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ARKANSAS POWER & LIGHT COMPANY

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ARKANSAS NUCLEAR ONE

STEAM ELECTRIC STATION

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UNIT ONE

STARTUP REPORT

TO THE

U.S. NUCLEAR REGULATORY COMMISSION

LICENSE NUMBER DPR-51

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FOR THE

PERIOD ENDING 19 DECEMBER 1974

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1.0 INTRODUCTION

Arkansas Nuclear One Unit One Technical Specification Number 6.12.2.1 requires that a Startup Report be prepared detailing the startup and power escalation testing performed on unit systems and components. Unit One FSAR, Section 13, describes the test program and indicates that the initial startup program was designed to meet the requirements of the AEC publication entitled "Guide for the Planning of Initial Startup Programs". The "Guide for the Planning of Initial Startup Programs" states in paragraph A: "This guide . . . provides information on the breadth and depth of initial fuel loading, startup and testing programs and procedures that have been found to be acceptable in the past". Paragraph A further states: "This guide and the interest of the AEC are limited to the safety related aspects of the fuel loading, initial startup and testing program". In keeping with this philosophy, the scope of this Startup Report encompasses tests that fulfill the requirements of the Guide and does not include a description of many other tests performed on systems and components that were of interest to AP&L from an economic or design point of view.

Four changes were made in the initial startup program described in the FSAR and/or the Guide as follows:

1. Hot no flow trip testing of reactor control rod drive mechanisms was not done as prescribed in the FSAR, Section 13, and the Guide. A full battery of tests were accomplished to assure that the control rods would enter the core under all reactor coolant pump and flow combinations. Trips under flow conditions are more conservative than trips under no flow conditions.

2. The Guide specifies that a turbine trip test be performed at 50% FP (Table 13-4 of the FSAR amends the power level to 40% FP). We chose to initiate the turbine trip by a manual reactor trip (which automatically trips the turbine within milliseconds) because it was felt that more useful information could be gained by initiating the trip in this manner. The decision to perform the test in this manner was supported by the fact that Regulatory Guide 1.68 (which has replaced the older Guide) does not require a turbine trip at 50% as the old Guide did. It was determined that the trip test at 100% FP would be initiated by manually tripping the turbine.

3. The FSAR states in paragraphs 4.1.1.2 and 7.2.3.1 that the plant can experience a turbine trip from 100% load without tripping the reactor. This condition was an acceptance criteria for successful completion of the turbine and generator trip tests at 100% FP. This acceptance criteria was deleted because planned and unplanned occurrences at other similar plants had demonstrated that the reactor would trip if turbine and/or generator tripped at 100% FP. The criteria was purely commercial and had nothing to do with safe operation of the plant.

4. We did not perform a generator trip test at 100% power since the turbine trip test at 100% power produced the same reactor transient. A turbine trip test at 100% FP and a generator trip test at 100% FP are required in FSAR and the Guide. (See Section 2.28)

The four changes in the test program listed above were reviewed by the NSSS supplier and the AP&L General Office Safety Review Committee to assure that no unreviewed safety questions resulted from the changes.

On May 21, 1974 the Atomic Energy Commission issued Facility Operating License DPR-51 to Arkansas Power and Light Company for Arkansas Nuclear One-Unit 1. The first fuel assembly was loaded into the core on May 29, 1974, and initial fuel loading was completed on June 5, 1974. ANO Unit 1 successfully achieved initial criticality on August 6, 1974.

Zero Power Physics testing, which commenced on August 6, 1974, was successfully completed on August 8, 1974. This program was conducted at a reactor coolant temperature of 532° F and below the nuclear heat power level to eliminate any temperature feedback effects.

Initial power escalation was begun on August 13, 1974. A testing program was carried out at several points during the power escalation with most of the tests performed at four major power plateaus:

Power Level (%FP)	Date
15	August 16, 1974
40	September 24, 1974
75	October 23, 1974
100	December 8, 1974

Planned power testing was interrupted on five occasions. These outages were the result of (1) a need for miscellaneous Balance of Plant repairs, (2) electrical short on the exciter shaft of main/turbine generator, (3) twice, because of excessive Reactor Coolant System leakage, and (4) due to an electrical short in the controller on one of the Reactor Building main chillers.

The startup and power escalation testing sequence was completed on December 11, 1974.

2.1 REACTOR COOLANT FLOW TEST

2.1.1 PURPOSE

The purposes of this test were:

A. To determine the functional flow rates of the reactor coolant system and the reactor coolant pumps for the following sequence of test conditions, each test being a prerequisite for each subsequent test:

Test 1 CF - Cold Flow Test prior to installation of the core
Test 2 HF - Hot Flow Test prior to installation of the core
Test 3 FC - Single Pump Flow Tests with core installed
Test 4 HFC - Hot Flow Test with core installed

B. To determine the reactor coolant flow for the required pump combinations in each of the four test conditions listed in A.

C. To determine the adequacy of the instrumentation used for the Reactor Coolant Flow Coastdown Test.

D. To demonstrate that the reactor coolant pumps perform consistently for extended periods at normal operating conditions.

2.1.2 TEST METHOD

A. Test 1 - CF - Cold Flow Test prior to installation of the core

Steady state data with all four reactor coolant pumps running was accumulated utilizing the plant computer and the B&W supplied Reactimeter.

B. Test 2 - HF - Hot Flow Test prior to installation of the core

Steady state data with all four reactor coolant pumps running was accumulated utilizing the plant computer and the B&W supplied Reactimeter. Reactor coolant temperatures during the test were approximately 520° F.

C. Test 3 - Single Pump Tests with core installed

Steady state data for single pump operation of each reactor coolant pump to determine individual flows was taken after reaching equilibrium conditions. The same relative reactor coolant system tempe. ature and pressure was used during each test. Data was taken utilizing the plant computer and the B&W supplied Reactimeter.

D. Test 4 - Hot Flow Test with core installed

Seventeen test cases of different pump combinations, trips and coastdowns were run. Test cases eleven (11) through seventeen (17) were run solely as part of this test. Test cases one (1) through ten (10) were run in conjunction with Reactor Coolant Flow Coastdown Test. Data was accumulated utilizing the plant computer and the B&W supplied Reactimeter. Descriptions of the test cases are listed below in the order that they were run:

Sequence Number	Test Case Number	Description
1	17	Four Pump Operation
2	6	One Pump Flow Coastdown From Four Pump Operation (Highest flow pump tripped)
3	7	Three Pump Flow Coastdown From Three Pump Operation (Trip three pumps simultaneously)
4	8	Two Pump Flow Coastdown From Four Pump Operation (Highest flow pump each loop tripped simultaneously)
5	11	Two Pump Operation (Lowest flow pump each loop)
6	1	One Pump Flow Coastdown From Two Pump Operation (Lowest flow pump each loop)
7	12	Two Pump Operation (Highest flow pump in one loop and lowest flow pump in other loop)
8	2	Two Pump Flow Coastdown From Two Pump Operation (High flow pump one loop, low flow pump other loop)
9	15	Three Pump Operation (Highest Flow Pump shutdown)
10	4	One Pump Flow Coastdown From Three Pump Operation (Case 15 conditions, trip highest flow pump in loop with two pumps on)
11	16	Three Pump Operation (Both pumps in loop with high flow pump and lowest flow pump in other loop)
12	5	One Pump Flow Coastdown From Three Pump Operation (Case 16 conditions, trip pump in loop with idle pump)
13	9	Two pump flow Coastdown From Four Pump Operation (Both pumps in highest flow loop tripped)

Sequence Number	Test Case Number	Description
14	13	Two Pump Operation (Both pumps in loop with lowest flow)
15	3	Two Pump Flow Coastdown From Two Pump Operation (Both pumps in loop with lowest flow)
16	10	Four Pump Flow Coastdown (Trip all four pumps simultaneously)
17	14	Two Pump Operation (Both pumps in loop with highest flow)

2.1.3 RESULTS AND EVALUATION

A. Test 1 CF - Cold Flow Test Prior to Installation of Core

With four pumps running, the flows determined after correction to 554° F were as follows: Loop A = 69.81 X 10^6 lb/hr; Loop B = 69.59 X 10^6 lb/hr; Total Flow = 139.40 X 10^6 lb/hr.

B. Test 2 HF - Hot Flow Test Prior to Installation of Core

With four pumps running, the flows determined after correction to 554° F were as follows: Loop A = 71.10 X 10^6 lb/hr; Loop B = 70.02 X 10^6 lb/hr; Total Flow - 141.12 X 10^6 lb/hr.

C. Test 3 [illegible] - Single Pump Flow Tests with Core Installed

Each pump was run individually with the following results:

Pump	Flow X 10^6 lb/hr
A	≈39.6
B	≈39.6
C	≈40.0
D	≈36.8

Reactimeter data showed actual flows such that highest to lowest flow pumps were as follows: C, A, B, D. Thus, the highest flow pump in Loop A is pump C, and the highest flow pump in Loop B is pump A.

D. Test 4 Hot Flow Test with Core Installed

Only Test Cases 11 through 17 were run as a part of this test procedure. Other Test Cases (1 through 10) were run as a part of the RC Flow Coastdown Test. Results of Test Cases 11 through 17 are tabulated below:

Test Case No.	Pumps On	Loop A Flow X 10^6 lb/hr	Loop B Flow X 10^6 lb/hr	Total Flow X 10^6 lb/hr	Total RC Flow Accept. Criteria X 10^6 lb/hr	
					Min.	Max.
11	A & D	34.9	34.8	69.7	67.8	153.4
12	A & C	34.7	35.6	70.3	67.8	153.4
13	C & D	0.4	79.6	80.0	62.4	153.4
14	A & B	79.9	0.2	80.1	62.4	153.4
15	A&B&D	30.0	75.3	105.3	103.2	153.4
16	B&C&D	76.0	30.3	106.3	103.2	153.4
17	A&B&C&D	71.5	71.0	142 5	138.5	153.4

2.1.4 CONCLUSIONS

All of the test objectives were met plus providing a good set of baseline data for future reference. The RC flow rates of Test Cases 11 through 17 were within the allowable limits of the acceptance criteria of the test.

2.2 REACTOR COOLANT FLOW COASTDOWN TEST

2.2.1 PURPOSE

A. To determine flow decay versus time for various (worst case) reactor coolant pump trip combinations.

B. To compare flow coastdown with minimum design flow coastdown on loss of flow.

2.2.2 TEST METHOD

A. One test was run at cold conditions (no core) with all four reactor coolant pumps initially on with a simultaneous trip of all pumps following. Coastdown times versus flows were monitored and recorded utilizing the plant computer, brush recorders, and the B&W supplied reactimeter.

B. Another test was run under hot conditions (no core) with all four reactor coolant pumps initially on with a simultaneous trip of all pumps following. The same data gathering techniques were used as with the cold test.

C. Test cases 1 through 10 were then performed utilizing the same data gathering techniques as in the tests described in A & B above. The tests were run in the following order:

Test Case No.	Description
6	A&B&C&D Running, Trip C
7	A&B&D Running, Trip A&B&D
8	A&B&C&D Running, Trip C&A
1	B&D Running, Trip B
2	A&D Running, Trip A&D
4	A&B&D Running, Trip A
5	B&C&D Running, Trip B
9	A&B&C&D Running, Trip C&D
3	A&B Running, Trip A&B
10	A&B&C&D Running, Trip A&B&C&D
8 (Rerun)	A&B&C&D Running, Trip C&A
9 (Rerun)	A&B&C&D Running, Trip C&D

2.2.3 RESULTS AND EVALUATION

A. Results of the cold coastdown test (no core) showed that flow was present for approximately eighty seconds after the trip of all pumps. The coastdown was smooth throughout the transient. No acceptance criteria were given for a no core condition.

B. Results of the hot coastdown test (no core) showed that flow was present for approximately one hundred and twenty seconds after the trip of all pumps, a considerably longer coastdown time than for cold conditions. As the specific weight of the water is smaller at higher temperatures it is expected that the coastdown time be longer. The coastdown was smooth throughout the transient. No acceptance criteria were given for a no core condition.

C. Summary of test cases 1 through 10 is provided utilizing Figures 2.2-1 through 2.2-10. The lower curve on each page represents the minimum time/flow relationship for each test case. As can be seen, with the retests of test cases 8 and 9, all cases met minimum requirements easily. Test case 6 required that "...flow must decrease to less than or equal to 95.3% of initial S.S. flow within 4.9 sec."

Initial steady state flow was 142.7 X 10^6 lb/hr and flow at + 4.9 seconds was 129.0 X 10^6 lb/hr, or 90.4% of the initial S.S. flow. Test cases 8 and 9 required "...flow must decrease to less than or equal to 95.3% of the initial steady state flow within 2.2 seconds after trip." Case 8 flow was 95.0% and case 9 was 94.9% of initial S.S. flow.

2.2.4 CONCLUSION

Test results show that all acceptance criteria were met. Test instrumentation used was sufficient and performed well. Valuable data was accumulated for proof of adequacy of system design parameters.

FIGURE 2.2-1

TOTAL REACTOR COOLANT SYSTEM FLOW RATE

Vs.

TIME AFTER PUMP TRIP

TEST CASE 1

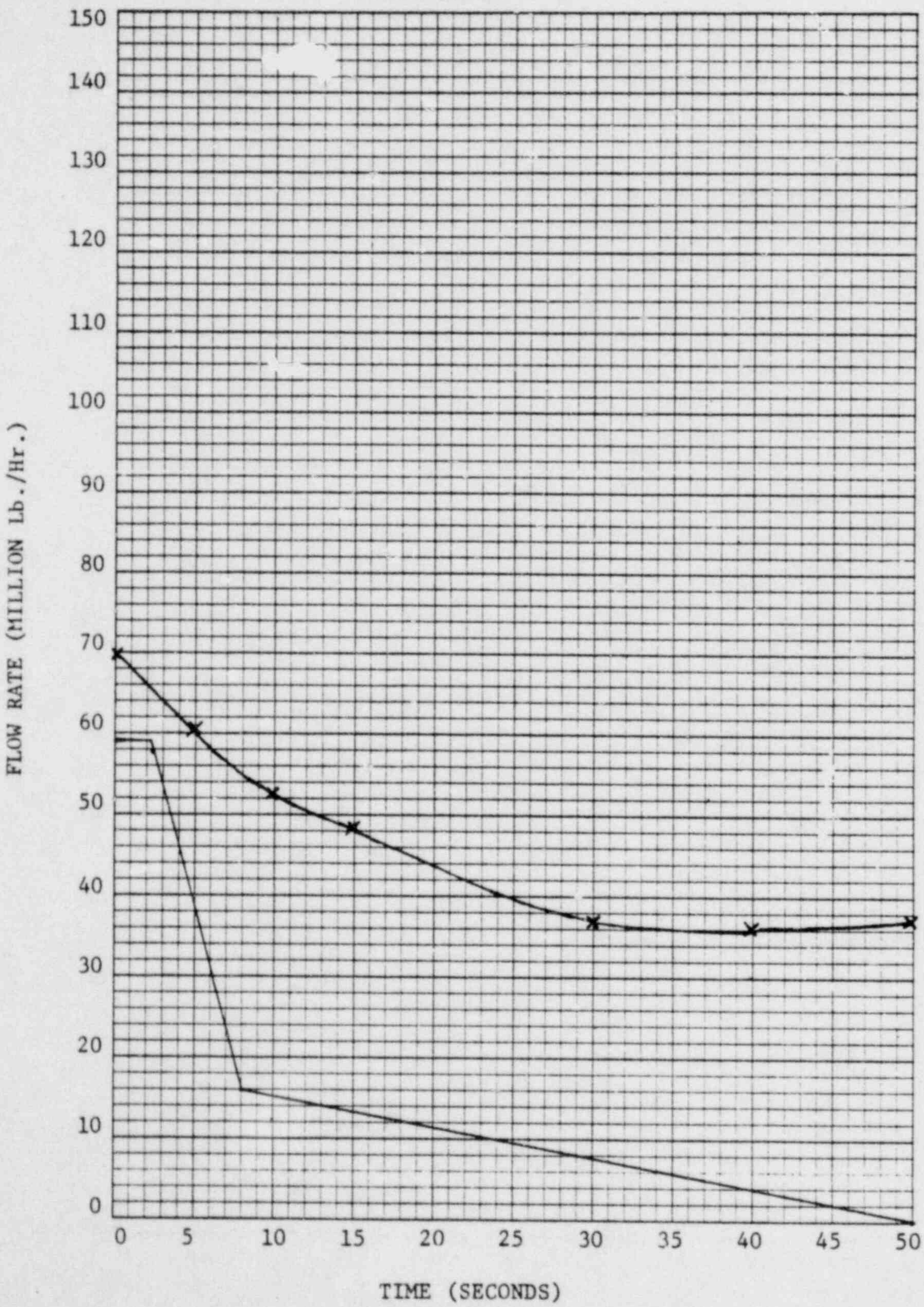


FIGURE 2.2-2

TOTAL REACTOR COOLANT SYSTEM FLOW RATE

Vs.

TIME AFTER PUMP TRIP

TEST CASE 2

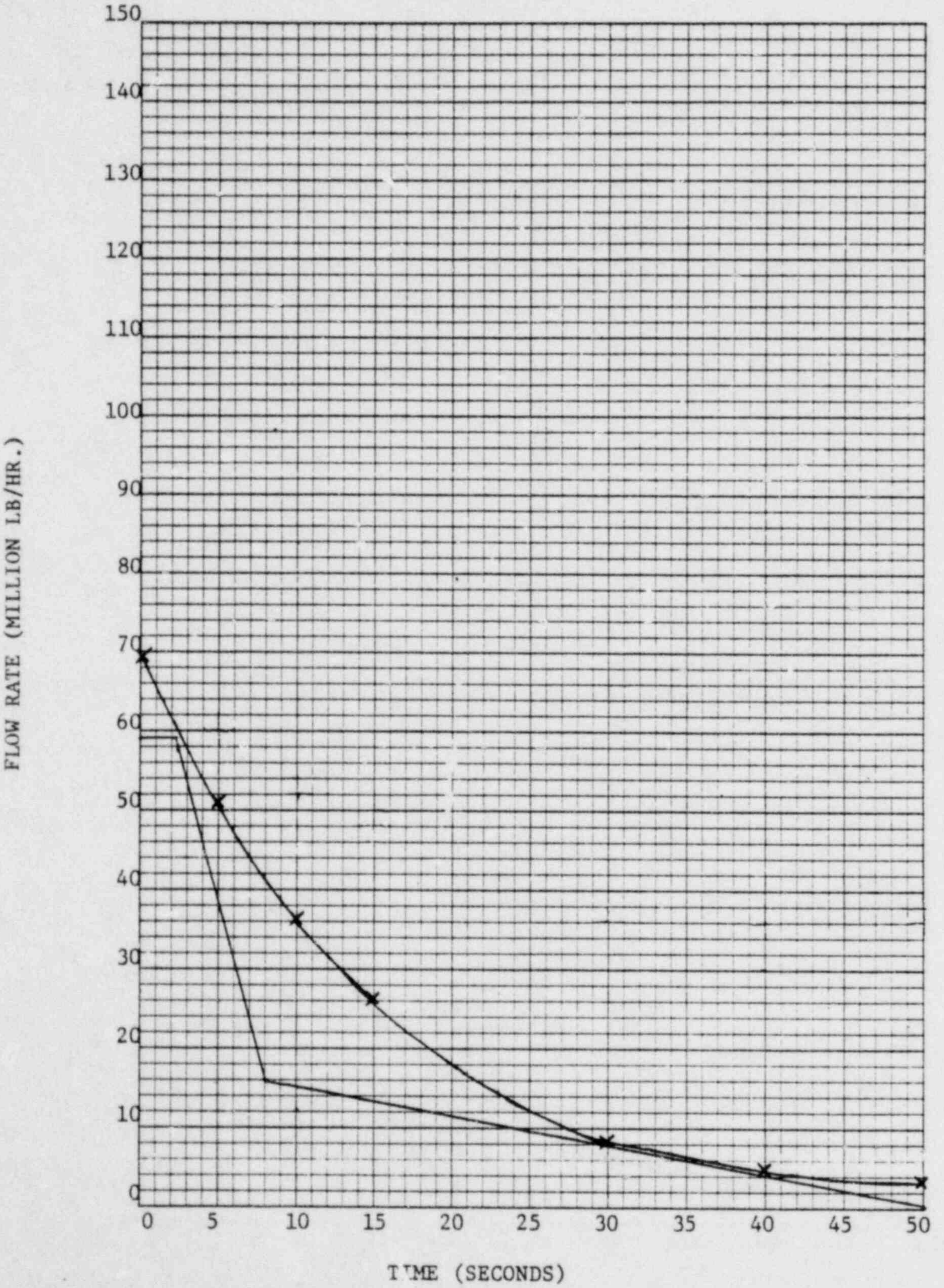


FIGURE 2.2-3

TOTAL REACTOR COOLANT SYSTEM FLOW RATE

Vs.

