ENRICO FERMI UNIT 2

REGULATORY REQUIREMENTS REVIEW REPORT

FOR

FERMI 2

DETROIT EDISON COMPANY

September, 1981

INTRODUCTION

This document provides a summary of the conformance of Enrico Fermi Unit 2 to the NRC regulations of 10 CFR Parts 20, 50, and 100. Those sections of Parts 20, 50 and 100 which impose compliance requirements on licensees are addressed.

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SECTION

10 CFR 20.101(a)

Statement of Section

(a) In accordance with the provisions of 20.102(a) and except as provided in paragraph (b) of this section no licensee shall possess, use, or transfer licensed material in such a manner as to cause any individual in a restricted area to receive in any period of one calendar quarter from radioactive material and other sources of radiation a total occupational dose in excess of the standards specified in the following table:

	Rems per calendar quarter
1. Whole body: head and trunk: active blood-forming organs: lens of eyes or gonads	1 1/4
2. Hands and forearms: feet and ankles	18 3/4
3. Skin of whole body	7 1/2

Evaluation

Fermi 2 Health Physics Procedure "Radiation, Contamination and Airborne Guides and Limits," (61.000.05), requires that the external radiation exposure to individuals in restricted areas shall not exceed the limits set forth in 10 CFR 20.101(a).

SECTION

10 CFR 20.101(b)

Statement of Section

(b) A licensee may permit an individual in a restricted area to receive a total occupational dose to the whole body greater than that permitted under paragraph (a) of this section provided:

1. During any calendar quarter the total occupational dose to the whole body shall not exceed 3 rems, and

2. The dose to the whole body when added to the accumulated occupational dose to the whole body shall not exceed 5 (N-18) rems where "N" equals the individual's age in years at his last birthday, and

3. The licensee has determined the individual's accumulated occupational dose to the whole body on Form NRC-4, or on a clear and legible record containing all the information required in that form; and has otherwise complied with the requirements of 20.102. As used in paragraph (b), "Dose to the whole body" shall be deemed to include any dose to the whole body, gonads, active bloodforming organs, head and trunk, or lens of eye.

Evaluation

Fermi 2 Health Physics Procedure "Radiation, Contamination and Airborne Guides and Limits," (61.000.05), allows an individual to exceed a quarterly whole body dose of 1,250 mrem provided the requirements of 10 CFR 20.101(b) are met.

SECTION

10 CFR 20.102 (a)

Statement of Section

(a) Each licensee shall require any individual, prior to first entry of the individual into the licensee's restricted area during each employment or work assignment under such circumstances that the individual will receive or is likely to receive in any period of one calendar quarter an occupational dose in excess of 25 percent of the applicable standards specified 20.101(a) and 20.104(a), to disclose in a written signed statement either (1) that the individual had no prior occupational dose during the current calendar quarter or (2) the nature and amount . any occupational dose which the individual may have received during that specifically identified current calendar quarter from sources of radiation possessed or controlled by other persons. Each licensee shall maintain records of such statements until the Commission authorizes their disposition.

Evaluation

Fermi 2 Health Physics Procedure "Radiation, Contamination and Airborne Guides and Limits," (61.000.05) requires that all personnel complete an equivalent to NRC Form 4 prior to being occupationally exposed to ionizing radiation. Record keeping requirements will also be addressed in site procedures.

SECTION

10 CFR 20.102(b)

Statement of Section

(b) Before permitting, pursuant to 20.101(b), any individual in a restricted area to receive an occupational radiation dose in excess of the standards specified in 20.101(a) each licensee shall:

(1) Obtain a certificate on Form NRC-4, or on a clear and legible record containing all the information required in that form, signed by the individual showing each period of time after the individual attained the age of 18 in which the individual received an occupational dose of radiation; and

(2) Calculate on Form NRC-4 in accordance with the instructions appearing therein or on a clear and legible record containing all the information required in that form, the previously accumulated occupational dose received by the individual and the additional dose allowed for that individual under 20.101(b).

Evaluation

Fermi 2 Health Physics Procedure "Radiation, Contamination and Airborne Guides and Limits" (61.000.05), requires that all personnel complete an equivalent to NRC Form 4 prior to being occupationally exposed to ionizing radiation. The Fermi NRC Form 4 equivalent requires that each individual provide a record of exposure history, and sign the form. Space is also provided for clearly calculating the dose allowed under 10 CFR 20.101(b).

SECTION

10 CFR 20.102(c)

Statement of Section

(c)(1) In the preparation of Form NRC-4 or a clear and legible record containing all the information required in that form, the licensee shall make a reasonable effort to obtain reports of the individual's previously accumulated occupational dose. For each period for which the licensee obtains such reports the licensee shall use the dose shown in the report in preparing the form. In any case where a licensee is unable to obtain reports of the individual's occupational dose for a previous complete calendar quarter it shall be assumed that the individual has received the occupational dose specified in whichever of the following columns apply:

Part of Body	Column 1 Assumed exposure in rems for calendar quarters prior to January 1, 1981	Column 2 Assumed exposure in rems for calendar quarters beginning on or after January 1, 1981
Whole body, gonads, active blood-forming organs, head and trunk, lens of eye.	3 3/4	1 1/4

(2) The licensee shall retain and preserve records used in preparing Form NRC-4 until the Commission authorizes their disposition.

If calculation of the individual's accumulated occupational dose for all periods prior to January 1, 1961 yields a result higher than the applicable accumulated dose value for the individual as of that date as specified in paragraph (b) of 20.101 the excess may be disregarded.

Evaluation

The requirements of this section are contained in Fermi 2 Health Physics Procedure, "Personnel Radiation Exposure Records and Reports" (61.000.51), except that record keeping requirements are not specified. Record keeping requirements will be addressed in Health Physics Procedures "Personnel Radiation Exposure Records" (61.000.61) and "Issuance of Personnel Dosimetry (62.000.61).

SECTION

10 CFR 20.103(a)

Statement of Section

(a)(1) No licensee shall possess, use, or transfer licensed material in such a manner as to permit any individual in a restricted area to inhale a quantity of radioactive material in any period of one calendar quarter greater than the quantity which would result from inhalation for 40 hours per week for 13 weeks at uniform concentrations of radioactive material in air specified in Appendix B Table I, Column 1[123]. If the radioactive material is of such form that intake by absorption through the skin is likely, individual exposures to radioactive material shall be controlled so that the uptake of radioactive material by any organ from either inhalation or absorption or both routes of intake in any calendar quarter does not exceed that which would result from inhaling such radioactive material for 40 hours per week for 13 weeks at uniform concentrations specified in Appendix B, Table I, Column 1.

(2) No licensee shall possess, use or transfer mixtures of U-234, U-235, and U-238 in soluble form in such a manner as to permit any individual in a restricted area to inhale a quantity of such material in excess of the intake limits specified in Appendix B, Table 1, Column 1 of this part. If such soluble uranium is of a form such that absorption through the skin is likely, individual exposures to such material shall be controlled so that the uptake of such material by any organ from either inhalation or absorption or both routes of intake[1] does not exceed that which would result from inhaling such material at the limits specified in Appendix B, Table 1, Column 1 and footnote 4 thereto.

(3) For purpose of determining compliance with the requirements of this section the licensee shall use suitable measurements of concentrations of radioactive materials in air for detecting and evaluating airborne radioactivity in restricted areas and in addition, as appropriate, shall use measurements of radioactivity in the body, measurements of radioactivity excreted from the body, or any combination of such measurements as may be necessary for timely detection and assessment of individual intakes of radioactivity by exposed individuals. It is assumed that an individual inhales radioactive material at the airborne concentration in which he is present unless he uses respiratory protective equipment pursuant to paragraph (c) of this section. When assessment of a particular individual's intake of radioactive material is necessary, intakes less than those which would result from inhalation for 2 hours in any one day or for 10 hours in any one week at uniform concentrations specified in Appendix B, Table I, Column 1 need not be included in such assessment provided that for any assessment in excess of these amounts the entire amount is included.

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[1] Since the concentration specified for tritium oxide vapor assumes equal intakes by skin absorption and inhalation, the total intake permitted is twice that which would result from inhalation alone at the concentration specified for H 3 S in Appendix B, Table I, Column 1 for 40 hours per week for 13 weeks.

[2] For radon-222, the limiting quantity is that inhaled in a period of one calendar year. For radioactive materials designated "Sub" in the "Isotope" column of the table the concentration value specified is based upon exposure to the material as an external radiation source. Individual exposures to these materials may be accounted for as part of the limitation on individual dose in 20.101. These nuclides shall be subject to the precautionary procedures required by 20.103(b)(1).

[3] Multiply the concentration values specified in Appendix B, Table I, column 1, by 6.3 x 10^3 ml to obtain the quarterly quantity limit. Multiply the concentration value

specified in Appendix B, Table I, Column 1, by 2.5 x 10^9 ml to obtain the annual quantity limit for Rn-222.

4 Significant intake by ingestion or injection is presumed to occur only as a result of circumstances such as accident, inadvertence, poor procedure, or similar special conditions. Such intakes must be evaluated and accounted for by techniques and procedures as may be appropriate to the circumstances of the occurrence. Exposures so evaluated shall be included in determining whether the limitation on individual exposures in 20.103(a)(1) has been exceeded.

5 Regulatory guidance on assessment of individual intakes of radioactive material is given in Regulatory Guide 3.9. "Acceptable Concepts, Models, Equations and Assumptions for a Bioassay Program," single copies of which are available from the Office of Standards Development U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 upon written request.

Evaluation

1. Fermi 2 Health Physics Procedure "Radiation, Contamination, and Airborne Guides and Limits (61.000.05) stipulates that no individual shall be permitted to receive more than 520 mpc hours in a calendar quarter. Limits are also provided for loose and fixed surface contamination which could be absorbed through the skin.

2. Fermi 2 does not possess mixtures of U-234, U-235 or U-238 in soluble form.

3. Fermi 2 Health Physics Procedure "MPC Hour Determination" (61.000.06), provides instructions for calculating the MPC ratio. Fermi 2 Health Physics Procedure "Airborne Radioactivity Survey Techniques" (62.002.30), provides instructions on the performance of airborne radioactivity surveys, and provides instructions for the conduct of measurements of radioactivity within the body. Fermi 2 Health Physics Procedure, "Bioassay Program" (62.001.30) also stipulates that whole body counting be used when an individual has been

exposed to airborne radioactivity in excess of 30 mpc-hours in a quarter. Guidance on the conduct of excrete collection and analysis is also provided.

SECTION

10 CFR 20.103(b)

Statement of Section

(b)(1) The licensee shall, as a precautionary procedure, use process or other engineering controls to the extent practicable to limit concentrations of radioactive materials in air to levels below those which delimit an airborne radioactivity area as defined in 20.203(d)(1)(ii).

(2) When it is impracticable to apply process or other engineering controls to limit concentrations of radioactive material in air below those defined in 20.203(d)(1)(ii), other precautionary procedures such as increased surveillance, limitation of working times, or provision of respiratory protective equipment shall be used to maintain intake of radioactive material by any individual within any period of seven consecutive days as far below that intake of radioactive material which would result from inhalation of such material for 40 hours at the uniform concentrations specified in Appendix B, Table 1, Column 1 as is reasonably achievable. Whenever the intake of radioactive material by any individual exceeds this 40 hour control measure the licensee shall make such evaluations and take such actions as are necessary to assure against recurrence. The licensee shall maintain records of such occurrences, evaluations, and actions taken in a clear and readily identifiable form suitable for summary review and evaluation.

Evaluation

Fermi 2 Health Physics Procedure, "ALARA Program" (61.000.02) provides general guidance on exposure reduction techniques including the use of process and engineering controls. Fermi 2 will meet the requirements of this section.

SECTION

10 CFR 20.103(e)(f)

Statement of Section

(e) The licensee shall notify in writing the Director of the appropriate Nuclear Regulatory Commission Inspection and Enforcement Regional Office listed in Appendix D at least 30 days before the date that respiratory protective equipment is first used under the provisions of this section.

(f) A licensee who was authorized to make allowance for use of respiratory protective equipment prior to December 29, 1976 shall bring his respiratory protective program into conformance with the requirements of paragraph (c) of this section within one year of that date and is exempt from the requirement of paragraph (e) of this section.

Evaluation

Fermi 2 is aware of and commits to the applicable requirements of these sections.

SECTION

10 CFR 20.104

Statement of Section

(a) No licensee shall possess, use, or transfer licensed material in such a manner as to cause any individual within a restricted area who is under 18 years of age to receive in any period of one calendar quarter from radioactive material and other sources of radiation in the licensee's possession a dose in excess of 10 percent of the limits specified in the table in paragraph (a) of 20.101.

(b) No licensee shall possess, use, or transfer licensed material in such a manner as to cause any individual within a restricted area who is under 18 years of age to be exposed to airborne radioactive material possessed by the licensee in an average concentration in excess of the limits specified in Appendix B, Table II of this part. For purposes of this paragraph concentrations may be averaged over periods not greater than a week.

(c) The provisions of 20.103(b) (2) and 20.103(c) shall apply to exposures subject to paragraph (b) of this section except that the reference in 20.103 (b)(2) and 20.103(c) to Appendix B, Table I, Column 1 shall be deemed to be references to Appendix B, Table II, Column 1.

Evaluation

Fermi 2 Health Physics Procedure "Radiation, Contamination and Airborne Guides and Limits" (61.000.05) states that persons under 18 years of age will not be occupationally exposed to ionizing radiation at Fermi 2. Fermi 2 will meet the requirements at this section before allowing any visitors under the age of 18 to areas of the site where they could be exposed to ionizing radiation.

SECTION

10 CFR 20.105(a)

Statement of Section

(a) There may be included in any application for a license or for amendment of a license proposed limits upon levels of radiation in unrestricted areas resulting from the applicant's possession or use of radioactive material and other sources of radiation. Such applications should include information levels and anticipated occupancy times for each unrestricted area involved. The Commission will approve the proposed limits if the applicant demonstrates that the proposed limits are not likely to cause any individual to receive a dose to the whole body in any period of one calendar year in excess of 0.5 rem.

Evaluation

Fermi 2 meets the requirements of 10 CFR 20.105(b) and is not applying for less restrictive limits.

SECTION

10 CFR 20.105(b) (c)

Statement of Section

(b) Except as authorized by the Commission pursuant to paragraph (a) of this section, no licensee shall possess, use, or transfer licensed material in such a manner as to create in any unrestricted area from radioactive material and other sources of radiation in his possession:

1. Radiation levels which, if an individual were continuously present in the area, could result in his receiving a dose in excess of two millirems in any one hour or

2. Radiation levels which, if an individual were continuously present in the area could result in his receiving a dose in excess of 100 millirems in any seven consecutive days.

(c) In addition to other requirements of this part, licensees engaged in uranium fuel cycle operations subject to the provisions of 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," shall comply with that part.

Evaluation

(b) Fermi 2 Health Physics Procedure "Radiation, Contamination and Airborne Guides and Limits" (61.000.05), states that external radiation exposure to individuals in an unrestricted area shall be limited to 500 mrem per year, 2 mrem per hour, or 100 mrem in any seven consecutive days.

(c) The requirements of this section will be included in the Radiological Effluent Technical specifications.

Fermi 2 has committed to the requirements of these sections.

SECTION

10 CFR 20.106

Statement of Section

(a) A licensee shall not possess, use, or transfer licensed material so as to release to an unrestricted area radioactive material in concentrations which exceed the limits specified in Appendix "B", Table II of this part, except as authorized pursuant to 20.302 or paragraph (b) of this section. For purposes of this section concentrations may be averaged over a period not greater than one year.

(b) An application for a license or amendment may include proposed limits higher than those specified in paragraph (a) of this section. The Commission will approve the proposed limits if the applicant demonstrates:

1. That the applicant has made a reasonable effort to minimize the radioactivity contained in effluents to unrestricted areas; and

2. That it is not likely that radioactive material discharged in the effluent would result in the exposure of an individual to concentrations or radioactive material in air or water exceeding the limits specified in Appendix "B", Table II of this part.

(c) An application for higher limits pursuant to paragraph (b) of this section shall include information demonstrating that the applicant has made a reasonable effort to minimize the radioactivity discharged in effluents to unrestricted areas, and shall include, as pertinent:

1. Information as to flow rates, total volume of effluent, peak concentration of each radionuclide in the effluent, and concentration of each radionuclide in the effluent averaged over a period of one year at the point where the effluent leaves a stack, tube, pipe, or similar conduit:

2. A description of the properties of the effluents, including:

(i) chemical composition;

(ii) physical characteristics, including suspended solids content in liquid effluents, and nature of gas or aerosol for air effluents;

(iii) the hydrogen ion concentrations (p^H) of liquid effluents; and

(iv) the size range of particulates in effluents released into air.

3. A description of the anticipated human occupancy in the unrestricted area where the highest concentration of radioactive material from the effluent is expected, and, in the case of a river or stream, a description of water uses downstream from the point of release of the effluent.

4. Information as to the highest concentration of each radionuclide in an unrestricted area, including anticipated concentrations averaged over a period of one year:

(i) In air at any point of human occupancy; or

(ii) In water at points of use downstream from the point of release of the effluent.

5. The background concentration of radionuclides in the receiving river or stream prior to the release of liquid effluent.

6. A description of the environmental monitoring equipment, including sensitivity of the system, and procedures and calculations to determine concentrations of radionuclides in the unrestricted area and possible reconcentrations of radionuclides.

7. A description of the waste treatment facilities and procedures used to reduce the concentration of radionuclides in effluents prior to their release.

(d) For the purposes of this section the concentration limits in Appendix "B", Table II of this part shall apply at the boundary of the restricted area. The concentration of radioactive material discharged through a stack, pipe, or similar conduit may be determined with respect to the point where the material leaves the conduit. If the conduit discharges within the restricted area, the concentration at the boundary may be determined by applying appropriate factors for dilution, dispersion, or decay between the point of discharge and the boundary.

(e) In addition to limiting concentrations in effluent streams, the Commission may limit quantities of radioactive materials released in air or water during a specified period of time if it appears that the daily intake of radioactive material from air, water, or food by a suitable sample of an exposed population group, averaged over a period not exceeding one year, would otherwise exceed the daily intake resulting from continuous exposure to air or water containing one-third the concentration of radioactive materials specified in Appendix "B", Table II of this part.

(f) The provisions of paragraphs (a) through (e) of this section do not apply to disposal of radioactive material into sanitary sewerage systems, which is governed by 20.303.

(g) In addition to other requirements of this part, licensees engaged in uranium fuel cycle operations subject to the provisions of 40 CFR Part 190, "Environmental Radiation Protection Standard for Nuclear Power Operations," shall comply with that part.

Evaluation

The requirements of these sections will be addressed in the Radiological Effluent Technical Specifications and associated implementing procedures.

Fermi 2 has committed to the requirements of these sections.

SECTION

10 CFR 20.202

Statement of Section

(a) Each licensee shall supply appropriate personnel monitoring equipment to, and shall require the use of such equipment by:

1. Each individual who enters a restricted area under such circumstances that he receives, or is likely to receive, a dose in any calendar quarter in excess of 25 percent of the applicable value specified in paragraph (a) of 20.101.

2. Each individual under 18 years of age who enters a restricted area under such circumstances that he receives, or is likely to receive a dose in any calendar quarter in excess of 5 percent of the applicable value specified in paragraph (a) of 20.101.

3. Each individual who enters a high radiation area.

(b) As used in this part,

1. "Personnel monitoring equipment" means devices designed to be worn or carried by an individual for the purpose of measuring the dose received (e.g., film badges, pocket chambers, pocket dosimeters, film rings, etc.).

2. "Radiation area" means any area, accessible to personnel, in which there exists radiation, originating in whole or in part within licensed material, at such levels that a major portion of the body could receive in any one hour a dose in

excess of 5 millirems, or in any 5 consecutive days a dose in excess of 100 millirems;

3. "High radiation area" means any area, accessible to personnel, in which there exists radiation originating in whole or in part within licensed material at such levels that a major portion of the body could receive in any one hour a dose in excess of 100 millirems.

Evaluation

FSAR Section 12.3.3.1.1 requires that all personnel who enter a radiation-restricted area are monitored with a TLD. Section 12.3.3.1.2 requires that all personnel who enter a high radiation area or a radiation area wear a self-reading dosimeter. Section 12.3.3.1.3 requires that a neutron film badge be worn by workers in areas where neutron radiation is present. Section 12.3.3.1.4 requires extremity and other special dosimetry in special cases. Section 12.3.3.2 requires the use of internal dosimetry at regular intervals and as conditions warrant. Fermi 2 is in compliance with the requirements of this section.

SECTION

10 CFR 20.203 (a)

Statement of Section

(a) General. (1) Except as otherwise authorized by the Commission, symbols prescribed by this section shall use the conventional radiation caution colors (magenta or purple on yellow background). The symbol prescribed by this section is the conventional three-bladed design:

RADIATION SYMBOL

1. Cross-hatched area is to be magenta or purple.

2. Background is to be yellow.

(2) In addition to the contents of signs and labels prescribed in this section, licensees may provide on or near such signs and labels any additional information which may be appropriate in aiding individuals to minimize exposure to radiation or to radioactive material.

Evaluation

The Fermi 2 FSAR on page A-65 states that Fermi 2 will conform to Regulatory Guide 8.1, "Radiation Symbol." This regulatory guide provides acceptable means to implement 10 CFR 20.203.

SECTION

10 CFR 20.203 (b) (c)

Statement of Section

(b) Radiation areas. Each radiation area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION[1]
RADIATION AREA

(c) High radiation areas. (1) Each high radiation area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION[1]
HIGH RADIATION AREA

2. Each entrance or access point to a high radiation area shall be:

(i) Equipped with a control device which shall cause the level of radiation to be reduced below that at which an individual might receive a dose of 100 millirems in 1 hour upon entry into the area; or

(ii) Equipped with a control device which shall energize a conspicuous visible or audible alarm signal in such a manner that the individual entering the high radiation area and the licensee or a supervisor of the activity are made aware of the entry; or

(iii) Maintained locked except during periods when access to the area is required, with positive control over each individual entry.

3. The controls required by subparagraph (2) of this paragraph shall be established in such a way that no individual will be prevented from leaving a high radiation area.

4. In the case of a high radiation area established for a period of 30 days or less, direct surveillance to prevent unauthorized entry may be substituted for the controls required by subparagraph (2) of this paragraph.

5. Any licensee, or applicant for a license, may apply to the Commission for approval of methods not included in subparagraphs (2) and (4) of this paragraph for controlling access to high radiation areas. The Commission will approve the proposed alternatives if the licensee or applicant demonstrates that the alternative methods of control will prevent unauthorized entry into a high radiation area, and that the requirement of subparagraph (3) of this paragraph is met.

6. Each area in which there may exist radiation levels in excess of 500 rems in one hour at one meter from a sealed radioactive source that is used to irradiate materials shall:

(i) Have each entrance or access point equipped with entry control devices which shall function automatically to prevent any individual from inadvertently entering the area when such radiation levels exist; permit deliberate entry into the area only after a control device is actuated that shall cause the radiation level within the area, from the sealed source to be reduced below that at which it would be possible for an individual to receive a dose in excess of 100 mrem in one hour; and prevent operation of the source if the source would produce radiation levels in the area that could result in a

dose to an individual in excess of 10 mrem in one hour. The entry control devices required by this paragraph (c) (6) shall be established in such a way that no individual will be prevented from leaving the area.

(ii) Be equipped with additional control devices such that upon failure of the entry control devices to function as required by paragraph (c) (6) (i) of this section the radiation level within the area, from the sealed source, shall be reduced below that at which it would be possible for an individual to receive a dose in excess of 100 mrem in one hour; and visible and audible alarm signals shall be generated to make an individual attempting to enter the area aware of the hazard and the licensee or at least one other individual who is familiar with the activity and prepared to render or summon assistance, aware of such failure of the entry control devices.

