July
Ju 31, 1998

Mr. Douglas R. Gipson
Senior Vice President
Nuclear Generation
Detroit Edison Company
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: FERMI 2 - ISSUANCE OF AMENDMENT RE: SAFETY/RELIEF VALVE (SRV) SETPOINT TOLERANCE CHANGE (TAC NO. MA0720)

Dear Mr. Gipson:

The Commission has issued the enclosed Amendment No. 123 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 28, 1998 (NRC-98-0011) as supplemented March 12 and June 9, 1998.

The amendment revises Technical Specification (TS) 3.4.2.1, "Safety/Relief Valves," changing the SRV setpoint tolerance from ±1 percent to ±3 percent. An associated footnote is revised to indicate that, although the as-found setpoint tolerance is now ±3 percent, the as-left settings of the SRVs shall be within ±1 percent of the specified setpoints prior to installation of the SRVs after testing. Bases section 3/4.4.2 is also revised.

Because full implementation of this amendment may not take place until the fall of 1998, until full implementation Detroit Edison should submit two sets of TS pages for any pages affected in future amendments by the issuance of this amendment. The TS pages should reflect the conditions before and after full implementation of this amendment so that the correct TS pages may be issued in any future amendments. The NRC also requests that you submit a letter informing the staff when this amendment is fully implemented.

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,

ORIG. SIGNED BY A. KUGLER

Andrew J. Kugler, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

DFO1

Docket No. 50-341
Enclosures: 1. Amendment No. 123 to NPF-43
2. Safety Evaluation
cc w/encl: See next page
DISTRIBUTION: See attached page

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DATED: July 31, 1998

AMENDMENT NO. 123 TO FACILITY OPERATING LICENSE NO. NPF-43 - FERMI 2

Docket File (50-341)
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DETROIT EDISON COMPANY

DOCKET NO. 50-341

FERMI 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 123
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 28, 1998, as supplemented March 12 and June 9, 1998, 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. 123, 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 with full implementation prior to restart from the sixth refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Andrew J. Kugler, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: July 31, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 123

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 vertical lines indicating the area of change.

REMOVE	INSERT
3/4 4-7	3/4 4-7
B 3/4 4-1a	B 3/4 4-1a

3/4.4.2 SAFETY/RELIEF VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 The safety valve function of at least 11 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*

5 safety/relief valves @ 1135 psig ±3%
5 safety/relief valves @ 1145 psig ±3%
5 safety/relief valves @ 1155 psig ±3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of less than 11 of the above safety/relief valves OPERABLE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 95°F, close the stuck open safety/relief valve(s); if unable to close the stuck open valve(s) within 2 minutes or if suppression pool average water temperature is 95°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief valve position indicators inoperable, restore the inoperable indicator(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1.1 The valve position indicator for each safety/relief valve shall be demonstrated OPERABLE with the pressure setpoint of each of the tail-pipe pressure switches verified to be 30 ± 5 psig by performance of a CHANNEL CALIBRATION at least once per 18 months.

4.4.2.1.2 At least 1/2 of the safety relief valves shall be set pressure tested at least once per 18 months, such that all 15 safety relief valves are set pressure tested at least once per 40 months.

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. Although the as-found lift setting tolerance is ±3%, the as-left lift settings shall be within ±1% of the specified setpoints prior to installation following testing.

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 77% 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. Although the safety/relief valves are tested to demonstrate that opening pressures are within ±3% of the nominal pressure setpoints, they are adjusted to within ±1% of the nominal pressure setpoints prior to reinstallation.

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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 123 TO FACILITY OPERATING LICENSE NO. NPF-43

DETROIT EDISON COMPANY

FERMI 2

DOCKET NO. 50-341

1.0 INTRODUCTION

By letter dated January 28, 1998 (NRC-98-0011) as supplemented March 12 and June 9, 1998, the Detroit Edison Company (DECo or the licensee) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License No. NPF-43 for Fermi 2. The proposed amendment would revise Technical Specification 3.4.2.1, "Safety/Relief Valves," changing the safety relief valve (SRV) setpoint tolerance from ±1 percent to ±3 percent. An associated footnote would be revised to indicate that, although the as-found setpoint tolerance is ±3 percent, the as-left settings of the SRVs shall be within ±1 percent of the specified set points prior to installation of the SRVs after testing. Bases Section 3/4.4.2 would also be revised. The March 12 and June 9, 1998, letters provided clarifying information that was within the scope of the original *Federal Register* notice and did not change the staff's initial proposed no significant hazards considerations determination.

