NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4) Inadequacies in Technical Specification Surveillances Found during Surveillance Review

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
1 0	0 8	8 7	8 7	— 0 4 8 —	0 4	0 6	0 7	8 8	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9): 4

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
Patricia Anthony, Compliance Engineer	3 1 3 5 8 6 - 1 6 1 7

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) X NO

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

The ongoing review of Technical Specification surveillance procedures has discovered additional violations of Technical Specifications. The time delay relays associated with the RWCU differential flow isolation function was not adequately tested. The Rod Block Monitor Functional surveillance procedure did not adequately test the bypass function of the Rod Block Monitor system at the required 30% power.

These conditions were causeu by incomplete or inadequate surveillance procedures. The corrective actions include revising the appropriate procedures. All Technical Specification surveillance procedures are scheduled to be reviewed by the end of 1988 as part of the Technical Specification Improvement Program.

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Previously, Detroit Edison committed to perform a review of Technical Specification surveillances as part of its Technical Specification Improvement Program. This report will describe the findings of that program that constitute violations of the Technical Specifications.

Description of the Event:

1) On October 8, 1987 at 1400 hours, it was discovered that the technical specification requirement to perform a channel check for the Reactor Protection System (RPS) drywell high pressure instruments (PT) was not being met. The plant was in Operational Condition 4 at the time with reactor power at zero percent, reactor pressure at 0 psig and reactor temperature at 133 degrees Fahrenheit. Table 4.3.1.1-1 item 7 and Table 4.3.2.1-1 item 1.b require that the channel check be performed for these channels at least once per 12 hours when in the specified Operational Conditions. None of the RPS drywell high pressure channels were included in surveillance procedure 24.000.02, "Shiftly, Daily, Weekly and Situation Required Surveillances".

   The Technical Specification Table 4.3.1.1-1 states that each RPS instrumentation channel shall be demonstrated operable by the performance of the channel check, channel functional test and channel calibration for certain operational conditions and at the frequences shown in the table.

   Table 4.3.1.1-1 item 7 requires that a channel check be performed for RPS drywell high pressure instruments in Operational Conditions 1 and 2.

   The Technical Specification Table 4.3.2.1-1 states that each isolation actuation instrumentation channel shall be demonstrated operable by the performance of the channel check and channel functional tests at the frequences shown in Table 4.3.2.1-1. Table 4.3.2.1-1 item 1.b requires that channel checks be performed for the RPS drywell high pressure instruments in Operational Conditions 1, 2 and 3.

The surveillance procedure 24.000.02 attachment 3 did not meet these requirements. There were no channel checks performed for pressure instruments C71-N650 A, B, C and D in Operational Conditions 1, 2 and 3.

The immediate action was to notify the Nuclear Shift Supervisor and to perform a channel check from instruments C71-N650 A, B, C and D to verify compliance with the channel check requirements in the Technical Specifications Table 4.3.1.1-1 item 7 and Table 4.3.2.1-1 item 1.b.

2) The Electrical Protection Assembly (EPA)(BKR) breakers calibration surveillance was reviewed as part of the improvement program. On February 3, 1988 at 1000 hours, the review revealed that a Technical Specification requirement had not been properly incorporated into the testing program. The plant was in Operational Condition 1 at the time with reactor power at 85 percent, reactor pressure at 970 psig and reactor temperature at 520 degrees Fahrenheit.

As defined in the Technical Specifications, a channel functional test is required as part of a channel calibration. Technical Specification 4.8.4.4.b requires a channel calibration be performed for the EPA breakers. After completion of procedure 42.610.02, "Electrical Protection Assembly Calibration" on June 23, 1986, the functional test procedure 42.610.01, "Electrical Protection Assembly Functional Test" was not performed since it was not indicated as necessary. The next channel functional test was completed on January 14, 1987, during the routinely scheduled surveillance. The last channel functional test had been performed on April 1, 1986.

3) On February 26, 1988 at 1250 hours, it was determined that the logic functional testing performed for the Emergency Core Cooling System (ECCS) actuation instrumentation associated with the loss of power logic for the 4160 volt emergency buses (EB) (BU) was inadequate. It did not verify the breaker (BKR) trips initiated by the loss of voltage or degraded grid voltage.

