Douglas R. Gipson
Senior Vice President
Nuclear Generation

Detroit Edison

Fermi 2
6400 North Dixie Highway
Newport, Michigan 48166
(313) 586-5249

September 20, 1995
NRC-95-0096

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

References: 1) Fermi 2
Docket No. 50-341
NRC License No. NPF-43

2) Detroit Edison Letter NRC-95-0083, "Request For One-Time Exemption from 10 CFR Part 50, Appendix J, Paragraphs III.D.2.a and III.D.3 Schedule Requirements," NRC-95-0083, dated September 1, 1995

Subject: One-Time Technical Specification Revision to Allow Extension of the Fermi 2 Operating Cycle

The Detroit Edison Company (Detroit Edison) hereby files an application to amend the Fermi 2 Technical Specifications. This application is filed to revise applicable Technical Specifications related to system testing, instrumentation calibration, component inspection, component testing, response time testing, and Logic System Functional Tests to allow a one-time extension of the surveillance intervals.

The proposed extensions are requested on a one-time only basis to support our current refueling outage schedule. Due to the lengthy turbine outage, startup and power ascension program following the December 25, 1993 turbine failure, Detroit Edison is postponing the spring 1996 refueling outage until September 27, 1996. This will allow targeted fuel burnup to be met, so Cycle 6 operation can be conducted as planned. Approval of the requested surveillance interval extensions will prevent a plant shutdown solely to perform surveillance tests. A shutdown to perform surveillance tests would be an unnecessary transient on the plant, provide unnecessary challenges to plant operators and would also result in additional

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radiation exposure to personnel, since many of the surveillances would need to be repeated during the refueling outage.

Also planned for the spring 1996 outage was replacement of the low pressure turbine rotors. The manufacturing schedule has slipped and the rotors will not be ready to support a March 1996 outage. The turbine modifications have now been postponed until the September 1996 refueling outage, which will provide further opportunity to complete the preparations for the turbine work.

The Fermi 2 low pressure turbines have pressure plates installed in place of the seventh and eighth stage blades. The NRC Project Manager, Mr. Tim Colburn, asked whether the vibration of the current turbine is such that it would preclude a six month addition to the operating cycle. Even with the current turbine configuration the vibration limits are established consistent with the manufacturer's recommendations and within the insurer's guidelines for continued operation of the turbine. Therefore, extending the cycle should not be precluded based on turbine vibration.

Similar surveillance extension requests to support postponement of refueling outages have previously been approved for River Bend, Nine Mile Point 2 and D.C. Cook. Also, several plants have received approvals to extend their operating cycle permanently to 24 months which allows a maximum surveillance interval of 30 months including the 25% allowance for scheduling flexibility. The longest extension expected (227 days) is approximately the same as the additional time period allowed by the permanent 24 month cycle.

Attachment 1 provides a table which lists the Technical Specifications for which extensions are being requested and the date required for extension. Where multiple components or divisions of equipment are tested to meet the surveillance requirement, the longest extension needed for any of the components tested is listed, though some of the components may have been tested more recently.

The table provides a cross reference to the Attachment 1 enclosure which justifies the extension. Additionally, Attachment 1 provides descriptions and discussions for Technical Specifications which require extension. Enclosures 1 through 39 to Attachment 1 provide the justification for the proposed extensions to the Technical Specifications. Attachment 2 provides the No Significant Hazards Consideration discussion. The revised Technical Specification pages are provided in Attachment 3.

The first surveillance requirement becomes due on March 29, 1996. However, approval is requested by December 31, 1995. Prompt review of this proposal is very

important because approval of this one- time extension greatly impacts our outage schedule preparations and work priorities as well as Detroit Edison "generating system" planning. Reference 2 proposes a one-time exemption from 10 CFR 50 Appendix J to extend type B and C leakage test intervals. Approval of the Reference 2 request is also needed to support the outage schedule.

This request meets the cost and safety criteria for a Cost Beneficial Licensing Action since it involves greater than $100,000 in savings and has a small effect on safety, as discussed in Attachment 1. However, Detroit Edison believes that this request meets the criteria for a priority 1 ranking since it is an action to prevent a plant shutdown.

Detroit Edison has evaluated the proposed Technical Specification extensions against the criteria of 10CFR50.92 and determined that No Significant Hazards Consideration is involved. The Fermi 2 Onsite Review Organization has approved and the Nuclear Safety Review Group has reviewed the proposed Technical Specification extensions and concurs with the enclosed determinations. In accordance with 10CFR50.91, Detroit Edison has provided a copy of this letter to the State of Michigan.

The following specific commitments are contained in this letter:

- Perform additional Preventative Maintenance items on the Diesel Generators as determined necessary by the manufacturer for extending SR 4.8.1.1.2.e.1. These additional items will be performed for each diesel before its current due date for SR 4.8.1.1.2.e.1.

- Should an outage occur prior to RFO5, Detroit Edison commits to evaluate performing surveillance testing during such an outage. The specific surveillance tests performed would be selected from the listing of surveillance requirements for which an extension is being requested. The selection will depend upon factors such as the outage length, available resources, the cause of the outage, and the plant conditions during the outage.

If you have any questions or comments, please contact Robert Newkirk of my staff at (313) 586-4211.

Sincerely,

Attachments
Enclosures

cc: T. G. Colburn
H. J. Miller
M. P. Phillips
A. Vegel
Supervisor, Electric Operators, Michigan
Public Service Commission - J. R. Padgett

I, DOUGLAS R. GIPSON, do hereby affirm that the foregoing statements are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

DRGipson
DOUGLAS R. GIPSON
Senior Vice President

On this 20th day of September, 1995 before me personally appeared Douglas R. Gipson, being first duly sworn and says that he executed the foregoing as his free act and deed.

Rosalie A. Armetta
Notary Public

ROSALIE A ARMETTA
NOTARY PUBLIC STATE OF MICHIGAN
MONROE COUNTY
MY COMMISSION EXP. NOV. 20,1995

SYSTEM TESTING, INSTRUMENTATION CALIBRATION, LOGIC SYSTEM FUNCTIONAL TESTING, COMPONENT INSPECTION, COMPONENT TESTING, AND RESPONSE TIME TESTING

ATTACHMENT 1
DETROIT EDISON COMPANY
ENRICO FERMI ATOMIC POWER PLANT
DOCKET 50-341/LICENSE NO. NPF-43

SYSTEM TESTING, INSTRUMENTATION CALIBRATION, LOGIC SYSTEM FUNCTIONAL TESTING, COMPONENT INSPECTION, COMPONENT TESTING, AND RESPONSE TIME TESTING

DOCUMENT INVOLVED: Technical Specifications

TECHNICAL SPECIFICATION SR	EXTENDED DATE	MAXIMUM DAYS EXTENSION	JUSTIFICATION LOCATION	DESCRIPTION OF SR REQUIREMENT
4.0.5			Enclosure 26	ISI/IST Program
4.1.3.1.4.a	10/05/96	52	Enclosure 1	Scram discharge vol. vent and drain valve operability
4.1.3.5.b.2	11/16/96	71	Enclosure 2	CR Accumulator Integrity Test (Check Valve Leakage)
4.1.5.d.1	11/16/96	138	Enclosure 3	SLCS operability Manual Initiation
4.1.5.d.2	11/16/96	151	Enclosure 3	SLCS pump Relief Valve operability
4.1.5.d.3	11/16/96	138	Enclosure 3	SLCS flow path demonstration
4.3.1.1, Table 4.3.1.1-1, Item 3	10/05/96	173	Enclosure 4	RPS Rx Steam Dome Press High cal
4.3.1.1, Table 4.3.1.1-1, Item 4	10/05/96	154	Enclosure 4	RPS Rx Low Water Level - Level 3 cal
4.3.1.1, Table 4.3.1.1-1, Item 5	10/05/96	62	Enclosure 4	RPS MSIV Closure cal
4.3.1.1, Table 4.3.1.1-1, Item 6	10/05/96	128	Enclosure 4	RPS Main Steam Line Radiation High cal
4.3.1.1, Table 4.3.1.1-1, Item 7	10/05/96	147	Enclosure 4	RPS Drywell Pressure High cal
4.3.1.1, Table 4.3.1.1-1, Item 11	11/16/96	146	Enclosure 5	RPS Rx Mode Switch shutdown position functional
4.3.1.2	11/16/96	173	Enclosure 5	RPS Logic System Functional Test
4.3.1.3	10/05/96	101	Enclosure 6	RPS Response Time Test
4.3.2.1, Table 4.3.2.1-1, Item 1.a.1	10/05/96	154	Enclosure 7	Pri Cont Isolation Actuation Rx Water Low Level - Level 3 cal
4.3.2.1, Table 4.3.2.1-1, Item 1.a.2	10/05/96	153	Enclosure 7	Pri Cont Isolation Actuation Rx Water Low Level - Level 2 cal
4.3.2.1, Table 4.3.2.1-1, Item 1.a.3	10/05/96	153	Enclosure 7	Pri Cont Isolation Actuation Rx Water Low Level - Level 1 cal
4.3.2.1, Table 4.3.2.1-1, Item 1.b	10/05/96	147	Enclosure 7	Pri Cont Isolation Actuation Drywell Press High cal

4.3.2.1, Table 4.3.2.1-1, Item 1.c.1	10/05/96	128	Enclosure 7	Pri Cont Isolation Actuation Main Steam Line Radiation High cal
4.3.2.1, Table 4.3.2.1-1, Item 1.c.2	10/05/96	149	Enclosure 7	Pri Cont Isolation Actuation Main Steam Line Press Low cal
4.3.2.1, Table 4.3.2.1-1, Item 1.d	10/05/96	183	Enclosure 7	Pri Cont Isolation Actuation Main Steam Line Tunnel Temp. High cal
4.3.2.1, Table 4.3.2.1-1, Item 1.e	10/05/96	182	Enclosure 7	Pri Cont Isolation Actuation Condenser Press High cal
4.3.2.1, Table 4.3.2.1-1, Item 1.f	10/05/96	183	Enclosure 7	Pri Cont Isolation Actuation Turbine Bldg. Area Temp. High cal
4.3.2.1, Table 4.3.2.1-1, Item 1.h	11/16/96	191	Enclosure 8	Pri Cont Isolation Actuation Manual Initiation Functional
4.3.2.1, Table 4.3.2.1-1, Item 2.d	11/16/96	138	Enclosure 8	RWCU - SLCS initiation channel functional test
4.3.2.1, Table 4.3.2.1-1, Item 2.e	10/05/96	153	Enclosure 7	RWCU Isolation Rx Water Low Level - Level 2 channel cal
4.3.2.1, Table 4.3.2.1-1, Item 2.g	10/05/96	131	Enclosure 8	RWCU Manual Initiation channel functional test
4.3.2.1, Table 4.3.2.1-1, Item 3.a.1	10/05/96	129	Enclosure 7	RCIC Steam Line Flow High DP channel cal
4.3.2.1, Table 4.3.2.1-1, Item 3.a.2	10/05/96	129	Enclosure 7	RCIC Steam Line Flow High Time Delay cal
4.3.2.1, Table 4.3.2.1-1, Item 4.a.1	10/05/96	132	Enclosure 7	HPCI Steam Line Flow High DP cal
4.3.2.1, Table 4.3.2.1-1, Item 4.a.2	10/05/96	132	Enclosure 7	HPCI Steam Line Flow High Time Delay cal
4.3.2.1, Table 4.3.2.1-1, Item 4.e	10/05/96	34	Enclosure 8	HPCI Manual Initiation functional test
4.3.2.1, Table 4.3.2.1-1, Item 5.a	10/05/96	154	Enclosure 7	RHR S/D Cooling Rx Water Level Low - Level 3 cal
4.3.2.1, Table 4.3.2.1-1, Item 5.c	11/16/96	170	Enclosure 8	RHR S/D Cooling Rx manual initiation functional test
4.3.2.1, Table 4.3.2.1-1, Item 6.a	11/16/96	195	Enclosure 7	Sec. Cont. Isolation - Rx Water Low Level - Level 2 cal
4.3.2.1, Table 4.3.2.1-1, Item 6.b	10/05/96	147	Enclosure 7	Sec. Cont. Isolation - Drywell Press High channel cal
4.3.2.2	11/16/96	183	Enclosure 8	Isolation Actuation Inst. LSFT
4.3.2.3	10/05/96	101	Enclosure 9	Isolation Actuation Inst. System Response Time
4.3.3.1, Table 4.3.3.1-1, Item 1.a	11/16/96	184	Enclosure 10	CS RPV Low Level 1 Cal
4.3.3.1, Table 4.3.3.1-1, Item 1.b	10/05/96	169	Enclosure 10	CS Drywell Press High Cal
4.3.3.1, Table 4.3.3.1-1, Item 1.c	11/16/96	223	Enclosure 10	CS Rx Steam Dome Press Low Cal
4.3.3.1, Table 4.3.3.1-1, Item 1.d	11/16/96	191	Enclosure 11	CS Manual Initiation
4.3.3.1, Table 4.3.3.1-1, Item 2.a	11/16/96	184	Enclosure 10	LPCI RPV Low Level 1 Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.b	10/05/96	169	Enclosure 10	LPCI Drywell Press High Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.c	11/16/96	223	Enclosure 10	LPCI Rx Steam Dome Press Low Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.d	11/16/96	184	Enclosure 10	LPCI Rx Low Level 2 Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.e	11/16/96	223	Enclosure 10	LPCI Rx Steam Dome Press Low Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.f	10/05/96	184	Enclosure 10	LPCI Riser Differential Pressure High Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.g	10/05/96	185	Enclosure 10	LPCI Recirc. Pump Differential Pressure High Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.h	11/16/96	185	Enclosure 11	LPCI Manual Initiation
4.3.3.1, Table 4.3.3.1-1, Item 3.a	11/16/96	184	Enclosure 10	HPCI RPV Low Level 2 Cal
4.3.3.1, Table 4.3.3.1-1, Item 3.b	10/05/96	169	Enclosure 10	HPCI Drywell Press High Cal
4.3.3.1, Table 4.3.3.1-1, Item 3.e	11/16/96	184	Enclosure 10	HPCI RPV High Level 8 Cal

4.3.3.1, Table 4.3.3.1-1, Item 3.f	10/05/96	34	Enclosure 11	HPCI Manual Initiation
4.3.3.1, Table 4.3.3.1-1, Item 4.a	11/16/96	184	Enclosure 10	ADS RPV Low Level 1 Cal
4.3.3.1, Table 4.3.3.1-1, Item 4.f	10/05/96	181	Enclosure 10	ADS RPV Low Level 3 Cal
4.3.3.1, Table 4.3.3.1-1, Item 4.h	11/16/96	184	Enclosure 10	ADS Drywell Pressure High Bypass Timer
4.3.3.1, Table 4.3.3.1-1, Item 4.i	10/05/96	72	Enclosure 11	ADS Manual Inhibit Functional Test
4.3.3.2	11/16/96	227	Enclosure 11	ECCS Logic System Functional Tests
4.3.3.3	11/16/96	191	Enclosure 12	ECCS Response Time Tests
4.3.4, Table 4.3.4-1, Item 1	11/16/96	184	Enclosure 13	RPV Low Water Level 2 Cal (ATWS)
4.3.4, Table 4.3.4-1, Item 2	10/05/96	169	Enclosure 13	RPV Press High Cal (ATWS)
4.3.4.2	11/16/96	184	Enclosure 14	ATWS Logic System Functional Test
4.3.5.1, Table 4.3.5.1-1, Item a	11/16/96	184	Enclosure 15	RPV Low Level 2 Cal (RCIC)
4.3.5.1, Table 4.3.5.1-1, Item b	11/16/96	184	Enclosure 15	RPV High Level 8 Cal (RCIC)
4.3.5.2	11/16/96	184	Enclosure 16	RCIC Logic System Functional Test
4.3.6, Table 4.3.6-1, Item 5.b	11/16/96	103	Enclosure 17	Scram Disc. Vol. Trip Bypass Funct. Test
4.3.6, Table 4.3.6-1, Item 7	11/16/96	146	Enclosure 17	Rx Mode Switch Shutdown Pos. Rod Block Funct. Test
4.3.7.4.1, Table 4.3.7.4.-1, Item 1	10/05/96	168	Enclosure 18	RPV Press Cal - Remote Shutdown
4.3.7.4.1, Table 4.3.7.4.-1, Item 2	11/16/96	184	Enclosure 18	RPV Level Cal - Remote Shutdown
4.3.7.5, Table 4.3.7.5-1, Item 1	10/05/96	164	Enclosure 19	RPV Press Cal Accident Mon.
4.3.7.5, Table 4.3.7.5-1, Item 2.a	10/05/96	182	Enclosure 19	RPV Fuel Zone Level Cal Accident Mon
4.3.7.5, Table 4.3.7.5-1, Item 2.b	11/16/96	184	Enclosure 19	RPV Wide Range Level Cal Accident Mon
4.3.7.5, Table 4.3.7.5-1, Item 11	10/05/96	74	Enclosure 19	SRV Position Indic Cal Accident Mon.
4.3.7.5, Table 4.3.7.5-1, Item 12	10/05/96	121	Enclosure 19	CTMT High Range Rad Monitoring Cal Accident Mon.
4.3.7.5, Table 4.3.7.5-1, Item 16	11/16/96	191	Enclosure 19	CTMT Isolation Valve Position Cal Accident Mon
4.3.7.10.c	10/05/96	16	Enclosure 20	Loose Part Detection System Cal
4.3.9.1, Table 4.3.9.1-1, Item a	10/05/96	181	Enclosure 21	RPV High Water Level 8 Cal FW/Main Turbine Trip
4.3.9.2	10/05/96	181	Enclosure 22	FW/Main Turbine Trip LSFT
4.3.11.1, Table 4.3.11.1-1, Item 7	10/05/96	168	Enclosure 23	Alt S/D system Rx Water Level instrument operability
4.3.11.1, Table 4.3.11.1-1, Item 8	10/05/96	168	Enclosure 23	Alt S/D system Rx Press instrument operability
4.4.2.1.1	10/05/96	74	Enclosure 24	SRV Tail Pipe Pressure Switch Cal
4.4.2.1.2	10/05/96	115	Enclosure 24	SRV lift set point test
4.4.2.2.b	10/05/96	169	Enclosure 24	SRV Low Low Set Pressure setpoint Cal and LSFT
4.4.3.1.b	10/05/96	112	Enclosure 25	Drywell Sump Flow/Lvl Monitoring Cal
4.4.3.2.2.a	10/05/96	181	Enclosure 25	RCS Pressure Isol Valve Leak Test
4.4.8			Enclosure 26	ISI/IST
4.5.1.c.1	11/16/96	191	Enclosure 27	ECCS System Functional Test

4.5.1.d.2.a	10/05/96	72	Enclosure 27	ADS System Functional Test
4.5.2.1			Attachment 1	ECCS Shutdown Operability
4.6.1.2.b	10/05/96	157	Reference 2	Type B and C LLRT's
4.6.1.2.d	10/05/96	170	Reference 2	MSIV Leak Test
4.6.1.2.g	10/05/96	183	Reference 2	Hydrostatic Leak Test ECCS/RCIC Cont Isol Valves
4.6.1.4.d.3	10/05/96	175	Enclosure 28	MSIV LCS Press Inst. Cal and DP Calibration
4.6.2.1.e	10/05/96	22	Enclosure 29	Suppression Chamber operability (visual inspection)
4.6.2.1.h	10/05/96	3	Enclosure 29	Suppression Chamber operability DW to torus bypass leak test
4.6.3.2	11/16/96	189	Enclosure 30	Primary Containment Isol Valve operability
4.6.3.4	10/05/96	16	Enclosure 30	Instr. Excess Flow Check operability
4.6.3.5.b	10/05/96	92	Enclosure 30	TIP Explosive Squib operability test
4.6.4.1.b.2.a	10/05/96	63	Enclosure 31	Torus/Drywell vacuum breaker setpoint operability
4.6.4.1.b.2.b	10/05/96	63	Enclosure 31	Torus/Drywell vacuum breaker position indication cal
4.6.4.1.b.2.c	10/05/96	63	Enclosure 31	Torus/Drywell vacuum breaker switch opening gap
4.6.4.2.b.2.a	10/05/96	60	Enclosure 31	RB/Torus Vacuum Breaker operability (setpoint)
4.6.4.2.b.2.b	10/05/96	60	Enclosure 31	RB/Torus Vacuum Breaker operability (visual)
4.6.4.2.b.2.c	10/05/96	60	Enclosure 31	RB/Torus Vacuum Breaker position indication operability
4.6.5.2.b	11/16/96	27	Enclosure 32	Secondary Containment Isolation Damper Actuation
4.7.1.2.b	11/16/96	91	Enclosure 33	ECCW Automatic Actuation
4.7.1.3.b	11/16/96	91	Enclosure 33	EESW Automatic Actuation
4.7.1.4.b	11/16/96	215	Enclosure 33	EDG Cooling Water Pump Automatic Actuation
4.7.2.1.c.1	11/16/96	146	Enclosure 34	CR Ventilation Filter Penetration
4.7.2.1.c.2	11/16/96	146	Enclosure 34	CR Ventilation Filter Charcoal Laboratory Analysis
4.7.2.1.c.3	11/16/96	146	Enclosure 34	CR Emergency Filtration System Flowrate
4.7.2.1.e.1	11/16/96	146	Enclosure 34	CR Ventilation Filter Pressure Drop
4.7.2.1.e.2	11/16/96	27	Enclosure 34	CR Emergency Filtration System Operational Mode Actuation
4.7.2.1.e.4	11/16/96	146	Enclosure 34	CR Emergency Makeup Inlet Heater Dissipation
4.7.2.1.h	06/01/98	48	Enclosure 34	CR Emergency Ventilation Duct Leakage
4.7.5.b			Attachment 1	Snubber Visual Inspection
4.7.5.e	11/16/96	105	Enclosure 35	Snubber Functional Test
4.7.11.4	10/05/96	82	Enclosure 36	Alternative Shutdown Control Circuit Functional Test
4.8.1.1.2.e.1	11/16/96	216	Enclosure 37	EDG Inspection
4.8.1.1.2.e.2	11/16/96	215	Enclosure 37	EDG Load Rejection (1666 kW)
4.8.1.1.2.e.3	11/16/96	215	Enclosure 37	EDG Load Rejection (2850 kW)
4.8.1.1.2.e.4.a	11/16/96	141	Enclosure 37	EDG LOP Load Shedding

4.8.1.1.2.e.4.b	11/16/96	141	Enclosure 37	EDG LOP Auto Start and Load Sequencing
4.8.1.1.2.e.5	11/16/96	215	Enclosure 37	EDG ECCS Auto Start
4.8.1.1.2.e.6.a	11/16/96	141	Enclosure 37	EDG LOP / ECCS Load Shedding
4.8.1.1.2.e.6.b	11/16/96	141	Enclosure 37	EDG LOP / ECCS Auto Start and Load Sequencing
4.8.1.1.2.e.7	11/16/96	215	Enclosure 37	EDG Non-essential Trip Bypass
4.8.1.1.2.e.8	11/16/96	131	Enclosure 37	EDG 24 Hour Run and Hot Fast Start
4.8.1.1.2.e.9	11/16/96	94	Enclosure 37	EDG Auto Connect Load Verification
4.8.1.1.2.e.10	11/16/96	94	Enclosure 37	EDG Restoration of Offsite Power
4.8.1.1.2.e.11	11/16/96	123	Enclosure 37	EDG Auto Load Sequencer Timer
4.8.1.1.2.e.12.a	11/16/96	215	Enclosure 37	EDG 4160-volt ESF Bus Lockout
4.8.1.1.2.e.12.b	11/16/96	215	Enclosure 37	EDG Differential Trip Lockout
4.8.1.1.2.e.12.c	11/16/96	215	Enclosure 37	EDG Shutdown Relay Trip Lockout
4.8.1.2			Attachment 1	Electrical Power System Shutdown Operability
4.8.2.1.c.3	11/16/96	190	Enclosure 38	130 VDC Battery Connections Resistance
4.8.2.1.c.4	11/16/96	190	Enclosure 38	130 VDC Battery Charger Functional Test
4.8.2.1.d	11/16/96	190	Enclosure 38	130 VDC Battery Capacity
4.8.2.2			Attachment 1	Battery Shutdown Operability
4.8.4.2.a.1.a	10/05/96	103	Enclosure 39	Primary Containment 4160 Volt Penetration Protective Relay Cal
4.8.4.2.a.1.b	10/05/96	103	Enclosure 39	Primary Containment 4160 Volt Penetration Protective Device Integrated Functional Test

REASON FOR REQUEST:

BACKGROUND

The amendment is being requested to permit an approximately 6 month extension in the Fermi 2 Operating Cycle from the previously scheduled end date of March 15, 1996 to an end date of September 27, 1996. An extended startup and power ascension program, caused by the December 25, 1993 turbine event, has extended the operating cycle length.

