AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-285
UNIT Fort Calhoun Station
DATE February 7, 1984
COMPLETED BY T. P. Matthews
TELEPHONE (402) 536-4733

MONTH January, 1984

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	456.2	17	454.6
2	456.4	18	454.5
3	455.9	19	454.6
4	458.5	20	454.5
5	458.4	21	454.7
6	457.8	22	454.6
7	456.4	23	455.2
8	454.4	24	455.0
9	454.7	25	454.9
10	455.4	26	455.1
11	455.7	27	455.0
12	455.9	28	454.7
13	455.2	29	454.3
14	454.5	30	454.6
15	454.8	31	454.7
16	454.9		

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 February 7, 1984
COMPLETED BY T. P. Matthews
TELEPHONE (402) 536.4733

OPERATING STATUS

1. Unit Name: Fort Calhoun Station
2. Reporting Period: January, 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	744	90,746.0
12. Number Of Hours Reactor Was Critical	744	744	70,637.9
13. Reactor Reserve Shutdown Hours	0.0	0.0	1,309.0
14. Hours Generator On-Line	744	744	70,146.5
15. Unit Reserve Shutdown Hours	0.0	0.0	0.0
16. Gross Thermal Energy Generated (MWH)	1,105,887.8	1,105,887.8	87,865,601.5
17. Gross Electrical Energy Generated (MWH)	355,562.0	355,562.0	28,673,131.0
18. Net Electrical Energy Generated (MWH)	338,780.8	338,780.0	27,418,648.7
19. Unit Service Factor	100.0	100.0	77.3
20. Unit Availability Factor	100.0	100.0	77.3
21. Unit Capacity Factor (Using MDC Net)	104.0	104.0	65.8
22. Unit Capacity Factor (Using DER Net)	95.0	95.0	63.5
23. Unit Forced Outage Rate	0.0	0.0	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

DOCKET NO. 50-285
UNIT NAME Fort Calhoun Station
DATE February 7, 1984
COMPLETED BY T. P. Matthews
TELEPHONE (402) 536-4733

REPORT MONTH January, 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
									No unit shut downs during the month of January, 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 January 1984 .

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 '84

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 February 1, 1984

OMAHA PUBLIC POWER DISTRICT
Fort Calhoun Station Unit No. 1

January, 1984
Monthly Operations Report

I. OPERATIONS SUMMARY

Fort Calhoun Station operated at a nominal 100% power throughout the month of January, 1984, continuing its record on-line run. New fuel for Cycle 9 core load began arriving on site January 26, 1984. The installation of the new spent fuel racks was completed during January. Preparations for the 1984 refueling outage continue.

No safety valve or PORV challenges occurred.

A. PERFORMANCE CHARACTERISTICS

LER Number	Deficiency
83-004, Rev. 1	During performance of surveillance test ST-ESF-3, F.2, "Containment Pressure Channel Check", pressure switches A/PC-742-1 and A/PC-742-2 were found to be initiating/actuating above the Tech. Spec. limit of 5 psig. During the time pressure switches A/PC-742-1 and A/PC-742-2 were considered inoperable, the remaining pressure switches feeding the B, C and D channels of the "A" and "B" Containment Pressure High Signal initiation matrices were operable, available and fully capable of initiating protective actions required to mitigate the consequences of an accident. This revision depicts changes made to the Corrective Action section (Attachment 2) and Failure Data section (Attachment 3) of the original LER submittal.
83-013	During performance of surveillance test ST-ESF-3, F.2, "Containment Pressure Channel Check", pressure switches A/PC-742-1 and A/PC-742-2 were found to be initiating/actuating above the Tech. Spec. limit of 5 psig. During the time pressure switches A/PC-742-1 and A/PC-742-2 were considered inoperable, the remaining pressure switches feeding the B, D and D channels of the "A" and "B" Containment Pressure High Signal (CPHS) initiation matrices were operable, available and fully capable of initiating designed protective actions required to mitigate the consequences of an accident.

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-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-ATCOR-1	Compare Electrical Drawings with As Built Condition. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 since it only involved work on a non-safety related system.
SP-CAPSULE-1	Separation of Surveillance Capsule Wall Assembly. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it only provided for disassembly of a surveillance capsule once it was removed from the reactor vessel.
SP-CAPSULE-2	Reactor Vessel Surveillance Capsule Installation. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it only provided for the safe and orderly insertion of a surveillance capsule assembly into the reactor vessel during refueling operation. The use of surveillance capsules to monitor the effect neutron flux has on the reactor vessel wall is a Technical Specification requirement.
SP-CTPC-1	Core Thermal Power Calculation. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it simply consisted of recording data from existing plant instrumentation.

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

Procedure	Description
SP-ECT-2	Eddy Current Testing of Shutdown Cooling Heat Exchange Tubes. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it only provided for eddy current testing of tubes in a shutdown cooling heat exchanger. The test was performed during refueling shutdown when the reactor core was off loaded and the shutdown cooling heat exchanger was not required to be in service.
SP-FE-10	Inspection of Spent Fuel Using CE Curb Mounted Equipment. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 since all fuel handling was in compliance with approved procedures and Technical Specification limitations. The existing safety analysis applied to this work.
SP-GCASK-1	Shipment of CNSI Cask. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 because this procedure only provides for loading a cask with a surveillance capsule. The cask was handled with a single failure proof crane and appropriate radiation protection measures were utilized.
SP-GCASK-1	Shipment of CNSI Cask. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 because this procedure only provides for shipment of radioactive material in an approved cask.
SP-NI-6	Nuclear Instrumentation Decalibration and Heat Rate Optimization Test. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 because all data was obtained during typical plant operating conditions covered by accident analyses and Technical Specifications.

