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: ONE-TIME TECHNICAL SPECIFICATION CHANGE TO ALLOW EXTENSION OF THE FERMI 2 OPERATING CYCLE (TAC NO. M93689)

Dear Mr. Gipson:

The Commission has issued the enclosed Amendment No. 106 to Facility Operating License No. NPF-43 for the Fermi-2 facility. The amendment consists of changes to the Technical Specifications (TS) in response to your letter dated September 20, 1995, as supplemented by letters dated December 18 and 22, 1995.

The amendment revises TS related to system testing, instrumentation calibration, component inspection, component testing, response time testing, and logic system functional tests to allow a one-time extension of the surveillance intervals to support extending the Fermi 2 operating cycle until September 27, 1996. The cycle had previously been scheduled to end with the start of refueling outage (RFO) 5 in March 1996.

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,

Original signed by:
Timothy G. Colburn, Sr. Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-341

Enclosures: 1. Amendment No. 106 to NPF-43
2. Safety Evaluation

cc w/encl: See next page

with additions to SE noted

DOCUMENT NAME: G:\WPDOCS\FE93689.AMD *See previous concurrence

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

March 1, 1996

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: ONE-TIME TECHNICAL SPECIFICATION CHANGE TO ALLOW EXTENSION OF THE FERMI 2 OPERATING CYCLE (TAC NO. M93689)

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Docket No. 50-341

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1. Amendment No. 106 to NPF-43
2. Safety Evaluation

cc w/encl: See next page

Mr. Douglas R. Gipson
Detroit Edison Company

Fermi-2

cc:

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Ms. Lynne S. Goodman
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Detroit Edison Company
Fermi-2
6400 North Dixie Highway
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DATED: March 1, 1996

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

Docket File
PUBLIC
PD3-1 Reading
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G. Hill, IRM (2)
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M. Jordan, RIII
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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. 106
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 September 20, 1995, as supplemented December 18 and 22, 1995, 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. 106 , 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 within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Timothy G. Colburn

Timothy G. Colburn, Sr. 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: March 1, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 106

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
xxii	xxii
3/4 0-1*	3/4 0-1*
3/4 0-2	3/4 0-2
3/4 0-3*	3/4 0-3*
--	3/4 0-4
--	3/4 0-5
--	3/4 0-6
--	3/4 0-7
3/4 6-7	3/4 6-7
3/4 6-8*	3/4 6-8*
3/4 7-10a	3/4 7-10a

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

LIST OF TABLES

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL CONDITIONS or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in an OPERATIONAL CONDITION in which the Specification does not apply by placing it, as applicable, in:

1. At least STARTUP within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

This Specification is not applicable in OPERATIONAL CONDITION 4 or 5.

3.0.4 Entry into an OPERATIONAL CONDITION or other specified condition shall not be made when the conditions for the Limiting Conditions for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL CONDITION or other specified condition may be made in accordance with the ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval. For the purpose of the fifth refueling outage, those Surveillance Requirements listed on Table 4.0.2-1 and 4.0.2-2 are extended to the date specified in the table.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

f. The Inservice Inspection (NDE) Program for piping identified in NRC Generic Letter 88-01, dated January 25, 1988, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping", shall be performed in accordance with the staff positions on schedule, methods and personnel, and sample expansion included in this generic letter.

TABLE 4.0.2-1

SURVEILLANCE TEST INTERVALS EXTENDED TO OCTOBER 5, 1996

SURVEILLANCE REQUIREMENT	DESCRIPTION
4.1.3.1.4.a	Scram discharge vol. vent and drain valve operability
4.3.1.1, Table 4.3.1.1-1, Item 3	RPS Rx Steam Dome Press High cal.
4.3.1.1, Table 4.3.1.1-1, Item 4	RPS Rx Low Water Level - Level 3 cal
4.3.1.1, Table 4.3.1.1-1, Item 5	RPS MSIV Closure cal
4.3.1.1, Table 4.3.1.1-1, Item 6	RPS Main Steam Line Radiation High cal
4.3.1.1, Table 4.3.1.1-1, Item 7	RPS Drywell Pressure High cal
4.3.1.3[a]	RPS Response Time Test
4.3.2.1, Table 4.3.2.1-1, Item 1.a.1	Pri Cont Isolation Actuation Rx Water Low Level - Level 3 cal
4.3.2.1, Table 4.3.2.1-1, Item 1.a.2	Pri Cont Isolation Actuation Rx Water Low Level - Level 2 cal
4.3.2.1, Table 4.3.2.1-1, Item 1.a.3	Pri Cont Isolation Actuation Rx Water Low Level - Level 1 cal
4.3.2.1, Table 4.3.2.1-1, Item 1.b	Pri Cont Isolation Actuation Drywell Press High cal
4.3.2.1, Table 4.3.2.1-1, Item 1.c.1	Pri Cont Isolation Actuation Main Steam Line Radiation High cal
4.3.2.1, Table 4.3.2.1-1, Item 1.c.2	Pri Cont Isolation Actuation Main Steam Line Press Low cal
4.3.2.1, Table 4.3.2.1-1, Item 1.d	Pri Cont Isolation Actuation Main Steam Line Tunnel Temp. High cal
4.3.2.1, Table 4.3.2.1-1, Item 1.e	Pri Cont Isolation Actuation Condenser Press High cal
4.3.2.1, Table 4.3.2.1-1, Item 1.f	Pri Cont Isolation Actuation Turbine Bldg. Area Temp. High cal
4.3.2.1, Table 4.3.2.1-1, Item 2.e	RWCU Isolation Rx Water Low Level - Level 2 channel cal
4.3.2.1, Table 4.3.2.1-1, Item 2.g	RWCU Manual Initiation channel functional test
4.3.2.1, Table 4.3.2.1-1, Item 3.a.1	RCIC Steam Line Flow High DP channel cal
4.3.2.1, Table 4.3.2.1-1, Item 3.a.2	RCIC Steam Line Flow High Time Delay cal
4.3.2.1, Table 4.3.2.1-1, Item 4.a.1	HPCI Steam Line Flow High DP cal
4.3.2.1, Table 4.3.2.1-1, Item 4.a.2	HPCI Steam Line Flow High Time Delay cal
4.3.2.1, Table 4.3.2.1-1, Item 4.e	HPCI Manual Initiation functional test
4.3.2.1, Table 4.3.2.1-1, Item 5.a	RHR S/D Cooling Rx Water Level Low - Level 3 cal
4.3.2.1, Table 4.3.2.1-1, Item 6.b	Sec. Cont. Isolation - Drywell Press High channel cal
4.3.2.3[a]	Isolation Actuation Inst. System Response Time
4.3.3.1, Table 4.3.3.1-1, Item 1.b	CS Drywell Press High Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.b	LPCI Drywell Press High Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.f	LPCI Riser Differential Pressure High Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.g	LPCI Recirc. Pump Differential Pressure High Cal
4.3.3.1, Table 4.3.3.1-1, Item 3.b	HPCI Drywell Press High Cal
4.3.3.1, Table 4.3.3.1-1, Item 3.f	HPCI Manual Initiation
4.3.3.1, Table 4.3.3.1-1, Item 4.f	ADS RPV Low Level 3 Cal
4.3.3.1, Table 4.3.3.1-1, Item 4.i	ADS Manual Inhibit Functional Test
4.3.4, Table 4.3.4-1, Item 2	RPV Press High Cal (ATWS)
4.3.7.4.1, Table 4.3.7.4.-1, Item 1	RPV Press Cal - Remote Shutdown
4.3.7.5, Table 4.3.7.5-1, Item 1	RPV Press Cal Accident Mon.
4.3.7.5, Table 4.3.7.5-1, Item 11	SRV Position Indic Cal Accident Mon.
4.3.7.5, Table 4.3.7.5-1, Item 12	CTMT High Range Rad Monitoring Cal Accident Mon.
4.3.7.5, Table 4.3.7.5-1, Item 2.a	RPV Fuel Zone Level Cal Accident Mon
4.3.7.10.c	Loose Part Detection System Cal
4.3.9.1, Table 4.3.9.1-1, Item a	RPV High Water Level 8 Cal FW/Main Turbine Trip
4.3.9.2	FW/Main Turbine Trip LSFT
4.3.11.1, Table 4.3.11.1-1, Item 7	Alt S/D system Rx Water Level instrument operability
4.3.11.1, Table 4.3.11.1-1, Item 8	Alt S/D system Rx Press instrument operability
4.4.2.1.1	SRV Tail Pipe Pressure Switch Cal
4.4.2.1.2	SRV lift set point test
4.4.2.2.b	SRV Low Low Set Pressure setpoint Cal and LSFT
4.4.3.1.b	Drywell Sump Flow/Lvl Monitoring Cal
4.4.3.2.2.a	RCS Pressure Isol Valve Leak Test
4.5.1.d.2.a	ADS System Functional Test
4.6.1.2.b	Type B and C LLRT's
4.6.1.2.d	MSIV Leak Test
4.6.1.2.g	Hydrostatic Leak Test ECCS/RCIC Cont Isol Valves
4.6.1.4.d.3	MSIV LCS Press Inst. Cal and DP Calibration

TABLE 4.0.2-1

SURVEILLANCE TEST INTERVALS EXTENDED TO OCTOBER 5, 1996 Cont'd

SURVEILLANCE REQUIREMENT	DESCRIPTION
4.6.2.1.e	Suppression Chamber operability (visual inspection)
4.6.2.1.h	Suppression Chamber operability DW to torus bypass leak test
4.6.3.4	Instr. Excess Flow Check operability
4.6.3.5.b	TIP Explosive Squib operability test
4.6.4.1.b.2.a	Torus/Drywell vacuum breaker setpoint operability
4.6.4.1.b.2.b	Torus/Drywell vacuum breaker position indication cal
4.6.4.1.b.2.c	Torus/Drywell vacuum breaker switch opening gap
4.6.4.2.b.2.a	RB/Torus Vacuum Breaker operability (setpoint)
4.6.4.2.b.2.b	RB/Torus Vacuum Breaker operability (visual)
4.6.4.2.b.2.c	RB/Torus Vacuum Breaker position indication operability
4.7.11.4	Alternative Shutdown Control Circuit Functional Test
4.8.4.2.a.1.a	Primary Containment 4160 Volt Penetration Protective Relay Cal
4.8.4.2.a.1.b	Primary Containment 4160 Volt Penetration Protective Device Integrated Functional Test

TABLE NOTATIONS

(a) The surveillance interval of channels within the same trip system required to be tested at least once every N times 18 months, where N is the total number of channels in the trip system, may be based upon the performance of the surveillance during the fifth refueling outage.