TIME AFTER PUMP TRIP

TEST CASE 3

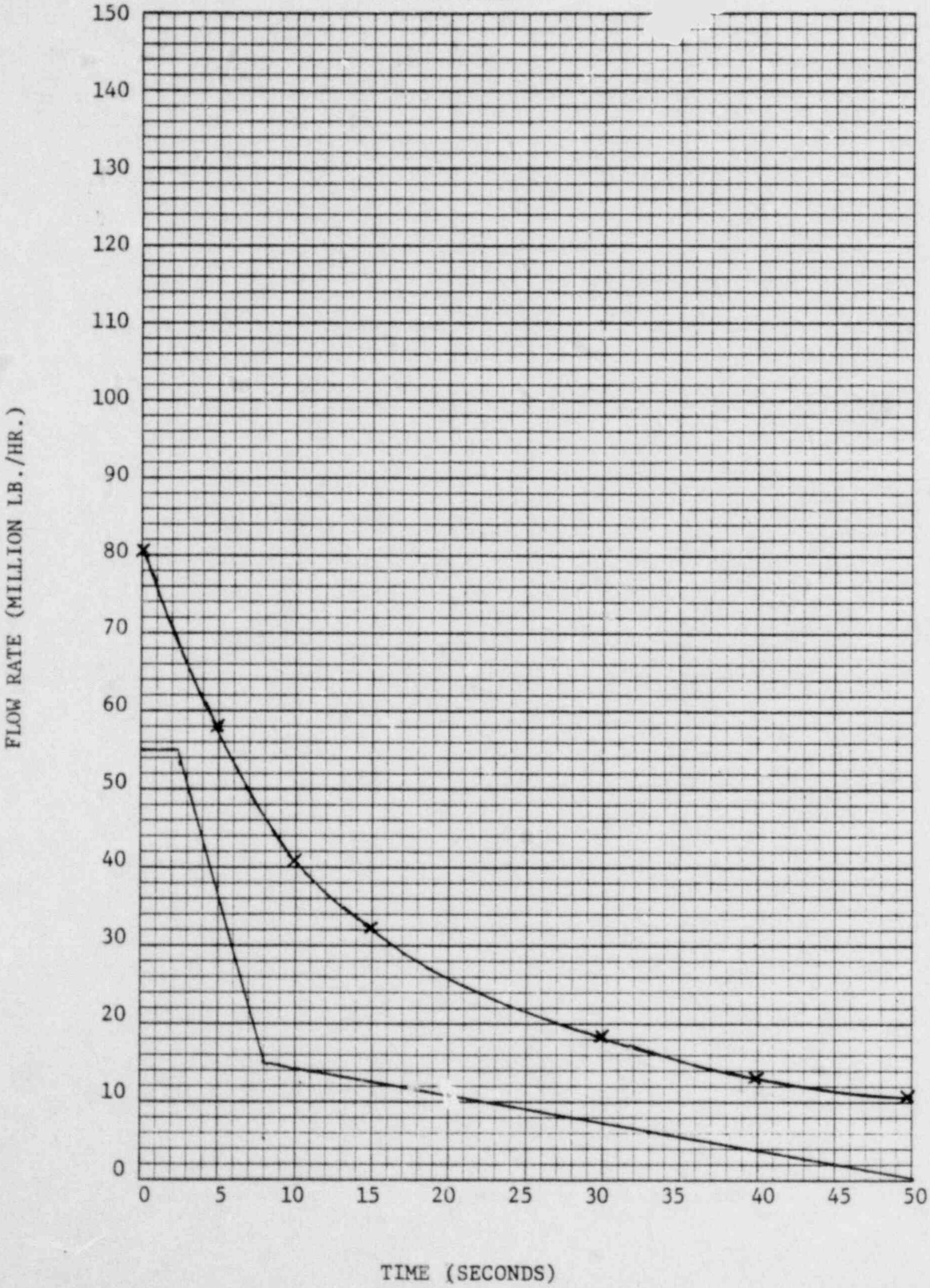


FIGURE 2.2-4

TOTAL REACTOR COOLANT SYSTEM FLOW RATE

Vs.

TIME AFTER PUMP TRIP

TEST CASE 4

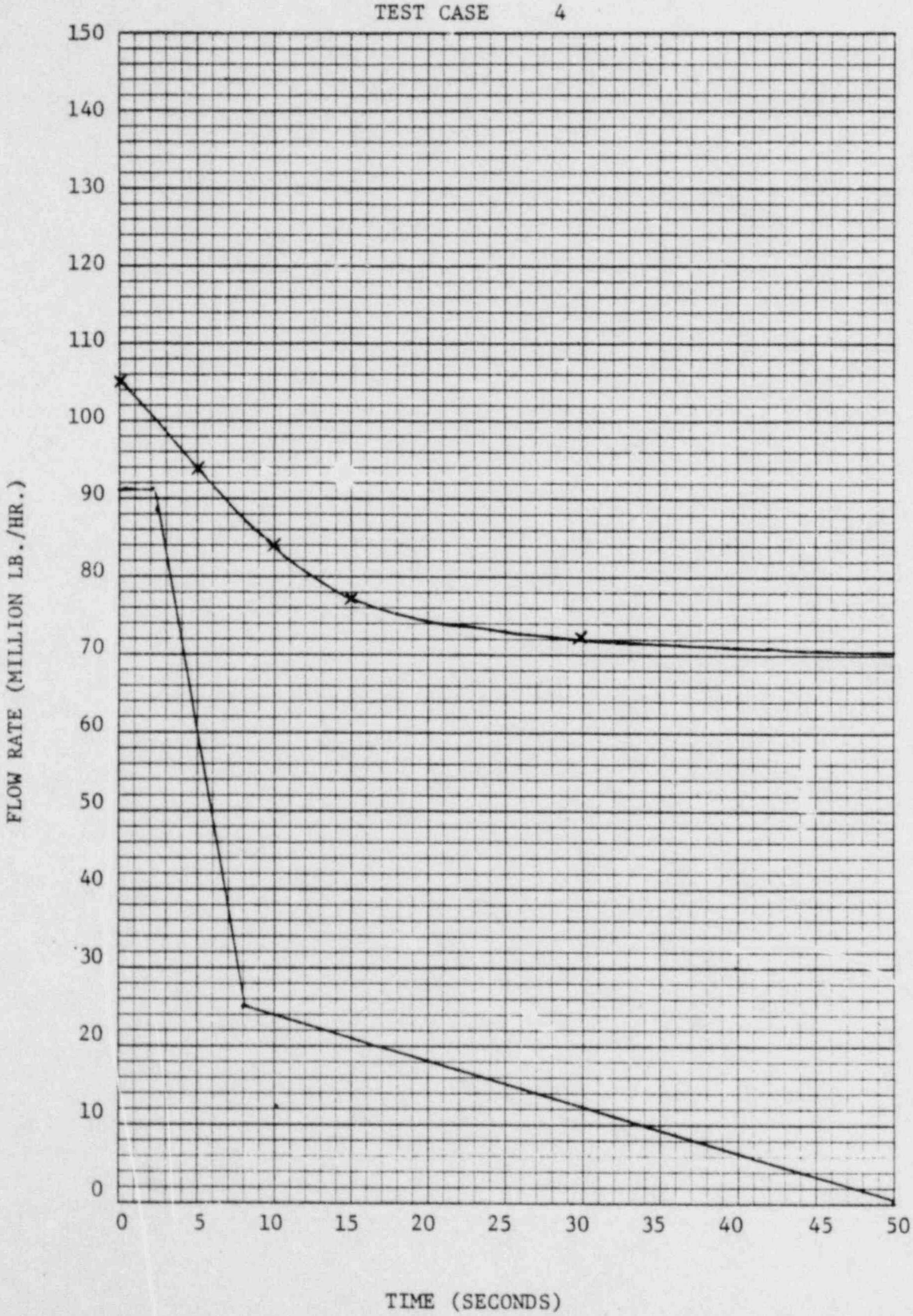


FIGURE 2.2-5

TOTAL REACTOR COOLANT SYSTEM FLOW RATE

Vs.

TIME AFTER PUMP TRIP

TEST CASE 5

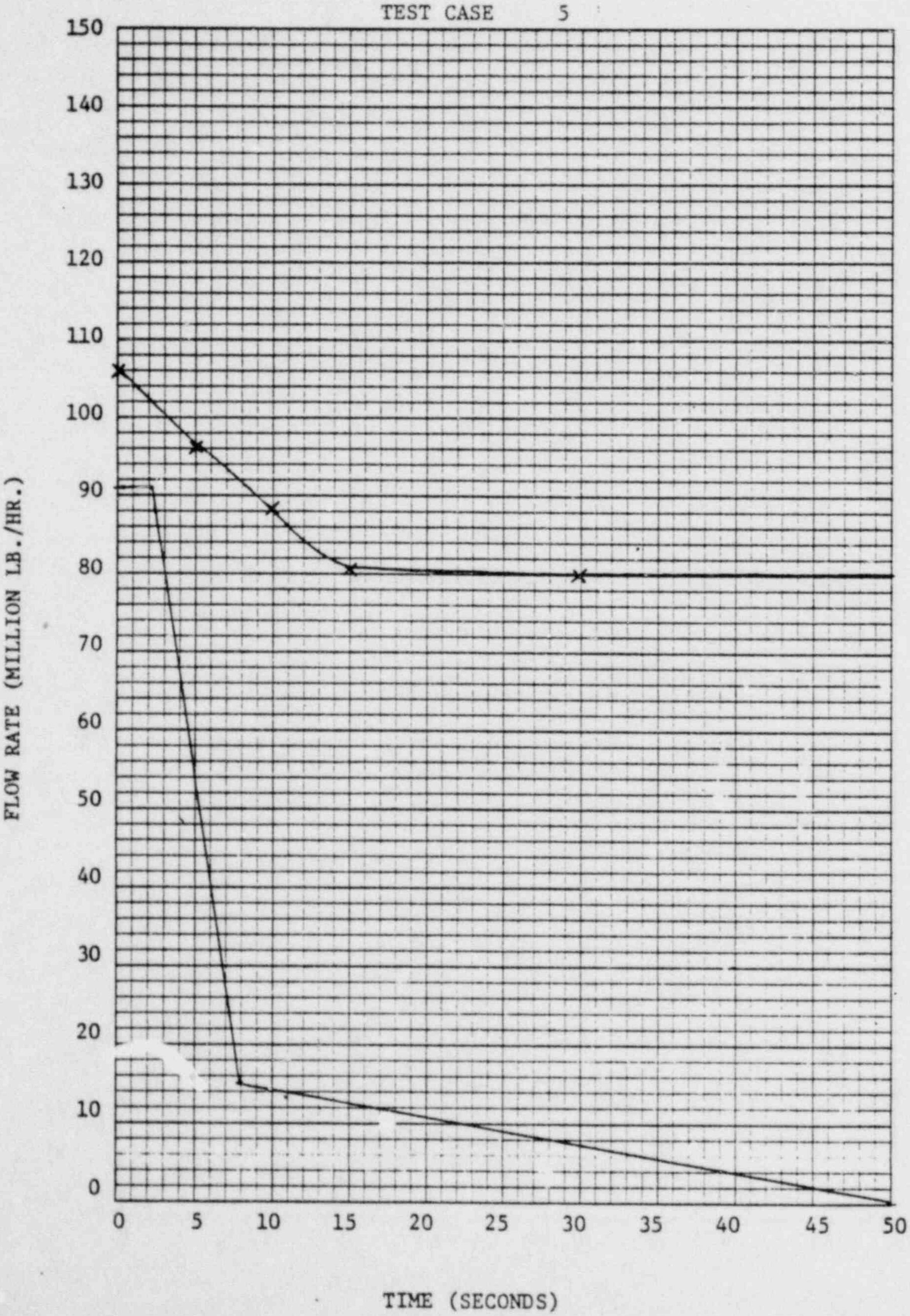


FIGURE 2.2-6

TOTAL REACTOR COOLANT SYSTEM FLOW RATE

Vs.

TIME AFTER PUMP TRIP

TEST CASE 6

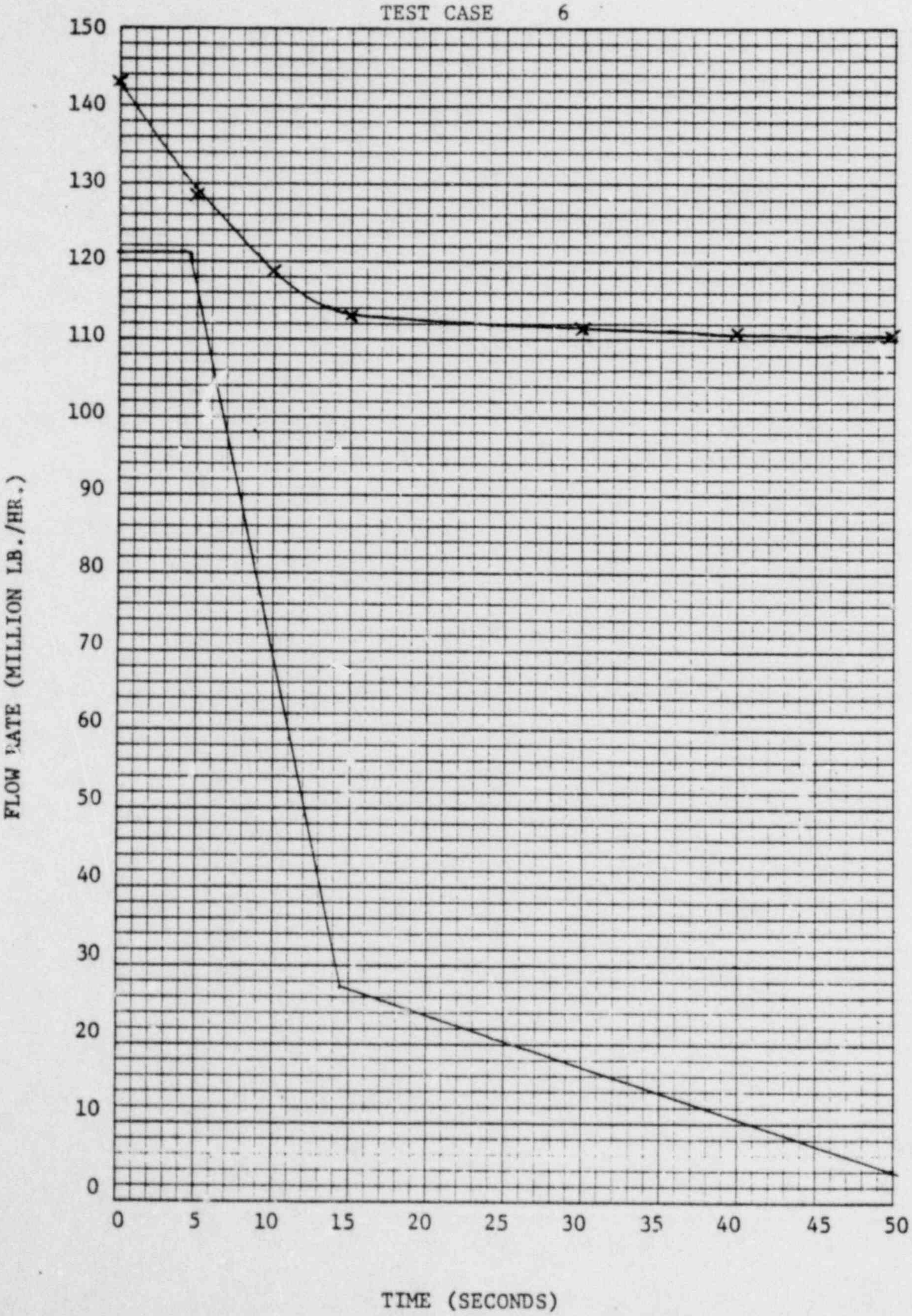


FIGURE 2.2-7

TOTAL REACTOR COOLANT SYSTEM FLOW RATE

Vs.

TIME AFTER PUMP TRIP

TEST CASE 7

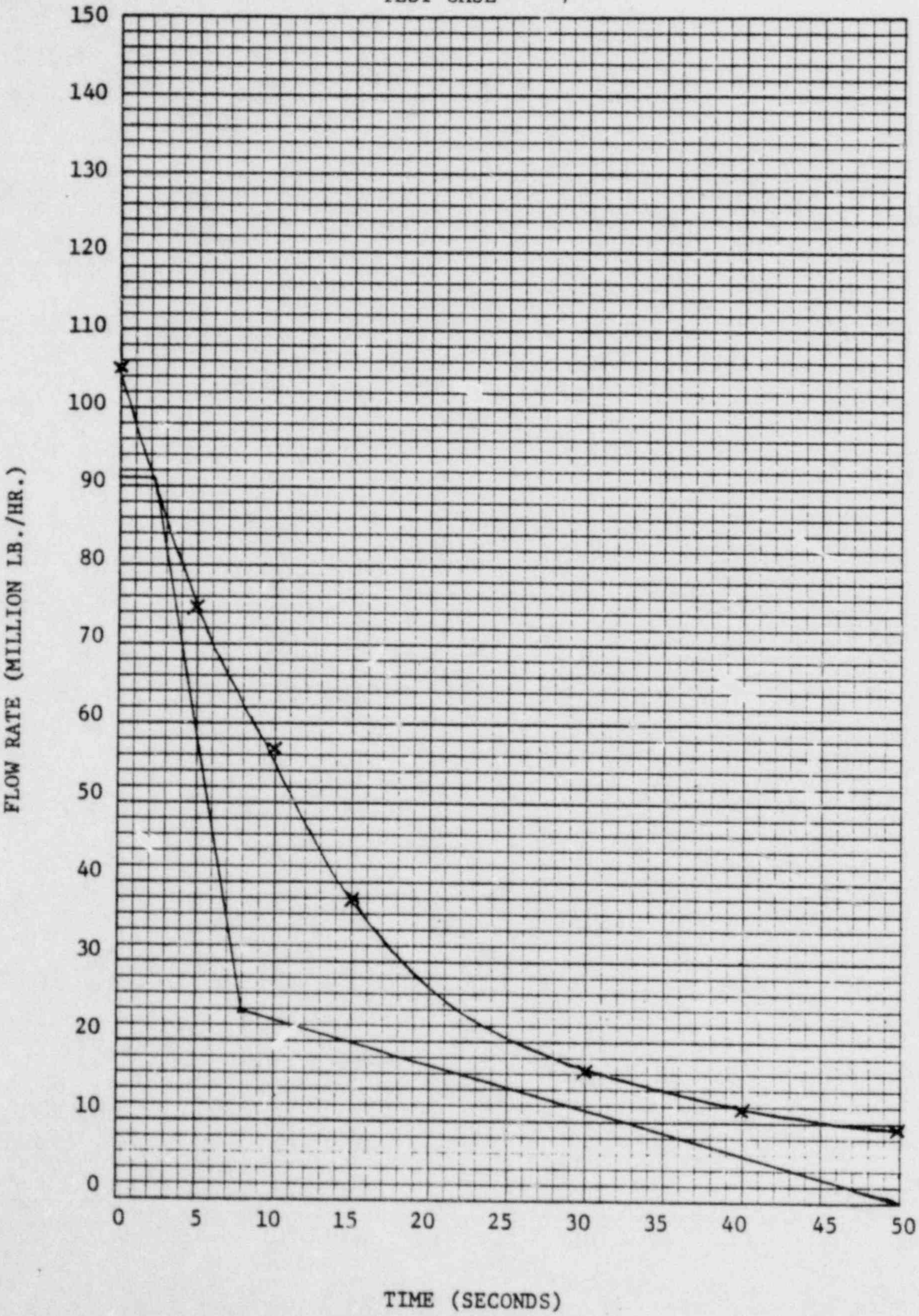


FIGURE 2.2-8

TOTAL REACTOR COOLANT SYSTEM FLOW RATE

Vs.

