William S. Orser
Senior Vice President

Detroit Edison

Fermi 2
6400 North Dixie Highway
Newport, Michigan 48166
(313) 586-5201

Nuclear Operations

10CFR50.73

April 15, 1992
NRC-92-0027

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

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

Subject: Licensee Event Report (LER) No. 92-002

Please find enclosed LER No. 92-002, dated April 15, 1992, for a reportable event that occurred on March 16, 1992. A copy of this LER is also being sent to the Regional Administrator, USNRC Region III.

If you have any questions, please contact Joseph M. Pendergast, Compliance Engineer, at (313) 586-1682.

Sincerely,

Enclosure: NRC Forms 366, 366A

cc: T. G. Colburn
A. B. Davis
R. W. DeFayette
S. Stasek
P. L. Torpey

Wayne County Emergency
Management Division

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)
Fermi 2	0 5 0 0 0 3 4 1	1 OF 0 4

TITLE (4): Low Pressure Coolant Injection Loop Select Logic Actuation During Channel Functional Surveillance Resulted in a Manual Scram

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 3	1 6	9 2	9 2	0 0 2	0 0	0 4	1 5	9 2		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 1 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)	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)	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	
Joseph M. Pendergast, Compliance Engineer	3 1 3	5 8 6 - 1 6 8 2

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 fifteen single-space typewritten lines) (16)

At 1709 hours on March 16, 1992, during the performance of a Technical Specification Channel Functional Surveillance Test on Division 2 Reactor Water Levels 1, 2 and 8, an inadvertent actuation of the Low Pressure Coolant Injection (LPCI) system Loop Select Logic (LSL) occurred. The LSL signal caused the Reactor Recirculation Pump (RRP) "B" discharge valve to close. The resulting core flow decrease led to entry into Region A of the power/flow map. No Thermal Hydraulic instability occurred. By procedure and the Technical Specification Action Statement, the reactor operator then manually tripped the reactor by placing the Mode Switch in the "Shutdown" position. The plant responded as expected to the resultant manual scram.

The LSL signal actuation occurred due to a short circuit condition across the K79 relay contacts for the LPCI LSL. This occurred when a Digital Multimeter (DMM) was connected to take the voltage readings required of the surveillance test. The input impedance on the meter will go to a relatively low value with the function switch slightly clockwise of the "Volts DC" position. The DMM in question, and all DMMs of the same model number and series have been taken out of service at the site.

NRC Form 366
(9-83)

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

Initial Plant Conditions:

Operational Condition:	1	(Power Operation)
Reactor Power:	100	Percent
Reactor Pressure:	1010	psig
Reactor Temperature:	520	Degrees Fahrenheit

Description of the Event:

At 1709 hours on March 16, 1992, during the performance of a Technical Specification Channel Functional Surveillance Test on Division 2 Reactor Water Levels 1, 2 and 8, Channel "D", an inadvertent actuation of the Low Pressure Coolant Injection [(LPCI) (BO)] system Loop Select Logic (LSL) occurred. At that time the reactor was operating in Mode 1 at approximately 100% power. The LSL signal caused the Reactor Recirculation Pump [(AD) (RRP)] "B" discharge valve (ISV) to close. The closure of this valve (F031B) results in a RRP "B" number 1 Limiter pump runback and subsequent pump trip. The transient caused by the rapid power decrease from the RRP "B" runback and trip resulted in a runback of RRP "A" to number 3 Limiter on a loss of heater drain (SK) flow (anticipated reduced feed flow). The resulting core flow decrease led to entry into Region A of the power/flow map (commonly referred to as the instability region). No Thermal Hydraulic instability occurred. By procedure and the Technical Specification 3.4.10 Action Statement, the reactor operator then manually tripped the reactor by placing the Mode Switch in the "Shutdown" position. The plant responded as expected to the manual scram. There was no anomalous plant or equipment behavior observed following the plant scram and recovery.

The Post Scram Investigation Group (PSIG) was activated. The PSIG determined the cause of the LPCI LSL Actuation that resulted in the manual Reactor Scram. The Onsite Safety Review Organization reviewed the Post Scram Evaluation and recommended reactor restart to the Plant Manager. The required surveillances were completed. At 2352 hours on March 17, the plant entered start up.

Cause of the Event:

The PSIG evaluation determined that the LSL signal actuation occurred because of a short circuit condition across the K79 relay (RLY) contacts for the LPCI LSL. This occurred when a Digital Multimeter [(DMM) (MTR)] was connected (using installed testability jacks) to take the voltage readings required of the surveillance test. Interviews with the Instrumentation & Controls (I&C) technicians who performed the test, and observation of a re-test of the surveillance determined that personnel error was not a factor during this activity. A review of the electrical prints and discussions with PSIC team members, systems and nuclear engineering personnel confirmed that a momentary short circuit condition caused the LSL actuation signal. Bench tests of the DMM conducted as part of the PSIG review determined that the DMM was the source of a momentary short circuit condition which caused the LSL actuation signal. The input impedance on the meter will go to a relatively low value (short circuit) with the function switch slightly clockwise of the "Volts DC" position.

Analysis of the Event:

The region of the power/flow operating domain bounded by the 80% load line and the 45% constant core flow line is an area of high reactor power/low reactor flow in which Boiling Water Reactors (BWR) are particularly susceptible to experiencing thermal hydraulic instabilities. A smaller subset of this region, Region "A" (above the 100% load line with core flow less than 40% of rated), represents the region where thermal hydraulic instability events have previously occurred at other BWRs. Two types of instabilities generally occur; core wide (in-phase oscillations) or regional (out-of-phase oscillations). General Electric (GE) and BWR Owners Group analyses initiated following the LaSalle-2 instability event of March 9, 1988, have indicated that the Safety Limit Minimum Critical Power Ratio could be violated during regional oscillations. Therefore, the prompt action of manually scramming (placing the mode switch in Shutdown) when Region "A" is entered was required per procedure and the Technical Specification 3.4.10 Action Statement in order to minimize the potential for a thermal hydraulic instability to occur. The Average Power Range Monitoring chart recorder traces and the GE Transient and Recording System traces were reviewed by Reactor Engineering personnel, and no evidence of a thermal hydraulic instability could be found.

The plant responded as expected to the manual scram. Reactor pressure control was maintained through the Turbine Bypass Valves to the Main Condenser. There was no anomalous plant or equipment behavior observed following the plant scram and recovery from the scram. The health and safety of the public and the plant were not compromised by this event.

Corrective Actions:

An accountability meeting was held to discuss the event with personnel involved and Senior Management. The DMM in question, a Fluke model 77 series II, and all DMMs of the same model and series number have been taken out of service at the site. Other models have been checked for similar characteristics. New DMMs of the rotary switch type will be checked for the same problem before they are released for use in the plant. I&C personnel will review this event during continuing training. The I&C continuing training began in the second quarter of 1992.

Previous Similar Events:

There have been no previous Licensee Event Reports where a DMM has caused this type of problem.