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 5

TITLE (4): Misconfiguration of Transversing Incore Probe (TIP) System

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

OPERATING MODE (9): 1

POWER LEVEL (10): 1 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)	[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)	Voluntary Report
[ ] 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 Technician	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 7/22/86, it was predicted the Unit 2 reactor Transversing Incore Probe (TIP) System Channel B, 8 and 10 tubing pathways (B-8 and B-10), were interchanged at the channel indexing mechanism. Unit 2 was at 100% with a 6% margin to reactor thermal limits. Following this discovery, reactor power was reduced to achieve an approximate 10% margin of reactor thermal limits.

The TIP misconfiguration is attributed to misidentification of the tubing which resulted in it being improperly connected. This occurred during the unit 1984 refuel/maintenance outage when the tubing was reconnected following temporary removal to permit preventive maintenance.

Software changes were made to the plant process computer to redirect the incoming signals from the B-8 and B-10 channels to ensure the core was being properly monitored. Appropriate procedural revisions have and will be taken concerning the event. Following proper reconfiguration of the B-8 and B-10 tubing, the subject process computer software changes, made following discovery of this event, will be removed and the TIP System will be tested for proper data acquisition.

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

At approximately 1600 on July 22, 1986, an evaluation of the axial flux profile (traces) of the Unit 2 reactor using Transversing Incore Probe (TIP) System Channel B revealed the traces for the path 10 (B-10) did not correlate as expected with traces using the number 10 tube paths of TIP Channels A, C, and D. Additional trace evaluations revealed the B-10 trace was similar to that of the C-5 trace, which is the appropriate symmetric partner of a B-8 trace. The B-8 trace was found to be similar to A-10, C-10, and D-10 traces. At the time of this discovery, Unit 2 was operating at 100 percent power with a 6 percent margin to reactor thermal limits. As the result of the suspected misconfiguration involving the B-10 and B-8 TIP trace tube paths, reactor power was reduced, at 1715, to obtain an approximate ten percent margin to thermal limits.

At approximately 2200, reactor power was reduced to 75 percent and TIP traces were taken for the B-10, A-10, and B-8 channels. A control rod was inserted to a shallow position adjacent to the common channel and postinsertion traces were taken for B-10, A-10, and B-8. This control rod was then fully withdrawn and another control rod was inserted adjacent to B-8. Postinsertion traces were taken for A-10, B-10, and B-8. This test revealed the following: When the control rod adjacent to the common channel was inserted, A-10 and B-8 traces showed a similar flux response. When the control rod adjacent to B-8 was inserted, the B-10 trace showed a flux response. Consequently, misconfiguration of B-8 and B-10 channels was verified. This test also provided high confidence that the common channel was properly configured for the other machines.

On July 23, 1986, a special procedure was written and approved to test for the misconfiguration of TIP tubing in addition to that of B-8 and B-10. Power was reduced to approximately 75 percent. TIP traces were taken of the 31 local power range monitor (LPRM) strings prior to any control rod motion for test purposes. After completion of these TIP runs, a control rod was selected for movement in close proximity to each TIP location. Out of sequence rods were chosen to maximize flux response. After the control rod was inserted to a shallow position adjacent to the TIP location, a second TIP trace was obtained to analyze the change in power distribution as a result of the rod insertion. This was repeated for each TIP location. The results of these tests indicated that each TIP location (other than B-8 and B-10) showed power distribution changes of a magnitude expected with an adjacent control rod inserted to a shallow position.

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

Following recognition of the B-8 and B-10 misconfiguration, a process computer software change was installed to redirect the B-8 and B-10 TIP information in the process computer. These changes required no hardware modification and were controlled and tested through the use of approved procedures. Software changes were made in the digital fast scan (DFS) programming and the TIP driver software to redirect B-8 and B-10 data. The DFS software change switched the incoming B-8 and B-10 data from the TIP machines. The TIP driver software change provides checks for TIP machine channel selection. Before and after process computer power distribution data was taken to verify the B-8 and B-10 TIP data had been properly switched within the process computer, allowing actual core power distributions to be properly calculated and monitored. The results of the software changes were found to be consistent with the calculational changes predicted with the use of the process computer off-line backup program performed with inputing B-8 and B-10 data in the proper configuration.

