August 4, 1998

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

SUBJECT: FERMI 2 - ISSUANCE OF AMENDMENT RE: ONE-TIME TECHNICAL SPECIFICATION REVISION TO ALLOW EXTENSION OF REFUELING OUTAGE SIX SURVEILLANCE REQUIREMENTS (TAC NO. MA2183)

Dear Mr. Gipson:

The Commission has issued the enclosed Amendment No. 124 to Facility Operating License No. NPF-43 for the Fermi 2 facility. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated June 26, 1998 (NRC-98-0040), as supplemented July 16 and July 23, 1998.

The amendment provides a one-time extension of the interval for a number of TS surveillance requirements that will be performed during the sixth refueling outage. TS 4.0.2 and Index page xxii are revised and TS tables 4.0.2-1 and 4.0.2-2 are replaced to reflect the extensions. By letter dated July 16, 1998, you made some editorial changes and withdrew the portion of the June 26, 1998, submittal related to TS 4.0.5 for the inservice testing of two valves. A change to the schedule for these valves will be handled within the Inservice Testing Program and a TS change is not necessary. By letter dated July 23, 1998, you added two instruments to Table 4.0.2-1 for Surveillance Requirement 4.3.1.3.

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:

Andrew J. Kugler, 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. 124 to NPF-43
2. Safety Evaluation
cc w/encl: See next page

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DATED: August 4, 1998

AMENDMENT NO. 124 TO FACILITY OPERATING LICENSE NO. NPF-43 - FERMI 2

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

DETROIT EDISON COMPANY

DOCKET NO. 50-341

FERMI 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 124
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 June 26, 1998, as supplemented July 16 and July 23, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

<u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 124, 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment: Changes to the Technical Specifications

Date of Issuance: August 4, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 124

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 0-4
3/4 0-5	3/4 0-5
3/4 0-6	3/4 0-6
3/4 0-7	3/4 0-7

*Overleaf pages 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 sixth 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:

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 SEPTEMBER 14, 1998

SURVEILLANCE REQUIREMENT	DESCRIPTION
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 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, Table 4.3.1.1-1, Item 2.b	APRM Flow Biased Thermal Power - High
4.3.1.3, Table 4.3.1.1-1, Item 2.c	APRM Fixed Neutron Flux - High
4.3.2.1, Table 4.3.2.1-1, Item 1.a.1	Pri Cont Isolation Actuation Rx Water Low - 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 2 cal
4.3.2.1, Table 4.3.2.1-1, Item 1.a.3	Pri Cont Isolation Actuation Rx Water Low - 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.c.3	Pri Cont Isolation Actuation Main Steam Line Flow High 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 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 2.e	RWCU Isolation Rx Water Low Level - Level 2 channel cal
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 5.c	RHR S/D Cooling Rx manual initiation functional test
4.3.2.1, Table 4.3.2.1-1, Item 6.b	Sec. Cont. Isolation - Drywell Press High channel cal
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 3.a	HPCI RPV Low Level 2 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 4.a	ADS RPV Low Level 1 Cal
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.h	ADS Drywell Pressure High Bypass Timer
4.3.4, Table 4.3.4-1, Item 1	RPV Low Water Level 2 Cal (ATWS)
4.3.4, Table 4.3.4-1, Item 2	RPV Press High 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
4.3.7.5, Table 4.3.7.5-1, Item 1	RPV Press Cal Accident Mon.
4.3.7.5, Table 4.3.7.5-1, Item 2.a	RPV Fuel Zone Level Cal Accident Mon
4.3.7.5, Table 4.3.7.5-1, Item 2.b	RPV Wide Range Level 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 16	CTMT Isolation Valve Position Cal Accident Mon
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
4.4.2.2.b	SRV Low Low Set Pressure setpoint Cal and LSFT
4.5.1.d.2.a	ADS System Functional Test
4.6.3.2	Primary Containment Isol Valve operability
4.7.4.c.1	RCIC Functional Test
4.8.4.2.a.1.a	Pri. Cont. Pen. Conductor Overcurrent Devices Functional Test
4.8.4.2.a.1.b	Pri. Cont. Pen. Conductor Overcurrent Devices Functional Test

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TABLE 4.0.2-2

SURVEILLANCE TEST INTERVALS EXTENDED TO OCTOBER 18, 1998

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 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
4.3.3.2	ECCS Logic System Functional Tests
4.3.3.3(a)	ECCS Response Time Tests
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
4.5.1.c.1	ECCS System Functional Test
4.6.5.1.d.1	Secondary Containment SGTS Test
4.6.5.1.d.2	Secondary Containment SGTS Test
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
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.d	130 VDC Battery Capacity
4.8.4.5.a	SLCS Circuit Breakers 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 sixth refueling outage.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 124 TO FACILITY OPERATING LICENSE NO. NPF-43

