OMAHA PUBLIC POWER DISTRICT
Fort Calhoun Station Unit No. 1

JULY 1992
Monthly Operating Report

OPERATIONS SUMMARY

Fort Calhoun Station (FCS) operated at full power July [illegible]. On July 2, a design deficiency was discovered whereby a single failure in the wiring of Heater Drain Pumps FW-5A/B/C could cause a fire in both switchgear rooms. Consequently, a one-hour notification to the NRC was made pursuant to 10 CFR 50.72. As a compensatory measure, hourly fire watches were instituted. To correct the potential problem, cable upgrade and rerouting is currently in progress.

At 2335 hours on July 3, during replacement of a degraded circuit board in non-safety related inverter No. 2, power was momentarily lost to the instrument bus that supplies power to the turbine electrohydraulic control system, causing the turbine control valves to close. The closing of the turbine control valves and resulting large mismatch between reactor power and steam demand caused a sharp increase in reactor coolant system (RCS) temperature and pressure. At 2336 hours the reactor protection system automatically tripped the reactor due to high pressurizer pressure. Subsequently, primary safety valve RC-142 failed open resulting in high pressure in the pressurizer quench tank. The pressure in the quench tank caused the tank's rupture disk to burst at 2355 hours, effecting the loss of approximately 21,500 gallons of RCS water to the containment building sump.

An ALERT emergency classification was declared at 2352 hours on July 3, due to a challenge to a fission product barrier (i.e., the RCS pressure boundary). The plant was stabilized and a controlled, expeditious cooldown/depressurization was commenced to bring the plant to cold shutdown. At 0[illegible]0 hours on July 4, the event was downgraded to a Notification of Unusual Event (NOUE). The plant was placed on shutdown cooling at 1312 hours, and at 1825 hours FCS entered cold shutdown. At 1840 hours the plant downgraded from the NOUE, terminating the emergency. Additional details of this event are contained in Licensee Event Report No. 92-023.

OPPD's recovery organization developed a 28 point Recovery/Restart Action Plan. Both primary safety valves were removed on July 7 and sent to an offsite laboratory for testing and recalibration. The root cause of the RC-142 safety valve failure was determined to be an adjusting bolt nut which apparently vibrated loose during the original pressure transient and allowed the adjusting bolt to back out. The change of position of the adjusting bolt effectively lowered the safety valve setpoint to approximately 1923 psia, causing the valve to lift during the normal post-trip repressurization.

Both non-safety related inverters were modified to provide an additional testing power source to the non-safety buses, so that the inverters can be removed from service and tested prior to their return to service. Another modification added a turbine trip whenever turbine control valve No. 1 closes, with provisions to bypass this trip under low power conditions. Also, positive mechanical locking devices for both pressurizer safety valve adjusting bolts were added.

OPERATIONS SUMMARY (continued)

Actions specified in the Recovery/Restart Action Plan were completed and reactor criticality was achieved at 2052 hours on July 22. The generator was synchronized to the grid at 0610 hours on July 23. Power was maintained at 30% for chemistry holds until July 25, when power was raised to 90%. Currently, FCS is operating at 100% power.

The following NRC Inspections took place during July:

IER No:	Title
92-14	Monthly Resident Inspection
92-18	Augmented Inspection Team
92-16	Generic Letter 88-17

The following LERs were submitted during July:

LER	Description
92-20	Failure to Obtain Grab Sample During Radiation Monitor Inoperability
92-21	Failure to Initiate a Fire Watch for an Inoperable Fire Door

A. SAFETY VALVES OR PORV CHALLENGES OR FAILURES WHICH OCCURRED

FCS experienced the previously described event which challenged both PORVs and one safety valve (RC-142). Both PORVs opened as designed on the loss of load event and both reseated properly. One of the two primary safety valves (RC-142) opened during the initial pressure transient of the event, closed, then reopened but did not properly reseat.

Surveillance testing was performed before plant restart to demonstrate operability.

B. RESULTS OF LEAK RATE TESTS

Due to the July 3 reactor trip, only 13 RCS leak rate tests were performed in July. Reactor power and Xenon concentration were changing during many of the leak rate tests, reducing the accuracy of the results. During the periods of relative stability, the total RCS leak rate was about 0.12 gpm, composed of approximately 0.03 to 0.06 gpm of "Known" leakage to the reactor coolant drain tank with the balance being "Unknown" leakage.

