UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report Nos.: 50-348/88-20 and 50-364/88-20

Licensee: Alabama Power Company
600 North 18th Street
Birmingham, AL 36291

Docket Nos.: 50-348 and 50-364 License Nos.: NPF-2 and NPF-8

Facility name: Farley 1 and 2

Inspection Conducted: May 16 - 20 and June 6 - 10, 1988

Inspector: Scott E. Sparks for P. T. Burnett 6-23-88
Date Signed

Approved by: Frank Jape 6/23/88
F. Jape, Section Chief
Test Programs Section
Engineering Branch
Division of Reactor Safety
Date Signed

SUMMARY

Scope: This routine unannounced inspection addressed the areas of witnessing post-refueling startup tests, review of completed core surveillance procedures, independent measurements of reactor thermal power and reactor coolant system leakage, and review of the licensee's related procedures.

Results: One violation was identified. The procedure used to calculate reactor coolant system inventory was inadequate in that the constant used to make corrections for changes in pressurizer level was neither correct nor conservative (Violation 348,364/88-20-03) - Paragraph 5.

Management made a commitment to evaluate the feasibility of moving the source range detectors to a region of lower flux so that criticality would occur below P-6 (Inspector Followup Item 348,364/88-20-01) - Paragraph 2.

The licensee made a commitment to upgrade the U1118 plant computer calculation of thermal power (Inspector Followup Item 348,364/88-20-02) - Paragraph 4.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

#R. G. Berryhill, Systems Performance and Planning Manager
J. A. Collier, Junior Engineer
#R. D. Hill, Assistant General Plant Manager - Plant Services
W. S. MacDonald, Reactor Engineer
#*R. H. Marlow, Technical Supervisor
#*C. D. Nesbitt, Technical Manager
#J. K. Osterholtz, Operations Manager
#*W. D. Shipman, Assistant General Plant Manager - Operations
#*J. J. Thomas, Maintenance Manager
*J. D. Woodard, Vice President

Other licensee employees contacted included, operations personnel, security force members, and office personnel.

Other Organization

L. Grobmeyer, Westinghouse

NRC Resident Inspectors

#*W. H. Bradford, Senior Resident Inspector
W. H. Miller, Resident Inspector

*Attended exit interview on May 20, 1988.
#Attended exit interview on June 6, 1988.

Acronyms and initialisms used throughout this report are listed in the final paragraph.

2. Post-Refueling Startup Tests (72700, 61708, 61710)

a. Test Witnessing - Unit 1

The inspector witnessed the initial criticality of Unit 1 to start Cycle 9.

Prior to pulling the shutdown banks, a statistical reliability check (Chi-squared test) of the two SRNIs, N31 and N32, was performed successfully using procedure FNP-0-ETP-3635 (Revision 1), Reliability Check of Source Range Instrumentation. The initial countrates were in excess of 500 cps.

After the two shutdown rod banks were withdrawn, countrate data were obtained for reference countrates for ICRR calculations. These calculations were performed and the results plotted every 50 steps as the control banks were withdrawn in overlap to a final configuration of D bank at 190 steps. After the first 50 steps increment, no additional rod withdrawal was performed until the ICRR plot for both SRNIs confirmed that criticality was not likely in the next increment. At the end of rod withdrawal, SRNI countrates were in excess of 3000cps.

While observing these activities, the inspector noted that the two data takers were not well prepared to understand and perform their duties. Both needed training on data collection and use of the data sheets. It was necessary for the test engineer on shift to provide instruction that more properly should have been performed as part of a pretest briefing. Before the next phase of the approach to criticality was performed, the data takers were briefed on the test and their responsibilities.

A dilution rate of 20gpm was begun using the alternate dilute mode, and with it a new set of ICRR plots was started. Once those plots showed the trend of reactivity increase, the dilution rate was increased successively to 30 and then 45gpm.

