AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-285

UNIT Fort Calhoun #1

DATE June 7, 1979

COMPLETED BY B. J. Hickle

TELEPHONE 402-536-4413

MONTH May, 1979

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	453.0	17	452.0
2	452.5	18	450.8
3	453.3	19	451.8
4	453.7	20	451.5
5	454.0	21	451.2
6	453.4	22	449.6
7	451.4	23	450.2
8	452.3	24	451.2
9	451.7	25	451.4
10	452.8	26	450.6
11	453.4	27	450.6
12	452.8	28	449.9
13	452.6	29	449.9
14	452.9	30	448.9
15	453.3	31	449.9
16	452.8		

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.

2285 028

(9/77)

7906120336

OPERATING DATA REPORT

DOCKET NO. 50-285
DATE June 7, 1979
COMPLETED BY B. J. Hickle
TELEPHONE 402-536-4413

OPERATING STATUS

1. Unit Name: Fort Calhoun Station Unit No. 1
2. Reporting Period: May, 1979
3. Licensed Thermal Power (MWt): 1420
4. Nameplate Rating (Gross MWe): 502
5. Design Electrical Rating (Net MWe): 457
6. Maximum Dependable Capacity (Gross MWe): 481
7. Maximum Dependable Capacity (Net MWe): 457

Notes

8. If Changes Occur in Capacity Ratings (Items Number 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	3,623.0	49,800.0
12. Number Of Hours Reactor Was Critical	744.0	3,614.3	39,579.3
13. Reactor Reserve Shutdown Hours	0.0	0.0	1,136.0
14. Hours Generator On-Line	744.0	3,602.3	38,668.4
15. Unit Reserve Shutdown Hours	0.0	0.0	0.0
16. Gross Thermal Energy Generated (MWH)	1,045,546.3	4,956,223.8	46,604,647.1
17. Gross Electrical Energy Generated (MWH)	352,903.8	1,677,029.9	15,481,291.6
18. Net Electrical Energy Generated (MWH)	336,110.9	1,595,783.2	14,617,188.9
19. Unit Service Factor	100.0	99.4	77.6
20. Unit Availability Factor	100.0	99.4	77.6
21. Unit Capacity Factor (Using MDC Net)	98.9	96.4	64.8
22. Unit Capacity Factor (Using DER Net)	98.9	96.4	64.2
23. Unit Forced Outage Rate	0.0	0.6	4.6

24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each):
    N/A
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		

2285 029

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-285
UNIT NAME Fort Calhoun #1
DATE June 7, 1979
COMPLETED BY B. J. Hickle
TELEPHONE 402-536-4413

REPORT MONTH May, 1979

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
	NONE								

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)

2285 030

Refueling Information
Fort Calhoun - Unit No. 1

Report for the month ending May 31, 1979.

1. Scheduled date for next refueling shutdown. — January 1, 1980

2. Scheduled date for restart following refueling. — March 1, 1980

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?

   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. — Stretch Power Application
   .Site Related Information, July, 1979
   .Non-Core Related Information October, 1979
   .Core Related Analysis and Tech. Spec. Changes, November, 1979

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 — 157 "
   c) spent fuel pool storage capacity — 483 "
   d) planned spent fuel pool storage capacity — 483 "

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 Gazer   Date June 7, 1979

2285 031

OMAHA PUBLIC POWER DISTRICT
Fort Calhoun Station Unit No. 1

May 1979
Monthly Operations Report

I. OPERATIONS SUMMARY

Fort Calhoun Station Unit No. 1 operated at essentially a nominal 100% power during the month of May.

Simulator training at the Combustion Engineering Simulator for four hot license candidates and senior and operator licensed personnel continued through the month.

Normal operations and surveillance tests were completed during the month of May.

