September 6, 1991

Docket No. 50-341

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

Dear Mr. Orser:

SUBJECT: AMENDMENT NO. 75 FACILITY OPERATING LICENSE NO. NPF-43: (TAC NO. 77680)

The Commission has issued the enclosed Amendment No.75 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 extends surveillance intervals and allowed out-of-service time for instrumentation associated with the reactor protection system, emergency core cooling system, control block function, and isolation function.

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,

C. E. Carpenter, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.75 to NPF-43
2. Safety Evaluation

cc w/enclosures:
See next page

LA/PD31	PM/PD31	PE:PDIII-1	PM:PDIII-1	D/PD31	OGC
PShuttleworth	CECarpenter	MGamberoni	JStang	LMarsh	
8/21/91	8/21/91	8/21/91	8/21/91	9/6/91	9/6/91

*FERMI AMEND 77680
OFFICIAL RECORD COPY

Mr. William Orser
Detroit Edison Company

Fermi-2 Facility

cc:
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Preparedness
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Monroe, Michigan 48161

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
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Ms. Lynne Goodman
Director - Nuclear Licensing
Detroit Edison Company
Fermi Unit 2
6400 North Dixie Highway
Newport, Michigan 48166

DATED: September 6, 1991

AMENDMENT NO. 75 TO FACILITY OPERATING LICENSE NO. NPF-43-FERMI UNIT 2

Docket File
NRC & Local PDRs
PDIII-1 Reading
B. Boger
J. Zwolinski
T. Marsh
P. Shuttleworth
J. Stang
C. Carpenter
OGC-WF
D. Hagan, 3302 MNBB
E. Jordan, 3701 MNBB
G. Hill (4), P-137
Wanda Jones, MNBB-7103
C. Grimes, 11/F/23
S. Newberry 11/D/23
S. Rhow 8/H/3
ACRS (10)
GPA/PA
OC/LFMB
B. Clayton, R-III
Fermi Plant File

cc: Plant Service list

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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. 75
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 20, 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. 75, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. DECo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

William O. Long

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 6, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 75

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
B 2-7	B 2-7
3/4 3-1	3/4 3-1
---	3/4 3-1a
3/4 3-2	3/4 3-2
3/4 3-5	3/4 3-5
3/4 3-6	3/4 3-6
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
3/4 3-9	3/4 3-9
3/4 3-11	3/4 3-11
3/4 3-13	3/4 3-13
3/4 3-14	3/4 3-14
3/4 3-14a	3/4 3-14a
3/4 3-20	3/4 3-20
3/4 3-21	3/4 3-21
3/4 3-22	3/4 3-22
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-37	3/4 3-37
3/4 3-38	3/4 3-38
3/4 3-39*	3/4 3-39*
3/4 3-40	3/4 3-40
3/4 3-45	3/4 3-45

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

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 Simulated Thermal Power - 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

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE channels per Trip System requirement for one trip system:

   1. Within 1 hour, verify that each Functional Unit within the affected trip system contains no more than one inoperable channel or place the inoperable channel(s) and/or that trip system in the tripped condition*.

   2. If placing the inoperable channel(s) in the tripped condition would cause a scram, the inoperable channel(s) shall be restored to OPERABLE status within 6 hours or the ACTION required by Table 3.3.1-1 for the affected Functional Unit shall be taken.

   3. If placing the inoperable channel(s) in the tripped condition would not cause a scram, place the inoperable channel(s) and/or that trip system in the tripped condition within 12 hours.

   The provisions of Specification 3.0.4 are not applicable.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within 1 hour and take the ACTION required by Table 3.3.1-1.

---

*An inoperable channel need not be placed in the tripped condition where this would cause a scram to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours after the channel was first determined to be inoperable or the ACTION required by Table 3.3.1-1 for that Functional Unit shall be taken.

