

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-285
 UNIT Fort Calhoun Station
 DATE January 13, 1984
 COMPLETED BY D. J. Munderloh
 TELEPHONE 402-536-4001

MONTH December, 1983

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	<u>458.2</u>
2	<u>456.8</u>
3	<u>456.3</u>
4	<u>456.7</u>
5	<u>456.5</u>
6	<u>456.3</u>
7	<u>455.9</u>
8	<u>455.7</u>
9	<u>455.8</u>
10	<u>455.9</u>
11	<u>455.7</u>
12	<u>456.1</u>
13	<u>455.8</u>
14	<u>455.4</u>
15	<u>454.9</u>
16	<u>456.6</u>

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
17	<u>457.0</u>
18	<u>456.8</u>
19	<u>456.9</u>
20	<u>456.9</u>
21	<u>457.0</u>
22	<u>457.1</u>
23	<u>457.1</u>
24	<u>457.3</u>
25	<u>456.6</u>
26	<u>457.2</u>
27	<u>457.1</u>
28	<u>457.3</u>
29	<u>456.8</u>
30	<u>456.7</u>
31	<u>456.5</u>

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)

8401170226 831231
 PDR ADOCK 05000285
 R PDR

IE24

OPERATING DATA REPORT

DOCKET NO. 50-285
 DATE January 13, 1984
 COMPLETED BY D. J. Munderloh
 TELEPHONE 402-536-4001

OPERATING STATUS

1. Unit Name: Fort Calhoun Station
2. Reporting Period: December, 1983
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): 461
7. Maximum Dependable Capacity (Net MWe): 438
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	8,761.0	90,002.0
12. Number Of Hours Reactor Was Critical	744.0	6,503.9	69,893.9
13. Reactor Reserve Shutdown Hours	0.0	0.0	1,309.5
14. Hours Generator On-Line	744.0	6,405.0	69,402.5
15. Unit Reserve Shutdown Hours	0.0	0.0	0.0
16. Gross Thermal Energy Generated (MWH)	1,108,677.3	9,143,165.3	86,759,713.7
17. Gross Electrical Energy Generated (MWH)	356,570.0	2,894,090.0	28,317,569.7
18. Net Electrical Energy Generated (MWH)	339,672.5	2,749,833.5	27,079,867.9
19. Unit Service Factor	100.0	73.1	77.1
20. Unit Availability Factor	100.0	73.1	77.1
21. Unit Capacity Factor (Using MDC Net)	104.2	71.7	65.6
22. Unit Capacity Factor (Using DER Net)	95.5	65.7	63.3
23. Unit Forced Outage Rate	0.0	0.8	3.6

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

1984 refueling outage scheduled to start around March 3, 1984

25. If Shut Down At End Of Report Period, Estimated Date of Startup: N/A

26. Units In Test Status (Prior to Commercial Operation):	Forecast	Achieved
INITIAL CRITICALITY	_____	_____
INITIAL ELECTRICITY	_____	_____
COMMERCIAL OPERATION	_____	_____

UNIT SHUTDOWNS AND POWER REDUCTIONS

REPORT MONTH December, 1983

DOCKET NO. 50-285
 UNIT NAME Fort Calhoun Station
 DATE January 13, 1984
 COMPLETED BY D. J. Munderloh
 TELEPHONE 402-536-4001

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
									No unit shutdowns during the month of December, 1983.

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

Refueling Information
Fort Calhoun - Unit No. 1

Report for the month ending December 1983.

1. Scheduled date for next refueling shutdown. March 1984
2. Scheduled date for restart following refueling. May 1984
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?

A Technical Specification Change

- 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. Tech. Specs. - February 1984
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	<u>133</u>	assemblies
b) in the spent fuel pool	<u>265</u>	"
c) spent fuel pool storage capacity	<u>483</u>	"
d) planned spent fuel pool storage capacity	<u>728</u>	"
7. The projected date of the last refueling that can be discharged to the spent fuel pool assuming the present licensed capacity. 1985

Prepared by

J. K. Gager

Date January 1, 1984

OMAHA PUBLIC POWER DISTRICT
Fort Calhoun Station Unit No. 1

December, 1983
Monthly Operations Report

I. OPERATIONS SUMMARY

Fort Calhoun Station operated at a nominal 100% power throughout the month of December, 1983.