(iii) Be equipped with control devices such that upon failure or removal of physical radiation barriers other than the source's shielded storage container the radiation level from the source shall be reduced below that at which it would be possible for an individual to receive a dose in excess of 100 mrem in one hour; and visible and audible alarm signals shall be generated to make potentially affected individuals aware of the hazard and the licensee or at least one other individual, who is familiar with the activity and prepared to render or summon assistance, aware of the failure or removal of the physical barrier. When the shield for the stored source is a liquid, means shall be provided to monitor the integrity of the shield and to signal, automatically, loss of adequate shielding. Physical radiation barriers that comprise permanent structural components, such as walls, that have no credible probability of failure or removal in ordinary circumstances need not meet the requirements of this paragraph (c) (6) (iii).

(iv) Be equipped with devices that will automatically generate visible and audible alarm signals to alert personnel in the area before the source can be put into operation and in sufficient time for any individual in the area to operate a clearly identified control device which shall be installed in the area and which can prevent the source from being put into operation.

(v) Be controlled by use of such administrative procedures and such devices as are necessary to assure that the area is cleared of personnel prior to each use of the source preceding which use it might have been possible for an individual to have entered the area.

(vi) Be checked by a physical radiation measurement to assure that prior to the first individual's entry into the area after any use of the source, the radiation level from the source in the area is below that at which it would be possible for an individual to receive a dose in excess of 100 mrem in one hour.

(vii) Have entry control devices required in paragraph (c) (6) (i) of this section which have been tested for proper functioning prior to initial operation with such source of radiation on any day that operations are not uninterruptedly continued from the previous day or before resuming operations after any unintended interruption, and for which records are kept of the dates, times, and results of such tests of function. No operations other than those necessary to place the source in safe condition or to effect repairs on controls shall be conducted with such source unless control devices are functioning properly. The licensee shall submit an acceptable schedule for more complete periodic tests of the entry control and warning systems to be established and adhered to as a condition of the license.

(viii) Have those entry and exit portals that are used in transporting materials to and from the irradiation area, and that are not intended for use by individuals, controlled by such devices and administrative procedures as are necessary to physically protect and warn against inadvertent entry by any individual through such portals. Exit portals for processed materials shall be equipped to detect and signal the presence of loose radiation sources that are carried toward such an exit and to automatically prevent such loose sources from being carried out of the area.

7. Licensees with, or applicants for, licenses for radiation sources that are within the purview of paragraph (c) (6) of this section, and that must be used in a variety of positions or in peculiar locations, such as open fields or forests, that make it impracticable to comply with certain requirements of paragraph (c) (6) of this section, such as those for the automatic control of radiation levels, may apply to the Director, Office of Nuclear Material Safety and Safeguards, U. S. Nuclear Regulatory Commission, Washington D. C., 20555, for approval, prior to use of safety measures that are alternative to those specified in paragraph (c) (6) of this section, and that will provide at least an equivalent degree of personnel protection in the use of such sources. At least one of the alternative measures must include an entry-preventing interlock control based on a physical measurement of radiation that assures the absence of high radiation levels before an individual can gain access to an area where such sources are used.

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1 This paragraph (c) (6) does not apply to radioactive sources that are used in teletherapy, in radiography, or in completely self-shielded irradiators in which the source is both stored and operated within the same shielding radiation barrier and, in the designed configuration of the irradiator, is always physically inaccessible to any individual and cannot create high levels of radiation in an area that is accessible to any individual. This paragraph (c) (6) also does not apply to sources from which the radiation is incidental to some

other use nor to nuclear reactor generated radiation other than radiation from byproduct, source, or special nuclear materials that are used in sealed sources in non-self-shielded irradiators.

2 These requirements apply after March 14, 1978. Each person licensed to conduct activities to which this paragraph (c) (6) applies and who is not in compliance with the provisions of this paragraph on March 14, 1978, shall file with the Director, Office of Nuclear Material Safety and Safeguards, U. S. Nuclear Regulatory Commission, Washington, D. C., 20555, on or before June 14, 1978, information describing in detail the actions taken or to be taken to achieve compliance with this paragraph by December 14, 1978, and may continue activities in conformance with present license conditions and the provisions of the previously effective 20.203 until such compliance is achieved. For such persons compliance must be achieved not later than December 14, 1978.

Evaluation

The requirements of this section will be addressed in Health Physics Procedure "Posting for Radiation Control" (61.001.15).

SECTION

10 CFR 20.203 (d)

Statement of Section

(d) Airborne radioactivity areas. (1) As used in the regulations in this part, "airborne radioactivity area" means (i) any room, enclosure, or operating area in which airborne radioactive materials, composed wholly or partly of licensed material, exist in concentrations in excess of the amounts specified in Appendix B, Table I, Column 1 of this part; or (ii) any room, enclosure, or operating area in which airborne radioactive material composed wholly or partly of licensed material exists in concentrations which, averaged over the number of hours in any week during which individuals are in the area, exceed 25 percent of the amounts specified in Appendix B, Table I, Column 1 of this part.

2. Each airborne radioactivity area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION[1]
AIRBORNE RADIOACTIVITY AREA

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[1] Or "Danger"

Evaluation

The requirements of this section will be addressed in Health Physics Procedure "Posting for Radiation Control" (61.000.15).

SECTION

10 CFR 20.203 (e)

Statement of Section

(e) Additional requirements. (1) Each area or room in which licensed material is used or stored and which contains any radioactive material (other than natural uranium or thorium) in an amount exceeding 10 times the quantity of such material specified in Appendix C of this part shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION[1]
RADIOACTIVE MATERIAL(S)

2. Each area or room in which natural uranium or thorium is used or stored in an amount exceeding one-hundred times the quantity specified in Appendix C of this part shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION[1]
RADIOACTIVE MATERIAL(S)

(f) Containers. (1) Except as provided in subparagraph (3) of this paragraph, each container of licensed material shall bear a durable, clearly visible label identifying the radioactive contents.

2. A label required pursuant to subparagraph (1) of this paragraph shall bear the radiation caution symbol and the words "CAUTION, RADIOACTIVE MATERIAL" or "DANGER, RADIOACTIVE MATERIAL". It shall also provide sufficient information[1] to permit individuals handling or using the containers, or

working in the vicinity thereof, to take precautions to avoid or minimize exposures.

3. Notwithstanding the provisions of subparagraph (1) of this paragraph, labeling is not required:

(i) For containers that do not contain licensed materials in quantities greater than the applicable quantities listed in Appendix C of this part.

(ii) For containers containing only natural uranium or thorium in quantities no greater than 10 times the applicable quantities listed in Appendix C of this part.

(iii) For containers that do not contain licensed materials in concentrations greater than the applicable concentrations listed in Column 2, Table I, Appendix B of this part.

(iv) For containers when they are attended by an individual who takes the precautions necessary to prevent the exposure of any individual to radiation or radioactive materials in excess of the limits established by the regulations in this part.

(v) For containers when they are in transport and packaged and labeled in accordance with regulations of the Department of Transportation.

(vi) For containers which are accessible[2] only to individuals authorized to handle or use them, or to work in the vicinity thereof, provided that the contents are identified to such individuals by a readily available written record.

(vii) For manufacturing or process equipment, such as nuclear reactors, reactor components, piping and tanks.

4. Each licensee shall, prior to disposal of an empty uncontaminated container to unrestricted areas, remove or deface the radioactive material label or otherwise clearly indicate that the container no longer contains radioactive materials.

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[1] As appropriate, the information will include radiation levels, kinds of material, estimate of activity, date for which activity is estimated, mass enrichment, etc.

[2] For example, containers in locations such as water-filled canals, storage vaults, or hot cells.

Evaluation

The requirements of this section will be addressed in Health Physics Procedure "Posting for Radiation Control," (61.000.15).

SECTION

10 CFR 20.205 (a)

Statement of Section

(a) (1) Each licensee who expects to receive a package containing quantities of radioactive material in excess of the Type A quantities specified in paragraph (b) of this section shall:

(i) If the package is to be delivered to the licensee's facility by the carrier, make arrangements to receive the package when it is offered for delivery by the carrier; or

(ii) If the package is to be picked up by the licensee at the carrier's terminal, make arrangements to receive notification from the carrier of the arrival of the package, at the time of arrival.

(2) Each licensee who picks up a package of radioactive material from a carrier's terminal shall pick up the package expeditiously upon receipt of notification from the carrier of its arrival.

Evaluation

Fermi 2 Health Physics Procedure "Procurement and Receipt of Radioactive Materials" (67.000.10) requires that all procurement of radioactive materials be coordinated through Health Physics and that Health Physics be notified prior to allowing a sole use vehicle on site.

Fermi 2 does not plan to pick up packages of radioactive material at a carrier's terminal. If such a package were picked up at a terminal, the requirements of this section would be met.

SECTION

10 CFR 20.205 (b)

Statement of Section

(b) (1) Each licensee, upon receipt of a package of radioactive material, shall monitor the external surfaces of the package for radioactive contamination caused by leakage of the radioactive contents, except;

(i) Packages containing no more than the exempt quantity specified in the table in this paragraph;

(ii) Packages containing no more than 10 millicuries of radioactive material consisting solely of tritium, carbon-14, sulfur-35, or iodine-125;

(iii) Packages containing only radioactive material as gases or in special form;

(iv) Packages containing only radioactive material in other than liquid form (including Mo-99/Tc-99m generators) and not exceeding the Type A quantity limit specified in the table in this paragraph; and

(v) Packages containing only radionuclides with half-lives of less than 30 days and a total quantity of no more than 100 millicuries.

The monitoring shall be performed as soon as practicable after receipt, but no later than three hours after the package is received at the licensee's facility if received during the licensee's normal working hours, or eighteen hours if received after normal working hours.

(2) If removable radioactive contamination in excess of 0.01 microcuries (22,000 disintegrations per minute) per 100 square centimeters of package surface is found on the external surfaces of the package, the licensee shall immediately notify the final delivering carrier and, by telephone and telegraph, mailgram, or facsimile, the appropriate Nuclear Regulatory Commission Inspection and Enforcement Regional Office shown in Appendix D.

TABLE OF EXEMPT AND TYPE A QUANTITIES

Transport Group[1]	Exempt quantity limit (in millicuries)	Type A quantity limit (in curies)
I	.01	0.001
II	0.1	0.050
III	1	3
IV	1	20
V	1	20
VI	1	1000
VII	25 000	1000
Special Form	1	20

[1] The definitions of "transport group" and "special form" are specified in 71.4 of this chapter.

Evaluation

Fermi 2 Health Physics Procedure "Procurement and Receipt of Radioactive Material" (67.000.10) will be modified to require that radioactive material be surveyed within three hours following receipt. The procedure requires that Health Physics

supervision must be notified if the limits found in site procedures are exceeded. Health Physics Procedure "Shipment of Radioactive Material" specifies a loose surface contamination limit of 22,000 dpm/100 cm^2.

Other requirements of this section will be added to this Health Physics Procedure.

SECTION

10 CFR 20.206

Statement of Section

Instructions required for individuals working in or frequenting any portion of a restricted area are specified in 19.12 of this chapter.

Evaluation

FSAR Section 12.4.1.1 states that the radiation training program meets the requirements of 10 CFR 19 and 20.

SECTION

10 CFR 20.207

Statement of Section

(a) Licensed materials stored in an unrestricted area shall be secured from unauthorized removal from the place of storage.

(b) Licensed materials in an unrestricted area and not in storage shall be tended under the constant surveillance and immediate control of the licensee.

Evaluation

FSAR paragraph 12.4.2 states that any material stored in an unrestricted area will be secured in accordance with 10 CFR 20.207.

SECTION

10 CFR 20.301

Statement of Section

No licensee shall dispose of licensed material except:

a. By transfer to an authorized recipient as provided in the regulations in Part 30, 40, 60, or 70 of this chapter, whichever may be applicable; or

b. As authorized pursuant to 20.302; or

c. As provided in 20.303, applicable to the disposal of licensed material by release into sanitary sewerage systems, or in 20.306 for disposal of specific wastes, or in 20.106 (Radioactivity in effluents to unrestricted areas).

Evaluation

Fermi 2 has committed to the requirements of this section.

SECTION

10 CFR 20.303

Statement of Section

No licensee shall discharge licensed material into a sanitary sewerage system unless:

a. It is readily soluble or dispersible in water; and

b. The quantity of any licensed or other radioactive material released into the system by the licensee in any one day does not exceed the larger of subparagraphs (1) or (2) of this paragraph:

1. The quantity which, if diluted by the average daily quantity of sewage released into the sewer by the licensee will result in an average concentration equal to the limits specified in Appendix B, Table I, Column 2 of this part; or

2. Ten times the quantity of such material specified in Appendix C of this part; and

c. The quantity of any licensed or other radioactive material released in any one month, if diluted by the average monthly quantity of water released by the licensee, will not result in an average concentration exceeding the limits specified in Appendix B, Table I, Column 2 of this part; and

d. The gross quantity of licensed and other radioactive material, excluding hydrogen-3 and carbon-14, released into the sewerage system by the licensee does not exceed one curie per year. The quantities of hydrogen-3 and carbon-14 released

SECTION

10 CFR 20.304

Statement of Section

No licensee shall dispose of licensed material by burial in soil unless:

a. The total quantity of licensed and other radioactive materials buried at any one location and time does not exceed, at the time of burial, 1,000 times the amount specified in Appendix C of this part; and

b. Burial is at a minimum depth of four feet; and

c. Successive burials are separated by distances of at least six feet and not more than 12 burials are made in any year.

Evaluation

The requirements of this section will be addressed in Health Physis Procedure "Health Physics Section Policy and Objectives" (61.000.01).

into the sanitary sewerage system may not exceed 5 curies per year for hydrogen-3 and 1 curie per year for carbon-14. Excreta from individuals undergoing medical diagnosis or therapy with radioactive material shall be exempt from any limitations contained in this section.

Evaluation

The requirements of this section will be addressed in Health Physics Procedure "Health Physics Section Policy and Objectives" (61.000.01).

SECTION

10 CFR 20.401

Statement of Section

(a) Each licensee shall maintain records showing the radiation exposures of all individuals for whom personnel monitoring is required under 20.202 of the regulations in this part. Such records shall be kept on Form NRC-5, in accordance with the instructions contained in that form or on clear and legible records containing all the information required by Form NRC-5. The doses entered on the forms or records shall be for periods of time not exceeding one calendar quarter.

(b) Each licensee shall maintain records in the same units used in this part, showing the results of surveys required by 20.201(b), monitoring required by 20.205(b) and 20.205(c), and disposals made under 20.302, 20.303 and 20.304.

(c) (1) Records of individual exposure to radiation and to radioactive material which must be maintained pursuant to the provisions of paragraph (a) of this section and records of bioassays, including results of whole body counting examinations, made pursuant to 20.108, shall be preserved until the Commission authorizes disposition.

2. Records of the results of surveys and monitoring which must be maintained pursuant to paragraph (b) of this section shall be preserved for two years after completion of the survey except that the following records shall be maintained until the Commission authorizes their disposition: (i) records of the results of surveys to determine compliance with 20.103(a); (ii) in the absence of personnel monitoring data, records of the results of surveys to determine external radiation dose; and (iii) records of the results of surveys used

to evaluate the release of radioactive effluents to the environment.

3. Records of disposal of licensed material made pursuant to 20.302, 20.303, and 20.304 are to be maintained until the Commission authorizes their disposition.

4. Records which must be maintained pursuant to this part may be 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.

5. 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 20.501, has granted a specific exemption from the record retention requirements specified in the regulations in this part.

Evaluation

FSAR section 12.3.1.3 requires records of all radiological surveys to be kept in accordance with 10 CFR 20.401, records of effluent releases in accordance with Regulatory Guide 1.21, and records of occupational exposure in accordance with Regulatory Guide 8.7.

SECTION

10 CFR 20.402

Statement of Section

(a) Each licensee shall report by telephone to the Director of the appropriate Nuclear Regulatory Commission Inspection and Enforcement Regional Office listed in Appendix D, immediately after its occurrence becomes known to the licensee, any loss or theft of licensed material in such quantities and under such circumstances that it appears to the licensee that a substantial hazard may result to persons in unrestricted areas.

(b) Each licensee who is required to make a report pursuant to paragraph (a) of this section shall, within thirty (30) days after he learns of the loss or theft make a report in writing to the appropriate NRC Regional Office listed in Appendix D with copies to the Director of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington D. C., 20555, setting forth the following information:

1. A description of the licensed material involved, including kind, quantity, chemical and physical form;

2. A description of the circumstances under which the loss or theft occurred;

3. A statement of disposition or probable disposition of the licensed material involved;

4. Radiation exposures to individuals, circumstances under which the exposures occurred, and the extent of possible hazard to persons in unrestricted areas;

5. Actions which have been taken, or will be taken, to recover the material; and

6. Procedures or measures which have been or will be adopted to prevent a recurrence of the loss or theft of licensed material.

(c) Subsequent to filing the written report the licensee shall also report any substantive additional information on the loss or theft which becomes available to the licensee, within 30 days after he learns of such information.

(d) Any report filed with the Commission pursuant to this section shall be so prepared that names of individuals who may have received exposure to radiation are stated in a separate part of the report.

Evaluation

Fermi 2 Health Physics Procedure "Reports of Incidents Involving NRC Licensed Radioactive Material" (61.000.70) requires that the NRC be immediately notified of any loss or theft of licensed radioactive material under such circumstances that it appears that a substantial hazard may result to persons in unrestricted areas.

The procedure also requires that a report to NRC involving the theft or loss of radioactive material must contain the information specified in 10 CFR 20.402(b).

SECTION

10 CFR 20.403

Statement of Section

(a) Immediate notification. Each licensee shall immediately notify by telephone and telegraph, mailgram, or facsimile, the Director of the appropriate NRC Regional Office listed in Appendix D of any incident involving byproduct, source, or special nuclear material possessed by him and which may have caused or threatens to cause:

1. Exposure of the whole body of any individual to 25 rems or more of radiation; exposure of the skin of the whole body of any individual of 150 rems or more of radiation; or exposure of the feet, ankles, hands or forearms of any individual to 375 rems or more of radiation; or

2. The release of radioactive material in concentrations which, if averaged over a period of 24 hours would exceed 5,000 times the limits specified for such materials in Appendix B, Table II; or

3. A loss of one working week or more of the operation of any facilities affected; or

4. Damage to property in excess of $200,000.

(b) Twenty-four hour notification. Each licensee shall within 24 hours notify by telephone and telegraph, mailgram, or facsimile, the Director of the appropriate NRC Regional Office listed in Appendix D of any incident involving licensed material possessed by him and which may have caused or threatens to cause:

1. Exposure of the whole body of any individual to 5 rems or more of radiation; exposure of the skin of the whole body of any individual to 30 rems or more of radiation; or exposure of the feet, ankles, hands, or forearms to 75 rems or more of radiation; or

2. The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 500 times the limits specified for such materials in Appendix B, Table II; or

3. A loss of one day or more of the operation of any facilities affected; or

4. Damage to property in excess of $2,000.

(c) Any report filed with the Commission pursuant to this section shall be prepared so that names of individuals who have received exposure to radiation will be stated in a separate part of the report.

(d) For nuclear power reactors licensed under 50.21 or 50.22, the incidents included in paragraph (a) and paragraph (b) in this section shall in addition be reported pursuant to 50.72.

Evaluation

FSAR Section 12.3.1.3 states that reports of radiological incidents will be made to the NRC in accordance with 10 CFR 20.403.

SECTION

10 CFR 20.405

Statement of Section

(a) In addition to any notification required by 20.403, each licensee shall make a report in writing within 30 days to the appropriate NRC Regional Office listed in Appendix D with a copy to the Director of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, of:

(1) each exposure of an individual to radiation in excess of the applicable limits in 20.101 or 20.104(a) or the license; (2) each exposure of an individual to radioactive material in excess of the applicable limits in 20.103(a)(1), 20.103(a)(2), 20.104(b) or the license; (3) levels of radiation or concentrations of radioactive material in a restricted area in excess of any other applicable limit in the license; (4) any incident for which notification is required by 20.403; and (5) levels of radiation or concentrations of radioactive material (whether or not involving excessive exposure of any individual) in an unrestricted area in excess of ten times any applicable limit set forth in this part or in the license. Each report required under this paragraph shall describe the extent of exposure of individuals to radiation or to radioactive material, including estimates of each individual's exposure as required by paragraph (b) of this section; levels of radiation and concentrations of radioactive material involved; the cause of the exposure, levels or concentrations; and corrective steps taken or planned to assure against a recurrence.

(b) Any report filed with the Commission pursuant to paragraph (a) of this section shall include for each individual exposed the name, social security number, and date of

birth, and an estimate of the individual's exposure. The report shall be prepared so that this information is stated in a separate part of the report.

(c) In addition to any notification required by 20.403, each licensee shall make a report in writing within 30 days to the appropriate NRC Regional Office listed in Apendix D, with a copy to the Director of Inspection and Enforcement., U. S. Nuclear Regulatory Commission, Washington, D. C., 20555, of levels of radiation or releases of radioactive material in excess of limits specified by 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," or in excess of license conditions related to compliance with 40 CFR Part 190. Each report required under this paragraph shall describe the extent of exposure of individual to radiation or to radioactive material; levels of radiation and concentrations of radioactive material involved; the cause of the exposure, levels or concentrations; and corrective steps taken or planned to assure against a recurrence including the schedule for achieving comformance with 40 CFR Part 190 and associated license conditions.

Evaluation

FSAR Section 12.3.1.3 states that reports of radiological incidents will be made to the NRC in accordance with 10 CFR 20.405.

SECTION

10 CFR 20.407

Statement of Section

Each person described in 20.408 of this part shall, within the first quarter of each calendar year, submit to the Director of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D. C., 20555, the reports specified in paragraphs (a) and (b) of this section covering the preceding calendar year.[1] All other persons specifically licensed by the Commission shall, within the first quarter of calendar years 1979 and 1980, submit to the Director of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D. C., 20555, the reports specified in paragraphs (a) and (b) of this section covering the preceding calendar years 1978 and 1979.[2]

(a) A report of either (1) the total number of individuals for whom personnel monitoring was required under 20.202(a) or 34.33(a) of this chapter during the calendar year; or (2) the total number of individuals for whom personnel monitoring was provided during the calendar year. Provided, however, that such total includes at least the number of individuals required to be reported under paragraph (a)(1) of this section. The report shall indicate whether it is submitted in accordance with paragraph (a)(1) or (a)(2) of this section.

If personnel monitoring was not required to be provided to any individual by the licensee under 20.202(a) or 34.33(a) of this chapter during the calendar year, the licensee shall submit a negative report indicating that such personnel monitoring was not required.

(b) A statistical summary report of the personnel monitoring information recorded by the licensee for individuals for whom personnel monitoring was either required or provided, as described in paragraph (a) of this section, indicating the number of individuals whose total whole body exposure recorded during the previous calendar year was in each of the following estimated exposure ranges:

Estimated whole body exposure range (rems)[1]	Number of individuals in each range
No measurable exposure	
Measurable exposure less than 0.1	
0.1 to 0.25	
0.25 to 0.5	
0.5 to 0.75	
0.75 to 1	
1 to 2	
2 to 3	
3 to 4	
4 to 5	
5 to 6	
6 to 7	
7 to 8	
8 to 9	
9 to 10	
10 to 11	
11 to 12	
13 + ...	

[1]Individual values exactly equal to the values separating exposure ranges shall be reported in the higher range.

The low exposure range data are required in order to obtain better information about the exposures actually recorded. This section does not require improved measurements.

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1 A licensee whose license expires or terminates prior to, or on the last day of the calendar year, shall submit reports at the expiration or termination of the license, covering that part of the year during which the license was in effect.

2 The Commission will evaluate the data obtained for 1978 and 1979 pursuant to this paragraph, and the benefits derived therefrom and may take action, including publication of notice of proposed rulemaking, to extend or otherwise modify this reporting requirement.