TABLE 4.0.2-2

SURVEILLANCE TEST INTERVALS EXTENDED TO END OF REFUELING OUTAGE 5

SURVEILLANCE REQUIREMENT	DESCRIPTION
4.1.3.5.b.2	CR Accumulator Integrity Test (Check Valve Leakage)
4.1.5.d.1	SLCS operability Manual Initiation
4.1.5.d.2	SLCS pump Relief Valve operability
4.1.5.d.3	SLCS flow path demonstration
4.3.1.1, Table 4.3.1.1-1, Item 11	RPS Rx Mode Switch shutdown position functional
4.3.1.2	RPS Logic System Function Test
4.3.2.1, Table 4.3.2.1-1, Item 1.h	Pri Cont Isolation Actuation Manual Initiation Functional
4.3.2.1, Table 4.3.2.1-1, Item 2.d	RWCU - SLCS initiation channel functional test
4.3.2.1, Table 4.3.2.1-1, Item 5.c	RHR S/D Cooling Rx manual initiation functional test
4.3.2.1, Table 4.3.2.1-1, Item 6.a	Sec. Cont. Isolation - Rx Water Low Level - Level 2 cal
4.3.2.2	Isolation Actuation Inst. LSFT
4.3.3.1, Table 4.3.3.1-1, Item 1.a	CS RPV Low Level 1 Cal
4.3.3.1, Table 4.3.3.1-1, Item 1.c	CS Rx Steam Dome Press Low Cal
4.3.3.1, Table 4.3.3.1-1, Item 1.d	CS Manual Initiation
4.3.3.1, Table 4.3.3.1-1, Item 2.a	LPCI RPV Low Level 1 Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.c	LPCI Rx Steam Dome Press Low Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.d	LPCI Rx Low Level 2 Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.e	LPCI Rx Steam Dome Press Low Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.h	LPCI Manual Initiation
4.3.3.1, Table 4.3.3.1-1, Item 3.a	HPCI RPV Low Level 2 Cal
4.3.3.1, Table 4.3.3.1-1, Item 3.e	HPCI RPV High Level 8 Cal
4.3.3.1, Table 4.3.3.1-1, Item 4.a	ADS RPV Low Level 1 Cal
4.3.3.1, Table 4.3.3.1-1, Item 4.h	ADS Drywell Pressure High Bypass Timer
4.3.3.2	ECCS Logic System Functional Tests
4.3.3.3[(a)]	ECCS Response Time Tests
4.3.4, Table 4.3.4-1, Item 1	RPV Low Water Level 2 Cal (ATWS)
4.3.4.2	ATWS Logic System Functional Test
4.3.5.1, Table 4.3.5.1-1, Item a	RPV Low Level 2 Cal (RCIC)
4.3.5.1, Table 4.3.5.1-1, Item b	RPV High Level 8 Cal (RCIC)
4.3.5.2	RCIC Logic System Functional Test
4.3.6, Table 4.3.6-1, Item 5.b	Scram Disc. Vol. Trip Bypass Funct. Test
4.3.6, Table 4.3.6-1, Item 7	Rx Mode Switch Shutdown Pos. Rod Block Funct. Test
4.3.7.4.1, Table 4.3.7.4.-1, Item 2	RPV Level Cal - Remote Shutdown
4.3.7.5, Table 4.3.7.5-1, Item 16	CTMT Isolation Valve Position Cal Accident Mon
4.3.7.5, Table 4.3.7.5-1, Item 2.b	RPV Wide Range Level Cal Accident Mon
4.5.1.c.1	ECCS System Functional Test
4.6.3.2	Primary Containment Isol Valve operability
4.6.5.2.b	Secondary Containment Isolation Damper Actuation
4.7.1.2.b	ECCW Automatic Actuation
4.7.1.3.b	EESW Automatic Actuation
4.7.1.4.b	EDG Cooling Water Pump Automatic Actuation
4.7.2.1.c.1	CR Ventilation Filter Penetration
4.7.2.1.c.2	CR Ventilation Filter Charcoal Laboratory Analysis
4.7.2.1.c.3	CR Emergency Filtration System Flowrate
4.7.2.1.e.1	CR Ventilation Filter Pressure Drop
4.7.2.1.e.2	CR Emergency Filtration System Operational Mode Actuation
4.7.2.1.e.4	CR Emergency Makeup Inlet Heater Dissipation
4.7.5.e	Snubber Functional Test
4.8.1.1.2.e.1	EDG Inspection
4.8.1.1.2.e.2	EDG Load Rejection (1666 kW)
4.8.1.1.2.e.3	EDG Load Rejection (2850 kW)
4.8.1.1.2.e.4.a	EDG LOP Load Shedding
4.8.1.1.2.e.4.b	EDG LOP Auto Start and Load Sequencing

TABLE 4.0.2-2

SURVEILLANCE TEST INTERVALS EXTENDED TO END OF REFUELING OUTAGE 5 (Cont'd)

SURVEILLANCE REQUIREMENT	DESCRIPTION
4.8.1.1.2.e.5	EDG ECCS Auto Start
4.8.1.1.2.e.6.a	EDG LOP / ECCS Load Shedding
4.8.1.1.2.e.6.b	EDG LOP / ECCS Auto Start and Load Sequencing
4.8.1.1.2.e.7	EDG Non-essential Trip Bypass
4.8.1.1.2.e.8	EDG 24 Hour Run and Hot Fast Start.
4.8.1.1.2.e.9	EDG Auto Connect Load Verification
4.8.1.1.2.e.10	EDG Restoration of Offsite Power
4.8.1.1.2.e.11	EDG Auto Load Sequencer Timer
4.8.1.1.2.e.12.a	EDG 4160-volt ESF Bus Lockout
4.8.1.1.2.e.12.b	EDG Differential Trip Lockout
4.8.1.1.2.e.12.c	EDG Shutdown Relay Trip Lockout
4.8.2.1.c.3	130 VDC Battery Connections Resistance
4.8.2.1.c.4	130 VDC Battery Charger Functional Test
4.8.2.1.d	130 VDC Battery Capacity

TABLE NOTATIONS

(a) The surveillance interval of channels within the same trip system required to be tested at least once every N times 18 months, where N is the total number of channels in the trip system, may be based upon the performance of the surveillance during the fifth refueling outage.

SURVEILLANCE REQUIREMENTS (Continued)

g. ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months.

h. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Specification 4.6.1.8.2.

i. The provisions of Specification 4.0.2 are not applicable to Specifications 4.6.1.2a., 4.6.1.2b.* and 4.6.1.2c.

* Extension of Specification 4.6.1.2b per Table 4.0.2-1 for the fifth refueling outage is allowed.

PRIMARY CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each primary containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 56.5 psig.

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

ACTION:

a. With one primary containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the primary containment air lock inoperable, except as a result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

h. At least once per 36 months## by verifying that the sections of Control Room Emergency Filtration System duct listed in Table 4.7.2.1-1, when leak tested in accordance with ASME N510-1989# exhibit inleakage less than the acceptance criteria listed in Table 4.7.2.1-1 for the associated pressures.

4.7.2.2 The portions of the Control Room Emergency Filtration System duct listed below, which are accessible during normal operation, shall be visually inspected at least once per 366 days for cracking, debonding, or other abnormal degradation of the applied silicone sealant. Any such cracking, debonding, or other abnormal degradation shall be reported in accordance with Specification 6.9.2 within 14 days in a Special Report describing the findings and giving the intended course of action, including evaluation of and justification for continued plant operation.

a. Normal intake between damper T4100F042 and the Control Room wall (Penetration V-430)

b. Normal exhaust between damper T4100F044 and the Control Room wall (Penetration V-429)

c. Discharge of recirculation fans T4100C047, 48 between the discharge flanges on filter train T4100D016 and the 5th Floor CCHVAC Equipment Room wall (Penetration V-504B)

d. Division II supply plenum between the Control Room wall (Penetration V-431) and the 4th Floor Aux. Building ceiling (Penetration V-9014)

e. Emergency intake between the discharge flange on filter train T4100D011 and the inlet flange on filter train T4100D016

f. Recirculation duct between the 5th Floor CCHVAC Equipment Room wall (Penetration V-504A) and the inlet flange on filter train T4100D016

Tests performed in accordance with ANSI N510-1980 prior to the implementation of this requirement satisfy this requirement until the next required performance of the test.

##This surveillance requirement may be extended on a one-time basis to June 1, 1998.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. NPF-43

DETROIT EDISON COMPANY

FERMI-2

DOCKET NO. 50-341

1.0 INTRODUCTION

By letter dated September 20, 1995, as supplemented December 18 and 22, 1995, 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 extend the surveillance interval on a one-time basis for TS surveillances related to system testing, instrumentation calibration, component inspection, component testing, response time testing, and logic system functional tests. Two new tables, Table 4.0.2-1 and Table 4.0.2-2 would be added to indicate those surveillances which would be extended to October 5, 1996, and the end of refueling outage (RFO) 5, respectively. The proposed amendment would extend all 18-month surveillances that cannot be performed at power, and the maximum extension would be 227 days, which would be approximately the same interval as would be allowed by a permanent change to a 24-month operating cycle. Licensees are required by 10 CFR 50.36(c)(3) to provide Technical Specifications for surveillance requirements to assure that the necessary quality of systems and components is maintained. Guidance on proposed TS changes to accommodate a 24-month fuel cycle were provided by Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. The December 18, 1995, letter corrects a typographical error on one of the proposed TS pages and provides a corrected Table of Contents page to reflect the addition of the new Tables. The December 22, 1995, letter provides additional information on the licensee's review of historical plant drift data. This information was within the scope of the original application and did not change the staff's initial no significant hazards consideration determination.

The licensee needs the requested extension due to the lengthy turbine repair outage and extended startup and power ascension program following the December 25, 1993, turbine failure. These activities resulted in lower than targeted fuel burnup, which, in turn, resulted in a postponement of the spring 1996 refueling outage until September 27, 1996. If the plant were required to shut down solely to perform these surveillance tests, these surveillances would still be required to be performed during RFO5 beginning September 1996. The shutdown would place an unnecessary transient on the plant, challenge operators, and require unnecessary radiation exposure since these surveillances would be repeated approximately 6 months later.

The licensee had also submitted for staff review a related schedular exemption request from the 2-year schedular requirement of 10 CFR Part 50, Appendix J, to perform Types B and C containment leakage testing. This exemption was granted on December 18, 1995.

2.0 EVALUATION

To accommodate the one-time extension of surveillance intervals, the licensee has modified TS 4.0.2 to reference two new tables (Table 4.0.2-1 and Table 4.0.2-2) added to identify those surveillances that had been extended and the new late completion date for those surveillances. Those surveillances listed in Table 4.0.2-1 are required to be completed by October 5, 1996, and include all surveillances for equipment not required to be operable during RF05 (applicable in Condition(s) 1 or 1 and 2, or 1, 2, and 3). Those surveillances listed in Table 4.0.2-2 are required to be completed prior to startup from RF05, currently scheduled to end November 20, 1996, and include surveillances applicable during Condition 4 or 5 or other shutdown situations.

2.1 Scram Discharge Volume (SDV) Vent and Drain Valve Operability

The licensee has requested a 52-day extension for TS 4.1.3.1.4.a which requires demonstrating that the SDV vent and drain valves are operable at least once per 18 months by verifying that the vent and drain valves close within 30 seconds after receipt of a reactor protection system (RPS) signal to scram the control rods and open when the scram is reset. The SDV is normally vented to atmosphere through the normally open redundant vent and drain valves. After a scram an RPS signal closes the vent and drain valves to isolate the SDV and prevent the release of potentially contaminated water from the scram exhaust. The SDV vent and drain valves are tested to verify operation of the fail safe actuators and are full stroke exercised to the closed position once every 92 days. Stroke times are measured and verified to be less than 15 seconds. The logic signal generated by the RPS on a scram signal and which reopens the valves after the scram is reset is the only portion of the surveillance that needs an extension.

The licensee has referenced industry studies provided by the BWR [boiling water reactor] Owners Group (NEDC-30936P, "BWR Owners Group Technical Specification Improvement Methodology (w/Demonstration for BWR ECCS Actuation Instrumentation)") that show that the overall safety system's reliabilities are not dominated by the reliabilities of the logic system but by that of the mechanical components. The report concludes that since the probability of a relay or contact failure is small relative to the probability of a mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability. The licensee performed a historical search of the 18-month surveillance tests for each valve for the last two refueling outages to identify all failed or partially failed tests. None were identified for this test. The licensee also evaluated historical valve performance for the last two refueling outages and found it to be within acceptable limits. The licensee concluded that the impact of the extension on the vent and drain valves is small.

The staff has reviewed the information provided by the licensee and concludes that the proposed change would not have a significant effect on safety and generally meets the guidance contained in GL 91-04; therefore, the proposed one-time surveillance interval extension for TS 4.1.3.1.4.a is acceptable.

2.2 Scram Accumulator

The scram accumulator is one of the basic components of the control rod drive (CRD) hydraulic control unit. The scram accumulator stores sufficient energy to fully insert a control rod at lower reactor pressure vessel (RPV) pressures. At higher RPV pressures, the accumulator pressure is assisted by RPV pressure. The licensee requested a 71-day extension to TS 4.1.3.5.b.2 which requires demonstrating that the accumulator is operable at least once per 18 months by measuring and recording the time for at least 10 minutes that each individual accumulator check valve maintains the associated accumulator pressure above the alarm setpoint with no CRD pump operating.

The licensee has performed a review of the accumulator check valves' performance for the last two refueling outages and verified that there have been no failures to maintain pressure. The licensee concluded that the probability that a check valve would fail if there were no CRD pump operating was low.

The staff has reviewed the information provided by the licensee and concludes that the proposed change would not have a significant effect on safety; therefore, the proposed one-time surveillance interval extension for TS 4.1.3.5.b.2 is acceptable.

2.3 Standby Liquid Control System (SLCS)

The licensee has requested 138-day, 151-day, and 138-day extensions to TS 4.1.5, items d.1, d.2, and d.3, respectively. The SLCS is manually initiated from the control room and injects boron neutron absorber solution into the reactor as a backup shutdown feature if the reactor cannot be shut down using the control rods. The SLCS is designed with a redundant loop.

TS 4.1.5.d.1 requires that the SLCS be demonstrated operable at least once per 18 months by initiating one of the SLCS loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor vessel is available by pumping demineralized water into the reactor vessel. TS 4.1.5.d.2 requires demonstrating that the pump relief valve setpoint is less than or equal to 1400 psig and verifying that the relief valve does not actuate during recirculation to the relief tank. TS 4.1.5.d.3 requires demonstrating that all piping between the storage tank and the explosive valve is unblocked by pumping from the storage tank to the test tank and then draining and flushing the piping with demineralized water. These surveillances test the relief valves that provide system overpressure protection from the discharge of the positive displacement pumps. The current surveillance testing frequency exceeds the requirements of ASME (American Society for Mechanical Engineers) Section XI/OM-1 which states that all valves of a particular type be tested at least once per 10 years and that 20 percent of a valve type be tested within 48 months. The licensee's proposed frequency remains within these requirements. Additionally, if one of the relief valves lifted at too low a pressure, the check valve in that discharge line would prevent the other pump's flow from recycling back to the storage tank.

