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 6

TITLE (4) Reactor Scram Due to Primary Containment Group 1 Isolation Along With Trip of Reactor Core Isolation Cooling System and Trip/Lockout of Diesel Generator No. 4

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 5 0 0 0
1 1	0 2	8 5	8 5	— 0 5 9	— 0 1	0 6	2 7	8 8		0 5 0 0 0

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

OPERATING MODE (9)	2											
POWER LEVEL (10)	0 0 6		20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
			20.405(a)(1)(i)		50.36(c)(1)	X	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 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)		
			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 - 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
X	S J	J X	G 0 9 0	Y	X	B N	S H V	S 0 7 5	Y
X	L A	P S	N 1 5 2	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 2326 on 11-02-85, Unit 1 primary containment Groups 1, 3, and 6 isolations occurred due to a reactor low level (LL) No. 2 signal. The High Pressure Coolant Injection System and the Reactor Core Isolation Cooling (RCIC) System auto-started and RCIC tripped. The Core Spray Subsystems (A and B) auto-started on a momentary reactor LL No. 3 signal. The Reactor Building Ventilation System auto-isolated and the Standby Gas Treatment System auto-started. The Residual Heat Removal System LPCI mode did not initiate because the LL No. 3 signal did not seal in. Emergency ac diesel generator (D/G) Nos. 1-4 auto-started and D/G No. 4 tripped. At 2332, the unit auto-scrammed due to high reactor pressure. A scram recovery was carried out. Reactor safety relief valve A was manually opened to control reactor pressure. The RCIC and Reactor Condensate Feedwater Systems were utilized to control reactor level.

The incurred low level signals resulted from reactor level instrumentation falsely sensing low reactor levels when a pressure spike occurred on the instruments' common reference leg due to opening the instrument drain valve of reactor pressure instrument C32-PT-N008.

The involved person was appropriately disciplined. Problems affecting operability of D/G No. 4 and RCIC were resolved and they were returned to service.

IE22 #1

Initial Conditions

Unit 1 was in startup at approximately 6 percent power for testing following a unit refueling/maintenance outage.

Event Description

On November 2, 1985, at 2315, an Auxiliary Operator (AO) was dispatched to the Unit 1 Reactor Building to investigate unexpected downscale indication shown by the reactor pressure narrow range indicator/recorder, C32-R609, located on the unit reactor turbine gauge board (RTGB). The AO performed a valve alignment verification of the R609 transmitter, C32-PT-N008. While checking the position of the PT-N008 instrument drain valve, C32-N008-6, he turned the valve handle approximately 1/8 turn in the open direction and immediately reclosed the valve after hearing flow noises. The time of this occurrence was approximately 2326 on November 2, 1985.

At 2326, Unit 1 primary containment Groups 1, 3, and 6 isolations occurred due to a reactor low level (LL) No. 2 signal. The High Pressure Coolant Injection (HPCI) System automatically started but did not inject and the Reactor Core Isolation Cooling (RCIC) System automatically started and tripped shortly thereafter. The reactor recirculation pumps tripped. The Reactor Building Ventilation System automatically isolated and the Standby Gas Treatment (SBGT) System trains A and B automatically started. A momentary LL No. 3 signal occurred but did not seal in, starting the Core Spray subsystems A and B; however, they did not inject due to high reactor pressure. Emergency ac diesel generator (D/G) Nos. 1-3 automatically started. D/G No. 4, upon receipt of the automatic start signal, began to roll during the diesel starting sequence but tripped and locked out. The lockout signal was subsequently reset and the diesel was successfully started approximately 20 minutes after the incurred trip and lockout. D/G Nos. 1-3 did not tie onto their respective emergency electrical buses as the normal power sources for the buses remained available. Attempts were begun to reset the incurred Group 1 isolation signal and reopen the reactor main steam isolation valves (MSIVs) by equalizing pressure around the valves. The Core Spray subsystems and the HPCI System were secured and returned to standby. Reactor pressure increased and an automatic reactor scram occurred due to high reactor pressure before the MSIVs could be reopened.

