CP&L

Carolina Power & Light Company

Brunswick Nuclear Project
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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

BRUNSWICK STEAM ELECTRIC PLANT UNIT 2
DOCKET NO. 50-324
LICENSE NO. DRP-62
LICENSEE EVENT REPORT 2-91-021

Gentlemen:

In accordance with Title 10 of the Code of Federal Regulations, the enclosed Licensee Event Report is submitted. This report fulfills the requirement for a written report within thirty (30) days of a reportable occurrence and is submitted in accordance with the format set forth in NUREG-1022, September 1983.

Very truly yours,

J. W. Spencer, General Manager
Brunswick Nuclear Project

GT/

Enclosure

cc: Mr. S. D. Ebneter
Mr. N. B. Le
BSEP NRC Resident Office

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NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Brunswick Steam Electric Plant Unit 2	05000324	1

TITLE (4) VOLTMETER INTERNAL FAILURE DURING TESTING RESULTS IN AN INADVERTENT HPCI INJECTION AND REACTOR SCRAM

EVENT DATE (5)			LER NUMBER (6)					REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR		SEQ. NO.		REV. NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
12	17	91	91	-	21	-	0	01	13	92		

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

OPERATING MODE (9)	2		20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	5		20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
			20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER (Specify in Abstract and Text)
			20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
			20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Glen M. Thearling, Regulatory Compliance Specialist	(919) 457-2038

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	BJ	EI	F137	N						

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

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single space typewritten lines) (16)

At 09:42, on December 17, 1991, Unit 2 was at 5% Reactor power. A startup was in progress on completion of a refueling outage. The Emergency Core Cooling Systems were operable. The Reactor Scrammed when the High Pressure Coolant Injection (HPCI) system inadvertently started and injected relatively cold water into the reactor vessel.

Instrumentation and Controls (I&C) technicians were performing a routine surveillance (2MST-RHR23M) on the HPCI Reactor vessel Low Level 2 (118") initiation logic. Per the surveillance, an "A" channel level instrument was actuated and its relay contacts were verified closed by checking the voltage across the open "B" channel contacts. This voltage check was performed with a Fluke 8600A-01 digital voltmeter (DVM), which has since been verified to have an internal fault. The fault allowed sufficient current flow to actuate an initiation of the HPCI system. The relatively cold water injected by the HPCI system caused the Reactor power to exceed the 15% Average Power Range Monitor (APRM) Scram setpoints. Once the HPCI initiation was verified as invalid, the HPCI system was manually shut down.

After the cause of the Unit 2 Reactor Scram had been identified, startup was recommenced at 02:23 on 12/18/91.

This isolated event posed minimal safety significance in that plant systems responded as designed.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQ NO.	REV NO.	PAGE (3)
Brunswick Steam Electric Plant Unit 2	05000324	91	21	0	2

TEXT (If more space is required, use additional NRC Form 366A's) (17)

INITIAL CONDITIONS

At ~09:42 on December 17, 1991, Unit 2 was in Startup at 5% Reactor power following completion of a refueling outage. Instrumentation and Controls (I&C) technicians were performing a routine surveillance 2MST-RHR23M on Low Level 2 (118") Emergency Core Cooling Systems (ECCS) initiation logic. The ECCS were operable.

EVENT NARRATIVE

During the High Pressure Coolant Injection (HPCI) portion of the surveillance, an "A" logic channel level instrument is actuated and the relay contacts are verified closed by checking the voltage across the open "B" channel relay contacts. The technicians saw a voltage lower than the ~125VDC they expected and stopped at this point to investigate the cause. Unknown to the technicians, an internal meter short allowed sufficient current flow to actuate the initiation relay for the HPCI system.

At 09:42:59 the HPCI system started and injected cold water (relative to the reactor coolant temperature) into the Reactor vessel at 4250 gpm. This caused Reactor power to exceed the 15% Average Power Range Monitor (APRM) Scram setpoints. All control rods fully inserted during the Scram and the plant systems functioned as designed. The control operators verified the HPCI initiation was invalid and began shutting down the HPCI system. Prior to completing the HPCI shutdown, the automatic high level trip signal (208") for HPCI, the reactor feed pumps, and the main turbine was reached.

