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10 CFR 50.73

February 6, 2017

TMI-17-009

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

THREE MILE ISLAND NUCLEAR STATION, UNIT 1 (TMI-1)
RENEWED FACILITY OPERATING LICENSE NO. DPR-50
DOCKET NO. 50-289

SUBJECT: LICENSEE EVENT REPORT (LER) NO. 2017-002-00
"Leak At High Pressure Connection on Reactor Coolant Pump "A" Thermal Barrier"

This report is being submitted in accordance with 10 CFR 50.73(a)(2)(ii)(A). For additional information regarding this LER, please contact Mike Fitzwater, Sr. Regulatory Engineer, TMI Unit 1 Regulatory Assurance at (717) 948-8228.

There are no regulatory commitments contained in this LER.

Respectively,

Thomas P. Haaf
Plant Manager, Three Mile Island Unit 1
Exelon Generation Co., LLC

cc: Administrator, NRC Region I
TMI Senior Resident Inspector
TMI-1 Project Manager
R. R. Janati, Chief, Division of Nuclear Safety, Pennsylvania Department of Environmental Protection, Bureau of Radiation Protection

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NRR

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(06-2016)

LICENSEE EVENT REPORT (LER)

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

(See NUREG-1022, R.3 for instruction and guidance for completing this form http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)

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4. TITLE

Leak At High Pressure Connection on Reactor Coolant Pump "A" Thermal Barrier

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 05000
12	07	2016	2017	- 002	- 00	02	06	2017	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
N	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☒ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
0%	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
		☐ 50.73(a)(2)(i)(C)	☐ OTHER	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Michael Fitzwater, TMI Unit 1 Regulatory Assurance Engineer	(717) 948-8228

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

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)*

On December 7, 2016 Three Mile Island Unit 1 was in the hot shutdown condition following a planned maintenance outage that replaced the Reactor Coolant Pump (RCP) 1A seal package when a reactor coolant system (RCS) leak was discovered on a welded connection of the Reactor Coolant Pump 1A thermal barrier. The identified leak was determined to be approximately 0.5 gpm, located on the RCS pressure boundary and nonisolable. Operators returned Unit 1 to a cold shutdown condition to repair the leak. The most probable cause for the leak is a latent weld defect that reduced the fatigue strength of the connection, coupled with RCP-1A startup vibration leading to failure. Immediate corrective action involved a weld repair modification to the leak location. Extent of condition applied to five similar locations that were examined with no weld defects identified. Additional corrective actions are planned to implement a weld repair modification to five similar connections on the RCPs during the Fall 2017 maintenance & refueling outage (T1R22). This event is reported as a degraded condition pursuant to 10 CFR 50.73(a)(2)(ii)(A). This event had no effect on public health and safety.

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A. EVENT DESCRIPTION

Plant Conditions before the event:

Babcock & Wilcox – Pressurized Water Reactor – 2568 MWth Core Power
Date/Time: December 7, 2016 / 16:00 hours
Power Level: 0%
Mode: Hot Shutdown

Three Mile Island Unit 1 (TMI-1) was in a hot shutdown condition following a planned maintenance outage that replaced the Reactor Coolant Pump (RCP) "A" seal package. A reactor coolant system (RCS) leak was discovered on Reactor Coolant Pump 1A (RC-P-1A) and estimated to be approximately 0.5 gpm. The location of the leak was identified at a welded connection between the pump thermal barrier and a 1.25 inch pipe that leads to a blank flange. The line was formerly used for instrumentation to monitor differential pressure across the RCP labyrinth seal but was disconnected and blanked in November 2015 during the maintenance and refueling outage (T1R21) as part of an RCP shaft seal modification. The leak location was determined to be nonisolable. In accordance with TMI-1 Technical Specification 3.1.6.4, this condition required the reactor to be shut down, and "a cooldown to the cold shutdown condition shall be initiated within 24 hours of detection." Unit 1 was promptly returned to the cold shutdown condition to perform the repair.

