May 15, 1991

Docket No. 50-341

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

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Dear Mr. Orser:

SUBJECT: AMENDMENT NO. 69 TO FACILITY OPERATING LICENSE NO. NPF-43: (TAC NO. 77676)

The Commission has issued the enclosed Amendment No. 69 to Facility Operating License No. NPF-43 for the Fermi-2 facility. This amendment consists of changes to the Plant Technical Specifications (TS) in response to your letter dated August 20, 1990.

The amendment revises the TS by implementing an expanded operating domain.

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:

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. 69 to NPF-43
2. Safety Evaluation

cc w/enclosures:
See next page

LA/PD31:DRP345	PM/PD31:DRP345	D/PD31:DRP345	OGC
PShuttleworth	JStang	LMarsh	
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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Senior Vice President - Nuclear
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Detroit Edison Company
6400 North Dixie Highway
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cc w/enclosures:
See next page

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DETROIT EDISON COMPANY

FERMI-2

DOCKET NO. 50-341

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 69
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 August 1, 1990, 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. 69, 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

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: May 15, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 69

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
*iii	*iii
iv	iv
*xi	*xi
xii	xii
2-4	2-4
B 2-7	B 2-7
*3/4 1-17	*3/4 1-17
3/4 1-18	3/4 1-18
3/4 2-5	3/4 2-5
3/4 2-5a	-
3/4 3-8	3/4 3-8
3/4 3-41	3/4 3-41
3/4 3-43	3/4 3-43
3/4 3-44	3/4 3-44
3/4 3-44a	3/4 3-44a
*3/4 3-45	*3/4 3-45
3/4 3-46	3/4 3-46
3/4 4-1	3/4 4-1
3/4 4-2	3/4 4-2
B 3/4 2-1a	B 3/4 2-1a
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4	B 3/4 2-4

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

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

TABLE 2.2.1-1
REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-Upscale		
1) During two recirculation loop operation:		
a. Flow Biased	≤ 0.66 W+64%, with	≤ 0.66 W+67%, with
b. High Flow Clamped	a maximum of ≤ 113.5% of RATED THERMAL POWER	a maximum of ≤ 115.5% of RATED THERMAL POWER
2) During single recirculation loop operation:		
a. Flow Biased	≤0.66W+58.7%,**	≤0.66W+61.7%,**
b. High Flow Clamped	NA	NA
c. Fixed Neutron Flux-Upscale	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	N.A.	N.A.
3. Reactor Vessel Steam Dome Pressure - High	≤ 1068 psig	≤ 1088 psig
4. Reactor Vessel Low Water Level - Level 3	≥ 173.4 inches*	≥ 171.9 inches

*See Bases Figure B 3/4 3-1.
**During single recirculation loop operation, rather than adjusting the APRM Flow Biased Setpoints to comply with the single loop values, the gain of the APRMs may be adjusted for a period not to exceed 72 hours such that the final APRM readings are at least 5.3% of rated power greater than 100% times FRTP, provided that the adjusted APRM readings do not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

BASES

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Fixed Neutron Flux-Upscale setpoint; i.e, for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow Biased Neutron Flux-High setpoint, a time constant of 6 ± 1 seconds is introduced into the flow biased APRM in order to simulate the fuel thermal transient characteristics. A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown. For single recirculation loop operation, the reduced APRM setpoints are based on a Δ W value of 8%. The Δ W value corrects for the difference in indicated drive flow (in percentage of drive flow which produces rated core flow) between two loop and single loop operation of the same core flow. The decrease in setpoint is derived by multiplying the slope of the setpoint curve by 8%. The High Flow Clamped Flow Biased Neutron Flux-High setpoint is not applicable to single loop operation as core power levels which would require this limit are not achievable in a single loop configuration.

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious

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REACTIVITY CONTROL SYSTEMS

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both rod block monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1 with:

a. THERMAL POWER greater than or equal to 30% of RATED THERMAL POWER and less than 90% of RATED THERMAL POWER and the MINIMUM CRITICAL POWER RATIO (MCPR) less than 1.71, or

b. THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER and the MCPR less than 1.40.