On December 19, 1994, the reactor was taken critical at the end of the fourth refueling outage (RFO4) which began December 25, 1993. RFO4 ended on January 18, 1995 when the main turbine-generator was synchronized to the grid. A slow power ascension program was implemented, with 95% reactor power not achieved until May 31, 1995. The extensive length of RFO4, the prolonged startup process, and forced outages during Operating Cycle 5 have impacted the 18-month Operating Cycle length. As a result Detroit Edison is postponing the spring, 1996 refueling outage until September 27, 1996. This will allow targeted fuel burnup to be met, so Cycle 6 operation can be conducted as planned. Approval of the requested surveillance interval extensions will prevent a plant shutdown solely to perform surveillance tests.

Also planned for the spring 1996 outage was replacement of the low pressure turbine rotors. The manufacturing schedule has slipped and the rotors will not be ready to support a March 1996 outage. The turbine modification has been postponed until the September 1996 refueling outage, which will provide further opportunity to complete the preparations for the turbine work.

The extended Operating Cycle length will result in surveillance tests, performed during the last refueling outage, exceeding the surveillance interval plus the allowable extension to the interval specified in TS 4.0.2. Detroit Edison is, therefore, requesting an extension in the surveillance intervals so a mid-cycle shutdown to perform surveillance testing can be prevented. A surveillance testing outage would cause an unnecessary transient on the plant, provide unnecessary challeges to plant operators and result in additional radiation exposure to personnel. The proposed one-time Technical Specification changes extend the testing interval for surveillance requirements.

In order to minimize the changes to the individual pages of the Technical Specifications, a new subsection is proposed for Section 4.0.2 which would reference two tables showing the affected specifications and new late completion date for these surveillances. Additionally, a specific extension for Control Room Emergency Filtration System duct leakage testing, Surveillance Requirement (SR) 4.7.2.1.h, is requested. The associated page for this SR is marked up accordingly. A note is also proposed for addition to SR

4.6.1.2.i to allow SR 4.0.2 to apply for extending Appendix J requirements for RFO5 only. The authority to delay such tests is being sought on a one-time only basis. The Fermi 2 Technical Specifications (TS) require the performance of Operating Cycle surveillance requirements (SRs), including: instrumentation calibration, response time testing and Logic System Functional Tests (LSFT). The TS to which these SRs apply include the Reactor Protection System Instrumentation, Isolation Actuation Instrumentation, Emergency Core Cooling System (ECCS) Actuation Instrumentation, Control Rod Block Instrumentation, Remote Shutdown Monitoring Instrumentation, Accident Monitoring Instrumentation, Plant Systems Actuation Instrumentation, Appendix R instrumentation and Reactor Protection System Electric Power Monitoring channels. The SRs specify that the required calibration, response time testing and/or LSFT be conducted nominally at refueling intervals but at least once every 18 months. TS 4.0.2 allows a 25% extension of the surveillance interval to 22.5 months, if required, to provide some flexibility in cycle lengths.

In addition, system testing involving valve operability or performance testing, including Primary Containment Isolation Valve actuation and leak testing and AC and DC system surveillances, are required to be performed normally at refueling intervals but at least once every 18 months. With the exception of Primary Containment Isolation Valves tested under the requirements of 10 CFR 50 Appendix J, TS 4.0.2 allows a 25% extension of the surveillance interval to 22.5 months, if required, to provide some flexibility in cycle lengths.

Specific TS sections require visual inspections or disassembly and inspection of components including Emergency Diesel Generators, accessible portions of the Torus, and power supply breakers. These inspections are required to be performed normally at refueling intervals but at least once every 18 months. TS 4.0.2 also allows a 25% flexibility in cycle lengths.

A one-time change is being requested to extend the surveillance intervals for the above cited TS SRs. Most of these SRs are currently due prior to September 27, 1996 and should not be conducted during power operations because of technical limitations, impact on plant safety, or an unacceptable risk of an unplanned plant transient. To require the plant to shut down solely to perform surveillance tests would cause an unnecessary thermal transient on the plant and result in additional radiation exposure to personnel, since many of the surveillances would need to be repeated during the refueling outage. Detroit Edison proposes to amend the cited TS contained in Appendix A to the Fermi Operating License, as discussed below and in the respective enclosures, to perform the subject surveillance tests during the fifth refueling outage (RFO5), presently scheduled to begin September 27, 1996. The extension is needed until at least October 5, 1996, in order to allow time for cooldown to Mode 4 when a majority of the Limiting Conditions for Operation for the respective SRs are not applicable. Though some such equipment is

only required to be operable in Mode 1 or Modes 1 and 2, rather than in Modes 1, 2, and 3, all surveillances for equipment not required to be operable during the outage are grouped together in a table extending the surveillance test intervals to October 5, 1996. This grouping reduces the complexity of the proposed change and eliminates the need to specify in the Technical Specification precisely on which date the mode changes will occur.

An additional extension is being requested for SRs wherein the applicability extends to Modes 4, 5 or other shutdown situations. These SRs associated with the TS are being extended into the refueling outage, scheduled to end November 16, 1996, to support the current schedule and system windows already established, to provide 'defense in depth' during the shutdown period and to provide flexibility in outage scheduling. Therefore, to maintain outage schedule flexibility and to ensure that an unnecessary outage extension does not result, the proposed surveillance interval extensions are needed. The proposed revisions to the TS are worded such that the surveillance tests may be extended to the completion of the fifth refueling outage, which is scheduled to begin September 27, 1996.

Also some specific line items that are not required to be operational during mode 4 or 5 have a requested extension date of the end of the outage (as an example Reactor Low Water Level-Level 2 for HPCI). 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 Modes 4 or 5. To prevent possible confusion associated with multiple extension dates for a single Surveillance Procedure, a complex revision to the surveillance tracking system, or the possible consequences of missed surveillances, the same extension period is being requested for several Technical Specification line items associated with a single surveillance procedure.

Additional discussions are provided as informational items (e.g., SR 4.0.5) where Detroit Edison does not believe that an extension is required. These items provide a discussion of the SR requirements, information stating the Detroit Edison position, and why performance of the activity in the fall of 1996 is acceptable.

It is also requested that the "N times 18 months" cumulative surveillance interval for various response time testing be baselined to this outage; i.e., the beginning of the "N times 18 months" interval be restarted at the respective response time testing dates to be performed during RFO5. 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 RFO5. The same justification provided in this amendment request for the individual response time surveillance interval extensions applies to the "N times 18 months" cumulative surveillance interval extensions because the cumulative surveillance interval would not be extended by more than that being requested for individual response time tests.

Extensions are being requested for surveillances which are due prior to the start of the scheduled outage for equipment only needed during plant operations. Extensions are being requested for surveillances due prior to the end of the outage for equipment needed during the outage.

Should the proposed changes not be granted by March 29, 1996 when the first 18 month SR becomes due, Fermi will be forced to implement a several week surveillance outage during this operating cycle.

DISCUSSION:

The following sections provide a discussion of each of the proposed TS changes cited above. Enclosures to this Attachment describe the justification for each proposed extension. Evaluations of instrument drift/calibration data and surveillance testing failure experience are included in these justifications where applicable.

The Surveillance Test history data base was reviewed to identify any tests during the last two refueling outages which were coded as either equipment failure or "partially complete". The partially complete category includes not only component failures causing an interruption of the test, but it would also include procedure problems, plant conditions that might have precluded further testing, etc. Based on the list of "failures" identified by this search, the surveillance test records were retrieved and each was evaluated. 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, further evaluations were conducted 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), PM activities, IST or other monitoring activities; (2) whether the failure was caused by a special event, occurrence or maintenance activity which 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.

ASME Code Class Testing

As stated in TS Basis Section 3/4.0, Section 4.0.5 establishes the requirement that 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 a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. These requirements apply except when relief has been provided in writing by the Commission.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout the TS and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

TS Section 4.0.5, requires that 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:

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 TS 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 SR.

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

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.

Where specific TS extensions are required for ASME components (i.e., excess flow check valves) these extensions are discussed and justified with the TS sections. Although not specifically identified in Fermi 2 Technical Specifications, the Code 2 year requirement is being treated as a 731 day surveillance frequency and is subject to a 25% extension. This has already been determined to be acceptable per NUREG-1482, Section 6.2. The surveillances associated with Technical Specification 4.0.5 are associated with the Pump and Valve testing, and in addition to Technical Specification 4.4.8, address piping, component support, system leakage, and weld examination. The ASME requirements for these surveillances specify a refueling or 24 month evaluation frequency. Based on these requirements a specific exemption is not required for Technical Specification 4.0.5. However, since the normal Fermi 2 operating cycle is 18 months and the inspections have been performed on a 18 month cycle, Enclosure 26 provides discussions which justify a minimal impact on plant safety for extending the operating cycle for these surveillances to the fall outage.

Scram Discharge Volume

As stated in UFSAR Section 4.5.2.2.3 and Section 7.2.1.1.5.4, water displaced by the Control Rod Drive pistons during a scram goes to the Scram Discharge Volume (SDV). During normal plant operation, the SDV is empty and vented to atmosphere through its open vent and drain valves. When a scram occurs, upon a signal from the reactor protection system, these vent and drain valves are closed to isolate the SDV which controls and terminates the release of potentially contaminated reactor water from the scram exhaust. Lights in the main control room indicate the position of these valves. Redundant vent and drain valves are provided to ensure against loss of reactor coolant from the SDV following a scram. Technical Specification 4.1.3.1.4.a requires demonstrating that the scram discharge vent and drain are operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by verifying that the drain and vent valves close within 30 seconds after receipt of a signal for control rods to scram and open when the scram is reset. This item becomes due on 8/14/96 and requires an extension of 52 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 1.

Scram Accumulator

As stated in UFSAR Section 4.5.2.2.2.3 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.

Technical Specification 4.1.3.5.b.2 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 item becomes due on 9/06/96 and requires an extension of 71 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 2.

Standby Liquid Control System

As stated in UFSAR Section 4.5.2.4.2 the Standby Liquid Control System (SLCS), which is manually initiated from the main control room, injects boron neutron absorber solution into the reactor if the operator believes the reactor cannot be shut down or kept shutdown with control rods. The SLCS function is a backup to control rods and is to maintain the reactor shutdown under all conditions without control rods. The boron solution tank, the test tank, the two positive-displacement pumps, the two explosive valves, and associated local valves and controls are mounted in the reactor building. Borated liquid is piped into the Reactor Pressure Vessel (RPV) and discharged near the bottom of the core shroud so it mixes with the cooling water rising through the core. The boron absorbs thermal neutrons and thereby terminates the nuclear fission chain reaction in the fuel.

Technical Specification 4.1.5.d items 1 through 3 state the standby liquid control system shall be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by:

1. Initiating one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one charge of that batch successfully fired. Both injection loops shall be tested in

36 months. This item becomes due on 7/01/96 and requires an extension of 138 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 3.

2. Demonstrating that the pump relief valve setpoint is less than or equal to 1400 psig and verifying that the relief valve does not actuate during recirculation to the test tank. This item becomes due on 6/18/96 and requires an extension of 151 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 3.

3. 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. This item becomes due on 7/01/96 and requires an extension of 138 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 3.

Reactor Protection System

The RPS instrumentation and control initiates an automatic reactor shutdown (scram) if monitored system variables exceed pre-established limits. This action prevents fuel damage and limits system pressure. Technical Specification 4.3.1.1 requires the each reactor protection system instrumentation channel be demonstrated operable by the performance of a Channel Functional test or a Channel Calibration for the operational conditions at the frequencies shown in Table 4.3.1.1-1. The following discussions are provided for the RPS signals of concern for this one-time Technical Specification extension.

Item 3 Reactor Pressure

As stated in UFSAR Section 7.2.1.1.5.4, a nuclear system pressure increase during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes increased core heat generation that could lead to fuel failure and system overpressurization. A scram counteracts a pressure increase by quickly reducing core fission heat generation. The nuclear system high-pressure scram setting is chosen slightly above the RPV maximum normal operating pressure to permit normal operation without spurious scram, yet provides a wide margin to the maximum allowable nuclear system pressure.

Item 3 for table 4.3.1.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/15/96 and requires an extension of 173 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 4.

Item 4 Reactor Vessel Low Water Level - Level 3

As stated in UFSAR Section 7.2.1.1.5.4, low water level in the RPV indicates that the fuel is in danger of being inadequately cooled. Decreasing the water level while the reactor is operating at power decreases the reactor coolant inlet subcooling. The effect is the same as raising feedwater temperature. Should Reactor Pressure Vessel water level decrease too far, fuel damage could result. A reactor scram protects the fuel by reducing the fission heat generation within the core.

Item 4 for table 4.3.1.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/04/96 and requires an extension of 154 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 4.

Item 5 MSIV Clos

As stated in UFSAR Section 7.2.1.1.5.4: The MSIV closure scram protects the reactor on loss of the heat sink. The MSIV closure initiates a scram earlier than the neutron monitoring system or nuclear system high pressure. Automatic closure of the MSIVs is initiated when conditions indicate a steam line break. The main steam line isolation scram setting is selected to give the earliest positive indication of isolation valve closure.

Item 5 for table 4.3.1.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 8/04/96 and requires an extension of 62 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 4.

Item 6 Main Steam Line High Radiation

As stated in UFSAR Section 7.2.1.1.5.4, high radiation in the vicinity of the main steam lines may indicate a gross fuel failure in the core. When high radiation is detected near the steam line, a scram is initiated to limit the release of fission products from the fuel.

Item 6 for table 4.3.1.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/30/96 and requires an extension of 128 days

to reach October 5, 1996. The justification for this extension is provided in Enclosure 4.

Item 7 Drywell Pressure

As stated in UFSAR Section 7.2.1.1.5.4, high pressure inside the primary containment may indicate a break in the nuclear system process barrier. It is prudent to scram the reactor in such a situation, to minimize the possibility of fuel damage and to reduce energy transfer from the core to the coolant.

Item 7 for table 4.3.1.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/11/96 and requires an extension of 147 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 4.

RPS Logic System Functional Tests (LSFT)

Technical Specification SR 4.3.1.2 RPS Logic System Functional Test

SR 4.3.1.2 requires a LSFT and simulated automatic actuation of all channels of the RPS at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). The Logic System Functional Test consists of performing several different plant procedures, which when combined test the complete logic system. The first of these plant procedures becomes due on 4/15/96. The longest extension required for any plant procedure supporting the Logic System Functional Test is 173 days. Since some components of the RPS system are required to be functional during modes 4 and 5 this extension is requested to the end of the refueling outage. The justification for this extension is provided in Enclosure 5.

Table 4.3.1.1-1 Item 11 Reactor Mode Switch In Shutdown.

As stated in UFSAR Section 7.2.1.1.5.4, when the mode switch is in Shutdown, the reactor is to be shutdown with all control rods inserted. This scram is not considered a protective function because it is not required to protect the fuel or nuclear system process barrier, and it bears no relationship to minimizing the release of radioactive material from any barrier.

Item 11 for table 4.3.1.1-1 requires channel functional test at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). The functional testing of this component is a portion of the system logic system functional test. This item becomes due on 6/23/96 and requires an extension of 146 days

to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 5.

RPS Response Time Testing

Technical Specification SR 4.3.1.3 requires the RPS response time testing of the functional units to be within the limits at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). The SR also requires that the response time test include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific trip system. This specification becomes due on 6/26/96 and requires an extension of 101 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 6.

Primary and Secondary Isolation

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. During normal plant operation, the isolation control system sensors and trip logic that are essential to safety are energized. When abnormal conditions are sensed, instrument channel relay contacts open and deenergize the trip logic and thereby initiate isolation. Once initiated, the CRVICS trip logics seal in and may be reset by the operator only when the initiating conditions return to normal. The following variables provide input to the CRVICS logics for initiation of reactor vessel and Containment isolation, as well as initiation trip for other plant functions when predetermined limits are exceeded:

Primary Containment

As stated in UFSAR Section 7.3.2.2.7.1, a low-water level in the RPV could indicate that reactor coolant is being lost through a breach in the nuclear system process barrier and that the core has the potential of becoming overheated as the reactor coolant inventory diminishes. Reactor vessel low-water level initiates closure of various Class A valves and Class B valves. The closure of Class A valves is intended to either isolate a breach in any of the lines in which valves are closed, or to conserve reactor coolant by closing off process lines. The closure of Class B valves is intended to prevent the escape of radioactive materials from the primary containment through process lines that are in communication with the primary containment free space. Three RPV low water level isolation trip settings are used to completely isolate the RPV and the primary containment. The level signals are defined as follows:

a. Level 3 is the highest of the three and also initiates the level scram and isolates RHR shutdown cooling.

b. Level 2 is the initiation level for the reactor core isolation cooling (RCIC) and HPCI systems and is selected to be less than the volume resulting from a void collapse occurring in the event of a scram from full power. Level 2 also closes certain containment isolation valves.

c. Level 1 is selected far enough above the top of the active fuel based on the time required for the RHR and Core Spray Systems to function in the event of a large break. Level 1 also isolates the MSIVs.

The following discussions are provided for Isolation Actuation Instrumentation (TS Table 4.3.2.1-1) surveillance interval extensions.

Item 1.a.1 Reactor Vessel Low Water Level-Level 3

Item 1.a.1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/04/96 and requires an extension of 154 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 7.

Item 1.a.2 Reactor Vessel Low Water Level-Level 2

Item 1.a.2 for table 4.3.2.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/05/96 and requires an extension of 153 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 7.

Item 1.a.3 Reactor Vessel Low Water Level-Level 1

Item 1.a.3 for table 4.3.2.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/05/96 and requires an extension of 153 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 7.

Item 1.b Drywell High Pressure

As stated in UFSAR Section 7.3.2.2.7.6, high pressure in the drywell could indicate a breach of the nuclear system process barrier inside the drywell. The

automatic closure of various Class B valves prevents the release of significant amounts of radioactive material from the primary containment. Item 1.b for table 4.3.2.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/11/96 and requires an extension of 147 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 7.

Item 1.c.1 Main Steam Line High Radiation

As stated in UFSAR Section 7.3.2.2.7.2, high radiation in the vicinity of the main steam lines could indicate a gross release of fission products from the fuel. High radiation near the main steam lines initiates isolation of all main steam lines, main steam line drain and reactor water sample line. Item 1.c.1 for table 4.3.2.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/30/96 and requires an extension of 128 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 7.

Item 1.c.2 Main Steam Line Pressure Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. Item 1.c.2 for table 4.3.2.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/09/96 and requires an extension of 149 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 7.

Item 1.d Main Steam Line Tunnel Temperature - High

High temperature in the space in which the main steam lines are located outside the primary containment could indicate a breach in a main steam line. The automatic closure of various Class A valves prevents both the excessive loss of reactor coolant and the release of significant amounts of radioactive material from the nuclear system process barrier. When high temperatures occur in the main steam line space, all four main steam lines and the main steam line drain are isolated. Item 1.d for table 4.3.2.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/05/96 and requires an extension of 183 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 7.

Item 1.e Condenser Pressure High

The Condenser Pressure High Function is provided to prevent over-pressurization of the main condenser in the event of a loss of the main condenser vacuum. As condenser pressure increases, an upper limit ensures that the MSIV closure occurs soon enough to prevent release of steam to the Turbine Building through the turbine shaft seals or from rupture of the turbine exhaust diaphragms. Item 1.e for table 4.3.2.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/06/96 and requires an extension of 182 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 7.

Item 1.f Turbine Building Area Temperature - High

High temperature in the space in which the main steam lines are located outside the primary containment could indicate a breach in a main steam line. The automatic closure of various Class A valves prevents both the excessive loss of reactor coolant and the release of significant amounts of radioactive material from the nuclear system process barrier. When high temperatures occur in the main steam line space, all four main steam lines and the main steam line drain are isolated. Item 1.f for table 4.3.2.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/05/96 and requires an extension of 183 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 7.

Reactor Water Cleanup System (RWCU) Isolation

Item 2.e Reactor Vessel Low Water Level - Level 2

See the Reactor Vessel Low Water Level discussion provided under Primary Containment above. Item 2.e for table 4.3.2.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/05/96 and requires an extension of 153 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 7.

Reactor Core Isolation Cooling (RCIC) System Isolation

Item 3.a.1 Steam Line Flow - High

High flow is provided to detect a break of the RCIC steam lines and initiate closure of the steam line isolation valves. Isolations are initiated on high flow to prevent excessive loss of reactor coolant and release of significant amounts of radioactive material. Item 3.a.1 for table 4.3.2.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/29/96 and requires an extension of 129 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 7.

Item 3.a.2 Steam Line Flow - High Timer

This instrument ensures system reliability by protecting against spurious high flow isolations which may occur during a RCIC System start-up. Item 3.a.2 for table 4.3.2.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/29/96 and requires an extension of 129 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 7.

High Pressure Coolant Injection (HPCI) System

Item 4.a.1 Steam Line Flow—High

High flow is provided to detect a break of the HPCI steam lines and initiate closure of the steam line isolation. Isolations are initiated on high flow to prevent excessive loss of reactor coolant and release of significant amounts of radioactive material. Item 4.a.1 for table 4.3.2.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/26/96 and requires an extension of 132 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 7.

Item 4.a.2 Steam Line Flow—High Timer

This instrument ensures system reliability by protecting against spurious high flow isolations which may occur during a HPCI System start-up. Item 4.a.2 for table 4.3.2.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2).

This item becomes due on 5/26/96 and requires an extension of 132 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 7.

RHR System Shutdown Cooling Mode

Item 5.a Reactor Vessel Low Water Level - Level 3

See the Reactor Vessel Low Water Level discussion provided under Primary Containment above. Item 5.a for table 4.3.2.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/04/96 and requires an extension of 154 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 7.

Secondary Containment

Item 6.a Reactor Vessel Low Water Level - Level 2

See the Reactor Vessel Low Water Level discussion provided under Primary Containment above. Item 6.a for table 4.3.2.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/05/96 and requires an extension of 195 days to the end of the refueling outage. The justification for this extension is provided in Enclosure 7.

Item 6.b Drywell Pressure High

See the Drywell Pressure High discussion provided under Primary Containment above. Item 6.b for table 4.3.2.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/11/96 and requires an extension of 147 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 7.

Isolation Logic System Functional Tests

Technical Specification SR 4.3.2.2 Isolation Logic System Function Test (LSFT)

SR 4.3.3.2 requires a LSFT and simulated automatic actuation of all channels of the Primary and Secondary Isolation system at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). The Logic System Functional Test consists of performing several different plant procedures, which when

combined test the complete logic system. The first of these plant procedures becomes due on 4/05/96. The longest extension required for any plant procedure supporting the Logic System Functional Test is 183 days. Since some components of the primary and secondary containment isolation system are required to be functional during modes 4 and 5 this extension is requested to the end of the refueling outage. The justification for this extension is provided in Enclosure 8.

Table 4.3.2.1-1 Item 1.h Manual Initiation

The manual initiation push button channels introduce signals into the isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific UFSAR safety analysis that takes credit for this function. It is retained for the overall redundancy and diversity of the isolation function as required by the NRC in the plant licensing basis. Item 1.h for table 4.3.2.1-1 requires a channel functional test at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/09/96 and requires an extension of 191 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 8.

Table 4.3.2.1-1 Item 2.d Standby Liquid Control System (SLCS) Initiation.

The isolation of the RWCU System is required when the SLCS System has been initiated to prevent dilution and removal of the boron solution by the RWCU System. SLCS System initiation signals are initiated from the two SLCS pump start signals. Item 2.d for table 4.3.2.1-1 requires channel functional at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 7/01/96 and requires an extension of 138 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 8.

Table 4.3.2.1-1 Item 2.g RWCU System - Manual Initiation

The manual initiation push button channels introduce signals into the isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific UFSAR safety analysis that takes credit for this function. It is retained for the overall redundancy and diversity of the isolation function as required by the NRC in the plant licensing basis. Item 2.g for table 4.3.2.1-1 requires a channel functional test at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/27/96 and requires an extension of 131 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 8.

Table 4.3.2.1-1 Item 4.e High Pressure Coolant Injection Manual Initiation

The manual initiation push button channels introduce signals into the isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific UFSAR safety analysis that takes credit for this function. It is retained for the overall redundancy and diversity of the isolation function as required by the NRC in the plant licensing basis. Item 4.e for table 4.3.2.1-1 requires channel functional test at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 9/01/96 and requires an extension of 34 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 8.