A. PERFORMANCE CHARACTERISTICS

LER Number	Deficiency
79-010	During the performance of surveillance test ST-SI/CS-1, containment spray pump SI-3B failed to start by control switch. The remaining two containment spray pumps were operable throughout this event. The unit was operating at approximately 100% steady state power.
79-011	During normal power operation it was noted that Channel A DC Sequencer S1-1 had de-energized Matrix lights for engineered safeguards loads supplied from 480 volt bus 1B3A. The AC Sequencer for Channel A and both AC and DC Sequencers for Channel B were operable throughout this event. The safeguards loads from the other plant 480 volt buses tied to Sequencer S1-1 were also considered operable.
79-012	During the performance of surveillance test ST-RPS-11 F.1 the matrix test failed for the BC matrix. This test is required by Technical Specification Table 3.1 Item 11. The failure resulted in only a slight degradation as 5 of the 6 logic units were working properly when the fault resulted in a loss of one of four trip relays in one of the six assemblies.
79-014	As directed by IE Bulletin 79-01, ASCO solenoid valves were found to lack sufficient environmental qualification data. These solenoid valves function to provide or vent air to containment safety related valve operations. The plant operators have been instructed to fail instrument air to containment during post-LOCA conditions which potentially cause solenoid failure. Failure of instrument air will ensure that these safety related valves are maintained in their safety position.

A. PERFORMANCE CHARACTERISTICS (Continued)

LER Number	Deficiency
79-015	On May 11, 1979, the District was informed by its A/E (Gibbs & Hill) that seismic support appeared to be inadequate for the component cooling water piping to containment cooling and filtering unit VA-3A. At 1630 on May 15, 1979, the preliminary indication was confirmed. Also at this time, containment cooling and filtering unit VA-3A was immediately declared inoperable in accordance with Technical Specification 2.4.

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 results in Operations Incidents and are not reported elsewhere in the report:

Operations Incident		Deficiency
OI-797	ST-ISI-CVCS-3	CH-4B discharge pressure high. Retested within specification.

Surveillance Test No.	Description
ST-ISI-CVCS-3 (F.1) Chemical and Volume Control System Pump Test	Boric Acid Pump CH-4B discharge pressure was 100 psig which is in excess of 1.03 x P_r. Pump to be repaired.

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

Procedure	Description
SP-CHEM-4-1	Spiked Radiological Samples for laboratory analysis comparison/completed per procedure remarks noted in comment section.

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

Procedure	Description
SP-RPS-5	Excore Detector Symmetric offset recalibration/ completed per procedure - no significant abnormal results.
DCR 74A-67	New Fuel Rack Modification (Boral Sheet)/ completed as designed.
DCR 79-35/ DCO 79-17	Security Building Diesel Generator Room lighting.
DCR 78-22	Turbine Building Jib Crane - Installed as designed.
DCR 77-45	Upgrading Security System - Completed as designed.

E. RESULTS OF LEAK RATE TESTS

None

F. CHANGES IN PLANT OPERATING STAFF

None

G. TRAINING

Training consisted of Simulator Training for the operators at Combustion Engineering.

Plant training included monitor team training, driver training, valve and pump packing, and procedure reviews.

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

None

Approved by *W.C. Andrews*
Manager-Fort Calhoun Station

2285 034

II. MAINTENANCE (Significant Safety Related)

M. O. #	Date	Description	Corrective Action
400	4-9-79	AC Matrix failure of Mercury Wetted Relay.	Cycled relay.
539	4-24-79	DC Sequender for Bus 1B3A Relay Failure.	Replaced relay.
593	4-30-79	Turbine Building to Auxiliary Building fire door closer isn't working.	Readjusted door.
380	4-4-79	DG-2 Recirculating pump has faulty coupling.	Replaced
568	4-30-79	Door 1007-16 will not close properly.	Adjusted
162	5-4-79	Emergency Diesel #2 radiator core leaks.	Repaired core and replaced gasket.
635	5-5-79	Emergency Diesel #2 safety valve on #2 air compressor.	Repair and completed.
645	5-7-79	Fire Protection - open floor penetration in Room 81.	Installed threaded pipe caps as needed.
637	5-8-79	Emergency Diesel #2 - Relief valve on #1 air compressor leaking through.	Replace relief valve.
632	5-4-79	Emergency Diesel #2 - crack in radiator water divider.	Repair and welded crack.
46	5-10-79	Heater Drain Pump (FW-5C) - install throttle bushing bleed off line to ensure proper lubrication.	Completed
733	5-17-79	AC-pipe supports - install U bolts per procedure.	Completed
723	5-16-79	A-046 Load New Fuel for shipment.	Completed
654	5-8-79	RPS Matrix relay BC-4 failure.	Removed and cleaned contacts.
723	5-16-79	A046 New Fuel Bundle load for shipment.	Completed

