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)
Leakage In Excess of The Allowable Found During Local Leak Rate Testing

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)
0 2	2 9	8 8	8 8	— 0 0 8	— 0 1	0 5	2 7	8 8	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9)	4	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 0 0	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)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
		20.405(a)(1)(iv)	X 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
Joseph Pendergast, Licensing Engineer	AREA CODE 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	
X	S B	I S V	A 5 8 5	Y							
X	J M	I S V	T 0 2 0	Y							

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

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

Periodic leak rate testing of primary containment isolation valves was completed as required by Technical Specification Surveillance Requirements during the Local Leak Rate Testing Outage. During the course of testing, the determination was made that the combined leakage limits as specified in Technical Specification Limiting Conditions for Operation were exceeded.

The basis of the surveillance requirement is to allow for the early detection of valve leakage due to normal wear and degradation during a specified time interval. The valve leakage was caused by normal degradation of valve components and/or contaminants on the valve seats.

Forty-four primary containment isolation valves were cleaned, reworked, refurbished and successfully retested to bring the total allowable leakage into compliance with Technical Specifications.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

Initial Conditions:

Operational Condition: 4 (Cold Shutdown)
Reactor Power: 0%
Reactor Pressure: 0 psig
Reactor Temperature: approximately 105 degrees Fahrenheit

Description of Event:

Periodic leak rate testing of primary containment isolation valves (ISV) was completed as required by Technical Specification Surveillance Requirement 4.6.1.2, during the Local Leak Rate Testing (LLRT) Outage. The primary containment isolation valves were subjected to tests as specified by 10 CFR 50 App. J. The Technical Specification leakage limit of 178 Standard Cubic Feet per Hour (SCFH) for Type B and C tests and the leakage limit for Main Steam Isolation Valves (MSIV) of 100 SCFH were exceeded. In all, a total of forty-four of the 237 valves tested significantly contributed to total combined leakage for the primary containment boundary. After the completion of testing, the determination was made that the combined leakage limits as specified in Technical Specification Limiting Condition for Operation (LCO) 3.6.1.2 b, c, d and e were exceeded.

The forty-four primary containment isolation valves were cleaned, reworked, refurbished and successfully retested to bring the total allowable leakage into compliance with Technical Specification 4.6.1.2.

Cause of the Event:

The basis of the surveillance requirement is to allow for the early detection of valve leakage due to normal wear and degradation during a specified time interval. The valve leakage was caused by normal degradation of valve components and/or contaminants on the valve seats.

Analysis of Event:

LLRT is performed to detect degradation in the leakage characteristics of containment penetrations. Any degradation found is corrected so that the total leakage of all containment penetrations is within Technical Specification Limits. The frequency of the testing is such that gross degradation of the containment penetration barriers does not occur between testing periods.

In the case of isolation in the Main Steam System (SB) following a postulated loss of coolant accident, both air operated sets of MSIVs, the turbine stop valves (FCV), the control valves (FCV) and the bypass valves (V) would close in order to seal the main steam lines. If required, the control room operators would initiate the Main Steam Isolation Valve Leakage Control System (MSIVLC) Isolation Valves and manually close a third set of motor operated MSIVs. The MSIVLC is a safety grade system designed to prevent leakage through the MSIVs by maintaining the volume between the MSIVs at a pressure greater than the containment pressure. Then any leakage through the MSIVs would be into the containment.

For all LLRT, penetration leakages measured during testing are determined using a maximum pathway method. That is, the maximum possible penetration leakage is used in determining Technical Specification acceptance. In reality, minimum path way leakages would occur and any leakage would be contained within process piping. Since no valves were found inoperable during this period, this assumption is reasonable. In addition, some of the systems would be in operation during accident conditions and therefore their corresponding penetrations would be in use. Thus, failure of the individual valves to meet LLRT requirements does not automatically mean a loss of primary containment integrity.

Corrective Actions:

Forty-four valves were cleaned, reworked and/or refurbished. The valves were then retested and leakage certified to be in compliance with Technical Specification Surveillance Requirement 4.6.1.2. The following table summarizes the present condition and test results of the primary containment isolation valves and penetrations.

Type of Test	As Left Condition	Technical Specification Limit
Type B and C *	88.25 SCFH	178 SCFH
MSIV	7.93 SCFH	100 SCFH
Hydrostatic	1.51 GPM	5 GPM

* For all containment isolation valves exclusive of the MSIVs.

Previous Similar Occurrences:

In Licensee Event Report 86-011, excessive leakage from the main steam isolation valves during local leak rate testing was reported.

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

May 27, 1988
NRC-88-0127

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 88-008-00 dated March 30, 1988, NRC-88-0087

Subject: Licensee Event Report (LER) No. 88-008-01

Please find enclosed LER No. 88-008-01, dated May 27, 1988, for a reportable event that occurred on February 29, 1988. This LER is being revised to describe the final findings of Local Leak Rate Testing conducted during outage 88-01. A copy of this LER is also being sent to the Regional Administrator, USNRC Region III.

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

Sincerely,

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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