Table 4.3.2.1-1 Item 5.c RHR Shutdown Cooling Mode Isolation Manual Initiation

The manual initiation push button channels introduce signals into the isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific UFSAR safety analysis that takes credit for this function. It is retained for the overall redundancy and diversity of the isolation function as required by the NRC in the plant licensing basis. Item 5.c for table 4.3.2.1-1 requires channel functional test at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/30/96 and requires an extension of 170 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 8.

Isolation System Response Time Testing

Technical Specification SR 4.3.2.3 requires the Primary and Secondary Isolation system response time testing of the functional units to be within the limits at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). The SR also provides that the response time test include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific trip system. This item becomes due on 6/26/96 and requires an extension of 101 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 9.

Emergency Core Cooling System

As stated in UFSAR Section 6.3.1 the objective of the emergency core cooling systems (ECCS), in conjunction with the containment, is to limit the release of radioactive materials should a LOCA occur, so that resulting radiation exposures are kept within the guideline values given in 10 CFR 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. For most anticipated operational occurrences and Design Basis Accidents (DBAs), a wide range of dependent and independent parameters are monitored. The following variables provide input to the ECCS logics for initiation of ECCS and ECCS support systems.

Core Spray System

Item 1.a Reactor Vessel Water Level Low - Level 1

Previously discussed under Primary and Secondary Containment Instrumentation. Item 1.a for table 4.3.3.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/16/96 and requires an extension of 184 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 10.

Item 1.b Drywell Pressure High

Previously discussed under Primary and Secondary Containment Instrumentation. Item 1.b for table 4.3.3.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/19/96 and requires an extension of 169 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 10.

Item 1.c Steam Dome Pressure Low

Reactor Steam Dome low pressure provides the permissive signal for ECCS injection. This signal prevents opening the injection valves until pressure is within the limitations of the injection system. This prevents the potential overpressurization of ECCS low pressure piping. Item 1.c for table 4.3.3.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/07/96 and requires an extension of 223 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 10.

Low Pressure Coolant Injection Mode of RHR System

Item 2.a Reactor Vessel Water Level Low - Level 1

Previously discussed under Primary and Secondary Containment Isolation Instrumentation. Item 2.a for table 4.3.3.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/16/96 and requires an extension of 184 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 10.

Item 2.b Drywell Pressure High

Previously discussed under Primary and Secondary Containment Isolation Instrumentation. Item 2.b for table 4.3.3.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/19/96 and requires an extension of 169 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 10.

Item 2.c Steam Dome Pressure Low

Previously discussed under Item 1.c above

Item 2.c for table 4.3.3.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/07/96 and requires an extension of 223 days to the end of the refueling outage. The justification for this extension is provided in Enclosure 10.

Item 2.d Reactor Vessel Water Level Low - Level 2

Previously discussed under Primary and Secondary Containment Isolation Instrumentation. Item 2.d for table 4.3.3.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/16/96 and requires an extension of 184 days to the end of the refueling outage. The justification for this extension is provided in Enclosure 10.

Item 2.e Steam Dome Pressure Low

Steam Dome Pressure low provides a permissive signal for LPCI loop select logic. The signal delays LPCI loop select logic during a small break loss of coolant accident until reactor pressure decreases to increase sensitivity of the loop selection process.

Item 2.e for table 4.3.3.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/07/96 and requires an extension of 223 days to the end of the refueling outage. The justification for this extension is provided in Enclosure 10.

Item 2.f Riser DP High and Item 2.g Recirculation Pump DP High

The entire LPCI system is activated by either high drywell pressure or reactor low water level. The recirculation pump differential switches set up the network logic in the optimum arrangement depending on whether one pump or two pumps are operating. If only one pump is operating, the pressure difference due to the pump flow tends to mask the pressure difference due to the break. To avoid this, the loop selection time is delayed (0.5 sec) to determine if either recirculation pump is shut down and to allow proper selection of the unbroken loop. If only one pump is operating, the pump is tripped by the logic circuit. After satisfaction of the reactor-pressure permissive mentioned above or if both recirculation pumps have indicated DP greater than the setpoint, the logic network is delayed about 2 sec to allow momentum effects to settle and system parameters to stabilize.
Finally, the loop selection is made. If loop A pressure is greater than that of loop B, then loop B is broken and injection will occur in loop A. If the pressure at loop A is not greater than that at loop B, the 0.5-sec timer will run out, causing loop B to be selected. The 0.5-sec time delay allows the loop selection logic to function. The DP is measured from each recirculation loop riser pipe to the corresponding riser pipe on the other recirculation loop.

Item 2.f for table 4.3.3.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/04/96 and requires an extension of 184 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 10.

Item 2.g for table 4.3.3.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/03/96 and requires an extension of

185 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 10.

High Pressure Coolant Injection System

Item 3.a Reactor Vessel Water Level Low - Level 2

Previously discussed under Primary and Secondary Containment Isolation Instrumentation. Item 3.a for table 4.3.3.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/16/96 and requires an extension of 184 days to the end of the refueling outage. The justification for this extension is provided in Enclosure 10.

Item 3.b Drywell Pressure High

Previously discussed under Primary and Secondary Containment Isolation Instrumentation. Item 3.b for table 4.3.3.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/19/96 and requires an extension of 169 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 10.

Item 3.e Reactor Vessel High Water Level - Level 8

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Further increase in level could result in excessive vessel level. The RPV high water level setting that closes the HPCI turbine steam supply valve is near the top of the steam separators and is sufficient to prevent excessive reactor vessel level increase. Item 3.e for table 4.3.3.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/16/96 and requires an extension of 184 days to the end of the refueling outage. The justification for this extension is provided in Enclosure 10.

Automatic Depressurization System (ADS)

Item 4.a Reactor Vessel Water Level Low - Level 1

Previously discussed under Primary and Secondary Containment Instrumentation. Item 4.a for table 4.3.3.1-1 requires channel calibration at least once per 18

months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/16/96 and requires an extension of 184 days to the end of the refueling outage. The justification for this extension is provided in Enclosure 10.

Item 4.f Reactor Vessel Water Level Low - Level 3

Previously discussed under Primary and Secondary Containment Instrumentation. Item 4.f for table 4.3.3.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/07/96 and requires an extension of 181 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 10.

Item 4.h Drywell Pressure - High Bypass Timer

One of the signals required for ADS initiation is Drywell Pressure—High. However, if the event requiring ADS initiation occurs outside the drywell (e.g., main steam line break outside containment), a high drywell pressure signal may never be present. Therefore, the Automatic Depressurization System Low Water Level Actuation Timer is used to bypass the Drywell Pressure—High Function after a certain time period has elapsed. Item 4.h for table 4.3.3.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/16/96 and requires an extension of 184 days to the end of the refueling outage. The justification for this extension is provided in Enclosure 10.

ECCS System Logic System Functional Tests

Technical Specification SR 4.3.3.2 ECCS System Logic System Functional Test

SR 4.3.3.2 requires a LSFT and simulated automatic actuation of all channels of the ECCS system at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). The Logic System Functional Test consists of performing several different plant procedures, which when combined test the complete logic system. The first of these plant procedures becomes due on 4/03/96. The longest extension required for any plant procedure supporting the Logic System Functional Test is 227 days. Since some components of the ECCS system are required to be functional during modes 4 and 5 this extension is requested to the end of the refueling outage. The justification for this extension is provided in Enclosure 11.

Table 4.3.3.1-1 Item 1.d Core Spray Manual Initiation

The manual initiation push button channels introduce signals into the ECCS start logic that are redundant to the automatic protective instrumentation and provide manual ECCS start capability. Item 1.d for table 4.3.3.1-1 requires channel functional tests at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/09/96 and requires an extension of 191 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 11.

Table 4.3.3.1-1 Item 2.h LPCI Manual Initiation

The manual initiation push button channels introduce signals into the ECCS start logic that are redundant to the automatic protective instrumentation and provide manual ECCS start capability. Item 2.h for table 4.3.3.1-1 requires channel functional test at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/15/96 and requires an extension of 185 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 11.

Table 4.3.3.1-1 Item 3.f HPCI Manual Initiation

The manual initiation push button channels introduce signals into the ECCS start logic that are redundant to the automatic protective instrumentation and provide manual ECCS start capability. Item 3.f for table 4.3.3.1-1 requires channel functional test at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 9/01/96 and requires an extension of 34 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 11.

Table 4.3.3.1-1 Item 4.i ADS Manual Inhibit

Manual inhibit switches are provided in the control room for the ADS; however, their function is not required for ADS operability (provided ADS is not inhibited when required to be Operable). Item 4.i for table 4.3.3.1-1 requires channel functional test at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 7/25/96 and requires an extension of 72 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 11.

ECCS System Response Time Testing

Technical Specification SR 4.3.3.3 requires the ECCS system response time testing of the functional units to be within the limits at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). The SR also provides that the response time test include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific trip system. This item becomes due on 5/09/96 and requires an extension of 191 days to the end of the refueling outage. The justification for this extension is provided in Enclosure 12.

ATWS Recirculation Pump Trip Instrumentation

The ATWS-RPT System initiates a recirculation pump trip (RPT), adding negative reactivity, following events in which a scram does not (but should) occur, to lessen the effects of an ATWS event. Tripping the recirculation pumps adds negative reactivity from the increase in steam voiding in the core area from core flow decrease. When Reactor Vessel Water Level low or Reactor Steam Dome Pressure high setpoint is reached, the recirculation pump drive motor breakers and motor-generator field breakers trip. Both of these initiating parameters have been discussed for the Primary and Secondary Containment Isolation or Reactor Protection System section above.

Item 1 for table 4.3.4-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/16/96 and requires an extension of 184 days to the end of the refueling outage. The justification for this extension is provided in Enclosure 13.

Item 2 for table 4.3.4-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/19/96 and requires an extension of 169 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 13.

ATWS Recirculation Pump Trip System Logic System Functional Test

Technical Specification SR 4.3.4.2 requires a LSFT and simulated automatic actuation of all channels of the ATWS Recirculation Pump Trip System at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). The Logic System Functional Test consists of performing several different plant procedures, which when combined test the complete logic system. The first of these plant procedures becomes due on 4/19/96. The longest extension required for any plant procedure supporting the Logic System Functional Test is 184 days. Since some components tested in these procedures are required to be functional during modes 4 and 5

this extension is requested to the end of the refueling outage. The justification for this extension is provided in Enclosure 14.

Reactor Core Isolation Cooling System

As stated in UFSAR Section 7.4.1.1, the reactor core isolation cooling (RCIC) system provides core cooling during reactor shutdown by pumping makeup water into the RPV in case of a loss main feedwater flow. The RCIC system is started either automatically upon receipt of a low reactor water level signal (Level 2) or manually by the operator. Water is pumped to the core by a turbine pump driven by reactor steam. High water level (Level 8) in the RPV indicates that the RCIC system has performed satisfactorily in providing makeup water to the RPV. All of these parameters have been discussed for the Primary and Secondary Containment Isolation or ECCS sections above.

Item a. for table 4.3.5.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/16/96 and requires an extension of 184 days to reach the end of the refueling. The justification for this extension is provided in Enclosure 15.

Item b. for table 4.3.5.1-1 requires channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 5/16/96 and requires an extension of 184 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 15.

Reactor Core Isolation Cooling Logic System Functional Test

Technical Specification SR 4.3.5.2 requires a LSFT and simulated automatic actuation of all channels of the Reactor Core Isolation Cooling System at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). The Logic System Functional Test consists of performing several different plant procedures, which when combined test the complete logic system. The first of these plant procedures becomes due on 5/16/96. The longest extension required for any plant procedure supporting the Logic System Functional Test is 184 days. Since some components tested in these procedures are required to be functional during modes 4 and 5 this extension is requested to the end of the refueling outage. The justification for this extension is provided in Enclosure 16.

Control Rod Block Instrumentation

The scram discharge high water level trip bypass is controlled by the manual operation of two keylocked switches, a bypass switch, and the mode switch. The mode switch must be in either the Shutdown or the Refuel position in order to bypass this trip. Four bypass

channels emanate from the four banks of the RPS mode switch and are each connected into the RPS logic. 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. In addition, actuating the bypass initiates a control rod block. Resetting the trip actuators opens the scram discharge volume vent and drain valves. An annunciator in the main control room indicates the bypass condition. Operation of both the Mode Switch and the Scram Discharge Volume were previously discussed in this submittal.

Item 5.b for table 4.3.6-1 requires a channel functional test at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 8/05/96 and requires an extension of 103 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 17.

Item 7 for table 4.3.6-1 requires a channel functional test at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 6/23/96 and requires an extension of 146 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 17.

Remote Shutdown System

As discussed in UFSAR Section 7.5.1.5 and Section 7.5.1.5.3, the Remote Shutdown System provides a means to carry out reactor shutdown and bring the reactor to cold shutdown from outside the main control room for conditions where the control room becomes uninhabitable. The assumed plant conditions for the use of the Remote Shutdown System include normal operating conditions, no accidents or transients are occurring, all plant personnel have been evacuated from the Main Control Room, and it is inaccessible for control of the plant. The initial event that causes the Main Control Room to become inaccessible is assumed to be such that the Reactor Operator can manually scram the reactor before leaving the Main Control Room. Specification SR 4.3.7.4.1, Table 4.3.7.4-1, Items 1 and 2, require that these instrumentation channels units be calibrated at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2).

Item 1 RPV Pressure, becomes due on 4/20/96 and requires an extension of 168 days to reach October 5, 1996. The justification for this surveillance interval extension is provided in Enclosure 18.

Item 2 RPV Level, becomes due on 5/16/96 and requires an extension of 184 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 18.

Post Accident Monitoring

As discussed in UFSAR Section 7.5.1.4 information readouts are provided to accommodate events up to and including a LOCA. These readouts are designed from the standpoint of operator action, information, and event tracking requirements, providing assurance that requirements for all other credible events or incidents will be covered. The process instrumentation provides information to the operator for his use in monitoring reactor conditions after a LOCA.

Technical Specification SR 4.3.7.5, Table 4.3.7.5-1, Items 1, 2.a , 2.b, 11, 12, and 16 require a channel calibration of the instrumentation at least once per 18 months (with an allowable surveillance interval extension of 4.5 months per TS 4.0.2). Technical Specification Table 4.3.7.5-1, Item 1 becomes due on 4/24/96 and requires an extension of 164 days to reach October 5, 1996. The justification for this surveillance interval extension is provided in Enclosure 19.

Item 2.a, becomes due on 4/06/96 and requires an extension of 182 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 19.

Item 2.b, becomes due on 5/16/96 and requires an extension of 184 days to the end of the refueling outage. The justification for this extension is provided in Enclosure 19.

Item 11, becomes due on 7/23/96 and requires an extension of 74 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 19.

Item 12, becomes due on 6/06/96 and requires an extension of 121 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 19.

Item 16, the primary containment isolation valve position verification, consists of performing several different plant procedures, which when combined verify all containment isolation valves. The first of these plant procedures becomes due on 3/29/96. The longest extension required for any plant procedure supporting the containment isolation valve position verifications is 191 days. Since components of the containment isolation system (i.e., secondary containment isolation) checked by some of the same procedures are required to be functional during modes 4 and 5, this extension is requested to the end of the refueling outage. The justification for this extension is provided in Enclosure 19.

Loose Parts Monitoring System

The purpose of the Loose Parts Monitoring system is to detect and annunciate unusual noises that may indicate a metallic loose part in the primary system in order to avoid or mitigate possible damage to the primary system components.

SR 4.3.7.10.c, requires that each channel of the loose-part detection system be demonstrated operable by performance of a channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item will becomes due on 9/19/96 and requires an extension of 16 days to reach October 5,1996. The justification for this extension is provided in Enclosure 20.

Feedwater Turbine Trip System

As stated in Technical Specification Basis Section 4.3.9, the feedwater/main turbine trip system actuation instrumentation is provided to initiate action of the feedwater system/main turbine trip system in the event of a high reactor vessel water level due to failure of the feedwater controller under maximum demand.

Technical Specification Section 4.3.9.1 requires each feedwater/main turbine trip system actuation instrumentation channel be demonstrated operable by the performance of Channel Calibrations for the operational conditions and at the frequencies shown in Table 4.3.9.1-1. Table 4.3.9.1-1 requires these tests at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2), for Reactor Vessel High Water Level -Level 8. This item becomes due on 4/07/96 and requires an extension of 181 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 21.

Feedwater Turbine Trip System Logic System Functional Test

Technical Specification Section 4.3.9.2, requires a Logic System Functional Test and simulated automatic operation of all channels at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/07/96 and requires an extension of 181 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 22.

Alternative Shutdown System

As stated in Technical Specification Basis Section 4.3.11, the operability of the alternative shutdown system ensures that a fire will not preclude achieving safe shutdown. The alternative shutdown system instrumentation is independent of areas where a fire could damage systems normally used to shutdown the reactor. Thus, the

system capability is consistent with General Design Criterion 3 and Appendix R to 10 CFR 50. Technical Specification Section 4.3.11.1, requires each alternative shutdown instrumentation channel be demonstrated operable by the performance of Channel Calibrations for the operational conditions and at the frequencies shown in Table 4.3.11.1-1. Table 4.3.11.1-1, requires these tests at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2).

Items 7 (Reactor Water Level) and 8 (Reactor Pressure) will become due on 4/20/96 and require an extension of 168 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 23.

Safety Relief Valves

The safety valve function of the safety relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. A total of 11 operable safety relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient. SR 4.4.2.1.1 requires that the valve position indicator for each safety relief valve be demonstrated operable with the pressure setpoint of each of the tail-pipe pressure switches verified to be 30 ± 5 psig by performance of a channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 7/23/96 and requires an extension of 74 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 24.

Additionally SR 4.4.2.1.2, requires at least 1/2 of the safety relief valves be set pressure tested at least once per 18 months, such that all 15 safety relief valves are set pressure tested at least once per 40 months. Only the portion of the surveillance requiring at least 1/2 of the safety relief valves be tested at least once per 18 months will require an extension. This portion of the SR becomes due on 6/12/96 and requires an extension of 115 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 24. All 15 safety relief valves were tested during the last refueling outage (RFO4) therefore the requested one-time extension will not affect the plant's ability to test all valves in 40 months.

SR 4.4.2.2.b requires the low-low set function pressure actuation instrumentation be demonstrated Operable by performance of a Channel Calibration, Logic System Functional Test and simulated automatic operation of the entire system at least once per 18 months. This item becomes due on 4/19/96 and requires an extension of 169 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 24.

Reactor Coolant System Leakage

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be pressure boundary leakage, the reactor will be shut down to allow further investigation and corrective action. The equipment drain sump collects only identified leakage and is equipped with high/low-level switches that control the sump drain. The floor drain sump collects unidentified leakage.

SR 4.4.3.1.b requires the primary containment sump flow and drywell floor drain sump level monitoring systems be demonstrated operable by performance of a channel calibration test at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 6/15/96 and requires an extension of 112 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 25.

Reactor Coolant Isolation Valve Leakage

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent inter-system LOCA. Leakage from the RCS pressure isolation valves is identified leakage and will be considered as a portion of the allowed limit.

SR 4.4.3.2.2.a, requires that each reactor coolant system pressure isolation valve specified in Technical Specification Table 3.4.3.2-1 be demonstrated operable by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit, at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/07/96 and requires an extension of 181 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 25.

ASME Code Inspection Program

As stated in Technical Specification Basis Section 3/4.4.8: The inspection programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through Summer, 1975. The inservice inspection program for ASME Code Class 1, 2, and 3 components will b performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i).

SR 4.0.5 provides Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components as discussed earlier.

Where specific Technical Specification exemptions are required for ASME components (i.e., excess flow check valves) these exemptions are discussed and justified with the referenced Technical Specification sections. Although not specifically identified in Fermi 2 Technical Specifications, a Code 2 year requirement is being treated as a 731 day surveillance frequency and is subject to a 25% extension. This has already been determined to be acceptable per NUREG-1482, Section 6.2. The surveillances associated with Technical Specification 4.4.8 are associated with the RPV system leakage test, piping and vessel NDE examinations, RPV internals examinations, Class 1 retaining bolt examinations and examination of pump and valve internals. The ASME requirements for these surveillances specify a refueling or 24 month evaluation frequency. Based on these requirements a specific exemption is not required for Technical Specification SR 4.4.8 or 4.0.5. However, since the normal Fermi 2 operating cycle is 18 months and the inspections have been performed on a 18 month cycle, Enclosure 26 provides discussions which justify a minimal impact on plant safety for extending the operating cycle for these surveillances to the fall outage.

Emergency Core Cooling System Operation

As stated in Technical Specification Bases Section 3/4.5.1 and 3/4.5.2, the core spray system (CSS), together with the LPCI mode of the RHR system, are provided to assure that the core is adequately cooled following a loss-of-coolant accident and provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the ADS. The CSS is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining. The surveillance requirements provide adequate assurance that the CSS will be Operable when required. 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 reactor shutdown.

SR 4.5.l.c.1, requires that the emergency core cooling systems be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) for the CSS, the LPCI system, and the HPCI system, by performing a system functional test which includes simulated automatic actuation of the system throughout 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 this test.

SR 4.5.1.c.1 for CSS, LPCI and HPCI becomes due on 5/09/96 and requires an extension of 191 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 27.

SR 4.5.l.d.2.a, requires that the emergency core cooling systems be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) for the ADS by performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

SR 4.5.1.d.2.a for ADS becomes due on 7/25/96 and requires an extension of 72 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 27.

SR 4.5.2.1 requires that at least the required ECCS be demonstrated operable for modes 4 and 5 per Surveillance Requirement 4.5.1, with the exception that, for the LPCI system, the cross-tie valve may be closed to isolate a subsystem if the operable subsystem is made capable of injection to the reactor vessel. The extension dates listed above for the 4.5.1 SRs include the time required to the end of the refueling outage. This SR does not require any additional extension other than the ones requested for section 4.5.1

Primary Containment Leakage

As stated in Technical Specification Bases Section 3/4.6.1.2 the limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure of P_a, 56.5 psig. Updated analysis demonstrates maximum expected pressure is less than 56.5 psig.

SR 4.6.1.2.b requires the primary containment leakage rates be demonstrated by Type B and C tests conducted with gas at P_a, 56.5 psig (unless a hydrostatic test is required), at intervals no greater than 24 months except for tests involving:

1. Air locks,
2. Main steam line isolation valves,
3. Penetrations using continuous leakage monitoring systems,

4. Valves pressurized with fluid from a seal system,
5. ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and
6. Purge supply and exhaust isolation valves with resilient material seals.

This item becomes due on 4/30/96 and requires an extension of 157 days to reach October 5, 1996. The justification for this extension is provided in Reference 2.

SR 4.6.1.2.d requires the primary containment leakage rates be demonstrated for Main Steam Isolation Valves by leak testing at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/18/96 and requires an extension of 170 days to reach October 5, 1996. The justification for this extension is provided in Reference 2.

SR 4.6.1.2.g requires the primary containment leakage rates be demonstrated for ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment by leak testing at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/05/96 and requires an extension of 183 days to reach October 5, 1996. The justification for this extension is provided in Reference 2.

MSIV Leakage Control System

As stated in Technical Specification Bases Section 3/4.6.1.4, calculated doses resulting from the maximum leakage allowance for the main steam line isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR Part 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIVs such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIVs when isolation of the primary system and containment is required.

SR 4.6.1.4.d.3 requires that each MSIV leakage control system subsystem be demonstrated operable by verifying the pressure control (pressure and differential pressure) instrumentation to be operable by performance of a channel calibration at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 4/13/96 and requires an extension of 175 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 28.

Pressure Suppression Systems

As stated in Technical Specification Bases Section 3/4.6.2, the suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1045 psig.

SR 4.6.2.1.e requires the suppression chamber be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by a visual inspection of the accessible interior and exterior of the suppression chamber. This item becomes due on 9/13/96 and requires an extension of 22 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 29.

SR 4.6.2.1.h requires the suppression chamber be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure of 1 psi and verifying that the differential pressure does not decrease by more than 0.20 inch of water per minute for a period of 10 minutes. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed. This item becomes due on 10/02/96 and requires an extension of 3 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 29.