Evaluation

FSAR section 12.3.1.3 states that personnel monitoring reports will be made in accordance with 10 CFR 20.407.

SECTION

10 CFR 20.408

Statement of Section

(a) This section applies to each person licensed by the Commission to:

1. Operate a nuclear reactor designed to produce electrical or heat energy pursuant to 50.21(b) or 50.22 of this chapter or a testing facility as defined in 50.2(r) of this chapter.

2. Possess or use byproduct material for purposes of radiography pursuant to Parts 30 and 34 of this chapter.

3. Possess or use at any one time, for purposes of fuel processing, fabricating, or reprocessing, special nuclear material in a quantity exceeding 5,000 grams of contained uranium-235, uranium-233, or plutonium or any combination thereof pursuant to Part 70 of this chapter;

4. Possess high-level radioactive waste at a geologic repository operations area pursuant to Part 60 of this chapter; or

5. Possess or use at any one time, for processing or manufacturing for distribution pursuant to part 30, 32, or 33 of this chapter, byproduct material in quantities, exceeding any one of the following quantities:

Radionuclide*	Quantity in curies
Cesium-137	1
Cobalt-60	1
Gold-198	100
Iodine-131	1
Iridium-192	10
Krypton-85	1,000
Promethium-147	10
Technetium-99m	1,000

*The Commission may require, as a license condition, or by rule, regulation or order pursuant to 20.502 reports from licensees who are licensed to use radionuclides not on this list in quantities sufficient to cause comparable radiation levels.

(b) When an individual terminates employment with a licensee described in paragraph (a) of this section, or an individual assigned to work in such a licensee's facility but not employed by the licensee, completes the work assignment in the licensee's facility, the licensee shall furnish to the Director of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D. C., 20555, a report of the individual's exposures to radiation and radioactive material, incurred during the period of employment or work assignment in the licensee's facility, containing information recorded by the licensee pursuant to 20.401(a) and 20.108. Such report shall be furnished within 30 days after the exposure of the individual has been determined by the licensee or 90 days after the date of termination of employment or work assignment, whichever is earlier.

Evaluation

FSAR section 12.3.1.3 states that reports of personnel monitoring on termination or completion of employment will be made in accordance with 10 CFR 20.408.

SECTION

10 CFR 20.409

Statement of Section

(a) Requirements for notifications and reports to individuals of exposure to radiation or radioactive material are specified in 19.13 of this chapter.

(b) When a licensee is required pursuant to 20.405 or 20.408 to report to the Commission any exposure of an individual to radiation or radioactive material, the licensee shall also notify the individual. Such notice shall be transmitted at a time not later than the transmittal to the Commission, and shall comply with the provisions of 19.13(a) of this chapter.

Evaluation

Fermi 2 Health Physics Procedure "Reports of Incidents Involving NRC Licensed Radioactive Material" (61.000.70) requires that copies of reports given to the NRC in accordance with the procedure be given to each individual involved, this includes overexposure incidents. Health Physics procedure "Personnel Radiation Exposure Records and Reports" (61.000.51) also requires that reports be made to individuals annually on request and upon termination of employment. The annual report will be provided on the Fermi 2 NRC Form-5 Equivalent, the termination of employment report will be made by completing a Report of Individual Occupational Radiation Exposure.

Other aspects of 19.13 will also be addressed in site Health Physics procedures.

SECTION

10 CFR 50.33 and 10 CFR 50.33a

Statement of Section

10 CFR 50.33 discusses, in part, the need for information to demonstrate the financial qualifications of an applicant. Further guidance is given in 10 CFR 50, Appendix C.

10 CFR 50.33a discusses the information needed by the Attorney General for antitrust review. Further guidance is given in 10 CFR 50, Appendix L.

Evaluation

Detroit Edison has complied with Appendices C and L of 10 CFR Part 50 by providing financial and antitrust information, including information submitted on behalf of co-applicants Wolverine Electric and Northern Michigan Electric Cooperatives, as required by 10 CFR Part 50.33 and 10 CFR Part 50.33a. Based on this information the Department of Justice (DOJ) was able to render its determination that the Enrico Fermi 2 project would not have any antitrust implications (see DOJ letters of August 16, 1971 and September 30, 1977).

SECTION

10 CFR 50.34(a)

Statement of Section

(a) Preliminary safety analysis report. Each application for a construction permit shall include a preliminary safety analysis report. The minimum information[1] to be included shall consist of the following:

1. 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. Such assessment shall 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 subparagraphs (2)-(8) of this paragraph, as well as the information required by this subparagraph, in support of the application for a construction permit.

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

3. The preliminary design of the facility, including:

(i) The principal design criteria for the facility.[2] 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 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 asurance 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; an identification and description of the research and development program which will be conducted to resolve any safety questions associated with such structures, systems, or components; and a schedule of the research and development program showing that such safety questions will be resolved at or before the latest date stated in the application for completion of construction of the facility.

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

10. A discusson 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.

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1 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.

2 General design criteria for chemical processing facilities are being developed.

Evaluation

A PSAR containing the required information was submitted to and reviewed by the AEC. A construction permit was issued following the review.

SECTION

10 CFR 50.34 (b)

Statement of Section

(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 therefore, 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 Cri-

teria 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 shall, as a minimum, meet the requirements for those programs contained in Appendix A of Part 55 of this chapter.

Evaluation

A FSAR containing the required information was submitted to and is under review by the NRC. Modifications to the FSAR continue to be made.

SECTION

10 CFR 50.34 (c)

Statement of Section

(c) Physical security plan. Each application for a license to operate a production or utilization facility shall include a physical security plan. The plan shall consist of two parts. Part I shall address vital equipment, vital areas, and isolation zones, and shall demonstrate how the applicant plans to comply with 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) of Part 11, at the proposed facility). Part II shall list tests, inspections, and other means to be used to demonstrate compliance with such requirements if applicable.

Regulatory Guide 1.17 dated June 1973 describes physical security criteria generally acceptable for the protection of nuclear power reactors against acts of industrial sabotage.

Evaluation

Fermi 2 has submitted a security plan to the NRC for review in accordance with the requirements of this section. The security plan is currently undergoing review by the NRC staff.

SECTION

10 CFR 50.34 (d)

Statement of Section

(d) Safeguards contingency plan. Each application for a license to operate a production or utilization facility that shall be subject to 73.50, 73.55, or 73.60 of this chapter shall 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 industrial 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.)[1]

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[1] 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.

Evaluation

Fermi 2 has submitted a safeguards contingency plan to the NRC for review in accordance with the requirements of this section. Detroit Edison is in compliance with the requirements of this section.

SECTION

10 CFR 50.34a (a)

Statement of Section

(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 provide numerical guidance on design objectives for light-water-cooled nuclear power reactors to meet the requirement 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.

Evaluation

This requirement addresses the information which should be included in a PSAR. The AEC staff has reviewed the Fermi 2 PSAR and has issued a construction permit.

SECTION

10 CFR 50.34a (b)

Statement of Section

(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 radionuclides expected to be released annually to unrestricted areas in liquid effluents produced during normal reactor operations; and

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

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

Evaluation

This requirement addresses information which is to be included in a PSAR. The AEC has reviewed the Fermi 2 PSAR and has issued a construction permit.

SECTION

10 CFR 50.34a (c)

Statement of Section

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

Evaluation

Chapter 11 of the FSAR, sections 11.2 and 11.3, contains the required information as descibed below. Sections 11.2 and 11.3 provide a description of the equipment and procedures for the control of gaseous and liquid effluents and for the use of equipment installed in the radioactive waste systems. Maintenance other than performance testing of such systems is not discussed. A revised estimate of expected releases and exposures is provided in sections 11.2 and 11.3. FSAR Appendix 11A provides information demonstrating compliance with the "as low as reasonably achievable" principle.

A revision to FSAR sections 11.2 and 11.3 was filed on July 31, 1981.

SECTION

10 CFR 50.36

Statement of Section

(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.

(b) Each license authorizing operation for 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 which are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor shall be snutdown. The licensee shall notify the Commission, review the matter and record the results of the review, including the cause of the conditions

and the basis for corrective action taken to preclude reccurrence. Operation shall not be resumed until authorized by the Commission.

(E) ***

(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 shall be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor. He 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.

(B) ***

(2) Limiting conditions for operation. 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 specification 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 specification until the condition can be met. In the case of either a nuclear reactor or a 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.

(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 the safety limits, and that the limiting conditions of 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 subparagraphs (1), (2), and (3) of this paragraph (c).

(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.

(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 where 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, 1979.

(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.

Evaluation

(a) The draft Technical Specifications for Fermi 2 dated August 1980 include the associated basis for each requirement.

(b) Fermi 2 acknowledges and will comply with the requirements of this section.

(c)(1)(i) The draft Technical Specifications contain safety limits as required. Actions to be taken in the event of exceeding a safety limit are specified in Technical Specification section 6.7.1.

(ii) The draft Technical Specifications contain limiting safety system settings as required. Reporting requirements regarding the failure of an automatic safety system to operate is addressed in technical specification section 6.9. The requirements to take appropriate corrective action in the event of such a failure is addressed in the Technical Specification action statements.

(2)-(5) The draft Technical Specifications contain the limiting conditions for operation, surveillance requirements, design features, and administrative controls required by these sections.

(d) This section is not applicable to Fermi 2.

SECTION

10 CFR 50.36a

Statement of Section

(a) In order to keep releases of radioactive materials to unrestricted areas during normal reactor operations, including expected operational occurrences, as low as is reasonably achievable, each license authorizing operation of a nuclear power reactor will include technical specifications that in addition to requiring compliance with applicable provisions of 20.106 of this chapter require:

1. That operating procedures developed pursuant to 50.34a(c) for the control of effluents be established and followed and that equipment installed in the radioactive waste system, pursuant to 50.34a(a) be maintained and used.

2. The submission of a report to the appropriate NRC Regional Office shown in Appendix D of Part 20 of this chapter within sixty (60) days after January 1 and July 1 of each year specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and in gaseous effluents during the previous six (6) months of operation, and such other information as may be required by the Commission to estimate maximum potential annual radiation doses to the public resulting from effluent releases. Copies of such report shall be

sent to the Director of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington, D. C., 20555.

If quantities of radioactive materials released during the reporting period are significantly above design objectives, the report shall cover this specifically. On the basis of such reports and any additional information the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such 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 specification described in this section will keep average annual releases of radioactive material in effluents at small percentages of the limits specified in 20.106 of this chapter and in the operating 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 operating conditions which may temporarily result in releases higher than such small percentages, but still within the limits specified in 20.106 of this chapter and the operating license. It is expected that in using this operational flexibility under unusual operating conditions, the licensee will exert his 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 radio-

active materials in effluents released to unrestricted areas be kept as low as is reasonably achievable.

Evaluation

The requirements of this section will be addressed in the Radiological Effluent Technical Specifications.

SECTION

10 CFR 50.44 (a)

Statement of Section

(a) Each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy cladding, shall, 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.

Evaluation

Fermi 2 has two thermal hydrogen recombiners for the control of hydrogen gas, as described in FSAR section 6.2.5. In addition, the Fermi 2 containment is inerted. Fermi 2 meets the requirements of this section.

SECTION

10 CFR 50.44 (b)

Statement of Section

(b) Each boiling or pressurized light-water reactor power reactor fueled with oxide pellets within cylindrical zircaloy cladding shall 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.

Evaluation

FSAR section 6.2.5.1 states that the design basis of the combustible gas control system includes the capability to measure hydrogen concentration and control combustible gas concentrations in the primary containment following a LOCA. FSAR section 6.2.5.3.1.3 describes the drywell cooling fans designed to ensure adequate mixing in the containment following a postulated LOCA. In addition, the Fermi 2 containment is inerted. Fermi 2 meets the requirements of this section.

SECTION

10 CFR 50.44 (c)

Statement of Section

(c) For each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy cladding, it shall be shown that during the time period following a postulated LOCA but prior to effective operation of the combustible gas control system, either (1) an uncontrolled hydrogen-oxygen recombination would not take place in the containment; or (2) the plant could withstand the consequences of uncontrolled hydrogen-oxygen recombination without loss of safety function. If neither of these conditions can be shown, the containment shall be provided with an inerted atmosphere or an oxygen deficient condition in order to provide protection against hydrogen burning and explosions during this time period.

Evaluation

FSAR section 6.2.5.3.1.3 demonstrates that following a postulated LOCA, an uncontrolled hydrogen-oxygen recombination would not take place in the contaiment prior to the effective operation of the combustible gas control system. Item 2 of this section does not apply to Fermi 2. Fermi 2 meets the requirements of this section.

SECTION

10 CFR 50.44 (d)

Statement of Section

(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.

Evaluation

FSAR section 6.2.5.3.1.3 states that the amount of hydrogen contributed by core metal-water reaction was assumed to be that amount which would result from the reaction of all the metal in the outside surfaces of the cladding to a depth of 0.00023 inch. Section 6.2.5.3.1.3 states that this assumed amount is greater than five times the total amount of hydrogen calculated in demonstrating compliance with 10 CFR 50.46

(b)(3). FSAR section 6.2.5.3.1.3c states that generation of hydrogen is assumed to occur within two minutes after the blowdown ends.

SECTION

10 CFR 50.44 (e)

Statement of Section

(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.

Evaluation

The notice of hearing on the application for a construction permit for Fermi 2 was published on March 26, 1971. FSAR section 6.2.5 states that thermal hydrogen recombiners are used for combustible gas control. FSAR section 6.2.5.2.5 states that the capability for a containment purge is provided for the purpose of removing fission product activity from the containment atmosphere.

SECTION

10 CFR 50.44 (f)-(h)

Statement of Section

(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 flamability limits. Combustible gas control system are of two types: (i) systems that allow controlled release from containment,

through filters if necessary, such as purging systems and repressurization system, 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 of 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.

Evaluation

The notice of hearing on the application for a construction permit was published on March 26, 1971. Therefore, these sections are not applicable to Fermi 2.

SECTION

10 CFR 50.46 (a)

Statement of Section

(a)(1) Except as provided in paragraph (a)(2) and (3) of this section each boiling and pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy cladding shall be provided with an emergency core cooling system (ECCS) which shall be designed such 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 shall be calculated in accordance with an acceptable evaluation model, and shall be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the entire spectrum of postulated loss-of-coolant accidents is covered. Appendix K, ECCS Evaluation Models, set forth certain required and acceptable features of evaluation models. Conformance with the criteria set forth in paragraph (b) of this section with ECCS cooling performance calculated in accordance with an acceptable evaluation model, may require that restrictions be imposed on reactor operation.

(2) With respect to reactors for which operating licenses have previously been issued and for which operating licenses may issue on or before December 28, 1974:

(i) The time within which actions required or permitted under this subparagraph (2) must occur shall begin to run on February 4, 1974.

(ii) Within six months following the date specified in paragraph (a)(2)(i) of this section an evaluation in accordance

with paragraph (a)(1) of this section shall have been submitted to the Director of Regulation of the Atomic Energy Commission. The evaluation shall have been accompanied by such proposed changes in technical specifications or license amendments as may be necessary to bring reactor operation in conformity with paragraph (a)(1) of this section

(iii) Any licensee may have requested an extension of the six-month period referred to in paragraph (a)(2)(ii) of this section for good cause. Any such request shall have been submitted not less than 45 days prior to expiration of the six-month period, and shall have been accompanied by affidavits showing precisely why the evaluation is not complete and the minimum time believed necessary to complete it. The Director of Regulation of the Atomic Energy Commission shall have caused notice of such a request to be published promptly in the Federal Register; such notice shall have provided for the submission of comments by interested persons within a time period established by the Director of Regulation. If, upon reviewing the foregoing submissions, the Director of Regulation concluded that good cause had been shown for an extension, he may have extended the six-month period for the shortest additional time which in his judgment will be necessary to enable the licensee to furnish the submissions required by paragraph (a)(2)(ii) of this section. Requests for extensions of the six-month period submitted under this subparagraph will have been ruled upon by the Director of Regulation prior to expiration of that period.

(iv) Upon submission of the evaluation required by paragraph (a)(2)(ii) of this section (or under paragraph (a)(2)(iii), if the six-month period is extended) the facility shall continue or commence operation only within the limits of both the proposed technical specifications or license amendments submitted in accordance with this paragraph (a)(2) and all technical specifications or license conditions previously imposed by the

Atomic Energy Commission, including the requirements of the Interim Policy Statement (June 29, 1971, 36 FR 12248) as amended December 18, 1971, 36 FR 24082).

(v) Further restrictions on reactor operation will be imposed if it is found that the evaluations submitted under paragraphs (a)(2)(ii) and (iii) of this section are not consistent with paragraph (a)(1) of this section and as a result such restrictions are required to protect the public health and safety.

(vi) Exemptions from the operating requirements of paragraph (a)(2)(iv) of this section may be granted for good cause. Requests for such exemption shall be submitted not less than 45 days prior to the date upon which the plant would otherwise be required to operate in accordance with the procedures of said paragraph (a)(2)(iv) of this section. Any such request shall be filed with the Secretary of the Commission, who shall cause notice of its receipt to be published promptly in the Federal Register; such notice shall provide for the submission of comments by interested persons within 14 days following Federal Register publication. The Director of Nuclear Reactor Regulation shall submit his views as to any requested exemption within five days following expiration of the comment period.

(vii) Any request for an exemption submitted under subparagraph (vi) of this subparagraph (2) must show, with appropriate affidavits and technical submissions, that it would be in the public interest to allow the licensee a specified additional period of time within which to alter the operation of the facility in the manner required by subparagraph (iv) of this subparagraph (2). The request shall also include a discussion of the alternatives available for establishing compliance with the rule.

(3) Construction permits may have been issued after December 28, 1973, but before December 28, 1974 subject to any applicable conditions or restrictions imposed pursuant to other regulations in this chapter and the Interim Acceptance Criteria for Emergency Core Cooling Systems published on June 29, 1971 (36 FR 12248) as amended (December 18, 1971, 36 FR 24082): Provided, however, that no operating license shall be issued for facilities constructed in accordance with construction permits issued pursuant to this paragraph, unless the Commission determines, among other things that the proposed facility meets the requirements of paragraph (a)(1) of this section.

Evaluation

Only subsection (a)(i) of this section applies to Fermi 2.

FSAR section 6.3 describes the Fermi 2 emergency core cooling system. Section 6.3.3 provides the ECCS performance evaluation as required by 10 CFR 50.46.

SECTION

10 CFR 50.46 (b)

Statement of Section

(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 fuel 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.

Evaluation

FSAR section 6.3.3.2 demonstrates compliance with the criteria set forth in 10 CFR 50.46(b).

SECTION

10 CFR 50.46 (c)(d)

Statement of Section

(c) As used in this section:

1. Loss-of-coolant accidents (LOCAs) 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 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 for 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.

Evaluation

No additional requirements are provided in this section.

SECTION

10 CFR 50.54 (a)-(h)

Statement of Section

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

(a) (Deleted 32 FR 2562.)

(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 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 will at any time before expiration of the license, upon request of the Commission submit written statements, signed under oath or affirmation, to enable the

Commission to determine whether or not the license should be modified, suspended or revoked.

(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.

Evaluation

These requirements are restrictions placed on an operating license. Detroit Edison acknowledges and will comply with these restrictions.

SECTION

10 CFR 50.54 (i)

Statement of Section

(i) Except as provided in 55.9 of this chapter, the licensee shall 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.

Evaluation

Fermi 2 site procedures provide for this requirement.

SECTION

10 CFR 50.54 (i-1)

Statement of Section

(i-1) Within three (3) months after issuance of an operating license, the licensee shall have in effect an operator requalification program which shall, as a minimum meet the requirements of Appendix A of Part 55 of this Chapter. Notwithstanding the provisions of 50.59 the licensee shall not, except as specifically authorized by the Commission, make a change in an approved operator requalification program by which the scope, time allotted for the program, or frequency in conducting different parts of the program is decreased.

Holders of operating licenses in effect on September 17, 1973 shall implement an operator requalification program which, as a minimum, meets the requirements of Appendix A of Part 55 of this chapter which was submitted for approval by the Atomic Energy Commission.

Evaluation

FSAR section 13.2.2 states that a requalification program for licensed operators and senior operators will be implemented no later than three months following the issuance of an operating license for the plant.

SECTION

10 CFR 50.54(j)-(m)

Statement of Section

(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) 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 shutdown or significant reduction in power, and refueling, or as otherwise prescribed in the facility license.

Evaluation

(j) Fermi 2 procedures require that except as allowed by 10 CFR 55.59, no one is allowed to manipulate the reactor controls of Fermi 2 unless he is licensed pursuant to 10 CFR 55

(reactor controls includes apparatus and mechanisms the manipulation of which directly affects reactor reactivity of power level).

(k) Fermi 2 procedures require that when fuel is in the reactor, the licensed operator is not permitted to leave the "at controls area" without proper relief except in an emergency.

(l) Fermi 2 procedures place the nuclear shift supervisor in charge of all shift activities, licensed and unlicensed. The nuclear shift supervisor must have a senior reactor operators license.

(m) Fermi 2 procedures require that a senior reactor operator be in the control room during any of the conditions referred to in this section.

SECTION

10 CFR 50.54(n)

Statement of Section

(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.

Evaluation

This requirement places a limitation on the construction permit and operating license. Detroit Edison acknowledges and complies with this limitation. Detriot Edison's current schedule calls for filing of proposed technical specifications in May, 1982.

SECTION

10 CFR 50.54(o)

Statement of Section

(o) Primary reactor containments for water cooled power reactors shall be subject to the requirements set forth in Appendix J.

Evaluation

Fermi 2 is in compliance with the requirements of this section.

SECTION

10 CFR 50.54(p)

Statement of Section

(p) The licensee shall prepare and maintain safeguards contingency plan Procedures in accordance with Appendix C of 10 CFR Part 73 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 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) of Part 73, as applicable, without prior approval of the Commission. A licensee desiring to make such a change shall submit an application for an amendment to his license pursuant to 50.90. The licensee may make changes to the security plan or to the safeguards contingency plan 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 of the change, and shall furnish to the Director of Nuclear Material Safety and Safeguards (for enrichment and reprocessing facilities) or to the Director of Nuclear Reactor Regulation (for nuclear reactors), U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate NRC Regional Office specified in Appendix A of Part 73 of this chapter, 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:

(1) All safeguards capabilities specified in the safeguards contingency plan available and functional.

(2) Detailed Procedures developed according to Appendix C to Part 73 available at the licensee's site, and

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

The licensee shall provide for the development, revision, implementation, and maintenance of his 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 shall 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 system along with commitments established for response by local law enforcement authorities. The results of the review and audit, along with recommendations for improvements, shall 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.

Evaluation

The following requirements will be provided in site procedures: 1) review proposed changes to the security plan and contingency plan to determine whether the proposed change decreases the effectiveness of the plan, 2) obtain NRC approval prior to implementing a security or contingency plan change which could decrease the effectiveness of the plan, 3) maintain records of changes made without prior NRC approval

and 4) submit a report to the NRC of each change made without prior NRC approval within two months of the change.

Fermi 2 acknowledges and will comply with the requirements regarding the contingency plan.

Section

10 CFR 50.54(q)-(u)

Statement of Section

(q) A licensee authorized to possess and/or 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 of this Part. The nuclear power reactor licensee may make changes to these plans without Commission approval only if such 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 of this Part. The research reactor licensee and/or the fuel facility licensee may make changes to these plans without Commission approval only if such changes do not decrease the effectiveness of the plans, and the plans as changed continue to meet the requirements of Appendix E of this Part. Proposed changes that decrease the effectiveness of the approved emergency plans shall not be implemented without application to and approval by the Commission. The licensee shall furnish 3 copies of each proposed change for approval; and/or if a change is made without prior approval, 3 copies shall be submitted within 30 days after the change is made or proposed to the Director of the appropriate NRC regional office specified in Appendix D, 10 CFR Part 20, with 10 copies to the Director of Nuclear Reactor Regulation, or, if appropriate, the Director of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

(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 500 kW 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 Nuclear Reactor Regulation for approval within one year from the effective date of this rule. Each licensee who is authorized power level less than 500 kW thermal, under a license of the type specified in 50.1(c), shall submit emergency plans complying with 10 CFR Part 50, Appendix E, to the Director of Nuclear Reactor Regulation for approval within two years from the effective date of this amendment.

