Progress Energy

May 18, 2006

SERIAL: BSEP 06-0055

10 CFR 50.73

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

Subject: Brunswick Steam Electric Plant, Unit No. 1
Docket No. 50-325/License No. DPR-71
Licensee Event Report 1-2006-002

Ladies and Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.73, Carolina Power & Light Company, now doing business as Progress Energy Carolinas, Inc., submits the enclosed Licensee Event Report.

Please refer any questions regarding this submittal to Mr. Randy C. Ivey, Manager – Support Services, at (910) 457-2447.

Sincerely,

B C W

B. C. Waldrep
Plant General Manager
Brunswick Steam Electric Plant

MAT/mat

Enclosure:

Licensee Event Report

Progress Energy Carolinas, Inc.
Brunswick Nuclear Plant
PO Box 10429
Southport, NC 28461

IE22

cc (with enclosure):

U. S. Nuclear Regulatory Commission, Region II
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Ms. Jo A. Sanford
Chair - North Carolina Utilities Commission
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NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@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
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4. TITLE

Cracking Found In B Loop Core Spray Header Piping

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
03	21	2006	2006	-- 002 --	00	05	18	2006	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more)			
5	☐ 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)
000	☐ 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 Specify in Abstract below or in NRC Form 366A
	☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Mark A. Turkal, Lead Engineer - Licensing	(910) 457-3066

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	MO	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

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

On March 15, 2006, as a result of visual examinations of reactor vessel internals during the Unit 1 refuel outage (i.e., B116R1), cracking was identified on a Core Spray (CS) system header piping weld (i.e., weld P3c-270). Subsequently, on March 21, 2006, Ultrasonic Test (UT) examinations of the entire weld were performed to determine the extent of the indication on the portions of the weld inaccessible to visual examinations and to more accurately size the indications. The UT examination results demonstrated that the as-found condition of the weld was unacceptable for operation without repair. This condition is being reported in accordance with 10 CFR 50.73(a)(2)(ii)(B), as an unanalyzed condition that significantly degraded plant safety and 10 CFR 50.73(a)(2)(i)(B) as operation prohibited by the plant's Technical Specifications.

The root cause of this event is intergranular stress corrosion cracking (IGSCC) in the heat affected zone (HAZ) of weld P3c-270.

On April 1, 2006, weld P3c-270 was repaired by installation of a clamp designed to structurally replace the weld. The clamp was designed in accordance with Boiling Water Reactor Vessel and Internals Project (BWRVIP) Report BWRVIP-19-A, "BWR Vessel and Internals Project, Internal Core Spray Piping and Sparger Repair Design Criteria," which includes BWRVIP-84, "BWR Vessel and Internals Project, Guidelines for Selection and Use of Materials for Repairs to BWR Internal Components."

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

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

INTRODUCTION

On March 15, 2006, as a result of visual examinations of reactor vessel internals during the Unit 1 refuel outage (i.e., B116R1), cracking was identified on a Core Spray (CS) system [BM] header piping weld (i.e., weld P3c-270). Subsequently, on March 21, 2006, Ultrasonic Test (UT) examinations of the entire weld were performed to determine the extent of the indication on the portions of the weld inaccessible to visual examinations and to more accurately size the indications. The UT examination results demonstrated that the as-found condition of the weld was unacceptable for operation without repair.

EVENT DESCRIPTION

Initial Conditions

At the time the condition was identified, Unit 1 was in Mode 5 for refueling outage B116R1.

Discussion

On March 15, 2006, during the Unit 1 refueling outage, routine visual inspections of the reactor internals in accordance with the BWR Vessel and Internals Project (BWRVIP), identified an indication on a CS header piping weld. Specifically, the B loop CS piping circumferential weld designated as P3c-270 was examined using a Wesdyne/Westinghouse automated Double-Up Inspection Tool with an ROS color camera calibrated to 0.0005-inch resolution (i.e., enhanced visual examination (EVT-1)) and a previously unidentified indication extending from the five o'clock to seven o'clock positions was found. Weld P3c-270 also contained an existing indication (i.e., starting at the one o'clock position and ending at the three o'clock position), discovered in 1993, which has been inspected every outage since 1993 with no evidence of growth. The new indication was not previously recorded because its location was not accessible during previous inspections with hand-held cameras. The use of the automated Double-Up Inspection Tool for this component permitted increased coverage; thus locating the flaw.

