CP&L

Carolina Power & Light Company

Brunswick Nuclear Project
P. O. Box 10429
Southport, NC 28461-0429
November 6, 1989

FILE: B09-13510C
SERIAL: BSEP/89-0997

10CFR50.73

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

BRUNSWICK STEAM ELECTRIC PLANT UNIT 2
DOCKET NO. 50-324
LICENSE NO. DPR-62
LICENSEE EVENT REPORT 2-89-017

Gentlemen:

In accordance with Title 10 to 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 in accordance with the format set forth in NUREG-1022, September 1983.

Very truly yours,

J. L. Harness, General Manager
Brunswick Nuclear Project

MJP/mcg

Enclosure

cc: Mr. S. D. Ebneter
Mr. E. G. Tourigny
BSEP NRC Resident Office

IE22
1/1

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 2	0 5 0 0 0 3 2 4	1 OF 0 7

TITLE (4) Unplanned Actuation of Reactor Low Level Instrumentation Due to Suspected Perturbation of Reactor Instrumentation Lines While Returning C32-PT-N008 to Service Following Calibration

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

OPERATING MODE (9)	5	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)					
POWER LEVEL (10)	0 0 0	20.402(b)	20.405(c)	X	50.73(a)(2)(iv)	73.71(b)	
		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 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

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 1525 hours on 10/10/89 inadvertent Unit 2 low water level (LL) No. 2 and 3 signals occurred causing a Group 1 isolation, Core Spray (CS) initiation signal, and auto-starting of the Units 1 and 2 common emergency diesel generators (DGs). The Unit 2 1989-1990 refuel/maintenance outage was ongoing with the Reactor (Rx) defueled. The CS, Residual Heat Removal (RHR), Rx Water Cleanup (RWCU), Standby Gas Treatment, and the Rx Building ventilation systems were under equipment clearance. Return to service of Rx feedwater/Rx pressure transmitter (PT) C32-PT-N008 was ongoing. The Control Operator became aware of this event through Control Room indication/alarm annunciation. An RHR/Low Pressure Coolant Injection (LPCI) initiation signal and a Group 3 of the RWCU System were not confirmed. The isolation/initiation signals were reset and by 1818 hours the DGs were returned to standby. This event had minimal safety significance.

The LL signals are attributed to perturbation of the instrument sensing leg common to N008 and Rx LL instrumentation when N008 was returned to service, per OPIC-PT001. An instruction/precautionary statement was not provided with the procedure to ensure the test pressure on N008 was equal to the system pressure conditions prior to return to service.

In addition to OPIC-PT001, two similar procedural problems will be resolved by 12/20/89. Prior to subsequent startup, appropriate testing will assure operability of the RHR/LPCI and RWCU Group 3 functions.

Event

Inadvertent Reactor low water level No. 2 and 3 signals caused a Unit 2 Primary Containment (PC) Group 1 isolation of Reactor water sample PC outboard isolation valve 2-B32-F020, Reactor Core Spray initiation signal, and automatic starting of the Units 1 and 2 common emergency alternating current diesel generators Nos. 1-4. The low level signal is attributed to unplanned actuation of Reactor level instrumentation due to a suspected perturbation of Reactor instrumentation lines during the return of Reactor feedwater/Reactor pressure instrument C32-PT-N008 to service following routine instrumentation calibration.

Initial Conditions

The Unit 2 1989-1990 refueling/maintenance outage was in progress with the Reactor (EIIS/AC) defueled and the Reactor cavity (EIIS/DF/**) flooded. The Reactor Core Spray System (EIIS/BM), Residual Heat Removal (RHR) (E11) System (EIIS/BO), Reactor Water Cleanup (RWCU) System (G31) (EIIS/CE), Standby Gas Treatment System (EIIS/BH), and the Reactor Building Ventilation System (EIIS/VA) were under equipment clearance for ongoing outage activities. In addition, the Units 1 and 2 common emergency alternating current (AC) diesel generators (DGs) (EIIS/EK/DG) were in standby readiness and normal off-site power was available. Maintenance technicians were in the process of returning Reactor feedwater (C32) (EIIS/SJ)/Reactor pressure transmitter (PT), 2-C32-PT-N008 (EIIS/SJ/PT) to service following a periodic preventative maintenance calibration of the instrument in accordance with the process instrument calibration (PIC) of General Electric type 551 and 552 PTs, OPIC-PT001.

