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Detroit Edison

10CFR50.73

May 12, 2000
NRC-00-0031

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. 00-007

Pursuant to 10CFR50.73(a)(2)(ii)(B), Detroit Edison is hereby submitting the enclosed Licensee Event Report (LER) No. 00-007. This LER addresses the main steam line drain valves which failed to meet the secondary containment bypass leakage limits of Technical Specification 3.6.1.3, "Primary Containment Isolation Valves," during the Seventh Refueling Outage.

No new commitments are being made in this LER.

Should you have any questions or require additional information, please contact Mr. Norman K. Peterson of my staff at (734) 586-4258.

Sincerely,

cc: J. Dyer
A. J. Kugler
M. A. Ring
M. V. Yudasz, Jr.
NRC Resident Office
Region III
Wayne County Emergency Management Division

IE22

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(6-1998)

LICENSEE EVENT REPORT (LER)

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

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If 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.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Fermi 2	05000341	1 of 5

TITLE (4)
Failure of Isolation Valves to Meet Secondary Containment Bypass Leakage Limits

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 NAME	DOCKET NUMBER
										05000
04	12	00	00	--0 0 7	0	05	12	00	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	5
POWER LEVEL (10)	0

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

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Jerome Flint – Licensing	(734) 586-5212

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	SB	ISV	P312	N					

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 15 single-spaced typewritten lines) (16)

Local leak rate testing of the main steam line drain valves was conducted during the Seventh Refueling Outage. Secondary containment bypass leakage through penetration X-8 exceeded the 0.04 La allowed by Technical Specification Surveillance Requirement 3.6.1.3.11. The leakage through B2103-F016, Main Steam Line Drain Inboard Isolation Valve, was not quantifiable. The leakage through B2103-F019, Main Steam Line Drain Outboard Isolation Valve, was 232.7 standard cubic feet per hour. Conservatively, the as found minimum path leakage value for penetration X-8 was determined to be 232.7 scfh. For the B2103-F016, the valve disc was found to provide an improved fit when it was rotated 180 degrees. Refurbishment work during the sixth refueling outage had left the B2103-F019 valve seats with insufficient margin. There was no evidence of steam cutting on the seating surfaces for either valve indicating that there was no active leakage through this pathway during the last operating cycle.

Lapping and polishing of the valve seats was performed during the current outage. The B2103-F019 in body valve seats were lapped and polished and a build up was performed on the valve disc to restore the valve to an acceptable condition. Both valves subsequently passed their local leak rate tests.

Initial Plant Conditions:

Mode	5 (Refueling)
Reactor Power	0 Percent
Reactor Pressure	0 psig
Reactor Temperature	89 Degrees Fahrenheit

Description of the Event:

On April 12, 2000, local leak rate testing (LLRT) was being performed in accordance with the provisions of Surveillance Requirement 3.6.1.3.11, secondary containment bypass leakage. The leakage acceptance criterion, 0.04 La, is 11.9 standard cubic feet per hour (scfh) at a test pressure of ≥ 56.5 psig. The as found leakage for penetration X-8, main steam line drain path, failed to meet this criteria. This penetration consists of two isolation valves, B2103-F016, Main Steam Line Drain Inboard Isolation Valve [SB][ISV], and B2103-F019, Main Steam Line Drain Outboard Isolation Valve [SB][ISV]. Test pressure could not be achieved for the inboard isolation valve, B2103-F016. Leakage for the outboard isolation valve, B2103-F019 was 232.7 scfh. Conservatively, the minimum path leakage value for penetration X-8 was determined to be 232.7 scfh.

Exceeding the secondary containment bypass leak rate Surveillance Requirement 3.6.1.3.11 is considered reportable pursuant to 10CFR50.73(a)(2)(ii)(B), "any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded, or that resulted in the nuclear plant being in a condition that was outside the design basis of the plant." On April 12, 2000, at 1039 hours this was reported as a four hour notification in accordance with 10CFR50.72(b)(2)(i), "any event, found while the reactor is shut down, that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant, including its principal safety barriers, being seriously degraded or being in an unanalyzed condition that significantly compromises plant safety," Event Notification Number 36873.

In addition to the leakage through the main steam line drain penetration, this LER is addressing Main Steam Isolation Valve (MSIV) [ISV] leakage, which was initially reported under the same event notification number on April 7, 2000 at 1715 hours for Main Steam Line C, and April 7, 2000 at 1836 hours for Main Steam Line D. The initial telephone notifications were made on April 7, 2000, because test pressure could not be achieved during LLRT testing and leakage could not be adequately attributed to inboard or outboard valves, nor could it be confirmed that leakage did not exceed the design capability of the MSIV Leakage Control System (MSIVLCS) [BD]. Based on this preliminary information, phone notifications were made for these conditions under 10CFR50.72(b)(2)(i). Subsequent investigation and testing determined that there was at least one MSIV in each main steam line was acceptable in the as found condition. The total as found minimum path leakage attributed to the four main steam lines was 61.57 scfh. MSIV leakage limits are given by Technical Specification (TS) Surveillance Requirement 3.6.1.3.12, "Verify combined MSIV leakage rate for all four main steam lines is ≤ 100 scfh when tested at ≥ 25 psig." Since the as found leakage did not exceed 100 scfh, this condition was determined to be not reportable under 10CFR50.73(a)(2)(i)(B), "any operation or condition prohibited by the plant's technical specifications." Furthermore, since this as found leakage is bounded by the capacity of the MSIVLCS, the as found

condition did not constitute "any event, found while the reactor is shut down, that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant, including its principal safety barriers, being seriously degraded or being in an unanalyzed condition that significantly compromises plant safety." Consequently, the initial notification made on April 7, 2000, (Event Notification Number 36873) under 10CFR50.72(b)(2)(i) was unnecessary. However, no retraction was made since the main steam line drain isolation valve secondary containment bypass leakage, which was reported under the same event number, remains reportable as discussed above.

