Exelon Generation Company, LLC
Quad Cities Nuclear Power Station
22710 206th Avenue North
Cordova, IL 61242-9740

www.exeloncorp.com

Exelon Nuclear

June 4, 2012

10 CFR 50.73

SVP-12-055

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

Quad Cities Nuclear Power Station, Unit 2
Renewed Facility Operating License No. DPR-30
NRC Docket No. 50-265

Subject: Licensee Event Report 265/2012-002-00, "Reactor Vessel Instrument Nozzle Leakage Due to Intergranular Stress Corrosion Cracking"

Enclosed is Licensee Event Report (LER) 265/2012-002-00, "Reactor Vessel Instrument Nozzle Leakage Due to Intergranular Stress Corrosion Cracking," for Quad Cities Nuclear Power Station, Unit 2.

This report is submitted in accordance with 10 CFR 50.73 (a)(2)(ii)(A) which requires the reporting of any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded.

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
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

NRC FORM 366
(10-2010)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

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4. TITLE
Reactor Vessel Instrument Nozzle Leakage Due to Intergranular Stress Corrosion Cracking

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
04	04	2012	2012	- 002 -	00	06	04	2012	FACILITY NAME N/A	DOCKET NUMBER N/A

9. OPERATING MODE: 4

10. POWER LEVEL: 000%

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	AD	NZL	C310	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 April 4, 2012, at 1716 hrs, while Unit 2 was shutdown for refueling outage Q2R21, leakage was identified exiting from a 2-inch reactor vessel instrumentation nozzle (N-11B) during a Reactor Pressure Vessel (RPV) pressure test. The leakage amount was approximately 60 drops per minute (dpm). The vessel penetration (N-11B) provides the connection point for the reference leg of the "B"-train of the Reactor Vessel Level Instrumentation System (RVLIS).

The leakage originated from the area where the nozzle penetrates the vessel wall. The nozzle is welded on the inside of the vessel, so the actual attachment weld could not be examined. The RPV pressure test was stopped and the reactor vessel depressurized to allow additional inspections and necessary repairs.

The most probable cause of the leakage was determined to be Intergranular Stress Corrosion Cracking (IGSCC) that was likely influenced by higher residual stresses that remained in the nozzle assembly following nozzle replacement in 1970, prior to the initial start-up of Unit 2.

Corrective actions included repairing the nozzle with IGSCC resistant material, and obtaining approval of a Relief Request from the NRC prior to startup to allow the flaw to remain for one operating cycle. Future corrective actions include inspecting other similar RPV nozzles, and performing a specialized flaw evaluation to support safe operation for continued operating cycles.

The safety significance of this event was minimal given the leakage was very small, was found while the reactor was shutdown, and if leaked during plant operation, did not exceed Technical Specification (TS) leakage limits for unidentified drywell leakage. Given the impact on the reactor vessel pressure boundary, this report is submitted in accordance with the requirements of 10 CFR 50.73 (a)(2)(ii)(A), which requires the reporting of any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded.

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NARRATIVE

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

Reactor Vessel Instrument Nozzle Leakage Due to Intergranular Stress Corrosion Cracking

A. CONDITION PRIOR TO EVENT

Unit: 2	Event Date: April 4, 2012	Event Time: 1716 hours
Reactor Mode: 4	Mode Name: Power Operation	Power Level: 000%

B. DESCRIPTION OF EVENT

On April 4, 2012, at 1716 hrs, while Unit 2 was in cold shutdown for refueling outage Q2R21, a 60 drops per minute (dpm) leak was observed at the insulation [ISL] surrounding the reactor vessel [AD] penetration associated with the N-11B instrument nozzle [NZL]. The leak was discovered during the Reactor Pressure Vessel (RPV) Class 1 pressure boundary system leakage test that is performed each refueling outage in accordance with ASME Section XI, IWB-2500, and station procedures. During the test, the vessel was water-solid and pressurized to 1005 psig.

Upon discovering the leak, the qualified VT-2 inspector requested the insulation be removed to allow a more detailed inspection. The leak was confirmed to be originating in the RPV annulus region between the N-11B nozzle and the RPV wall. Based on the design, any leakage into the annulus region between the nozzle body and the RPV wall would likely be associated with a through-wall failure of the nozzle assembly, either at the nozzle body or at the RPV attachment weld. Since no additional RPV leaks were found, the RPV pressure test was stopped and the reactor vessel was depressurized.

On April 5, 2012, an extent of condition review was performed by removing insulation on similar nozzles and re-pressurizing the RPV to allow additional inspections. Since the RPV was confirmed to have no additional leakage, the RPV was depressurized to allow the necessary repairs.

With the leakage confirmed to be originating from the N-11B nozzle assembly, a repair team was mobilized, consisting of station, corporate, and industry experts. The station determined that the appropriate repair method would be a half nozzle repair which consisted of replacing the outer portion of the existing nozzle with a new Inconel Alloy 690 nozzle that would be welded to the outside of the RPV versus the inside. The original RPV attachment weld and a remnant section of the original nozzle in which the through-wall failure occurred were effectively abandoned in place.