2285 035

II. MAINTENANCE (Significant Safety Related)

M. O. #	Date	Description	Corrective Action
733	5-17-79	Replace U-bolts on CCW piping to containment air coolers that were left off during construction.	U-bolts installed per detailed work instructions.
604	5-1-79	Retest of CH-4B.	Retested per ST-ISI-CVCS-3 (F.1). Discharge press. = 98 psig, within specification.
777	5-25-79	Seismic Support FWS-83	Install new U-bolts.

2285 036

NUCLEAR ENERGY LIABILITY INSURANCE

MUTUAL ATOMIC ENERGY LIABILITY UNDERWRITERS

1. Amendment of Advance Premium Endorsement

2. Standard Premium and Reserve Premium Endorsement

3. Additional Premium Due

1. Advance Premium

It is agreed that the Amended Advance Premium due the companies for the calendar year 1979 is $96,870.95.

2. Standard Premium and Reserve Premium

Subject to the provisions of the Industry Credit Rating Plan, it is agreed that the Standard Premium and Reserve Premium for the calendar year designated above are:

Standard Premium $96,870.95.

Reserve Premium $72,946.92.

3. Additional Premium $4,726.93.

Effective Date of this endorsement January 1, 1979

To form a part of Policy No. MF-57

Issued to Virginia Electric and Power Company

Date of Issue May 10, 1979

For the Subscribing Companies

MUTUAL ATOMIC ENERGY LIABILITY UNDERWRITERS

By ______________________

~~Endorsement No.~~ 42 42

Countersigned by ______________________
Authorized Representative

...CERTIF... COPY OF ... 42 ...CE PART OF NUCLEAR ENERGY POLICY ...ACILITY FORM) NO. MF-57 ...NO ...SURANCE IS AFFORDED UNDER THIS ...UE COPY.

Theodore Geras

THEODORE GERAS, SECRETA...
MUTUAL ATOMIC ENERGY LIABILITY UNDERWR...

2285 037

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J

TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

716 Edney Build'ng

June 6, 1979

Mr. James P. O'Reilly, Director
U.S. Nuclear Regulatory Commission
Office of Inspection and Enf   ent
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Dear Mr. O'Reilly:

TENNESSEE VALLEY AUTHORITY - BROWNS FERRY NUCLEAR PLANT UNIT 2 - DOCKET NO. 50-260 - FACILITY OPERATING LICENSE DPR-52 - REPORTABLE OCCURRENCE REPORT BFRO-50-260/7911

The enclosed report provides details concerning the first rod, 26-27, in group 3 which was inadvertently withdrawn three notches while pulling control rods to achieve initial criticality for cycle 3. This report is submitted in accordance with Browns Ferry unit 2 Technical Specification 6.7.2.a.4.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

H. S. Fox
Director of Power Production

Enclosure (3)
cc (Enclosure):
Director (3)
Office of Management Information and Program Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Director (40)
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

R. F. Sullivan, NRC Inspector, Browns Ferry

2285 039

POOR ORIGINAL

7906120329

NRC FORM 366
(7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | | (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

01 | LICENSEE CODE: A L B R F 2 (2) | LICENSE NUMBER: 0 0 - 0 0 0 0 0 - 0 0 (3) | LICENSE TYPE: 4 1 1 1 1 (4) | CAT: (5)

CON'T

01 | REPORT SOURCE: L (6) | DOCKET NUMBER: 0 5 0 0 0 2 6 0 (7) | EVENT DATE: 0 5 2 6 7 9 (8) | REPORT DATE: 0 6 0 6 7 9 (9)

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

02 While pulling control rods to achieve initial criticality for cycle 3, the first
03 rod, 26-27, in group 3 was inadvertently withdrawn three notches. The rod moved from
04 position 02 to position 08 rather than from position 02 to 04 as prescribed in the
05 rod move sheets. The reactor scrammed on hi-hi IRM flux. Based on evaluation of
06 process neutron monitoring instrumentation, no safety limit was exceeded but is
07 reported in accordance with T. S. 6.[illegible].a([illegible]) No hazard existed to the health and
08 safety of the public. Previous occurrence: 259/7901.