On March 21, 2006, additional Ultrasonic Test (UT) examinations of the entire weld were performed to determine full extent of the indication. The UT results revealed that, with the two flaws, a total of approximately 81 percent of the weld was flawed. Based on these results, it was determined that repair was required for continued operation of the unit and that, without extensive evaluation, confirmation that the B loop CS piping would remain intact during a design basis accident was not possible.

This condition is being reported in accordance with 10 CFR 50.73(a)(2)(ii)(B), as an unanalyzed condition that significantly degraded plant safety. Although it cannot be determined when the B loop of CS piping was rendered inoperable, it is evident that this condition existed in excess of the seven day Completion Time of Technical Specification 3.5.1, "ECCS - Operating," Condition A. As such, this condition is also

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EVENT DESCRIPTION (continued)

being reported in accordance with 10 CFR 50.73(a)(2)(i)(B) as operation prohibited by the plant's Technical Specifications.

EVENT CAUSE

The root cause of this event is intergranular stress corrosion cracking (IGSCC) in the heat affected zone (HAZ) of weld P3c-270.

During B116R1, invessel inspections were performed using a Wesdyne/Westinghouse automated Double-Up Inspection Tool with an ROS color camera calibrated to 0.0005-inch resolution. The use of the improved inspection equipment permitted increased coverage; thus locating the flaw.

The P3c-270 weld is unique to Unit 1, B loop CS, and was likely made to assist in fit-up of the piping during installation. Neither Unit 1 CS loop A or Unit 2 have a corresponding weld location.

SAFETY ASSESSMENT

The safety significance of this condition is considered minimal and was characterized as having very low safety significance in Brunswick Steam Electric Plant - NRC Integrated Inspection Report Nos. 05000325/2006002 and 05000324/2006002, dated April 30, 2006.

The as-found condition of the Unit 1 B loop CS pipe weld P3c-270 resulted in the Unit 1 B loop CS subsystem being inoperable for an indeterminate amount of time. Although the CS system mitigates several core damage sequences, B loop CS was still capable of mitigating all of the sequences with the exception of large break loss of coolant accident (LOCA). General Electric NEDO-20566A, "GE Analytical Model for Evaluation of LOCA Analysis in Accordance with 10 CFR 50 Appendix K," provides the basis for this conclusion. NEDO-20566A describes the design features required in various accident conditions, and concludes that for all pipe breaks, other than breaks in the recirculation system, the core would remain flooded over the entire fuel length, and injection from any one Emergency Core Cooling System pump is sufficient to maintain adequate core cooling. Therefore, for pipe breaks other than on the recirculation lines, it is not necessary to maintain a core spray pattern over the top of the fuel because it would not be uncovered.

Making the conservative assumption that the flawed weld would not remain intact, B loop CS was capable of delivering flow to the reactor vessel outside the shroud. Therefore, the vulnerable accidents can be limited to the design-basis double-ended break of a recirculation line, which is quantified as a Large LOCA accident sequence in the BSEP Probabilistic Safety Analysis (PSA). If the PSA model is quantified with

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

SAFETY ASSESSMENT (continued)

B loop CS unavailable for Large LOCA initiating events, the change in core damage frequency (CDF) would be sufficiently small to conclude that the safety significance of the CS piping crack is minimal. The estimated frequency of a severe seismic event is sufficiently small to conclude that consideration of a seismic event would have a negligible effect on CDF.

CORRECTIVE ACTIONS

On April 1, 2006, weld P3c-270 was repaired by installation of a clamp designed to structurally replace the weld. The clamp was designed in accordance with BWRVIP Report BWRVIP-19-A, "BWR Vessel and Internals Project, Internal Core Spray Piping and Sparger Repair Design Criteria," which includes BWRVIP-84, "BWR Vessel and Internals Project, Guidelines for Selection and Use of Materials for Repairs to BWR Internal Components."

PREVIOUS SIMILAR EVENTS

A review of events which have occurred within the past three years has not identified any previous similar occurrences.

COMMITMENTS

No regulatory commitments are contained in this report.