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Exelon Nuclear

10 CFR 50.73

May 18, 2011
Byron Ltr 2011-0082
File # 1.10.0101

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

Byron Station, Unit 1
Facility Operating License No. NPF-37
NRC Docket No. STN 50-454

Subject: Licensee Event Report 2011-002-00, "Unit 1 Reactor Pressure Vessel Head Penetration Nozzle Weld Flaws Attributed to Primary Water Stress Corrosion Cracking"

The enclosed Licensee Event Report (LER) is being submitted in accordance with 10 CFR 50.73, "Licensee event report system." The LER involves the identification of weld flaws on the Unit 1 reactor head that required repairs prior to returning the reactor head to service.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this submittal, please contact Mr. David Gudger, Regulatory Assurance Manager, at (815) 406-2800.

Respectfully,

Timothy J. Tulon
Site Vice President
Byron Station

Enclosure: LER Number 2011-002-00

NRC FORM 366 (10-2010)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013

LICENSEE EVENT REPORT (LER)

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

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4. TITLE
Unit 1 Reactor Pressure Vessel Head Penetration Nozzle Weld Flaws Attributed to Primary Water 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: NA	DOCKET NUMBER: NA
03	19	2011	2011	- 002 -	00	05	18	2011	FACILITY NAME: NA	DOCKET NUMBER: NA

9. OPERATING MODE: 6

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)*
David T. Gudger, Regulatory Assurance Manager	(815) 406-2800

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)*

During the Byron Station Unit 1 spring 2011 refueling outage, Exelon Generation Company, LLC performed volumetric and surface examinations of the reactor Vessel Head Penetration (VHP) nozzles in accordance with 10 CFR 50.55a(g)(6)(ii)(D). Examination results identified flaws for VHP nozzles 64 and 76, as well as 31 and 43 that did not meet the applicable acceptance criteria and therefore required repair prior to returning the reactor head to service. Some of the flaws in VHP nozzles 31 and 43 were considered in the Reactor Coolant system pressure boundary region. The flaws were identified and repaired prior to through wall leakage occurring using an alternate weld overlay method in accordance with NRC approved Westinghouse WCAP-15987-P, Revision 2-P-A. The cause of the recordable indications is attributed to Primary Water Stress Corrosion Cracking (PWSCC). The frequency of Unit 1 reactor head penetrations examinations will be accelerated to each refuel outage and future preemptive repairs on VHPs nozzles that may be more susceptible to PWSCC will be evaluated. This condition had no actual safety consequences.

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NARRATIVE

A. Plant Operating Conditions Prior to the Event

Unit 1 was in Mode 6 – Refueling
Reactor Coolant (RC) [AB] system was at ambient temperature and depressurized.

No structures, systems, or components were inoperable at the start of this event that contributed to the initiation or mitigation of this event.

B. Description of the Event

During the Byron Station Unit 1 spring 2011 refueling outage, Exelon Generation Company, LLC (EGC) performed volumetric and surface examinations of the Byron Station Unit 1 reactor Vessel Head Penetration (VHP) nozzles in accordance with 10 CFR 50.55a(g)(6)(ii)(D). Examination results identified flaws for VHP nozzles 64 and 76, as well as 31 and 43 that did not meet the applicable acceptance criteria and therefore required repair prior to returning the reactor head to service. Some of the flaws in VHP nozzles 31 and 43 were considered in the Reactor Coolant system pressure boundary region.

VHP nozzle 31 had two recordable flaws: one linear flaw 0.600 inches in length with a through-wall depth of 0.169 inches, and a rounded flaw 0.10 inches in width and 0.20 inches into the J-groove weld (i.e. RC system pressure boundary region).

VHP 43 nozzle had four recordable flaws: one linear flaw 0.840 inches in length with a through-wall depth of 0.294 inches, a rounded flaw of 0.15 inches in width at the junction of the J-groove weld to penetration tube, one linear flaw of 0.10 inches in length 0.10 inches into the J-groove weld, and one flaw 0.20 inches in size 0.25 inches into the J-groove weld (i.e. RC system pressure boundary region).

Additionally:
VHP 64 nozzle had one axially oriented recordable flaw 0.52 inches in length with a through-wall depth of 0.177 inches, and an axial flaw of 0.15 inches in length, which did not meet acceptance criteria.

VHP 76 nozzle has two recordable flaws: one recordable flaw 0.52 inches (axially oriented) with a through-wall depth of 0.194 inches and a recordable flaw 0.84 inches (circumferentially oriented) in length and a through-wall depth of 0.187 inches, which did not meet acceptance criteria.

A full bare metal visual reactor head inspection did not identify any indications of through wall leakage. The flaws were repaired using an alternate weld overlay method in accordance with NRC approved Westinghouse WCAP-15987-P, Revision 2-P-A.

This condition is reportable to the NRC in accordance with 10 CFR 50.73 (a)(2)(ii), as a condition that resulted in a principle safety barrier being seriously degraded.

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NARRATIVE

C. Cause/s of the Event

The cause of the recordable indications is attributed to Primary Water Stress Corrosion Cracking (PWSCC).

D. Safety Significance

This condition had no actual safety consequences. The flaws were identified in a timely manner and repaired prior to through wall leakage occurring. The flaws were identified as part of required periodic inspections. Potentially, if a flaw remained undetected it could have over time propagated through the alloy 600 weld material to form a leak path through the RC system boundary.

E. Corrective Actions

The flaws were repaired using an alternate weld overlay method in accordance with NRC approved Westinghouse WCAP-15987-P, Revision 2-P-A

The frequency of Unit 1 reactor head penetration nozzle examinations will be accelerated to each refuel outage.

Future preemptive repairs on VHPs nozzles that may be more susceptible to PWSCC will be evaluated. This condition had minimal actual safety consequences.

F. Previous Occurrences

Byron Station Unit 2 LER 455-20070001-00, "Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzle Weld Indication Due to an Initial Construction Weld Defect Allowing the Initiation of Primary Water Stress Corrosion Cracking".

The identification of PWSCC was unexpected on the Unit 1 reactor head. The head was classified as having low susceptibility to PWSCC.