Progress Energy

JUN 1 6 2005

SERIAL: BSEP 05-0069

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 1-2005-003

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. 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,

David Hinds
Plant General Manager
Brunswick Steam Electric Plant

WRM/wrm

Enclosure: Licensee Event Report

IE22

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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
Inappropriate Use of Technical Specification 3.0.5 During Control Rod Manipulations

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	20	2005	2005	-- 003 --	00	06	16	2005	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)
William R. Murray, Lead Licensing Engineer	(910) 457-2842

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 April 20, 2005, Unit 1 was in Mode 5 for a fuel bundle replacement outage. Following completion of maintenance to the position indicating probe (PIP) for control rod 46-15, a PIP position simulator, being used to provide full-in indication to the one-rod-out interlock, was not removed prior to performing single control rod subcriticality testing and operability testing for the control rod. Plant operators determined that performance of single control rod subcriticality testing was a prerequisite for performing operability testing of the control rod. Operators incorrectly used Technical Specification (TS) 3.0.5 to return control rod 46-15 to service before beginning the single control rod subcriticality testing. Following single control rod subcriticality testing for two control rods and encountering problems with the operability testing for control rod 46-15, operators continued to use TS 3.0.5 while performing testing of another control rod. This resulted in a failure to comply with Technical Specification required actions. To prevent recurrence of this event, training on the proper use of TS 3.0.5 will be provided to licensed operators. In addition, enhancements are planned to the plant procedures which govern the installation and removal of PIP jumpers and to plant procedures to prohibit use of the PIP position simulator as a jumper for the one-rod-out interlock.

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

INTRODUCTION

On April 20, 2005, Unit 1 was in Mode 5, having been shut down on April 15, 2005, for a fuel bundle replacement outage. Following completion of maintenance to the position indicating probe (PIP) for control rod [AA] 46-15, a PIP position simulator, being used to provide full-in indication to the one-rod-out interlock, was not removed prior to performing single control rod subcriticality testing and operability testing for control rod 46-15. Following maintenance to the PIP for control rod 46-15, plant operators determined that performance of single control rod subcriticality testing was a prerequisite for performing operability testing of control rod 46-15. Operators incorrectly used Technical Specification 3.0.5 to return control rod 46-15 to service before beginning performance of the single control rod subcriticality testing. Following single control rod subcriticality testing for two control rods and encountering problems with the operability testing for control rod 46-15, operators continued with testing of one additional control rod before it was determined that the PIP position simulator was still installed, thus resulting in a failure to comply with Technical Specification requirements. This condition was determined to be reportable in accordance with 10 CFR 50.73(a)(2)(i)(B) as an operation or condition prohibited by the Technical Specifications.

EVENT DESCRIPTION

Initial Conditions

On April 15, 2005, Unit 1 was shut down for a fuel bundle replacement outage. Following diagnostic testing, it was determined that the PIP for control rod 46-15 required replacement. Control rod 46-15 was fully inserted and disarmed.

Technical Specification 3.9.4, "Control Rod Position Indication," requires the "full-in" position indication channel for each control rod to be operable while in Mode 5. In addition, Technical Specification 3.9.2, "Refuel Position One-Rod-Out Interlock," requires the refuel position one-rod-out interlock to be operable when in Mode 5 with the mode switch in the refuel position when any control rod is withdrawn.

In order to allow fuel replacement activities to continue, a PIP position simulator was installed. The PIP position simulator provided a full-in indication to the one-rod-out interlock for control rod 46-15, allowing the refueling bridge to move over the core.