Fort Calhoun Station participated in the annual emergency exercise. The exercise involved full participation of the states and the NRC.

The areas of fire protection, radwaste transportation and design changes were audited by the Regional office of the NRC. The annual SALP review was conducted and a public meeting was held during December.

The spent fuel pool rerack continues.

Outage preparations continue for the 1984 refueling outage.

No safety valve or PORV challenges occurred.

A. PERFORMANCE CHARACTERISTICS

<u>LER Number</u>	<u>Deficiency</u>
83-008 Rev. 1	While at approximately 100% power and during the performance of monthly surveillance test ST-ESF-6, Section F.2, "Diesel Generator Check", diesel generator DG-2 started and was run up to operating speed. However, the generator field failed to flash and the control room had no indication that the diesel was running (that is, the diesel intake air dampers did not open, the diesel stop light did not go out, and the control room diesel tachometer had no RPM indication). Subsequently, diesel generator DG-2 was shutdown and the redundant diesel generator DG-1 was satisfactorily started to verify operability. The 161 KV and 345 KV supplies were also available throughout the incident. Revision 1 involved a change to Attachment 2, Corrective Action only.
83-010	During the performance of surveillance test, ST-ESF-5, Section F.1 (the monthly AC sequencer timer test), timer SI-1A (corresponding to the low pressure safety injection pump) on sequencer SI-2 failed to time out within the prescribed limit. During the time of the failure, the DC sequencer circuitry which activates SI-1A as well as the AC and DC sequencer circuitry actuating the redundant low pressure safety injection pump (SI-1B) was operable.

A. PERFORMANCE CHARACTERISTICS (Continued)

<u>LER Number</u>	<u>Deficiency</u>
83-011	While operating at approximately 100% power and during performance of annual surveillance test ST-ESF-6, F.5, "Diesel Generator Inspection", diesel generator DG-1 failed to field flash. Diesel generator DG-2 had been proven operable at the start of the surveillance test, ST-ESF-6, F.5. The 161 KV and 345 KV supplies were available throughout the incident.

B. CHANGES IN OPERATING METHODS

None

C. RESULTS OF SURVEILLANCE TESTS AND INSPECTIONS

<u>Operation Incident</u>		<u>Deficiency</u>
OI-1786	ST-ESF-14, F.2	ST-ESF-14, Section F.2 was not performed within the specified time frame.
OI-1787	ST-FP-9, F.1	During the performance of ST-FP-9, Section F.1, it was noticed that a bolt was missing from 4160 volt breaker SI-1A.
OI-1791	ST-ESF-3, F.2	During the performance of ST-ESF-3, Section F.2, it was found that pressure switches were out of tolerance.

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

<u>Procedure</u>	<u>Description</u>
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.

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

<u>Procedure</u>	<u>Description</u>
------------------	--------------------

SP-VA-80	Hydrogen Purge System Test.
----------	-----------------------------

This procedure is a quarterly test to verify the operability of the Hydrogen Purge System fans, valves and filters. It does not constitute an unreviewed safety question as defined by 10CFR50.59 since it does not adversely affect containment integrity; remote valve operation capability is maintained for one fan at all times; no Technical Specification basis is affected; and the system has no automatic emergency functions which are bypassed.

System Acceptance Committee Packages for December, 1983:

<u>Package</u>	<u>Description/Analysis</u>
----------------	-----------------------------

EEAR FC-83-53	Concentrate Line Improvements.
---------------	--------------------------------

This modification provided the installation of unions in the concentrate line from the drumming station to the concentrate tanks providing points of access to open the line for cleaning. This modification has no adverse effect on the safety analysis.

EEAR FC-81-135	QA Records Vault (QA Office).
----------------	-------------------------------

This modification provided the installation of an insulated false wall and suspended ceiling. This modification does not effect safety related equipment, therefore, it has no adverse effect on the safety analysis.

DCR 73-22	HI-LO Level Alarm on SIRWT.
-----------	-----------------------------

This modification increased the level of safety due to the fact that the level of the SIRWT is now more easily monitored. This modification has no adverse effect on the safety analysis.

