AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-285
UNIT Fort Calhoun Station
DATE March 8, 1983
COMPLETED BY T. P. Matthews
TELEPHONE (402)536-4733

MONTH February, 1983

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	
14	0.0	30	
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 March 8, 1983
COMPLETED BY T. P. Matthews
TELEPHONE (402)536-4733

OPERATING STATUS

1. Unit Name: Fort Calhoun Station
2. Reporting Period: February, 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): 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	672.0	1,416.0	82,657.0
12. Number Of Hours Reactor Was Critical	0.0	0.0	64,110.5
13. Reactor Reserve Shutdown Hours	0.0	0.0	1,309.5
14. Hours Generator On-Line	0.0	0.0	62,947.5
15. Unit Reserve Shutdown Hours	0.0	0.0	0.0
16. Gross Thermal Energy Generated (MWH)	0.0	0.0	77,616,548.4
17. Gross Electrical Energy Generated (MWH)	0.0	0.0	25,735,333.5
18. Net Electrical Energy Generated (MWH)	0.0	0.0	24,330,034.4
19. Unit Service Factor	0.0	0.0	76.2
20. Unit Availability Factor	0.0	0.0	76.2
21. Unit Capacity Factor (Using MDC Net)	0.0	0.0	63.9
22. Unit Capacity Factor (Using DER Net)	0.0	0.0	63.6
23. Unit Forced Outage Rate	0.0	0.0	3.9

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: March 20, 1983
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 March 8, 1983
COMPLETED BY T. P. Matthews
TELEPHONE (402)536-4733

REPORT MONTH February, 1983

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
82-06	921206	S	2022	C	4	N/A	XX	XXXXXX	1982/1983 refueling outage commenced December 6, 1982.

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 February 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.	Methodology - Dec. 1983 Tech. Specs.- Feb. 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	133 assemblies
	b) in the spent fuel pool	265 "
	c) spent fuel pool storage capacity	483 "
	d) planned spent fuel pool storage capacity	728 "
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 Gasper Date March 1, 1983

OMAHA PUBLIC POWER DISTRICT
Fort Calhoun Station Unit No. 1

February, 1983
Monthly Operations Report

I. OPERATIONS SUMMARY

Fort Calhoun Station continued refueling outage operations during the month of February.

Several important jobs which were completed or are in progress during February include:

1. The ten year reactor vessel internals inspection was completed. First review of the ultrasonic and visual inspections has not identified any defects.
2. Two new neutron fluence surveillance capsules were installed in the reactor vessel. These surveillance capsules will provide more accurate data on the effectiveness of the new core loading to reduce neutron fluence on the reactor vessel. Installation efforts were delayed a couple of days due to improper fitting of the surveillance capsules.
3. The core support barrel has been re-installed in the reactor vessel.
4. The turbine generator low pressure turbine has been re-assembled. The high pressure turbine rotor has arrived on site.
5. Eddy current and profilometry tests were performed on selected control element assemblies.
6. Two reactor coolant pump motors were pulled.
7. Eddy current tests were performed on a shutdown cooling heat exchanger. One tube was plugged.

No safety valve or PORV challenges occurred.

A. PERFORMANCE CHARACTERISTICS

LER Number	Deficiency
82-020	During performance of ST-MSSV-1, F.1, Main Steam Safety Valves Test, it was discovered that 4 out of 10 main steam safety valves had lift setpoints that were out-of-tolerance. Technical Specification 2.1.6(3) is applicable. Two of the 4 safety valves (MS-275 and MS-280) were discovered to have lower than required lift setpoints and the other 2 valves (MS-278 and MS-282) had higher than required lift setpoints.

B. CHANGES IN OPERATING METHODS

None.

