April 11, 1991

Docket No. 50-341

Mr. William S. Orser
Senior Vice President - Nuclear
Operations
Detroit Edison Company
6400 North Dixie Highway
Newport, Michigan 48166

DISTRIBUTION

Docket File	JStang
NRC & Local PDRs	GHill(4)
PD31 R/F	Wanda Jones
EJordan	JCalvo
JZwolinski	ACRS (10)
PShuttleworth	GPA/PA
OGC	ARM/LFMB
DHagan	

Dear Mr. Orser:

SUBJECT: AMENDMENT NO. 64 TO FACILITY OPERATING LICENSE NO. NPF-43: (TAC NO. 77832)

The Commission has issued the enclosed Amendment No. 64 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 dated March 26, 1990 (NRC-89-0299) as supplemented by letter dated February 21, 1991, (NRC-91-0019).

The amendment revises the Technical Specifications by removing the existing cycle-specific parameters and placing them in a Core Operating Limits Report under the control of the licensee. The proposed changes are consistent with guidance provided in NRC Generic Letter 88-16.

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,

/s/

John F. Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 64 to NPF-43
2. Safety Evaluation

cc w/enclosures:
See next page

LA/PD31:DRP345	PM/PD31:DRP345	D/PD31:DRP345	OGC
PShuttleworth	JStang/tg	LMarsh	
3/12/91	3/18/91	4/9/91	3/18/91

OFFICIAL RECORD COPY

BC/SRXB
R. Jones
3/11/91

NRC FILE CENTER COPY

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

April 11, 1991

Docket No. 50-341

Mr. William S. Orser
Senior Vice President - Nuclear
Operations
Detroit Edison Company
6400 North Dixie Highway
Newport, Michigan 48166

Dear Mr. Orser:

SUBJECT: AMENDMENT NO. 64 TO FACILITY OPERATING LICENSE NO. NPF-43: (TAC NO. 77832)

The Commission has issued the enclosed Amendment No. 64 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 dated March 26, 1990 (NRC-89-0299) as supplemented by letter dated February 21, 1991, (NRC-91-0019).

The amendment revises the Technical Specifications by removing the existing cycle-specific parameters and placing them in a Core Operating Limits Report under the control of the licensee. The proposed changes are consistent with guidance provided in NRC Generic Letter 88-16.

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,

John F. Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 64 to NPF-43
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. William Orser
Detroit Edison Company

Fermi-2 Facility

cc:
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 30195
Lansing, Michigan 48909

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

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

Ms. Lynne Goodman
Supervisor - Licensing
Detroit Edison Company
Fermi Unit 2
6400 North Dixie Highway
Newport, Michigan 48166

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

DETROIT EDISON COMPANY

FERMI-2

DOCKET NO. 50-341

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 64
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 26, 1990, as supplemented by letter dated February 21, 1991, 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. 64, 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

William O. Long

for L. B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 11, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 64

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 a vertical line indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
i	i
ii*	ii*
xi*	xi*
xii	xii
xix*	xix*
xx	xx
xxi	xxi
xxv	xxv
1-2	1-2
-	1-2a
3/4 2-1	3/4 2-1
3/4 2-2	-
3/4 2-3	-
3/4 2-4	-
3/4 2-4a	-
3/4 2-6	3/4 2-6
3/4 2-6a	-
3/4 2-7	3/4 2-7
3/4 2-8	-
3/4 2-8a	-
3/4 2-8b	-
3/4 2-9	-
3/4 2-10	3/4 2-10
3/4 4-1	3/4 4-1
B 3/4 2-1	B 3/4 2-1
B 3/4 2-1a	B 3/4 2-1a
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4	B 3/4 2-4
B 3/4 2-4a	-
B 3/4 2-4b	-
B 3/4 2-5	-
6-21	6-21
6-22	6-22
-	6-22a