Throughout the rod withdrawal phase and in the early stages of dilution, the inspector made independent checks of the reliability of the SRNI by performing Chi-squared tests based upon three to five observations of countrate. All tests were successful until the countrates exceeded 10,000cps, after which excessively large values of Chi squared were calculated, probably the result of high resolving time losses in the system. P6, the source range block permissive, was obtained well in advance of criticality, and the remainder of the approach to criticality was monitored solely by the two IRMs. N36 appeared to be noisy, but responding, while N35 was well-behaved. At one point, prior to P6, instrument technicians took N36 out of service to adjust the compensating voltage in an attempt to reduce its irratic reading. Dilution continued; since no mode change was imminent. Criticality was announced after about a four-fold increase in flux above P6.

It appears the SRNIs are too sensitive, or located in a region of too high flux, to satisfactorily monitor the entire approach to criticality. Thus, as in this instance, the final stage was monitored by the IRMs, which had yet to demonstrate acceptable performance. An initial countrate of 0.5cps has been found acceptable (Regulatory Guide 1.68), although a rate one or two orders of magnitude higher does facilitate data collection. If the Unit 1 source range detectors could be moved to a location where the flux is a factor of

four to ten lower, then both the initial and at-critical countrate requirements could be satisfied. The current core is designed for low leakage; so it does not seem likely that future cores will provide a significantly reduced source.

At the exit interview, management made a commitment to evaluate the feasibility of moving the source range detectors to a region of lower flux so that criticality would occur below P-6 (Inspector Followup Item 348,364/88-20-01).

b. Review of Completed Procedures - Unit 1

The following completed startup test procedures were reviewed:

(1) FNP-1-ETP-3601 (Revision 5), Zero Power Physics Test Procedure, was used to perform boron endpoint measurements at ARO and reference bank-in configurations, determine the ITC and MTC, and measure control rod reactivity worth. For the ITC measurement, both the procedure and the vendor recommend a temperature change of at least 4°F. In practice, the heatup was 2.8°F and the cooldown was 3.3°F. For this core the MTC is far from the limiting value, and the potentially reduced precision of the measurement is not of concern. However, the procedure guidance should be followed more closely in the future to assure the MTC is better resolved when it approaches the limiting value. The reactivity computer trace from the reference bank worth measurement was spot checked. In all cases, the inspector's determination of the reactivity increment was slightly greater than that reported by the licensee. Thus it appears the licensee's reported control rod worths are conservatively low, but in satisfactory agreement with design values.

(2) FNP-1-STP-3605 (Revision 6), Startup and Power Ascension Procedures.

(3) FNP-1-STP-29.3 (Revision 0), Shutdown Margin Verification with Control Rods at the Rod Insertion Limit.

(4) FNP-1-STP-114 (Revision 7), Determination of Moderator Temperature Coefficient at All Rods Out Zero Power and at 70% Power, was performed on May 19, 1988. Using the ITC measured in the zero power tests, the procedure confirmed by appropriate adjustment of design parameters that the MTC limits of TS 3.1.1.3 were satisfied at ARO and at 70% RTP. This revision of the procedure reflects a significant improvement over the last time this area was inspected.

(5) FNP-1-STP-111 (Revision 6), Overall Reactivity Balance.

No violations or deviations were identified.

3. Core Power Distribution Monitoring (61702)

Although the procedures are essentially the same, those for Unit 2 were chosen for review; since that unit is currently in its fifth month of the current cycle, and review of data for an ongoing cycle was of more interest than for a completed cycle. The documents reviewed were:

a. FNP-2-STP-110 (Revision 12), Determination of Limiting Hot Channel Factors, FQ(Z) and FdHN;

b. FNP-2-STP-121 (Revision 16), Power Range Axial Offset Calibration, and;

c. Unit 2 Flux Map Log, which contains copies of the data sheets generated in performing the above surveillances during the current cycle.

The logged flux maps, numbers 152 to 158, were all full core maps, and with the exception of the first, which was at 32% RTP, were obtained at full power. All maps were obtained by traversing 49 to 50 instrument thimbles; the minimum allowable, TS 3.3.3.2, is 38. The maximum exposure between maps was 30.8 EFPD, and the typical span was about 22 EFPD. TS 4.2.2.2.d(1)(b) and 4.2.3.1.b require these surveillances on 31 EFPD intervals.

