AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-285

UNIT Fort Calhoun Station

DATE June 13, 1984

COMPLETED BY T. P. Matthews

TELEPHONE (402) 536-4733

MONTH May, 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	0.0
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)

IE24

OPERATING DATA REPORT

DOCKET NO 50-285
DATE June 13, 1984
COMPLETED BY T. P. Matthews
TELEPHONE (402) 536-4733

OPERATING STATUS

1. Unit Name: Fort Calhoun Station
2. Reporting Period: May, 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): 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	3,647.0	93,649.0
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	40.8	75.7
20. Unit Availability Factor	0.0	40.8	75.7
21. Unit Capacity Factor (Using MDC Net)	0.0	41.1	64.6
22. Unit Capacity Factor (Using DER Net)	0.0	37.7	62.3
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: June 24, 1984
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 June 13, 1984
COMPLETED BY T. P. Matthews
TELEPHONE (402) 536-4733

REPORT MONTH May, 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	2157	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 May 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?

      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. 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 June 1, 1984

OMAHA PUBLIC POWER DISTRICT
Fort Calhoun Station Unit No. 1

May, 1984
Monthly Operations Report

I. OPERATIONS SUMMARY

Fort Calhoun Station was performing system startups for a return to power. During a hydrostatic leak test of the reactor coolant system on May 16, a steam generator tube ruptured causing a primary to secondary leak. Startup has been delayed while extensive eddy-current testing is performed on the steam generator tubes.

Two incidents occurred on May 21. A 4160/480 volt transformer failed causing a fire in the switchgear room. Later, a Blair City mower threw a downed wire into the 161 KV transmission line which feeds Fort Calhoun Station causing a temporary partial loss of offsite power.

Three operators are in hot license training. They will take the NRC Reactor Operator exam in early June.

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-2A).
	This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it only provides for plugging five tubes in "A" steam generator. Appropriate radiation protection and quality requirements have been addressed.

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

Procedure	Description
SP-SGAI-1	Steam Generator Annulus Inspection. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it only involved an inspection inside steam generator "A" while the plant was in a refueling outage.
SP-SGAI-1	Steam Generator Annulus Inspections. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it only involved an inspection inside steam generator "B" while the plant was in a refueling outage.
SP-SGDLT-1	Steam Generator Dye Leak Test. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it only provided for looking for a suspected leaking tube in "B" steam generator. The dye which was used met appropriate chemical requirements. Pressurization of the steam generator was in accordance with technical specifications and ASME codes.
SP-VA-80	Hydrogen Purge System Test. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it provides for flow measurement of the hydrogen purge system. No technical specifications are involved.
SP-UF6-1	Uranium Hexafluoride Storage Cylinder External Visual Inspection. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it provides for a routine inspection of storage cylinders and area.
SP-DW-2	Salt Cleaning of Anion Bed Resin. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 since the procedure was carried out using normal plant practices and procedure was adequate for work involved.

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

Procedure	Description
SP-SL-RC-2	Steam Generator Tubesheet Sludge Lancing.
	This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it only involved removal of sludge from the secondary side of steam generator "A" while the plant was in a refueling outage.
SP-SL-RC-2	Steam Generator Tubesheet Sludge Lancing.
	This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it only involved removal of sludge from the secondary side of steam generator "B" while the plant was in a refueling outage.
SP-TG-DROP-1	Generator Drop Test.
	This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it is not safety related equipment.

System Acceptance Committee Packages for May, 1984:

Package	Description/Analysis
EEAR FC-82-12	Power Receptacle.
	This modification provided for the installation of two 120V power receptacles in Room 19 and does not affect any safety equipment. This modification has no adverse effect on the safety analysis.
EEAR FC-82-137	Reactor Stud Tensioner Repair Air Line Addition.
	This modification provided for the installation of a compressed air line to the reactor stud tensioner/snubber repair area in Room 69 of the auxiliary building. This modification does not involve safety related equipment so it has no adverse effect on the safety analysis.

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

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

Package	Description/Analysis
EEAR FC-83-121	Gaitronics for Room 23. This modification provided for the installation of a gaitronics station in Room 23. This modification does not involve safety related equipment so it has no adverse effect on the safety analysis.
EEAR FC-83-179	Clarifier Desludge Isolation Valve Installation. This modification provided for the installation of an isolation valve to the clarifier desludge valves. This modification does not involve safety related equipment so it has no adverse effect on the safety analysis.
EEAR FC-80-10	Drain Valve Between SI-169 and SI-170. This modification provided for the installation of a drain valve to expedite draining of portions of the shutdown cooling system. This modification has no adverse effect on the safety analysis.
EEAR FC-80-99	Replacement of FCV-326 Valve. This modification replaced the existing valve with a superior quality valve. This modification has no adverse effect on the safety analysis.
EEAR FC-81-172	Access Platforms for Reactor Coolant Pump Motors RC-3A, RC-3B, RC-3C and RC-3D. This modification provided for the fabrication and installation of platforms to provide maintenance access to RC-3A, RC-3B, RC-3C and RC-3D reactor coolant pump motors. 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 May, 1984: (Continued)