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

DEFINITIONS

SECTION

1.0 DEFINITIONS PAGE

1.1 ACTION..... 1-1

1.2 AVERAGE PLANAR EXPOSURE..... 1-1

1.3 AVERAGE PLANAR LINEAR HEAT GENERATION RATE..... 1-1

1.4 CHANNEL CALIBRATION..... 1-1

1.5 CHANNEL CHECK..... 1-1

1.6 CHANNEL FUNCTIONAL TEST..... 1-1

1.7 CORE ALTERATION..... 1-2

1.7a CORE OPERATING LIMITS REPORT..... 1-2

1.8 CRITICAL POWER RATIO..... 1-2

1.9 DOSE EQUIVALENT I-131..... 1-2

1.10 $\bar{E}$-AVERAGE DISINTEGRATION ENERGY..... 1-2

1.11 EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME..... 1-2a

1.12 FRACTION OF LIMITING POWER DENSITY..... 1-2a

1.13 FRACTION OF RATED THERMAL POWER..... 1-2a

1.14 FREQUENCY NOTATION..... 1-3

1.15 IDENTIFIED LEAKAGE..... 1-3

1.16 ISOLATION SYSTEM RESPONSE TIME..... 1-3

1.17 LIMITING CONTROL ROD PATTERN..... 1-3

1.18 LINEAR HEAT GENERATION RATE..... 1-3

1.19 LOGIC SYSTEM FUNCTIONAL TEST..... 1-3

1.20 MAXIMUM FRACTION OF LIMITING POWER DENSITY..... 1-3

1.21 MEMBER(S) OF THE PUBLIC..... 1-3

1.22 MINIMUM CRITICAL POWER RATIO..... 1-4

1.23 OFF-GAS TREATMENT SYSTEM..... 1-4

1.24 OFFSITE DOSE CALCULATION MANUAL..... 1-4

1.25 OPERABLE - OPERABILITY..... 1-4

INDEX

DEFINITIONS

SECTION

DEFINITIONS (Continued) PAGE

1.26 OPERATIONAL CONDITION - CONDITION..... 1-4

1.27 PHYSICS TESTS..... 1-4

1.28 PRESSURE BOUNDARY LEAKAGE..... 1-4

1.29 PRIMARY CONTAINMENT INTEGRITY..... 1-4

1.30 PROCESS CONTROL PROGRAM..... 1-5

1.31 PURGE-PURGING..... 1-5

1.32 RATED THERMAL POWER..... 1-5

1.33 REACTOR PROTECTION SYSTEM RESPONSE TIME..... 1-6

1.34 REPORTABLE EVENT..... 1-6

1.35 ROD DENSITY..... 1-6

1.36 SECONDARY CONTAINMENT INTEGRITY..... 1-6

1.37 SHUTDOWN MARGIN..... 1-6

1.38 SITE BOUNDARY..... 1-6

1.39 SOLIDIFICATION..... 1-7

1.40 SOURCE CHECK..... 1-7

1.41 STAGGERED TEST BASIS..... 1-7

1.42 THERMAL POWER..... 1-7

1.43 TURBINE BYPASS SYSTEM RESPONSE TIME..... 1-7

1.44 UNIDENTIFIED LEAKAGE..... 1-7

1.45 UNRESTRICTED AREA..... 1-7

1.46 VENTILATION EXHAUST TREATMENT SYSTEM..... 1-8

1.47 VENTING..... 1-8

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION	PAGE
3/4.11 RADIOACTIVE EFFLUENTS	
3/4.11.1 LIQUID EFFLUENTS	
Concentration	3/4 11-1
Dose	3/4 11-5
Liquid Waste Treatment	3/4 11-6
Liquid Holdup Tanks	3/4 11-7
3/4.11.2 GASEOUS EFFLUENTS	
Dose Rate	3/4 11-8
Dose-Noble Gases	3/4 11-12
Dose-Iodine 131, Iodine 133, Tritium, and Radionuclides in Particulate Form	3/4 11-13
Off-Gas Treatment System	3/4 11-14
Ventilation Exhaust Treatment System	3/4 11-15
Explosive Gas Mixture	3/4 11-16
Main Condenser	3/4 11-17
Venting or Purging	3/4 11-18
3/4.11.3 SOLID RADIOACTIVE WASTE TREATMENT	3/4 11-19
3/4.11.4 TOTAL DOSE	3/4 11-20
3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING	
3/4.12.1 MONITORING PROGRAM	3/4 12-1
3/4.12.2 LAND USE CENSUS	3/4 12-13
3/4.12.3 INTERLABORATORY COMPARISON PROGRAM	3/4 12-14