DETROIT EDISON COMPANY

FERMI 2

DOCKET NO. 50-341

1.0 INTRODUCTION

By letter dated June 26, 1998 (NRC-98-0040), as supplemented July 16 (NRC-98-0096) and July 23 (NRC-98-117), 1998, the Detroit Edison Company (DECo or the licensee) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License No. NPF-43 for Fermi 2. The proposed amendment would provide a one-time extension of the interval for a number of TS surveillance requirements (SRs) that will be performed in the sixth refueling outage. TS 4.0.2 and Index page xxii would be revised and TS tables 4.0.2-1 and 4.0.2-2 would be replaced to reflect the extensions. The proposed amendment would extend all 18-month surveillances that cannot be performed at power, and the maximum extension would be 61 days. The letter dated July 16, 1998, made some editorial changes and withdrew the portion of the June 26, 1998, submittal related to TS 4.0.5 for the inservice testing of two valves. A change to the schedule for these valves will be handled within the Inservice Testing Program and a TS change is not necessary. The letter dated July 23, 1998, added an additional surveillance requirement for two instruments to the amendment. The information in these two supplements was within the scope of the original *Federal Register* notice and did not change the staff's initial proposed no significant hazards considerations determination.

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. Revised TS tables 4.0.2-1 and 4.0.2-2 would indicate those surveillances that would be extended to September 14, 1998 (by which time the plant will be in Operational Condition 4, Cold Shutdown), and October 18, 1998 (the planned end of the refueling outage), respectively. The surveillances that are extended to September 14, 1998, are those that support equipment that is required to be operable in Operational Condition 1, 2, or 3. Once the plant enters Operational Condition 4 in the refueling outage, this equipment will not be required to be operable until the unit is restarted at the end of the outage. In general, the surveillances that are extended to October 18, 1998, are those that support equipment that is required to be operable in Operational Conditions 4 or 5 or other shutdown situations. Also, some specific line items that are not required to be operational during Operational Condition 4 or 5 have a requested extension date of the end of the outage. The reason for this is that the surveillance procedure that tests these components also fulfills the surveillance requirements for components that are required to be operational during Operational Condition 4 or 5. In addition, the licensee requested that the "N times 18 months" cumulative surveillance interval for various response time testing be baselined to the sixth refueling outage using the dates the response time tests are actually performed in the outage.

2.0 BACKGROUND

Fermi 2 experienced an extended shutdown at the beginning of the current operating cycle. As a result, a large number of the SRs that will be performed during the sixth refueling outage will reach the end of their surveillance intervals (including the 25-percent extension) in late August 1998. In its June 26, 1998, submittal, the licensee expressed its concern that a plant shutdown in August could jeopardize the reliability of the electrical grid in its service area and possibly beyond. This concern is discussed in more detail in the North American Electric Reliability Council report, "1998 Summer Assessment: Reliability of Bulk Electricity Supply in North America," published in May 1998. The licensee has proposed a one-time extension of the surveillance intervals for a large number of SRs that will be performed during the sixth refueling outage. The proposed extensions would allow the licensee to delay the refueling outage until early September, by which time the shutdown of Fermi 2 would not be expected to adversely affect grid reliability.

The licensee requested a similar amendment for a one-time extension of the interval for a number of TS SRs in a submittal dated September 20, 1995, as supplemented December 18 and December 22, 1995. Almost all of the SRs involved in the present request were also a part of the 1995 request. However, for the SRs common to both submittals, in all cases the length of the extension requested in the 1995 submittals was longer than that requested in the 1998 submittal. The NRC approved the 1995 amendment request in Amendment No. 106 dated March 1, 1996. As discussed in the July 16, 1998, submittal, the licensee considered the information provided in the 1995 submittals during its review of the current amendment request. The staff also considered this information during its review.

3.0 EVALUATION

As part of the development of the justification for this amendment, the licensee reviewed its surveillance test history data base to identify any tests during the last (fifth) refueling outage that were coded as either equipment failure or "partially complete." The licensee indicated that the partially complete category includes not only component failures causing an interruption of the test, but also procedure problems, plant conditions that might have precluded further testing, etc. The licensee then retrieved and evaluated the surveillance test records of the list of "failures" identified by this search. The evaluation categorized the failure modes for the components. Failures that would have no effect on the safety function were eliminated. Where the failure may have impacted the safety function, the licensee conducted further evaluations that considered factors such as (1) whether a similar failure would be detected during the extended cycle by other, more frequently conducted tests (such as functional tests), preventive maintenance activities, inservice testing, or other monitoring activities, (2) whether the failure was caused by a special event, occurrence or maintenance activity that has not occurred during the current cycle, and (3) whether or not the failure could have been time dependent and thus be relevant to the proposed extension. As discussed in Section 2.0, additional historical information that was provided in a submittal dated September 20, 1995, as supplemented

December 18 and December 22, 1995, formed the bases for a similar interval extension for many of these same SRs that was approved by the staff in Amendment No. 106 dated March 1, 1996. The staff considered this additional historical information during the evaluation of this amendment.