C. RESULTS OF SURVEILLANCE TESTS AND INSPECTIONS

Surveillance tests as required by the Technical Specifications Section 3.0 and Appendix B, were performed in accordance with the annual surveillance test schedule. The following is a summary of the surveillance tests which resulted in Operation Incidents and are not reported elsewhere in the report:

Operation Incidents		Deficiency
OI-1642	ST-ENV-1	River Cooling Water Temperature Recorder Malfunction
OI-1635	ST-SI/CS-1, F.1	After performance of ST-SI/CS-1, F.1 the test was not reviewed within 96 hours after the test was completed.
OI-1645	ST-NDWC-1, F.2	During the performance of surveillance test ST-NDWC-1, F.2 temperature elements TE-732A, TE-733A, TE-734A failed to open circuits at the RTD.
OI-1633	ST-DG-2, F.4 and F.6	During performance of surveillance test, the diesel speed sensing switch interlock was found to be operating out-of-specification tolerance as was the computer trip setpoint for DG-2 fuel oil transfer pump No. 2, high pressure alarm.
OI-1654	ST-ESF-6, F.2	During the performance of ST-ESF-6, F.2, Appendix E diesel generator No. 1 did not start within the required 10 seconds.

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

See Page Four.

E. RESULTS OF LEAK RATE TESTS

All leak rate tests performed during January and February will be reported to the PRC per a special report (which concerns itself with all CONT-2, 3, 7 and associated tests). This special report will be submitted to the PRC prior to startup.

F. CHANGES IN PLANT OPERATING STAFF

One operator with a Reactor Operators License resigned in February.

G. TRAINING

Training included operator training on Cycle 8 modifications. Fire brigade training and drills and several general employee classes to support outage personnel. Maintenance training on hydrotesting and crane operations were conducted for various crafts.

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

Amendment No. 67 revises limits for reactor coolant and steam generator coolant radioactivity.

Amendment No. 68 revises limiting conditions for operations and surveillance requirements for containment purge isolation valves.

II. MAINTENANCE (Significant Safety Related)

Refueling Outage Maintenance will be submitted as one package in April, 1983.

System Acceptance Meeting Packages 1983:

Package	Description/Analysis
EEAR/DCR FC-81-106	Circulating Water System Sanding-Sheet Pile Alignment Wall The Alignment Wall will serve to decrease the amount of sanding of the intake structure which will increase the ability to maintain plant cooling. This modification will have no effect on plant safety or the safety analysis.
EEAR/DCR FC-82-46	FW-10 Mechanical Trip Linkage Position Indication This addition is for the back pressure trip alarm on FW-10, this will add an alarm in the control room. System and plant safety should be increased This modification will increase plant safety and consequently no effect on the safety analysis.
EEAR/DCR FC-79-182, A, B, D	Plant Shielding for Post Accident Operation The purpose of the shield walls in the Control Room complex, pipe penetration area (Room 59) and the RWD control area is to increase access time in post-accident conditions. The walls have been seismically designed and classified as Catagory I structure in accordance with FSAR Volume II, Section 5.2, 5.11 and Appendix F. This modification will increase plant safety and consequently no effect on the safety analysis.
EEAR/MR FC-81-159	VA-66 Chronic Flow Problems The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in Section 14.18 of the Updated Safety Analysis Report has not changed. The capability of the Auxiliary Building ventilation system is to handle airborne radiation as described in Section 9.10.2.1 and 9.10.3.1 of USAR has been increased by this modification. This modification will increase plant safety and consequently no effect on the safety analysis.

System Acceptance Meeting Packages 1983 (continued)

