August 8, 1996

Mr. Douglas R. Gipson
Senior Vice President
Nuclear Generation
Detroit Edison Company
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: FERMI-2 - ISSUANCE OF AMENDMENT RE: IMPLEMENTATION OF 10 CFR PART 50 APPENDIX J OPTION B (TAC NO. M94366)

Dear Mr. Gipson:

The Commission has issued the enclosed Amendment No. 108 to Facility Operating License No. NPF-43 for the Fermi-2 facility. The amendment consists of changes to the Technical Specifications (TS) in response to your letter dated December 21, 1995 (NRC-95-0133).

The amendment revises the TS and Bases to implement 10 CFR Part 50, Appendix J, Option B, by referring to Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program." Specifically, changes have been made to TS 3/4.6.1.2, "Primary Containment Leakage," TS 3/4.6.1.3, "Primary Containment Air Locks," TS 3/4.6.1.5, "Primary Containment Structural Integrity," TS 6.0, "Administrative Controls," and their associated Bases.

A copy of our Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by:
Andrew J. Kugler, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-341

Enclosures: 1. Amendment No. 108 to NPF-43
2. Safety Evaluation

cc w/encl: See next page

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DATED: August 8, 1996

AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. NPF-43-FERMI-2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DETROIT EDISON COMPANY

DOCKET NO. 50-341

FERMI-2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 108
License No. NPF-43

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Detroit Edison Company (the licensee) dated December 21, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-43 is hereby amended to read as follows:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 108 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. DECo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance with full implementation within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Kevin A. Connaughton

Andrew J. Kugler, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 8, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 108

FACILITY OPERATING LICENSE NO. NPF-43

DOCKET NO. 50-341

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
3/4 0-3*	3/4 0-3*
3/4 0-4	3/4 0-4
xiv	xiv
xx	xx
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
3/4 6-5	3/4 6-5
3/4 6-6	3/4 6-6
3/4 6-7	3/4 6-7
3/4 6-8*	3/4 6-8*
3/4 6-9	3/4 6-9
3/4 6-10*	3/4 6-10*
3/4 6-11	3/4 6-11
3/4 6-12*	3/4 6-12*
B 3/4 6-1	B 3/4 6-1
B 3/4 6-1a	B 3/4 6-1a
B 3/4 6-2	B 3/4 6-2
- - -	B 3/4 6-2a
B 3/4 6-3	B 3/4 6-3
6-16b	6-16b
- - -	6-16c

*Overleaf page provided to maintain document completeness. No changes contained on these pages.

