Docket No. 50-341

JUL 17 1987

Mr. B. Ralph Sylvia
Group Vice President
Nuclear Operations
Detroit Edison Company
6400 North Dixie Highway
Newport, Michigan 48166

Dear Mr. Sylvia:

SUBJECT: AMENDMENT NO. 8 TO FACILITY OPERATING LICENSE NO. NPF-43 (TAC NO. 64835)

The Commission has issued the enclosed Amendment No. 8 to Facility Operating License No. NPF-43 for the Fermi-2 facility. This amendment consists of changes to the Plant Technical Specifications in response to your letter (VP-NO-87-0020) dated March 9, 1987.

The amendment revises the Plant Technical Specifications to make editorial and typographical error corrections to Specification 3/4.8.4.3 (MOV Thermal Overload Protection), Specification 3/4.5.1 (ECCS-Operating) and Specification 3/4.6.1.2 (the Bases for Primary Containment Leakage).

A copy of the Safety Evaluation supporting this amendment is enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register Notice.

Sincerely,

Original signed by

John J. Stefano, Project Manager
Project Directorate III-1
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. 8 to License No. NPF-43
2. Safety Evaluation

cc w/enclosures:
See next page

LA:PD31	PM:PD31	D:PD31	OGC-BETH
RIngram	JStefano:lt	MVirgilio	
6/26/87	6/30/87	7/1/87	7/6/87

8707310252 870717
PDR ADOCK 05000341
P PDR

Mr. B. Ralph Sylvia
Detroit Edison Company

Fermi-2 Facility

cc:
Mr. Harry H. Voigt, Esq.
LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N.W.
Washington, D.C. 20036

John Flynn, Esq.
Senior Attorney
Detroit Edison Company
2000 Second Avenue
Detroit, Michigan 48226

Nuclear Facilities and Environmental
Monitoring Section Office
Division of Radiological Health
P. O. Box 30035
Lansing, Michigan 48909

Mr. Steve Frost
Supervisor-Licensing
Detroit Edison Company
Fermi Unit 2
6400 North Dixie Highway
Newport, Michigan 48166

Mr. Thomas Randazzo
Director, Regulatory Affairs
Detroit Edison Company
Fermi Unit 2
6400 North Dixie Highway
Newport, Michigan 48166

Mr. Walt Rogers
U.S. Nuclear Regulatory Commission
Resident Inspector's Office
6450 W. Dixie Highway
Newport, Michigan 48166

Monroe County Office of Civil
Preparedness
963 South Raisinville
Monroe, Michigan 48161

Ronald C. Callen
Adv. Planning Review Section
Michigan Public Service Commission
6545 Mercantile Way
P. O. Box 30221
Lansing, Michigan 48909

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Distribution/Concurrence for Letter to Sylvia Dated - 7/17/87

DISTRIBUTION:

Docket File	JStefano	Wanda Jones
NRC PDR	OGC-Beth	EButcher
Local PDR	DHagan	ACRS(10)
PD31 Gray File	EJordan	GPA/PA
GHolahan	JPartlow	ARM/LFMB
RIngram	TBarnhart(4)	

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

DETROIT EDISON COMPANY

WOLVERINE POWER SUPPLY COOPERATIVE, INCORPORATED

DOCKET NO. 50-341

FERMI-2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 8
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 March 9, 1987, 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 the 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. 8, 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.

8707310258 870717
PDR ADOCK 05000341
P PDR

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Martin J. Virgilio

Martin J. Virgilio, Acting Director
Project Directorate III-1
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: July 17, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 8

FACILITY OPERATING LICENSE NO. NPF-43

DOCKET NO. 50-341

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
3/4 5-2	3/4 5-2
3/4 8-24	3/4 8-24
B3/4 6-1	B3/4 6-1

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 The emergency core cooling systems shall be OPERABLE with:

a. The core spray system (CSS) consisting of two subsystems with each subsystem comprised of:

   1. Two OPERABLE CSS pumps, and
   2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

b. The low pressure coolant injection (LPCI) system of the residual heat removal system consisting of two subsystems with each subsystem comprised of:

   1. Two OPERABLE LPCI (RHR) pumps, and
   2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.***

c. The high pressure cooling injection (HPCI) system consisting of:

   1. One OPERABLE HPCI pump, and
   2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

d. The automatic depressurization system (ADS) with at least five OPERABLE ADS valves.

APPLICABILITY: OPERATIONAL CONDITION 1, 2* ** # and 3* **.

---

*The HPCI system is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 150 psig.

**The ADS is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 150 psig.

***Upon receipt of an LPCI initiation signal, operator action is required to manually open the torus suction valves to facilitate LPCI operation if the LPCI system is in the RHR shutdown cooling mode of operation per Specification 3.4.9.1.

