Exelon Nuclear

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10 CFR 50.73

SVP-11-063

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

Quad Cities Nuclear Power Station, Unit 1
Renewed Facility Operating License No. DPR-29
NRC Docket No. 50-254

Subject: Licensee Event Report 254/2011-002-00, "Unit 1 Manual Reactor Scram Due to Steam Leak"

Enclosed is Licensee Event Report (LER) 254/2011-002-00, "Unit 1 Manual Reactor Scram Due to Steam Leak," for Quad Cities Nuclear Power Station, Unit 1.

This report is submitted in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50.73(a)(2)(iv)(A), which requires the reporting of any event or condition that resulted in manual or automatic actuation of the reactor protection system (RPS), including reactor scram or reactor trip.

There are no regulatory commitments contained in this letter.

Should you have any questions concerning this report, please contact Mr. W. J. Beck at (309) 227-2800.

Respectfully,

Tim Hanley

Tim Hanley
Site Vice President
Quad Cities Nuclear Power Station

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station

IE22
NRR

LICENSEE EVENT REPORT (LER)

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

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4. TITLE
Unit 1 Manual Reactor Scram Due to Steam Leak

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
06	13	2011	2011	- 002 -	00	08	12	2011	FACILITY NAME N/A	DOCKET NUMBER N/A

9. OPERATING MODE: 1

10. POWER LEVEL: 34%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 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)
☐ 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)
☐ 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

FACILITY NAME	TELEPHONE NUMBER *(Include Area Code)*
Tom Petersen – Regulatory Assurance	(309) 227-2825

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	SB	N/A	N/A	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	■ NO		N/A	N/A	N/A

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On June 13, 2011, Quad Cities Unit 1 was performing startup from refueling outage Q1R21. During power ascension while at 61% power (560 MWe) a steam leak was identified on the main steam line downstream of turbine control valve #1 (1-5699-CV1). Reactor power was reduced and the reactor was manually scrammed. The steam leak occurred at a recently repaired sensing line that had previously leaked. A forced outage was initiated to investigate and performs repairs.

The reactor and turbine responded as designed. Operators performed required actions safely and in accordance with procedures and training. There were no complications during the reactor scram and turbine trip, and all systems functioned as required.

The root cause for the manual scram due to a steam leak from a pressure sensing line repair was that the capped pipe stub repair failed due to a fatigue induced flaw that was not known to exist at the time of the repair.

Corrective actions included replacing the Unit 1 sensing line sock-o-lets with gamma plugs. Future corrective actions include replacement of the equivalent Unit 2 sensing line sock-o-lets with gamma plugs.

The safety significance of this event was minimal. This event is reportable per 10 CFR 50.73(a)(2)(iv)(A), as any event or condition that resulted in manual or automatic actuation of the reactor protection system (RPS), including reactor scram.

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PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor, 2957 Megawatts Thermal Rated Core Power

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

EVENT IDENTIFICATION

Unit 1 Manual Reactor Scram Due to Steam Leak

A. CONDITION PRIOR TO EVENT

Unit: 1	Event Date: June 13, 2011	Event Time: 0510 hours
Reactor Mode: 1	Mode Name: Power Operation	Power Level: 34%

B. DESCRIPTION OF EVENT

Scram Event

On 6/13/11 at 0445, with Unit 1 operating at 61% power (560 MWe), station personnel reported a steam leak in the Unit 1 low pressure heater bay near the main turbine [TA] control valves [FCV]. The turbine building [NM] was evacuated of non-essential personnel, and Radiation Protection personnel established access control to the turbine building. There were no indications of abnormal radiation levels or contamination.

At 0456, Operations began an emergency load drop to approximately 34% power.

At 0510, Operations personnel inserted a manual scram at 34% power.

At 0515, after the scram and the subsequent turbine trip, the leak was reported to have diminished significantly. It was determined that the leak was located downstream of the turbine control valve (CV #1) where a newly installed pipe stub and capped assembly had been installed as a repair to an earlier leak on a pressure instrument sensing line for the Pressure Transmitter [PT]-13 location. The pipe stub assembly was completely separated from the sock-o-let [PSF], but the sock-o-let remained attached to the main steam [SB] piping [PSP]. The affected equipment was related to instrumentation taps for the Unit 1 turbine upgrade project.