(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 govermental 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 copies of the above plans shall be forwarded to the Director of Nuclear Reactor Regulation with 3 copies to the Director 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 appropriate to protect the food ingestion pathway.

(2) For operating power reactors, the licensee, State, and local emergency response plans shall be implemented by April 1, 1981, except as provided in Section N, D-3 of Appendix E of this part. If after April 1, 1981, the NRC finds that the state of emergency preparedness does not provide reasonable assurance that appropriate protective measures can and will be taken in the event of a radiological emergency and if the deficiencies are not corrected within four months of that finding, the Commission will determine whether the reactor shall be shut down until such deficiencies are remedied or whether other enforcement action is appropriate. In determining whether a shutdown or other enforcement action is appropriate, the Commission shall take into account, among other factors, whether the licensee can demonstrate to the Commission's satisfaction that the deficiencies in the plan are not significant for the plant in question, or that adequate interim compensating actions have been or will be taken promptly, or that there are other compelling reasons for continued operation.

(3) The NRC will base its finding on a review of the FEMA findings and determinations as to whether State and local emergency plans are adequate and capable of being implemented, and on the NRC assessment as to whether the licensee's emergency plans are adequate and capable of being implemented. Nothing in ths paragraph shall be construed as limiting the authoriy 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 of this Part.

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[1]Emergency Planning Zones (EPZs) are discussed in NUREG-0396; EPA 520/178-016. "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.

Evaluation

(q) The Fermi 2 Emergency Plan is under review by the NRC. The Emergency Plan when approved will be followed and maintained in effect in accordance with the requirements of this section. Facility procedures will address the requirements of this section regarding changes to the emergency plan.

(r) The requirements of this section are not applicable to Fermi 2.

(s) State emergency plans have been submitted to the NRC. Local emergency plans have not been developed. DECO will submit these plans when available.

(t) Fermi 2 acknowledges and will implement these requirements. Section P.2 of the Fermi 2 emergency plan provides for the review of the emergency preparedness program every twelve months.

(u) This section is not applicable to Fermi 2.

SECTION

10 CFR 50.55a(a)(1)

Statement of Section

Each operating license for a boiling or pressurized water-cooled nuclear power facility shall be subject to the conditions in paragraph (g) and each construction permit for a utilization facility shall be subject to the following conditions in addition to those specified in 50.55,

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

Evaluation

Section 3.2 of the FSAR provides the classification of structures, components, and systems based on the importance of the safety functions they perform.

Table 3.2-1 provides the quality group classification for structures, systems, and components and Table 3.2-2 provides minimum code requirements for each quality group classification.

Fermi 2 is in compliance with the requirements of this section.

SECTION

10 CFR 50.55a(a)(2)

Statement of Section

(2) As a minimum, the systems and components of boiling and pressurized water-cooled nuclear power reactors specified in paragraphs (c), (d), (e), (f), (g), and (i) of this section shall meet the requirements described in those paragraphs, except that the American Society of Mechanical Engineers (hereinafter referred to as ASME) Code N-symbol need not be applied, and the protection systems of nuclear power reactors of all types shall meet the requirements described in paragraph (h) of this section, except as authorized by the Commission or the Atomic Energy Commission upon demonstration by the applicant for or holder of a construction permit that:

(i) Design, fabrication, installation, testing, or inspection of the specified system or component, is to the maximum extent practical, in accordance with generally recognized codes and standards, and compliance with the requirements described in paragraphs (c) through (i) of this section or portions thereof would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety; or

(ii) Proposed alternatives to the described requirements of portions thereof will provide an acceptable level of quality and safety. For example, the use of inspection or survey systems other than those required by the specified ASME Codes and addenda may be authorized under this subparagraph provided that an acceptable level of quality and safety in design, fabrication, installation, and testing is achieved.

Evaluation

Compliance with paragraphs (c), (d), (e), (f), (g), (h) and (i) of 10 CFR 50.55a is discussed in subsequent subsections. Detroit Edison requested a waiver from certain of the code requirements of Section 50.55a in a letter to the AEC dated May 3, 1973 (EF2-17172). The waiver and approval by the AEC of the codes used are documented in a letter from the AEC to Detroit Edison dated July 12, 1973. The code differences are shown in FSAR Table 3.2-3.

SECTION

10 CFR 50.55a(b)

Statement of Section

(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 on January 1, 1981. 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 Nuclear Regulatory Commission's Public Document Room, 1717 H St., N.W., Washington, D.C.

(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 editions through the 1977 Edition and addenda through the Summer 1979 Addenda.

(2) As used in this section, references to Section XI of the ASME Boiler and Pressure Vessel Code refer to Section XI, Division 1 and include editions through the 1977 edition and addenda through the Summer 1979 Addenda, subject to the following limitations and modifications:

(i) Applicability of 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.

(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 IX 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 shall be 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, shall be examined. The extent of examination for these systems shall 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.

(B) For a nuclear power plant whose application for a construction permit is docketed prior to July 1, 1976, 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.

Evaluation

The construction permit for Fermi 2 was issued on September 26, 1972 and therefore the 1974 edition of the code through the summer 1975 Addenda apply.

Details of compliance are discussed in sections 10 CFR 50.55a (c) through (h) below.

Section

10 CFR 50.55a(c)

Statement of Section

(c) Pressure vessels:

(1) For constuction permits issued before January 1, 1971, for reactors not licensed for operation, pressure vessels which are part of the reactor coolant pressure boundary[2] shall meet the requirements for Class A vessels set forth in section III of the ASME Boiler and Pressure Vessels Code, applicable Code Cases, and Addenda[3] in effect[4] on the date of order[3] of the vessel. The pressure vessels may meet the requirements set forth in editions of this Code, applicable Code Cases, and Addenda which have become effective after the date of vessel order.

(2) For construction permits issued on or after January 1, 1971, but before July 1, 1974, pressure vessels which are part of the reactor coolant pressure boundary[2] shall meet the requirements for Class A or Class 1 vessels set forth in editions of section III of the ASME Boiler and Pressure Vessel Code and Addenda[6] in effect[4] on the date of order[5] of the pressure vessel: Provided, however, that if the pressure vessel is ordered more than 18 months prior to the date of issuance of the construction permit, compliance with the requirements for Class A or Class 1 vessels set forth in editions of section III of the ASME Boiler and pressure Vessel Code and Addenda in effect 18 months prior to the date of issuance of the construction permit is required. The pressure vessels may meet the requirements set forth in editions of this Code and Addenda which have become effective after the date of vessel order or after 18 months prior to the date of issuance of the construction permit.

(3) For construction permits issued on or after July 1, 1974, pressure vessels which are part of the reactor coolant pressure boundary[2] shall meet the requirements for Class 1 components set forth in Section III[3 4 5 6] of the ASME Boiler and Pressure Vessel Code: Provided that the ASME Code provisions applied to the pressure vessels shall be no earlier than those of the Summer 1972 Addenda of the 1971 edition.

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[2]Components which are connected to the reactor coolant system and are part of the reactor coolant pressure boundary defined in 50.2(v) need not meet these requiements, provided:

(a) 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 only, or

(b) The component is or can be isolated from the reactor coolant system by two valves (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]Copies may be obtained from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. Copies are available for inspection at the Commission's Public Document Room, 1717 H St., N.W., Washington, D.C.

[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 which have been determined suitable for use by the Commission staff are listed in NRC Regulatory Guide 1.84, "Code Case Acceptability--ASME Section III Materials." The use of other Code cases may be authorized by the Commission upon request pursuant to 50.55a(a)(2)(ii).

Evaluation

The construction permit for Fermi 2 was issued on September 26, 1972 and therefore paragraph (c)(2) of 10 CFR 50.55a applies. The reactor pressure vessel was ordered in January 1967. Detroit Edison requested an exemption from the AEC from the requirements of this Section to allow the application of the ASME Boiler and Pressure Code Section III with addenda through summer 1969. The AEC accepted this alternative in a letter to Detroit Edison dated July 12, 1973.

SECTION

10 CFR 50.55a(d)

Statement of Section

(d) Piping

(1) For construction permits issued before January 1, 1971, for reactors not licensed for operation, piping which is part of the reactor coolant pressure boundary[2] shall meet the requirements set forth in:

(i) The American Standard Code for Pressure Piping (ASA B31.1), Addenda, and applicable Code Cases[3] or the U.S.A. Standard Code for Pressure Piping (USAS B31.1.0), Addenda, and applicable Code Cases[3] in effect[4] on the date of order[5] of the piping and

(ii) The nondestructive examination and acceptance standards of ASA B31.1 Code Cases N7, N9, and N10, except that the acceptance standards of Class I piping of the U.S.A. Standard Code for Pressure Piping (USAS B31.7) may be applied.

The piping may meet the requirements set forth in editions of ASA B31.1, USAS B31.1.0, and USAS B31.7, Addenda, and Code Cases which became effective after the date of order of the piping.

(2) For construction permits issued on or after January 1, 1971, but before July 1, 1974 piping which is part of the reactor coolant pressure boundary[2] shall meet the requirements for Class I piping set forth in editions of (i) the USA Standard Code for Pressure Piping (USAS B31.7) and Addenda[6] in effect on the date of order of the piping, or (ii) the requirements applicable to Class 1 piping of editions of

section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of the order of the piping: Provided, however, that if the piping is ordered more than 6 months prior to the date of issuance of the construction permit, compliance with the requirements for Class I of Class 1 piping set forth in editions of USAS B31.7 of section III of the ASME Boiler and Pressure Vessel Code and Addenda[5] in effect 6 months prior to the date of issuance of the construction permit is required. The piping may meet the requirement set forth in editions of these Codes and Addenda[6] which have become effective after the date of piping order or after 6 months prior to the date of issuance of the construction permit.

(3) For construction permits issued on or after July 1, 1974, piping which is part of the reactor coolant pressure boundary[2] shall meet the requirements for Class 1 components set forth in Section III[3 4 5 6] of the ASME Boiler and Pressure Vessel Code: Provided that the ASME Code provisions applied to the piping shall be no earlier than those of Winter 1972 Addenda of the 1971 edition.

Evaluation

The construction permit for Fermi 2 was issued on September 26, 1972, and therefore paragraph (d) (2) of 10 CFR 50.55a applies. Detroit Edison requested an exemption from the AEC in a letter dated May 31, 1973 to allow the use of alternate codes for the reactor coolant pressure boundary. The request for an exemption was granted in a letter from the AEC to Detroit Edison dated July 12, 1973. A tabulation showing the code applied and the code required by 10 CFR 50.55a is shown on FSAR Table 3.2-3.

SECTION

10 CFR 50.55a(e)

Statement of Section

(e) Pumps:

(1) For construction permits issued before January 1, 1971, for reactors not licensed for operation, pumps which are part of the reactor coolant pressure boundary[2] shall meet--

(i) The requirements for Class I pumps set forth in the Draft ASME Code for Pumps and Valves for Nuclear Power Addenda, and Code Cases[3] in effect[4] on the date of order[5] of the pumps, or

(ii) The nondestructive examination and acceptance standards set forth in ASA B31.1 Code Cases N7, N9, and N10, except that the acceptance standards for Class I pumps set forth in the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda in effect on the date of order of the pumps may be applied.

The pumps may meet the requirements set forth in editions of the Draft ASME Code for Pumps and Valves for Nuclear Power, Addenda, and Code Cases which became effective after the date of order of the pumps.

(2) For construction permits issued on or after January 1, 1971, but before July 1, 1974, pumps which are part of the reactor coolant pressure boundary[2] shall meet the requirements for Class I pumps set forth in editions of (i) the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda[6] in effect[4] on the date of order[5] of the pumps and the requirements applicable to pumps set forth in articles 1 and 8 of section III of the ASME Boiler and Pressure Vessel Code and

Addenda in effect on the date of order of the pumps, or (ii) the requirements applicable to Class 1 pumps of editions of section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of order of the pumps: Provided however, that if the pumps are ordered more than 12 months prior to the date of issuance of the construction permit, compliance with the requirements for Class I pumps set forth in the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda[6] and the requirements applicable to pumps set forth in articles 1 and 8 of editions of section III of the ASME Boiler and Pressure Vessel Code and Addenda, or for Class 1 pumps of editions of section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect 12 months prior to the date of issuance of the construction permit is required. The pumps may meet the requirement set forth in editions of these Codes or Addenda which have become effective after the date of pump order or after 12 months prior to the date of issuance of the construction permit.

(3) For construction permits issued on or after July 1, 1974, pumps which are part of the reactor coolant pressure boundary[2] shall meet the requirements for Class 1 components set forth in Section III[3 4 5 6] of the ASME Boiler and Pressure Vessel Code: Provided that the ASME Code provisions applied to the pumps shall be no earlier than those of the Winter 1972 Addenda of the 1971 edition.

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[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 shall be the date of docketing of the information required by 2.101(a-1) (2) or (3), whichever is later.

Evaluation

The construction permit for Fermi 2 was issued on September 26, 1972, and therefore paragraph (e)(2) of 10 CFR 50.55a applies. FSAR Table 3.2-3 provides a comparison of the codes used for primary pressure boundary components and the codes required by 10 CFR 50.55a. Detroit Edison requested an exemption from the AEC in a letter dated May 31, 1973 to allow the use of codes other than those required by 10 CFR 50.55a. The exemption was granted by the AEC in a letter dated July 12, 1973.

SECTION

10 CFR 50.55a (f)

Statement of Section

(f) Valves:

(1) For construction permits issued before January 1, 1971, for reactors not licensed for operation, valves which are part of the reactor coolant pressure boundary[2] shall meet the requirements set forth in

(i) The American Standard Code for Pressure Piping (ASA B31.1), Addenda, and applicable Code Cases, or the USA Standard Code for Pressure Piping (USAS B31.1), Addenda, and applicable Code Cases, in effect[4] on the date of order[5] of the valves or the Class I section of the Draft ASME Code for Pumps and Valves for Nuclear Power,[3] Addenda, and Code Cases in effect on the date of order of the valves; or

(ii) The nondestructive examination and acceptance standards of ASA B31.1 Code Cases N2,N7, N99, and N10, except that the acceptance standards for Class I valves set forth in the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda in effect on the date of order of the valves may be applied.

The valves may meet the requirements set forth in editions of ASA B31.1, USAS B.31.1.0, and the Draft ASME Code for Pumps and Valves for Nuclear Power, Addenda, and Code Cases, which became effective after the date of order of the valves.

(2) For construction permits issued on or after January 1, 1971, but before July 1, 1974 valves which are part of the reactor coolant pressure boundary[2] shall meet the requirements for Class I valves set forth in editions of (i) the Draft ASME

Code for Pumps and Valves for Nuclear Power and Addenda[8] in effect on the date of order of the valves, of (ii) the requirements applicable to Class 1 valves of section II of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of order of the valve; Provided however that if the valves are ordered more than 12 months prior to the date of issuance of the construction permit, compliance with the construction permit, compliance with the requirements for Class I valves set forth in editions of the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda[6] and the requirements applicable to valves set forth in articles 1 and 8 of editions of section III of the ASME Boiler and Pressure Vessel Code and Addenda, or for Class 1 valves of section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect 12 months prior to the date of issuance of the construction permit is required. The valves may meet the requirements set forth in editions of these Codes or Addenda which have become effective after the date of valve order or after 12 months prior to the date of issuance of the construction permit.

(3) For construction permits issued on or after July 1, 1974, valves which are part of the reactor coolant pressure boundary[2] shall meet the requirements set forth in Section III[3 4 5 6] of the ASME Boiler and Pressure Vessel Code: Provided that the ASME Code provisions applied to the valves shall be no earlier than those of the Winter 1972 Addenda of the 1971 edition.

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[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 the purposes of this section shall be the date of docketing of the information required by 2.101(a-1)(2) or (3), whichever is later.

Evaluation

The construction permit for Fermi 2 was issued on September 26, 1972, and therefore paragraph (f)(2) of 10 CFR 50.55a applies. FSAR Table 3.2-3 provides a comparison of the codes used for primary pressure boundary components and the codes required by 10 CFR 50.55a. Detroit Edison requested an exemption from the AEC in a letter dated May 31, 1973 to allow the use of codes other than those required by 10 CFR 50.55a. The exemption was granted by the AEC in a letter dated July 12, 1973.

SECTION

10 CFR 50.55.a(g)(1) through (3)

Statement of Section

(g) Inservice inspection requirements:

(1) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued prior to January 1, 1971, components (including supports) shall meet the requirements of paragraphs (g)(4) and (g)(5) of this section to the extent practical. Components which are part of the reactor coolant pressure boundary[2] and their supports shall meet the requirements applicable to components which are classified as ASME Code Class 1. Other safety-related pressure vessels, piping, pumps, and valves shall 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 through 2 shall be designed and be provided with access to enable the performance of (i) inservice examination of such components (including supports) and (ii) tests for operational readiness of pumps and valves, and shall meet the preservice examination requirements set forth in editions of Secton XI of the ASME Boiler and Pressure Vessel Code and Addenda[3, 6] in effect 6 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 therein.

(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 shall be designed and be provided with access to enable the performance of inservice examination of such components and shall meet the preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda[3, 6] applied to the construction of the particular component in accordance with paragraph (c), (d), (e), or (f) of this section.

(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 shall be designed and be provided with access to enable the performance of inservice examination of such components and shall meet the preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda[3, 6] applied to the construction of the particular component.

(iii) Pumps and valves which are classified as ASME Code Class 1 shall 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[3, 6] applied to the construction of the particular component.

(iv) Pumps and valves which are classified as ASME Code Class 2 and Class 3 shall be designed and be provided with access to enable the performance of inservice testing of the pumps and valves or assessing operational readiness set forth in Section XI of editions of the Boiler and Pressure Vessel

Code and Addenda[3,6] applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

(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.

Evaluation

The construction permit for Fermi 2 was issued on September 26, 1972, and therefore paragraph (g)(2) of 10 CFR 50.55a applies. FSAR section 5.2.8 states that the inservice inspection program will comply as required by this regulation with ASME section XI, 1974 edition through the 1975 summer addenda. Section 4.0.5 of the Fermi 2 draft technical specifications states that the requirements of 10 CFR 50.55a(g) will be met except where specific written relief has been granted pursuant to 10 CFR 50.55a(g)(6)(i).

SECTION

10 CFR 50.55a(g)(4)

Statement of 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 shall 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 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 examinations of components, inservice tests to verify operational readiness of pumps and valves whose function is required for safety, and system pressure tests, conducted during the initial 120-month inspection interval shall 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 examinations of components, inservice tests to verify operational readiness of pumps and valves whose function is required for safety, and system pressure tests, conducted during successive 120-month inspection intervals shall 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) For a facility whose operating license was issued prior to March 1, 1976, at the start of the next one-third of a 120 month inspection interval. During that third of an inspection interval and the remainder of the inspection interval, the inservice examinations of components, tests to verify operational readiness of pumps and valves whose function is required for safety, and system pressure tests, for such facilities shall 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 start of that third of an inspection interval, subject to the limitations and modifications listed in paragraph (b) of this section.

(iv) Inservice examinations of components, tests of pumps and valves, 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, 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.

Evaluation

FSAR section 5.2.8.6 states that the inservice testing program for pumps and valves complies as required by this regulation with ASME Section XI, 1974 Edition through the 1975 Summer Addenda. The Inservice Inspection Pump and Valve Program was submitted to the NRC with FSAR Amendment 15. Section 4.0.5 of the Fermi 2 draft technical specifications states that the requirements of 10 CFR 50.55a(g) will be met except where specific written relief has been granted pursuant to 10 CFR 50.55a(g)(6)(i).

SECTION

10 CFR 50.55a(g)(5)(6)

Statement of Section

(5)(i) The inservice inspection program for a boiling or pressurized water-cooled nuclear power facility shall 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. This application shall be submitted at least 6 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 his facility, the licensee shall notify the Commission and submit information to support his determinations.

(iv) Where an examination of test 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 shall 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 or test 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 relieve 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.

Evaluation

(5) Draft Technical Specification section 4.0.5 requires the implementation of an inservice inspection program which meets the requirements of 10 CFR 50.55a(g). The inservice inspection program is still under development. Requests for exemptions have not been submitted.

(6) This section provides direction to the NRC and is not applicable to Fermi 2.

SECTION

10 CFR 50.55a(h)

Statement of Section

(h) Protection systems: For construction permits issued after January 1, 1971, protection systems shall 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 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.

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 Street, New York, NY 10017. A copy is available for inspection at the Commission's Public Document Room, 1717 H Street N.W., Washington, D.C.

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[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 the purposes of this section shall be the date of docketing of the information required by 2.101(a-1)(2) or (3), whichever is later.

Evaluation

The construction permit for Fermi 2 was issued on September 26, 1972. FSAR section 7.1.2.1.1 1 states that the reactor protection system complies with the requirements of IEEE 279-1971.

SECTION

10 CFR 50.55a(i)

Statement of Section

(i) Fracture toughness requirements: Pressure-retaining components of the reactor coolant pressure boundary shall meet the requirements set forth in Appendices G and H to this part.

Evaluation

Compliance with Appendices G and H to 10 CFR 50 is discussed in the evaluation of compliance section for these requirements in this report.

SECTION

10 CFR 50.55a(j)

Statement of Section

(j) Power reactors for which a notice of hearing on an application for a provisional construction permit or a construction permit has been published on or before December 31, 1970, may meet the requirements of paragraphs (c)(1), (d)(1), (e)(1), and (f)(1) of this section instead of paragraphs (c)(2), (d)(2), (e)(2), and (f)(2) of this section, respectively.

Evaluation

The notice of a hearing on the application for a construction permit was published on March 26, 1971. Therefore this section is not applicable to Fermi 2.

SECTION

10 CFR 50.59

Statement of Section

(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 prososed 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 evaluatd 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) The licensee shall maintain records of changes in the facility and of changes in procedures made pursuant to this section, to the extent that such changes constitute changes in the facility as described in the safety analysis report or 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 shall include a written safety evaluation which provides the basis for the determination that

the change, test, or experiment does not involve an unreviewed safety question.

The licensee shall furnish to the appropriate NRC Regional Office shown in Appendix D of Part 20 of this chapter with a copy to the Director of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, annually or at such shorter intervals as may be specified in the license, a report containing a brief description of such changes, tests, and experiments, including a summary of the safety evaluation of each. Any report submitted by a licensee pursuant to this paragraph will be made a part of the public record of the licensing proceeding. In addition to a signed original, 39 copies of each report of changes in a facility of the type described in 50.21(b) of 50.22 or a testing facility, and 12 copies of each report of changes in any other facility, shall be filed.

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.

Evaluation

Fermi 2 has committed to the requirements of this section. These requirements will be included in facility procedures.

SECTION

10 CFR 50.70

Statement of Section

(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 Inspection and Enforcement, 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 onsite space is suggested as a guide. For sites containing multiple power reactor units or fuel facilities, additional space may be requested to accommodate additional full-time inspector(s). The office space that is provided shall be subject to the

approval of the Director, Office of Inspection and Enforcement. 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 Director as likely to inspect the facility, immediate unfettered access, equivalent to access provided regular plant employees, following proper identification and compliance with applicable access control measures for security, radiological protection and personal safety.

Evaluation

(a) Site procedures provide for inspection of the Fermi 2 site by authorized NRC inspectors.

(b)(1) Fermi 2 provides the NRC resident inspectors rent free office space in accordance with the requirements of this section.

(2) An office trailer on site is being provided to the resident NRC inspectors in accordance with the requirements of this section.