**The trip system need not be placed in the tripped condition if this would cause a scram to occur. When a trip system can be placed in the tripped condition without causing a scram to occur, place the trip system with the most inoperable channels in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel 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 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	ACTION
1. Intermediate Range Monitors[(b)]:			
a. Neutron Flux - High	2 3, 4 5[(c)]	3 3 3[(d)]	1 2 3
b. Inoperative	2 3, 4 5	3 3 3[(d)]	1 2 3
2. Average Power Range Monitor[(e)]:			
a. Neutron Flux - High, Setdown	2 3 5[(c)]	2 2 2[(d)]	1 2 3
b. Flow Biased Simulated Thermal Power - High	1	2	4
c. Fixed Neutron Flux - High	1	2	4
d. Inoperative	1, 2 3 5[(c)]	2 2 2[(d)]	1 2 3
3. Reactor Vessel Steam Dome Pressure - High	1, 2[(f)]	2	1
4. Reactor Vessel Low Water Level - Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	1[(g)]	4	4

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) This function shall be automatically bypassed when the reactor mode switch is in the Run position.

(c) Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn.*

(d) When the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 4 APRMs, 6 IRMs and per Specification 3.9.2, 2 SRMs.

(e) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(g) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) This function shall be automatically bypassed when turbine first stage pressure is $\leq$ 154 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

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*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - High, Setdown	NA
b. Flow Biased Simulated Thermal Power - High	6 ± 1**
c. Fixed Neutron Flux - High	$\leq$ 0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 0.55
4. Reactor Vessel Low Water Level - Level 3	$\leq$ 1.05
5. Main Steam Line Isolation Valve - Closure	$\leq$ 0.06
6. Main Steam Line Radiation - High	NA
7. Drywell Pressure - High	NA
8. Scram Discharge Volume Water Level - High	
a. Float Switch	NA
b. Level Transmitter	NA
9. Turbine Stop Valve - Closure	$\leq$ 0.06
10. Turbine Control Valve Fast Closure	$\leq$ 0.08***
11. Reactor Mode Switch Shutdown Position	NA
12. Manual Scram	NA
13. Deleted	

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.

**Including simulated thermal power time constant.

***Measured from deenergization of K-37 relay which inputs the turbine control valve closure signal to the RPS.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S,(b) S	S/U[c], W W	SA SA	2 3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor[f]:				
a. Neutron Flux - High, Setdown	S/U,S,(b) S	S/U[c], W W	SA SA	2 3, 5
b. Flow Biased Simulated Thermal Power - High	S	S/U[c], Q	W[d][e],SA,R[h]	1
c. Fixed Neutron Flux - High	S	S/U[c], Q	W[d], SA	1
d. Inoperative	NA	Q	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	S	Q[k]	R	1, 2
4. Reactor Vessel Low Water Level - Level 3	S	Q[k]	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
6. Main Steam Line Radiation - High	S	Q	R	1, 2[i]
7. Drywell Pressure - High	S	Q[k]	R	1, 2

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	Q(k)	R	1, 2, 5(j)
9. Turbine Stop Valve - Closure	NA	Q	R	1
10. Turbine Control Valve Fast Closure	NA	Q	NA	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	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.
(k) Includes verification of the trip setpoint of the trip unit.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2 and with ISOLATION SYSTEM RESPONSE TIME as shown in Table 3.3.2-3.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, 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 System requirement for one trip system:

   1. If placing the inoperable channel(s) in the tripped condition would cause an isolation, the inoperable channel(s) shall be restored to OPERABLE status within 6 hours or the ACTION required by Table 3.3.2-1 for the affected trip function shall be taken.

   2. If placing the inoperable channel(s) in the tripped condition would not cause an isolation, the inoperable channel(s) and/or that trip system shall be placed in the tripped condition within:

      a) 12 hours for trip functions common to RPS Instrumentation; and
      b) 24 hours for trip functions not common to RPS Instrumentation.

   The provisions of Specification 3.0.4 are not applicable.

c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system* in the tripped condition within one hour and take the ACTION required by Table 3.3.2-1.