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP SUBCODE	VALVE SUBCODE
09	R C (11)	X (12)	Z (13)	Z Z Z Z Z Z (14)	Z (15)	Z (16)

(17) LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	7 9	—	0 1 1	/	0 [illegible]	T	—	0

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
Z (18)	(19)	C (20)	C (21)	0 0 0 0	Y (23)	N (24)	Z (25)	Z 9 9 9 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

10 The withdrawal of the rod from position 02 to position 08 is unexplained. Functional
11 testing of the rod after the event revealed that the rod would not travel three
12 positions when given a notch withdraw signal. Prior to restart, a review of the
13 event was conducted and no safety limits were exceeded. A subsequent rod pull,
14 using the same sequence, resulted in a period of approximately 43 seconds.

15 | FACILITY STATUS: B (28) | % POWER: 0 0 0 (29) | OTHER STATUS (30): N/A | METHOD OF DISCOVERY: A (31) | DISCOVERY DESCRIPTION (32): Operator Observation

16 | ACTIVITY RELEASED: Z (33) | CONTENT OF RELEASE: Z (34) | AMOUNT OF ACTIVITY (35): N/A | LOCATION OF RELEASE (36): N/A

PERSONNEL EXPOSURES

17 | NUMBER: 0 0 0 (37) | TYPE: Z (38) | DESCRIPTION (39): N/A

PERSONNEL INJURIES

18 | NUMBER: 0 0 0 (40) | DESCRIPTION (41): N/A

LOSS OF OR DAMAGE TO FACILITY (43)

19 | TYPE: Z (42) | DESCRIPTION: N/A

2285 039

PUBLICITY

20 | ISSUED: N (44) | DESCRIPTION (45): N/A

7906120330

NRC USE ONLY

NAME OF PREPARER ______ PHONE ______

POOR ORIGINAL

Tennessee Valley Authority
Browns Ferry Nuclear Plant

Form BF-17
BF 15.2
1/10/79

LER SUPPLEMENTAL INFORMATION

BFRO-50- 260 / 7911 Technical Specification Involved 6.7.2.a.4

Reported Under Technical Specification 6.7.2.a.4

Date of Occurrence 5/26/79 Time of Occurrence 1928 Unit 2

Identification and Description of Occurrence:

BFNP unit 2 scrammed at 1928 on May 26, 1979, while pulling control rod 26-27, the first rod in RWM group 3, during an approach to critical. The cause of the scram was high neutron flux on IRM's C, D, and F.

The reactor period was calculated to be 5.5 seconds.

Conditions Prior to Occurrence:

Normal startup procedure following the spring 1979 refueling outage. This was the first approach to critical.

Action specified in the Technical Specification Surveillance Requirements met due to inoperable equipment. Describe.

N/A

Apparent Cause of Occurrence:

Movement of control rod 26-27 three positions.

Analysis of Occurrence:

A reactivity insertion of approximately 0.287 percent delta K/K resulted when control rod 26-27 was withdrawn from position 02 to position 08. The resulting reactor period by the SRM recorder was about one second. It was calculated to be 5.5 seconds.

Corrective Action: Prior to initial criticality following a refueling outage, any control rod that has a single-notch worth which is capable of producing a period of equal to or less than 60 seconds, or any control rod that has a double-notch worth capable of producing a period of equal to or less than 30 seconds, will be withdrawn on a notch basis. The nuclear engineer will be present in the control room during the period of time from initial rod pull until criticality is achieved. During this period of time, he will confirm acceptable core behavior.

Failure Data:

N/A

*Retention: Period - Lifetime; Responsibility - Administrative Supervisor

*Revision:

2285 040