(3) Site procedures will provide for access for NRC resident inspectors and inspectors identified by the Regional Director as required by this regulation.

SECTION

10 CFR 50.71

Statement of Section

(a) Each licensee and each holder of a construction permit shall maintain such records and make such reports, in connection with the licensed activity, as may be required by the conditions of the license or permit or by the rules, regulations, and orders of the Commission effectuating the purposes of the Act, including section 105 of the Act.

(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, upon each issuance of its annual financial report, including the certified financial statements, file a copy thereof with the **Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.**

(c) Records which are required by the regulations in this part, by license condition, or by technical specification, shall be maintained for the period specified by the appropriate regulation, license condition, or technical specification. If a retention period is not otherwise specified, such records shall be maintained until the Commission authorizes their disposition.

(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 b authorized personnel and the microform is capable of producing a clear and legible copy after storage for the period specified by Commission regulations.

(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, 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 shall update periodically, as provided in paragraphs (e)(3) and (e)(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) Revisions containing updated information shall be submitted on a replacement-page basis and shall be accompanied by a list which identifies the current pages of the FSAR following page replacement. One signed original and 12 additional copies of the required information shall be filed with the

Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

(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 shall be filed no less frequently than annually and shall reflect all changes up to a maximum of 6 months prior to the date of filing.

(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 acutally changed, and a page change identification (date of change or change number or both).

Evaluation

(a)-(d) The requirements of these sections will be included in facility procedures.

(e) Fermi 2 acknowledges and will comply with the requirements of this section.

SECTION

10 CFR 50, Appendix A, General Design Criteria

Evaluation

The General Design Criteria for Nuclear Power Plants are discussed in detail in FSAR Section 3.1. Additional detail is provided throughout the FSAR in the system design descriptions.

SECTION

10 CFR 50, Appendix B, Quality Assurance Criteria

Evaluation

The 18 Quality Assurance criteria are discussed in the following FSAR sections:

A17.1:	Design and Construction (DECo)
B17.1:	Design and Construction (General Electric)
17.2:	Preoperational testing, startup, operation, maintenance, and modifications

SECTION

10 CFR 50, Appendix E

Evaluation

Detroit Edison submitted its Radiological Emergency Response Plan as FSAR Annex H.I. This plan is currently undergoing review by the NRC staff. When the fully upgraded emergency preparedness program is required to be in place, the requirements set forth in Appendix E will be met. The following analysis of regulatory requirements related to Appendix E consists of information which has been submitted and is presently being evaluated by the NRC staff.

SECTION

10 CFR 50, Appendix E, Paragraph III

Statement of Section

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)[2] to an extent sufficient to demonsrate that the plans provide reasonable assurance that appropriate measures can and will be taken in the event of an emergency.

Evaluation

The requirements of this regulation are set forth in the EF2 FSAR, Annex H.1 (March 1981), entitled "Enrico Fermi Atomic Power Plant Unit 2 Radiological Emergency Response Plan."

SECTION

10 CFR 50, Appendix E, Paragraph IV (Introduction)

Statement of Section

IV. Content of Emergency Plans

The applicant's emergency plans shall contain, but not necessarily be limited to, information needed to demonstrate compliance with the elements set forth below, i.e., organization for coping with 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 Section 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.

Evaluation

The Radiological Response Plan has been submitted as Annex H.1 to the EF2 FSAR. Preliminary evacuation time estimates are in Appendix 3 to Annex H.1.

SECTION

10 CFR 50, Appendix E, Paragraph IV.A

Statement of Section

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 of an onsite emergency coordinator who shall be in charge of the exchange of information with offsite authorities responsible for coordinating and implementing offsite emergency measures.

3. A description, by position and function to be performed, of the licensee's headquarters personnel who will be sent to the plant site to augment the onsite emergency organization.

4. Identification, by position and function to be performed, of persons within the licensee organization who will be

responsible for making offsite dose projections, and a description of how these projections will be made and the results transmitted to State and local authorities, NRC, and other appropriate governmental entities.

5. Identification, by position and function to be performed, of other employees of the licensee with special qualifications for coping with emergency conditions that may arise. Other persons with special qualifications, such as consultants, who are not employees of the licensee and who may be called upon for assistance for emergencies shall also be identified. The special qualifications of these persons shall be described.

6. A description of the local offsite services to be provided in support of the licensee's emergency organization.

7. 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.

Evaluation

Information concerning requirements of 10 CFR 50, Appendix E, Paragraph IV.B is discussed in the FSAR references cited below:

10 CFR 50 Appendix E Section	EF2 FSAR Annex H.1 Reference(s)	Subject
IV.A.1	B.1.1, Figure B-1	Normal plant organization.
IV.A.2.a	B.2.1, Table B-2	Emergency Director
IV.A.2.b	Table B-2	Plant assignments.
IV.A.2.c	B.2.1, Table B-2	EOF Coordinator
IV.A.3	Table B-2	Headquarters assignments at site
IV.A.4	Table B-2	Dose projection responsibility
	I.3	Dose projections
	B.5, F	Communications
IV.A.5	B.4	Medical, firefighting, etc., offsite assistance
	C.1, C.2	Government and industry support
	L	Medical support details
IV.A.6	B.4	Local offsite services
	L	Local medical support

10 CFR 50 Appendix E Section	EF2 FSAR Annex H.1 Reference(s)	Subject
IV.A.7	Figure A-1 Appendix 1	State/local responsibilities Agreements (Note: several agreements are under development.)
IV.A.8	A	State/local planning (Note: local planning is under development.)

SECTION

10 CFR 50, Appendix E, Paragraph IV.B

Statement of Section

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 in-plant 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.

Evaluation

Information concerning requirements of 10 CFR 50, Appendix E, Paragraph IV.B is discussed in the FSAR Annex H.1 references cited below:

10 CFR 50 Appendix E Section	EF2 FSAR Annex H.1 Reference(s)	Subject
IV.B	I	Dose assessment and field monitoring.

10 CFR 50 Appendix E Section	EF2 FSAR Annex H.1 Reference(s)	Subject
	D	Emergency action levels based on system and effluent parameters
	P	Emergency planning and agreement review

SECTION

10 CFR 50, Appendix E, Paragraph IV.C

Statement of Section

C. Activation of Emergency Organization

The entire spectrum of emergency conditions that involve the alerting of 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.

Evaluation

Information concerning requirements of 10 CFR 50, Appendix E, Paragraph IV.C is discussed in the FSAR Annex H.1 references cited below:

10 CFR 50 Appendix E Section	EF2 FSAR Annex H.1 Reference(s)	Subject
IV.C	D	Emergency classification systems and Tables per NUREG-0654.

10 CFR 50 Appendix E Section	EF2 FSAR Annex H.1 Reference(s)	Subject
	E.1	Message verification (Note: procedure to be established.)

SECTION

10 CFR 50, Appendix E, Paragraph IV.D

Statement of Section

D. Notification Procedures

1. Administrative and physical means for notifying local, State, and Federal officials and agencies and agreements reached with these officials and agencies for the prompt notification of the public and for public evacuation or other protective measures, should they become necessary, shall be described. This description shall include identification of the appropriate officials, by title and agency, of the State and local government agencies within the EPZs.

2. Provisions shall be described for yearly dissemination to the public within the plume exposure pathway EPZ of basic emergency planning information, such as the methods and times required for public notification and the protective actions planned if an accident occurs, general information as to the nature and effects of radiation, and a listing of local broadcast stations that will be used for dissemination of information during an emergency. Signs or other measures shall also be used to disseminate to any transient population within the plume exposure pathway EPZ appropriate information that would be helpful if an accident occurs.

3. A licensee shall have the capability to notify responsible State and local governmental agencies within 15 minutes after declaring an emergency. The licensee shall demonstrate that the State/local officials have the capability to make a public notification decision promptly on being informed by the licensee of an emergency condition. By July 1, 1981, the nuclear power reactor licensee shally demonstrate that

administrative and physical means have been established for alerting and providing prompt instructions to the public within the plume exposure pathway EPZ. The design objective 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 judgement whether or not to activate the public notification system. Where there is a decision to activate the notification system simultaneously or in a graduated or staged manner. The responsibility for activating such a public notification system shall remain with the appropriate government authorities.

Evaluation

Information concerning requirements of 10 CFR 50, Appendix E, Paragraph IV.D is discussed in the FSAR Annex H.1 references cited below:

10 CFR 50 Appendix E Section	EF2 FSAR Annex H.1 Reference(s)	Subject
IV.D.1	E	Notification of offsite organizations and the general public
IV.D.2	G.1	Education and information program
	G.2	Public awareness
IV.D.3	E.2	Notification of general public

EF2 presently complies with 10 CFR 50, Appendix E, Paragraph IV.D except for certain parts of IV.D.1 (letters of agreement and local plans are under development) and IV.D.3 (demonstation of public notification capability). An exercise will be conducted to demonstrate responce and notification capabilities.

Section

10 CFR 50, Appendix E, Paragraph IV.E

Statement of Section

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 onsite;

6. Arrangements for transportation of contaminated injured individuals from the site to specifically identified treatment facilities outside the site boundary;

7. Arrangements for treatment of individuals injured in support of licensed activities on the site at treatment facilities outside the site boundary;

8. A 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 communication systems shall be tested annually.

c. Provision for communications among the nuclear power reactor control room, the onsite technical support center, and the nearsite emergency operations facility; and among the nuclear facility, the principal State and local emergency operations centers, and the field assessment teams. Such communication systems shall be tested annually.

d. Provisions for communications by the licensee with NRC Headquarters and the appropriate NRC Regional Office Operations Center form the nuclear power reactor control room, the onsite technical suport center, and the near-site emergency operations facility. Such commnuications shall be tested monthly.

Evaluation

Information concerning the requirements of 10 CFR 50, Appendix E, Paragraph IV.E is discussed in the FSAR Annex H.1 references cited below:

10 CFR 50 Appendix E Section	EF2 FSAR Annex H.1 Reference(s)	Subject
IV.E.1	H.4	Emergency equipment and supplies; personnel monitoring.
IV.E.2	H.2, H.3	Meterological monitoring, radiation monitoring, and lab facilities.
IV.E.3	H.4	Emergency equipment and supplies; decontamination
IV.E.4	L.2	Onsite first-aid capability
IV.E.5	L.1.2	Medical services
IV.E.6	L.3	Transportation of contaminated injured personnel
IV.E.7	L.1.1	Hospital arrangements
IV.E.8	H.1.1 H.1.3	Technical Support Center Emergency Operations Facility
IV.E.9.a	F, N.2.1	Emergency communications; state and local
IV.E.9.b	N.2.1	Emergency communications; Federal

10 CFR 50 Appendix E Section	EF2 FSAR Annex H.1 Reference(s)	Subject
IV.E.9.c	F, Table F-1	Emergency communications; between CR, EDF, TSC, field teams and state/local respresentatives.
IV.E.9.d	Table F-1	Emergency communications; NRC

Section

10 CFR 50, Appendix E, Paragraph IV.F

Statement of Section

F. Training

The program to provide for (1) 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 (2) 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:

a. Directors and/or coordinators of the plant emergency organization;

b. Personnel responsible for accident assessment, including control room shift personnel;

c. Radiological monitoring teams;

d. Fire control teams (fire brigades);

e. Repair and damage control teams;

f. First aid and rescue teams;

g. Medical support personnel;

h. Licensee's headquarters support personnel;

i. Security personnel.

In addition, a radiological orientation training program shall be made available to local services personnel, e.g., local Civil Defense, local law enforcement personnel, local news media persons.

The plan shall describe provisions for the conduct of emergency preparedness exercises. Exercises shall test the adequacy of timing and content of implementing procedures and methods, test emergency equipment and communication networks, test the public notification system, and ensure that emergency organization personnel are familiar with their duties. Each licensee shall exercise at least annually the emergency plan for each site at which it has one or more power reactors licensed for operation. Both full-scale and small-scale exercises shall be conducted and shall include participation by appropriate State and local government agencies as follows:

1. A full-scale exercise which tests as much of the licensee, State, and local emergency plans as is reasonably achievable without mandatory public participation shall be conducted;

a. For each site at which one or more power reactors are located and licensed for operation, at least once every five years and at a frequency which will enable each State and local government within the plume exposure pathway EPZ to participate in at least one full-scale exercise per year and which will enable each State within the ingestion pathway to participate in at least one full-scale exercise every three years.

b. For each site at which a power reactor is located for which the first operating license for that site is issued

after the effective date of this amendment, within one year before the issuance of the operating license for full power, which will enable each State and local government within the plume exposure EPZ and each State withir the ingestion pathway EPZ to participate.

2. The plan shall also describe provisions for involving Federal emergency response agencies in a full-scale emergency preparedness exercise for each site at which one or more power reactors are located and licensed for operation at least once every 5 years;

3. A small-scale exercise which tests the adequacy of communication links, establishes that response agencies understand the emergency action levels, and tests at least one other component (e.g., medical or offsite monitoring) of the offsite emergency response plan for licensee, State, and local emergency plans for jurisdictions within the plume exposure pathway EPZ shall be conducted at each site at which one or more power reactors are located and licensed for operation each year a full-scale exercise is not conducted which involves the State(s) within the plume exposure pathway EPZ.

All training, including exercises, shall provide for formal critiques in order to identify weak areas that need corrections. Any weaknesses that are identified shall be corrected.

Evaluation

Information concerning requirements of 10 CFR 50, Appendix E, Paragraph IV.F is discussed in the FSAR Annex H.1 references cited below:

10 CFR 50 Appendix E Section	EF2 FSAR Annex H.1 Reference(s)	Subject
IV.F	O	Radiological emergency response tra ing (Note: detailed information on emergency response training is under development.
	G.5	News media acquaintance program
	N	Exercises and drills
	N.3.3	Critiques and corrective action

Detroit Edison is in the process of responding to a July 6, 1981 NRC request for a description of specialized initial training and retraining programs for emergency personnel.

Details concerning the conduct of exercises have not yet been formulated, but Detroit Edison and Monroe County have addressed the subject from a general veiwpoint. Schedular commitments for conducting exercises are under development.

Section

10 CFR 50, Appendix E, Paragraph IV.G

Statement of Section

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.

Evaluation

FSAR Annex H.1, Section P, describes the emergency planning update responsibilities.

Section

10 CFR 50, Appendix E, Paragraph IV.H

Statement of Section

H. Recovery

Criteria to be used to determine when, following an accident, re-entry of the facility would be appropriate or when operation could be resumed shall be described.

Evaluation

FSAR Annex H.1, Section M, describes re-entry, recovery, and post-accident operations.

Section

10 CFR 50, Appendix E, Paragraph V

Statement of Section

V. Implementation Procedures

No less than 180 days prior to scheduled issuance of an operating license for a nuclear power reactor or a license to possess nuclear material, 3 copies of each of the applicant's detailed implementing procedures for its emergency plan shall be submitted to the Director of the appropriate NRC Regional Office with 10 copies to the Director of Nuclear Reactor Regulation or, if appropriate, the Director of Nuclear Material Safety and Safeguards. In cases where a decision on an operating license is scheduled less than one year after the effective date of this rule, such implementing procedures shall be submitted as soon as practicable but before full power operation is authorized. Prior to March 1, 1981, licensees who are authorized to operate a nuclear power facility shall submit 3 copies each of the licensee's emergency plan implementing procedures to the Drector of the appropriate NRC Regional Office with 10 copies to the Director of Nuclear Reactor Regulation. Three copies each of any changes to maintain these implementing procedures up to date shall be submitted to the same NRC Regional Office with 10 copies to the Director of Nuclear Reactor Regulation or, if appropriate, the Director of Nuclear Material Safety and Safeguards within 30 days of such changes.

Evaluation

The emergency plan has been submitted as FSAR Annex H.1. Procedures will be submitted on a schedule commensurate with the fuel load date and with the regulation (Appendix E.V).

SECTION

10 CFR 50, Appendix G, Subparagraph III.A

A. To demonstrate compliance with the minimum fracture toughness requirements of sections IV and V of this appendix, ferritic materials shall be tested in accordance with the ASME Code, section 2300, "Fracture toughness requirements for materials." Both unirradiated and irradiated ferritic materials shall be tested for fracture toughness properties by means of the Charpy V-notch test specified by paragraph NB-2321.2 of the ASME Code. In addition, when required by the ASME Code, unirradiated ferritic materials shall be tested by means of the dropweight test specified by paragraph NB-2321.1 of the ASME Code. Provision shall be made for supplemental tests in crucial situations such as that described in Section V.C.

Evaluation

Compliance with 10 CFR 50, Appendix G is related to compliance with 10 CFR 50.55a. Detroit Edison requested a waiver from certain of the 50.55a requirements in a letter to the AEC dated May 3, 1973 (EF2-17172). AEC approval of the ASME Codes used by Detroit Edison is documented in a letter from the AEC dated July 12, 1973. The differences between the Codes required by 10 CFR 50.55a and the Codes authorized for use by Detroit Edison are shown in FSAR Table 3.2-3. The Code applied to the reactor pressure vessel was ASME Code Section III, Summer 1969 Addenda.

Fracture toughness testing performed on the reactor vessel material at EF-2 consisted of either drop-weight tests or longitudinally oriented CVN tests in accordance with ASME Code Section III including the Summer of 1969 Addenda.

SECTION

10 CFR 50, Appendix G, Subparagraph III.B.1

Statement of Section

B. Charpy V-notch (CVN) impact tests and dropweight tests shall be conducted in accordance with the following requirements:

1. Location and orientation of impact test specimens shall comply with the requirements of paragraph NB-2322 of the ASME Code.

Evaluation

Compliance with 10 CFR 50, Appendix G, is related to compliance with 10 CFR 50.55a. Detroit Edison requested a waiver from certain 50.55a requirements in a letter to the AEC dated May 3, 1973 (EF2-17172). AEC approval of the ASME Codes used by Detroit Edison is documented in a letter from the AEC dated July 12, 1973. The differences between the Codes required by 10 CFR 50.55a and the Codes authorized for use by Detroit Edison are shown in FSAR Table 3.2-3. The Code applied to the reactor pressure vessel was ASME Code Section III, Summer 1969 Addenda.

CVN tests of reactor pressure vessel material were conducted with longitudinal specimen orientations. Correlations were then derived to relate potential transverse orientations to the longitudinal orientations. FSAR Table 5.2-9 provides additional comments and explanation relevant to Subparagraph III.B.1.

Additional data was submitted to the NRC in transmittal letter EF2-54193, dated July 29, 1981 providing justification for the

correlations used to show that EF-2 complies with the Appendix G criteria.

SECTION

10 CFR 50, Appendix G, Subparagraph III.B.2

Statement of Section

2. Materials used to prepare test specimens shall be representative of the actual materials of the finished component as required by the applicable rules of the construction code under which the component is built pursuant to paragraph 50.55a, except that ferritic materials intended for the reactor vessel beltline region shall comply with the additional requirements of Section III.C of this appendix.

Evaluation

Compliance with 10 CFR 50, Appendix G, is related to compliance with 10 CFR 50.55a. Detroit Edison requested a waiver from certain 50.55a requirements in a letter to the AEC dated May 3, 1973 (EF2-17172). AEC approval of the ASME Codes used by Detroit Edison is documented in a letter from the AEC dated July 12, 1973. The differences between the Codes required by 10 CFR 50.55a and the Codes authorized for use by Detroit Edison are shown in FSAR Table 3.2-3. The Code applied to the reactor pressure vessel was ASME Code Section III, Summer 1969 Addenda.

EF2 obtained test specimen materials in accordance with the governing codes at the time of construction. Longitudinal versus transverse specimens were used.

Correlations were applied that added 30°F to the 50 ft-lb transition temperature determined on basis of longitudinal Charpy V-notch test data. Additional data was submitted to the NRC in EF2-54193, to support this correlation.

SECTION

10 CFR 50, Appendix G, Subparagraph III.B.3

Statement of Section

A. Calibration of temperature instruments and Charpy-V-Notch impact test machines used in impact testing shall comply with the requiremnts of NB-2360 of the ASME Code.

Evaluation

Compliance with 10 CFR 50, Appendix G, is related to compliance with 10 CFR 50.55a. Detroit Edison requested a waiver from certain 50.55a requirements in a letter to the AEC dated May 3, 1973 (EF2-17172). AEC approval of the ASME Codes used by Detroit Edison is documented in a letter from the AEC dated July 12, 1973. The differences between the Codes required by 10 CFR 50.55a and the Codes authorized for use by Detroit Edison are shown in FSAR Table 3.2-3.

ASME Section III Subsection NB-2360 was not in effect at the time calibration of instruments was conducted. The requirements of the 1968 edition of ASME Code Section III including Winter 1970 Addenda were met. An NRC acceptance discussion is provided on FSAR page 5.2-41. Calibration data is not currently available, however, reference to several Regulatory Guides, a General Electric position, and the ANSI QA manual indicate compliance with accepted standards for retention of calibration data.

SECTION

10 CFR 50, Appendix G, Subparagraph III.B.4

Statement of Section

Individuals performing fracture toughness tests shall be qualified by training and experience and shall have demonstrated competency to perform the tests in accord with written procedures of the component manufacturer.

Evaluation

Written procedures for performing fracture toughness tests were not in existence when the EF2 reactor vessel was purchased. At that time personnel were qualified by on-the-job training and previous experience.

SECTION

10 CFR 50, Appendix G, Subparagraph III.B.5

Statement of Section

5. Fracture toughness test results shall be recorded and shall include a certification by the licensee or person performing the tests for the licensee that:

a. The tests have been performed in compliance with the requirements of this appendix.

b. The test data are correctly reported and identified with the material intended for a pressure-retaining component.

c. The tests have been conducted using machines and instrumentation with available records of periodic calibration, and

d. Records of the qualifications of the individuals performing the tests are available upon request.

Evaluation

Refer to the preceding Evaluations for Subparagraphs III.B.1 through III.B.4 for additional references regarding compliance with fracture toughness testing. FSAR Table 5.2-9 also indicates compliance with this Subparagraph.

SECTION

10 CFR 50, Appendix G, Subparagraph III.C.1

Statement of Section

C. In addition to the test requirements of section III.A of this appendix, tests on materials of the reactor vessel beltline shall be conducted in accordance with the following minimum requirements;

1. Charpy V-notch (C.) impact tests shall be conducted at appropriate temperatures over a temperature range sufficient to define the C. test curves (including the upper-shelf levels) in terms of both fracture energy and lateral expansion of specimens. Location and orientation of impact test specimens shall comply with the requirements of paragraph NB-2322 of the ASME Code.

Evaluation

Compliance with 10 CFR 50, Appendix G, is related to compliance with 10 CFR 50.55a. Detroit Edison requested a waiver from certain 50.55a requirements in a letter to the AEC dated May 3, 1973 (EF2-17172). AEC approval of the ASME Codes used by Detroit Edison is documented in a letter from the AEC dated July 12, 1973. The differences between the Codes required by 10 CFR 50.55a and the Codes authorized for use by Detroit Edison are shown in FSAR Table 3.2-3. The Code applied to the reactor pressure vessel was ASME Code Section III, Summer 1969 Addenda.

As stated previously in the evaluation of Subparagraphs III.A and III.B.2 correlations were derived for vessel materials including vessel beltline materials to arrive at conservative estimates of RT_{NDT}. The location and orientation of vessel

beltline region material were obtained in accordance with the governing codes at the time of the component construction.

Additional data to demonstrate the conservatism of the RT_{NDT} estimates was submitted to the NRC in EF2-54193.

SECTION

10 CFR 50, Appendix G, Subparagraph III.C.2

Statement of Section

2. Materials used to prepare test specimens for the reactor vessel beltline region shall be taken directly from excess material and welds in the vessel shall course(s) following completion of the production longitudinal weld joint, and subjected to a heat treatment that produces metallurgical effects equivalent to those produced in the vessel material throughout its fabrication process, in accordance with paragraph NB-2211 of the ASME Code. Where seamless shell forgings are used, or where the same welding process is used for longitudinal and circumferential welds in plates, the test specimens may be taken from a separate weldment provided that such a weldment is prepared using excess material from the shell forging(s) or plates, as applicable, the same heat of filler material, and the same production welding conditions as those used in joining the corresponding shell materials.