ACTION:

a. With one RBM channel inoperable:

1. Verify that the reactor is not operating on a LIMITING CONTROL ROD PATTERN, and

2. Restore the inoperable RBM channel to OPERABLE status within 24 hours.

Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within 1 hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and for the OPERATIONAL CONDITIONS specified in Table 4.3.6-1.

b. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating on a LIMITING CONTROL ROD PATTERN.

SPECIFICATION 3/4.2.2 DELETED

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High				
a. Float Switch	NA	Q	R	1, 2, 5(j)
b. Level Transmitter	S	M	R	1, 2, 5(j)
9. Turbine Stop Valve - Closure	NA	M	R	1
10. Turbine Control Valve Fast Closure	NA	M	NA	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	M	NA	1, 2, 3, 4, 5
13. Deleted.				

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least ½ decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least ½ decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Within 24 hours prior to startup, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.
(g) Deleted.
(h) This calibration shall consist of verifying the 6 ± 1 second simulated thermal power time constant.
(i) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
(j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

INSTRUMENTATION

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod block instrumentation channel trip setpoint* less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

*The APRM Flow Biased Neutron Flux-High instrumentation need not be declared inoperable upon entering single reactor recirculation loop operation provided the setpoints are adjusted within 4 hours per Specification 3.4.1.1.

TABLE 3.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION STATEMENTS

ACTION 60 - Declare the RBM inoperable and take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within 1 hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within 1 hour.

ACTION 63 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, initiate a rod block.

TABLE NOTATIONS

* When (1) THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER and less than 90% of RATED THERMAL POWER and MCPR is less than 1.71, or (2) THERMAL POWER is greater than or equal to 90% of RATED THERMAL POWER and MCPR is less than 1.40.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(a) The RBM shall be automatically bypassed when a peripheral control rod is selected or the reference APRM channel indicates less than 30% of RATED THERMAL POWER.

(b) This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range 3 or higher.

(c) This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

(d) This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

(e) This function shall be automatically bypassed when the IRM channels are on range 1.

(f) These two Source Range Monitors shall be OPERABLE as required by Specification 3.9.2.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	As specified in the CORE OPERATING LIMITS REPORT	As specified in the CORE OPERATING LIMITS REPORT
b. Inoperative	NA	NA
c. Downscale	$\geq$ 5% of RATED THERMAL POWER	$\geq$ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - High		
1) During two recirculation loop operation	< 0.66 W + 58%* with a maximum of 108%	$\leq$ 0.66 W + 61%* with a maximum of 110%
2) During single recirculation loop operation	$\leq$ 0.66 W + 52.7%#*	$\leq$ 0.66 W + 55.7%#*
b. Inoperative	NA	NA
c. Downscale	$\geq$ 5% of RATED THERMAL POWER	$\geq$ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Setdown	$\leq$ 12% of RATED THERMAL POWER	$\leq$ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq 1.0 \times 10^5$ cps	$\leq 1.6 \times 10^5$ cps
c. Inoperative	NA	NA
d. Downscale	$\geq$ 3 cps**	$\geq$ 2 cps**

*The APRM rod block function is varied as a function of recirculation loop drive flow (W).

**May be reduced to $\geq$ 0.7 cps provided the signal-to-noise ratio $\geq$ 20.

#During single recirculation loop operation, rather than adjusting the APRM Flow Biased Setpoints to comply with the single loop values, the gain of the APRMs may be adjusted for a period not to exceed 72 hours such that the final APRM readings are at least 5.3% of rated power greater than 100% times FRTP, provided that the adjusted APRM readings do not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS (Continued)

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	≤ 589'11½"	≤ 591'0"
b. Scram Trip Bypass	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 110/125% of rated flow	≤ 113/125% of rated flow
b. Inoperative	NA	NA
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	NA