*Place one trip system (with the most inoperable channels) in the tripped condition. The trip system need not be placed in the tripped condition when this would cause the isolation to occur.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM(a)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Low Water Level				
1) Level 3 (e)##	13, 15	2	1, 2, 3	20
2) Level 2 (d)	2, 12, 14, 16, 17, 18	2	1, 2, 3	20
3) Level 1	1	2	1, 2, 3	20
b. Drywell Pressure - High (j)##	2, 12, 13, 14, 15, 16, 17, 18	2	1, 2, 3	20
c. Main Steam Line				
1) Radiation - High##	1, 2	2	1, 2, 3	21
2) Pressure - Low	1	2	1	22
3) Flow - High	1	2	1, 2, 3	21
d. Main Steam Line Tunnel Temperature - High	1	2(c)	1, 2, 3	21
e. Condenser Pressure - High	1	2	1, 2**, 3**	21
f. Turbine Bldg. Area Temperature - High	1	2	1, 2, 3	21
g. Deleted				
h. Manual Initiation	1, 2, 3, 5, 12, 13, 14 15, 16, 17, 18	1/valve	1, 2, 3	26

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Flow - High				
1. Differential Pressure	6	1	1, 2, 3	23
2. Time Delay	6	1	1, 2, 3	23
b. HPCI Steam Supply Pressure - Low	6, 7 (g)	2	1, 2, 3	23
c. HPCI Turbine Exhaust Diaphragm Pressure - High	6	2	1, 2, 3	23
d. HPCI Equipment Room Temperature - High	6	1	1, 2, 3	23
e. Manual Initiation	6, 7	1/valve	1, 2, 3	26
5. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Low Water Level - Level 3##	4(e)	2	1, 2, 3	25
b. Reactor Vessel (Shutdown Cooling Cut-in Permissive Interlock) Pressure - High	4	1	1, 2, 3	25
c. Manual Initiation	4	1/valve	1, 2, 3	26
6. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Low Water Level - Level 2 (b) (d)	***	2	1, 2, 3, and *	24
b. Drywell Pressure - High (b) (j)##	***	2	1, 2, 3	24
c. Fuel Pool Ventilation Exhaust Radiation - High (b)	***	2	1, 2, 3, and *	24
d. Manual Initiation (b)	***	1(i)	1, 2, 3, and *	27

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

ACTION STATEMENTS

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Be in at least STARTUP within 6 hours.

ACTION 23 - Close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 25 - Disable in the closed position the affected system isolation valves within 1 hour and declare the shutdown cooling mode of RHR inoperable.

ACTION 26 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable.

ACTION 27 - Restore the manual initiation function to OPERABLE status within 8 hours or establish SECONDARY CONTAINMENT INTEGRITY with the Standby Gas Treatment System operating.

TABLE NOTATIONS

* When handling irradiated fuel in the secondary containment, during CORE ALTERATIONS, or during operations with a potential for draining the reactor vessel.

** The high condenser pressure input to the isolation actuation instrumentation may be bypassed during reactor shutdown or for reactor startup when condenser pressure is above the trip setpoint.

*** Actuates dampers shown in Table 3.6.5.2-1.

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the channel or trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter. In addition, for the HPCI system and RCIC system isolation, provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is OPERABLE and all required actuation instrumentation for that valve is OPERABLE, one channel may be placed in an inoperable status for up to 8 hours for required surveillance without placing the channel or trip system in the tripped condition.

(b) Also starts the standby gas treatment system.

(c) A channel is OPERABLE if 2 of 4 detectors in that channel are OPERABLE.

(d) This level signal actuates Groups 2, 10, 11, 12, 14, 16, 17, 18, and ***.

(e) This level signal actuates Groups 4, 13 and 15.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATIONS (Continued)

(f) Isolates with simultaneous RCIC Steam Supply Pressure-Low (Isolation Instrumentation) and Drywell Pressure-High (ECCS Actuation Instrumentation).

(g) Isolates with simultaneous HPCI Steam Supply Pressure-Low (Isolation Actuation Instrumentation) and Drywell Pressure-High (ECCS Actuation Instrumentation).

(h) Reserved.

(i) Secondary Containment Isolation Push-buttons.

(j) This pressure signal actuates Groups 2, 12, 13, 14, 15, 16, 17, 18, and ***.

With time delay of 45 seconds.