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

Procedure	Description
SP-PRCPT-1	Post Refueling Core Physics Testing. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 because all testing was performed in accordance with Technical Specifications and was consistent with startup during previous cycles.
SP-RCS-LE-1	Visual Inspection of Class I Piping in 10 Year ISI. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 since it only provided guidelines for performing a visual leak examination of RCS piping hydrostatically tested under a surveillance test.
SP-RRC-3	Reactivity Computer and Reactor Physics Constants Adjustments. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 because the procedure simply consisted of recording data from existing plant instrumentation.
SP-SHAP-1	Shape Annealing Factor Verification. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 because all reactor power and axial shape maneuvers were performed within limits described by the Station's Technical Specifications and safety analysis report.
SP-UF6-2	Off Loading of UF6 Containers in Storage Areas. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it only provided a safe and orderly procedure for off-loading UF6 storage cylinders into the UF6 storage area which has been approved by the NRC.

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

Package	Description/Analysis
SP-WDS-16	Spent Resin Disposal. This procedure did not constitute an unreviewed safety question as defined by 10CFR50.59 as it only provided an alternate method for sampling a waste gas decay tank. Technical Specifications dealing with radioactive gas release have been addressed.

System Acceptance Committee Packages for January, 1984:

Package	Description/Analysis
DCR 74A-27	Main Steam Isolation Valves. This modification completed a manufactuer's request to modify the main steam isolation valves to prevent stress created upon power closure. This modification has no adverse effect on the safety analysis.
DCR 74A-78	Bantam Crane Seismic Tiedown. This modification installed hold downs for the bantam crane on top of the containment. This modification has no adverse effect on the safety analysis.
DCR 77-11	Request for a Bathroom in Warehouse. This modification installed a bathroom in the warehouse and is not safety related. This modification has no adverse effect on the safety analysis.
DCR 77-106	Installation of FM Radio System. This modification installed an emergency FM radio communications system. All equipment is powered by converience outlets and is not safety related. This modification has no adverse effect on the safety analysis.
EEAR FC-83-95	Circuit Noise on LC-101X and LC-101Y. This modification reduced circuit noise only. This modification has no adverse effect on the safety analysis.

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

Package	Description/Analysis
EEAR FC-83-98	PIC-1538 Contact Rearrangement. This modification involved a high pressure alarm indicator and rearranging the contacts did not change its operation. This modification has no adverse effect on the safety analysis.
EEAR FC-83-102	Access Platforms for Inlet Cells to Intake Structure. This modification provided access platforms for inlet cells to the intake structure. This modification has no adverse effect on the safety analysis.
DCR 74A-29	Automatic Gas Analyzer. This modification installed an automatic waste gas analyzer that is no longer used. This modification has no adverse effect on the safety analysis.
DCR 74B-01	FRC-269X Modification. This modification changed a response based on a square root to a linear response and no operational aspects were changed. This modification has no adverse effect on the safety analysis.
DCR 75A-36	SIRWT Level RAS Switches. This modification improved the repeatability of the SIRWT RAS level switches. This modification has no adverse effect on the safety analysis.
DCR 76-105	Steam Generator Blowdown Flow Transmitters. This modification installed blowdown flow transmitters which improved the control system due to better indication of the flow through the system. This modification has no adverse effect on the safety analysis.

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

Package	Description/Analysis
DCR 78-36	Noise Spikes in Plant Instrumentation. This modification only installed noise spike suppressing devices across the relay coils of specific valves. Operation of the valves or associated system was not effected. This modification has no adverse effect on the safety analysis.
EEAR FC-79-111	Instrument and Control Shop Modification. This modification expanded the Instrument and Control Shop only and is not safety related. This modification has no adverse effect on the safety analysis.
EEAR FC-80-111	Waste Holdup Tank Instrument Line. This modification is intended to provide a recirculation path so that the gas space may be circulated through a charcoal filter. This would provide a method of removing noble gases from the gas space so the tank could be opened to the atmosphere for inspection. This modification has no adverse effect on the safety analysis.
EEAR FC-80-138	Storeroom Power Cable. This modification rerouted a power cable in the storeroom in conduit. This modification has no adverse effect on the safety analysis.

E. RESULTS OF LEAK RATE TESTS

Procedure	Results
ST-CONT-2, F.2	PAL 60 psig test - 3800 cc's (700 cc's greater than previous test).

F. CHANGES IN PLANT OPERATING STAFF

None

G. TRAINING

Training for January was conducted as scheduled in the areas of operator requalification (licensed and non-licensed operators), fire brigade, maintenance, crane operator and the emergency plan. General employee initial and requalification training program schedule was increased to accommodate the additional manpower required for the upcoming Fort Calhoun Station refueling outage. Operator training was conducted on the ERF computer being installed in the control room and Technical Support Center.

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

Package	Description
Amendment No. 76	The amendment incorporates administrative changes which correct terminology in the basis section concerning pressurizer operability, clarify the basis section for the diesel generator fuel oil inventory, clarify the basis section of shock suppressors (snubbers) specifications, clarify the scope of the inservice inspection program, correct references to DNB parameters and environmental sampling data, remove reference to an offsite organization figure which was deleted in a prior amendment, changes the title of a Safety Audit and Review Committee member, and changes the titles of other OPPD support staff members. The amendment also increases the audit frequency of the Emergency Plan, Site Security Plan, and Safeguards Contingency Plan from at least once per two years to at least once every twelve months.

II. MAINTENANCE (Significant Safety Related)

None

W Gary Gates

W. Gary Gates
Manager
Fort Calhoun Station

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

February 14, 1984
LIC-84-049

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:

January Monthly Operating Report

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

Sincerely,

WC 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

IE24
1/1