Additionally, it failed to verify the emergency diesel generator (EDG) (EK) actuation initiated by the load shedding loss of voltage or degraded grid voltage. This is contrary to the requirements of Technical Specification 4.3.3.2.

At the time of this discovery, the plant was in Operational Condition 1 at 81 percent reactor power with reactor pressure at 975 psig and reactor temperature at 522 degrees Fahrenheit. Since this placed all of the emergency diesel generators in a questionable operability status, a shutdown commenced in accordance with Technical Specification 3.0.3. The plant was placed in shutdown condition at 0039 hours on February 27, 1988. A sequence of events test was performed on February 28, 1988 in order to verify the operability of the Division I 4160 volt emergency buses undervoltage load shedding circuits up to the relays which must energize in order to start the emergency diesel generators in an emergency condition. Completion of this testing placed the plant in compliance with Technical Specification 3.3.3.

4) On March 16 ,1988, the plant was in Operational Condition 4 with 0 percent reactor power, reactor pressure of 0 psig and reactor temperature at 126 degrees Fahrenheit. It was discovered that the response time testing for the Low Pressure Coolant Injection System (LPCI) (BO), the Core Spray System (CSS) (BM) and the High Pressure Coolant Injection System (HPCI) (BG) had not fully met the requirement of Technical Specification 4.3.3.3 and Technical Specification definition 1.11. The Technical Specification requires that response time for ECCS be measured from when the initiating set point is exceeded to the point where the equipment is performing its safety function. The term, "equipment is performing its safety function", is defined as meaning the valve travel is complete, pump discharge pressure has reached its required value, etc.

TEXT (If more space is required, use additional NRC Form 366A's) (17)

Contrary to this, none of the previous surveillance testing had verified that the design basis pump (P) discharge pressures were achieved in the required time for these systems. Additionally, the CSS response time testing did not verify that the required design basis flow was developed and the LPCI injection valve (V) response times were not verified.

5) On April 14, 1988, the plant was in Operational Condition 4 with reactor temperature at 126 degrees Fahrenheit. It was discovered that the time delay relays associated with the Reactor Water Cleanup System (RWCU) differential flow isolation function were not adequately tested during the channel calibration, test Table 4.3.2.1 - 1, Item 2.A. The time delay relay was not time tested to ensure that it actuates within the design limit of 45 seconds.

6) On April 14, 1988, the plant was in Operational Condition 4 with the reactor temperature at 126 degrees Fahrenheit. It was discovered that the Rod Block Monitor calibration test procedure could result in a non-conservative setpoint being established. The Rod Block Monitoring System is to be automatically bypassed at < 30% thermal power. The setpoint used in the calibration test contains a tolerance that could result in the rod block function being bypassed at a setpoint in excess of 30% thermal power contrary to the requirements of Technical Specification Table 3.3.6-1. Trip Function 1, Table notation (A).

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Cause of the Event:

1) The cause of these events was an incomplete or inadequate surveillance procedures. The "Shiftly, Daily, Weekly and Situation Required Surveillances" procedure did not require performance of the channel check for instruments C71-N650 A, B, C, and D at least every 12 hours. These instruments indicate drywell pressure.

2) The channel functional test was not incorporated in the channel calibration for the EPA breakers. In one instance, the channel calibration requirement was assumed to be met, but the channel functional test was not performed for approximately six months.

3) The operability requirements for the ECCS actuation instrumentation had not been considered in the development of the 4160 volt bus procedure.

4) In the case of the ECCS response time testing, the achievement of the design basis pump performance parameters was not specified as the end point for response time.

5) The cause of the event was an incomplete surveillance test procedure. The requirement to time check the 45 second time delay relay associated with the RWCU System Differential Flow Isolation Function was not considered in the development of the channel calibration procedure.

6) The Rod Block Monitor calibration test procedure did not adequately establish the unbypass function of the rod block monitors at the required > 30% thermal power because the setpoint used in the calibration test contained a tolerance that could result in the rod block function being bypassed at a setpoint in excess of 30% thermal power. The potential impact of the tolerance affecting the procedure acceptance criteria was not considered in the development of the calibration test procedure.