On 6/15/11, the leaking sensing line PT-13 location was replaced with a gamma plug [CON] (radiograph port). All of the components from the initial instrument line assembly, including the sock-o-let, were removed and replaced by a gamma plug as the corrective action to prevent recurrence of the failure.

On 6/15/11 at 1110, the generator [TB] was re-synchronized to grid.

The failure mechanism of the capped pipe stub repair was determined in a failure analysis performed by Exelon Power Labs. An existing unanticipated fatigue induced flaw in the reused sock-o-let already attached to the PT-13 location was determined to have caused the failure and subsequent steam leak. The repair to the earlier leak required a visual inspection technique for the prepared surface of the sock-o-let, and although specified per ASME Code requirements, failed to detect this flaw in the reused sock-o-let. This inspection did not detect the existing

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fatigue induced flaw since that flaw had initiated from the internal diameter and ran essentially parallel to the prepared surface. Because the potential for an existing flaw at the base of the weld was not anticipated during the repair to an earlier leak on this pressure instrument sensing line location, the design did not specify any additional special requirements in performing the repair.

Background

On 10/29/10, a permanent design change was approved to install test instrumentation (sensing line assemblies for pressure transmitters) for a thermal performance test to be performed after a planned low pressure turbine [TRB] rotor retrofit during the refueling outage for Unit 1, Q1R21 (May/June 2011).

On 5/20/11, installation was completed for two sensing line assemblies for pressure transmitters. The sensing lines were connected to the steam headers downstream of the turbine control valves. The inlets of the assemblies were connected to the main steam lines by short pipe stubs welded to new sock-o-lets that were welded to the steam line piping. Each assembly consisted of an isolation valve [ISV] that transitioned to tubing [TBG] through a coupling [CPLG], then attached to pressure transmitters and associated supports [SPT] and tubing, including the flex hoses [FCON] to the sensing line assemblies.

On 6/9/11 at 2153, the Unit 1 generator synchronized to the grid; Q1R21 ended.

On 6/10/11 at 0930, the main control room [NA] received reports of steam leaks in the Unit 1 low pressure heater bay. Walk-downs identified that the steam leaks were from failures of the newly installed sensing line assemblies installed downstream of the turbine control valves at locations: PT-13 (located downstream of CV #1), and PT-14 (located downstream of CV #2).

On 6/11/11, the next morning, maintenance personnel entered the low pressure heater bay to evaluate repair methods for the steam leaks.

On 6/11/11 at 1017, the Unit 1 turbine was tripped due to the steam leaks. Engineering personnel then developed a repair method that involved removing the failed sensing line assemblies and welding a new capped pipe stub assembly in each of the existing sock-o-lets.

On 6/12/11 at 0500, repairs were completed at the PT-13 and PT-14 sensing line locations. These repairs were performed by removing the instrument lines from the sock-o-lets that were welded to the steam line, then at each location, welding in a short section of new pipe containing a seal welded cap.

On 6/12/11 at 1300, the Unit 1 turbine generator was synchronized to the grid.

Initial Design Issues

Since the PT-13 sock-o-let had failed on 6/13/11 and resulted in the Unit 1 scram, the cause of the previous failure/leak at the PT-13 location on 6/10/11 was also reviewed; and was determined to be due to vibration induced fatigue failure. The Owner's Acceptance Review did not adequately challenge the engineering contractor design of the original instrument lines with respect to piping vibrations at the installation location.

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Outcome

The original design resulted in developing an instrument sensing line failure due to high cycle fatigue, and a flaw was created in the sock-o-let attached to PT-13 that went undetected when reused in the PT-13 repair on 6/12/11. Hence on 6/13/11 at 0445, with Unit 1 operating at 61% power, station personnel identified a steam leak that was ultimately traced to a fatigue induced flaw in the reused sock-o-let at the PT-13 instrument sensing line location. Unit 1 was subsequently scrammed at 0510.

The safety significance of this event was minimal. This event is reportable per 10 CFR 50.73(a)(2)(iv)(A), which requires the reporting of any event or condition that resulted in manual or automatic actuation of the reactor protection system (RPS), including reactor scram or reactor trip.