Primary Containment Isolation Valves

As stated in Technical Specification Bases Section 3/4.6.3, 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 and is consistent with the requirements of GDC 54 through 57 of Appendix A of 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

SR 4.6.3.2 requires each primary containment automatic isolation valve be demonstrated operable during COLD SHUTDOWN or REFUELING at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by verifying that on a containment isolation test signal each automatic isolation valve

actuates to its isolation position. The isolation valve operability verification consists of performing several different plant procedures, which when combined verify operability of all isolation valves. The first of these plant procedures becomes due on 4/05/96. The longest extension required for any plant procedure supporting the isolation valve operability is 189 days. Since components of the isolation system tested by some of the same procedures are required to be functional during modes 4 and 5 this extension is requested to the end of the refueling outage. The justification for this extension is provided in Enclosure 30.

SR 4.6.3.4 requires each reactor instrumentation line excess flow check valve shall be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by verifying that the valve checks flow. This item becomes due on 9/19/96 and requires an extension of 16 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 30.

SR 4.6.3.5.b requires that each traversing in-core probe system explosive isolation valve be demonstrated operable: at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2), by removing the explosive squib from at least one explosive valve such that the explosive squib in each explosive valve will be tested at least once per 90 months, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life or operating life, as applicable. This item becomes due on 7/05/96 and requires an extension of 92 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 30.

Vacuum Relief

As stated in Technical Specification Bases Section 3/4.6.4, vacuum relief breakers are provided to equalize the pressure between the suppression chamber and drywell and between the Reactor Building and suppression chamber. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures. There are valves to provide redundancy so that operation may continue for up to 72 hours with redundant vacuum breakers inoperable in the closed position. The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident.

SR 4.6.4.1.b.2 items a, b, c require each suppression chamber - drywell vacuum breaker to be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by:

a) Verifying the opening setpoint, from the closed position, to be less than or equal to 0.5 psid, and
b) Verifying both position indicators Operable by performance of a Channel Calibration.
c) Verify the opening gap for switch actuation to be less than or equal to 0.03 inches.

These items become due on 8/03/96 and require an extension of 63 days to reach October 5, 1996. The justification for these extensions is provided in Enclosure 31.

SR 4.6.4.2.b.2 items a, b, c require each Reactor Building - suppression chamber vacuum breaker be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by:

a) Demonstrating that the force required to open each vacuum breaker does not exceed the equivalent of 0.5 psid.

b) Visual inspection.

c) Verifying the position indicator Operable by performance of a Channel Calibration.

These items become due on 8/06/96 and require an extension of 60 days to reach October 5, 1996. The justification for these extensions is provided in Enclosure 31.

Secondary Containment

As stated in Technical Specification Bases Section 3/4.6.5, secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The reactor building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times, the drywell may be open and, when required, secondary containment integrity is specified.

SR 4.6.5.2.b requires each secondary containment ventilation system automatic isolation damper shown in Technical Specification Table 3.6.5.2-1 be demonstrated operable during cold shutdown or refueling at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by verifying that on a containment isolation test signal each isolation damper actuates to its isolation position. This item becomes due on 10/20/96 and requires an extension of 27 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 32.

Service Water Systems

As stated in Technical Specification Bases Section 3/4.7.1, the OPERABILITY of the service water systems ensures 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 the emergency equipment cooling water system be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by verifying that each automatic valve servicing non-safety-related equipment actuates to its isolation position and the associated EECW pump automatically starts on an automatic actuation test signal. This item becomes due on 8/17/96 and requires an extension of 91 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 33.

SR 4.7.1.3.b requires the emergency equipment service water system be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by verifying the EESW pump automatically starts upon receipt of an actuation test signal. This item becomes due on 8/17/96 and requires an extension of 91 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 33.

SR 4.7.1.4.b requires each of the diesel generator service water subsystems be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by verifying that each DGSW pump starts automatically upon receipt of a start signal for the associated diesel generator. This item becomes due on 4/15/96 and requires an extension of 215 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 33.

Control Room Emergency Filtration System

As stated in Technical Specification Bases Section 3/4.7.2, the operability of the control room emergency filtration system ensures that (1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and (2) the control room will remain habitable for operations personnel during and following all design basis accident conditions. Continuous operation of the system with heaters operable for 10 hours during each 31-day period is sufficient to reduce the buildup of moisture on the charcoal adsorbers. The operability of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50.

SR 4.7.2.1.c, items 1, 2, 3 require that the control room emergency filtration system be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by:

1. Verifying that the system satisfies the in-place penetration testing acceptance criteria of less than 1.0% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, while operating the system at a flow rate of 1800 cfm ± 10% through the makeup filter and 3000 cfm ± 10% through the recirculation filter.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March l978, for a methyl iodide penetration of less than 1.0%.

3. Verifying a system flow rate of 3000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

These items becomes due on 6/23/96 and requires an extension of 146 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 34.

SR 4.7.2.1.e items 1, 2 & 4 require that the control room emergency filtration system be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by:

1. Verifying that the pressure drop across the recirculation train and across the makeup train combined HEPA filters and charcoal adsorber banks is less than 8 inches and 6 inches water gauge respectively while operating the system at a flow rate of 3000 cfm ± 10% through the recirculation filter train and 1800 cfm ± 10% through the makeup filter train. This item (SR 4.7.2.1.e.1) becomes due on 6/23/96 and requires an extension of 146 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 34.

2. Verifying that the system will automatically switch to the recirculation mode of operation on each of the below actuation test signals and verifying that on any one of the below recirculation mode actuation test signals, the system automatically switches to the recirculation mode of operation, the isolation

valves close within 5 seconds and the control room is maintained at a positive pressure of at least 0.125 inch water gauge relative to the outside atmosphere during system operation at a flow rate less than or equal to 1800 cfm through the emergency makeup air filter:

a) Control center inlet radiation monitor.
b) Fuel pool ventilation exhaust radiation monitor.
c) Low reactor water level.
d) High drywell pressure.

This item (SR 4.7.2.1.e.2) becomes due on 10/20/96 and requires an extension of 27 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 34.

4. Verifying that each of the emergency makeup inlet air heaters dissipate 12.0 ± 2.0 kW when tested in accordance with ANSI N510-1980. This item (SR 4.7.2.1.e.4) becomes due on 06/23/96 and requires an extension of 146 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 34.

SR 4.7.2.1.h requires that the control room emergency filtration system be demonstrated operable at least once per 36 months by verifying that the sections of Control Room Emergency Filtration System duct listed in Table 4.7.2.1-1, when leak tested in accordance with ASME N510-1989 exhibit inleakage less than the acceptance criteria listed in Technical Specification Table 4.7.2.1-1 for the associated pressures. This SR verifies the integrity of the duct sealing compound and duct overall integrity. This SR was performed during RFO4 and an extension will be needed so it will not need to be repeated during RFO5. Its critical date is April 15, 1998 so it should not need to be performed during RFO5. However, based on the length of Operating Cycle 5, there is little flexibility to ensure the test does not fall due before RFO6 and so require a plant shutdown for this one test. Therefore, Detroit Edison requests explicitly that the next due date for this surveillance be extended to June 1, 1998, an extension of 48 days, so that the test can be performed during RFO6. The justification for this extension is provided in Enclosure 34.

Snubbers

As stated in Technical Specification Bases Section 3/4.7.5: All snubbers are required Operable to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on non-safety-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.b requires each snubber be demonstrated operable by performance of visual inspections. Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.7.5-1. The visual inspection interval for each category of snubber shall be determined based upon the criteria provided in Table 4.7.5-1. The first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before Amendment 84.

The visual inspection requirements for snubbers is based on Generic Letter 90-09 requirements implemented in RFO3 (Technical Specification change), based on the RFO2 inspection results with NRC approval. As a result, visual inspection of the entire population of snubbers was not performed during RFO3. All snubbers were visually inspected during RFO4, and based up these on inspection results do not require inspection until RFO6. A self imposed inspection requirement (non-Technical Specification) of selected N30 system snubbers will be performed during RFO5 (fall 1996). This SR does not require an extension.

SR 4.7.5.e requires each snubber be demonstrated operable by performing Functional Tests. At least once per 18 months during shutdown, a representative sample of snubbers shall be tested 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 selected prior to the test period or the sample plan used in the prior test period shall be implemented. Detroit Edison has elected to again use the 10% sample plan.

This item becomes due on 8/03/96 and requires an extension of 105 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 35.

Appendix R Alternative Shutdown Auxiliary Systems

As discussed in the basis for Technical Specifications Section 3/4.7.11, the systems identified in this section are those utilized for Appendix R Alternative shutdown but not included in other sections of the Technical Specifications. The ACTION statements assure that the auxiliary systems will be OPERABLE or that acceptable alternative means are established to achieve the same objective.

SR 4.7.11.4 requires that each alternative shutdown system control circuit be demonstrated OPERABLE by verifying its capability to perform its intended function(s)

at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2). This item becomes due on 7/15/96 and requires an extension of 82 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 36.

A.C. Sources, D.C. Sources And Onsite Power Distribution Systems

As stated in Technical Specification Bases Sections 3/4.8.1, 3/4.8.2 and 3/4.8.3 the Operability of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criteria 17 of Appendix "A" to 10 CFR 50.

The surveillance requirements for demonstrating the operability of the diesel generators are in accordance with the recommendations of Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies", December 1979; Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants", Revision 1, August 1977; and Regulatory Guide 1.137, "Fuel-Oil Systems for Standby Diesel Generators", Revision 1, October 1979.

Emergency Diesel Generators

SR 4.8.1.1.2.e.1 requires each of the diesel generators be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service. This item becomes due on 4/14/96 and requires an extension of 216 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 37.

SR 4.8.1.1.2.e.2 requires each of the diesel generators be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) 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% of the difference between nominal speed and the overspeed trip setpoint or 115% of nominal speed, whichever is lower. This item becomes due on 4/15/96 and requires an extension of 21[illegible] days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 37.

SR 4.8.1.1.2.e.3 requires each of the diesel generators be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by verifying the diesel generator capability to reject a load of 2850 kW without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection. This item becomes due on 4/15/96 and requires an extension of 215 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 37.

SR 4.8.1.1.2.e.4.a requires each of the diesel generators be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by simulating a loss-of-offsite power by itself, and verifying deenergization of the emergency busses and load shedding from the emergency busses. This item becomes due on 6/28/96 and requires an extension of 141 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 37.

SR 4.8.1.1.2.e.4.b requires each of the diesel generators be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by simulating a loss-of-offsite power by itself, and verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test. This item becomes due on 6/28/96 and requires an extension of 141 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 37.

SR 4.8.1.1.2.e.5 requires each of the diesel generators be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by verifying that on an ECCS actuation test signal, without loss-of-offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test. This item becomes due on 4/15/96 and requires an extension of 215 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 37.

SR 4.8.1.1.2.e.6.a requires each of the diesel generators be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) 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. This item becomes due on 6/28/96 and requires an

extension of 141 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 37.

SR 4.8.1.1.2.e.6.b requires each of the diesel generators be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) 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 the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test. This item becomes due on 6/28/96 and requires an extension of 141 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 37.

SR 4.8.1.1.2.e.7 requires each of the diesel generators be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by verifying that all automatic diesel generator trips, except overspeed, generator differential, low lube oil pressure, crankcase overpressure, and failure to start are automatically bypassed for an emergency start signal. This item becomes due on 4/15/96 and requires an extension of 215 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 37.

SR 4.8.1.1.2.e.8 requires each of the diesel generators be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) 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 Surveillance Requirement 4.8.1.1.2.a.4). This item (SR 4.8.1.1.2.e.8) becomes due on 7/08/96 and requires an extension of 131 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 37.

SR 4.8.1.1.2.e.9 requires each of the diesel generators be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 3100 kW. This item becomes due on 8/14/96 and

requires an extension of 94 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 37.

SR 4.8.1.1.2.e.10 requires each of the diesel generators be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by verifying the diesel generator's capability to synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power, transfer its loads to the offsite power source, and be restored to its standby status. This item becomes due on 8/14/96 and requires an extension of 94 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 37.

SR 4.8.1.1.2.e.11 requires each of the diesel generators be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by verifying that the automatic load sequence timer is Operable with the interval between each load block within ± 10% of its design interval. This item becomes due on 7/16/96 and requires an extension of 123 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 37.

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 (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

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

These items become due on 4/15/96 and require an extension of 215 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 37.

SR 4.8.1.2 requires that at least the required A.C. electrical power sources shall be demonstrated Operable per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2, and 4.8.1.1.3, except for the requirement of 4.8.1.1.2.a.5. Only the applicable portions of 4.8.1.1.2.e discussed above will require extension of the SRs through the Refueling outage. The extension dates listed above for the 4.8.1.1.2 e SRs include the time required to the end of the refueling outage. This SR does not require any additional extension other than the ones requested for section 4.8.1.1.2.e.

Battery

The surveillance requirements for demonstrating the OPERABILITY of the unit batteries are in accordance with the recommendations of Regulatory Guide 1.129 "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1972, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

SR 4.8.2.1.c items 3 & 4 require that each of the required 130-volt batteries and chargers be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by verifying that:

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

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

These items become due on 5/10/96 and require extensions of 190 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 38.

SR 4.8.2.1.d requires that each of the required 130-volt batteries and chargers shall be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by verifying that either:

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

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

    a) Batteries 2PA and 2PB greater than or equal to 710 amperes during the initial 6 seconds of the test.
    b) Batteries 2PA and 2PB greater than 182 amperes during the next 42 seconds of the test.

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

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

This item becomes due on 5/10/96 and requires an extension of 190 days to reach the end of the refueling outage. The justification for this extension is provided in Enclosure 38.

SR 4.8.2.2 requires at least the required battery and chargers be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1. Only the applicable portions of 4.8.2.1 discussed above will require extension of the SRs through the Refueling outage. The extension dates listed above for the 4.8.2.1 SRs include the time required to the end of RFO5. This SR does not require any additional extension other than the ones requested for section 4.8.2.1.

Electrical Equipment Protective Devices

As stated in Technical Specification Bases Sections 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.

SR 4.8.4.2.a.1.a requires each of the primary containment penetration conductor overcurrent protective devices shown in Technical Specification Table 3.8.4.2-1 be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by performing a channel calibration of the associated 4.16-kV circuit protective relays. This item becomes due on 6/24/96 and requires an extension of 103 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 39.

SR 4.8.4.2.a.1.b requires each of the primary containment penetration conductor overcurrent protective devices shown in Technical Specification Table 3.8.4.2-1 be demonstrated operable at least once per 18 months (with a maximum allowable surveillance interval extension of 4.5 months per TS 4.0.2) by an integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and overcurrent control circuits function as designed. This item becomes due on 6/24/96 and requires an extension of 103 days to reach October 5, 1996. The justification for this extension is provided in Enclosure 39.

Revised Technical Specifications

The requested revisions to the Technical Specifications identified above are shown on Attachment 3. The revisions are one-time only extensions of the surveillance intervals to allow the surveillance testing to be performed during the fifth refueling outage scheduled to begin September 27, 1996. A revision is also requested to reset the "N times 18 months" cumulative surveillance interval for various response time testing to the dates performed during RFO5. Also there is a one time only change to SR 4.7.2.1.h to clearly state that the next time this surveillance is due will be by June 1, 1998. A note is also proposed for an addition to SR 4.6.1.2.i to allow SR 4.0.2 to apply for extending Appendix J requirements for RFO5 only. The authority to delay such tests is being sought on a one-time only basis.

ENCLOSURE 1
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.1.3.1.4.a
SCRAM DISCHARGE VOLUME VENT AND DRAIN VALVE OPERABILITY

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months: "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30937P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, 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 SDV Vent and Drain Valves are tested to verify operation of the fail safe actuators and are full stroke exercised to the close position once per 92 days by the surveillance program, thereby verifying that the valves are capable of closing and opening. Stroke times are measured on the full stroke exercise test and verified to be less than 15 seconds. Therefore the logic signal generated by the RPS on an actual scram to close the valves within 30 seconds and open the valves is the only remaining portion of the SR which requires extension. In that this logic is the subject of the logic system functional discussion above, and the valves are cycled periodically during the operating cycle, the extension of the surveillance intervals is justified.

To verify this conclusion, a historical search of the 18 month surveillance tests for each valve for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a component would be unavailable. No failures were identified in this review. The review also evaluated the historical performance of the valves for the last two refueling outages and found that performance was within acceptable limits. The results of this review supports the above conclusion that the impact on safety, if any, is small as a result of this one-time extension in the subject surveillance interval.

Based on the above evaluation, it is concluded that the impact on vent and drain valve availability, if any, is small as a result of the one-time surveillance interval change. Therefore the one-time extension is justified.

ENCLOSURE 2
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.1.3.5
SCRAM ACCUMULATOR

Justification for extending SR 4.1.3.5.b.2

The scram accumulator check valves are only required to maintain pressure assuming that no control rod drive pump is operating. A review of check valve performance for the last two refueling outages has indicated no failures to maintain pressure. Therefore the probability that the check valve would fail at a time when no control rod drive pump was available is considered low.

Based on the above discussion, including the historical failure review the requested extension is justified.

ENCLOSURE 3
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.1.5
STANDBY LIQUID CONTROL SYSTEM OPERABILITY

Justification for extending Items d.1, d.2, and d.3

An evaluation of the system functional testing performed during the last two refueling outages indicated no failures to meet acceptance criteria. In addition, explosive charge continuity and valve position verification are performed monthly and pump testing is performed quarterly in accordance with the plant surveillance program. The explosive valves (squibs) are purchased in lots with samples tested prior to installation. The triggers for both explosive valves have sufficient service life for the requested extension.

This SR tests the relief valves that provide system overpressure protection from the discharge of the positive displacement pumps. The SLCS is designed with two redundant loops. If one relief valve 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. In addition, the current TS surveillance frequency significantly exceeds the ASME XI/OM-1 requirements. The OM-1 requirement is that all valves of a type be tested within 10 years with a minimum of 20% tested within any 48 months. The Fermi 2 IST program requires a 5 yea period on these valves. The SR delineates the testing requirements to ensure operability of the SLCS.

The SLCS is a backup to the control rods. The system is also designed with a redundant loop. In addition, functional testing of the SLCS pump is performed on a quarterly basis throughout the operating cycle and the charges in the explosive valves are monitored for circuit continuity in the control room. An alarm sounds when the circuit is opened. During the functional testing system pressure is raised to 1215 psig. Since the SLCS relief valves are exposed to this pressure, any significant relief valve setpoint drift would be detected during the performance of this more frequent test performed during the operating cycle. The overall impact on system availability, if any, of extending the operating cycle on this one-time basis is small. This conclusion is based on the fact that more frequent testing is performed, there are control room alarms which verify circuit continuity in the explosive valves, and active component redundancy.

A review of the history of the SR results indicates that previous testing of the relief valves was performed in situ with a wide tolerance because of the difficulty in determining the lifting pressure with a positive displacement pump. The RFO4 test was the first test where the valves were removed and bench tested. One of the valves passed and the other relief valve failed high though it did lift. The function of the SLCS important to reactor safety is to inject sodium penteborate at high reactor pressure.

Failure of the relief valve high has no impact on the operability of the SLCS to perform its safety function. Further review of the calibration history indicates that the valves had not required adjustment since initial plant startup prior to this bench test. This result demonstrates that the relief valves can be installed for a significant time period without having an adverse effect on the SLCS operability. No additional failures were identified by the historical review.

The temperature of the sodium pentaborate solution and the temperature of the SLCS pumps suction piping is verified once every 24 hours. The minimum tank and piping temperature requirement for operation is 48 degrees. Therefore, it is highly unlikely that the piping would become blocked.

The proposed one-time Technical Specification SR extension has little or no effect on the Standby Liquid Control System availability since the routine surveillances which include pump testing, temperature monitoring, and explosive valve continuity verification provide assurance of system operability. Therefore, the requested extension is justified.

ENCLOSURE 4
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.1.1 Table 4.3.1.1-1
REACTOR PROTECTION SYSTEM INSTRUMENTATION

Justification for extending Item 3, 4 and 7

The Reactor Vessel Dome Pressure, Reactor Vessel Level - Level 3 and Drywell Pressure setpoints were analyzed using the GE Instrument Setpoint Methodology, NEDC-31336. The specific calculation for the Reactor Vessel Dome Pressure instrument channels was submitted to the NRC during the power uprate review that was accepted by NRC Safety Evaluation Report dated September 9, 1992 on Fermi-2 docket. The GE Instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket. In addition, Rosemount published a report in February 1990, "30 Month Stability Specification For Rosemount 1152, 1153, 1154 Pressure Transmitters" (Rosemount Report D89000126, Revision A) which was accepted by NRC Safety Evaluation Report dated 8/2/93 on Peach Bottom Atomic Power Station docket. This report supported the extension of the calibration interval for the transmitters from 18 months to 30 months based on a reduction in the drift allowance.

The Fermi 2 design calculations contain a drift allowance for Rosemount 1153 transmitter equal to at least three consecutive six month vendor drift intervals. The previously published model 1153 six month drift interval value is of the same order of magnitude as the currently published thirty month drift interval value. Therefore the calculations contain at least 30 months drift allowance. Associated loop trip units are checked on a quarterly basis. This check confirms that the trip units are maintained within allowances. As a result, the existing Fermi-2 design calculation includes an adequate allowance for drift and the extension has no potential impact on the plant safety analyses, i.e., no analytic limit changes are required. Plant calibration data was also reviewed and found to be within the design allowances. Therefore, the extension of the surveillance interval is justified.

Justification for extending Item 5

Limit switches are mechanical devices that require mechanical adjustment only; drift is not applicable to these devices. Therefore, an increase in the surveillance interval to accommodate a one time extended operating cycle does not affect the limit switches with respect to drift. A historical search of the 18 month surveillance tests for these switches for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests, each failed or partially failed test was to be identified, reviewed and evaluated. The purpose of this evaluation was to demonstrate that the

increased calibration surveillance interval would not increase the period an instrument would be unavailable. No failures were identified in this review. Therefore, the requested extension is justified

Justification for extending Item 6

For the Main Steam Line (MSL) Radiation - High instruments, correct operation is confirmed by a quarterly channel functional check which does not include the detector, and a channel check every 12 hours. These tests will identify any potential drift of these instruments (except for the detector and cable) during this period of time. The quarterly and twice daily surveillance tests, monitoring by plant staff, and the fact that the ion chamber detectors and cable are not considered susceptible to drift ensure that there is no potential impact on the subject instrument availability by allowing the additional time interval of source calibration of the existing radiation monitors.

The surveillance allowance for the MSL Radiation monitoring instrument is 30 percent of span. A review of the surveillance tests indicates that drift for the MSL Radiation monitors was 19 percent for the worst case condition. A further review of the MSL radiation monitors calibration history indicates that the detectors have never been adjusted to compensate for drift. Therefore, the drift history for these components documents the ability of the MSL monitors to perform for multiple cycles without adjustment and with performance remaining within expected bounds.

A historical search of the 18 month surveillance tests for the MSL radiation monitors for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests, each failed or partially failed test was to be reviewed and evaluated. The purpose of this evaluation was to demonstrate that the increased calibration surveillance interval would not increase the period an instrument would be unavailable. One failure was identified in this review. However the detailed evaluation determined that this failure (test cable for a spare monitor) would not have affected device performance. Therefore, the requested extension is justified.

The instruments used for MSL Radiation High are identical to the instruments approved by the NRC for a permanent 24 month cycle extension in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months.

Based on the above discussion, the NRC's prior approval of extensions for identical instrumentation and the fact that historical data for the radiation monitors shows that recalibration has not been required for a period of time greater than 24 months, it is concluded that the impact on instrument availability, if any, is small as a result of the one-time surveillance interval extension and the requested extension is justified.

ENCLOSURE 5
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.1.2
LOGIC SYSTEM FUNCTIONAL TEST AND SIMULATED AUTOMATIC OPERATION
REACTOR PROTECTION SYSTEM INSTRUMENTATION

The equipment and components used in the design of systems requiring LSFT was chosen based on reliability as demonstrated by years of service in both the nuclear and non-nuclear industries. A review of all of the surveillance test history for the subject SR was performed to detect evidence of excessive random equipment or component failure rates and no such evidence was found. Based on this review and the redundant equipment in each of the subject systems, it was concluded that the impact of reducing the LSFT frequency on system availability is insignificant. This conclusion was supported by two independent studies. The first study was completed for the BWR Owners Group in 1989 and the second was completed for the Nuclear Regulatory Commission (NRC) in 1988. An evaluation of LSFT completed by General Electric Co. for the BWR Owners Group (BWR Owners Group Report EAS 25-0489, Evaluation of Logic System Functional Testing methods, July 1989) has shown that circuit unavailability due to removal from service for testing at power, is the largest contributor to total circuit unavailability. This study analyzed the effect of different surveillance intervals or various basic logic configurations with the net result being that for most logic configurations changing from a 6 month to a refuel cycle surveillance improves the total circuit availability. With the exception of ADS, in no case is the unavailability increased by an appreciable amount. This conclusion is based on the determination that for the subject system the unavailability is only increased appreciably after increasing the test interval beyond 90 months. For ADS, unavailability caused by logic system failure (assuming a test interval of 36 months) remains two orders of magnitude less than currently accepted unavailability caused by valve actuation solenoid failure. Therefore, the additional unavailability due to increased test interval is insignificant in comparison to the existing system unavailability.

