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

Report Nos.: 50-338/88-10 and 50-339/88-10

Licensee: Virginia Electric and Power Company
Richmond, VA 23261

Docket Nos.: 50-338 and 50-339 License Nos.: NPF-4 and NPF-7

Facility Name: North Anna 1 and 2

Inspection Conducted: April [illegible], 1988

Inspectors: P. T. Burnett 5-2-88
P. T. Burnett Date Signed

P. T. Burnett for 5-2-88
S. E. Sparks Date Signed

Approved by: Frank Jape 5/2/88
F. Jape, Section Chief Date Signed
Engineering Branch
Division of Reactor Safety

SUMMARY

Scope: This routine unannounced inspection addressed the areas of post refueling startup tests, shutdown margin surveillance, reactor coolant system leakage surveillance, and thermal power monitoring.

Results: No violations or deviations were identified.

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   *P. B. Boulden, Plant Engineering
   *R. F. Driscoll, Manager, Quality Assurance
   *L. L. Edwards, Superintendent, Nuclear Training
   *R. C. Enfinger, Assistant Station Manager, Operations and Maintenance
   *R. Garver, Reactor Engineer
   *S. Hamill, Supervisor, In-Service Inspection Engineering
   *E. Hendrixson, Acting Supervisor, S & T Engineering
   *M. R. Kansler, Superintendent, Maintenance
   *J. H. Lebesstein, Engineer
   *D. B. Roth, Nuclear Specialist

   Other licensee employees contacted included engineers, technicians, operators, security force members, and office personnel.

   NRC Resident Inspectors

   *J. L. Caldwell, Senior Resident Inspector
   L. P. King, Resident Inspector

   *Attended exit interview

2. Exit Interview

   The inspection scope and findings were summarized on April 8, 1988, with those persons indicated in paragraph 1 above. The inspectors described the areas inspected and discussed in detail the inspection findings. No dissenting comments were received from the licensee. Proprietary material was reviewed in the course of the inspection, but is not included in this report. One commitment was received from the licensee:

   Inspector Followup Item 338/339/88-10-01: Institute, by October 31, 1988, a program to perform Chi-Squared Tests on the source range nuclear instruments to ensure their proper functioning prior to and during fuel loading and initial criticality following refueling - Paragraph 5.

3. Licensee Action on Previous Enforcement Matters

   This subject was not addressed in the inspection.

4. Unresolved Items

   No unresolved items were identified.

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

The records of the most recent series of post-refueling startup tests were reviewed for each unit. Essentially, identical procedures are used. Initial criticality was attained following procedure 1(2)-OP-1.5 as appropriate. Subsequent testing and power escalation was performed in accordance with 1(2)-PT-94.0, Refueling Nuclear Design Check Tests.

From review of these tests and discussions with licensee personnel, it was determined that no tests are performed on the SRNIs to assure they are responding primarily and proportionally to neutrons before being used to monitor fuel loading or the succeeding startup. This is not accomplished by the surveillances required by the Technical Specifications since those tests exclude the neutron detectors and amount to no more than setting bistables in response to clean test signals. The licensee agreed that a more certain method of determining SRNI reliability was desirable and committed to establish a program of Chi-Squared tests to that end. This commitment was confirmed at the exit interview with a completion date of October 31, 1988, which is prior to the next scheduled refueling outage. Appropriate times for performing the Chi-Squared tests would be after loading the source-bearing assemblies during fuel loading, prior to pulling shutdown banks or renormalizing the inverse count rate ratio during the first startup on a new core, and any time those activities are interrupted for eight hours or more.

a. Unit 1 Test Results

The test to determine the critical boron concentration for the all rods out condition (1-PT-94.0, Sequence Step No. 4) was performed 6-29-87. The measured value of 1969 ppm agreed well when compared with the design value of 1995 ± 50 ppm.

The checkout of the reactivity computer satisfied the acceptance criterion that the reactivity derived directly from period measurement and use of the inhour equation agree within four percent of the reactivity computer solution for both positive and negative periods.

