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DTE Energy

10 CFR 50.73

May 24, 2006
NRC-06-0037

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington D C 20555-0001

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

Subject: Licensee Event Report No. 2006-001, "Excessive Feedwater Check Valve Leakage at Containment Penetration"

Pursuant to 10 CFR 50.73(a)(2)(ii)(A), Detroit Edison is hereby submitting the enclosed Licensee Event Report (LER) No. 2006-001. This LER documents a containment minimum-pathway leak rate for a reactor feedwater line (Penetration X-9A) that exceeded the limiting conditions for operation in the plant Technical Specifications.

No commitments are made in this LER:

Should you have any questions or require additional information, please contact Mr. Ronald W. Gaston of my staff at (734) 586-5197.

Sincerely,

Donald K. Cobb

cc: D. H. Jaffe
T. J. Kozak
C. A. Lipa
NRC Resident Office
Regional Administrator, Region III
Supervisor, Electric Operators,
Michigan Public Service Commission

IE22

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: No. 3150-0104 Expires 6/30/2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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4. TITLE
Excessive Feedwater Check Valve Leakage at Containment Penetration

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
04	01	2006	2006	- 001 -	00	05	24	2006	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 5

10. POWER LEVEL: 0%

11. THIS REPORT SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☒ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Robert J. Salmon – Principal Licensing Engineer	(734) 586-4273

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	SJ	ISV	A391	Y	B	SJ	ISV	A585	Y

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

At 0039 hours on 4/01/2006, Fermi 2 feedwater line check valves, B2100F010A and B2100F076A, failed their Local Leak Rate Test (LLRT) test. The air leakage rate of the inboard check valve B2100F010A was 324.21 standard cubic feet per hour (SCFH), and the leakage rate of the outboard check valve B2100F076A was above the measurement capability of the leak rate monitor. The penetration (X-9A) minimum-pathway air leakage value was determined to be 324.21 SCFH which is greater than the allowable containment leakage rate (La) value of 296.3 SCFH per Technical Specification 5.5.12 and higher than the allowable secondary containment bypass leakage rate of 0.1 La or 29.63 SCFH per Technical Specification Surveillance Requirement 3.6.1.3.11. The B2100F076A failure was attributed to soft seat degradation which was primarily caused by extending its service time to three operating cycles. The B2100F010A valve failure was attributed to soft seat degradation due to a slight misalignment of the valve disc to the in-body seat exasperated by wear between the internal shaft and valve disc. The slight misalignment caused the soft seat along the top portion of the disc to contact the seat first resulting in a scraping action as the disc flexed to its full seat position. For both valves, the soft seats were replaced, and the soft seat service time has been limited to two operating cycles. The internal shaft for the B2100F010A valve was replaced, and the alignment between the disc and the valve seat was adjusted. Both valves were retested and met their associated LLRT acceptance criteria prior to restart of the unit.

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

Initial Plant Conditions:

Mode 5

Reactor Power 0 percent

Description of the Event

At 0039 hours on 4/01/2006, Fermi 2 feedwater [SJ] line check valves B2100F010A and B2100F076A failed their primary containment local leak rate test (LLRT). The air leakage rate of inboard isolation valve B2100F010A [ISV] was 324.21 standard cubic feet per hour (SCFH), and the leakage rate of outboard isolation valve B2100F076A was above the measurement capability of the leak rate monitor. The penetration (X-9A) minimum-pathway air leakage value was thus determined to be 324.21 SCFH which is greater than the allowable containment leakage rate (La) value of 296.3 SCFH per Technical Specification 5.5.12, and higher than the allowable secondary containment bypass leakage rate of 0.1 La or 29.63 SCFH per Technical Specification Surveillance Requirement 3.6.1.3.11.

There are two functions of the B2100F076A valve. One function is to stop reverse flow into the feedwater lines on High Pressure Coolant Injection (HPCI) [BJ] system injection. It shares that function with another inline check valve (B2100F032A) that is located between the HPCI injection point and the B2100F076A valve. The second function is to minimize containment out-leakage in the event of a loss of coolant accident (LOCA) after the line drains. It is this second "containment isolation" function that the LLRT test is intended to verify.

