

Omaha Public Power District
1623 Harney Omaha, Nebraska 68102-2247
402/536-4000

April 16, 1990
LIC-90-0316

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

- References:
1. Docket No. 50-285
 2. Letter from NRC (L. M. Muntzing) to OPPD (T. E. Short) dated September 22, 1977

Gentlemen:

SUBJECT: March Monthly Operating Report (MOR)

Pursuant to Technical Specification Section 5.9.1, and 10 CFR Part 50.4(b)(1), please find enclosed one copy of the March 1990 Monthly Operating Report for the Fort Calhoun Station Unit No. 1.

In accordance with Reference 2, Omaha Public Power District (OPPD) includes a summary of major safety-related maintenance as part of the MOR. In recent MORs, OPPD has provided a computer printout of maintenance data as an attachment to the report. Since a description of significant safety-related maintenance is included in the Operations Summary, these maintenance printouts are not considered necessary and will no longer be included in the MOR.

If you should have any questions, please contact me.

Sincerely,

W. G. Gates

W. G. Gates
Division Manager
Nuclear Operations

WGG/pjc

c: LeBoeuf, Lamb, Leiby & MacRae
R. D. Martin, NRC Regional Administrator, Region IV
P. H. Harrell, NRC Senior Resident Inspector
R. M. Caruso - Combustion Engineering
R. J. Simon - Westinghouse
Office of Management & Program Analysis (2)
Nuclear Safety Analysis Center
INPO Records Center
American Nuclear Insurers

9004190251 900331
PDR ADJOCK 05000285
R PIC

TE24
!!!

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-285
 UNIT Fort Calhoun Station
 DATE April 12, 1990
 COMPLETED BY D. L. Stice
 TELEPHONE (402)636-2474

MONTH March 1990

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	0	17	0
2	0	18	0
3	0	19	0
4	0	20	0
5	0	21	0
6	0	22	0
7	0	23	0
8	0	24	0
9	0	25	0
10	0	26	0
11	0	27	0
12	0	28	0
13	0	29	0
14	0	30	0
15	0	31	0
16	0		

INSTRUCTIONS

On this form, list the average daily unit power level in MWe-Net for each day in the reporting month. Compute to the nearest whole megawatt.

OPERATING DATA REPORT

DOCKET NO. 50-285
 UNIT Fort Calhoun Station
 DATE April 12, 1990
 COMPLETED BY D. L. Stice
 TELEPHONE (402)636-2474

OPERATING STATUS

- | | |
|--|-------|
| 1. Unit Name: Fort Calhoun Station | Notes |
| 2. Reporting Period: March 1990 | |
| 3. Licensed Thermal Power (Mwt): 1500 | |
| 4. Nameplate Rating (Gross MWe): 502 | |
| 5. Design Electrical Rating (Net MWe): 478 | |
| 6. Maximum Dependable Capacity (Gross MWe): 502 | |
| 7. Maximum Dependable Capacity (Net MWe): 478 | |
| 8. If changes occur in Capacity Ratings (Item Numbers 3 through 7) Since Last Report, Give Reasons:
N/A | |
| <hr/> | |
| 9. Power Level to Which Restricted, If Any (Net MWe): N/A | |
| 10. Reasons for Restrictions, If Any: N/A | |
| <hr/> | |

	This Month	Yr-to-Date	Cumulative
11. Hours in Reporting Period	744.0	2,160.0	144,770.0
12. Number of Hours Reactor was Critical	0.0	1,142.1	112,308.4
13. Reactor Reserve Shutdown Hours	0.0	0.0	1,309.5
14. Hours Generator On-Line	0.0	1,130.6	111,136.0
15. Unit Reserve Shutdown Hours	0.0	0.0	0.0
16. Gross Thermal Energy Generated (MWH)	0.0	1,675,900.7	145,291,011.9
17. Gross Electrical Energy Generated (MWH)	0.0	546,866.0	47,756,974.2
18. Net Electrical Energy Generated (MWH)	0.0	521,906.2	45,589,459.0
19. Unit Service Factor	0.0	52.3	76.8
20. Unit Availability Factor	0.0	52.3	76.8
21. Unit Capacity Factor (Using MDC Net)	0.0	50.5	68.4
22. Unit Capacity Factor (Using DER Net)	0.0	50.5	66.8
23. Unit Forced Outage Rate	0.0	0.0	2.9
24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each): The 12th Refueling Outage commenced on February 17, 1990 with startup scheduled for May 13, 1990.			
25. If Shut Down at End of Report Period, Estimated Date of Startup: May 13, 1990			
26. Units In Test Status (prior to Commercial Operation):	Forecast	Achieved	

