NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

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 3

TITLE (4) Inoperability of Reactor Core Spray and Residual Heat Removal Low Pressure Coolant Injection Systems During Periodic Testing

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)
1 0	2 9	8 5	8 5	— 0 5 8 —	0 0	1 1	2 7	8 5	Brunswick Unit 2	0 5 0 0 0 3 2 4
										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)	[X] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[X] 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
M. J. Pastva, Jr., Regulatory Technician	AREA CODE 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)

On 10/29/85, a determination was made that during performance of the Core Spray Simulated Automatic Actuation and Logic Functional Test, Periodic Test (PT) 07.1.9, both loops of the low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System were rendered incapable of reactor injection should receipt of an actual LPCI initiation signal occur. The PT is used to satisfy requirements of Technical Specification 4.3.3.2. The procedural problem applied to Units 1 and 2 and was initially discovered during a technical review of the PT. Unit 1 was in a refuel/maintenance outage and Unit 2 was at 90 percent power.

The procedural problem resulted from a step in the procedure which deenergized relays in the reactor low pressure permissive logic to core spray initiation instrumentation. It was not recognized this also prevented the RHR LPCI initiation logic from sensing reactor pressure. The cause of this oversight is attributed to inadequate technical review during prior revisions to the PT.

The PT procedure for Unit 1 was appropriately revised to ensure LPCI operability during performance of the testing. The respective Unit 2 procedure is currently being revised in a likewise manner. By 1/31/86, a procedure developed by the ongoing Maintenance Surveillance Test Rewrite program will be implemented to replace the PT on each unit with a procedure which only affects operability of the core spray loop under test.

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☆U.S.GPO:1984-0-454-451/18759

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Brunswick Steam Electric Plant Unit 1	0 5 0 0 0 3 2 5	8 5	— 0 5 8 —	0 0	0 2 OF 0 3

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

On October 29, 1985, it was determined that performance of the Core Spray Simulated Automatic Actuation and Logic Functional Test, Periodic Test (PT) 07.1.9, rendered both loops of the low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System incapable of reactor injection should receipt of an actual LPCI initiation signal occur. PT-07.1.9 is used to satisfy the requirements of Technical Specification 4.3.3.2 for a logic system functional test of the Core Spray System to be performed at least once per 18 months. The procedural problem was determined when an attempt was made to gain permission from Operations for a scheduled performance of the PT for Unit 1 on October 24, 1985. The unit Shift Foreman requested a determination of whether both Core Spray System loops (A and B) are rendered inoperable by performance of the PT. Further review of the procedure by personnel of the Maintenance Surveillance Test (MST) Rewrite program revealed that, in addition to both Core Spray System loops being rendered inoperable, both RHR System LPCI loops were also rendered inoperable for the duration of the test performance. Prior to this determination, it was believed that only the Core Spray System loops were affected by performance of the PT. An appropriately revised procedure was then generated for the Unit 1 scheduled performance. The procedural problem affecting PT-07.1.9, which applies to both units, was discovered while Unit 1 was in a refueling/maintenance outage and Unit 2 was operating at 90 percent power.

The test procedure (PT-07.1.9) lifted leads which deenergized relays in the reactor low pressure permissive logic to the Core Spray System initiation logic. This was intended to simulate a high reactor pressure so that the reactor low pressure permissive circuitry could be thoroughly tested in this procedure. However, it was not recognized that deenergization of the subject relays also inhibits the reactor low pressure permissive logic to the opening logic of the LPCI loops' injection valves, E11-F015 and F017. A review of prior revisions to PT-07.1.9 showed this condition existed during earlier performances of the test on each unit. The inadequacy of PT-07.1.9 is attributed to inadequate technical review during prior revisions of the PT.

An assessment of possible negative effects and consequences resulting from performance of the PT was conducted. It was determined that the severity of possible consequences resulting from performance of the test can increase significantly if other conditions occur during the test, which necessitate the use of the RHR System to maintain adequate core cooling and/or reactor inventory. A feasible situation would involve an inadvertent loss of RHR reactor shutdown cooling. Such an event could be initiated from various valid or spurious trip signals (i.e., a primary containment Group 8 isolation) which would initiate an automatic trip of the RHR pumps. Due to the configuration of the LPCI initiation logic resulting from performance of the PT, reestablishment of RHR reactor shutdown cooling would be prohibited. Only the realignment of the circuitry affected by this PT back to normal would allow a restart of the RHR pumps for shutdown cooling. Another feasible situation which could possibly occur during performance of PT-07.1.9 is a small-break reactor LOCA while in the shutdown cooling mode of RHR System operation. In this case, timely Control Operator response would mitigate the situation. The Control

Operator would be required to realign the suction supply source of the RHR pumps from the reactor to the suppression pool before reactor level decreased to where an automatic isolation of the LPCI injection valves occurred. If not, LPCI injection would be possible only after restoring normal alignment of subject reactor pressure permissive instrumentation affected by performance of the PT.

Although performance of PT-07.1.9 effectively negates analyzed protective sequences outlined in the plant Final Safety Analysis Report, the actual response to the aforementioned situations would have depended on the availability of alternate flow paths for reactor cooling and inventory reestablishment. A review of plant conditions during the last four performances of PT-07.1.9 has disclosed that, at least once, the unit undergoing testing was depending on RHR shutdown cooling alone to remove decay heat from the vessel. However, this is not the typical case, as at another time, the test was performed while the Reactor Condensate System and Main Turbine Condenser Cooling System were aligned to utilize the main condenser directly for removal of decay heat. Other performances of this test have included a run where reactor decay heat generation was low enough to allow securing the reactor shutdown cooling for the duration of the test and another performance that was done when the vessel was flooded up, fuel pool gates were removed, and the torus was drained (i.e., ECCS was not required).

With normal availability of the Reactor Condensate/Feedwater System flow paths to the vessel, concern for maintenance of reactor level is minimized as this system can readily be aligned to inject to the reactor. In addition, other systems with the capability of reactor injection can be aligned as required. These systems include the Control Rod Drive System and the Standby Liquid Control System.

Following the determination of the subject procedural deficiency, the respective procedure for Unit 1 was appropriately revised to ensure operability of both RHR System LPCI loops during performance of the PT. The respective Unit 2 procedure is currently in a similar revision process.

Procedures have already been developed as part of the ongoing Maintenance Surveillance Test Rewrite program to replace PT-07.1.9 with others that better perform the required subject testing. These procedures will only affect the operability of the Core Spray System loop under test. Expected implementation of these procedures is January 31, 1986.

CP&L

Carolina Power & Light Company

Brunswick Steam Electric Plant
P. O. Box 10429
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November 27, 1985

FILE: B09-13510C
SERIAL: BSEP/85-1976

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

BRUNSWICK STEAM ELECTRIC PLANT UNIT 1
DOCKET NO. 50-325
LICENSE NO. DPR-71
LICENSEE EVENT REPORT 1-85-058

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/ag

Enclosure

cc: Dr. J. N. Grace

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