Based on inspections, the B2100F076A valve moved freely and closed fully. Metal to metal seating was observed with minimal relative gaps around the circumference of the valve which would improve under high pressure conditions where the valve disc is pressured into its seat. The metal to metal seating surface is relied upon to provide the high-pressure seating function, and the fluid leak rate under these conditions was judged to be inconsequential when compared to High Pressure Coolant Injection requirements. Additionally the other check valve (B2100F032A) in series with the B2100F076A valve is capable of performing the HPCI check function either by itself or in conjunction with the B2100F076A valve. Therefore the ability to inject water to the reactor using the HPCI system was not affected by this event.

The primary containment isolation function is needed for conditions where the feedwater line is drained. This function relies on the soft seats in outboard containment isolation valve B2100F076A and inboard containment isolation valve B2100F010A to minimize the leakage through penetration X-9A in the event of a loss of coolant accident. Since the feedwater lines also penetrate the secondary containment, this penetration is also considered a source of secondary containment bypass leakage. That is, a fraction of the primary containment leakage could bypass the secondary containment and be released to the environment after a LOCA without first being treated by the Standby Gas Treatment System. Lower leakage limits are required for lines associated with secondary containment bypass leakage in addition to primary containment leakage limits. A soft seat O-ring is provided on these feedwater check valves in addition to the metal to metal seating surfaces to attain leak tight closure under air conditions such as those used in the LLRT air test. The soft seat O-rings for both valves were observed to be at least partially recessed in their retaining grooves and were not effective in achieving the isolation function. The valves were reworked prior to resuming operation at power.

Immediate notification was made to the NRC in accordance with 10 CFR 50.72 at 02:45 ET on April 1, 2006 (EN 42465).

This event is being reported under 50.73(a)(2)(ii)(A), as an event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded. A report is required by this section of the regulation when containment leak rate testing identifies a minimum-pathway leak rate that exceeds the limiting condition for operation in the plant Technical Specifications.

Cause of the Event

The cause of the event is a failure of primary containment isolation valves B2100F076A and B2100F010A to meet their respective leakage criteria during LLRT air testing such that the penetration X-9A pathway leakage exceeded Technical Specification requirements for normal plant operation.

The B2100F076A failure was attributed to soft seat degradation which was primarily caused by extending its service time from a nominal two operating cycles to three operating cycles. Extending the service life for a third operating cycle was evaluated and justified based on the predicted radiation and temperature effects. Benefits included reduction in personnel dose and avoidance of previously identified valve bonnet sealing issues. However, it has since been determined that the high flow condition of the feedwater system also affects the life of the seats resulting in limiting the effectiveness of the soft seats to no more than two operating cycles.

The B2100F010A valve failure was attributed to soft seat degradation due to a slight misalignment of the valve disc to the in-body seat exasperated by wear between the internal shaft and valve disc. The slight misalignment caused the soft seat along the top portion of the disc to contact the seat first resulting in a scraping action as the disc flexed to its full seat position.

Analysis of the Event

This event involves a failure to meet the acceptance criteria associated with primary containment isolation and secondary containment bypass combined leak rate for penetration X-9A. The B2100F076A valve leakage was greater than measurable by the test apparatus, whereas, the B2100F010A valve leaked at 324.21 SCFH. The valves are in the same line, so the combined penetration (X-9A) leakage value was determined to be 324.21 SCFH which is greater than the allowable containment leakage rate (La) value of 296.3 SCFH per Technical Specification 5.5.12 and higher than the allowable secondary containment bypass leakage rate of 0.1 La or 29.6 SCFH per Technical Specification Surveillance Requirement 3.6.1.3.11.

While this path is considered a secondary containment bypass leakage path that could release directly to the environment, a line break would be required in the portion of the feedwater system located in the Turbine Building combined with a LOCA in order for the leakage to bypass the secondary containment. That is, if a postulated feedwater line break occurs in the reactor building, the secondary containment would not be bypassed, and the leakage would be treated by the Standby Gas Treatment System [BH] prior to release. If a LOCA were to occur without a feedwater line break, the feedwater line leakage flowpath is through multiple balance of plant (BOP) valves, pumps and demineralizers [SF] to the main condenser [SG]. This path would allow radioactive iodine plateout which would reduce radioactive releases to the main condenser. While the main condenser is not

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relied upon under accident conditions, operation of the off-gas system, if available, would further reduce any radioactive release due to its charcoal adsorber [ADS] capability. In the event the off-gas system was not capable of operation, the scrubbing and hold up volume provided by the main condenser itself would limit release to the environment. Although the plant is not credited for maintaining non-safety related systems such as the feedwater system intact after a LOCA, that is the most likely scenario.