The test to determine the isothermal temperature coefficient for the all rods out condition (1-PT-94.0, Sequence Step No. 5) was performed 6-29-87. The measured value of -0.45 pcm/°F at a boron concentration of 1965 ppm agreed well with the design value of -0.67 ± 3.0 pcm/°F. In addition, the measured value was within the Technical Specification 3.1.1.4 value of less than or equal to 6.0 pcm/°F.

Control bank and shutdown bank worth measurements were performed 6-29-87. The measured and design value worths were as follows:

Control Bank	Measured Worth	Design Value Worth
A	343 pcm	321 ± 100 pcm
B	1323 pcm	1338 ± 134 pcm
C	766 pcm	780 ± 117 pcm
D	766 pcm	807 ± 121 pcm

Shutdown Bank	Measured Worth	Design Value Worth
A	1054 pcm	1056 ± 158 pcm
B	902 pcm	930 ± 140 pcm

The measured value for the total rod worths of 5154 pcm compares favorably with the design value of 5232 ± 523 pcm. All measured control bank and shutdown bank worths were within the design value worths.

The test to determine the hot zero power, boron worth coefficient was performed 6-29-87. The measured value of -7.27 pcm/ppm compares well with the design value of -7.25 ± 0.73pcm/ppm. That was determined during the reactivity worth measurement of control bank B during continuous boron dilution. (All other rod bank reactivity worths were determined by rod swap with control bank B, the reference bank).

In addition, the inspectors performed an independent review of control bank B worth based on a detailed review of test data from the reactivity computer strip chart records. Attachment 2 provides a graphical comparison of the licensee's differential rod worth data to inspector generated data and shows excellent agreement. However, the inspectors noted that the licensee's published results as contained in the Cycle 7 Startup report for the differential worth of control bank B do not coincide with the original results generated during the test. There is an apparent smoothing out of the integral worth profile at approximately rod position step 150, with no accompanying explanation as to why this smoothing out was performed. Although this has an no effect on the integral worth of control bank B or the other bank worths determined by comparison with it, the inspectors expressed concern about the publishing of test results which do not coincide with actual test data without explanation of the adjustments made. This concern was discussed with the licensee, as an example of poor practice, at the exit interview. By changing one rod insertion by two steps from the annotated value on the strip chart, the inspectors were able to obtain the same smoothing of their data.

b. Unit 2 Test Results

Unit 2 startup testing was performed in the period October 19, to December 3, 1987.

Following criticality with D bank partially inserted, the checkout of the reactivity computer satisfied the acceptance criterion that the reactivity derived directly from period measurement and use of the inhour equation agree within four percent of the reactivity computer solution for both positive and negative periods.

The ARO critical boron concentration was 1982 ppmB, which was in good agreement with the design value for the actual conditions of 1994 ± 50 ppmB.

The isothermal and moderator temperature coefficients for ARO were -0.6 pcm/°F and 1.13 pcm/°F respectively. The predicted ITC for the actual core conditions was -0.94±3.0 pcm/°F. The maximum MTC allowed by Technical Specification 3.1.1.4 is +6.0pcm/°F. Hence, all acceptance criteria were satisfied.

The reference bank, control bank B, measured reactivity worth of 1282 pcm was less than the predicted value of 1367 ± 137 pcm, but satisfied the design tolerance of ± 10%. The sum of all rod worths, the remainder were measured by rod swap, was less than 6% below the predicted sum, and hence, satisfied the design tolerance.

The measured and predicted boron worth coefficients were -6.82 pcm/ppm and -7.27 pcm/ppm respectively, and the design tolerance was satisfied.

During power escalation, flux maps were taken using the moveable detector system at 28, 47, and 100% RTP. In all cases, F_Q satisfied Technical Specification 3.2.2, F_{dH} satisfied Technical Specification 3.2.3, and QPTR satisfied Technical Specification 3.2.4.

No violations or deviations were identified.

