Progress Energy

DEC 29 2006

SERIAL: BSEP 06-0133 10 CFR 50.73

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

Subject: Brunswick Steam Electric Plant, Unit No. 2
Docket No. 50-324/License No. DPR-62
Licensee Event Report 2-2006-001

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,

BCW

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

MAT/mat

Enclosure:

Licensee Event Report

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IE22

cc (with enclosure):

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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
Brunswick Steam Electric Plant (BSEP), Unit 2	05000324	1 of 6

4. TITLE
Loss of Startup Auxiliary Transformer Results in Unit 2 Manual Reactor Protection System Actuation

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
11	01	2006	2006	-- 001 --	00	12	29	2006	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100

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

☐ 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)
☐ 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 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 November 1, 2006, at approximately 18:23 (EST) Unit 2 experienced a loss of the unit's Startup Auxiliary Transformer resulting in a loss of reactor forced circulation and initiation of a manual Reactor Protection System actuation. All control rods properly inserted when the manual reactor scram was performed. Due to the loss of the Startup Auxiliary Transformer offsite power to the unit's power buses was lost when the Unit 2 main generator tripped. The four Emergency Diesel Generators (EDGs) properly started and EDGs 3 and 4 supplied the Unit 2 emergency buses. Appropriate Primary Containment Isolation System (PCIS) isolations occurred upon the loss of power to the unit's power buses. An expected Reactor Pressure Vessel coolant shrink resulted in the coolant level decreasing below the Reactor Vessel Water Level - Low Level 1, with appropriate PCIS isolations. Additionally, coolant level subsequently reached Low Level 2, at which point an additional PCIS isolation and system actuations appropriately occurred.

Loss of the SAT was caused by an ineffective mechanical connection between the SAT's x-winding non-segregated bus and the associated bus bar. The root cause of the event is the ineffective connection combined with the fact that the condition monitoring performed for this bus was not adequate to detect the latent problem. Corrective actions include implementation of enhanced predictive maintenance requirements that will include thermography inspections and trending of all Unit 1 and Unit 2 non-segregated buses.

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

Initial Conditions

At the time of this event, Unit 2 was in Mode 1, at approximately 100 percent of rated thermal power. All required safety-related systems were operable.

Reportability Criteria

This event resulted in a manual Reactor Protection System (RPS) [JC] actuation, initiation of the High Pressure Coolant Injection (HPCI) [BJ] system, manual initiation of the Reactor Core Isolation Cooling (RCIC) [BN] system, various Primary Containment Isolation system (PCIS) [JM] initiations, and starting of the Emergency Diesel Generators (EDGs) [EK]. As such, this condition is being reported in accordance with 10 CFR 50.73(a)(2)(iv)(A) as an event that resulted in valid actuation of systems listed in 10 CFR 50.73(a)(2)(iv)(B). The NRC was initially notified of this event at 1910 EST on November 1, 2006 (i.e. Event Number 42955).

Event Description

On November 1, 2006, at approximately 18:23 (EST) Unit 2 experienced a loss of the unit's Startup Auxiliary Transformer (SAT) [EA] and an associated loss of the reactor recirculation pumps [AD]. A manual reactor scram was performed, as required by station Abnormal Operating Procedures, to preclude the potential for thermal-hydraulic instability. All control rods properly inserted.

Due to the loss of the SAT and subsequent manual reactor scram, a loss of offsite power resulted to the unit's power buses. This is consistent with BSEP electrical design. When the unit is running, power is supplied to the onsite electrical distribution system via the Unit Auxiliary Transformer (UAT) [EA]. When the main generator is taken offline, an automatic quick dead bus transfer to the SAT occurs. In this event, the SAT was faulted. The four site EDGs started. EDGs 3 and 4 supplied the Unit 2 emergency buses. Consistent with plant design, the loss of power to the unit's power buses resulted in the following automatic actuations and isolations.

Group 1 Main Steam Isolation Valves (MSIVs)

Group 3 Reactor Water Cleanup Isolation Valves

Group 10 Drywell pneumatic Isolation Valves

Reactor Building Ventilation [VA] system isolation

Standby Gas Treatment [BH] system initiation

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

Event Description (continued)

The loss of offsite power resulted in declaration of an Unusual Event. The Unusual Event was declared at 1823 EST on November 1, 2006, and terminated at 1745 EST on November 2, 2006, when offsite power was restored to the Unit 2 emergency buses via backfeed through the UAT. Although actions could have been taken to establish the backfeed faster, to ensure personnel safety, a clearance was developed which specifically accounted for existing plant conditions. Since the plant can remain on the EDGs for several days, the timing of the manual transfers to the alternate circuits did not adversely impact any plant safety design limits.

