July 27, 1990

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. 53 TO FACILITY OPERATING LICENSE NO. NPF-43: (TAC NO. 69074)

The Commission has issued the enclosed Amendment No. 53 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 August 4, 1988 as supplemented by letter dated August 18, 1989.

The amendment revises the Technical Specifications to allow extended operation of Fermi-2 at reduced power with a single recirculation loop in operation.

A copy of the Safety Evaluation supporting this amendment is also enclosed. 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 & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 53 to NPF-43
2. Safety Evaluation

cc w/enclosures:
See next page

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

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

Fermi-2 Facility

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

DETROIT EDISON COMPANY

DOCKET NO. 50-341

FERMI-2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 53
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 4, 1988 as supplemented August 18, 1989, 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. 53, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. DECo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert C. Pierson

Robert C. Pierson, Acting Director
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 27, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 53

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.

REMOVE	INSERT
v*	v*
vi	vi
xiii	xiii
xxi	xxi
2-1	2-1
2-3	2-3
2-4	2-4
-	2-4a
B 2-7	B 2-7
-	B 2-7a
B 2-8*	B 2-8*
3/4 2-1	3/4 2-1
3/4 2-5	3/4 2-5
-	3/4 2-5a
3/4 3-41	3/4 3-41
3/4 3-42*	3/4 3-42*
3/4 3-44	3/4 3-44
-	3/4 3-44a
3/4 4-1	3/4 4-1
3/4 4-2	3/4 4-2
3/4 4-3	3/4 4-3
3/4 4-4	3/4 4-4
3/4 4-5	3/4 4-5
3/4 4-6	3/4 4-6
-	3/4 4-6a
-	3/4 4-30
-	3/4 4-31
3/4 10-3*	3/4 10-3*
3/4 10-4	3/4 10-4
B 3/4 2-1	B 3/4 2-1
-	B 3/4 2-1a
B 3/4 2-3	B 3/4 2-3
B 3/4 4-1	B 3/4 4-1
-	B 3/4 4-1a
B 3/4 4-2*	B 3/4 4-2*
-	B 3/4 4-8
-	B 3/4 4-9

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

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION | PAGE

INSTRUMENTATION (Continued)

MONITORING INSTRUMENTATION (Continued)

LIST OF FIGURES

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than the Safety Limit MCPR of 1.07 for two recirculation loop operation and shall not be less than the Safety Limit MCPR of 1.08 for single loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than the Safety Limit MCPR of 1.07 for two recirculation loop operation or less than the Safety Limit MCPR of 1.08 for single loop operation and with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint* less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

*The APRM flow biased instrumentation need not be declared inoperable upon entering single recirculation loop operation provided the setpoints are adjusted within 4 hours per Specification 3.4.1.1.

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	≤ 0.58 W+59%, with a maximum of	≤ 0.58 W+62%, with a maximum of
b. High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
2) During single recirculation loop operation:		
a. Flow Biased	≤0.58W+54.4%,**	≤0.58W+57.4%,**
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.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
5. Main Steam Line Isolation Valve - Closure	$\leq$ 8% closed	$\leq$ 12% closed
6. Main Steam Line Radiation - High	$\leq$ 3.0 x full power background	$\leq$ 3.6 x full power background
7. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Float Switch	$\leq$ 594'8"	$\leq$ 596'0"
b. Level Transmitter	$\leq$ 592'6"	$\leq$ 596'0"
9. Turbine Stop Valve - Closure	$\leq$ 5% closed	$\leq$ 7% closed
10. Turbine Control Valve Fast Closure	Initiation of fast closure	N.A.
11. Reactor Mode Switch Shutdown Position	N.A.	N.A.
12. Manual Scram	N.A.	N.A.
13. Deleted		

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. The flow referenced trip setpoint must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when MFLPD is greater than or equal to FRTP. For single recirculation loop operation, the reduced APRM setpoints are based on a ΔW value of 8%. The ΔW 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

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

3. Reactor Vessel Steam Dome Pressure-High (continued)

trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine stop valve closure trip is bypassed. For a turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

LIMITING SAFETY SYSTEM SETTINGS

BASES

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Low Water Level-Level 3

The reactor vessel water level trip setpoint was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

5. Main Steam Line Isolation Valve-Closure

The main steam line isolation valve closure trip was provided to limit the amount of fission product release for certain postulated events. The MSIV's are closed automatically from measured parameters such as high steam flow, high steam line radiation, low reactor water level, high steam tunnel temperature, and low steam line pressure. The MSIV's closure scram anticipates the pressure and flux transients which could follow MSIV closure and thereby protects reactor vessel pressure and fuel thermal/hydraulic Safety Limits.

6. Main Steam Line Radiation-High

The main steam line radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected, a trip is initiated to reduce the continued failure of fuel cladding. At the same time the main steam line isolation valves are closed to limit the release of fission products. The trip setting is high enough above background radiation levels to prevent spurious trips yet low enough to promptly detect gross failures in the fuel cladding. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

7. Drywell Pressure-High

High pressure in the drywell could indicate a break in the primary pressure boundary systems or a complete loss of drywell cooling. The reactor is tripped in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant. The trip setting was selected as low as possible without causing spurious trips.

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 Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, and 3.2.1-4 for the applicable bundle type.

The above limits shall be multiplied by a factor of 0.90 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.