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	S/U(b),M	Q	1*
b. Inoperative	NA	S/U(b),M	NA	1*
c. Downscale	NA	S/U(b),M	Q	1*
2. APRM				
a. Flow Biased Neutron Flux - High	S	S/U(b),M	SA	1
b. Inoperative	NA	S/U(b),M	NA	1, 2, 5
c. Downscale	S	S/U(b),M	SA	1
d. Neutron Flux - Upscale, Setdown	S	S/U(b),M	SA	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U(b),W	NA	2***, 5
b. Upscale	S	S/U(b),W	SA	2***, 5
c. Inoperative	NA	S/U(b),W	NA	2***, 5
d. Downscale	S	S/U(b),W	SA	2***, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U(b),W	NA	2, 5
b. Upscale	S	S/U(b),W	SA	2, 5
c. Inoperative	NA	S/U(b),W	NA	2, 5
d. Downscale	S	S/U(b),W	SA	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	Q	R	1, 2, 5**
b. Scram Trip Bypass	NA	R	NA	2, 5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U(b),M	Q	1
b. Inoperative	NA	S/U(b),M	NA	1
c. Comparator	NA	S/U(b),M	Q	1
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	R	NA	3, 4

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) Within 24 hours prior to startup, if not performed within the previous 7 days.

* When (1) THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER and less than 90% of RATED THERMAL POWER and MCPR is less than 1.71, or (2) THERMAL POWER is greater than or equal to 90% of RATED THERMAL POWER and MCPR is less than 1.40.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

*** With IRMs on Range 2 or less.

3/4.4 REACTOR COOLANT SYSTEM
3/4.4.1 RECIRCULATION SYSTEM
RECIRCULATION LOOPS
LIMITING CONDITION FOR OPERATION

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 Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation# per Specifications 2.2.1 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 Flow Biased Setpoints to comply with the single loop values for a period of up to 72 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 Each pump discharge valve shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each STARTUP* prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.

4.4.1.1.2 Each pump MG set scoop tube mechanical and electrical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 110% and 107%, respectively, of rated core flow, at least once per 18 months.

4.4.1.1.3 With one reactor coolant system recirculation loop not in operation, at least once per 12 hours verify that:

a. THERMAL POWER is less than or equal to 70% of RATED THERMAL POWER, and
b. The individual recirculation pump flow controller for the operating recirculation pump is in the Manual mode, and
c. The speed of the operating recirculation pump is less than or equal to 75% of rated pump speed.

4.4.1.1.4 With one reactor coolant system loop not in operation with THERMAL POWER less than or equal to 30% of RATED THERMAL POWER or with recirculation loop flow in the operating loop less than or equal to 50% of rated loop flow, verify the following differential temperature requirements are met within no more than 15 minutes prior to either THERMAL POWER increase or recirculation flow increase:

a. Less than or equal to 145°F between reactor vessel steam space coolant and bottom head drain line coolant, and
b. Less than or equal to 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel**, and
c. Less than or equal to 50°F between the reactor coolant within the loop not in operation and the operating loop.**

*If not performed within the previous 31 days.

**Requirement does not apply when the recirculation loop not in operation is isolated from the reactor pressure vessel.

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

For plant operation with a single recirculation loop, the above MAPLHGR limits are multiplied by a factor specified in the CORE OPERATING LIMITS REPORT (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.

Power and flow dependent adjustments are provided in the COLR to assure that the fuel thermal-mechanical design criteria are preserved during abnormal transients initiated from off-rated conditions.

3/4.2.2 APRM SETPOINTS

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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).

3/4.2.3 MINIMUM CRITICAL POWER RATIO (Continued)

Details on how evaluations are performed, on the methods used, and how the MCPR limit is adjusted for operation at less than rated power and flow conditions are given in References 1 and 3 and the CORE OPERATING LIMITS REPORT.

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 rate 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 CORE OPERATING LIMITS REPORT.

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-A (the approved version at the time the reload analyses are performed shall be identified in the COLR).

