Progress Energy

May 15, 2003

SERIAL: BSEP 03-0077

10 CFR 50.73

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

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-324/LICENSE NO. DPR-62
LICENSEE EVENT REPORT 2-03-001

Ladies and Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.73, Progress Energy Carolinas, Inc. submits the enclosed Licensee Event Report. This report fulfills the requirement for a written report within sixty (60) days of a reportable occurrence.

Please refer any questions regarding this submittal to Mr. Edward T. O'Neil, Manager – Support Services, at (910) 457-3512.

Sincerely,

W. G. Noll
Plant General Manager
Brunswick Steam Electric Plant

SFT/sft

Enclosure: Licensee Event Report

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

IE22

cc (with enclosure):

U. S. Nuclear Regulatory Commission, Region II
ATTN: Mr. Luis A. Reyes, Regional Administrator
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, GA 30303-8931

U. S. Nuclear Regulatory Commission
ATTN: NRC Resident Inspector
8470 River Road
Southport, NC 28461-8869

U. S. Nuclear Regulatory Commission
ATTN: Ms. Brenda L. Mozafari (Mail Stop OWFN 8G9) **(Electronic Copy Only)**
11555 Rockville Pike
Rockville, MD 20852-2738

Ms. Jo A. Sanford
Chair - North Carolina Utilities Commission
P.O. Box 29510
Raleigh, NC 27626-051

NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
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 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@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 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

Main Steam Line Drain Isolation Valve Local Leak Rate Test Failures

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
03	21	03	2003 --	01 --	00	05	15	2003	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	5
10. POWER LEVEL	0

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)	X	50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)	X	50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Steve Tabor – Lead Engineer Technical Support Specialist	(910) 457-2178

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	JM	ISV	A391	Y					

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MO	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

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

On March 21, 2003, during the Unit 2 B216R1 refueling outage, the results of local leak rate testing of the main steam line drain inboard and outboard isolation valves, 2-B21-F016 and 2-B21-F019, determined that the valves would not pressurize and consequently, the Technical Specification primary containment leakage rate limit was exceeded. Following the performance of valve maintenance activities and prior to restart of the unit, post-maintenance leakage rate tests were performed which verified valve leakage rates were within allowable limits. The failure of the valves to minimize leakage rates within allowable limits is attributed to a less than optimum valve design. Corrective actions include future replacement of the valves with a valve design which is more suited for the valve application. Review of Licensee Event Reports within the past three years identified no similar events.

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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 21, 2003, the Unit 2 reactor was shutdown in day 14 of refueling outage B216R1. Local leak rate testing (LLRT) of the main steam line (MSL) [SE] drain inboard and outboard isolation valves [ISV], 2-B21-F016 and 2-B21-F019, had been completed to satisfy Technical Specification (TS) primary containment leakage rate surveillance requirements. TS 5.5.12, Primary Containment Leakage Rate Testing Program, establishes the overall primary containment leakage rate acceptance criteria as less than or equal to 1.0 L_a when pressurized to P_a, 49 psig.

EVENT DESCRIPTION

On March 9, 2003, MSL drain outboard isolation valve, 2-B21-F019, failed LLRT requirements. The as-found leakage rate was documented as a would not pressurize (WNP) condition, indicating that the valve's leakage rate exceeded the ability to be measured with the normal flow meter range of 0 to 78 standard cubic feet per hour (scfh). A work order was generated to repair the valve. Corrective maintenance on the valve included replacement of the valve disc and stem components. On March 17, 2003, a post-maintenance LLRT verified that the as-left 2-B21-F019 leakage rate (i.e., 0.12 scfh) was within the allowable limit.

