CP&L

Carolina Power & Light Company
P.O. Box 10429
Southport, NC 28461-0429

SEP 27 1996

SERIAL: BSEP 96-0347
10 CFR 50.73

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

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NO. 50-325 AND 50-324/LICENSE NO. DPR-71 and DPR-62
LICENSEE EVENT REPORT 2-96-003

Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.73, Carolina Power & Light Company submits the enclosed Licensee Event Report. This report fulfills the requirement for a written report within thirty (30) days of a reportable occurrence.

Please refer any questions regarding this submittal to Mr. Mark Turkal at (910) 457-3066.

Sincerely,

W. Levis, Director - Site Operations
Brunswick Nuclear Plant

SFT/sft

Enclosures
1. Licensee Event Report
2. Summary of Commitments

cc: Mr. S. D. Ebneter, Regional Administrator, Region II
Mr. D. C. Trimble, NRR Project Manager - Brunswick Units 1 and 2
Mr. C. A. Patterson, Brunswick NRC Senior Resident Inspector
The Honorable H. Wells, Chairman - North Carolina Utilities Commission

IE22

NRC FORM 366 (4.95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Brunswick Steam Electric Plant, Unit 2	05000324	1 OF 6

TITLE (4)

Operation In Excess Of Maximum Power Level Specified In Operating License

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 NAME	DOCKET NUMBER
08	28	96	96	-- 03 --	00	09	27	96		05000
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1
POWER LEVEL (10)	92

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

20.2201(b)	20.2203(a)(2)(v)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)
20.2203(a)(1)	20.2203(a)(3)(i)	50.73(a)(2)(ii)	50.73(a)(2)(x)
20.2203(a)(2)(i)	20.2203(a)(3)(ii)	50.73(a)(2)(iii)	73.71
20.2203(a)(2)(ii)	20.2203(a)(4)	50.73(a)(2)(iv)	OTHER
20.2203(a)(2)(iii)	50.36(c)(1)	50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 366A
20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Steve Tabor, Sr. Analyst, Regulatory Affairs	(910) 457-2178

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 15 single-spaced typewritten lines) (16)

On August 28, 1996, with Unit 2 operating at approximately 100% power, nuclear engineering personnel discovered that temperature compensation had been omitted from the Unit 2 Plant Process Computer (PPC) feedwater flow algorithm during a PPC upgrade that occurred in March of 1994. The PPC error resulted in unit operation in excess of the core thermal power limit of 2436 MWT during those periods since Unit 2 startup in July of 1994 when the reactor was operating at 100% indicated power and feedwater temperature was less than 420°F. The maximum core thermal power reached during any condition was 102.4%. Operational history and the conservatisms applied in the fuel analysis are such that the safety significance of this event is considered minimal.

In addition, the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) thermal limit as specified in the Technical Specifications was exceeded in December of 1995 during unit coastdown while operating at reduced reactor power and feedwater temperature. The MAPLHGR thermal limit is bounded by fuel thermal-mechanical performance during operation and not LOCA performance. The at least 10% conservatism present in the fuel thermal-mechanical performance limits imposed on MAPLHGR and the B2C11 and B2C12 fuel operational history are such that the MAPLHGR thermal limit violation is not safety significant.

The discovered condition resulted from the failure to revalidate the model provided by the vendor. Corrective actions include correction of the Unit 2 PPC feedwater flow algorithm, review of the other PPC compensated algorithms to ensure accuracy, and the training of PPC personnel on the lessons learned from this event.

Exceeding the MAPLHGR thermal limit constitutes operation prohibited by Technical Specifications and is therefore reportable in accordance with the requirements of 10 CFR 50.73(a)(2)(i).

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant, Unit 2	05000324	96	-- 03 --	00	2 OF 6

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

TITLE

Operation In Excess Of Maximum Power Level Specified In Operating License

INITIAL CONDITIONS

On August 28, 1996, Unit 2 was in Operational Condition 1 with reactor power at approximately 100%.

EVENT NARRATIVE

On August 28, 1996, at approximately 1500 hours, while researching the core thermal power calculation for the Power Uprate Project, engineering personnel discovered that the Unit 2 Plant Process Computer (PPC) algorithm for feedwater flow did not include temperature compensation. The lack of feedwater temperature compensation resulted in the core thermal power (CTP) calculated and displayed by the PPC to be less than actual CTP and consequently created the potential for unit operation at power levels in excess of established limits.

Following discovery of the condition, reactor power was reduced to avoid exceeding the Operating License CTP limit. By approximately 1725 hours, Unit 2 feedwater temperature compensation was corrected and verified. By 2155 hours, the reactor was restored to the normal operating power level.