INDEX

BASES

SECTION		PAGE
3/4.0 APPLICABILITY		B 3/4 0-1
3/4.1 REACTIVITY CONTROL SYSTEMS		
3/4.1.1	SHUTDOWN MARGIN	B 3/4 1-1
3/4.1.2	REACTIVITY ANOMALIES	B 3/4 1-1
3/4.1.3	CONTROL RODS	B 3/4 1-2
3/4.1.4	CONTROL ROD PROGRAM CONTROLS	B 3/4 1-3
3/4.1.5	STANDBY LIQUID CONTROL SYSTEM	B 3/4 1-4
3/4.2 POWER DISTRIBUTION LIMITS		
3/4.2.1	AVERAGE PLANAR LINEAR HEAT GENERATION RATE	B 3/4 2-1
3/4.2.2	APRM SETPOINTS	B 3/4 2-2
3/4.2.3	MINIMUM CRITICAL POWER RATIO	B 3/4 2-3
3/4.2.4	LINEAR HEAT GENERATION RATE	B 3/4 2-4
3/4.3 INSTRUMENTATION		
3/4.3.1	REACTOR PROTECTION SYSTEM INSTRUMENTATION	B 3/4 3-1
3/4.3.2	ISOLATION ACTUATION INSTRUMENTATION	B 3/4 3-2
3/4.3.3	EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION	B 3/4 3-2
3/4.3.4	ATWS RECIRCULATION PUMP TRIP SYSTEM ACTUATION INSTRUMENTATION	B 3/4 3-3
3/4.3.5	REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION	B 3/4 3-3
3/4.3.6	CONTROL ROD BLOCK INSTRUMENTATION	B 3/4 3-3
3/4.3.7	MONITORING INSTRUMENTATION	
	Radiation Monitoring Instrumentation	B 3/4 3-3
	Seismic Monitoring Instrumentation	B 3/4 3-4

INDEX

ADMINISTRATIVE CONTROLS

SECTION		PAGE
6.1 RESPONSIBILITY		6-1
6.2 ORGANIZATION		6-1
6.2.1	OFFSITE	6-1
6.2.2	UNIT STAFF	6-1
6.2.3	INDEPENDENT SAFETY ENGINEERING GROUP	6-6
	FUNCTION	6-6
	COMPOSITION	6-6
	RESPONSIBILITIES	6-6
	AUTHORITY	6-6
6.2.4	SHIFT TECHNICAL ADVISOR	6-6
6.3 UNIT STAFF QUALIFICATIONS		6-7
6.4 TRAINING		6-7
6.5 REVIEW AND AUDIT		6-7
6.5.1	ONSITE REVIEW ORGANIZATION (OSRO)	
	FUNCTION	6-7
	COMPOSITION	6-7
	ALTERNATES	6-8
	MEETING FREQUENCY	6-8
	QUORUM	6-8
	RESPONSIBILITIES	6-8
	RECORDS	6-9
6.5.2	NUCLEAR SAFETY REVIEW GROUP (NSRG)	6-9
	FUNCTION	6-9
	COMPOSITION	6-10
	ALTERNATES	6-10
	CONSULTANTS	6-10
	MEETING FREQUENCY	6-10
	QUORUM	6-10