#See Special Test Exception 3.10.6.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. For the core spray system:

1. With one CSS subsystem inoperable, provided that at least one LPCI pump in each LPCI subsystem is OPERABLE, restore the inoperable CSS subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With both CSS subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. For the LPCI system:

1. With one LPCI pump in either or both LPCI subsystems inoperable, provided that at least one CSS subsystem is OPERABLE, restore the inoperable LPCI pump(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one LPCI subsystem otherwise inoperable, provided that both CSS subsystems are OPERABLE, restore the inoperable LPCI subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With a LPCI system cross-tie valve closed, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

4. With both LPCI subsystems otherwise inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.*

5. The provisions of Specification 3.0.4 are not applicable for up to 4 hours for the purpose of establishing the RHR system in the LPCI mode once the reactor vessel pressure is greater than the RHR cut-in permissive setpoint.

c. For the HPCI system, provided the CSS, the LPCI system, the ADS and the RCIC system are OPERABLE:

1. With the HPCI system inoperable, restore the HPCI system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to $\leq$ 150 psig within the following 24 hours.

*Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

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.

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. 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 surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50 with the exception of exemptions granted for main steam isolation valve leak testing and testing the airlocks after each opening and analyzing the Type A test data.

Appendix J to 10 CFR Part 50, Paragraph III.A.3, requires that all Type A tests be conducted in accordance with the provisions of N45.4-1972, "Leakage-Rate Testing of Containment Structures for Nuclear Reactors." N45.4-1972 requires that Type A test data be analyzed using point-to-point or total time analytical techniques. Specification 4.6.1.2a. requires use of the mass plot analytical technique. The mass plot method is considered the better analytical technique, since it yields a confidence interval which is a small fraction of the calculated leak rate; and the interval decreases as more data sets are added to the calculation. The total time and point-to-point techniques may give confidence intervals, which are large fractions of the calculated leak rate, and the intervals may increase as more data sets are added.

The mass plot method is endorsed by ANSI/ANS 56.8-1981 (Containment System Leakage Requirements) which superseded N45.4-1972.

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.

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 Type A leakage tests 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 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

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.

TABLE 3.8.4.3-1 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

	VALVE NUMBER	SYSTEM(S) AFFECTED
	E41-F022	HPCI
	E41-F041	HPCI
	E41-F042	HPCI
	E41-F059	HPCI
	E41-F075	HPCI
	E41-F079	HPCI
	E41-F600	HPCI
7.	E51-F001	Reactor Core Isolation Cooling System (RCIC)
	E51-F002	RCIC
	E51-F007	RCIC
	E51-F008	RCIC
	E51-F010	RCIC
	E51-F012	RCIC
	E51-F013	RCIC
	E51-F019	RCIC
	E51-F022	RCIC
	E51-F029	RCIC
	E51-F031	RCIC
	E51-F045	RCIC
	E51-F046	RCIC
	E51-F059	RCIC
	E51-F062	RCIC
	E51-F084	RCIC
8.	G1154-F018	Drywell Floor Drain System
	G1154-F600	Drywell Floor Drain System
9.	G33-F001	Reactor Water Clean-Up System (RWCU)
	G33-F004	RWCU
10.	G51-F600	Torus Water Management System (TWMS)
	G51-F601	TWMS
	G51-F602	TWMS
	G51-F603	TWMS
	G51-F604	TWMS
	G51-F605	TWMS
	G51-F606	TWMS
	G51-F607	TWMS
11.	N11-F607	Main Steam System
	N11-F608	Main Steam System
	N11-F609	Main Steam System
	N11-F610	Main Steam System

TABLE 3.8.4.3-1 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER		SYSTEM(S) AFFECTED
12.	DELETED	
13.	P44-F601A	Emergency Equipment Cooling Water (EECW)
	P44-F601B	EECW
	P44-F602A	EECW
	P44-F602B	EECW
	P44-F603A	EECW
	P44-F603B	EECW
	P44-F604	EECW
	P44-F605A	EECW
	P44-F605B	EECW
	P44-F606A	EECW
	P44-F606B	EECW
	P44-F607A	EECW
	P44-F607B	EECW
	P44-F608	EECW
	P44-F613	EECW
	P44-F614	EECW
	P44-F615	EECW
	P44-F616	EECW
14.	P50-F603	Compressed Air Systems
	P50-F604	Compressed Air Systems
15.	T48-F601A	Containment Atmosphere Control System
	T48-F601B	Containment Atmosphere Control System
	T48-F602A	Containment Atmosphere Control System
	T48-F602B	Containment Atmosphere Control System
	T48-F603A	Containment Atmosphere Control System
	T48-F603B	Containment Atmosphere Control System
	T48-F604A	Containment Atmosphere Control System
	T48-F604B	Containment Atmosphere Control System
	T48-F605A	Containment Atmosphere Control System
	T48-F605B	Containment Atmosphere Control System
	T48-F606A	Containment Atmosphere Control System
	T48-F606B	Containment Atmosphere Control System
	T4803-F601	Containment Atmosphere Purging System
	T4803-F602	Containment Atmosphere Purging System
16.	T49-F601	Primary Containment Pneumatic Supply System
	T49-F602	Primary Containment Pneumatic Supply System