3.1 Scram Accumulator, SR 4.1.3.5.b.2

The licensee has requested a 17-day extension for SR 4.1.3.5.b.2 which requires demonstrating that each control rod scram accumulator is operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) 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 set point with no control rod drive pump operating. This SR becomes overdue on October 2, 1998.

The Control Rod Drive (CRD) hydraulic system supplies and controls the pressure and flow to and from the drives through hydraulic control units (HCUs). The basic components in each HCU are manual, pneumatic, and electrical valves; an accumulator; related piping; electrical connections; filters; and instrumentation. 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 or replaced by RPV pressure.

The scram accumulator check valves are required only to maintain pressure assuming that no CRD pump is operating. A review of check valve performance for the fifth refueling outage has indicated no failures to maintain pressure. In addition, in its September 20, 1995, submittal the licensee indicated that there were no failures in the third and fourth refueling outages. Therefore, the licensee has concluded that the probability that the check valve would fail at a time when no CRD pump was available is low. In addition, the requested extension is for an SR that will not become overdue until the unit enters the refueling outage. During refueling all rods in cells containing fuel are required to be inserted, except for a single rod that can be withdrawn under the one-rod-out-interlock provided shutdown margin requirements have been satisfied. In that case, failure of the associated accumulator has no consequence. For multiple rod withdrawal, the associated cells must be defueled, eliminating the need for accumulator operability. Therefore, accumulator operability is not required for refueling activities.

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 SR 4.1.3.5.b.2 is acceptable.

3.2 Standby Liquid Control System (SLCS), SR 4.1.5.d items 1 through 3

The licensee has requested extensions of 51, 25, and 51 days for SR 4.1.5.d items 1, 2, and 3, respectively. Items 1 and 3 become overdue on August 29, 1998, and item 2 becomes overdue on September 24, 1998. 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.

SR 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. SR 4.1.5.d.2 requires demonstrating that the pump relief valve set point is less than or equal to 1400 psig and verifying that the relief valve does not actuate during recirculation to the relief tank. SR 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 fifth refueling outage. 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 the fourth refueling outage 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 that in general the licensee meets the guidance in Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals To Accommodate a 24-Month Fuel Cycle." Therefore, the proposed one-time surveillance interval extensions for SR 4.1.5.d items 1, 2, and 3 are acceptable.

3.3 Calibration Interval Extensions

3.3.1 Reactor Protection System (RPS), Table 4.3.1.1-1, Items 3, 4, 6, and 7

The licensee has requested one-time calibration interval extensions of 13 to 22 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 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.

3.3.2 Isolation Actuation Instrumentation, Table 4.3.2.1-1, Items 1.a.1, 2, and 3, 1.b, 1.c.1, 2, and 3, 1.d, 1.e, 1.f, 2.e, 5.a, 6.a, and 6.b

The licensee has requested one-time calibration interval extensions of 19 to 57 days for several isolation actuation instrument channels 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 instrument channels affected by this request include reactor water low level - Level 3, Level 2, and Level 1; drywell pressure high; main steam line radiation high, pressure low, flow high, and tunnel temperature high; condenser pressure high; and turbine building area temperature high. Other instrument channels affected include the reactor water cleanup (RWCU) isolation on reactor water low level - Level 2, residual heat removal shutdown cooling reactor water low level - Level 3, and the secondary containment isolations on reactor water low level - Level 2 and drywell pressure high.

3.3.3 Emergency Core Cooling System (ECCS) Actuation Instrumentation, Table 4.3.3.1-1, Items 1.a, b, and c, 2.a, b, c, d, and e, 3.a, 3.b, and 4.a, f, and h

The licensee has requested one-time calibration interval extensions of 10 to 57 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 set points, 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 high-pressure safety injection (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; and the CS and LPCI reactor steam dome pressure low calibrations (both for ECCS permissive injection and LPCI loop select logic).

3.3.4 Anticipated Transient Without Scram (ATWS) Recirculation Pump Trip (RPT) Actuation Instrumentation, Table 4.3.4-1, Items 1 and 2

The licensee has requested one-time calibration interval extensions of 20 and 22 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 instrument channels affected are RPV low level - Level 2 and RPV pressure high.

3.3.5 Reactor Core Isolation Cooling (RCIC) Actuation Instrumentation, Table 4.3.5.1-1, Items a and b

The licensee has requested one-time interval extensions of 20 days for two RCIC actuation instrument channels 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 instrument channels affected are the RPV low level - Level 2 and RPV high level - Level 8.

3.3.6 Post-Accident Monitoring System, Table 4.3.7.5-1, Items 1, 2.a, 2.b, 12, and 16

The licensee has requested one-time calibration interval extensions of 11 to 22 days for several post-accident monitoring system instrument channels 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 affected instrument channels include RPV pressure, RPV fuel zone level, RPV wide range level, containment isolation valve position, and containment high range radiation.