When the technicians heard the Unit 2 Scram announced, everything was as it had been left when they stopped the test. After discussions with the management and Operations, the surveillance was repeated using the same test equipment and the HPCI initiation signal was again received. This supported a faulty Fluke 8600A-01 digital voltmeter (DVM) as the cause of the HPCI initiation. The surveillance was performed a third time, but with a new meter, and the results of the surveillance were satisfactory. Resistance checks of the meter later showed the existence of a low resistance value across the input terminals.

After the cause of the Unit 2 Reactor Scram had been identified, the startup was recommenced at 02:23 on 12/18/91.

CAUSE OF EVENT

The internal failure of the Fluke 8600A-01 DVM resulted in low resistance values at the input terminals for the voltage scales. Normal resistance values (on the order of 10 mega-ohms) were expected, but tests found only 500 ohms on the voltage scales. The low resistance value meant that a current path through the meter existed. The meter was returned to John Fluke Manufacturing Co. for repair and evaluation. The instrument was found to have been subjected to an overload to the input terminals. The overload appears to have been in excess of the 1000 volt maximum common mode voltage specified for using the meter. While the exact source of the overload could not be determined, the vender indicated that historically it has occurred while measuring the voltage across a coil as it was being de-energized. This results in a voltage spike that can momentarily exceed the maximum common mode voltage of the meter. The effect of the overload was an arc within the instrument that left a conductive carbon path between the input terminals. With the carbon path between the terminals, the meter acted as a short around whatever component it was to measure a voltage across. In this condition the meter would not indicate greater than ~50 volts on any voltage scale. This meter problem would have been obvious during the next use of the voltage scales.

NRC FORM 366A U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQ NO.	REV NO.	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

The vendor was aware of the potential for this type of failure and issued Fluke Product Change Notice (PCN) 888 Rev. 1 on 7/28/89 that referenced potential damage if voltages exceeded 1000 volts. The meter's technical manual cautions the user not to exceed the maximum allowable DC input overload values of 1200 volts DC and 1700 volts peak AC. The PCN was only sent to subscribers of this information service as it was not a personnel safety issue. BSEP did not subscribe to this service and therefore was not aware of this problem until Fluke's evaluation of the meter failure was completed on 12/20/91. When CP&L was notified of the PCN, the 37 remaining meters were inspected. The inspection found that unknown to CP&L, 26 of the meters that had been returned to Fluke for unrelated repairs had been modified to increase the gap (originally < 1/32") between the printed circuit board land (conductor printed on the circuit board) and the input pin. The remaining 11 meters were removed from the meter issue room pending modification or replacement. One of the 11 unmodified meters had indication of an arc but satisfactorily passed the resistance test.

CORRECTIVE ACTIONS

A maintenance policy has been initiated to require Fluke 8600 DVMs to be checked for low resistance readings at the input terminals for the voltage scales prior to issuance for use in Maintenance Surveillance Tests (MST). This will continue until the meters are properly modified to increase the gap where the arcing occurred.

All Fluke 8600 DVMs in the field were returned to the I&C instrument shop or checked in the field for proper resistance prior to the Unit 2 restart.

The procedure for calibration of Fluke 8600 DVMs will be revised to include resistance checks.

The unmodified Fluke 8600 DVMs were removed from the meter issue room pending modification or replacement.

An evaluation of the methods used by test equipment vendors to notify CP&L of problems that may impact plant operation will be conducted.

SAFETY ASSESSMEMT

This event posed minimal safety significance in that the plant systems responded as designed. Per the Updated Final Safety Analysis Report inadvertent HPCI injection is not a limiting Abnormal Operating Occurrence for this operating cycle.

PREVIOUS SIMILAR EVENTS

None

EIIS COMPONENT IDENTIFICATION

System/Component	EIIS Code
High Pressure Coolant Injection System	BJ
Volt Meter (test equipment)	EI