Discussion

The PIP for control rod 46-15 was replaced on April 18, 2005. On April 19, 2005, at the beginning of the night work shift, a senior reactor operator (SRO) from the outage organization was dispatched to the control room to assist in the performance of control rod testing in accordance with plant procedure 0PT-90.2, "Friction Testing of Control Rods." At that time, the PIP for control rod 46-15 was still considered inoperable and the limiting condition for operation (LCO) of Technical Specification 3.9.4 was being satisfied by Required Actions A.2.1 and A.2.2 of Technical Specification 3.9.4 (i.e., control rod 46-15 was fully inserted and disarmed). A review by the SRO determined that the work order to repair or replace the PIP for control rod 46-15 was still open, testing to demonstrate the operability of control rod 46-15 had not yet been performed, and equipment controls to disarm control rod 46-15 remained in effect. The SRO contacted the Instrumentation & Control (I&C) group to determine the status of the work order.

An I&C Supervisor incorrectly determined, based on review of the work order and discussion with an I&C technician involved, that replacement of the PIP had been completed and that the PIP position simulator had been removed. No field verification was performed to confirm the status of the work order. This resulted in incorrect information being provided to the SRO assigned to assist with the performance of control rod testing, who processed the work order as complete.

Based on completion of the work order for control rod 46-15, the SRO prepared to perform testing to demonstrate the operability of control rod 46-15 in accordance with plant procedure 0PT-90.2, "Friction Testing of Control Rods." The SRO correctly determined that Technical Specification 3.0.5 would allow re-arming of control rod 46-15 for the purpose of testing that control rod. Technical Specification 3.0.5 allows equipment that has been removed from service or declared inoperable to comply with Technical Specification Actions to be returned to service, under administrative controls, solely to perform testing required to demonstrate operability of the equipment.

However, a prerequisite of plant procedure 0PT-90.2 requires that a reactor subcriticality test be performed in accordance with plant procedure 0FH-11, "Refueling," if this testing is being performed following an initial core reload. The reactor subcriticality test is referenced as the "single control rod subcriticality test" in step 7.2.15 of plant procedure 0FH-11. The reactor engineer recommended that two control rods, 14-27 and 26-15, be tested to satisfy the prerequisite in plant procedure 0PT-90.2 for the reactor subcriticality test.

The SRO assisting with control rod testing then asked the unit SRO to independently assess the use of Technical Specification 3.0.5. Both SROs incorrectly determined that Technical Specification 3.0.5 allowed re-arming of control rod 46-15 for the purpose of completing prerequisite single rod subcriticality testing to establish operability of control rod 46-15. In fact, the single control rod subcriticality testing could have been performed without use of Technical Specification 3.0.5 to re-arm control rod 46-15. The one-rod-out interlock would have allowed individual selection and withdrawal of the control rods necessary for the single control rod subcriticality test and, therefore, use of Technical Specification 3.0.5 was not necessary.

At 0330 hours on April 20, 2005, control rod 46-15 was re-armed for the testing. Single control rod subcriticality testing was then individually performed on control rods 14-27 and 26-15 by withdrawing each control rod from position 00 to position 48 and then inserting each control rod from position 48 back to position 00. At 0515 hours, testing for control rod 46-15 began. When operators attempted to move control rod 46-15, the green full-in indicator light remained lit and the position indication stayed on position 00. Additional attempts were made to move control rod 46-15 with the same results. Operators concluded that control rod 46-15 was difficult to move and then used Section 8.1, "Control

Rod Difficult to Withdraw, Control Rod at Position 00," of plant procedure OP-07, "Reactor Manual Control System Operating Procedure," to attempt movement of the control rod.

To verify the control rod drive system was functioning properly, operators decided to stop testing control rod 46-15 and to perform friction testing of control rod 26-35. Testing of control rod 26-35 began at 0536 hours. At 0548 hours, operators successfully withdrew control rod 26-35 to position 48 and reinserted the control rod to position 00.

Having confirmed proper functioning of the control rod drive system, Operators then returned to control rod 46-15 and again made numerous attempts to move the control rod from position 00. These attempts were not successful. At 0609 hours, operators concluded an undetermined problem existed with control rod 46-15 and terminated control rod testing. Investigation revealed the PIP position simulator was still installed for control rod 46-15. At that time, operators reassessed compliance with Technical Specifications. Operators determined that compliance with the Required Actions of Technical Specifications 3.9.2 and 3.9.4 had not been satisfied when control rods 14-27, 26-15, and 26-35 had been withdrawn for the single control rod subcriticality testing, but that the facility was currently in compliance with Technical Specifications 3.9.2 and 3.9.4. The PIP position simulator was subsequently removed. When the PIP position simulator was removed and the circuit restored, the control room position indication displayed position 00 for control rod 46-15. Testing of control rod 46-15 was then completed satisfactorily.