3. "Fermi 2 Maximum Extended Operating Domain Analysis," NEDC-31843P, July 1990.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 69 TO FACILITY OPERATING LICENSE NO. NPF-43

DETROIT EDISON COMPANY

FERMI-2

DOCKET NO. 50-341

1.0 INTRODUCTION

By letter dated August 20, 1990, (Reference 1), the Detroit Edison Company (DECo or the licensee) requested amendment to the Technical Specifications (TSs) appended to Facility Operating License No. NPF-43 for Fermi-2. The proposed amendment would change Technical Specifications (TS) and associated Bases to provide for the implementation of operation in an expanded region of the power-flow map called the Maximum Extended Operating Domain (MEOD), which for this submittal includes operation in partial feedwater heating (PFH) conditions and a group of "Average Power Range Monitor (APRM)/Rod Block Monitor (RBM) Technical Specification" (ARTS) improvements. MEOD involves (1) operation above the 100 percent (control) rod line below 100 percent flow with operation permitted at 100 percent power down to 75 percent flow in the Maximum Extended Load Line Limit (MELLL) region, and (2) operation with increased flow (to 105 percent at full power) in the Increased Core Flow (ICF) region. PFH includes feedwater heaters out-of-service (FWHOS) before the end of cycle and final feedwater temperature reduction (FFWTR) beyond the end of the normal fuel cycle, both of which would be implemented only to a maximum of 50°F feedwater temperature reduction in this proposal. MELLL and ICF involve changes to the APRM rod block and thermal power scram trip setpoints. ARTS involves implementation of power and flow dependent fuel thermal limits to eliminate APRM trip setdown and supports power dependent RBM trips rather than the current flow dependent trips, (3) reconfiguration of Local Power Range Monitor (LPRM) inputs to the RBM and new trip logic, and (4) redefinition of RBM operability requirements.

In support of its request the licensee has submitted (1) a description and evaluation of the change, (2) the proposed new TS, (3) a sample Core Operating Limits Report (COLR) for the revised operation, and (4) General Electric topical report NEDC-31843P (Ref. 2) describing the changes, and presenting the analysis and evaluations used to justify the changes for Fermi 2, and operation with the revised analytical and physical systems and within the new limits resulting from the changes.

2.0 EVALUATION

These proposed changes for Fermi 2 are not unique or new for GE reactors. They have all become part of standard "Operating flexibility options" and are listed as such in the GE standard application for reactor fuel (GESTARII) NEDE-24011-P-A-9. Extended operating regions, increased flow and reduced feedwater temperature have been approved on many BWRs over the past decade, and several ARTS upgrades have been reviewed and approved, beginning with Hatch in 1984. The methodologies used for safety analyses involved in the justifications for the changes and determination of new operating limits have been previously reviewed and approved by the staff. The proposed new operating regions, and modes and limits of operation, and changes to the APRM and RBM systems are all bounded by changes previously approved for other reactors.

Since the changes proposed for Fermi 2 have become standard improvements or operating options which have been described and approved in past staff reviews, only a brief discussion of the changes and the GE analyses justifying the changes will be provided here. Descriptions and explanations of the changes, accompanying analyses and results and limits provided by the analysis are provided in NEDC-31843P, which in turn draws on results from NEDC-31515. The NEDC-31843P analysis were based on Fermi 2 Cycle 2 conditions, although the MEOD-ARTS changes will not be implemented until Cycle 3. Some of the analyses and results are generic and some are cycle specific and must be reexamined for each reload. Sample Core Operating Limits Report (COLR) TS values based on Cycle 2 were submitted with the proposal. The first implementation of the TS changes, however, is expected to be for Cycle 3, and reload specific analyses and COLR values will be prepared for that reload.

The MEOD-ARTS improvements, in addition to the power-flow operating region expansion previously indicated, consists primarily of the following changes.

(1) The APRM flow-biased rod block and thermal power scram trip setpoints are changed (increased) to allow operation in the expanded region. This change includes hardware changes for clamping the trips in the high flow region to provide appropriate setpoints in the region from 75 to 105 percent flow. These trip functions are the same as those approved for, other Boiling Water Reactors e.g., Grand Gulf 1.

