Fermi 2

Annual Operating Report

March 20 - December 31, 1985

Detroit Edison Company

NRC Docket No. 50-341

Facility Operating License No. NPF-43

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1.0 Introduction

The Fermi 2 Nuclear Power Plant site is located on the western shore of Lake Erie in Frenchtown Township, Monroe County, Michigan. The Nuclear Steam Supply System is a General Electric BWR 4, with a pressure-suppression Mark I containment. The unit is rated at 3292 MWt. The plant is owned jointly by the Detroit Edison Company (90 percent) and the Wolverine Power Supply Cooperative, Incorporated (10 percent). Detroit Edison has exclusive responsibility and control over the operation and maintenance of the facility.

2.0 Summary of Operating Experience

2.1 Summary of Operations

This operations summary covers the period beginning with issuance of Facility Operating License NPF-33 on March 20, 1985, through December 31, 1985. NPF-33 authorized operation of Fermi 2 to power levels not to exceed 5 percent of full power, or 165 MWt.

Fuel loading commenced immediately upon receipt of the operating license, and was completed 15 days later on April 4, 1985. Open vessel tests were then conducted and completed by mid-June. Initial criticality was achieved on June 21, 1985, marking the start of the Heatup Testing Phase. On July 15, 1985, Facility Operating License NPF-43 was issued authorizing operation of Fermi 2 at power levels not in excess of 3292 MWt, or 100 percent. Startup testing continued at power levels not exceeding 5 percent of full power through October 1985. A Confirmatory Action Letter dated July 16, 1985, was issued to Detroit Edison that restricted operating the plant above 5 percent.

On October 11, 1985, the plant entered a planned maintenance and modification outage, which continued through the end of the year. The primary planned activities for the outage included modifications to equipment to comply with 10CFR 50.49 Environmental Qualification of Electrical Equipment, and installation of additional dedicated shutdown capability as described in 10CFR 50 Appendix R, paragraph III.L.

Startup testing remained incomplete at the end of 1985, but will resume upon plant start up.

2.2 Summary of Outages and Forced Reductions Greater Than 20 Percent of Full Power

March 20 - June 20, 1985

Facility Operating (Low Power) License NPF-33 issued. Performed fuel load and open vessel testing.

June 21 - June 30, 1985

Achieved initial criticality. Commenced heatup testing.

Plant Shutdowns:

June 28, forced shutdown - 20.8 hours.

A manual scram was initiated in accordance with procedure when a CRD pump tripped due to a low suction pressure transient. The transient occurred when a Torus Water Management System (TWMS) valve, in the same supply line feeding the CRD pump, was opened to raise the water level in the Torus. Reported as LER 85-029, transmitted July 24, 1985, by NP-85-831.

July

Facility Operating (Full Power) License NPF-43 issued. Continued low power testing.

Plant Shutdowns:

July 1, forced shutdown - 10.3 hours.

Reactor scram caused by a spurious reactor water level 3 signal generated when an instrument valve was opened during surveillance testing of a narrow range reactor water level instrument. Reported as LER 85-030, transmitted July 31, 1985, by NP-85-854, and 85-030-01, transmitted December 12, 1985, by NP850248.

July 5, forced shutdown - 9.6 hours.

Reactor trip resulted from a water level transient that occurred when the turbine bypass valve controller malfunctioned causing the bypass valves to close. Reported as LER 85-033, transmitted August 14, 1985, by NP-85-0007.

Plant Shutdowns:

July 9, forced shutdown - 9.9 hours.

Reactor trip resulting from a slow reactor water level decrease that started when reactor pressure was manually increased. Reported as LER 85-035, transmitted August 8, 1985, by NP-85-0016.

July (Continued)

July 24 - July 31, scheduled shutdown - 176.8 hours.

South Reactor Feedpump turbine thrust bearing failure occurred. To avoid a similar failure of the north turbine, the reactor was shut down and the south turbine repaired and the north inspected. Other major activities completed at this time were surveillance tests of Anticipated Transients Without Scram (ATWS) instrumentation, and replacement of vent valves on insert and withdrawal lines of hydraulic control units in the Control Rod Drive system.

August

Continued low power testing.

Plant Shutdowns:

August 1 - August 10, scheduled shutdown - 227.02 hours.

Continuation of outage that began on July 24.

September

Continued low power testing.

Plant Shutdowns:

September 3, forced shutdown - 247.15 hours.

Improper refill of a condenser pressure sensing line caused a Group 1 isolation; resulting in closure of all Main Steam Isolation Valves (MSIV's). This produced an increase in reactor pressure which caused a reactor high pressure trip. Reported as LER 85-059, transmitted October 3, 1985, by NP850124. While the plant was shut down, maintenance was performed on valves in the HPCI, RHR, Primary Containment Monitoring, and Standby Gas Treatment Systems.

Plant Shutdowns:

September 27, forced shutdown - 27.85 hours.

Reactor trip due to excessive noise signal received by run-up module of the west turbine bypass valve. Valve went full closed while main turbine run-up testing was in progress. Reported as LER 85-066, transmitted October 25, 1985, by NP850171.

October

Low power testing continued until the start of a scheduled outage that began October 11, 1985.

Plant Shutdowns:

October 1, forced shutdown - 42.9 hours.

An automatic reactor scram occurred when a discontinuity in the motorized potentiometer that derives the reactor pressure setpoint signal caused both turbine bypass valves to go full open. Reactor pressure was being increased for testing at the time of the scram. Reported as LER 85-068, transmitted October 30, 1985, by NP-85-0187.

October 11, forced shutdown

Reactor water level 3 reactor trip. Cause unknown. Reactor tripped while plant was shutting down for a scheduled maintenance outage and is still in the outage at this time. Reported as LER 85-071, transmitted November 7, 1985, by NP-85-0199.

October 11 - December 31, 1985 - 1965.9 hours

Scheduled plant shutdown. This was the plant's first major planned outage. Major items completed during this outage included equipment environmental qualification modifications, installation and testing of a panel that provides additional dedicated shut down capability, Main Steam Bypass Lin replacement, South Reactor Feedpump turbine repairs and modifications, Main Condenser cleaning, painting and inspection, Emergency Diesel Generator repair and testing, Residual Heat Removal pump motor "B" replacement and Thermal Recombiner seal replacement. This outage continued through the remainder of the reporting period.

2.3 Fuel Performance

Initial loading of the Fermi 2 core began on March 20, 1985 and was completed April 4, 1985. Initial criticality was achieved on June 21, 1985. During the period from June 21, 1985 through October 11, 1985 the core was operated at various power levels up to 164 MWt (5% of rated capacity). The reactor has been shut down since October 11, 1985 for a planned outage.

Low Power Physics Test conducted subsequent to fuel loading demonstrated the shutdown margin to be above the Technical Specification minimum requirement of 0.38% $\Delta k/k$, and a reactivity anomaly which is within the allowable range of $\pm$ 1% $\Delta k/k$.

The fuel has experienced a few power transients during startup testing. The maximum power level obtained during transients was 207 MWt (6.3% of rated). However, monitoring of chemistry samples has not produced any indication of fuel failure, and offgas activity has essentially remained below the lower level of detection.

2.4 Shore Barrier Survey

A survey of the Fermi 2 shore barrier was completed on September 6, 1985, per procedure 43.000.01 "Shore Barrier Surveillance", and as required by Technical Specification 3.7.3. The results of the survey indicated no damage, or significant movement or deterioration of the barrier. All of the survey point elevations were within the tolerance specified in Technical Specification Table 3.7.3-1. Civil Engineering drawings 6C721-44 through 49 were revised to incorporate the survey data. No unusual events occurred in 1985 that would have required additional surveillances.