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 8 TO FACILITY OPERATING LICENSE NO. NPF-43

DETROIT EDISON COMPANY

WOLVERINE POWER SUPPLY COOPERATIVE, INCORPORATED

FERMI-2

DOCKET NO. 50-341

1.0 INTRODUCTION

By letter dated March 9, 1987, Detroit Edison requested an amendment to Facility Operating License No. NPF-43 for Fermi-2. The proposed changes are to Technical Specification 3/4.8.4.3 (Motor-Operated Valves Thermal Overload Protection); Specification 3/4.5.1 (Emergency Core Cooling System - Operating); and Bases 3/4.6.1.2 (Primary Containment Leakage).

The proposed changes are intended to correct either editorial or typographical errors which the licensee has identified in the Fermi-2 Technical Specifications.

2.0 EVALUATION

2.1 Motor-Operated Valves Thermal Overload Protection (3/4.8.4.3)

The existing Motor-Operated Valves Thermal Overload Protection Table 3.8.4.3-1, Item 12, currently specifies a Condensate Storage and Transfer System valve (P11-F616). The proposed change would delete this line item from the Technical Specifications. This change is proposed because the Condensate Storage and Transfer System valve (P11-F616) is no longer applicable to the design. In 1984, prior to the issuance of the Fermi-2 Facility Operating License, a design change was implemented and the valve was replaced with a spectacle flange. At that time, all design documents and drawings were changed and the Final Safety Analysis Report updated to reflect the change to the spectacle flange. This change should have been reflected in Tables 3.6.3-1 and 3.8.4.3-1. However, because of an oversight, Table 3.8.4.3-1 remained unchanged. We find this correction, to make the Technical Specifications consistent with the plant design, acceptable.

2.2 Emergency Core Cooling Systems-Operating (3/4.5.1)

The existing ECCS-Operating, ACTIONS b.2 and b.3 (on page 3/4 5-2) are out of sequence.

8707310263 870717
PDR ADOCK 05000341
P PDR

The proposed change would resequence these two Action statements by replacing ACTION b.2 with ACTION b.3 and by replacing ACTION b.3 with ACTION b.2. This editorial correction is administrative in nature which we find to be acceptable.

2.3 Primary Containment Leakage (Bases 3/4.6.1.2)

The existing Bases reference, in two separate places, an American National Standard document N45.5-1972, "Leakage-Rate Testing of Containment Structures for Nuclear Reactors." The proposed change would correct two identical typographical errors by replacing "N45.5-1972" with "N45.4-1972" which is the correct reference. This correction is administrative in nature, required to correct a reference, and is therefore acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes to requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. We have 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 nor environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSIONS

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: F. Skopec, NRR

Dated: July 17, 1987

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

July 17, 1987

MEMORANDUM FOR: Sholly Coordinator

FROM: John J. Stefano, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE (TAC NO. 64835)

Detroit Edison Company, Docket No. 50-341, Fermi-2, Monroe County, Michigan

Date of application for amendment: March 9, 1987

Brief description of amendment: This amendment revises the Fermi-2 Technical Specifications to make editorial and typographical error corrections to Specification 3/4.8.4.3 (MOV Thermal Overload Protection), Specification 3/4.5.1 (ECCS-Operating), and Specification 3/4.6.1.2 (the Bases for Primary Containment Leakage). The change to Specification 3/4.8.4.3 deletes reference in Table 3.8.4.3-1 of Item 12, Valve No. P11-F616, which is no longer applicable for the Condensate Storage and Transfer System design; the change to Specification 3/4.5.1 resequences ECCS-Operating ACTIONS b.2 and b.3 under 3.5.1 by replacing ACTION b.2 with ACTION b.3 and by replacing ACTION b.3 with ACTION b.2; the change to Bases 3/4.6.1.2 corrects two typographical errors that reference an American National Standard document N45.4-1972, "Leakage-Rate Testing of Containment Structures for Nuclear Reactors." (This document was incorrectly referenced as "N45.5-1972.")

8707300489 XA
2pp

Date of Issuance: 7/17/87

Effective Date: 7/17/1987

Amendment No.: 8

Facility Operating License No. NPF-43: Amendment revises the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: April 8, 1987 (52 FR 11359)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 17, 1987

No significant hazards consideration comments received: No.

Local Public Document Room location: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161.

John J. Stefano

John J. Stefano, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V and Special Projects