BASES

ADMINISTRATIVE CONTROLS

TABLE 4.0.2-1

SURVEILLANCE TEST INTERVALS EXTENDED TO OCTOBER 5, 1996

SURVEILLANCE REQUIREMENT	DESCRIPTION
4.1.3.1.4.a	Scram discharge vol. vent and drain valve operability
4.3.1.1, Table 4.3.1.1-1, Item 3	RPS Rx Steam Dome Press High cal.
4.3.1.1, Table 4.3.1.1-1, Item 4	RPS Rx Low Water Level - Level 3 cal
4.3.1.1, Table 4.3.1.1-1, Item 5	RPS MSIV Closure cal
4.3.1.1, Table 4.3.1.1-1, Item 6	RPS Main Steam Line Radiation High cal
4.3.1.1, Table 4.3.1.1-1, Item 7	RPS Drywell Pressure High cal
4.3.1.3[a]	RPS Response Time Test
4.3.2.1, Table 4.3.2.1-1, Item 1.a.1	Pri Cont Isolation Actuation Rx Water Low Level - Level 3 cal
4.3.2.1, Table 4.3.2.1-1, Item 1.a.2	Pri Cont Isolation Actuation Rx Water Low Level - Level 2 cal
4.3.2.1, Table 4.3.2.1-1, Item 1.a.3	Pri Cont Isolation Actuation Rx Water Low Level - Level 1 cal
4.3.2.1, Table 4.3.2.1-1, Item 1.b	Pri Cont Isolation Actuation Drywell Press High cal
4.3.2.1, Table 4.3.2.1-1, Item 1.c.1	Pri Cont Isolation Actuation Main Steam Line Radiation High cal
4.3.2.1, Table 4.3.2.1-1, Item 1.c.2	Pri Cont Isolation Actuation Main Steam Line Press Low cal
4.3.2.1, Table 4.3.2.1-1, Item 1.d	Pri Cont Isolation Actuation Main Steam Line Tunnel Temp. High cal
4.3.2.1, Table 4.3.2.1-1, Item 1.e	Pri Cont Isolation Actuation Condenser Press High cal
4.3.2.1, Table 4.3.2.1-1, Item 1.f	Pri Cont Isolation Actuation Turbine Bldg. Area Temp. High cal
4.3.2.1, Table 4.3.2.1-1, Item 2.e	RWCU Isolation Rx Water Low Level - Level 2 channel cal
4.3.2.1, Table 4.3.2.1-1, Item 2.g	RWCU Manual Initiation channel functional test
4.3.2.1, Table 4.3.2.1-1, Item 3.a.1	RCIC Steam Line Flow High DP channel cal
4.3.2.1, Table 4.3.2.1-1, Item 3.a.2	RCIC Steam Line Flow High Time Delay cal
4.3.2.1, Table 4.3.2.1-1, Item 4.a.1	HPCI Steam Line Flow High DP cal
4.3.2.1, Table 4.3.2.1-1, Item 4.a.2	HPCI Steam Line Flow High Time Delay cal
4.3.2.1, Table 4.3.2.1-1, Item 4.e	HPCI Manual Initiation functional test
4.3.2.1, Table 4.3.2.1-1, Item 5.a	RHR S/D Cooling Rx Water Level Low - Level 3 cal
4.3.2.1, Table 4.3.2.1-1, Item 6.b	Sec. Cont. Isolation - Drywell Press High channel cal
4.3.2.3[a]	Isolation Actuation Inst. System Response Time
4.3.3.1, Table 4.3.3.1-1, Item 1.b	CS Drywell Press High Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.b	LPCI Drywell Press High Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.f	LPCI Riser Differential Pressure High Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.g	LPCI Recirc. Pump Differential Pressure High Cal
4.3.3.1, Table 4.3.3.1-1, Item 3.b	HPCI Drywell Press High Cal
4.3.3.1, Table 4.3.3.1-1, Item 3.f	HPCI Manual Initiation
4.3.3.1, Table 4.3.3.1-1, Item 4.f	ADS RPV Low Level 3 Cal
4.3.3.1, Table 4.3.3.1-1, Item 4.i	ADS Manual Inhibit Functional Test
4.3.4, Table 4.3.4-1, Item 2	RPV Press High Cal (ATWS)
4.3.7.4.1, Table 4.3.7.4.-1, Item 1	RPV Press Cal - Remote Shutdown
4.3.7.5, Table 4.3.7.5-1, Item 1	RPV Press Cal Accident Mon.
4.3.7.5, Table 4.3.7.5-1, Item 11	SRV Position Indic Cal Accident Mon.
4.3.7.5, Table 4.3.7.5-1, Item 12	CTMT High Range Rad Monitoring Cal Accident Mon.
4.3.7.5, Table 4.3.7.5-1, Item 2.a	RPV Fuel Zone Level Cal Accident Mon
4.3.7.10.c	Loose Part Detection System Cal
4.3.9.1, Table 4.3.9.1-1, Item a	RPV High Water Level 8 Cal FW/Main Turbine Trip
4.3.9.2	FW/Main Turbine Trip LSFT
4.3.11.1, Table 4.3.11.1-1, Item 7	Alt S/D system Rx Water Level instrument operability
4.3.11.1, Table 4.3.11.1-1, Item 8	Alt S/D system Rx Press instrument operability
4.4.2.1.1	SRV Tail Pipe Pressure Switch Cal
4.4.2.1.2	SRV lift set point test
4.4.2.2.b	SRV Low Low Set Pressure setpoint Cal and LSFT
4.4.3.1.b	Drywell Sump Flow/Lvl Monitoring Cal
4.4.3.2.2.a	RCS Pressure Isol Valve Leak Test
4.5.1.d.2.a	ADS System Functional Test
4.6.1.4.d.3	MSIV LCS Press Inst. Cal and DP Calibration