TIME AFTER PUMP TRIP

TEST CASE 8

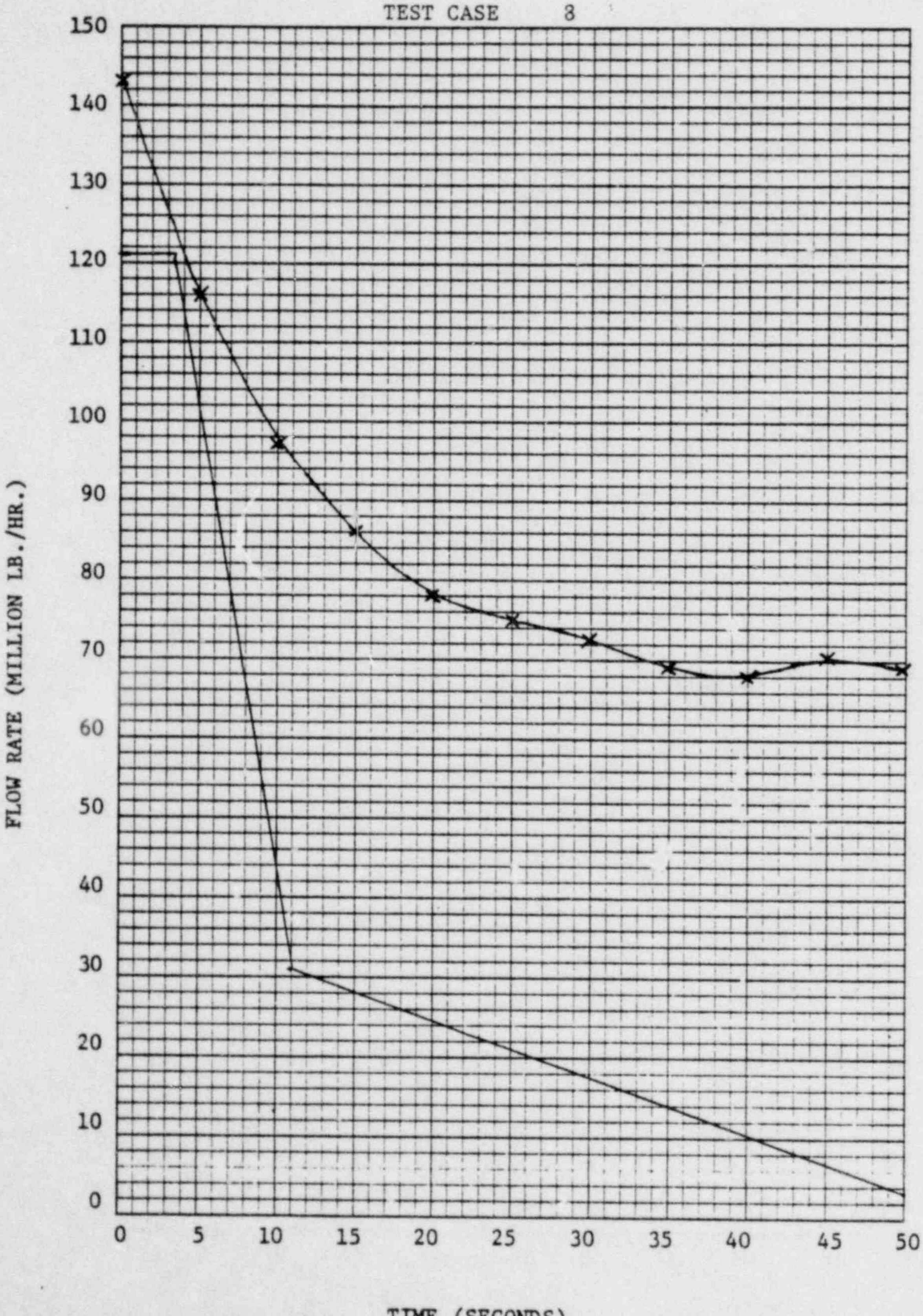


FIGURE 2.2-9

TOTAL REACTOR COOLANT SYSTEM FLOW RATE

Vs.

TIME AFTER PUMP TRIP

TEST CASE 9

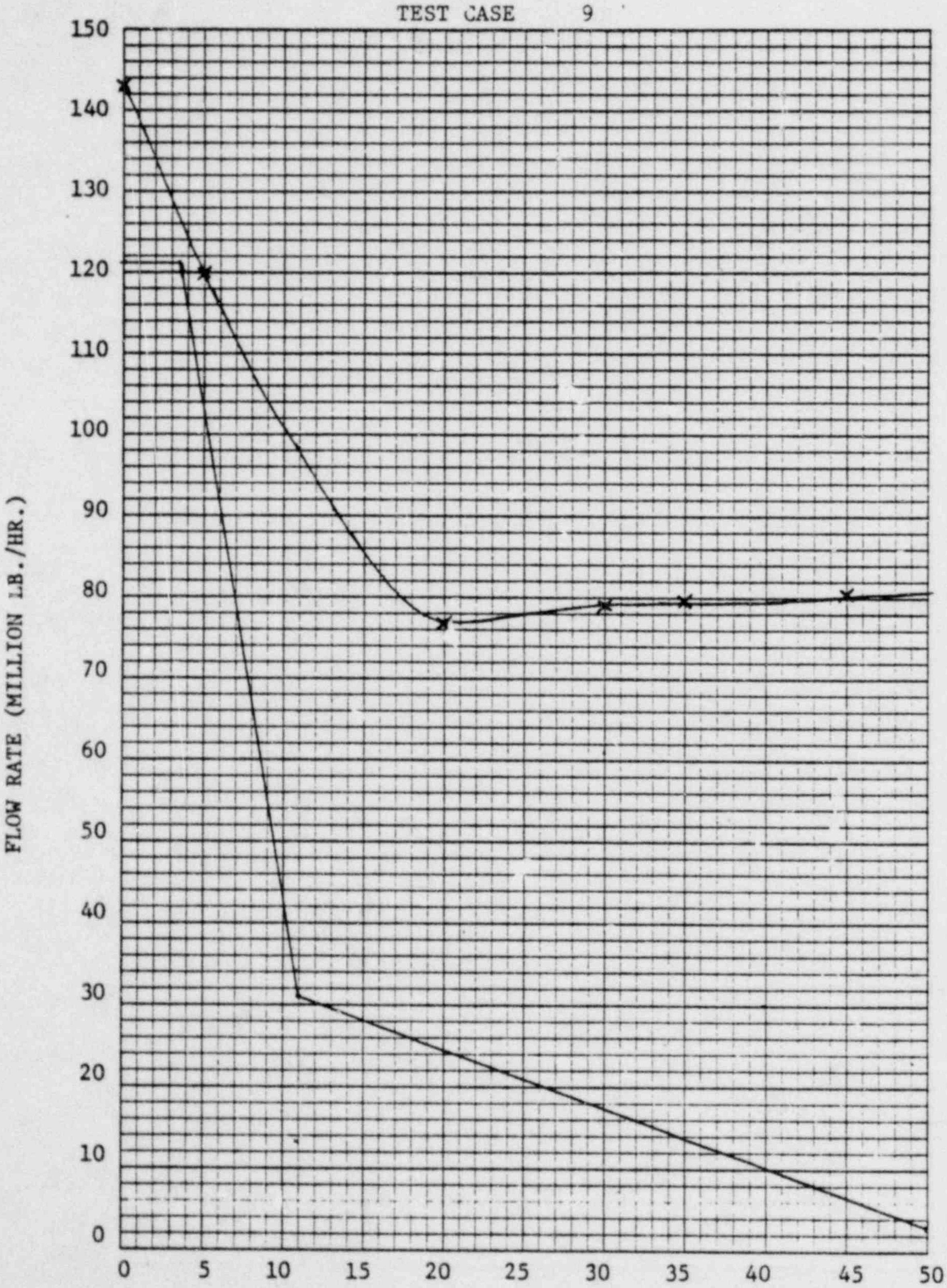


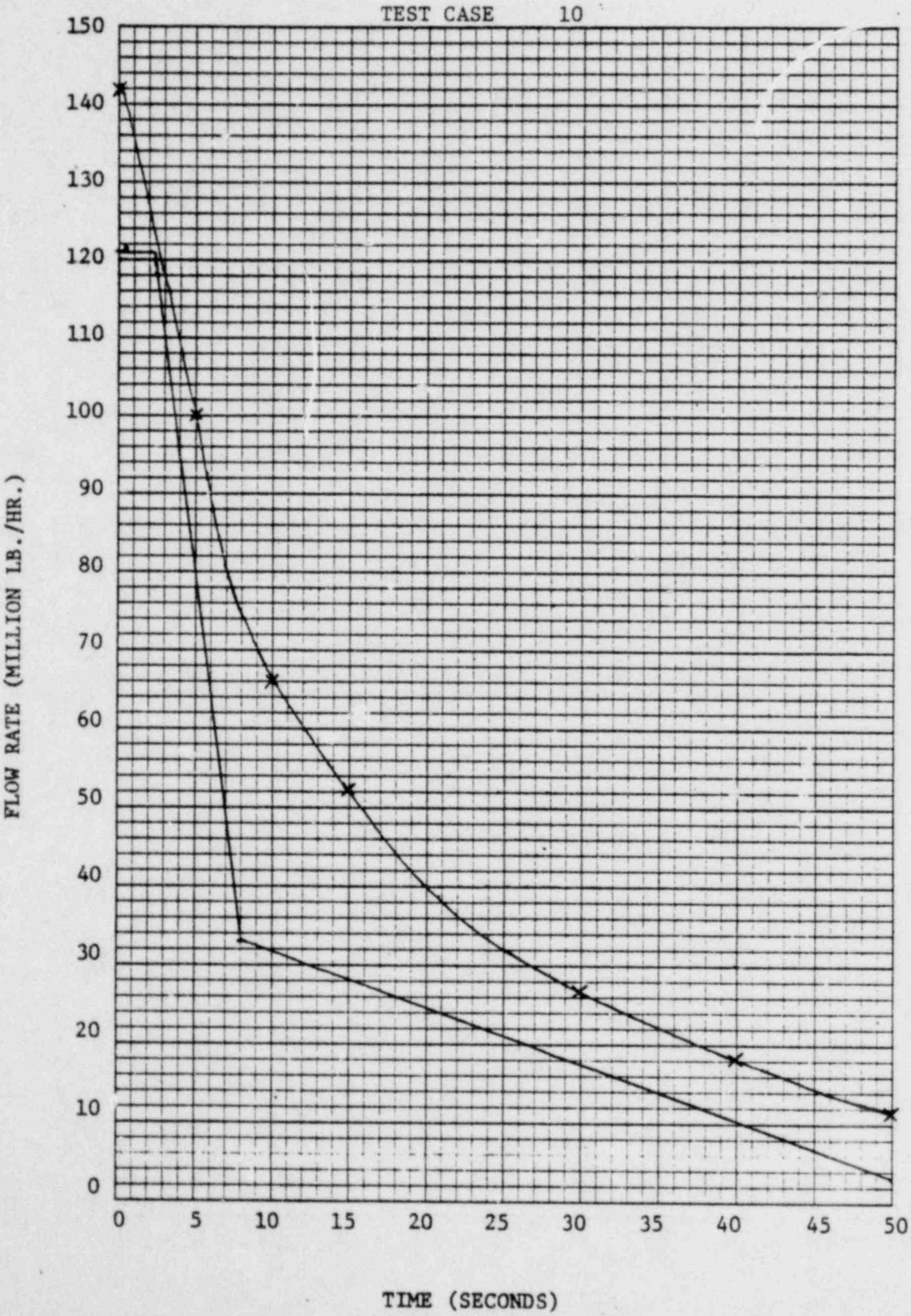
FIGURE 2.2-10

TOTAL REACTOR COOLANT SYSTEM FLOW RATE

Vs.

TIME AFTER PUMP TRIP

TEST CASE 10



2.3 PRESSURIZER OPERATION AND SPRAY FLOW TEST

2.3.1 PURPOSE

The purposes of the Pressurizer Operation and Spray Flow Test are listed below:

A. Evaluate calibration of pressurizer level setpoints and heater interlock.

B. Evaluate proper functioning of pressurizer level controls.

C. By heat balance of the pressurizer, to set the spray flow.

D. By heat balance of the pressurizer, to set the bypass spray flow.

E. Evaluate calibration and proper functioning of pressurizer spray valve and heater controls and setpoints.

2.3.2 TEST METHOD

A. The pressurizer level setpoints and heater interlocks were checked by varying the pressurizer level under cold and no pressure conditions. The actual level was monitored via clear tubing used as a sightglass.

The actual levels were compared with indicated levels and with level switch actuation points. Data was recorded at all points.

B. The pressurizer level controls were tested by inducing transients and monitoring and recording data on the recovery response of the controls.

C. The pressurizer spray flow was set before fuel loading. This initial setting was accomplished during hot functional testing with the bypass spray flow valve closed. Since spray flow is not measured directly, a heat balance was used to determine the actual spray flow. Heater input was measured using accurate voltage and amperage measuring equipment, then calculating the power input. When a proper flow was achieved by manually opening the spray control valve and calculating heater input to balance it, the intermediate position switches were set to stop the valve at that point when automatically actuated.

D. The bypass spray valve was set before and after initial fuel loading by utilizing a method very similar to that used in setting the main spray flow. Upon determination of the proper setting for the valve, it was fixed in place.

E. The pressurizer spray valve and heater setpoints were tested by taking hand control of the spray valve to check heater setpoints and by taking hand control of a large heater bank to check the spray valve setpoint. Pressures were varied and switch actuation points were recorded.

2.3.3 RESULTS AND EVALUATION

Pressurizer controls, setpoints, and general performance were tested per this procedure. All proved to be acceptable.

The following is a listing of test data and acceptance criteria:

A. Pressurizer Level Setpoints and Heater Interlock

	Test Value	Acceptance Criteria
1. Low-Low Level Alarm	37.75 inches	40 ± 6.4 inches
2. Low-Low Level Interlock of Heaters	Functioned Properly	Function Properly
3. Low Level Alarm	160.3 inches	160 ± 6.4 inches
4. High Level Alarm	217.5 inches	220 ± 6.4 inches
5. High-High Level Alarm	272.8 inches	275 ± 6.4 inches

B. Pressurizer Level Controls

1. Increased Letdown Flow	Level cycles but remained within acceptance criteria.	Constant, ± 6 inches
2. Change Setpoint	Level increased and stabilized at new setpoint.	Reach constant level with no valve hunting.

C. Setting of Spray Flow

	Test Value	Acceptance Criteria
1. Before Fuel Load with 3 RCP's	160 gpm	170 ± 10 gpm
2. After Fuel Load with 4 RCP's	192 gpm	190 + 19, -6 gpm

D. Setting of Bypass Spray Flow

1. Before Fuel Load	1.04 gpm	1.0 + 0.5,-0.25 gpm
2. After Fuel Load	1.10 gpm	1.0 + 0.5,-0.25 gpm

E. Heater and Spray Control Valve Setpoints

1. Heater Bank No. 1:		
Completely On At	2135 psig	2135 ± 16 psig
Completely Off At	2170 psig	2155 ± 16 psig
2. Heater Bank No. 2:		
Completely On At	2135 psig	2135 ± 16 psig
Completely Off At	2170 psig	2155 ± 16 psig
3. Heater Bank No. 3:		
On At	2131 psig	2135 ± 16 psig
Off At	2154 psig	2155 ± 16 psig
4. Heater Bank No. 4:		
On At	2116 psig	2120 ± 16 psig
Off At	2139 psig	2140 ± 16 psig
5. Heater Bank No. 5:		
On At	2101 psig	2105 ± 16 psig
Off At	2125 psig	2125 ± 16 psig

	Test Value	Acceptance Criteria
6. Spray Control Valve:		
Opens At	2204 psig	2205 $\pm$ 16 psig
7. Heater Banks Come On and Off Sequentially (No. 1 and No. 2 Operate together.)	Yes	Yes

2.3.4 CONCLUSION

From analysis of all test data, it was concluded that the operational functions, controls and instrumentation, and spray flow are all acceptable and satisfactory.

2.4 POST HOT FUNCTIONAL TEST INTERNALS INSPECTION

2.4.1 PURPOSE

The purpose of this inspection was to visually determine that there was no unacceptable wear, galling, cracking, fretting, loose parts, or distortion of the reactor vessel internals following hot functional testing.

2.4.2 TEST METHOD

The following areas of the reactor vessel were visually inspected:

All major load bearing elements of the reactor internals relied upon to retain the core structure in place; the lateral, vertical, and torsional restraints provided within the vessel; those locking and bolting devices whose failure could adversely affect the structural integrity of the internals; and the interior of the reactor vessel for evidence of loose parts or foreign material.

2.4.3 RESULTS AND EVALUATION

The condition of the internals was found to be good with the exception of one surveillance specimen holder tube. One of the hinges on the specimen holder was improperly installed and the holder would not adequately lock in the retracted position. This was corrected by the Babcock and Wilcox Company.

There were numerous minor dents, scratches, and galls found but none were considered significant. No bad welds, cracks, or loose parts were found.

2.4.4 CONCLUSION

The acceptance criteria for this test were met in full without deficiencies.

2.5 INITIAL FUEL LOADING

2.5.1 INTRODUCTION

At 12:45 a.m. on May 29, 1974, the first fuel assembly, 1C25, was removed from location B-9 in the spent fuel pool and loaded into location N-14 in the ANO-1 core. Fuel loading was completed on June 5, 1974 with the loading of fuel assembly 1C02 and visual verification of the core loading. Figure 2.5-1 shows the core configuration as verified after loading.

During the fuel loading procedure the core neutron flux was measured by two temporary incore detectors and one out-of-core source range channel NI-2. All the detectors were of the BF-3 proportional type. Only two detectors are actually required to be operating by the fuel loading procedure. Plots of the inverse neutron count rate ratio were kept independently for each detector channel. The plots of 1/m versus number of fuel assemblies for the three detector channels are given in Figures 2.5-2, 2.5-3 and 2.5-4. New base count rates were established for a detector whenever the detector or a neutron startup source was moved. The step-by-step loading procedure is shown in Table 2.5-1.

2.5.2

Fuel loading began after calibrating and source checking the neutron detectors. The boron concentration in the reactor vessel, spent fuel pool and reactor coolant system was greater than 1850 ppm. The first two assemblies containing the neutron startup sources were placed in core positions N-14 and D-2. Fuel loading was continued through fuel assembly 1C45 (Step No. 5). At this point fuel loading was suspended while the fuel transfer mechanism was being repaired. Two divers were sent into the deep end of the fuel transfer canal to attach the fuel transfer tube cover to the fuel transfer tube. After the divers emerged from the fuel transfer canal, the fuel transfer tube and fuel tilt pit were drained and personnel entered to work on the fuel transfer mechanism. Repairs were made and the area was reflooded.

Fuel loading was resumed with fuel assembly 1C55 (Step No. 6) and continued routinely through fuel assembly 1B46 (Step No. 25). After 1B46 was loaded, auxiliary detector B was removed from location P-10 for relocation to position M-7. At this point the holder for auxiliary detector B was discovered to be leaning approximately 30° from the vertical. Using procedure 1701.03, <u>Retrieval of Auxiliary Detector Holder for Inspection</u>, the detector was uprighted and removed for inspection. The detector holder and the top of the reactor grid plate were inspected and no damage was observed. The detector holder was replaced in position M-7. All detectors were source checked and a new base count rate was established. Refueling resumed with fuel assembly 1A49 (Step No. 26).