These trip function(s) are common to the RPS trip function.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Low Water Level-				
1) Level 3	S	$Q^{\#}$	R	1, 2, 3
2) Level 2	S	$Q^{\#}$	R	1, 2, 3
3) Level 1	S	$Q^{\#}$	R	1, 2, 3
b. Drywell Pressure - High	S	$Q^{\#}$	R	1, 2, 3
c. Main Steam Line				
1) Radiation - High	S	Q	R	1, 2, 3
2) Pressure - Low	S	$Q^{\#}$	R	1
3) Flow - High	S	$Q^{\#}$	R	1, 2, 3
d. Main Steam Line Tunnel Temperature - High	S	$Q^{\#}$	R	1, 2, 3
e. Condenser Pressure - High	S	$Q^{\#}$	R	1, 2**, 3**
f. Turbine Bldg. Area Temperature - High	S	$Q^{\#}$	R	1, 2, 3
g. Deleted				
h. Manual Initiation	NA	R	NA	1, 2, 3

TABLE 4.3.2.1-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
2. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	Q	R	1, 2, 3
b. Heat Exchanger/Pump/High Energy Piping Area Temperature - High	S	Q	R	1, 2, 3
c. Heat Exchanger/Pump/Phase Separator Area Ventilation Δ Temperature - High	S	Q	R	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2, 3
e. Reactor Vessel Low Water Level - Level 2	S	Q#	R	1, 2, 3
f. Deleted				
g. Manual Initiation	NA	R	NA	1, 2, 3
3. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High				
1. Differential Pressure	S	Q#	R	1, 2, 3
2. Time Delay	NA	Q	R	1, 2, 3
b. RCIC Steam Supply Pressure - Low	S	Q#	R	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	S	Q#	R	1, 2, 3
d. RCIC Equipment Room Temperature - High	S	Q#	R	1, 2, 3
e. Manual Initiation	NA	R	NA	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Flow - High				
1. Differential Pressure	S	Q#	R	1, 2, 3
2. Time Delay	NA	Q	R	1, 2, 3
b. HPCI Steam Supply Pressure - Low	S	Q#	R	1, 2, 3
c. HPCI Turbine Exhaust Diaphragm Pressure - High	S	Q#	R	1, 2, 3
d. HPCI Equipment Room Temperature - High	S	Q#	R	1, 2, 3
e. Manual Initiation	NA	R	NA	1, 2, 3
5. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Low Water Level - Level 3	S	Q#	R	1, 2, 3
b. Reactor Vessel (Shutdown Cooling Cut-in Permissive Interlock) Pressure - High	S	Q#	R	1, 2, 3
c. Manual Initiation	NA	R	NA	1, 2, 3
6. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Low Water Level - Level 2	S	Q#	R	1, 2, 3, and *
b. Drywell Pressure - High	S	Q#	R	1, 2, 3
c. Fuel Pool Ventilation Exhaust Radiation - High	S	Q	R	1, 2, 3, and *
d. Manual Initiation	NA	R	NA	1, 2, 3, and *

* When handling irradiated fuel in the secondary containment, during CORE ALTERATIONS, and during operations with a potential for draining the reactor vessel.
** May be bypassed under administrative control.
\# Includes verification of the trip setpoint of the trip unit.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
4. AUTOMATIC DEPRESSURIZATION SYSTEM#			
a. Reactor Vessel Low Water Level - Level 1	2	1, 2, 3	30
b. Drywell Pressure - High	2	1, 2, 3	30
c. ADS Timer	1	1, 2, 3	31
d. Core Spray Pump Discharge Pressure - High (Permissive)	1/pump	1, 2, 3	31
e. RHR LPCI Mode Pump Discharge Pressure - High (Permissive)	1/pump	1, 2, 3	31
f. Reactor Vessel Low Water Level - Level 3 (Permissive)	1	1, 2, 3	31
g. Manual Initiation	1/valve	1, 2, 3	33
h. Drywell Pressure - High Bypass Timer	2	1, 2, 3	31
i. Manual Inhibit	1	1, 2, 3	33