An engineering modification was implemented as a permanent repair to the leak location by removing the labyrinth seal differential pressure (dP) high pressure tap. The welded pipe/flange connection was completely removed and a welded plug was installed to seal this connection in the RC-P-1A thermal barrier. This repair was done in accordance with ASME Section III, 1968 Edition as a Category D weld as defined in Paragraph N-460. The configuration geometry and associated weld repair rendered it impractical to remove the region in question without potential damage. Because no physical evidence could be retained a detailed metallurgical analysis of the failed part was not performed.

B. CAUSE OF EVENT

A root cause investigation was conducted to determine the cause of the RC-P-1A leak, as well as contributing factors that may have led to the failure.

A definitive root cause was not determined. The most probable cause was determined to be a latent weld defect in the socket-weld which reduced the fatigue strength of the connection. Two contributing causes were identified:

- Contributing Cause #1: Short-term high-amplitude vibration resulting from elevated vibrations experienced during pump start-up initiated a flaw at the existing weld defect. This flaw propagated through the pipe wall resulting in a leak.

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- Contributing Cause #2: The vibrations driven by RCP operation had an amplified effect on the piping due to a RCP seal package modification made in 2015 reducing the margin between the natural frequency of the pressure taps and the driving frequencies of the pump.

An extent of condition review was performed. The extent of condition included visual inspections of the remaining labyrinth seal differential pressure (dP) low pressure tap on RC-P-1A and the labyrinth seal dP high and low pressure taps on RC-P-1B and RC-P-1D. RC-P-1C was not included in the extent of condition exams as the new pump installed in 1999 did not have the labyrinth seal dP taps. The RC-P-1A labyrinth seal dP low pressure tap included both external and internal VT-1 examinations. Due to operational sequence of the RCPs and the associated higher vibrations on RC-P-1A relative to RC-P-1B/C/D, the three remaining thermal barrier welded connections on RC-P-1A were inspected (seal injection/make-up location, the Intermediate Closed Cooling Water (ICCW) heat exchanger (HX) outlet, and the ICCW heat exchanger (HX) inlet). No indications were observed during these VT-1 examinations. The ICCW HX outlet/inlet and Makeup seal injection connections are a different configuration, and were not modified by the seal package modification and are therefore less susceptible to vibration fatigue. As a result, those connections are not included in the root cause corrective action extent of condition.

During start-up following repairs, when the plant reached normal operating pressure and temperature, VT-2 examinations were performed on all labyrinth seal connections. No leakage was observed.

The most probable root cause of the RC-P-1A high side labyrinth seal dP tap leak was an existing weld defect from original construction that decreased the fatigue strength of the connection. Specifically, this weld defect created a stress concentration that allowed vibration fatigue to initiate/propagate a flaw. While all welds (socket or buttwelds) are susceptible to weld defects, socket welded connections are the most susceptible to fatigue failures (Contributing Cause #1). Additionally, high cycle fatigue (HCF) would be expected to cause failure within the first couple of operating cycles of a component, and older, unmodified connections are not susceptible (Contributing Cause #2). Therefore, this extent of cause can be limited to the RCP labyrinth seal dP taps by use of the contributing causes.

C. ANALYSIS / SAFETY SIGNIFICANCE

This event resulted in a nonisolable leak in the RCS pressure boundary. The leak was not a threat to the safety of the reactor, since it was already in a safe shutdown condition. The leak extended a planned maintenance shutdown that had an occupational radiological dose implication of approximately 3.579 Rem.

The actual safety consequence was minimal. The leak was approximately 0.5 gpm and the unit was in a shutdown condition at the time of discovery of the leak.

A leak at one of the remaining Reactor Coolant Pump labyrinth seal dP pressure taps was assessed. Independent analysis supports that the failure likely would not have occurred due to vibrational effects alone. Although high amplitude vibrations experienced during startup are required to initiate a flaw, visual inspections after startup did not identify any leakage.

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D. CORRECTIVE ACTIONS

Immediate action was taken to remove the RC-P-1A labyrinth seal differential pressure (dP) high pressure tap. The welded pipe/flange connection was completely removed and a welded plug was installed to seal this connection in the RC-P-1A thermal barrier.

The extent of condition will be addressed by implementing a design change to eliminate the vulnerability of a similar failure occurring on the socket welds for the remaining labyrinth seal dP taps on the three applicable RCPs (five locations) during the Fall 2017 refueling and maintenance outage.

