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Detroit Edison

Fermi 2
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10CFR50.73

January 27, 1997
NRC-97-0013

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

References: 1) Fermi 2
NRC Docket No. 50-341
NRC License No. NPF-43

Subject: Licensee Event Report (LER) No. 96-024

Pursuant to 10CFR50.73, Detroit Edison is submitting the enclosed LER No. 96-024. This LER addresses an automatic reactor scram due to perturbations in the Reactor Pressure Vessel Water Level Indicating System reference leg backfill system while placing it in service.

There are no commitments made in this LER.

If you have any questions, please contact Ron Wittschen at (313) 586-1267.

Sincerely,

cc: A. B. Beach
M. J. Jordan
A. J. Kugler
C. O'Keefe
M. V. Yudasz, Jr.
Region III
Wayne County Emergency Management Division

LICENSEE EVENT REPORT (LER)

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TITLE (4) Automatic Reactor Scram Due to Perturbations in the Reference Leg Backfill System While Placing the System In Service

EVENT DATE (5) MON	DAY	YR	LER NUMBER (6) YR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MON	DAY	YR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER (S)
										0 5 0 0 0
12	28	96	96	- 0 2 4 -	0 0	01	27	97		0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 1 8

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (11)

X 10 CFR 50.73(a)(2)(iv)

OTHER -

(Specify in Abstract below and in text, NRC Form 366A)

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER AREA CODE	
Ron Wittschen - Compliance Engineer	313	586-1267

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 (16)

On December 28, 1996, at 1314 hours, the Division 1 reference leg backfill system was being placed in service. As the isolation valve from the backfill system to the reference leg was opened, a rapid pressure spike occurred in the reactor pressure vessel (RPV) water level instrument reference leg which induced several actuations from the false water level signals. A High Level 8 actuation on Division 1 level instrumentation occurred that tripped the operating reactor feed pump. This was immediately followed by a RPV water Low Level 2 and Low Level 3 actuations. The reactor scrammed properly on the false Level 3 signal and all rods fully inserted. Due to the short duration of the Division 1 reference leg level spike, not all Level 2 trip channels actuated. These are expected responses due to the short signal duration.

The causes of this event are design deficiencies and procedural inadequacies. The design of the backfill system did not facilitate complete venting of entrained air in a two foot section of Division 1tubing. The procedure used to place the backfill system in service did not adequately address the potential for a pressure spike when placing the system back in service after filling and venting. If the backfill system is not adequately repressurized following venting, a transient may occur in the common reference leg while placing the system in service.

Corrective actions included revising the procedure for placing the backfill system in service to ensure the system is completely filled and vented and that the differential pressure between the backfill system and the reference leg is minimized prior to placing it in service. On Division 1, a design modification was developed and implemented to relocate a test valve used for system venting closer to the backfill isolation valve. The test valves for both divisions were elevated to facilitate proper venting of the lines.

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Initial Plant Condition:

Operational Condition:	1 (Power Operation)
Reactor Power:	18 Percent
Reactor Pressure:	960 psig
Reactor Temperature:	540 degrees Fahrenheit

Description of the Event:

A. Background

Three reactor pressure vessel (RPV) low water level isolation trip [JE] settings are used to completely isolate the RPV and the primary containment. The level signals are defined as follows:

a. Level 3 (L3) is the highest of the three. It initiates the reactor pressure vessel water level scram; isolates the residual heat removal (RHR) system [BO]; actuates the traversing in-core probe (TIP) system [IG] withdrawal and isolation; and isolates the drywell equipment and floor drains.

b. Level 2 (L2) is selected to be less than the level resulting from a void collapse occurring in the event of a scram from full power. It is the initiation level for the reactor core isolation cooling (RCIC)[BN], the high pressure coolant injection (HPCI)[BJ] systems, and the alternate rod insertion (ARI) logics. The L2 signal also isolates the majority of the nuclear pressure boundary lines and the primary and secondary containment paths.

c. The final isolation level is Level 1 (L1). The L1 setpoint is far enough above the top of the active fuel and is selected based on the time required for the RHR [BO] and Core Spray (BG) systems to function in the event of a large break. This level setting provides automatic isolation of the RHR and Core Spray systems, which penetrate the primary containment, if they are open. Level 1 also closes the main steam line isolation valves (MSIVs)[ISV].

A fourth low water-level setpoint also provides an anticipatory alarm and is designated Level 4 (L4), and actuates prior to the L3 setpoint on decreasing RPV water level.

Following an accident, High water level (Level 8) in the RPV indicates that the makeup systems have performed satisfactorily in providing makeup water to the RPV. Further increase in level could result in turbine [TRB] damage caused by gross carryover of moisture. Therefore, the Level 8 (L8) signal closes various turbine steam supply valves (RCIC, HPCI, Main Turbine stop valves[ISV]) and also will isolate the standby feedwater (SBFW) injection valve [INV] and trip the reactor feed pumps [P] if they are in operation.

Fermi 2 is a BWR/4 design that uses only non-heated reference columns that are maintained full of condensate by the condensing chambers for all reactor vessel level measurements. A backfill system is installed on each instrument reference leg. The system provides a metered flow of water from the control rod drive system to each leg. The backfill is designed to prevent the accumulation of dissolved noncondensible gases in the reference legs.

B. Event Description

On December 28, 1996, at 1314 hours, while performing procedure 46.000.046, "Filling Reactor Instrumentation Sensing Lines and Operation of the Reactor Reference Leg Back Fill System," a pressure perturbation was received in the Division 1 reference leg instrumentation. This pressure perturbation created false high and low reactor water level actuations. As a result, attendant monitoring instruments generated near simultaneous false Level 2, 3, 4, and 8 signals of a very short duration.

