AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-285

UNIT Fort Calhoun Statio

DATE August 3, 1984

COMPLETED BY T. P. Matthews

TELEPHONE (402) 536-4733

MONTH July, 1984

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	0.0	17	158.3
2	0.0	18	172.0
3	0.0	19	244.1
4	0.0	20	275.0
5	0.0	21	275.7
6	0.0	22	309.7
7	0.0	23	74.1
8	0.0	24	377.0
9	0.0	25	453.7
10	0.0	26	455.2
11	0.0	27	456.7
12	0.0	28	456.3
13	75.6	29	456.1
14	95.9	30	456.8
15	97.4	31	444.6
16	123.6		

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 August 3, 1984
COMPLETED BY T. P. Matthews
TELEPHONE (402) 536-4733

OPERATING STATUS

1. Unit Name: Fort Calhoun Station
2. Reporting Period: July, 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: N/A

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	744.0	5,111.0	95,113.0
12. Number Of Hours Reactor Was Critical	543.9	2,034.1	71,928.0
13. Reactor Reserve Shutdown Hours	0.0	0.0	1,309.0
14. Hours Generator On-Line	442.2	1,931.7	71,334.2
15. Unit Reserve Shutdown Hours	0.0	0.0	0.0
16. Gross Thermal Energy Generated (MWH)	457,374.1	2,610,171.0	89,369,884.7
17. Gross Electrical Energy Generated (MWH)	104,784.0	831,042.0	29,148,611.0
18. Net Electrical Energy Generated (MWH)	130,985.5	787,522.0	27,867,390.7
19. Unit Service Factor	59.4	37.8	75.0
20. Unit Availability Factor	59.4	37.8	75.0
21. Unit Capacity Factor (Using MDC Net)	36.8	32.2	63.9
22. Unit Capacity Factor (Using DER Net)	36.8	32.2	61.6
23. Unit Forced Outage Rate	2.9	0.7	3.5

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: N/A
26. Units In Test Status (Prior to Commercial Operation): N/A

	Forecast	Achieved
INITIAL CRITICALITY		
INITIAL ELECTRICITY		
COMMERCIAL OPERATION		

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-285
UNIT NAME Fort Calhoun Station
DATE August 3, 1984
COMPLETED BY T. P. Matthews
TELEPHONE (402) 536-4733

REPORT MONTH July, 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	3162.5	C	4	N/A	XX	XXXXXX	1984 refueling outage commenced March 3, 1984. Unit returned to service July 12, 1984.
84-02	840722	F	16.3	H	3	N/A	XX	XXXXXX	The reactor tripped when spikes on two of the RPS channels were of sufficient magnitude to cause TM/LP trips. Noise suppressors and administrative controls were implemented as corrective action.

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 July 1984.

1. Scheduled date for next refueling shutdown. December 1985

2. Scheduled date for restart following refueling. February 1986

3. Will refueling or resumption of operation thereafter require a technical specification change or other license amendment? Unknown at this time

   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. November 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 August 1, 1984

OMAHA PUBLIC POWER DISTRICT
Fort Calhoun Station Unit No. 1

July, 1984
Monthly Operations Report

I. OPERATIONS SUMMARY

Fort Calhoun Station went critical for Cycle 9 on July 8, 1984. After several days of low power physics testing the generator was put online on July 12. Power was increased until the reactor tripped at 85% on July 22. The plant was back online on July 23 and reached 97% power on July 25. Power was reduced to 93% on July 30 for more reactor physics testing.

Operator Control Room Design Review interviews were completed during July, 1984.

No safety valve or PORV challenges occurred.

A. PERFORMANCE CHARACTERISTICS

LER Number	Deficiency
84-006	VIAS Actuation.
84-007	VIAS Actuation.
84-009	10 CFR 50.49 Testing of Conax Electrical Penetrations.

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-GT-2	Gai-Tronics and Nuclear Emergency Alarm Audibility Test for Containment. This procedure did not constitute an unviewed safety question as defined by 10CFR50.59 as it only involved personnel performing an audibility test of the nuclear emergency horns in containment.