ADMINISTRATIVE CONTROLS

SECTION		PAGE
	REVIEW	6-10
	AUDITS	6-11
	RECORDS	6-12
6.5.3	TECHNICAL REVIEW AND CONTROL	6-12
	ACTIVITIES	6-12
	REVIEW	6-13
	SAFETY EVALUATIONS	6-13
	QUALIFICATIONS	6-13
	RECORDS	6-13
6.6	REPORTABLE EVENT ACTION	6-13
6.7	SAFETY LIMIT VIOLATION	6-14
6.8	PROCEDURES AND PROGRAMS	6-14
6.9	REPORTING REQUIREMENTS	6-16
6.9.1	ROUTINE REPORTS	6-16
	STARTUP REPORT	6-16
	ANNUAL REPORTS	6-17
	MONTHLY OPERATING REPORTS	6-18
	ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT	6-18
	SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT	6-19
6.9.2	SPECIAL REPORTS	6-21
6.9.3	CORE OPERATING LIMITS REPORT	6-21
6.10	RECORD RETENTION	6-21
6.11	RADIATION PROTECTION PROGRAM	6-22a
6.12	HIGH RADIATION AREA	6-22a
6.13	PROCESS CONTROL PROGRAM	6-23
6.14	OFFSITE DOSE CALCULATION MANUAL	6-24
6.15	MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS	6-24

LIST OF FIGURES

FIGURE		PAGE
3.1.5-1	SODIUM PENTABORATE VOLUME/ CONCENTRATION REQUIREMENTS.....	3/4 1-21
3.4.1.4-1	THERMAL POWER VS CORE FLOW	3/4 4-6a
3.4.6.1-1	MINIMUM REACTOR PRESSURE VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE	3/4 4-21
3.4.10-1	THERMAL POWER VS. CORE FLOW.....	3/4 4-31
4.7.5-1	SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST ..	3/4 7-21
B 3/4 3-1	REACTOR VESSEL WATER LEVEL	B 3/4 3-7
B 3/4.4.6-1	FAST NEUTRON FLUENCE (E>1MeV) AT 1/4 T AS A FUNCTION OF SERVICE LIFE	B 3/4 4-7
B 3/4.6.2-1	LOCAL POOL TEMPERATURE LIMIT	B 3/4 6-5
B 3/4.7.3-1	ARRANGEMENT OF SHORE BARRIER SURVEY POINTS ..	B 3/4 7-6
5.1.1-1	EXCLUSION AREA	5-2
5.1.2-1	LOW POPULATION ZONE	5-3
5.1.3-1	MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS	5-4

LIST OF TABLES (Continued)

TABLE		PAGE
4.8.2.1-1	BATTERY SURVEILLANCE REQUIREMENTS	3/4 8-12
3.8.4.2-1	PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES	3/4 8-19
3.8.4.3-1	MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION	3/4 8-21
3.8.4.5-1	STANDBY LIQUID CONTROL SYSTEM ASSOCIATED ISOLATION DEVICES 480 V MOTOR CONTROL CENTERS...	3/4 8-27
4.11.1.1.1-1	RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM	3/4 11-2
4.11.2.1.2-1	RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM	3/4 11-9
3.12.1-1	RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM ..	3/4 12-3
3.12.1-2	REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES	3/4 12-9
4.12.1-1	DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS	3/4 12-10
B3/4.4.6-1	REACTOR VESSEL TOUGHNESS	B 3/4 4-6
5.7.1-1	COMPONENT CYCLIC OR TRANSIENT LIMITS	5-7
6.2.2-1	MINIMUM SHIFT CREW COMPOSITION	6-5

CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement of SRMs, IRMs, TIPs, or special movable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.7a The CORE OPERATING LIMTS REPORT (COLR) is a plant specific document that provides selected core operating limits for the current reload cycle. These cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.3. Plant operation within these core operating limits is addressed in individual specifications.

CRITICAL POWER RATIO

1.8 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of an NRC approved critical power correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.9 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.10 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.11 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

FRACTION OF LIMITING POWER DENSITY

1.12 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.

FRACTION OF RATED THERMAL POWER

1.13 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) shall not exceed:

a. The MAPLHGR limit which has been approved for the respective fuel and lattice type as a function of the average planar exposure (as determined by the NRC approved methodology described in GESTAR-II), or

b. When hand calculations are required, the most limiting lattice type MAPLHGR limit as a function of the average planar exposure shown in the CORE OPERATING LIMITS REPORT (COLR) for the applicable fuel type.

