NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

FACILITY NAME (1): Fermi 2
DOCKET NUMBER (2): 05000 341
PAGE (3): 1 OF 7

TITLE (4): Unexpected Reactor Water Level 2 and ESF Actuations after Planned Scram

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
04	09	95	95	-- 004 --	00	05	09	95		05000
										05000

OPERATING MODE (9): 1
POWER LEVEL (10): 080

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

	20.402(b)		20.405(c)	X	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)		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: Joseph M. Pendergast - Compliance Engineer
TELEPHONE NUMBER (Include Area Code): (313) 586-1682

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	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 15 single-spaced typewritten lines) (16)

At 1501 hours, on April 9, 1995 a planned manual reactor scram was inserted at 80 percent power to obtain main turbine vibration data through the critical coastdown speed of 860 to 840 revolutions per minute. Approximately fourteen seconds after the scram insertion, Reactor water level approached Level 2 causing partial actuation of Level 2 safety functions. The low water level was of a short duration, and reactor vessel water level recovered just as the Emergency Core Cooling System (ECCS), Nuclear Steam Supply Shutoff System (NSSSS), and Alternate Rod Insertion (ARI) / Reactor Recirculation Pump Trip (RPT) instrumentation set points for reactor water Level 2 were reached. As a result, Reactor Water Clean Up and Division 1 Drywell Pneumatics isolated, Reactor Building Heating Ventilation and Air Conditioning isolated, Standby Gas Treatment System automatically started, and Control Center Heating Ventilation and Air Conditioning shifted to recirculation mode.

The Reactor Recirculation Pumps tripped, High Pressure Coolant Injection (HPCI) System started but did not inject and Reactor Core Isolation Cooling initiated. An Unusual Event was conservatively declared because the HPCI System initiated although HPCI System injection did not occur. The reactor scram was reset at 1534 hours, and the Unusual Event was terminated at 1659 hours after Reactor Recirculation System had been restarted. A review of the Post Scram Feedwater Logic will be conducted to determine if improvements are possible.

REQUIRED NUMBER OF DIGITS/CHARACTERS
FOR EACH BLOCK

BLOCK NUMBER	NUMBER OF DIGITS/CHARACTERS	TITLE
1	UP TO 46	FACILITY NAME
2	8 TOTAL 3 IN ADDITION TO 05000	DOCKET NUMBER
3	VARIES	PAGE NUMBER
4	UP TO 76	TITLE
5	6 TOTAL 2 PER BLOCK	EVENT DATE
6	7 TOTAL 2 FOR YEAR 3 FOR SEQUENTIAL NUMBER 2 FOR REVISION NUMBER	LER NUMBER
7	6 TOTAL 2 PER BLOCK	REPORT DATE
8	UP TO 18 -- FACILITY NAME 8 TOTAL -- DOCKET NUMBER 3 IN ADDITION TO 05000	OTHER FACILITIES INVOLVED
9	1	OPERATING MODE
10	3	POWER LEVEL
11	1 CHECK BOX THAT APPLIES	REQUIREMENTS OF 10 CFR
12	UP TO 50 FOR NAME 14 FOR TELEPHONE	LICENSEE CONTACT
13	CAUSE VARIES 2 FOR SYSTEM 4 FOR COMPONENT 4 FOR MANUFACTURER NPRDS VARIES	EACH COMPONENT FAILURE
14	1 CHECK BOX THAT APPLIES	SUPPLEMENTAL REPORT EXPECTED
15	6 TOTAL 2 PER BLOCK	EXPECTED SUBMISSION DATE

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

Initial Plant Condition

Operational Condition:	1	Power Operation
Reactor Power:	80	Percent
Reactor Pressure:	1004	psig
Reactor Temperature:	540	degrees Fahrenheit

Description of the Event:

At 1501 hours, on April 9, 1995 a planned manual reactor scram was inserted to obtain main turbine (TA) vibration data through the critical coastdown speed of 860 to 840 Revolutions Per Minute (RPM). Reactor water level was at the normal level of 197 inches above the Top of Active Fuel (TAF) when the manual scram was inserted. Reactor water level decreased because of the steam void collapse in the reactor vessel. Reactor Water Level 3 (173 inches above the TAF) safety system (JE) isolations and actuations occurred as were expected. The control room operator (utility licensed) started Standby Feedwater [(SBFW)(SJ)] when reactor water level reached approximately 135 inches above the TAF. Approximately fourteen seconds after the manual scram was inserted, some actuations for reactor water Level 2 occurred when the water level approached (i.e. reached some instrument setpoints) the Level 2 setpoint (110.8 inches above the TAF). Reactor water level began to increase just as the Emergency Core Cooling System (ECCS), Nuclear Steam Supply Shutoff System (NSSSS), and Alternate Rod Insertion (ARI) / Reactor Recirculation Pump Trip (RPT) instrumentation set points for reactor water Level 2 were approached. Specifically, partial reactor water Level 2 signals caused division 1 NSSSS to isolate valves (ISV) but the division 2 NSSSS isolation valves did not isolate. While the division 1 ARI/RPT logic did not actuate, the division 2 ARI/RPT logic actuated which resulted in the reactor recirculation pumps tripping. The lowest recorded reactor water level reached was 111 inches on Control Room Recorder (LR) B21R623B. As a result of Level 2 initiation signals, High Pressure Coolant Injection [(HPCI)(BJ)] System actuated, Reactor Core Isolation Cooling [(RCIC)(BN)] System actuated, Reactor Water Clean up [(RWCU)(CE)] isolated, Drywell Pneumatics Division 1 isolated (LF), Reactor Building Heating Ventilation and Air Conditioning [(RBHVAC)(VA)] isolated, Standby Gas Treatment System [(SGTS)(BH)] automatically started, and Control Center Heating Ventilation and Air Conditioning [(CCHVAC)(VI)] shifted to recirculation mode.

The HPCI System operated as expected. HPCI initiated but did not inject into the reactor vessel because reactor water level had increased and was above the Level 2 setpoint before the HPCI injection valve could receive an open signal. The RCIC System initiated and injected into the reactor vessel. RCIC and SBFW injected into the reactor vessel to raise reactor vessel water level to normal. Much of the cold water added as a result of the RCIC and SBFW injection settled near the bottom of the vessel because the Reactor Recirculation Pumps(AD) tripped and RWCU isolated. With only natural circulation in the reactor vessel at this time, the bottom head area exceeded the Technical Specification cooldown limit of 100 degrees Fahrenheit in any hour. The maximum reactor water cooldown rate was 168 degrees Fahrenheit in 63 minutes per control room recorders.

An Unusual Event was conservatively declared in accordance with the Fermi 2 Emergency Plan because the HPCI System initiated, although HPCI System injection did not occur. At 1520 hours the RWCU System was restored to service. Once the RWCU System was returned to service, flow through the vessel bottom head drain line was re-established past the vessel bottom head drain line thermocouple (TW). The reactor scram was reset at 1534 hours, and the Unusual Event was terminated at 1659 hours following the restart of the reactor recirculation pumps. Due to mixing via RWCU and Reactor Recirculation Systems, the reactor vessel bottom head drain line thermocouple indicated a heatup rate of 173 degrees Fahrenheit in 60 minutes. Most of this occurred in the few minutes following the start of reactor recirculation pumps (the first pump started at 1638 hours) due to rapid mixing of hot (approximately 520 degree Fahrenheit) bulk reactor coolant with the cool (approximately 390 degree Fahrenheit) stratified layer near the bottom head.

Cause of the Event:

The probable cause of reaching Reactor Vessel Water Level 2 during this event is the selection of parameters used in the Post Scram Feedwater Logic modification per Engineering Design Package (EDP) 9207 installed during RF04. The purpose of the modification is to prevent a reactor water Level 8 (214 inches above the TAF) trip, following postulated component failure. The post scram reactor water level setdown parameters were changed from 173 inches to 150 inches and time delay from 7 seconds to 3 seconds. The post scram feedwater logic automatically lowers the setpoint of the controller from 197 inches to 150 inches after a time delay of 3 seconds. This, combined

with the large feedwater - steam flow mismatch following a scram, drives the controller's output down, forcing the Reactor Feed Pumps (RFP) to slow down, and lowering feedwater flow. The post scram parameters were chosen to prevent reaching a Level 2 (110.8 inches above the TAF), or a Level 8. The changes necessary to prevent Level 8 reduced the margin for Level 2 avoidance. These parameters were based on reactor transient analysis (RETRAN) model. This prediction by RETRAN was based on the model response that feedwater flow rate would decrease fast enough to initiate the number 2 recirculation pump speed limiter within the first few seconds following the scram. During this event feedwater flow did not decrease as fast as predicted by RETRAN. Therefore, no recirculation pump run back was required to be initiated prior to the trip of the Reactor Recirculation Pumps when Level 2 was reached.

Additionally, the feedwater demand limiter (C32K610) was modified to automatically increase the RFP speed from 1600 to between 2600 to 2700 RPM one minute after the scram. The actual RFP speed increase was only to approximately 2300 RPM and operator intervention was required to increase the pump speed to the desired 2600 to 2700 RPM. The feedwater demand limiter bounds the output of the master level controller between approximately 28 and 30 percent. The actual operating speed of the Feedwater Pumps within these limits depends upon plant conditions prior to and during the scram. This discrepancy did not contribute to the cause of reaching Level 2 setpoint and did not significantly impact the recovery efforts of the operators.