2.0 BACKGROUND

The proposed change does not alter the SRV lift set points, the SRV lift setpoint test frequency, or the number of SRVs required to be operable. Also, the proposed change requires the as-left safety valve function settings to be within ±1 percent of the specified nominal lift set points prior to installation after testing. The staff has previously granted plant-specific approval to individual boiling water reactors (BWRs) to increase the as-found SRV tolerance to ±3 percent. The bases for the approval of these plant-specific amendments were the plant-specific submittals and a staff safety evaluation (SE) dated March 8, 1993, for licensing topical report (LTR) NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report" (Proprietary information. Not publicly available.), dated February 1990, evaluating the setpoint tolerance increase. The staff SE included six conditions which must be addressed on a plant-specific basis for licensees applying for the increased SRV setpoint tolerance:

(a) Transient analysis of all abnormal operational occurrences as described in NEDC-31753P should be performed utilizing a ±3 percent tolerance for the safety mode of spring safety valves (SSVs) and SRVs. In addition, the standard reload methodology (or other method approved by the staff) should be used for this analysis.

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(b) Analysis of the design-basis overpressurization event using the 3 percent tolerance limit is required to confirm that the vessel pressure does not exceed the American Society of Mechanical Engineers (ASME) pressure vessel code upset limit.

(c) The plant-specific analysis described in items (a) and (b) should assure that the number of SSVs and SRVs, and relief valves (RVs) included in the analyses correspond to the number of valves required to be operable in the TS.

(d) Reevaluation of the performance of high pressure systems (pump capacity, discharge pressure, etc.,), motor-operated valves, and vessel instrumentation and associated piping must be completed, considering the 3 percent tolerance limit.

(e) Evaluation of the effect of the ±3 percent tolerance on any plant-specific operating modes (e.g., increased core flow, extended operating domain, etc.) should be completed.

(f) Evaluation of the effect of the 3 percent tolerance limit on the containment response during loss-of-coolant accidents (LOCAs) and the hydrodynamic loads on the SRV discharge lines and containment should be completed.

3.0 EVALUATION

The safety objective of the SRVs is to prevent overpressurization of the nuclear system. This protects the nuclear system process barrier from failure which could result in the uncontrolled release of fission products. The pressure relief system at Fermi 2 includes fifteen SRVs, arranged into three setpoint groupings of five each: one group of SRVs set at 1135 psig, a second group of SRVs set at 1145 psig, and a third group of SRVs set at 1155 psig. Existing TS require a ±1 percent as-found setpoint tolerance and ±1 percent as-left setpoint tolerance. The proposed TS changes would require a ± 3 percent as-found setpoint tolerance and a ±1 percent as-left setpoint tolerance. The licensee's submittal was evaluated against the generic SE described above.

3.1 Transient Analysis/Reload Methodology

The licensee must consider the impact of the tolerance increase on abnormal operational transients (AOTs). For Fermi 2, the licensee performed analyses of AOTs using the ±3 percent tolerance assuming 11 of the total 15 SRVs in service (the minimum required by TS). The transient that generates the limiting reduction in critical power ratio is the feedwater controller failure with the turbine bypass system and moisture separator reheater out of service and reduced feedwater temperature.

Based on the analysis, the thermal limits of the limiting transient would not be affected by the relaxation of SRV setpoint tolerance because the limiting value of the thermal limit occurs before the SRVs open. Further, other transient events remain non-limiting and bounded by the above event. The NRC-approved licensing analysis methodology (as documented in the licensee's October 14, 1997, letter forwarding the Fermi 2 Cycle 6 Core Operating Limits Report) was used for the analysis. The staff reviewed the results of these analyses and finds them acceptable.

The licensee also evaluated the uncertainty associated with the testing of the SRVs and its effect on the transient analysis to support this TS change. The licensee stated that ANSI/ASME Standard PTC-25.3-1976 requires a setpoint testing uncertainty of less than 0.5 percent and that vendors who perform the testing typically use instruments with accuracies of ±0.1 percent to ensure that this requirement is met. The licensee further stated that this uncertainty equates to approximately 1 psi of instrument inaccuracy which is small compared to the allowed positive as-found setpoint drift of approximately 34 psi. Therefore, the licensee concluded that it is not necessary to account for this uncertainty in the analysis. The staff reviewed the licensee's evaluation and finds that the setpoint testing uncertainty is small compared to both the allowable tolerance and the margin in the analysis and that this is acceptable.