(2) The APRM flow biased scram setpoint setdown requirement of Fermi 2 TS 3/4.2.2, which lowered the setpoint when local power peaking is such that the Maximum Fraction of Limiting Power Density is greater than the Fraction of Rated Thermal Power, is removed. This is replaced by power and flow dependent limits on Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Minimum Core Power Ratio (MCPR). Multipliers, K_p and $MAPFAC_p$, as a function of power and MCPR(F) limits and $MAPFAC_F$ multipliers as a function of flow to modify the basic operating limit MCPR (OLMCPR) and MAPLHGRs are developed from the analysis of the core-wide operating occurrences described in the Fermi Updated Final Safety Analysis Report (UFSAR). These become a part of the Fermi 2 TS (via the COLR). At any given power/flow state all four limits must be determined, with the most limiting governing the operation. GE loss of coolant accident (LOCA) analyses, including results from the

review of NEDC-31515, indicate that the combination of the elimination of the setdown requirement and the new MAPLHGR limit still provide for LOCA requirements for Fermi 2.

(3) The changes to provide the Fermi 2 ARTS RBM are identical to those approved in previous ARTS improvement reviews (e.g., Hatch and Monticello). The RBM is used to block rod motion to prevent exceeding MCPR limits during control rod withdrawal at power. For ARTS there is a change in the assignment of LPRMs to RBM channels. This change provides better and more uniform sensitivity to rod motion. ARTS has a more direct trip logic including a calibration to a fixed reference upon rod selection rather than calibration to the APRM, and an upscale trip level which is a step function of core power rather than a flow biased trip. A downscale trip is still used to detect abnormally low signals. The changes to the system require a reevaluation of the rod withdrawal event and new analyses to provide setpoints for the system.

The previous deterministic analysis is replaced with a statistical analysis using a large number of calculations of various operating states and giving results valid for all Fermi 2 cores using GE fuel through types GE9. Currently approved methods were used in the calculations. The calculations give, as a function of RBM trip setting, values of an initial MCPR which assure that 95 percent of withdrawal errors do not violate the MCPR safety limit (1.07) with a 95 percent confidence level. This can be used to select setpoints for the RBM, chosen with respect to the other ARTS APRM limits so that the withdrawal event is not limiting. Calculations have been done for Fermi 2 with both GEXL and GEXL-PLUS. The setpoints (for the Cycle 2 example) for power intervals of 27-62, 62-82, and 82-100 percent power are 114, 108.2, and 104.4 percent of the reference signal respectively. Calculations were done to examine the sensitivity of the results for core periphery rods (with fewer LPRM strings) and for LPRM failures (up to a 30 percent failure rate). The results indicated that the setpoints were suitable. An analysis of the effects of filters and time delays in the system was made. Use of filters requires a reduction of setpoints, and values for adjustments required, if used, is given in NEDC-31843P. The above setpoints used values provided in the Cycle 2 sample COLR.

The data base described above has been used to determine operating limit MCPRs such that no rod withdrawal error could lead to exceeding limits. Two MCPR values are defined, for below 90 percent power and for above 90 percent power, which are 1.70 and 1.40 respectively. When the operating MCPR is below these values the plant is on a "limiting control rod pattern" and the RBM system must be operable. When above these values bypass is allowed. For single loop operation with a MCPR safety limit of 1.08 the 1.70 value is increased to 1.71.

To justify operation of Fermi 2 with MELLL, ICF, ARTS and PFH and to determine values for the limits previously described, GE has evaluated the FSAR core-wide transients, considering the extremes of operation within this MEOD scope. The evaluation determined the events most significantly impacted by MEOD, Fermi 2 Cycle 2 parameters and approved GEMINI methods were used in

most of the analyses. Extended Cycle 2 parameters at ICF and FFWTR conditions were used for the primary impacted events turbine generator trip (TGT) and feedwater controller failure (FWCF), and Cycle 3 parameters for the third event, rod withdrawal error (RWE), in conjunction with the core simulator. The relevant analyses also considered combinations of "Moisture Separator Reheater" and "Turbine Bypass" inoperable conditions, and the sample COLR MCPR and MAPLHGR limits are a function of the applicable combination.

