June 23, 2008

Mr. Jack M. Davis
Senior Vice President and Chief Nuclear Officer
Detroit Edison Company
Fermi 2 - 210 NOC
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: FERMI 2 - ISSUANCE OF AMENDMENT RE: CONTROL ROD NOTCH SURVEILLANCE TEST FREQUENCY - TSTF-475 (TAC NO. MD7952)

Dear Mr. Davis:

The Commission has issued the enclosed Amendment No.179 to Facility Operating License No. NPF-43 for the Fermi 2 facility. The amendment consists of changes to the Technical Specifications in response to your application dated January 15, 2008.

The amendment revises the Technical Specifications (TS) Surveillance Requirement frequency in TS 3.1.3, "Control Rod OPERABILITY" from "7 days after the control rod is withdrawn and THERMAL POWER is greater than the [Low Power Setpoint] LPSP of [Rod Worth Minimizer] RWM" to "31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM" and revises Example 1.4-3 in Section 1.4 "Frequency" to clarify the applicability of the 1.25 surveillance test interval extension.

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,

/RA/

Justin C. Poole, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-341

Enclosures:
1. Amendment No.179 to NPF-43
2. Safety Evaluation

cc w/encls: See next page

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Package Accession Number: ML081500551
Amendment Accession Number: ML081490420
TS Accession Number: ML081790436 *Per Memo dated 5/16/08

OFFICE	NRR/LPL3-1/PM	NRR/LPL3-1/LA	DIRS/ITSB	OGC	NRR/LPL3-1/BC
NAME	JPoole	THarris	RElliott*	KWinsberg	LJames(PTam for)
DATE	6/11/08	6/11/08	5/16/08	6/13/08	6/23/08

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DETROIT EDISON COMPANY

DOCKET NO. 50-341

FERMI 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 179
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 January 15, 2008, 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. 179 , 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 and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by P.Tam for/

Lois M. James, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License and Technical Specifications

Date of Issuance:
June 23, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 179

FACILITY OPERATING LICENSE NO. NPF-43

DOCKET NO. 50-341

Replace the following pages of the Facility Operating License and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
License Page 3	License Page 3
1.4-4	1.4-4
1.4-5	1.4-5
3.1-8	3.1-8
3.1-10	3.1-10
3.1-11	3.1-11
3.1-14	3.1-14

(4) DECo, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material such as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) DECo, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) DECo, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

DECo is authorized to operate the facility at reactor core power levels not in excess of 3430 megawatts thermal (100% power) in accordance with conditions specified herein and in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment 179 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. DECo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

DECo shall abide by the agreements and interpretations between it and the Department of Justice relating to Article I, Paragraph 3 of the Electric Power Pool Agreement between Detroit Edison Company and

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 179 FACILITY OPERATING LICENSE NO. NPF-43

DETROIT EDISON COMPANY

FERMI 2

DOCKET NO. 50-341

1.0 INTRODUCTION

By application dated January 15, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML 080230716), the Detroit Edison Company (DECo or the licensee) requested changes to the Technical Specifications (TSs) for Fermi 2. The proposed changes would: (1) revise the TS surveillance requirement (SR) frequency in TS 3.1.3, "Control Rod OPERABILITY" from "7 days after the control rod is withdrawn and THERMAL POWER is greater than the [Low Power Setpoint] LPSP of [Rod Worth Minimizer] RWM" to "31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM" and (2) revise Example 1.4-3 in Section 1.4 "Frequency" to clarify the applicability of the 1.25 surveillance test interval extension.

These changes are based on Technical Specifications Task Force (TSTF) change traveler TSTF-475, Revision 1, that proposes revisions to the reference Standard TS by: (1) revising the frequency of SR 3.1.3.2, notch testing of each fully withdrawn control rod, from "7 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of RWM" to "31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM" and (2) revising Example 1.4-3 in Section 1.4 "Frequency" to clarify that the 1.25 surveillance test interval extension in SR 3.0.2 is applicable to time periods discussed in NOTES in the "SURVEILLANCE" column in addition to the time periods in the "FREQUENCY" column.