The repair was completed on April 15, 2012 and the post-repair leakage test immediately followed, which validated the integrity of the repair and the RPV Class 1 pressure boundary.

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On April 4, 2012, ENS Notification # 47806 was made in accordance with 10 CFR 50.72 (b)(3)(ii)(A). This event has been classified as a Maintenance Rule Functional Failure for the RPV.

Given the impact on the reactor vessel pressure boundary, this report is submitted in accordance with the requirements of 10 CFR 50.73 (a)(2)(ii)(A), which requires the reporting of any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded.

C. CAUSE OF EVENT

The most probable root cause of the N-11B instrument nozzle weld crack and leak was determined to be Intergranular Stress Corrosion Cracking (IGSCC). Contributing to crack development were higher residual stresses that remained in the nozzle assembly following nozzle replacement in 1970, prior to the initial start-up of Unit 2. A most probable cause was determined because the repair resulted in requiring the degraded nozzle remnant to remain installed in the RPV penetration, hence, the specific characterization of the flaw causing the leak could not be determined.

The root cause investigation identified that the N-11B was originally replaced during initial RPV construction in 1970. Since the 1970 replacement nozzle was installed after the original RPV heat treatment, no additional post weld heat treatment was pursued at that time, and as such, both the N-11B nozzle body and the attachment weld likely had higher residual stresses than other similar nozzles installed during initial RPV construction which were heat treated. Based on the higher residual stresses that likely existed, and since the nozzle and nozzle attachment weld were both fabricated from susceptible materials (Inconel Alloy 600 for the nozzle, and Inconel Alloy 182 for the weld metal), it was concluded that the observed leak was most likely caused by IGSCC. This cause is supported by previous industry events (primarily Pressurized Water Reactors (PWRs)) where similar cracking has occurred in the same Inconel alloy materials.

The extent of condition of this leakage event is limited to the N-11B nozzle. Attached to each Unit RPV are 36 nozzles, 177 Control Rod Drive (CRD) [AA] penetrations, and 53 flux monitor penetrations. Each of the RPV nozzles and penetrations are visually inspected for leakage during the RPV Class 1 system pressure test that occurs each refuel outage. Based on the most recent system pressure test on both units (Q1R21 and Q2R21) there were no other RPV penetrations identified exhibiting evidence of similar through-wall failures. Additionally, following the N-11B leak during Q2R21, the insulation was removed from similar N-11 and N-12 nozzles, and no leakage was observed.

The extent of cause is IGSCC, and is likely limited to the N-11B nozzle, since the N-11B nozzle is unique in that following the replacement of the nozzle in 1970 no post weld heat treatment was performed. All other nozzles installed in the Unit 1 and Unit 2 RPVs were heat treated along with the vessel, and as such would likely have lower residual stresses. This assumption is supported by industry experience given there have been no similar RPV nozzle failures at other Boiling Water Reactors (BWRs). Previous industry events are either specific to PWRs, are associated with external weld failures, or are non-IGSCC related. In addition, it is possible that as a result of the N-11B nozzle replacement in 1970, IGSCC was able to initiate earlier, or propagate more quickly, due to the shallower than normal J-groove weld depth along the nozzle to weld interface. Based on the possibility of IGSCC as a potential cause, the extent of cause will be expanded to include inspections of other nozzles of similar design/material on Units 1 and 2.

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D. SAFETY ANALYSIS

System Design

The purpose of the RPV and its appurtenances, such as the N-11B nozzle, is to retain the reactor core coolant-moderator within the RPV and to serve as a high integrity barrier against leakage steam used for power production and leakage of radioactive materials to the drywell [NH] during all modes of plant operation. The N-11B nozzle serves as the connection point for the 2-inch diameter "reference" leg associated with the "B" train of RVLIS. The purpose of RVLIS is to provide initiation signals to the Emergency Core Cooling System (ECCS) [JE] based on reactor water level, and also provide initiation signals to trip functions in the Anticipated Transient Without Scram (ATWS) system.

The N-11B nozzle is located on the RPV, 35 inches above normal reactor water level (+30 inches), and during plant operation is exposed to the steam section of the RPV. The N-12B nozzle, which is of similar construction as the original N-11B nozzle, is located on the RPV below the normal reactor water level, which serves as the connection point for the 2-inch "variable" leg for the B-train of RVLIS, and during plant operation is exposed to the water section of the RPV.

Safety Impact

The safety significance of this event was minimal given the leakage was very small, was found while the reactor was shutdown for refueling, and if leaked during plant operation, did not exceed Technical Specification (TS) leakage limits for unidentified drywell leakage.