As a result of the scram and loss of offsite power, reactor water level reached both the Reactor Vessel Water Level - Low Level 1 (LL1) and Low Level 2 (LL2) setpoints. The LL1 signal properly resulted in the following primary containment isolations.

Group 2	Drywell Equipment and Floor Drain, Traversing In-Core Probe, Residual Heat Removal (RHR) Discharge to Radwaste, and RHR Process Sample Isolation Valves
Group 6	Containment Atmosphere Control/Dilution, Containment Atmosphere Monitoring, and Post Accident Sampling System Isolation Valves
Group 8	RHR Shutdown Cooling Suction and RHR Inboard Injection Isolation Valves

All LL1 isolations occurred as designed.

Prior to reactor water level reaching the LL2 setpoint, the RCIC system was manually started for level control. Subsequently, the LL2 setpoint was momentarily reached, resulting in the following actuation.

HPCI system actuation and injection

These HPCI system actuation occurred per design.

At the time of the event, it was believed that the Control Room Emergency Ventilation (CREV) [VI] system and the Alternate Rod Insertion (ARI) system did not operate appropriately and control room operators took manual actions to initiate these systems. Typically, these systems actuate at LL2. Engineering has evaluated the calibration data, as-left setpoints, and plant transient response records and, based on this data, determined that the CREV and ARI automatic initiation logic were never completed since indicated level did not get low enough to ensure trip unit actuation. In addition, surveillance tests were performed on CREV and ARI that validated both systems were operable and would have performed their normal safety function as required if preemptive manual actions had not been taken.

During the transient response and subsequent stabilization and cooldown, reactor water level and reactor pressure were controlled by intermittent use of HPCI, RCIC, Safety/Relief valves (SRVs) A, B, E, F, G, and L, and the Control Rod Drive (CRD) [AA] system. During this event, SRV 2F spuriously closed twice with the control switch in the on position. The first occurred with reactor pressure at approximately 830 psi for

Event Description (continued)

approximately 1.5 minutes. The second occurred with reactor pressure at approximately 927 psi for approximately 1 minute. Engineering performed a fault tree analysis of this condition and all active components which were identified as having the potential to cause the failure were replaced. Additionally, connections in the control room back panels and the drywell were confirmed to be tight and the results of continuity and voltage measurements were all satisfactory. No other SRVs demonstrated abnormal response during the event and stroking of all SRVs has been observed with no deficiencies noted. Post-maintenance testing confirmed the operability of all SRVs prior to startup of the unit.

As stated above, this event resulted in the four site EDGs starting with EDGs 3 and 4 supplying the Unit 2 emergency buses. EDGs 1 and 2 continued to run unloaded, per design, until, at approximately 0400 EST on November 2, 2006 (i.e., after approximately 9 hours and 37 minutes of run time), EDG 1 tripped. EDGs 3 and 4 were unaffected by the tripping of EDG 1 and continued to supply the Unit 2 emergency buses until offsite power was restored to the Unit 2 emergency buses via backfeed through the UAT.

EDG 1 tripped as the result of lube oil duplex strainer swapping. The trip occurred because the procedure for cleaning, filling and swapping an EDG lube oil duplex filter does not have special instructions for restoring a cleaned, empty filter basket when the EDG is in service and the in-service strainer has a high differential pressure (dP). The instructions require the empty basket to be filled slowly with the process fluid by opening the vent valve on the empty basket lid and then slowly adjusting the valve handle to allow a portion of the process flow to fill the empty basket; while maintaining flow through the in-service strainer to the lube oil header. The diverting of the oil flow to the path of least resistance (i.e. the empty strainer basket versus the high dP strainer) results in a momentary drop in lube oil header pressure. This pressure drop was sufficient to result in a low lube oil header pressure trip. LER 1-2006-007 (i.e., Serial: BSEP 06-0137) addresses this condition as well as inoperability of EDG 1 due to foreign material (i.e., a cleaning towel) which was left in the lube oil system after maintenance activities conducted from October 23 through 27, 2006.