Package	Description/Analysis
EEAR FC-83-159 (SRDCO 84-32)	Steam Generator A Rim Cut No. 8 Support Plate. This modification provided for removing the outer rim and lugs of the No. 8 drilled tube support plates in the steam generators. Such a design change would provide stress relief for the plates and associated tubes such that the steam generator tube denting rate can be decreased. This modification has been reviewed by the NSSS supplier. This modification has no adverse effect on the safety analysis.
EEAR FC-83-159 (SRDCO 84-33)	Steam Generator B Rim Cut No. 8 Support Plate. This modification provided for removing the outer rim and lugs of the No. 8 drilled tube support lplates in the steam generators. Such a design change would provide stress relief for the plates and associated tubes such that the steam generator tube denting rate can be decreased. This modification has been reviewed by the NSSS supplier. This modification has no adverse effect on the safety analysis.
EEAR FC-81-99 Part 14	Installation of Heated Junction Thermocouple (HJTC) Handling Canister Mounting Bracket and Assembly of HJTC Handling Canisters. This modification provided for the assembly and installation of the HJTC handling canister mounting bracket and handling canisters. This modification has no adverse effect on the safety analysis.
EEAR FC-82-96	Auxiliary Building Room 4A Ventilation. This modification provided for additional ventilation duct in Room 4A. 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 May, 1984: (Continued)

Package	Description/Analysis
EEAR FC-83-51	HCV-1103/1104/1385/1386 Valve Operators Motor Brakes. This modification was essentially a one-for-one replacement of valve operator motor brakes on HCV-1385 and HCV-1386 and the replacement of valve operator motor brakes operating coils on HCV-1103 and HCV-1104 in order to enhance reliability. This modification has no adverse effect on the safety analysis.
EEAR FC-78-74	HCV-1041/1042 Control Circuitry. This modification provided for the addition of a separate control switch for HCV-1041 and HCV-1042. This modification has no adverse effect on the safety analysis.
EEAR FC-83-144	PAL Door Hinges. This modification provided for the installation and adjustment of Personnel Air Lock (PAL) door hinges. This modification has no adverse effect on the safety analysis.
EEAR FC-83-69	Hydrogen Purge Filter Test Port. This modification provided for the installation of a test port on the hydrogen purge filter housing (VA-82) located in Room 59 of the auxiliary building. This modification has no adverse effect on the safety analysis.
EEAR FC-82-115	Remove Concrete Interference to FW-5A Suction Piping. This modification provided for the removal of floor concrete to widen the suction piping trenches to heater drain pumps FW-5A, 5B, and 5C (located in the basement of the turbine building); and then fabricated and installed a replacement structural steel floor system. This modification did not affect safety related equipment so it has no effect on the safety analysis.

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

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

Package	Description/Analysis
EEAR FC-80-35	PORV Internals.
	This modification replaced the existing PORV internals with those of a different seat design to improve leak tightness of the valves. The operation of the valves was not changed. This modification has no adverse effect on the safety analysis.
EEAR FC-84-74	Fuse Protection of Certain Limit Switch Circuits.
	This modification enhanced reliability of valve operation during accident conditions by allowing the valve to be operated even though position indication may be lost. This modification has no adverse effect on the safety analysis.
EEAR FC-84-004	Upgrade Limit Switches in Room 81.
	This modification replaced the existing limit switches with switches of superior quality. This modification has no adverse effect on the safety analysis.
EEAR FC-79-81	Variable Setpoint for PORV Actuation.
	This modification provides for the design and installation of the variable setpoint for PORV actuation to ensure limiting conditions are not violated, provides for more flexibility in operating conditions thereby decreasing the chances of unnecessarily lifting the PORV's and retains the high reliability in the RCS overpressurization protection function of the PORV circuitry. This modification has no adverse effect on the safety analysis.
EEAR FC-81-99, Part 13	Installation of Heated Junction Thermocouple (HJTC) Probe.
	This modification provided for the receipt inspection or preoperational checkout and installation of unirradiated HJTC probes and the proper makeup and venting of the HJTC Grayloc flange in preparation for operation of the plant. 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 May, 1984: (Continued)

Package	Description/Analysis
EEAR FC-83-01	Replace Panel Mounted Indicators with Qualified Instruments. This modification provided for a one-for-one functional replacement for the existing system. This modification has no adverse effect on the safety analysis.
EEAR FC-83-44	Differential Steam Generator Pressure Trip Module. This modification provided for the installation of new components and modification of the reactor protective system to provide the new Asymmetric Steam Generator Transient (ASGT) trip functions. This modification has no adverse effect on the safety analysis.
DCR 74A-21F	Steam Generator Blowdown-Condensate 1" Bypass Valve. This modification provided for the installation of the new bypass valve to aid the plant operators to fill the piping for the condensate supply to the heater exchanger FW-44. This modification did not involve safety related equipment so it has no effect on the safety analysis.
EEAR FC-83-32	Qualification of Foxboro Transmitters. This modification replaced the existing amplifier assemblies in Foxboro transmitters with one-for-one function replacements. This modification has no adverse effect on the safety analysis.
EEAR FC-83-146	Relocation of PT-105. This modification relocated Foxboro transmitter PT-105 and renumbered GEMAC transmitter PT-105 and its instrument loop to avoid confusion of two devices with the same tag. 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

During May, general employee training was increased to support outage requirements. Operations department received Cycle 9 modifications training. NRC license candidates received increased training in preparation of an exam to be administered in June. Advanced Technologies, Inc., was awarded a contract to support the accredidation process for training required by INPO. Three programs in the operations area are being upgraded in 1984. Seven other programs are being scheduled for upgrade at a later time.

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

Amendment No.	Description
Amendment 78	Incorporates administrative changes which achieve agreement between the Fort Calhoun Technical Specifications and new regulations which became effective January 1, 1984. This rule change affects Technical Specifications in the area of shift manning and the same amendment also changes to whom the Quality Control personnel report. This change permits Quality Control personnel to report to the Technical Supervisor.
Amendment 79	This amendment provides an up-to-date identification of the accessibility of safety related system hydraulic snubbers (Table 2-6A).

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

June 12, 1984
LIC-84-177

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:

May Monthly Operating Report

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

Sincerely,

W C Jones

W. C. Jones
Division Manager
Production Operations

WCJ/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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