3.2 Analysis of the Design-Basis Overpressurization Event

The licensee is required to reevaluate the limiting design-basis pressurization transient using the 3-percent tolerance limit to confirm that the vessel pressure does not exceed the ASME pressure vessel code upset limit. The ASME Boiler and Pressure Vessel Code Section III permits pressure transients up to 10 percent over design pressure (110 percent x 1250 psig = 1375 psig). The limiting pressurization AOT analyzed is a main steam isolation valve (MSIV) closure event occurring at the end of full power life without credit for reactor trip on MSIV position sensing. The licensee analyzed the MSIV closure event with the 3-percent tolerance and calculated the maximum vessel pressure to be 1315 psig. This is within the 1375-psig ASME limit and is acceptable to the staff.

3.3 TS Operability Statement for SRVs

The licensee has stated that all plant-specific analyses have been conducted with the number of SRVs included in the analyses corresponding to the number of valves required to be operable in TS. The analyses took credit only for 11 of the 15 SRVs required by the TS. This is acceptable to the staff.

As reported in several Licensee Event Reports (LERs), the Fermi 2 SRVs have experienced several occurrences of positive setpoint drift in excess of the ±3 percent used in the licensee's analysis to support this TS change. As a corrective action, the licensee has stated that it will install platinum ion-beam bombarded pilot valve disks in the plant SRVs during the next (sixth) refueling outage currently scheduled to begin in August 1998. The licensee believes this modification will allow the plant SRVs to meet the proposed TS setpoint tolerance of ±3 percent during future refueling cycles. The licensee also states that, should the TS limits be exceeded, it would be required to analyze the resultant potential overpressurization transient and notify the NRC via an LER. The platinum ion-beam bombarded pilot valve disks have significantly reduced the SRV setpoint drift experienced at another BWR plant site having SRVs of the same design as those at Fermi 2. The staff reviewed the licensee's planned near-term corrective action and finds that it is appropriate and consistent with the proposed TS change to the SRV setpoint tolerance.

3.4 Reevaluation of the Performance of High Pressure Systems

3.4.1 System Performance

The licensee must also reevaluate performance of high pressure systems (pump capacity, discharge pressure, etc.), considering the 3-percent tolerance limit. Fermi 2 has three systems whose performance could be affected by the increased pressure and that are required to inject to the vessel at high pressure conditions: high pressure coolant injection (HPCI), reactor core isolation cooling (RCIC) and standby liquid control (SLC). The most significant impact is the increased reactor pressure specified for system operation. The systems' performances were evaluated for the new reactor pressure of 1169 psig from 1146 psig. The HPCI system was determined to have the capability to inject its design flow of 5000 gallons per minute to the vessel at the new maximum pressure of 1169 psig without any changes. The RCIC turbine/pump maximum speed is increased from 4550 rpm to 4600 rpm in order for the RCIC system to perform at the new maximum reactor operating pressure. The increased speed reduces the overspeed margin from 125 percent to 122.3 percent. The licensee considered the reduction in margin acceptable due to the system modifications made in the turbine start feature. The licensee also determined that the SLC system was capable of injecting boron into the vessel at its design flow rate.

3.4.2 Evaluation of Motor-Operated and Air-Operated Valves

In support of the SRV and SV tolerance increase from ±1 percent to ±3 percent, the licensee reviewed the motor-operated valves (MOVs) and the air-operated valves (AOVs) in the HPCI, RCIC, and SLC systems. The licensee determined that all such MOVs and AOVs can operate within their performance capability.

3.4.3 Evaluation of Piping

The licensee also evaluated the effects of the higher pressures associated with the increased setpoint tolerance on the instrumentation and piping for the HPCI, RCIC, and SLC systems. The licensee determined that, with the exception of the calibration change for the increased speed of the RCIC turbine, no changes to instrumentation will be required. The licensee also determined that the impact of the higher pressure on system piping and other components was negligible.

3.4.4 Conclusion for High Pressure Systems

The staff reviewed the licensee's evaluation of the effect of the proposed change in SRV setpoint tolerance on high pressure systems and finds that, with the change in the speed of the RCIC turbine, the HPCI, RCIC, and SLC systems will remain capable of performing their design functions considering the higher pressures associated with the change in the SRV setpoint tolerance.