**EIIS component description unavailable

Event Description

At 1525 hours on October 10, 1989, when the technicians opened the instrument isolation valve, 2-C32-PT-N008-3 (EIIS/SJ/ISV) in accordance with Step 7.4.3 of the procedure, Reactor low water level Nos. 2 and 3 signals occurred. These low level signals initiated a Primary Containment Group 1 isolation signal, a Core Spray initiation signal, and automatic starting of the units' emergency AC DGs. Per design, the DGs did not tie onto their respective emergency (E) buses, E1-E4 (EIIS/EK/BU) as normal off-site power remained available. The Unit 2 Control Operator (CO) became aware of this event through appropriate Control Room indication and alarm annunciation. Following this event it was verified that the Reactor water sample outboard primary containment isolation valve (PCIV), 2-B32-F020 (EIIS/AC/ISV), which had been opened prior to the event, had closed as required due to the Group 1 signal. At 1533 hours, following verification that the Reactor low level alarm annunciations were false, the isolation/initiation signals were reset. In addition, the DGs were

secured from operation and returned to standby readiness (DGs Nos. 1 and 2 at 1610 hours) (DGs Nos. 3 and 4 at 1818 hours). It could not be determined if the RHR/LPCI signal had been received at the time of the event. In addition, as the RWCU System was under equipment clearance with the System inlet outboard PCIV, 2-G31-F004 (EIIS/CE/ISV) closed, visual confirmation of the required closure signal to the valve, due to the incurred low level No. 2, was unavailable.

Event Investigation

Following the event, interviews with the involved maintenance technicians were conducted in order to reconstruct their actions relative to the event. Based upon these interviews, it was concluded that the technicians had properly adhered to the procedure. Therefore, personnel error by the involved technicians was ruled out as a root cause or contributing factor to the event.

The procedure isolates the instrument from its respective instrument sensing line (EIIS/SJ/PSX) through closure of the instrument isolation valve, 2-C32-PT-N008-3 (EIIS/SJ/ISV), in order to perform a five point (0%, 25%, 50%, 75%, and 100%) calibration of the instrument at corresponding applied test pressures ranging from 814-1114 pounds per square inch gauge (psig). At the time of the event, with only the preexisting static hydraulic head pressure on the instrument sensing line, the instrument had been satisfactorily calibrated following application of the 1114 psig test pressure and the technicians were in the process of reopening 2-C32-PT-N008-3 in order to return the instrument to service in accordance with Step 7.4.3 of the procedure.

An adequacy review of OPIC-PT001 determined the procedure did not provide for the contingency of valving in the transmitter to a depressurized reference leg. Consequently, when 2-C32-PT-N008 was valved back into service the calibration test pressure was applied to the already depressurized instrument sensing leg.

The investigation concluded the low level signals resulted from actuation of Reactor low level instrumentation on the common referencing leg with 2-C32-PT-N008, due to a momentary perturbation of the leg when the instrument was returned to service. Instrumentation, which shares this common reference sensing leg, is listed in Table 1.

During a review of plant documentation to examine the applicability of prior similar occurrences to this event, the appropriateness of corrective actions for LER 1-87-017 as they relate to this event were evaluated. The event reported in LER 1-87-017, which occurred on June 12, 1987, involved valving-in depressurized test equipment to the pressurized instrument reference leg common to 2-C32-PT-N008. The investigation of the event in 1987 determined it had resulted from a momentary perturbation of the reference leg which caused an automatic Reactor Scram due to actuation of the affected Reactor level instrumentation. As part of corrective action to the event, a review/revision was conducted of plant procedures involving valving-in instruments on a common reference sensing leg. This effort was focused to ensure the instruments are properly pressurized prior to valving-in during Reactor power operation. Reexamination of the approach taken in achieving the corrective actions of LER 1-87-017 has determined the actions did not address restoration to a depressurized reference leg, in that only procedures for valving-in devices to a pressurized Reactor System were considered.