2.5 Safely Relief Valve Challenges

There were no Safety Relief Valve challenges during 1985.

2.6 Personnel Monitoring and Exposure

Pursuant to 10CFR 20.407(a)(2), a tabulation of the number of individuals for whom monitoring was provided is shown in Table 2.6-1.

Table 2.6-2 provides a breakdown of radiation exposure by work and job function as required by Technical Specification 6.9.1.5(a).

Table 2.6-1

Statistical Summary Report
of the Number of Individuals for Whom
Personnel Monitoring was Provided.

For the Period
January 1, 1985 to December 31, 1985

NUMBER OF INDIVIDUALS IN EACH RANGE	ESTIMATED WHOLE BODY EXPOSURE RANGE (Rems)
3245	NO MEASURABLE EXPOSURE
323	MEASURABLE EXPOSURE 0.1
2	0.10 TO 0.25
1	0.25 TO 0.50
0	0.50 TO 0.75
0	0.75 TO 1.00
0	1.00 TO 2.00
0	2.00 TO 3.00
0	3.00 TO 4.00
0	4.00 TO 5.00
0	5.00 TO 6.00
0	6.00 TO 7.00
0	7.00 TO 8.00
0	8.00 TO 9.00
0	9.00 TO 10.00
0	10.00 TO 11.00
0	11.00 TO 12.00
0	12.00+

Table 2.6-2

ANNUAL REPORT
NUMBER OF PERSONNEL AND MAN-REM BY WORK AND JOB FUNCTION

For the Period January 1, 1985 to December 31, 1985

Work & Job Function	Number of Personnel [> 100 mrem]			Total Man-Rem		
	Station Employees	Utility Employees	Contractors and Others	Station Employees	Utility Employees	Contractors and Others
Reactor Oper & Surveillance						
Maintenance Personnel	0	0	0	0.474	0.000	0.510
Operating Personnel	0	0	1	1.407	0.043	0.980
Health Physics Personnel	0	0	0	0.171	0.000	0.307
Supervisory Personnel	0	0	0	0.117	0.000	0.091
Engineering Personnel	0	0	0	0.445	0.030	1.220
Routine Maintenance						
Maintenance Personnel	0	0	0	0.456	0.002	1.878
Operating Personnel	0	0	0	0.004	0.000	0.100
Health Physics Personnel	0	0	0	0.008	0.000	0.000
Supervisory Personnel	0	0	0	0.003	0.000	0.093
Engineering Personnel	0	0	0	0.101	0.000	0.200
Inservice Inspection						
Maintenance Personnel	0	0	0	0.005	0.000	0.004
Operating Personnel	0	0	0	0.000	0.000	0.000
Health Physics Personnel	0	0	0	0.000	0.000	0.015
Supervisory Personnel	0	0	0	0.000	0.000	0.000
Engineering Personnel	0	0	0	0.000	0.000	0.015
Special Maintenance						
Maintenance Personnel	0	0	0	0.134	0.000	0.221
Operating Personnel	0	0	0	0.015	0.000	0.124
Health Physics Personnel	0	0	0	0.045	0.000	0.058
Supervisory Personnel	0	0	0	0.017	0.000	0.008
Engineering Personnel	0	0	2	0.022	0.000	0.485
Waste Processing						
Maintenance Personnel	0	0	0	0.084	0.000	0.022
Operating Personnel	0	0	0	0.023	0.000	0.493
Health Physics Personnel	0	0	0	0.007	0.000	0.058
Supervisory Personnel	0	0	0	0.015	0.000	0.012
Engineering Personnel	0	0	0	0.000	0.000	0.000
Refueling						
Maintenance Personnel	0	0	0	0.002	0.000	0.000
Operating Personnel	0	0	0	0.005	0.000	0.000
Health Physics Personnel	0	0	0	0.001	0.000	0.001
Supervisory Personnel	0	0	0	0.000	0.000	0.000
Engineering Personnel	0	0	0	0.008	0.000	0.009
TOTAL						
Maintenance Personnel	0	0	0	1.155	0.002	2.635
Operating Personnel	0	0	1	1.454	0.043	1.697
Health Physics Personnel	0	0	0	0.232	0.000	0.439
Supervisory Personnel	0	0	0	0.152	0.000	0.204
Engineering Personnel	0	0	2	0.576	0.030	1.929
Grand Total	0	0	3	3.569	0.075	6.904

2.7 ECCS Outages

Pursuant to Fermi 2 Technical Specification 6.9.1.5.c. a summary of ECCS System Outages which occurred between March 20, 1985 and December 31, 1985 is provided. The tabulation of total outage hours (Table 2.7-1) includes both forced and planned outages.

As shown in Figure 2.7-1, equipment failures which would have forced a system outage but were discovered during a planned outage have been reported as separate outages.

ECCS Outage Hours
March 20, 1985 through December 31, 1985

ECCS System	Forced Hours	Total Hours
Division I LPCI	25.0	1,434.2
Division II LPCI	913.9	2,494.4
Division I Core Spray	20.8	313.0
Division II Core Spray	152.8	670.0
HPCI	958.8	2,423.3
ADS	0.0	0.0

Table 2.7-1

ECCS System: Division I & II LPCI
Out of Service From 0930 Hrs. 10-11-85 to 2200 Hrs. 10-19-85
Duration 204.5 hrs. Forced ___ Planned X

Outage Summary

Both divisions of RHR were declared inoperable to facilitate installation of a temporary modification. This temporary modification defeated Division II LPCI loop select logic, and would have caused Division I LPCI to be selected if the LPCI subsystem would have been required to function. The temporary modification was installed while a permanent modification was made to the LPCI loop select logic; which was a change to alleviate relay chatter. The temporary modification was removed after the permanent modification was completed, and the RHR system was returned to service.

ECCS System: Division I & II LPCI
Out of Service From 0500 Hrs. 10-27-85 to 1356 Hrs. 11-12-85
Duration 394 hrs. Forced ___ Planned X

Outage Summary

Both divisions of RHR were removed from service for an extended system outage. During the outage, modifications to facilitate the installation of a dedicated shutdown panel were completed. In addition, the LPCI Loop Selection Logic was modified and motor operators were replaced on the RHR Loop A Containment Spray/Test Isolation Valve, the LPCI Loop A Outboard Injection Isolation Valve, and the LPCI Loop A Inboard Isolation Valve. Upon completion of the above modifications, the systems were tested and returned to service.

ECCS System: Division I LPCI
Out of Service From 1330 Hrs. 4-3-85 to 1530 Hrs. 4-13-85
Duration 242 hrs. Forced ___ Planned X

Outage Summary

Division I of the RHR system was removed from service for an extended system outage. During the outage, several snubbers were removed for flushing and functional testing. Upon completion of outage work items the system was returned to service.

ECCS System: Division I LPCI
Out of Service From 1900 Hrs. 5-19-85 to 0340 Hrs. 6-11-85
Duration 536.7 hrs. Forced ___ Planned X

Outage Summary

Division I RHR was removed from service to repair a leaking return line check valve. The return line check valve was repacked and its limit switch repaired.

ECCS System: Division I LPCI
Out of Service From 1200 Hrs. 11-27-85 to 1300 Hrs. 11-28-85
Duration 25 hrs. Forced X Planned ___

Outage Summary

During a system walkdown conducted while vibration testing the RHR piping system, a broken rod was discovered on a spring support for Loop A RHR discharge line. Examination of other pipe supports on the system revealed no damage. The broken rod was replaced and the system returned to service.