Cause of the Event:

The B2103-F016 valve was disassembled and inspected. There was no evidence of steam cutting of the seats. This valve was replaced with a new valve from stock during the sixth refueling outage (RF06). The valve was successfully bench tested prior to installation and was then welded into the main steam line drain piping. During bench testing it had been observed that the seating surfaces were slightly offset axially from one another; however, the valve passed a blue check of its seating surfaces and a bench test LLRT. During assembly the gate disc was likely installed with the gate disc 180 degrees from the preferred orientation. Although the valve passed the post installation LLRT, proper seating may not have been repeatable due to the differing angles of the in body and disc seating surfaces. Turning the disc 180 degrees was found to provide satisfactory seat alignment. The likely cause of the LLRT failure was a marginal fit of the new valve seating surfaces. After repair, the valve passed its as left LLRT.

Upon disassembly and inspection of the B2103-F019 valve it was identified, by blue check, that the disc seat was not making contact for the entire 360 degree seating surface. It was also determined that the in body valve guides were offset which caused an uneven wear of the valve over time. This wear led to machining of the valve seating surfaces in RF06 and associated difficulty achieving a satisfactory fit. As a result the valve seats had been left with minimal remaining seating margin (overlap between disc and body seats) to accommodate normal wear and variance in seat alignment when the valve is operated. The cause of the LLRT failure was inadequate margin left after maintenance. After appropriate refurbishment, the valve passed its as left LLRT.

Analysis of the Event:

During normal plant operation the B2103-F016, Main Steam Line Drain Inboard Isolation Valve, and B2103-F019, Main Steam Line Drain Outboard Isolation Valve, are normally closed. Disassembly and inspection indicated no evidence of steam cutting of the valve seating surfaces on either valve. Therefore, a leakage path likely did not exist through this pathway during the last operating cycle, and the valves were tightly seated at operating temperature and pressure.

During the B2103-F016 valve LLRT testing, test pressure is applied to the valve body under the disc and pressurizes the space around the disc and between the two body seats. This test method tests both the upstream seat and downstream seat simultaneously and does not distinguish between the leakage paths. The total leakage measured by this test is the aggregate of the individual contributions of each seat and overestimates the leakage that would be obtained through the valve if measured by pressurizing the upstream side of the valve alone. In addition,

during routine operation and under accident conditions, only the upstream side of the valve would be pressurized. This applies additional seating force to the disc against the downstream seat. The bonnet test, in contrast, does not force the disc into the seat. The absence of steam cutting indications that would be evident had leakage of the magnitude measured during the test actually existed indicates that the valve was properly seated during plant operation. The actual minimum path leakage was most likely significantly less than the measured value.

While this path is considered a secondary containment bypass leakage path that would release directly to the environment, the main steam line drain flowpath is through multiple balance of plant (BOP) valves to the main condenser [SG]. During operation one BOP valve is maintained closed with a small bypass flowpath open to the condenser. This drain path would allow limited leakage 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 main condenser itself would provide some scrubbing, provide a hold up volume and thus limit release to the environment.

The radiological response description of the UFSAR section 15.6.5, Lost of Coolant Accident Inside Containment, is conservative. The criteria and assumptions conservatively overestimate the radiological consequences. Among the assumptions that are used is the primary containment is postulated to leak at a constant rate of 0.5 percent per day of the primary containment air volume for 30 days after the postulated accident. This assumption results in primary containment pressure to remain near its design value over the entire 30 day period. An analysis has shown that the pressure within 2 days after a postulated design basis loss of coolant accident (LOCA) would be a small fraction of the maximum design pressure.

The radiological consequence analysis in the UFSAR assumes a conservatively large source term (e.g., 50 percent of the iodine and 100 percent of the noble gases are instantaneously released to the containment environment). No credit is taken for scrubbing by the reactor coolant. Furthermore, this degree of release from the fuel implies widespread fuel damage beyond what would result assuming operation of the Emergency Core Cooling System. 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.

This event was reviewed against the criteria of NRC Inspection Manual Chapter 0609, Significance Determination Process, Appendix H, Containment Integrity Significance Determination Process. Since the MSIV steam line drain isolation valves are components whose functions are unrelated to the prevention or mitigation of core damage, but are part of a potential containment bypass leakage path, this event would be considered a Type B Finding. Fermi 2 has a BWR Mark I containment design. Table 3 of NRC Inspection Manual Chapter 0609, Appendix H, identifies structures, systems, and components (SSC) which could potentially influence the Large Early Release Frequency. Since the main steam line drain isolation valves are not SSC included in Table 3 for BWR Mark I containments, this event would constitute a finding of low risk significance.

Corrective Actions:

This event was documented in the Fermi 2 Corrective Action Program. The B2103-F016 valve disc was turned 180 degrees to achieve improved seating. The B2103-F019 valve disc was built up to allow the in body and valve disc seating surfaces to match and the valve guides were machined to remove the offset. The valve seats of both valves were lapped and polished.

Further corrective actions are being considered within the established processes and priorities of the Fermi 2 Corrective Action Program.

Additional Information:

A. Failed Components

Component: B2103-F016, Main Steam Line Drain Inboard Isolation Valve
Description: Motor operated three inch gate valve
Manufacturer: Wm. Powell Company
Type: 19023WE

Component: B2103-F019, Main Steam Line Drain Outboard Isolation Valve
Description: Motor operated three inch gate valve
Manufacturer: Wm. Powell Company
Type: 19023WE

B. Previous LERs on Similar Problems

No similar LERs were identified within the past two years. This is the first time this penetration has failed to meet the requirements of secondary containment bypass leakage.