After fuel assembly 1A24 (Step No. 46) was loaded, detector A was moved from location H-14 to E-9 and a new base count rate was established. At this time detector A was moved from location E-9 to F-10 as per a temporary change to the fuel loading procedure due to the low count rate at location E-9. A new base count rate was established here. After loading fuel assembly 1B38 (Step No. 47), a mechanic was called in to repair the hydraulic pump on the reactor building side of the fuel upender. The pump was repaired.

Fuel loading was resumed; but when an attempt was made to seat 1B36 (Step No. 51) in location E-8, it would not seat. After repeated unsuccessful attempts to seat 1B36 in location E-8, an attempt was made to seat 1B36 in location D-8. This attempt failed and fuel assembly 1B36 was returned to the storage rack in the deep end of the fuel transfer canal. The dummy fuel assembly was successfully seated in location E-8. The dummy assembly was removed and another attempt was made to seat 1B36. Fuel assembly 1B36 could not be seated and was returned to the spent fuel pool for inspection.

A temporary procedure change was used to bypass Step No. 51 and continue with fuel loading. Fuel assemblies 1A19, 1A18, 1B54 and 1B19 were loaded to completely surround location E-8. Then, fuel assembly 1B36 was successfully loaded into location E-8. Fuel loading continued routinely through fuel assembly 1A51 (Step No. 80). Fuel loading was halted when it was discovered that a cable on the fuel mast and transfer mechanism had been sheared. The cable was replaced and fuel loading continued without incident through fuel assembly 1C14 (Step No. 91). Fuel assembly 1C57 (Step No. 92) failed to seat on several attempts. The dummy fuel assembly was seated in location K-2 which enabled a successful seating of 1C57. Auxiliary detector B was moved from location M-7 to location L-1 and a new initial count rate was established. Fuel loading continued through fuel assembly 1C01 (Step No. 123). Auxiliary detector A was moved from F-10 to E-10 using a temporary procedure change. A new initial count rate was established. Fuel loading continued through fuel assembly 1A43 (Step No. 132). Fuel assembly 1B52 would not seat until the dummy assembly was loaded into location E-13 as a guide. Similar loading difficulties were encountered with fuel assemblies 1C11 and 1B13.

A discovery was made that the indexing on the bridge was approximately 1/8" off. This was corrected and fuel assembly 1C44 was loaded. Fuel loading continued routinely through fuel assembly 1C22 (Step No. 171). Auxiliary detector A was relocated to location F-15 (Step No. 142A). New count rates were determined for the neutron detectors. Fuel assembly 1C50 was loaded using the dummy assembly as a guide. Fuel loading continued through fuel assembly 1C05 (Step No. 177). Auxiliary detector B was removed from the core and placed in the storage rack in the deep end of the fuel transfer canal. Fuel assembly 1C41 (Step No. 178) was placed in location L-1. Auxiliary detector A was removed and fuel assembly 1C02 (Step No. 179) was loaded into location F-15. Fuel loading was complete. The fuel assembly identification numbers and locations were visually verified.

2.5.3 RESULTS AND EVALUATION

The results of the initial fuel loading of Arkansas Nuclear One - Unit 1 are detailed in section 2.5.2. Figures 2.5-1 through 2.5-4 summarize the results of fuel loading. Figure 2.5-1 is the core configuration as visually verified following fuel loading. The 1/m plots maintained during fuel loading are shown in Figures 2.5-2 through 2.5-4. During fuel loading count rates greater than 2 cps were maintained on the auxiliary detectors. After fuel loading was complete, the final count rate on the permanently installed nuclear instrumentation was greater than 3 cps.

2.5.4 CONCLUSION

The initial fuel loading for ANO-1 was carried out in a safe and orderly manner. The technique of using a dummy assembly as a guide for an assembly which would not seat initially was used as well as "boxing in" a location with other assemblies before trying to reseat an assembly which would not seat initially. These techniques allowed all assemblies to be loaded with minimum delay.

TABLE 2.5-1

STEP NUMBER	FUEL ASSEMBLY I.D. NO.	CONTAINS CRA, APSR BPRA, ORA	FROM *	TO CORE POSITION
1	1C25	032 (s)	SP B-9	N-14
2	1C04	033 (s)	SP A-8	D-2
3	1C18	—	SP A-22	O-13
4	1C49	B47	SP E-22	N-13
5	1C45	B52	SP H-21	O-12
6	1C55	—	SP D-23	M-14
7	1A39	C10	SP F-18	N-12
8	1A28	C09	SP K-16	M-13
9	1B14	B66	SP F-6	M-12
10	1B21	B32	SP G-7	L-13
11	1B12	B68	SP D-6	N-11
12	1A56	A03	SP K-20	L-12
13	1A10	C29	SP E-14	M-11
14	1B04	B57	SP F-4	L-11
15	1B28	B40	SP H-8	K-12
16	1B55	B65	SP K-6	M-10
17	1A03	C43	SP E-13	L-10
18	1A13	C50	SP H-14	K-11
19	1B42	B08	SP D-11	K-10
20	1B02	B09	SP D-4	L-9
21	1B47	B37	SP C-12	H-11
22	1A20	A04	SP H-15	N-10
23	1A27	C44	SP H-16	M-9
24	1B41	B28	SP C-11	N-9
25	1B46	B43	SP H-11	M-8
25A	DetectorB		RX P-10	M-7
26	1A49	C11	SP K-19	O-11
27	1B49	B51	SP E-12	O-10
28	1C09	020	SP A-13	P-12
29	1C28	—	SP B-12	P-11
30	1A07	C56	SP K-13	K-9
31	1A25	C57	SP F-16	L-8
32	1A21	C55	SP K-15	H-10
33	1B30	B04	SP D-9	K-8
34	1B56	B02	SP K-7	H-9
35	1B26	C61	SP F-8	H-8

* SP = SPENT FUEL POOL ; Rx = REACTOR CORE

TABLE 2.5-1

STEP NUMBER	FUEL ASSEMBLY I.D. NO.	CONTAINS CRA, APSR BPRA, ORA	FROM *	TO CORE POSITION
36	1B27	B10	SP G-8	L-7
37	1B31	B07	SP E-9	G-10
38	1A12	C60	SP G-14	K-7
39	1A14	C54	SP K-14	G-9
40	1B34	B03	SP H-9	H-7
41	1B33	B01	SP G-9	G-8
42	1A44	C58	SP D-19	G-7
43	1B01	B12	SP C-4	K-6
44	1B40	B06	SP H-10	F-9
45	1A02	C59	SP D-13	H-6
46	1A24	C53	SP E-16	F-8
46A	Detector A		Rx H-14	E-9
47	1B38	B11	SP F-10	G-6
48	1B25	B05	SP E-8	F-7
49	1A42	C51	SP K-18	F-6
50	1B45	B36	SP G-11	H-5
51	1B36	B30	SP D-10	E-8
52	1A19	C42	SP G-15	G-5
53	1A18	C52	SP F-15	E-7
54	1B54	B59	SP K-5	F-5
55	1B19	B62	SP E-7	E-6
56	1A11	C35	SP F-14	E-5
57	1B43	B33	SP E-11	G-4
58	1B06	B27	SP H-4	D-7
59	1A33	A07	SP G-17	F-4
60	1A26	A08	SP G-16	D-6
61	1B58	B55	SP K-9	E-4
62	1B16	B67	SP H-6	D-5
63	1A35	C22	SP K-17	D-4
64	1B50	B31	SP F-12	F-3
65	1B48	B22	SP D-12	C-6
66	1A41	C21	SP H-18	E-3
67	1C48	B26	SP D-22	D-3
68	1C52	——	SP H-22	E-2
69	1A30	C23	SP D-17	C-5
70	1C26	B21	SP B-10	C-4

* SP = SPENT FUEL POOL ; Rx = REACTOR CORE

TABLE 2.5-1

STEP NUMBER	FUEL ASSEMBLY I.D. NO.	CONTAINS CRA, APSR BPRA, ORA	FROM *	TO CORE POSITION
71	1C33	—	SP B-17	C-3
72	1C24	003	SP B-8	B-4
73	1C54	—	SP C-23	B-5
74	1A43	C47	SP C-19	L-6
75	1A05	C48	SP G-13	K-5
76	1B05	B61	SP G-4	L-5
77	1A06	C49	SP H-13	H-4
78	1B57	B39	SP K-8	K-4
79	1A15	A06	SP C-15	L-4
80	1A51	C34	SP D-20	G-3
81	1B08	B35	SP E-5	H-3
82	1A38	C33	SP E-18	K-3
83	1B51	B41	SP G-12	L-3
84	1C13	C20	SP A-17	F-2
85	1B24	B16	SP D-8	G-2
86	1C19	C19	SP A-23	H-2
87	1B60	B19	SP K-11	K-2
88	1C10	C18	SP A-14	L-2
89	1C43	—	SP F-21	F-1
90	1C37	—	SP B-21	G-1
91	1C14	—	SP A-18	H-1
92	1C57	—	SP F-23	K-1
92A	Detector B		Rx M-7	L-1
93	1C17	C14	SP A-21	P-10
94	1C03	—	SP A-7	R-10
95	1A01	C30	SP C-13	O-9
96	1B61	B13	SP K-12	P-9
97	1C60	—	SP K-23	R-9
98	1A45	C45	SP E-19	N-8
99	1B23	B38	SP C-8	O-8
100	1C23	C13	SP B-7	P-8
101	1C30	—	SP B-14	R-8
102	1A55	C46	SP H-20	M-7
103	1B37	B45	SP E-10	N-7
104	1A47	C31	SP G-19	O-7
105	1B29	B14	SP C-9	P-7

* SP = SPENT FUEL POOL ; Rx = REACTOR CORE

TABLE 2.5-1

STEP NUMBER	FUEL ASSEMBLY I.D. NO.	CONTAINS CRA, APSR BPRA, ORA	FROM *	TO CORE POSITION
106	1C40	—	SP C-21	R-7
107	1B03	B64	SP E-4	M-6
108	1A53	A05	SP F-20	N-6
109	1B22	B49	SP H-7	O-6
110	1C16	C02	SP A-20	P-6
111	1C42	—	SP E-21	R-6
112	1A52	C32	SP E-20	M-5
113	1B39	B53	SP G-10	N-5
114	1A46	C05	SP F-19	O-5
115	1C53	—	SP K-22	P-5
116	1B15	B63	SP G-6	M-4
117	1A29	C16	SP C-17	N-4
118	1C47	B48	SP C-22	O-4
119	1A37	C17	SP D-18	M-3
120	1C46	B29	SP K-21	N-3
121	1C32	—	SP B-16	O-3
122	1C51	—	SP G-22	M-2
123	1C01	—	SP A-5	N-2
124	1A22	C39	SP C-16	F-10
125	1A17	C40	SP E-15	G-11
126	1B35	B60	SP C-10	F-11
127	1A08	C41	SP C-14	H-12
128	1B32	B34	SP F-9	G-12
129	1A32	A02	SP F-17	F-12
130	1A36	C28	SP C-18	K-13
131	1B09	B50	SP F-5	H-13
132	1A48	C27	SP H-19	G-13
133	1B52	B42	SP H-12	F-13
134	1C11	C08	SP A-15	L-14
135	1B10	B20	SP G-5	K-14
136	1C20	C07	SP B-4	H-14
137	1B13	B15	SP E-6	G-14
138	1C06	C06	SP A-10	F-14
139	1C44	—	SP G-21	L-15
140	1C36	—	SP B-20	K-15
141	1C15	—	SP A-19	H-15

* SP = SPENT FUEL POOL ; Rx = REACTOR CORE

TABLE 2.5-1

STEP NUMBER	FUEL ASSEMBLY I.D. NO.	CONTAINS CRA, APSR BPRA, ORA	FROM *	TO CORE POSITION
142	1C59	——	SP H-23	G-15
142A	Detector A		Rx E-9	F-15
143	1C12	C24	SP A-16	B-6
144	1C34	——	SP B-18	A-6
145	1A23	C36	SP D-16	C-7
146	1B20	B18	SP F-7	B-7
147	1C58	——	SP G-23	A-7
148	1A16	C37	SP D-15	D-8
149	1B07	B23	SP D-5	C-8
150	1C38	C01	SP B-22	B-8
151	1C21	——	SP B-5	A-8
152	1A09	C38	SP D-14	E-9
153	1B44	B46	SP F-11	D-9
154	1A40	C25	SP G-18	C-9
155	1B59	B17	SP K-10	B-9
156	1C56	——	SP E-23	A-9
157	1B18	B56	SP D-7	E-10
158	1A54	A01	SP G-20	D-10
159	1B53	B24	SP K-4	C-10
160	1C31	C12	SP B-15	B-10
161	1C35	——	SP B-19	A-10
162	1A50	C26	SP C-20	E-11
163	1B17	B54	SP C-7	D-11
164	1A04	C03	SP F-13	C-11
165	1C39	——	SP B-23	B-11
166	1B11	B58	SP C-6	E-12
167	1A31	C04	SP E-17	D-12
168	1C27	B25	SP B-11	C-12
169	1A34	C15	SP H-17	E-13
170	1C29	B44	SP B-13	D-13
171	1C22	——	SP B-6	C-13
172	1C50	——	SP F-22	E-14
173	1C07	——	SP A-11	D-14
174	1C25	032 (s)	Rx N-14	P-4
175	1C08	——	SP A-12	N-14
176	1C04	033 (s)	Rx D-2	B-12

* SP = SPENT FUEL POOL ; Rx = REACTOR CORE

TABLE 2.5-1

STEP NUMBER	FUEL ASSEMBLY I.D. NO.	CONTAINS CRA, APSR BPRA, ORA	FROM *	TO CORE POSITION
177	1C05	—	SP A-9	D-2
177A	Detector B		Rx L-1	Remove
178	1C41	—	SP D-21	L-1
178A	Detector A		Rx F-15	Remove
179	1C02	—	SP A-6	F-15

* SP = SPENT FUEL POOL ; Rx = REACTOR CORE

FIGURE 2.5-1

CORE LOADING PLAN

1XXX YYY

1XXX : FUEL ASSEMBLY IDENTIFICATION NUMBER

YYY : CONTROL ROD, AXIAL POWER SHAPING ROD, ORIFICE ROD, OR BURNABLE POISON ROD ASSEMBLY IDENTIFICATION NUMBER (*** - NO ROD INSERTED).

-X- FUEL TRANSFER CANAL ↱

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						1C34 ***	1C58 ***	1C21 ***	1C56 ***	1C35 ***					
B				1C24 003	1C54 ***	1C12 C24	1B20 B18	1C38 C01	1B59 B17	1C31 C12	1C39 ***	1C04 033			
C			1C33 ***	1C26 B21	1A30 C23	1B48 B22	1A23 C36	1B07 B23	1A40 C25	1B53 B24	1A04 C03	1C27 B25	1C22 ***		
D		1C05 ***	1C48 B26	1A35 C22	1B16 B67	1A26 A08	1B06 B27	1A16 C37	1B44 B46	1A54 A01	1B17 B54	1A31 C04	1C29 B44	1C07 ***	
E		1C52 ***	1A41 C21	1B58 B55	1A11 C35	1B19 B62	1A18 C52	1B36 B30	1A09 C38	1B18 B56	1A50 C26	1B11 B58	1A34 C15	1C50 ***	
F	1C43 ***	1C13 C20	1B50 B31	1A33 A07	1B54 B59	1A42 C51	1B25 B05	1A24 C53	1B40 B06	1A22 C39	1B35 B60	1A32 A02	1B52 B42	1C06 C06	1C02 ***
G	1C37 ***	1B24 B16	1A51 C34	1B43 B33	1A19 C42	1B38 B11	1A44 C58	1B33 B01	1A14 C54	1B31 B07	1A17 C40	1B32 B34	1A48 C27	1B13 B15	1C59 ***
H	1C14 ***	1C19 C19	1B08 B35	1A06 C49	1B45 B36	1A02 C59	1B34 B03	1B26 C61	1B56 B02	1A21 C55	1B47 B37	1A08 C41	1B09 B50	1C20 C07	1C15 ***
K	1C57 ***	1B60 B19	1A38 C33	1B57 B39	1A05 C48	1B01 B12	1A12 C60	1B30 B04	1A07 C56	1B42 B08	1A13 C50	1B26 B40	1A36 C28	1B10 B20	1C36 ***
L	1C41 ***	1C10 C18	1B51 B41	1A15 A06	1B05 B61	1A43 C47	1B27 B10	1A25 C57	1B02 B09	1A03 C43	1B04 B57	1A56 A03	1B21 B32	1C11 C08	1C44 ***
M		1C51 ***	1A37 C17	1B15 B63	1A52 C32	1B03 B64	1A55 C46	1B46 B43	1A27 C44	1B55 B65	1A10 C29	1B14 B66	1A28 C09	1C55 ***	
N		1C01 ***	1C46 B29	1A29 C16	1B39 B53	1A53 A05	1B37 B45	1A45 C45	1B41 B28	1A20 A04	1B12 B68	1A39 C10	1C49 B47	1C08 ***	
O			1C32 ***	1C47 B48	1A46 C05	1B22 B49	1A47 C31	1B23 B38	1A01 C30	1B49 B51	1A49 C11	1C45 B52	1C18 ***		
P				1C25 032	1C53 ***	1C16 C02	1B29 B14	1C23 C13	1B61 B13	1C17 C14	1C28 ***	1C09 020			
						1C42 ***	1C40 ***	1C30 ***	1C60 ***	1C03 ***					

FIGURE -2

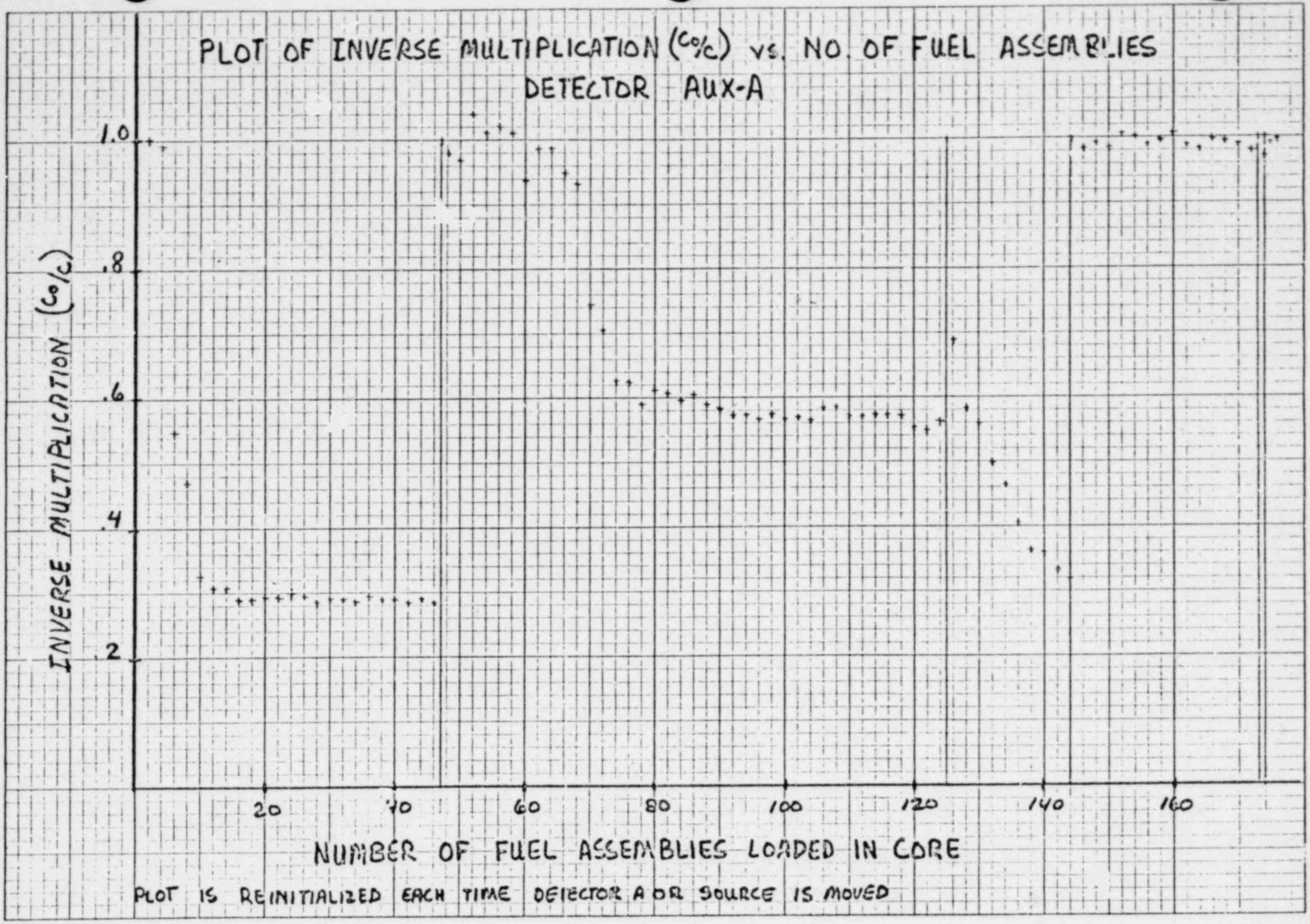

PLOT OF INVERSE MULTIPLICATION (Co/C) vs. NO. OF FUEL ASSEMBLIES
DETECTOR AUX-A
INVERSE MULTIPLICATION (Co/C)
1.0
.8
.6
.4
.2
20
40
60
80
100
120
140
160
NUMBER OF FUEL ASSEMBLIES LOADED IN CORE
PLOT IS REINITIALIZED EACH TIME DETECTOR A OR SOURCE IS MOVED