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased neutron flux-high scram trip setpoint (S) and flow biased neutron flux-high control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

	TRIP SETPOINT	ALLOWABLE VALUE
1. During two recirculation loop operation:		
	$S \leq (0.58W + 59\%)T$	$S \leq (0.58W + 62\%)T$
	$S_{RB} \leq (0.58W + 50\%)T$	$S_{RB} \leq (0.58W + 53\%)T$
2. During single recirculation loop operation:		
	$S \leq (0.58W + 54.4\%)T$	$S < (0.58W + 57.4\%)T$
	$S_{RB} \leq (0.58W + 45.4\%)T$	$S_{RB} \leq (0.58W + 48.4\%)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER,
W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 100 million lbs/hr, at 100% of RATED THERMAL POWER
T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY. T is applied only if less than or equal to 1.0

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

ACTION:

With the APRM flow biased neutron flux-high scram trip setpoint and/or the flow biased neutron flux-high control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint value*# within 6 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

*With MFLPD greater than the FRTP during power ascension up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times MFLPD, 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. With MFLPD greater than FRTP and a single recirculation loop in operation, if the APRM flow biased setpoints have not been adjusted to their single loop values, then the minimum required APRM reading must be increased by an additional 5.3% of rated power.

#During single recirculation loop operation with FRTP greater than or equal to MFLPD, rather than adjusting the APRM setpoints to comply with the single loop values, the APRM gain 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% RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP and the MFLPD for each class of fuel shall be determined, the value of T calculated, and the most recent actual APRM flow biased neutron flux-high scram and flow biased neutron flux-high control rod block trip setpoints verified to be within the above limits or adjusted, or the APRM gain readings shall be verified as indicated below*#, as required:

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 MFLPD greater than or equal to FRTP.

d. The provisions of Specification 4.0.4 are not applicable.

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*With MFLPD greater than the FRTP during power ascension up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times MFLPD, 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. With MFLPD greater than FRTP and a single recirculation loop in operation, if the APRM flow biased setpoints have not been adjusted to their single loop values, then the minimum required APRM reading must be increased by an additional 5.3% of rated power.

#During single recirculation loop operation with FRTP greater than or equal to MFLPD, rather than adjusting the APRM setpoints to comply with the single loop values, the APRM gain 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% RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

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 and Rod Block Monitor 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

CONTROL ROD BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD BLOCK MONITOR[a]			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. APRM			
a. Flow Biased Neutron Flux - High	4	1	61
b. Inoperative	4	1, 2, 5	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Setdown	4	2, 5	61
3. SOURCE RANGE MONITORS			
a. Detector not full in[b]	3 2[f]	2 5	61 61
b. Upscale[c]	3 2[f]	2 5	61 61
c. Inoperative[c]	3 2[f]	2 5	61 61
d. Downscale[d]	3 2[f]	2 5	61 61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale[e]	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5**	62
b. Scram Trip Bypass	2	2, 5**	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. Comparator	2	1	62
7. REACTOR MODE SWITCH SHUTDOWN POSITION	2	3, 4	63

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale		
1) During two recirculation loop operation	≤ 0.66 W + 40%	≤ 0.66 W + 43%
2) During single recirculation loop operation	≤ 0.66 W + 34.7%#	≤ 0.66 W + 37.7%#
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - High		
1) During two recirculation loop operation	≤ 0.58 W + 50%*	≤ 0.58 W + 53%*
2) During single recirculation loop operation	≤ 0.58 W + 45.4%#*	≤ 0.58 W + 48.4%#*
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Setdown	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 1.0×10^5 cps	≤ 1.6×10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps**	≥ 2 cps**

*The APRM rod block function is varied as a function of recirculation loop drive flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

**May be reduced to ≥ 0.7 cps provided the signal-to-noise ratio ≥ 20.

#During single recirculation loop operation, rather than adjusting the APRM and RBM 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	$\leq$ 108/125 divisions of full scale	$\leq$ 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ 5/125 divisions of full scale	$\geq$ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	$\leq$ 589'11½"	$\leq$ 591'0"
b. Scram Trip Bypass	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	$\leq$ 108/125% of rated flow	$\leq$ 111/125% of rated flow
b. Inoperative	NA	NA
c. Comparator	$\leq$ 10% flow deviation	$\leq$ 11% flow deviation
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	NA

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 to a value of 0.90 times the two recirculation loop operation 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 Text 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.

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 105% and 102.5%, 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.

THERMAL POWER VERSUS CORE FLOW

FIGURE 3.4.1.1-1 - DELETED

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.2 With THERMAL POWER greater than 25% of RATED THERMAL POWER, each of the above required jet pumps shall be demonstrated OPERABLE at least once per 24 hours* by determining operating recirculation loop flow(s), total core flow, and diffuser-to-lower plenum differential pressure for each operating jet pump and verifying that no two of the following conditions occur:

a. The indicated operating recirculation loop flow(s) differs by more than 10% from the established pump speed-loop flow characteristics.

b. The indicated total core flow differs by more than 10% from the established total core flow value derived from recirculation loop flow measurements.

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump in an operating loop differs from the mean of all jet pump differential pressures in the same loop by more than 20% deviation from its normal deviation.

*The provisions of Specification 4.0.4 are not applicable provided that this surveillance is performed within 12 hours after exceeding 25% of RATED THERMAL POWER.

