with the application for a manufacturing license, but shall contain, in addition to the information and analyses otherwise required by §§50.34(a) and 50.34a, sufficient information to demonstrate that the site on which the reactor(s) is to be operated falls within the postulated site parameters specified in the relevant manufacturing license application.

10. The Commission may issue a permit to construct a nuclear power reactor(s) which is the subject of an application for a manufacturing license pursuant to this appendix M if the Commission (a) finds that the site on which the reactor is to be operated falls within the postulated site parameters specified in the relevant application for a manufacturing license and (b) makes the findings otherwise required by this part. In no event will a construction permit be issued until the relevant manufacturing license has been issued.

11. An operating license for a nuclear power reactor(s) that has been manufactured under a Commission license issued pursuant to this appendix M may be issued by the Commission pursuant to §50.57 and subpart A of part 51 of this chapter except that the Commission shall find, pursuant to §50.57(a)(1), that construction of the reactor(s) has been substantially completed in conformity with both the manufacturing license and the construction permit and the applications therefor, as amended, and the provisions of the Act, and the rules and regulations of the Commission. Notwithstanding the other provisions of this paragraph, no application for an operating license for a nuclear power reactor(s) that has been manufactured under a Commission license issued pursuant to this appendix M will be docketed until the application for an amendment to the relevant manufacturing license required by paragraph 7 has been docketed.

12. In making the findings required by this part for the issuance of a construction permit or an operating license for a nuclear power reactor(s) that has been manufactured under a Commission license issued pursuant to this appendix, or an amendment to such a manufacturing license, construction permit, or operating license, the Commission will treat as resolved those matters which have been resolved at an earlier stage of the licensing process, unless there exists significant new information that substantially affects the conclusion(s) reached at the earlier stage or other good cause.

[38 FR 30253, Nov. 2, 1973, as amended at 49 FR 9404, Mar. 12, 1984; 49 FR 35754, Sept. 12, 1984; 50 FR 18853, May 3, 1985; 51 FR 40311, Nov. 6, 1986]

APPENDIX N TO PART 50—STANDARDIZATION OF NUCLEAR POWER PLANT DESIGNS: LICENSES TO CONSTRUCT AND OPERATE NUCLEAR POWER REACTORS OF DUPLICATE 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, 50.33a, 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.

For the technical information required by §§50.34(a) (1) through (5) and (8) and 50.34a (a) and (b), reference may be made to a single preliminary safety analysis of the design[2] which, for the purposes of §50.34(a)(1) includes one set of site parameters postulated for the design of the reactors, and an analysis and evaluation of the reactors in terms of such postulated site parameters. Such single preliminary safety analysis shall also include information pertaining to design features of the proposed reactors that affect plans for coping with emergencies in the operation of the reactors, and shall describe the quality assurance program with respect to aspects of design, fabrication, procurement and construction that are common to all of the reactors.

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]

APPENDIX O TO PART 50—STANDARDIZATION OF DESIGN: STAFF REVIEW OF STANDARD DESIGNS

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

1. Any person may submit a proposed preliminary or final standard design for a nuclear power reactor of the type described in §50.22 to the regulatory staff for its review. Such a submittal may consist of either the preliminary or final design for the entire reactor facility or the preliminary or final design of major portions thereof.

2. The submittal for review of the standard design must be made in the same manner and in the same number of copies as provided in §§50.4 and 50.30 for license applications.

3. The submittal for review of the standard design shall include the information described in §50.33(a) through (d) and the applicable technical information required by §§50.34 (a) and (b), as appropriate, and 50.34a (other than that required by §§50.34(a) (6) and (10), 50.34(b)(1), (6)(i), (ii), (iv), and (v) and 50.34(b) (7) and (8)). The submittal shall also include a description, analysis and evaluation of the interfaces between the submitted design and the balance of the nuclear power plant. With respect to the requirements of

[1] If the design for the power reactor(s) proposed in a particular application is not identical to the others, that application may not be processed under this appendix and subpart D of part 2 of this chapter.

[2] As used in this appendix, the design of a nuclear power reactor included in a single referenced safety analysis report means the design of those structures, systems and components important to radiological health and safety and the common defense and security.

§§50.34(a)(1), the submittal for review of a standard design shall include the site parameters postulated for the design, and an analysis and evaluation of the design in terms of such postulated site parameters. The information submitted pursuant to §50.34(a)(7) shall be limited to the quality assurance program to be applied to the design, procurement and fabrication of the structures, systems, and components for which design review has been requested and the information submitted pursuant to §50.34(a)(9) shall be limited to the qualifications of the person submitting the standard design to design the reactor or major portion thereof. The submittal shall also include information pertaining to design features that affect plans for coping with emergencies in the operation of the reactor or major portion thereof.

4. Once the regulatory staff has initiated a technical review of a submittal under this appendix, the submittal will be referred to the Advisory Committee on Reactor Safeguards (ACRS) for a review and report.

