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

August 28, 1991

Docket No. 50-341

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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. 74 TO FACILITY OPERATING LICENSE NO. NPF-43:
(TAC NO. 77678)

The Commission has issued the enclosed Amendment No. 74 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 3, 1990.

The amendment revises the TS by changing the Emergency Core Cooling System response time requirements for the Low Pressure Core Injection mode of the Residual Heat Removal system.

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-I
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 74 to NPF-43
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	:LA/PD31	:PM/PD31	:DRIS	:OGC	:D/PD31
NAME	:MShuttleworth	:CCarpenter	:RFrahm	:	:LBMarsh
DATE	:8/15/91	:8/9/91	:8/13/91	:8/16/91	:8/28/91

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

DETROIT EDISON COMPANY

FERMI-2

DOCKET NO. 50-341

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 74
License No. NPF-43

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Detroit Edison Company (the licensee) dated August 1, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-43 is hereby amended to read as follows:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 74, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. DECo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 28, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 74

FACILITY OPERATING LICENSE NO. NPF-43

DOCKET NO. 50-341

Replace the following page of the Appendix "A" Technical Specifications with the attached page. The revised page is identified by Amendment number and contains a vertical line indicating the area of change.

REMOVE	INSERT
3/4 3-29	3/4 3-29

TABLE 3.3.3-3
EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

TRIP FUNCTION	RESPONSE TIME (Seconds)
1. CORE SPRAY SYSTEM	
a. Reactor Vessel Low Water Level - Level 1	≤ 30
b. Drywell Pressure - High	≤ 30
c. Reactor Steam Dome Pressure - Low	NA*
d. Manual Initiation	NA
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM	
a. Reactor Vessel Low Water Level - Level 1	≤ 55
b. Drywell Pressure - High	≤ 55
c. Reactor Steam Dome Pressure - Low	NA*
d. Reactor Vessel Low Water Level - Level 2	NA
e. Reactor Steam Dome Pressure - Low	NA
f. Riser Differential Pressure - High	NA
g. Recirculation Pump Differential Pressure - High	NA
h. Manual Initiation	NA
3. HIGH PRESSURE COOLANT INJECTION SYSTEM	
a. Reactor Vessel Low Water Level - Level 2	≤ 30
b. Drywell Pressure - High	NA
c. Condensate Storage Tank Level - Low	NA
d. Reactor Vessel Water Level - High, Level 8	NA
e. Suppression Pool Water Level - High	NA
f. Manual Initiation	NA
4. AUTOMATIC DEPRESSURIZATION SYSTEM	
a. Reactor Vessel Low Water Level - Level 1	NA
b. Drywell Pressure - High	NA
c. ADS Timer	NA
d. Core Spray Pump Discharge Pressure - High	NA
e. RHR LPCI Mode Pump Discharge Pressure - High	NA
f. Reactor Vessel Low Water Level - Level 3	NA
g. Manual Initiation	NA
h. Drywell Pressure - High Bypass Timer	NA
i. Manual Inhibit	NA
5. LOSS OF POWER	
a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	NA
b. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	NA

*These are permissive signals only. They do not activate ECCS initiation.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 74 TO FACILITY OPERATING LICENSE NO. NPF-43

DETROIT EDISON COMPANY

FERMI-2

DOCKET NO. 50-341

1.0 INTRODUCTION

By letter dated August 2, 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 revise the emergency core cooling system (ECCS) response time requirements for the low pressure core injection (LPCI) mode of the residual heat removal (RHR) system. The new response time requirements reflect the results of a recalculation of the applicable accident analysis by General Electric that includes a five second time delay for re-energizing the Swing Bus that was not specifically accounted for in the accident analysis.

While performing a modification change to enhance the LPCI Swing Bus to address a degraded voltage concern (this modification was the subject of the Fermi-2 Licensee Event Report 87-045-01, dated November 6, 1989), it was determined that a five second time delay in the re-energization of the LPCI Swing Bus, which is required in some accident situations, had not been specifically accounted for in the accident analysis. The licensee stated that the original analysis at the time of the above determination contained enough conservatism to bound the five second Swing Bus time delay even though it did not specifically consider it. The accident situation which is applicable to re-energization of the Swing Bus is discussed below.