REACTOR COOLANT SYSTEM

RECIRCULATION PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation pump speed shall be maintained within:

a. 5% of each other with core flow greater than or equal to 70% of rated core flow.

b. 10% of each other with core flow less than 70% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*, during two recirculation loop operation.

ACTION:

With the recirculation pump speeds different by more than the specified limits, either:

a. Restore the recirculation pump speeds to within the specified limit within 2 hours, or

b. Shutdown one of the recirculation loops and take the ACTION required by Specification 3.4.1.1.

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

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation pump speed shall be verified to be within the limits at least once per 24 hours.

*See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.4 An idle recirculation loop shall not be started unless THERMAL POWER is within the unrestricted zone of Figure 3.4.1.4-1 and the temperature differential between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is less than or equal to 145°F, and:

a. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is less than or equal to 50°F, or

b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is less than or equal to 50°F and the operating loop flow rate is less than or equal to 50% of rated loop flow.

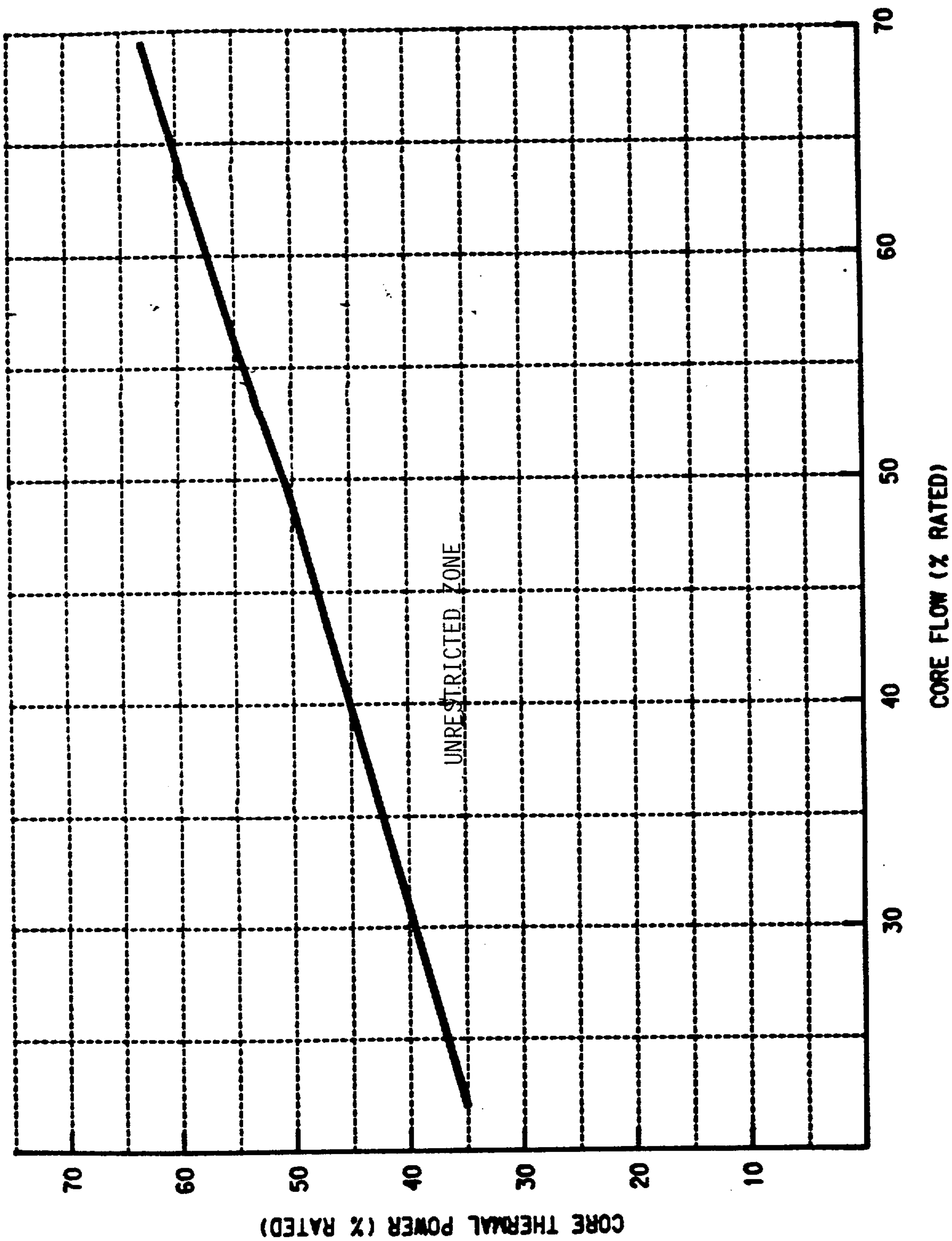
APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With THERMAL POWER, temperature differences and/or flow rates exceeding the above limits, suspend startup of any idle recirculation loop.

SURVEILLANCE REQUIREMENTS

4.4.1.4 The THERMAL POWER, temperature differentials and flow rate shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.



THERMAL POWER VERSUS CORE FLOW

FIGURE 3.4.1.4-1

REACTOR COOLANT SYSTEM

3/4.4.10 CORE THERMAL HYDRAULIC STABILITY

LIMITING CONDITION FOR OPERATION

3.4.10 The Reactor core shall not be operated in Region A or Region B of Figure 3.4.10-1.

APPLICABILITY: OPERATIONAL CONDITION 1

ACTION:

a. With the Reactor operating in Region A of Figure 3.4.10-1, immediately place the Reactor Mode Switch in the SHUTDOWN position.

