Omaha Public Power District
444 South 16th Street Mall
Omaha, Nebraska 68102-2247
402/636-2000

September 14, 1992
LIC-92-297R

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Reference: Docket No. 50-285

Gentlemen:

SUBJECT: August 1992 Monthly Operating Report (MOR)

Enclosed is the August 1992 MOR for Fort Calhoun Station (FCS) Unit No. 1 as required by FCS Technical Specification Section 5.9.1.

If you should have any questions, please contact me.

Sincerely,

W. G. Gates

W. G. Gates
Division Manager
Nuclear Operations

WGG/grc

Enclosures

c: LeBoeuf, Lamb, Leiby & MacRae
J. L. Milhoan, NRC Regional Administrator, Region IV
R. P. Mullikin, NRC Senior Resident Inspector
S. D. Bloom, NRC Acting Project Manager
R. T. Pearce, Combustion Engineering
R. J. Simon, Westinghouse
Office of Management & Program Analysis (2)
INPO Records Center
American Nuclear Insurers

IE24 /

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO.	50-285
UNIT	FORT CALHOUN STATION
DATE	SEPTEMBER 08 1992
COMPLETED BY	[illegible]. R. CAVANAUGH
TELEPHONE	(402) 636-2474

MONTH AUGUST 1992

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	471	17	477
2	472	18	475
3	4[illegible]	19	473
4	470	20	472
5	353	21	472
6	14	22	23
7	302	23	0
8	459	24	0
9	462	25	0
10	461	26	0
11	468	27	0
12	470	28	0
13	472	29	0
14	474	30	0
15	475	31	0
16	476		

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.

OPERATING DATA REPORT

DOCKET NO.	50-285
UNIT	FORT CALHOUN STATION
DATE	SEPTEMBER 08 1992
COMPLETED BY	G. R. CAVANAUGH
TELEPHONE	(402) 636-2474

OPERATING STATUS

1. Unit Name: FORT CALHOUN STATION
2. Reporting Period: AUGUST 1992

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:
   NA
9. Power Level to which restricted, if any (Net MWe): NA
10. Reasons for restrictions, if any:
    NA

	THIS MONTH	YR-TO-DATE	CUMULATIVE
11. Hours in Reporting Period...........	744.0	5855.0	165985.0
12. Number of Hours Reactor was Critical	505.8	2964.7	127783.4
13. Reactor Reserve Shutdown Hours......	.0	.0	1309.5
14. Hours Generator On-line.............	487.9	2874.2	126251.3
15. Unit Reserve Shutdown Hours.........	.0	.0	.0
16. Gross Thermal Energy Generated (MWH)	700842.0	3785203.6	165408929.3
17. Gross Elec. Energy Generated (MWH)..	231608.0	1259591.0	54435717.2
18. Net Elec. Energy Generated (MWH)....	219931.7	1194368.7	51928120.1
19. Unit Service Factor.................	65.6	49.1	76.1
20. Unit Availability Factor............	65.6	49.1	76.1
21. Unit Capacity Factor (using MDC Net)	61.8	42.7	68.0
22. Unit Capacity Factor (using DER Net)	61.8	42.7	66.2
23. Unit Forced Outage Rate.............	34.4	21.1	4.4

24. Shutdowns scheduled over next 6 months (type, date, and duration of each):
    NONE.
25. If shut down at end of report period, estimated date of startup: 09/05/92
26. Units in test status (prior to comm. oper.):

		Forecast	Achieved
INITIAL CRITICALITY			
INITIAL ELECTRICITY	N/A		
COMMERCIAL OPERATION			

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-285
UNIT NAME Fort Calhoun St.
DATE September 9, 1992
COMPLETED BY G. R. Cavanaugh
TELEPHONE (402) 536-2474

