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 4

TITLE (4) Control Rod 10-39 Unknowingly Withdrawn With Reactor Protection System Shorting Links Instalied

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 3	0 8	8 8	8 8	0 0 6	0 0	0 4	0 6	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5
POWER LEVEL (10): 0 0 0

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

20.402(b)	20.405(c)	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)	X 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)

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

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

During the Unit 2 1988 refuel/maintenance outage, with the reactor core reload complete, the reactor head removed, and the reactor cavity flooded, from 0350 hours on 3/8/88 until event discovery at 1948 hours on 3/8/88, control rod 10-39 was unknowingly in the withdrawn position with the Reactor Protection System shorting links installed. This condition constituted a violation of Technical Specification Table 3.3.1-1, which requires removal of the shorting links prior to and during removal of any control rod while in the refuel mode. Following verification of actual rod position at 2014 hours, the rod was inserted (manually scrammed) at 2052 hours on 3/8/88. During this event, no activities were in progress involving work over the reactor core. This event had minimal impact upon the safety of the unit.

Following control rod withdrawal for testing, the operator failed to reinsert the rod as required by procedure. Subsequent verification of rod position was not adequate to detect that the rod was not full in.

Involved personnel have been counseled to be constantly aware of the plant status. A standing instruction has been issued alerting Operations personnel to recent changes to process computer software, and appropriate procedure revisions will be implemented to specify the method of control rod position verification and to provide for the use of sign-offs for documentation of rod position.

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Initial Condit. ons

The Unit 2 1988 refueling/maintenance outage was in progress with the reactor core reload complete, the reactor head removed, and the reactor cavity flooded. No activities were in progress involving work over the reactor core.

Event Description

From 0350 hours until 1948 hours on March 8, 1988, control rod 10-39 (EIIS/AA/ROD) was unknowingly in the fully withdrawn position and the Reactor Protection System (RPS) shorting links (EIIS/JC/CBL) installed with the reactor in mode 5 (refuel). This constituted a violation of table notation b of Technical Specification Table 3.3.1-1 which requires the RPS shorting links be removed from the RPS circuitry prior to and during the time any control rod is withdrawn while in reactor mode 5. Removal of the shorting link allows any single RPS trip signal to initiate a full RPS trip (reactor scram).

At 2025 hours on March 7, 1988, the RPS shorting links were removed to permit performance of Control Rod Operability Check Periodic Test (PT)-14.1 and Control Rod Coupling Check and Control Rod Drive (CRD) (EIIS/AA) Testing PT-14.1A. CRD movement, required by the tests, was completed at 0029 hours on March 8, 1988; however, control rod 10-39 remained at position 48 (fully withdrawn). The shorting links were installed at 0350. The tests, which were completed at 0505 hours on March 8, 1988, were performed to meet postmaintenance testing requirements (PMTRs) following repairs to various control rod position indication probes (PIPs) (EIIS/JD/33).

Checks of control rod positions at 0037 and 0215 hours using the main process computer on demand (OD-7) new scan (Option 2) of control rod position (OD-7 Option 2) had shown the control rods were inserted (00 position), with the exception of rod 10-39, which showed a blank position indication. A subsequent OD-7 Option 2 performed at 0534 hours again indicated a blank position indication for rod 10-39.

At 0640 hours and again at 0720 hours by the CO of the day shift, the OD-7 Option 2 was performed with rod 10-39 showing a blank indication for position. At approximately 0900 hours, an equipment clearance was placed on the reactor Rod Worth Minimizer (RWM) (EIIS/JD/MON) to allow performance of a plant modification. This prevented selection of any rods via the Reactor Manual Control System (RMCS) (EIIS/JD).

Rod position is frequently verified by utilizing the process computer's on demand Option 7-2 which prints a matrix of rod positions. The matrix shows the rod position pattern on the full core display with the position printed denoting the actual rod position (there are 49 reed switches over the normal range of rod travel). A recent modification of the Unit 2 computer software in December 1987 deleted position 48 from printing out. A rod at 48 shows up as a blank versus a number. This is desirable when operating as it provides a less cluttered printout (many rods are at 48 when at power) for the Nuclear Engineers when assessing control rod changes at power. New options, OD 7-3 and 7-4, were created for Unit 2 at the same time, which provide printouts of rod positions including position 48. The Unit 1 process computer software has not yet been modified to provide the OD 7-3 and 7-4 options. The Unit 1 OD 7-2 can be manipulated by the Nuclear Engineers so as not to print position 48, which is done normally while at power.

In addition to the process computer, the full core display has a full in (green) and full out (red) indication for each of 137 control rods. A rod fully out during shutdown is very prominent as a single red light is easily distinguished. In this case rod 10-39, which was full out, had a defective full out indicator. Also, the involved Operations personnel failed to notice that the full in indicating light for rod 10-39 was not on.