A historical search of the 18 month surveillance tests for these components for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests, each failed or partially failed test was to be reviewed and evaluated. The purpose of this evaluation was to demonstrate that the increased functional test surveillance interval would not increase the period a function would be unavailable. No failures of logic system components were identified in this review.

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

> "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 system, 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 evaluation above is applicable to Fermi 2, therefore, the surveillance interval extension is justified.

The Reactor Mode Switch as well as most manual actuation and manual inhibit switches are considered to be system logic components since their failure modes and generally failure history are more closely related to circuit components than standard mechanical components. Based on this classification the justification for the Reactor Mode Switch is presented as follows.

Table 4.3.1.1-1 Item 11

This test verifies the Reactor Mode Switch Shutdown Scram and Scram Bypass logic associated with the RPS function properly. The reactor mode switch scram function is not required to protect the fuel or nuclear boundaries. The RPS functions independently from the mode switch. Based on the above discussion and the reliability of logic it is concluded that the impact, if any, on the mode switch availability, is small as a result of this change. A review of the history of the SR results demonstrates that there is no evidence of any failures which would invalidate this conclusion.

ENCLOSURE 6
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.1.3
RESPONSE TIME TESTING
REACTOR PROTECTION SYSTEM INSTRUMENTATION

Regulatory Guide 1.118 (Revision 2) states:

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

On January 14,1994, the BWR Owner's Group submitted a Licensing Topical Report (LTR) prepared by the General Electric Company, NEDO-32291, "System Analyses For Elimination Of Selected Response Time Testing Requirements", January 1994 for NRC review. The NRC issued the Safety Evaluation Report for this topical report December 28, 1994. Based on the review of the GE Topical Report the NRC concluded that significant degradation of instrument response times, i.e., delays greater than about 5 seconds, can be detected during the performance of other surveillance tests, principally calibration, if properly performed. The NRC Staff concluded that response time testing could be eliminated from Technical Specifications for the selected instrumentation identified in the topical report and accepted NEDO-32291 for reference in license amendment applications for all boiling water reactors. Detroit Edison's Fermi 2, as one of the lead plants for the Topical Report, has verified the applicability of the generic analysis of NEDO-32291. The information contained in the Topical Report justifies a one-time extension of the surveillance requirement intervals for response time testing.

Certain functions were not included in the GE Topical Report as evaluated for the elimination of response time testing. These functions are APRM flow biased simulated thermal power high, APRM fixed neutron flux high, Reactor Vessel Steam Dome Pressure High (trip unit and logic only), Reactor Vessel Low Water Level Level 3 (trip unit and logic only), Main Steam Line Isolation Valve closure, Turbine Stop Valve closure, and Turbine Control Valve fast closure. The following justification is provided for instruments supporting these functions:

The extension would have no substantial measurable effect on plant safety because:

a. There are redundant sensors that can initiate the scram operation.

b. Redundancy exists for every individual instrument channel within each trip function. Only one of the channels per trip system is required to be tested during each surveillance period.

c. The instrumentation failure probability is a very small fraction of the total control rod insertion (scram failure probability).

d. There are several redundant and diverse instrument channels which can detect and generate a scram signal (e.g., flux, pressure, etc.).

e. The failure of instrumentation in the sluggish responding mode is a small fraction of its overall failure.

Extension of the Peach Bottom Atomic Power Station Units 2 and 3 surveillance intervals for RPS response time testing from 18 to 24 months was accepted by the NRC in the Safety Evaluation Report dated August 2, 1993. The justification cited above is very similar to that provided in the NRC Safety Evaluation Report 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."

A historical search of the 18 month surveillance for response time testing for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a function would be unavailable. No RPS response time test failures were identified in this review.

Based on the above, a one-time extension of the RPS Instrumentation SR is justified.

ENCLOSURE 7
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.2.1 Table 4.3.2.1-1
ISOLATION ACTUATION INSTRUMENTATION

Justification for extending Items 1.a.1, 1.a.2, 1.a.3, 1.b, 2.e, 5.a, 6.a, and 6.b

The Reactor Vessel Level - Level 1, 2, 3 and 8 and drywell pressure setpoints were analyzed using the GE Instrument Setpoint Methodology, NEDC-31336. The method of calculation is similar to the calculations submitted to the NRC during the power uprate review that was accepted by NRC Safety Evaluation Report dated September 9, 1992 on Fermi-2 docket. The GE Instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket. In addition, Rosemount published a report in February 1990, "30 Month Stability Specification For Rosemount 1152, 1153, 1154 Pressure Transmitters" (Rosemount Report D89000126, Revision A) which was accepted by NRC Safety Evaluation Report dated August 2, 1993 on Peach Bottom Atomic Power Station docket. This report supported the extension of the calibration interval for the transmitters from 18 months to 30 months based on a reduction in the drift allowance.

The Fermi 2 design calculations contain a drift allowance for Rosemount 1153 transmitter equal to at least three consecutive six month vendor drift intervals. The previously published model 1153 six month drift interval value is of the same order of magnitude as the currently published thirty month drift interval value. Therefore the calculations contain at least 30 months drift allowance. Associated loop trip units are checked on a quarterly basis. This check confirms that the trip units are maintained within allowances. As a result, the existing Fermi-2 design calculation includes an adequate allowance for drift and the extension has no potential impact on the plant safety analyses, i.e., no analytic limit changes are required. Plant calibration data was also reviewed and found to be within the design allowances. Therefore, the extension of the surveillance interval is justified.

Justification for extending Item 1.c.1

For the Main Steam (MSL) Line Radiation - High instruments, correct operation (except the detector) is confirmed by a quarterly channel functional check and a channel check every 12 hours. These tests will identify any potential drift of these instruments (except for the detector and cable) during this period of time. The quarterly and twice daily surveillance tests, monitoring by plant staff, and the fact that the ion chamber detectors and cable are not considered susceptible to drift ensure that there is no potential impact on the subject instrument availability by allowing the additional time interval of source calibration of the existing radiation monitors.

The surveillance allowance for the MSL radiation instrument is 30 percent of span. A review of the surveillance tests indicates that Drift for the MSL radiation monitors was 19 percent for the worst case condition. A further review of the MSL radiation monitors calibration history indicates that the detectors have never been adjusted to compensate for drift. Therefore, the drift history for these components documents the ability of the MSL monitors to perform for multiple cycles without adjustment and with performance remaining within expected bounds.

The instruments used for MSL Radiation High are identical to the instruments approved by the NRC for a permanent 24 month cycle extension in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months.

Based on the above discussion, the NRC's prior approval of identical instrumentation and the fact that historical data for the radiation monitors shows that recalibration has not been required for a period of time greater than 24 months, it is concluded that the impact on instrument availability, if any, is small as a result of the one-time surveillance interval extension and the requested extension is justified

Justification for extending Items 1.c.2 and 1.e

The Main Steam Line Pressure and Condenser Pressure setpoints were analyzed for an extended surveillance interval using the GE Instrument Setpoint Methodology, NEDC-31336. The method of calculation is similar to the calculations submitted to the NRC during the power uprate review that was accepted by NRC Safety Evaluation Report dated September 9, 1992 on Fermi-2 docket. The GE Instrument Setpoint Methodology, NEDC-31336 has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket.

The evaluation conservatively considered six consecutive 6 months drift intervals for the Rosemount 1151 transmitters. Therefore the calculations contain at least 36 months drift allowance. Associated loop trip units are checked on a quarterly basis. This check confirms that the trip units are maintained within allowances. As a result, the existing Fermi-2 setpoint includes an adequate allowance for drift and the extension has no potential impact on the plant safety analyses, i.e., no analytic limit changes are required. Plant calibration data was also reviewed and found to be within the design allowances. Therefore, the extension of the surveillance interval is justified.

Justification for extending Item 1.d and 1.f

The Main Steam Line Tunnel and Turbine Building Area Temperature setpoints were analyzed for an extended surveillance interval using the GE Instrument Setpoint Methodology, NEDC-31336. The method of calculation is similar to the calculations submitted to the NRC during the power uprate review that was accepted by NRC Safety Evaluation Report dated September 9, 1992 on Fermi-2 docket. The GE setpoint methodology has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket.

The Main Steam Line Tunnel and Turbine Building Area Temperature calibration procedure performs an operability check of the RTD's and calibrates the remaining loop components (Tech Spec Definition 1.4 states "...Calibration of instrument channels with resistance temperature detectors (RTD) or Thermocouple sensors shall consist of verification of operability of the sensing element and adjustment, as necessary, of the remaining adjustable devices in the channel."). The loop operability is also verified in shiftly channel checks. Associated loop trip units are checked on a quarterly basis. This check confirms that the trip units are maintained within allowances. As a result, the Fermi-2 setpoint includes an adequate allowance for drift and has no potential impact on plant safety analyses, i.e., no analytic limit changes are required. Plant calibration data was reviewed and found to be within the design allowances. Therefore, the extension of the surveillance interval is justified.

Justification for extending Item 3.a.1 and 3.a.2

The RCIC Steam Line Flow setpoints were analyzed for an extended surveillance interval using the GE Instrument Setpoint Methodology, NEDC-31336. The method of calculation is similar to the calculations submitted to the NRC during the power uprate review that was accepted by NRC Safety Evaluation Report dated September 9, 1992 on Fermi-2 docket. The GE Instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket.

The evaluation conservatively considered six consecutive 6 months drift intervals for the Rosemount 1151 transmitters and found the results to be acceptable. A time delay relay accuracy/drift allowance of two times reference accuracy was conservatively evaluated and was found to be acceptable. Associated loop trip units are checked on a quarterly basis. This check confirms that the trip units are maintained within allowances. As a result, the existing Fermi-2 setpoints include an adequate allowance for drift and have no potential impact on plant safety analyses, i.e., no analytic limit changes are required. Plant calibration data was reviewed and found to be within the design allowances. Therefore, the extension of the surveillance interval is justified.

Justification for extending Items 4.a.1 and 4.a.2

The HPCI Steam Line Flow setpoints were analyzed for an extended surveillance interval using the GE Instrument Setpoint Methodology, NEDC-31336. The method of calculation is similar to the calculations submitted to the NRC during the power uprate review that was accepted by NRC Safety Evaluation Report dated September 9, 1992 on Fermi-2 docket. The GE Instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket. In addition, Rosemount published a report in February 1990, "30 Month Stability Specification For Rosemount 1152, 1153, 1154 Pressure Transmitters" (Rosemount Report D89000126, Revision A) which was accepted by NRC Safety Evaluation Report dated 8/2/93 on Peach Bottom Atomic Power Station docket. This report supported the extension of the calibration interval for the transmitters from 18 months to 30 months based on a reduction in the drift allowance.

The Fermi 2 design calculations contain a drift allowance for Rosemount 1153 transmitter equal to at least three consecutive six month vendor drift intervals. The previously published model 1153 six month drift interval value is of the same order of magnitude as the currently published thirty month drift interval value. Therefore the calculations contain at least 30 months drift allowance. Associated loop trip units are checked on a quarterly basis. This check confirms that the trip units are maintained within allowances. A time delay relay accuracy/drift allowance of two times reference accuracy was conservatively evaluated and was found to be acceptable. As a result, the existing Fermi-2 setpoints include an adequate allowance for drift and have no potential impact on plant safety analyses, i.e., no analytic limit changes are required. Plant calibration data was reviewed and found to be within the design allowances. Therefore, the extension of the surveillance interval is justified.

A historical search of the 18 month surveillance tests for these instruments for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased calibration surveillance interval would not increase the period a component would be unavailable. Failures of components associated with the extended interval were identified; however, the type of failures would be detected during tests that are performed more frequently. Therefore, these failures have no effect on component availability for the extended operating cycle. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

ENCLOSURE 8
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.2.2
LOGIC SYSTEM FUNCTIONAL TEST AND SIMULATED AUTOMATIC OPERATION
ISOLATION ACTUATION INSTRUMENTATION SR 4.3.2.:

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

> "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 system, 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."

Justification for extending Table 4.3.2.1-1 Items 1.h, 2.d, 2.g, 4.e, and 5.c

The Reactor Mode Switch as well as most manual actuation and manual inhibit switches are considered to be system logic components since their failure modes and generally failure history are more closely related to circuit components than standard mechanical components. Based on this classification the justification for the extension of the surveillance for manual initiation and manual inhibit switches is identical to the above discussion.

The evaluation above is applicable to Fermi 2. An historical search of the 18 month surveillance tests for these components for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased functional test surveillance interval would not increase the period a function would be unavailable. This evaluation determined that one clamp-on valve position indication and two limit switch valve position indication failures had occurred during the last two refueling outages. The evaluation also determined that one valve had exceeded its stroke time during the isolation manual initiation and that one relay did not de-energize within allowable time limits. The valve position indication failures were repaired and retested satisfactorily, additionally this type of failure does not degrade the valves' safety function. The valve stroke time failure was corrected by spring adjustment. The valve has been stroked successfully three times

since the failure. This valve is also tested on a more frequent basis than the 18 month surveillance. Therefore, the failure would have been detected by this more frequent testing. The relay closure failure was associated with a relay type (GE model CR 120) which has been evaluated under a Fermi 2 Deviation Event Report and a Preventative Maintenance task assigned to replace all of this type of relay. The failed relay was replaced and retested satisfactory. The cycling of the relay, but not all relay contact positions, is also verified at least every 13 weeks by other functional tests. Additional failures of components associated with the extended interval were identified; however, the type of failures would be detected during tests that are performed more frequently. Therefore these failures have no effect on component availability for the extended operating cycle. Based on the previous discussion, the historical failure review, and the small impact on safety function availability, the requested extension is justified.

ENCLOSURE 9
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.2.3
RESPONSE TIME TESTING
ISOLATION ACTUATION INSTRUMENTATION

Regulatory Guide 1.118 (Revision 2) states:

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

On January 14,1994, the BWR Owner's Group submitted a Licensing Topical Report (LTR) prepared by the General Electric Company, NEDO-32291, "System Analyses For Elimination Of Selected Response Time Testing Requirements", January 1994 for NRC review. The NRC issued the Safety Evaluation Report for this topical report December 28, 1994. Based on the review of the GE Topical Report the NRC concluded that significant degradation of instrument response times, i.e., delays greater than about 5 seconds, can be detected during the performance of other surveillance tests, principally calibration, if properly performed. The NRC Staff concluded that response time testing could be eliminated from Technical Specifications for the selected instrumentation identified in the topical report and accepted NEDO-32291 for reference in license amendment applications for all boiling water reactors. Detroit Edison, as one of the lead plants for the Topical Report, has verified the applicability of the generic analysis of NEDO-32291. The information contained in the Topical Report justifies a one-time extension of the surveillance requirement intervals for response time testing.

Some functions were not included in the GE Topical Report as evaluated for the elimination of response time testing. These functions are primary containment isolation reactor vessel low water level level 1 (trip unit and logic only) and primary containment isolation main steam line flow high (trip unit and logic only). The following justification is provided for instruments supporting these functions:

The extension would have no substantial measurable effect on plant safety because:

a. There are redundant trip units and logics that can initiate the isolation.

b. Redundancy exists for every individual instrument channel within each of these isolation functions. Only one of the redundant channels per trip system is required to be tested during each surveillance period.

c. The instrumentation failure probability is a very small fraction of the total primary containment isolation function failure probability.

d. There are several redundant and diverse instrument channels which can detect and generate an isolation signal.

e. The failure of instrumentation in the sluggish responding mode is a small fraction of its overall failure.

Extension of the Peach Bottom Atomic Power Station Units 2 and 3 surveillance intervals for response time testing from 18 to 24 months was accepted by the NRC in the Safety Evaluation Report dated August 2, 1993. The justification cited above is very similar to that provided in the NRC Safety Evaluation Report.

A historical search of the 18 month surveillance for response time testing for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a function would be unavailable. No Isolation instrumentation response time test failures were identified in this review.

Based on the above, a one-time extension of the instrument response time testing surveillance interval is justified.

ENCLOSURE 10
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.3.1 Table 4.3.3.1-1
EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

Justification for extending Item 1.a, 1.b, 1.c, 2.a, 2.b, 2.c, 2.d, 2.e, 2.f, 2.g, 3.a, 3.b, 3.e, 4.a, 4.f

The Reactor Vessel Level - Levels 1, 2, and 8, Drywell Pressure, Reactor Vessel Dome Pressure, RHR (LPCI Mode) Riser Differential Pressure, and Recirculation Pump Differential Pressure setpoints were evaluated using the GE Instrument Setpoint Methodology, NEDC-31336. The method of calculation is similar to the calculations submitted to the NRC during the power uprate review that was accepted by NRC Safety Evaluation Report dated September 9, 1992 on the Fermi-2 docket. The GE Instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket. In addition, Rosemount published a report in February 1990, "30 Month Stability Specification For Rosemount 1152, 1153, 1154 Pressure Transmitters" (Rosemount Report D89000126, Revision A) which was accepted by NRC Safety Evaluation Report dated 8/2/93 on the Peach Bottom Atomic Power Station docket. This report supported the extension of the calibration interval for the transmitters from 18 months to 30 months based on a reduction in the drift allowance.

The Fermi 2 design calculations contain a drift allowance for Rosemount 1153 transmitters equal to at least three consecutive six month vendor drift intervals. The previously published model 1153 six month drift interval value is of the same order of magnitude as the currently published thirty month drift interval value. Therefore, the calculations contain at least 30 months drift allowance. Associated loop trip units are checked on a quarterly basis. This check confirms that the trip units are maintained within allowances. As a result, the existing Fermi-2 design calculation includes an adequate allowance for drift and the extension has no potential impact on the plant safety analyses, i.e., no analytic limit changes are required. Plant calibration data was also reviewed and found to be within the design allowances. Therefore, the extension of the surveillance interval is justified.

Justification for extending Item 4.h

The ECCS Drywell Pressure - High Bypass Timer setpoint was analyzed for an extended surveillance interval using the GE Instrument Setpoint Methodology, NEDC-31336. The method of calculation is similar to the calculations submitted to the NRC during the power uprate review that was accepted by NRC Safety Evaluation Report dated September 9, 1992 on the Fermi-2 docket. The GE Instrument Setpoint Methodology,

NEDC-31336 has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket.

A setpoint evaluation was performed that analyzed an accuracy/drift allowance of two times the vendor accuracy and found the results to be acceptable. Vendor performance specifications do not define a separate drift allowance and the setpoint is not considered to be dependent upon the surveillance interval; however, an evaluation was performed assuming that the increased interval would degrade the setpoint repeatability. As a result, the existing Fermi-2 setpoint includes an adequate allowance for drift. Plant calibration data was also reviewed and found to be within the design allowances. Therefore, the extension of the surveillance interval is justified.

An historical search of the 18 month surveillance tests for these instruments for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a function would be unavailable. This evaluation determined that one failure of a Drywell Pressure High transmitter to calibrate was caused by water in the transmitter sensing lines. The transmitter sensing line was drained, and the transmitter was retested. The transmitter was recalibrated satisfactorily. A Deviation Event Report (DER) was generated to document and evaluate the water in the transmitter. Other transmitters were investigated to determine if water was present. These investigations indicated that two other transmitter also contained water. The apparent source of the water was the previous response time testing of these transmitters with water and inadequate draining of the lines after the test was completed. Procedures were revised to correct the problem and prevent reoccurrence. Based on the previous discussion, the historical failure review, and the small impact on safety function availability, the requested extension is justified.

ENCLOSURE 11
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.3.2
LOGIC SYSTEM FUNCTIONAL TEST AND SIMULATED AUTOMATIC OPERATION
EMERGENCY CORE COOLING SYSTEM

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

> "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 system, 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."

Justification for extending Table 4.3.3.1-1 items 1.d, 2.h, 3.f, and 4.i

Most manual actuation and manual inhibit switches are consider to be system logic components since their failure modes and generally failure history are more closely related to circuit components than standard mechanical components. Based on this classification the justification for the extension of the surveillance for manual initiation and manual inhibit switches is identical to the above discussion.

The evaluation above is applicable to Fermi 2. A historical search of the 18 month surveillance tests for these instruments for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased functional test surveillance interval would not increase the period a function would be unavailable. This evaluation determined that one failure of a Drywell Pressure High transmitter to calibrate was caused by water in the transmitter. Additionally a failure of the Diesel Generator output breaker was caused by the Diesel Generator not reaching rated speed and voltage within sufficient time, and a RPV level 8 transmitter could not be calibrated due to failed components. The Dry well Pressure High transmitter failure is the same failure discussed in Enclosure 10. The root cause analysis for the Diesel Generator output breaker determined that the EDG did not reach rated speed and voltage in sufficient time for the breaker close permissive signal to allow timely breaker closure. The EDG governor was repaired to reduce

the time for the EDG to reach rated speed and voltage. The ability of the EDG to reach rated speed and voltage is tested on a more frequent basis (i.e. every 184 days per SR 4.8.1.1.2.a.4). While this test does not verify that the output breaker will close within the required time, this test does verify that the EDG reaches rated frequency and voltage in less than 10 seconds. Therefore this failure or a similar one, has no reliability implications for the extension of the surveillance because it would be detected by testing that is performed more frequently. The RPV Level 8 transmitter amplifier and calibration boards were replaced and the transmitter was recalibrated satisfactorily. This failure was determined not to have a time based mode. Therefore the surveillance interval extension will have no effect on availability of the component. Based on the previous discussion, the historical failure review, and the small impact on safety function availability, the requested extension is justified.

ENCLOSURE 12
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.3.3
RESPONSE TIME TESTING
EMERGENCY CORE COOLING SYSTEM

Regulatory Guide 1.118 (Revision 2) states:

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

On January 14,1994, the BWR Owner's Group submitted a Licensing Topical Report (LTR) prepared by the General Electric Company, NEDO-32291, "System Analyses For Elimination Of Selected Response Time Testing Requirements", January 1994 for NRC review. The NRC issued the Safety Evaluation Report for this topical report December 28, 1994. Based on the review of the GE Topical Report the NRC concluded that significant degradation of instrument response times, i.e., delays greater than about 5 seconds, can be detected during the performance of other surveillance tests, principally calibration, if properly performed. The NRC Staff concluded that response time testing could be eliminated from Technical Specifications for the selected instrumentation identified in the topical report and accepted NEDO-32291 for reference in license amendment applications for all boiling water reactors. Detroit Edison, as one of the lead plants for the Topical Report, has verified the applicability of the generic analysis of NEDO-32291. The information contained in the Topical Report justifies a one-time extension of the surveillance requirement intervals for response time testing.

The GE Topical Report only evaluated the elimination of sensor and trip unit response time testing. The system response time testing is still required. For the Core Spray System, High Pressure Coolant Injection and Low Pressure Coolant Injection pump, valve and flow testing is performed on a more frequent basis in accordance with the Inservice Testing Program during the operating cycle to ensure that this assumed ECCS function is available. Although this test does not ensure the response times for the ECCS initiation, this testing in combination with Emergency Diesel Generator Testing would indicate any significant system slow responses.

The extension would have no substantial measurable effect on plant safety because:

a. There are redundant ECCS systems powered by different methods which can perform the required safety functions.

b. The response time failure probability is a very small fraction of the total ECCS failure probability.

c Failure of ECCS components in the sluggish responding mode does not invalidate the components' ability to perform its safety function.

Extension of the Peach Bottom Atomic Power Station Units 2 and 3 surveillance intervals for response time testing from 18 to 24 months was accepted by the NRC in the Safety Evaluation Report dated August 2, 1993.

A historical search of the 18 month surveillance for response time testing for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a function would be unavailable. No ECCS response time test failures were identified in this review.

Based on the above, a one-time extension of the instrument and system response time testing surveillance interval is justified.