During normal power operation, this nozzle is exposed to a reactor steam environment, and not the water solid conditions of the Class 1 pressure boundary system leakage test that is performed each refueling outage in accordance with ASME Section XI, IWB-2500.

Had a worst case scenario occurred, in which the 2-inch diameter N-11B connection line to RVLIS failed completely such that there was a 2-inch diameter opening in the RPV, the consequences would have been minimal. The 2-inch diameter line break (0.02 sq ft in area) would be bounded by the line breaks of up to 0.12 sq ft in area, as discussed in Chapter 15 of the Updated Failure Safety Analysis Report (UFSAR) which provides that the High Pressure Coolant Injection (HPCI) [BJ] system could supply sufficient coolant to depressurize the vessel and cool the core for line breaks of up to 0.12 sq ft in area.

Additionally, the leakage through the N-11B nozzle was small and had no impact on the ability of the RVLIS system to monitor RPV level. Since the leak was located on the reference leg of the RVLIS system, where the reference leg is maintained full by backfilling from the CRD system, the leak at the nozzle would not impact the RVLIS reference leg unless a complete failure of the 2-inch line had occurred. Furthermore, even if a complete line failure had occurred, the redundant "A" train of RVLIS would have provided the necessary level indication for safety system actuations.

Risk Insights

Prior to and during this event, the capability of RVLIS was not lost since there were no identified RVLIS leaks during the prior operating cycle. This condition did not create any actual plant or safety consequences since the Unit was not in an accident or transient condition requiring use of RVLIS initiation signals to ECCS or ATWS during this period of time or prior operating cycle. The nozzle leak was discovered during system pressure testing which is designed to identify any RPV leakage before the unit is placed into power operation.

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There is no makeup rate required to mitigate an RPV leakage of 60 dpm. N-11B is a 2 inch penetration. Even a complete failure of the penetration would only result in Small LOCA (SLOCA). The break diameter for a SLOCA is between 0.5 and 2 inches and would be well within the capability of HPCI or one RHR pump. 60 dpm is negligible compared to any SLOCA.

The N-11B instrument nozzle, which is attached to the wall of the pressure vessel, provides the connection point for the reference leg of the "B" train of the RVLIS. Steam from the RPV travels through piping to the condensing pot and maintains a constant level in the reference leg. The 60 dpm when converted to steam would become a very small flow of leakage steam, and would not result in diverting any measurable steam flow needed in the piping for the reference leg condensing pot.

Based on the above, the impact on the RPV and RVLIS of a 60 dpm leak is negligible, therefore, considering the impact of this condition on the Plant Probabilistic Risk Assessment (PRA), the change in Core Damage Frequency (CDF) due to the observed leakage will be less than 1.0E-06/yr. In conclusion, the overall safety significance and impact on risk of this event was minimal.

E. CORRECTIVE ACTIONS

Immediate:

1. A Relief Request was submitted to the NRC since the defect in N-11B nozzle would not be removed, and since a qualified technique to perform volumetric non-destructive examination (NDE) of the partial penetration weld for characterizing the flaw and determining flaw growth in the specific configuration does not exist. The NRC subsequently approved the Relief Request prior to Unit startup.
2. The degraded N-11B nozzle was partially removed to allow a half-nozzle repair to be installed. The half-nozzle design was welded to the outer vessel wall, effectively moving the pressure boundary from the degraded nozzle to the new externally attached nozzle. The new half-nozzle assembly was fabricated from Inconel Alloy 690 and attached to the RPV with Inconel Alloy 52M weld metal. Both materials are considered IGSCC resistant.
3. A failure assessment and flaw evaluation were completed prior to startup to demonstrate the acceptability of leaving the original partial penetration attachment weld, with a maximum postulated flaw, in place for one operating cycle.

Follow-up:

1. N-11A/B and N-12A/B nozzles will be inspected from inside the RPV using enhanced VT-1 methods during the next U1 and U2 refuel outage. Additional corrective actions will be identified as necessary as a result of these inspections.
2. Perform an Elastic Plastic Fracture Mechanics (EPFM) flaw evaluation for the N-11B remnant J-groove weld to support continued operation beyond the next refuel outage for Unit 2.

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 RPV nozzle attachment weld leak associated with IGSCC. There were no previous similar occurrences identified at Quad Cities Nuclear Power Station that involved an event of this type.

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G. COMPONENT FAILURE DATA

The failed component was the N-11B instrument nozzle which was fabricated by Chicago Bridge and Iron in 1970 after the original nozzle was replaced due to damage that occurred after initial construction of the RPV. The original construction nozzle, along with the other nozzles installed in Unit 2, were fabricated by Babcock and Wilcox.

The N-11B nozzle is a 2 inch diameter RPV nozzle located in the steam section of the vessel approximately 35 inches above normal reactor water level. Materials of construction are, Inconel Alloy 600 for the nozzle, and Inconel Alloy 182 for the weld metal.

This event has been reported to EPIX as Failure Report No. 1152.