The results of the surveillances were:

Maximum FQ(Z) = 1.79 (TS 3.2.2 limit = 2.32)
Maximum FdHN = 1.48 (TS 3.2.3 limit = 1.55)
Incore QPTRs were all less than 1%.

STP-110 does not require that flux maps be obtained during power escalation if the measured Fxy does not exceed the limiting Fxy of the first map taken, No. 152 at 32%. However, not taking intermediate power maps before reaching 100% RTP is contrary to practice at other similar facilities. Prudence would argue for at least one intermediate map at a power level between 50 and 90% RTP.

At the start of the cycle, an incore-excore nuclear instrument correlation was performed at 32% RTP in accordance with section 10 of STP-121. The correlation was based upon seven flux maps (146 to 152), of which only the first and last were full core maps and the remainder were quarter core maps. Incore measured axial offsets ranged from +27.8% to -27.1%. Individual chamber currents ranged from 44 to 60 microamperes.

STP-121 permits additional incore-excore correlation data to be taken during power escalation to enhance the fit, but no additional maps were made.

The inspector independently analyzed the incore-excore correlation data using a least squares spreadsheet with the microcomputer program SUPERCALC3 (Release 2.1). For the eight neutron chambers involved, the correlation coefficients ranged from 0.973 to 0.994. The inspector is accustomed to finding all coefficients in excess of 0.99 when the data for the correlation were obtained in the range of 50 to 75% RTP. Attachment 1 is a typical plot of correlation data and results for the pair of chambers comprising Unit 2 PRNI N42.

In the second week of the inspection, incore-excore correlation data were available from the recent startup of Unit 1 (Procedure FNP-1-STP-121). A similar analysis of those data yielded correlation coefficients in the range 0.985 to 0.999. Attachment 2 is a typical plot of correlation data and results for the pair of chambers comprising Unit 1 PRNI N44. As power was escalated on Unit 1, additional flux maps were obtained at 50 and 80% RTP. Those results were used to adjust rather than augment the earlier correlations. The adjustments affected only the zero offset currents and not the slopes of the offset versus current relationships. The methodology used and the bases for those adjustments will be reviewed in later inspection.

The currents produced by the chambers are only 40 to 50% of those observed for similar instruments at other facilities. Hence at the relatively low powers, about 30% RTP, that the licensee uses for the incore-excore correlation, chamber currents are only 40 to 60 microamperes. Thus the observed currents are resolved to only two significant figures instead of the three available at other facilities. The low currents appear to have a direct and negative impact on the quality of the correlations. The inspector referred the licensee to another facility that has been successful in increasing the currents produced by the PRNIs.

Unit 1 limiting hot channel factors were satisfactory at 32 and 48% RTP as analyzed using FNP-1-STP-110 (Revision 17) during startup operations.

No violations or deviations were identified.

4. Thermal Power Determination (61706)

The NRC independent measurement program for determination of reactor thermal power is described in NUREG1167, TPDWR2: Thermal Power Determination for Westinghouse Reactors, Version 2. To customize the program for use at Farley 1 and 2, the necessary system parameters were obtained by review of the FSAR and vendor documents. The parameters for insulation losses were adjusted to approximate the licensee's measured losses on Unit 1. To obtain Unit 2 data for use with the microcomputer program TPDWR2, the operators adjusted one of the standard plant computer edits to output the required process data every 15 minutes. Simultaneously, the inspector collected feedwater flow data manually from the Barton

differential pressure meters. (The Bartons are the source of feedwater flow measurements for the licensee's calorimetric procedure). The data obtained, although sufficient for use in TPDWR2, were not in the order or, in all cases, in the units required for input to that program. A SUPERCALC3 spreadsheet was created to facilitate ordering and conversion of the data for input to TPDWR2. The customized plant parameters for Unit 2 (Unit 1's are identical) and a typical set of input data are given in Attachment 3.