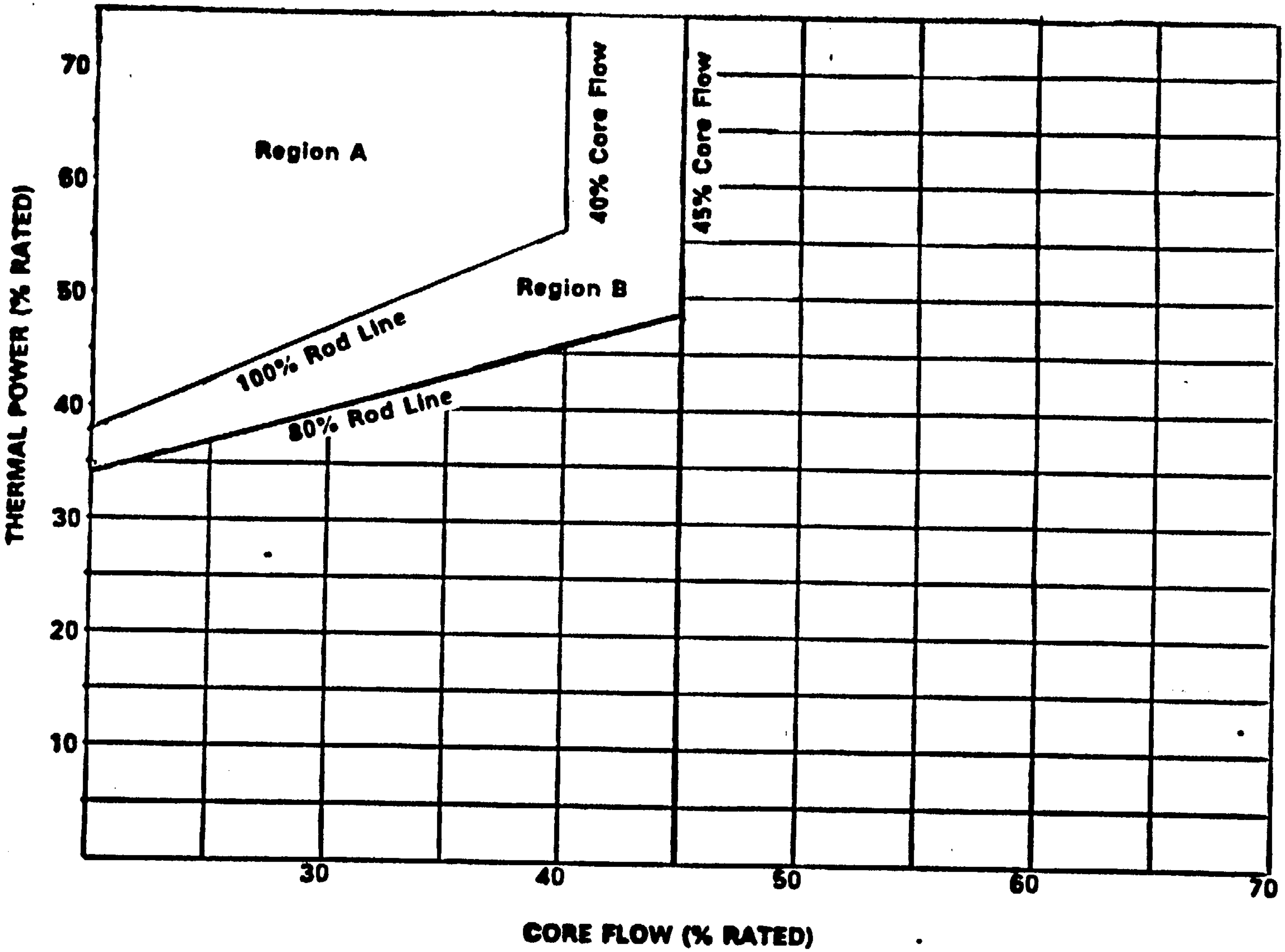
b. With the Reactor operating in Region B of Figure 3.4.10-1, immediately initiate action to exit Region B by inserting control rods.

c. If, while exiting Region B, core thermal hydraulic instability occurs as evidence by APRM readings oscillating by greater than or equal to 10% of RATED THERMAL POWER peak to peak or LPRM readings oscillating greater than or equal to 30 watts/cm^2 peak to peak, immediately place the Reactor Mode Switch in the SHUTDOWN position.

SURVEILLANCE REQUIREMENTS

4.4.10.1 The provisions of Specification 4.0.4 are not applicable.

4.4.10.2 With THERMAL POWER greater than 30% of RATED THERMAL POWER and Core Flow less than 50% of Rated Core Flow, verify that the reactor core is not operating in Region A or Region B of Figure 3.4.10-1 at least once every 4 hours.



THERMAL POWER VERSUS CORE FLOW

FIGURE 3.4.10 -1

SPECIAL TEST EXCEPTIONS

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

3.10.3 The provisions of Specification 3.9.1, Specification 3.9.3, and Table 1.2 may be suspended to permit the reactor mode switch to be in the Startup position and to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

a. The source range monitors are OPERABLE per Specification 3.9.2 with the RPS circuitry "shorting links" removed.

b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1, or conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.

c. The "rod-out-notch-override" control shall not be used during out-of-sequence movement of the control rods.

d. No other CORE ALTERATIONS are in progress.