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ECCS System: Division I LPCI
Out of Service From 2030 Hrs. 12-10-85 to 0430 Hrs. 12-12-85
Duration 32 hrs. Forced ___ Planned X

Outage Summary

The RHR Pump C motor was removed from service for inspection. Upon completion of motor inspection the pump was returned to service.

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ECCS System: Division II LPCI
Out of Service From 0900 Hrs. 4-17-85 to 1530 Hrs. 5-2-85
Duration 366.5 hrs. Forced ___ Planned X

Outage Summary

Division II RHR was removed from service to allow several snubbers to be removed from the system. The snubbers were removed for reservoir flushing and functional testing. In addition, RHR Pump D was reterminated with new cement, and a RHR Room Cooler thermocouple was recalibrated. Upon completion of the above items the snubbers were replaced and the RHR System returned to service.

ECCS System: Division II LPCI
Out of Service From 1900 Hrs. 5-19-85 to 0500 Hrs. 6-14-85
Duration 610 hrs. Forced ___ Planned X

Outage Summary

Division II RHR was removed from service to allow repairs to be made to a leaking check valve in the RHR return line. The check valve was repacked and its limit switch repaired.

ECCS System: Division II LPCI
Out of Service From 1510 Hrs. 9-8-85 to 0001 Hrs. 9-10-85
Duration 32.9 hrs. Forced X Planned ___

Outage Summary

Division II LPCI was declared inoperable because of a Division II LPCI Loop Select Logic failure. The logic failure was caused by a defective capacitor in a signal conditioning card. The defective card was replaced and the Loop Select Logic functionally tested. After successful testing, the Division II LPCI System was returned to service.

ECCS System: Division II LPCI
Out of Service From 1000 Hrs. 10-11-85 to 0400 Hrs. 10-20-85
Duration 210 hrs. Forced ___ Planned X

Outage Summary

Division II of the RHR System was removed from service to permit changing the motor operators for the RHR Heat Exchanger "B" Relief Valve, LPCI Loop B Inboard Isolation Valve, LPCI Loop B Outboard Injection Isolation Valve, and RHR Loop B Containment Spray/Test Isolation Valve. Local Leak Rate Testing was also performed on these valves. Also during the outage the Reactor Steam Dome Pressure Transmitter was replaced, and the RHR Division II Room Cooler was removed from service to permit thermocouple recalibration. Upon completion of these items, the system was returned to service.

ECCS System: Division II LPCI
Out of Service From 0700 Hrs. 11-25-85 to 2400 Hrs. 12-31-85
Duration 881 hrs. Forced X Planned ___

Outage Summary

While operating in the shutdown cooling mode, RHR pump B motor tripped on overcurrent. The cause of the overcurrent was a phase to phase short. The short circuit occurred when pieces of metal (two fingers) from the rotor assembly broke and contacted the phase windings. An investigation showed that the failure was caused by a lack of process control during manufacture followed by low amplitude cyclic stress during operation. The RHR Pump B motor was replaced with a motor obtained from the Browns Ferry plant. The remaining three RHR pump motors at Fermi 2 were all built about 10 years before the B pump motor. Inspection of one of these determined that these pump motors are acceptable.

ECCS System: Division II LPCI
Out of Service From 1730 Hrs. 12-11-85 to 1430 Hrs. 12-13-85
Duration 50 hrs. Forced ___ Planned X

Outage Summary

The RHR Pump D Motor was removed from service for inspection. Upon completion of the motor inspection the pump was returned to service.

ECCS System: Division II LPCI
Out of Service From 1030 Hrs. 12-29-85 400 Hrs. 12-31-85
Duration 61.5 hrs. Forced ___ Planned X

Outage Summary

RHR Pump D was removed from service for preventive maintenance on the Pump D Breakers. Upon completion of the breaker maintenance the pump was returned to service.

ECCS System: Division I Core Spray
Out of Service From 2125 Hrs. 4-10-85 to 0630 Hrs. 4-11-85
Duration 9 hrs. Forced X Planned ___

Outage Summary

The Division I Core Spray System was declared inoperable after the Division I Core Spray Room Cooler failed to start when required. The cause of the failure was a defective relay. The defective relay was replaced and the room cooler returned to service.

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ECCS System: Division I Core Spray
Out of Service From 1500 Hrs. 6-18-85 to 0246 Hrs. 6-19-85
Duration 11:75 hrs. Forced X Planned ___

Outage Summary

During performance of a routine system surveillance Core Spray Pump C failed to start. The cause of the failure was a dirty contact on a relay. The contact was cleaned and the surveillance reperformed. Upon successful completion of the surveillance, the system was returned to service.

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ECCS System: Division I Core Spray
Out of Service From 0600 Hrs. 10-20-85 to 2130 Hrs. 10-29-85
Duration 231.5 hrs. Forced ___ Planned X

Outage Summary

While the reactor was in cold shutdown for a plant outage. Division I of the Core Spray System was removed from service for preventive maintenance and modification to the 64C Bus. The system was returned to service upon completion of these items.

ECCS System: Division I Core Spray
Out of Service From 0700 Hrs. 12-7-85 to 1940 Hrs. 12-9-85
Duration 60.7 hrs. Forced ___ Planned X

Outage Summary

Division I Core Spray was removed from service to allow the oil to be changed in pump motors A and C. Upon completion of the oil change, Division I Core Spray was returned to service.

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ECCS System: Division II Core Spray
Out of Service From 2300 Hrs. 4-17-85 to 0930 Hrs. 5-2-85
Duration 346.5 hrs. Forced ___ Planned X

Outage Summary

Division II Core Spray was removed from service to allow recalibration and functional testing of a room cooler thermocouple. In addition, several snubbers were removed for hydraulic reservoir flushing and filling. Upon completion of the recalibration of the thermocouple and functional testing and reinstallation of the snubbers, the Core Spray system was returned to service.

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ECCS System: Division II Core Spray
Out of Service From 1345 Hrs. 4-25-85 to 2230 Hrs. 5-1-85
Duration 152.75 hrs. Forced X Planned ___

Outage Summary

While Division II Core Spray was removed from service for a planned system outage, a broken thermocouple was discovered on Core Spray Pump D Motor. The broken thermocouple was repaired, however, the system remained out of service until the completion of the planned system outage.

ECCS System: Division II Core Spray
Out of Service From 0642 Hrs. 11-4-85 to 1235 Hrs. 11-10-85
Duration 150 hrs. Forced ___ Planned X

Outage Summary

While the plant was in a cold shutdown condition, Division II of the Core Spray System was removed from service for preventative maintenance. Upon completion of the maintenance item the system was returned to service.

ECCS System: Division II Core Spray
Out of Service From 0620 Hrs. 12-14-85 to 0300 Hrs. 12-15-85
Duration 20.7 hrs. Forced ___ Planned X

Outage Summary

The Core Spray System was removed from service to allow inspection of Pump Motors B and D. The pump motors were inspected and the system returned to service.

ECCS System: HPCI
Out of Service From 1400 Hrs. 6-21-85 to 1400 Hrs. 6-23-85
Duration 48 hrs. Forced ___ Planned X

Outage Summary

The HPCI system was removed from service to install spectacle flange in pump flow orifice. Reactor pressure was maintained below 150 psig while the spectacle flange was installed. Upon completion of spectacle flange installation the system was returned to service.