In addition to the transient analyses, GE examined the effect of MEOD-ARTS on overpressure protection, thermal-hydraulic stability, LOCA, containment response, feedwater nozzle and sparger fatigue, flow induced loads and vibration, and standard relevant events in the accident category. These evaluations considered the areas examined, and followed the methods used in previous staff approved reviews of similar operation extensions. It was determined that with the limits used and derived for the MEOD-ARTS improvement all required safety associated limits within the above areas are met. The feedwater nozzle and sparger review, as in previous similar reviews provided a GE recommendation for seal refurbishment. For thermal-hydraulic stability, Fermi 2 is operating under the "Interim Recommendations for Stability Actions" of NRC Bulletin No. 88-07, Supplement 1. These operations are applicable to the Fermi operations with MEOD-ARTS, as is the case for other operating reactors with similar systems.

These evaluations and transient analyses have examined the same areas examined in previous MEOD-ARTS related reviews by the staff. The method used in the evaluation have been previously approved by the staff and the results and conclusion of the evaluation fall within expected ranges. There are no significantly new regions of operation or parameter values or limits compared to those previously examined and approved. This review has concluded that the safety evaluation results presented in the GE report NEDC-31843P justify the proposed MEOD-ARTS improvements and operation within the power-flow and temperature boundaries and operating limits described in the report. As in previous reviews of similar ARTS programs for other reactors the instrumentation changes, analyses, methods used, criteria and setpoints proposed are acceptable.

3.0 TECHNICAL SPECIFICATIONS

There are changes to limits and operability requirements necessary for MEOD-ARTS. These include (1) deletion of the current setdown requirements, (2) new power and flow dependent MCPR and MAPLHGR limits, (3) changes to the APRM and RBM flow-biased scram, (4) new RBM limits and operability requirements.

There are a number of TS changes required to implement these changes. However, most of the TS related are proposed to be in the Fermi 2 Core Operating Limits Report. The values used for the RBM and the changes to the MCPR and MAPLHGR limits, i.e., the new multipliers and limits which are, or may be, cycle specific, are proposed to be in the COLR, which is designed for such parameters. The standard MCPR and MAPLHGR values already appear in the COLR. Directions for applying the multiplying factors and new limit curves are also provided in the COLR along with the corresponding limits. NEDC-31843, which describes the changes and methodology, becomes a reference in the COLR and the TS Bases.

The COLR presented with this submittal is only a sample. The values in it have been calculated for Fermi 2 Cycle 2. It is not intended that the MEOD-ARTS changes and the COLR changes be implemented until the beginning of Cycle 3. The reload period for Cycle 3 will be used to make the changes to the plant for ARTS. The COLR for Cycle 3 will be based on calculations done specifically for Cycle 3. However, the sample COLR provides a suitable example for this review.

The review has indicated that the parameters proposed for the COLR are likely to be cycle specific and are directly related to parameters already approved for the Fermi 2 COLR. They arise directly from TSs 3/4.2.1, .2.3, .2.4, and .3.6. It is appropriate that the instructions for using these parameters to provide the MCPR or MAPLHGR limits for a given reactor state point also be directly associated with the limits in the COLR. It is thus concluded that the placement of these values in the COLR is acceptable. This review has also indicated that the material, organization of the material and values (for the sample for Cycle 2) in the COLR are reasonable and provide an acceptable change to the COLR.

There are also several direct changes to the TS supporting the MEOD-ARTS changes.

(1) The Index is changed to delete TS 3/4.2.2. This TS on APRM setpoint setdown is deleted since, as previously discussed, this action is no longer needed with the ARTS improvement.