The investigation determined that the most probable reason why an RHR/LPCI initiation signal was not received as a result of this event is the suspected relatively short duration of the Reactor low level signal, which did not allow both the RHR Reactor low level channel relays, 2-E11-K7B and K8B (EIIS/BO/RLY), and the RHR/LPCI seal-in relay, 2-E11-K9B (EIIS/BO/RLY), to energize for a sufficient time to allow the normally open K9B seal-in contacts to close.

Action Taken

The failure to review procedures in order to provide for return to service condition regardless of Reactor pressure, should be corrected through improved root cause/corrective action determinations at the Brunswick station. The corrective action program is viewed as an appropriate measure to ensure that adequate attention is devoted to ensure the proper identification and focus of corrective action.

As a result of this event, a review was conducted to identify Reactor instrumentation and maintenance procedures involving instrument sensing lines which are sensitive to pressure perturbations, in order to ensure adequate procedural controls are provided regarding a pressurized or depressurized Reactor System.

This effort revealed two additional instrumentation calibration instructions were inadequate to ensure the Reactor System pressure is taken into account during return of the involved instruments to service.

Actions to be Taken

By December 20, 1989, OPIC-PT001, as well as the two additional procedures identified during the subject procedural review, will be appropriately revised to provide for consideration of the state of the Reactor System (depressurized or pressurized) during valving-in of instrumentation or other test equipment/ devices.

In addition, prior to startup of Unit 2 from the current refueling/maintenance outage, appropriate testing will be performed to ensure operability of the RHR/LPCI actuation instrumentation. It is felt the Group 3 isolation capability of the RWCU System will be verified through postmaintenance testing requirements which follow the cancellation of the involved equipment clearance on the system.

Event Assessment

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

In addition to LER 1-87-017, prior events involving instrument perturbations while valving-in instrumentation have been reported in LERs 1-85-039, 1-85-047, and 2-86-020.

TABLE 1

Instrument No.	Function
B21-LT-N025A-1 (EIIS/SB/LT)	Initiates signal for closure of main steam isolation valves (MSIVs) (EIIS/SB/ISV), 2-B32-F020, main steam line drain outboard PCIV 2-B21-F019 (EIIS/SB/ISV), RWCU System inlet outboard PCIV, G31-F004 (EIIS/CE/ISV), isolation of the Reactor Building Ventilation System, and automatic starting of the SBGT trains A and B (EIIS/BH/**).
B21-LT-N025B-1 (EIIS/SB/LT)	Initiates signal for closure of MSIVs, 2-B32-F020, 2-B21-F019, G31-F004, isolation of the Reactor Building Ventilation System, and automatic starting of the SBGT Trains A and B.
B21-LT-N025A-2 (EIIS/SB/LT)	Initiates an anticipated transient without a scram (ATWS) (EIIS/JC) trip of reactor recirculation pump A (EIIS/AD/P).
B21-LT-N025B-2 (EIIS/SB/LT)	Initiates an ATWS trip of reactor recirculation pump B (EIIS/AD/P).
B21-LT-N031B (EIIS/SB/LT)	Initiates automatic initiation of the Reactor Core Isolation Cooling (RCIC) System (EIIS/BN), permissive for actuation of the Automatic Depressurization (ADS) System (EIIS/*), initiation of the RHR low pressure coolant injection (LPCI) mode, Reactor Core Spray System, and automatic starting of units' common emergency AC DGs.
B21-LT-N031D (EIIS/SB/LT)	Initiates RCIC System automatic initiation, permissive for actuation of the ADS, initiation of the RHR/LPCI mode, Reactor Core Spray System, and automatic starting of the units' common emergency AC DGs.
B21-PT-N021B *** (EIIS/SB/PT)	Provides 410 psig Core Spray injection permissive as well as permissive for the 310 psig Reactor Recirculation pump discharge valve closure.

TABLE 1 (Cont'd)

B21-PT-N023C *** (EIIS/SB/PT)	Initiates the Reactor Vessel High pressure automatic scram
B21-PT-N023D *** (EIIS/SB/PT)	Initiates the Reactor Vessel High pressure automatic scram

*EIIS system description not available

**EIIS component description not available

***It is believed the magnitude of the incurred sensing leg perturbation was not sufficient to have affected this instrument.