ENCLOSURE 13
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.4 Table 4.3.4-1
ATWS RECIRCULATION PUMP TRIP INSTRUMENTATION

Justification for extending Item 1

The Reactor Vessel Level - Levels 1, 2, and 8 setpoints were analyzed using the GE Instrument Setpoint Methodology, NEDC-31336. The method of calculation is similar to the calculations submitted to the NRC during the power uprate review that was accepted by NRC Safety Evaluation Report dated September 9, 1992 on Fermi-2 docket. The GE Instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket. In addition, Rosemount published a report in February 1990, "30 Month Stability Specification For Rosemount 1152, 1153, 1154 Pressure Transmitters" (Rosemount Report D89000126, Revision A) which was accepted by NRC Safety Evaluation Report dated 8/2/93 on the Peach Bottom Atomic Power Station docket. This report supported the extension of the calibration interval for the transmitters from 18 months to 30 months based on a reduction in the drift allowance.

The Fermi 2 design calculations contain a drift allowance for Rosemount 1153 transmitters equal to at least three consecutive six month vendor drift intervals. The previously published model 1153 six month drift interval value is of the same order of magnitude as the currently published thirty month drift interval value. Therefore, the calculations contain at least 30 months drift allowance. Associated loop trip units are checked on a quarterly basis. This check confirms that the trip units are maintained within allowances. As a result, the existing Fermi-2 design calculation includes an adequate allowance for drift and the extension has no potential impact on the plant safety analyses, i.e., no analytic limit changes are required. Plant calibration data was also reviewed and found to be within the design allowances. Therefore, the extension of the surveillance interval is justified.

Justification for extending Item 2

The high Reactor Vessel pressure setpoints were analyzed using the GE Instrument Setpoint Methodology, NEDC-31336. The specific calculation for Reactor pressure instrument channels was submitted to the NRC during the power uprate review that was accepted by NRC Safety Evaluation Report dated September 9, 1992 on Fermi-2 docket. The GE Instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket. In addition, Rosemount published a report in February 1990, "30 Month Stability Specification For Rosemount 1152, 1153, 1154 Pressure Transmitters" (Rosemount

Report D89000126, Revision A) which was accepted by NRC Safety Evaluation Report dated August 2, 1993 on the Peach Bottom Atomic Power Station docket. This report supported the extension of the calibration interval for the transmitters from 18 months to 30 months based on a reduction in the drift allowance.

The Fermi 2 design calculations contain a drift allowance for Rosemount 1153 transmitter equal to at least three consecutive six month vendor drift intervals. The previously published model 1153 six month drift interval value is of the same order of magnitude as the currently published thirty month drift interval value. Therefore the calculations contain at least 30 months drift allowance. Associated loop trip units are checked on a quarterly basis. This check confirms that the trip units are maintained within allowances. As a result, the existing Fermi-2 design calculation includes an adequate allowance for drift and the extension has no potential impact on the plant safety analyses, i.e., no analytic limit changes are required. Plant calibration data was also reviewed and found to be within the design allowances. Therefore, the extension of the surveillance interval is justified.

A historical search of the 18 month surveillance tests for these instruments for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased calibration surveillance interval would not increase the period a component would be unavailable. Failures of components associated with the extended interval were identified; however, the type of failures would be detected during tests that are performed more frequently. Therefore, these failures have no effect on component availability for the extended operating cycle. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

ENCLOSURE 14
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.4.2
LOGIC SYSTEM FUNCTIONAL TEST AND SIMULATED AUTOMATIC OPERATION
ATWS RECIRCULATION PUMP TRIP INSTRUMENTATION

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

> "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 system, 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."

A historical search of the 18 month surveillance tests for these components for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a component would be unavailable. Failures of components associated with the extended interval were identified; however, the type of failures would be detected during tests that are performed more frequently. Therefore, these failures have no effect on component availability for the extended operating cycle. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

ENCLOSURE 15
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.5.1 Table 4.3.5.1-1
REACTOR CORE ISOLATION COOLING INSTRUMENTATION

Justification for extending Item a and b

The Reactor Vessel Level - Levels 1, 2, and 8 setpoints were analyzed using the GE Instrument Setpoint Methodology, NEDC-31336. The method of calculation is similar to the calculations submitted to the NRC during the power uprate review that was accepted by NRC Safety Evaluation Report dated September 9, 1992 on Fermi-2 docket. The GE Instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket. In addition, Rosemount published a report in February 1990, "30 Month Stability Specification For Rosemount 1152, 1153, 1154 Pressure Transmitters" (Rosemount Report D89000126, Revision A) which was accepted by NRC Safety Evaluation Report dated 8/2/93 on the Peach Bottom Atomic Power Station docket. This report supported the extension of the calibration interval for the transmitters from 18 months to 30 months based on a reduction in the drift allowance.

The Fermi 2 design calculations contain a drift allowance for Rosemount 1153 transmitter equal to at least three consecutive six month vendor drift intervals. The previously published model 1153 six month drift interval value is of the same order of magnitude as the currently published thirty month drift interval value. Therefore, the calculations contain at least 30 months drift allowance. Associated loop trip units are checked on a quarterly basis. This check confirms that the trip units are maintained within allowances. As a result, the existing Fermi-2 design calculation includes an adequate allowance for drift and the extension has no potential impact on the plant safety analyses, i.e., no analytic limit changes are required. Plant calibration data was also reviewed and found to be within the design allowances. Therefore, the extension of the surveillance interval is justified.

A historical search of the 18 month surveillance tests for these instruments for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased calibration surveillance interval would not increase the period a component would be unavailable. Failures of components associated with the extended interval were identified; however, the type of failures would be detected during tests that are performed more frequently. Therefore, these failures have no effect on component availability for the extended operating cycle. Based on the previous discussion, the historical failure review, and the small impact on safety function availability, the requested extension is justified.

ENCLOSURE 16
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.5.2
LOGIC SYSTEM FUNCTIONAL TEST AND SIMULATED AUTOMATIC OPERATION
REACTOR CORE ISOLATION COOLING

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

> "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 system, 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."

A historical search of the 18 month surveillance tests for these components for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a component would be unavailable. Failures of components associated with the extended interval were identified; however, the type of failures would be detected during tests that are performed more frequently. Therefore, these failures have no effect on component availability for the extended operating cycle. Based on the previous discussion, the historical failure review, and the small impact on safety function availability, the requested extension is justified.

ENCLOSURE 17
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.6 Table 4.3.6-1
CONTROL ROD BLOCK INSTRUMENTATION

Justification for extending Item 5.b and 7

The Reactor Mode Switch as well as most manual actuation and manual inhibit switches are considered to be system logic components since their failure modes and generally failure history are more closely related to circuit components than standard mechanical components (i.e., pumps and valves). Based on this classification the justification for the extension of the surveillance for manual initiation and manual inhibit switches is identical to the Logic System Functional Test discussion provided in Enclosure 16. A review of the surveillance history during RFO3 and RFO4 for these components did not indicate any failures.

ENCLOSURE 18
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.7.4.1 Table 4.3.7.4-1
REMOTE SHUTDOWN MONITORING SYSTEM INSTRUMENTATION

Justification for extending Item 1

The Remote Shutdown Reactor Pressure monitoring instrumentation was analyzed using the GE Instrument Setpoint Methodology, NEDC-31336. The GE Instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket. In addition, Rosemount published a report in February 1990, "30 Month Stability Specification For Rosemount 1152, 1153, 1154 Pressure Transmitters" (Rosemount Report D89000126, Revision A) which was accepted by NRC Safety Evaluation Report dated 8/2/93 on Peach Bottom Atomic Power Station docket. This report supported the extension of the calibration interval for the transmitters from 18 months to 30 months based on a reduction in the drift allowance.

The Fermi 2 design calculations contain a drift allowance for Rosemount 1152 transmitter equal to at least three consecutive six month vendor drift intervals. The previously published model 1152 six month drift interval value is of the same order of magnitude as the currently published thirty month drift interval value. Therefore, the calculations contain at least 30 months drift allowance. The transmitter provides input to a recorder. Channel check is performed monthly. This check confirms that the indicating channel performs within allowances. As a result, the existing Fermi-2 design calculation includes an adequate allowance for drift and the extension has no potential impact on the plant safety analyses. Plant calibration data was also reviewed and found to be within the design allowances. Therefore, the extension of the surveillance interval is justified.

Justification for extending Item 2

The Reactor Vessel Level monitoring instrumentation was analyzed using the GE Instrument Setpoint Methodology, NEDC-31336. The GE Instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket. In addition, Rosemount published a report in February 1990, "30 Month Stability Specification For Rosemount 1152, 1153, 1154 Pressure Transmitters" (Rosemount Report D89000126, Revision A) which was accepted by NRC Safety Evaluation Report dated 8/2/93 on Peach Bottom Atomic Power Station docket. This report supported the extension of the calibration interval for the transmitters from 18 months to 30 months based on a reduction in the drift allowance.

The Fermi 2 design calculations contain a drift allowance for Rosemount 1153 transmitter equal to at least three consecutive six month vendor drift intervals. The

previously published model 1153 six month drift interval value is of the same order of magnitude as the currently published thirty month drift interval value. Therefore, the calculations contain at least 30 months drift allowance. The transmitter provides input to an indicator. Channel check is performed monthly. This check confirms that the indicating channel performs within allowances. As a result, the existing Fermi-2 design calculation includes an adequate allowance for drift and the extension has no potential impact on the plant safety analyses, i.e., no analytic limit changes are required. Plant calibration data was also reviewed and found to be within the design allowances.

A historical search of the 18 month surveillance tests for these instruments for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased calibration surveillance interval would not increase the period a component would be unavailable. Failures of components associated with the extended interval were identified; however, the type of failures would be detected during tests that are performed more frequently. Therefore, these failures have no effect on component availability for the extended operating cycle. Therefore, the extension of the surveillance interval is justified.

ENCLOSURE 19
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.7.5 Table 4.3.7.5-1
POST ACCIDENT MONITORING INSTRUMENTATION

Justification for extending Item 1

The Accident Monitoring Reactor Pressure instrumentatio. was analyzed using the GE Instrument Setpoint Methodology, NEDC-31336. The GE instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket. In addition, Rosemount published a report in February 1990, "30 Month Stability Specification For Rosemount 1152, 1153, 1154 Pressure Transmitters" (Rosemount Report D89000126, Revision A) which was accepted by NRC Safety Evaluation Report dated 8/2/93 on the Peach Bottom Atomic Power Station docket. This report supported the extension of the calibration interval for the transmitters from 18 months to 30 months based on a reduction in the drift allowance.

The Fermi 2 design calculations contain a drift allowance for Rosemount 1152 transmitter equal to at least three consecutive six month vendor drift intervals. The previously published model 1152 six month drift interval value is of the same order of magnitude as the currently published thirty month drift interval value. Therefore, the calculations contain at least 30 months drift allowance. The transmitter provides input to recorders. Channel check is performed monthly. This check confirms that the indicating channel performs within allowances. As a result, the existing Fermi-2 design calculation includes an adequate allowance for drift and the extension has no potential impact on the plant safety analyses. Plant calibration data was also reviewed and found to be within the design allowances. Therefore, the extension of the surveillance interval is justified.

Justification for extending Item 2.a

The Reactor Fuel Zone indication is supplied by the same instruments which supply the Reactor Vessel Level - Levels 1, 2, and 8 setpoints this transmitters drift was analyzed using the GE Instrument Setpoint Methodology, NEDC-31336. The GE Instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket. In addition, Rosemount published a report in February 1990, "30 Month Stability Specification For Rosemount 1152, 1153, 1154 Pressure Transmitters" (Rosemount Report D89000126, Revision A) which was accepted by NRC Safety Evaluation Report dated 8/2/93 on the Peach Bottom Atomic Power Station docket. This report supported the extension of the calibration interval for the transmitters from 18 months to 30 months based on a reduction in the drift allowance.

The Fermi 2 design calculations contain a drift allowance for Rosemount 1153 transmitter equal to at least three consecutive six month vendor drift intervals. The previously published model 1153 six month drift interval value is of the same order of magnitude as the currently published thirty month drift interval value. Therefore, the calculations contain at least 30 months drift allowance. The transmitter provides input to recorder/indicators. Channel check is performed monthly. This check confirms that the indicating channel performs within allowances. As a result, the existing Fermi-2 design calculation includes an adequate allowance for drift and the extension has no potential impact on the plant safety analyses. Plant calibration data was also reviewed and found to be within the design allowances. Therefore, the extension of the surveillance interval is justified.

Justification for extending Item 2.b

The Reactor Vessel Water Level instrumentation was analyzed using the GE Instrument Setpoint Methodology, NEDC-31336. The GE Instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket. In addition, Rosemount published a report in February 1990, "30 Month Stability Specification For Rosemount 1152, 1153, 1154 Pressure Transmitters" (Rosemount Report D89000126, Revision A) which was accepted by NRC Safety Evaluation Report dated 8/2/93 on the Peach Bottom Atomic Power Station docket. This report supported the extension of the calibration interval for the transmitters from 18 months to 30 months based on a reduction in the drift allowance.

The Fermi 2 design calculations contain a drift allowance for Rosemount 1153 transmitter equal to at least three consecutive six month vendor drift intervals. The previously published model 1153 six month drift interval value is of the same order of magnitude as the currently published thirty month drift interval value. Therefore, the calculations contain at least 30 months drift allowance. The transmitter provides input to recorders. Channel check is performed monthly. This check confirms that the indicating channel performs within allowances. As a result, the existing Fermi-2 design calculation includes an adequate allowance for drift and the extension has no potential impact on the plant safety analyses. Plant calibration data was also reviewed and found to be within the design allowances. Therefore, the extension of the surveillance interval is justified.

Justification for extending Item 11

The PCI pressure switches are used to detect SRV position. The switches trip on increasing pressure and provide an alarm in the control room. The switch setpoint was analyzed for the extended surveillance interval using the GE Instrument Setpoint Methodology, NEDC-31336. The method of calculation is similar to the calculations submitted to the NRC during the power uprate review that was accepted by the NRC

Safety Evaluation Report dated September 9, 1992 on the Fermi-2 docket. The GE Instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket.

The evaluation conservatively considered two (18 months) drift intervals. As a result, the Fermi-2 setpoint includes an adequate allowance for drift. Plant calibration data was reviewed and found to be within the required design limits. Therefore, the extension of the surveillance interval is justified.

Justification for extending Item 12

For the Containment High Range Radiation Monitor (CHRRM) instruments, correct operation is confirmed by a monthly channel functional check (except the detector). These tests will identify any potential drift of these instruments (except for the detector and cable) during this period of time. The quarterly and twice daily surveillance tests, monitoring by plant staff, and the fact that the ion chamber detectors and cable are not considered susceptible to drift ensure that there is no potential impact on the subject instrument availability by allowing the additional time interval of source calibration of the existing radiation monitors.

The surveillance allowance for the CHRRM instrument is 20 percent of span. A review of the surveillance tests indicates that the drift for the CHRRM was 23 percent for the worst case condition. This drift was isolated to a single data point and when removed as an outlier the remaining data points indicated a drift of 18 percent. A further review of the CHRRM calibration history indicates that the detectors have never been adjusted to compensate for drift. Therefore, the drift history for these components documents the ability of the CHRRM to perform for multiple cycles without adjustment and with performance remaining within expected bounds.

The instruments used for CHRRM are identical to the instruments approved by the NRC for a permanent 24 month cycle extension in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months.

Based on the above discussion, the NRC's prior approval of identical instrumentation and the fact that historical data for the radiation monitors shows that recalibration has not been required for a period of time greater than 24 months, it is concluded that the impact on instrument availability, if any, is small as a result of the one-time surveillance interval extension and the requested extension is justified.

Justification for extending Item 16

Limit switches, which provide the position indication for these valves, are mechanical devices that require mechanical adjustment only; drift is not applicable to these devices. Therefore, an increase in the surveillance interval to accommodate one time extended operating cycle does not affect the limit switches with respect to drift.

A historical search of the 18 month surveillance tests for these instruments for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests, each failed or partially failed test was to be reviewed and evaluated. The purpose of this evaluation was to demonstrate that the increased calibration surveillance interval would not increase the period an instrument would be unavailable. This evaluation determined that one clamp-on valve position indication and three limit switch valve position indication failures had occurred during the last two refueling outages. The clamp-on valve position indication and two of the three limit switch position indication failures are the same failures discussed in Enclosure 8. The third Limit switch position indication failure caused a valve stroke time requirement to fail because locally the valve indicated full close and remotely the valve indicated mid position. Additionally an RPV level 8 transmitter could not be calibrated due to failed components. The Level 8 transmitter failure is the same failure discussed in Enclosure 11. The valve position indication failure was repaired and retested satisfactorily. A review of maintenance history did not find significant position indication failures. Based on the previous discussion, the historical failure review, and the small impact on safety function availability, the requested extension is justified.

ENCLOSURE 20
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.7.10
LOOSE PARTS DETECTION SYSTEM

The Fermi Loose Parts Detection System has incorporated features that allow periodic checks of the functional operational capability and calibration to be accomplished. Channel checks can be performed by observing the analog readout in the control center. Channel functional tests can be performed by using the preamplifier bias check feature to confirm the ability of the channel to function in the required manner. The detectors for this system do not have a drift characteristic similar to analog instrumentation. The system does not perform a direct safety related function. A historical review for this SR for RFO3 and RFO4 indicated no failures. Therefore, since system failure does not require plant shutdown for the short extension time requested the surveillance interval extension is justified.

ENCLOSURE 21
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.9.1 Table 4.3.9.1-1
FEEDWATER TURBINE TRIP SYSTEM INSTRUMENTATION

Justification for extending Item a

The Reactor Vessel Level - Level 3 and 8 setpoints were analyzed using the GE Instrument Setpoint Methodology, NEDC-31336. The method of calculation is similar to the calculations submitted to the NRC during the power uprate review that was accepted by NRC Safety Evaluation Report dated September 9, 1992 on the Fermi-2 docket. The GE Instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket. In addition, Rosemount published a report in February 1990, "30 Month Stability Specification For Rosemount 1152, 1153, 1154 Pressure Transmitters" (Rosemount Report D89000126, Revision A) which was accepted by NRC Safety Evaluation Report dated August 2, 1993 on the Peach Bottom Atomic Power Station docket. This report supported the extension of the calibration interval for the transmitters from 18 months to 30 months based on a reduction in the drift allowance.

The Fermi 2 design calculations contain a drift allowance for Rosemount 1153 transmitter equal to at least three consecutive six month vendor drift intervals. The previously published model 1153 six month drift interval value is of the same order of magnitude as the currently published thirty month drift interval value. Therefore, the calculations contain at least 30 months drift allowance. Associated loop trip units are checked on a quarterly basis. This check confirms that the trip units are maintained within allowances. As a result, the existing Fermi-2 design calculation includes an adequate allowance for drift and the extension has no potential impact on the plant safety analyses, i.e., no analytic limit changes are required. Plant calibration data was also reviewed and found to be within the design allowances.

A historical search of the 18 month surveillance tests for these instruments for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased calibration surveillance interval would not increase the period an instrument would be unavailable. This evaluation determined that an RPV level 8 transmitter could not be calibrated due to failed components. The Level 8 transmitter failure is the same one discussed in Enclosure 11. This was an isolated failure with no time dependent failure mode. Therefore this failure has no effect on availability of the instrument for the extended operating cycle. Based on the previous discussion, the

historical failure review, and the small impact on safety function availability the requested extension is justified.

ENCLOSURE 22
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.9.2
LOGIC SYSTEM FUNCTIONAL TEST AND SIMULATED AUTOMATIC OPERATION
FEEDWATER TURBINE TRIP SYSTEM

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

> "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 system, 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 evaluation above is applicable to Fermi 2. A historical search of the 18 month surveillance tests for these instruments for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period an instrument would be unavailable. This evaluation determined that an RPV level 8 transmitter could not be calibrated due to failed components. The Level 8 transmitter failure is the same one discussed in Enclosure 11. This was an isolated failure with no time dependent failure mode. Therefore this failure has no effect on availability of the instrument for the extended operating cycle. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

ENCLOSURE 23
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.3.11.1 Table 4.3.11.1-1
ALTERNATIVE SHUTDOWN SYSTEM

Justification for extending Item 7

Alternate Shutdown Reactor Water Level Indication was analyzed using the GE Instrument Setpoint Methodology, NEDC-31336. The GE Instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket. In addition, Rosemount published a report in February 1990, "30 Month Stability Specification For Rosemount 1152, 1153, 1154 Pressure Transmitters" (Rosemount Report D89000126, Revision A) which was accepted by NRC Safety Evaluation Report dated 8/2/93 on the Peach Bottom Atomic Power Station docket. This report supported the extension of the calibration interval for the transmitters from 18 months to 30 months based on a reduction in the drift allowance.

The Fermi 2 design calculations contain a drift allowance for Rosemount 1153 transmitters equal to at least three consecutive six month vendor drift intervals. The previously published model 1153 six month drift interval value is of the same order of magnitude as the currently published thirty month drift interval value. Therefore, the calculations contain at least 30 months drift allowance. The transmitter provides input to an indicator. Channel check is performed monthly. This check confirms that the indicating channel performs within allowances. As a result, the existing Fermi-2 design calculation includes an adequate allowance for drift and the extension has no potential impact on the plant safety analyses. Plant calibration data was also reviewed and found to be within the design allowances. Therefore, the extension of the surveillance interval is justified.

Justification for extending Item 8

The Alternative Shutdown System Reactor Pressure instrumentation was analyzed using the GE Instrument Setpoint Methodology, NEDC-31336. The GE Instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket. In addition, Rosemount published a report in February 1990, "30 Month Stability Specification For Rosemount 1152, 1153, 1154 Pressure Transmitters" (Rosemount Report D89000126, Revision A) which was accepted by NRC Safety Evaluation Report dated 8/2/93 on the Peach Bottom Atomic Power Station docket. This report supported the extension of the calibration interval for the transmitters from 18 months to 30 months based on a reduction in the drift allowance.

The Fermi 2 design calculations contain a drift allowance for Rosemount 1153 transmitters equal to at least three consecutive six month vendor drift intervals. The previously published model 1153 six month drift interval value is of the same order of magnitude as the currently published thirty month drift interval value. Therefore, the calculations contain at least 30 months drift allowance. The transmitter provides input to a recorder. Channel check is performed monthly. This check confirms that the indicating channel performs within allowances. As a result, the existing Fermi-2 design calculation includes an adequate allowance for drift and the extension has no potential impact on the plant safety analyses, i.e., no analytic limit changes are required. Plant calibration data was also reviewed and found to be within the design allowances.

A historical search of the 18 month surveillance tests for these instruments for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased calibration surveillance interval would not increase the period a component would be unavailable. No failures were identified by this review. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

ENCLOSURE 24
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.4.2.1.1, 4.4.2.1.2, AND 4.4.2.2
SAFETY RELIEF VALVES

Justification for extending SR 4.4.2.1.1

As stated in Fermi 2 Technical Specification 4.4.2.1.1, the valve position indicator for each safety/relief valve (SRV) shall be demonstrated operable with the pressure setpoint of each of the tail-pipe pressure switches verified to be 30 ± 5 psig by performance of a Channel Calibration at least once per 18 months. The SRV position indicator essentially serves two monitoring functions. The first function is to provide indication to the plant operator, if an SRV has inadvertently opened. The second function is that SRV position indicator provides a verification of the operator's actions to open an individual SRV. The position indicators provide the ability to determine specifically which SRV has lifted. In addition to the SRV position indicators, the operator can determine that an SRV has lifted by a variety of plant indications, including decreased power generation or main turbine throttle position, and increasing suppression pool temperature. In addition, the SRV tail pipe temperature can be used to identify specifically which SRV has lifted. With alternate means of detecting whether a relief valve lifts, the extension of this surveillance will have a small, if any, impact on the ability of the operator to determine the position of an SRV.

The PCI pressure switches are used to detect SRV position. The switches trip on increasing pressure and provide an alarm in the control room. The switch setpoint was analyzed for the extended surveillance interval using the GE Instrument Setpoint Methodology, NEDC-31336. The method of calculation is similar to the calculations submitted to the NRC during the power uprate review that was accepted by the NRC Safety Evaluation Report dated September 9, 1992 on Fermi-2 docket. The GE Instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket.

The evaluation conservatively considered two (18 months) drift intervals and found the results to be acceptable. As a result, the Fermi-2 setpoint includes an adequate allowance for drift. Plant calibration data was reviewed and found to be within the required design limits. Therefore, the extension of the surveillance interval is justified.