The radiological consequence analysis of a design basis LOCA in the UFSAR utilizes alternative radiological source terms in accordance with Regulatory Guide 1.183. The analytical results were evaluated against the criteria contained in 10 CFR 50.67. Each criterion and fundamental assumption used in the analysis is, in itself, appropriately conservative. However, these criteria and assumptions are used collectively and likely result in substantially overestimating potential exposures. Among the assumptions used is that the primary containment is postulated to leak at a constant rate of 0.5 percent volume per day for the first 24 hours and then at 0.25 percent per day for the remainder of the 30 day event. LLRT testing is performed at the maximum design pressure of 56 psig. An analysis has shown that a design basis large break LOCA pressure peak of about 50 psig is achieved in about 5 seconds, is reduced to less than half that pressure in about 30 minutes, and continues to decline such that containment pressure for most of the 30-day accident duration after a postulated design basis LOCA would be a small fraction of the maximum design pressure. It is the containment pressure that primarily drives the containment leakage. Therefore the leakage rate measured during LLRT testing is conservatively high when considering these analyzed post accident conditions. Furthermore, the degree of radiological release during an accident specified by the regulations implies widespread fuel damage beyond what would result assuming operation of the Emergency Core Cooling System. A more realistic analysis of a LOCA has been performed, and it was determined that the temperature and pressure transients are insufficient to cause perforation of the fuel cladding. Therefore no fuel damage results from this accident. Consequently, the as found leakage would be expected to have little impact on the health and safety of the public had a postulated design bases loss of coolant accident occurred.

Corrective Actions

The soft seats were replaced in valves B2100F076A and B2100F010A. The soft seat service time for both valves has been limited in the preventive maintenance program to two operating cycles. The internal shaft for the B2100F010A valve was replaced, and the alignment between the disc and the valve seat was adjusted. Both valves were retested and met their associated LLRT acceptance criteria before the restart of the plant.

This event has been documented and continues to be evaluated in the Fermi 2 corrective action program, CARD 06-21751. Other corrective actions were performed or are being considered that are not directly related to the cause of the failure to meet the LLRT acceptance criteria. The evaluations include consideration of disc replacement for the B2100F010A valve. Corrective actions identified as a result of these evaluations will be tracked and implemented by the corrective action program.

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Additional Information

A. Failed Components:

Component: Feedwater Outboard Containment Isolation Valve B2100F076A
Function: 20-inch Swing Check Valve
Manufacturer: Anchor Darling
Model Number: 2276-3
Failure Cause: Degraded soft seat

Component: Feedwater Inboard Containment Isolation Valve B2100F010A
Function: 20-inch Swing Check Valve
Manufacturer: Atwood & Morrill
Model Number: 21389-H
Failure Cause: Degraded soft seat

B. Previous LERs on Similar Problems:

There have been no LERs in the last decade related to the excessive leak rates in feedwater penetration containment isolation valves. In the last six years there has been one instance where 10CFR50 Appendix J air leakage limits have been reported in excess of that allowed by the plant technical specifications:

LER 00-007: This LER describes a condition where the secondary containment bypass limit through Main Steam Drain Penetration X-8 exceeded the technical specification requirements. The valves involved in this LER were 3-inch motor operated gate valves, and the cause of the failure was attributed to over machining of valve seating surfaces during maintenance resulting in inadequate overlap between the disc and body seats. Other than being containment isolation valves, the valves discussed in LER 00-007 have little in common with the 20-inch swing check valves involved in the current event. The cause of the current event (20-inch swing check valve soft seat leakage) is unrelated to leaving sufficient overlap margin when performing corrective valve maintenance on 3-inch hard gate valve seats. Therefore, the corrective actions relating to the previous event would not be expected to be effective is precluding the current event.