DCR 76-61	Waste Evaporator Concentrate Cooler.
-----------	--------------------------------------

This modification provided the installation of a bypass around the concentrate cooler. This modification has no adverse effect on the safety analysis.

<u>Package</u>	<u>Description/Analysis</u>
DCR 77-89	<p>Change Out Valve Design on Steam Seal System.</p> <p>This modification replaced a steam valve with a valve that would not erode. This modification did not affect safety related equipment, therefore, it has no adverse effect on the safety analysis.</p>
EEAR FC-79-11	<p>PORV Low Pressure Operation Failure Mode.</p> <p>This modification eliminates the possibility of a LOCA caused by the opening of the PORV's due to a loss of power to the pressure recorders. This modification has no adverse effect on the safety analysis.</p>
DCR 77-76	<p>Electrical Penetration Leak Test Connection.</p> <p>This modification only moved the leak test gauges outside the penetration covers. This modification has no adverse effect on the safety analysis.</p>
DCR 74A-18	<p>Automatic Hydrazine Analyzer.</p> <p>This modification added an automatic analyzer for hydrazine off of the discharge of No. 6 heater and improves the chemistry monitoring for the steam generators. This modification did not affect safety related equipment, therefore, it has no adverse effect on the safety analysis.</p>
DCR 75A-58	<p>Tertiary Treatment Plant (Sewage Lagoons).</p> <p>This modification installed the tertiary treatment plant which is not safety related. This modification has no adverse effect on the safety analysis.</p>
EEAR FC-80-125	<p>Control of Gas Leaks from WD-32.</p> <p>This modification is only an extension of WD-32 and does not involve a safety related system or component. This modification has no adverse effect on the safety analysis.</p>

<u>Package</u>	<u>Description/Analysis</u>
EEAR FC-81-126	Warehouse Gas Cylinder Dock. This modification to the warehouse gas cylinder dock does not involve a safety related system or component. This modification has no adverse effect on the safety analysis.
DCR 78-35	Noise Spikes on RPS. This modification increased the reliability of the reactor protective system. This modification has no adverse effect on the safety analysis.
EEAR FC-83-12	Conversion of Turbine to Full Arc Admission. This modification to the turbine was made for ease of operation only. This modification did not affect safety related equipment, therefore, it has no adverse effect on the safety analysis.
DCR 74A-4D	Process Radiation Monitor RM-055A. This modification installed a second overboard discharge monitor, RM-055A, a redundant monitor. This modification has no adverse effect on the safety analysis.
DCR 74A-4G	Process Radiation Monitor RM-059. This modification replaced a process radiation monitor that had failed due to overheating with a better one and relocated it away from the evaporator. This modification has no adverse effect on the safety analysis.
EEAR FC-79-36	Drain Line for Suction Piping to Feedwater Pumps. This modification is to put drain lines on the suction headers to the feedwater pumps FW-4A, 4B and 4C. This is not a safety related section of piping. This modification has no adverse effect on the safety analysis.

E. RESULTS OF LEAK RATE TESTS

There are no new leak rate results to report.

F. CHANGES IN PLANT OPERATING STAFF

Effective December 15, 1983, Mr. James J. Fisicaro transferred to the Licensing Department. Mr. Michael E. Kallman, formerly a Shift Technical Advisor, replaced Mr. Fisicaro as Supervisor-Administrative Services and Security.

G. TRAINING

Training for December was conducted as scheduled in the areas of operator requalification (licensed and non-licensed operators), fire brigade, maintenance, crane operator, emergency plan and general employee training. Seven NRC licensed candidates participated in the Commission examination.

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

None

II. MAINTENANCE (Significant Safety Related)

None



W. Gary Gates
Manager
Fort Calhoun Station

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

January 13, 1984
LIC-84-016

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

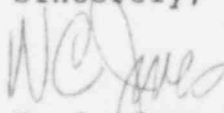
Reference: Docket No. 50-285

Dear Mr. DeYoung:

Monthly Operating Report
December, 1983

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

Sincerely,


W. C. Jones
Division Manager
Production Operations

WCJ/DJM: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
1/1