Evaluation

Compliance with 10 CFR 50, Appendix G, is related to compliance with 10 CFR 50.55a. Detroit Edison requested a waiver from certain 50.55a requirements in a letter to the AEC dated May 3, 1973 (EF2-17172). AEC approval of the ASME Codes used by Detroit Edison is documented in a letter from the AEC dated July 12, 1973. The differences between the Codes required by 10 CFR 50.55a and the Codes authorized for use by Detroit Edison are shown in FSAR Table 3.2-3. The Code applied to the reactor pressure vessel was ASME Code Section III, Summer 1969 Addenda.

Weld material toughness test coupons were made with the exact same weld filler metal and procedure as in the actual vessel weld. However, these weld deposits were not necessarily made on the exact same heat of base plate as in the vessel. Base plate of the same specification was employed for this purpose. This small difference in base plate would not affect the testing of the weld metal since the Charpy specimen would be in the weld metal. Toughness testing of the exact base plate in the vessel was done separately.

SECTION

10 CFR, Appendix G, Subparagraph IV.A.2.a

Statement of Section

IV. Fracture Toughness Requirements

IV.A Requirements during Hydrostatic Tests and any Condition of Normal Operation, Including Anticipated Operational Occurrences

2. For vessels, exclusive of bolting or other fasteners:

a. Calculated stress intensity factors shall be lower than the reference stress intensity factors by the margins specified in the ASME Code Appendix G, "Protection Against Non-Ductile Failure." The calculation procedures shall comply with the procedures specified in the ASME Code Appendix G, but additional and alternative procedures may be used if the Commission determines that they provide equivalent margins of safety against fracture making appropriate allowance for all uncertainties in the data and analyses.

Evaluation

Compliance with 10 CFR 50, Appendix G, is related to compliance with 10 CFR 50.55a. Detroit Edison requested a waiver from certain 50.55a requirements in a letter to the AEC dated May 3, 1973 (EF2-17172). AEC approval of the ASME Codes used by Detroit Edison is documented in a letter from the AEC dated July 12, 1973. The differences between the Codes required by 10 CFR 50.55a and the Codes authorized for use by Detroit Edison are shown in FSAR Table 3.2-3. The Code applied to the reactor pressure vessel was ASME Code Section III, Summer 1969 Addenda.

FSAR Table 5.2-9 indicates that EF2 complies with this requirement.

SECTION

10 CFR 50, Appendix G, Subparagraph IV.A.2.b

Statement of Section

b. For Nozzles, flanges and shell regions near geometric discontinuities, the data and procedures required in addition to those specified in the ASME Code shall provide margins of safety comparable to those required for shells and heads remote from discontinuities.

Evaluation

Compliance with 10 CFR 50, Appendix G, is related to compliance with 10 CFR 50.55a. Detroit Edison requested a waiver from certain 50.55a requirements in a letter to the AEC dated May 3, 1973 (EF2-17172). AEC approval of the ASME Codes used by Detroit Edison is documented in a letter from the AEC dated July 12, 1973. The differences between the Codes required by 10 CFR 50.55a and the Codes authorized for use by Detroit Edison are shown in FSAR Table 3.2-3.

EF2 main flange discontinuity was considered by the addition of 60°F to the RT_{NDT} to establish minimum bolt-up temperature and pressurization. The minimum bolt-up temperature was 71°F based on an initial RT_{NDT} of 11°F for the shell plate connected to the closure-flange bolting. The RT_{NDT} is estimated in the same way as the vessel plate material.

The feedwater nozzle forgings CVN results indicate a RT_{NDT} of 12°F and the purchase specification for the forgings required a maximum of 40°F. Feedwater-nozzle discontinuites were considered by adjusting the results of a BWR/6 reactor discontinuity analysis to the Fermi-2 reactor. The adjustment involved increasing the minimum temperatures of operation by

the difference between the Fermi-2 and BWR/6 feedwater nozzle forging RT_{NDT}'s. The adjustment was based on a feedwater nozzle RT_{NDT} of $40^{o}F$.

Additional data to support the methods used in the estimated RT_{NDT} was submitted to the NRC on July 27, 1981 in EF2-54193.

SECTION

10 CFR 50, Appendix G, Paragraph IV.A.2.C

Statement of Section

c. Whenever the core is critical, the metal temperature of the reactor vessel shall be high enough to provide an adequate margin of protection against fracture, taking into account such factors as the potential for overstress and thermal shock during anticipated operational occurrences in the control of reactivity. In no case when the core is critical (other than for the purpose of low-level physics tests) shall the temperature of the reactor vessel be less than the minimum permissible temperature for the inservice system hydrostatic pressure test nor less than 40°F above that temperature required by section IV.A.2.a.

Evaluation

Compliance with 10 CFR 50, Appendix G, is related to compliance with 10 CFR 50.55a. Detroit Edison requested a waiver from certain 50.55a requirements in a letter to the AEC dated May 3, 1973 (EF2-17172). AEC approval of the ASME Codes used by Detroit Edison is documented in a letter from the AEC dated July 12, 1973. The differences between the Codes required by 10 CFR 50.55a and the Codes authorized for use by Detroit Edison are shown in FSAR Table 3.2-3.

Specific temperature limits for operation when the core is critical are based on a proposed modification to 10 CFR 50, Appendix G, Paragraph IV.A.2.C. The proposed modification and the justification for it are given in GE Licensing Topical Report NEDO-21778-A.

SECTION

10 CFR 50, Appendix G, Subparagraph IV.A.2.d

Statement of Section

d. If there is no fuel in the reactor during the initial preoperational system leakage and hydrostatic pressure tests, the minimum permissible test temperature shall be determined in accordance with paragraph G2410 of the ASME Code except that the factor of safety applied to each term making up the calculated stress intensity factor may be reduced to 1.0. In no case shall the test temperature be less than $RT_{NDT}+60^{\circ}F$.

Evaluation

Compliance with 10 CFR 50, Appendix G, is related to compliance with 10 CFR 50.55a. Detroit Edison requested a waiver from certain 50.55a requirements in a letter to the AEC dated May 3, 1973 (EF2-17172). AEC approval of the ASME Codes used by Detroit Edison is documented in a letter from the AEC dated July 12, 1973. The differences between the Codes required by 10 CFR 50.55a and the Codes authorized for use by Detroit Edison are shown in FSAR Table 3.2-3.

Based on 10 CFR 50 Appendix G, the preoperational system hydrostatic test at 1563 psig may be performed at $150^{\circ}F$, and this temperature is based on the feedwater nozzle.

SECTION

10 CFR 50, Appendix G, Subparagraph IV.A.3

Statement of Section

Materials for piping, pumps and valves shall meet the requirements of paragraph NB-2332 of the ASME Code. Materials for bolting and other fasteners shall meet the requirements of NB-2333 of the ASME Code.

Evaluation

Compliance with 10 CFR 50, Appendix G, is related to compliance with 10 CFR 50.55a. Detroit Edison requested a waiver from certain 50.55a requirements in a letter to the AEC dated May 3, 1973 (EF2-17172). AEC approval of the ASME Codes used by Detroit Edison is documented in a letter from the AEC dated July 12, 1973. The differences between the Codes required by 10 CFR 50.55a and the Codes authorized for use by Detroit Edison are shown in FSAR Table 3.2-3.

Main steam piping complies with USAS B-31.7, the main steam line isolation valve complies with the draft of the 1968 ASME Code and Addenda for pumps and valves, and SRV's comply with the 1968 ASME Code and the 1970 Summer Addenda. Current toughness requirements for closure head studs are met at 10°F. Table 5.2-9 of the Fermi 2 FSAR contains pertinent information.

SECTION

10 CFR 50, Appendix G, Subparagraph IV.B

Statement of Section

B. Reactor vessel beltline materials shall have minimum upper-shelf energy, as determined from Charpy V-notch tests on unirradiated speciments in accordance with paragraph NB-2322.2(a) of the ASME Code of 75 ft lbs unless it is demonstrated to the Commmission by appropriate data and analyses that lower values of upper-shelf fracture energy still provide adequate margin for deterioration from irradiation.

Evaluation

Compliance with 10 CFR 50, Appendix G, is related to compliance with 10 CFR 50.55a. Detroit Edison requested a waiver from certain 50.55a requirements in a letter to the AEC dated May 3, 1973 (EF2-17172). AEC approval of the ASME Codes used by Detroit Edison is documented in a letter from the AEC dated July 12, 1973. The differences between the Codes required by 10 CFR 50.55a and the Codes authorized for use by Detroit Edison are shown in FSAR Table 3.2-3.

Appendix G requires a minimum of 75 Ft-Lbs tranverse upper shelf CVN energy for beltline material. In accordance with MTEB No.2, 70 Ft-Lbs is adequate for the fluence which the Fermi-2 vessel is expected to receive. All beltline plates achieve a minimum of 70 Ft-Lbs CVN when the longitudinal upper shelf results are converted to tranverse results in accordance with the procedures described in MTEB No. 2. There were no upper shelf measurements made on beltline welds. Charpy V-Notch tests were conducted at a single test temperature of 10°F. These test results when compared to other similar BWR vessel test results indicate that beltline weld upper shelf energy will be adequate.

SECTION

10 CFR 50, Appendix G, Section V, Subparagraphs V.A, V.B, V.C, and V.D

Section V Inservice Requirements--Reactor Vessel Beltline Material

Evaluation

Surveillance of reactor vessel beltline materials is discussed below in the evaluations of 10 CFR 50, Appendix H. Also refer to FSAR Table 5.2-10.

SECTION

10 CFR 50, Appendix H, Subparagraph II.A

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 (E = 1MeV) at the end of the design life of the vessel will not exceed 10^{17} n/cm^2.

Evaluation

Edison will provide a materials surveillance program for the EF-2 reactor vessel. This program will monitor the neutron radiation effects on the reactor pressure vessel base metals, the weld heat affected zone metal, and the weld metal from a steel joint that simulates a welded joint in the reactor pressure vessel beltline.

SECTION

10 CFR 50, Appendix H, paragraph II.B

Statement of Section

B. Reactor vessels constructed of ferritic materials which do not meet the conditions of Section II.A shall have their beltline regions monitored by a surveillance program complying with the American Society for Testing and Materials (ASTM) Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels. ASTM Designation: E-185-73, except as modified by this appendix.

Evaluation

The EF-2 ferritic materials irradiation surveillance program is not strictly in accordance with ASTM-E-185-73, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels", in that the surveillance specimens are not necessarily from the limiting beltline material. This resulted from the fact that the fabrication of the EF-2 vessel predated the initiation of formal ferritic materials surveillance programs. Specimens are from representative material, however, and are used to predict behavior of the limiting material heat and heat/lot number from surveillance materials that are supplied.

The weld material has a copper content of 0.32 wt%, and is very close to being the limiting material in the vessel beltline. It may actually be limiting since the copper for the limiting material in seams 2-30 is not known but is probably lower than 0.32. The plate materials are very close to being the limiting beltline plates since they are only 8-10^{o}F lower EOL RT_{NDT} than the limiting plate.

SECTION

10 CFR 50, Appendix H, Paragraph II.C.1

Statement of Section

C. The surveillance program shall meet the following requirements:

1. Surveillance specimens shall be taken from locations alongside the fracture toughness test specimens required by Section III of Appendix G. The specimen types shall comply with the requirements of Section III.A of Appendix G (except that drop weight specimens are not required).

Evaluation

Compliance with 10 CFR 50, Appendix G, is related to compliance with 10 CFR 50.55a. Detroit Edison requested a waiver from certain 50.55a requirements in a letter to the AEC dated May 3, 1973 (EF2-17172). AEC approval of the ASME Codes used by Detroit Edison is documented in a letter from the AEC dated July 12, 1973. The differences between the Codes required by 10 CFR 50.55a and the Codes authorized for use by Detroit Edison are shown in FSAR Table 3.2-3.

The surveillance specimens were not taken from alongside the ASME NB-2300 specimens. This is not considered critical since they are as representative of the material in the vessel as the NB-2300 specimens. This requirement has been dropped from the current proposed revision (November 1980) to paragraph II.C.

SECTION

10 CFR 50, Appendix H, Paragraph II.C.2

Statement of Section

2. Surveillance specimen capsules shall 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 shall be done according to the requirements for permanent structural attachments to reactor vessels given in the ASME Code, Sections III and XI. The design and location of the capsules shall permit insertion of replacement capsules. Accelerated irradiation capsules may be used in addition to the required number of surveillance capsules specified in paragraph II.C.3.

Evaluation

The program will consist of three baskets, each containing tensile and CVN specimens hermetically sealed in an inert gas environment in thin-wall austenitic stainless steel capsules. The capsules are not buoyant and this presents no handling problems. The three baskets will be placed near the core midplane adjacent to the RPV wall where the neutron flux and temperature history closely approximate that of the inner surface of the RPV wall. In-service examination will be conducted as required.

SECTION

10 CFR 50, Appendix H, Paragraphs II.C.3.a, II.C.3.b, and II.C.3.c

Statement of Section

3. The required number of surveillance capsules and their withdrawal schedules are as follows:

a. For reactor vessels for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steel, making appropriate allowances for all uncertainties in the measurements, that the adjusted reference temperature established in accordance with Section III.B will not exceed 100°F at the end of the service lifetime of the reactor vessel, at least three surveillance capsules shall be provided for subsequent withdrawal as follows:

First capsule:	One-fourth service life
Second capsule:	Three-fourths service life
Third capsule:	Standby

In the event that the surveillance specimens exhibit, at one-quarter of the vessel's service life, a shift of the reference temperature greater than originally predicted for similar material as recorded in the applicable technical specification the remaining withdrawal schedule shall be modified as follows:

Revised
Withdrawal Schedule

Second capsule:	One-half service life
Third capsule:	Standby

b. For reactor vessels which do not meet the conditions of Section II.C.3.a but for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steels that the adjusted reference temperature will not exceed 200°F at the end of the service lifetime of the reactor vessel, at least four surveillance capsules shall be provided for the subsequent withdrawal as follows:

Withdrawal Schedule

First capsule: At the time when the predicted shift of the adjusted reference temperature is approximately 50°F or at one-fourth service life, whichever is earlier.

Second Capsule: At approximately one-half of the time interval between first and third capsule withdrawal.

Third capsule: Three-fourths service life.

Fourth capsule: Standby.

c. For reactor vessels which do not meet the conditions of Section II.C.3.b., at least five surveillance capsules shall be provided for subsequent withdrawal as follows:

Withdrawal Schedule

First capsule: At the time when the predicted shift of the adjusted reference temperature is approximately 50°F or at one-fourth service life, whichever is earlier.

Second and third capsules: At approximately one-third and two-thirds of the time interval between first and fourth capsule withdrawal.

Fourth capsule: Three-fourths of service life.

Fifth capsule: Standby.

Evaluation

The EF-2 program is required to conform to paragraph C.3.a of Section II because the adjusted reference temperature for the reactor vessel will not exceed 100°F. The capsule withdrawal schedule complies with that specified in paragraph II.C.3.a.

SECTION

10 CFR 50, Appendix H, Pargaraph II.C.3.d

Statement of Section

d. Provision shall also be made for additional surveillance tests to monitor the effects of annealing and subsequent irradiation.

Evaluation

The EF2 program includes additional specimens, contained in a standby capsule, that could be used to monitor annealing and subsequent irradiation or for other needs that may arise.

SECTION

10 CFR 50, Appendix H, Paragraphs II.C.3.e, II.C.3.f and II.C.3.g

Statement of Section

e. Withdrawal schedules may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedule.

f. If accelerated irradiation capsules are employed in addition to the minimum required number of surveillance capsules, the withdrawal schedule may be modified, taking into account the test results obtained from testing of the specimens in the accelerated capsules. The proposed modified withdrawal schedule in such cases shall be approved by the Commission on an individual case basis.

g. Proposed withdrawal schedules that differ from those specified in paragraphs a. through f. shall be submitted with a technical justification therefore, to the Commission for approval. The proposed schedule shall not be implemented without prior Commission approval.

Evaluation

The withdrawal schedule for the EF-2 reactor vessel material surveillance capsules are as specified in paragraph C.3.a.

SECTION

10 CFR 50, Appendix H, Paragraph II.C.4

Statement of Section

4. For multiple reactors located at a single site, an integrated surveillance program may be authorized by the Commission on an individual case basis, depending on the degree of commonality and the predicted severity of irradiation.

Evaluation

The EF-2 unit will be the only operating unit at the site. Unit 1 has been decommissioned.

SECTION

10 CFR 50, Appendix H, Section III - Fracture Toughness Tests

Paragraph III.A and III.B

A. Fracture toughness testing of the specimens withdrawn from the capsules shall be conducted in accordance with the requirements of Section III of Appendix Q, "Fracture Toughness Requirements."

B. The adjusted reference temperatures for the base metal, heat-affected zone, and weld metal shall be obtained from the test results by adding to the reference temperature the amount of the temperature shift in the Charpy test curves between the unirradiated material and the irradiated material, measured at the 50 foot-pound level or that measured at the 35 mil lateral expansion level, whichever temperature shift is greater. The highest adjusted reference temperature and the lowest upper-shelf energy level of all the beltline materials shall be used to verify that the fracture toughness requirements of Section V.B. of Appendix G are satisfied.

Evaluation

The EF2 program incorporates these requirements. Future adjusted reference temperatures will conform to the stated criteria.

SECTION

10 CFR 50, Appendix H, Section IV. - Report of Test Results

Statement of Section

Paragraph IV.A, IV.B and IV.C

A. Each capsule withdrawal and the results of the fracture toughness tests shall be the subject of a summary technical report to be provided to the Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. The report shall include a schematic diagram of the capsule locations in the reactor vessel, identification of specimens withdrawn, the test results, and the relationship of the measured results to those predicted for the reactor vessel beltline region.

B. The report shall also include the dosimetry measurements performed at each specimen withdrawal, analyses of the results which yield the calculated neutron fluence which the reactor vessel beltline region has received at the time of the tests, and comparisons with the originally predicted values of fluence.

C. The operating pressure and temperature limitations established for the period of operation of the reactor vessel between any two surveillance specimen withdrawals shall be specified in the report including any changes made in operational procedures to assure meeting such temperature limitations.

Evaluation

The EF2 program incorporates these requirements. Applicable limitations on operating pressure and temperature, if any,

will be reported and any changes in operating procedures will be documented in accordance with technical specification revision procedures, as needed.

SECTION

10 CFR 50, Appendix I, Paragraph II

Statement of Section

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 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.

B.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 form such radioactive iodine and radioactive material in particulate form for any individual in an unrestricted area for all path ways 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

an… $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.

Evaluation

Compliance with 10 CFR 50, Appendix I, Paragraph II is discussed in FSAR Appendix 11A (Evaluation of Fermi 2 to Demonstrate Compliance with Section II of Appendix I to 10 CFR Part 50) as described below:

10 CFR 50 Appendix I Section	EF2 FSAR Appendix 11A Discussion
II.A	0.23 mrem whole body and 0.48 mrem maximum organ dose (bone) from liquid effluents. Regulatory Guides 1.109, 1.112, and 1.113 are referenced.
II.B.1	4.74 mrad gamma and 2.68 mrad beta fromm airborne activity. Regulatory Guides 1.109 and 1.111 are referenced.
II.B.2	0.72 mrem whole body and 3.65 mrem skin dose from gaseous effluents.

10 CFR 50 Appendix I Section	EF2 FSAR Appendix 11A Discussion
II.C	11.19 mrem thyroid dose from iodine and particulates.
II.D	EF2 Construction Permit application date: April 29, 1969. NUREG-0389 concludes that radwaste augmentation cannot be performed in a cost-beneficial manner to plants with CP docketing before January 2, 1971.

SECTION

10 CFR 50, Appendix I, Paragraph III

Statement of Section

III. Implementation.

A.1. Conformity with the guides on design objectives of Section II shall be demonstrated by calculational procedures based upon models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated, all uncertainties being considered together. Account shall be taken of the cumulative effect of all sources and pathways within the plant contributing to the particular type of effluent being considered. For determination of design objectives in accordance with the guides of Section II, the estimation 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 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.

Evaluation

Compliance with 10 CFR 50, Appendix E, Paragraph III is discussed in FSAR Appendix 11A as described below:

10 CFR 50 Appendix I Section	EF2 FSAR Appendix 11A Discussion
III.A.1	FSAR Appendix 11A, Section III, lists conservative assumptions used in liquid effluent calculations. Section IV lists conservative assumptions used in gaseous effluent calculations.

10 CFR 50 Appendix I Section	EF2 FSAR Appendix 11A Discussion
	FSAR Appendix 11A, Annex A, gives data needed for radiation source term calculations. Annex B gives the atmospheric transport and dispersion model used for 10 CFR 50, Appendix I calculations.
	Radiological effluent surveillance will be per Technical Specifications.
III.A.2	Modeling and calculation process follows Regulatory Guides and uses NRC models and codes.
III.B	Not applicable because calculated exposures are lower than II.C guidelines.

SECTION

10 CFR 50, Appendix I, Paragraph IV

Statement of Section

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 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 releases 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 operation, compatible with considerations of health and safety, to assure that the public is provided a dependable

source of power even under unusual operating conditions which may temporarily result in releases higher than such 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 operating 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:

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 to the appropriate NRC Regional Office shown in Appendix D of Part 20 of this chapter, with a copy to the Director of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, 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 for operation in a license authorizing operation of a light-water-cooled nuclear power reactor.

Evaluation

Technical Specifications will be formally submitted to the NRC.

SECTION

10 CFR 50, Appendix I, Paragraph V

Statement of Section

V. Effective dates.

A. The guides for limiting conditions for operation set forth in this Appendix shall be applicable in any case in which an application was filed on or after January 2, 1971, for a 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.

Evaluation

Compliance with 10 CFR 50, Appendix I, Paragraph V is discussed below:

10 CFR 50 Appendix I Section	EF2 Compliance Discussion
V.A	Not applicable because EF2 CP application date was April 29, 1969.
V.B	This section applies to Fermi 2. EF2 does not have an Operating License, however, Appendix I information has been docketed as Appendix 11A to the FSAR and will be updated to reflect ongoing radwaste system modifications.

SECTION

10 CFR 50, Appendix J

Statement of Section

I. Introduction

Appendix J to 10 CFR 50 provides test requirements for pre-operational and periodic verification of the leak tight integrity of the primary reactor containment, and systems and components which penetrate the containment, and establishes the acceptance criteria for such tests.

The tests assure that leakage through the primary containment and systems and components penetrating the primary containment shall not exceed allowable leak rate values specified in Tech. Specs. or associated Bases and that periodic surveillance of containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment and associated penetrations and isolation valves.

II. Explanation of Terms

Not applicable to this evaluation

III. Leakage Testing Requirements

The introduction states the requirement that a program be developed consisting of a schedule for conducting Type A, B, and C tests for the purpose of leak testing the containment and related systems and components penetrating the containment pressure boundary.

Evaluation

Type A, B, and C tests are described in detail in FSAR Section 6.2.4.4. Additional supporting data is contained in FSAR Sections 6.2.4.2.2.1, 6.2.4.2.6, 6.2.1.2.1, Tables 6.2-1 and 6.2.2 and the EF-2 Tech. Specs.

SECTION

10 CFR 50, Appendix J, Subparagraph III.A

Statement of Section

This subparagraph describes Type A tests in the areas of pretest requirements, conduct of tests, test methods, preoperational leakage rate tests, periodic leakage rate tests, test schedules and acceptance criteria.

Evaluation

EF2 Type A tests are to be conducted in accordance with Appendix J requirements supplemented by ANSI-N45.5, "Leakage Rate Testing of Containment Structures for Nuclear Reactors." EF2 Type A leak test acceptance criteria are in accordance with the requirements of Appendix J.