6. Determination of Reactor Shutdown Margin - Units 1 and 2 (61707).

The inspector reviewed the Unit 1 performance test procedure, 1-PT-10, Determination of Shutdown Margin, completed 11-23-87. A recently completed procedure for Unit 2 shutdown margin was not available, however, determination of reactor shutdown margin for both units is essentially the same. The review consisted of verification of technical adequacy, compliance with procedural requirements, and compliance with station Technical Specifications. The inspector verified that xenon worth curves, rod worths, boron worths, and temperature defect values were properly transcribed from the correct version of the Station Curve Book. The calculated shutdown margin of -3207 pcm satisfied the Technical Specification shutdown margin of equal to or more negative than -1770 pcm.

In addition, the inspector reviewed operating procedure 2-OP-1C, Estimated Critical Position, for the initial startup of Unit 2 Cycle 6, completed 11-3-87. The procedure allows for the estimated critical condition to be calculated either manually or by a computer program. The estimated critical condition for this startup was determined manually. For a boron concentration of 1928 ppm, the predicted control rod positions, which include an administrative span of ± 400 pcm were as follows:

Bank C	Bank D	Worth
197 steps	69 steps	-876 pcm
228 steps	194 steps	-76 pcm

Actual critical conditions of 1922 ppmB, Bank C at 228 steps, and Bank D at 98 steps, agreed well with the predicted rod positions.

7. 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. To customize the program for use at North Anna, plant-specific parameters for each unit were obtained from review of the following documents: Updated FSAR, vendor manuals for steam generators and pressurizers, station curve books and internal memoranda. The parameter list for Unit 2 (Unit 1 is identical) is given in Attachment 3.

To obtain data for use with RCSLK9 and TPDWR2, which is discussed below, the licensee established Group Review 11 on the plant computer to monitor and print out the required data at fifteen minute intervals. Because of makeup to the VCT, the longest span of time for the calculation was 1.25 hours. Over that period, the results from RCSLK9 were acceptable. The licensee's calculational program, which was performed in parallel with the inspection activities, uses ten-minute-averaged data for the beginning and ending points. Using the averaged data provided by the licensee, the results from RCSLK9 were in good agreement with the licensee value of 0.78 and 0.26 gpm identified and unidentified leakage respectively. The output from RCSLK9 is given in Attachment 4.

No violations or deviations were identified.

8. Thermal Power Determination (61706)

The NRC independent measurement program for determination of reactor thermal power is described in NUREG-1167, TPDWR2: Thermal Power Determination for Westinghouse Reactors, Version 2. To customize the program for use at North Anna 1 and 2 the necessary system parameters were obtained by review of the documents listed in paragraph 7 above. The parameters for insulation losses were adjusted to duplicate the licensee's measured losses on Unit 1. To obtain data for use with the microcomputer program TPDWR2, the inspectors again made use of the Group Review 11 output. 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 5.

Both units have both feedwater and steam flow venturis, and the latter are used in the licensee's calculation of thermal power. TPDWR2 was first run in its designed mode of using feedwater flow, and the agreement in its result was within 0.1% of licensee values on average. The input data were then adjusted to simulate the steam flow venturi data, and the results were in even better agreement with licensee values.

Typical results for Unit 2, corresponding to the input data in Attachment 5, are given in Attachment 6.

No violations or deviations were identified.

Attachments:
1. List of Acronyms and Initialisms
2. Unit 1, Bank B, Differential Reactivity Worth
3. RCSLK9 Parameters for Unit 2
4. RCSLK9 Results for Unit 2
5. TPDWR2 Parameters and Data for Unit 2
6. TPDWR2 Results for Unit 2