ECCS System: HPCI
Out of Service From 0300 Hrs. 7-2-85 to 0600 Hrs. 7-3-85
Duration 27 hrs. Forced ___ Planned X

Outage Summary

The HPCI System was declared inoperable while a damaged ribbon cable on the HPCI turbine controls was being replaced. During the outage the reactor was in the startup mode and pressure was maintained below 150 psig. The cable replacement was completed and the system returned to service.

ECCS System: HPCI
Out of Service From 0415 Hrs. 7-4-85 to 0200 Hrs. 7-5-85
Duration 21.75 hrs. Forced ___ Planned X

Outage Summary

The HPCI System was removed from service to allow the spectacle flange to be changed. The reactor pressure was maintained below 150 psig. Spectacle flange was changed and system returned to service.

ECCS System: HPCI
Out of Service From 0300 Hrs. 7-11-85 to 0600 Hrs. 8-13-85
Duration 795 hrs. Forced X Planned ___

Outage Summary

The HPCI system was declared inoperable while steam balance chamber pressure adjustments were being performed. The reactor was maintained at rated pressure to facilitate HPCI, and control rod testing. Upon completion of this testing reactor pressure was maintained below 150 psig until the HPCI System was returned to service.

ECCS System: HPCI
Out of Service From 1058 Hrs. 7-16-85 to 1505 Hrs. 7-16-85
Duration 4.0 hrs. Forced X Planned ___

Outage Summary

The HPCI System was removed from service after an oil leak was discovered at the operator of the turbine stop valve during demonstration testing of the HPCI speed control governor. The leak was caused by a loose flange between the pilot valve assembly and the hydraulic cylinder of the HPCI turbine stop valve operator. The bolts at the flange were tightened about a quarter turn and the leak stopped. The system was then returned to service. The event was reported as LER 85-039 in transmittal letter NP850034, dated August 15, 1985.

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ECCS System: HPCI
Out of Service From 1940 Hrs. 7-21-85 to 1420 Hrs. 7-22-85
Duration 18.7 hrs. Forced X Planned ___

Outage Summary

During demonstration testing of the HPCI speed control governor, the HPCI low oil pressure alarm annunciated in the control room and the HPCI turbine was manually tripped. The cause of the low oil pressure was a blank gasket in the line between the discharge of the oil cooler and one inlet of the oil temperature control valve. The blank gasket was installed as part of a temporary modification and was inadvertently left in the line when the modification was removed. The gasket was removed from the line and an inspection was made of the inlet flange for a similar gasket. None was found. Proper oil flow through the system was verified and the oil system was restored. The event was reported as LER 85-041 in transmittal letter NP850042, dated August 21, 1985.

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ECCS System: HPCI
Out of Service From 0830 Hrs. 8-13-85 to 2138 Hrs. 8-15-85
Duration 61.75 hrs. Forced ___ Planned X

Outage Summary

HPIC removed from service to place high pressure orifices in the test return line, and also to install diodes on several valves. The above modifications were completed and the system returned to service.

ECCS System: HPCI
Out of Service From 2257 Hrs. 8-28-85 to 0655 Hrs. 8-31-85
Duration 56 hrs. Forced X Planned ___

Outage Summary

The HPCI System was declared inoperable after receiving an isolation signal from a high steam flow trip unit which had failed downscale. The downscale failure was caused by a poor connection between an amplifier card and the pins into which the card plugs. The amplifier card was not properly seated. The card was securely reseated and an operability surveillance performed on the HPCI Testability Loop. The HPCI system was then returned to service. The event was reported in LER 85-055 in transmittal letter NP850104, dated September 27, 1985.

ECCS System: HPCI
Out of Service From 1745 Hrs. 8-31-85 to 0530 Hrs. 9-5-85
Duration 107.75 hrs. Forced X Planned ___

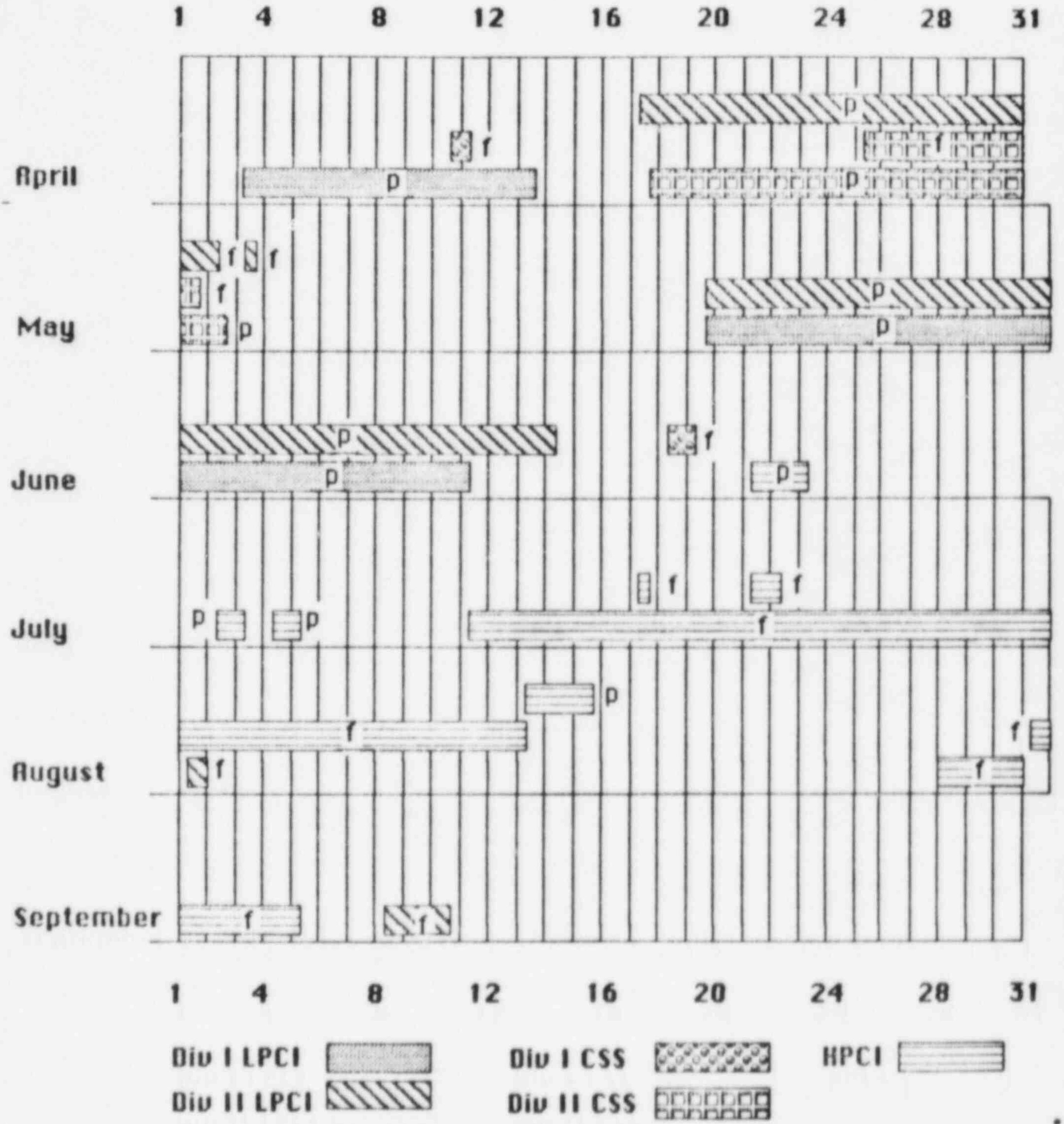
Outage Summary

After a HPCI test run, the suction relief valve was found to be leaking. The valve could not be reseated. The leaking valve was then isolated. Upon disassembly a small section of stainless steel wire was discovered lodged between the seat and disk. The valve was reworked and reseated.