REPORT MONTH August 1992

No.	Date	Type[1]	Duration (Hours)	Reason[2]	Method of Shutting Down Reactor[3]	Licensee Event Report #	System Code[4]	Component Code[5]	Cause & Corrective Action to Prevent Recurrence
92-06	920805	F	17.8	A	4	None	EJ	BRK	On August 5, 1992, while the plant was operating at 100% power, during a normal preventative maintenance activity, a burning odor was traced to the main switch of AI-41A (125 VDC BUS-1 Panel). The switch was found to be hot to the touch and had visual indications of heat damage. Due to the adverse consequences of a potential loss of DC power at 100% power, the decision was made to do the repairs off-line. A power reduction was initiated at 1250 hours on August 5, and the turbine was taken off-line at 2250 hours. Reactor power was then reduced to approximately 1% before installing a jumper to keep the DC panel energized while repairs were made. The generator was put back on-line on August 6, at 1639 hours.
92-07	920822	F	238.4	A	3	LER-92-028	EC	JX	Failure of a power supply in the Electro-hydraulic Control System caused the turbine control valves to partially close. This caused the Reactor Coolant System pressure to increase. Although the pressure did not reach the setpoint, RC-142 lifted prematurely to relieve pressure. The valve reseated but pressure dropped to the reactor trip setpoint for Thermal Margin/Low Pressure and an automatic reactor trip occurred. The plant remained shutdown for the remainder of the month. See LER-92-028 for further details.

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
G-Operational Error (Explain)
H-Other (Explain)

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

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

5
Exhibit I - Same Source

(9/77)

Refueling Information
Fort Calhoun - Unit No. 1

Report for the month ending August 1992

1. Scheduled dated for next refueling shutdown. — September 1993

2. Scheduled date for restart following refueling. — November 1993

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? — Incorporate specific requirements resulting from reload safety analysis.

   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. — June 1993

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 procedure — New fuel supplier; New LOCA analysis

6. The number of fuel assemblies:
   a) in the core — 133 Assemblies
   b) in the spent fuel pool — [illegible]29 Assemblies
   c) spent fuel pool storage capacity — 729 Assemblies
   d) planned spent fuel pool storage capacity — Planned to be increased with higher density spent fuel racks.

7. The projected date of the last refueling that can be discharged to the spent fuel pool assuming the present licensed capacity. — 1995*

* Capability of full core offload of 133 assemblies lost. Reracking to be performed between the 1993 and 1995 Refueling Outages.

Prepared by Ma[illegible] Date 9/9/92

OMAHA PUBLIC POWER DISTRICT
Fort Calhoun Station Unit No. 1

AUGUST 1992
Monthly Operating Report

I. OPERATIONS SUMMARY

Fort Calhoun Station (FCS) operated at nominal full power August 1 through August 5. During a normal preventive maintenance activity, a burning odor was noted within one of the Control Room panels. The odor was traced to the main switch of AI-41A (125 VDC BUS-1 Panel). The switch was hot to the touch and had visual indications of heat damage. Due to the adverse consequences of a potential loss of DC power at 100% power, the decision was made to do the repairs off-line. A power reduction was initiated at 1250 hours on August 5, 1992, and the turbine was taken off line at 2250 hours. Reactor power was then reduced to approximately 1% before installing a jumper to keep the DC panel energized while repairs were made. Following repairs, the turbine-generator was synchronized to the grid at 1639 hours on August 6, 1992.

During synchronization of the turbine-generator on August 6, the 3451-5 Breaker (one of two breakers connecting FCS to the 345 kV grid) would not close. An investigation revealed 3 defective coils. Repairs were successfully completed and the breaker was reclosed at 1540 hours on August 13, 1992.

Full power operation continued until 0152 hours on August 22, 1992. Failure of a power supply in the Electro-Hydraulic Control (EHC) System caused the turbine control valves to close from approximately 40% open to 22% open. The primary to secondary load mismatch caused reactor coolant system pressure to increase to approximately 2398 psia (reactor trip setpoint is <2400 psia). At approximately 2398 psia, RC-142 (pressurizer code safety valve, setpoint approximately 2500 psia) lifted prematurely and relieved pressure. The valve reseated, but pressure dropped to the reactor trip setpoint for Thermal Margin/Low Pressure (TM/LP) and an automatic reactor trip occurred. RC-142 is the same valve which lifted prematurely and did not properly reseat during the July 3, 1992 trip (see July 1992 Monthly Operating Report). The plant was maintained in hot shutdown while *in situ* testing of RC-141 and RC-142 was completed.