Plant cooldown continued, per existing plant procedures, and Unit 2 entered Mode 4 on November 4, 2006, at approximately 0624 EST.

Event Cause

The Unit 2 SAT was lost due to a fault on the X-winding non-segregated bus. The SAT non-segregated bus consists of two aluminum bus bars connected by a spacer for each of the three phases. The pair of buses for each phase is connected to an electrical penetration in the Turbine Building wall by a pair of flexible links, one on the top and one on the bottom. The A phase flexible link was the source of the fault. Inspection and analysis of the flexible link and the bus bar connections found they had been operating at an unusually high temperature (i.e., 500 to 800 degrees F) for an extended period of time, as indicated by extensive corrosion of the copper flexible link laminations and signs of severe overheating of the insulating tape wrapping. The most likely source of the heat is inadequate flexible link to bus bar contact. The cause of the inadequate

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

Event Cause (continued)

contact is most likely due to inadequate bolt torquing, the presence of tape or some other foreign material between the bolted surfaces, or relaxation of the bolted connection from the heatup and cooldown cycles, or some combination of these factors. It is not possible to identify the specific mechanism causing the inadequate contact as the materials were consumed by the fault. Based on these indications, it was determined that the original design and/or installation of this joint did not ensure adequate connection between the flexible connection and the bus bar.

The root cause of the event was determined to be ineffectiveness of the original installation combined with the fact that the condition monitoring performed for this bus was not adequate to detect the latent problem with overheating which had existed for an extended period of time. The intrusiveness and/or frequency of the existing preventive maintenance task was not sufficient to detect the extended operation at high temperature.

Safety Assessment

The safety significance of this condition is considered minimal.

The loss of the Unit 2 SAT resulted in a manual RPS actuation; however, during this event, systems directly supporting Unit 2 functioned as expected and operator response to the event was appropriate, resulting in low safety significance. The spurious closure of SRV 2F did not have a significant impact on the ability to control reactor pressure. Prudent actions were implemented to re-establish offisite power to the Unit 2 emergency buses via backfeed through the UAT. Since the plant can remain on the EDGs for several days, the timing of the manual transfers to the alternate circuits did not adversely impact any plant safety design limits. The loss of EDG 1 did not result in a loss of power to emergency bus 1 since this bus was not affected by the loss of the Unit 2 SAT and continued to be supplied via offiste power.

Corrective Actions

The affected flexible link and bus bar connection have been repaired and the adjacent flexible links were inspected to confirm they were not damaged.

The following corrective actions to prevent recurrence has been established as a result of this event.

- Predictive maintenance requirements will be revised to include thermography of all Unit 1 and Unit 2 non-segregated buses and to provide appropriate trending. The revision is currently scheduled to be completed by January 12, 2007.

- Thermography of energized Unit 1 and Unit 2 non-segregated buses has been completed with no areas of excessive heating identified.

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Corrective Actions (continued)

- Unit 1 and Unit 2 non-segregated bus flexible link connections will be determinated, cleaned, and inspected. A portion of the Unit 2 non-segregated bus flexible link connections will be completed during the B218R1 outage (i.e., currently scheduled to begin in March 2007), if necessary, the remaining Unit 2 flexible link connections will be completed during the B219R1 refueling outage for Unit 2 (i.e., currently scheduled to begin in February 2009). The Unit 1 flexible link connections will be completed during the B117R1 refueling outage (i.e., currently scheduled to begin in March 2008).

- Infrared (IR) windows, allowing for at-power thermography, will be installed at non-segregated bus flexible link connections on Unit 1 during the B117R1 refueling outage (i.e., currently scheduled to begin in March 2008) and on Unit 2 during the B218R1 refueling outage (i.e., currently scheduled to begin in March 2007).

Previous Similar Events

A review of LERs and corrective action program condition reports for the past three years identified the following similar events.

- LER 1-2005-005, dated September 12, 2005, documents a Unit 1 Main Turbine trip followed by an automatic RPS actuation. The cause of the Main Turbine trip was the failure of the B phase of the Main Generator No Load Disconnect Switch (NLDS). The root cause of the event reported in LER 1-2005-005 was inadequate design and testing of the NLDS by the vendor; resulting in the NLDS not meeting its nameplate design rating. The corrective actions associated with LER 1-2005-005 could not have reasonably been expected to prevent the condition reported in this LER.

Commitments

No regulatory commitments are contained in this report.