	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
5. LOSS OF POWER					
1. 4.16 kV Emergency Bus Under-voltage (Loss o' Voltage)	2/bus	1/bus	1/bus	1, 2, 3, 4**, 5**	35
2. 4.16 kV Emergency Bus Under-voltage (Degraded Voltage)	2/bus	1/bus	1/bus	1, 2, 3, 4**, 5**	35

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.
(b) Also actuates the associated emergency diesel generators.
(c) One trip system. Provides signals to HPCI and RCIC suction valves.
(d) One trip system. Provides signal to HPCI pump suction valves only.
(e) On 2 out of 2 logic, provides a signal to trip the HPCI turbine.
* When the system is required to be OPERABLE per Specification 3.5.2.
** Required when ESF equipment is required to be OPERABLE.
\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 150 psig.
\#\# Individual component controls.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION STATEMENTS

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement:

a. For one trip system, place that trip system in the tripped condition within 24 hours* or declare the associated ECCS inoperable.

b. For both trip systems, declare the associated ECCS inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, declare the associated ADS Trip System inoperable within 24 hours.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place the inoperable channel in the tripped condition within 24 hours.

ACTION 33 - Restore the manual initiation and/or manual inhibit function to OPERABLE status within 24 hours or declare the associated ECCS or ADS Trip System inoperable.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place at least one inoperable channel in the tripped condition within 24 hours*, align the HPCI system to take suction from the suppression pool, or declare the HPCI system inoperable.

ACTION 35 - With the number of OPERABLE channels:

a. One less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 72 hours or declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2, as appropriate.

b. Less than the Minimum Channels OPERABLE requirement, declare the associated diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2, as appropriate.

*The provisions of Specification 3.0.4 are not applicable.

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. CORE SPRAY SYSTEM				
a. Reactor Vessel Low Water Level - Level 1	S	Q##	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	Q##	R	1, 2, 3
c. Reactor Steam Dome Pressure - Low	S	Q##	R	1, 2, 3, 4*, 5*
d. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM				
a. Reactor Vessel Low Water Level - Level 1	S	Q##	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	Q##	R	1, 2, 3
c. Reactor Steam Dome Pressure - Low	S	Q##	R	1, 2, 3, 4*, 5*
d. Reactor Vessel Low Water Level - Level 2	S	Q##	R	1, 2, 3, 4*, 5*
e. Reactor Steam Dome Pressure - Low	S	Q##	R	1, 2, 3, 4*, 5*
f. Riser Differential Pressure - High	S	Q##	R	1, 2, 3
g. Recirculation Pump Differential Pressure - High	S	Q##	R	1, 2, 3
h. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
3. HIGH PRESSURE COOLANT INJECTION SYSTEM#				
a. Reactor Vessel Low Water Level - Level 2	S	Q##	R	1, 2, 3
b. Drywell Pressure - High	S	Q##	R	1, 2, 3
c. Condensate Storage Tank Level - Low	S	Q##	R	1, 2, 3
d. Suppression Pool Water Level - High	S	Q##	R	1, 2, 3
e. Reactor Vessel High Water Level - Level 8	S	Q##	R	1, 2, 3
f. Manual Initiation	NA	R	NA	1, 2, 3

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
4. AUTOMATIC DEPRESSURIZATION SYSTEM#				
a. Reactor Vessel Low Water Level - Level 1	S	Q##	R	1, 2, 3
b. Drywell Pressure - High	S	Q##	R	1, 2, 3
c. ADS Timer	NA	Q	R	1, 2, 3
d. Core Spray Pump Discharge Pressure - High	S	Q##	R	1, 2, 3
e. RHR LPCI Mode Pump Discharge Pressure - High	S	Q##	R	1, 2, 3
f. Reactor Vessel Low Water Level - Level 3	S	Q##	R	1, 2, 3
g. Manual Initiation	NA	R	NA	1, 2, 3
h. Drywell Pressure - High Bypass Timer	NA	Q##	R	1, 2, 3
i. Manual Inhibit	NA	R	NA	1, 2, 3
5. LOSS OF POWER				
a. 4.16 kV Emergency Bus Under-voltage (Loss of Voltage) (Division 1 and Division 2)	NA	M	R	1, 2, 3, 4**, 5**
b. 4.16 kV Emergency Bus Under-voltage (Degraded Voltage) (Division 1 and Division 2)	NA	M	R	1, 2, 3, 4**, 5**

* When the system is required to be OPERABLE per Specification 3.5.2.