3.3.7 Feedwater/Main Turbine Trip System, Table 4.3.9.1-1, Item a

The licensee has requested a one-time interval extension of 15 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.

3.3.8 Alternate Shutdown System, Table 4.3.11.1-1, Items 7 and 8

The licensee has requested a one-time calibration interval extension of 14 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.

3.3.9 Safety Relief Valves (SRVs), SR 4.4.2.2.b (Partial)

The licensee has requested a one-time calibration interval extension of 22 days for the SRV low-low set pressure setpoint calibration. The SRVs' function is 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. SR 4.4.2.2.b requires that the low-low set function pressure actuation instrumentation be demonstrated operable by performance of a channel calibration.

3.3.10 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 referenced a Rosemount, Inc., Report D89000126, Revision A, "30 Month Stability Specification For Rosemount 1152, 1153, 1154 Pressure Transmitters," dated February 1990, which was accepted by the NRC in a safety evaluation dated August 2, 1993, for the Peach Bottom Atomic Power Station. This report supports the extension of the calibration interval for the transmitters from 18 to 30 months based on a reduction in the drift allowance. Based on the setpoint methodology and drift allowance used at Fermi-2, the licensee indicates that the calculations for these transmitters contain at least 30 months drift allowance. This information applies to all of the pressure and level instrument channels affected by this interval extension.

The licensee performed a historical search of instrument 18-month surveillance test and calibration data from the previous (fifth) refueling outage. No failures were noted and in all cases the data were found to be within design tolerances. The licensee concluded that the impact of the extensions of the affected instrument surveillance tests and calibrations on operability would be small. In addition, the historical information provided in submittal dated September 20, 1995, as supplemented December 18 and December 22, 1995, formed the bases for a similar interval extension for many of these same SRs that was approved by the staff in Amendment No. 106 dated March 1, 1996. Finally, the licensee 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.

3.4 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.

3.4.1 Reactor Mode Switch Shutdown Position Functional Test and RPS Logic System Functional Test, Table 4.3.1.1-1, Item 11, and SR 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 14 days for the channel functional test required by TS Table 4.3.1.1-1.

The RPS instrumentation 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 56 days for the RPS LSFT required by SR 4.3.1.2.

3.4.2 Isolation Actuation Logic System Functional Tests, Table 4.3.2.1-1, Items 1.h, 2.d, and 5.c, and SR 4.3.2.2

The 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 be reset by the operator only when initiating conditions have returned to normal. The licensee has requested a one-time surveillance interval extension of 57 days for the isolation actuation instrument LSFT required by SR 4.3.2.2.

The primary containment isolation actuation and residual heat removal (RHR) shutdown cooling mode isolation manual actuation 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 18 and 15 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 17 days from the RWCU - SLCS initiation channel functional test required by TS Table 4.3.2.1-1.

3.4.3 ECCS Logic System Functional Tests, Table 4.3.3.1-1, Items 1.d and 2.h, and SR 4.3.3.2

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 set points, logic signals are generated to initiate ECCS and ECCS support systems. The licensee has requested a one-time surveillance interval extension of 57 days for the ECCS LSFT required by SR 4.3.3.2.

The CS and LPCI 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 licensee has requested one-time surveillance interval extensions of 52 and 49 days, respectively, for the channel functional tests required by TS Table 4.3.3.1-1.

3.4.4 ATWS-RPT Logic System Functional Test, SR 4.3.4.2

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 licensee has requested a one-time surveillance interval extension of 22 days for the ATWS-RPT LSFT required by SR 4.3.4.2.

3.4.5 RCIC Logic System Functional Test, SR 4.3.5.2

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 20 days for the RCIC LSFT required by SR 4.3.5.2.

3.4.6 Control Rod Block Instrumentation, Table 4.3.6-1, Items 5.b and 7

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 scram discharge volume (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 14 days each for the channel functional tests of the SDV-scram trip bypass and reactor mode switch required by TS Table 4.3.6-1.

3.4.7 Feedwater/Main Turbine Trip Logic System Functional Test, SR 4.3.9.2

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 15 days for the LSFT required by SR 4.3.9.2.

3.4.8 SRV Logic System Functional Test, SR 4.4.2.2.b (Partial)

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 22 days from the requirement of SR 4.4.2.2.b. The related calibration interval extension for this surveillance has been previously discussed above.

3.4.9 ECCS Functional Test, SRs 4.5.1.c.1 and 4.5.1.d.2.a

The CS system, together with the LPCI mode of the RHR system, is 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. SR 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 that 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 56 days for the ECCS functional test required by SR 4.5.1.c.1.

SR 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 20 days for the ADS functional test required by SR 4.5.1.d.2.a.

