10CFR50.73

Virginia Electric and Power Company
North Anna Power Station
P. O. Box 402
Mineral, Virginia 23117

May 18, 2012

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

Serial No.: 12-264
NAPS: MPW
Docket No.: 50-338
License No.: NPF-4

Dear Sirs:

Pursuant to 10CFR50.73, Virginia Electric and Power Company hereby submit the following Licensee Event Report applicable to North Anna Power Station Unit 1.

Report No. 50-338/2012-001-00

This report has been reviewed by the Facility Safety Review Committee and will be forwarded to the Management Safety Review Committee for its review.

Sincerely,

FOR

Gerald T. Bischof
Site Vice President
North Anna Power Station

Enclosure

Commitments contained in this letter: None

cc: United States Nuclear Regulatory Commission
Region II
Marquis One Tower
245 Peachtree Center Ave., NE, Suite 1200
Atlanta, Georgia 30303-1257

NRC Senior Resident Inspector
North Anna Power Station

IE22
NRR

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)	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 Section (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.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
North Anna Power Station , Unit 1	05000 338	1 OF 4

4. TITLE
Degraded Reactor Coolant System Piping Due 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	DOCUMENT NUMBER 05000
03	24	2012	2012	-- 001 --	00	05	18	2012	FACILITY NAME	DOCUMENT 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)(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)
	☐ 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
Refueling	☐ 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*)
F. Mladen, Director Station Safety and Licensing	(540) 894-2108

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 March 24, 2012, at 1855 hours, with Unit 1 in Mode 6, refueling, two through-wall cracks were identified after machining the Unit 1 'B' Steam Generator (SG) Hot Leg Nozzle. The defects were identified after machining approximately 0.7" to 1.1" of material from the Nozzle-to-Safe End weld in preparation of full structural weld overlays (FSWOL). Work was being performed to support Relief Request N1-14-CMP-001 which permitted the application of FSWOL to mitigate the potential for primary water stress corrosion cracking (PWSCC) susceptibility at North Anna Unit 1. The cracking was determined to be caused by PWSCC and confirmed to be fully contained within the Dissimilar Metal (DM) Alloy 600 weld and butter. Subsequently, the 'B' Steam Generator and associated Reactor Coolant System piping were drained and the through-wall cracks were seal welded under the provisions of Relief Request N1-14-CMP-001. This event is reportable per 10 CFR 50.73(a)(2)(ii)(A) for a condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded. On March 25, 2012, at 0144, a non-emergency 8-hour report was made to the NRC Operations Center, in accordance with 10CFR50.72(b)(3)(ii)(A) for the same condition. The health and safety of the public were not affected by the event.

NARRATIVE

1.0 DESCRIPTION OF THE EVENT

Dominion submitted Relief Request N1-14-CMP-001 to mitigate the primary water stress corrosion cracking (PWSCC) susceptibility of North Anna, Unit 1, Steam Generator (SG) (EIIS System-AB, Component-SG) hot leg nozzle Dissimilar Metal (DM) welds. A design change was developed which installs full structural weld overlays (FSWOL) on all three nozzles. The weld overlays were being performed to mitigate the consequences for PWSCC in the nozzle-to-safe-end dissimilar metal (DM) welds and to strengthen the DM weld region. As a part of the overlay process, the outer surface of the nozzle (carbon steel, DM weld region, and safe end) was being machined to provide a flat surface for welding and to provide a configuration that would better facilitate future inspections.

On 3/24/2012, the two through-wall cracks were observed after machining approximately 0.7" to 1.1" of material from the Nozzle-to-Safe End weld in preparation of the FSWOL on the 'B' SG. Machining of the nozzle (EIIS Component-NZL) was necessary in order to eliminate the taper which would interfere with the final volumetric ultrasonic (UT) examinations. The material thickness of the DM weld after final machining was approximately 4.1 inches. Because the leakage from the DM weld was not noticed until completion of the machining process (due to masking effects of the machining oils and lubricants) the exact through-wall lengths of the cracks are unknown. However, based on visual and surface examinations, the two through-wall cracks were axially oriented and fully contained within the Alloy 82/182 DM weld. Also, it was concluded that the cracks did not propagate through the entire weld based on the as-found condition of the pipe (i.e., no boric acid). Further investigation, using conventional manual UT examination methods, located the two through-wall cracks as well as three additional indications with varying degrees of propagation. No unacceptable indications were identified in the 'A' or 'C' Hot Leg nozzle DM welds (post machining) using conventional UT methods.

After the discovery of the five indications in the 'B' SG DM weld additional UT examinations were performed using manual phased array equipment. This allowed for further characterization of the indications such as orientation and shape. The data collected from the phased array UT examination was provided to Electric Power Research Institute (EPRI) and they compiled images of the sector scans from each of the five indication areas. The phased array scans revealed a stacked pattern (i.e. highly branched), axial cracking which is a typical response for PWSCC. The cracking was confirmed to be fully contained within the DM weld and butter.