The operating reactor feed pump [P] tripped as a result of the Level 8 signal. A reactor trip resulted from the false Level 3 signal and all control rods [ROD] fully inserted. The false Level 3 signal also resulted in actuation for containment isolation Groups 4, 13 and 15. The Level 2 signal resulted in partial actuations of containment isolation Groups 2, 10, 12, 14, 16, 17, 18; the automatic start of a Division 1 control air compressor (CAC)[CMP] unit; the automatic start of the standby gas treatment system (SGTS); trip and isolation of the reactor building heating ventilation and air conditioning (RBHVAC); and shift of the control center HVAC (CCHVAC) system to the recirculation mode of operation. The Level 2 alternate rod insertion (ARI) logic designed to mitigate the potential consequences of an anticipated transient without scram (ATWS) was also activated during this event. This resulted in the trip of both reactor recirculation pumps and initiation of the ARI logic. A standby feedwater (SBFW) pump was manually started to maintain reactor vessel level in accordance with procedures. The lowest actual reactor vessel level was 164 inches, which is well above the Level 2 actuation setpoint of 110 inches.

A review of the Sequence-of-Events Recorder (SOER) [IQ] data, General Electric Transient Analysis Recorder System (GETARS) data, and wide range level recorder [LR] data for the automatic scram was performed. Based on the initiating conditions and the plant response, the reactor scram was determined to be consistent with a pressure transient on the reference leg from the backfill system, which resulted in false level indication and associated level trip signals. The data indicated that the Division 2 level recorder [LR], which was not affected by the perturbation, did not indicate a significant change in reactor vessel level until the reactor feed pump was tripped and the automatic reactor scram from the false Level 3 trip signal occurred. The remainder of the SOER data indications were consistent with a reactor SCRAM at low power levels.

On December 28, 1996, at 1426 hours, a four-hour notification was made to the NRC because a reactor protection system actuation is an engineered safety feature (ESF) actuation. Additional

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information and clarification of the original notification was made on December 30, at 1835 hours.

Several key components/systems did not actuate on the level trip signals received. These components/systems were:

- Closure of the Nitrogen Inerting Isolation Valve, T48-F455 [ISV]
- Automatic start of HPCI
- Automatic start of RCIC
- Actuation of the Low Pressure Core Injection (LPCI) Loop Select logic

These are all Level 2 actuated components. Subsequent investigation determined that T48-F455 did not go closed due to the short time the actuation signal was present, approximately 56 milliseconds, relative to the time required to actuate the relays that complete the isolation logic. Furthermore, due to the limited duration of the level perturbation signals (56 milliseconds), it was determined that the relaying for RCIC, HPCI, and LPCI Loop Select logic sequence never completed. The LPCI Loop Select annunciator, which is actuated from an auxiliary relay for the Level 2 signal transmitter, was activated during the event. It requires only one side of the logic to initiate the alarm. Surveillances 44.030.251 and 44.030.253 "ECCS - Reactor Vessel Water Level (Level 1, 2, and 8) Division 1, Channel Functional Tests" were completed to verify the logic actually was functioning properly. The ARI initiation signals are designed to seal in the initiation logic, however it takes approximately 15 seconds to depressurize the scram pilot valve air header. Both the Level 3 and manual SCRAM were initiated prior to the 15 seconds required for ARI to initiate control rod insertion.

Cause of the Event:

The causes of this event are design deficiencies AND procedural inadequacies. The design of the backfill system did not facilitate complete venting of entrained air in a two foot section of tubing for the Division 1 arrangement. Additionally, Procedure 46.000.046 did not address the potential for a pressure spike due to the high differential pressure when restoring the system to service after the backfill system was filled and vented. With the backfill system de-pressurized, there is a large differential pressure across the backfill system to the reference leg isolation valve when it is placed into service (approximately 950 psig). The potential entrained air void in the two foot section of tubing, combined with the large differential pressure across the isolation valve when it was opened, created a pressure spike which induced the level perturbations seen in the reference leg.

The cause for the lack of response of some systems/components to the event is that those systems/components which did not respond within the 56 millisecond time frame of the pressure perturbation required a longer response time for their relay logic to complete. Therefore, they were not expected to actuate for the reference leg backfill perturbation.

Analysis of the Event:

The equipment and components actuated during this event performed their intended safety functions. The subsequent scram of the reactor and plant shutdown was according to design. Chapter 15 of the Updated Final Safety Analysis Report (UFSAR) bounds this event in that a reactor scram on a Level 3 signal is initiated for a loss of all feedwater flow, which is a consequence of the Level 8 trip of the operating reactor feed pump. Therefore, the health and safety of the public were not adversely affected by this event.

Corrective Actions:

A. Immediate Corrective Actions

Partial surveillances were performed to verify that T48-F455 would isolate due to a valid signal and that HPCI, RCIC, and LPCI Loop Select logic would initiate upon a valid signal.

B. Corrective Actions to Prevent Recurrence

1. Procedure 46.000.046 was revised to ensure the backfill system is completely filled and vented prior to placing it in service and to ensure differential pressure across the backfill isolation valve is minimized prior to opening of the valves.

2. On Division 1, a design modification was developed and implemented to relocate a test valve used for system venting, closer to the backfill isolation valve. This eliminated a length of tubing which could not previously be vented and where potential voiding could occur. Additionally, the test valves for both divisions were elevated to facilitate proper venting of the lines. Backflushing of both divisions of the backfill system was completed after the design modification was implemented.

Following completion of the design modification and using the revised procedure, the backfill system was successfully placed in service during the next plant restart with no level perturbation seen.

Additional Information:

A. Failed Components

None.

B. Previous LERs on Similar Problems

None.