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Detroit Edison

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Nuclear Generation

10CFR50.73

May 4, 1990
NRC-90-0081

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

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

Subject: Licensee Event Report (LER) No. 90-004-00

Please find enclosed LER No. 90-004-00, dated May 4, 1990. A copy of this LER is also being sent to the Regional Administrator, USNRC Region III.

If you have any questions, please contact Lynne Goodman at (313) 586-4211.

Sincerely,

Enclosure: NRC Forms 366, 366A

cc: A. B. Davis
J. R. Eckert
R. W. DeFayette/W. L. Axelson
W. G. Rogers
J. F. Stang

Wayne County Emergency
Management Division

IE22
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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): Fermi 2

DOCKET NUMBER (2): 0 5 0 0 0 3 4 1

PAGE (3): 1 OF 6

TITLE (4): Residual Heat Removal System Small Bore Connections

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 4	0 4	9 0	9 0	0 0 4	0 0	0 5	0 4	9 0		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9):

POWER LEVEL (10): 0 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(a)(1)(i)
- [ ] 20.405(a)(1)(ii)
- [ ] 20.405(a)(1)(iii)
- [ ] 20.405(a)(1)(iv)
- [ ] 20.405(a)(1)(v)
- [ ] 20.405(c)
- [ ] 50.36(c)(1)
- [ ] 50.36(c)(2)
- [ ] 50.73(a)(2)(i)
- [X] 50.73(a)(2)(ii)
- [ ] 50.73(a)(2)(iii)
- [ ] 50.73(a)(2)(iv)
- [ ] 50.73(a)(2)(v)
- [ ] 50.73(a)(2)(vii)
- [ ] 50.73(a)(2)(viii)(A)
- [ ] 50.73(a)(2)(viii)(B)
- [ ] 50.73(a)(2)(x)
- [ ] 73.71(b)
- [ ] 73.71(c)
- [X] OTHER (Specify in Abstract below and in Text, NRC Form 366A)

LICENSEE CONTACT FOR THIS LER (12)

NAME: Lynne Goodman, Director Nuclear Licensing

TELEPHONE NUMBER: AREA CODE 3 1 3 | 5 8 6 - 4 2 1 1

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)

On April 4, 1990, an initial evaluation was completed of analytical results on vibratory stress experienced by some small bore connections to the Residual Heat Removal (RHR) System in the RHR pump rooms. Five RHR connections were calculated to experience stress above the ASME Section III 1971 code requirement, which was the committed design requirement. The calculations were conservative using hand held accelerometer data and simple beam models.

Short term modifications were made to the affected RHR welds. Permanent modifications will be made to the connections. Other small bore test, vent and drain connections on the RHR system are being inspected. Vibration data will be taken on small bore lines that appear susceptible to high vibratory stress.

Additionally, on April 13, 1990, a small crack was observed on the Division I Low Pressure Coolant Injection high point vent in the drywell. The connection was modified and all other RHR small bore connections in the drywell were visually inspected and found to be acceptable.

Plant Conditions at Time of Discovery:

Operational Condition :	1 (power operation)
Reactor Power:	100%
Reactor Pressure:	1005 psig (approx.)
Reactor Temperature:	525 degrees F (approx.)

Description of the Event:

In late 1987 and early 1988 a vibration monitoring and analysis program was established for small bore piping connections to the Residual Heat Removal (RHR) System (BO) in the RHR pump (P) rooms due to vibration related concerns. Vibration related problems had been experienced with motor electrical termination boxes, a hydraulic snubber (SNB) and a small drain connection. Data on small bore connections was taken in 1988 and 1989. On April 3, 1990 the Detroit Edison Engineering Research Department, which was evaluating the data, provided the first numerical results to Fermi 2. An initial evaluation of the analytical results was completed on April 4, 1990. The report indicated that 5 existing RHR small bore connections were experiencing vibratory stress levels above the design standard (ASME Section III, 1971) of 12,500 psi (at 10E6 cycles). All were calculated to be below 20,000 psi. Additionally, two Emergency Equipment Cooling Water (EECW) (BI) connections to the pumps were calculated to experience stresses above 12,500 psi, though one was only slightly above at 12,800 psi. The vibratory stress levels were calculated using measured accelerations or displacements and simple beam or finite element mathematical models. These simple models are conservative in nature.

Additionally on April 13, 1990, while the plant was in a cold shutdown, leakage from a crack was observed by an operator doing a system valve (V) lineup verification. The crack was located in the heat affected zone of a weld on the Division I Low Pressure Coolant Injection (LPCI) piping high point vent in the drywell. The RHR pumps supply the LPCI system. The cause of the failure was determined to be fatigue induced and had initiated at the outside surface, as expected with a fatigue failure. This small bore connection was cantilevered, with two 1500 psi (12 lb) valves located along approximately 17 inches of small bore pipe. This configuration resulted in a large moment arm on the vent pipe/connection and was not supported. As a result, over the long-term, it failed as noted above.

