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

November 14, 1991
LIC-91-310R

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

Reference: Docket No. 50-285

Gentlemen:

SUBJECT: October Monthly Operating Report (MOR)

Enclosed is the October 1991 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/sel

Enclosures

c: LeBoeuf, Lamb, Leiby & MacRae
R. D. Martin, NRC Regional Administrator, Region IV
R. P. Mullikin, NRC Senior Resident Inspector
D. K. Sentell, Combustion Engineering
R. J. Simon, Westinghouse
Office of Management & Program Analysis (2)
INPO Records Center
American Nuclear Insurers

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R PDR

Employment with Equal Opportunity
Male/Female

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO.	50-285
UNIT	Fort Calhoun Station
DATE	November 11, 1991
COMPLETED BY	G. R. Cavanaugh
TELEPHONE	(402)636-2474

MONTH October 1991

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	0	17	483
2	0	18	381
3	0	19	52
4	0	20	209
5	0	21	368
6	31	22	480
7	105	23	484
8	179	24	485
9	410	25	390
10	456	26	35
11	456	27	394
12	456	28	478
13	456	29	482
14	456	30	486
15	470	31	487
16	479		

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 November 11, 1991
COMPLETED BY G. R. Cavanaugh
TELEPHONE (402)636-2474

OPERATING STATUS

1. Unit Name: Fort Calhoun Station — Notes
2. Reporting Period: October 1991
3. Licensed Thermal Power (MWt): 1500
4. Nameplate Rating (Gross MWe): 502
5. Design Electrical Rating (Net MWe): 478
6. Maximum Dependable Capacity (Gross MWe): 502
7. Maximum Dependable Capacity (Net MWe): 478
8. If changes occur in Capacity Ratings (Item Numbers 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	745.0	7,296.0	158,666.0
12.	Number of Hours Reactor was Critical	628.5	6,566.0	123,354.7
13.	Reactor Reserve Shutdown Hours	0.0	0.0	1,309.5
14.	Hours Generator On-Line	571.5	6,483.2	121,913.1
15.	Unit Reserve Shutdown Hours	0.0	0.0	0.0
16.	Gross Thermal Energy Generated (MWH)	735,387.1	8,152,359.2	159,435,848.5
17.	Gross Electrical Energy Generated (MWH)	244,986.0	2,680,730.0	52,430,856.2
18.	Net Electrical Energy Generated (MWH)	231,537.2	2,536,761.2	50,021,537.5
19.	Unit Service Factor	76.7	88.9	76.8
20.	Unit Availability Factor	76.7	88.9	76.8
21.	Unit Capacity Factor (Using MDC Net)	65.0	72.7	68.2
22.	Unit Capacity Factor (Using DER Net)	65.0	72.7	66.8
23.	Unit Forced Outage Rate	23.3	11.1	3.9

24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each):
    Refueling outage scheduled to begin February 1, 1992 and last approximately three months.
25. If Shut Down at End of Report Period, Estimated Date of Startup: NA
26. Units In Test Status (Prior to Commercial Operation):

		Forcast	Achieved
INITIAL CRITICALITY			
INITIAL ELECTRICITY	N/A		
COMMERCIAL OPERATION			

Refueling Information
Fort Calhoun - Unit No. 1

Report for the month ending October 1991

1. Scheduled date for next refueling shutdown. — February 1, 1992

2. Scheduled date for restart following refueling. — April 29, 1992

3. Will refueling or resumption of operation 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. — November 1991

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. — New fuel supplier; New LOCA Analysis

6. The number of fuel assemblies:
   a) in the core — 133 Assemblies
   b) in the spent fuel pool — 477 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 Marcus Johnson for KCH Date 11/14/91

Fort Calhoun Station
November 1991
Cavalier
[illegible]

UNIT SHUTDOWNS AND POWER REDUCTIONS

REPORT MONTH OCTOBER 1991

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
91-05	910912	F	131.2	[illegible]	1	91-18	EC	BATTRY	After completing the outage to replace the station batteries, Fort Calhoun Station was [illegible] critical on October 5, 1991 at [illegible] hours. The generator was put on line at 1114 hours on October 6, 1991.
91-06	911018	F	32.2	A	4	N/A	HA	VALVEX	Fort Calhoun Station was taken off line on October 18, 1991 at 0307 hours to repair a steam leak on the drain line from the No. 2 control valve. The reactor remained critical and the generator was placed back on line at 1116 hours on October 19, 1991.
91-07	911025	F	10.1	A	4	N/A	HA	XXXXXX	Fort Calhoun Station was taken off line on October 25, 1991 at 2204 hours to repair a steam leak from an instrument tap on the turbine high pressure shell. The reactor remained critical and the generator was placed back on line at 0810 on October 26, 1991.