FIGURE 3

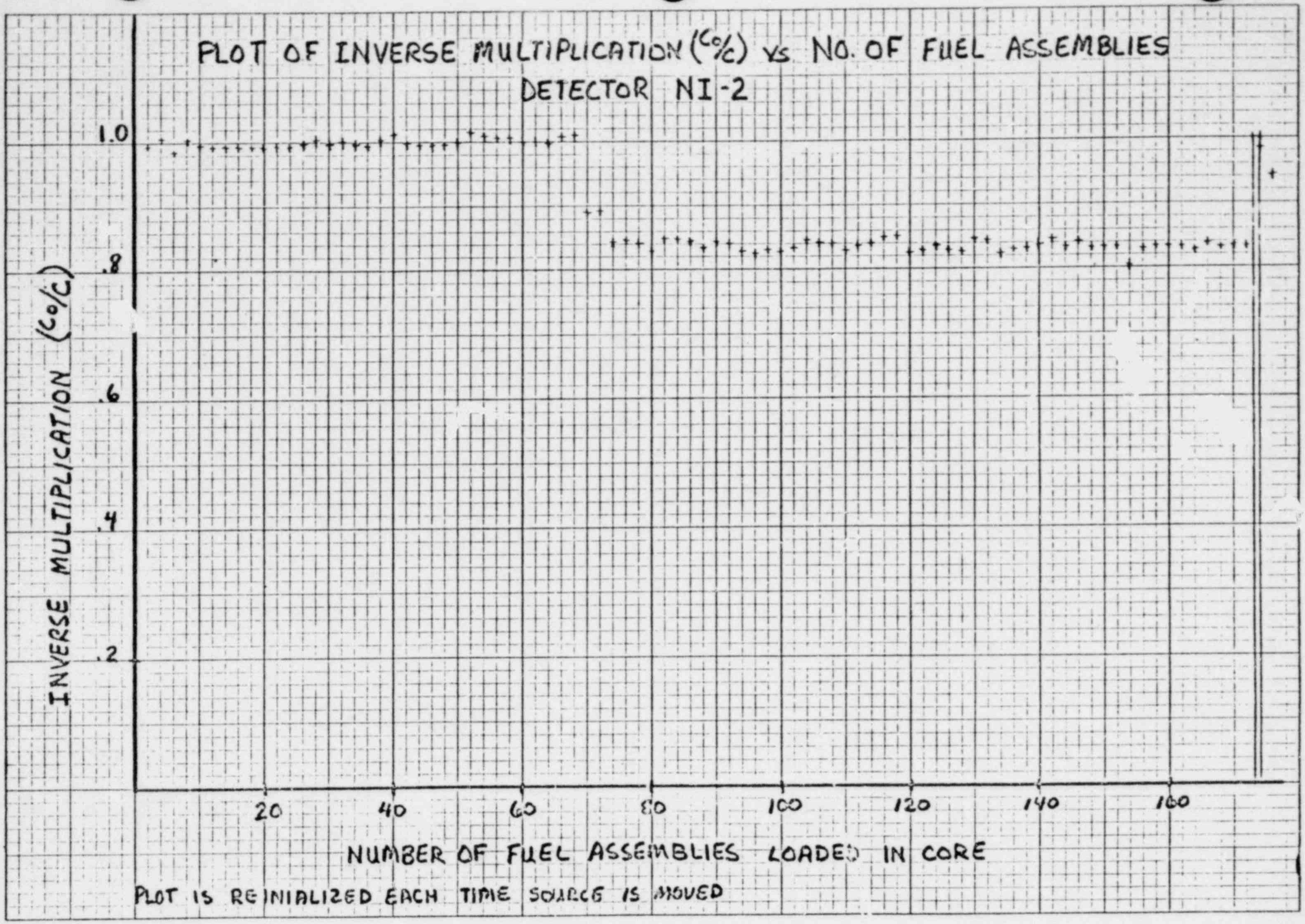
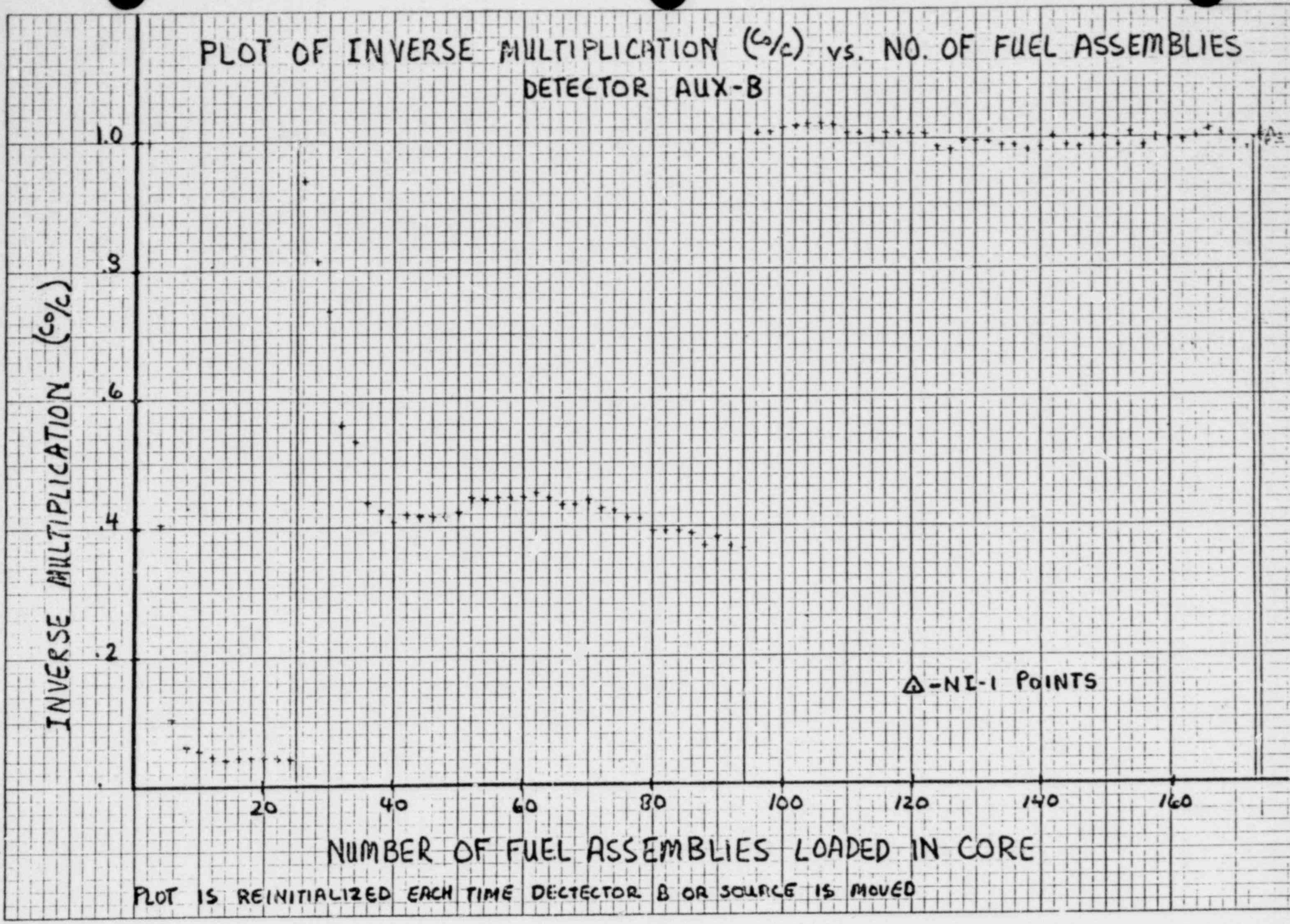

PLOT OF INVERSE MULTIPLICATION (C_0/C) vs NO. OF FUEL ASSEMBLIES
DETECTOR NI-2
1.0
.8
.6
.4
.2
INVERSE MULTIPLICATION (C_0/C)
20
40
60
80
100
120
140
160
NUMBER OF FUEL ASSEMBLIES LOADED IN CORE
PLOT IS REINIALIZED EACH TIME SOURCE IS MOVED

FIGURE 4


PLOT OF INVERSE MULTIPLICATION (Co/c) vs. NO. OF FUEL ASSEMBLIES
DETECTOR AUX-B
1.0
.8
.6
.4
.2
INVERSE MULTIPLICATION (Co/c)
20
40
60
80
100
120
140
160
NUMBER OF FUEL ASSEMBLIES LOADED IN CORE
Δ-NI-1 POINTS
PLOT IS REINITIALIZED EACH TIME DECTECTOR B OR SOURCE IS MOVED

2.6 CONTROL ROD DRIVE DROP TIME TEST

2.6.1 PURPOSE

The purpose of the Control Rod Drive Drop Time Test was to verify the integrated, functional trip capability of the Control Rod Drive System and to determine for each control rod assembly, the total elapsed drop time from the initiation of a trip signal until the control rod assembly was three-fourths inserted.

2.6.2 TEST METHOD

This test was conducted at various combinations of reactor coolant flow, pressure and temperature as follows:

Test Condition	Flow	Pressure	Temperature
1	No Flow	$\geq$350 psig	$\geq$400° F
2	One Pump Each Loop	$\geq$350 psig	Test 1 ±10°F
3	Four Pumps	2155 ± 30 psig	532 ± 10° F

At each condition, control Rod Groups 1 through 7 were driven, sequentially, to the fully withdrawn position. A manual trip of all control rod drives was then initiated and, coincidentally, a time signal was provided to the data logging equipment. As each control rod assembly reached the three-fourths insertion position, a second time signal was provided to the data logging devices from a switch located on each control rod drive's position indicator tube. The total elapsed time from the initiation of a trip signal until three-fourths insertion was then determined for each control rod drive from the data acquired.

The total drop time for each rod by core location in milliseconds was recorded along with the date, time, number of reactor coolant pumps operating, reactor coolant flow, reactor coolant temperature, and reactor coolant pressure.

2.6.3 RESULTS AND EVALUATION

An analysis of the drop times indicated that at Test Condition 1 rod H-4 was fastest at 1.012 seconds and rod D-12 was the slowest at 1.056 seconds. For Test Condition 2, rods H-4 and K-7 were fastest at 1.078 seconds, and H-14 was slowest at 1.106 seconds. For Test Condition 3, rod L-2 was fastest at 1.139 seconds, and L-8 was slowest at 1.177 seconds.

Rod L-8 (the slowest) and L-2 (the fastest) were dropped an additional 10 times and produced drop times within 23 ms and 14 ms, respectively.

2.6.4 CONCLUSIONS

The rod drop times were well below the acceptance criteria stated in Section 4.7 of the Technical Specifications, of 1.66 seconds at full flow and 1.40 seconds at no flow conditions. Also, as would be expected, all drop times under flow conditions were longer than under no flow conditions.

2.7 ZERO POWER PHYSICS TEST

2.7.1 PURPOSE

The purpose of the Zero Power Physics Test was to verify the nuclear design parameters used in the safety analysis, the Technical Specification limits, and the operational parameters. This test was conducted after initial fuel loading and before power escalation. Testing was performed at 532° F and 2155 psig. The test included the following measurements:

a) Initial criticality.
b) Nuclear instrumentation overlap between the source and intermediate range.
c) "All Rods Out" critical boron concentration.
d) Differential and integral control rod worth.
e) Differential boron worth.
f) Ejected control rod worth.
g) Stuck control rod worth.
h) Temperature coefficients as a function of boron concentration.

2.7.2 TEST METHOD

Initial criticality was achieved by control rod withdrawal and boron dilution of the Reactor Coolant System after system conditions had been established at 532°F and 2155 psig. During the initial approach to criticality a plot of inverse neutron count rate ratio versus boron concentration and time was maintained by using channels NI-1 and NI-2 of the nuclear instrumentation. After achieving criticality, nuclear power was increased and the source and intermediate range nuclear instrumentation overlap was verified to be in excess of one decade. During this same increase in power, the sensible heating point was determined and the upper power limit for Zero Power Physics testing was established.

Physics testing was then conducted including the following measurements:

a) "All Rods Out" critical boron concentration.
b) Temperature coefficient of reactivity at four boron concentrations.
c) Differential and integral rod worth of Control Rod Groups 8, 7, 6, 5 and part of Group 4 by the rod versus boron swap technique.
d) Integral rod worth of Control Rod Groups 1, 2, and 3 and part of Control Rod Group 4 by the rod drop technique.
e) Stuck rod worth by the rod drop technique.
f) Ejected rod worth by the rod swap and rod drop technique.
g) Differential boron worth.

The procedure for the approach to initial criticality from the initial condition of 1848 ppmB was as follows:

a) Control Rod Group withdrawal:

Group 1	100% Withdrawn
Group 2	100% Withdrawn
Group 3	100% Withdrawn
Group 4	100% Withdrawn
Group 8	100% Withdrawn
Group 5	100% Withdrawn
Group 6	100% Withdrawn
Group 7	75% Withdrawn

b) Deboration from 1848 ppmb to the critical boron concentration using a feed and bleed rate of approximately 20 gallons per minute. When criticality was imminent, Group 7 was withdrawn to 82% to establish a positive startup rate and verify criticality.

Throughout the approach to criticality, plots of the inverse neutron count rate versus Boron concentration and versus deboration time were maintained. These plots were used to project the Boron concentration and time at which criticality would be achieved.

2.7.3 RESULTS AND EVALUATION

2.7.3.1 Initial Criticality

Initial criticality was achieved August 6, 1974 at 0602 hours at reactor coolant conditions of 532°F and 2155 psig. Plots of inverse neutron count rate ratio versus time and versus boron concentration during the approach to initial criticality are presented in Figures 2.7 -1 through 2.7 -4. Initial criticality was achieved in a safe and orderly manner. Analyzed results indicate good agreement between predicted and measured criticality end points.

2.7.3.2 Nuclear Instrumentation Overlap

Technical Specifications require that prior to operation in the intermediate nuclear instrumentation range, at least a one decade overlap between the source range and intermediate range must be observed.

To satisfy the overlap requirements, after initial criticality was reached, core power was slowly increased until the intermediate range channels came on scale. Detector signal response was then recorded for both the intermediate and source range channels until the required decade of overlap was achieved. The results of the nuclear instrumentation overlap data is listed in table 2.7 -1.

2.7.3.3 Determination of Sensible Heat

The power level at which sensible heat is produced is important to the physics test program in that by restricting reactor power operation to below the sensible heat level the effects of temperature feedback are eliminated in the measurement of the physics parameters.

The determination of sensible heat from the core was accomplished by raising the neutron flux in small increments and observing neutron flux. RCS temperature, pressurizer level and turbine bypass valve position. When an increase in these parameters was indicated, the neutron flux level was designated as the level at which sensible neutron produced heat is present. the average intermediate range current reading was 9×10^{-8} amps. This level was reduced by a factor of 5 and established as the upper level for ZZPT.

2.7.3.4 "All Rods Out" Critical Boron Concentration

The "All Rods Out" critical boron concentration measurement was made with Control Rod Group 7 partially inserted. The measured boron concentrations were adjusted to the "All Rods Out" condition using the results of rod worth measurements to determine the reactivity worth, in terms of boron concentration, of the inserted control rods. The measured Boron concentration for all rods out was 1602 ppmB which compared favorably with the predicted value of 1634 ppmB and is within the acceptance criterion of ±100 ppmB.

2.7.3.5 Control Rod Group Worths

The configuration of control rod groups in the Reactor Core is shown in Figure 2.7 -5.

Calculated and measured beginning of life (BOL) control rod group reactivity worths for the normal calculations were made using the PDQ-7 computer code with either a two or three-dimensional description of the core. The rod/boron swap method was used to determine integral and differential worth for Rod Groups 8, 7, 6, 5 and part of 4. This method consisted of establishing a boration or deboration rate and compensating for the change in reactivity by small step changes in rod group positions.

The rod drop method was used to determine the worth of control rod groups not measured by boron swap. For each measurement, the reactor was adjusted to criticality with all of the control rod groups to be measured out of the core and at a power level near the Zero Power Physics test upper power limit. The control rod groups being measured were then tripped.

Besed on previous experience of Oconee Units 1 and 2 and TMI Unit 1, it was predicted that rod drop measurements would yield values approximately 74 percent of the correct value when considerably more the 1% ΔK/K was being inserted. The results were consistent with these expectations. Table 2.7 -2 compares the calculated and the corrected measured results (1.35 correction factor applied) for the rod drop from Group 5 at 23% withdrawn. The results show that the corrected measured value compares favorably with the predicted value.

The results of both the predicted control rod group worths and the measured control rod group worths are tabulated in Table 2.7-3. Control Rod Groups 1 through 4 were tabulated as one group, since the groups were measured as one unit. The total worth of Control Rod Groups 1 through 4 were tabulated as one group, since the groups were measured by taking the data of the control rod drops from 24 and 23 percent withdrawn on Control Rod Group 5 and from 40 percent on Control Rod Group 4. The data from the boron swap method was then used to calculate the total worth of Control Rod Groups 1 through 4. The tabulated number is the average of the three measurements. The results are within the acceptance criteria for rod worth measurements.

The results of the measured differential control rod groups worths are plotted in Figure 2.7-6 for Control Rod Group 8 position at 23.5%.

The shape of the integral curve for Control Groups 5 through 7 is shown in Figure 2.7-7. APSR's are 23.5% withdrawn.

2.7.3.6 Soluble Poison Worths

Measurements of the soluble poison differential worths were made at two different boron concentrations at 532°F. The measured values were determined by summing the incremental reactivity values measured during the rod worth measurements over a known boron concentration range.

Measured differential soluble poison worths are compared with predicted results in Table 2.7-4 and Figure 2.7-8. The measured values were within 3.5 percent of the calculated worths.

2.7.3.7 Ejected Control Rod Worths

Pseudo ejected control rod reactivity worth was measured for rod L-14. The purpose of this measurement was to verify the safety analysis calculations relating to the assumed accidental ejection of the most reactive control rod during power operation. The acceptance criterion for this measurement was that the reactivity worth of the most reactive control rod does not exceed .9% ΔK/K at 532°F, 2155 psig, zero power conditions. The criteria was reduced from the 1.0% ΔK/K predicted value to .9% ΔK/K in the conservative direction to allow for measurement errors.

The ejected rod worth was measured using rod swap and rod drop techniques. The calculated and measured ejected rod worths are tabulated in Table 2.7 -5. The maximum ejected control rod worth was determined to be .664% ΔK/K.

