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DTE Energy

10 CFR 50.73

November 5, 2015
NRC-15-0094

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 (LER) No. 2015-006

Pursuant to 10 CFR 50.73 (a)(2)(iv)(A) and (a)(2)(v)(C), DTE Electric Company (DTE) is submitting LER No. 2015-006, Reactor Scram Due to Loss of Turbine Building Closed Cooling Water.

No commitments are being made in this LER.

Should you have any questions or require additional information, please contact Mr. Christopher R. Robinson of my staff at (734) 586-5076.

Sincerely,

Vito A. Kaminskas
Site Vice President

Enclosure

cc: NRC Project Manager
NRC Resident Office
Reactor Projects Chief, Branch 5, Region III
Regional Administrator, Region III
Michigan Public Service Commission
Regulated Energy Division (kindschl@michigan.gov)

Enclosure to
NRC-15-0094

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

LER 2015-006, Reactor Scram Due to Loss of Turbine Building Closed Cooling Water

NRC FORM 366 (02-2014) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 01/31/2017

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 FOIA, Privacy and Information Collections Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to Infocollects.Resource@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

Reactor Scram Due to Loss of Turbine Building Closed Cooling Water

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
									N/A	05000 N/A
09	13	2015	2015	- 006	- 00	11	05	2015	FACILITY NAME N/A	DOCKET NUMBER 05000 N/A

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
1	☐ 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)
10. POWER LEVEL	☐ 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)
100	☐ 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 THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Christopher R. Robinson - Manager, Nuclear Licensing	(734) 586-5076

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	KB	HX	Y030	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 2305 EDT on September 13, 2015, a manual reactor scram was initiated in response to a loss of Turbine Building Closed Cooling Water (TBCCW). A tube leak occurred in the East TBCCW heat exchanger which ultimately tripped the running TBCCW pumps. The loss of TBCCW caused all Station Air Compressors (SACs) to trip on loss of cooling. The loss of SACs resulted in the Instrument Air header pressure to degrade and the Secondary Containment (SC) isolation dampers to drift closed. This resulted in a positive SC differential pressure. At 2325 EDT, Operators started the Standby Gas Treatment system and manually inserted a SC isolation signal. SC vacuum was promptly restored to within Technical Specification limits.

At 0409 EDT on September 14, 2015, the Reactor Core Isolation Cooling (RCIC) system was manually actuated due to the identification of an unisolable leak in the Standby Feedwater (SBFW) System. The unisolable leak occurred due to a failed weld associated with a SBFW drain valve. At 1847 EDT, an automatic Reactor Protection System actuation occurred due to Reactor Water Level (RWL) 3 while shutdown in MODE 3. Operators were manually controlling Reactor Pressure Vessel (RPV) level and pressure with RCIC and by cycling Safety Relief Valves (SRVs); however, at 1847 EDT the RPV level went below the Level 3 setpoint. Operators promptly restored RPV level with manual operation of RCIC.

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 01/31/2017

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 FOIA, Privacy and Information Collections Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to Infocollects.Resource@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.

NARRATIVE

Initial Plant Conditions

Mode 1
Reactor Power 100 percent

Description of the Event

At 2305 EDT on September 13, 2015, a manual reactor scram was initiated in response to a loss of all Turbine Building Closed Cooling Water (TBCCW) [[KB]] in accordance with plant procedures. All control rods were fully inserted and the lowest Reactor Water Level (RWL) reached was 137 inches above Top of Active Fuel which is below the RWL 3 setpoint of 173 inches. All isolations and actuations for RWL 3 occurred as expected. Decay heat was initially being removed through the Main Turbine Bypass System [[JI]] to the Main Condenser [[COND]], however, as a result of the loss of TBCCW, the Reactor Feed Pumps lost cooling and had to be secured. At 2310 EDT, the Standby Feedwater (SBFW) [[SJ]] system was initiated. A field investigation later verified that a tube leak occurred in the East TBCCW heat exchanger [[HX]], causing General Service Water (GSW) to flow into the lower pressure TBCCW system. This resulted in a TBCCW head tank [[TK]] level increase and water flowing from the TBCCW head tank relief valve [[RV]]. The interaction of TBCCW system pressure fluctuations with the TBCCW tank instrumentation ultimately caused a trip of the running TBCCW pumps and a loss of TBCCW.