The licensee performed an evaluation of the system functional testing for the previous two refueling outages. No failures of the acceptance criteria were noted. Routine surveillances are performed including daily temperature monitoring of the sodium pentaborate solution and pump suction piping to detect insolubility of the solution, monthly valve position verification and explosive charge continuity verification, and quarterly pump testing. The explosive valves (squibs) are purchased in lots with samples tested prior to installation. The triggers for the explosive valves have service life beyond that of the requested extension. The daily temperature monitoring ensures that the minimum tank and piping temperature remains above 48 degrees F making it highly unlikely that the piping would become blocked. The quarterly pump testing raises system pressure to 1215 psig. It is expected that relief valve setpoint drift low would be detected by this more frequent pump test.

During RF04, the relief valves were bench tested for the first time to provide more accurate testing results than the previous in situ testing. One valve passed but the other lifted at a higher pressure than acceptable. The licensee reviewed previous calibration history and verified that the valves had required no previous adjustments since initial startup. Despite the failure, the relatively short duration of the extension does not create a significant probability that the valve would lift at higher pressure and, therefore, would not create a significant overpressurization concern with respect to piping integrity. Any failure to lift until higher pressure would not affect the primary safety function to inject sodium pentaborate solution into the reactor vessel.

The staff has reviewed the information provided by the licensee and has determined that the proposed changes would not have a significant effect on safety and generally meet the guidance contained in GL 91-04; therefore, the proposed one-time surveillance interval extensions for TS 4.1.5.d.1-3 are acceptable.

2.4 Calibration Interval Extensions

2.4.1 Reactor Protection System (RPS)

The licensee has requested one-time calibration interval extensions of 62 to 173 days for several RPS instrument channel calibrations required by TS Table 4.3.1.1-1. These include the RPS reactor steam dome pressure HIGH, reactor low water level, LEVEL 3, main steam isolation valve closure, main steam line radiation HIGH, and drywell pressure HIGH instrumentation functional units. These signals initiate an automatic reactor shutdown (scram) if the monitored system variables exceed pre-established limits in order to prevent fuel damage and limit system pressure.

2.4.2 Isolation Actuation Instrumentation

The licensee has requested one-time calibration interval extensions of 128 to 183 days for several isolation actuation instrument channel calibrations required by TS Table 4.3.2.1-1. The containment and reactor vessel isolation control system (CRVICS) includes the instrument channels, trip logics, and actuation circuits that automatically initiate valve closure providing isolation of the containment and/or reactor vessel and initiation of systems provided to limit the release of radioactive materials. When abnormal conditions are sensed, instrument channel relay contacts open and deenergize the normally energized trip logic and thereby initiate isolation. Once initiated, the CRVICS trip logics seal in and may only be reset by the operator when initiating conditions have returned to normal. The specific primary containment isolation actuation calibrations affected by this request include the reactor water low level, LEVEL 3, LEVEL 2, and LEVEL 1 calibrations; the main steam line radiation HIGH, pressure LOW, and tunnel temperature HIGH calibrations; the drywell pressure HIGH calibration; the condenser pressure HIGH calibration; and the turbine building area temperature HIGH calibration. Other calibrations affected include the reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) steam line flow HIGH DP [differential pressure] and time delay calibrations; the reactor water cleanup (RWCU) isolation on reactor water low level LEVEL 2 channel calibration; and the residual heat removal shutdown cooling reactor water level low LEVEL 3 calibration. In addition, secondary containment isolation calibrations on reactor water low level LEVEL 3 and drywell pressure HIGH are affected by the proposed change.

2.4.3 Emergency Core Cooling System (ECCS) Actuation Instrumentation

The licensee has requested one-time calibration interval extensions of 169 to 223 days for several ECCS actuation instrument channel calibrations required by TS Table 4.3.3.1-1. The objective of the ECCS, in conjunction with the containment, is to limit the release of radioactive materials should a loss-of-coolant accident (LOCA) occur so that resulting radiation levels are kept within the guideline values given in 10 CFR Part 100. The purpose of the ECCS instrumentation is to initiate appropriate responses from the systems to ensure that the fuel is adequately cooled in the event of a design-basis accident or transient. If the monitored parameters exceed the established setpoints, logic signals are generated to initiate ECCS and ECCS support systems. The specific calibrations affected include those for core spray (CS), low pressure coolant injection (LPCI), and the automatic depressurization system (ADS) RPV water level low LEVEL 1 calibrations; LPCI and HPCI RPV low level LEVEL 2 calibrations; the ADS RPV low level LEVEL 3 calibration; CS, LPCI, and HPCI drywell pressure HIGH, and the ADS drywell pressure HIGH bypass timer calibrations; CS and LPCI reactor steam dome pressure LOW calibrations (both for ECCS permissive injection and LPCI loop select logic); the LPCI Riser DP HIGH and recirc pump DP HIGH calibrations; and the HPCI RPV high LEVEL 8 calibration.

2.4.4 ATWS (Anticipated Transient Without Scram) Recirculation Pump Trip (RPT) Actuation Instrumentation

The licensee has requested one-time calibration interval extensions of 169 and 184 days for the two ATWS RPT actuation instrument channel calibrations required by TS Table 4.3.4-1. The ATWS RPT system initiates an RPT which decreases core flow, resulting in increased steam voiding and the addition of negative reactivity. This mitigates the effects of an ATWS event in which a scram does not (but should) occur. The calibrations affected are the RPV low level LEVEL 2 and RPV pressure HIGH calibrations.

2.4.5 Reactor Core Isolation Cooling (RCIC) Actuation Instrumentation

The licensee has requested one-time calibration interval extensions of 184 days for two RCIC actuation instrument channel calibrations required by TS Table 4.3.5.1-1. The RCIC system provides core cooling during reactor shutdown by pumping makeup water into the RPV in the event of a loss of main feedwater flow. The system includes a steam turbine driven pump and can be started either manually or automatically upon receipt of a reactor vessel low water level (LEVEL 2) signal. The system will automatically trip upon receipt of a high reactor vessel level (LEVEL 8) signal. The calibrations affected are the RPV low level LEVEL 2 and high level LEVEL 8 calibrations.

2.4.6 Remote Shutdown System

The licensee has requested one-time calibration interval extensions of 168 and 184 days for two remote shutdown actuation instrument channel calibrations required by TS Table 4.3.7.4-1. The remote shutdown system provides a means for conducting a reactor shutdown outside the main control room if the control room becomes uninhabitable. It is assumed that prior to the evacuation of the control room, plant conditions are normal with no accidents or transients in progress, and the reactor operator has time to manually scram the plant. The calibrations affected are the RPV pressure and water level calibrations.

2.4.7 Post-Accident Monitoring System

The licensee has requested one-time calibration interval extensions of 74 to 191 days for several post-accident monitoring system instrument channel calibrations required by TS Table 4.3.7.5-1. The information provided by the post-accident monitoring system is designed to accommodate events up to and including a LOCA to facilitate operator action, information, and event tracking. The calibrations affected include RPV pressure, fuel zone level, and wide range level calibrations; safety/relief valve (SRV) position indicator and containment isolation valve position calibrations; and the containment high range radiation monitoring calibration.

2.4.8 Loose Parts Detection System

The licensee has requested a one-time calibration interval extension of 16 days for the channel calibration of the loose parts detection system required by TS 4.3.7.10.c. The loose parts detection system detects and annunciates unusual noises that may indicate a metallic loose part in the primary system in order to avoid or mitigate potential damage of primary system components.

2.4.9 Feedwater/Main Turbine Trip System

The licensee has requested a one-time calibration interval extension of 181 days for the automatic feedwater/main turbine trip on high RPV water level (LEVEL 8) actuation instrumentation channel calibration required by TS Table 4.3.9.1-1. The feedwater/main turbine trip system is provided to initiate action in the event of a high RPV level due to failure of the feedwater controller under maximum demand.

2.4.10 Alternative Shutdown System

The licensee has requested a one-time calibration interval extension of 168 days for the alternative shutdown reactor water level and pressure instrument channel calibrations required by TS Table 4.3.11.1-1. The alternative shutdown system ensures that a fire will not preclude achieving safe shutdown. The alternative shutdown system instrumentation is independent of areas where fire could damage systems normally used to shut down the reactor.

2.4.11 Safety Relief Valves

The licensee has requested one-time calibration interval extensions of 74 and 169 days for the SRV tail pipe pressure switch and low low set pressure setpoint calibrations. The SRVs function to prevent the reactor coolant system (RCS) from being pressurized above the safety limit of 1325 psig. A total of 11 of 15 SRVs are required to be operable to limit worst-case transient reactor pressure. TS 4.4.2.1.1 and 4.4.2.2.b, respectively, require (1) that the valve position indicator for each SRV be demonstrated operable with the pressure setpoint of each tailpipe pressure switch verified by performance of a channel calibration and (2) that the low-low set function pressure actuation instrumentation be demonstrated operable by performance of a channel calibration.

2.4.12 Drywell Floor Drain Sump Flow/Level Monitoring Systems

The licensee has requested a one-time calibration interval extension of 112 days for the primary containment sump flow and drywell floor drain sump level monitoring instrument calibrations required by TS 4.4.3.1.b. The TS specify allowable values for identified and unidentified leakage of reactor coolant. If the leakage values are exceeded or the leakage is determined to be pressure boundary leakage, the reactor must be shut down. The equipment drain sump collects only identified leakage and the drywell floor drain sump collects unidentified leakage.

2.4.13 Main Steam Isolation Valve (MSIV) Leakage Control System (LCS)

The licensee has requested a one-time calibration interval extension of 175 days for the MSIV LCS pressure control instrument (pressure and DP) calibrations required by TS 4.6.1.4.d.3. The MSIV LCS ensures that leakage beyond the MSIVs during a postulated LOCA would be treated such that doses from the maximum allowed leakage would result in only a small fraction of the 10 CFR Part 100 guidelines.

2.4.14 Justification and Discussion

The above calibrations are to be performed at least once per 18 months (with a maximum surveillance interval extension of 4.5 months per TS 4.0.2). The licensee has provided justification for the above one-time calibration interval extensions. The licensee stated that in many cases, more frequent surveillances, such as quarterly channel functional tests and twice daily or shiftly channel checks, would be expected to identify potential drift of the instruments except for the detector and cable. The licensee also performed a historical search of plant calibration data from the previous two refueling outages. In all cases, the data were found to be within design tolerances. The licensee concluded that the impact on the availability of the instrumentation as a result of the one-time surveillance interval extension would be small.

Additionally, the licensee performed a historical search of instrument 18-month surveillance tests for the previous two refueling outages. Where failures were noted, it was determined in almost all cases that the types of failures would be detected during tests that are performed more frequently, or in the case of a main steam line radiation spare monitor test cable failure, would not have impacted device performance. In the case of a failed drywell pressure transmitter, the cause was investigated and concluded to result from a procedural problem with previous response time testing which has since been corrected. In the case of a failed RPV level transmitter, the transmitter could not be calibrated because of failed components and was replaced. It was determined that this was an isolated failure with no time-dependent failure mode.

The licensee also performed a historical review of surveillance tests from the previous two refueling outages for mechanical devices such as limit switches and determined there were no significant numbers of failures. A few valve position indication failures occurred for some containment isolation valve limit switches and one clamp-on position indication, but the valves performed their safety function during the surveillances in which the failures were discovered. The lone exception was one valve stroke time failure due to failed limit switch position indication. In all cases the failures were corrected and retested satisfactorily. In the case of pressure switch valve position indicators for the Target Rock two-stage SRVs, a number of alternate means of detecting valve position are available to operators. In addition, a review of historical calibration data determined that drift was within acceptable limits.

In response to staff questions, the licensee provided additional information by letter dated December 22, 1995, on the use of historical plant data for validating nominal trip setpoints for the requested extended surveillance intervals. The licensee used historical as-found and as-left surveillance data to conservatively calculate the drift for the 18-month interval. The licensee verified that the as-left and as-found drift data was bounding for the proposed interval extensions. In all cases, the licensee concluded that the safety impact of extending the surveillance intervals was small.