E. PREVIOUS OCCURENCES

Previous Events	Previous Event Review
LER 1995-003-00 Reactor Coolant Leak Due To A Cracked Weld In A Cold Leg Drain Line	On September 9, 1995 TMI-1 was in Hot Shutdown and in the process of cooling down for the Cycle 11 Refueling Outage. The Reactor Coolant System (RCS) was at 535 degrees F and 2000 psig. At approximately 7:00 am, a Reactor Building radiation monitor indicated an increase in iodine activity and at approximately 10:15 am a leak was found in a weld on a nonisolable 2-inch diameter cold leg drain line. The leak was estimated at approximately 20 drops per second. This event was reported in accordance with 10 CFR 50.73.a.2.ii. There were no adverse safety consequences or safety implications that resulted from this event, and this event did not affect the health and safety of the public. Metallurgical evaluations have determined that the failure was fatigue induced and it appears that the fatigue occurred over a long period of time. The most probable root cause for growth of the crack from an initial flaw is reactor coolant turbulent penetration into the drain line coupled with thermal stratification causing fatigue cycles in the weld. The failed line was replaced. Welds in similar lines have been inspected and evaluated to be acceptable for operation. Actions included insulation of the Reactor Coolant Pump suction drain lines, additional inspections, testing, and evaluation to confirm the adequacy of GPU Nuclear response to this event.
LER 2012-003-00 PRESSURIZER HEATER BUNDLE LEAK	On August 22, 2012 Three Mile Island (TMI) Unit 1 discovered an unisolable leak in the upper pressurizer heater bundle diaphragm plate and shut down the reactor. The cause of the leak was Primary Water Stress Corrosion Cracking (PWSCC). The root cause was determined to be "*The use of Alloy 600 materials in high temperature locations was a design weakness in the construction of the TMI station.*" The corrective actions included replacement of the upper pressurizer heater bundle (completed September 2012) and subsequent replacement of the remaining Alloy 600 susceptible pressurizer heater bundle. The leak was not a threat to the safety of the reactor and did not

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	represent a reduction in the public health and safety. A previous PWSCC condition was reported in LER 50-289/2003-003-00. This LER was submitted pursuant to 10 CFR 50.73(a)(2)(i)(A).
LER 2013-001-00 REACTOR COOLANT "B" COLD LEG DRAIN LINE FLAW	On November 7, 2013 TMI-1 was in a refueling shutdown mode for the planned T1R20 refueling and maintenance outage. During a planned ISI volumetric examination of the reactor coolant "B" cold leg drain line a flaw in the pipe weld was discovered. The flaw was located in a 2-inch drain line elbow to pipe weld. The flaw was determined to not meet acceptance standards under ASME Section XI, IWB-3600, "Analytical Evaluation of Flaws", and the RCS strength boundary was considered degraded. This condition required reporting under 10 CFR 50.72(b)(3)(ii)(A) as a non-emergency degraded condition. The eight hour report was made at 13:02 on November 07, 2013 documented under EN# 49512. An extent of condition and ISI scope expansion was performed. Similar pipe configurations were examined and their structural integrity to meet ASME code requirements was confirmed. The flawed section of "B" cold leg drain line was cut out and replaced. This LER was supplemented after receipt of the destructive laboratory test results of the flawed section. There was no actual breach of the RCS that resulted in leakage. There were no adverse safety consequences or safety implications that resulted from this event and this event did not affect the health and safety of the public.
LER 2013-001-01 REACTOR COOLANT "B" COLD LEG DRAIN LINE FLAW	LER 2013-001-01 Supplement submitted June 20, 2014. A destructive laboratory examination and finite element analysis (FEA) was performed on the removed pipe section. The root cause of the crack is unknown but believed to be the result of geometry induced focusing of lower levels of stress, not capable of inducing cracking alone, but when combined at a geometric feature such as a lack of fusion (LOF) at a natural notch, initiated the crack. The lower energy requirements of propagation then governed the crack growth.

* Energy Industry Identification System (EIIS), System Identification (SI) and Component Function Identification (CFI) Codes are included in brackets, [SI/CFI] where applicable, as required by 10 CFR 50.73 (b)(2)(ii)(F).