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JUN 0 7 2018

10 CFR 50.73

Serial: HNP-18-052

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

Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400/Renewed License No. NPF-63

Subject: Licensee Event Report 2018-002-00

Ladies and Gentlemen:

Duke Energy Progress, LLC, submits the enclosed Licensee Event Report 2018-002-00 in accordance with 10 CFR 50.73 for Shearon Harris Nuclear Power Plant, Unit 1. This report details two rejected indications associated with a control rod drive mechanism nozzle penetration, identified during the reactor vessel closure head inspection completed during the last refueling outage. All rejected indications have been restored to code compliance.

This document contains no regulatory commitments. Please refer any questions regarding this submittal to Jeffrey Robertson, Manager – Regulatory Affairs, at (919) 362-3137.

Sincerely,

Tanya M. Hamilton

Tanya M. Hamilton

Enclosure: Licensee Event Report 2018-002-00

cc: J. Zeiler, NRC Senior Resident Inspector, HNP
M. Barillas, NRC Project Manager, HNP
C. Haney, NRC Regional Administrator, Region II

NRC FORM 366
(04-2018)

U.S. NUCLEAR REGULATORY COMMISSION

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/)

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 03/31/2020

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 Information Services Branch (T-2 F43), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by 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.

1. Facility Name	2. Docket Number	3. Page
Shearon Harris Nuclear Power Plant – Unit 1	05000 400	1 OF 3

4. Title

Reactor Pressure Vessel Closure Head Penetration Nozzle Indications 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	Docket Number
										05000
04	11	2018	2018	- 002	- 00	06	07	2018	Facility Name	Docket Number 05000

9. Operating Mode	11. This Report is Submitted Pursuant to the Requirements of 10 CFR §: *(Check all that apply)*			
6	☐ 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)
000	☐ 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)
	☐ 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)
Jeffrey Robertson, Manager – Regulatory Affairs	(919) 362-3137

13. Complete One Line for each Component Failure Described in this Report

Cause	System	Component	Manufacturer	Reportable to ICES		Cause	System	Component	Manufacturer	Reportable to ICES
B	AB	RPV	CB&I	Yes						

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 14 single-spaced typewritten lines)

On April 11, 2018, while the Shearon Harris Nuclear Power Plant was shut down for a scheduled refueling outage, the reactor vessel head was being examined in accordance with the Inservice Inspection Program. Ultrasonic examinations identified two rejectable indications in head penetration nozzle 33.

The two indications associated with nozzle 33 were attributed to primary water stress corrosion cracking (PWSCC), with the indications having an axial extent of 0.223 in. and 0.260 in. with a through-wall extent of 0.100 in. (16%) and 0.095 in. (15%), respectively. The indications were located on the downhill side of the nozzle at the lower toe of the J-groove weld.

A leak path assessment and a bare metal visual examination of the reactor vessel head was completed, with no leakage identified. The two PWSCC indications were repaired using the inside diameter temper bead weld method. The repair was completed prior to exiting the refueling outage.

LICENSEE EVENT REPORT (LER) CONTINUATION SHEET

(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/)

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Shearon Harris Nuclear Power Plant – Unit 1	05000-400	2018	- 002	- 00

NARRATIVE

Note: Energy Industry Identification System (EIIS) codes are identified in the text within brackets [ ].

A. Background

Event Date: April 11, 2018 Mode: 6 Reactor Power: 0 percent

On April 11, 2018, the Shearon Harris Nuclear Power Plant (SHNPP) was in a scheduled refueling outage (RFO) for cycle 21 (RFO-21). During the outage, the inspection of the reactor vessel closure head (RVCH) [RPV] control rod drive mechanisms (CRDM) [DRIV] penetration nozzles [NZL] occurred. The RVCH was manufactured by Chicago Bridge and Iron, Serial Number T40.

No Structures, Systems or Components (SSCs) were inoperable at the start of this event that contributed to the event. No change in plant mode or in reactor power occurred as a result of this event.

This condition is reportable in accordance with 10 CFR 50.73(a)(2)(ii)(A), as an event or condition that resulted in the condition of the nuclear power plant, including its principal barriers, being degraded.

B. Event Description

Nondestructive examinations (NDE) identified two rejectable indications in penetration nozzle 33. The two indications associated with nozzle 33 were identified using ultrasonic examination and were attributed to primary water stress corrosion cracking (PWSCC), with the indications having an axial extent of 0.223 in. and 0.260 in. with a through-wall extent of 0.100 in. (16%) and 0.095 in. (15%), respectively. The indications were located on the downhill side of the nozzle at the lower toe of the J-groove weld.

The 2007 Edition with 2008 Addenda of the ASME Code Section XI Acceptance Criteria in Table IWB-3663-1 General Note (a) states, "Linear surface flaws of any size in the partial penetration nozzle to vessel (J-groove) welds are not acceptable." A rejectable flaw in a partial penetration nozzle weld in the RVCH does not meet the acceptance standards referenced per ASME Code Case N-729-4. Thus, the rejectable PWSCC indications required repair and were reportable as a degraded barrier.

LICENSEE EVENT REPORT (LER) CONTINUATION SHEET

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Shearon Harris Nuclear Power Plant – Unit 1	05000- 400	2018	- 002	- 00

NARRATIVE

C. Causal Factors

The cause of the indications in nozzle 33 was attributed to PWSCC, which occurs under conditions of high tensile stresses (operating and/or residual), conducive environment (temperature and chemistry), and susceptible material. The CRDM nozzles in the SHNPP RVCH were originally constructed from Alloy 600 tubing and Alloy 82/182 weld metal. There is widespread industry operating experience that documents PWSCC of Alloy 600 dissimilar metal weld configurations.

D. Corrective Actions

The two PWSCC indications were repaired using the inside diameter temper bead weld method. All RVCH CRDM nozzles were inspected, as required by ASME Code Case N-729-4, due to previously identified PWSCC indications.

E. Safety Analysis

After PWSCC was identified in RFO-17, inspections of the RVCH are required every refueling outage in accordance with ASME Code Case N-729-4, as conditioned by 10 CFR 50.55a. These inspections include NDE for all RVCH penetrations to identify indications, and are supplemented by bare metal visual examinations of the RVCH. If rejectable indications are found, repairs are required in accordance with both ASME Code and with relief requests submitted to the NRC on a case-by-case basis. This ensures indications are identified and repaired before there are any significant impacts on the integrity of the RVCH.

The In-Service Inspection examinations performed on the RVCH did not reveal any through-wall leakage. An ultrasonic leak path assessment and a bare metal visual examination of the reactor vessel head were completed, with no leakage identified. There was not a breach in the fission product barrier, and the structural integrity of the reactor vessel was not significantly compromised. Therefore, there was no significant impact to the health and safety of the public.

F. Additional Information

PWSCC has previously been detected in nozzles 5, 17, 38, 49, and 63 (RFO-17), 37 (RFO-18), 14, 18, and 23 (RFO-19), and 30, 40, and 51 (RFO-20). LERs 2013-001-00, 2013-003-00, 2015-003-00, and 2016-006-00 all document previous experience with indications in the RVCH CRDM penetration nozzles.