TABLE 4.0.2-1

SURVEILLANCE TEST INTERVALS EXTENDED TO OCTOBER 5, 1996 Cont'd

SURVEILLANCE REQUIREMENT	DESCRIPTION
4.6.2.1.e	Suppression Chamber operability (visual inspection)
4.6.2.1.h	Suppression Chamber operability DW to torus bypass leak test
4.6.3.4	Instr. Excess Flow Check operability
4.6.3.5.b	TIP Explosive Squib operability test
4.6.4.1.b.2.a	Torus/Drywell vacuum breaker setpoint operability
4.6.4.1.b.2.b	Torus/Drywell vacuum breaker position indication cal
4.6.4.1.b.2.c	Torus/Drywell vacuum breaker switch opening gap
4.6.4.2.b.2.a	RB/Torus Vacuum Breaker operability (setpoint)
4.6.4.2.b.2.b	RB/Torus Vacuum Breaker operability (visual)
4.6.4.2.b.2.c	RB/Torus Vacuum Breaker position indication operability
4.7.11.4	Alternative Shutdown Control Circuit Functional Test
4.8.4.2.a.1.a	Primary Containment 4160 Volt Penetration Protective Relay Cal
4.8.4.2.a.1.b	Primary Containment 4160 Volt Penetration Protective Device Integrated Functional Test

TABLE NOTATIONS

(a) The surveillance interval of channels within the same trip system required to be tested at least once every N times 18 months, where N is the total number of channels in the trip system, may be based upon the performance of the surveillance during the fifth refueling outage.

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to: L_a, 0.5 percent by weight of the containment air per 24 hours at P_a, 56.5 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for primary containment penetrations and primary containment isolation valves subject to Type B and C tests when pressurized to P_a in accordance with the Primary Containment Leakage Rate Testing Program described in Specification 6.8.5.g, except for main steam line isolation valves* and primary containment isolation valves which are hydrostatically tested.

c. *Less than or equal to 100 scf per hour for all four main steam lines when tested at 25.0 psig.

d. A combined leakage rate of less than or equal to 5 gpm for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 P_a, 62.2 psig.

e. Less than or equal to 1 gpm times the number of valves per penetration not to exceed 3 gpm per penetration for any line penetrating containment and hydrostatically tested at 1.10 P_a, 62.2 psig.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:
With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a, or
b. The measured combined leakage rate for primary containment penetrations and primary containment isolation valves subject to Type B and C tests in accordance with the Primary Containment Leakage Rate Testing Program, except for main steam line isolation valves* and primary containment isolation valves which are hydrostatically tested, exceeding 0.60 L_a, or
c. The measured leakage rate exceeding 100 scf per hour for all four main steam lines, or
d. The measured combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 5 gpm, or
e. The leakage rate of any hydrostatically tested line penetrating primary containment exceeding 1 gpm per isolation valve times the number of containment isolation valves per penetration or greater than 3 gpm per penetration,

prior to increasing reactor coolant system temperature above 200°F, restore:

a. The overall integrated leakage rate(s) to less than or equal to 0.75 L_a, and

*Exemption to Appendix J of 10 CFR Part 50.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. The combined leakage rate for primary containment penetrations and primary containment isolation valves subject to Type B and C tests in accordance with the Primary Containment Leakage Rate Testing Program, except for main steam line isolation valves* and primary containment isolation valves which are hydrostatically tested, tests to less than or equal to 0.60 L_a, and

c. The leakage rate to less than or equal to 100 scf per hour for all four main steam lines, and

d. The combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 5 gpm, and

e. The leakage rate of any hydrostatically tested line penetrating primary containment to less than 1 gpm per isolation valve times the number of containment isolation valves per penetration or less than 3 gpm per penetration.