Early in July, higher than normal leak rates were attributed to leakage from the "B" charging pump discharge valve (CH-192) packing. Therefore, CH-192 was closed and RCS leakage returned to normal until the July 3 plant trip. After power and Xenon levels stabilized following startup on July 29, the RCS leak rate tests again yielded normal values.

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

Amendment No.	Description
None	

D. SIGNIFICANT SAFETY RELATED MAINTENANCE FOR THE MONTH OF JULY

The following transmitters were inspected for water/foreign material intrusion:

A/DPT-114X (RCS Loop 1B Cold Leg Differential Pressure)

C/DPT-114X (RCS Loop 1 Hot Leg and 1A Cold Leg Differential Pressure Transmitter)

B/PT-120 (Pressurizer RC-4 Pressure Transmitter)

C/PT-120 (Pressurizer RC-4 Pressure Transmitter)

The following transmitters were inspected and calibrations performed:

PT-105 (Wide Range Pressure Transmitter for RC-4)

C/DPT-114W (RCS Loop 1 Hot Leg and 1B Cold Leg Differential Pressure)

A/LT-911 (Steam Generator RC-2A Wide Range Level Transmitter)

A/PT-913 (Steam Generator RC-2A Wide Range Pressure Transmitter)

LT-2904X (Safety Injection Tank SI-6A Wide Range Level Transmitter)

FT-328 (Low Pressure Safety Injection to RC Loop 1B Flow Transmitter)

FC-313 (High Pressure Safety Injection to RC Loop 1B Flow Transmitter)

LT-599 & LT-600 (Containment Sump Level Transmitters)

PT-130 & PT-131 (Pressurizer Quench Tank RC-5 Narrow Range Pressure Transmitters)

LT-132 (Pressurizer Quench Tank Level Transmitter)

PT-3194 (Reactor Coolant Pump RC-3B Gasket Housing Pressurizer Transmitter)

D. SIGNIFICANT SAFETY RELATED MAINTENANCE FOR THE MONTH OF JULY (continued)

Inspected and tested HCV-150 and HCV-151 (Power Operated Relief Isolation Valves for Pressurizer RC-4).

Removed, refurbished and reinstalled RC-141 and RC-142 (Pressurizer RC-4 Code Safety Valves).

Repaired leaking flange gasket on RC-142.

Replaced rupture disc on RC-5 (Pressurizer Quench Tank).

Inspected limit switches and meggered the motor of HCV-314 (HPSI to RC Loop 1A Isolation Valve) and HCV-327 (LPSI to RC-Loop 1B Isolation Valve).

Performed setpoint and leakage test of AC-336 (Charging Pump CH-1A Oil Cooler Component Cooling Water Inlet Relief Valve) and AC-337 (Charging Pump CH-1B Oil Cooler Component Cooling Water Inlet Relief Valve).

Removed HCV-2881B (Component Cooling Heat Exchanger AC-1B Raw Water Outlet Valve) and installed an approved replacement.

Obtained resistance readings for each control element drive mechanism clutch coil.

Replaced 52/STA and 52/HH switches for 4160V cubicle/breaker 1A4-9 (Feeder for Transformer T1B-4B).

Replaced 94-4/1045 relay (Time Delay Relay in Auto Start Circuit for YCV-1045A and Steam Feed for Auxiliary Feed Water Pump FW-10).

Replaced packing cooling water pump and motor for CH-1B (Charging Pump 1B).

Replaced FW-1444 (Auxiliary Feed Water to RC-2B Header Relief Valve).

OPERATING DATA REPORT

DOCKET NO.	50-285
UNIT	FORT CALHOUN STATION
DATE	AUGUST 06 1992
COMPLETED BY	M. L. EDWARDS
TELEPHONE	(402) 636-2451

OPERATING STATUS

1. Unit Name: FORT CALHOUN STATION
2. Reporting Period: JULY 1992

NOTES

3. Licensed Thermal Power (MWt): 1500
4. Nameplate Rating (Gross MWe): 502
5. Design Elec. Rating (Net MWe): 478
6. Max. Dep. Capacity (Gross MWe): 502
7. Max. Dep. Capacity (Net MWe): 478