INITIAL CRITICALITY			
INITIAL ELECTRICITY	N/A	_____	_____
COMMERCIAL OPERATION		_____	_____

Refueling Information
Fort Calhoun - Unit No. 1

Report for the month ending March 1990

1. Scheduled date for next refueling shutdown. February 17, 1990
2. Scheduled date for restart following refueling. May 13, 1990
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?
 - 1) Decrease the total unrodded integrated radial peaking factor from 1.80 to 1.70.
 - 2) Decrease the total unrodded planar radial peaking factor from 1.80 to 1.75.
 - 3) Decrease the PLHGR from 15.22 to 14.4 Kw/ft
 - 4) Change the TM/LP P_{var} equation to reflect a change in the gamma term from -11350 to -11240.
 - 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. N/A
 - c. If no such review has taken place, when is it scheduled? N/A
4. Scheduled date(s) for submitting proposed licensing action and support information. Submitted Jan. 1990
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. None
6. The number of fuel assemblies:

a) in the core	<u>133</u>	assemblies
b) in the spent fuel pool	<u>437</u>	"
c) spent fuel pool storage capacity	<u>729</u>	"
d) planned spent fuel pool storage capacity	May be increased via fuel pin consolidation or dry cask storage	
7. The projected date of the last refueling that can be discharged to the spent fuel pool assuming the present licensed capacity. 1994*

*Capability of full core offload of 133 assemblies lost.

Prepared by *K. J. Hill* Date April 11, 1990

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-285

UNIT NAME Fort Calhoun Station

DATE April 12, 1990

COMPLETED BY D. L. Stice

TELEPHONE (402) 636-2474

REPORT MONTH March, 1990

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
90-01	900217	S	1029.4	C	1	N/A	XX	XXXXXX	On February 17, 1990, the 12th Fort Calhoun Station Refueling Outage commenced.

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
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 1 - Same Source

OMAHA PUBLIC POWER DISTRICT
Fort Calhoun Station Unit No. 1

March 1990
Monthly Operating Report

I. OPERATIONS SUMMARY

Fort Calhoun Station continued with its twelfth refueling and maintenance outage. The 60 PSIG Integrated Leak Rate Test (ILRT) was performed on the containment structure. Individual penetration leak rate testing continued through the month. Outages for trip testing and preventive maintenance were conducted on 161 KV offsite power, two inhouse 4160 V buses, and several 480 V buses. The #1 battery discharge test was performed. The annual inspection on emergency diesel generator DG-1 is complete and it was returned to service.

The reactor vessel head was removed and the core was off-loaded to the spent fuel pool where all Control Element Assembly (CEA) and neutron source moves were completed. A temporary spent fuel pool cooling system was placed in service using chiller units to cool the pool. With this temporary cooling system in service, the component cooling water system and the raw water system were removed from service and drained for extensive maintenance.

Inspection of the reactor coolant pump bearings found that all four upper guide bearings showed signs of excessive wear. The bearings are being rebuilt. Eddy current testing (ECT) of the U-tubes in both steam generators indicated no tubes required plugging. ECT was also conducted on "A" condenser and several feedwater heaters. Motor operated valve apparatus testing (MOVAT), modifications and preventive maintenance are being performed on motor operated valves. The primary water storage tank interior was stripped, repaired and repainted. The turbine generator high pressure rotor first stage blading has been replaced. The retaining rings were replaced on the main generator.