3.4.10 Primary Containment Isolation Valves, SR 4.6.3.2

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 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. SR 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 23 days for the primary containment isolation valve operability demonstration required by SR 4.6.3.2.

3.4.11 Secondary Containment Drawdown Test and Isolation Damper Actuation, SRs 4.6.5.1.d.1 and 2 and 4.6.5.2.b

Secondary containment is designed to minimize any ground level release of radioactive material that 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. SR 4.6.5.1.d.1 and 4.6.5.1.d.2 require testing to verify the capability of the standby gas treatment system to attain an adequate vacuum in a specified time in the secondary containment and to maintain the vacuum. The licensee has requested one-time surveillance interval extensions of 19 days each from the secondary containment draw-down test requirements of SR 4.6.5.1.d.1 and d.2.

SR 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 17 days from the secondary containment isolation damper actuation requirement of SR 4.6.5.2.b.

3.4.12 Safety-Related Service Water Systems, SRs 4.7.1.2.b, 4.7.1.3.b, and 4.7.1.4.b

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. SR 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. SR 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 one-time surveillance interval extensions of 50 days each from the EECW and EESW automatic actuation tests required by SR 4.7.1.2.b and 4.7.1.3.b. SR 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 53 days from the EDGSW pump automatic actuation test required by SR 4.7.1.4.b.

3.4.13 RCIC System Functional Test, SR 4.7.4.c.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. SR 4.7.4.c.1 requires the performance of a system functional test of the RCIC system. This functional test includes simulated automatic actuation and restart and verification that each automatic valve in the flow path actuates to its correct position. The licensee has requested a one-time surveillance interval extension of 20 days for the RCIC system functional test required by SR 4.7.4.c.1.

3.4.14 Justification and Discussion

As stated previously, the above LSFTs 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 affected LSFTs and actuation tests for the fifth refueling outage 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. Additional historical information that was provided in a submittal dated September 20, 1995, as supplemented December 18 and December 22, 1995, formed the bases for a similar interval extension for many of these same

SRs that was approved by the staff in Amendment No. 106 dated March 1, 1996. This information also supports the licensee's conclusion.

The licensee also referenced an industry study (Boiling Water Reactor (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 report indicated that, with the exception of ADS, system unavailability is increased appreciably only after increasing the test interval beyond 90 months. The report also indicated that for ADS, assuming a 36-month test interval, unavailability caused by logic system failure remains two orders of magnitude less than the currently accepted unavailability caused by valve actuation solenoid failure. 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, therefore, concluded that the safety impact of extending the surveillance interval for this test would also be small.

The secondary containment drawdown tests (SR 4.6.5.1.d.1 and d.2) and the RCIC functional test (SR 4.7.4.c.1) were not discussed in the licensee's submittal dated September 20, 1995, as supplemented December 18 and December 22, 1995, and Amendment No. 106 dated March 1, 1996. In the June 26, 1998, submittal the licensee indicated it had performed a historical review of past testing of the standby gas treatment system for SR 4.6.5.1.d.1 and d.2 and found no indication of degradation or failure. Concerns during this testing have only been raised when the atmospheric pressure transmitters were affected by wind gusts that produced spikes in the differential pressure recorder. However, the licensee indicated that these spikes are not representative of bulk differential pressure conditions between the reactor building and atmosphere and that the drawdown of secondary containment has consistently been accomplished within the 567-second time requirement of SR 4.6.5.1.d.1 and a 0.25-inch water gauge vacuum has been consistently maintained during previous tests. In the June 26, 1998, submittal the licensee indicated it had performed a historical review of past testing of the RCIC system for SR 4.7.4.c.1 for the last three refueling outages and found no indication of degradation or failure. The licensee, therefore, concluded that the safety impact of extending the surveillance intervals for these tests would also be small.

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 Generic Letter 91-04. Therefore, the proposed one-time interval extensions are acceptable.

3.5 <u>RPS Response Time Tests, SR 4.3.1.3</u>

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 period 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 a one-time surveillance interval extension of 18 days for the RPS response time testing required by SR 4.3.1.3 for average power range monitor (APRM) items 2.b (APRM flow-biased thermal power - high) and 2.c (APRM fixed neutron flux - high) in Table 4.3.1.1-1.

The licensee indicated that (1) there are diverse instrument channels that can generate a scram signal, (2) there are redundant channels for these two functions for which this SR will remain current, (3) the instrument failure probability is a small fraction of the total scram failure probability, and (4) sluggish response is a small contributor to the overall instrument failures.

In addition, the licensee referenced the NRC safety evaluation dated August 2, 1993, for the extension of the Peach Bottom Atomic Power Station Units 2 and 3 surveillance intervals from 18 months to 24 months, which states:

> The RPS system consists of two independent trip systems with at least two subchannels of a parameter per trip system. The logic of the RPS system is such that either subchannel can trip a trip system and that both trip systems must trip to cause a reactor trip. The logic is such that a single failure will neither cause nor prevent a required reactor scram. The licensee states that, based on the inherent redundancy in the RPS system, the impact of extending the response time surveillance interval on system availability is small.

and,

> The staff reviewed the information presented by the licensee and concluded that the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed change to TS Table 4.1.2 concerning response time testing, is acceptable.