In addition, EPRI performed an independent evaluation of the fully automated phased array UT examinations performed prior to the machining operations which were looking for circumferential indications. There was no evidence of circumferential cracking in any of the Unit 1 SG Hot Leg nozzles.

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

A comparison is made between a bounding axial flaw length and the allowable flaw length limit given in Appendix C to ASME Section XI to demonstrate stability of a bounding through-wall axial flaw. Assuming PWSCC as the only crack growth mechanism, this evaluation demonstrates that the existence of a bounding axial through-wall flaw spanning the entire length of the PWSCC susceptible material (~2.0 inches) could not have reached an unstable axial length during previous operations.

The applied hoop stress is less than the projected allowable stress in ASME Section XI requirements considering a ligament of 0.57 inches (machined depth of 4.1 inches). However, a postulated flaw depth of 4.1" (depth-to-thickness ratio a/t=0.88) exceeds the limits of applicability in the code of a/t=0.75. And since the true depths of the uncovered axial flaws prior to machining are unknown, it is reasonable to assume they extended to a higher a/t ratio. Therefore, the as-found flaws, while potentially within acceptable stress limits, did not meet the ASME Section XI code requirements.

Since pressure loading is the only stressor considered for stability of a component containing axial flaws, the only postulated events that would have resulted in any additional challenges to the structural integrity of the nozzle, other than the normal operation experienced, would have been pressure transients. For the Reactor Coolant System (RCS), pressure transients are not significant relative to the normal operating pressure. Therefore, there would have been no significant challenges to the structural integrity of the nozzle connection related to the discovered axial flaws during previous operation.

The assessments provide reasonable assurance that gross structural failure of the “B” hot leg inlet nozzle would not have occurred during previous operations for any postulated design basis events including Design Basis Earthquake. This is supported by using ASME Section XI, Appendix C methods for evaluating hoop stress and stability of a bounding axial through-wall flaw. Had PWSCC continued to the point of an axial through-wall leak, the consequences of that outcome would be minimized given that the overall nozzle connection would not have been significantly challenged. As such, the event posed no significant safety implications and the health and safety of the public were not affected by the event.

3.0 CAUSE

The Direct Cause determined the leakage developed due to PWSCC in a susceptible material (i.e. Alloy 82/182). After review of the fabrication records, it is apparent that the primary water stress corrosion cracks were caused by extensive ID weld repairs performed only on the ‘B’ Hot Leg nozzle. There were no ID weld repairs performed on the ‘A’ or ‘C’ Hot Leg nozzles. Repairs performed during SG fabrication to the ‘B’ Hot Leg DM weld were made with a susceptible material (i.e. Alloy 182) in a sequence which would result in the formation of adverse residual tensile stresses. A combination of tensile stresses and low

NARRATIVE

chromium weld material on the ID of the DM weld in a high temperature, primary water environment directly led to the formation of stress corrosion cracks. Because NDE has characterized the cracks as being branched and fully contained within the DM weld; the identified cracks are the result of PWSCC and not some other mechanical or thermal loading mechanisms.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

After confirmation that the water which was slowly accumulating along the length of the indication was reactor coolant, the 'B' SG was drained and the through-wall cracks were seal welded under the provisions of Relief Request N1-14-CMP-001.

5.0 ADDITIONAL CORRECTIVE ACTIONS

Extent of Condition volumetric examinations were performed on the Unit 1 SG Cold Leg nozzles during the Spring 2012 refueling outage per the Section XI program. No flaws were identified. The FSWOLs were completed on all three Unit 1 SG Hot Leg nozzles with satisfactory tests results.

Unit 2 SG hot and cold leg nozzles will have UT examinations performed during the 2013 refueling outage. Note that Unit 2 SG nozzles contain an alloy 52/152 weld inlay that significantly reduces susceptibility to PWSCC.

6.0 ACTIONS TO PREVENT RECURRENCE

The potential that PWSCC could be present in other systems/components is limited to whether Alloy 600 material was used. Continued inspections and testing performed by the ISI program will identify any future cases of defects propagating in similar components.

7.0 SIMILAR EVENTS

LER 50-339/02-001-00 documents the Unit 2 nozzle through-wall leakage of three reactor vessel head penetrations.

LER 50-339/01-003-00 and Supplemental LER 50-339/01-003-01 documents the Unit 2 nozzle through-wall leakage of three reactor vessel head penetrations.

8.0 ADDITIONAL INFORMATION

Unit 2 was operating in Mode 1, 100 percent power on March 24, 2012.