The purpose of the surveillance is to confirm control rod insertion capability which is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. Control rods and control rod drive (CRD) mechanism (CRDM), by which the control rods are moved, are components of the CRD System, which is the primary reactivity control system for the reactor. By design, the CRDM is highly reliable with a tapered design of the index tube which is conducive to control rod insertion.

A stuck control rod is an extremely rare event and industry review of plant operating experience did not identify any incidents of stuck control rods while performing a rod notch surveillance test.

The purpose of these revisions is to reduce the number of control rod manipulations and, thereby, reduce the opportunity for reactivity control events.

The purpose of the change to Example 1.4-3 in Section 1.4 "Frequency" is to clarify the applicability of the 25 percent allowance of SR 3.0.2 to time periods discussed in NOTES in the "SURVEILLANCE" column as well as to time periods in the "FREQUENCY" column.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, General Design Criterion (GDC) 29, Protection against anticipated operational occurrences, requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in an event of anticipated operational occurrences. The design relies on the CRD System to function in conjunction with the protection systems under anticipated operational occurrences, including loss of power to all recirculation pumps, tripping of the turbine generator, isolation of the main condenser, and loss of all offsite power. The CRD System provides an adequate means of inserting sufficient negative reactivity to shut down the reactor and prevent exceeding acceptable fuel design limits during anticipated operational occurrences. Meeting the requirements of GDC 29 for the CRD System prevents occurrence of mechanisms that could result in fuel cladding damage such as severe overheating, excessive cladding strain, or exceeding the thermal margin limits during anticipated operational occurrences. Preventing excessive cladding damage in the event of anticipated transients ensures maintenance of the integrity of the cladding as a fission product barrier.

3.0 TECHNICAL EVALUATION

In order to perform this Safety Evaluation, the Nuclear Regulatory Commission (NRC) staff reviewed the following information provided by the TSTF to justify the submitted license amendment request to revise the weekly control rod notch frequency to monthly, and revise the discussion of the applicability of the 25 percent allowance in Example 1.4-3. Specifically, the following documents were reviewed during the NRC staff's evaluation:

- TSTF letter TSTF-04-07 (Reference 1) - Provided a description of the proposed changes in TSTF-475 that changes the weekly rod notch frequency to monthly and clarify the applicability of the 25 percent allowance in Example 1.4-3.

- TSTF letter TSTF-06-13 (Reference 8) - Provided responses to NRC staff request for additional information (RAI) on (1) industry experience with identifying stuck rods, (2) tests that would identify stuck rods, (3) continue compliance with SIL 139, (4) industry experience on collet failures, and (4) applying the 25 percent grace period to the 31 day control rod notch SR test frequency.

- Boiling-Water Reactor (BWR) Owners Group (BWROG) letter BWROG-06036 (Reference 9) – Provided the General Electric (GE) Nuclear Energy Report, "CRD Notching Surveillance Testing for Limerick Generating Station," in which CRD notching frequency and CRD performance were evaluated.

- TSTF letter TSTF-07-19 (Reference 10) - Provided response to NRC staff RAI on CRD performance in Control Cell Core designed plants, including TSTF-475, Revision 1.

The CRD System is the primary reactivity control system for the reactor. The CRD System, in conjunction with the reactor protection system, provides the means for the reliable control of reactivity changes to ensure under all conditions of normal operation, including anticipated operational occurrences that specified acceptable fuel design limits are not exceeded. Control rods are components of the CRD System that have the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System.

The CRD System consists of a CRDM, by which the control rods are moved, and a hydraulic control unit (HCU) for each control rod. The CRDM is a mechanical hydraulic latching cylinder that positions the control blades. The CRDM is a highly reliable mechanism for inserting a control rod to the full-in position. The collet piston mechanism design feature ensures that the control rod will not be inadvertently withdrawn. This is accomplished by engaging the collet fingers, mounted on the collet piston, in notches located on the index tube. Due to the tapered design of the index tube notches, the collet piston mechanism will not impede rod insertion under normal insertion or scram conditions.

