CP&L

Carolina Power & Light Company
PO Box 10429
Southport NC 28461-0429

William R. Campbell
Vice President
Brunswick Nuclear Plant

SERIAL: BSEP 96-0138

April 12, 1996

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNITS NOS. 1 AND 2
DOCKET NOS. 50-325 & 50-324/LICENSE NOS. DPR-71 & DPR-62
NRC GENERIC LETTER 89-10, "SAFETY-RELATED MOTOR-OPERATED VALVE TESTING AND SURVEILLANCE"

Gentlemen:

On June 28, 1989, the NRC issued Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance." The generic letter recommended that licensees' motor-operated valve (MOV) programs include testing, inspection, and maintenance of MOVs necessary to provide assurance that the MOVs will function as required when subjected to the design basis conditions that are considered during both normal operation and abnormal events. By letter dated December 27, 1989, Carolina Power & Light Company (CP&L) responded to NRC GL 89-10, indicating that the Company intended to implement a MOV Program at each of its nuclear plants which would meet the schedule and recommendations of the generic letter. Clarifications to CP&L's MOV Program were provided in a letter dated June 6, 1991. CP&L provided an update on the status of the Brunswick Plant GL 89-10 Program by letter dated August 23, 1994.

Reporting Requirement m. of NRC GL 89-10 requires licensees to notify the NRC in writing within 30 days after the actions described in the first paragraph of Item i. of GL 89-10 have been completed. Item i. of the GL 89-10 states that licensees should complete all design-basis reviews, analyses, verifications, tests, and inspections that have been instituted in order to comply with Items a. through h. within 5 years or three refueling outages of the date of the letter. This letter provides notification to the NRC staff that CP&L's Brunswick Steam Electric Plant, Unit 2, has completed implementing an MOV Program, in accordance with NRC GL 89-10, Items a. through h., consistent with the commitments contained in CP&L's December 27, 1989 and June 6, 1991 response letters. Documentation of completion of these activities on Unit 1 was provide in CP&L's June 21, 1995 response.

A064

Please refer any questions regarding this letter to Mr. George Honma at (910) 457-2741.

Sincerely,

William R. Campbell

GMT/gmt

Enclosure:

1. List of Regulatory Commitments

William Levis, Director Site Operations, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, and agents of Carolina Power & Light Company.

Gwendolyn R. Midgette
Notary (Seal)

My commission expires: August 12, 1996

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

OMAHA PUBLIC POWER DISTRICT
Fort Calhoun Station Unit No. 1

MARCH 1996
Monthly Operating Report

1. OPERATIONS SUMMARY

During the month of March 1996, the Fort Calhoun Station (FCS) operated at a nominal 100% power until March 14, 1996, when power was reduced to start a scheduled maintenance outage. The pre-planned maintenance outage began on March 15, 1996 at 1910 hours, when the plant was taken off-line and Mode 4 (Cold Shutdown) was entered at 1813 hours on March 18th. During control rod insertion, the Control Element Drive Mechanism (CEDM) #15 seal failed, resulting in an increased Reactor Coolant System (RCS) leak rate and causing a significant increase in containment activity. While on-line during March, normal plant maintenance, surveillance, equipment rotation activities and scheduled on-line modifications were performed.

Major activities scheduled and completed during the pre-planned maintenance outage included: replacement/rebuild of three CEDM seals; performance of minor maintenance on the Reactor Coolant Pumps (RCPs); reduction of operator work-arounds; and repair of Control Room Deficiencies (CRDs). A scheduled modification replaced seven of thirteen 480 VAC breaker trip devices, with the remaining six motor feeder breaker trip devices scheduled to be replaced on-line.

On March 14th at 0843 hours, a 480 VAC Motor Control Center MCC-4A2 was removed from service for maintenance, and an eight-hour Limiting Condition for Operation (LCO) was entered in accordance with Technical Specification (TS) 2.7(2)g. At 0858 hours, while troubleshooting instrument loop noise, a technician shorted the AC leads causing the AI-40B instrument bus inverter to transfer power to the bypass transformer. The inverter on bypass required entry into an eight-hour LCO in accordance with TS 2.7(2)h. However, TS 2.7 only allows one condition to exist at a time. Per emergency procedure EPIP-OSC-1, *Emergency Classification*, a forced shutdown required by a TS requires a Notification of Unusual Event (NOUE). The AI-40B instrument bus was returned to its normal supply at 0905 hours. A notification was made to the NRC, as well as to the states of Nebraska and Iowa, that conditions existed for declaration of a NOUE, but the NOUE was not declared since the condition was corrected in an expedient time frame. Subsequently, a Technical Specification Interpretation of Section 2.7 was generated and approved and the NRC notification was retracted.