Cause of Event:

Vibratory stresses were calculated by a simplified conservative model to be greater than expected. The design bases for connections to meet vibration standards is not an exact science since the vibratory loadings are not known at the time the connections are designed. The connections were built to a Detroit Edison standard. During preoperational testing, a Piping Vibration and Dynamic Effects Testing (PVDET) program was implemented. This involved measurement of vibration on specified large bore piping systems and visual inspection of small bore connections to determine whether they were in resonance with system vibration or posed any danger to personnel or potential damage to the system. The inspections were performed by qualified contractor inspectors with expertise in this area. During that inspection the connections now determined to be of concern were not identified as problems. These inspections were visual in nature, as required by the PVDET for small bore piping. Due to the vibration related concerns experienced in the pump rooms, a further monitoring program was initiated which included acceleration and displacement measurements on small bore piping. The results of this testing show some connections experience greater vibratory stresses than desired.

With regards to the vent line in the drywell, the design of the small bore piping, which was cantilevered to avoid building structural steel, combined with the heavy unsupported valves, provided a large moment arm which ultimately resulted in the fatigue failure.

Analysis of Event:

Between the time the data was taken and the results of the calculations received, two connections in the pump rooms had experienced failure. Both of these were calculated to have experienced stress levels significantly above the 1971 code allowable value of 12,500 psi. All failures were detected by observation of minor leakage. If a breach had occurred it would have been detected by sump level monitoring, the floor flood detectors or the temperature leak detection system (IJ). Catastrophic failures have not been experienced, nor are they expected to occur.

If the RHR connections calculated to be experiencing stress above 12,500 psi had broken at the time the system was needed to mitigate the consequences of an accident, the pumps could still have provided adequate cooling flow to the core or torus. Additionally, the core spray system (BM) alone can adequately cool the core. These RHR connections are all on the suction side of the pumps, with one of the 4 pumps having all analyzed small bore connections less than 12,500 psi. If a tap had broken while the system was in service, there would have been insignificant offsite radiological consequences, unless it happened at the same time as a core damage accident. Even if a connection had fully broken during a 100% fuel damage accident, the Part 100 limit at the site boundary would have been met. If all 5 taps had broken during the postulated accident, using the conservative licensing basis source term calculations assuming 100% fuel damage, the lines with the broken taps would have had to be isolated within 1 hour to prevent the Part 100 limit from being exceeded at the site boundary. The lines are isolated using remote manual valves on the suction side of each pump and the discharge side check valve. The leak detection systems would have alerted the operators to the presence of the break in the event of such an improbable scenario.

While calculations have not been completed for all the data taken, the two connections that appeared potentially susceptible to high stress were temporarily modified as described in the corrective action section to cover any possible problems in the pump room.

Regarding the cracked high point vent line in the drywell, if it had broken, the consequences would have been bounded by the analysis for a small line break inside containment. When found, there was only a small amount of seepage through the approximately 120 degree crack.

The analysis performed was conservative, using accelerometer and displacement data and simple beam models. The actual stresses that would be be experienced are expected to be smaller than those calculated.

Corrective Actions:

Following initial evaluation of the calculated stress values, the five RHR connections reported as experiencing stress levels in excess of 12,500 psi had their welds modified. The welds joining the nipples to the headers were "buttered" to a 2:1 slope and blended to reduce stress concentrations. This action provided an additional level of assurance that these connections would not fail. Additionally, two other RHR connections which had not yet been analyzed, but appeared to be susceptible to higher stress values based on their location and geometry in comparison to those taps analyzed, were "buttered." An EECW connection was also modified. This work was completed April 8, 1990.

Information on the stress levels was extensively discussed with NRC representatives. On April 9, 1990, an Engineering Functional Analysis was submitted to the NRC by letter NRC-90-0053. Meetings were held at NRC offices on April 16 (Region III) and April 24, 1990 (NRR/ Headquarters).

As a conservative measure all the RHR connections which have been analyzed to have stresses above 10,000 psi will be modified to reduce the stresses to no greater than 7690 psi, which is the current code limit and the acceptance criteria used for large bore piping during the PVDET program. The welds that were "buttered" will also be modified. As a minimum, the modification of the 11 taps meeting this criterion is expected to be completed in early May, depending on the length of time it takes to modify each connection and any Technical Specification restrictions due to other equipment. Additionally, four EECW taps will be modified.

Analytical evaluations of the remaining vibration data taken in the RHR pump rooms is expected to be completed in mid-May. The data will be promptly evaluated to determine if any other modifications are necessary.

While pump induced vibration primarily affects the piping in the pump rooms, it has been decided to visually inspect potentially susceptible welds on vents, drains and test connections on the remainder of the RHR system. This inspection has started. Where a susceptible connection configuration is identified, vibration measurements will be taken on the connection and a dye penetrant test performed on the susceptible weld. Any modifications required as a result of these activities will be completed by the end of the second refueling outage.

The cracked vent line in the drywell was replaced with a shorter connection containing only one valve and a welded cap. Post-modification testing resulted in a stress level well within the acceptance criteria of 7690 psi. All other RHR small-bore connection welds in the drywell were visually inspected with no other problems noted. Vibration data was taken on the high point vent on Division II of LPCI. The conservatively calculated stress value was 10,500 psi. This connection will be modified prior to startup from the second refueling outage.

Similar Occurrences:

Licensee Event Report 86-016, "Misinterpretation of Computer Data Results in Calculated Pipe Stress to Exceed ASME Code", describes an event in which piping stresses were determined to exceed ASME code allowable values.