The collet retainer tube (CRT) is a short tube welded to the upper end of the CRD which houses the collet mechanism which consist of the locking collet, collet piston, collet return spring and an unlocking cam. The collet mechanism provides the locking/unlocking mechanism that allows the insert/withdraw movement of the control rod. The CRT has three primary functions: (a) to carry the hydraulic unlocking pressure to the collet piston, (b) to provide an outer cylinder, with a suitable wear surface for the metal collet piston rings, and (c) to provide mechanical support for the guide cap, a component which incorporates the cam surface for holding the collet fingers open and also provides the upper rod guide or bushing.

According to the BWROG, at the time of the first CRT crack discovery in 1975, each partially or fully withdrawn operable control rod was required to be exercised one notch at least once each week. It was recognized that notch testing provided a method to demonstrate the integrity of the CRT. Control rod insertion capability was demonstrated by inserting each partially or fully withdrawn control rod at least one notch, and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal.

It was determined that during scrams, the CRT temperature distribution changes substantially at reactor operating conditions. Relatively cold water moves upward through the inside of the CRT and exits via the flow holes into the annulus on the outside. At the same time, hot water from the reactor vessel flows downward on the outside surface of the CRT. There is very little mixing of the cold water flowing from the three flow holes into the annulus and the hot water flowing downward. Thus, there are substantial through wall and circumferential temperature gradients during scrams which contribute to the observed CRT cracking.

Subsequently, many BWRs have reduced the frequency of notch testing for partially withdrawn control rods from weekly to monthly. The notch test frequency for fully withdrawn control rods are still performed weekly. The change, for partially withdrawn control rods, was made because of the potential power reduction required to allow control rod movement for partially withdrawn control rods, the desire to coordinate scheduling with other plant activities, and the fact that a large sample of control rods are still notch tested on the weekly basis. The operating experience related to the changes in CRD performance also provided additional justification to reduce the notch test frequency for the partially withdrawn control rods.

In response to the NRC staff RAIs and to support their position to reduce the CRD notch testing frequency, the BWROG provided plant data and GE Nuclear Energy report, CRD Notching Surveillance Testing for Limerick Generating Station. The GE report provided a description of the cracks noted on the original design CRT surfaces. These cracks, which were later determined to be intergranular, were generally circumferential, and appeared with greatest frequency below and between the cooling water ports, in the area of the change in wall thickness. Subsequently, cracks associated with residual stresses were also observed in the vicinity of the attachment weld. Continued circumferential cracking could lead to 360 degree severance of the CRT that would render the CRD inoperable which would prevent insertion, withdrawal or scram. Such failure would be detectable in any fully or partially withdrawn control rod during the surveillance notch testing required by the TSs. To a lesser degree, cracks have also been noted at the welded joint of the interim design CRT, but no cracks have been observed in the final improved CRT design. In a RAI, BWROG response of being unable to find a collet housing failure since 1975 supported the NRC staff review of not finding a collet housing failure. To date, operating experience data shows no reports of a severed CRT at any BWR. No collet housing failures have been noted since 1975. On a numerical basis for instance, based on BWROG assumption that there are 137 control rods for a typical BWR/4 and 193 control rods for a typical BWR/6, the yearly performance would be 6590 rod notch tests for a BWR/4 plant and 9284 for a BWR/6 plant. For example, if all BWRs operating in the U.S. are taken into consideration, the yearly performances of rod notch data would translate into approximately 240,000 rod notch tests without detecting a failure.

In addition, the intergranular stress-corrosion cracking (IGSCC) crack growth rates were evaluated, at Limerick Generating Station, using GE's PLEDGE model with the assumption that the water chemistry condition is based on GE recommendations. The model is based on fundamental principles of stress corrosion cracking which can evaluate crack growth rates as a function of water oxygen level, conductivity, material sensitization and applied loads. It was determined that the additional time of 24 days represented an additional 10 mils of growth in total crack length. The small difference in growth rate would have little effect on the behavior between one notch test and the next subsequent test. Therefore, from the materials perspective based on low crack growth rates, a decrease in the notch test frequency would not affect the reliability of detecting a CRDM failure due to crack growth.

