AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-285

UNIT Fort Calhoun Statio

DATE July 6, 1984

COMPLETED BY T. P. Matthews

TELEPHONE (402) 536-4733

MONTH June, 1984

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	0.0	17	0.0
2	0.0	18	0.0
3	0.0	19	0.0
4	0.0	20	0.0
5	0.0	21	0.0
6	0.0	22	0.0
7	0.0	23	0.0
8	0.0	24	0.0
9	0.0	25	0.0
10	0.0	26	0.0
11	0.0	27	0.0
12	0.0	28	0.0
13	0.0	29	0.0
14	0.0	30	0.0
15	0.0	31	
16	0.0		

INSTRUCTIONS

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

(9/77)

OPERATING DATA REPORT

DOCKET NO. 50-285
DATE July 6, 1984
COMPLETED BY T. P. Matthews
TELEPHONE (402) 536-4733

OPERATING STATUS

1. Unit Name: Fort Calhoun Station
2. Reporting Period: June, 1984
3. Licensed Thermal Power (MWt): 1500
4. Nameplate Rating (Gross MWe): 501
5. Design Electrical Rating (Net MWe): 478
6. Maximum Dependable Capacity (Gross MWe): 501
7. Maximum Dependable Capacity (Net MWe): 478
8. If Changes Occur in Capacity Ratings (Items Number 3 Through 7) Since Last Report, Give Reasons: 6 and 7 were restored to their previous values to reflect replacement of the first stage blading in the high pressure turbine.

Notes

9. Power Level To Which Restricted, If Any (Net MWe): N/A
10. Reasons For Restrictions, If Any: None

	This Month	Yr.-to-Date	Cumulative
11. Hours In Reporting Period	720.0	4,367.0	94,369.0
12. Number Of Hours Reactor Was Critical	0.0	1,490.2	71,384.1
13. Reactor Reserve Shutdown Hours	0.0	0.0	1,309.0
14. Hours Generator On-Line	0.0	1,489.5	70,892.0
15. Unit Reserve Shutdown Hours	0.0	0.0	0.0
16. Gross Thermal Energy Generated (MWH)	0.0	2,152,796.9	88,912,510.6
17. Gross Electrical Energy Generated (MWH)	0.0	690,258.0	29,007,827.0
18. Net Electrical Energy Generated (MWH)	0.0	656,536.5	27,736,405.2
19. Unit Service Factor	0.0	34.1	75.1
20. Unit Availability Factor	0.0	34.1	75.1
21. Unit Capacity Factor (Using MDC Net)	0.0	31.5	64.1
22. Unit Capacity Factor (Using DER Net)	0.0	31.5	61.8
23. Unit Forced Outage Rate	0.0	0.0	3.5

24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each):

25. If Shut Down At End Of Report Period, Estimated Date of Startup: July 10, 1984
26. Units In Test Status (Prior to Commercial Operation): N/A

	Forecast	Achieved
INITIAL CRITICALITY		
INITIAL ELECTRICITY		
COMMERCIAL OPERATION		

DOCKET NO. 50-285
UNIT NAME Fort Calhoun Station
DATE July 6, 1984
COMPLETED BY T. P. Matthews
TELEPHONE (402) 536-4733

UNIT SHUTDOWNS AND POWER REDUCTIONS

REPORT MONTH June, 1984

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
84-01	840303	S	2877	C	4	N/A	XX	XXXXX	1984 refueling outage commenced March 3, 1984.

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 June 1984.

1. Scheduled date for next refueling shutdown. September 1985

2. Scheduled date for restart following refueling. November 1985

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?

   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. ________

   c. If no such review has taken place, when is it scheduled? ________

4. Scheduled date(s) for submitting proposed licensing action and support information. August 1985

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.

6. The number of fuel assemblies:
   a) in the core 133 assemblies
   b) in the spent fuel pool 305 "
   c) spent fuel pool storage capacity 729 "
   d) planned spent fuel pool storage capacity * "

   *May be increased via fuel pin consolidation

7. The projected date of the last refueling that can be discharged to the spent fuel pool assuming the present licensed capacity. 1996

Prepared by J K Gasper Date July 2, 1984

OMAHA PUBLIC POWER DISTRICT
Fort Calhoun Station Unit No. 1

June, 1984
Monthly Operations Report

I. OPERATIONS SUMMARY

Fort Calhoun Station has been shutdown through June for testing and repair of the steam generators. Plant startup began on June 28 and the plant should go online in July.

Three operators sat for NRC reactor operator license examinations in June.

The Missouri River reached a high level of 1002.4' MSL on June 27 flooding the chemical waste lagoons and part of the parking lot.

No safety valve or PORV challenges occurred.

A. PERFORMANCE CHARACTERISTICS

None

B. CHANGES IN OPERATING METHODS

None

C. RESULTS OF SURVEILLANCE TESTS AND INSPECTIONS

None

D. CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL

Procedure	Description
SP-RC-2-1	Plugging Steam Generator Tubes (RC-2B).
	This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it only provides for installation of two tube plugs in "B" steam generator. Tubes were plugged in accordance with Combustion Engineering guidelines.

D. CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL (Continued)

Procedure	Description
SP-RC-2-1	Plugging Steam Generator Tubes (RC-2B). This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it only provides for installation of three tube plugs in "B" steam generator. Tubes were plugged in accordance with Combustion Engineering guidelines.
SP-RC-2-1	Plugging Steam Generator Tubes (RC-2B). This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it only provides for installation of four tube plugs in "B" steam generator. Tubes were plugged in accordance with Combustion Engineering guidelines.
SP-VA-80	Hydrogen Purge System Test. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it only provides for running of blowers and measurement of differential pressure. No technical specifications govern the use of this equipment.
SP-MS-4	Testing of Steam Dump Valves TCV-909-1, 909-2, 909-3 and 909-4. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 because the steam dump valves are not considered to be safety related and even though they may help to mitigate the consequences of specific accidents, they are not credited for helping to mitigate the consequences of any accident previously evaluated in the USAR.

System Acceptance Committee Packages for June, 1984:

Package	Description/Analysis
EEAR FC-82-91	Steam Generator Blowdown Monitor Relocation. This modification provided for sufficient shielding of the steam generator blowdown monitors. Samplers with six inch thick lead shield were installed because of high background radiation within the sampling room 60. Therefore, the installation of the new samplers will improve the function of the monitors. This modification has no adverse effect on the safety analysis.
EEAR FC-79-66	Qualification of Backup Containment Instrumentation for Post LOCA. This modification provided for the relocation of existing transmitters and the one-for-one functional replacement of RTD's. Channel redundancy, equipment separation and power supply independence are consistent with the existing systems. Measurement sensitivity is equivalent to the existing equipment. This modification has no adverse effect on the safety analysis.
EEAR FC-82-84	Capillary Tube Replacement on RCDT. This modification provided for the replacement of a capillary tube on RCDT. This modification has no adverse effect on the safety analysis.
EEAR FC-83-50	Turbine Drain Valve Controls. This modification provided for the installation of a drain valve to operate automatically upon turbine trip and to help prevent water induction in the HP section of the turbine. This modification has no adverse effect on the safety analysis.
EEAR FC-83-56	Hydrogen Purge Valves VA-289, VA-280. This modification provided for the replacement of existing valves with valves designed for a better seal. This modification has no adverse effect on the safety analysis.

D. CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL (Continued)

System Acceptance Committee Packages for June, 1984: (Continued)

Package	Description/Analysis
DCR 74B-38	Pressurizer Channel 101Y Output Oscillation. This modification replaced transmitters on pressurizer level channels. This modification has no adverse effect on the safety analysis.
DCR 75B-16	PT-235 Vibration Damage. This modification improved the reliability of the system by moving the charging pump pressure transmitter and protecting it from vibration. This modification has no adverse effect on the safety analysis.
DCR 76-49	Flow Meters for Radiation Monitors. This modification replaced rotameters with flow meters. This modification has no adverse effect on the safety analysis.
DCR 77-39	CEDM Cooling Ducts. This modification provided for the relocation of a joint in the ductwork to a more accessible location. This modification has no adverse effect on the safety analysis.
DCR 78-7	Generator Core Monitoring. This modification added a means to detect early stages of generator core overheating. This modification has no adverse effect on the safety analysis.
EEAR FC-78-29	RPS Noise Spikes. This modification improved the reliability of the RPS system by eliminating spurious trips caused by the operation of certain valves. This modification has no adverse effect on the safety analysis.

D. CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL (Continued)

System Acceptance Committee Packages for June, 1984: (Continued)

Package	Description/Analysis
EEAR FC-79-149	Warehouse Decking Fire Hazard. This modification did not effect a safety related system; therefore, has no adverse effect on the safety analysis.
EEAR FC-79-153	FW Flow Transmitter Replacement. This modification replaced existing FW flow transmitters to improve operability. This modification has no adverse effect on the safety analysis.

E. RESULTS OF LEAK RATE TESTS

The Fort Calhoun Station is currently performing B and C penetration tests. A report will be sent out at the end of the refueling outage.

F. CHANGES IN PLANT OPERATING STAFF

None

G. TRAINING

Training for the month of June for operators was directed toward procedure and administrative changes due to the steam generator tube rupture and the trip two, leave two logic for reactor coolant pumps. Maintenance training was held on flood control procedures, crane operation and systems. Three candidates sat for NRC reactor operator license examinations.

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

None

II. MAINTENANCE (Significant Safety Related)

A report will be submitted at the end of the refueling outage.

W. Gary Gates

W. Gary Gates
Manager
Fort Calhoun Station

Omaha Public Power District
1623 Harney Omaha, Nebraska 68102
402/536-4000

July 9, 1984
LIC-84-213

Mr. Richard C. DeYoung, Director
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Reference: Docket No. 50-285

Dear Mr. DeYoung:

June Monthly Operating Report

Please find enclosed ten (10) copies of the June Monthly Operating Report for the Fort Calhoun Station Unit No. 1.

Sincerely,

R L Andrews

R. L. Andrews
Division Manager
Nuclear Production

RLA/TPM:jmm

Enclosures

cc: NRC Regional Office
Office of Management & Program Analysis (2)
Mr. R. R. Mills - Combustion Engineering
Mr. T. F. Polk - Westinghouse
Nuclear Safety Analysis Center
INPO Records Center
NRC File

IE24
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