EVENT CAUSE

The cause of this occurrence was the misapplication of Technical Specification 3.0.5. Technical Specification 3.0.5 allows for the return of equipment to service in order to verify operability. Although performance of the single control rod subcriticality test for control rods 14-27 and 26-15 was a procedural prerequisite for testing control rod 46-15, the single control rod subcriticality test was not required in order to demonstrate the operability of control rod 46-15 and re-arming of control rod 46-15 was not necessary to perform the single control rod subcriticality testing. Therefore, the allowance of Technical Specification 3.0.5 should have been used only to return control rod 46-15 to service for operability testing and should not have been used to return control rod 46-15 to service for performance of single control rod subcriticality testing of control rods 14-27, 26-15, and subsequent friction testing of control rod 26-35.

The assessment and determination of the acceptability to use Technical Specification 3.0.5 was conducted independently by two senior reactor operators. However, interviews of other licensed operators determined that a larger population of operators could have misapplied the allowance of Technical Specification 3.0.5 in this situation.

A contributing cause of this event was the failure to remove the PIP position simulator. Because the PIP position simulator was installed instead of a jumper, operators received normal position indications (i.e., a green "full-in" light and the display of position 00 on the four rod display) from control

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

rod 46-15. Installation of a jumper instead of the PIP position simulator would have provided a green "full-in" light to operators, but the four rod position indication would have been blank. With the green "full-in" light and the display of position 00 on the four rod display, operators initially concluded that, with normal indications, the problem with control rod 46-15 was a difficult-to-move control rod.

CORRECTIVE ACTIONS

1. To prevent recurrence of this event, training on the proper use of Technical Specification 3.0.5 will be provided to licensed operators. This is scheduled to be completed by July 18, 2005.
2. Lessons learned for this event will be reviewed with licensed operators and I&C personnel.
3. Refresher training on the proper use of Technical Specification 3.0.5 and outage-related LCOs will be provided to licensed operators.
4. Appropriate I&C personnel involved with this event have been counseled.
5. The controls governing the installation and removal of PIP jumpers will be changed from the plant work order process to being controlled through a plant procedure. Applicable plant procedures will be revised to prohibit use of the PIP position simulator as a jumper for the one-rod-out interlock.

SAFETY ASSESSMENT

The significance of this occurrence is considered minimal. Although manipulation of control rods with the PIP position simulator installed potentially allowed withdrawal of more than one control rod, the control rod manipulations were conducted using plant procedures that only allowed individual control rod manipulation. Therefore, only two control rods could have been withdrawn at the same time. An analysis was performed, assuming that control rod 46-15 and a face adjacent control rod (i.e., control rod 46-19 was used) were fully withdrawn simultaneously. The analysis demonstrates that the reactor would have remained subcritical at 0.611% shutdown margin. This analysis bounds the potential combined withdrawal of control rods 46-15 and 26-35.

PREVIOUS SIMILAR EVENTS

A review of events occurring within the past three years identified an occurrence which was reported in LER 2-2005-001, Compliance with Single Control Rod Withdrawal - Cold Shutdown Technical Specification. This LER discussed an event where operators placed equipment control tags on the wrong hydraulic control unit during replacement of a control rod drive, resulting in the failure to comply with TS 3.10.4 Required Actions. The root cause and corrective actions associated with LER 2-2005-001 could not reasonably be expected to have prevented the condition being reported herein.

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COMMITMENTS

No regulatory commitments are contained in this report. Those actions discussed in this submittal will be implemented in accordance with corrective action program requirements.