2.7.3.8 Stuck Control Rod Worth

The purpose of the stuck rod worth measurements at zero power were to verify that the calculated stuck rod worths are conservative compared to the measured results. The method used in measuring the simulated stuck rod worths involves taking the difference between a rod drop during which all rods (except Grp. 8) that are withdrawn are dropped from criticality and another rod drop in which all the above rods except the stuck rod are dropped.

The results of both predicted and measured stuck rod worths are given in Table 2.7-6. The low value of the eight measurements was discarded. The average of the seven remaining measurements is 3.47% ΔK/K. The difference between the measured and predicted worth is attributed to the use of the rod drop measurement method. This does not, however, affect unit safety, since the maximum predicted stuck rod worth of 3.91% ΔK/K is used in determining available shutdown margin.

2.7.3.9 Temperature Coefficient of Reactivity

The temperature coefficient of reactivity is defined as the fractional change in the reactivity of the core per unit change in core temperature. Temperature coefficients were measured for various soluble poison concentrations.

The measurements were made by initially decreasing the reactor coolant temperature by 5°F and then increasing the temperature by approximately 10°F. The temperature was then decreased to 532°F. The change in the position of the inserted control rod group was recorded and converted to a change in reactivity.

The results of both the predicted and measured temperature coefficients are plotted in Figure 2.7-9. This curve also contains the measured and predicted moderator coefficient of reactivity which is defined as the fractional change in the reactivity of the core per unit change in moderator temperature.

All measured temperature coefficients of reactivity were within the acceptance criteria of $\pm 0.4 \times 10^{-4}$ $\Delta K/K/°F$ of the predicted value. In addition, calculation of the moderator coefficient indicates that it is well within the requirements of less than $+.5 \times 10^{-4}$ $\Delta K/K/°F$ per Technical Specifications.

2.7.4 CONCLUSIONS

Zero Power Physics Testing commenced on August 6, 1975 and was completed on August 8, 1974, with good agreement between measured and predicted results. A summary of each of the measurements performed during the Zero Power Physics Test is given below.

(a) Initial Criticality

Initial Criticality was achieved at 0602 hours on August 6, 1974. The approach to criticality was performed in a safe and orderly manner.

(b) Nuclear Instrumentation Overlap

Nuclear instrumentation overlap was verified to be in excess of one decade between the source and intermediate range. The minimum acceptable overlap is one decade.

(c) "All Rods Out" Boron Concentration

The measured "All Rods Out" boron concentration was 1602 ppmB at 532°F as compared to the predicted value of 1634 ppmB.

(d) Control Rod Group Worths

Control rod group integral and differential reactivity worths were calculated by rod/boron swap and rod drop measurements. The measurement of rod worth by rod/boron swap indicated good agreement between the measured and calculated group worths. Good correlation was obtained for rod drop measurements when compared with expected measured worths.

(e) Soluble Poison Worths

Measured differential boron reactivity worths of 1.024% $\Delta K/K$/100ppm at 1492 ppm and 1.036% $\Delta K/K$/100 ppm at 1354 ppm were determined. Comparison of these values to the predicted values showed that the measured values were within 3.5 percent of calculated worths.

(f) Ejected Control Rod Worths

The maximum ejected control rod worth was determined to be 0.664% $\Delta K/K$, which ensured that Technical Specification 3.5.2 will be met at zero power.

(g) Stuck Control Rod Worth

The measured stuck rod worth of 3.47% $\Delta K/K$ was less than the calculated value due to the rod drop measurement method utilized. This does not, however, affect unit safety, since the maximum calculated stuck rod worth of 3.91% $\Delta K/K$ is used in determining available shutdown margin.

(h) Temperature Coefficient of Reactivity

Measured temperature coefficient of reactivity at 532°F was within the acceptance criteria of $\pm$ 0.4×10^{-4} $\Delta K/K/°F$ of the predicted value. In addition, calculation of the oderator coefficient indicates that it is well within the requirements of Technical Specification 3.1.7.

NUCLEAR INSTRUMENTATION RANGE OVERLAP DATA

FIGURE 2.7-

TIME 24 HOUR CLOCK	RCS T-AVE. DEG-F	SOURCE RANGE				INTERMEDIATE RANGE			
		NI-1 CPS		NI-2 CPS		NI-3 AMPS		NI-4 AMPS	
		CONSOLE	CABINET	CONSOLE	CABINET	CONSOLE	CABINET	CONSOLE	CABINET
0807	532	$4.5x10^3$	$3.5x10^3$	$5x10^3$	$4x10^3$	off scale	off scale	off scale	off scale
0820	532	$1.0x10^4$	$9x10^3$	$1.1x10^4$	$1.2x10^4$	$1.5x10^{-11}$	$1.9x10^{-11}$	off scale	off scale
0838	532	$1.8x10^4$	$1.7x10^4$	$2x10^4$	$1.9x10^4$	$2.5x10^{-11}$	$3x10^{-11}$	$2.5x10^{-11}$	$3.5x10^{-11}$
0852	532	$1.8x10^5$	$1.7x10^5$	$2x10^5$	$2x10^5$	$3x10^{-10}$	$3x10^{-10}$	$3x10^{-10}$	$3x10^{-10}$

SOURCE RANGE HIGH VOLTAGE CUTOFF CURRENT (APPROX. $1x10^{-9}$ AMPS)	$1x10^{-9}$	$1x10^{-9}$
SOURCE RANGE HIGH VOLTAGE CUT-ON CURRENT (APPROX. $5x10^{-10}$ AMPS)	$5x10^{-10}$	$5x10^{-10}$

TABLE 2

COMPARISON OF PREDICTED AND CORRECTED MEASURED CONTROL ROD GROUP REACTIVITY WORTHS

Moderator Temperature at 532°F, APSR's at 23.5% wd (ROD DROP)

Rod Group Number	Position Interval % wd	Predicted Worth, % Δk/K		Corrected Measured Worth, % Δk/k	Deviation from Calculated Worth
1	0 → 100	.89			
2	0 → 100	3.01			
3	0 → 100	.74	6.70	6.48	-3.32%
4	0 → 100	1.86			
5	0 → 23	.20			

TABLE 2.7-3

COMPARISON OF PREDICTED AND MEASURED CONTROL ROD GROUP REACTIVITY WORTH

Moderator Temperature at 532F, APSR's at 23.5% wd

Group	No. Rods	Predicted Worth, %ΔK/K	Measured Worth, %ΔK/K
1	8		
2	8	-6.50	-6.15
3	8		
4	8		
5	12	-1.07	-1.09
6	8	-1.22	-1.14
7	9	-1.20	-1.05
8	8	-0.38	-0.37
Total	69	-10.37	-9.80

TABLE 2.7-4

DIFFERENTIAL BORON REACTIVITY WORTH AT 532°F MODERATOR TEMPERATURES

Temp F	Critical Conditions Rod Position	Boron Conc. ppm	Avg. Boron Concentration ppm	Δ Boron Concentration ppm	Δρ % Δk/k	Measured Differential Boron Worth % Δk/k/100 ppm	Predicted Differential Boron Worth % Δk/k/100 ppm
532	CRG 7 76% wd CRG 8 23.5% wd	1549					
532	CRG 6 71.5% wd CRG 8 23.5% wd	1435	1492	114	1.167	1.024	1.06
532	CRG 6 76.0% wd CRG 8 23.5% wd	1444					
532	CRG 5 19.5% wd CRG 8 23.5% wd	1261	1354	183	1.895	1.036	1.07

TABLE 2.7-5

EJECTED CONTROL ROD REACTIVITY AT ZERO POWER, 532°F

Rod Ejected	Predicted Ejected Rod Worth	Control Rod Group Positions	Measured Ejected Rod Worth
Transient Rod L-14 (7-4)	1.00% ΔK/K	CRG 8 23.5% wd CRG 4 at 47% wd	.664% ΔK/K

TABLE 2.7 -6

STUCK CONTROL ROD REACTIVITY AT ZERO POWER, 532°F

STUCK ROD	Predicted Stuck Rod Worth %ΔK/K	Control Rod Group Positions	Measured Stuck Rod Worth %ΔK/K
Rod H-2 (4-7)	3.91	CRG 8 23.5% wd CRG 7 72.0% wd	4.121
Rod H-14 (4-3)	3.91	CRG 8 25.0% wd CRG 7 76.0% wd	3.102
Rod H-2 (4-7)	3.91	CRG 8 24.5% wd CRG 6 73% wd	3.662
Rod H-14 (4-3)	3.91	CRG 8 23% wd CRG 6 74% wd	2.861
Rod H-2 (4.7)	3.91	CRG 8 24% wd CRG 5 23% wd	3.04
Rod H-14 (4-3)	3.91	CRG 8 24% wd CRG 5 24% wd	2.64
Rod H-2 (4-7)	3.91	CRG 8 23.5% wd CRG 3 61% wd	3.74
Rod H-14 (4.3)	3.91	CRG 8 23.5% wd CRG 3 62.0% wd	3.74
Average	3.91	Average *	3.47

* Lowest value (2.64) discarded for extra conservatism.

FIGURE -1

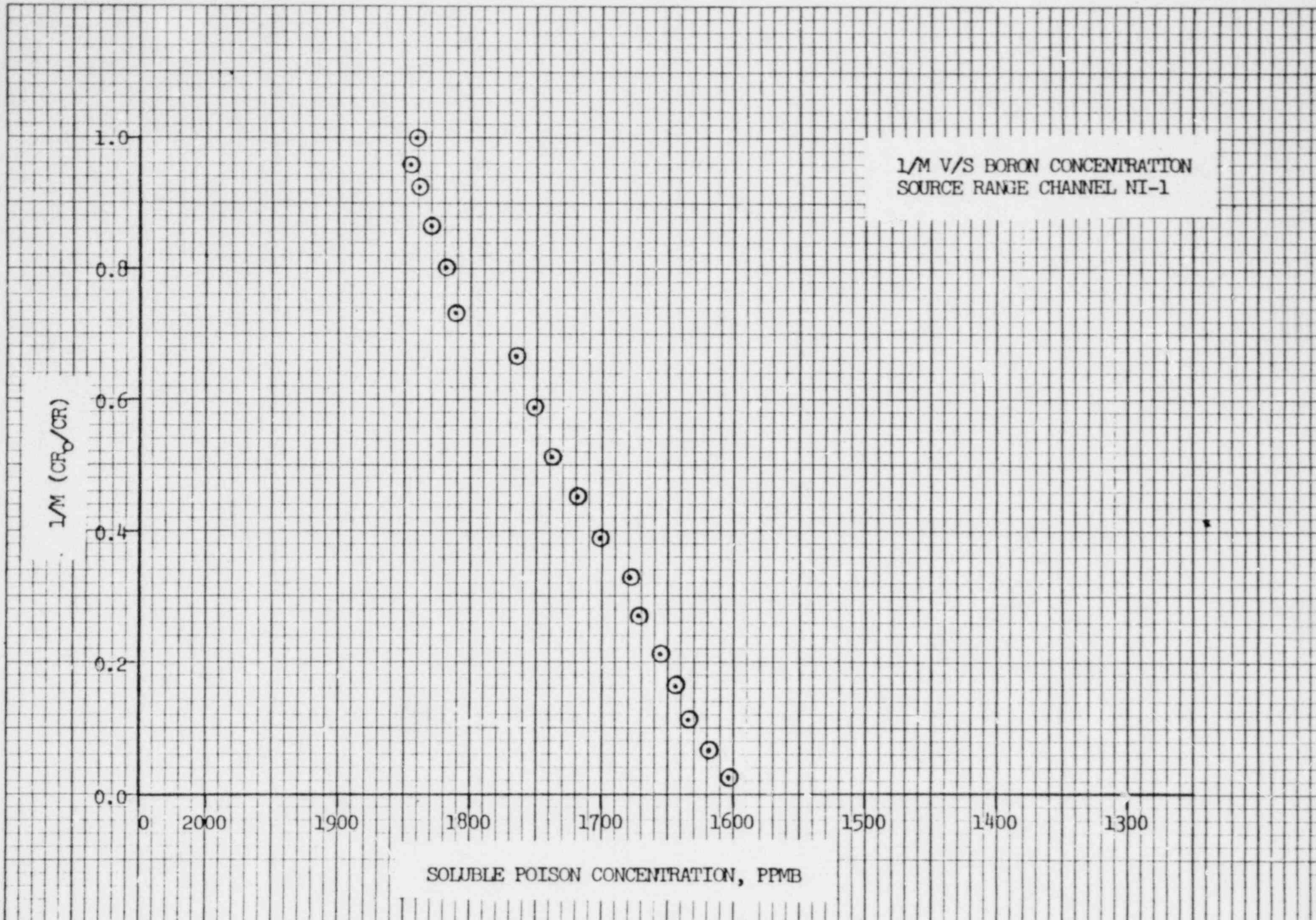
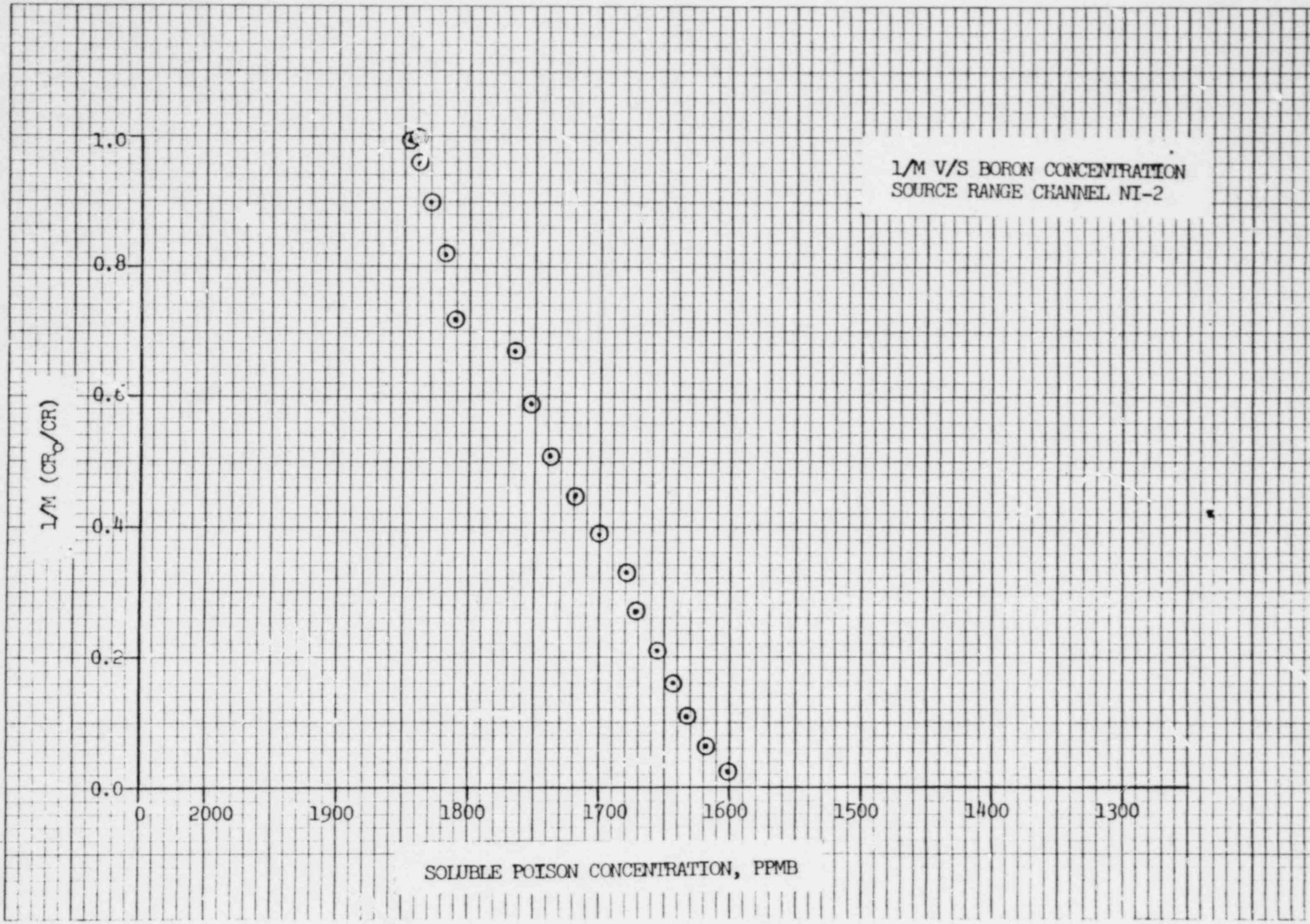
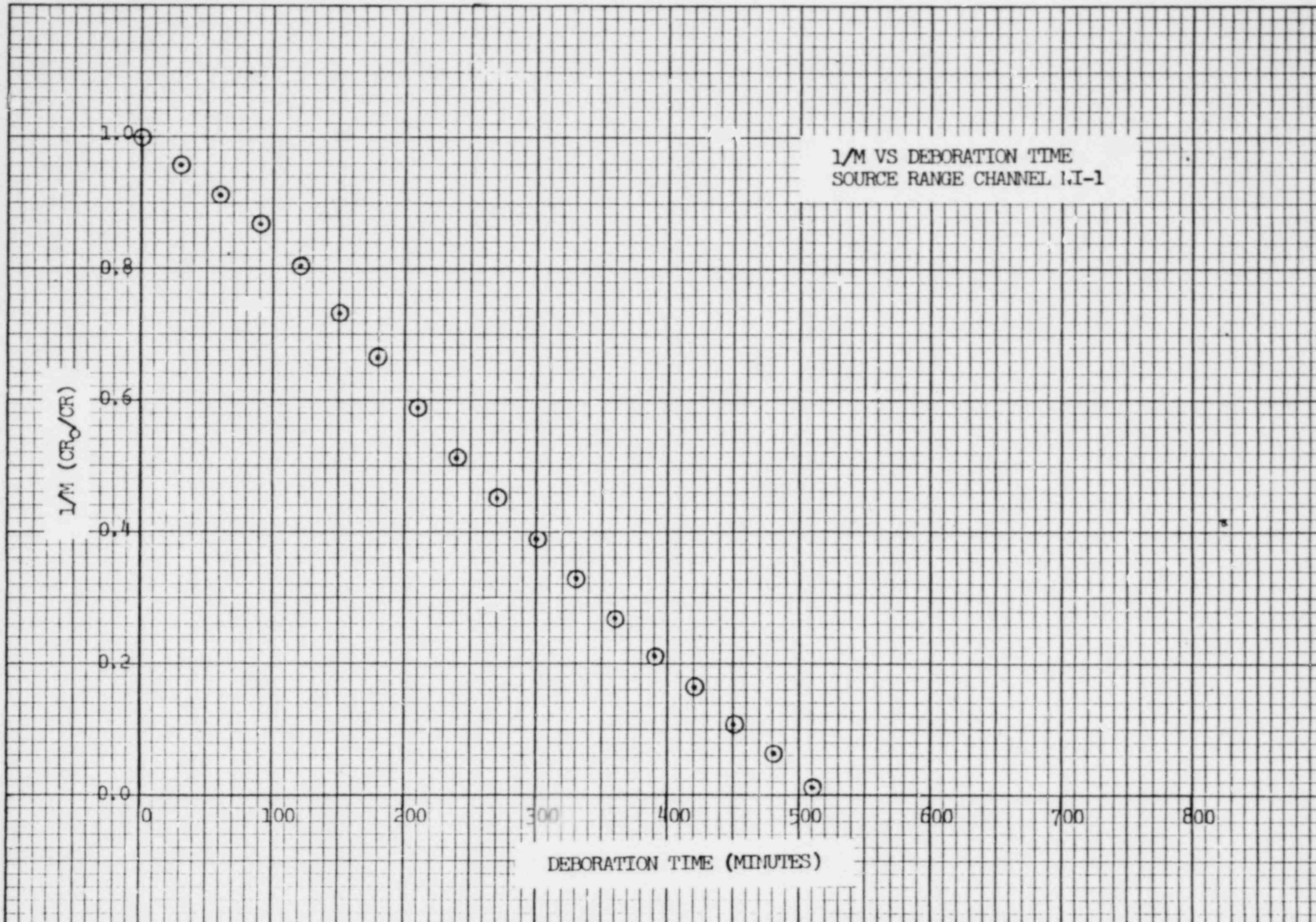