A team comprised of PPC and nuclear engineering personnel was assembled to determine the cause of the PPC error, the impact of the error on past operation, and whether other compensated PPC points were in error. Based on this review it was determined that temperature compensation had been omitted from the Unit 2 Plant Process Computer (PPC) feedwater flow algorithm during a PPC upgrade in March of 1994. The PPC error resulted in unit operation in excess of the CTP limit of 2436 MWT during those periods since Unit 2 startup in July of 1994 when the reactor was operating at 100% indicated power and feedwater temperature was not equal to 420°F. Specifically, from July 5, 1994 through September 6, 1995 feedwater temperature averaged 410 °F resulting in an actual power of 100.4% or 2446 MW CTP. With one exception during this period being on February 26, 1995 for approximately 7 hours when power was 101% or 2460 MW CTP. Additionally, from March 26, 1996 through August 28, 1996 feedwater temperature averaged 417 degrees resulting in an actual power of 100.2% or 2441 MW CTP. Exceptions during this period were from April 17, 1996 for approximately 9 days when actual power was approximately 102.3% or 2492 MW CTP and July 19 through July 26, 1996 when actual power was approximately 102.4% or 2494 MW CTP.

In addition, the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) thermal limit as specified in the Technical Specifications was exceeded during the Unit 2 B2C11 fuel cycle due to operation with the PPC error during unit coastdown. Based on data review, the MAPLHGR limit is believed to have been exceeded during the period of December 10 through 20, 1995.

Exceeding the MAPLHGR thermal limit constitutes operation prohibited by Technical Specifications and is therefore reportable in accordance with the requirements of 10 CFR 50.73(a)(2)(i).

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant, Unit 2	05000324	96	-- 03 --	00	3 OF 6

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

CAUSE OF EVENT

In 1992 and 1993, the Unit 1 and Unit 2 PPC replacement modifications were designed respectively. To reduce the differences between the units and redundant validations, the design team determined that the best course of action was to copy the configuration of the Unit 1 PPC to the Unit 2 PPC. During 1993, the Unit 1 PPC modification was installed and accepted. The Unit 2 PPC modification was then installed and accepted in 1994. The copying strategy resulted in an error in the configuration of the Unit 2 PPC database of points and compensation formulas. Specifically, the two feedwater flow process points were not correctly compensated in the Unit 2 PPC.

The error occurred in the internal representation of the data points. In the process point database each process point is numbered. To create the Unit 2 PPC database, the configuration of the Unit 1 PPC database was copied and then database unloaded, the names of the Unit 1 points changed to Unit 2 points and then loaded in bulk in the Unit 2 PPC database. The Unit 1 PPC database feedwater points are numbered 231 and 232. However, the Unit 2 PPC database feedwater points are numbered 420 and 421. Therefore, the feedwater process point numbers differ between the two units.

Of all the process points in the database, only the feedwater process points require compensation. When calculating a compensated value, the PPC accepts a value from a temperature sensor and then retrieves the compensation formula by process point number and temperature. In the Unit 2 compensation database, there were no compensation formulas for points numbered 420 and 421. When the PPC failed to find compensation values in the compensation database, the PPC applied a value of one. At the normal operating temperature of 412 degrees, the compensation factor for feedwater flow is approximately 1.003. Using unity as a compensation factor results in a 0.3% error in heat balance.

Several acceptance tests failed to identify this error. These tests did not check that actual process point numbering was identical in both units nor that the relationships between process points and compensation formulas were preserved. In addition, the calibration tests were applied upstream of the error. Each Unit 2 PPC process point was calibrated by injecting a signal and reading the value from the data acquisition subsystem, a unit that passes the values to the PPC once every second. Only a check of the compensation factors produced by the PPC would have revealed a variance from expected values. Finally, the error was too small to be found during checks on the heat balance of the system. Nuclear Engineers calculate the heat balance in two ways and compare the values to validate the system. One method utilizes PPC temperature and flow values to perform a hand calculation of heat balance. The second utilizes control room instrumentation temperature and flow values to perform the hand calculation. The allowed error between the two methods is two percent. The error caused by the improper compensation factor was 0.3% at normal operating conditions. That error was lost in the larger error expected from the accuracy of the instruments for the hand calculation.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant, Unit 2	05000324	96	-- 03 --	00	4 OF 6

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

In conclusion, the cause of the Unit 2 PPC error is attributed to the assumptions made by the team involved in installing the PPC upgrade project. The team did not recognize that errors could occur as a result of database copying. Consequently, the team decided that credit could be taken for some of the Unit 1 PPC testing and inappropriately waived some of the testing specified in the Unit 2 PPC modification acceptance tests. One of the tests which was not performed verified the feedwater temperature compensation factor.

CORRECTIVE ACTIONS

Feedwater temperature compensation was corrected on the Unit 2 PPC.

A review was performed to determine whether other points in the Unit 2 PPC database were affected by the same data loading error. No other discrepancies were identified.

A review of the current procedures for the implementation and control of plant modifications and software changes was performed. Existing processes are considered adequate for ensuring proper validations of software changes. In addition, the current process includes an independent design verification team review which was not in effect at the time of the PPC replacements. This team review provides an additional barrier for ensuring proper independent validations of software changes.

PPC personnel will be trained by December 15, 1996, on the lessons learned from this event to ensure that personnel involved in validating future software changes are aware of the potential problems associated with the validation of replicated software.