ECCS System: HPCI
Out of Service From 1728 Hrs. 10-11-85 to 0330 Hrs. 12-5-85
Duration 1,306 hrs. Forced ___ Planned X

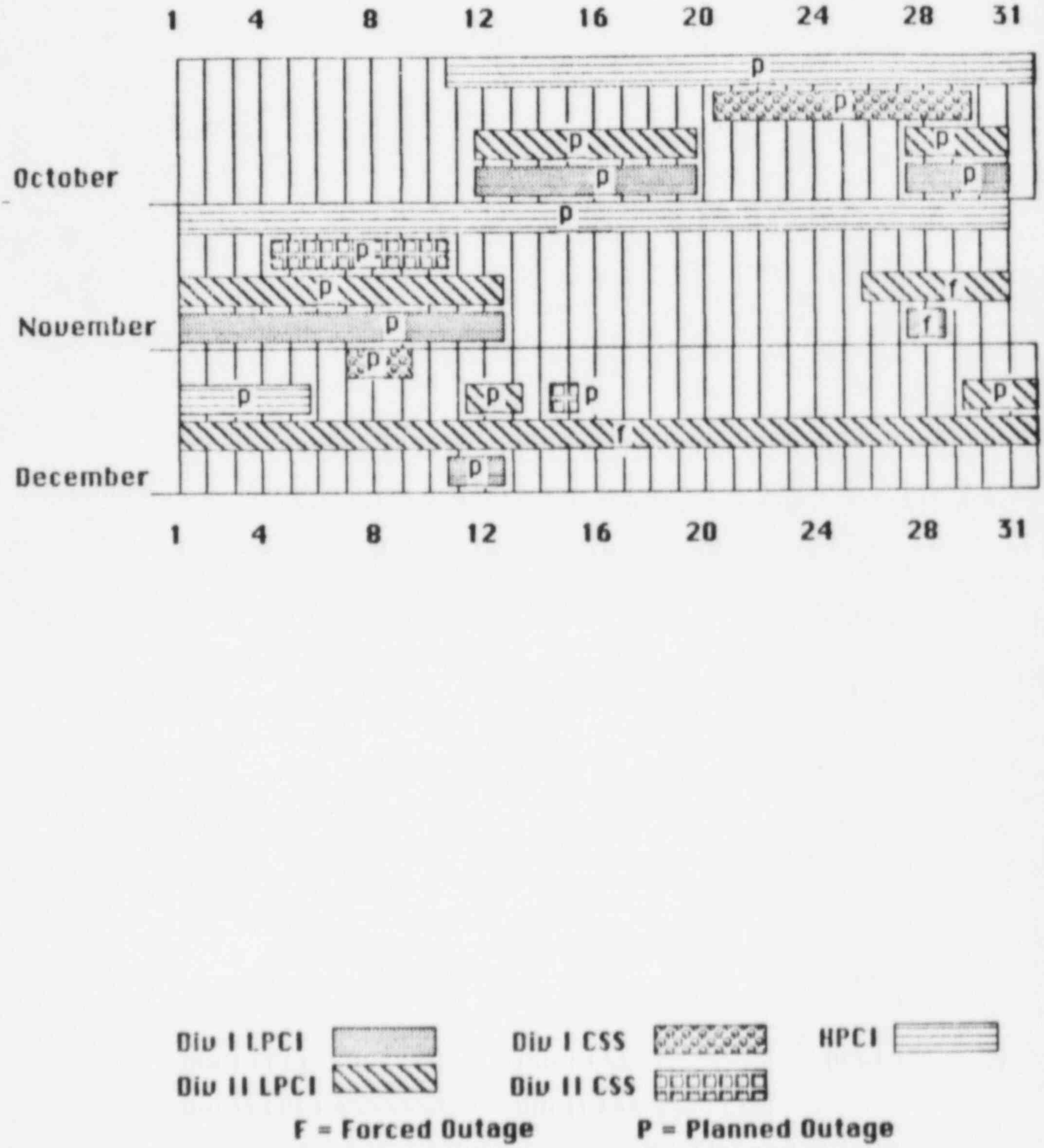
Outage Summary

While the reactor was in cold shutdown for an extended plant outage the HPCI System was removed from service to perform a cold alignment and to allow measurement of the vane tip clearances on the HPCI booster pump. The HPCI System was realigned and returned to service.



ECCS Outages

Fig. 2.7 - 1



ECCS Outages

Fig. 2.7 - 1

3.0 Facility Changes, Tests and Experiments

This section is provided in accordance with the requirements of 10CFR 50.59(b). This regulation allows licensees to make changes in the facility as described in the Safety Analysis Report, make changes in procedures as described in the Safety Analysis Report, and conduct tests and experiments not described in the Safety Analysis Report, without prior NRC approval, provided the change, test or experiment does not involve a change in the Technical Specifications or an unreviewed safety question. 10CFR 50.59(b) requires that a brief description of such changes, tests and experiments, including a summary of the safety evaluation of each, be reported on an annual basis.

3.1 Design Changes

There were no design changes during 1985 which introduced an unreviewed safety question and, therefore, prior NRC approval was not required.

The following summary of design changes includes those significant modifications that were completed between March 20 and December 31, 1985. Many of these design changes are not specifically required to be reported by 10CFR 50.59(b) since they do not constitute a change in the facility as described in the Safety Analysis Report. However, they are considered to be of significance, warranting mention in this report.

System T82-00 Implementing Document No. EDP-2943 Rev. A

Title of Change: Reactor Bldg. Fifth Floor Infrared Fire Detection System

Summary:
This modification replaced the ionization fire detector system located on the 5th floor Reactor Building with an infrared fire detector system. Rather than change the ionization fire detector system, the infrared detector system was installed instead, to conform with the National Fire Protection Association (NFPA) Code Section 72E.

Safety Evaluation:

Upgrade of the detection system to NFPA 72E assures proper operation of the fire detection system in accordance with FSAR commitments.

System B21, E11, E41, E51, G33 Implementing Document No. EDP-1860 Rev. 0

Title of Change: Modify Power Supply Feed to Steam Leak Detection Temperature Monitors

Summary:
The Steam Leak Detection temperature monitor modules were found to generate a trip or alarm when the 120VAC power to them was interrupted and then reapplied. The existing power feed was disconnected and the temperature monitors were reconnected to the Division I B21B-K801A and Division II B21B-K801B inverters to reduce the number of power interruptions. Additional fuse protection was also provided.

Safety Evaluation:
This modification will provide a more reliable power source to the Steam Leak Detection System and will reduce challenges to safety systems. Additional fusing provides protection to the inverters.

System T4102 Implementing Document No. EDP-2838 Rev. A

Title of Change: Modification of Mechanical Equipment Room Return Air Damper Control

Summary:
This EDP changed the operation of the mechanical equipment room return air damper from closed position to open position while operating the Control Center HVAC system in the chlorine and recirculation modes. This change will prevent the mechanical equipment room from becoming overpressurized while operating the Control Center HVAC in the recirculation or chlorine mode. Overpressurization was occurring because the air supply from the HVAC system was greater than leakage from the room.