1/M V/S BORON CONCENTRATION
SOURCE RANGE CHANNEL NI-1
1/M (CR0/CR)
1.0
0.8
0.6
0.4
0.2
0.0
0
2000
1900
1800
1700
1600
1500
1400
1300
SOLUBLE POISON CONCENTRATION, PPMB

FIGURE 2.7



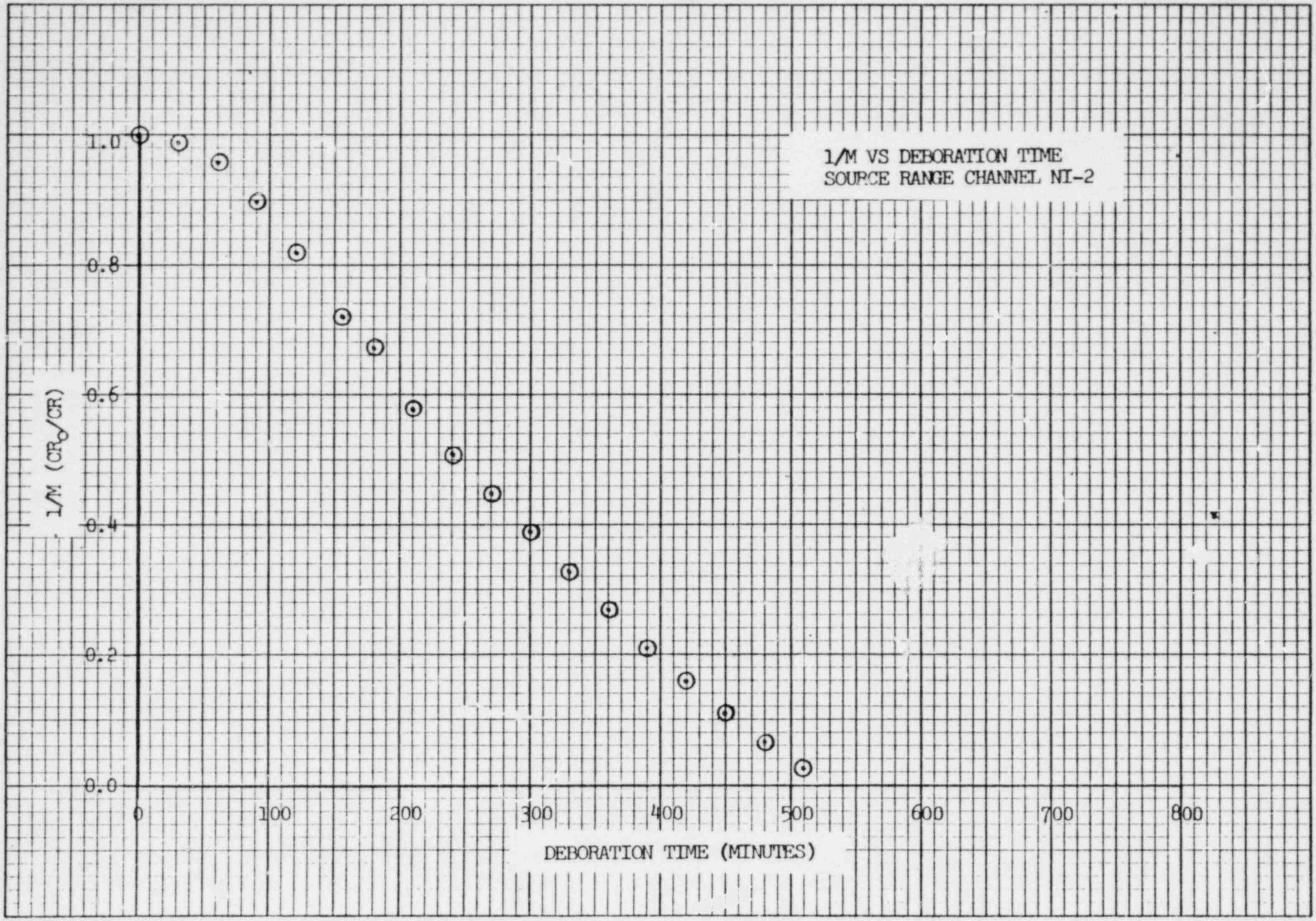
1/M V/S BORON CONCENTRATION
SOURCE RANGE CHANNEL NI-2
1/M (CR_o/CR)
1.0
0.8
0.6
0.4
0.2
0.0
0
2000
1900
1800
1700
1600
1500
1400
1300
SOLUBLE POISON CONCENTRATION, PPMB

FIGURE [illegible]7-3



1/M VS DEBORATION TIME
SOURCE RANGE CHANNEL NI-1
1/M (CR_o/CR)
1.0
0.8
0.6
0.4
0.2
0.0
0
100
200
300
400
500
600
700
800
DEBORATION TIME (MINUTES)

FIGURE 2.



1/M VS DEBORATION TIME
SOURCE RANGE CHANNEL NI-2
1/M (CR_0/CR)
1.0
0.8
0.6
0.4
0.2
0.0
0
100
200
300
400
500
600
700
800
DEBORATION TIME (MINUTES)

FIGURE 2.7-5

CONTROL ROD GROUP LOCATIONS

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A															
B						(7)		(4)		(7)					
C					(5)		(3)		(3)		(5)				
D				(4)		(8)		(6)		(8)		(4)			
E			(5)		(6)		(1)		(1)		(6)		(5)		
F		(7)		(8)		(2)		(2)		(2)		(8)		(7)	
G			(3)		(1)		(5)		(5)		(1)		(3)		
H		(4)		(6)		(2)		(7)		(2)		(6)		(4)	
K			(3)		(1)		(5)		(5)		(1)		(3)		
L		(7)		(8)		(2)		(2)		(2)		(8)		(7)	
M			(5)		(6)		(1)		(1)		(6)		(5)		
N				(4)		(8)		(6)		(8)		(4)			
O					(5)		(3)		(3)		(5)				
P						(7)		(4)		(7)					
R															

(X) ← Control Rod Group Number

FIGURE 2.7-6

Differential Rod Worth - vs -
Position for Zero Power Conditions
and APSR's at 23.5% WD (Groups 5-7)

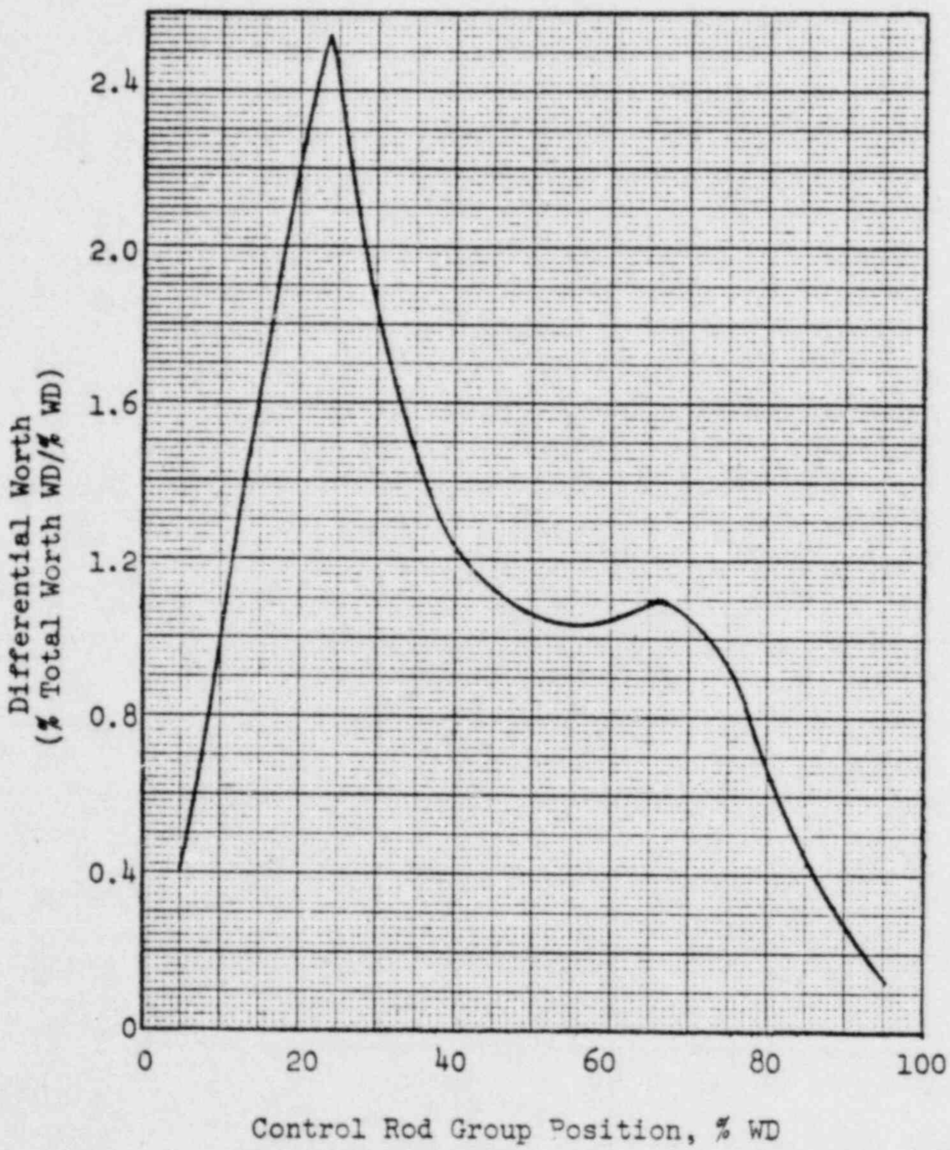
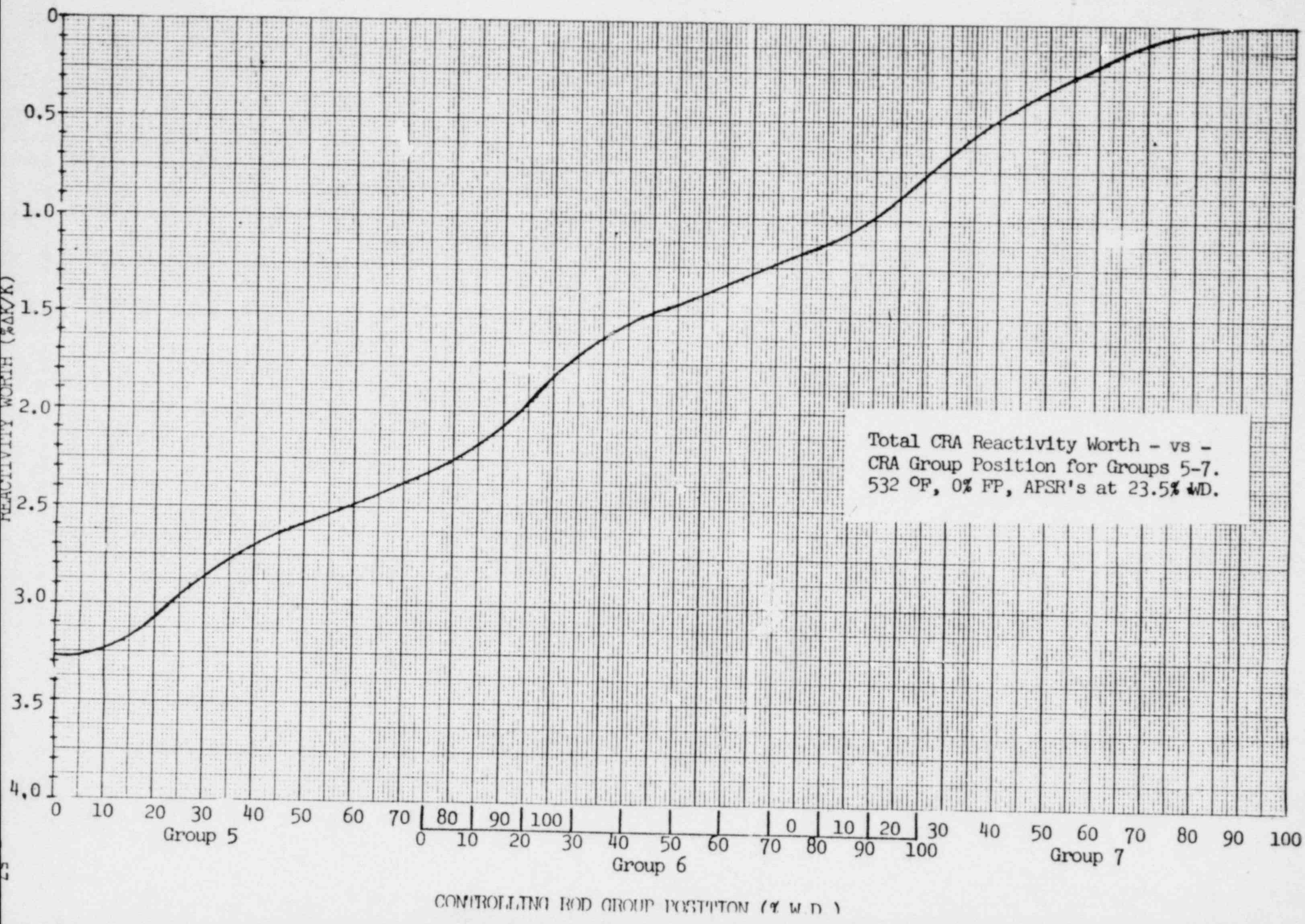


FIGURE 2.7-7



Total CRA Reactivity Worth - vs -
CRA Group Position for Groups 5-7.
532 °F, 0% FP, APSR's at 23.5% WD.
REACTIVITY WORTH (%ΔK/K)
0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0
0 10 20 30 40 50 60 70 80 90 100
Group 5
0 10 20 30 40 50 60 70 80 90 100
Group 6
0 10 20 30 40 50 60 70 80 90 100
Group 7
CONTROLLING ROD GROUP POSITION (% W.D.)

FIGURE 2.7-8

Differential Reactivity Worth of Soluble Boron Vs

Boron Concentration for Moderator Temperature of 532F

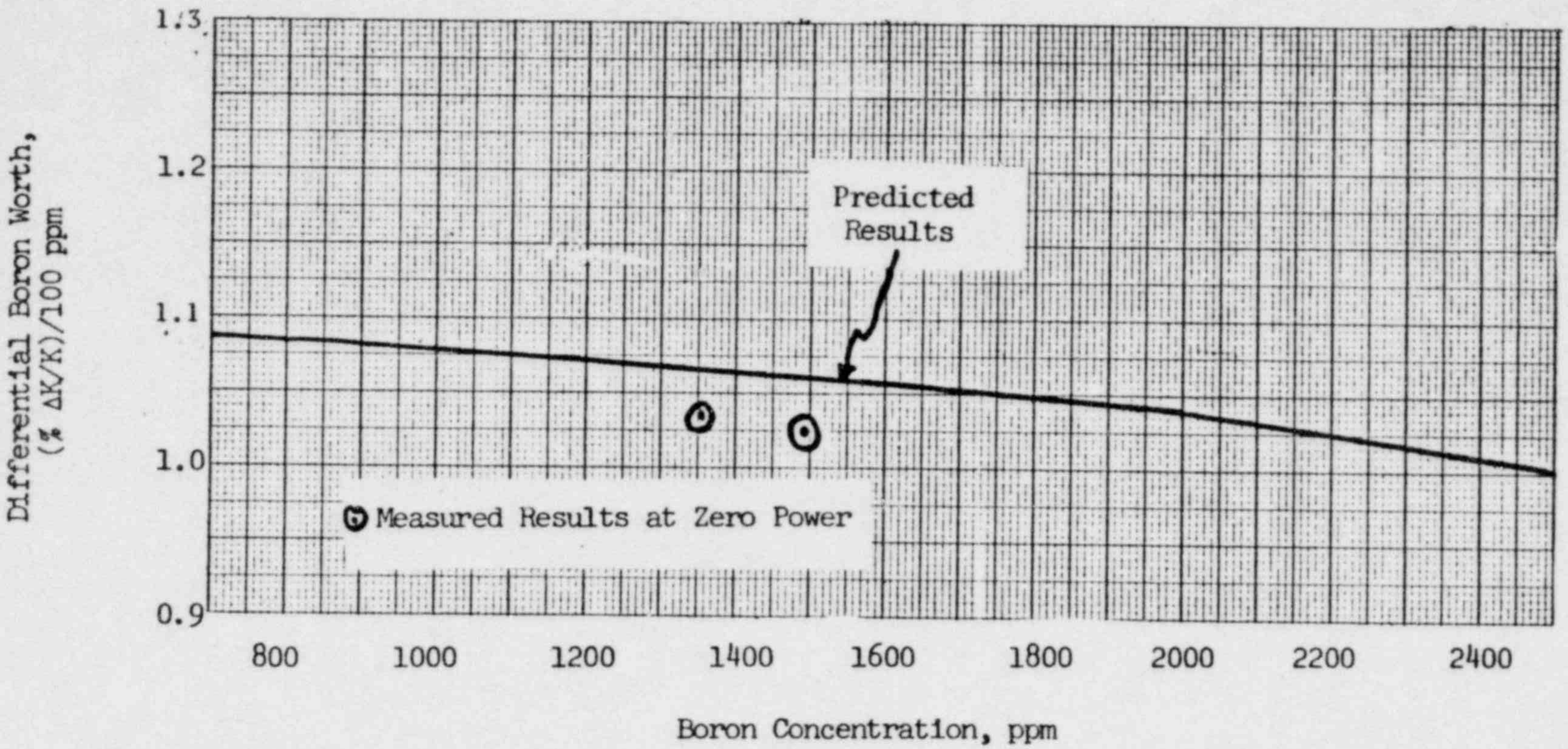
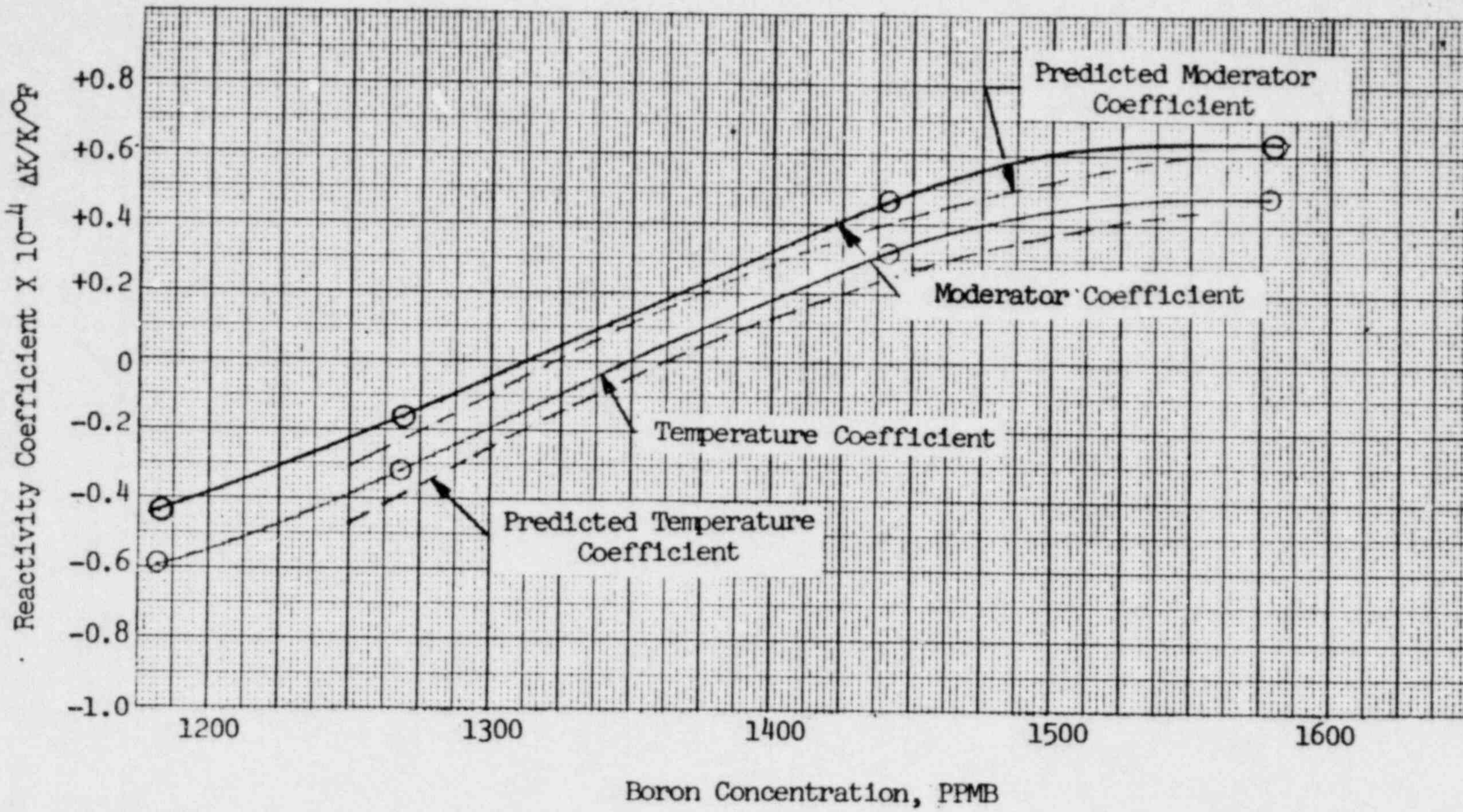


FIGURE 2.7-9

Temperature Coefficients of Reactivity Vs Boron Concentration at 532 °F, 2155 psi, 0 EFPD



2.8 LOSS OF OFF-SITE POWER TEST

2.8.1 PURPOSE

The purpose of this test was to demonstrate that the unit could safely sustain a loss of off-site power without exceeding any plant safety limits while maintaining the main turbine generator on line supplying auxiliary house load.