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Analysis of Event:

1) The RPS drywell high pressure channel check was performed successfully on the first attempt. The channel functional and channel calibration surveillance requirements had been performed since the receipt of the operating license. This ensured a level of reliability in the instrumentation. This indicated that the instruments were functioning properly. In addition drywell pressure channel check surveillance requirements have routinely been performed for Emergency Core Cooling and Accident Monitoring Systems (IP). As a result, this condition did not affect the safe operation of the plant. This event was not contributed to by any components, systems, structures or conditions of the workplace.

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2) The channel functional testing for the EPA breakers is being performed on a six month basis as required by Technical Specification 4.8.4.4.a. The channel calibration testing is specified to be performed on an eighteen month basis. This channel testing was scheduled and performed as required, but the required channel functional testing was not performed. While the operability testing requirements were not met, the availability of the EPA breaker was ensured when the subsequent testing that was performed satisfactorily confirmed the EPA breakers' operability.

3) While the emergency diesel generators had not been properly surveillance tested per the Technical Specifications, the circuitry had been tested during the pre-operational testing and the loss of offsite power test. The integrity of the circuit up to the relays (RLY), which must energize in order for the diesels to start in an emergency condition, had been verified previously. Control of maintenance activities maintained the integrity of the circuits.

Therefore, the emergency diesel generators were functional and available for service as proven by other testing even though the required testing for operability had not been completed.

4) Even though the verification of the response time for the required design basis flow and discharge pressure was not in the response time testing procedures, other procedures have verified pump performance in the past. However, these other procedures did not include the requirement to response time test the systems. The LPCI pump discharge pressure is verified to be greater than 230 psig when flow is 10,750 gpm by procedures 24.204.01 and 24.204.06, the Division I and II LPCI and Suppression Pool Cooling/Spray Pump and Valve Operability Tests. The injection valves, E11-F015A and E11-F015B, stroke times are verified in procedure 24.204.04, "RHR Shutdown Cooling and Head Spray Valve Operability". The procedures 24.203.02 and 24.203.05, the Division I and II CSS Pump and Valve Operability Tests, verify that the CSS pumps develop 6600 gpm when discharge pressure is greater than 270 psig.

The HPCI pump performance is verified by measuring discharge pressure when flow is 5200 gpm by procedure 24.202.01, "HPCI Pump Operability and Flow Test at 1000 PSI and Valve Operability". These procedures are performed every 92 days while the plant is in Operational Conditions 1, 2 and 3.

5) The channel calibration of the Reactor Water Cleanup System differential flow isolation function is being performed on a 18 month basis as required by Technical Specification Table 4.3.2.1-1, Trip Function 2, item a. However, the time delay relay associated with the isolation function was not timed. The availability of the isolation function occurring within the required 45 seconds was ensured when the subsequent testing was performed satisfactorily.

6) The Rod Block Monitor channel calibration test was being performed every 3 months as required by Technical Specification Table 4.3.6-1, Trip Function 1. However, the possible impact of the tolerance on the setpoint was not fully recognized. Using the worst case setpoint, the Rod Block Monitor system could be bypassed up to 31.625% of thermal power which is in excess of the Technical Specification limit of 30% thermal power. The difference in the worst case setpoint and the required setpoint is within the Fermi 2 Accident Analysis described in the UFSAR, Section 15.4.2.3.3.

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Upon evaluation of the above information, there is reasonable assurance that no degradation to plant safety resulted from the failure to perform correctly the surveillances described above.

Corrective Actions:

1) The corrective action was to revise the Technical Specification surveillance procedure 24.000.02, "Shiftly, Daily, Weekly and Situation Required Surveillances" for Operational Conditions 1 through 4 to include C71-N650 A, B, C and D as part of the drywell pressure channel check for Reactor Protection System/Nuclear Steam Supply Shutoff System.

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As part of the enhancement program, surveillance procedures 24.000.02 and 24.000.03, "Mode 5 Shiftly, Daily, Weekly and Situation Required Surveillances", were reviewed in order to verify that other appropriate Technical Specification surveillance requirements were met.