Safety Evaluation:
This change allows the Mechanical Equipment Room to have a slightly positive pressure while operating the Control Center HVAC in the chlorine or recirculation modes. Operating with a slight positive pressure was the original design intent.

System G33 Implementing Document No. EDP-4154 Rev. A

Title of Change: Addition of a 0.5 Second Time Delay to the Room Ambient and Differential Temperature String of the RWCU Steam Leak Detection System

Summary:
This EDP provided instruction to replace existing control relays with time delay relays to introduce a 0.5 second time delay in the response of the Reactor Water Cleanup system ambient and differential temperature isolation scheme. This change was performed to eliminate isolations of inboard and outboard primary containment isolation valves G33-F001 and G33-F004 that are caused by spurious signals.

Safety Evaluation:
This change was reviewed against the Pipe Rupture Analysis Outside Containment contained in Appendix C of the FSAR. The added time delay did not change the results of the safety analysis. In addition, this change will reduce challenges to safety systems.

System B31 Implementing Document No. EDP-3871 Rev. A

Title of Change: Electrical De-energization of the Hatch ATWS Trip on the Recirculation Pump Motor Generator Drive Breakers

Summary:
This design change de-energized the Hatch ATWS trip. This trip functioned by tripping the breakers to the drive motor of the Reactor Recirculation Motor-Generator sets. The change was accomplished by removing the fuses and tagging the fuse holders for that portion of the logic. This change was performed to electrically separate safety-related and nonsafety-related equipment.

Safety Evaluation:
This change results in increased electrical separation. No change in transient analysis will result from removal of this trip. The ATWS trip on the M-G set field breaker is unaffected by this change.

System T23-04 Implementing Document No. EDP-1996 Rev. A

Title of Change: Test, Vent and Drain Connection Caps

Summary:
Test, vent and drain connections that are in piping which is part of the Primary Containment boundary, require caps to be installed. This EDP identified the connections requiring caps and provided instructions for their installation.

Safety Evaluation:
Caps will provide additional assurance of non-leakage through test, drain and vent connections.

System N30-12 Implementing Document No. EDP-4729 Rev. 0

Title of Change: East & West Bypass Valve Module Failure Defeat

Summary:
During initial testing and operation of the Main Turbine pressure control system, a number of bypass valve trips occurred due to "module failure" protection. This has usually occurred when the bypass valves have been stroked by a fast oscillating pressure demand signal. This EDP defeated the tripping action of the bypass valve "module failure" circuit so that the valves are not tripped on valve position error.

Safety Evaluation:
Removing the module failure trip feature from the bypass valve controls improves pressure regulation during transients by preventing spurious valve closure on position error, thereby reducing overall pressure transients, and reducing challenges to safety systems.

System D11 Implementing Document No. EDP-1795 Rev. B

Title of Change: Installation of Heat Trace on the Sample Lines to the SGTS Effluent Radiation Monitors

Summary:
This EDP provided instruction for the installation of electric heat tracing on the sample lines to the Standby Gas Treatment System (SGTS) effluent radiation monitors. This included sample lines to the SGTS Radiation Monitoring system (RMS) and the SGTS Post-Accident Radiation Monitoring system (PARMS). The heat tracing was added to prevent condensation from forming within the lines, which could affect radiation monitoring equipment performance.

Safety Evaluation:
Heat tracing provides no control function. The heat tracing of these sample lines assures radiation monitoring equipment performance.

System C36 Implementing Document No. EDP-1697 Rev. 0

Title of Change: Dedicated Shutdown System Control Panel H21-P623 Fabrication and Installation

Summary:
This EDP provided the design and installation of controls and instrumentation in the dedicated shutdown control panel (H21-P623). This panel was installed to provide dedicated shutdown capability for fire zones defined in FSAR 9B.3.4.

Safety Evaluation:
The panel has only nonsafety-related equipment installed. Interaction of this equipment with safety-related systems was reviewed. It was determined that this design does not adversely affect the results of transients analyzed in Chapter 15B of the FSAR.

System C36 Implementing Document No. EDP-1701 Rev. A

Title of Change: Dedicated Shutdown System - BOP Switchgear and MCC Modification, Lighting, Communications, Supervisory Control Fermi I Relay Room Modifications, 120 KV Switchyard Modifications

Summary:
This Engineering Design Package provided the design to interface Balance of Plant (BOP) equipment associated with the dedicated shutdown control panel. This interface allows the operation of switchgear, motor control centers and other equipment necessary to operate the dedicated shutdown system.

Safety Evaluation:
The modifications in this EDP are limited to nonsafety-related systems.

System C36 Implementing Document No. EDP-1702 Rev. B

Title of Change: Dedicated Shutdown System - Class IE MCCs and 4160 V Switchgear Modification, Instrumentation, Installation and SRV Control Modification

Summary:
The EDP provided the design to interface safety related equipment associated with the dedicated shutdown control panel. This interface allows the operation of switchgear, motor control centers, valves, and other equipment necessary to operate the dedicated shutdown system.

Safety Evaluation:
The changes made in this design maintained the integrity of safety-related equipment. This modification does not adversely affect the results of transients analyzed in Chapter 15B of the FSAR.

System G41 Implementing Document No. EDP-1946 Rev. B

Title of Change: Addition of Isolation Valve on Condensate Supply Header to Fuel Pool Cleanup and Cooling Deminerlizers

Summary:
This EDP directed the installation of an additional manual operated valve in the Condensate supply header to the Fuel Pool Cooling and Cleanup (FPCCU) demineralizers. This change will provide the capability of isolating the condensate supply to the FPCCU demineralizers without isolating other equipment supplied by the header.

Safety Evaluation:
This is a nonsafety-related system, and the change does not affect any safety functions.

System T41, P80 Implementing Document No. EDP-2190 Rev. A

Title of Change: Auxiliary Building Fire Dampers

Summary:
This EDP provided instruction for the installation of six additional, three hour rated fire dampers in the Control Center and reactor/auxiliary building HVAC systems. All installations were in the auxiliary building. The addition of the dampers was required to complete fire barrier requirements per FSAR Section 9B and in response to Detroit Edison's Final Report of 10CFR50.55(e) Item 147 (Letter EF2-70396 Dated February 28, 1985).

Safety Evaluation:
This modification was required to complete fire barrier features as described in FSAR Section 9B.

System D11-10 Implementing Document No. EDP-2217 Rev. 0

Title of Change: Automatic Reset Control Center Emergency Air Inlet Radiation Monitors

Summary:
This EDP converted Control Center Emergency Air Inlet Radiation Monitors (D11-K836A, D11-K836B, D11-K837A, D11-K837B) to automatic reset thereby precluding potentially undesirable operation of Emergency Air Inlet Dampers following high radiation trips. This change required the installation of an automatic reset jumper in each of the bistable trip unit circuit boards (4 total) contained in the above modules. The modules are identical radiation analyzer readout modules (Model RP-30A) manufactured by General Atomic Company.

Safety Evaluation:
Minor modifications were made to provide automatic reset of the trip units. No changes to performance characteristics (accuracy, drift, etc.) were made. Therefore, the margin of safety has not been decreased.

System E41-01 Implementing Document No. EDP-3444 Rev. 0

Title of Change: HPCI Turbine Overspeed Trip Modification

Summary:
The modification consisted of replacing the existing ball tappet assembly and piston with a solid mushroom head tappet assembly and slotted piston. Installing the new design required modification to the trip valve body.