The loss of TBCCW also caused all Station Air Compressors (SACs) [[CMP]] to trip on loss of cooling. The loss of SACs caused the Instrument Air header pressure to degrade to the point at which the Secondary Containment (SC) isolation dampers [[DMP]] drifted closed. This resulted in the Reactor Building [[NG]] pressure going positive and exceeding the Technical Specification minimum requirement of -0.125 inches water column. At 2325 EDT, Operators started the Standby Gas Treatment system [[BH]] and manually inserted a SC isolation signal. SC vacuum was restored to within Technical Specification limits. The Technical Specification limit was exceeded for approximately 3 minutes and 43 seconds and the maximum pressure recorded was 1.932 inches water column. Additionally, Operators were monitoring for expected Main Steam Isolation Valve (MSIV) [[ISV]] drift due to the degraded Instrument Air header pressure. When outboard MSIVs were observed to be drifting, Operators closed the outboard and inboard MSIVs at 2345 EDT. At 2352 EDT, Low-Low set Safety Relief Valves (SRVs) [[RV]] reached their setpoint and began automatic cycling to control reactor pressure.

At 0405 EDT on September 14, 2015, a Radiation Protection Technician discovered an unisolable leak in a weld associated with a Standby Feedwater (SBFW) drain valve [[V]] and the system was declared non-functional. At 0409 EDT, the Reactor Core Isolation Cooling (RCIC) [[BN]] system was placed in service due to the unisolable leak in the SBFW System. Reactor water level and pressure were then being controlled though the RCIC system and SRVs.

At 1847 EDT on September 14, 2015, a valid automatic Reactor Protection System (RPS) actuation occurred due to Reactor Water reaching Level 3. Operators were manually controlling Reactor Pressure Vessel (RPV) level and pressure with RCIC and SRVs. While operators were cycling SRVs, the RPV level went below the Level 3 setpoint. Operators promptly restored RPV level by manual operation of RCIC. The Level 3 actuations and associated isolations were verified to occur as expected.

The manual scram is reportable under 10 CFR 50.73(a)(2)(iv)(A), as an event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B). A 4-hour event notification (No. 51391) was made to the NRC based on meeting the reporting criteria of 10 CFR 72(b)(2)(iv)(B).

The loss of SC function is reportable under 10 CFR 50.73(a)(2)(v)(C) as an event or condition that could have prevented the fulfillment of a safety function needed to control the release of radioactive material. An 8-hour event notification was made to the NRC based on meeting the reporting criteria of 10 CFR 50.72(b)(3)(v)(C).

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NARRATIVE

The manual actuation of the RCIC system and automatic RPS actuation due to RPV water Level 3 are reportable under 10 CFR 50.73 (a)(2)(iv)(A), as an event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B). Follow-up notifications were made to the NRC based on meeting the reporting criteria of 10 CFR 50.72(b)(3)(iv)(A) for this ongoing event.

There were no radiological releases associated with this event.

Significant Safety Consequences and Implications

There were no significant safety consequences associated with this event. At no time during this event was there a potential for endangering the public health and safety.

Manual Scram

The safety significance with respect to the manual scram is considered less than minimal, since a manual scram is inherently a safety function. No safety-related equipment was out of service at the time of the event and all offsite power sources were adequate and available throughout the duration of the event.

Loss of Secondary Containment

The specified safety function of the SC is to contain, dilute, and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). SC in conjunction with SGTS is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment. For the SC to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained.

During this particular event, SC vacuum briefly exceeded the Technical Specification limit when the isolation dampers drifted closed. When the dampers drifted closed, Operations started the SGTS system, manually inserted a SC isolation signal, and restored SC vacuum within the Technical Specification limit. In Chapter 15 of the Fermi 2 Updated Final Safety Analysis Report (UFSAR), Reactor Building Heating, Ventilation, and Cooling (RBHVAC) is assumed lost at the onset of a Loss of Coolant Accident (LOCA) concurrent with a Loss of Offsite Power. As a result, calculations show that the SC would be pressurized until the SGTS restores vacuum. For this particular event, the SC vacuum degraded when the SACs tripped on a loss of cooling causing the SC isolation dampers to drift closed. The structural integrity (i.e., leak tightness) of the SC was re-confirmed when the safety-related SGTS restored vacuum to greater than 0.125 inches of water column.