SURVEILLANCE REQUIREMENTS

4.6.1.2 Perform required primary containment leakage rate testing in accordance with the Primary Containment Leakage Rate Program described in Specification 6.8.5.g.**

*Exemption to Appendix J of 10 CFR Part 50

**Except for LPCI Loop A and B Injection Isolation valves, which are hydrostatically tested in accordance with Specification 4.4.3.2.2 in lieu of this requirement.

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SURVEILLANCE REQUIREMENTS

4.6.1.3 Each primary containment air lock shall be demonstrated OPERABLE:

a. Within 7 days following each closing, except when the air lock is being used for multiple entries, then at least once per 30 days, by verifying seal leakage rate less than or equal to 5 scf per hour when the gap between the door seals is pressurized to P_a, 56.5 psig.

b. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when the air lock has been opened during periods when containment integrity was not required. The demonstration shall verify a seal leakage rate less than or equal to 5 scf per hour when the gap between the door seals is pressurized to P_a, 56.5 psig, unless the air lock is tested pursuant to Specification 4.6.1.3.c.2.

c. By conducting an overall air lock leakage test at P_a, 56.5 psig, and by verifying that the overall air lock leakage rate is within its limit:

   1. Prior to initial fuel loading and at 30 months* intervals thereafter,

   2. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when the air lock has been opened during periods when containment integrity was not required, if maintenance which could affect the leak tight integrity of the doors has been performed since the last successful test pursuant to Specification 4.6.1.3.c.1.

d. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.**

*The provisions of Specification 4.0.2 are not applicable.

**Except that the inner door need not be opened to verify interlock OPERABILITY when the primary containment is inerted, provided that the inner door interlock is tested within 8 hours after the primary containment has been deinerted.

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of those surfaces. This inspection shall be performed prior to the Type A containment leakage rate test and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.5.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days. This report shall include a description of the condition of the structure, the inspection procedure, the inspection criteria, and the corrective actions taken.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

PRIMARY CONTAINMENT INTEGRITY is demonstrated by leak rate testing and by verifying that all primary containment penetrations not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by locked valves, blank flanges or deactivated automatic valves secured in the closed position. For test, vent and drain connections which are part of the containment boundary, a threaded pipe cap with acceptable sealant in addition to the containment isolation valve(s) provides protection equivalent to a blank flange.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure of 56.5 psig, P_a. Updated analysis demonstrates maximum expected pressure is less than 56.5 psig. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The maximum allowable leakage rate for the primary containment (L_a) is 0.5 percent by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a) of 56.5 psig.

A Primary Containment Leakage Rate Testing Program has been established to implement the requirements of 10 CFR Part 50 Appendix J, Option B. The Primary Containment Leakage Rate Testing Program conforms with NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Rate Testing Program", Revision 0, dated September 1995, and Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J", Revision 0, dated July 26, 1995. NRC Regulatory Guide 1.163, Revision 0 endorses NEI 94-01 which in turn identifies ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements" as an acceptable standard regarding leakage-rate test methods, procedures, and analyses.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE (Continued)

The measured leakage rate acceptance criteria of $\leq 0.60\ L_a$ for the combined Type B and C tests and as-left acceptance criterion of $\leq 0.75\ L_a$ for the Type A test ensures a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses. Primary containment operability is maintained by limiting leakage to $\leq 1.0\ L_a$.

Individual leakage rates specified for the primary containment air lock are addressed in Specification 3.6.1.3.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50 Option B with the exception of exemptions granted for main steam isolation valve leak testing and testing the Low Pressure Coolant Injection Inboard Isolation Valves. The program as defined in Specification 6.8.5.g eliminates the need for the previous exemptions granted concerning analyzing the Type A test data and testing airlocks after each opening.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.6.1.1 and

PRIMARY CONTAINMENT AIR LOCKS (Continued)

3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment. In the event of an inoperable door interlock, locking shut the inner door will ensure containment integrity while permitting access to the lock for maintenance and surveillance testing.