APPLICABILITY: OPERATIONAL CONDITION 5, during shutdown margin demonstrations.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown or Refuel position.

SURVEILLANCE REQUIREMENTS

4.10.3 Within 30 minutes prior to and at least once per 12 hours during the performance of a shutdown margin demonstration, verify that;

a. The source range monitors are OPERABLE per Specification 3.9.2,

b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1 or a second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown demonstration procedures, and

c. No other CORE ALTERATIONS are in progress.

SPECIAL TEST EXCEPTIONS

3/4.10.4 RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.4.1.1 and 3.4.1.3 that recirculation loops be in operation with matched pump speed may be suspended for up to 24 hours for the performance of PHYSICS TESTS, provided that THERMAL POWER does not exceed 5% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITION 2, during PHYSICS TESTS.

ACTION:

a. With the above specified time limit exceeded, insert all control rods.

b. With the above specified THERMAL POWER limit exceeded during PHYSICS TESTS, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The time during which the above specified requirement has been suspended shall be verified to be less than 24 hours at least once per hour during PHYSICS TESTS.

4.10.4.2 THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

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 shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3 and 3.2.1-4.

The Technical Specification 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.

Only the most limiting MAPLHGR values are shown in the Technical Specification figures for multiple lattice fuel. When hand calculations are required, these Technical Specifications MAPLHGR figure 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 0.90. The constant factor of 0.90 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.

Reference

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (latest approved revision).

BASES TABLE B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters:

Core THERMAL POWER................ 3430 MWt* which corresponds to 105% of rated steam flow

Vessel Steam Output............... 14.86×10^6 lbm/hr which corresponds to 105% of rated steam flow

Vessel Steam Dome Pressure........ 1055 psia

Design Basis Recirculation Line Break Area for:

a. Large Breaks $4.1\ ft^2$

b. Small Breaks $0.1\ ft^2$

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kW/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	8 x 8	13.4	1.4	1.18**
First Reload	8 x 8	14.4	1.4	1.18**

A more detailed listing of input of each model and its source is presented in Section II of Reference 1 and subsection 6.3 of the UFSAR.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

**For single recirculation loop operation, loss of nucleate boiling is assumed at 0.1 second after LOCA regardless of initial MCPR.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

The impact of single recirculation loop operation upon plant safety is assessed and shows that single-loop operation is permitted at power level is up to 70% of RATED THERMAL POWER if the MCPR fuel cladding safety limit is increased as noted by Specification 2.1.2. APRM scram and control rod block setpoints (or APRM gains) are adjusted as noted in Tables 2.2.1-1 and 3.3.6-2, respectively. MAPLHGR limits are decreased by the factor given in Specification 3.2.1. A time period of 4 hours is allowed to make these adjustments following the establishment of single loop operation since the need for single loop operation often cannot be anticipated. MCPR operating limits adjustments in Specification 3.2.3 for different plant operating situations are applicable to both single and two recirculation loop operation.

To prevent potential control system oscillations from occurring in the recirculation flow control system, the operating mode of the recirculation flow control system must be restricted to the manual control mode for single-loop operation.

Additionally, surveillance on the pump speed of operating recirculation loop is imposed to exclude the possibility of excessive core internals vibration. The surveillance on differential temperatures below 30% THERMAL POWER or 50% rated recirculation loop flow is to prevent undue thermal stress on vessel nozzles, recirculation pump and vessel bottom head during a power or flow increase following extended operation in the single recirculation loop mode.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation.

Recirculation pump speed mismatch limits are in compliance with the ECCS LOCA analysis design criteria for two recirculation loop operation. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA.

In the case where the mismatch limits cannot be maintained during two loop operation, continued operation is permitted in a single recirculation loop mode.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM (Continued)

Sudden equalization of a temperature difference greater than 145°F between the reactor vessel bottom head coolant and the coolant in the upper region of the reactor vessel by increasing core flow rate would cause undue stress in the reactor vessel bottom head.

Requirements are imposed to prohibit idle loop startup above the 80% rod line to minimize the potential for initiating core thermal-hydraulic instability.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. A total of 11 OPERABLE safety/relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient.

Demonstration of the safety/relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

The low-low set system ensures that a potentially high thrust load (designated as load case C.3.3) on the SRV discharge lines is eliminated during subsequent actuations. This is achieved by automatically lowering the closing setpoint of two valves and lowering the opening setpoint of two valves following the initial opening. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action. Service sensitive reactor coolant system Type 304 and 316 austenitic stainless steel piping; i.e., those that are subject to high stress or that contain relatively stagnant, intermittent, or low flow fluids, requires additional surveillance and leakage limits.

The purpose of the RCS interface valves leakage pressure monitors (LPMs) is to provide assurance of the integrity of the Reactor Coolant System pressure isolation valves which form a high/low pressure boundary. The LPM is designed to alarm on increasing pressure on the low pressure side of the high/low pressurc interface to provide indication to the operator of abnormal interface valve leakage.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

3/4.4.10 CORE THERMAL HYDRAULIC STABILITY

BWR cores typically operate with the presence of global flux noise in a stable mode which is due to random boiling and flow noise. As the power/flow conditions are changed, along with other system parameters (pressure, subcooling, power distribution, etc.) the thermal hydraulic/reactor kinetic feedback mechanism can be enhanced such that random perturbations may result in sustained limit cycle or divergent oscillations in power and flow.