A reactor scram recovery was carried out. The RCIC System was subsequently placed into service and utilized for reactor level control. Reactor level was maintained at a minimum of 174 inches to a maximum of 199.5 inches. Reactor pressure, with a maximum recorded pressure of 1050 psig, was controlled by manual opening of reactor safety/relief valve B21-F013A. The scram signal was reset, the D/Gs were secured and returned to standby, the MSIVs were reopened, and the RCIC System was secured and returned to standby. Reactor level was maintained by use of the Reactor Condensate/Feedwater Systems

steam-driven feed pump B. The SBGT trains were secured and returned to standby, and the Reactor Building Ventilation System was returned to service.

Event Investigation

Following the reactor scram recovery, investigations were conducted to determine the cause of the incurred LL Nos. 2 and 3 signals and the trips of the RCIC System and D/G No. 4. In addition, the absence of a Residual Heat Removal (RHR) System low pressure cooling injection (LPCI) mode initiation, due to the LL No. 3 signal, was assessed to determine whether LPCI initiation logic was functioning properly.

(LL Nos. 2 and 3 Signals)

The LL Nos. 2 and 3 signals resulted from actuation of reactor level instruments B21-LT-N025A-1 and A-2, B21-LT-N025B-1 and B-2, and B21-LT-N031B and D. These instruments share a common reference sensing leg with C32-PT-N008. When the position of the N008-6 drain valve of PT-N008 was checked, a slight pressure spike occurred in the reference sensing leg due to reclosing the valve. The subject reactor level instruments detected false low level conditions and consequently initiated the LL Nos. 2 and 3 signals. The false LL No. 3 condition was not sensed for a sufficient time to seal in the LL No. 3 initiation signal to the RHR LPCI mode instrumentation. On November 3, 1985, a subsequent operability check of the RHR System LPCI initiation logic showed the logic was functioning as required.

(RCIC System Trip)

During the unit scram recovery, it was noted the RCIC System trip occurred without accompanying RTGB alarm annunciation. It was suspected the trip latch mechanism of the RCIC turbine trip and throttle valve, E51-V8, may have inadvertently released while the turbine was operating at rated speed following the automatic initiation on low reactor level. As a result, V-8 closed causing the turbine to shut down. This conclusion is supported due to the fact that no overspeed trip reset of the turbine was required prior to restarting the turbine. The V-8 trip latch mechanism and trip initiation coil were inspected for irregularities which may have led to a spurious release of the V-8 trip latch mechanism (Schutte-Koeting Drawing No. 69-XE-4h). The full-closed limit switch (EIIS/BN/33) for the motor operator (EIIS/BN/84) on V-8 was found set close to the latch point for valve reset. The switch was adjusted to ensure more positive latch actuating would occur before position indication changed. It could not be determined if this close setting affected this event, but the possibility does exist. No other problems were found which could be associated with the suspected cause of the V-8 closure. Following a satisfactory completion of an operability test, the RCIC System was returned to service.

The inadvertent trip of diesel generator No. 4 is attributed to a low lube oil pressure lockout of the diesel. A calibration check of the actuation time setpoint of the diesel start time delay (STR) relay, 2-DG4-STR, found the relay actuating in 26.81 seconds. The specified setpoint for actuation of the relay is 30.0 ± 4 seconds. DG4-STR functions to permit bypassing the diesel low lube oil pressure lockout logic during the initial 30-second starting time frame so normal lube oil pressure may be achieved. During the initial 30 seconds of starting sequence, actual lube oil pressure is sensed by lube oil pressure switch DG-PS-6534-4, which will actuate when pressure reaches 27 ± 1 psig preventing DG4-STR relay circuit from locking out the D/G after the 30-second time delay. Investigation of the diesel trip and lockout revealed the diesel lube oil cooler temperature control valve, 2-LO-TCV-2100, was in the fully closed position. The temperature control valve is adjusted during operation of the diesel to help maintain proper lube oil temperatures. In the fully closed position, the majority of the lube oil is sent through the lube oil cooler. Sending the oil through the cooler will develop a higher system pressure drop than when bypassing the cooler. Jacket water is utilized to cool the lube oil. If jacket water temperatures are not maintained high enough, the lube oil passing through the cooler will be overcooled, increasing the viscosity of the oil. If the viscosity of the oil gets too high, satisfying pressure switch 6534-4 in 30 seconds can be impossible. Troubleshooting of the D/G No. 4 lockout was performed and resolutions presented to PNSC via letter dated January 30, 1986, from J. O'Sullivan. This letter concluded that the failure of the No. 4 diesel was directly attributed to the following:

1. Failure to maintain proper lube oil and jacket water temperatures in the shutdown condition.

2. The position of the temperature control valve has a significant effect on pressure drops, especially at lower lube oil temperatures.