Finally, the licensee performed a historical search of the 18 month surveillances for response time testing for the last three refueling outages to identify and evaluate all failed or partially failed tests. The licensee did not identify any RPS response time test failures during this review.

Rebaselining of the (N x 18)-month period for these surveillances, as discussed further in section 3.6 of this safety evaluation, is not necessary for these SRs because these functions will be eliminated when the new power range neutron monitoring system is installed in the sixth refueling outage.

Based on the information provided by the licensee, the staff 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 for time response testing required by SR 4.3.1.3 is acceptable.

3.6 ECCS Response Time Tests, SR 4.3.3.3

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 response time testing of only one channel of a given function during any given 18-month surveillance period 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 a one-time surveillance interval extension of 52 days for the ECCS response time testing required by SR 4.3.3.3.

The licensee indicated that the response time failure probability is a very small fraction of the total ECCS failure probability. The licensee also stated that the NRC had approved the extension of the ECCS response time testing interval from 18 to 24 months for the Peach Bottom Atomic Power Station Units 2 and 3 in a safety evaluation dated August 2, 1993. Finally, the licensee performed a review of the surveillance test history for the ECCS instrumentation response time tests for the fifth refueling outage and found no evidence of test failures that would invalidate the conclusion that the surveillance extension is acceptable. Additional historical information was provided in submittals dated September 20, 1995, as supplemented December 18 and December 22, 1995, that formed the bases for a similar interval extension for this same SR that was approved by the staff in Amendment No. 106 dated March 1, 1996. Based on its evaluation, the licensee concluded that the interval extension is justified.

In addition, the licensee requested that the (N x 18)-month cumulative surveillance interval for affected response time testing be baselined to this outage; i.e., the beginning of the (N x 18)-month interval be restarted at the respective response time testing dates to be performed during the sixth refueling outage. The licensee indicated that this re-establishment of the baseline will ensure that future response time testing intervals, with respect to the cumulative "N times 18 months" interval, will not become late due to the interval extensions that are required for the sixth refueling outage. The licensee also stated that the same justification provided in the June 26, 1998, submittal for the individual response time surveillance interval extensions would apply to the (N x 18)-month cumulative surveillance interval extensions because the cumulative surveillance interval would not be extended by more than that being requested for individual response time tests.

Based on the information provided by the licensee, the staff 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 for time response testing required by

SR 4.3.3.3 is acceptable. The staff also concludes that the baselining of the (N x 18)-month interval to the sixth refueling outage for the affected response time tests is acceptable.

3.7 Control Room Emergency Filtration System, SR 4.7.2.1.c.1, 2, and 3 and 4.7.2.1.e.1, 2, and 4

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 high-efficiency particulate air (HEPA) filters and charcoal adsorbers. 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.

SR 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 cubic feet per minute plus or minus 10 percent, when tested in accordance with American National Standards Institute/American Society of Mechanical Engineers (ANSI/ASME) Standard N510-1980, "Testing of Nuclear Air Cleaning Systems." SR 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 adsorbers 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. The licensee has requested a one-time surveillance interval extension of 28 days from the requirements of SR 4.7.2.1.c and SR 4.7.2.1.e items 1 and 4. The licensee has requested a one-time surveillance interval extension of 17 days from the requirements of SR 4.7.2.1.e.2.

The licensee has indicated that other, more frequent tests demonstrate operability of the CREFS such as SR 4.7.2.1.b which requires operability of the main control room fans and verification of flow through the HEPA filters and charcoal adsorbers 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 licensee also indicated that the extension of the logic testing required by SR 4.7.2.1.e.2 is supported by 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 was discussed in Section 3.4.14 of this safety evaluation. The licensee stated that it performed a review of the surveillance test history for these tests for the fifth refueling outage and found no evidence of test failures that would invalidate the conclusion that the surveillance extension is acceptable. Based on this review and the redundant equipment in the subject systems, the licensee concluded that the safety impact of the extension on system availability was small. Additional historical information that was provided in a submittal dated September 20, 1995, as supplemented December 18 and December 22, 1995, formed the bases for a similar interval extension for these same SRs that was approved by the staff in Amendment No. 106 dated March 1, 1996. This information also supports the licensee's conclusion.

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 SR 4.7.2.1.c items 1, 2, and 3, and SR 4.7.2.1.e items 1, 2, and 4 are acceptable.

3.8 Snubbers, SR 4.7.5.e

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.