Two major modes of oscillations have been observed in BWRs. The first mode is the fundamental or core-wide oscillation mode in which the entire core oscillates in phase in a given axial plane. The second mode involves regional oscillation in which one half of the core oscillates 180 degrees out of phase with the other half. Studies have indicated that adequate margin to the Safety Limit Minimum Critical Power Ratio (SLMCPR) may not exist during regional oscillations.

Region A and B of Figure 3.4.10-1 represent the least stable conditions of the plant (high power/low flow). Region A and B are usually entered as the result of a plant transient (for example, recirculation pump trips) and therefore are generally not considered part of the normal operating domain. Since all stability events (including test experience) have occurred in either Region A or B, these regions are avoided to minimize the possibility of encountering oscillations and potentially challenging the SLMCPR. Therefore, intentional operation in Regions A or B is not allowed. It is recognized that during certain abnormal conditions within the plant, it may become necessary to enter Region A or B for the purpose of protecting equipment which, were it to fail, could impact plant safety or for purpose of protecting a safety or fuel operating limit. In these cases, the appropriate actions for the region entered would be performed as required.

Most oscillations that have occurred during testing and operation have occurred at or above the 100% rod line with core flow near natural circulation. This behavior is consistent with analysis which predict reduced stability margin with increasing power or decreasing flow. As core flow is increased or power decreased, the probability of oscillations occurring will decrease. Region A of Figure 3.4.10-1 bounds the majority of the stability events and tests observed in GE BWRs. Since Region A represents the least stable region of the power/flow operating domain, the potential to rapidly encounter large magnitude core thermal hydraulic oscillations is increased. During transients, the operator may not have sufficient time to manually insert control rods to mitigate the oscillations before they reach an unacceptable magnitude. Therefore, the prompt action of manually scramming the plant when Region A is entered is required to ensure protection of the SLMCPR.

3/4.4.10 CORE THERMAL HYDRAULIC STABILITY (Continued)

Based on test and operating experience, the frequency of core thermal hydraulic oscillations is less in Region B than in Region A. Decay ratios are expected and predicted to be lower in this region since Region B covers a lower power and higher flow range than Region A. Also, the margin to the SLMCPR will typically be larger in Region B than in Region A. With more margin to SLMCPR and a lower probability of oscillations, exiting Region B by control rod insertion is justified. However, if oscillations are observed while exiting Region B, the reactor will be manually scrammed.

The potential for core thermal hydraulic oscillations to occur outside of Regions A and B is very small and therefore special requirements are not necessary outside of these regions.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 53 TO FACILITY OPERATING LICENSE NO. NPF-43

DETROIT EDISON COMPANY

FERMI-2

DOCKET NO. 50-341

1.0 INTRODUCTION

By letter dated August 4, 1988 as supplemented August 18, 1989, 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 change the Technical Specifications to allow extended operation of Fermi-2 at reduced power with a single recirculation loop in operation (SLO).

2.0 EVALUATION

2.1 Single Loop Operation

Current Fermi-2 TS require shutdown of the reactor within 12 hours when only one recirculation loop is in operation. Proposals for TS changes to allow extended operating time under SLO conditions have been accepted in recent years for a large number of BWRs with similar restrictions. Thermal hydraulic stability (THS) concerns have been largely resolved for SLO by the introduction of TS requiring avoidance of potentially unstable regions of the power flow map and surveillance in neighboring regions. More recently, interim operating procedures requirements described in NRC Bulletin No. 88-07 and Supplement 1 to that bulletin (Ref. 4) have been issued to address THS. NRC Generic Letters 86-02 and 86-09 (Refs. 6 and 7) present staff positions in the areas of SLO and related THS surveillance requirements.

In addition to the THS changes, it is necessary to reexamine the analyses of abnormal operational transient and accident events under SLO conditions and provide for required changes, including TS changes, of trip set points and operating and safety limits resulting from the changed reactor conditions. The necessary analyses are provided in the GE report (Ref. 3), and DECo has proposed the necessary changes to the TS. These analyses and changes are similar to those approved in previous reviews of SLO operations.

GE has provided (Ref. 3) the results of the reexamination, and where required reanalysis, of transients and accidents relevant to SLO. The events examined are the same as those considered and approved in previous staff reviews of SLO. These include the abnormal operational transients involving flow increase, flow decrease, cold water injection, pressurization and rod withdrawal events. Events requiring analysis have been analyzed with standard, staff approved methodology as described in GESTAR II (Ref. 5). For SLO, these events begin at a maximum power level about 30 percent less than that for two loop operation (TLO). Thus maximum transient conditions are for the most part less severe than those analyzed for TLO. Average Power Range Monitor (APRM) and Rod Block Monitor (RBM) trip set point equations require adjustment for these analysis and for operations under SLO conditions to account for the changes in actual core flow versus measured flow as a result of backflow in idle jet pumps. These changes in trip equations in turn require TS changes for SLO.

As is normally the case, the reexamination of these transient events by GE resulted in change to safety limits. The operating limit minimum critical power ratio (MCPR) remains unchanged as determined by TLO. These results are expected and acceptable.

The safety limit MCPR does change, however. Two of the uncertainties involved in the determination of this safety limit are increased by SLO. These are (1) the random noise for the neutron flux detector readings (called the TIP readings) used for power determination, and (2) core flow measurement uncertainty. The analyses for these increased uncertainties are the same as has been presented and accepted in previous SLO reviews. The result is an increase of 0.01 in the safety limit MCPR. This is a reasonable and acceptable change.