Safety Evaluation:
This design improves the performance of the HPCI overspeed (mechanical) trip assembly. Therefore, the reliability of this trip has increased, and has improved the availability of the HPCI turbine.

System N30-17 Implementing Document No. EDP-4410 Rev. F

Title of Change: Repairs on the East and West Main Steam Bypass Lines

Summary:
The main steam bypass lines beyond the east and west dump valves experienced high cyclic fatigue cracking. The cracks were confined to areas of discontinuities (i.e. lugs, taps). As a result, all 24 inch and 30 inch diameter pipe was replaced. The wall thickness of the 24 inch and 30 inch diameter pipe was increased from 3/8 inch to 1-1/2 inch and 1 inch, respectively.

Also installed were dual pressure breakdown orifices in the two 30 inch diameter steam bypass lines. The orifices were installed to increase back pressure at the bypass valves, which will lower the vibrations experienced by the piping. The new pipe was installed in exactly the same location.

Safety Evaluation:

Increasing the bypass pipe wall thickness from 3/8 inch to 1 inch reduces the possibility of cracking in the line.

The replacement of the bypass piping does not change the ability of the bypass lines to flow 13 percent of rated steam in each line. As a consequence, this design change does not affect the results of the accident analyses performed in Chapter 15B of the FSAR.

Design Change Summary Fermi 2

System C36 Implementing Document No. EDP-4797 Rev. 0

Title of Change: Fire Wrap Wireway RI-069 and Conduit RI-005-2P

Summary:
Install a 1 hour fire barrier around wireway RI-069 and conduit RI-005-2P. The fire barrier has two (2) layers of interface firewrap to prevent heat transfer. This was done to meet existing fire barrier requirements.

Safety Evaluation:
The installation of the fire wrap allows for the required circuit to be protected in case of fire. The fire barrier did not degrade the circuits and safety is improved.

Design Change Summary Fermi 2

System N21-02 Implementing Document No. EDP-1629 Rev. 0

Title of Change: Replacement of Feedpump Turbine Governor Signal Converter N21-KA03A, B, C and D with Convertors Equipped with High and Low Speed Demand Limiter

Summary:
Before implementing this change, the Reactor Feedpump Turbine governor signal convertors had a "HIGH" demand limit only. The original design required a "LOW" demand limit. It has been determined that both "HIGH" and "LOW" demand limits are necessary to obtain the required system performance capability. This modification replaced 4 high limit signal convertors with signal convertors that have high and low limit control - these are used in the reactor feed pump turbine system.

Safety Evaluation:
The changes made do not affect any safety functions.

System A30 Implementing Document No. EDP-2889 Rev. 0

Title of Change: Pipe Break Outside Containment Water Shields

Summary:
To minimize the effects of moderate energy line break and to assure a safe shutdown, the following changes were made:

4th Floor - Auxiliary Building - Column/Row H/11-12
1) A shroud was installed around 3 inch diameter fire protection lines (horizontal and vertical runs) and a 2 inch demineralized water line (horizontal run) located near an equipment hatch and Motor Control Center 72C-3B.
2) A metal curb (sealed) with handrail was installed around the perimeter of the equipment hatch.

3rd Floor - Auxiliary Building - Column/Row H/11-12
A sealed shroud with drain line was installed around the 4 inch diameter Reactor Building Closed Cooling Water and Emergency Equipment Cooling Water supply and return lines (horizontal run) located near Motor Control Centers 2PA-1 and 2PB-1.

Safety Evaluation:
The installation of the spray shields will provide added assurance that safety related electrical equipment will operate under moderate energy line break conditions.

Design Change Summary Fermi 2

System C71-01 Implementing Document No. EDP-1774 Rev. A

Title of Change: Modify the Scram Discharge Vent and Drain System by Installing a Time Delay

Summary:
This EDP provided the modification necessary to delay opening (for at least 5 seconds) scram discharge volume vent and drain isolation valves C11F010 and C11F011 until after valves C11F180 and C11F181 open. C11F180 and C11F181 are redundant vent and drain isolation valves installed in series with C11F010 and C11-F011.

Safety Evaluation:
The five second time delay for the sequenced opening of the vent and drain valves was part of the original system design. The five second time delay was unachievable with the originally installed equipment. This modification allowed obtaining the required 5 second time delay.

Environmental Qualification Design Changes

A number of design changes were completed in 1985 to environmentally qualify ~quipment to the requirements of 10CFR 50.49. These design changes are summerized below.

A safety evaluation was completed for each Engineering Design Package (EDP) listed. The safety evaluations found that the design change enhanced the ability of the equipment to perform under adverse environmental conditions. In each case the change did not result in a condition previously unanalyzed in the FSAR, nor did it reduce the margin of safety as defined in the basis of any Technical Specification. None of the design changes constituted an unreviewed safety question.

The following equipment was replaced:

EDP No. 1400 Electronic Flow Controllers T46K006 A and B; Standby Gas Treatment Exhaust Fan Controls, Division I and II.

Electronic to Pneumatic Converters T46K008 A and B; Standby Gas Treatment Exhaust Fan Vortex Damper Controls, Division I and II.

EDP No. 1407 Thermocouples:
T46N009 A and B; Standby Gas Treatment Air Heater Temperature, Division I and II.
T46N103 A and B; Standby Gas Treatment Charcoal Absorber Temperature, Division I and II.

EDP No. 1404 Electronic Flow Transmitters E11N015 A and B; Residual Heat Removal Loop A and B Flow to Reactor Pressure Vessel.

EDP No. 1405 Electronic Flow Transmitters E21N003 A and B; Core Spray Loop A and B Pump Outlet Flow.

EDP No. 1406 Electronic Flow Transmitter E41N008; High Pressure Coolant Injection Pump Discharge.

EDP No. 1421 Solenoid Valves:
T50F401 A and B - F408 A and B; Primary Containment Monitoring System Gas Sample Line Pilot Valves.
T50F420 A and B; Primary Containment Monitoring System Drywell Valve, Division I and II valves.
T50F422 A; Primary Containment Monitoring System Torus Valve, Division I.

EDP No. 1422 Hydrogen/Oxygen Monitor for Primary Containment Monitoring System, Division I and II.

EDP No. 1402 Current Transformers for 480V Motor Control Centers 72C and 72CF.

Environmental Qualification Design Changes

EDP No. 1428 1432 1437 1440 1443	Terminal Blocks, Internal Wiring, and Terminations for panels and skids in safety-related systems.
EDP No. 1426	Temperature Switches T46N006 A and B; Standby Gas Treatment System Air Heater, Division I and II.
EDP No. 1419	Valve Position Switches T50N451 A and B through T50N458 A and B; Primary Containment Monitoring System Gas Sample Line Containment Isolation Valve Position Switches.
EDP No. 1401	Motor Operator E4150F002; High Pressure Coolant Injection (HPCI) Turbine Steam Supply Inboard Isolation Valve.
EDP No. 1403	Motor Operators B3105F031 A and B; Recirculation Pump A and B Discharge Valves.
EDP No. 1425	Motor Operators: E1150F015 A and B; Low Pressure Coolant Injection (LPCI) Loops A and B Inboard Injection Isolation Valve: E11-50F017 A and B; Low Pressure Coolant Injection (.PCI) Loop A and B Outboard Injection Isolation Valve. E1150F028 A and B; Residual Heat Removal (RHR) Loop A and B Containment Spray/Test Isolation Valve.
EDP No. 1439	Motor Operator E2150F005 A and B; Core Spray Loop A and B Inboard Injection Isolation Valve.
EDP No. 1427	Electronic Transmitters: B21N080 A-D; Reactor Level Scram (Narrow Range), Division I and II. B21N081 A-D; Reactor Level 1 Trip (Wide Range), Division I and II. B21N094 A-H; Drywell Pressure High, Division I and II. B21N095 A&B; Reactor Level 3 Trip (Narrow Range), Division I and II. B21N110 A-D; Steam Dome Pressure - RHR Permissive, Division I and II. B21N111 A-D; Reactor Recirculation Pump A and B High Pressure

Environmental Qualification Design Changes

EDP No. 1433 Electronic Pressure Transmitters:
B31N110 A-D; Jet Pump Supply Line Pressure, Division I and II. B31N112A&B through B31N115A&B; Recirculation Pump Head B3101C001 A and B.