The licensee also provided information related to previous approvals of 24-month surveillance intervals for other licensees and favorable comparisons of its justification with that of those licensees. The staff has reviewed the information provided by the licensee and has determined that approval of the proposed one-time calibration interval extensions would not have a significant impact on safety; therefore, the proposed one-time calibration interval extensions are acceptable.

2.5 Logic System Functional Tests (LSFTs)/Actuation Tests

LSFTs are surveillance tests that verify the operability of all relays and contacts from the sensor through the actuated device for a system's control logic. The LSFT consists of performing several plant procedures, which when combined, test the complete logic system. Simulated automatic actuation tests verify the ability of a system to perform its design automatic function by confirming the proper operation of the electrical, electronic, and mechanical components of a system. These tests are required to be performed once per 18 months (with a maximum surveillance interval extension of 4.5 months per TS 4.0.2). The licensee requested a one-time extension of the surveillance interval for several logic system functional tests and automatic actuation tests that would normally be performed during the refueling outage.

2.5.1 RPS Logic System Functional Test and Reactor Mode Switch Shutdown Position Functional Test

The RPS instrumentation and control initiates an automatic reactor shutdown (scram) if the monitored system variables exceed pre-established limits in order to prevent fuel damage and limit system pressure. The licensee has requested a one-time surveillance interval extension of 173 days for the RPS LSFT required by TS 4.3.1.2.

When the reactor mode switch is in the SHUTDOWN position, the reactor is to be shut down with all control rods inserted. The functional testing of this switch is a portion of the LSFT and verifies that the reactor mode switch shutdown scram and bypass logic associated with the RPS functions properly. The licensee has stated that this scram is not considered a protective function because it is not required to protect the fuel or nuclear system process barrier, or to minimize the release of radioactive material from any barrier. The RPS functions independently of the mode switch. The licensee has requested a one-time surveillance interval extension of 146 days for the channel functional test required by TS Table 4.3.1.1-1.

2.5.2 Isolation Actuation Logic System Functional Tests

The containment and reactor vessel isolation control system (CRVICS) includes the instrument channels, trip logics, and actuation circuits that automatically initiate valve closure providing isolation of the containment and/or reactor vessel, and initiation of systems provided to limit the release of radioactive materials. When abnormal conditions are sensed, instrument channel relay contacts open and deenergize the normally energized trip logic and thereby initiate isolation. Once initiated, the CRVICS trip logics seal in and may only be reset by the operator when initiating conditions have returned to normal. The licensee has requested a one-time surveillance interval extension of 183 days for the isolation actuation instrument LSFT required by TS 4.3.2.2.

The primary containment isolation actuation, residual heat removal shutdown cooling mode isolation actuation, RWCU, and HPCI, manual initiation channel functional tests introduce signals to the isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. No safety analyses take credit for these functions. They are retained for overall redundancy and diversity of the isolation function as required by the licensing basis. The licensee has requested a one-time surveillance interval extension of 191, 170, 131, and 34 days, respectively, for these manual initiation channel functional tests required by TS Table 4.3.2.1-1.

The RWCU - SLCS initiation channel functional test produces SLCS initiation signals from the two SLCS pump start signals to test the isolation of the RWCU system. The RWCU system isolation is required whenever the SLCS has been initiated to prevent dilution and removal of the boron solution by the RWCU system. The licensee has requested a one-time surveillance interval extension of 138 days from the RWCU - SLCS initiation channel functional test required by TS Table 4.3.2.1-1.

2.5.3 ECCS Logic System Functional Tests

The objective of the ECCS, in conjunction with the containment, is to limit the release of radioactive materials should a LOCA occur so that resulting radiation levels are kept within the guideline values given in 10 CFR Part 100. The purpose of the ECCS instrumentation is to initiate appropriate responses from the systems to ensure that the fuel is adequately cooled in the event of a design-basis accident or transient. If the monitored parameters exceed the established setpoints, logic signals are generated to initiate ECCS and ECCS support systems. The licensee has requested a one-time surveillance interval extension of 227 days for the ECCS LSFT required by TS 4.3.3.2.

The CS, LPCI, and HPCI manual initiation channel functional tests initiate signals to the ECCS start logic that are redundant to the automatic protective instrumentation and provide manual ECCS start capability. The ADS manual inhibit channel functional test verifies the function of the ADS manual inhibit switches in the control room. Their function is not required for ADS operability (provided ADS is not inhibited when required to be operable). The licensee has requested one-time surveillance interval extensions of 191, 185, 34, and 72 days, respectively, for the channel functional tests required by TS Table 4.3.3.1-1.

2.5.4 ATWS Recirculation Pump Trip Logic System Functional Test

The ATWS-RPT system initiates an RPT which decreases core flow, resulting in increased steam voiding and the addition of negative reactivity. This mitigates the effects of an ATWS event in which a scram does not (but should) occur. Tripping the RPT adds negative reactivity from the increase in steam voiding in the core area from the core flow decrease. The licensee has requested a one-time surveillance interval extension of 184 days for the ATWS LSFT required by TS 4.3.4.2.

2.5.5 RCIC Logic System Functional Test

The RCIC system provides core cooling during reactor shutdown by pumping makeup water into the RPV in the event of a loss of main feedwater flow. The system includes a steam turbine driven pump and can be started either manually or automatically upon receipt of a reactor vessel low water level (LEVEL 2) signal. The system will automatically trip upon receipt of a high reactor vessel level (LEVEL 8) signal. The licensee has requested a one-time surveillance interval extension of 184 days for the RCIC LSFT required by TS 4.3.5.2.

2.5.6 Control Rod Block Instrumentation

The scram discharge high water level trip bypass is controlled by the manual operation of two keylocked switches, a bypass switch, and the reactor mode switch. The reactor mode switch must be in either the SHUTDOWN or the REFUEL position in order to bypass this trip. Four bypass channels are each connected to the RPS logic from the reactor mode switch. This bypass allows the operator to reset the RPS scram relays so that the system is restored to operation while the operator drains the SDV. In addition, actuating the bypass initiates a control rod block. Resetting the trip actuators opens the SDV vent and drain valves. An annunciator in the main control room indicates the bypass condition. The licensee has requested a one-time surveillance interval extension of 103 and 146 days, respectively, for the channel functional tests of the SDV-scram trip bypass and reactor mode switch required by TS Table 4.3.6-1.

2.5.7 Feedwater/Main Turbine Trip Logic System Functional Test

The feedwater/main turbine trip system is provided to initiate action in the event of a high RPV level due to failure of the feedwater controller under maximum demand. The licensee has requested a one-time surveillance interval extension of 181 days for the LSFT required by TS 4.3.9.2.

2.5.8 SRV Logic System Functional Test

The SRVs function to prevent the RCS from being pressurized above the safety limit of 1325 psig. A total of 11 of 15 SRVs are required to be operable to limit worst-case transient reactor pressure. The purpose of this test is to perform an LSFT and simulated automatic operation of the entire low-low set function pressure actuation instrumentation. The low-low set logic is designed with redundancy and single-failure criteria; that is, no single electrical failure will prevent any low-low set valve from opening or cause inadvertent seal-in of low-low set logic. The licensee has requested a one-time surveillance interval extension of 169 days from the requirement of TS 4.4.2.2.b. The related calibration interval extension for this surveillance has been previously discussed above.

2.5.9 ECCS Functional Test

The CS system together with the LPCI mode of the residual heat removal system, are provided to assure that the core is adequately cooled following a LOCA and provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor circulation line break, and for smaller breaks following the depressurization by the ADS. The CS system is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding the core in case of accidental draining. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires a reactor shutdown. TS 4.5.1.c.1 requires that the ECCS be demonstrated operable for the CS, LPCI, and HPCI systems by performing a system functional test which includes simulated automatic operation of the system through its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from the test. The licensee has requested a one-time surveillance interval extension of 191 days for the ECCS functional test required by TS 4.5.1.c.1.

TS 4.5.1.d.2.a requires that the ECCSs be demonstrated operable for the ADS by performing a system functional test that includes simulated automatic operation of the system throughout its emergency operating sequence but excluding actual valve actuation. The licensee has requested a one-time surveillance interval extension of 72 days for the ADS functional test required by TS 4.5.1.d.2.a.

TS 4.5.2.1 requires that the ECCS be demonstrated operable for modes 4 and 5 per TS 4.5.1 with the exception that for the LPCI system, the cross-tie valve may be closed to isolate a subsystem if the subsystem is made capable of injection into the reactor vessel. The licensee has indicated that the extension for this surveillance is covered by the request for TS 4.5.1.

2.5.10 Primary Containment Isolation Valves

The operability of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment as specified in General Design Criteria (GDC) 54 through 57 of Appendix A of 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the accident analyses for a LOCA. TS 4.6.3.2 requires that each primary containment automatic isolation valve be demonstrated operable by verifying that on a containment isolation signal each valve actuates to its isolation position. The licensee has requested a one-time surveillance interval extension of 189 days from the primary containment isolation valve operability demonstration required by TS 4.6.3.2.

2.5.11 Secondary Containment Isolation Damper Actuation

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The reactor building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment is specified. TS 4.6.5.2.b requires each secondary containment ventilation system automatic isolation damper in TS Table 3.6.5.2-1 be demonstrated operable by verifying that on a containment isolation test signal, each damper actuates to its isolation position. The licensee has requested a one-time surveillance interval extension of 27 days from the secondary containment isolation damper actuation requirement of TS 4.6.5.2.b.

2.5.12 Safety-Related Service Water Systems

The operability of the safety-related service water systems ensures availability of the ultimate heat sink and that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. TS 4.7.1.2.b requires that the emergency equipment cooling water (EECW) system be demonstrated operable by verifying that each automatic valve servicing nonsafety-related equipment actuates to its isolation position and the associated EECW pump starts on an automatic actuation test signal. TS 4.7.1.3.b requires that the emergency equipment service water (EESW) system be demonstrated operable by verifying that the EESW pump starts upon receipt of an automatic actuation test signal. The licensee has requested a one-time surveillance interval extension of 91 days from the EECW and EESW automatic actuation tests required by TS 4.7.1.2.b and 4.7.1.3.b, respectively. TS 4.7.1.4.b requires that each of the emergency diesel generator service water (EDGSW) subsystems be demonstrated operable by verifying that each EDGSW pump starts upon receipt of a start signal for the associated EDG. The licensee has requested a one-time surveillance interval extension of 215 days from the EDGSW pump automatic actuation test required by TS 4.7.1.4.b.

2.5.13 Alternative Shutdown Auxiliary Systems

The alternative shutdown system ensures that a fire will not preclude achieving safe shutdown. The alternative shutdown system instrumentation is independent of areas where fire could damage systems normally used to shut down the reactor. Alternative shutdown auxiliary systems are those utilized for Appendix R alternative shutdown but not included in other sections of the TS. These systems must be operable or acceptable alternative means must be established to achieve the same objective. TS 4.7.11.4 requires that each alternative shutdown system control circuit be demonstrated operable by verifying its capability to perform its intended function(s). The licensee has requested a one-time surveillance interval extension of 82 days from the alternative shutdown control circuit functional test required by TS 4.7.11.4.

2.5.14 Justification and Discussion

As stated previously, the above logic system functional tests and actuation tests are to be performed during each cold shutdown or refueling outage at least once per 18 months (with a maximum surveillance interval extension of 4.5 months per TS 4.0.2). The licensee has provided justification for the requested surveillance extensions. The licensee performed a review of the surveillance test history for the RPS LSFT and found no evidence of excessive random equipment or component failure rates. Based on this review and the redundant equipment in each of the subject systems, the licensee concluded that the safety impact of the extension on system availability was small.

The licensee also referenced an industry study (BWR Owners Group Report EAS 25-0489, "Evaluation of Logic System Functional Test Methods," dated July 1989) which concluded that for most logic configurations, the total circuit availability improved by changing from a 6-month to 18-month surveillance interval and that with the exception of ADS, in no case did unavailability increase any appreciable amount. The licensee also referenced the August 2, 1993, NRC Safety Evaluation relating to the Peach Bottom Atomic Power Plant Units 2 and 3, extension of surveillance intervals from 18 to 24 months which contained the following conclusion:

> "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic systems, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis....Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability."

The licensee stated that the above statement was applicable to Fermi 2 and concluded that the safety impact of increasing the surveillance intervals for logic system functional tests was small.

The licensee further stated that the reactor mode switch as well as most manual actuation and inhibit switches are considered to be system logic components since their failure modes and failure history resemble circuit components rather than mechanical components. The licensee stated also that the RPS functions independently of the mode switch. The licensee concluded based on the above discussion, that the impact of the changes in surveillance intervals on the availability of the reactor mode switch and the other manual actuation and inhibit switches would be small.