Justification for extending SR 4.4.2.1.2

The SRVs are Target Rock Corp. two-stage pilot operated dual function safety/relief valves. In the safety mode, the valve opens solely by mechanical means when pressure at the inlet of the valve reaches the set pressure of the valve. This provides the code

required overpressure protection of the reactor coolant pressure boundary. In the relief mode, the valve is remotely opened by a solenoid valve manifold/pneumatic operator assembly to provide controlled depressurization of the reactor coolant pressure boundary. There are a total of 15 SRVs that all function in the safety mode and have the capability to operate in the relief mode via manual actuation from the Main Control Room. Five (5) of the SRVs are allocated to the automatic depressurization system (ADS) which can automatically operate the valves in the depressurization mode to reduce reactor pressure and thus allow the low pressure Emergency Core Cooling Systems (ECCS) to cool the reactor. This SR only pertains to the self-actuating safety mode of the SRVs. The SRVs are tested to lift at nominal setpoint at a ± 1 percent tolerance established by TS 3.4.2.1.

Every Fermi 2 LLRT and refueling outage has experienced a sufficient number of SRVs that failed the setpoint SR to result in an LER. These failures have also resulted in all 15 SRV pilots being tested each outage. These failures are predominantly due to upward setpoint drift, which is consistent with industry results for these valves. In all cases, subsequent analysis of the safety valve capability has demonstrated that required vessel overpressure protection has been available throughout each operating cycle.

An evaluation of Fermi 2 set pressure surveillance data for RFO3 and RFO4, as well as industry data from the BWROG on Target Rock two-stage SRVs, does not indicate a trend toward downward drift (decreasing set pressure). Therefore, the proposed increase in the testing interval duration will not impact the probability of occurrence of an inadvertent SRV opening.

The primary problem in the surveillance failures for the SRV setpoint is upward setpoint drift. Historical evaluation by the BWROG and GE of the Target Rock two stage SRV setpoint surveillances indicates that setpoint drift is not closely associated with plant operating time. This is consistent with knowledge gained from research on metallic oxide formation and valve corrosion, as applicable to the SRV. Limited oxide formation and consequent seat-to-disc bonding occurs rapidly on the freshly refurbished metal seating surfaces. The oxide layers protect the underlying base metal, passivating the seating surface so that further bond formation is very slow. Therefore, upward setpoint drift during the proposed extension of the surveillance interval should not significantly increase.

The following discussion concerning upward setpoint drift was provided in the Limerick Generating Station, Units 1 and 2 Technical Specification Change Request of May 8, 1991 as justification for a permanent change of the SRV test interval to 24 months. It is provided here for reference:

> "General Electric (GE) proprietary topical report NEDE-30476, "Setpoint Drift Investigation of Target Rock Two-Stage Safety/Relief Valve (Final Report)," as a

result of an extensive SRV testing program funded by the Boiling Water Reactor (BWR) Owner's Group, identified that Target Rock two-stage SRVs experience an upward drift (i.e., increase) in set pressure due to corrosion induced bonding of the pilot disc and seat. The bonding process occurs due to a high oxygen environment corroding the Stellite pilot disc and seat surfaces combined with their close contact. The oxides from both surfaces grow together to form a bond. The force required to break this bond increases the effective initial set pressure since the pilot disc must lift to actuate the main disc. SRV set pressures which have drifted due to pilot disc/seat bonding return to near their nominal set point after the first actuation. Another cause of upward set pressure drift identified in NEDE-30476 is labyrinth seal induced friction due to insufficient clearances between the pilot rod and the pilot guide.

As discussed in Section 3.9.3.4 of Supplement 3 to NUREG-0991, "Safety Evaluation Report Related to the Operation of Limerick Generating Station, for Units 1 and 2," dated October 1984, the NRC recognized the generic upward set pressure drift problem exhibited by Target Rock two-stage SRVs as applicable to LGS, Units 1 and 2, but concluded that LGS can be operated with no adverse effect on the health and safety of the public until the NRC reaches a final generic solution for setpoint based on the reasons described below.

1) LGS has implemented the recommendations of all applicable supplements to GE Service Information Letter (SIL) No. 196, including Supplement 14, "Target Rock 2-Stage SRV Setpoint Drift." GE incorporated the NEDE-30476 report recommendations for improved SRV maintenance and refurbishment into Supplement 14 of GE SIL No. 196. The NRC concluded that implementation of these recommendations adequately address setpoint drift due to labyrinth seal induced friction. Additionally, the NRC concluded, based on available two-stage SRV data, that setpoint drift resulting from pilot disc/seat bonding occurs less frequently then that caused by labyrinth seal induced friction. The proposed change in the frequency of SRV testing will not impact the implementation of these recommendations.

2) The 14 SRVs installed at LGS provide considerably more relieving capacity than is required by the applicable edition of the American Society of Mechanical Engineers (ASME) Code. Only 11 of the 14 SRVs are required to be operable in accordance with TS Section 3.4.2. The proposed change in the frequency of SRV testing will not impact the design of the SRVs.

3) The TS SRV set pressure testing at LGS exceeds the current ASME Code Section XI requirements, i.e., at least 50% of the SRVs are tested each refueling outage in accordance with TS versus 20% in accordance with ASME Code Section XI. ANSI/ASME OM-1-1981, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices," as invoked by ASME Code Section XI, requires all SRVs to be tested within a 60 month period with a minimum of 20% tested within any 24 months. The proposed change to TS Section 4.4.2.2 would require 50% of the SRVs to be tested in 24 months, and all SRVs to be tested within a maximum period of 54 months (i.e., 48 months with a six month grace period)."

The above discussion and conclusions apply to Fermi 2. Fermi 2 has implemented the recommendations of SIL No. 196 including Supplement 14. The 15 SRVs installed at Fermi 2 provide considerable relieving capacity above the 11 SRVs needed to meet ASME Code requirements. The Fermi 2 TS SRV testing requirements also exceed the ASME Code requirements by requiring 50 percent of the SRVs to be tested during each refueling outage.

Based upon this evaluation, Detroit Edison concludes that the one time extension will not have any affect on the ability of the SRVs to perform their safety function. Therefore, this extension is justified.

Justification for extending SR 4.4.2.2.b

The purpose of this test is to perform a Logic System Functional test, 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 (1) prevent any low-low set valve from opening, and (2) cause inadvertent seal-in of low-low set logic.

Based on the above discussion, the logic design of the low-low pressure set provides redundancy and reliability which ensures the function will remain available during the extended operating cycle.

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

"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 system, 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."

A historical search of the 18 month surveillance tests for these surveillance requirements for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a component would be unavailable. Failures associated with the SRV's identified during this review, have been discussed in the above text for this Enclosure. Instrument related problems encountered during these SR's were accounted for in the instrument drift evaluation discussed in this enclosure or would not have had any effect on the system's ability to perform its safety function. Therefore, these failures have no effect on component availability for the extended operating cycle. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

ENCLOSURE 25
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.4.3
REACTOR COOLANT SYSTEM LEAKAGE

Justification for extending 4.4.3.1.b

Drywell sumps are monitored to determine fill up and pump out rates to indirectly determine Reactor Coolant System leakage rates. Since drift is a small change over a long duration (months) and leakage is determined over a short duration (hours) as a differential, the Reactor Coolant Leakage Detection Systems will not be affected by instrument drift resulting from an increase in calibration interval. Functional testing is done monthly on these systems. Therefore, the extension of the surveillance interval is justified.

A historical search of the 18 month surveillance tests for these instruments for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests, each failed or partially failed test was to be reviewed and evaluated. The purpose of this evaluation was to demonstrate that the increased calibration surveillance interval would not increase the period an instrument would be unavailable. The only failure identified by this review 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 the small impact on safety function availability the requested extension is justified.

Justification for extending SR 4.4.3.2.2.a

All Pressure Isolation Valves (PIV) have passed both RF03 and RF04 surveillances, and all are well below the leakage limit. The highest recorded leakage for any PIV was 0. 14 gpm. This review also shows no increasing leakage trends for any of the PIVs and it appears there is very little correlation between time and leakage. Extension of RF05 to the fall of 1996 will have little effect on test results.

Based on the abcve discussion the extension of this SR is justified.

ENCLOSURE 26
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.4.8 AND 4.0.5
ASME CODE INSPECTION PROGRAM (ISI/IST)

Discussion for SR 4.4.8 and 4.0.5

The purpose of the system leakage test is to identify pressure boundary leakage. Through wall leakage (i.e., cracked welds, leaking pipe, etc.) is cause for failure of this test and subsequent repair. During the test the entire Class I pressure boundary is walked down and inspected for leakage. Packing, gaskets, seals, and through seat leakage rates for all components are recorded, evaluated, and repaired if required.

The only through wall leakage identified to date was on a 3/4" vent valve connection on the Recirculation system during RFO3. The system leakage test was performed twice following RF04 and all leakage noted was from valve packing, gaskets and rework of valve E41FOO2. During normal plant operation plant/drywell leakage is consistently monitored (Tech Spec 3.4.3.2). Any degradation of the ASME Class 1 piping would be detected and required actions taken.

The Reactor Pressure Vessel System leakage test is required to be performed each refueling outage prior to plant start-up (Table IWB-2500,-1 Category B-P). In addition, a system leakage test is required following the opening and re-closing of any Class 1 component. This requirement does not have a set periodicity (i.e., 18 or 24 months) but is tied to the Reactor Refueling Outage or maintenance activities on Class 1 components. Therefore, no scheduling extension is required if the cycle outage is extended to 24 months.

The ISI-NDE Program Plan identifies all components requiring inspection over the 10 year inspection interval and the refueling outage in which each component is to be inspected. Procedure 43.000.016 implements administrative control of these ISI-NDE inspections. Fermi has completed 2 inspection periods and is in compliance with code requirements for percent complete. Extending the current cycle from March to October will have no affect on ISI Program compliance. As this program is a sampling program of inspections over the interval, it is not anticipated that the 6 month extension will have a significant affect on the RF05 inspection results.

A review of historical results from RF04 was performed. This includes the results for piping and vessel welds, RPV internals, and for component supports. During RF04 16 of 48 shroud head bolts (SHB's) were found to have indications. All 16 bolts, and one additional bolt, were replaced with improved, more crack resistant designed bolts. While

these 16 SHB's had indications, complete separation did not occur and these bolts were still capable of providing structural integrity. The shroud head bolts will be inspected during RF05 and any indications will be evaluated and the bolts will be replaced as necessary.

Based on the above there is no reason the inspection scheduled for RF05 cannot be performed in the fall of 1996.

ISI-NDE Program Plan requirements for reactor pressure vessel/vessel internals visual examinations are implemented per 43.000.017 each refuel outage. This procedure provides additional detailed inspection points for components included in the ISI-NDE Program. Inspection of Reactor internals is required to be performed each Reactor Refueling Outage per ASME Section XI and augmented inspection requirements imposed by SIL's, RICSIL's, etc. Commitments have been made based on RF04 inspection results to re-inspect components during the next Refuel Outage.

A historical review of RPV inspections performed and indications observed during the previous outages is summarized below.

Prior to RF04

- During RF01 and RF02, a number of minor indications were noted on internal components. No action was taken to repair these indications and they have been reinspected each outage since with no signs of further degradation.

- During RF03, two cracks were identified on the steam dryer. Following an evaluation it was decided to repair the hood-to-end-panel weld. In addition, all similar configuration joints on the dryer were reinforced. Reinspection during RF04 verified that the repair was successful and that the unrepaired indication did not degrade further.

Based on the reinspection results mentioned, inspection in the fall of 1996 is acceptable.

RF04

- During RF04 an extensive visual inspection of the Core Shroud was performed on the shroud welds, as recommended by GE and the NRC, using the latest available techniques. This inspection was intended to serve as a baseline for future inspections. Only two very small indications were identified on the H-2 weld on the inside surface. Based on Boiling Water Reactor Vessel Internals Project (BWRVIP) guidelines, Fermi 2 is classified as a category "A" plant because of fabrication materials, conductivity, and time in operation. BWRVIP

recommendation for category "A" plants is inspection after 8 effective full power years of operation. This cumulative operating time is expected by RF06 or RF07. Therefore, performance of follow-up inspections in the fall of 1996 is acceptable.

- Corrosion deposits on the vessel wall, steam dryer and feedwater nozzles (attributable to the reactor water chemistry excursion following the turbine failure of December, 1993) were hydrolazed to remove them. This effort was highly successful and it should be noted that there was no base metal attack. Evaluation by GE has concluded that no deleterious effects are expected. Reinspection of these areas during the next refueling outage is planned however. Performance in the fall of 1996 is acceptable.

- During RF04 indications were identified on the steam dryer assembly. This included protruded and cracked tie-rod washer/nut tack welds and 2 linear indications on the shroud support ring. A review of video tapes from RF03 of the dryer confirmed that these conditions were present in RF03 and there was minimal change. The linear indications on the support ring are of no structural significance and are typically found on all BWR dryers after operation.

The visual inspections of the RPV internals are performed each refuel outage to detect degradation of internal components. As in the past, all identified conditions are evaluated and repaired as required. While time in service contributes to the initiation of Intergrannular Stress Corrosion Cracking (IGSCC), there is no reason to expect a significant change/degradation by extending the refueling outage until the fall of 1996.

The ISI-NDE Program requires visual examination (VT- 1) of Class I pressure retaining bolts be performed per 43.000.014. As a result, components are selected each refuel outage such that all the components are inspected over the standard ten year inspection interval, and at least a minimum number of components are inspected to satisfy the percentage requirements for each inspection period. In addition to the scheduled outage scope, all replacement bolting utilized during valve or component refurbishment is also inspected. Fermi 2 has completed 2 inspection periods and is in compliance with code requirements for percent complete.

Based on the above, performance inspections in the fall of 1996 is acceptable.

43.000.010, "ISI Check Valve Inspection Procedure," calls for the disassembly and inspection of the following valves: E1100F046A, B, C, and D; E2100F003A, B, C, and D; E2100F006A and B; E2100F038A, B, C, and D; E4100F045; and P4400F051, 116A, 116B, and 165. These are already specified in the appropriate Relief Requests as refueling periodicity. The valve having the longest time period since last being inspected is the one to be inspected during the next outage (RF05). During RF03 and RF04 there

were no failures in the sampled population of check valves in the IST Program. Therefore, there is no increased risk by extending the inspection to the fall of 1996.

Note: Other ISI/IST components (i.e., instrument line excess flow check valves) are discussed and justified in their appropriate Technical Specification Section enclosures.

ENCLOSURE 27
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.5.1
EMERGENCY CORE COOLING SYSTEM OPERATION

Justification for extending SR 4.5.1.c.1

For the Core Spray System, a pump, valve and flow test is performed on a more frequent basis in accordance with the Inservice Testing Program during the operating cycle to ensure that this assumed ECCS function is available. Although this test does not ensure the operability of the entire Core Spray logic, it does ensure the functional ability of the Core Spray Pumps, and a majority of the Core Spray System to produce the required ECCS flow at the required pressure. The portion of the system not tested on the more frequent basis is equivalent to the logic system and the testing would be equivalent to a logic system functional test.

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

> "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 system, 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 evaluation above is applicable to the Fermi 2 surveillance interval extension. A historical search of the 18 month surveillance tests for these components for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a component would be unavailable. No failures were identified by this review. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

For the Low Pressure Coolant Injection (LPCI) System, a pump, valve and flow test is performed on a more frequent basis in accordance with the Inservice Testing Program during the operating cycle to ensure that this assumed ECCS function is available. Although this test does not ensure the operability of the entire LPCI logic, it does ensure

the functional ability of the LPCI Pumps, and a majority of the LPCI System to produce the required ECCS flow at the required pressure. The portion of the system not tested on the more frequent basis is equivalent to the logic system and the testing would be equivalent to a logic system functional test.

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

> "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 system, 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."

A historical search of the 18 month surveillance tests for these components for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a component would be unavailable. No failures of the components were identified by this review. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

For the High Pressure Coolant Injection (HPCI) System, a pump, valve and flow test is performed on a more frequent basis in accordance with the Inservice Testing Program during the operating cycle to ensure that this assumed ECCS function is available. Although this test does not ensure the operability of the entire HPCI logic, it does ensure the functional ability of the HPCI Pump, and a majority of the HPCI System to produce the required ECCS flow at the required pressure. The portion of the system not tested on the more frequent basis is equivalent to the logic system and the testing would be equivalent to a logic system functional test.

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

> "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 system, 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 evaluation above is applicable to the Fermi 2 surveillance interval extension. A historical search of the 18 month surveillance tests for these components for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a component would be unavailable. No failures were identified by this review. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

Justification for extending SR 4.5.1.d.2.a

The logic design of the Automatic Depressurization System (ADS) provides redundancy and reliability which ensures the function will remain available during the extended operating cycle. SR 4.5.1.d.2.a requires a simulated automatic actuation of the system, which excludes actual valve operation. No extension in the surveillance interval is needed or requested for SR 4.5.1.d.2.b, in which the valves are actually opened.

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

> "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 system, 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 evaluation above is applicable to the Fermi 2 surveillance interval extension. A historical search of the 18 month surveillance tests for these components for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a component would be unavailable. No failures were identified by this review. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

ENCLOSURE 28
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.6.1.4
MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM

Justification for extending SR 4.6.1.4.d.3

The MSIV Leakage Control System Pressure Control setpoints were analyzed using the GE Instrument Setpoint Methodology, NEDC-31336. The method of calculation is similar to the calculations submitted to the NRC during the power uprate review that was accepted by NRC Safety Evaluation Report dated September 9, 1992 on Fermi-2 docket. The GE Instrument Setpoint Methodology, NEDC-31336, has been accepted by NRC Safety Evaluation Report dated 7/18/95 on the Perry Nuclear Power Plant docket. In addition, Rosemount published a report in February 1990, "30 Month Stability Specification For Rosemount 1152, 1153, 1154 Pressure Transmitters" (Rosemount Report D89000126, Revision A) which was accepted by NRC Safety Evaluation Report dated 8/2/93 on Peach Bottom Atomic Power Station docket. This report supported the extension of the calibration interval for the transmitters from 18 months to 30 months based on a reduction in the drift allowance.

The Fermi 2 design calculations contain a drift allowance for Rosemount 1153 transmitters equal to at least three consecutive six month vendor drift intervals. The previously published model 1153 six month drift interval value is of the same order of magnitude as the currently published thirty month drift interval value. Therefore, the calculations contain at least 30 months drift allowance. Associated loop trip units are checked on a quarterly basis. This check confirms that the trip units are maintained within allowances. As a result, the existing Fermi-2 design calculation includes an adequate allowance for drift and the extension has no potential impact on the plant safety analyses, i.e., no analytic limit changes are required. Plant calibration data was also reviewed and found to be within the design allowances. Therefore, the extension of the surveillance interval is justified.

A historical search of the 18 month surveillance tests for these instruments for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased calibration surveillance interval would not increase the period an instrument would be unavailable. There were no failures identified for this SR. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

ENCLOSURE 29
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.6.2.1
DEPRESSURIZATION SYSTEMS

Justification for extending SR 4.6.2.1.e

The purpose of this inspection is to determine that there is no evidence of corrosion of painted surfaces which could result in the unevaluated degradation of the coating system during the next operating cycle. During plant operation all surfaces required to be inspected by these requirements are normally in an inerted environment. The inerted environment will help to reduce the corrosion from occurring at an excessive rate in all areas other than the underwater area of the torus. In addition, the original surveillance interval between inspections of the drywell and the torus is based on the accessibility to the containment interior not on a specific time based requirement.

Detailed inspections of the torus immersion area, vapor phase area and exterior surface were conducted during RFO4 to identify and evaluate any current or incipient coating failure, and the effects of corrosion resulting from coating deficiencies.

Immersion Area (Below Water)

Qualitative visual inspections were performed on the torus pressure boundary immersion surfaces and submerged structures in all sixteen bays. Overall, the coating system was found to be intact. No significant evidence of current or incipient general coating failure was identified. No significant changes in coating integrity were identified when compared to the results of inspection performed during RFO1 and RFO2. Corrosion is taking place only in extremely small areas where fractured blisters are present and where nicks, dings, scrapes, etc. have damaged the coating substrate. No significant pitting corrosion was noted at these locations. Underwater coating repairs were performed on localized corrosion in all sixteen bays to mitigate corrosion.

Vapor Phase Area (Above Water)

The vapor phase coating inspection revealed isolated minor mechanical damage on pressure boundary and internal structures. Corrosion of the exposed substrate is minimal with no measurable pitting corrosion. Coating repair was performed on most of the identified locations and the remainder are to be inspected and evaluated during the next refueling outage. No significant changes in coating integrity were identified when compared to the results of inspection performed during RFO1 and RFO2.

Torus Exterior Surface

The items (two) noted in the RFO4 inspection report were not detrimental to the structural integrity of the pressure boundary and will be repaired or re-evaluated during the next refueling outage. No significant changes in coating integrity were identified when compared to the results of inspection performed during RFO1, RFO2 and RFO3.

The containment environment is inerted and any "as found" degradation of the protective coating has been evaluated and determined acceptable for continued operation for the extended Operating Cycle. Therefore, the impact, if any, on the containment integrity from the change to the surveillance interval for the subject requirements is small.

Justification for extending SR 4.6.2.1.h

The drywell -to-suppression chamber leakage test is performed to ensure that the pressure suppression function of the primary containment is maintained. Excessive leakage from the drywell directly to the suppression chamber could result in a failure of the primary containment during a design basis accident. The only active component in this barrier is the drywell-to-suppression chamber vacuum breaker. This vacuum breaker is normally in the closed position, and is verified to be in the closed position every 7 days in accordance with SR 4.6.4.1.a. The design of the Mark 1 containment minimizes the potential for drywell to suppression chamber leakage. Based on the design of the Fermi 2 containment, and the verification of the position of the vacuum breaker on a more frequent basis, the impact, if any, on system availability is small from the extended operating cycle.

A historical search of the 18 month surveillance tests results for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a component or system would be unable to perform its function. This evaluation determined that a failure did occur during the test performed in RFO4. Overall leakage was slightly in excess of the acceptance criteria. The reason for the failure was attributed primarily to leakage past the seat of a single drywell-to-suppression chamber vacuum breaker. The vacuum breaker was repaired and the retest successful. Historically the bypass leakage rate has been low. Based on the historical evaluation and the short extension requested for this SR (for 3 days from 10/2/96 to 10/5/96) the extension is justified.

ENCLOSURE 30
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.6.3
PRIMARY CONTAINMENT ISOLATION VALVE FUNCTIONAL TEST

Justification for extending SR 4.6.3.2

Most Primary Containment Isolation Valves (PCIVs) are cycled on a more frequent basis in accordance with the ISI/IST program during the operating cycle to ensure that the valves are capable of performing their isolation function. Although this test does not ensure the operability of the entire PCIVs Logic, it does ensure the functional ability of the PCIVs as assumed in the offsite dose calculations. The remaining portions of the Isolation valve circuit that is not tested during the ISI/IST program is equivalent to a logic system and the testing of these components would be equivalent to a logic system functional test.

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

> "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliability are not dominated by the reliabilities of the logic system, 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 evaluation above is applicable to Fermi 2. An historical search of the 18 month surveillance tests for these valves for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a component would be unavailable. This evaluation determined that one relay did not de-energize within allowable time limits. This is the same relay failure previously discussed in Enclosure 8. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

Justification for extending SR 4.6.3.4

Instrument piping which is connected to the reactor coolant system, and which exits the primary containment, is designed with excess flow check valves. The instrument piping ends at the instrument connection. Each line contains a 0.25 inch restricting orifice and a manual isolation valve upstream of the excess flow check valve. The test that is performed on the excess flow check valves is a modified leak rate test. ASME code Section XI states that leak rate testing of check valves be performed every 2 years. Fermi Relief request VR-009 allows for conducting these tests at a refueling outage periodicity.

An evaluation of any failures identified for RFO3 and RFO4 was performed to determine if the failure modes contained time based elements which could impact the one-time extension of the surveillance interval. The failure rate for these valves was determined to be less than two percent. This evaluation also determined that any failures identified would have little or no effect on the availability of the instrument to perform its intended safety function.

The NRC has approved a permanent extension of excess flow check valve test for the Oyster Creek Nuclear Generating Station (SER issued January 10, 1991).

Based on the previous discussion, the historical failure review, the NRCs previous acceptance of an extension for these valves and the small impact on safety function availability the requested extension is justified.