EDP No. 1434 Electronic Pressure Transmitters E11N055 A-D, E11N056 A-D; Residual Heat Removal Pumps Discharge Pressure Permissive Auto Blowdown.

EDP No. 1435 Electronic Pressure Transmitters E21N055 A & B, E21N062 A & B; Core Spray System Pump A, B, C, D Discharge Permissive for ADS Auto Actuation.

EDP No. 1436 Electronic Pressure Transmitters:
E41N055 A-D; HPCI Turbine Exhaust Diaphram High Pressure Isolation.
E41N057 A&B; HPCI High Differential Pressure Isolation Signal.
E41N058 A-D; HPCI Steam Line Low Pressure Isolation Signal.

EDP No. 1424 Motor Operators:
E4150F004; HPCI Booster Pump Suction From Condensate Storage Tank Isolation Valve.
E4150F006; HPCI Pump Discharge Inboard Isolation Valve.
E4150F012; HPCI Pump Test Return To Suppression Pool Isolation Valve.
E4150F041 and E4150F042; HPCI Booster Pump Suction From Suppression Pool Outboard and Inboard Isolation Valves.

The following equipment was added:

EDP No. 1840 Addition of solenoid valve in the 100 psig Non-Interruptable Air System to the damper T4100F037 for the Standby Gas Treatment System, Division I.

The following equipment was relocated:

EDP No. 1400 Electronic Flow Transmitters T46N013 A and B; Standby Gas Treatment Exhaust Fan Flow, Division I and II.

EDP No. 1408 Panels H21P295 A and B; Standby Gas Treatment System, Division I and II.

3.2 Procedure Changes

There were no procedure revisions between March 20 and December 31, 1985 that introduced an unreviewed safety question or that changed procedures as described in the Safety Analysis Report.

3.3 Tests and Experiments

The following special tests were performed on the Emergency Diesel Generators following replacement and inspection of failed internal components:

49.307.01 Special Test of Lube Oil System EDG #11
49.307.02 Special Test of Lube OIl System EDG #12
49.307.03 Special Test of Lube Oil System EDG #13
49.307.04 Special Test of Lube Oil System EDG #14

PURPOSE: 1) To determine the length of time required to adequately prelutricate each EDG. 2) Determine the static pressure the sta dby lube oil pump supplies to it's EDG when the lube oil system .s in normal standby conditions at various lube oil temperatures. This test was done to collect data to aid in evaluation of EIG bearing failures.

49.307.05	Operation of EDG #13 After Vendor Inspection
49.307.06	Operation of EDG #14 After Vendor Inspection
49.307.07	Operation of EDG #12 After Engine Bearing Replacement
49.307.09	Operation of EDG #11 After Engine Bearing Replacement
49.307.19T	Operation of EDG #13 After Crankshaft Bearing Replacement
49.307.20T	Operation of EDG #14 After Crankshaft Bearing Replacement
49.307.21T	Operation of EDG #11 After Crankshaft Bearing Replacement
49.307.22T	Operation of EDG #12 After Crankshaft Bearing Replacement

PURPOSE: These test procedures were performed to verify proper operation of the Emergency Diesel Generator internal engine components following maintenance activities. The maintenance activities were disassembly of engine for vendor inspections and replacement of bearings and/or cylinder lines found to be bad as a result of the inspections. In addition a 100 hour run was performed to "season" the crankshaft bearings per vendor recommendation. These procedures were performed at various times in 1985 due to diesel failures. Each procedure identifies the specific component replacement for which the test was run.

49.307.08 Special Test for EDG #12 Surge Suppressor (CR8)
49.307.10 Special Test for EDG #11 Surge Suppressor (CR8)

PURPOSE: 1) To verify correct operation of the new CR8 surge suppressor installed on EDG #11 and #12. 2) Record operating temperatures of CR8. 3) Record reverse bias leakage current of CR8 after engine run. 4) Record baseline data including kilowatts, kilovars, field current, and field voltage.

49.307.11 Special Test for EDG #12 Lube Oil Pressure Response

PURPOSE: 1) Verify lube oil pressure response to EDG #12 bearings during a fast start. 2) Compare and record engine start time and increasing speed to the actual lube oil pressure at the bearing supply lines.

49.307.12 Special Test for EDG #11 Linear Reactor - LR3

PURPOSE: Verify correct operation of new LR3 (Linear Reactor) which was installed in the exiter voltage regulator feedback circuit, and to record resistance, impedance, and insulation values of the LR3.

49.307.15T Verification of EDG #11 Slow Start Capability - Special Test
49.307.16T Verification of EDG #12 Slow Start Capability - Special Test
49.307.17T Verification of EDG #13 Slow Start Capability - Special Test
49.307.18T Verification of EDG #14 Slow Start Capability - Special Test

PURPOSE: 1) Verify proper operation of EDG slow start capability following installation of exciter bypass switch to allow slow start. 2) Adjust the lower speed limit of the mechanical governor speed control. 3) Record generator bearing vibration data at various engine speeds between 300-950 RPM.

49.307.23T Verification of EDG #13 Crankshaft Bearing Integrity Special Test

PURPOSE: Verify that crankshaft bearing damage did not occur following a 100 hour run followed by 35 successive slow starts and 3 fast (10 second) starts. The 100 hour continuous run will "season" the crankshaft bearing as per vendor recommendations.

SAFETY EVALUATION: Performance of each test required the associated Emergency Diesel Generator to be declared inoperable. A test could only be performed in accordance with operability requirements of the plant's Technical Specifications. Therefore, no unreviewed safety question exists per the requirements of 10CFR 50.59.

DMB

Frank E. Agosti
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Nuclear Operations

February 28, 1986
VP860016

PRIORITY ROUTING

Mr. James G. Keppler
Regional Administrator
Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Dear Mr. Keppler:

Reference: Fermi 2
NRC Docket No. 50-341
NRC License No. NPF-43

Subject: Annual Operating Report

In accordance with Fermi 2 Technical Specification 6.9.1.4 and U.S. NRC Regulatory Guide 1.16, the Detroit Edison Company is submitting the Annual Operating Report for the Fermi 2 Nuclear Power Plant for the period of March 21, 1985 through December 31, 1985.

The 1985 Fermi 2 Annual Operating Report also satisfies the reporting requirements of 10CFR20.407, Personnel Monitoring Report, Technical Specification 6.9.1.5.b (Safety/Relief Valve Challenges) and Technical Specification 6.9.1.5.c (Emergency Core Cooling System Outages).

If you have any questions or comments about this report, please contact Mr. Lewis Bregni at (313) 586-5313.

Sincerely,

F. E. Agosti

F. E. Agosti
Vice President
Nuclear Operations

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MAR 5 1986

cc: W. G. Rogers
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