Results of this testing indicate that RC-142 was lifting approximately 100 psia early. *In situ* testing was also conducted for RC-141 (the other code safety valve) with similar results. OPPD determined that valve body temperature is a significant contributor to the setpoint and consequently, previous setpoints may have been inappropriately low. After concluding the *in situ* testing, the plant commenced a cooldown, initiating shutdown cooling at 1059 hours on August 27, 1992. RC-141 and RC-142 were removed on August 28, 1992 and shipped off-site to Wyle Laboratories for inspection and additional testing. Further details of the testing and the root cause will be detailed in LER-92-028. The pressurizer manway was removed and RCS level was reduced to midloop for Reactor Coolant Pump RC-3D seal replacement on August 30, 1992.

The safety valves were returned to the site on August 31, 1992 and were reinstalled September 1, 1992. The failed EHC System power supply was removed. The EHC System was modified to receive its power directly from the power bus within Panel AI-50 (where the EHC is located). Panel AI-50 receives its power from the Permanent Magnet Generator with back-up from the Station 120 VAC.

The following NRC Inspections took place during August 1992:

IER No.	Title
92-15	Monthly Resident Inspection
92-17	EOP Follow-up Inspection
92-21	August 22, 1992 Plant Shutdown Special Inspection

The following LERs were submitted during August 1992:

LER No.	LER Date	Description
92-022	08/03/92	Inadequately Sized Heater Drain Pump Cables
92-023	08/03/92	Reactor Trip Due to Inverter Malfunction and Subsequent Pressurizer Safety Valve Leak
92-024	08/17/92	Failure to Comply with Linear Heat Rate Technical Specifications During Alarm Inoperability
92-025	08/21/92	Inadvertent Manual Start of Emergency Diesel Generator at the Local Control Panel
92-026	08/24/92	Incore Detector Alarm Limits Non-Conservative for Monitoring Peak Linear Heat Rate

A. SAFETY VALVES OR PORV CHALLENGES OR FAILURES WHICH OCCURRED

Code Safety Valve RC-142 lifted below its expected setpoint, resulting in the automatic reactor trip on August 22, 1992. The valve reseated after lifting. For further details, see Section I, Operations Summary.

B. RESULTS OF LEAK RATE TESTS

RCS leakage during the month of August, 1992 was generally low. During periods of relative stability, the total RCS leakage averaged less than 0.1 gpm. This leakage was composed of approximately one-half "Known" leakage to the Reactor Coolant Drain Tank and one-half "Unknown" leakage. "Unknown" leakage is the arithmetic difference between Known leakage and Total leakage.

On August 22, 1992, the plant experienced an automatic reactor trip as noted in Section I. This and the subsequent cooldown and depressurization resulted in anomalous leakrate test results on August 22 and August 26. With the plant in the cold shutdown mode, no leakrate tests could be performed after August 26.

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

Amendment No.	Description
146	This amendment revises the Technical Specifications to increase the maximum allowable setpoints from ±1% to +3%/-2% for the MSSVs and to specify lift settings for all MSSVs and for the two pressurizer safety valves.
147	This amendment makes changes to Technical Specification 2.7, "Electrical Systems", to correct inconsistencies and to provide further guidance on equipment necessary for the 161 Kv power supply. Additionally, administrative changes are incorporated for Technical Specification 2.7 and Table 2-10.

D. SIGNIFICANT SAFETY RELATED MAINTENANCE FOR THE MONTH OF AUGUST 1992

A1TS-120 (Test switch for Channel 'A' of the diversified scram system) did not operate correctly and was replaced.

AC-189 and AC-190 (discharge check valves for spent fuel pool circulation pumps AC-5A and AC-5B) were replaced.

Damage due to overheating was discovered to the main switch of AI-41A (125 VDC BUS-1 panel). A temporary modification instructed electrical maintenance to remove the main switch and install jumpers in its place.

Packing was replaced on Charging Pump CH-1B.

Repairs were performed on HCV-2805B (raw water strainer AC-12B, backwash control valve).

Repairs were performed on component cooling heat exchanger AC-1D component cooling water inlet and outlet valves HCV-492A and HCV-492B.

New supports were welded in place and new fittings installed into primary containment penetrations M-73 and M-84.

Toxic gas monitors YIT-6288A and YIT-6288B were repaired due to evidence of water intrusion.

A 52/STA switch was installed into 1A3-18 (spare feeder in the 4160V switch gear).