On March 18th, while starting a containment purge release, a valid Ventilation Isolation Actuation Signal (VIAS) occurred. A four-hour non-emergency notification was made to the NRC pursuant to 10 CFR 50.72(b)(2)(ii) for Engineered Safety Feature (ESF) Actuation. The Containment Stack Radiation Monitor (RM-052) initiated the VIAS while monitoring the Auxiliary Building Ventilation Stack. The RM-052 count rate exceeded its setpoint; however, through operator action to reduce the purge flowrate, the RM-052 count rate decreased below its setpoint and VIAS was reset. No release limits were exceeded at the site boundary. This event is described in Licensee Event Report (LER) 96-001.

On March 21st, the circulating water outfall to the Missouri River was sampled for hydrazine, which is used as an oxygen scavenger in the condensate system, and was found to be 143 ppb. The National Pollution Discharge Elimination System (NPDES) permit limit is 100 ppb. A four-hour non-emergency notification was made to the NRC pursuant to 10 CFR 50.72(b)(2)(vi) due to the notification of other government agencies. The condensate system flush, which was the source of the hydrazine release, was terminated. A second sample was taken and indicated the release level dropped to 3 ppb. The cause for the high level of hydrazine is currently being investigated.

The maintenance outage was completed and the reactor was taken critical on March 24th. On March 25th at 0457 hours, the turbine was placed on-line. A nominal 100% power was achieved on March 28th.

Following return to power operations, indications of a condenser tube leak appeared. On March 29th, a power reduction from 99% to 50% was started to allow a condenser to be isolated in order to troubleshoot and repair the suspected tube leak(s). At 2000 hours, during the power reduction, the condenser tube leakage increased significantly, causing the plant to enter steam generator chemistry Action Level 2, requiring a power reduction to 30% power. At 2048 hours, a NOUE was declared to heighten management's awareness of a degrading plant condition. At 2108 hours, notification of the NOUE was made to the NRC pursuant to 10 CFR 50.72(a)(1)(i). At 2200 hours, the condenser inleakage increased to a Chemistry Action Level 3, requiring the plant to reduce power below 5%. On March 29th, at 2235 hours, the reactor was manually tripped due to lowering condenser vacuum. The NRC was notified of the termination of the NOUE and the manual reactor trip at 2312 hours on March 29th per 10 CFR 50.72(b)(2)(ii). The condenser tubes were tested; the leaking tube was identified and plugged; and the plant was placed on-line on March 31st at 0442 hours.

Five additional incore nuclear detectors failed in March 1996, rendering seven of the twenty-eight detector strings inoperable. All failures have occurred in detectors that were installed during the 1995 refueling outage. These failures are under investigation with assistance from ABB/CE and the incore detector vendor.

2. <u>SAFETY VALVES OR PORV CHALLENGES OR FAILURES WHICH OCCURRED</u>

During the month of March, no power operated relief valve (PORV) or primary system safety valve challenges or failures occurred. In preparation for the March maintenance outage, two valve cycle tests were satisfactorily completed during the month: OP-ST-RC-3002, *Reactor Coolant System (RCS) Category B Valve Exercise Test*, tested the block valves HCV-150 AND HCV-151; and OP-ST-RC-3004, *Power Operated Relief Valves (PORVs) Low Temperature Low Pressure Exercise Test (PCV-102-1 and PCV-102-2)*, tested the PORVs PCV-102-1 and PCV-102-2.

3. <u>RESULTS OF LEAK RATE TESTS</u>

The March RCS leak rate was relatively steady at approximately 0.2 to 0.3 gpm, until the plant was shutdown on March 15th for a planned maintenance outage. During shutdown, after the control rods were fully inserted, the mechanical seal on CEDM #15 failed. Leakage from the CEDM approached 5 gpm as the plant was taken to Cold Shutdown. CEDM #15 was the major source of RCS leakage since the August 24, 1995 reactor trip, and was the primary reason for the planned maintenance outage this month.