SAFETY ASSESSMENT

Since the replacement PPC began operation during the B2C11 fuel cycle, operation during the B2C11 and B2C12 fuel cycles was reviewed to estimate the impact of the incorrect core power calculation on fuel thermal limits. The bulk of the operation during B2C11 and B2C12 fuel cycles had a minimal error in the calculation of core power (less than 1%). Power calculation errors of this magnitude have been previously evaluated as being within the uncertainty used in the reload licensing analysis, and relative to licensing basis transients are not safety significant. In addition, the Brunswick Loss Of Coolant Accident (LOCA) analysis was performed at 110% power which bounds all operation during the B2C11 and B2C12 fuel cycles.

During the B2C11 fuel cycle coastdown, CTP was 2.5% higher than the indicated value. During the B2C12 fuel cycle, a period of low feedwater temperature operation caused actual CTP levels to be approximately 2.4% higher than the indicated value. These periods of operation were examined in closer detail because the error in calculated core power was greater during these periods.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant, Unit 2	05000324	96	-- 03 --	00	5 OF 6

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

B2C11 Coastdown

Actual CTP was 2.5% higher than the indicated power during the end-of-cycle (EOC) coastdown, but the core power at this time was less than 100%. The coastdown power level, although indicated incorrectly, is bounded by the full power EOC transient analysis.

The MAPLHGR thermal limit was slightly exceeded during the coastdown due to the erroneous feedwater temperature. In December of 1995, the MAPLHGR value during the coastdown is estimated to have been 1.008. Based on a review of available data, the MAPLHGR limit was probably exceeded during the period of December 10 through December 20, 1995. Since Brunswick has been evaluated using the General Electric (GE) SAFER/GESTR-LOCA methodology, the MAPLHGR limit is established by fuel thermal-mechanical considerations, not LOCA performance. The MAPLHGR values have power adjustment factors applied at reduced power or flow operation to ensure that a power excursion will not result in nodal power levels in the fuel which will challenge the thermal/mechanical performance criteria of 1% cladding plastic strain listed in GE's licensing methodology document. The MAPLHGR values which were exceeded during this coastdown were exceeded because of the reduced power MAPLHGR limit adjustment factors, not due to the actual nodal power generated in the fuel. The adjustment factors are conservative because Brunswick has always used generic, cycle independent MAPLHGR adjustment factors. These curves are verified each cycle to ensure that they conservatively bound the adjustment factors which would have been calculated if they were developed on a cycle-specific basis. The Brunswick Power Uprate transient analysis indicates that, even including the 5% power uprate, the MAPLHGR power dependent adjustment factor is at least 10% conservative at the power levels being generated during December of 1995. This conservatism is more than enough to ensure that the power generated in the fuel would not have exceeded the fuel design bases even in the event of a transient.

B2C12

A period of low feedwater temperature operation during the B2C12 fuel cycle occurred causing core power to be approximately 102.4%. This power level is within the 110% power level used in the Brunswick LOCA analysis, but is outside the normal 2% uncertainty in core power level used in the transient analysis. This operation outside the 2% uncertainty has not been specifically analyzed, but due to the margin to the fuel thermal limits present during B2C12 operation, this operation was not a safety concern and did not violate the fuel design bases provided in GE document number NEDE-24011, "General Electric Standard Application for Reactor Fuel, GESTAR-II".

The core was operated such that the full power values of MAPRAT and MFLCPR were below the operating limits by approximately 10%. Therefore, even though the core power was slightly above the value used in the licensing analysis, the amount of margin to the MAPRAT and MFLCPR limits indicate that there was sufficient design basis margin to ensure proper fuel performance had an Anticipated Operational Occurrence (AOO) occurred.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant, Unit 2	05000324	96	-- 03 --	00	6 OF 6

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

The core designs developed for each reactor cycle are also analyzed to ensure that the AOO of a Main Steam Isolation Valve closure will not over pressurize the reactor vessel. The 2.4% increase of reactor power above the rated thermal power would not have caused this value to be violated due to the large margin between the Technical Specification requirements for vessel pressure and the core reload analysis result. Therefore, the 2.4% overpower is not significant with respect to the vessel pressure analysis.

PREVIOUS SIMILAR EVENTS

A previous similar event was reported in LER 2-93-008. This event involved exceeding CTP limits due to inaccuracies in feedwater flow instrumentation which is believed to have resulted from erosion/corrosion of the feedwater flow elements.

EIIS COMPONENT IDENTIFICATION

System/Component	EIIS Code
Feedwater System	SJ
Plant Process Computer	CPU

Enclosure
List of Regulatory Commitments

The following table identifies those actions committed to by Carolina Power & Light Company in this document. Any other actions discussed in the submittal represent intended or planned actions by Carolina Power & Light Company. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Manager-Regulatory Affairs at the Brunswick Nuclear Plant of any questions regarding this document or any associated regulatory commitments.

Commitment	Committed date or outage
Training of PPC personnel will be performed on the lessons learned from this event to ensure that personnel involved in validating future software changes are aware of the potential problems associated with the validation of replicated software.	12/15/96