** Required OPERABLE when ESF equipment is required to be OPERABLE.

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 150 psig.

\## Includes verification of the trip setpoint of the trip unit.

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	ACTION
a. Reactor Vessel Low Water Level - Level 2	2	50
b. Reactor Vessel High Water Level - Level 8	2[(b)]	50
c. Condensate Storage Tank Water Level - Low	2[(c)]	51
d. Manual Initiation	1/valve	52

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(b) One trip system with two-out-of-two logic.

(c) One trip system with one-out-of-two logic.

TABLE 3.3.5-1 (Continued)

REACTOR CORE ISOLATION COOLING SYSTEM

ACTION STATEMENTS

ACTION 50 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement:

a. For one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition within 24 hours or declare the RCIC system inoperable.

b. For both trip systems, declare the RCIC system inoperable.

ACTION 51 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place at least one inoperable channel in the tripped condition within 24 hours* or align RCIC to take suction from the suppression pool or declare the RCIC system inoperable.

ACTION 52 - Restore the manual initiation function to OPERABLE status within 24 hours or declare the RCIC system inoperable.

*The provisions of Specification 3.0.4 are not applicable.

TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Low Water Level - Level 2	$\geq$ 110.8 inches*	$\geq$ 103.8 inches
b. Reactor Vessel High Water Level - Level 8	$\leq$ 214 inches*	$\leq$ 219 inches
c. Condensate Storage Tank Level - Low	$\geq$ 3 inches (27 inches above tank bottom)	$\geq$ 0 inches (24 inches above tank bottom)
d. Manual Initiation	NA	NA

*See Bases Figure B 3/4 3-1.

TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Low Water Level - Level 2	S	Q#	R
b. Reactor Vessel High Water Level - Level 8	S	Q#	R
c. Condensate Storage Tank Level - Low	S	Q#	R
d. Manual Initiation	NA	R	NA

Includes verification of the trip setpoint of the trip unit.

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],Q	Q	1*
b. Inoperative	NA	S/U[b],Q	NA	1*
c. Downscale	NA	S/U[b],Q	Q	1*
2. APRM				
a. Flow Biased Neutron Flux - High	S	S/U[b],Q	SA	1
b. Inoperative	NA	S/U[b],Q	NA	1, 2, 5
c. Downscale	S	S/U[b],Q	SA	1
d. Neutron Flux - Upscale, Setdown	S	S/U[b],Q	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],Q	Q	1
b. Inoperative	NA	S/U[b],Q	NA	1
c. Comparator	NA	S/U[b],Q	Q	1
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	R	NA	3, 4

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 75 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, 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 extend surveillance intervals and allow out-of-service time for instrumentation associated with the related protection system, emergency core cooling system, control rod block function, and isolation function.

2.0 EVALUATION

The BWR Owners' Group (BWROG), of which Detroit Edison is a member, sponsored studies by General Electric (GE) to apply probabilistic analytical methods in order to justify an increase in surveillance test intervals (STI) and allowable out-of-service times (AOST) for various BWR instrumentation. These studies resulted in a series of Licensing Topical Reports (LTR) which have been generically reviewed and approved by the NRC. However, the approval was conditional based on a list of plant-specific conditions that the licensees should follow. The plant-specific conditions are as follows. This proposed amendment applies these generic results to Fermi-2.

(1) Confirm the applicability of the generic analyses for the Licensing Topical Reports to its plant.

(2) Demonstrate by use of current drift information provided by the equipment vendor or plant-specific data that the drift characteristics for instrumentation used in the plant are bounded by the assumptions used in the Licensing Topical Reports when the functional test interval is extended from monthly to quarterly.