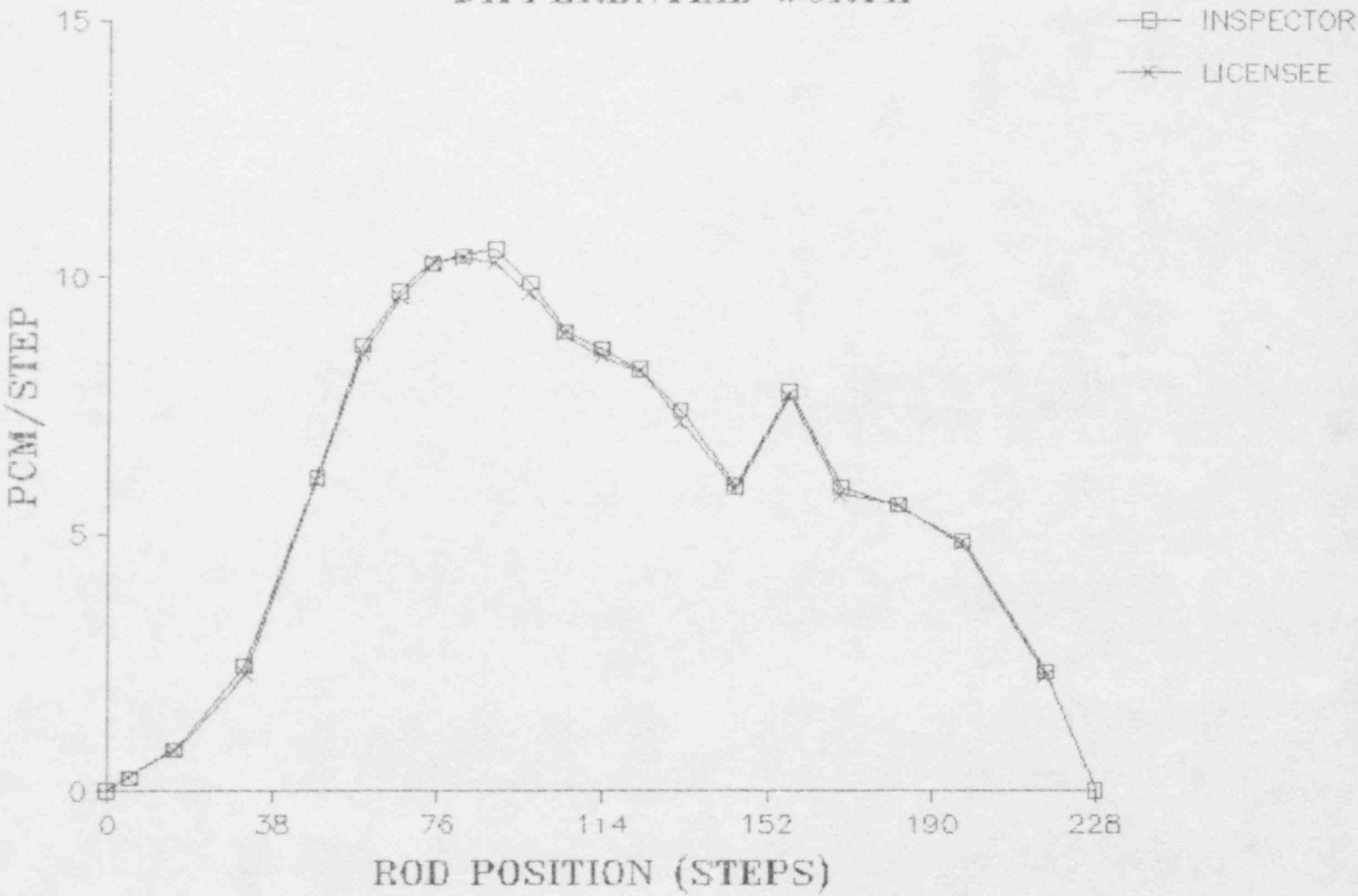
Attachment 1

Acronyms and Initialisms

ARO - All Rods Out
F_{dH} - Nuclear Enthalpy Rise Hot Channel Factor
F_Q - Total Heat Flux Hot Channel Factor
FSAR - Final Safety Analysis Report
gpm - gallon per minute
ITC - Isothermal Temperature Coefficient
MTC - Moderator Temperature Coefficient
OP - Operating Procedure
pcm - Percent Millirho (unit of reactivity)
ppm - Parts Per Million
ppmB - Parts per Million Boron
PT - Periodic Test
QPTR - Quadrant Power Tilt Ratio
RTP - Rated Thermal Power
SRNI - Source Range Nuclear Instrument
VCT - Volume Control Tank