The licensee also concluded that for the system actuation tests, pump and valve flow tests and damper cycling are performed on a more frequent basis in accordance with the Inservice Testing Program requirements and failures would be detected by these more frequent tests. The portions of the systems not tested on a more frequent basis are equivalent to the logic system and testing of those portions of the systems would be equivalent to an LSFT for those systems. The licensee concluded that extending the surveillance intervals for those system actuation tests would therefore have a small safety impact. The licensee also concluded that the 18-month surveillance test of the alternative shutdown operability when assessed with the more frequent surveillance testing of alternative shutdown system components, is effectively a logic system test for the controls on the alternative shutdown panel. The licensee, therefore, concluded that the safety impact of extending the surveillance interval for this test would also be small.

The licensee also performed a historical search of the 18-month surveillance tests for the above LSFTs and actuation tests and determined that some failures occurred, but in general, they did not degrade the component's ability to function or would have been detected by more frequent surveillances. One clamp-on valve position indication and two limit switch valve position indication failures occurred during the last two refueling outages during performance of the primary containment isolation actuation functional test. Also, one valve exceeded its stroke time test. These are the same failures described above in Section 2.4.15. Additionally, one relay did not de-energize within the allowable time limit. This relay is cycled quarterly, but not through all relay contact positions, by other functional tests. The failed relay was replaced and tested satisfactorily and all relays of this type are being replaced.

The licensee also identified during the ECCS LSFT 18-month surveillance history review, a failure of a drywell pressure transmitter to calibrate due to water ingress and the failure of an RPV level transmitter to calibrate due to failed components. These failures are discussed in Section 2.4.15. The licensee also identified the failure of an EDG output breaker to close caused by the EDG not reaching rated speed and voltage within sufficient time. The EDG governor was repaired to correct this failure. The licensee noted that subsequent failures would be detected because the EDG ability to reach rated speed and voltage is tested more frequently (semiannually). Based on the above discussion, the licensee concluded that the safety impact on LSFTs and system actuation tests would be small as a result of the proposed changes.

The staff has reviewed the licensee's justification and determined that the safety impact of extending the above surveillance intervals for the above LSFTs and actuation tests on a one-time basis would be small, and that in general the licensee meets the guidance in GL 91-04; therefore, the proposed changes are acceptable.

2.6 Response Time Tests

The measurement of response times for systems at the specified frequencies provides assurance that the protective functions associated with each channel will be completed within the time limit assumed in the safety analyses. The licensee's TS typically require only one channel to be tested for the response time testing during any given 18-month surveillance such that all channels would be tested during an (N x 18)-month period, where N is the number of channels being tested. The licensee has requested one-time surveillance interval extensions of 101, 101, and 191 days, respectively, from the RPS, isolation actuation instrumentation system, and ECCS response time testing required by TS 4.3.1.3, 4.3.2.3, and 4.3.3.3.

Regulatory Guide 1.118 (Revision 2) states:

> "Response time testing of all safety-related equipment, per se, is not required if, in lieu of response time testing, the response time testing of the safety equipment is verified by functional testing, calibration checks or other tests, or both. This is acceptable if it can be demonstrated that changes in response time beyond acceptable limits are accompanied by changes in performance characteristics which are detectable during routine tests."

With this in mind, on December 28, 1994, the NRC staff completed a review of the BWR Owners Group Licensing Topical Report, NEDO-32291, "Systems Analyses for Elimination of Selected Response Time Testing Requirements," dated January 1994, prepared by General Electric Company. Based on its review, the staff concluded that significant degradation of instrument response times, i.e., delays greater than about 5 seconds, can be detected during the performance of other surveillance tests, principally calibration, if properly performed. The NRC staff concluded that for selected instrumentation identified in the topical report, response time testing could be eliminated from the TS. The licensee has reviewed the generic analysis contained in the topical report and has verified its applicability to Fermi 2. On this basis the licensee concluded that the safety impact of extending the surveillance intervals for the subject selected response time testing was small.

Additionally, the licensee has identified several other instrument functions not included in the topical report analysis for elimination of response time testing that it believes are justified for the requested one-time surveillance interval extension. These include the following RPS functions: average power range monitor (APRM) flow-biased simulated thermal power high; APRM fixed neutron flux high; reactor vessel steam dome pressure high and low water level LEVEL 3 (trip units and logic only); MSIV closure; turbine stop valve closure; and turbine control valve fast closure. The licensee stated that the surveillance interval extensions would have no substantial measurable safety impact because there are redundant sensors that can initiate the scram operation. There are several redundant and diverse instrument channels which can detect and generate a scram signal. Redundancy exists for every individual instrument channel within each trip function and only one of the channels per trip system is required to be tested during each surveillance interval. The overall instrumentation failure probability is a very small fraction of the total control rod insertion (scram) failure probability and the failure of instrumentation in the sluggish responding mode is a small fraction of the overall instrumentation failure probability. On this basis the licensee concluded that the safety impact of the surveillance interval extensions would be small.

The licensee provided similar justifications of redundancy, diversity, and failure probability for the isolation actuation instrumentation surveillance interval extensions. The specific functions not included in the topical report analysis for elimination of response time testing are primary containment isolation reactor vessel low water level LEVEL 1 and main steam line flow HIGH (trip units and logic only). The licensee concluded that the surveillance interval extensions would also have no substantial measurable safety impact for these functions.

For the ECCS response time testing, the licensee acknowledged that the topical report only evaluated elimination of the sensor and trip unit response time testing and that system response time testing would still be required. The licensee stated that more frequent pump and valve testing required by the inservice testing program coupled with EDG testing would indicate any significant slow responses in ECCS components. The licensee provided similar arguments for redundancy and failure probabilities as mentioned previously and concluded that the surveillance interval extensions would have no substantial measurable safety impact. For all of these response time tests, the licensee performed a historical review of 18-month surveillances. No failures were identified.

Based on the information provided by the licensee, the staff has determined that the safety impact of the proposed one-time surveillance interval extensions would be small. Therefore, the proposed one-time surveillance interval extensions for time response testing required by TS 4.3.1.3, 4.3.2.3, and 4.3.3.3 are acceptable.

2.7 <u>Inservice Inspection (ISI)/Inservice Testing (IST)</u>

The licensee has reviewed surveillance requirements for TS 4.4.8 and 4.0.5 related to implementation of the ASME Code ISI/IST programs. TS 4.4.8 states that no requirements other than 4.0.5 apply for ensuring reactor coolant system ASME Code (Code) Class 1, 2, and 3 component structural integrity. TS 4.0.5 delineates the ISI surveillance requirements for Code Class 1, 2, and 3 components and IST surveillance requirements for Code Class 1, 2, and 3 pumps and valves. TS 4.0.5 basically refers to the applicable portions of Section XI of the Code and applicable addenda for specific requirements and schedules.

The ISI NDE [nondestructive examination] Program Plan identifies all components requiring inspection over the 10-year inspection interval and the refueling outage in which each component is to be inspected. The licensee has completed two inspection periods at Fermi 2 and is in compliance with Code requirements for percent complete. Extending the current operating cycle from March 1996 to October 1996 will not affect ISI program compliance. As the inspections are spaced over the entire 10-year interval, the licensee states that the 6-month cycle extension is not expected to have a significant effect on the refueling outage inspection results. The licensee has stated that the RPV system leakage test required by the Code is required to be performed only during each refueling outage prior to plant startup or following the opening and reclosing of any Class 1 component (no specific periodicity is assigned).

The licensee stated that no scheduling extension is required in order to extend the operating cycle.

The licensee reviewed previous leakage test findings and determined that the only noted through-wall leakage was repaired during RF03 and has been repaired and subsequently tested. Any degradation of Class 1 piping would be detected by plant/drywell leakage monitoring requirements and limited before significant degradation could take place.

The licensee also reviewed the results of RF04 inspections for RPV internals, vessel welds, piping, and component supports. The licensee evaluated the effect of delaying followup RF05 inspections committed to as a result of minor indications found during the RF04 inspections and determined that the delay would be acceptable and any impact would be small. The licensee also evaluated the impact of delaying scheduled visual inspections of internal components and pressure-retaining bolts and determined that no significant degradation/change would be expected as a result of extending the operating cycle. The licensee also verified that there were no failures in the sample population for IST check valves inspected during RF03 and RF04. The licensee does not expect any significant impact as a result of delaying these inspections.

The licensee has concluded that no specific TS changes are required to TS 4.0.5 and no additional reliefs are required to the ISI or IST programs to support the extension of the operating cycle. Also, the licensee has concluded that the impact of extending the operating cycle on the structures, components, pumps, and valves would be small. The staff has reviewed the licensee's determination and agrees that no specific reliefs or changes to the TS are required and that the impact of extending the operating cycle on ISI/IST structures, components, pumps, and valves would be small; therefore, the proposed extension will not impact the inspection schedule for the ISI/IST programs.

2.8 <u>Snubbers</u>

Snubbers are required to be operable to ensure that the structural integrity of the RCS and all other safety-related systems is maintained during and following a seismic event or other event initiating dynamic loads. Snubbers are categorized as inaccessible or accessible during reactor operation. Snubbers excluded from the inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

TS 4.7.5.b requires each snubber to be demonstrated operable by performance of visual inspections in accordance with the schedule and criteria determined by TS Table 4.7.5-1. The licensee has stated that the last TS-required inspection was performed during RF04, and based on the results, the next inspection is not required until RF06. The licensee has indicated that a self-imposed inspection of selected N30 snubbers will be performed during performed during RF05; however, this requires no TS surveillance inspection as it is not required by TS.

TS 4.7.5.e requires each snubber be demonstrated operable by performing functional tests. A representative sample shall be tested at least once per 18 months (with a maximum surveillance inspection interval extension of 4.5 months per TS 4.0.2) during shutdown using one of the provided sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan to be used or the sample plan from the previous test period shall be used. The licensee has elected to again use the 10 percent sample plan and has requested a one-time surveillance interval extension of 105 days from the requirement of TS 4.7.5.e.

The licensee has stated that the primary snubber failure modes experienced at Fermi have been due to temperature and vibration induced degradation. Snubber degradation is not simply time dependent, but a function of operating time and time at elevated temperature for those snubbers in high temperature areas (e.g., drywell, steam tunnel). The licensee projects a total of 19.7 months operation for the current extended operating cycle. The licensee experienced only three snubber failures during the RF04 inspection. The licensee believes that the slight increase in operational time at elevated temperature will not have a significant impact on snubber failure rates and that the safety impact will be small.

The staff has reviewed the licensee's justification and agrees that the projected number of operating months falls within the maximum range expected for an operating cycle (22.5 months) currently allowed by the TS. As the failure rates are predominantly temperature and vibration induced, it is not expected that the time shutdown during RF04 will contribute significantly to the failure rates for any snubbers required to be tested. The staff finds that the expected snubber performance would be comparable to that which would be seen during the maximum currently allowed TS surveillance interval; therefore, the proposed surveillance interval is acceptable.

2.9 SRV Lift Set Point Test

The SRVs function to prevent the RCS from being pressurized above the safety limit of 1325 psig. A total of 11 of 15 SRVs are required to be operable to limit worst-case transient reactor pressure. TS 4.4.2.1.2 requires at least one-half of the SRVs be set-pressure tested at least once per 18 months such that all 15 SRVs be set-pressure tested at least once per 40 months. The licensee has requested a one-time surveillance interval extension of 115 days for the portion of the surveillance requiring at least one-half of the SRVs be tested at least once per 18 months. The licensee stated that all 15 SRVs were set-pressure tested during RF04.

The SRVs are Target Rock Corp. two-stage pilot-operated dual function safety/relief valves. In the safety mode, the valve opens solely by mechanical means when pressure at the inlet of the valve reaches the set pressure of the valve. This provides overpressure protection of the reactor coolant pressure boundary. In the relief mode, the valve is remotely opened by a solenoid valve manifold/pneumatic operator assembly to provide controlled depressurization from the control room. This surveillance requirement pertains only to the safety mode.

The licensee has indicated that the failure mode of previous SRV failures (which have resulted in all 15 SRVs being tested each refueling outage) has primarily been due to upward setpoint drift which is consistent with industry results. Analyses have shown that in no case has the required vessel overpressure protection been unavailable during any operating cycle as a result of these failures. A review of RF03 and RF04 data shows no tendency toward downward setpoint drift which could result in an inadvertent opening of an SRV.

The licensee provided a discussion on the impact of the surveillance interval extension as a result of the upward drift problem. The licensee cited Section 3.9.3.4 of NUREG-0991, "Safety Evaluation Report [SER] Related to the Operation of Limerick Generating Station [LGS], for Units 1 and 2," dated October 1984, in support of its position that the one-time surveillance interval extension will not impact the ability of the SRVs to perform their safety function. The phenomenon of upward drift is recognized to occur as a result of two primary causes: labyrinth seal-induced friction and pilot disc/seat bonding. The NRC staff concluded that implementation of recommendations for improved SRV maintenance and refurbishment contained in GE Service Information Letter (SIL) No. 196, "Target Rock Two-Stage Setpoint Drift," including Supplement 14, will adequately address setpoint drift due to labyrinth seal-induced friction. The licensee has implemented these recommendations for Fermi 2.