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
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 1 - Same Source

(9/77)

OMAHA PUBLIC POWER DISTRICT
Fort Calhoun Station Unit No. 1

OCTOBER 1991
Monthly Operating Report

I. OPERATIONS SUMMARY

Fort Calhoun Station remained shutdown for battery replacement until October 5, 1991. At that time the reactor was taken critical and power increased, with nominal 100% power operation being attained on October 17, 1991.

On October 18, 1991, power was reduced from 100% to 10% due to a steam leak on a turbine control valve drain line. Repairs were completed on October 19, 1991. Power was increased, with nominal 100% power operation being attained October 22, 1991.

The plant remained at 100% until October 25, 1991, when power was decreased from 100% to 10% due to a steam leak from an instrument tap on the high pressure turbine. Repairs were completed on October 26, 1991. Power was increased, with nominal 100% power operation being attained October 29, 1991.

The plant operated at a nominal 100% power for the remainder of the month.

On October 4, 1991, during the heat-up of the RCS prior to criticality, an inadvertent actuation of "A" train of Containment Isolation Actuation Signal (CIAS) occurred during the performance of Engineered Safeguards testing. The cause of the event was determined to be improper contact stack-up in the CIAS override test switch. A four hour NRC report was made due to the event being an unplanned actuation of Engineered Safeguards. Corrective actions included placing a caution tag on the override switch to ensure that it is firmly put in the "TEST" position. Disassembly of the override switch for inspection is planned during the next refueling outage.

From October 22, through October 30, 1991, the Nuclear Regulatory Commission administered Licensed Operator Requalification, SRO Upgrade and Initial License Examinations to selected plant personnel. The exams consisted of written, oral and dynamic (simulator) evaluations. Three Requalification Crews (total of 12 individuals), seven (7) SRO Candidates and seven (7) Initial License candidates participated in the exam process.

The following NRC inspection took place in October:

IR-91-23 Routine Residents' Inspection

The following NRC LERs were submitted:

LER No.	Date	Description
91-18	10/15/91	Inoperable Station Batteries Due to Inadequate Design of Terminal Post Seals
91-19	10/15/91	Approved Procedure Could Have Prevented Containment Spray From Fulfilling Design Function
91-20	10/21/91	Unmonitored Steam Generator Release to Missouri River.

A. SAFETY VALVES OR PORV CHALLENGES OR FAILURES WHICH OCCURRED

On October 2, 1991, during the performance of the PORV ISI test (OP-ST-RC-3004), PCV-102-1 (PORV #1) failed to open within the prescribed time limit. No problem was encountered with PCV-102-2 (PORV #2). Per Technical Specification 2.1.6, PCV-102-1 was declared inoperable and its associated block valve HCV-151 was closed. Upon further investigation it was determined that PCV-102-1 had been leaking. The leakage may have caused some thermal expansion which prevented the PORV from opening. With the block valve closed the leakage was stopped. On October 3, 1991, the block valve was opened, and PCV-102-1 was successfully opened. The PORV was cycled successfully two more times before the surveillance test was deemed satisfactory.

B. RESULTS OF LEAK RATE TESTS

Following the performance of the first post-outage (i.e., battery outage) RCS leak rate test on October 4, 1991, the reactor has been cycled from 10 to 100% power three times. This has complicated the process of obtaining meaningful RCS leak rate test results.

The Total RCS leak rate for October, 1991, averaged approximately 0.25 gpm. This was greater than that experienced prior to the start of the battery outage in mid September 1991. "Known" leakage to the Reactor Coolant Drain Tank (RCDT) following the battery outage increased to an average of 0.3 gpm. The unusual situation of having a larger "Known" leak rate than "Total" leak rate resulted in having negative "Unknown" leak rate values for most of the tests performed in October.

It appears that there is some leakage in addition to Reactor Coolant leaking into the RCDT. This accounts for the negative "Unknown" leak rate test results. This discrepency is currently under investigation.

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

None

D. SIGNIFICANT SAFETY RELATED MAINTENANCE FOR THE MONTH OF OCTOBER 1991:

- Adjusted the impeller lift on the raw water pump (AC-10D) and obtained new baseline data.
- Installed new "O" rings and gaskets on the component cooling outlet valve actuator (HCV-490B) for CCW heat exchanger (AC-1B).
- Replaced a circuit board on the Qualified Safety Parameter Display System Panel (AI-208A) regarding heated junction thermocouple temperature values.

- Adjusted the heated junction thermocouple heater controllers (A&B) for the Qualified Safety Parameter Display System Panels (AI-208A and AI-208B).

- Replaced a temperature control switch for the immersion heater on Emergency Diesel Generator No. 1.

- Replaced the roll pins on the radiator exhaust damper for the Emergency Diesel Generators (DG-1, DG-2).