<u>Analysis of the Event</u>

While reactor water level was not expected to decrease to the Level 2 setpoint following the manual scram from 80 percent power, plant response to this event is bounded by expected transients analyzed as described in chapter 15 of the UFSAR. It can be concluded that this transient does not have any safety significance in terms of affecting the plant or the health and safety of the public.

The HPCI System E4150-F006 discharge valve did not open and allow the system to inject water into the reactor vessel. This was because the reactor water Level 2 signal was not in long enough to allow the E4150-F006 valve to open. The E4150-F001 steam supply valve and E4100-F067 turbine stop valve must clear full closed in conjunction with the Level 2 signal before the E4150-F006 receives an open signal. Opening the

E4150-F006 valve starts approximately seven seconds from the initiation of the ECCS Level 2 signal. The permissive to open the E4150-F006 did not exist since the Level 2 signal was confirmed to have cleared in 2.5 seconds. Prior to reactor restart the E4150-F006 valve was stroked to assure proper operability. The valve stroked properly. The conclusion reached from review of the timing of the post scram data and the proper valve stroking was that the Level 2 ECCS signal cleared before the E4150-F006 received a permissive to open for HPCI System injection.

Shortly after the scram, some of the isolations and actuations required to occur at Level 2 occurred when reactor water Level 2 was approached. Because the actual water level did not decrease below the required setpoint for Level 2, it was concluded that normal variation of actual instrument setpoints had resulted in Level 2 signals from some of the Level 2 instruments. The observed actuations were consistent with the actuated logic recorded by the plant sequence of events recorder. The instruments which did not actuate on Level 2 were functionally checked to verify their operability. Based on these checks, the instruments were determined to be operable and would have actuated within tolerance as required.

Reactor Recirculation Pump trip and RWCU isolation on Level 2, and recovery resulted in cooldown and heatup rates which exceeded limits prescribed in Technical Specification 3.4.6.1. Review of the temperature data and the evaluation of potential damage to susceptible components was required by Technical Specification 3.4.6.1. The bottom head drain temperature is representative of the bottom head region only when sufficient reactor water mixing and flow exists through the bottom head drain. RWCU flow was stopped from 1502 to 1520 hours. This period of time corresponded to the time frame in which the majority of cold water introductions to the shroud region from RCIC and SBFW occurred. The first representative value of bottom head temperature occurred at 1520 hours after restart of RWCU. The operators controlled the cooldown from this point until cooldown was stopped at 1605 hours. The maximum cooldown rate recorded was 168 degrees Fahrenheit in 63 minutes. The number of core components that experienced the 168 degrees Fahrenheit cooldown was limited by the reduced flow in the bottom head region after loss of forced circulation. The operators achieved a heatup of approximately 40 degrees Fahrenheit from 1605 to 1635 hours. When a Recirculation Pump was started at 1638 hours, temperature in the bottom head area quickly equalized with the bulk coolant temperature. This resulted in an overall heatup of 173 degrees

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Fahrenheit in 60 minutes. The number of core components that experienced the 173 degree Fahrenheit heatup was limited to the bottom head components since other portions of the reactor vessel were exposed to water at various higher temperatures than the bottom head drain line temperature value of 398 degrees Fahrenheit. The recirculation loops were at 515 degrees Fahrenheit and the core barrel, separator and annulus were at some value that approached the steam dome saturation temperature of 535 degrees Fahrenheit. As specified by Technical Specification 3.4.6.1 an engineering evaluation was completed. Detroit Edison engineering and General Electric concluded that the transient is bounded by previously analyzed events.

Control rod 34-31 had a slower than expected scram time of 4.08 seconds. The Technical Specification limit is 7 seconds. Although the control rod scram time was within the Technical Specification limit for an individual control rod, a Deviation Event Report (95-0259) was written to investigate the unexpectedly slow scram time of this rod.

Corrective Actions

The post scram feedwater logic modification, EDP 9207 will be reviewed by Detroit Edison. Engineering will analyze for various initial conditions to optimize the water level response for controllable events. Improvements will be made to the logic based on recommendations from the review. Prior to reactor restart Engineering Change Request 9207-4 was issued to document the speed the Reactor Feed Pumps would reach (approximately 2300 RPM) following a scram.

Operating shifts were informed by the Nuclear Shift Supervisors about the post scram feed water logic problems, and Abnormal Operating Procedure 20.000.21, "Reactor Scram" was revised to address the high power scram and reactor water level recovery. Although there was training in the past, operator training for scrams where Level 2 is reached will be reemphasized during operator requalification training and will include recovery from recirculation pump trip/controlling heatup rate.

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Previous Similar Events:

Licensee Event Reports (LERs) 92-012 and 93-010 described conditions where reactor water Level 2 was reached and for LER 93-010 where heatup and cooldown rates were exceeded. HPCI injected for these events. These LERs are similar to this event only because Level 2 was reached and in all instances safety systems responded appropriately.