The following modifications are in progress: DC sequencer replacement; control room indication for diesel generator malfunction; redundant power supply for raw water/component cooling interface valves; increasing the minimum flow of the main feedwater pumps; replacement of area radiation monitors; redistribution of loads on the DC buses and inverters; replacement of valves; control room heating, ventilation, and air conditioning (HVAC) habitability upgrade; containment equipment hatch crane; addition of third auxiliary feedwater pump; RCS narrow range mid-loop level indication; steam generator blowdown tank erosion; feedwater regulating valve supports; alternate shutdown panel wiring for fire protection; and, limiter valve operator thermal overload bypasses.

The following modifications for control room design review for human factors are in progress: reactor regulating system/steam dump and bypass alarms; meter scales; replacement of oddly shaped switch handles; labeling, demarcation, and meter banding.

A review of all large bore Critical Quality Element piping for Thermal Anchor Motion, nozzle loadings, Seismic Anchor Motion, and zero period

acceleration consideration determined the following:

Steam generator blowdown piping near the steam generator had an overstress condition due to thermal growth of the generators. Nineteen supports on the main steam piping in containment and safety injection piping were found to exceed their design capacity. Modifications have been initiated to resolve the concerns. The supports on the main steam piping outside containment cannot adequately withstand seismic and safety relief valve discharge loadings. Failure of these supports would create overstress conditions in the main steam piping.

An analysis showed that in the event of a main steam line break inside containment, the pressure in the auxiliary feedwater piping between each pair of the containment penetration isolation valves would exceed design limits. The resolution of this issue is to be addressed in upcoming Licensee Event Report 90-009.

The following NRC inspections took place in March:

- IR 90-04 System Entry and Retest (SERT) Team Inspection
- IR 90-07 Technical Specifications Calibration Verification
- IR 90-09 ILRT Program
- IR 90-10 Compliance with Appendix J Local Leak Rate Test (LLRT) Requirements
- IR 90-13 Residents' Monthly Inspection (to continue through April 14, 1990)
- IR 90-14 Procedures Program
- IR 90-15 Inservice Inspection (ISI) and Snubber Program
- IR 90-16 Welding Program
- IR 90-17 Maintenance & Test Equipment (M & TE) Program
- IR 90-18 Outage Occupational Exposure
- IR 90-19 Environmental Monitoring Program
- IR 90-21 Special - Loss of Off-site Power
- IR 90-22 ISI Data and Records

The following LERs were submitted:

		Date Submitted
89-024R1	Containment Spray Piping Outside Design Basis	March 30, 1990
90-002	Inadvertent Actuation of Containment Isolation Actuation Signal	March 28, 1990
90-003	Auxiliary Feedwater Piping Outside Design Basis Due to Design Deficiency	March 19, 1990
90-004	Main Steam Safety Valves Outside Setpoint Acceptance Criteria	March 19, 1990
90-005	Spent Fuel Pool Ventilation System Outside Design Basis	March 26, 1990
90-006	Loss of Off-Site Power and Diesel Generator Actuation	March 28, 1990
90-007	Main Feedwater Piping outside Design Basis	March 30, 1990

Monthly Operations Report
Page Three

A. SAFETY VALVES OR PORV CHALLENGES OR FAILURES WHICH OCCURRED

No PORV or Safety Valve challenges or failures occurred.

B. RESULTS OF LEAK RATE TESTS

During the month of March 1990, the reactor was in refueling shutdown condition (Mode 5). Therefore, the Reactor Coolant System Leak Rate test, ST-RLT-3 F.1, was not performed.

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

<u>Amendment No.</u>	<u>Description</u>
No. 125	This amendment changes the Fort Calhoun Technical Specification to place operability and surveillance requirements on the alternate shutdown panels AI-179, 185, and 212. These Technical Specification changes include additions to section 2.15, Limiting Conditions for Operation (LCO) and Table 3-3a.

II. MAINTENANCE (Significant Safety Related)

See Section I Operations Summary

Gary R. Peterson
Manager-Fort Calhoun Station