SR 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 15 days for the requirement of SR 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 16.6 months operation for the current operating cycle. This would be comparable to, if not shorter than, the operating time during a normal operating cycle. Therefore, the licensee has concluded that the requested extension will have no impact on safety. Additional historical information that was provided in a submittal dated September 20, 1995, as supplemented December 18 and December 22, 1995, formed the bases for a similar interval extension for this same SR that was approved by the staff in Amendment No. 106 dated March 1, 1996. This information also supports the licensee's conclusion.

The staff has reviewed the information provided by the licensee and concludes that the expected snubber performance for the current operating cycle would be comparable to that which would be seen during the maximum currently allowed TS surveillance interval. Therefore, the proposed surveillance interval extension is acceptable.

3.9 Electrical Components Surveillance

3.9.1 Emergency Diesel Generators (EDGs) SRs

3.9.1.1 SR 4.8.1.1.2.e.1

SR 4.8.1.1.2.e.1 requires each of the EDGs be demonstrated operable at least once per 18 months by subjecting the EDG to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service. The licensee has requested a one-time surveillance interval extension of 61 days for the requirement of SR 4.8.1.1.2.e.1.

The EDGs are subjected to operational testing every 31 days and fast-start testing every 184 days. This testing demonstrates the ability of the EDGs to start and run under various loading conditions during the operating cycle. Since these tests have indicated no diesel generator degradation in the past cycles, the licensee is confident of the EDGs' operability and capability to perform their intended function. Although the licensee has replaced several components during SR 4.8.1.1.2.e.1 inspections in the past, the licensee indicated that such replacements were done for preventive measures, not because of degradation.

Based on the satisfactory results of the periodic EDG testing performed during the operating cycle and past diesel SR inspections, the licensee has concluded 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. The licensee has also indicated that the SR may be performed on-line before the outage for one or more of the EDGs using the extended EDG allowed outage time approved by the NRC in Amendment No. 119 dated June 2, 1998. The requested extension will not be necessary for any of the EDGs for which the SR is completed before the outage.

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.

3.9.1.2 SRs 4.8.1.1.2.e.2, 3, 4.a, 4.b, 5, 6.a, 6.b, 7, 10, 11, 12.a, 12.b, and 12.c

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 set point or 115 percent of nominal speed, whichever is lower. The licensee has requested a one-time surveillance interval extension of 53 days for the requirement of SR 4.8.1.1.2.e.2.

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. The licensee has requested a one-time surveillance interval extension of 53 days for the requirement of SR 4.8.1.1.2.e.3.

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. The licensee has requested a one-time surveillance interval extension of 50 days for the requirement of SR 4.8.1.1.2.e.4.a.

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. The licensee has requested a one-time surveillance interval extension of 20 days for the requirement of SR 4.8.1.1.2.e.4.b.

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. The licensee has requested a one-time surveillance interval extension of 54 days for the requirement of SR 4.8.1.1.2.e.5.

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 deenergization of the emergency busses and load shedding from the emergency busses. The licensee has requested a one-time surveillance interval extension of 50 days for the requirement of SR 4.8.1.1.2.e.6.a.

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. The licensee has requested a one-time surveillance interval extension of 54 days for the requirement of SR 4.8.1.1.2.e.6.b.

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. The licensee has requested a one-time surveillance interval extension of 53 days for the requirement of SR 4.8.1.1.2.e.7.

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. The licensee has requested a one-time surveillance interval extension of 5 days for the requirement of SR 4.8.1.1.2.e.10.

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. The licensee has requested a one-time surveillance interval extension of 23 days for the requirement of SR 4.8.1.1.2.e.11.

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 engineered safety features (ESF) bus lockout
b) Differential trip
c) Shutdown relay trip

The licensee has requested a one-time surveillance interval extension of 53 days for the requirements of SR 4.8.1.1.2.e.12.

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 reactor water level low - Level 2, drywell pressure high, 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 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. In addition, certain conditions result in a lockout of the EDG as a protective measure. 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 indicated that the same reasoning applied as for other logic system function tests. See Section 3.4 of this safety evaluation. The licensee has concluded that the diesel control logic components are less likely to fail to function than diesel mechanical components. The licensee also indicated that the historical test data for the EDGs has proven their ability to start and operate under various load conditions. 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. In addition, SRs 4.8.1.1.2.e.2 and 3 may be performed on-line before the outage for one or more of the EDGs using the extended EDG allowed outage time approved by the NRC in Amendment No. 119 dated June 2, 1998. The requested extension will not be necessary for any of the EDGs for which these SRs are completed before the outage.

The staff has reviewed the information supplied by the licensee and concludes that the above proposed one-time extension of SRs 4.8.1.1.2.e.2, 3, 4.a, 4.b, 5, 6.a, 6.b, 7, 10, 11, and 12 is acceptable.

3.9.1.3 <u>SR 4.8.1.1.2.e.8</u>

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. The licensee has requested a one-time surveillance interval extension of 56 days for the requirement of SR 4.8.1.1.2.e.8.