The NRC staff also concluded in the LGS SER that, based on available two-stage SRV data that setpoint drift occurring from pilot disc/seat bonding occurs less frequently than that caused by labyrinth seal-induced friction. Additionally, historical data indicates that limited oxide formation and consequent pilot disc/seat bonding occurs rapidly on freshly refurbished metal seating surfaces and then further bond progression is slow as the initial oxide layers passivate and protect the seating surface. Also, SRV set pressures that drift due to pilot disc/seat bonding return to near their nominal setpoint after the first actuation. The licensee states that the slow progression would result in no significant impact as a result of the proposed extension.

The licensee also indicated that similarly to LGS, the proposed one-time surveillance interval extension will still result in more frequent testing than the ASME Code Section XI requirements invoked from ANSI/ASME OM-1-1981, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices," of 20 percent of SRVs tested within 24 months and all SRVs tested within 60 months. Additionally, the relieving capacity at Fermi 2 is similar to that at LGS, and is considerably more than required by the applicable edition of the ASME Code. Only 11 of 15 SRVs (11 of 14 at LGS) are required to be operable to meet the ASME Code requirements. The NRC determined in the LGS SER that LGS can be operated with no adverse effect on the health and safety of the public until the NRC reaches a final generic solution for setpoint drift.

Based on the above the staff finds that the safety impact of the one-time surveillance interval extension for the SVR lift setpoint test required by TS 4.4.2.1.2 is small; therefore, the proposed surveillance interval extension is acceptable.

2.10 Containment Isolation Valves Leakage Testing

Appendix J to 10 CFR Part 50 of the Commission's regulations specifies primary reactor containment leakage testing requirements for water-cooled power reactors. The testing requirements consist of Types A, B,and C tests. Type A tests are intended to measure the primary reactor containment overall integrated leakage rate after the containment has been completed and at periodic intervals thereafter. Type B tests are intended to detect local leaks and to measure leakage for containment penetrations whose design incorporates resilient seals, gaskets, or sealing compounds, and certain piping and electrical penetrations. Additionally, air-lock door seals and doors with resilient seals or gaskets are Type B tested. Type C tests measure containment isolation valve leakage rates. These requirements are also incorporated into the Fermi 2 TS.

TS 4.6.1.2.b requires that the primary containment leakage rates be demonstrated less than the limits specified in TS 3.6.1.2, in part by performance of Types B and C containment integrated leak rate tests (ILRT). Types B and C tests shall be performed with gas at P_a, 56.5 psig (unless a hydrostatic test is required) at intervals no greater that 24 months, except for tests involving air locks, MSIVs, penetrations using continuous monitoring systems, valves pressurized with fluid from a seal system, ECCS and RCIC containment isolation valves in hydrostatically tested lines, and purge supply and exhaust isolation valves with resilient material seals. Additionally, TS 4.6.1.2.d requires that MSIVs shall be leak tested at least once per 18 months. TS 4.6.1.2.g requires that ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months. The above two surveillances have a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.5. The licensee has requested a one-time surveillance interval extension of 157, 170, and 183 days, respectively, from the requirements for containment leakage testing in TS 4.6.1.2.b, d and g.

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at peak accident pressure of P_a, 56.5 psig. The analyses in the Updated Final Safety Analysis Report demonstrate that the maximum expected pressure is less than 56.5 psig. The licensee had previously, by letter dated September 1, 1995, requested a one-time exemption from the schedule requirements of 10 CFR Part 50, Appendix J, paragraphs III.D.2.a and III.D.3. This exemption was granted on December 18, 1995.

In its exemption request, the licensee stated that the NRC was proposing to amend its regulations for containment ILRTs in 10 CFR Part 50, Appendix J (60 FR 9634, dated February 21, 1995). That proposed rule was made final on September 12, 1995. The rule establishes performance-based Types B and C test intervals, with factors for establishing extended test intervals of up to 10 years for Type B tests and 5 years for Type C tests based on regulatory guide and industry guidance. The licensee proposed a surveillance interval of 24 months (with an allowed extension of 25 percent) which is within the most limiting surveillance interval in the rule (30 months with an allowed extension of 25 percent). The NRC staff in approving the licensee's request stated that extending the Appendix J intervals by a small amount to reach the next refueling outage will not significantly impact the integrity of the containment boundary, and will not significantly impact the consequences of any accident or transient that may occur within that time.

Based on the above, the staff has determined that the proposed one-time surveillance interval extension will not significantly impact containment integrity or leakage rates and that the impact on safety is small. Therefore, the proposed one-time surveillance interval extension is acceptable.

2.11 <u>RCS Pressure Isolation Valve (PIV) Leak Test</u>

TS 4.4.3.2.2.a requires that each RCS PIV in TS Table 3.4.3.2-1 be demonstrated operable by leak testing pursuant to TS 4.0.5 and verifying that the leakage of each valve is within specified limits at least once per 18 months (with a maximum surveillance interval extension of 4.5 months per TS 4.0.5). The surveillance requirement to leak test RCS PIVs, provides added assurance of valve integrity thereby reducing the probability of gross valve failure and consequential intersystem LOCA. Leakage past the RCS PIVs (but not valve body leakage which is considered pressure boundary leakage) is identified leakage and is considered as a portion of the allowable limit. The licensee has requested a one-time surveillance interval extension of 181 days from the RCS PIV leak test requirements of TS 4.4.3.2.2.a.

The licensee stated that all RCS PIVs had passed their RF03 and RF04 surveillances and are well below the leakage limit; nor were there any increasing leakage trends. The highest recorded leakage of any PIV was 0.14 gallon per minute (gpm). The observed leakage was determined not to be time dependent. Therefore the licensee concluded that the safety impact of extending the surveillance interval was small.

Based on our review of the information provided above, the staff has determined that the one-time surveillance interval extension will not have a significant impact on the RCS PIV leakage rates and will not significantly decrease valve integrity or increase the probability of gross valve failure and consequential intersystem LOCA. Therefore the safety impact is small and the proposed one-time surveillance interval extension for the RCS PIV leakage tests of TS 4.4.3.2.2.a is acceptable.

2.12 Pressure Suppression Chamber

TS 4.6.2.1.e requires that the suppression chamber be demonstrated operable at least once per 18 months. The purpose of this inspection is to determine that there is no evidence of corrosion of painted surfaces which could result in the unevaluated degradation of the coating system during the next operating cycle. During plant operation all surfaces required to be inspected by these requirements are normally in an inverted environment. The inverted environment will help reduce the corrosion rate in all areas other than the underwater area of the torus. The original surveillance interval between inspections is based on accessibility to the containment interior rather than a specific time-based requirement. The licensee has requested a one-time surveillance interval extension of 22 days from the visual inspection requirement of TS 4.6.2.1.e.

TS 4.6.2.1.h requires that the suppression chamber be demonstrated operable at least once per 18 months by conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure (DP) of 1 psid and verifying that the DP does not decrease by more than 0.20-inch water gage per minute for a period of 10 minutes. The drywell-to-suppression chamber leakage test is performed to ensure that the pressure suppression function of the primary containment is maintained. Excessive leakage from the drywell directly to the suppression chamber could result in a failure of the primary containment during a design-basis accident. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the Commission must review and approve subsequent schedules, and if any two consecutive bypass leak tests fail, tests shall be performed at least every 9 months until two consecutive tests pass, after which time the normal interval may be resumed. The licensee has requested a one-time surveillance interval extension of 3 days for the bypass leakage test required by TS 4.6.2.1.h. The above surveillance tests have a maximum surveillance interval extension of 4.5 months per TS 4.0.2.

The licensee conducted detailed inspections of the torus immersion area, vapor phase area, and exterior surface during RF04 to identify and evaluate any current or incipient coating failure and the effects of corrosion resulting from any coating deficiencies. No significant evidence of current or incipient general coating failure was identified in any of the 16 bays. Any small nicks, dings, scrapes, and subsequent localized corrosion were repaired. The vapor phase coating inspection revealed isolated minor mechanical damage on the pressure boundary and internal structures with no measurable pitting. This was repaired. The torus exterior surface inspection did not note any items detrimental to the structural integrity of the pressure boundary. In all cases, no significant differences were noted from the results of inspections during RF01, RF02, or RF03. Based on the results of the licensee's previous inspections and the short duration of the extension interval, the staff has determined that the one-time surveillance interval extension for TS 4.6.2.1.e will not impact safety and is therefore acceptable.

The only active component in the drywell-to-suppression chamber barrier is the vacuum breaker which is normally closed and verified closed once every 7 days per TS 4.6.4.1.a. The licensee performed a historical search of the surveillance data from RFO3 and RFO4. One failure of a vacuum breaker was identified during RFO4 resulting in leakage slightly in excess of the allowable limit. This vacuum breaker was repaired and retested successfully. No other failures were noted. Otherwise, the historical leakage has been low. Based on the licensee's historical surveillance data and the short duration of the extension, the staff has determined that the safety impact is small, and the proposed surveillance interval extension of the drywell-to-suppression chamber bypass leakage test required by TS 4.6.2.1.h is acceptable.

2.13 Vacuum Relief System

TS 4.6.4.1.b.2 items a, b, and c, require that each suppression chamber-drywell vacuum breaker be demonstrated operable at least once per 18 months by verifying the opening setpoint from the closed position to be less than or equal to 0.5 psid, verifying both position indicators operable by performing a channel calibration, and verifying the opening gap for switch actuation to be less than or equal to 0.03 inch. TS 4.6.4.2.b.2 items a, b, and c require each reactor building-suppression chamber vacuum breaker be demonstrated operable at least once per 18 months by demonstrating that the force required to open each vacuum breaker does not exceed the equivalent of 0.5 psid, by visual inspection and by verifying the position indicator operable by performing a channel calibration. These surveillances have a maximum surveillance interval extension of 4.5 months per TS 4.0.2. The vacuum relief breakers are provided to equalize the pressure between the suppression chamber and the drywell and between the reactor building and the suppression chamber. This maintains structural integrity of the primary containment under conditions of large differential pressures. The licensee has requested one-time surveillance interval extensions of 63 and 60 days, respectively, from the requirements of TS 4.6.4.1.b.2 and 4.6.4.2.b.2.

Vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. The inherent redundancy of the system allows operation to continue for up to 72 hours if a vacuum breaker were inoperable in the closed position. There are two reactor building-suppression chamber vacuum breakers but only one is necessary to provide vacuum relief and there are 12 suppression chamber-drywell vacuum breakers but only 9 are needed to provide vacuum relief.

The reactor building-suppression chamber vacuum breakers are stroked monthly during operation to demonstrate their performance. The licensee has reviewed the RFO3 and RFO4 failure history for these surveillances and stated that there have been no failures of the reactor building-suppression chamber vacuum breakers and no experience with the suppression chamber-drywell vacuum breakers that would indicate an opening force problem. There have been some problems with test actuators; however, test actuator problems do not affect valve performance since the actuator is only used during valve testing and has no impact on valve operation once testing is complete.

Closed position indication is checked every 7 days to ensure vacuum breakers are closed. The reactor building-suppression chamber vacuum breaker open limit switches are checked during monthly cycling and there have been no failures during the last two operating cycles. There have been some problems with the suppression chamber-drywell vacuum breaker limit switches following maintenance or cycling; however, these have not affected the valves' safety function in either the opening or closing direction. The licensee has stated that the extension of the surveillance interval for the visual inspection of the reactor building-suppression chamber vacuum breakers is unlikely to be of concern because of the previous surveillance history good performance. On this basis the licensee has indicated that the safety impact of extending the vacuum relief system surveillances would be small.

The staff has reviewed the information provided by the licensee and has determined that the impact of the one-time surveillance interval extensions for TS 4.6.4.1.b.2 and 4.6.4.2.b.2 would be small; therefore, the proposed one-time surveillance interval extensions are acceptable.

2.14 Control Room Emergency Filtration System

The control room emergency filtration system (CREFS) provides a suitable environment for continuous personnel occupancy and ensures the operability of the control room equipment and instruments during accident conditions. The system is normally in standby condition to minimize fouling and plugging of the HEPA [high-efficiency particulate air] filters and charcoal absorbers. There are redundant filter trains and fans which ensure system availability in the event of a failure of one of the active components. The design considerations are that the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment cooled by this system and the control room will remain habitable for operators during and following all design-basis accident conditions.