NORTH ANNA 1, CYCLE 7 BANK B

DIFFERENTIAL WORTH



PARAMETER LIST

Unit Identification:	
Plant Name	NORT" ANNA
Unit Number	2
Docket Number	50-339
Nuclear Steam System Supplier	Westinghouse
Vessel and Piping:	
Volume	8557.2 cubic feet
Pressurizer:	
Level Units	%
Temperature Compensated	No
Calibration Curve	
Slope	354 pounds per %
Upper Level Limit	100 %
Lower level Limit	0 %
Relief	Relief Tank
Volume Control Tank:	
Level Units	%
Calibration Curve	
Slope	116.7 pounds per %
Upper Level Limit	100 %
Lower level limit	0 %
Geometric Method Available	No
Drain Tank:	
Level Units	%
Calibration Curve	
Slope	70 pounds per %
Upper Level Limit	76.5 %
Lower level limit	32.5 %
Geometric Method Available	No
Relief Tank:	
Level Units	%
Calibration Curve	
Slope	921 pounds per %
Upper Level Limit	70 %
Lower level limit	30 %
Geometric Method Available	No

NRC

INDEPENDENT MEASUREMENTS PROGRAM

REACTOR COOLING SYSTEM LEAK RATES

STATION: NORTH ANNA
UNIT : 2
DOCKET : 50-339

TEST DATE : APRIL 6, 1988
START TIME:
DURATION 1.333 hours

TEST DATA

	Initial	Final
System Parameters		
Pressure, psia	2257.32	2257.32
T Ave, degrees F	586.75	586.66
Water Levels		
Pressurizer, %	62.54	62.85
Relief Tank, %	50	50
Volume Control Tank, %	47.77	40.34
Drain Tank, %	28.98	36.67

Water Charged = 0 gal   Water Drained = 0 gal

TEST RESULTS

Change in Water Inventory in pounds:

Vessel & Piping	61	Relief Tank (1)	0
Pressurizer	110	Drain Tank (1)	538
Volume Control Tank (1)	-867		------
Less: Water Charged	0	Collected Leakage	538
Plus: Water Drained	0		

Cooling System	-697		

Leak Rates in gpm (3):

Gross	1.05
Identified	0.81
Unidentified	0.24

(1) Determined from tank calibration curve.
(2) Determined from tank dimensions.
(3) The density used for converting inventory change to leak rate was 62.31 pounds/cubic foot based on standard conditions.

HEAT BALANCE DATA
NORTH ANNA 2
4-6-88

PLANT PARAMETERS:

REACTOR COOLANT SYSTEM

Pump Power (MW each)	5.9
Pump Efficiency (%)	90.0
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	0.100
Moisture Carry-over (%) in B	0.100
Moisture Carry-over (%) in C	0.100

REFLECTIVE INSULATION

Inside Surface Area (sq ft)	26,000
Heat Loss Coefficient (BTUs/hr sq ft)	110.00

NONREFLECTIVE INSULATION

Inside Surface Area (sq ft)	9,423
Thickness (inches)	[illegible].0
Thermal Conductivity (BTUs/hr ft F)	0.100

LICENSED THERMAL POWER (MWt) 2893

DATA:

	SET 1	SET 2
TIME	1059	1108

STEAM GENERATOR A

	SET 1	SET 2
Steam Pressure (psia)	895.2	894.1
Feedwater Flow (E6 lb/hr)	4.286	4.297
Feedwater Temperature (F)	439.7	439.9
Surface Blowdown (gpm)	0.0	0.0
Bottom Blowdown (gpm)	30.0	30.0
Water Level (inches)	63.7	62.7

STEAM GENERATOR B

	SET 1	SET 2
TIME	1059	1108
Steam Pressure (psia)	890.7	890.0
Feedwater Flow (E6 lb/hr)	4.269	4.272
Feedwater Temperature (F)	441.1	441.3
Surface Blowdown (gpm)	0.0	0.0
Bottom Blowdown (gpm)	30.0	30.0
Water Level (inches)	65.1	64.1