TPDWR2 was first run using feedwater flow as measured through the plant computer and then using flow calculated using the Barton dP cells. In general, the steam generator power calculations were lower using the Barton flows than with the installed flowmeters, but only the differences for Steam Generator B appeared to be significant, about 1.4% greater averaged over four sets of calculations. Typical results for Unit 2, corresponding to the input data in Attachment 3, are given in Attachment 4. This calculation used the same feedwater flow data as used in the licensee's routine surveillance (FNP-2-STP-109), which gave a result of 2654.1 megawatts thermal. The difference from the TPDWR2 result of 2655.3 is not considered significant. The licensed power level is 2652 Mwth. The indicated 0.1% overpower is a typical and tolerable variation from the setpoint

Reading the Barton dP meters installed for Unit 2 is not as straightforward operation as it should be. The installed meters are scaled for 0 to 450 inches of water column, and that two-inch-increment scale is backed by a mirror so needle position may be read without parallax. On Steam Generator A, the scale has been extended to 475 inches of water, and the calibrated 475-inch point is scribed on the solid face of the meter. The actual operating point of the meter is about 460 inches, but that point must be determined by the reader without the aid of scribed subdivisions between 450 and 475 inches and without the aid of the mirror to avoid parallax in the reading. For Steam Generator C, two insulated pipes cover the face of the meter with less than two feet of clearance. Reading the meter accurately is difficult whether the viewer places his head between the pipes and the meter or attempts to read the meter through the space between the pipes.

At the exit interview, the licensee stated that three new Barton meters with 500 inch scales were on order and that it was their intent to install them where needed as soon as they arrive. Changing meters does not require and outage. Management also took notice of the comments on the difficulty in reading the meters.

The installed flowmeters also provide input to the plant computer calculation of thermal power at computer point U1118. That point does not provide a reliable estimate of power because no attempt has been made to keep the installed flowmeters calibrated to the precision required for measurement. (The licensee contends the flow meter calibration is

sufficient to support the safety function and level control). Management agreed that having an unreliable indicator on or available for display was poor practice. At the exit interview, the licensee made a commitment to upgrade the U1118 calculation (Inspector Followup Item 348,364/88-20-02).

No violations or deviations were identified.

5. Reactor Coolant System Leakage Measurements (61728)

The microcomputer program RCSLK9, which was developed by the NRC Independent Measurements Program, is described in NUREG-1107, RCSLK9: Reactor Coolant System Leakage Determination for PWRs. Farley uses FNP-1/2-STP-9.0 for surveillance of RCS leakage. Data obtained by the licensee for recent surveillances on both units were analyzed using RCSLK9. The comparisons revealed that one constant used by the licensee to correct for changes in pressurizer level was in error and was non-conservtive. As a result, volumetric changes in pressurizer inventory were equated with those in the VCT without correcting for the near factor of two difference in density. A one-percent increase in pressurizer level would lead to underestimating RCS losses by about 25 standard gallons. For a surveillance period of 30 to 60 minutes the leak rate would be underestimated by 1 to 0.5gpm. The maximum allowed unidentified leak rate is 1gpm. Although no violation of the leakage limit was identified, the inability of the procedure to perform the required surveillance with acceptable precision under permitted and anticipated conditions has been identified as a violation (Violation 348,364/88-20-03). The licensee initiated corrective action to revise the procedure as soon as the problem was identified, and on June 14, 1988, the licensee reported the corrected procedures would be implemented the following day.

6. Acronyms and Initialism Used in This Report

AFD - Axial flux difference
AO - Axial offset
ARO - all rods out
cps - counts per second
dP - differential pressure
ETP - Engineering technical procedure
EFPD - Effective full power day
FdHN - Nuclear enthalpy rise hot channel factor
FQ(Z)- Heat flux hot channel factor
Fxy - Radial (planar) peaking factor
gpm - gallon per minute
ICRR - inverse count rate ratio
IRM - intermediate range monitor
ITC - Isothermal temperature coefficient
MTC - Moderator temperature coefficient
pcm - Percent millirho (unit of reactivity)

ppmB - Parts per million boron
QPTR - Quadrant power tilt ratio
RCS - Reactor coolant system
RTP - Rated thermal power
SRNI - source range nuclear instrument
STP - Surveillance test procedure
TS - Technical Specification
VCT - volume control tank