The MAPLHGR limit shall be multiplied by a factor provided in the COLR during single loop operation.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the above limits, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits required by Specification 3.2.1:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit specified in the CORE OPERATING LIMITS REPORT (COLR), with:

$$\tau = \frac{(\tau_{ave} - \tau_B)}{\tau_A - \tau_B}$$

where:

τ_A = 1.096 seconds, control rod average scram insertion time limit to notch 36 per Specification 3.1.3.3,

$$\tau_B = 0.813 + 1.65 \sqrt{\frac{N_1}{\sum_{i=1}^{n} N_i}} \quad 0.018,$$

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i}$$

n = number of surveillance tests performed to date in cycle,

N_i = number of active control rods measured in the ith surveillance test,

τ_i = average scram time to notch 36 of all rods measured in the ith surveillance test, and

N_1 = total number of active rods measured in Specification 4.1.3.2.a.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

APPLICABILITY:

OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION

a. With MCPR less than the applicable MCPR limit in the COLR, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

b. With the main turbine bypass system inoperable and/or the moisture separator reheater inoperable per Specification 3.7.9, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that, within one hour, MCPR is determined to be equal to or greater than the applicable MCPR limit in the COLR.

SURVEILLANCE REQUIREMENTS

4.2.3.1 MCPR, with :

a. τ = 1.0 prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.1.3.2, or

b. τ as defined in Specification 3.2.3 used to determine the limit within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2,

shall be determined to be equal to or greater than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT (COLR):

a. At least once per 24 hours,

b. Within 12 hours after completion of the a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

d. The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 Prior to the use of a MCPR limit which is based upon a specific control rod pattern and whenever Surveillance Requirement 4.2.3.1 is performed while using a MCPR limit based upon a specific control rod pattern, the required control rod pattern shall be verified.

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the LHGR limit specified in the CORE OPERATING LIMITS REPORT (COLR) for the applicable fuel type.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the LHGR of any fuel rod exceeding the applicable limit, initiate corrective action within 15 minutes and restore the LHGR to within the applicable limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGRs shall be determined to be equal to or less than the applicable limit:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.

d. The provisions of Specification 4.0.4 are not applicable.

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*.
ACTION:

a. With one reactor coolant system recirculation loop not in operation:

   1. Within 4 hours:

      a) Place the individual recirculation pump flow controller for the operating recirculation pump in the Manual mode.

      b) Reduce THERMAL POWER to less than or equal to 70% of RATED THERMAL POWER.

      c) Limit the speed of the operating recirculation pump to less than or equal to 75% of rated pump speed.

      d) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to 1.08 per Specification 2.1.2.

      e) Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit per Specification 3.2.1.

      f) Reduce the Average Power Range Monitor (APRM) Scram and Rod Block and Rod Block Monitor Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation# per Specifications 2.2.1, 3.2.2 and 3.3.6.

      g) Perform Surveillance Requirement 4.4.1.1.4 if THERMAL POWER is less than or equal to 30% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is less than or equal to 50% of rated loop flow.

   2. The provisions of Specification 3.0.4 are not applicable.

   3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant system recirculation loop in operation while in OPERATIONAL CONDITION 1, immediately place the Reactor Mode Switch in the SHUTDOWN position.

c. With no reactor coolant system recirculation loops in operation, while in OPERATIONAL CONDITION 2, initiate measures to place the unit in at least HOT SHUTDOWN within the next 6 hours.

*See Special Test Exception 3.10.4
#APRM gain adjustments may be made in lieu of adjusting the APRM and RBM Flow Biased Setpoints to comply with the single loop values for a period of up to 72 hours.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is provided in the CORE OPERATING LIMITS REPORT (COLR).

The COLR MAPLHGR value is the most limiting composite of the fuel mechanical design analysis MAPLHGR and the ECCS MAPLHGR.

Fuel Mechanical Design Analysis: NRC approved methods (specified in Reference 1) are used to demonstrate that all fuel rods in a lattice, operating at the bounding power history, meet the fuel design limits specified in Reference 1. This bounding power history is used as the basis for the fuel design analysis MAPLHGR value.

LOCA Analysis: A LOCA analysis is performed in accordance with 10 CFR 50 Appendix K to demonstrate that the MAPLHGR values comply with the ECCS limits specified in 10 CFR 50.46. The analysis is performed for the most limiting break size, break location, and single failure combination for the plant. (Reference 2)

Only the most limiting MAPLHGR values are provided in the COLR for multiple lattice fuel. When hand calculations are required, these COLR MAPLHGR values for that fuel type are used for all lattices in that bundle.