Prior to March 15th, approximately 0.1 to 0.15 gpm of the total RCS leakage was classified as "Known" leakage. This leakage was collected in both the Reactor Coolant Drain Tank (RCDT) and the Pressurizer Quench Tank (PQT). CEDM #15 was the major "Known" leakage source with the Reactor Coolant Gas Vent System also contributing leakage with increases in PQT level. Prior to the maintenance outage, the Gas Vent System Leakage was an estimated 0.06 gpm with leakage collected in the Containment Sump and the PQT. Leakage to the Containment Sump is considered "Unknown" leakage.

During the maintenance outage, the mechanical seal on CEDM #15 was replaced and as an additional precaution, the seals on CEDM #4 and #16 were also replaced/repaired. To repair the Reactor Coolant Gas Vent System, isolation valves HCV-180 and HCV-181 were replaced. With the completion of these maintenance activities and the return of the plant to power operations, RCS leakage sources inside of containment have been almost completely eliminated.

The current RCS total leak rate trend is less than 0.15 gpm with most of the leakage coming from the Chemical Volume and Control System. Most of this leakage is attributed to minor charging pump packing leaks.

4. CHANGES, TESTS AND EXPERIMENTS REQUIRING NUCLEAR REGULATORY COMMISSION AUTHORIZATION PURSUANT TO 10CFR50.59

Amendment No.	Description
None	

5. SIGNIFICANT SAFETY RELATED MAINTENANCE FOR THE MONTH OF MARCH 1996

- Rebuilt and repacked charging pump CH-1A
- Rebuilt/Replaced mechanical seals on CEDMs RC-10-04, RC-10-15 and RC-10-16
- Took various measurements and repaired oil leaks on all four RCPs

6. OPERATING DATA REPORT

Attachment I

7. AVERAGE DAILY UNIT POWER LEVEL

Attachment II

8. UNIT SHUTDOWNS AND POWER REDUCTIONS

Attachment III

9. REFUELING INFORMATION, FORT CALHOUN STATION UNIT NO. 1

Attachment IV

ATTACHMENT I
OPERATING DATA REPORT

DOCKET NO.	50-285
UNIT	FORT CALHOUN STATION
DATE	APRIL 08, 1996
COMPLETED BY	D. L. LIPPY
TELEPHONE	(402) 533-6843

OPERATING STATUS

1. Unit Name: FORT CALHOUN STATION
2. Reporting Period: MARCH 1996

NOTES

3. Licensed Thermal Power (MWt): 1500
4. Nameplate Rating (Gross MWe): 502
5. Design Elec. Rating (Net MWe): 478
6. Max. Dep. Capacity (Gross MWe): 502
7. Max. Dep. Capacity (Net MWe): 478

8. If changes occur in Capacity Ratings (3 through 7) since last report, give reasons:
N/A

9. Power Level to which restricted, if any (Net MWe): N/A

10. Reasons for restrictions, if any:
N/A

	THIS MONTH	YR-TO-DATE	CUMULATIVE
11. Hours in Reporting Period...........	744.0	2184.0	197378.0
12. Number of Hours Reactor was Critical	509.7	1949.7	155657.7
13. Reactor Reserve Shutdown Hours......	.0	.0	1309.5
14. Hours Generator On-line.............	488.1	1928.1	153908.6
15. Unit Reserve Shutdown Hours.........	.0	.0	.0
16. Gross Thermal Energy Generated (MWH)	641616.2	2796125.8	205482434.1
17. Gross Elec. Energy Generated (MWH)..	214702.0	948162.0	67881887.2
18. Net Elec. Energy Generated (MWH)....	203763.5	904546.6	64761915.4
19. Unit Service Factor.................	65.6	88.3	78.0
20. Unit Availability Factor............	65.6	88.3	78.0
21. Unit Capacity Factor (using MDC Net)	57.3	86.6	70.9
22. Unit Capacity Factor (using DER Net)	57.3	86.6	69.3
23. Unit Forced Outage Rate.............	5.8	1.5	4.0

24. Shutdowns scheduled over next 6 months (type, date, and duration of each):
REFUELING OUTAGE SCHEDULED TO COMMENCE ON SEPTEMBER 21, 1996, WITH A PLANNED DURATION OF 42 DAYS.