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J as established in the Primary Containment Leakage Rate Testing Program, which has been established to implement 10 CFR Part 50, Appendix J, Option B. The Primary Containment Leakage Rate Testing Program conforms with NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Rate Testing Program" Revision 0, dated September 1995 and Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J", Revision 0, dated July 26, 1995.

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR Part 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIVs such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIVs when isolation of the primary system and containment is required.

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 56.5 psig in the event of a LOCA. A visual inspection in conjunction with the Primary Containment Leakage Testing Program is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the containment peak pressure of less than 56.5 psig does not exceed the maximum allowable pressure of 62 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 2 psid.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The drywell and suppression chamber purge supply and exhaust isolation valves are maintained closed during a majority of the plant operating time. Maintaining these valves closed (even though they have been qualified to close against the buildup of pressure in primary containment in the event of DBA/LOCA) reduces the potential for release of excessive quantities of radioactive material.

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM (Continued)

Purging or venting through the Standby Gas Treatment System (SGTS) imposes a vulnerability factor on the integrity of the SGTS. Should a LOCA occur while the purge pathway is through the SGTS the associated pressure surge, before the purge valves close, may adversely affect the integrity of the SGTS charcoal filters. Therefore, PURGING or VENTING through the SGTS is limited to 90 hours per 365 days. This time limit is not imposed when venting through the SGTS with the 1-inch valves or when PURGING or VENTING through the Reactor Building Ventilation System with any of the purge valves.

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

The 6, 10, 20, and 24 inch purge valves are generally configured in a three (3) valve arrangement at each of the associated purge penetrations. The valves are leak tested by pressurizing between the three valves and a total leakage is determined as opposed to a single valve leakage. Verifying that the measured leakage rate is less than 0.05 L_a for this multi-valve arrangement is more conservative than a limit of 0.05 L_a for a single valve.

3/4.6.2 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the maximum allowable pressure of 62 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1045 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss-of-coolant accident, the pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

PROCEDURES AND PROGRAMS (Continued)

f. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and,

3) Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

g. Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(O) and 10 CFR 50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 56.5 psig.

The maximum allowable primary containment leakage rate, L_a, shall be 0.5% of primary containment air weight per day at P_a.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

6.9.1.2 The startup report shall address each of the tests identified in Subsection 14.1.4.8 of the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. NPF-43

DETROIT EDISON COMPANY

FERMI-2

DOCKET NO. 50-341

1.0 INTRODUCTION

On September 12, 1995, the U.S. Nuclear Regulatory Commission (NRC) approved issuance of a revision to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The NRC added Option B "Performance-Based Requirements" to allow licensees to voluntarily replace the prescriptive testing requirements of 10 CFR Part 50, Appendix J, with testing requirements based on both overall leakage rate performance and the performance of individual components.

By application dated December 21, 1995, Detroit Edison Company, (the licensee) requested changes to the Operating License and Technical Specifications (TS) for the Fermi 2 plant. The proposed changes would permit implementation of 10 CFR Part 50, Appendix J, Option B. The licensee has established a "Containment Leakage Rate Testing Program" and proposed adding this program to the TS. The program references Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, which specifies a method acceptable to the NRC for complying with Option B.

2.0 BACKGROUND

Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TS and Bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. Appendix J of 10 CFR Part 50 was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which became effective on October 26, 1995. The revision added Option B, "Performance-Based Requirements," to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide 1.163 was developed as a method acceptable to the NRC staff for implementing Option B. This RG states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that the RG or other implementation document used by a licensee to develop a performance-based leakage rate testing program must be included, by general reference, in the plant TS. The licensee has referenced RG 1.163 in the Fermi 2 TS.