In the event of a break or loss-of-coolant accident (LOCA) in one of the two reactor recirculation system loops, logic is provided to sense the broken loop and to inject full LPCI flow into the unbroken loop from both divisions of LPCI. Thus, the flow from two LPCI divisions are interconnected by valving and depend on individual valves whose failure could completely inhibit LPCI flow.

Since electrical power to each LPCI Division is separated (Division I and II), Fermi-2 has a LPCI Swing Bus arrangement which permits essential LPCI recirculation system valves that could be disabled by a failure of a divisional electrical supply to be energized by either electrical division. Under loss-of-offsite-power (LOSP) conditions, the Swing Bus is normally re-energized from the Division II Emergency Diesel Generators (EDGs) without further time delay when this power source becomes available.

However, if this power source is not available because of single failure, the logic to allow re-energization from the Division I EDGs has an inherent five second time delay. Because this five second time delay was not specifically covered in the applicable analysis of record, Detroit Edison had General Electric recalculate the ECCS response time to allow for this additional time delay and to provide for additional margin, as described below. The response time was also recalculated to verify that delaying LPCI would not exceed the accident consequences of the most limiting analysis of record.

A recalculation of the applicable analysis of the LPCI mode of RHR was performed delaying LPCI for an additional 12 seconds to account for the loading time to re-energize the LPCI Swing Bus, as discussed above, and to justify an increase in the response time. The revised analysis determined that the limiting analysis of record was not exceeded by the additional time delay associated with LPCI. The limiting analysis of record is associated with a LPCI injection valve failure which completely disables all LPCI flow and relies on the core spray and high pressure coolant injection (HPCI) systems for core flooding.

2.0 EVALUATION

Detroit Edison had General Electric recalculate the appropriate ECCS accident response calculations accounting for a longer (by 12 seconds) LPCI injection time which, conservatively, bounds the additional five seconds required for the Swing Bus to re-energize as previously described. This calculation was done using an approved ECCS Evaluation Model in accordance with Appendix K to 10 CFR Part 50 to determine the peak cladding temperature (PCT) of the reactor fuel. This PCT is compared to the acceptance criterion of 2200°F PCT specified in 10 CFR 50.46. The previous worst-case PCT for events involving the response of the LPCI mode of RHR has increased from less than 1800°F to less than 1900°F when evaluated with a new response time of 55 seconds for the LOCA/LOSP event. The most limiting PCT for the large break LOCA/LOSP still occurs for an event where the LPCI mode of RHR does not respond (LPCI injection valve failure). This case is unaffected by the increased response times discussed above and thus the limiting PCT remains unchanged at 2084°F. The other acceptance criteria of 10 CFR 50.46 are not affected by the increased response time since the limiting PCT remained unchanged. The General Electric analysis also, conservatively, assumed that the last initiation signal (low water level 1) rather than the first initiation signal (high drywell pressure) initiates LPCI. This assumption delays the initiation of LPCI more than the delay from the single failure which causes the Swing Bus transfer.

In summary, a margin of over 300°F remains between the new calculated PCT value for LPCI mode of RHR and the 10 CFR 50.46 PCT criterion of 2200°F with the delayed response time. Additionally, the limiting PCT (2084°F) remains unchanged. For these reasons, we find the proposed change to be acceptable.

3.0 SUMMARY

We have reviewed the results of the analysis submitted by Detroit Edison Company for Fermi-2 with a delayed LPCI response time. We find that acceptable methods (10 CFR Part 50, Appendix K) were utilized for the calculations and that the criteria of 10 CFR 50.46 have been satisfied. For these reasons, we find the proposed change of 55 seconds instead of 43 seconds delay time to be acceptable.

4.0 STATE CONSULTATION

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

5.0 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. 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 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.

6.0 CONCLUSION

The Commission 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: R. Frahm

Date: August 28, 1991