8. If changes occur in Capacity Ratings (3 through 7) since last report, give reasons:
N/A

9. Power Level to which restricted, if any (Net MWe): N/A

10. Reasons for restrictions, if any:
N/A

	THIS MONTH	YR-TO-DATE	CUMULATIVE
11. Hours in Reporting Period..........	744.0	5111.0	165241.0
12. Number of Hours Reactor was Critical	290.7	2458.9	127277.6
13. Reactor Reserve Shutdown Hours......	.0	.0	1309.5
14. Hours Generator On-line............	281.4	2386.3	125763.4
15. Unit Reserve Shutdown Hours.........	.0	.0	.0
16. Gross Thermal Energy Generated (MWH)	334189.7	3084361.6	164708087.3
17. Gross Elec. Energy Generated (MWH)..	108894.0	1027983.0	54204109.2
18. Net Elec. Energy Generated (MWH)....	102680.1	974437.0	51708188.4
19. Unit Service Factor.................	37.8	46.7	76.1
20. Unit Availability Factor............	37.8	46.7	76.1
21. Unit Capacity Factor (using MDC Net)	28.9	39.9	68.1
22. Unit Capacity Factor (using DER Net)	28.9	39.9	66.2
23. Unit Forced Outage Rate............	62.2	17.7	4.2

24. Shutdowns scheduled over next 6 months (type, date, and duration of each):
NONE

25. If shut down at end of report period, estimated date of startup: ________

26. Units in test status (prior to comm. oper.):

		Forcast	Achieved
INITIAL CRITICALITY			
INITIAL ELECTRICITY	N/A		
COMMERCIAL OPERATION			

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO.	50-285
UNIT	FORT CALHOUN STATION
DATE	AUGUST 06 1992
COMPLETED BY	M. L. EDWARDS
TELEPHONE	(402) 636-2451

MONTH JULY 1992

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	476	17	0
2	477	18	0
3	469	19	0
4	0	20	0
5	0	21	0
6	0	22	0
7	0	23	48
8	0	24	91
9	0	25	138
10	0	26	357
11	0	27	428
12	0	28	427
13	0	29	430
14	0	30	466
15	0	31	470
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.

Refueling Information
Fort Calhoun - Unit No. 1

Report for the month ending July 1992

1. Scheduled dated for next refueling shutdown.		September 1993
2. Scheduled date for restart following refueling.		November 1993
3. Will refueling or resumption of operations thereafter require a technical specification change or other license amendment?		Yes
a. If answer is yes, what, in general, will these be?		Incorporate specific requirements resulting from reload safety analysis.
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.		June 1993
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.		New fuel supplier New LOCA analysis
6. The number of fuel assemblies:	a) in the core	133 Assemblies
	b) in the spent fuel pool	529 Assemblies
	c) spent fuel pool storage capacity	729 Assemblies
	d) planned spent fuel pool storage capacity	Planned to be increased with higher density spent fuel racks.
7. The projected date of the last refueling that can be discharged to the spent fuel pool assuming the present licensed capacity.		1995*

* Capability of full core offload of 133 assemblies lost. Reracking to be performed between the 1993 and 1995 Refueling Outages.

Prepared by K. Holthaus Date 8-6-92

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-285
UNIT NAME Fort Calhoun St.
DATE August 6, 1992
COMPLETED BY M. L. Edwards
TELEPHONE (402) 636-2451

REPORT MONTH July 1992

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
92-05	920703	F	462.6	B	3	92-023	ED	GENERA	On July 3, 1992, at 2336, while the plant was operating at 100% power, the Reactor Protection System automatically tripped the reactor due to high pressurizer pressure. The event was initiated as a result of maintenance on a non-safety related inverter. During replacement of a degraded circuit board, power was momentarily lost to the instrument bus that supplies power to the Turbine Electrohydraulic Control System, resulting in closure of the turbine control valves. A subsequent failure of a pressurizer code safety valve (RC-142) resulted in high pressure in the pressurizer quench tank that blew the tank's rupture disk and resulted in the loss of approximately 21,500 gallons of contaminated water to the containment building sump. The consequences of the event are bounded by the Fort Calhoun Station Updated Safety Analysis Report. The root cause of the momentary loss of power to the instrument bus was determined to be the inability to isolate and test the non-safety related inverters after maintenance without potentially losing power to the respective 120V AC instrument buses. The root cause of the malfunction of RC-142 was determined to be the adjusting bolt nut that loosened and allowed the set pressure adjusting bolt to back out. Corrective actions included: a modification to enhance the ability to test the non-safety related inverters off-line, the addition of a positive mechanical locking device for the pressurizer safety valve adjusting bolts and completion of a comprehensive Recovery/Restart Action Plan.

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 1 - Same Source

(9/77)