The only accident event, other than LOCA, relevant to SLO is the recirculation pump seizure accident. It was not specifically reanalyzed for Fermi-2. However, previous reviews for a range of reactors (including a large BWR4 similar to Fermi-2) requesting SLO have provided analyses, approved by the staff, that have shown that the event results in a MCPR significantly above the SLO safety limit. It is concluded that this result is also applicable to Fermi-2.

SLO affects LOCA calculations primarily by decreasing core flow more rapidly than for TLO and thus decreasing the time to departure from nucleate boiling. To examine this and other effects of SLO, LOCA analyses were performed using standard, approved methodology and covering a spectrum of large break sizes. These analyses result in a required reduction factor of 0.90 for Fermi-2 current fuel assembly maximum average planar linear heat generation rate (MAPLHGR) limits. With this limit reduction factor, the large and small break LOCA results remain within required limits of 10 CFR 50.46. This type of analysis and the reduction factor (of similar magnitude) have been reviewed and approved in previous SLO reviews and is acceptable for Fermi-2.

In addition to relevant transients and accidents, GE has examined several other areas possibly associated with SLO. These include containment analysis, ATWS, fuel mechanical performance, and pressure vessel internal vibrations.

As in previous SLO reviews, because of the more limited power flow region of SLO, the reactor conditions associated with these areas for Fermi-2 SLO generally fall within previously analyzed TLO bounds, with results within required limits. However, to assure conservatism, and in keeping with previous SLO submittals and approvals, there are proposed TS requiring (1) limitations on power (70 percent of rated) and flow (one pump speed 75 percent of rated), (2) additional surveillance on recirculation loop differential temperature to prevent stratification and associated stresses, and (3) manual flow control to prevent possible control oscillations. These are measures approved in previous SLO reviews and are acceptable for Fermi-2.

2.2 Thermal Hydraulic Stability (THS)

The proposed Technical Specification changes are intended to avoid problems with thermal hydraulic instability, which have been a focus of NRC attention following the LaSalle instability event of March 1988. This attention has resulted in the issuance of NRC Bulletin 88-07 and Supplement 1 to that bulletin (Ref. 4). These provide NRC action requests for utilities to provide operator training, instrumentation verification and operating procedures intended to minimize instability potential or consequences. The requested operating procedures of Supplement 1 are based primarily on the General Electric (GE) "Interim Recommendations for Stability Actions" (IRSA). The IRSA are presented in an attachment to the supplement. Supplement 1 also requests (1) that plants without effective automatic scram protection for regional oscillations (IRSA group 2 plants) should initiate manual scram upon loss of both recirculation pumps, and (2) that the region boundaries of IRSA be reevaluated and justified for core loading with fuel other than that supplied by GE.

The IRSA, along with the other NRC staff requests presented in the supplement, constitute current NRC recommendations for BWR THS related operations. They are the result of calculations and reviews by the NRC, GE, the BWR Owners' Group and associated consultants. The bulletin supplement requested that licensees implement the IRSA (and other associated requests) by modifying relevant procedures. Modification of TS was not specifically requested since it is expected that long term solution implementation, to replace the interim recommendations, will begin soon. However, several licensees have modified their stability TS to correspond to the bulletin requests.

Fermi-2 currently has THS restrictions procedures and relevant power-flow map regions, but they differ in many details from those requested in Supplement 1. Furthermore, Fermi-2 is a IRSA group 2 plant (having a filtered APRM flow biased neutron flux signal to provide a simulated thermal power monitor), and has current and future core fuel loadings consisting of GE P8x8R and GE 8x8B fuel assemblies. This requires changes in Fermi-2 operations and power flow map region boundaries to comply with the Supplement 1 requests. DECo has therefore submitted proposed TS changes and justifications to provide specifications which comply with the NRC requested changes.

The IRSA specify three regions (A, B, C) on the power-flow map involving different degrees of allowed or prohibited operation. These are bounded by constant flow lines or control rod lines (lines of flow variation with other reactor parameters, particularly control rod position, held constant). Region A is above the 100 percent rod line (intercepts 100 percent rated power at 100 percent rated flow) and below 40 percent flow. Region B is between the 80 and 100 percent rod lines and below 40 percent flow. Region C is above the 80 percent rod line and between 40 and 45 percent flow. Deliberate entry into regions A and B is not permitted, and if it occurs, immediate exit is required. For a group 2 plant (such as Fermi-2) immediate scram is required in region A, while the region B control rod insertion or flow increase may be used to exit. Operations may be conducted in region C, with suitable surveillance, if required during "start-ups" to prevent fuel damage. If during operations in B or C instability occurs, the reactor shall be scrammed, with evidence for instability coming from Average Power Range Monitor (APRM) oscillation greater than 10 percent or Local Power Range Monitor (LPRM) upscale or downscale alarms.

The new stability requirements are independent of the number of recirculation loops operating. The new Fermi-2 regions are displayed in the Figure 3.4.10-1. Immediate scram is required if region A is entered. The IRSA regions B and C are combined into a single Region B. The IRSA operating restrictions of region B were conservatively applied throughout Fermi-2 Region B. In Region B, control rods insertion is required immediately. The proposed regions provide a conservative representation of the Supplement 1 request and are acceptable.