Justification for extending SR 4.6.3.5.b

The Transversing In-Core Probe (TIP) shear valve is provided as a back-up isolation device for the TIP guide tube isolation valve. The isolation valve closes automatically upon receipt of a containment isolation signal after the TIP cable has been retracted. The shear valve can isolate the line if the TIP cannot be retracted or if the automatic isolation valve fails to close. The shear valves are explosive type valves designed to shear the cable and seal the guide tube. Actuation is by operator action from the Control Center. Continuity of the TIP shear valve squib firing circuits is continuously monitored in the control room to provide additional assurance that the TIP shear valves will operate as designed. In accordance with ASME Section XI Inservice Testing (IST) Program, the automatic isolation valve is given a full stroke exercise test, a fail safe test, and a stroke time test quarterly to verify its operability as the main isolation valve for these lines. Also, the proposed increase in the explosive charge testing interval would still comply with the IST requirements of ASME Section XI. Specifically ASME Section XI, paragraph IWV-3610 requires testing of the explosive charges in at least 20% of the explosive valves every two years and that charges shall not be older than 10 years. Based on the above discussion, the proposed change to the surveillance frequency will have a negligible impact on the ability of the TIP shear valve explosive

charges to function as designed. A review also confirmed that the life for each squib valve is sufficient for the extended surveillance interval.

An evaluation of any failures identified for RFO3 and RFO4 was performed to determine if the failure modes contained time based elements which could impact the one-time extension of the surveillance interval. This evaluation determined that there were no failures associated with SR 4.6.3.5.b. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

ENCLOSURE 31
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.6.4
VACUUM RELIEF SYSTEM

Justification for extending SR 4.6.4.1.b.2.a and 4.6.4.2.b.2.a

The vacuum breaker pallet is held closed by a magnet to insure the valve is fully closed. Surveillances 4.6.4.1.b.2.a and 4.6.4.2.b.2.a test that the valves are operable and not binding. The test demonstrates that the opening force to overcome the magnet and other forces such as friction is less than or equal to 0.5 psid. The justification for extending this SR is the fact that typically magnets do not strengthen over time and that the operating/maintenance experience of these valves does not indicate an adverse history of binding although there have been some problems with limit switches and test actuators as discussed below.

With regard to the magnets, if they weaken during the surveillance extension period, then the setting would be more conservative than needed; i.e. it would take less force to operate the valve. If the magnet becomes too weak, the valves could drift off their seat, however, the vacuum breakers are verified closed every 7 days using the position indication in the control room. Magnets on 2 of the suppression chamber-drywell vacuum breakers were replaced during RFO4, however, work had been performed on those vacuum breakers earlier in the outage.

With regard to their operating history, the suppression chamber-drywell vacuum breakers have experienced some problems with limit switches and test actuators. (The actuator is only used during valve testing and after a successful test it has no impact on valve operation.) These valves are tested during cold shutdown periods and not stroked during operations except after a discharge of steam to the suppression pool from a safety relief valve discharge. Therefore, no change in the pallet seating or seating position that might affect the magnet setting or limit switches is expected during the cycle extension period. If a cold shutdown period occurs during the extension period resulting in a test and subsequent failure, then repairs can be accomplished at that time. The reactor building-suppression chamber vacuum breakers are stroked monthly during operation to demonstrate their performance.

During the past 2 refueling outages there have been no failures of the reactor building-suppression chamber vacuum breakers and no experience with the suppression chamber-drywell vacuum breakers that would indicate an opening force problem.

Justification for extending SR 4.6.4.1.b.2.b and c and 4.6.4.2.b.2.c

Surveillances 4.6.4.1.b.2.b and c and 4.6.4.2.b.2.c check position indication. Closed position indication is checked every 7 days to ensure vacuum breakers are closed. The open limit switch is checked for the reactor building-suppression chamber vacuum breakers during their monthly cycling. There have been no failures of the reactor building-suppression chamber position indicators during the past 2 cycles. There have been some problems with the suppression chamber-drywell vacuum breaker limit switches that have been identified after work has been done or following cycling. However, the limit switches do not affect the ability of the vacuum breakers to perform their safety function in the opening or closing direction.

Note also that the suppression chamber-drywell vacuum breakers were last cycled in June, 1995 during a cold shutdown test thus demonstrating that the valves were not stuck closed. One of the valves required a limit switch replacement after it was stroked. Post maintenance testing verified that its position indication was functioning properly. Additionally, a test actuator was replaced. This test is within 18 months of the scheduled September 1996 refueling outage.

Justification for extending SR 4.6.4.2.b.2.b

Surveillance 4.6.4.2.b.2.b is a visual inspection of the reactor building - suppression chamber vacuum breakers. Based on the good performance history of the surveillance, the requested increased interval is not expected to have any impact on system availability.

Additionally, the amount of redundancy in the vacuum relief system makes it unlikely that the extended surveillance interval will have any effect on the ability of the system to perform its function upon demand. There are 2 reactor building-suppression chamber vacuum breakers. Only one is necessary to provide the vacuum relief function. There are 12 suppression chamber-drywell vacuum breakers but only 9 are needed for the system to perform its function.

Therefore, extension of the surveillance requirements is justified.

ENCLOSURE 32
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.6.5
SECONDARY CONTAINMENT

Justification for extending SR 4.6.5.2.b

Secondary containment dampers are cycled on a more frequent basis in accordance with the Inservice Testing Program during the operating cycle to ensure that the dampers are capable of performing their safety function. Although this test does not ensure the operability of the entire circuit logic, it does ensure the functional ability of the dampers to actuate and establish the conditions in the secondary containment assumed in the offsite dose calculations. The remaining portions of the Isolation damper circuit that is not tested during the ISI/IST program is equivalent to a logic system and the testing of these components would be equivalent to a logic system functional test.

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

> "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 system, 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."

A historical search of the 18 month surveillance tests for these valves for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a component would be unavailable. No failures were identified by this review. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

ENCLOSURE 33
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.7.1
SERVICE WATER SYSTEMS

Justification for extending SR 4.7.1.2.b

For the Emergency Equipment Cooling Water (EECW) System, a pump, valve and flow test is performed on a more frequent basis in accordance with the Inservice Testing Program during the operating cycle to ensure that this support safety function is available. Although this test does not ensure the operability of the entire EECW System logic it does ensure the functional ability of the EECW Pump, and a majority of the EECW System to produce the required EECW cooling flow to supported systems.

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

> "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 system, 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."

A historical search of the 18 month surveillance tests for this system, for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a component would be unavailable. No failures were identified by this review. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

Justification for extending SR 4.7.1.3.b

For the Emergency Equipment Service Water (EESW) System, a pump, valve and flow test is performed on a more frequent basis in accordance with the Inservice Testing Program during the operating cycle to ensure that this support safety function is available. Although this test does not ensure the operability of the entire EESW System logic, it

does ensure the functional ability of the EESW Pump, and a majority of the EESW System to produce the required EESW cooling flow to supported systems.

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

> "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 system, 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 evaluation above is applicable to Fermi 2. A historical search of the 18 month surveillance tests for this system, for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a component would be unavailable. No failures were identified by this review. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

Justification for extending SR 4.7.1.4.b

For the Diesel Generator Service Water (DGSW) System, a pump, valve and flow test is performed on a more frequent basis in accordance with the Inservice Testing Program during the operating cycle to ensure that this support safety function is available. Although this test does not ensure the operability of the entire DGSW System logic, it does ensure the functional ability of the DGSW Pump, and a majority of the DGSW System to produce the required DGSW cooling flow to supported systems.

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

> "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 system, 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 evaluation above is applicable to Fermi 2. A historical search of the 18 month surveillance tests for this system, for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a component would be unavailable. No failures were identified by this review that would adversely effect the system's ability to perform its safety function. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

ENCLOSURE 34
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.7.2
CONTROL ROOM EMERGENCY FILTRATION SYSTEM

Justification for extending SR 4.7.2.1.c.1, c.2, c.3, e.1, e.4

The Control Room Emergency Filtration System (CREFS) provides a suitable environment for continuous personnel occupancy and ensures the operability of control room equipment and instruments under accident conditions. Proper operation of the system (i.e., operation with proper flow rates for makeup and exhaust) verifies the operability of the control room pressure boundary. The system is normally in standby condition, thus gross plugging or fouling of the HEPA filters and charcoal adsorbers will be minimized. In addition, the CREFS has redundant filter trains and fans which will ensure system availability in the event of a failure of one of the system components. TS 4.7.2.1.b requires operability of the main control room fans and verification of flow through the HEPA filter and charcoal adsorbers for 10 hours every 31 days. During this testing, performance of the CREFS is demonstrated. This test would identify significant failures affecting CREFS operability, including failures to automatically initiate. Furthermore, the CREFS system is normally in standby, and as required by TS SR 4.7.2.1.c, if there is any condition such as painting or maintenance on the filter train which would impact the operability of the CREFS system, the test to verify the operability of the CREFS system will be performed. Therefore, it is concluded that because of the redundancy in the CREFS and the other required SRs the impact, if any, of postponing these surveillance tests on system availability is small. A review of the history of the SR results for RFO3 and RFO4 demonstrates that there is no evidence of any failures which would invalidate this conclusion. Therefore, this extension is justified.

Justification for extending SR 4.7.2.1.e. 2

The logic used to initiate the CREFS has been previously justified to allow its testing to be extended based on reliability studies presented in the BWR Owners Group topical report NEDC-30936P and has been accepted by the NRC (Reference NRC Safety Evaluation Report, dated August 2, 1993, for the Peach Bottom Atomic Power Station, Units 2 and 3).

The automatic actuation surveillance interval is justified to be extended based on the redundancy of the system involved, and the Inservice Testing performed during the operating cycle.

The evaluation above is applicable to Fermi 2. A historical search of the 18 month surveillance tests for this system, for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a component would be unavailable. No failures were identified by this review. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

Justification for extending SR 4.7.2.1.h

Trend analysis of past surveillances indicate no adverse trend exists and extension of the surveillance frequency should have no impact on acceptance criteria parameters. Surveillance history indicates that there has been no degradation of the silicone sealant used on this ductwork which is leak tested. Surveillance tests will be performed every 12 months as scheduled to inspect the sealant. Any deviations noted will be evaluated for impact. The 36-month frequency was established to correspond to every other refueling outage. This extension is consistent with this intention. Based on this discussion and the short period of time (48 days) requested for extension, this extension to perform duct leak testing on portions of recirculation, intake, exhaust, and Div. 2 supply plenum ductwork during RFO6 is justified.

ENCLOSURE 35
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.7.5
SNUBBERS

Justification for extending SR 4.7.5.e

Tech Spec 4.7.5.e requires functional testing of snubbers at least once per 18 months, during plant shutdowns. Fermi has chosen to use the ten percent (10%) sample plan per T. S. section 4.7.5.e. 1, therefore SRs 4.7.5.e.2 and 4.7.5.e.3 do not apply.

ASME XI, subsection IWF-5000 requires "(snubbers)... shall be tested each inspection period." The ASME Code does not specify a time frame for testing.

The primary snubber failure modes experienced at Fermi have been attributed to temperature and vibration induced degradation. Snubber degradation is therefore not simply time dependent, but rather a function of system operating time and time at elevated temperatures for snubbers in high temperature areas (e.g. Drywell, steam tunnel).

The reactor startup from RFO4 began in December 1994. Between December 1994 and the end of July 1995, the reactor has had 5.7 effective months of operation. Projecting continuous power operations between August 1, 1995 and September 27, 1996, adds an additional 14 months of operation; which provides a total of 19.7 months of plant operation for cycle 5.

During RF04, functional failures were limited to only three snubbers. The slight increase in operating time at elevated temperatures will have a negligible impact on potential failure rates. Therefore, extending the functional testing period will have little or no impact on plant safety. Therefore, the extension is justified.

ENCLOSURE 36
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.7.11
ALTERNATE SHUTDOWN OPERABILITY

In accordance with Technical Specification SR 4.7.11.1, the SBFW system is demonstrated operable every 31 days by verifying the system filled and that each valve is in its correct position, At least every 92 day the operability of or e SBFW pump is verified. SR 4.7.11.2 verifies the operation of CTG 11 Unit 1 at least once every 31 days. SR 4.7.11.3 verifies that at least one Drywell Cooling Unit is operable and available at least once every 92 days. The 18 month surveillance requirement, when assessed with the more frequent test of the surveillance program, is effectively a logic system test for the controls on the Alternate shutdown panel.

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

> "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 system, 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 evaluation above is applicable to Fermi 2. A historical search of the 18 month surveillance tests for this system, for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests for further review and evaluation. The purpose of this evaluation was to demonstrate that the increased surveillance interval would not increase the period a component would be unavailable. No failures were identified by this review. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

ENCLOSURE 37
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.8.1
DIESEL GENERATORS

Justification for extending SR 4.8.1.1.2.e.1

During the operating cycle, the diesel generators are subjected to operational testing every 31 days and fast start testing every 184 days. This testing provides confidence of diesel generator operability and the capability to perform its intended function. This testing during the past cycles has not indicated any degradation of the diesel which would negate the extension of the surveillance as requested. In addition, the past internal inspections conducted in accordance with TS SR 4.8.1.1.2.e.1 have not revealed any degradation which would necessitate replacement of internal components, although, as a matter of course, several components have been replaced as preventive measures. Therefore, the impact, if any, on system reliability will be small from the one time extension.

Fermi 2 has contacted the Diesel vendor and the vendor has concurred with the extension provided that certain additional conditions are met. The vendor has requested the following: (1) that certain additional inspections and measurements, that can be performed without a plant outage, be completed; (2) that operating data since the last inspection be provided to the vendor for trending purposes, and (3) that a vendor representative be allowed to witness a regularly scheduled full load diesel run in the future. Fermi 2 has agreed to these conditions which will provide additional confidence that the diesels are capable of performing their safety function. With these additional measures taken, the vendor agrees that the impact of the extension of overall diesel generator reliability will be minimal.

Historical testing and surveillance testing during operations have proven the ability of the Diesel engines to start and operate under various load conditions. An evaluation of failures identified for RFO3 and RFO4 was performed to determine if the failure modes contained time based elements which could impact the one-time extension of the surveillance interval. This evaluation determined that identified failures did not contain time based elements that would invalidate the conclusion that the increased operating cycle will have a small, if any, impact on system reliability. There is no reason to believe, based on diesel generator history and testing that the extension of this surveillance will have any effect on reliability. Therefore, this extension is justified.

Justification for extending SR 4.8.1.1.2.e.2, 3, 4.a, 4.b, 5, 6.a, 6.b, and 7

The design of the offsite power to the plant essential busses provides a decreased likelihood that a total loss of offsite power will occur. However, if a total loss of offsite

power were to occur and operation of the diesel generators was required, the requested extension would have minimal impact on the system failure probability. The extension of the surveillance interval for the diesel generator logic testing has in itself the same rationale for extension as LSFTs on other systems/components. Since the failure probability of the logic (relays, contacts, etc.) is reasoned, as documented in NEDC-30936P, to be less than the failure probability for the mechanical equipment (pumps, valves, etc.), the extension of the surveillance interval for the logic has minimal impact on the failure to function. And, since the mechanical components (diesel generators) are tested on a more frequent basis (i.e., monthly and 184 day by SR 4.8.1.1.2.a), the probability of failure to function is further minimized.

Historical testing and surveillance testing during operations have proven the ability of the Diesel engines to start and operate under various load conditions. An evaluation of failures identified for RFO3 and RFO4 was performed to determine if the failure modes contained time based elements which could impact the one-time extension of the surveillance interval. This evaluation determined that one failure of the Diesel Generator output breaker was caused by the Diesel Generator not reaching rated speed and voltage within sufficient time. This is the same Diesel Generator output breaker failure previously discussed in Enclosure 11. There is no reason to believe, based on diesel generator history and testing that the extension of this surveillance will have any effect on reliability. Therefore, this extension is justified.

Justification for extending SR 4.8.1.1.2.e.8

During the operating cycle, the diesel generators are subjected to operational testing every 31 days and fast start testing every 184 days. This testing provides confidence of diesel generator operability and the capability to perform its intended function. This testing during the past cycles has not indicated any degradation of the diesel which would negate the extension of the surveillance as requested. In addition, the past internal inspections conducted in accordance with TS SR 4.8.1.1.2.e.1 have not revealed any degradation which would necessitate replacement of internal components, although, as a matter of course, several components have been replaced as preventive measures.

Historical testing and surveillance testing during operations have proven the ability of the Diesel engines to start and operate under various load conditions. An evaluation of failures identified for RFO3 and RFO4 was performed to determine if the failure modes contained time based elements which could impact the one-time extension of the surveillance interval. No failures which would have prevented the EDGs from performing their safety function were identified. There is no reason to believe, based on diesel generator history and testing that the extension of this surveillance will have any effect on reliability. Therefore, this extension is justified.

Justification for extending SR 4.8.1.1.2.e.9

Auto connected loads for the diesel generators have not substantially changed since RF04. All changes to diesel generator loading have been evaluated in accordance with the appropriate section of the Diesel Generator loading calculation. The tested loads during RFO4 combined with any changes since the refueling outage do not exceed the rating requirements of Regulatory Guide 1.9 Rev. 2. There is no reason to believe, based on diesel generator history, the Fermi 2 diesel loading calculation and testing that the extension of this surveillance will have any affect on availability.

Historical testing and surveillance testing during operations have proven the ability of the Diesel engines to start and operate under various load conditions. An evaluation of failures identified for RFO3 and RFO4 was performed to determine if the failure modes contained time based elements which could impact the one-time extension of the surveillance interval. No failures were identified for this SR by the evaluation. There is no reason to believe, based on diesel generator history and testing that the extension of this surveillance will have any effect on reliability. Therefore, this extension is justified.

Justification for extending SR 4.8.1.1.2.e.10, 11, 12

The design of the offsite power to the plant essential busses provides a decreased likelihood that a total loss of offsite power will occur. However, if a total loss of offsite power were to occur and operation of the diesel generators was required, the requested extension would have minimal impact on the system failure probability. The extension of the surveillance interval for the diesel generator logic testing has in itself the same rationale for extension as LSFTs on other systems/components. Since the failure probability of the logic (relays, contacts, etc.) is reasoned, as documented in NEDC-30936P, to be less than the failure probability for the mechanical equipment (pumps, valves, etc.), the extension of the surveillance interval for the logic has minimal impact on the failure to function. And, since the mechanical components (diesel generators) are tested on a more frequent basis (i.e., monthly and 184 day by SR 4.8.1.1.2.a), the probability of failure to function is further minimized. An historical review of the load sequencer operation indicated that the timing for the load sequencer for both divisions has been found to always be in tolerance. Based on this good performance, surveillance extension to fall of 1996 is recommended. Therefore, the extensions of the surveillance intervals to reach the end date of RFO5 has minimal impact on the failure probability.

Historical testing and surveillance testing during operations have proven the ability of the Diesel engines to start and operate under various load conditions. An evaluation of failures identified for RFO3 and RFO4 was performed to determine if the failure modes contained time based elements which could impact the one-time extension of the

surveillance interval. No failures were identified by this review that adversely effect the system's ability to perform its safety function. There is no reason to believe, based on diesel generator history and testing that the extension of this surveillance will have any affect on reliability. Therefore, this extension is justified.

ENCLOSURE 38
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.8.2
BATTERIES

Justification for extending 4.8.2.1.c.3

Test results for both RF03 and RF04 indicated no significant corrosion which affected battery connection resistance. All resistance measurements were far below the maximum allowed. In addition, if corrosion is observed during the 90 day surveillance, a battery connection resistance test is required to verify resistance is below the Tech Spec limit of 150 microhms.

An evaluation of any failures identified for RFO3 and RFO4 was performed to determine if the failure modes contained time based elements which could impact the one-time extension of the surveillance interval. This evaluation did not identify any failures. Based on the previous discussion, the historical failure review, and the small impact on safety function availability the requested extension is justified.

Justification for extending 4.8.2.1.c.4

RFO1 through RF04 results showed no problems were encountered during charger tests. The chargers performed as expected. Based on previous good performance, the indicated surveillance extensions are justified.

Justification for extending SR 4.8.2.1.d.1, d.2

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. Capacity factors were greater than 100% for all batteries tested. Based on these capacity tests and the service tests during the other refueling outages, the service life has been determined by extrapolation. Based on the extrapolated rate of degradation for the worst case battery, the batteries will not reach 90% of the manufacturer's rating until their 15th year of service (2001). Based on this, and the fact that the capacity test bounds the service test, the indicated surveillance extensions are justified.

ENCLOSURE 39
JUSTIFICATION FOR SURVEILLANCE INTERVAL EXTENSION
TECHNICAL SPECIFICATION SR 4.8.4
BREAKER AND CONDUCTOR PROTECTION

Justification for extending SR 4.8.4.2.a.1.a

The Primary Containment Penetration Conductor Overcurrent Protective Devices for the reactor recirculation pumps penetration protection (overcurrent relays) were also evaluated for a 30 month period using a GE extrapolation method. The evaluated drift for these devices was found to be within Technical Specification requirements. Therefore, this one time surveillance interval extension is justified for the Primary Containment Penetration Conductor Overcurrent Protective Devices.

For the channel calibration an evaluation of any failures identified for RFO3 and RFO4 was performed to determine if the failure modes contained time based elements which could impact the one-time extension of the surveillance. The only failure identified by this review 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 the small impact on safety function availability the requested extension is justified

Justification for extending SR 4.8.4.2.a.1.b

A historical search review of the 18 month surveillance tests for SR 4.8.4.2.a.1.b for the last two refueling outages was performed. The search criteria was to identify all failed or partially failed tests. Each failed or partially failed test was to be reviewed and evaluated. The purpose of this evaluation was to demonstrate that the increased calibration surveillance interval would not increase the period a component would be unavailable. No failures were identified by this review. There were no failures of the breakers to open during these surveillance tests.

Excluding the breakers, the integrated system test of the primary containment penetration conductor overcurrent protective devices is essentially a Logic System functional test therefore the following discussion is provided.

As stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

> "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 system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability."

Based on the surveillance performance history review and the conclusions above regarding logic system functional testing, extension of this surveillance interval is justified.

NO SIGNIFICANT HAZARDS EVALUATION

ATTACHMENT 2
NO SIGNIFICANT HAZARDS EVALUATION

Information Supporting a Finding of No Significant Hazards Consideration

Detroit Edison has concluded that the proposed changes to the Fermi 2 TS, to facilitate a one-time extension in the Fermi 2 operating cycle, do not involve a Significant Hazards Consideration. In support of this determination, an evaluation of each of the three standards set forth in 10CFR50.92 is provided below.

1. The proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed TS changes involve a one-time only change in the surveillance testing intervals to facilitate a one-time only change in the Fermi 2 operating cycle. The proposed TS changes do not physically impact the plant nor do they impact any design or functional requirements of the associated systems. That is, the proposed TS changes do not significantly degrade the performance or increase the challenges of any safety systems assumed to function in the accident analysis. The proposed TS changes affect only the frequency of the surveillance requirements and do not impact the TS surveillance requirements themselves. In addition, the proposed TS changes do not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to the change in the frequency of surveillance testing. Also, the proposed TS changes do not significantly affect the availability of equipment or systems required to mitigate the consequences of an accident because of other, more frequent testing or the availability of redundant systems or equipment. Furthermore, a historical review of surveillance test results supports the above conclusions. Therefore, the proposed TS changes do not significantly increase the probability or consequences of an accident previously evaluated.

2. The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed TS changes involve a one-time only change in the surveillance testing intervals to facilitate the one-time only change in the Fermi 2 operating cycle. The proposed TS changes do not introduce any failure mechanisms of a different type than those previously evaluated since there are no physical changes being made to the facility. In addition, the surveillance test requirements themselves will remain unchanged. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed TS changes do not involve a significant reduction in a margin of safety.

   Although the proposed TS changes will result in an increase in the interval between some surveillance tests, the impact, if any, on system availability is small based on other, more frequent testing or redundant systems or equipment, and there is no evidence of any time dependent failures that would impact the availability of the systems. Therefore, the assumptions in the licensing basis are not impacted, and the proposed TS changes do not significantly reduce a margin of safety.

Based on the above, Detroit Edison has determined that the proposed amendment does not involve a significant hazards consideration.

ENVIRONMENTAL IMPACT

Detroit Edison has reviewed the proposed Technical Specification changes against the criteria of 10CFR51.22 for environmental considerations. The proposed change does not involve a significant hazards consideration, nor significantly change the types or significantly increase the amounts of effluents that may be released offsite. In addition the proposed changes will reduce occupational radiation exposure and so do not involve a significant increase in individual or cumulative occupational radiation exposures. Based on the foregoing, Detroit Edison concludes that the proposed Technical Specifications do meet the criteria given in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.

CONCLUSION

Based on the evaluation above: 1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and 2) such activities will be conducted in compliance with the Commission's regulations and proposed amendments will not be inimical to the common defense and security or to the health and safety of the public.