SECTION

10 CFR 50, Appendix J, Subparagraph III.B

Statement of Section

This subparagraph presents acceptable means of performing pre-operational and periodic Type B tests. The paragraph includes the subjects of test methods, test pressure and acceptance criteria.

Evaluation

EF2 Type B tests are to be conducted in accordance with Appendix J requirements. The leakage rate acceptance criteria specified for EF2 Type B tests are in accordance with those specified in Appendix J. The testing schedule is in accordance with the requirements of Appendix J.

SECTION

10 CFR 50, Appendix J, Subparagraph III.C

Statement of Section

This subparagraph presents acceptable methods and acceptance criteria for performance of Type C tests. It presents acceptable test methods, test pressure and acceptance criteria.

Evaluation

Type C tests will be conducted in accordance with the requirements of Appendix J. The test methods, test pressure, acceptance criteria, and test schedules are in accordance with the requirements of Appendix J. EF2 complies with the Type C test methods, test pressure, test interval and acceptance criteria as specified in Subparagraph III.c of Appendix J.

SECTION

10 CFR 50, Appendix J, Subparagraph III.D

Statement of Section

This subparagraph provides a periodic test schedule for Type A, B and C tests.

Evaluation

As previously noted in the evaluations of Subparagraphs III.A, B, and C the test intervals specified for those tests are in accordance with the requirements of Appendix J. Leak rates for air locks will be specified in the Technical Specifications.

SECTION

10 CFR 50, Appendix J, Subparagraphs V, A and B

Statement of Section

Subparagraph V, A and B provide details with regards to containment inspection and its reporting requirements and the report of test results.

Evaluation

This paragraph provides specific reporting requirements on the tests and inspections and additionally provides corrective action for tests which fail the acceptance criteria specified in Appendix J. Surveillance and reporting requirements are covered by the technical specifications.

SECTION

10 CFR 50, Appendix K, ECCS Evaluation Models

Evaluation

Compliance with 10 CFR 50, Appendix K is discussed in Section 6.3.3 of the EF2 FSAR:

> ECCS performance is determined by applying 10 CFR 50, Appendix K evaluation models and then showing conformance with 10 CFR 50.46 acceptance criteria. NEDO-20556, entitled "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR Part 50, Appendix K," contains the complete description of methods used by the NSSS vendor to perform ECCS calculations. The NEDO-20566 preface says the model has been found by NRC to comply with Appendix K. The EF2 FSAR contains a summary description of the model and the calculations.

SECTION

10 CFR 50, Appendix R, Fire Protection Program

Evaluation

10 CFR 50, Appendix R is not currently applicable to EF2 because EF2 does not have an Operating License. However, NRC Question 021.33 states that compliance with Appendix R will be made a license condition. The DECo response is contained in letter EF2-53498 of June 9, 1981 and states that a fire hazards analysis has been conducted for EF2 and a point-by-point comparison with Appendix A to BTP APCSB 9.5-1 is documented in FSAR Appendix 9B. It is Detroit Edison's belief that EF2 complies with Appendix R and references a presentation to that effect made to the NRC in Bethesda, Maryland on May 27, 1981. EF2 is continuing its discussions with the NRC on the fire protection issue.

SECTION

10 CFR 100.10, Factors to be considered when evaluating sites.

Statement of Section

Factors considered in the evaluation of sites include those relating both to the proposed reactor design and the characteristics peculiar to the site. It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in release of significant quantities of radioactive fission products. In addition, the site location and the engineered features included as safeguards against the hazardous consequences of an accident, should one occur, should insure a low risk of public exposure. In particular, the Commission will take the following factors into consideration in determining the acceptability of a site for a power or testing reactor:

(a) Characteristics of reactor design and proposed operation including:

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

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

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

(4) 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.

(b) Population density and use characteristics of the site environs, including the exclusion area, low population zone, and population center distance.

(c) Physical characteristics of the site, including seismology, meteorology, geology and hydrology.

(1) Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," describes the nature of investigations required to obtain the geologic and seismic data necessary to determine site suitability and to provide reasonable assurance that a nuclear power plant can be constructed and operated at a proposed site without undue risk to the health and safety of the public. It describes procedures for determining the quantitative vibratory ground motion design basis at a site due to earthquakes and describes information needed to determine whether and to what extent a nuclear power plant need be designed to withstand the effects of surface faulting.

(2) Meteorological conditions at the site and in the surrounding area should be considered.

(3) Geological and hydrological characteristics of the proposed site may have a bearing on the consequences of an escape of radioactive material from the facility. Special precautions should be planned if a reactor is to be located at a site where a significant quantity of radioactive effluent might accidentally flow into nearby streams or rivers or might find ready access to underground water tables.

(d) Where unfavorable physical characteristics of the site exist, the proposed site may nevertheless be found to be acceptable if the design of the facility includes appropriate and adequate compensating engineering safeguards.

Evaluation

Compliance with 10 CFR 100.10 is discussed in the references cited below:

10 CFR 100.10 Subsection	EF2 FSAR Reference(s)	Subject
(a)	Chapter 4	Reactor design chacteristics.
	Chapters 5, 6 7,8,9, and 10	NSSS design characteristics.
	Chapter 13	Operational characteristics.
(a)(1)	1.1	3430 Mwt/1154 Mwe (rated) for the commercial supply of electricity.
	1.2.1.1.1	To generate electricity in a safe and reliable manner.
	11.2	Radioactive material inventory in the radwaste system.
a(2)	3	Engineering standards, e.g., General Design Criteria, safety classifications, detailed design information.
	5	Engineering standards for RCPB materials.
	Appendix A	Conformance with Regulatory Guides.
	17	Quality Assurance Program

10 CFR 100.10 Subsection	EF2 FSAR Reference(s)	Subject
a(3)	1, 2	Site features are discussed throughout Chapters 1 and 2.
a(4)	6	Engineered Safety Features.

SECTION

10 CFR 100.11, Determination of exclusion area, low population zone, and population center distance.

Statement of Section

(a) As an aid in evaluating a proposed site, an applicant should assume a fission product release from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following:

(1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(2) A low population zone of such size that an individual located at any point on its outer boundary 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 total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(3) A population center distance of at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. In applying this guide, the

boundary of the population center shall be determined upon consideration of population distribution. Political boundaries are not controlling in the application of this guide. Where very large cities are involved, a greater distance may be necessary because of total integrated population dose consideration.

(b) For sites for multiple reactor facilities consideration should be given to the following:

(1) If the reactors are independent to the extent that an accident in one reactor would not initiate an accident in another, the size of the exclusion area, low population zone and population center distance shall be fulfilled with respect to each reactor individually. The envelopes of the plan overlay of the areas so calculated shall then be taken as their respective boundaries.

(2) If the reactors are interconnected to the extent that an accident in one reactor could affect the safety of operation of any other, the size of the exclusion area, low population zone and population center distance shall be based upon the assumption that all interconnected reactors emit their postulated fission product releases simultaneously. This requirement may be reduced in relation to the degree of coupling between reactors, the probability of concomitant accidents and the propability that an individual would not be exposed to the radiation effects from simultaneous releases. The applicant would be expected to justify to the satisfacion of the Commission the basis for such a reduction in the source term.

(3) The applicant is expected to show that the simultaneous operation of multiple reactors at a site will not result in total radioactive effluent releases beyond the allowable limits of applicable regulations.

Note: For further guidance in developing the exclusion area, the low population zone, and the population center distance, reference is made to Technical Information Document 14844, dated March 23, 1962, which contains a procedural method and a sample calculation that result in distances roughly reflecting current siting practices of the Commission. The calculations described in Technical Information Document 14844 may be used as a point of departure for consideration of particular site requirements which may result from evaluation of the characteristics of a particular reactor, its purpose and method of operation.

Evaluation

Compliance with 10 CFR 100.11 is discussed in the references cited below:

10 CFR 100.11 Subsection	EF2 FSAR Reference(s)	Subject
(a)	Appendix 15A	Dose calculation models.
(a)(1)	2.1.2	An exclusion area of 915 meter from the center line of the containment building is described in this section.
(a)(2)	2.1.3.3	3 mile Low Population Zone with 1970 census data and projected demographic data.

10 CFR 100.11 Subsection	EF2 FSAR Reference(s)	Subject
(a)(3)	2.1.3.3	Monroe is the only large town closer than 10 miles; all other towns close-in have fewer than 3500 residences. Population Center Distance = 1 1/3 x LPZ = 4 miles; nearest Monroe corporate boundary is 5.5 miles.
(b)	--	Not applicable; EF2 is not a multiple reactor site.

SECTION

10 CFR 100, Appendix A, Paragraph IV(a)

Statement of Section

(a) Required Investigation for Vibratory Ground Motion. The purpose of the investigations required by this paragraph is to obtain information needed to describe the vibratory ground motion produced by the Safe Shutdown Earthquake. All of the steps in paragraphs (a)(5) through (a)(8) of this section need not be carried out if the Safe Shutdown Earthquake can be clearly established by investigations and determinations of a lesser scope. The investigations required by this paragraph provide an adequate basis for selection of an Operating Basis Earthquake. The investigations shall include the following:

(1) Determination of the lithologic, stratigraphic, hydrologic, and structural geologic conditions of the site and the region surrounding the site, including its geologic history;

(2) Identification and evaluation of tectonic structures underlying the site and the region surrounding the site, whether buried or expressed at the surface. The evaluation should consider the possible effects caused by man's activities such as withdrawal of fluid from or addition of fluid to the subsurface, extraction of minerals, or the loading effects of dams or reservoirs;

(3) Evaluation of physical evidence concerning the behavior during prior earthquakes of the surficial geologic materials and the substrata underlying the site from the lithologic, stratigraphic, and structural geologic studies;

(4) Determination of the static and dynamic engineering properties of the materials underlying the site. Included

should be properties needed to determine the behavior of the underlying material during earthquakes and the characteristics of the underlying material in transmitting earthquake-induced motions to the foundations of the plant, such as seismic wave velocities, density, water content, porosity, and strength;

(5) Listing of all historically reported earthquakes which have affected or which could reasonably be expected to have affected the site, including the date of occurrence and the following measured or estimated data: magnitude or highest intensity, and a plot of the epicenter or location of highest intensity. Where historically reported earthquakes could have caused a maximum ground acceleration of at least one-tenth the acceleration of gravity (0.1g) at the foundations of the proposed nuclear power plant structures, the acceleration of intensity and duration of ground shaking at these foundations shall also be estimated. Since earthquakes have been reported in terms of various parameters such as magnitude, intensity at a given location, and effect on ground, structures, and people at a specific location, some of these data may have to be estimated by use of appropriate empirical relationships. The comparative characteristics of the material underlying the epicentral location or region of highest intensity and of the material underlying the site in transmitting earthquake vibratory motion shall be considered;

(6) Correlation of epicenters or locations of highest intensity of historically reported earthquakes, where possible, with tectonic structures any part of which is located within 200 miles of the site. Epicenters or locations of highest intensity which cannot be reasonably correlated with tectonic structures shall be identified with tectonic provinces any part of which is located within 200 miles of the site;

(7) For faults, any part of which is within 200 miles of the site and which may be of significance in establishing the Safe Shutdown Earthquake, determination of whether these faults are to be considered as capable faults. This determination is required in order to permit appropriate consideration of the geologic history of such faults in establishing the Safe Shutdown Earthquake. For guidance in determining which faults may be of significance in determining the Safe Shutdown Earthquake, Table 1 of this appendix presents the minimum length of fault to be considered versus distance from site. Capable faults of lesser length than those indicated in Table 1 and faults which are not capable faults need not be considered in determining the Safe Shutdown Earthquake, except where unusual circumstances indicate such consideration is appropriate;

TABLE 1

Distance from the site (miles):	Minimum length[a]
0 to 20	1
Greater than 20 to 50	5
Greater than 100 to 150	10
Greater than 150 to 200	40

[a]Minimum length of fault (miles) which shall be considered in establishing Safe Shutdown Earthquake.

(8) For capable faults, any part of which is within 200 miles of the site and which may be of significicnace in establishing the Safe Shutdown Earthquake, determination of:

(i) The length of the fault;

(ii) The relationship of the fault to regional tectonic structures; and

(iii) The nature, amount, and geologic history of displacements along the fault including particularly the estimated amount of the maximum Quaternary displacement related to any one earthquake along the fault.

Evaluation

Compliance with 10 CFR 100, Appendix A, Paragraph IV(a) is discussed in the FSAR references cited below:

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference (s)	Subject
IV (a) (1)	2.5.1.1.2, 2.5.1.2	The Fermi 2 site lies in the Michigan Basin Region. The rock units in the region consist of sedimentary strata of Cambrian through Jurassic Periods overlying igneous and/or metamorphic complex of Precambrian aged rocks. At the site the rocks are of Cambrian through Devonian age and consist of 2500 to 3000 feet of limestones, dolomites, sandstones and shales. Rocks of later periods were either not deposited in the area or were eroded away. Dolomite of the Bass Islands Group is the uppermost rock stratum and is overlain by Pleistocene deposits of lacustrine clay, till and sand. These soil deposits are approximately 25 feet thick.
	2.4,2.5.4.6	The site is located on the western shore of Lake Erie. Surface drainage and groundwater gradients are toward

10 CFR 100 Appendix A Subsection	E'2 FSAR Reference (s)	Subject
		Lake Erie. Borings taken at the site showed small artesian flow at shallow depth with increased artesian heads at greater depth.
	2.5.1.1.3, 2.5.1.2.3	The Fermi 2 site is located within the Central Stable Region tectonic province which is characterized by a thick sequence of sedimentary strata overlying the Precambrian basement rocks. During Paleozoic and Mesozoic time, the area was subjected to vertical crustal movements forming basins and arches. These arches and basins have been modified by local folding and faulting. No faulting was disclosed at the site and differential elevations were investigated and have been attributed to a shallow synclinal fold whose axis passes through the site.
	2.5.1.1.5	The geologic history from Precambrian through Pleistocene periods is described.
IV (a) (2)	2.5.1.1.3	The Structural geology of the Central Stable Region is described and consists of basins and arches which have been modified by folding and faulting.

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference (s)	Subject
	2.5.1.1.3.1 2.5.1.2.3.2	Regional and site geologic foldings are investigated. The site lies in a shallow synclinal fold.
	2.5.1.1.3.2, 2.5.1.2.3.3	Regional and site geologic faulting are investigated. There are four major faults identified within 100 miles of the site, however none of these fault lines are within 25 miles of the site.
	2.5.1.2.6	Neither hydrocarbon production nor subsurface gas storage are believed to have great potential within the site vicinity.
	2.5.1.2.7	There is no solution mining of salt within 17 miles of the site. There are no known salt deposits within 15 miles of the site. There are no anticipated threats of gas blowouts at the site.
	2.5.4.5	A test blast program was conducted to control excavation blasting for Fermi 2 relative to Fermi 1 and blasting criteria were developed.
	2.5 4.6	The groundwater conditions at the site were monitored. Dewatering for quarry operations and plant excavations

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference (s)	Subject
		caused a reversal in groundwater gradient from toward Lake Erie to away from the lake. Note: There are no permanent dewatering systems in operation. There are no dams and reservoirs which could cause loading effects in the site vicinity. Quarrying was only conducted to obtain rock for use as backfill for plant structures.
IV (a) (3)	2.5.2.3	There is no physical evidence at the Fermi 2 site of prior earthquake activity.
IV (a) (4)	2.5.1.2.7	The dolomite of the Bass Island group is considered satisfactory to support the plant structures.
	2.5.1.2.8	Tests borings ranging in depth from 12 to 325 feet were drilled to determine lithology, structure and physical properties of the subsurface materials. Water pressure tests were conducted on 5 borings to establish criteria for dewater [illegible] and foundation grouting. Static [illegible] dynamic soil and rock properties were determined from laboratory tests.

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference (s)	Subject
	2.5.1.2.9	Geophysical exploration at the site consisted of: -Borehole logging in 3 deep borings to measure compressional wave velocity and bulk density at one foot intervals. Shear wave velocity and Poisson's ratio were then computed based on these results. -Seismic Refraction Survey was conducted to measure compression wave velocities in underlying rock. Results were compared to other geophysical refraction work in Southeast Michigan. Vibration levels during these surveys were measured within the existing Fermi 1 plant by a blast monitoring program. -Ambient Vibration Measurements were made at two locations using micromotion equipment. At one site the response was very low indicative of hard rock. At the second location the response was so low that it was obscured by machinery noise.

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference (s)	Subject
	2.5.1.2.10	Laboratory tests on rock cores were conducted to measure physical properties. This testing included: Density tests Unconfined compression tests Shockscope tests and Resonant column tests.
	2.5.4.2	The test borings, geophysical exploration and laboratory testing results were used together with published properties of similar materials to generate the properties of underlying materials which are given in this section.
IV (a) (5)	2.5.2.5	A listing of historic earthquakes is presented with epicenter location, epicenter intensity and estimated site intensity. Table 2.5-14 gives all earthquakes with epicenters within 50 miles of the site and all earthquakes of Intensity V or greater within 200 miles of the site. Table 2.5-13 gives larger earthquakes greater than 200 miles from the site. None of the historic earthquakes has an estimated intensity greater than IV at the site. Additionally in response to a directive from NRC, DECo report

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference (s)	Subject
		EF2-53332, Supplementary Seismic Evaluation Report, discusses data in terms of Richter magnitudes. The report considers a 5.3±0.5 magnitude earthquake and provides confirmation of the adequacy of EF2 seismic design basis.
IV (a) (6)	2.5.2.6	The majority of the significant earthquakes in the region can be associated with well-defined geologic structural zones as shown in Figure 2.5-64. The 1947 Intensity VI southcentral Michigan and the 1943 Intensity V Lake Erie shocks are the largest earthquakes not positively related to specific tectonic features.
IV (a) (7)	2.5.2.7	There are no known capable faults within 200 miles of the Fermi 2 site.
IV (a) (8)	2.5.2.7	Not applicable: There are no known capable faults within 200 miles of the site.

SECTION

10 CFR 100, Appendix A, Paragraph IV(b)

Statement of Section

(b) Required Investigation for Surface Faulting. The purpose of the investigations required by this paragraph is to obtain information to determine whether and to what extent the nuclear power plant need be designed for surface faulting. If the design basis for surface faulting can be clearly established by investigations of a lesser scope, not all of the steps in paragraphs (b)(4) through (b)(7) of this section need be carried out. The investigations shall include the following:

(1) Determination of the lithologic, stratigraphic, hydrologic, and structural geologic conditions of the site and the area surrounding the site, including its geologic history;

(2) Evaluation of tectonic sturctures underlying the site, whether buried or expressed at the surface, with regard to their potential for causing surface displacement at or near the site. The evaluation shall consider the possible effects caused by man's activities such as withdrawal of fluid from or addition of fluid to the subsurface, extraction of minerals, or the loading effects of dams or reservoirs;

(3) Determination of geologic evidence of fault offset at or near the ground surface at or near the site;

(4) For faults greater than 1000 feet long, any part of which is within 5 miles of the site, determination of whether these faults are to be considered as capable faults;

(5) Listing of all historically reported earthquakes which can reasonably be associated with capable faults greater than 1000 feet long, any part of which is within 5 miles[5] of the site, including the date of occurrence and the following measured or estimated data: magnitude or highest intensity, and a plot of the epicenter or region of highest intensity;

(6) Correlation of epicenters or locations of highest intensity of historically reported earthquakes with capable faults greater than 1000 feet long, any part of which is within 5 miles of the site;

(7) For capable faults greater than 1000 feet long, any part of which is within 5 miles of the site, determination of:

(i) The length of the fault;

(ii) The relationship of the fault to regional tectonic structures;

(iii) The nature, amount, and geologic history of displacements along the fault, including particlarly the estimated amount of the maximum Quaternary displacement related to any one earthquake along the fault, and

(iv) The outer limits of the fault established by mapping Quarternary fault traces for 10 miles along its trend in both directions from the point of its nearest approach to the site.

Evaluation

Compliance with 10 CFR 100, Appendix A, Paragraph IV(b) is discussed in the FSAR references cited below:

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference(s)	Subject
IV (b) (1)	2.5.1, 2.5.2&2.5.3	Reference to lithologic, stratagraphic, hydrologic and geologic discussions is given in response to 10 CFR 100, Appendix A, Section IV (a) (1).
IV (b) (2)	2.5.1, 2.5.2	Tectonic features are discussed. There are four major faults identified within 100 miles of the site, however none are within 25 miles of the site.
	2.5.3	No surface or subsurface faults exist within 25 miles of the site.
	2.5.3.9	Surface faulting is not part of the design basis.
IV (b) (3)	2.5.3.2	There are no faults so there is no geologic evidence of fault offset at or near the ground surface at or near the site.
IV (b) (4) through IV (b) (7)	2.5.3.3	There are no faults within 25 miles of the site, therefore, there can be no capable faults within 5 miles of the site.

SECTION

10 CFR 100, Appendix A, Paragraph IV(c)

Statement of Section

(c) Required Investigation for Seismically Induced Floods and Water Waves.

(1) For coastal sites, the investigations shall include the determination of:

(i) Information regarding distantly and locally generated waves or tsunami which have affected or could have affected the site. Available evidence regarding the runup and drawdown associated with historic tsunami in the same coastal region as the site shall also be included;

(ii) Local features of coastal topography which might tend to modify tsunami runup or drawdown. Appropriate available evidence regarding historic local modifications in tsunami runup or drawdown at coastal locations having topography similar to that of the site shall also be obtained; and

(iii) Appropriate geologic and seismic evidence to provide information for establishing the design basis for seismically induced floods or water waves from a local offshore earthquake, from local offshore effects of an onshore earthquake, or from coastal subsidence. This evidence shall be determined, to the extent practical, by a procedure similar to that required in paragraphs (a) and (b) of this section. The probable slip characteristics of offshore faults shall also be considered as well as the potential for offshore slides in submarine material.

(2) For sites located near lakes and rivers, investigations similar to those required in paragraph (c)(1) of this section shall be carried out, as appropriate, to determine the potential for the nuclear power plant to be exposed to seismically induced floods and water waves as, for example, from the failure during an earthquake of an upstream dam or from slides of earth or debris into a nearby lake.

Evaluation

Compliance with 10 CFR 100, Appendix A, Paragraph IV(c) is discussed in the FSAR references cited below:

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference(s)	Subject
IV (c) (1)	---	Not applicable: EF2 is not a coastal (ocean) site.
IV (c) (2)	2.4.4	Potential dam failures (seismically induced) - there are no dams in southeastern Michigan the failure of which could affect the site.
	2.4.6	Probable maximum tsunami flooding - there have been no recorded tsunamis at the site, and seismic activity is too low to make tsunamis a concern.

SECTION

10 CFR 100, Appendix A, Paragraph V(a)

Statement of Section

(a) Determination of Design Basis for Vibratory Ground Motion. The design of each nuclear power plant shall take into account the potential effects of vibratory ground motion caused by earthquakes. The design basis for the maximum vibratory ground motion and the expected vibratory ground motion should be determined through evaluation of the seismology, geology, and the seismic and geologic history of the site and the surrounding region. The most severe earthquakes associated with tectonic structures or tectonic provinces in the region surrounding the site should be identified, considering those historically reported earthquakes that can be associated with these structures or provinces and other relevant factors. If faults in the region surrounding the site are capable faults, the most severe earthquakes associated with these faults should be determined by also considering their geologic history. The vibratory ground motion at the site should be then determined by assuming that the epicenters or locations of highest intensity of the earthquakes are situated at the point of the tectonic structures or tectonic provinces nearest to the site. The earthquake which could cause the maximum vibratory ground motion at the site should be designated the Safe Shutdown Earthquake. The specific procedures for determining the design basis for vibratory ground motion are given in the following paragraphs.