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 generators' 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. The licensee has also indicated that the SR may be performed on-line before the outage for one or more of the EDGs using the extended EDG allowed outage time approved by the NRC in Amendment No. 119 dated June 2, 1998. The requested extension will not be necessary for any of the EDGs for which the SR is completed before the outage.

Based on the above, the staff concludes that the one-time extension is not expected to have any 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.

3.9.1.4 <u>SR 4.8.1.1.2.e.9</u>

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. The licensee has requested a one-time surveillance interval extension of 5 days for the requirement of SR 4.8.1.1.2.e.9.

The auto-connected loads for the diesel generators at Fermi 2 have not substantially changed since the fifth 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 the fifth refueling outage combined with any additional changes since that 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.

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.

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

3.9.2.1 <u>SR 4.8.2.1.c.3</u>

SR 4.8.2.1.c.3 requires that each of the required 130-volt batteries and chargers be demonstrated operable at least once per 18 months by verifying that the resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohm. The licensee has requested a one-time surveillance interval extension of 55 days for the requirement of SR 4.8.2.1.c.3.

The licensee reviewed the records for the battery terminal connection resistance SR for the third and fourth refueling outages and found no significant corrosion that would affect the terminal resistance. The licensee also reviewed the failure data for this SR for the fifth refueling outage and determined there were no failures with time-based elements that could affect the one-time extension of the surveillance interval. The licensee concluded that the extension would have a small impact on safety function availability.

Based on the information provided by the licensee, the staff concludes that the one-time extension of SR 4.8.2.1.c.3 is acceptable.

3.9.2.2 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.

The licensee has requested a one-time surveillance interval extension of 55 days for the requirement of SR 4.8.2.1.d.

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. The capacity factors were greater than 100 percent for all batteries tested. 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). Based on the results of the past battery capacity and service, the licensee finds that the extension of the battery service test can be justified. The staff agrees that the proposed one-time extension of SR 4.8.2.1.d is acceptable.

3.9.3 Primary Containment Penetration Conductor Overcurrent Protective Devices

As stated in TS Bases Section 3/4.8.4, primary containment electrical penetrations and penetration conductors are protected by either de-energizing circuits not required during reactor operation or demonstrating the operability of primary and backup overcurrent protection circuit breakers by periodic surveillance.

3.9.3.1 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 (reactor recirculation pumps) be demonstrated operable at least once per 18 months by performing a channel calibration of the associated 4.16-kV circuit protective relays. The licensee has requested a one-time surveillance interval extension of 20 days for the requirement of SR 4.8.4.2.a.1.a.

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 General Electric Company 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 the fifth refueling outage. The review identified one failure that was associated with excessive instrument drift and was evaluated as a part of the instrument drift evaluation. Based on the previous discussion, the historical failure review and evaluation, and the small impact on safety function availability, the licensee concluded that the requested extension is justified.

Based on the information provided by the licensee, the staff concludes that the requested extension of SR 4.8.4.2.a.1.a is acceptable.

3.9.3.2 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 (reactor recirculation pumps) 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. The licensee has requested a one-time surveillance interval extension of 20 days for the requirement of SR 4.8.4.2.a.1.b.

The licensee performed a historical search of the 18-month surveillance tests for SR 4.8.4.2.a.1.b for the fifth refueling outage 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.

Additional historical information that was provided in a submittal dated September 20, 1995, as supplemented December 18 and December 22, 1995, formed the bases for a similar interval extension for this same SR that was approved by the staff in Amendment No. 106 dated March 1, 1996. This information also supports the licensee's conclusion.

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

3.9.3.3 SLCS Isolation Devices, SR 4.8.4.5.a

SR 4.8.4.5.a requires that the SLCS isolation devices (circuit breakers) be demonstrated operable by performance of a channel calibration of the associated protective relays and a channel functional test of each breaker at least once per 18 months (with a maximum allowable extension of 4.5 months per TS 4.0.2). The channel functional test includes simulated actuation of the system and verification that each relay and associated circuit breaker and overcurrent circuit functions as designed. The licensee has requested a one-time surveillance interval extension of 51 days for the requirement of SR 4.8.4.5.a.

The licensee performed a historical search of the 18-month surveillance tests for the calibration and functional testing of the SLCS protective relays and circuit breakers to identify and evaluate any failed or partially failed tests. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a component would be unavailable. Based on this review, the licensee concluded that the increased interval would have no effect on the availability of the safety function. In addition, the licensee reviewed the channel functional test history of the protective devices and concluded that no failures have been previously experienced, and the function will not be affected by the short increase in the surveillance interval. The licensee concluded that the proposed extension is justified.

Based on the information provided by the licensee, the staff finds that the one-time extension of SR 4.8.4.5.a is acceptable.

3.10 Summary Conclusion

For the reasons stated above, the staff finds that the requested one-time extensions for surveillances of the components and systems described in this safety evaluation 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 (63 *FR* 36273). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

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