7. Exit Interview

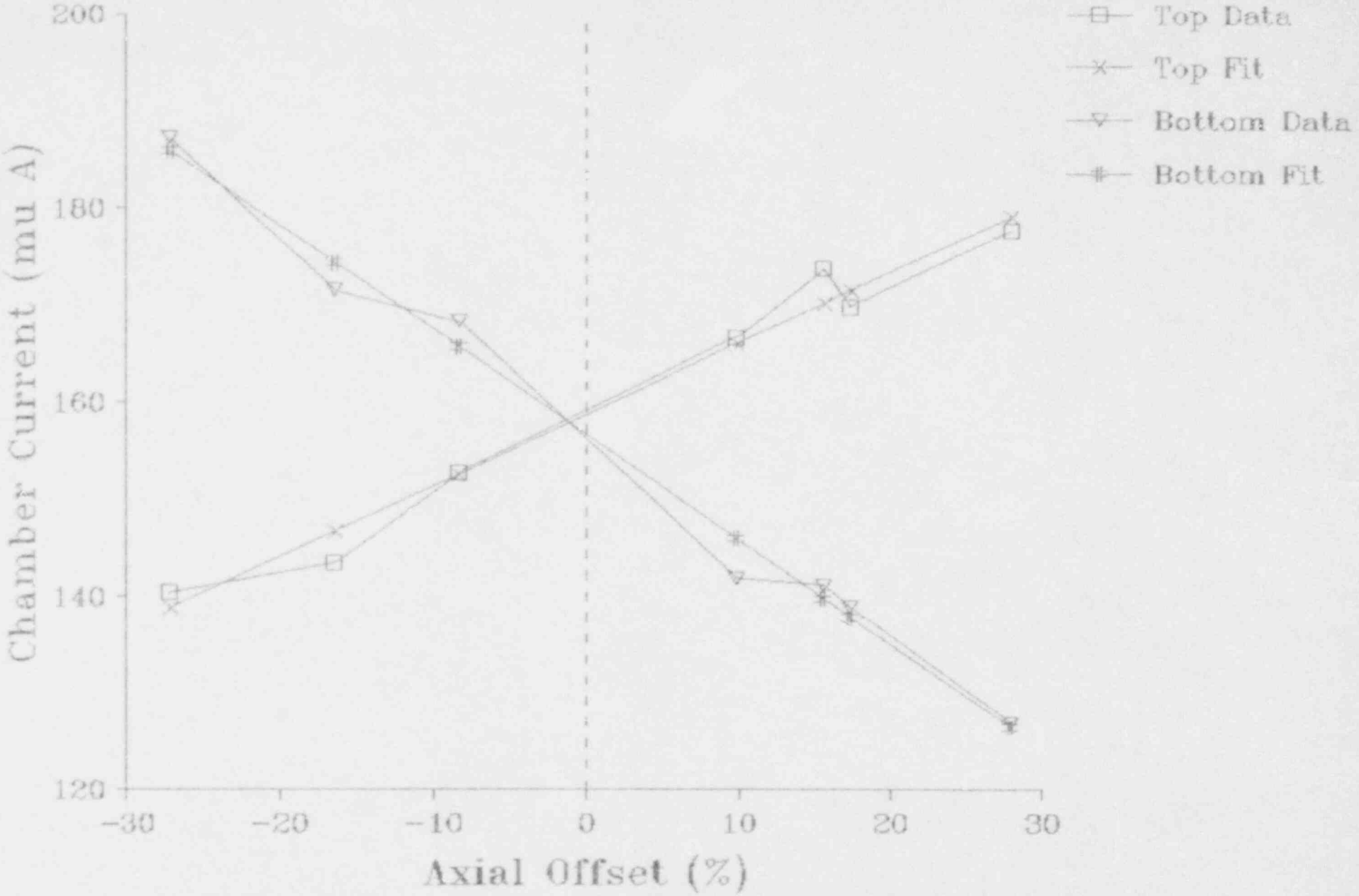
The inspection scope and findings were summarized on May 20, and June 10, 1988 with those persons indicated in paragraph 1 above. The inspector described the areas inspected and discussed in detail the inspection findings. Proprietary material was reviewed by the inspector during this inspection, but is not incorporated in this report. Dissenting comments were not received from the licensee.