5. Upon completion of their review of a submittal under this appendix, the regulatory staff shall publish in the FEDERAL REGISTER a determination as to whether or not the preliminary or final design is acceptable, subject to such conditions as may be appropriate, and make available in the Public Document Room an analysis of the design in the form of a report. An approved design shall be utilized by and relied upon by the regulatory staff and the ACRS in their review of any individual facility license application which incorporates by reference a design approved in accordance with this paragraph unless there exists significant new information which substantially affects the earlier determination or other good cause.

6. The determination and report by the regulatory staff shall not constitute a commitment to issue a permit or license, or in any way affect the authority of the Commission, Atomic Safety and Licensing Appeal Panel, Atomic Safety and Licensing Panel, and other presiding officers in any proceeding under subpart G of part 2 of this chapter.

7. The Commission may, on its own initiative or in response to a petition for rule making, approve the design in a rulemaking proceeding and in that event, the approved design will be subject to challenge only as provided in §2.758 of this chapter. An environmental impact statement may be prepared for such a rule making action in accordance with §§51.20(b)(13) and 51.85 of this chapter. If an environmental impact statement is prepared, the Commission may require the petitioner for rulemaking to submit information to the Commission to aid the Commission in the preparation of the environmental impact statement.

8. Information requests to the approval holder regarding an approved design shall be evaluated 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 evaluation performed by the NRC staff shall be in accordance with 10 CFR 50.54(f) and shall be approved by the Executive Director for Operations or his or her designee prior to issuance of the request.

[40 FR 2977, Jan. 17, 1975, as amended at 49 FR 9405, Mar. 12, 1984; 50 FR 38112, Sept. 20, 1985; 51 FR 40311, Nov. 6, 1986]

APPENDIX 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 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 staff and, if appropriate by the ACRS, of a submittal under this appendix, the 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 a Staff Site Report, the Staff shall publish a notice of the availability of the report in the FEDERAL REGISTER and shall place copies of the report in the Commission's Public Document Room at 2120 L Street, NW., Washington, DC 20555, and in a Local Public Document Room(s) located near the site identified in the Staff Site Report. The 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]

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.[1]

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.

[1] Clarification and guidance with respect to permissible alternatives to satisfy Appendix A to BTP APCSB 9.5–1 has been provided in four other NRC documents:

''Supplementary Guidance on Information Needed for Fire Protection Evaluation,'' dated October 21, 1976;

''Sample Technical Specification,'' dated May 12, 1977;

''Nuclear Plant Fire Protection Functional Responsibilities, Administrative Control and Quality Assurance,'' dated June 14, 1977;

''Manpower Requirements for Operating Reactors,'' dated May 11, 1978.

A Fire Protection Safety Evaluation Report that has been issued for each operating plant states how these guidelines were applied to each facility and identifies open fire protection issues that will be resolved when the facility satisfies the appropriate requirements of appendix R to part 50.

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,[2] independent of cables, systems or components in the area, room or zone under consideration, shall 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.

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[2] Alternative shutdown capability is provided by rerouting, relocating or modificating of existing systems; dedicated shutdown capability is provided by installing new structures and systems for the function of post-fire shutdown.

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 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[3] conditions for a PWR (hot shutdown[3] 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, of 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] As defined in the Standard Technical Specifications.

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 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.[4]

M. *Fire barrier cable penetration seal qualification.* Penetration seal designs shall utilize only noncombustible materials and shall be qualified by tests that are comparable to tests used to rate fire barriers. The acceptance criteria for the test shall include:

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

[4] 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.

design basis accident conditions and that there is reasonable assurance that the system will withstand the Safe Shutdown Earthquake.[5]

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 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]

APPENDIX S TO PART 50—EARTHQUAKE ENGINEERING CRITERIA FOR NUCLEAR POWER PLANTS

GENERAL INFORMATION

This appendix applies to applicants for a design certification or combined license pursuant to part 52 of this chapter or a construction permit or operating license pursuant to part 50 of this chapter on or after January 10, 1997. However, for either an operating license applicant or holder whose construction permit was issued prior to January 10, 1997, the earthquake engineering criteria in Section VI of appendix A to 10 CFR part 100 continues to apply.

I. INTRODUCTION

(a) Each applicant for a construction permit, operating license, design certification, or combined license is required by §50.34 (a)(12), (b)(10), 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 pursuant to subpart C of part 52 of this chapter.

Design Certification means a Commission approval, issued pursuant to subpart B of part 52 of this chapter, of a standard design for a nuclear power facility. A design so approved may be referred to as a ''certified standard design.''

The *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.

A *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.

The *Safe Shutdown Earthquake Ground Motion* (SSE) is the vibratory ground motion for which certain structures, systems, and components must be designed to remain functional.

The *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.

[5] See Regulatory Guide 1.29—''Seismic Design Classification'' paragraph C.2.

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.

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