NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Brunswick Steam Electric Plant Unit 1	0 5 0 0 0 3 2 5	1 OF 0 4

TITLE (4) Inoperability of High Pressure Coolant Injection (HPCI) System (E41) Resulting From Failure of HPCI Pump Suppression Pool Suction Supply Outboard Isolation Valve E41-F041

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 4	2 0	8 8	8 8	— 0 1 1	— 0 1	0 6	2 9	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
1		20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
1 0 0		20.405(a)(1)(ii)	50.36(c)(2)	X 50.73(a)(2)(vii)	X OTHER (Specify in Abstract below and in Text, NRC Form 366A)
		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)	10CFR21
		20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
	AREA CODE
M. J. Pastva Jr., Regulatory Compliance Specialist	9 1 9 — 4 5 7 - 1 2 3 1 5

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	
B	B J	M O	L 2 0 0	Y							

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 0930 hours on 4/20/88, performance of Maintenance Surveillance Test 1MST-HPCI27M and the Reactor Core Isolation Cooling (RCIC) System revealed the High Pressure Coolant Injection (HPCI) System (E41) pump suction from the suppression pool, E41-F041, would not open. Unit 1 was at 100%. The HPCI System was declared inoperable. The RCIC, residual heat removal/low pressure coolant injection, Automatic Depressurization Systems, and the A and B core spray subsystems were operable. E41-F041 would not open due to a failure of the valve direct current (DC) powered motor, Limitorque Corp. Part No. 150-B56-189. The failed motor was replaced and the valve was returned to service. The HPCI System was returned to standby readiness at 0857 hours on 4/22/88.

Analysis of the failed motor at the CP&L test facility concluded the most probable cause of the failure was breakdown of the motor-winding insulation caused by high inductive voltage surges across the motor shunt coil which occurs when the motor power supply circuit breaker is opened. Other failures of DC-powered motors have been reported in LERs 1-87-023 and 2-87-001. By 1-20-89, plant modifications will be implemented to install surge protection within the shunt coil circuitry of DC motor control circuitry on Units 1 and 2.

NRC Region II was notified of this design deficiency via a 10CFR21 Reportability Determination (Serial No. BSEP/88-0472).

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

Initial Conditions

Unit 1 was operating at 100% power. The instrument channel calibration and functional test of the unit condensate storage tank (CST) (EIIS/KA/TK) low water level interlock functions of the High Pressure Coolant Injection (HPCI) System (E41) (EIIS/BJ) and the Reactor Core Isolation Cooling (RCIC) System (EIIS/BN) was in progress in accordance with Maintenance Surveillance Test 1MST-HPCI27M. This test verifies the capability for automatic switch over of each system's pump suction supply, which is normally aligned from the CST, to the suppression pool (EIIS/BT/TK) upon a low level in the CST. The RCIC and HPCI Systems, Residual Heat Removal (RHR) System (EIIS/BO), Automatic Depressurization System (EIIS/*), and the A and B core spray subsystems (EIIS/BM) were operable.

Event Description

Performance of this test revealed at 0930 hours on April 20, 1988, that the HPCI pump suction Primary Containment Isolation System outboard isolation valve, 1-E41-F041, (EIIS/JM/ISV) would not open in response to a test-simulated CST low level. The valve showed a dual position indication on the Control Room reactor turbine gauge board (RTGB) (EIIS/IU/MCBD). In response to this discovery, the Reactor Building (EIIS/NG) Auxiliary Operator (AO) was dispatched to check the status of the valve operator 250 volt (V) direct current (DC) motor breaker (EIIS/EJ/BKR), which is located on motor control center (MCC) 1XDA (EIIS/EJ/MCC). The AO found the breaker had tripped on magnetic (current) overload.

Effective at 0930 hours, the HPCI System was declared inoperable and a limiting condition for operation (LCO) was established due to the inoperable flow path from the suppression pool.

Event Investigation

A review of the valve historical file found that the actuator for the F041 valve had been rebuilt and the valve reworked during the 1987 outage. The motor was replaced during the rebuild, because the nameplate did not list the insulation class.