STEAM GENERATOR C

	SET 1	SET 2
Steam Pressure (psia)	885.6	885.9
Feedwater Flow (E6 lb/hr)	4.231	4.235
Feedwater Temperature (F)	441.1	441.2
Surface Blowdown (gpm)	0.0	0.0
Bottom Blowdown (gpm)	30.0	30.0
Water Level (inches)	62.4	62.8

LETDOWN LINE

	SET 1	SET 2
Flow (gpm)	85.8	85.6
Temperature (F)	553.8	553.7

CHARGING LINE

	SET 1	SET 2
Flow (gpm)	41.8	41.4
Temperature (F)	483.8	483.8

PRESSURIZER

	SET 1	SET 2
Pressure (psia)	2245.5	2246.1
Water Level (inches)	244.1	245.7

REACTOR

	SET 1	SET 2
T ave (F)	586.7	586.7
T cold (F)	553.5	553.5

HEAT BALANCE
NORTH ANNA 2
4-6-88

DATA SET 1 OF 2 1059 hours	ENTHALPY (BTUs/lb)	FLOW (E6 lb/hr)	POWER (E9 BTUs/hr)	POWER (MWt)
STEAM GENERATOR A				
Steam	1195.9	4.277	5.114	
Feedwater	419.0	-4.286	-1.796	
Surface Blowdown	525.9	0.00000	0.00000	
Bottom Blowdown	470.9	0.01200	0.00565	

Power Dissipated			3.3241	973.5
STEAM GENERATOR B				
Steam	1196.0	4.260	5.095	
Feedwater	420.6	-4.269	-1.796	
Surface Blowdown	525.2	0.00000	0.00000	
Bottom Blowdown	471.3	0.01199	0.00565	

Power Dissipated			3.3049	967.9
STEAM GENERATOR C				
Steam	1196.2	4.218	5.045	
Feedwater	420.6	-4.231	-1.780	
Surface Blowdown	524.3	0.00000	0.00000	
Bottom Blowdown	470.9	0.01200	0.00565	

Power Dissipated			3.2714	958.1
OTHER COMPONENTS				
Letdown Line	551.9	0.03208	0.01770	
Charging Line	469.0	-0.01696	-0.00796	
Pressurizer	643.0	0.00141	0.00091	
Pumps			-0.05458	
Insulation Losses			0.00424	

Power Dissipated			-0.03968	-11.6

REACTOR POWER				2887.9

HEAT BALANCE
NORTH ANNA 2
4-6-88

DATA SET 2 OF 2 1108 hours	ENTHALPY (BTUs/lb)	FLOW (E6 lb/hr)	POWER (E9 BTUs/hr)	POWER (MWt)
STEAM GENERATOR A				
Steam	1195.9	4.288	5.128	
Feedwater	419.3	-4.297	-1.802	
Surface Blowdown	525.7	0.00000	0.00000	
Bottom Blowdown	470.9	0.01200	0.00565	

Power Dissipated			3.3318	975.8
STEAM GENERATOR B				
Steam	1196.0	4.263	5.098	
Feedwater	420.8	-4.272	-1.798	
Surface Blowdown	525.1	0.00000	0.00000	
Bottom Blowdown	471.4	0.01199	0.00565	

Power Dissipated			3.3064	968.4
STEAM GENERATOR C				
Steam	1196.2	4.222	5.050	
Feedwater	420.7	-4.235	-1.782	
Surface Blowdown	524.4	0.00000	0.00000	
Bottom Blowdown	471.0	0.01200	0.00565	

Power Dissipated			3.2740	958.9
OTHER COMPONENTS				
Letdown Line	551.8	0.03201	0.01766	
Charging Line	469.0	-0.01680	-0.00788	
Pressurizer	642.9	0.00141	0.00091	
Pumps			-0.05458	
Insulation Losses			0.00424	

Power Dissipated			-0.03965	-11.6

REACTOR POWER				2891.4