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DTE Energy



10 CFR 50.73

December 6, 2007
NRC-07-0057

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. 2007-001, "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. 2007-001. This LER documents a containment minimum-pathway leak rate for a reactor feedwater line (Penetration X-9B) 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,

A handwritten signature in cursive script that reads "Joseph H. Plona".

cc: NRC Project Manager
NRC Resident Office
Reactor Projects Chief, Branch 4, Region III
Regional Administrator, Region III
Supervisor, Electric Operators,
Michigan Public Service Commission

JE22
NRC

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: 80 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, NE0B-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
10	07	2007	2007	001	00	12	06	2007	FACILITY NAME	DOCKET NUMBER
										05000
										05000

9. OPERATING MODE 5	11. THIS REPORT SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: <i>(Check all that apply)</i>			
	<input type="checkbox"/> 20.2201(b)	<input type="checkbox"/> 20.2203(a)(3)(i)	<input type="checkbox"/> 50.73(a)(2)(i)(C)	<input type="checkbox"/> 50.73(a)(2)(vii)
	<input type="checkbox"/> 20.2201(d)	<input type="checkbox"/> 20.2203(a)(3)(ii)	<input checked="" type="checkbox"/> 50.73(a)(2)(ii)(A)	<input type="checkbox"/> 50.73(a)(2)(viii)(A)
	<input type="checkbox"/> 20.2203(a)(1)	<input type="checkbox"/> 20.2203(a)(4)	<input type="checkbox"/> 50.73(a)(2)(ii)(B)	<input type="checkbox"/> 50.73(a)(2)(viii)(B)
	<input type="checkbox"/> 20.2203(a)(2)(i)	<input type="checkbox"/> 50.36(c)(1)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(iii)	<input type="checkbox"/> 50.73(a)(2)(ix)(A)
10. POWER LEVEL 0%	<input type="checkbox"/> 20.2203(a)(2)(ii)	<input type="checkbox"/> 50.36(c)(1)(ii)(A)	<input type="checkbox"/> 50.73(a)(2)(iv)(A)	<input type="checkbox"/> 50.73(a)(2)(x)
	<input type="checkbox"/> 20.2203(a)(2)(iii)	<input type="checkbox"/> 50.36(c)(2)	<input type="checkbox"/> 50.73(a)(2)(v)(A)	<input type="checkbox"/> 73.71(a)(4)
	<input type="checkbox"/> 20.2203(a)(2)(iv)	<input type="checkbox"/> 50.46(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(v)(B)	<input type="checkbox"/> 73.71(a)(5)
	<input type="checkbox"/> 20.2203(a)(2)(v)	<input type="checkbox"/> 50.73(a)(2)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(v)(C)	<input type="checkbox"/> OTHER
	<input type="checkbox"/> 20.2203(a)(2)(vi)	<input type="checkbox"/> 50.73(a)(2)(i)(B)	<input type="checkbox"/> 50.73(a)(2)(v)(D)	

Specify in abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME Robert J. Salmon – Principal Licensing Engineer	TELEPHONE NUMBER (Include Area Code) (734).586-4273
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13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	SJ	ISV	A391	Y	B	SJ	ISV	A585	Y

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
<input type="checkbox"/> YES (If yes, complete 15. EXPECTED SUBMISSION DATE)	<input checked="" type="checkbox"/> NO			

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

At 0114 hours on 10/07/2007, Fermi 2 feedwater line check valves, B2100F010B and B2100F076B, failed their local leak rate test (LLRT). The air leakage rate of the inboard check valve B2100F010B was 297.3 standard cubic feet per hour (SCFH), and the leakage rate of the outboard check valve B2100F076B was above the measurement capability of the leak rate monitor. The penetration (X-9B) minimum-pathway air leakage value was determined to be 297.3 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 LLRT leakage measured for this penetration also caused the calculated as-found integrated leak rate test (ILRT) to exceed acceptance criteria. The valve failures were primarily attributed to soft seat erosion by feedwater flow to the point that the seats were not providing an effective seal. It was also determined that the soft seat replacement frequency of two operating cycles was less than adequate to reliably ensure LLRT requirements are met. The soft seats were replaced for all four of the LLRT feedwater check valves during the recently completed refueling outage (RF12). The soft seat replacement frequency has also been increased to require a replacement of all LLRT feedwater check valve soft seats each refueling outage. All four 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 0114 hours on 10/07/2007, it was identified that both Fermi 2 feedwater [SJ] line check valves B2100F010B and B2100F076B failed their primary containment local leak rate test (LLRT). The air leakage rate of inboard isolation valve B2100F010B [ISV] was 297.3 standard cubic feet per hour (SCFH), and the leakage rate of outboard isolation valve B2100F076B was above the measurement capability of the leak rate monitor. The penetration (X-9B) minimum-pathway air leakage value was thus determined to be 297.3 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 safety functions of the B2100F076B valve. One function is to stop reverse flow into the feedwater lines on Reactor Core Isolation Cooling Injection (RCIC) [BN] system injection. It shares that function with another inline check valve (B2100F032B) located between the RCIC injection point and the B2100F076B 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 B2100F076B valve moved freely and closed fully. Metal to metal seating was observed with minimal relative gaps around the circumference of the valve disc 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 RCIC flow requirements. Additionally, the other check valve (B2100F032B) in series with the B2100F076B valve is capable of stopping reverse flow either by itself or in conjunction with the B2100F076B valve. Therefore, the ability of the RCIC system to perform its design function was not affected by this event.