TS 4.7.2.1.c items 1, 2, and 3 require demonstrating operability of the CREFS at least once per 18 months by satisfying the testing guidance and acceptance criteria of Regulatory Guide 1.52 "Design, Testing, and Maintenance Criteria for Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants" for in-place penetration testing and carbon sample analysis, and verifying a system flow rate of 3000 cfm plus or minus 10 percent, when tested in accordance with ANSI [American National Standards Institute] N510-1980. TS 4.7.2.1.e items 1, 2, and 4 require demonstrating operability of the CREFS at least once per 18 months by verifying the pressure drop across the recirculation and makeup trains combined HEPA filters and charcoal absorbers meets acceptance criteria; verifying automatic switching to the emergency mode upon receipt of the listed actuation signals with valve closure and positive control room pressure within acceptance criteria; and verifying emergency makeup inlet air heaters dissipate the proper amount of heat in accordance with ANSI N510-1980. These surveillances have a maximum surveillance interval extension of 4.5 months per TS 4.0.2. The licensee has requested a one-time surveillance interval extension of 146 days from the requirements of TS 4.7.2.1.c and TS 4.7.2.1.e items 1 and 4. The licensee has requested a one-time surveillance interval extension of 27 days from the requirements of TS 4.7.2.1.e.2.

The licensee has indicated that other more frequent tests demonstrate operability of the CREFS such as TS 4.7.2.1.b which requires operability of the main control room fans and verification of flow through the HEPA filters and charcoal absorbers for 10 hours once per 31 days. Additionally, inservice testing is performed during the operating cycle for CREFS components. The licensee states that these tests would identify significant failures affecting CREFS operability, including failures to automatically initiate. The logic used to initiate CREFS is tested under TS 4.7.2.1.e.2 and the licensee states that reliability studies by the industry discussed in Section 2.5 above support extension of this surveillance. In addition, the licensee states that this extension is similar to that approved for Peach Bottom Units 2 and 3 in the staff's August 2, 1993, safety evaluation. The licensee concluded that the safety impact of extending the surveillances on a one-time basis would be small. The licensee has also performed a review of the CREFS surveillance history for RFO3 and RFO4 and there is no evidence of any failures that would invalidate this conclusion.

Additionally, TS 4.7.2.1.h requires that the CREFS be demonstrated operable at least once per 36 months by verifying that sections of the CREFS duct listed in TS Table 4.7.2.1-1 exhibit inleakage within the acceptance criteria when tested in accordance with ASME N510-1989. This verifies the integrity of the duct sealing compound and overall duct integrity. This surveillance was last performed during RFO4 and is designed to be performed once every other RFO. The licensee requests a one-time surveillance interval extension of 48 days in order to avoid having to perform this surveillance during RFO5.

The trend analysis of past surveillances indicates no adverse trend exists for this surveillance. The licensee submitted a special report dated January 23, 1996, in accordance with TS 4.7.2.2, indicating limited debonding which had been observed on very small portions (on the order of 4 linear inches per occurrence) during visual inspections. The licensee evaluated the cause and effect of the debonding and determined that it was most likely the result of poor application technique, was not representative of the overall condition of the bonded sealant (when compared to hundreds of linear feet inspected), and did not affect gasket or joint integrity. The licensee repaired the damaged sections and has committed to increased frequency (quarterly) inspections of the affected areas until the next scheduled annual inspection.

The staff has reviewed the licensee's justification for the above one-time surveillance interval extensions and based on its review, has determined that the safety impact would be small. Therefore, the proposed one-time surveillance interval extensions for TS 4.7.2.1.c items 1, 2, and 3, TS 4.7.2.1.e items 1, 2, and 4, and TS 4.7.2.1.h are acceptable.

2.15 Miscellaneous Surveillance Extensions

2.15.1 Instrument Excess Flow Check Valves

TS 4.6.3.4 requires each reactor instrumentation excess flow check valve shall be demonstrated operable at least once per 18 months by verifying that the valve checks flow. Instrument piping which is connected to the RCS, and which exits the primary containment, is designed with excess flow check valves. The instrument piping ends at the instrument connection. Each line contains a 0.25-inch restricting orifice and a manual isolation valve upstream of the excess flow check valve. A modified leak test is performed on the excess flow check valves and in accordance with ASME Code Section XI would be required every 2 years. The licensee received relief VR-009 allowing tests to be performed at a refueling outage periodicity. The licensee has requested a one-time surveillance interval extension of 16 days from the requirements of TS 4.6.3.4 to perform a test of the excess flow check valves.

The licensee reviewed RF03 and RF04 failure history for these valves to determine if there were any time-based elements associated with any failures. The licensee determined that the failure rate was less than 2 percent and that none of the failures would have affected the availability of the associated instrument to perform its intended safety function. Based on its review, the licensee determined that the impact on safety from the one-time short duration extension would be small.

The staff has reviewed the licensee's justification, and based on the information provided, has determined that the safety impact of the short duration one-time surveillance interval extension would be small. Therefore, the proposed one-time surveillance interval extension of the requirements of TS 4.6.3.4 to perform a test of the excess flow check valves is acceptable.

2.15.2 Traversing In-core Probe (TIP) Explosive Squib Operability

TS 4.6.3.5.b requires that each TIP system explosive isolation valve be demonstrated operable at least once per 18 months by removing the explosive squib from at least one explosive valve and initiating the explosive squib, such that the explosive squib in each explosive valve will be tested at least once per 90 months. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf life or operating life, as applicable. The (TIP) shear valve is provided as a backup isolation device for the TIP guide tube isolation valve. The isolation valve closes automatically upon receipt of a containment isolation signal after the TIP cable has been retracted. The shear valve can isolate the line if the TIP cannot be retracted or the automatic isolation valve fails to close. The shear valves are explosive type valves designed to shear the cable and seal the guide tube and are actuated by the control room operators. The licensee has requested a one-time surveillance interval extension of 92 days from the requirements of TS 4.6.3.5.b.

Continuity of the TIP shear valve firing circuits is continuously monitored in the control room to provide additional assurance that the valves will operate as designed. The primary automatic isolation valve is full-stroke exercised, stroke-time tested, and fail-safe tested quarterly in accordance with the IST program. The proposed one-time surveillance interval extension would still comply with the ASME Code Section XI paragraph IWV-3610 requirements for explosive charge testing on explosive valves. Based on the above, the licensee has concluded that the safety impact of the proposed one-time surveillance interval extension would be small.

The staff has reviewed the information provided by the licensee and based on its review, has determined that the safety impact of the proposed one-time surveillance interval extension would be small. Therefore, the proposed one-time surveillance interval extension of the requirements of TS 4.6.3.5.b to perform a test of the TIP explosive squib valves is acceptable.

3.0 Electrical Components Surveillance

3.1 Emergency Diesel Generators

3.1.1 Surveillance Requirement (SR) 4.8.1.1.2.e.1

SR 4.8.1.1.2.e.1 requires each of the diesel generators be demonstrated operable at least once per 18 months by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service. This surveillance becomes due on April 14, 1996. The licensee proposes that this surveillance be extended 216 days to reach the end of the refueling outage (i.e., RF05).

The diesel generators are subjected to operational testing every 31 days and fast-start testing every 184 days. This testing demonstrates the ability of the diesel engines to start and run under various loading conditions during the operating cycle. Since these tests have indicated no diesel generator degradation in the past cycle, the licensee is confident of the diesel generator's operability and capability to perform its intended function. Although the licensee has replaced several components under this proposed SR inspection in the past, the licensee contends that such replacements were done for preventive measures, not because of any degradation.

To provide additional assurance, the licensee has contacted the diesel vendor, who has concurred with the licensee's proposed SR extension provided that (1) certain additional inspections and measurements, which can be performed without disabling the diesel, be completed, (2) operating data since the last inspection be provided to the vendor for trending purposes, and (3) a vendor representative be allowed to witness a regularly scheduled full-load diesel run in the future.

Based on the satisfactory results of the periodic diesel generator testing performed during the operating cycle and past diesel SR inspections, the licensee finds that the proposed one-time extension of the SR would not result in any degradation that would negate the extension of the surveillance as requested, and that the additional measures recommended by the diesel vendor would reduce the impact on the diesel reliability even more.

The staff concurs with the licensee that the impact of the surveillance extension on the overall diesel generator reliability would be minimal. Therefore, the staff concludes that the proposed one-time extension of SR 4.8.1.1.2.e.1 is acceptable.

3.1.2 SRs 4.8.1.1.2.e.2, 3, 4.a, 4.b, 5, 6.a, 6.b, and 7

SR 4.8.1.1.2.e.2 requires each of the diesel generators be demonstrated operable at least once per 18 months by verifying the diesel generator capability to reject a load of greater than or equal to 1666 kW while maintaining engine speed less than the nominal speed plus 75 percent of the difference between nominal speed and the overspeed trip setpoint or 115 percent of nominal speed, whichever is lower. This surveillance becomes due on April 15, 1996. The licensee proposes this surveillance be extended 215 days to reach the end of the refueling outage.

SR 4.8.1.1.2.e.3 requires each of the diesel generators be demonstrated operable at least once per 18 months by verifying the diesel generator capability to reject a load of 2850 kW without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection. This surveillance becomes due on April 15, 1996. The licensee proposes this SR be extended 215 days to reach the end of the refueling outage.

SR 4.8.1.1.2.e.4.a requires each of the diesel generators be demonstrated operable at least once per 18 months by simulating a loss-of-offsite power by itself, and verifying deenergization of the emergency buses and load shedding from the emergency busses. This surveillance becomes due on June 28, 1996. The licensee proposes this SR be extended 141 days to reach the end of the refueling outage.

SR 4.8.1.1.2.e.4.b requires each of the diesel generators be demonstrated operable at least once per 18 months by simulating a loss-of-offsite power by itself, and verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected loads through the load sequencer, and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160±420 volts and 60 ±1.2 Hz during this test. This surveillance becomes due on June 28, 1996. The licensee proposes this SR be extended 141 days to reach the end of the refueling outage.

SR 4.8.1.1.2.e.5 requires each of the diesel generators be demonstrated operable at least once per 18 months by verifying that on an ECCS actuation test signal, without loss-of-offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160±420 volts and 60±1.2 Hz within 10 seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test. This surveillance becomes due on April 15, 1996. The licensee proposes this SR be extended 215 days to reach the end of the refueling outage.

SR 4.8.1.1.2.e.6.a requires each of the diesel generators be demonstrated operable at least once per 18 months by simulating a loss-of-offsite power in conjunction with an ECCS actuation test signal, and verifying deenerization of the emergency busses and load shedding from the emergency busses. This surveillance becomes due on June 28, 1996. The licensee proposes this SR be extended 141 days to reach the end of the refueling outage.

SR 4.8.1.1.2.e.6.b requires each of the diesel generators be demonstrated operable at least once per 18 months by simulating a loss-of-offsite power in conjunction with an ECCS actuation test signal, and verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160±420 volts and 60±1.2 Hz during this test. This surveillance becomes due on June 28, 1996. The licensee proposed this SR be extended 141 days to reach the end of the refueling outage.

SR 4.8.1.1.2.e.7 requires each of the diesel generators be demonstrated operable at least once per 18 months by verifying that all automatic diesel generator trips, except overspeed, generator differential, low lube oil pressure, crankcase overpressure, and failure to start are automatically bypassed for an emergency start signal. This surveillance becomes due on April 15, 1996. The licensee proposed this SR be extended 215 days to reach the end of the refueling outage.

The emergency ac power distribution system at Fermi 2 consists of the offsite power source and the onsite power source. The onsite power source for each emergency bus is a dedicated diesel generator. The diesel generators start automatically on a safety injection signal or on an emergency bus degraded voltage or undervoltage signal. Following the trip of offsite power, a sequencer/undervoltage signal strips nonpermanent loads from the emergency bus. When the diesel generator is tied to the emergency bus, the engineered safety feature (ESF) loads are then sequentially connected to their respective bus by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to ESF load breakers to prevent overloading the diesel generator. The above SRs demonstrate that each EDG operates as designed by testing the diesel generator control logic system.

The diesel generators are tested regularly (i.e., monthly and 184 days) to demonstrate the ability of the diesel engines to start and run under various load conditions during operations. This ensures the operability of diesel engine mechanical components.

For the operability of the diesel generator's control logic system (relays and contacts, etc.), the licensee referenced the NRC safety evaluation dated August 2, 1993, which related to the surveillance interval extension of the Peach Bottom Atomic Power System, Units 2 and 3, from 18 to 24 months where it stated, in part, that:

> Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic systems, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis....Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability.

Based on the above, the licensee took the position that the diesel control logic components are less likely to fail to function than diesel mechanical components.