3.5 Alternate Operating Modes

The licensee must also evaluate the increased tolerance on any plant-specific alternate operating modes (e.g., increased core flow, extended operating domain, etc.) The licensee's analyses considered the currently approved operating domains: maximum extended operating domain, increased core flow and single-loop operation. The staff finds that the consideration of these alternate operating modes in the analyses is acceptable.

3.6 Containment Response / Hydrodynamic Loads

The licensee must also evaluate the effect of the increased tolerance limit on (1) the containment hydrodynamic loads during LOCAs and (2) the hydrodynamic loads on the SRV discharge lines and the suppression chamber.

The licensee examined the potential effects of the proposed amendment on the containment design limits. The containment design-basis accident is a double-ended break at the suction of a recirculation pump. For this event, the reactor coolant system depressurizes very rapidly and thus, the SRVs are not challenged. Also, the reactor coolant system inventory and primary system heat sources that would contribute to the containment mass and energy are not increased. The setpoint tolerance thus has no effect on the capability of the containment to perform its design-basis safety function (i.e., the containment peak temperature and pressure loads would not be adversely affected). The staff notes that small break LOCAs also would not lead to increased reactor coolant system pressure and subsequent SRV challenges.

An increase in SRV setpoint tolerance involves a potential increase in SRV discharge dynamic and hydrodynamic loads on the SRV discharge piping and the torus. The licensee analyzed the loads and compared the increases to the margins determined in the Fermi 2 Plant Unique Analysis Report. The results demonstrated that the increased torus loads are acceptable for all SRVs. Similarly, the increases in the loads on the SRV piping and main steam lines due to the increased SRV setpoint tolerance were evaluated and found to be acceptable.

The staff reviewed the information provided by the licensee concerning the containment response and the dynamic and hydrodynamic loads on the SRV discharge piping and torus and finds that the results are acceptable.

3.7 Emergency Core Cooling System (ECCS) LOCA Performance

The Fermi 2 LOCA analyses (NEDC-31982P, "Fermi-2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," July 1991, proprietary information - not publicly available) were reviewed by the licensee to determine the effect of an increase in SRV opening pressures on ECCS performance. The limiting break LOCA, the design-basis accident reactor recirculation line break, the small break and the steam line break outside containment events were evaluated to determine the effects of the increased SRV setpoint tolerance. Peak cladding temperatures for the small break and the steam line break outside containment are not changed significantly and they are non-limiting. The increased SRV setpoint tolerance has no effect on the recirculation line break analyses because the SRVs do not actuate in that event. The acceptance criteria given in 10 CFR 50.46 are still satisfied for all break sizes and locations. The staff reviewed the information provided by the licensee and concludes that the setpoint tolerance change is acceptable with respect to ECCS-LOCA considerations.

3.8 Effect on Anticipated Transient Without Scram (ATWS) Events

The main steam isolation valve closure under ATWS conditions was reevaluated to support the current condition of 11 of the 15 SRVs operable with the requested relaxation in SRV setpoint tolerance to ±3 percent. The results of the analysis, using the ODYN code, show that the vessel pressure reaches a maximum of 1457 psig, which is within the vessel overpressure criterion of 1500 psig for ATWS events. The staff determined that the use of the ODYN code for this application is acceptable in this case. The long-term effect on suppression pool temperature due to ±3 percent SRV tolerance is negligible because there is little change in the total energy discharged to the pool. The staff concludes that the results of the analyses are acceptable.

3.9 Technical Specification Changes

In TS 3.4.2.1, "Safety/Relief Valves," the setpoint tolerance for the SRVs is changed from ±1 percent to ±3 percent. In addition, the following is added to the existing footnote:

> Although the as-found lift setting tolerance is ±3%, the as-left lift settings shall be within ±1% of the specified setpoints prior to installation following testing.

These changes are acceptable to the staff as described in this SE.

In TS Bases 3/4.4.2, the following sentence is added:

> Although the safety/relief valves are tested to demonstrate that opening pressures are within ±3% of the nominal pressure setpoints, they are adjusted to within ±1% of the nominal pressure setpoints prior to reinstallation.

This change is acceptable to the staff as described in this SE.

3.10 Conclusion

Based on the information provided by the licensee, the staff concludes that the plant will continue to satisfy the acceptance criteria for the limiting pressurization transient, AOTs, and design-basis accidents. The staff concludes that the proposed changes are 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 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 (63 *FR* 9600). 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 Contributors: G. Thomas
C. Hammer

Date: July 31, 1998