Regulatory Guide 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TS to implement Option B. After some discussion, the staff and NEI agreed on final TS which were attached to a letter from C. Grimes (NRC) to D. Modeen (NEI) dated November 2, 1995. These TS are to serve as a model for licensees to develop plant-specific TS in preparing amendment requests to implement Option B.

For a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B, and C tests have been met. In addition, the licensee must maintain performance comparisons of the overall containment system and individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 <u>EVALUATION</u>

In its December 21, 1995, letter, the licensee proposed establishing a "Primary Containment Leakage Rate Testing Program" and proposed adding this program to the TS. The program references RG 1.163, which specifies a method acceptable to the NRC for complying with Option B. The proposal requires a change to existing TS Section 3/4.6.1.2, "Primary Containment Leakage," TS 3/4.6.1.3, "Primary Containment Air Locks," TS 3/4.6.1.5, "Primary Containment Structural Integrity," and the addition of the "Primary Containment Leakage Rate Testing Program" to TS 6.0, "Administrative Controls." Corresponding bases would also be modified.

Option B permits a licensee to choose Type A, or Type B and C, or Type A, B, and C testing to be done on a performance basis. The licensee has elected to perform Type A, B, and C testing on a performance basis.

The TS changes proposed by the licensee are in compliance with the requirements of Option B and consistent with the guidance of RG 1.163, and the generic TS of the November 2, 1995, letter and are, therefore, acceptable to the staff.

Paragraph V.B.1 of 10 CFR Part 50, Appendix J, Option B, states that specific exemptions to Option A are still applicable to Option B, if necessary, unless specifically revoked by the NRC.

The licensee evaluated the existing exemptions from Option A against the new requirements of Option B and determined that two of the exemptions will be retained. A previously approved exemption to the original Appendix J requirements concerning reduced pressure for main steam isolation valve testing has been retained in Section 3.6.1.2.c. In addition, an approved exemption to test the LPCI Loop A and B injection isolation valves in accordance with TS Section 4.4.3.2.2 in lieu of the Type B and C Appendix J local leak-rate test requirements has been maintained in the proposed changes.

References to four previously granted exemptions have been deleted from the proposed Fermi 2 TS because these exemptions are not required under the new 10 CFR Part 50 Appendix J, Option B, regulations. These exemptions include two one-time schedule exemptions, the exemption for Type A data analysis methods, and the exemption for testing of airlocks after each opening. The latter two are no longer needed due to the added flexibility afforded by RG 1.163 and the NEI 94-01 methodology.

The staff has reviewed the licensee's proposed disposition of its existing (Option A) Appendix J exemptions as they relate to the Option B requirements, and pursuant to the provisions of 10 CFR Part 50, Appendix J, Option B, paragraph V.B.1, finds it acceptable.

Finally, a modification was proposed for TS Table 4.0.2-1. Due to the extended plant outage resulting from the 1993 turbine generator failure, the licensee rescheduled the fifth refueling outage from spring 1996 to fall 1996. In order to support the revised plant outage schedule, the licensee requested a one-time change to extend a number of surveillance test intervals. This was proposed in order to avoid shutting down the plant in mid-cycle to perform surveillances. License Amendment No. 106, issued on March 1, 1996, granted this request. TS Table 4.0.2-1, "Surveillance Test Intervals Extended To October 5, 1996," which was introduced in Amendment No. 106, identified each of the surveillances whose test intervals were extended. However, due to the implementation of Option B to 10 CFR Part 50 Appendix J, several surveillances included in TS Table 4.0.2-1 are being eliminated. Since surveillance requirements 4.6.1.2.b, 4.6.1.2.d and 4.6.1.2.g are being eliminated as a part of this license amendment, the licensee has proposed eliminating these items from the table. Because these surveillances are no longer included in the TS, the staff finds their removal from TS Table 4.0.2-1 acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan state official was notified of the proposed issuance of the amendment. The state official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (61 FR 7551). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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