For some fuel bundle designs MAPLHGR depends only on bundle type and burnup. Other fuel bundles have MAPLHGRs that vary axially depending upon the specific combination of enriched uranium and gadolinia that comprises a fuel bundle cross section at a particular axial node. Each particular combination of enriched uranium and gadolinia, for these fuel bundle types, is called a lattice type. These particular fuel bundle types have MAPLHGRs that vary by lattice (axially) as well as with fuel burnup.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

For plant operation with a single operating recirculation loop, the above MAPLHGR limits are multiplied by a factor specified in the COLR. The COLR factor is derived from LOCA analysis initiated from single loop operation to account for earlier boiling transition at the limiting fuel node compared to the standard LOCA analysis.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limiting MCPRs at steady-state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transients analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limiting MCPR of Specification 3.2.3 is obtained and presented in the CORE OPERATING LIMITS REPORT (COLR).

The evaluation of a given transient begins with the system initial parameters shown in UFSAR Table B.15.0-1 that are input to a GE-core dynamic behavior transient computer program. The codes used to evaluate transients are described in GESTAR II (Reference 1). The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the K_f factor in the COLR is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow the MCPR limit is the product of the MCPR limit at rated flow and the K_f factor. The K_f factors assure that the Safety Limit MCPR will not be violated during a flow increase transient resulting from a motor-generator speed control failure. The K_f factors may be applied to both manual and automatic flow control modes.

The K_f factor values in the COLR were developed generically and are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to RATED THERMAL POWER at rated core flow, although they are applicable for the extended operating region.

For the manual flow control mode, the K_f factors were calculated such that for the maximum flow rate, as limited by the pump scoop tube setpoint and the corresponding THERMAL POWER along the rated flow control line, the limiting bundle's relative power was adjusted until the MCPR changes with different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the K_f.

3/4.2.3 MINIMUM CRITICAL POWER RATIO (Continued)

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at RATED THERMAL POWER and rated core flow.

The K_f factors in the CORE OPERATING LIMITS REPORT (COLR) are conservative for the General Electric plant operation because the operating limit MCPRs in the COLR are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

At THERMAL POWER levels less than or equal to 25 percent of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial startup testing of the plant, a MCPR evaluation will be made at 25 percent of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25 percent of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

3.4.2.4 LINEAR HEAT GENERATION RATE

The thermal expansion rates of UO_2 pellets and Zircalloy cladding are different in that, during heatup, the fuel pellet could come into contact with the cladding and create stress. If the stress exceeds the yield stress of the cladding material, the cladding will crack. The LHGR limit assures that at any exposure, 1% plastic strain on the clad is not exceeded. This limit is a function of fuel type and is presented in the COLR.

References:

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (the approved version at the time the reload analyses are performed shall be identified in the COLR).

2. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K," NEDE-20566-P (the approved version at the time the reload analyses are performed shall be identified in the COLR)

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

CORE OPERATING LIMITS REPORT

6.9.3 Selected cycle specific core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in General Electric Company reports NEDE-24011-P-A and NEDE-20566-P. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The COLR, including any mid-cycle revisions or supplement thereto, shall be submitted upon issuance to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector prior to use.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. ALL REPORTABLE EVENTS.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10 RECORD RETENTION - (continued)

6.10.3 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of quality assurance activities required by the Operational Quality Assurance Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the OSRO and the NSRG.

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.5 including the date at which the service life commences and associated installation and maintenance records.

m. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

*Health physics personnel or personnel escorted by health physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 64 TO FACILITY OPERATING LICENSE NO. NPF-43

DETROIT EDISON COMPANY

FERMI-2

DOCKET NO. 50-341

1.0 INTRODUCTION

By letter dated March 26, 1990, as supplemented by letter dated February 21, 1991, the Detroit Edison Company (DECo or the licensee) requested amendment to the Technical Specifications (TS) appended to Facility Operating License No. NPF-43 for Fermi-2. The proposed amendment would modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to the "FERMI-2 CORE OPERATING LIMITS REPORT" for the values of those limits. The proposed changes also include the addition of the Core Operating Limits Report to the Definitions section and to the reporting requirements of the Administrative Controls section of the TS. Guidance on the proposed changes was developed by the NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket that was endorsed by the Babcock and Wilcox Owners Group. This guidance was provided to all power reactor licensees and applicants by Generic Letter (GL) 88-16, dated October 4, 1988.