Attachments

a. Unit 2, N42, Incore-Excore Correlation
b. Unit 1, N44, Incore-Excore Correlation
c. Heat Balance Parameters and Data
d. Heat Balance Results

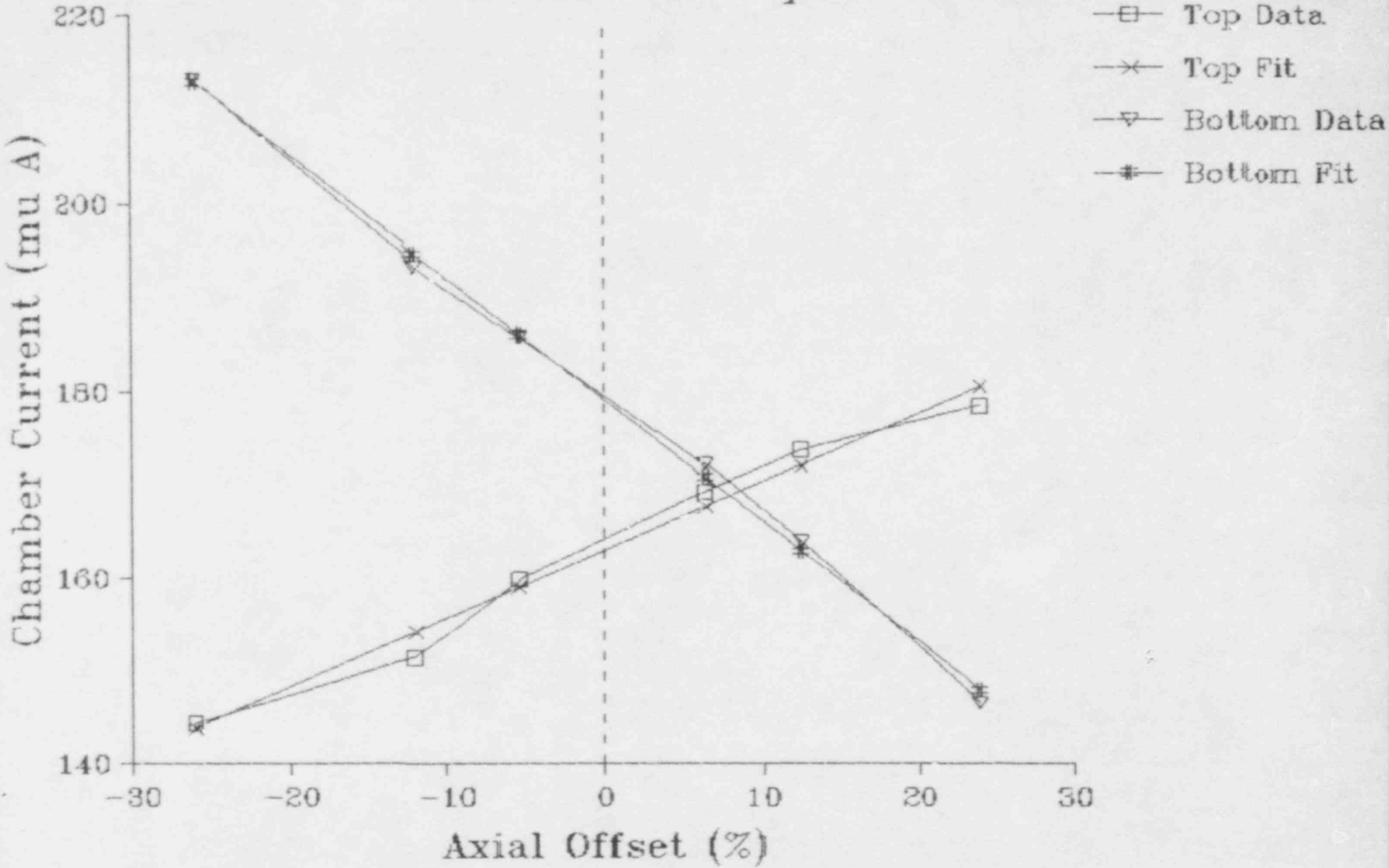
Data and Least Squares Fit

FNP 2, N42



FNP 1, N44

Data and Least Squares Fit



ATTACHMENT 3 Rpt. 348, 364/88-20

HEAT BALANCE DATA
FARLEY 2
6/7/88

PLANT PARAMETERS:

REACTOR COOLANT SYSTEM	
Pump Power (MW each)	4.5
Pump Efficiency (%)	93.0
Pressurizer Inside Diameter (inches)	84.0
STEAM GENERATORS	
Dome Inside Diameter (inches)	168.50
Riser Outside Diameter (inches)	56.75
Number of Risers	3
Moisture Carry-over (%) in A	0.040
Moisture Carry-over (%) in B	0.040
Moisture Carry-over (%) in C	0.040

REFLECTIVE INSULATION	
Inside Surface Area (sq ft)	34,000
Heat Loss Coefficient (BTUs/hr sq ft)	220.00
NONREFLECTIVE INSULATION	
Inside Surface Area (sq ft)	24,000
Thickness (inches)	4.0
Thermal Conductivity (BTUs/hr ft F)	0.140
LICENSED THERMAL POWER (MWt)	2652

DATA:

TIME	1700

STEAM GENERATOR A		STEAM GENERATOR B	
Steam Pressure (psia)	803.5	Steam Pressure (psia)	805.9
Feedwater Flow (E6 lb/hr)	3.972	Feedwater Flow (E6 lb/hr)	3.869
Feedwater Temperature (F)	438.9	Feedwater Temperature (F)	438.3
Surface Blowdown (gpm)	0.0	Surface Blowdown (gpm)	0.0
Bottom Blowdown (gpm)	15.2	Bottom Blowdown (gpm)	16.0
Water Level (inches)	48.5	Water Level (inches)	47.8

STEAM GENERATOR C	
Steam Pressure (psia)	806.8
Feedwater Flow (E6 lb/hr)	3.812
Feedwater Temperature (F)	439.2
Surface Blowdown (gpm)	0.0
Bottom Blowdown (gpm)	12.8
Water Level (inches)	48.2

LETDOWN LINE		CHARGING LINE	
Flow (gpm)	130.1	Flow (gpm)	94.4
Temperature (F)	539.9	Temperature (F)	453.9

PRESSURIZER		REACTOR	
Pressure (psia)	2242.5	T ave (F)	574.5
Water Level (inches)	98.9	T cold (F)	542.3

ATTACHMENT 4 Rpt. 348, 364/88-20

HEAT BALANCE
FARLEY 2
6/7/88

DATA SET 1 OF 1 1700 hours	ENTHALPY (BTUs/lb)	FLOW (E6 lb/hr)	POWER (E9 BTUs/hr)	POWER (MWt)
STEAM GENERATOR A				
Steam	1199.0	3.966	4.756	
Feedwater	418.1	-3.972	-1.661	
Surface Blowdown	510.4	0.00000	0.00000	
Bottom Blowdown	463.1	0.00612	0.00283	

Power Dissipated			3.0976	907.2
STEAM GENERATOR B				
Steam	1198.9	3.862	4.630	
Feedwater	417.4	-3.869	-1.615	
Surface Blowdown	510.8	0.00000	0.00000	
Bottom Blowdown	463.0	0.00644	0.00298	

Power Dissipated			3.0186	884.1
STEAM GENERATOR C				
Steam	1198.9	3.807	4.564	
Feedwater	418.4	-3.812	-1.595	
Surface Blowdown	511.0	0.00000	0.00000	
Bottom Blowdown	463.5	0.00515	0.00239	

Power Dissipated			2.9714	870.2
OTHER COMPONENTS				
Letdown Line	534.7	0.04956	0.02650	
Charging Line	435.7	-0.03939	-0.01716	
Pumps			-0.04287	
Insulation Losses			0.01226	

Power Dissipated			-0.02126	-6.2

REACTOR POWER				2655.3