The overall conclusion of the review is that the proposed TS changes and the material submitted to support the changes are acceptable. It should be noted, however, that the NRC staff, its consultants, the BWR Owners' Group (BWROG), GE, and others are continuing the review of THS concerns. The BWROG is developing several long term solutions for the problem. Any new requirements resulting from the continuing generic review of THS concerns and BWROG long term solutions will be applicable to Fermi-2 and may impact some of the operations, systems surveillance, or TS found to be acceptable in this review.

3.0 TECHNICAL SPECIFICATIONS

DECo has proposed that the following TS be changed to provide for SLO and THS requirements. For the most part the reasons for these changes have already been discussed and staff approval indicated.

Specification 2.1.2. - The safety limit MCPR is changed to 1.08 for SLO. It remains at 1.07 for TLO. This increase of 0.01 because of increased power and flow noise and uncertainty, as previously discussed, is acceptable.

Specification 2.2.1 - Added footnote to specify time allowance of 4 hours to comply with SLO requirements. The staff finds four hours is acceptable for not declaring the APRM flow biased instrumentation inoperable.

Table 2.2.1-1 - The APRM trip set point change for SLO is added. This change to account for the difference in measured and actual core flow, as previously discussed, is acceptable. No high flow clamp is required since flow levels are not attainable during SLO.

Specification 3.2.1 - The reduction of 0.90 for the SLO MAPLHGRs is provided. As discussed above in Section 2.1 the reduction of 0.90 for the SLO MAPLHGRS keeps the large and small break LOCA analysis within the limits of 10 CFR 50.46. Therefore the staff finds the change acceptable.

Specification 3.2.2. - This also changes the APRM trip set point equation and control rod block set points to account for lower core flow conditions under SLO conditions. The staff finds this acceptable.

<u>Table 3.3.6-2.</u> - Along with the scram trip set points, the rod block type set points are added to the TS for both the APRM and RBM to account for SLO trips. The staff finds this acceptable.

<u>Specification 3.4.1.1.</u> - This specification receives a number of additions and changes to account for requirements for SLO. These include, in the Action section, requirements for flow control in Local Manual mode, power level not above 70 percent, MAPLHGR limit reduced by 0.90, pump speed not above 75 percent, differential temperature surveillance for power not above 30 percent or for recirculation flow above 50 percent, and reduction of APRM and RBM trip set points. These action items have been previously discussed and are acceptable. The associated action times are similar to those previously reviewed and approved for SLO and are reasonable and appropriate for Fermi-2. The required surveillance and frequencies are reasonable and generally in accord with previous reviews and are acceptable.

<u>Specification 3.4.1.4.</u> - The proposed change includes an additional restriction for idle loop start-up which is based upon thermal hydraulic instability concerns. During an idle loop start-up, there is a potential for thermal hydraulic instability if thermal power is not restricted to below the 80% rod line shown in the proposed Figure 3.4.1.4-1. Accordingly, the LCO, action and surveillance requirements have been modified to prohibit idle loop start-up in the region of concern. The proposed change is acceptable.

<u>New Specification. 3/4.4.10 Core Thermal Hydraulic Stability</u>

The proposed TS adds a new core flow map defining operability boundaries. The proposed TS complies with the staff approved interim stability corrective actions for BWRs using GE fuel and hence is acceptable.

<u>Specification 3/4.10 Special Test Exceptions Recirculation Loops</u>

This specification has been modified to reflect that the provisions which allowed application of the exception to operation condition-1 is no longer available. This is currently allowed during the start-up test program; however, the start-up test program has been completed. Therefore, it is proposed to simplify the specification by eliminating reference to the expired provisions. This is acceptable.

There are minor changes to Bases 2.2.1, 3/4.2.1, Bases Table B 3.2.1-1, 3/4.4.1 and the new 3/4.4.10 addressing the above TS changes. They suitably reflect the TS changes and are acceptable.

We have reviewed the reports submitted by DECo for Fermi-2 proposed TS changes relating to SLO and THS. Based on the above review, the staff concludes that appropriate documentation was submitted and that the proposed changes satisfy staff positions and requirements in these areas. Extended SLO operation and THS monitoring in the manner thus described, and as augmented by compliance to the requests of NRC Bulletin 88-07 and Supplement 1, is acceptable.

4.0 <u>ENVIRONMENTAL CONSIDERATION</u>

This amendment involves changes 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. We have 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 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.

5.0 CONCLUSIONS

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (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.

6.0 REFERENCES

1. Letter and enclosures from B.R. Sylvia, to NRC, dated August 18, 1989, "Supplement to TS change License Amendment to include Single Loop Operations."

2. Letter from William S. Orser, to NRC, dated August 4, 1988, "Proposed Technical Specification changes to include Single Loop Operation (SLO).

3. GE report MDE-56-0386, "Fermi-2 Single Loop Operation Analysis," April 1987.

4. NRC Bulletin No. 88-07: Power Oscillations in Boiling Water Reactors (BWRs), June 15, 1988 and NRC Bulletin No. 88-07, Supplement 1, December 30, 1988.

5. GE report GESTAR II, NEDE-24011-P-A-6, dated April 1983.

6. Generic Letter No. 86-02, "Technical Resolution of Generic Issue B-19 Thermal Hyraulic Stability," January 23, 1986.

7. Generic Letter No. 86-09, "Technical Resolution of Generic Issue No. B-59-N-1 Loop Operation in BWRs and PWRs," March 31, 1986.

DATE: July 27, 1990