The primary containment isolation function is needed for accident conditions where the feedwater line is drained. This function relies on the soft seats in outboard containment isolation valve B2100F076B and inboard containment isolation valve B2100F010B to minimize the leakage through penetration X-9B 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 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 significantly worn and were not effective in achieving the isolation function. The soft seats were replaced, and the valves were successfully retested prior to resuming operation at power.

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An integrated leak rate test (ILRT) was also performed during this twelfth refueling outage (RF12). As a direct result of the feedwater check valve penetration X-9B minimum pathway leakage, the as-found ILRT leakage was calculated to be 0.775451% per day or 459.5 SCFH which exceeds the plant administrative Technical Specification requirement of less than 0.5% per day (La). Without the penetration X-9B, minimum pathway leakage, the as-found ILRT would have met requirements with a leakage rate of 0.55 La.

Immediate notification was made to the NRC in accordance with 10 CFR 50.72 at 05:56 ET on October 7, 2007 (EN 43699).

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 B2100F076B and B2100F010B to meet their respective leakage criteria during LLRT air testing such that the penetration X-9B minimum pathway leakage exceeded the Technical Specification requirements for normal plant operation. These failures were primarily attributed to soft seat erosion by feedwater flow to the point that the seats were not providing an effective seal. These failures occurred after plant operation over two complete operating cycles. It was also determined that the PM frequency of the B2100F076A/B and B2100F010A/B soft seat material was based on faulty data and did not include actual Fermi operational experience. This resulted in the soft seat material being used beyond its useful service life.

The soft seats were replaced for the penetration X-9A valves B2100F076A and B2100F010A during the eleventh (RF11) refueling outage. The as-found leakage for these valves met LLRT acceptance criteria this outage (RF12). This provides a measure of confidence that if the soft seats are replaced every refueling outage, LLRT leakage criteria can be met. It was determined that the feedwater check valves have not been able to reliably meet their LLRT requirements for more than one cycle after replacement of the soft-seating material.

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-9B. The B2100F076B valve leakage was greater than measurable by the test apparatus, whereas, the B2100F010B valve leaked at 297.3 SCFH. The valves are in the same line, so the combined penetration (X-9B) leakage value was determined to be 297.3 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,

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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 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 (RG) 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 peak calculated accident pressure of ≥ 56.5 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 measured leak rates applied over the duration of a LOCA or as specified in RG 1.183 are conservative given the analyzed accident pressure profile. 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 penetration X-9B valves B2100F076B and B2100F010B during RF12. The X-9A penetration valves passed with as-found leakage rates of 0.85 and 1.25 SCFH for a minimum pathway leakage of 0.85 SCFH. The soft seats for those valves were replaced during the RF11 outage. To ensure proper performance over the next cycle, the soft seats in penetration X-9A valves B2100F076A and B2100F010A were also replaced during RF12. The soft seat service time for all LLRT feedwater check valves has been limited in the preventive maintenance program to one operating cycle. A technical evaluation has been performed for all LLRT check valves that rely on seat material for sealing similar to these feedwater check valves. The evaluation determined that the seats in all of the LLRT check valves are acceptable for use in the upcoming operating cycle. This event is documented and evaluated in the Fermi 2 corrective action program.

Other evaluations and corrective actions are being considered that are not directly related to the primary causes of the failure to meet the LLRT acceptance criteria. 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 B2100F076B
 Function: 20-inch Swing Check Valve
 Manufacturer: Anchor Darling
 Model Number: 2276-3
 Failure Cause: Degraded soft seat

Component: Feedwater Inboard Containment Isolation Valve B2100F010B
 Function: 20-inch Swing Check Valve
 Manufacturer: Atwood & Morrill
 Model Number: 21389-H
 Failure Cause: Degraded soft seat

B. Previous LERs on Similar Problems:

LER 06-001 describes a similar condition which occurred on the penetration X-9A isolation valves B2100F076A and B2100F010A. The combined leakage of that penetration also exceeded the allowable containment leakage rate (La) value of 296.3 SCFH. The corrective actions focused on restoration of the penetration involved, and that penetration passed its LLRT test this outage. Those corrective actions were successful in eliminating excessive leaks from the X-9A penetration. One action planned as a result of that event could have been effective in precluding this event, however, since the action was identified after completion of the last outage, it was not planned to be implemented until the current (RF12) refueling outage. That action was to stagger soft seat replacements to ensure that at least one feedwater check valve per feedwater penetration had a soft seat that was replaced during the previous outage. Because this action relating to the previous event was developed after the RF11 refueling outage, it would not be expected to be effective in precluding the current event.

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 in precluding the current event.