3. Low STR relay setpoint due to setpoint drift.

During the troubleshooting activities, it was noted that the pressure sensing line to pressure switch DG-PS-6534-4 was found to have a leaking connection. It is suspected to have allowed air to enter the line. If air did enter the line and the leak was present at the time of the D/G start, an increase in response time for the pressure switch would result.

Both the jacket water and lube oil systems have heaters that maintain the fluids around 155 degrees (F), when in standby. If jacket water passing through the lube oil cooler is maintained around 155 degrees and lube oil temperatures are also maintained in the acceptable range, the position of the lube oil temperature control valve will not cause a low lube oil pressure lockout 30 seconds after a diesel start. The temperature switches (MUD-TS-6549 and LO-TS-6575) for both the lube oil and jacket water systems were not on any calibration program at the time of the incident.

An investigation was performed to determine if the lube oil and jacket water temperatures on the other three diesels were low enough to cause a diesel lockout on low lube oil pressure. The results of the testing revealed that all three remaining diesels satisfied the pressure requirement during a diesel start in under 16 seconds.

It has been concluded that the remaining diesels could have been susceptible to low lube oil pressure lockouts only if lube oil and jacket water temperatures were not maintained high enough while in standby, combined with the lube oil temperature control valve in the manual closed position, and drifting of the STR relay downscale. (A leak in the instrument line will also impact the results.) Since the D/Gs are tested monthly to ensure operability, this is not considered a feasible scenario and, therefore, not reportable per 10CFR50.73(a)(2)(v).

On November 4, 1985, following the satisfactory completion of the diesel generator operability test, diesel generator No. 4 was returned to service. An Engineering Work Request was initiated to evaluate relocating the lube oil pressure switch closer to the engine to improve response times. This request was evaluated and determined not to be the most desirable alternative. It was determined that increasing the time delay of the diesel generator lockout from 30 to 45 seconds would be a better solution. A prior similar event involving the 6534-2 lube oil pressure switch of D/G No. 2 was reported in LER 1-82-125 on January 21, 1983.

An investigation was also conducted to determine the cause of the indication problem with the R609 reactor pressure indicator/recorder. It revealed the incurred downscale indication resulted from a blown fuse in the PT-N008 power supply, 1-C32-K620, General Electric Co. Model No. 570-06. The fuse was replaced, PT-N008 was calibrated, and R609 was returned to service.

Corrective Actions

The AO who initiated the event was appropriately disciplined. In addition, follow-up meetings with shift personnel were held by the Manager - Operations and Real-Time Training personnel to review the event and reemphasize the requirements for proper valve manipulations.

The blown power supply fuse to PT-N008 was replaced and R609 was returned to service.

Diesel generator operating and test procedures have been revised to include the manual positioning of the temperature control valves during diesel operation if and only if the valves will not perform the function automatically. The operating procedure (OP-39) valve lineup has also been revised to ensure the lube oil TCVs are put in the thermostatic position. In addition, PM 85-123 increased the time delay period of the diesel generator lockout for low lube oil pressure from 30 to 45 seconds. The calibration

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

frequency of pressure switch PS-6534 on all four diesel generators has been increased from every 18 months to every 6 months to prevent large setpoint drifts. The thermostats for the lube oil and jacket water heaters were placed on a yearly calibration schedule.

Event Assessment

The consequences of this event under other reasonable and credible alternative conditions would not have been more severe.

CP&L

Carolina Power & Light Company

Brunswick Steam Electric Plant
P. O. Box 10429
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~~July~~ 27, 1988
June

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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 1-85-059

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. G. Bisley for

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

MJP/srg

Enclosure

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

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