Exelon Generation

LaSalle County Station
2601 North 21st Road
Marseilles, Illinois 61341

10 CFR 50.73

RA15-065

October 6, 2015

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

LaSalle County Station, Unit 2
Facility Operating License No. NPF-18
NRC Docket No. 50-374

Subject: Licensee Event Report 2015-003-00, "Reactor Recirculation Loop Discharge Isolation Valve Vent Line Leak Due to Weld Defect"

In accordance with 10 CFR 50.73(a)(2)(ii)(A), Exelon Generation Company (EGC), LLC, is submitting Licensee Event Report Number 2015-003-00 for LaSalle County Station Unit 2.

There are no regulatory commitments in this letter. Should you have any questions concerning this report, please contact Mr. Guy V. Ford, Regulatory Assurance Manager, at (815) 415-2800.

Respectfully,

Harold T. Vinyard
Plant Manager
LaSalle County Station

Enclosure: Licensee Event Report

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – LaSalle County Station

NRC FORM 366 (02-2014) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 01/31/2017
LICENSEE EVENT REPORT (LER) (See Page 2 for required number of digits/characters for each block)	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 Recirculation Loop Discharge Isolation Valve Vent Line Leak Due to Weld Defect

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: N/A	DOCKET NUMBER: N/A
08	07	2015	2015 -	003	- 00	10	06	2015	FACILITY NAME: N/A	DOCKET NUMBER: N/A

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
3	☐ 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)
000	☐ 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)*
John Kowalski, Engineering Director	815-415-3800

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 August 7, 2015, Unit 2 was in Mode 3 for a planned maintenance outage. At 1300 hours, during the initial drywell entry, a steam leak was observed on the Reactor Recirculation (RR) system line 2RR94AB-3/4", which is upstream of valve 2B33-F080B (RR Pump Discharge Valve 2B33-F067B Inspection Port - Reactor Side Upstream Stop Valve). At 1345 hours, the leak was determined to be pressure boundary leakage. Technical Specification 3.4.5, "RCS Operational Leakage," Required Actions C.1 and C.2 were entered, which require the unit to be in Mode 3 within 12 hours and in Mode 4 in 36 hours, respectively. Unit 2 entered Mode 4 at 2209 hours on August 7, 2015.

This condition was reported (EN# 51300) on August 7, 2015 to the NRC in accordance with 10 CFR 50.72(b)(3)(ii)(A) for the pressure boundary leakage as a principal safety barrier being in a seriously degraded condition.

The cause of the steam leak was determined to be poor weld quality and vibration induced fatigue. The weld was repaired during the maintenance outage.

NARRATIVE

LaSalle County Station Unit 2 is a General Electric Company Boiling Water Reactor with 3546 Megawatts Rated Core Thermal Power.

A. CONDITION PRIOR TO EVENT:

Unit(s): 2	Event Date: August 7, 2015	Event Time: 1345 CST
Reactor Mode(s): 3	Mode(s) Name: Hot Shutdown	Power Level: 0 percent

B. DESCRIPTION OF EVENT:

On August 7, 2015, Unit 2 was in Mode 3 for a planned maintenance outage. At 1300 hours, during the initial drywell entry, a steam leak was observed on the Reactor Recirculation (RR)[AD] system line 2RR94AB-3/4", which is upstream of valve 2B33-F080B (RR Pump Discharge Valve 2B33-F067B Inspection Port - Reactor Side Upstream Stop Valve). At 1345 hours, the leak was determined to be pressure boundary leakage. Technical Specification 3.4.5, "RCS Operational Leakage," Required Actions C.1 and C.2 were entered, which require the unit to be in Mode 3 within 12 hours and in Mode 4 in 36 hours, respectively. Unit 2 entered Mode 4 at 2209 hours on August 7, 2015.

This condition was reported (EN# 51300) on August 7, 2015, to the NRC in accordance with 10 CFR 50.72(b)(3)(ii)(A) for the pressure boundary leakage as a principal safety barrier being in a seriously degraded condition.

C. CAUSE OF EVENT:

The cause for the steam leak on line 2RR94AB-3/4" was determined to be poor weld quality and vibration induced fatigue due to Reactor Recirculation system operation.

D. SAFETY ANALYSIS:

The safety significance of the event was minimal. Makeup capability was adequate to compensate for the leak. All Emergency Core Cooling Systems (ECCS) were operable and capable of fulfilling their intended safety functions during the period of excessive leakage. The event did not constitute a safety system functional failure.

E. CORRECTIVE ACTIONS:

The leak was repaired by replacing line 2RR94AB-3/4", which included removal of the inspection/vent valves and installation of a pipe cap.

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NARRATIVE

F. PREVIOUS OCCURRENCES:

LER 373-2013-005-00

On April 27, 2013, LaSalle Unit 1 was in Mode 2 (Startup) following a forced outage. At 1800 hours CDT, during a walk down of the drywell, a steam leak was observed coming from the Reactor Core Isolation Cooling Steam Supply Inboard Isolation Bypass/Warm up Valve (1E51-F076), a normally-closed, one inch, motor operated valve. The leak was determined to be on the valve bonnet extension-to-bonnet upper seal weld. At 2124 hours CDT the leak was classified as reactor coolant pressure boundary leakage, and Technical Specification (TS) 3.4.5 Condition C was entered. TS 3.4.5 Required Actions C.1 and C.2 require that the unit be in Mode 3 within 12 hours, and Mode 4 within 36 hours.

The apparent cause was a weld defect or discontinuity from the original weld construction (i.e., manufacturing, installation/construction errors, etc.) of the upper seal weld that propagated through wall as a result of system loading and conditions (i.e., high pressure steam) during normal plant operations. Corrective actions included repair of the defective seal weld area.

LER 373-2011-002-00

On February 9, 2011, LaSalle Unit 1 was in Mode 2 (Startup) following a forced outage. A steam leak was observed coming from the Reactor Core Isolation Cooling Steam Supply Inboard Isolation Bypass/Warm up Valve (1E51-F076), a normally-closed, one inch, motor operated valve. The leak was determined to be on the valve bonnet extension-to-bonnet upper seal weld. At 1804 hours, the leak was classified as pressure boundary leakage, and Technical Specification (TS) 3.4.5 Condition C was entered. TS 3.4.5 Required Action C.1 and C.2 require that the unit be in Mode 3 within 12 hours and Mode 4 within 36 hours.

The equipment apparent cause evaluation determined that the cause was a weld defect or discontinuity from the original weld construction (i.e., manufacturing, installation/construction errors, etc.) of the upper seal weld that propagated through wall as a result of system loading and conditions (i.e., high pressure steam) during normal plant operations. Corrective action included repair of the defective upper seal weld area.

LER 374-2005-002-00

On March 12, 2005, during a scheduled refueling outage on Unit 2, a pinhole leak in a Class 1 weld on the outboard Main Steam Isolation Valve drain line (2B21-F028D) was discovered during a hydrostatic test of the reactor coolant pressure boundary. The apparent cause of the leak was a weld inclusion or defect from a Class 1 weld made in 1995.

The weld was repaired, non-destructive surface examination performed, and the hydrostatic test was re-performed successfully within acceptance criteria.

G. COMPONENT FAILURE DATA:

No component failures occurred during this event.