2) In the case of the EPA breaker channel calibration, the immediate remedial corrective action taken was to revise the Surveillance Performance Form for the channel calibration to include a requirement that the channel functional test also be completed. As long term corrective action procedure 42.610.02 will be revised to include a channel functional test following the channel calibration. The procedure revision will be completed by June 30, 1988.

3) The testing performed under a sequence of events test verified the operability of the Division I circuits. Procedures 42.302.04, "Calibration and Logic System Functional Testing of Division II 4160 Volt Emergency Bus Undervoltage Circuits" was revised to verify the operability of the Division II circuits and performed on March 25, 1988. Procedure 42.302.02, "Calibration of Division I 4160 Volt Emergency Bus Undervoltage Circuits" was revised to verify the operability of the Division I circuitry and performed on April 8, 1988.

4) In order to satisfy the response time testing requirements, the following actions have been taken. An evaluation of previous testing was made as part of the investigation of this problem. As a result, a revised LPCI response time test procedure 24.204.03,"LPCI Simulated Automatic Actuation Test and Valve Operability Test", was performed on April 27, 1988.

This will take credit for the overlap with procedures 24.204.01, 24.204.04 and 24.204.06 to fully meet the Technical Specification requirement for response time testing. Procedures 24.203.03 and 24.203.04, Division I and II CSS Simulated Automatic Actuation Tests, were revised to incorporate the appropriate criteria for CSS. The procedures were successfully completed on March 27, 1988. Credit is being taken for the overlap with 24.203.02, 24.203.05 and 24.203.06 in order to meet the response time testing requirement in Technical

Specifications. A revision to the HPCI procedure 24.202.04, "HPCI System Automatic Actuation/Suction Valve Auto Transfer", was made and successfully completed on May 9, 1988.

5) For the 45 second time delay associated with the Reactor Water Cleanup differential flow isolation function the immediate remedial corrective action was to issue work requests to calibrate the time delay relays. The work was completed on April 21, 1988. As part of the long term corrective action the associated channel calibration surveillance procedure was revised to include the requirement to calibrate the 45 second time delay relay, this was completed on May 9, 1988.

6) The Corrective Action was to revise the Technical Specification surveillance procedures 44.010.151 and 44.010.152 to change the setpoint associated with the Rod Block Monitor system to ensure that the worst case impact of the $\pm$ setpoint tolerance would not cause the Rod Block Monitor system to be bypassed in excess of 30% thermal power. The revised test procedures were satisfactorily completed on April 27, 1988.

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All Technical Specification surveillance procedures are scheduled to be reviewed to ensure Technical Specification compliance as part of the Technical Specification Improvement Program. This program is described in the Detroit Edison letter to the Nuclear Regulatory Commission dated April 6, 1988. The activity in this area is currently targeted to be completed by the end of 1988.

Previous Similar Events:

Licensee Event Reports 85-018, 85-036, 85-037, 85-040, 86-004, 86-008, 86-010, 86-022, 86-039, 87-029, 87-044, and this report have reported instances where inadequate or incorrect procedures caused violations of the Technical Specifications.

William S. Orser
Vice President
Nuclear Operations

Detroit Edison

Fermi 2
6400 North Dixie Highway
Newport, Michigan 48166
(313) 586-5300

10CFR50.73

Nuclear Operations

June 7, 1988
NRC-88-0130

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

Reference: (1) Fermi 2
NRC Docket No. 50-341
Facility Operating License No. NPF-43

(2) Transmittal of Licensee Event Report 87-048-03 dated April 15, 1988, NRC-88-0097

Subject: <u>Licensee Event Report (LER) No. 87-048-04</u>

Please find enclosed LER No. 87-048-04, dated June 7, 1988, for the reportable findings of a review of Technical Specification surveillances. This report is being amended to reflect further finding of the review. A copy of this LER is also being sent to the Regional Administrator, USNRC Region III.

If you have any questions, please contact Patricia Anthony at (313) 586-1617.

Sincerely,

WSOrser

Enclosure: NRC Forms 366, 366A

cc: A. B. Davis
J. R. Eckert
R. C. Knop
T. R. Quay
W. G. Rogers

Wayne County Emergency
Management Division

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