While performing control rod timing in accordance PT-14.1, the control rod is withdrawn to measure the time and determine adjustments. Following correct adjustment and timing, the rod is then inserted. In this case, the operator failed to reinsert the last control rod which was timed (10-39). Although the CO ran a process computer option to printout rod position and verify all rods were inserted, the CO and the Shift Foreman failed to properly interpret the printout and consequentially believed all rods were fully inserted.

At 1948 hours on March 8, 1988, the CO on the 1900-0700 shift performed an OD-7 Option 2 and recognized that rod 10-39 was not fully inserted (due to no 00 indication). Following this discovery, the SRO directed that an OD-7 Option 3 or 4 be performed to determine actual position of the rod. This was done and at 2014 hours, it was confirmed that the rod was fully withdrawn.

As control rods could not be selected from the RMCS due to an equipment clearance on the RWM, rod 10-39 was manually scrammed using the RPS Select Rod Insert System control switch (EIIS/JD/33) for the rod.

<u>Event Investigation</u>

The initial failure to reinsert the control rod is believed to have occurred when the CO was distracted from completing the final step of the procedure - the last rod timed (10-39). This is a personnel error due to inattention. Although the operator initiated a computer printout to verify all rods were in, he misinterpreted the printout. Subsequent failure to

interpret the process computer printout is attributed to inadequate training on the computer options and to the operator's inattention. A contributing factor in not detecting the rod full out was the failed full out indicator on the RTGB.

The "full-out" position red light indicator for the rod had previously been identified on December 15, 1987, as being inoperable and, in accordance with Technical Specification 3.1.3.7, a daily surveillance to verify actual position of the rod had been established in the CO Daily Surveillance Report (DSR). In addition, inoperability of the "full-out" position light for rod 10-39 had been documented in the performances of PTs-14.1 and 14.1A on March 7.

A significant factor in this event was reliance of the CO upon the OD-7 Option 2 function despite the fact this function only provides verification for rods that are not fully withdrawn. Discovery of this event was facilitated by awareness by the involved SRO of a modification installed on the Unit 2 process computer on December 30, 1987, which provides for OD-7 Options 3 and 4. Prior to this event, Operations personnel had not received training concerning use of these functions to verify "full-out" position.

Corrective Actions

On March 18, 1988, the "full-out" red light position indication for control rod 10-39 was repaired.

As a result of this event, involved Operations personnel who failed to detect this event have been counseled to be aware of the need for constant attention to detail. A standing instruction has been issued to alert personnel to changes in software and the potential for missing position 48. In addition, PTs-14.1 and 14.1A will be revised to specify the method of rod position verification, including reference to use of plant process computer options and to include sign-offs that control rods are fully inserted. The process of software modification will be reviewed to evaluate changes necessary to ensure formal operator training requirements are identified.

Prior to the next Unit 1 refueling/maintenance outage, the unit process computer software will be modified to provide the OD 7-3 and 7-4 options for determining control rod position.

Event Assessment

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

Prior events, involving a failure of Operations personnel to recognize a plant condition and take appropriate action, have been reported in LERs 1-84-005, 2-84-005, 1-85-009, and 2-87-010.

CP&L

Carolina Power & Light Company

Brunswick Steam Electric Plant
P. O. Box 10429
Southport, NC 28461-0429
April 6, 1988

FILE: B09-13510C
SERIAL: BSEP/88-0388

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 2-88-006

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,

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

MJP/cjd

Enclosure

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

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ATTACHMENT (1)

System	Weight Stress (1)	Longitudinal Pressure Stress (2)	OBE Stress (3)	DBE Stress (4)	Total Stress (1 + 2 + 4)	Allowable Stress for OBE (5)	Allowable Stress for DBE (6)	Additional Stress Due to Postulated Setpoint Error (7)	Percent Damping (8)
Safety Injection (USAS/ANSI B31.7 - 1969)	578	4971	6525	12234	17783	18664	22700	100	0.5
Boric Acid Pump Suction and Discharge to Charging System (USAS/ANSI B31.7 - 1969)	7001	192	2028	3803	10996	19860	28250	79	0.5
Saltwater (USAS/ANSI B31.1 - 1967)	5023	963	1551	2908	8894	18840	37680	866	5.0
Service Water (USAS/ANSI B31.1 - 1967)	556	729	3945	7397	8682	18000	31783	386.5	0.5
Component Cooling Water (USAS/ANSI B31.1 - 1967)	4393	431	8251	15471	20295	18000	34070	298	0.5

Note: All stresses listed in (psi)