The licensee also has evaluated failure modes with the time-based elements (e.g., wear, fatigue, and aging) for the diesel generator system for RF03 and RF04 by searching work orders, the surveillance tracking data system, and deviation event reports. This is to determine if it could impact the one-time extension of the above surveillance intervals. The evaluation revealed one incident in which the diesel generator output breaker failed because the diesel generator did not reach its rated speed and voltage within sufficient time. The root cause analysis determined that the diesel generator governor did not allow the diesel generator to reach rated speed and voltage in time to close the breaker. Subsequently, the governor was repaired to reduce the time the diesel takes to reach rated speed and voltage, and the timing for the diesel generator to reach rated speed and voltage is being verified every 184 days per SR 4.8.1.1.2.a.4.

Based on the inherent reliability of the diesel generator logic control system and a review of the historical failure data, the licensee finds that there is no reason to believe that the proposed surveillance extension would have any effects on diesel performance and availability. The staff concludes that the above proposed one-time TS extension of SRs 4.8.1.1.2.e.2, 3, 4.a, 4.b, 5, 6.a, 6.b, and 7 is acceptable.

3.1.3 SR 4.8.1.1.2.e.8

SR 4.8.1.1.2.e.8 requires each of the diesel generators be demonstrated operable at least once per 18 months by verifying the diesel generator operates for at least 24 hours. During the first 22 hours of this test, the diesel generator shall be loaded to greater than or equal to an indicated 2500-2600 kW and during the remaining 2 hours of this test, the diesel generator shall be loaded to an indicated 2800-2900 kW. The generator voltage and frequency shall be 4160±420 volts and 60±1.2 Hz within 10 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, the surveillance requires performance of SR 4.8.1.1.2.a.4. This surveillance becomes due on July 8, 1996. The licensee proposes this SR be extended 131 days to reach the end of the refueling outage.

Because the diesel generators are subjected to operational testing every 31 days and fast-start testing every 184 days, the ability of the diesel engines to start and run under various loading conditions has been demonstrated during the operating cycle. Since these tests have not indicated major diesel generator degradation, the licensee is confident of the diesel generator's operability and capability to perform their intended function. The licensee has also reviewed the diesel generator manufacturer prescribed inspection performed under TS SR 4.8.1.1.2.e.1 and finds that the past inspections have not revealed any major degradation that would negate the extension of the surveillance as requested.

In addition, the licensee has reviewed failure modes with the time-based elements (e.g., wear, fatigue, and aging) associated with the above SR for RF03 and RF04 by searching work orders, the surveillance tracking data system, and deviation event reports for actual failures. The review identified no failures which would have prevented the diesel generators from performing their safety function.

Based on the above, the staff finds that the one-time extension would have no effect on diesel generator performance. Therefore, the staff concludes that the proposed one-time extension of SR 4.8.1.1.2.e.8 is acceptable.

3.1.4 SR 4.8.1.1.2.e.9

SR 4.8.1.1.2.e.9 requires each of the diesel generators be demonstrated operable at least once per 18 months by verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 3100 kW. This test becomes due on August 14, 1996. The licensee proposed that SR 4.8.1.1.2.e.9 be extended 94 days to reach the end of the refueling outage.

The auto-connected loads for the diesel generators at Fermi 2 have not substantially changed since the last refueling outage. For those loads that have changed, the licensee has reevaluated loadings in accordance with the appropriate sections of the diesel generator load calculation. The licensee finds that the loads tested during RF04 combined with any additional changes since the last refueling outage do not exceed the diesel generator rating. Therefore, the licensee concludes that there is no reason to believe that the extension of this surveillance would have any effect on the diesel load-carrying capability.

In addition, the licensee has performed an evaluation of failures identified for RF03 and RF04 to determine if the failure modes contained time-based elements. No failures were identified for this SR. Based on the above information, the staff finds that the proposed one-time extension of SR 4.8.1.1.2.e.9 is acceptable.

3.1.5 SRs 4.8.1.1.2.e.10, 11, and 12

SR 4.8.1.1.2.e.10 requires each of the diesel generators be demonstrated operable at least once per 18 months by verifying the diesel generator's capability to synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power, transfer its loads to the offsite power source, and be restored to its standby status. This surveillance is due on August 14, 1996. The licensee proposed SR 4.8.1.1.2.e.10 be extended 94 days to reach the end of the refueling outage.

SR 4.8.1.1.2.e.11 requires each of the diesel generators be demonstrated operable at least once per 18 months by verifying that the automatic load sequence timers are operable with the interval between each load block within ±10 percent of its design interval. This surveillance is due on July 16, 1996. The licensee proposed SR 4.8.1.1.2.e.11 to be extended 123 days to reach the end of the refueling outage.

SR 4.8.1.1.2.e.12 items a, b, and c require each of the diesel generators be demonstrated operable at least once per 18 months by verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

a) 4160-volt ESF bus lockout
b) Differential trip
c) Shutdown relay trip

These surveillances are due on April 15, 1996. The licensee proposed SR 4.8.1.1.2.e.12 be extended 215 days to reach the end of the refueling outage.

For the mechanical diesel components, past surveillance testing history has demonstrated the ability of the diesel engines to start and run under various load conditions during normal operations. With regards to testing the diesel generator control logic system, the licensee again referenced that the control logic (relays and contacts, etc) components, as documented in NEDC-30936P, are less likely to fail than mechanical components (pumps and valves, etc). As an example, a historical review of the load sequencer operation found that the timing for the load sequencer for both divisions was always in tolerance. The extension of the surveillance interval for the above logic tests will therefore have minimal impact on the diesel generator's ability to function.

The licensee also reviewed failures identified in RF03 and RF04 to determine if the failure modes contained time-based elements which could impact the one-time extension of the surveillance interval. The licensee's review found no failures that would adversely affect the system's ability to perform its safety function.

Based on the above, the staff finds that extending the surveillance intervals to reach the end date of RF05 has minimal impact on the system's failure probability and its reliability. Therefore, the staff concludes that the proposed extension of SRs is acceptable.

3.2 <u>130-Volt DC Station Batteries and Battery Chargers</u>

3.2.1 SR 4.8.2.1.C items 3 and 4

SR 4.8.2.1.C items 3 and 4 require that each of the required 130-volt batteries and chargers be demonstrated operable at least once per 18 months by verifying that:

Item 3. The resistance of each cell-to-cell and terminal connection is less than or equal to 150 x 10^{-6} ohm, and

Item 4. The battery charger will supply at least 100 amperes at a minimum of 129 volts for at least 4 hours.

These items become due on May 10, 1996. The licensee proposes SR 4.8.2.1.C items 3 and 4 to be extended 190 days to reach the end of the refueling outage.

For the extension of SR 4.8.2.1.C item 3, the licensee has reviewed RF03 and RF04 records for the battery terminal connection resistance and found no significant corrosion that would affect the terminal resistance. All resistance measurements were far below the maximum allowed. Should corrosion be observed during the battery quarterly (i.e., every 92 days) surveillance test, a battery terminal connection resistance test will be performed to verify resistance remains below the TS limit of 150 microhm.

For the extension of SR 4.8.2.1.C item 4, the licensee has reviewed battery charger test records for RF01 through RF04 and finds no failures related to the battery chargers.

Based on successful review of the past battery terminal connection resistance and the battery charger tests, the licensee determined that the one-time extension of surveillance interval would not impact the availability of the battery and its charger. On this basis, the staff concludes that the one-time extension of SR 4.8.2.1.C items 3 and 4 is acceptable.

3.2.2 SR 4.8.2.1.d

SR 4.8.2.1.d requires that each of the required 130-volt batteries and chargers be demonstrated operable at least 18 months by verifying that either:

1. The battery capacity is adequate to supply and maintain in operable status all of the actual emergency loads for the design duty cycle (4 hours) when the battery is subjected to a battery service test, or

2. The battery capacity is adequate to supply a dummy load of the following profile while maintaining the battery terminal voltage greater than or equal to 105 or 210 volts, as applicable:

   a) Batteries 2PA and 2PB greater than or equal to 710 amperes during the initial 6 seconds of the test.

   b) Batteries 2PA and 2PB greater than 182 amperes during the next 42 seconds of the test.

   c) Batteries 2PA and 2PB greater than or equal to 54 amperes during the next 4 hours of the test.

   d) Batteries 2PA and 2PB greater than or equal to 480 amperes during the last 6 seconds of the test.

This battery service test becomes due on May 10, 1996. The licensee proposes SR 4.8.2.1.d be extended 190 days to reach the end of the refueling outage.

The Fermi 2 division 1 and 2 batteries were capacity tested in May and June, respectively, of 1986 and April and May, respectively, of 1991. No changes in the battery capacity were detected with all batteries tested having capacity factors greater than 100 percent. Also, the licensee estimated the battery service life by extrapolating the degradation rate for the worst case. It showed that the batteries will not reach TS degradation level (90 percent of the manufacturer's rating) until their 15th year of service (2001). The licensee has also reviewed its dc distribution system loads and finds that no loads have been added to the batteries since the last test. In view of the fact that there were no failures in the battery capacity and service tests in the past and no changes in the battery loads, the licensee finds that the extension of the battery service test can be justified. The staff concurs with the licensee that the proposed one-time extension of SR 4.8.2.1.d is acceptable.

3.3 <u>Primary Containment Penetration Conductor Overcurrent Protective Devices</u>

3.3.1 SR 4.8.4.2.a.1.a

SR 4.8.4.2.a.1.a requires each of the primary containment penetration conductor overcurrent protective devices shown in TS Table 3.8.4.2-1 be demonstrated operable at least once per 18 months by performing a channel calibration of the associated 4.16-kV circuit protective relays. This item becomes due on June 24, 1996. The licensee proposed SR 4.8.4.2.a.1.a be extended 103 days to October 5, 1996.

To support the extension of SR 4.8.4.2.a.1.a for the primary containment penetration conductor overcurrent protective devices, the licensee evaluated the reactor recirculation pumps penetration protection (i.e., overcurrent relays) for a 30-month period using a GE extrapolation method. The drift for these relays was found to be within TS requirements. The licensee also performed an evaluation of the channel calibration failures identified during RF03 and RF04. The review identified only one relay failure.

Based on the above, the licensee finds that one-time surveillance interval extension is justified for the primary containment penetration conductor overcurrent protection devices. Therefore, the staff concludes that the requested extension of SR 4.8.4.2.a.1.a is acceptable.

3.3.2 SR 4.8.4.2.a.1.b

SR 4.8.4.2.a.1.b requires each of the primary containment penetration conductor overcurrent protective devices shown in TS Table 3.8.4.2-1 be demonstrated operable at least once per 18 months by an integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and overcurrent control circuits function as designed. This test is due on June 24, 1996. The licensee proposed SR 4.8.4.2.a.1.b be extended 103 days to October 5, 1996.

The licensee performed a historical search of the 18-month surveillance tests for SR 4.8.4.2.a.1.b for the last two refueling outages to identify all failed or partially failed tests. This review identified no failures. The review also identified no failures of the breakers to open during these surveillance tests.

The licensee believes that the integrated system test of the primary containment penetration conductor overcurrent protective devices is essentially a logic system functional test. Since the probability of a relay or contact failure is smaller than that of mechanical component failure, the proposed extension of the integrated system test represents no significant change in the overall safety system unavailability.

Based on the historical search of the surveillance tests and the finding regarding the logic system functional testing, the staff concludes that the extension of SR 4.8.4.2.a.1.b is justified.

3.4 <u>AC Electrical Power System Shutdown Operability</u>

During shutdown, SR 4.8.1.2 requires that at least the required ac electrical power sources shall be demonstrated operable per SRs 4.8.1.1.1, 4.8.1.1.2, and 4.8.1.1.3, except for the requirement of 4.8.1.1.2.a.5.

The proposed extension of SRs submitted with this amendment request applies only to diesel generator tests performed every 18 months while the unit is in a refueling outage. Since the proposed SR 4.8.1.2 is performed when the unit is shut down and since no extension was requested, the staff finds that no evaluation is necessary.

3.5 Battery Shutdown Operability

During shutdown, SR 4.8.2.2 requires that at least the required battery and chargers be demonstrated operable per SR 4.8.2.1. The proposed extension of SRs submitted with this amendment request applies only to battery tests performed every 18 months while the unit is in a refueling outage. Since the proposed SR 4.8.2.2 is performed when the unit is shut down and since no extension was requested, the staff finds that no evaluation is necessary.

3.6 SUMMARY CONCLUSION

For the reaons stated above, the staff finds that the requested extensions of time for surveillances of the components and systems described above comply with 10 CFR 50.36(c)(3) and 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 surveillance requirements. 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 (60 FR 58400). 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 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.

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