(2) Table 2.2.1-1 of TS 2.2 is changed. The flow-biased APRM trip setpoints and corresponding allowable values are changed (increased) to allow operation in the expanded MELLL power-flow operating region. This is changed for both two loop and single loop operation. This change is necessary for MEOD operation, and is similar to previously approved MEOD setpoints. It is acceptable.

(3) Bases 2.2.1 is changed to reflect the deletion of TS 3.2.2. It is acceptable.

(4) TS 3.1.4.3, Applicability, is changed to reflect the new power level/MCPR values for which the operability of the RBM is not required. This has been previously discussed along with the ARTS RBM changes and is acceptable.

(5) TS 3/4.2.2, APRM Setpoints, is deleted as no longer necessary (see above). This is acceptable.

(6) Table 4.3.1.1-1, note (d) of TS 3/4.3.1 is partially deleted. It referred to changes related to APRM setpoint setdown, and is therefore no longer needed. The change is acceptable.

(7) TS 3.3.6, footnote, has reference to flow-biased RBM deleted since the RBM is no longer flow-biased. The change is acceptable.

(8) Table 3.3.6-1 of TS 3.3.6 has a footnote changed similar to that discussed above for TS 3.1.4.3. The change is acceptable.

(9) Table 3.3.6-2 of TS 3.3.6, has the RBM and APRM rod block trip setpoints changed similarly to that for the APRM scrams in TS 2.2.1 discussed above. The rod block changes are acceptable, as they were for the scram changes. The footnote referring to TS 3.2.2 is deleted since TS 3.2.2 has been deleted.

(10) Table 3.3.6-2, has the recirculation flow upscale rod block trip setpoints changed to reflect the expanded operating domain. It is acceptable.

(11) Table 4.3.6-1, TS 3/4.3.6, has a footnote changed to be consistent with TS 3.1.4.3, discussed above. The change is acceptable.

(12) TS 3/4.4.1, Action 1.f and footnote are changed to reflect elimination of RBM flow bias and deletion of TS 3.2.2. The changes are acceptable.

(13) TS 4.4.1.1.2 has the recirculation pump MG set scoop tube mechanical and electrical stop overspeed setpoints increased to provide for the increased operating domain. The changes are acceptable.

(14) Bases 3/4.2.1. A discussion of the MAPLHGR multipliers which have been added to the COLR is added to the Bases. It is acceptable.

(15) Bases 3/4.2.2. The Bases are deleted since the specification has been deleted. This is acceptable.

(16) Bases 3/4.2.3. The discussion of the k_f factor is deleted since this factor has been removed from the TS. Reference is made to the COLR and to NEDC-31843. This is acceptable.

4.0 CONCLUSIONS

The Detroit Edison Company has proposed MELLL, ICF, and PFH changes to the allowed operating region for Fermi 2 and operating and physical changes to change to ARTS operation, along with TS and COLR changes to implement these changes. We have reviewed the information, including the GE reports, submitted with these proposed changes. Based on this review we conclude that appropriate material has been submitted to justify the changes, that the changes fall within the scope and bounds of past staff reviews in these areas, and that the changes to and values proposed for the Fermi TS and sample COLR are acceptable. The use of the GE report NEDC-31843P in the TS Bases and in the COLR as a reference to the MEOD-ARTS changes, methodology and analyses approved here is also acceptable.

5.0 REFERENCES

1. Letter and enclosures from W. Orser, Detroit Edison Company, to USNRC, dated August 20, 1990, "Proposed Technical Specification Changes - Maximum Expanded Operating Domain."

2. NEDC-31843P, "Fermi-2 Maximum Extended Operating Domain Analysis," July 1990.

3. NEDC-31515, Rev. 1, "Maximum Extended Load Line Limit and Feedwater Heater Out-of-Service Analysis for Enrico Fermi Atomic Power Plant Unit 2," August 1989.

6.0 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.

7.0 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 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 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 Section 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 this amendment.

8.0 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public. The staff therefore concludes that the proposed changes are acceptable.

Principal Contributor: H. Richings

Date: May 15, 1991