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

Procedure	Description
SP-RRC-3	Reactivity Computer and Reactor Physics Constants Adjustments. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 since the procedure involved calibration of the reactivity computer used during physics testing.
SP-EEQ-1	LOCA Qualified Electrical Equipment Identification Check. This procedure did not constitute an unreviewed safety question because the procedure is simply an inventory list of qualified equipment and indications located on various panels and control boards in the plant. No plant operations or evolutions are involved.
SP-RC-2-1	Plugging Steam Generator Tubes. This procedure did not constitute an unreviewed safety question because the procedure only provided for the installation of plugs in tubes in "A" steam generator.
SP-WDS-18	NUPAC 14D-2.0 Cask Loading. This procedure did not constitute an unreviewed safety question because it only provided for loading material in a cask with appropriate radiation protection measures utilized.
SP-FAUD-1	Fuel Assembly Uplift Condition Detection. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 since it only involved the evaluation of data from a surveillance test to verify that a fuel assembly uplift condition did not exist.
SP-ECT-1	Eddy Current Testing of Heat Exchanger Tubes. This procedure did not constitute an unreviewed safety question because it only involved eddy current testing on non safety related feedwater heater numbers 4A, 4B, 5B and 6A.

System Acceptance Committee Packages for July, 1984:

Package	Description/Analysis
DCR 03-73	Replacement of Waste Gas Release Valves. This modification replaced a portion of the waste gas release system with new equipment that will automatically control and record the flow of waste gas vented to the atmosphere. This modification has no adverse effect on the safety analysis.
EEAR FC-79-205	Redundant Level Indication and Alarms for Auxiliary Feedwater System. This modification installed redundant level indication taps on the strong back to the auxiliary feedwater tank. This modification has no adverse effect on the safety analysis.
EEAR FC-80-88	Addition of Air Operated Valves to HCV-2908, 2918, 2928, 2937 and 2947. This modification provided for the addition of air operated valves on HCV-2908, 2918, 2928, 2937 and 2947 and modified the control scheme. This modification has no adverse effect on the safety analysis.
EEAR FC-81-09	Upgrading of Harsh Environment Safety Related Solenoids. This modification provided for upgrading safety related solenoids with qualified replacements. This modification has no adverse effect on the safety analysis.
EEAR FC-81-17B	Qualification of HCV-238 and HCV-239 Solenoids. This modification provided for raising these solenoids above flood level to ensure qualification of the valves. This modification has no adverse effect on the safety analysis.
EEAR FC-82-69	Installation of Gai-tronics. This modification provided for the installation of a gai-tronics station in the auxiliary building near the hand and foot portal monitor. 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 July, 1984: (Continued)

Package	Description/Analysis
EEAR FC-82-169	N2 Blanket on EFWST. This modification provided for supplying nitrogen to the emergency feedwater storage tank from another source. This modification has no adverse effect on the safety analysis.
EEAR FC-83-86	"D" RPS Channel APD Dual Potentiometer Failure. This modification provided for a new dual potentiometer module equivalent or better than the existing module. This modification has no adverse effect on the safety analysis.
EEAR FC-84-002	VA-43 A/B Heating Coils Replacement. This modification did not effect a safety related system; therefore, has no adverse effect on the safety analysis.
EEAR FC-84-69	MOV-SV-5 Replacement. This modification did not effect a safety related system; therefore, has no adverse effect on the safety analysis.

E. RESULTS OF LEAK RATE TESTS

Results of the B and C penentration tests will be submitted under separate cover.

F. CHANGES IN PLANT OPERATING STAFF

None

G. TRAINING

Training for the month of July included fire brigade training for operations and security personnel, annual requalification exam training for operators and systems training for maintenance personnel. The STA training program is in progress.

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

Package	Description/Analysis
Amendment No. 80	This amendment adds limiting conditions for operation and surveillance requirements for reactor coolant system vents and administrative requirements for sampling and analysis of plant effluents.
Amendment No. 81	This amendment adds operability and surveillance requirements for the containment wide range radiation monitors, wide range noble gas stack monitors and main steam line radiation monitor.

II. MAINTENANCE (Significant Safety Related)

Will be submitted in the August Monthly Operations Report.

W Gary Gates

W. Gary Gates
Manager
Fort Calhoun Station

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

August 3, 1984
LIC-84-263

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:

July Monthly Operating Report

Please find enclosed ten (10) copies of the July 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

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