If the Design Basis Accident (DBA) LOCA for SC concurrent with a Loss of Offsite Power were to occur during the time when SC pressure exceeded the Technical Specification limit, the SC was sufficiently leak tight such that the SGTS would still have established and maintained vacuum greater than the Technical Specification required value.

The radiological consequences of the DBA LOCA for SC contained in Chapter 15 of the Fermi 2 UFSAR result in doses that are below 10 CFR 50.67. The SC is assumed to be at a pressure of -0.125 inches water column at the onset of the LOCA. For this particular event, had the DBA LOCA for SC actually occurred, the increase in magnitude of radiological dose as a result of increased draw-down time from starting at 1.932 vice -0.125 inches of water column, would be minimal and negated by conservative assumptions in the existing analysis (e.g., 100% exfiltration from SC during the first 15 minutes of drawdown with SGTS in operation, 10% exfiltration from SC with SGTS in operation throughout the remaining 30 day duration of the accident, no holdup time in SC throughout the 30 day duration of the accident, and all exfiltration and filtered releases are at ground level). These conservative assumptions are not reflective of actual plant conditions and configuration.

NARRATIVE

This qualitative evaluation concludes that no actual loss of safety function occurred. This LER is required because the reporting threshold is "could have" prevented fulfillment of a safety function, which was valid at the time that SC was declared INOPERABLE.

Manual Actuation of RCIC System and Automatic RPS Actuation

The RCIC system was manually actuated to allow for Remote manual operation to control flow that matches decay heat steam generation after shutdown. This resulted in the reactor core being successfully cooled down to Mode 4 and transitioned to Shutdown Cooling system operation. Therefore, no safety consequences were attributed to the manual actuation of RCIC.

RPV water level briefly dropped below Level 3 and caused a valid automatic RPS actuation signal. Since all control rods were already fully-inserted into the core, the RPS safety function was already fulfilled.

No safety-related equipment was out of service at the time of the event and all offsite power sources were adequate and available throughout the duration of the event.

Cause of the Event

• Manual Scram - A tube leak occurred in the East TBCCW heat exchanger, causing GSW to flow into the lower pressure TBCCW system ultimately causing a trip of the running TBCCW pumps which resulted in a loss of TBCCW.
• Loss of SC - The loss of TBCCW also caused all SACs to trip on loss of cooling. The loss of SACs caused the Instrument Air header pressure to degrade to the point at which the SC isolation dampers drifted closed. This resulted in a degradation of the Reactor Building vacuum.
• Manual Actuation of RCIC and Automatic Actuation of RPS – When SBFW was removed from service due to an unisolable leak, RCIC was manually started per plant procedures to control RPV water level. While level and pressure were being manually controlled with RCIC and SRVs, a Reactor Operator did not timely correct a RPV level oscillation to prevent going below the Level 3 setpoint.

Corrective Actions

A work order has been generated to inspect and repair the East TBCCW heat exchanger. A work order was also generated to inspect the West TBCCW heat exchanger and performs repairs as necessary.

A work order has been generated to rework the failed weld associated with the SBFW drain valve.

These events were documented and are being evaluated in the Fermi 2 Corrective Action Program.

Additional Information

A. Component Failures

Failed Component: East TBCCW Heat Exchanger
Function: Transfer heat between systems
Manufacturer: Yuba
Model Number: AEL
Primary Failure Cause: Stress Corrosion Cracking

B. Previous Licensee Event Reports (LERs) for Similar Events:

LER 2013-001 involved a loss of SC function due to an RBHVAC system equipment malfunction. The cause of that event was related to damper sequencing. Therefore, the corrective actions for that event would not have precluded this event.

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LER 2013-003 involved a loss of SC function due to an RBHVAC system equipment malfunction. The cause of that event was related a RBHVAC system trip caused by the lack of steam flow through a heating coil due to inadequate draining of the downstream steam trap. Therefore, the corrective actions for that event would not have precluded this event.

LER 2015-001 involved a loss of SC function due to an RBHVAC system trip caused by a valid actuation of a freeze protection device. Therefore, the corrective actions for that event would not have precluded this event.

LER 2015-004 involved the loss of SC function due to reverse rotation of the RBHVAC center exhaust fan during post-maintenance testing. The cause of the event was reversed electrical leads. Therefore, the corrective actions for that event would not have precluded this event.

LER 2015-005 involved a loss of SC function due to an RBHVAC system equipment malfunction. The cause of that event was premature failure of a relay. Therefore, the corrective actions for that event would not have precluded this event.