Also, the BWR scram system has extremely high reliability. In addition to notch testing, scram time testing can identify failure of individual CRD operation resulting from IGSCC-initiated cracks and mechanical binding. Unlike the CRD notch tests, these single rod scram tests cover the other mechanical components such as scram pilot solenoid operated valves, the scram inlet and outlet air operated valves, and the scram accumulator, as well as operation of the control rods. Thus, the primary assurance of scram system reliability is provided by the scram time testing since it monitors the system scram operation and the complete travel of the control rod.

Also, the HCUs, CRD drives, and control rods are tested during refueling outages, approximately every 18-24 months. Based on the data collected during the preceding cycle of operation, selected control rod drives, are inspected and, as required, their internal components are replaced. Therefore, increasing the CRD notch testing frequency to monthly would have very minimal impact on the reliability of the scram system.

The NRC staff has reviewed the TSTF-475 proposal to amend the TS SR 3.1.3.2, "Control Rod OPERABILTY" from 7 days to monthly. Based on the following evaluation condition: (1) slow crack growth rate of the CRT; (2) the improved CRT design; (3) a higher reliable method (scram time testing) to monitor CRD scram system functionality; (4) GE chemistry recommendations; and (5) no known CRD failures have been detected during the notch testing exercise, the NRC staff concluded that the changes would reduce the number of control rod manipulations thereby reducing the opportunity for potential reactivity events while having a very minimal impact on the extremely high reliability of the CRD system.

The NRC staff has reviewed the TSTF-475 proposal to amend Example 1.4-3 in Section 1.4 “Frequency,” to make the 1.25 provision in SR 3.0.2 to be equally applicable to time periods specified in the “FREQUENCY” column and in the NOTE in the “SURVEILLANCE” column. The NRC staff finds this change acceptable since the revision would make it consistent with the definition of specified “Frequency” provided in the second paragraph of Section 1.4 which states that the specified “Frequency” is referred to throughout this section and each of the Specifications of Section 3.0, SR Applicability. The specified “Frequency” consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.”

3.1 Summary

The NRC staff has reviewed the licensee’s proposal to amend existing TS Sections SR 3.1.3.2, “Control Rod OPERABILTY,” and Example 1.4-3, “Frequency” applicable to SR 3.0.2. The NRC staff has concluded that the licensee’s proposed changes, in accordance with TSTF-475, are acceptable for Fermi 2 because, as described above, the TS revisions will have a minimal affect on the high reliability of the CRD system while reducing the opportunity for potential reactivity events; thus, meeting the requirement of 10 CFR Part 50, Appendix A, GDC 29, and will clarify the 1.25 provision in SR 3.0.2. Therefore, the NRC staff concludes that the amendment request is 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 and changes SRs. 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 that 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 (73 FR 10296). 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.

7.0 REFERENCES

1. Letter TSTF-04-07 from the Technical Specifications Task Force to the NRC, TSTF-475 Revision 0, “Control Rod Notch Testing Frequency and SRM Insert Control Rod Action,” August 30, 2004, ADAMS Accession No. ML042520035.

2. Deleted.

3. Deleted.

4. Deleted.

5. NUREG-1433, “Standard Technical Specifications General Electric Plants, BWR/4, Revision 3,” August 31, 2003.

6. Deleted.

7. Letter TSTF-07-19, Response from the TSTF to the NRC, “Request for Additional Information (RAI) Regarding TSTF-475 Revision 0, “Control Rod Notch Testing Frequency and SRM Insert Control Rod Action,” dated February 28, 2007, (TSTF-475 Revision 1 is an enclosure), ADAMS Accession No. ML071420428.

8. Letter TSTF-06-13 from the Technical Specifications Task Force to the NRC, “Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0,” dated July 3, 2006, ADAMS Accession No. ML0618403421.

9. Letter BWROG-06036 from the BWR Owners Group to the NRC, “Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0,” dated November 16, 2006, with Enclosure of the GE Nuclear Energy Report, “CRD Notching Surveillance Testing for Limerick Generating Station,” dated November 16, 2006, ADAMS Accession No. ML063250258.

10. Letter TSTF-07-19 from the Technical Specifications Task Force to the NRC, “Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0,” dated May 22, 2007, ADAMS Accession No. ML071420428.

Principal Contributor: Ravi Grover, NRR/DIRS/ITSB

Date: June 23, 2008

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June 22, 2007

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