25. If shut down at end of report period, estimated date of startup: ________

26. Units in test status (prior to comm. oper.):

		Forecast	Achieved
INITIAL CRITICALITY			
INITIAL ELECTRICITY	N/A		
COMMERCIAL OPERATION			

ATTACHMENT II
AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO.	50-285
UNIT	FORT CALHOUN STATION
DATE	APRIL 08, 1996
COMPLETED BY	D. L. LIPPY
TELEPHONE	(402) 533-6843

MONTH MARCH 1996

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	486	17	0
2	486	18	0
3	486	19	0
4	486	20	0
5	486	21	0
6	486	22	0
7	486	23	0
8	486	24	0
9	486	25	60
10	487	26	118
11	486	27	399
12	486	28	477
13	485	29	424
14	480	30	0
15	164	31	50
16	0		

INSTRUCTIONS

On this form, list the average daily unit power level in MWe-Net for each day in the reporting month. Compute to the nearest whole megawatt.

ATTACHMENT III
UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-285
UNIT NAME Fort Calhoun St.
DATE April 9, 1996
COMPLETED BY D. L. Lippy
TELEPHONE (402) 533-6843

REPORT MONTH March 1996

No.	Date	Type[1]	Duration (Hours)	Reason[2]	Method of Shutting Down Reactor[3]	Licensee Event Report No.	System Code[4]	Component Code[5]	Cause & Corrective Action to Prevent Recurrence
96-01	960315	S	226.8	B	1	N/A	ZZ	ZZZZZZ	At 1910 hours on March 15, 1996, the Fort Calhoun Station (FCS) commenced a pre-planned maintenance outage to repair/replace Control Element Drive Mechanism (CEDM) mechanical seals and perform other maintenance activities. On March 25, 1996, the turbine was placed on-line at 0457 hours. A nominal 100% power was achieved on March 28th.
96-02	960329	F	30.1	A	2	96-002	HC	HTEXCH	At 2235 hours on March 29th, the reactor was manually tripped due to lowering condenser vacuum. The condenser tubes were tested, leaks identified and tubes were plugged. The plant was placed on-line on March 31st at 0442 hours.

1
F: Forced
S: Scheduled

2
Reason:
A-Equipment Failure (Explain)
B-Maintenance or Test
C-Refueling
D-Regulatory Restriction
E-Operator Training & License Examination
F-Administrative
H-Other (Explain)

3
Method:
1-Manual
2-Manual Scram
3-Automatic Scram
4-Other (Explain)

4
Exhibit F - Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File (NUREG-0161)

5
Exhibit H - Same Source

Attachment IV
Refueling Information
Fort Calhoun Station - Unit No. 1

Report for the month ending March 31, 1996

1. Scheduled date for next refueling shutdown.	September 21, 1996
2. Scheduled date for restart following refueling.	November 2, 1996
3. Will refueling or resumption of operations thereafter require a technical specification change or other license amendment?	Yes
a. If answer is yes, what, in general, will these be?	Enrichment limit of spent fuel racks is to be increased to at least 4.5 w/o from 4.2 w/o. This is necessary based upon the preliminary Cycle 17 core pattern development.
b. If answer is no, has the reload fuel design and core configuration been reviewed by your Plant Safety Review Committee to determine whether any unreviewed safety questions are associated with the core reload.	N/A
c. If no such review has taken place, when is it scheduled?	N/A
4. Scheduled date(s) for submitting proposed licensing action and support information.	Spent fuel rack enrichment limit change was submitted February 1, 1996.
5. Important licensing considerations associated with refueling, e.g., new or different fuel design or supplier, unreviewed design or performance analysis methods, significant changes in fuel design, new operating procedures.	N/A
6. The number of fuel assemblies:	
a) in the core	133 Assemblies
b) in the spent fuel pool	618 Assemblies
c) spent fuel pool storage capacity	1083 Assemblies
7. The projected date of the last refueling that can be discharged to the spent fuel pool assuming the present licensed capacity.	2007 Outage

Prepared by J. Boettman

Date 4/8/96