(1) Determination of Safe Shutdown Earthquake. The Safe Shutdown Earthquake shall be identified through evaluation of seismic and geologic information developed pursuant to the requirements of paragraph IV(a), as follows:

(i) The historic earthquakes of greatest magnitude or intensity which have been correlated with tectonic structures pursuant to the requirements of paragraph (a)(6) of Section IV shall be determined. In addition, for capable faults, the information required by paragraph (a)(8) of Section IV shall also be taken into account in determining the earthquakes of greatest magnitude related to the faults. The magnitude or intensity of earthquakes based on geologic evidence may be larger than that of the maximum earthquakes historically recorded. The accelerations at the site shall be determined assuming that the epicenters of the earthquakes of greatest magnitude or the locations of highest intensity related to the tectonic structures are situated at the point on the structures closest to the site;

(ii) Where epicenters or locations of highest intensity of historically reported earthquakes cannot be reasonably related to tectonic structures but are identified pursuant to the requirements of paragraph (a)(6) of Section IV with tectonic provinces in which the site is located, the accelerations at the site shall be determined assuming that these earthquakes occur at the site.

(iii) Where epicenters or locations of the highest intensity of historically reported earthquakes cannot be reasonably related to tectonic structures but are identified pursuant to the requirements of paragraph (a)(6) of Section IV with tectonic provinces in which the site is not located, the accelerations at the site shall be determined assuming that the epicenters or locations of highest intensity of these earthquakes are at the closest point to the site on the boundary of the tectonic province;

(iv) The earthquake producing the maximum vibratory acceleration at the site, as determined from paragraphs (a)(1)(i) through (iii) of this section shall be designated the Safe

Shutdown Earthquake for vibratory ground motion, except as noted in paragraph (a)(1)(v) of this section. The characteristics of the Safe Shutdown Earthquake shall be derived from more than one earthquake determined from paragraphs (a)(1)(i) through (iii) of this section, where necessary to assure that the maximum vibratory acceleration at the site throughout the frequency range of interest is included. In the case where a causative fault is near the site, the effect of proximity of an earthquake on the spectral characteristics of the Safe Shutdown Earthquake shall be taken into account.

The procedures in paragraphs (a)(1)(i) through (a)(1)(iii) of this section shall be applied in a conservative manner. The determinations carried out in accordance with paragraphs (a)(1)(ii) and (a)(1)(iii) shall assure that the Safe Shutdown Earthquake intensity is, as a minimum, equal to the maximum historic earthquake intensity experienced within the tectonic province in which the site is located. In the event that geological and seismological data warrant, the Safe Shutdown Earthquake shall be larger than that derived by use of the procedures set forth in Sections IV and V of the Appendix.

The maximum vibratory accelerations of the Safe Shutdown Earthquake at each of the various foundation locations of the nuclear power plant structures at a given site shall be determined taking into account the characteristics of the underlying soil material in transmitting the earthquake-induced motions, obtained pursuant to paragraphs (a)(1), (3), and (4) of section IV. The Safe Shutdown Earthquake shall be defined by response spectra corresponding to the maximum vibratory accelerations as outlined in paragraph (a) of section VI; and

(v) Where the maximum vibratory accelerations of the Safe Shutdown Earthquake at the foundations of the nuclear power plant structures are determined to be less than one-tenth the acceleration of gravity (0.1g) as a result of the steps

required in paragraphs (a)(1)(i) through (iv) of this section, it shall be assumed that the maximum vibratory accelerations of the Safe Shutdown Earthquake at these foundations are at least 0.1g.

(2) Determination of Operating Basis Earthquake. The Operating Basis Earthquake shall be specified by the applicant after considering the seismology and geology of the region surrounding the site. If vibratory ground motion exceeding that of the Operating Basis Earthquake occurs, shutdown of the nuclear power plant will be required. Prior to resuming operations, the licensee will be required to 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. The maximum vibratory ground acceleration of the Operating Basis Earthquake shall be at least one-half the maximum vibratory ground acceleration of the Safe Shutdown Earthquake.

Evaluation

Compliance with 10 CFR 100, Appendix A, Paragraph V(a) is discussed in the FSAR references cited below:

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference(s)	Subject
V (a) (1)	2.5.2.9	The effects at the site of a possible future earthquake similar to a large historical shock have been investigated. The 1937 Intensity VIII Lima, Ohio shock placed at the confluence of the Findlay, Cincinnati and Kankakee Arches would attenuate to less than

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference(s)	Subject
		0.05 g at the site. This same shock if located at the nearest approach to the site of the Findlay Arch axis on the Bowling Green Fault would attenuate to less than 0.10 g. The 1811-1812 Intensity XII New Madrid, Missouri shock if placed at the closest approach of the Kentucky River-Rough Creek fault complex (about 350 miles) would attenuate to less than 0.05 g at the site. Shocks similar to the 1947 South Central Michigan Intensity VI and 1943 Lake Erie Intensity V earthquakes which cannot be definitely associated with a tectonic structure would generate a conservatively estimated shock of less than 0.10 g for an epicenter located near the site.
	2.5.2.10	A conservative evaluation is made of the remote possibility of ground motion considering the seismic history and underlying geological structure at the site. The Safe Shutdown Earthquake (SSE) has been established as a maximum horizontal ground acceleration of 0.15 g at foundation level for Category I structures.

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference(s)	Subject
	1.2.2.3.4	The Seismic Design Basis is that Category I systems will be designed so that the plant can be safely shut down for a maximum horizontal acceleration of 0.15 g and a vertical acceleration of 0.10 g. The vertical and horizontal acceleration will be considered to act simultaneously.
V (a) (2)	2.5.2.11	There will probably be no significant ground motion at the site during the plant's lifetime, however, Category I structures are conservatively designed for an Operating Basis Earthquake (OBE) of 0.08 g, which is more than half the maximum vibratory ground acceleration of the Safe Shutdown Earthquake (SSE).

SECTION

10 CFR 100, Appendix A, Paragaph V(b)

Statement of Section

(b) Determination of Need to Design for Surface Faulting. In order to determine whether a nuclear power plant is required to be designed to withstand the effects of surface faulting, the location of the nuclear power plant with respect to capable faults shall be considered. The area over which each of these faults has caused surface faulting in the past is identified by mapping its fault traces in the vicinity of the site. The fault traces are mapped along the trend of the fault for 10 miles in both diretions from the point of its nearest approach to the nuclear power plant because, for example, traces may be obscured along portions of the fault. The maximum width of the mapped fault traces, called the control width, is then determined from this map. Because surface faulting has sometimes occurred beyond the limit of mapped fault traces or where fault traces have not been previously recognized, the control width of the fault is increased by a factor which is dependent upon the largest potential earthquake related to the fault. This larger width delineates a zone, called the zone requiring detailed faulting investigation, in which the possibility of surface faulting is to be determined. The following paragraphs outline the specific procedures for determining the zone requiring detailed faulting investigation for a capable fault.

(1) Determination of Zone Requiring Detailed Faulting Investigation. The zone requiring detailed faulting investigation for a capable fault which was investigated pursuant to the requirement of paragraph (b)(7) of Section IV shall be determined through use of the following table:

TABLE 2

Determination of Zone Requiring Detailed Faulting Investigation

Magnitude of earthquake:	Width of zone requiring detailed faulting investigation
Less than 5.5	1 x control width
5.5-6.4	2 x control width
6.5-7.5	3 x control width
Greater than 7.5	4 x control width

The largest magnitude earthquake related to the fault shall be used in Table 2. This earthquake shall be determined from the information developed pursuant to the requirements of paragraph (b) of Section IV for the fault, taking into account the information required by paragraph (b)(7) of Section IV. The control width used in Table 2 is determined by mapping the outer limits of the fault traces from information developed pursuant to paragraph (b)(7)(iv) of section IV. The control width shall be used in Table 2 unless the characteristics of the fault are obscured for a significant portion of the 10 miles on either side of the point of nearest approach to the nuclear power plant. In this event, the use in Table 2 of the width of mapped fault traces more than 10 miles from the point of nearest approach to the nuclear power plant may be appropriate.

The zone requiring detailed faulting investigation, as determined from Table 2, shall be used for the fault except where:

(i) The zone requiring detailed faulting investigation from Table 2 is less than one-half mile in width. In this case the zone shall be at least one-half mile in width; or

(ii) Definitive evidence concerning the regional and local characteristics of the fault justifies use of a different value. For example, thrust or bedding-plane faults may require an increase in width of the zone to account for the projected dip of the fault plane; or

(iii) More detailed three-dimensional information, such as that obtained from precise investigative techniques, may justify the use of a narrower zone. Possible examples of such techniques are the use of accurate records from closely spaced drill holes or from closely spaced, high-resolution offshore geophysical surveys.

In delineating the zone requiring detailed faulting investigation for a fault, the center of the zone shall coincide with the center of the fault at the point of nearest approach of the fault to the nuclear power plant.

Evaluation

Compliance with 10 CFR 100, Appendix A, Paragraph V(b) is discussed in the FSAR references cited below:

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference(s)	Subject
V (b)	2.5.3.9	Not applicable; surface faulting is not part of the design basis.
	2.5.2.7	There are no capable faults within 200 miles of the site.
	2.5.3.7	There is no known geologic basis for the possible existence of faulting in the site area.

SECTION

10 CFR 100, Appendix A, Paragraph V(c)

Statement of Section

(c) Determination of Design Bases for Seismically Induced Floods and Water Waves. The size of seismically induced floods and water waves which could affect a site from either locally or distantly generated seismic activity shall be determined, taking into consideration the results of the investigation required by paragraph (c) of section IV. Local topographic characteristics which might tend to modify the possible runup and drawdown at the site shall be considered. Adverse tide conditions shall also be taken into account in determining the effect of the floods and waves on the site. The characteristics of the earthquake to be used in evaluating the offshore effects of local earthquakes shall be determined by a procedure similar to that used to determine the characteristics of the Safe Shutdown Earthquake in paragraph V(a).

Evaluation

Compliance with 10 CFR 100, Appendix A, Paragraph V(c) is discussed in the FSAR references cited below:

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference(s)	Subject
V (c)	1.2.2.3.5	Maximum still water elevation for the plant site was established on the basis of the probable maximum meteorological event.

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference(s)	Subject
	2.4.4	As presented in response to Section IV(c) (1) of 10 CFR 100, Appendix A, there are no dams in southeastern Michigan the failure of which could affect the site.
	2.5.6	There is no design basis for seismically induced flooding or wave runup. There is not enough seismic activity (historically) to generate a tsunami.

SECTION

10 CFR 100, Appendix A, Paragraph V(d)

Statement of Section

(d) Determination of Other Design Conditions.

(1) Soil Stability. Vibratory ground motion associated with the Safe Shutdown Earthquake can cause soil instability due to ground disruption such as fissuring, differential consolidation, liquefaction, and cratering which is not directly related to surface faulting. The following geologic features which could affect the foundations of the proposed nuclear power plant structures shall be evaluated, taking into account the information concerning the physical properties of materials underlying the site developed pursuant to paragraphs (a)(1), (3), and (4) of Section IV and the effects of the Safe Shutdown Earthquake:

(i) Areas of actual or potential surface or subsurface subsidence, uplift, or collapse resulting from:

(a) Natural features such as tectonic depressions and cavernous or karst terrains, particularly those underlain by calcareous or other soluble deposits:

(b) Man's activities such as withdrawal of fluid from or addition of fluid to the subsurface, extraction of minerals, or the loading effects of dams or reservoirs; and

(c) Regional deformation.

(ii) Deformational zones such as shears, joints, fractures, folds, or combinations of these features.

(iii) Zones of alteration of irregular weathering profiles and zones of structural weakness composed of crushed or disturbed materials.

(iv) Unrelieved residual stresses in bedrock.

(v) Rocks or soils that might be unstable because of their mineralogy, lack of consolidation, water content, or potentially undesirable response to seismic or other events. Seismic response characteristics to be considered shall include liquefaction, thixotropy, differential consolidation, cratering, and fissuring.

(2) Slope Stability. Stability of all slopes, both natural and artificial, the failure of which could adversely affect the nuclear power plant, shall be considered. An assessment shall be made of the potential effects of erosion or deposition and of combinations of erosion or deposition with seismic activity, taking into account information concerning the physical property of the materials underlying the site developed pursuant to paragraph (a)(1), (3), and (4) of Section IV and the effects of the Safe Shutdown Earthquake.

(3) Cooling Water Supply. Assurance of adequate cooling water supply for emergency and longterm shutdown decay heat removal shall be considered in the design of the nuclear power plant, taking into account information concerning the physical properties of the materials underlying the site developed pursuant to paragraphs (a)(1), (3), and (4) of section IV and the effects of the Safe Shutdown Earthquake and the design basis for surface faulting. Consideration of river blockage or diversion or other failures which may block the flow of cooling water, coastal uplift or subsidence, or tsunami runup and drawdown, and failure of dams and intake structures shall be included in the evaluation, where appropriate.

(4) Distant Structures. Those structures which are not located in the immediate vicinity of the site but which are safety related shall be designed to withstand the effect of the Safe Shutdown Earthquake and the design basis for surface faulting determined on a comparable basis to that of the nuclear power plant, taking into account the material underlying the structures and the different location with respect to that of the site.

Evaluation

Compliance with 10 CFR 100, Appendix A, Paragraph V(d) is discussed in the FSAR references cited below:

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference (s)	Subject
V (d) (1)	2.5.4.1	The site is underlain by competent bedrock strata. There are no zones of solution weathering in the site area. There are some zones of extensively fractured or highly vugged rocks. The foundation rock under the structures was pressure grouted to assure that these zones are not horizontally continuous across the site.
	2.5.4.2	Subsurface properties are presented for use in design and analysis.
	2.5.1.1, 2.5.1.2	Site and regional geologic conditions are described and discussed.

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference (s)	Subject
	2.5.4.5	Excavations in rock for plant foundations and backfill of plant structures and site area to final grade with rock quarried on site are described. Blasting criteria are given. All Category I buildings are supported on bedrock; no Category I buildings are placed on soil overburden or crushed rock fill.
	2.4.5.6	Effects of dewatering for plant excavations and quarry operations on the groundwater were monitored. After spring of 1971, the quarry operation was restricted to the south portion. The north portion was diked off and the water allowed to rise to its normal level. This functioned as a groundwater recharge pit. Quarry operation stopped in June, 1972.
	2.5.4.8	All Category I structures are supported within the Bass Islands Group on dolomite rock which is not subject to liquefaction. Note: The crushed rock backfill also is not subject to liquefaction.

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference (s)	Subject
V (d) (2)	2.5.5	There are no slopes whose failure could adversely affect the safe operation of the plant. During excavation, including blasting, no instances of instability of excavation slopes occurred.
V (d) (3)	1.2.2.3.6	Natural draft cooling towers are the normal heat sink. If these are lost, mechanical draft cooling towers in the RHR complex can be used as a heat sink to shut down the reactor and maintain it in the shutdown condition.
	1.2.2.9.21 9.2.2	The Emergency Equipment Cooling Water System (EECWS) cools safe shutdown systems. The EECWS is designed as Category I and has an emergency backup connection to the RHR complex.
V (d) (4)	Table 3.2-1	All Category I structures, systems and equipment are located onsite.

SECTION

10 CFR 100, Appendix A, Paragraph VI(a)

Statement of Section

(a) Vibratory Ground Motion.

(1) Safe Shutdown Earthquake. The vibratory ground motion produced by the Safe Shutsdown Earthquake shall be defined by response spectra corresponding to the maximum vibratory accelerations at the elevations of the foundations of the nuclear power plant structures determined pursuant to paragraph (a)(1) of Section V. The response spectra shall relate the response of the foundations of the nuclear power plant structures to the vibratory ground motion, considering such foundations to be single-degree-of-freedom damped oscillators and neglecting soil-structure interaction effects. In view of the limited data available on vibratory ground motions of strong earthquakes, it usually will be appropriate that the response spectra be smoothed design spectra developed from a series of response spectra related to the vibratory motions caused by more than one earthquake.

The nuclear power plant shall be designed so that, if the Safe Shutdown Earthquake occurs, certain structures, systems, and components will remain functional. These structures, systems, and components are those 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 condition, or (iii) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of this part. In addition to seismic loads, including aftershocks, applicable concurrent functional and accident-induced loads shall be taken into account in the design of these safety related

structures, systems, and components. The design of the nuclear power plant shall also take into account the possible effects of the Safe Shutdown Earthquake on the facility foundations by ground disruption, such as fissuring, differential consolidation, cratering, liquefaction, and landsliding, as required in paragraph (d) of section V.

The engineering method used to ensure that the required safety functions are maintained during and after the vibratory ground motion associated with the Safe Shutdown Earthquake shall involve the use of either a suitable qualification test to demonstrate that structures, systems and components can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate conservatism.

The analysis or test shall 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 and under the postulated concurrent conditions, provided that the necessary safety functions are maintained.

(2) Operating Basis Earthquake. The Operating Basis Earthquake shall be defined by response spectra. 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 shall be designed to remain functional and within applicable stress and deformation limits when subjected to the effects of the vibratory motion of the Operating Basis Earthquake in combination with normal operating loads. The engineering method used to ensure that these structures, systems, and components are capable of withstanding the effects of the Operating Basis Earthquake shall involve the use of either a suitable dynamic analysis or a suitable quali-

fication test to demonstrate that the structures, systems and components can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate conservatism. The analysis or test shall take into account soil-structure interaction effects and the expected duration of vibratory motion.

(3) Required Seismic Instrumentation. Suitable instrumentation shall be provided so that the seismic response of nuclear power plant features important to safety can be determined promptly to permit comparison of such response with that used as the design basis. Such a comparison is needed to decide whether the plant can continue to be operated safely and to permit such timely action as may be appropriate.

These criteria do not address the need for instrumentation that would automatically shut down a nuclear power plant when an earthquake occurs which exceeds a predetermined intensity. The need for such instrumentation is under consideration.

Evaluation

Compliance with 10 CFR 100, Appendix A, Paragraph VI(a) is discussed in the FSAR references cited below:

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference(s)	Subject
VI (a) (1)	Figure 2.5-71 & 3.7-3	Design response spectra for Safe Shutdown Earthquake at foundation level are given for horizontal acceleration =0.15g. Vertical ground motion is taken to be 2/3 of the maximum horizontal ground acceleration.

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference(s)	Subject
	2.5.4.7 & 3.7.1.1	The site response spectra for the operating basis earthquake (OBE) and safe shutdown earthquake (SSE) for the horizontal direction are shown in Figures 3.7-2 and 3.7-3, respectively. Vertical ground motions are taken to be 2/3 of the corresponding maximum horizontal ground acceleration.
	3.7.1.6	Soil structure interaction effects are insignificant because the foundations are founded on rock whose shear wave velocity is 7600 fps. The Fermi 2 foundations and underlying material behave as a rigid foundation.
	3.7.2	The dynamic response of the plant complex including buried electrical ducts and piping due to earthquake loading is discussed. Structure and equipment modeling is described and the mathematical models used to compute responses are presented.
	3.8	The design of Category I structures is described. Various load combinations are given in Tables 3.8-18 and 3.8-19. These loads include the SSE induced loadings.

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference(s)	Subject
	3.7.5	Category I systems and components are designed to function during and after the specified earthquakes. The seismic performance specifications and acceptance criteria are described.
	3.2	The seismic classification for structures, components and system are delineated.
VI (a) (2)	Figure 2.5-72 & 3.7-2	Design response spectra for Operating Basis Earthquake at foundation level are given for horizontal acceleration =0.08g. Vertical ground motion is taken to be 2/3 of the maximum horizontal ground acceleration.
	2.5.4.7 & 3.7.1.2	Design response spectra derivation.
	3.7	The dynamic response of the plant and associated systems to the OBE earthquake loads are presented.
	3.8	The design of Category I structures is described. Various load combinations are given in Tables 3.8-18 and 3.8-19. These loads include OBE induced loadings.

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference(s)	Subject
VI (a) (3)	3.7.4.1	An instrumentation program conforming to Regulatory Guide 1.12 to monitor and record input motion and behavior from an earthquake has been provided.
	3.7.4.2	Active earthquake recording instrumentation has been provided to measure the basic ground motion time history acceleration and seismic motion of the primary containment elements. Triaxial accelerometers responding to accelerations in three orthogonal directions have been placed in the reactor/auxiliary building sub-basement, at the bottom of the RPV pedestal and near the top of the sacrificial shield. This system is activated by a seismic trigger preset to activate at 0.01g. This trigger activates recording equipment and notifies the control room operator.
		Passive instrumentation consisting of triaxial response spectrum recorders responding to accelerations in three orthogonal axes have been installed. These units have 12 sensing elements with nominal frequencies of 2 to 25 Hz. The units have been installed at three locations in the reactor/auxiliary building and three locations in the RHR complex. One unit in the

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference(s)	Subject
		reactor/auxiliary building is located adjacent to the active device at the sub-basement level. The spectrum from this device will be utilized for direct comparison with the spectrum generated from the active device.
	3.7.4.3	This section describes the data reduction and data examination. The control room operator is notified that the seismic trigger has been activated. If the active device at the sub-basement of the reactor/auxiliary building shows an acceleration of 0.05g or greater, the plant is shut down. An operator also examines the response spectrum from the passive devices and if the response spectrum exceed the OBE response, the plant is shut down.
	3.7.4.4	The data from all recording systems is obtained and reduced. The measured responses are compiled into an earthquake data report and are then compared with the response spectra for the structures, systems and components.

SECTION

10 CFR 100, Appendix A, Paragraph VI(b)

Statement of Section

(b) Surface Faulting.

(1) If the nuclear power plant is to be located within the zone requiring detailed faulting investigation, a detailed investigation of the regional and local geologic and seismic characteristics of the site shall be carried out to determine the need to take into account surface faulting in the design of the nuclear power plant. Where it is determined that surface faulting need not be taken into account, sufficient data to clearly justify the determination shall be presented in the license application.

(2) Where it is determined that surface faulting must be taken into account, the applicant shall, in establishing the design basis for surface faulting on a site take into account evidence concerning the regional and local geologic and seismic characteristics of the site and from any other relevant data.

(3) The design basis for surface faulting shall be taken into account in the design of the nuclear power plant by providing reasonable assurance that in the event of such displacement during faulting certain structures, systems, and components will remain functional. These structures, systems, and components are those 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 the guideline exposures of this part. In addition to seismic loads, including aftershocks, appli-

cable concurrent functional and accident-induced loads shall be taken into account in the design of such safety features. The design provisions shall be based on an assumption that the design basis for surface faulting can occur in any direction and azimuth and under any part of the nuclear power plant, unless evidence indicates this assumption is not appropriate, and shall take into account the estimated rate at which the surface faulting may occur.

Evaluation

Compliance with 10 CFR 100, Appendix A, Paragraph VI(b) is discussed in the FSAR references cited below.

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference(s)	Subject
VI (b)	2.5.3.9	Not applicable; surface faulting is not part of the design basis.
	2.5.2.7	There are no capable faults within 200 miles of the site.

SECTION

10 CFR 100, Appendix A, Paragraph VI(c)

Statement of Section

(c) Seismically Induced Floods and Water Waves and Other Design Conditions. The design basis for seismically induced floods and water waves from either locally or distantly generated seismic activity and other design conditions determined pursuant to paragraphs (c) and (d) of Section V, shall 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.

Evaluation

Compliance with 10 CFR 100, Appendix A, Paragraph VI(c) is discussed in the FSAR reference cited below:

10 CFR 100 Appendix A Subsection	EF2 FSAR Reference(s)	Subject
VI (c)	2.4.4	As presented in response to Section IV(c) (1) of 10 CFR 100, Appendix A, there are no dams in southeastern Michigan the failure of which could affect the site.
	2.4.6	Not applicable; there is no design basis for seismically induced flooding or wave runup. There is not enough seismic activity (historically) to generate a tsunami.