The licensee by letter dated February 21, 1991 supplemented the original March 26, 1990 application. The supplemental information indicated the licensee's intention to keep the equations for the minimum critical power ratio (MCPR) in the TS. The change represents resolution, with the NRC staff of the details inherent in the implementation of GL 88-16. The initial findings in the licensee's March 26, 1990 application concerning no significant hazards consideration remain unchanged based on the supplemental information in the February 21, 1991 letter. Therefore, the NRC staff finds the no significant hazards consideration published in the Federal Register on November 14, 1990 (55 FR 47460) remains unchanged and encompasses the supplemental information and changes to the TS pages in the licensee's February 21, 1991 letter.

2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by GL 88-16 and are addressed below.

(1) The Definitions section of the TS was modified to include a definition of the FERMI-2 CORE OPERATING LIMITS REPORT that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with an NRC-approved methodology that maintains the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

(2) The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the FERMI-2 CORE OPERATING LIMITS REPORT that provides these limits.

OPERATING LIMIT	TECHNICAL SPECIFICATION
APLHGR	3/4.2.1
MCPR	3/4.2.3
LHGR	3/4.2.4

(3) Specification 6.9.3 Core Operating Limits Report, was added to the reporting requirement of the Administrative Controls section of the TS. This specification requires that the Fermi-2 Core Operating Limits Report be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, this specification requires that the values of these limits be established using NRC-approved methodology in, "General Electric Standard Application for Reactor Fuel, (GESTAR II)," NEDE-24011-P-A and US Supplement, Revision 9, September 1988 or latest approved version, and consistent with all applicable limits of the safety analysis. Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the Fermi-2 Core Operating Limits Report before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in GL 88-16 on modifying cycle-specific parameter limits in the TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using an NRC-approved methodology, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds the proposed changes 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 has no comments.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in 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 in surveillance requirements. The staff has determined that this amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents which may be released offsite, and that there is no significant increase in individual or

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 this amendment will not be inimical to the common and security or to the health and safety of the public.

Principal Contributor: Daniel Fieno, SRXB/DEST
Thomas Dunning, OTSB/DOEA
John Stang, PD31/DRP345

Date: April 11, 1991

April 11, 1991

Docket No. 50-341

Mr. William S. Orser
Senior Vice President - Nuclear
Operations
Detroit Edison Company
6400 North Dixie Highway
Newport, Michigan 48166

DISTRIBUTION
Docket File
NRC & Local PDRs
PD31 R/F
EJordan
JZwolinski
PShuttleworth
OGC
DHagan
JStang
GHill(4)
Wanda Jones
JCalvo
ACRS (10)
GPA/PA
ARM/LFMB

Dear Mr. Orser:

SUBJECT: AMENDMENT NO. 64 TO FACILITY OPERATING LICENSE NO. NPF-43: (TAC NO. 77832)

The Commission has issued the enclosed Amendment No. 64 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 dated March 26, 1990 (NRC-89-0299) as supplemented by letter dated February 21, 1991, (NRC-91-0019).

The amendment revises the Technical Specifications by removing the existing cycle-specific parameters and placing them in a Core Operating Limits Report under the control of the licensee. The proposed changes are consistent with guidance provided in NRC Generic Letter 88-16.

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,

/s/

John F. Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 64 to NPF-43
2. Safety Evaluation

cc w/enclosures:
See next page

LA/PD31:DRP345	PM/PD31:DRP345	D/PD31:DRP345	OGC
PShuttleworth	JStang:tg	LMarsh	
3/12/91	3/18/91	4/9/91	3/18/91

OFFICIAL RECORD COPY

BC/SRXB
R. Jones
3/11/91