Package	Description/Analysis
EEAR FC81-177/ SRDCO 82-55	SI-1B Casing Vent Valve The probability of occurrence or consequences of accidents or malfunctions of equipment important to the safety previously evaluated in Section 14.15 of the Updated Safety Analysis Report will be unchanged. The existing vent valve (SI-335) and the added valve (SI-372) are in series, and are normally closed unless the lower pressure safety injection pump (SI-1B) is being primed after being drained. The existing valve (SI-335) is defined as a Class II pressure boundary. Leakage during pump operation will move this pressure boundary to the added valve (SI-372) will be the new Class II pressure boundary. This modification will have no effect on plant safety or the safety analysis.
EEAR FC-79-190B/ DCO 81-528	Wide Range Radiation Monitor/Access Platform This platform is a non-critical quality element, however, it has been designed to Fort Calhoun Station seismic criteria as defined in the FSAR. This modification will have no effect on plant safety or the safety analysis.
EEAR FC-79-190B/ DCO 81-516	High Range Noble Gas and Effluent Monitor The High Range Noble Gas Monitor and Sampling System for particulates and iodines is required by NUREG-0737, Section II.F.1, and the the system is not safety related. However, the isokinetic nozzle assembly which is attached to the ventilation stack is seismically designed and classified as Category I structure in accordance with FSAR Volume II, Section 5.2, 5.11, and Appendix F. This modification will have no effect on plant safety or the safety analysis.

System Acceptance Meeting Packages (Continued)

PACKAGE	Description/Analysis
EEAR FC-82-05	Relay Test Plugs in AI-46 This modification provides a permanent DC source for relay testing which increases personal safety but in no way effect plant safety or the safety analysis.
DCR 74B-21	Stator Cooling Piping Restraints This change effects only the Stator Cooling Piping This modification will have no effect on plant safety or the safety analysis.
DCR 74B-19	Condenser Recirculation Restraints This change effects the heater drain pump recirculation line and has no effect on plant safety or the safety analysis.
DCR 75B-34	Chemical Basket Relocation in Containment This modification only moved the basket to a different location. There was no chemical removed that was not replaced, therefore, no effect on the safety analysis. or plant safety.
DCR 75A-38	CEA Flow Plug Modification This modification only filed off the edges of CEA flow plugs to eliminate interference with grapple of the fuel handling machine. This modification will have no effect on plant safety or the safety analysis.
DCR 75A-19	Waste Filters WD-17A/B This modification replaced a 20 micron filter element with a 50 micron element so that they would not plug up as quickly. This modification will have no effect on plant safety or the safety analysis.

System Acceptance Meeting Packages (Continued)

Package	Description/Analysis
DCR 75A-15	Refueling Machine FH-1 Wire Modification This modification improves the operation of the refueling machine during various stages of its operation and has no effect on plant safety or the safety analysis.
DCR 75A-3	Radiation Exposure Near RC-2B Entrance This change involves installing shield walls for radiation purposes only. This modification will have no effect on plant safety or the safety analysis.
EEAR FC-82-104	ILRT Manifold Improvements This piping system will only be used and functionally connected to the reactor containment when the reactor is in a shutdown condition. Following the completion of the Type A test, the air supply piping will be disconnected from the containment and the containment penetration isolation. This modification will have no effect on plant safety or the safety analysis.
EEAR FC-82-29	Removal of Sigma II Computer The modification did not involve any safety related equipment and will have no effect on plant safety or the safety analysis.
EEAR FC-82-42	Refueling Machine Underwater TV System This new TV system is identical to the existing system and does not require any changes in the Fort Calhoun Station structure or the components. It is a replacement with better functional and operational system. This modification will have no effect on plant safety or the safety analysis.

System Acceptance Meeting Packages (Continued)

Package	Description/Analysis
EEAR FC-82-49	Downstream Isolation for HCV-2500/3 This modification will allow isolation of valves HCV-2500, -2501, -2502, -2503, for repair in the event of valve packing or seal leaking, thus, the probability of occurence or the consequences of an accident or malfunction of equipment previously evaluated in the Safety Analysis Report will not be increased nor will plant safety be effected.
EEAR FC-82-156	Laundry Room Modification This modification did not involve changes to a safety related system. Completion of the mod significantly reduced the amount of liquid wastes produced by the laundry room. This modification will have no effect on plant safety or the safety analysis.

W. G. Gates

W. G. Gates
Manager
Fort Calhoun Station