Initial troubleshooting found that the fuses were not blown, but that the breaker was tripped. The motor, Limitorque part No. 150-B56-189 was bridged and meggered and the series field was found shorted to ground. The breaker was removed and tested with no problems found. The actuator was inspected and found in a closed position. The valve was opened using a torque wrench; however, 30 ft-lbs of torque was required to pull the valve out of the seat.

*EIIS system description unavailable.

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

The normal torque range of this actuator should have been 12-17 ft-lbs of torque to seat the valve with a torque wrench. After the valve was pulled from the seat several attempts were made to manually seat the valve with a torque wrench. The torque switch opened at 10 ft-lbs repeatedly. The spring pack and torque switch were removed and bench-tested with satisfactory results. Another motor was installed and the actuator was tested using the motor actuator characterizer (MAC) test equipment. The initial MAC test with the as-found torque switch settings obtained a unit output torque of 155 ft-lbs. The unit output torque range for this actuator is 172 to 234 ft-lbs. The torque switch was adjusted and another MAC test was performed, obtaining a satisfactory output torque of 200.5 ft-lbs. A review of the MAC traces by Maintenance and Technical Support found no evidence of actuator or valve problems that may have caused this failure.

Failure analysis of the 1-E41-F041 actuator motor at the Carolina Power & Light Company Harris E&E Center determined the root cause of the failure was high voltage transients in the motor shunt coil that occur when the motor breaker is opened. The control logic for DC motors at the Brunswick station is designed such that the motor's electrical shunt fields are continuously energized with no voltage surge protection utilized. As a result of this design, when the power supply breaker to the motor is opened, a momentary high voltage transient of several thousand volts is experienced in the shunt coil due to the collapsing electrical field in the coil. Documentation has shown this momentary voltage transient typically exceeds the rating of the insulation of the coil windings. This voltage transient is believed to cause significant damage to the insulation with subsequent electrical shorting and failure of the shunt coil. Loss of the shunt coil thereby results in a loss of speed control for the motor.

Discussions with Limitorque and other nuclear sites indicate that the most common logic for those sites, which may not have the surge protection, deenergizes the shunt field each time the value is cycled; whereas, at Brunswick, the shunt field is only deenergized when the breaker is open as previously noted. The non-Brunswick design would experience a larger number of voltage spikes and, thus, would likely experience a higher failure rate.

As a result of this concern, operability checks of DC valve motors on both units were made, which showed the valves are operable. In addition, a plant standing instruction was implemented, which requires verification of DC valve operability following activities involving restoration of power to DC-powered valve motors. This action will ensure that motor failures due to this failure mechanism are identified prior to the valve being restored to operable status.

Corrective Action

The failed HPCI E41-F041 actuator motor was removed, a replacement motor was installed, and the valve was returned to service. The involved LCO was subsequently canceled and the HPCI System was returned to standby readiness at 0857 hours on April 22, 1988. To correct this identified design deficiency, plant modifications will be implemented to install surge protection within the shunt coil circuitry of the DC motor control circuitry to prevent high voltage transients in the shunt coil. The anticipated completion date for installation of these modifications is January 20, 1989.

On May 6, 1988, notification of this design deficiency was made to NRC Region II via a 10CFR21 Reportability Determination (Serial No. BSEP/88-0472).

Event Assessment

This event would not have been more severe under reasonable and creditable conditions as the supply from the CST would have been available. The CST can be supplied from the other units CST or from the makeup water system, thus ensuring a supply to the HPCI suction.

Other failures of DC-powered motors have been reported in LERs 1-87-023 and 2-87-001.

CP&L

Carolina Power & Light Company

Brunswick Steam Electric Plant
P. O. Box 10429
Southport, NC 28461-0429

June 29, 1988

FILE: B09-13510C 10CFR50.73
SERIAL: BSEP/88-0655

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

BRUNSWICK STEAM ELECTRIC PLANT UNIT 1
DOCKET NO. 50-325
LICENSE NO. DPR-71
SUPPLEMENT TO LICENSEE EVENT REPORT 1-88-011

Gentlemen:

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

Very truly yours,

S. A. Bishop for

C. R. Dietz, General Manager
Brunswick Steam Electric Plant

MJP/ah

Enclosure

cc: Dr. J. N. Grace
Mr. E. D. Sylvester
BSEP NRC Resident Office

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