(3) Confirm that the differences between the parts of the reactor protection system (RPS) that perform the trip functions in the plant and those of the base case plant were included in the analysis. The procedures of Appendix K of NEDC-30851P should be used or provide a plant-specific analysis to demonstrate that there is not an appreciable change in RPS availability or public risk.

In accordance with these conditions by letter dated August 20, 1990, the Detroit Edison Company proposed changes to the TS related to the instrumentation for Fermi Unit 2.

The application of each generic LTR to Fermi-2 is evaluated individually below:

(1) (a) The generic study in NEDC-30851P provides a technical basis to modify the surveillance test frequencies and allowable out-of-service time of the RPS. The generic study also provides additional analyses of different RPS configurations to support the application of the generic basis on a plant-specific basis. The plant-specific evaluation for modifying the surveillance test frequencies and allowable out-of-service time of the RPS was performed by GE for Fermi-2. The evaluation utilized the generic basis and the additional analyses documented in NEDC-30851P. The results indicate that the RPS configuration for Fermi Unit 2 is different when compared to the RPS configuration in the generic evaluation. However, these differences do not affect the results of the generic analysis. Therefore, the generic analysis in NEDC-30851P is applicable to Fermi-2.

(b) The staff generic safety evaluation report (SER) of May 27, 1987, on GE Licensing Topical Reports NEDC-30844 and NEDC-30851P states the NRC's requirement for confirmation of instrument set point drift allowance. By a letter to the BWR Owners' Group from C. Rossi (NRC) dated April 27, 1988, the NRC requested licensees to confirm that the set point drift which could be expected under the extended STI has been studied and either (1) has been shown to remain within the existing allowance in the RPS and engineered safety fracture actuation system (ESFAS) instrument set point calculation or (2) that the allowance and set point have been adjusted to account for the additional expected drift. No additional information needs to be provided for staff review. However, records showing the actual set point calculation and supporting data should be retained on-site for possible future staff audit. The licensee provided a sample of actual drift data. The licensee has demonstrated that drift data of the affected instrumentation remained within the existing allowance in the RPS and ESFAS instrument set point calculation when considered over the extended period.

(c) The AOST and STI depicted in TS 3/4.3.1 reflect the generic TS changes included with the NRC's generic SER. Included is the increase in testing frequency from monthly to weekly for the manual scram trip function. The generic TS changes were based upon a base plant TS requirement to verify the trip unit set point during each functional test. Therefore, the licensee proposes a new footnote (k) to TS Table 4.3.1.1-1 which indicates that the quarterly functional test includes verification of the trip unit set point.

(d) The requirement to take the action required by Table 3.3.1-1 within 6 hours is appended to a 12-hour requirement to place the inoperable channel(s) and/or trip system in the tripped condition. The proposed TS has been restructured to include this provision in the action statement itself without changing the technical requirements. Further, the term "scram" is used to replace the term "trip function" since it is clearer and better conveys the meaning of the provision.

(e) In addition, this proposal changes the nomenclature for the Average Power Range Monitor (APRM) Flow-Biased instrument. The current nomenclature is inconsistent between TS 3/4.3.1 which uses the term "Flow-Biased Neutron Flux," and TS Table 2.2.1-1, which uses the term "Flow-Biased Simulated Thermal Power." The term "Flow-Biased Simulated Thermal Power" is correct and, therefore, this proposal replaces the incorrect terminology with the correct terminology to reduce the possibility of confusion.

(2) (a) GE Report RE-014, Rev. 1, followed the procedures of LTR NEDC-30936P-A; Part 2, Appendix F to identify and evaluate the differences between the Fermi-2 emergency core cooling system (ECCS) configuration and the ECCS configuration used in the generic analysis. The results of our review indicate that while the ECCS configuration for Fermi-2 has several differences compared to the generic configuration, the differences and their impact do not affect the applicability of the TS changes developed by the generic efforts of these LTR. Therefore, the generic analysis in NEDC-30936 P-A, Part 2, is applicable to Fermi-2.

(b) The ECCS actuation instrumentation channel drift characteristics are considered due to the extended STI. As in the RPS instrumentation discussed above, the licensee provided a sample of actual drift data and demonstrated that drift data of the affected instrumentation remained within the existing allowance in the ECCS instrument set point calculation when considered over the extended period.