C. CAUSE OF EVENT

The root cause for the manual scram due to a steam leak from a pressure sensing line repair was that the capped pipe stub repair failed due to a fatigue induced flaw that was not known to exist at the time of the repair. A contributing cause was that the Owner's Acceptance Review did not adequately challenge the engineering contractor design of the original instrument lines with respect to piping vibrations at the installation location. Another contributing cause to the event was that the inspection technique selected for the repair met applicable requirements, but was not capable of detecting the fatigue flaw that was present from the earlier instrument line failure.

The extent of condition is limited to two similar installations on Unit 2. Unit 1 had another installation similar to the one that failed, however, that installation was successfully repaired during the time the failed location was repaired.

D. SAFETY ANALYSIS

The safety significance of this event was minimal. The reactor [AC] and turbine responded as designed. All safety functions performed as expected including the control rod drives [AA] and primary containment isolation [JM] functions. Operators performed required actions safely and in accordance with procedures and training. There were no complications during the reactor scram and turbine trip, and all safety systems functioned as required.

The event contributed to a minor personnel safety risk since personnel were in the general vicinity performing monitoring activities when the steam leak occurred, however, there were no personnel injuries from this event. The radiological risk from the event was minimal since there were no indications of abnormal radiation levels or contamination.

A risk assessment of the manual scram due to the steam leak was performed. A manual shutdown contributes less than 2% to the base core damage frequency. Neither the main turbine nor the turbine control valves are explicitly modeled in the probabilistic risk assessment. While there is an initiator for a main steam line break, a leak of that magnitude is far greater than a steam leak on an instrument line and is therefore not a valid comparison.

In conclusion, the steam leak on the main steam line downstream of 1-5699-CV1 was not risk significant.

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E. CORRECTIVE ACTIONS

Immediate:

1. Replaced Unit 1 sensing line sock-o-lets with gamma plugs.

Follow-up:

1. Replace Unit 2 sensing line sock-o-lets with gamma plugs during the next outage of sufficient duration.
2. Develop and perform training for Engineering personnel on the technical review requirements when performing Owner's Acceptance Reviews.
3. Determine if additional guidance on NDE or welding requirements should be provided when parts are reused after a failure.

F. PREVIOUS OCCURRENCES

The station events database, LERs, EPIX, and NPRDS were reviewed for similar events at Quad Cities Nuclear Power Station. This event was a reactor scram due to a steam leak caused by failure to adequately design for the piping vibration at the sensing line instrumentation installation location. There were no previous similar occurrences identified at Quad Cities Nuclear Power Station within the previous 10 years that involved a reactor scram due to small bore line failure from high cycle fatigue, nor were there failures similar to the capped pipe stub failure.

Based on the causes of this event and associated corrective actions, the events listed below, although similar in topic, are not considered significant station experiences that would have directly contributed to preventing this event.

- Station Events Database - Quad Cities IR 102091 (04/02/02), Unit 2 Main Steam Piping Low Point Drain Line Fails Due to Vibration Related to Extended Power Uprate (EPU) Modifications. Turbine was tripped to effect repairs but the reactor remained at low power. This event was caused by vibration resonance induced high cycle fatigue. The Root Cause of the event was determined to be development of an inadequate vibration monitoring program which was not structured to detect incipient failure conditions prior to the actual failure. Changes were made to the Exelon design input and configuration impact screening (used by both Exelon personnel and contractors in performing design work) to address high cycle fatigue including vibration monitoring. The event caused by the 04/02/02 failure is not directly applicable to this LER event since a scram was not involved. However, the corrective action for the 04/02/02 failure was not adequately implemented during the development of the design for the sensing line installation (10/29/2010) since the Exelon reviews did not adequately challenge the engineering contractor design of the original instrument lines with respect to piping vibrations at the installation location. This review issue is included in the corrective actions of this LER.
- EPIX/ NPRDS – No similar events identified for Quad Cities.
- LER – No similar events identified for Quad Cities.

G. COMPONENT FAILURE DATA

The area that failed was in a weld that attached a commercially procured non-safety grade ¾ inch 3000 psi carbon steel sock-o-let.

This event has been reported to EPIX as Failure Report No. 1117.