On March 21, 2003, the MSL drain inboard isolation valve, 2-B21-F016, was leak rate tested. The 2-B21-F016 valve is a double-disc gate valve and, due to its configuration, can only be tested by pressurizing between the valve's inboard and outboard discs. At the time of testing, the main steam lines were flooded. Initial test results were considered inconclusive since bubbles were observed escaping from the reactor vessel, indicating a leak path on the inboard side of the valve, but with no identified leakage at the vent path on the outboard side of the valve. A second informational leak rate test was performed with the same result. The results of these two tests indicated that the valve's outboard disc was holding pressure; however, the amount of pressure could not be quantified. The decision was made to drain the steam lines and retest the valve's leakage rate. During this test water and air were observed escaping through the vent path and the leakage rate was considered to be a WNP condition. A work order was generated to repair the valve. The valve's limit switch was adjusted and the post-maintenance LLRT verified that the as-left 2-B21-F016 leakage rate (i.e., 0.02 scfh) was within the allowable limit.

This condition is being reported in accordance with the requirements of 10 CFR 50.73(a)(2)(ii)(A) in that the loss of containment function or integrity, including containment leak rate tests where the total containment as-found minimum pathway leak rate exceeds the limiting condition for operation in the Technical Specification and, therefore, resulted in the condition of the plant, including its principal safety barriers, being seriously degraded. In addition, this condition satisfies the reporting criteria of 10 CFR 50.73(a)(2)(v)(C) in that the leakage of the MSL drain isolation valves in excess of the overall limits established by the TS represents a condition that alone could have prevented the fulfillment of the primary containment isolation system to control the release of radioactive material.

EVENT CAUSE

The cause of this condition is attributed to a valve design that is not well suited for the application. Originally, the 2-B21-F016 and 2-B21-F019 valves were flex-wedge torque seated valves manufactured by Anchor Darling. These valves were susceptible to thermal binding and exhibited poor leak rate test performance and consequently, were replaced (i.e., Unit 2 valves in 1990 and Unit 1 valves in 1991) with double-disc gate valves manufactured by Anchor

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

Darling. This valve type was considered to be the best design available for containment isolation valves in high temperature service at that time. However, subsequent leak rate test performance did not significantly improve. To improve valve performance, the valves were changed from torque seated to limit seated in 1995. Although limit seating has improved performance, the current configuration is considered less than optimum for the long term.

CORRECTIVE ACTIONS

1. The 2-B21-F016 and 2-B21-F019 valves were repaired and post-maintenance LLRTs performed prior to unit startup. The post-maintenance LLRTs confirmed leakage rates within the allowable limit.

2. A team will be assembled for the purpose of reviewing applicable industry experience on similar valve applications to determine an optimum valve design. This team will develop a comprehensive valve improvement plan including the consideration of new valve design. The plan will incorporate improvements for the Unit 1 MSL drain valves which are also double-disc gate valves of similar design and have experienced previous leak rate performance concerns. The valve improvement plan will be implemented in accordance with the Maintenance Rule action plan.

SAFETY ASSESSMENT

The actual safety significance of this condition is considered minimal in that the main steam line drain piping inside primary containment, the reactor building, and extending through the turbine building to the condenser was not ruptured during the last operating cycle. The potential safety significance of this condition is considered minimal based on the physical evidence obtained during valve repair activities. During the post LLRT maintenance that was performed on each of the valves, no evidence of major valve internal wear or degradation was identified. The relatively minor defects that were identified are not considered the result of a total failure of either of the valves to isolate or the expected damage that would most likely occur from a 1000 psi differential pressure being applied across the valve seats during an operating cycle. Consequently, based on the evidence, although the valves were not leak tight, the valves were capable of closing and reducing the release of radioactive gases.

PREVIOUS SIMILAR EVENTS

A review of reportable events for the past three years did not identify any previous similar events.

COMMITMENTS

Those actions committed to by Progress Energy Carolinas, Inc. in this document are identified below. Any other actions discussed in this submittal represent intended or planned actions by Progress Energy Carolinas, Inc. They are described for the NRC's information and are not regulatory commitments. Please notify the Manager - Support Services at BSEP of any questions regarding this document or any associated regulatory commitments.

> A team will be assembled for the purpose of reviewing applicable industry experience on similar valve applications to determine an optimum valve design. This team will develop a comprehensive improvement plan, including the consideration of a new valve design, by July 30, 2003.