(c) The 24-hour AOST is proposed to be moved from TS 3.3.3 Action Statement to TABLE 3.3.3-1 individual actions (Action 30, Action 31,...). The recommended modification as written in the Licensing Topical Reports implies a 24-hour AOST before taking any action listed in TABLE 3.3.3-1. The proposed change does not . ter the intent of allowed out-of-service time and prevents the AOST extension from being applied to the EDG/Loss-Of-Power instrumentation which was not covered by the Topical Reports.

(d) A number of instruments covered by this TS utilize an associated trip unit to generate a channel trip signal. As in the RPS discussed above, it is the practice to verify the trip unit set point during each channel functional test. The licensee proposed to add a footnote that the functional test includes verification of the trip set point. This practice ensures the intent of generic TS changes.

(3) Licensing Topical Report NEDC-30851P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," provides a basis for extending STI for the control rod block function (CRBF) instrumentation. The CRBF shares common instrumentation with the RPS. To gain the full safety benefits of the proposed RPS instrumentation TS changes a corresponding STI extension for the CRBF is necessary. The generic analysis was performed by GE for the BWR Owners' Group and resulted in the above LTR. The NRC staff has reviewed and approved this Licensing Topical Report (LTR). Detroit Edison has applied the generic analysis to Fermi-2 by completing the conditions for plant specific application of the CRBF instrumentation TS changes.

(4) Licensing Topical Report, NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation" provides a basis for extending STI and AOST for isolation actuation instrumentation common to RPS and ECCS instrumentation. The necessary generic analysis was performed by GE for the BWR Owners' Group and resulted in this LTR, which the NRC staff has reviewed and approved.

Detroit Edison has applied the generic analysis to Fermi-2 by completing the conditions for a plant specific application contained in the NRC's generic SER. The licensee provided an analysis that shows that any increase in instrument drift due to the extended STI is properly accounted for in the set point calculation methodology.

A number of instruments covered by this TS utilize an associated trip unit to generate a channel trip signal. To verify that the trip unit trip set point is checked during each channel functional test, Detroit Edison has proposed to add a footnote to indicate, where appropriate, that the quarterly functional test includes verification of the trip unit set point. This will ensure consistency with the RPS and the intent of generic TS changes.

(5) Licensing Topical Report NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," provides a basis for extending STI and AOST for isolation actuation instrumentation not covered by the LTR instrumentation common to RPS and ECCS instrumentation. The NRC has reviewed and approved this LTR. Detroit Edison has applied the generic analysis to Fermi-2 by completing the conditions for plant specific application of the TS changes contained in the NRC's SER. The conditions are: (1) to confirm the applicability of the generic analyses to the plant, and (2) to confirm that any increase in instrument drift due to the extended STI is properly accounted for in the set point calculation methodology.

A number of instruments covered by this TS change utilize an associated trip unit to generate a channel trip signal. To meet the conditions, Detroit Edison proposes to add a footnote to indicate, where appropriate, that the quarterly functional test includes verification on the trip unit set point. This will ensure consistency with both the RPS changes and the intent of the generic TS changes.

The proposed changes extend STI and AOST for instrumentation which has been justified using probabilistic analytical methods. The affected instrumentation is associated with the RPS, ECCS, CRBF, and isolation function. The changes have been the subject of generic Licensing Topical Reports which the NRC has reviewed and approved. The changes also include an administrative change to the nomenclature for an RPS instrumentation functional unit, and a clarification that certain channel functional tests include a verification of the trip set point of the trip unit. Detroit Edison performed the required plant specific analysis and justified the application of the generic analysis to the Fermi-2 plant specific design. The information for set point drift supports the conclusion that instrument drift is not a concern in extending the functional test interval from monthly to quarterly. Therefore, we have found the proposed changes to the Fermi-2 Technical Specifications acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The NRC staff has determined that the 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 the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the 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 the amendment.

5.0 CONCLUSION

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

Principal Contributor: Sang Rhow, SCIB

Date: September 6, 1991