During the Unit 2 1984 refueling/maintenance outage the TIP indexers were removed from the drywell area for preventive maintenance. At this time, it was recognized that TIP tubing was not adequately identified to support reinstallation; i.e., some tubing was not marked, others were illegible. Stick-on markers were used prior to tubing removal to identify tubing for reinstallation. However, due to drywell environment--heat, humidity, and general activity--some of the labels had fallen off prior to reinstallation. An attempt was made to identify each when the tubing was reconnected. A Work Request & Authorization (WR&A) was initiated to support/assist with TIP cold alignment. When the cold alignment was completed, a WR&A was written on channels D-7, D-8, and D-9, because core tops did not agree with previous values. Channels D-7, D-8, and D-9 were found misconfigured at the indexer and were reconfigured and the cold alignment completed. The B-8 and B-10 TIP tubes are of identical lengths (within one inch). Consequently, the misconfiguration of the B-8 and B-10 TIPs was not identified during the 1984 outage. It is felt the misconfiguration of B-8 and B-10 occurred during this time and went unnoticed due to identical tubing lengths. TIP traces from January 1983 (Operating Cycle 5) do not show the asymmetry and the B-10 trace matches the A-10, C-10, and D-10 traces. The B-8 trace shows adequate difference in shape such that proper TIP configuration was assumed. Therefore, review of TIP trace data from Cycle 5 supports the prediction that the misconfiguration occurred during the 1984 refueling/maintenance outage and thus affected Operating Cycles 6 and 7.

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

Due to problems with the B TIP machine, a WR&A was initiated to check the flux probing monitor for linearity. No problems with the linearity were found and an IV curve showed no problems with the detector. The only other evidence of a problem with the B machine was in May 1985 when a WR&A was initiated indicating that B-10 did not compare with other TIP traces. It is felt the troubleshooting during Cycle 6 indicated no problems because deep insertion of the center rod for most of the cycle masked the problem.

As a result of this event, appropriate procedural revisions have been made to add independent verification on tubing reconnects and training of appropriate personnel will be performed. A method of retagging will be implemented. A hands-on physical verification walkdown of the TIP tubing will be performed, at which time retagging of the TIP tubing will be accomplished. The misconfiguration of Channels B-8/B-10 will be corrected during next scheduled or forced outage of sufficient length which requires a drywell entry. Following completion of the TIP System physical reconfiguration, the subject process computer software change will be removed and the system tested for proper TIP data acquisition. A new postrefueling startup periodic test (PT) will also be developed to test for proper configuration with rod movements during the medium power plateau.

The Incore Analysis Unit of the Carolina Power & Light Company Nuclear Fuel Section conducted evaluations of off-line process computer calculations at various statepoints during Cycle 7 operation. These evaluations indicated that due to the normalization process in the power distribution calculation, the TIP misconfiguration was allocating more power to the B TIP controlled LPRM strings and respective fuel bundles. Non-B TIP controlled fuel (i.e., controlled by A, C, or D TIPs) was receiving less power than actual conditions within the core. This resulted in thermal limits being calculated conservatively for B TIP controlled fuel and nonconservatively for A, C, and D TIP controlled fuel. It was noted during these studies that the degree of the nonconservative A, C, and D TIP calculational error would range in magnitude from one third to one quarter of the B TIP calculational error. This in turn caused the majority of operation to have leading thermal limit locations in the core controlled by the B TIP. Studies indicated that the nonconservative A, C, and D TIP calculations did not cause a core thermal limit to be exceeded. There were process computer calculations examined with other than B TIP controlled fuel producing the leading thermal limit, however, these situations existed at low enough power levels such that an indicated margin to thermal limits of greater than 20 percent existed. This large margin provided assurance that even with the nonconservatisms of the A, C, and D TIP calculations, no thermal limits were exceeded.

Cycle 6 was also analyzed for the impact of the TIP misconfiguration on thermal limit calculations. Similar off-line calculational techniques were used to analyze the misconfiguration of B-8 and B-10 TIP channels. It was concluded by this study that the magnitude of the calculational error in Cycle 6 was smaller than that during Cycle 7. This was due to the operating philosophy during Cycle 6 of having the center rod inserted to deep positions. When the calculational errors were applied to the statepoints analyzed, it was determined that sufficient margin existed such that no thermal limits were predicted to be violated.

This event was originally reported to Region II on July 23, 1986 (Serial Number BSEP/86-1138). This report is being submitted as an informational LER and was not submitted within 30 days of the event date due to the time necessary to complete the investigation.

CP&L
Carolina Power & Light Company

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

September 17, 1986

FILE: B09-13510C
SERIAL: BSEP/86-1259

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

BRUNSWICK STEAM ELECTRIC PLANT UNIT 2
DOCKET NO. 50-324
LICENSE NO. DPR-62
INFORMATIONAL LICENSEE EVENT REPORT 2-86-019

Gentlemen:

In accordance with Title 10 to the Code of Federal Regulations, the enclosed Licensee Event Report is submitted. This report 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/jlh

Enclosure

cc: Dr. J. N. Grace

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