2.8.2 TEST METHOD

The test was performed with the main turbine generator on line at a steady state power level of 15% full power and with main feedwater pump, turbine control system, turbine bypass valves, and diamond control rod drive system control stations in manual. The unit was then isolated from all off-site power by isolating the startup transformers from the plant and then separating the unit from the grid.

2.8.3 RESULTS AND EVALUATION

When the circuit breakers were opened the turbine header pressure increased rapidly from approximately 887 to 932 psig. The rapid increase in header pressure caused reactor temperature to increase about 7°F. With a Boron concentration of approximately 1284 ppm and moderator coefficient of $+0.18 \times 10^{-4}$ $(\Delta K/K)/°F$ the increase in reactor temperature caused reactor power to increase by about 3% full power. This, in turn, caused a further increase of average reactor temperature to 594°F. The pressurizer level increased about 97 in. to 282 in. which was 8 in. below the administrative upper limit of 290 in. This pressurizer level increase resulted in an increase of reactor pressure to approximately 2265 psig. The transient was manually terminated by the operator by decreasing power to approximately 14% full power to decrease steam flow and allow the turbine bypass valves to control header pressure.

2.8.4 CONCLUSIONS

The startup transformers were not required to carry plant load during this test and no safety limits were exceeded. Therefore; the test demonstrated satisfactorily that the unit could sustain a loss of offsite power as required.

2.9 REMOTE SHUTDOWN TEST

2.9.1 PURPOSE

The purpose of this test was to demonstrate that the unit could be brought safely to hot shutdown conditions from outside the control room.

2.9.2 TEST METHOD

The test was initiated at the 15% power level with normal system lineups for that power level by opening the CRD breakers manually. Operations personnel were dispatched to remote shutdown stations to monitor and perform required operations.

2.9.3 RESULTS AND EVALUATION

It was anticipated that if all plant systems functioned automatically, steam generator levels would stabilize at approximately 30 inches, pressurizer level at approximately 180 inches, reactor coolant pressure at approximately 2155 psig, and reactor coolant temperature at approximately 555°F. During the conduct of the test, the reactor coolant temperature and pressure, pressurizer level, and steam header pressure dropped more than anticipated. This was caused by an atmospheric dump valve which had stuck open. Additional makeup flow was necessary to maintain pressurizer level and reactor coolant pressure, which dropped to minimum values of approximately 22 inches and 1850 psig respectively. Reactor coolant temperature was stabilized at 526°F. The atmospheric dump valve was isolated, control of the pressurizer heaters was regained, and system parameters began to return to normal for hot shutdown. During the plant cooldown, after completion of this test, it was discovered that a code safety valve had stuck open on the other steam header which further contributed to the unexpected cooldown rate.

2.9.4 CONCLUSION

The test was performed according to procedure, crew actions were observed to be correct and well coordinated, and plant response would have been essentially as predicted had the steam relief valves not stuck open.

2.10 NUCLEAR INSTRUMENTATION CALIBRATION AT POWER

2.10.1 PURPOSE

The purpose of Nuclear Instrumentation Calibration at Power was to calibrate the power range nuclear instrumentation indication to within ±2%FP of the reactor thermal power as determined by a heat balance and to within ±5 percent incore axial offset as determined by the incore monitoring system. Additional purposes during the power escalation program were as follows:

(a) To adjust the high power level trip setpoint when required by the power escalation procedure.

(b) To verify that at least one decade overlap existed between the intermediate and power range nuclear instrumentation.

Two acceptance criteria are specified for nuclear instrumentation calibration at power as listed below.

(1) The power range nuclear instrumentation indicates the power level within ±2%FP of the power level indicated by heat balance and within ±5 percent incore axial offset as determined by the incore detectors.

(2) The high power level trip bistable is set to trip at the desired value within +0.0, -0.3%FP.

2.10.2 TEST METHOD

As required during power escalation, the top and bottom linear amplifier gains were adjusted in order that the power range nuclear instrumentation channels would indicate the power calculated by heat balance, and the axial offset as determined by the incore monitoring system.

During initial adjustments, data were also taken to verify overlap between the intermediate and power range channels. The required overlap was a minimum of one decade between these two nuclear instrumentation ranges.

When directed by the power escalation procedure, the high flux trip bistable setpoint was adjusted. The major settings during power escalation are given below:

Test Plateau %FP	Bistable Setpoint %FP
15	$\leq$35
40	$\leq$50
75	$\leq$85
100	$\leq$105.5

2.10.3 RESULTS AND EVALUATION

An analysis of test results indicated that changes in Reactor Coolant System boron concentration, changes in control rod configuration, and xenon buildup or burnout affected the power as observed by the nuclear instrumentation. This was as expected since the power range nuclear instrumentation measures reactor neutron leakage which is directly related to the above changes in system conditions. Changes in these system conditions resulted in a nuclear power range increase or decrease of approximately 3 to 5% FP. Each time that it was necessary to calibrate the power range nuclear instrumentation, the acceptance criteria of calibration to within ± 2.0%FP of the heat balance power was met without difficulty. Also, each time it was necessary to calibrate the power range nuclear instrumentation, the axial power offset acceptance criteria as determined by the incore monitoring system was also met. Table 2.10-1 is a summary of the data taken during calibration at different power levels during power escalation testing. In all cases, the nuclear instrumentation was adjusted to within ± 2.0%FP of the heat balance and to within ± 5% incore axial power offset.

The high flux level trip bistable was adjusted to 35, 50, 85 and 105.5% FP prior to escalation of power to 15, 40, 75 and 100% FP, respectively. Acceptance criteria of adjusting the setpoint to the above values within +0.0, -0.3% FP was met each time without difficulty. The maximum trip error observed was approximately 0.1% FP.

The overlap measured during the startup program included the total span of the power range, exceeding the one-decade overlap requirement. Figure 2.10-1 shows the overlap of all three nuclear instrumentation channels.

2.10.4 CONCLUSIONS

The power range channels were calibrated to within two percent of total power several times during the startup program. These calibrations were required due to power level, boron, and/or control rod configuration changes during the program. Acceptance criteria for nuclear instrumentation calibration at power were met in all instances.

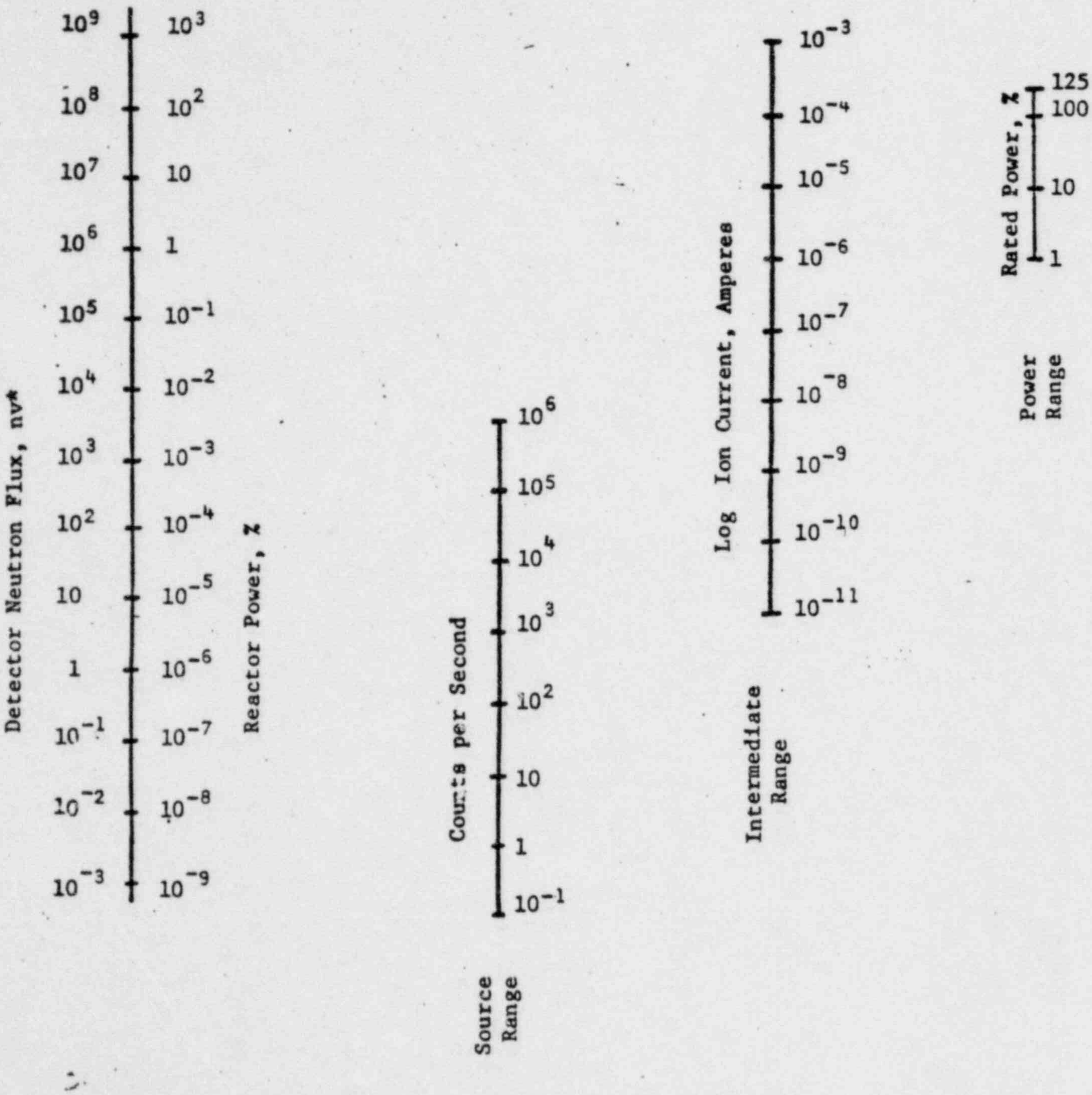
TABLE 2.10-1

Summary of Nuclear Instrumentation Calibrations at Power

Core Power (%FP)	Incore Offset (%)	Maximum Quad Tilt (%)	Power Before and After Calib.(%FP)				Offset Before and After Calib.(%)			
			NI-5	NI-6	NI-7	NI-8	NI-5	NI-6	NI-7	NI-8
12.6	- 2.5	-0.2	15.2 12.6	15.2 12.6	15.3 12.6	15.5 12.6	-22.7 - 2.4	-20.2 - 2.4	-21.1 - 2.4	-22.5 - 2.4
26.0	-10.9	-0.08	25.8 25.8	26.2 26.2	25.8 25.8	26.3 26.3	-10.6 -10.6	-11.2 -11.2	-10.4 -10.4	-10.2 -10.2
38.6	-14.0	-0.08	34.5 38.8	34.7 38.8	34.4 38.8	35.0 38.8	-19.3 -13.1	-20.4 -13.1	-18.2 -13.1	-18.4 -13.1
40.0	-13.0	+0.08	40.0 40.0	40.0 40.0	40.0 40.0	39.7 40.0	- 7.6 -12.0	- 7.3 -11.5	- 8.1 -11.5	- 7.9 -12.0
40.8	-15.4	+0.08	39.1 40.8	39.3 40.8	38.9 40.6	39.2 40.6	-11.4 -15.0	-10.8 -15.0	-11.1 -14.8	-11.5 -15.0
60.1	-10.5	±0.08	59.1 59.8	59.4 59.8	59.1 59.1	59.2 58.8	-17.5 -10.9	-18.0 - 9.7	-17.3 -11.8	-17.4 -12.0
77.0	-22.6	±0.08	74.2 77.3	74.0 76.6	73.4 76.7	72.6 76.8	-14.3 -21.2	-12.6 -21.6	-15.0 -21.1	-14.9 -22.1
84.0	+2.3	±0.04	83.4 84.1	83.8 84.0	83.5 84.0	83.7 84.1	- 2.8 1.8	- 2.9 1.9	- 2.8 1.8	- 2.6 1.9
95.3	- 0.7	±0.04	94.2 95.3	93.8 95.3	94.1 95.3	93.8 95.3	0.9 - 0.6	1.0 - 0.5	1.0 - 0.6	0.9 - 0.5
99.6	- 2.0	±0.12	98.5 98.6	97.8 98.8	98.2 99.2	98.1 99.7	- 2.1 - 3.0	- 1.9 - 3.1	- 2.0 - 2.6	- 1.8 - 3.1

FIGURE 2.10-1

NUCLEAR INSTRUMENTATION FLUX RANGES



* From Power Range
Detector Sensitivity

2.11 NUCLEAR STEAM SUPPLY SYSTEM HEAT BALANCE

2.11.1 PURPOSE

The Nuclear Steam Supply System Heat Balance was performed to satisfy two objectives. These objectives are to verify the plant computer calculation of core thermal power and to verify the Operating Procedure 1103.16, Heat Balance Calculation, calculational method.

2.11.2 TEST METHOD

Primary and secondary heat balances were performed at all power levels specified by the power escalation testing sequence. At each power level steady-state conditions were first established as follows:

(a)	Reactor Power as Constant as Possible	
(b)	RCS Pressure Constant	±50 psig
(c)	RCS Average Temperature Constant	± 2°F
(d)	Feedwater Temperature Constant	±10°F
(e)	Feedwater Flow Constant	±1.0%
(f)	Turbine Header Pressure Constant	±30 psig

Once these conditions were met all the data required were obtained as directed by the procedure. The data for the hand and off-line minicomputer calculations were raw values of the required temperatures, flows and pressures. These values were taken every minute for five minutes. Similarly the plant computer heat balance was determined by requesting printouts during the data acquisition period. Data was also obtained for calculation of an approximate indication of reactor power by logging various console temperatures and flows at three different times during the data acquisition period.

Upon completion of data acquisition, detailed primary and secondary heat balance calculations were performed both by hand and by an off-line minicomputer program. Also calculated by hand was the approximate indication of reactor power. The plant computer results were averaged before comparing to the hand calculated values.

2.11.3 RESULTS AND EVALUATION

The results from the various heat balance calculations from 15 through 100% Full Power are presented in Table 2.11-1. This information shows the percent of full power calculated by the plant computer, by hand calculation and by the minicomputer program. Comparison of the hand calculated values with the respective computer calculated values in all cases shows agreement within the acceptance criteria of $\pm 2\%$ FP. The acceptance criteria which applies to the agreement between the approximate indication of reactor power and the plant computer primary heat balance is a function of a weighted average of the hand calculated primary and secondary heat balances. This acceptance criteria between 15% FP and 100% FP varies linearly between 25 and 5 percent. The weighted average of the hand calculated primary and secondary heat balances were calculated as follows;

$$\overline{\%FP} = (\text{Sec.} + \text{Pri.})/2,$$

$$Y = 0.0117647\ (\overline{\%FP} - 15),$$

and

$$\text{weighted average} = (1-y)\text{Pri.} + Y\ \text{Sec.}$$

In all cases this acceptance criteria was met.

At low power levels the plant computer did not agree as closely with the hand calculation or the off-line minicomputer calculation as it did when the power level was increased. In cases where the primary and secondary heat balances differed, the more conservative value was used.

2.11.4 CONCLUSIONS

The plant computer calculation agreed with hand calculated values of primary and secondary power within ± 2% FP. Therefore, the plant computer calculation was shown to be an acceptable indication of reactor power and is now used for calibration of the nuclear instrumentation.

SUMMARY OF HEAT BALANCES PERFORMED DURING POWER ESCALATION

TABLE 2.11-1

Heat Balance Number	Date M-D-Y	Time Hr.:Min.	Core ΔT Power % FP	Primary Heat Balance % FP			Secondary Heat Balance % FP		
				Plant Computer	Nova 1200 Computer	Hand Calc.	Plant Computer	Nova 1200 Computer	Hand Calc.
7	8-20-74	03:23	14.9	16.0	14.5	14.6	16.7	15.1	15.1
10	9-8-74	12:47	26.7	27.2	26.9	27.2	29.6	29.5	29.5
13	9-24-74	11:24	43.0	39.0	38.2	38.1	41.1	41.7	41.7
15	10-15-74	12:25	58.4	56.6	57.9	58.0	62.3	62.4	62.4
16	10-23-74	20:11	73.8	72.1	72.1	72.3	77.1	76.9	76.9
17	12-5-74	05:29	82.6	79.1	78.5	78.6	84.3	84.0	84.0
18	12-5-74	16:41	92.8	90.4	90.7	90.7	95.3	95.3	95.3
19	12-9-74	01:37	96.7	94.7	96.6	96.6	100.1	100.1	100.0

2.12 BIOLOGICAL SHIELD SURVEY

2.12.1 PURPOSE

The purpose of this survey was to measure and record the radiation levels in accessible areas of the plant at various power levels to: (1) ascertain that the design criteria in regard to the radiation fields was met; (2) indicate locations where the shielding was defective; and (3) insure that plant personnel will not be subjected to unnecessary exposure to radiation as a result of inadequate shielding.

2.12.2 TEST METHOD

Surveys of radiation levels were made at power levels 0, 15, 40, and 100% of rated reactor power as follows:

Background survey measurements were made at selected locations in the reactor building and reactor auxiliary building prior to fuel loading. A planned survey outside the reactor building at 0% of rated reactor power was discontinued due to the absence of any appreciable radiation level.

General area surveys were made of the reactor shielding inside the reactor building for gamma and neutron radiation. Readings were taken at preselected readily assessible locations (floors, stairways, etc.) with slow scans between these locations. Areas of no access were not checked in this portion of the survey.

Penetrations through the secondary reactor shield inside the reactor building were surveyed for gamma and neutron radiation except that penetrations greater than 6 feet above solid footing were surveyed for gamma radiation only.

Gamma and neutron radiation surveys were made adjacent to the reactor building wall in all accessible areas at elevations of 2, 4 and 6 feet above the floor. General area surveys at approximately 3 feet above the floor were also made in these areas at some distance from the wall.

Penetrations to enclosed areas through the reactor building wall were surveyed for gamma and neutron radiation except that penetrations greater than 6 feet above solid footing were surveyed for gamma radiation only.

Surveys for gamma and neutron radiation were made adjacent to the unenclosed reactor building wall at 2, 4 and 6 feet above the ground level. General area surveys were made around the reactor building, reactor auxiliary building, turbine building, and the turbine auxiliary building.

The reactor auxiliary building elevator was surveyed for gamma and neutron radiation at approximately 2 to 3 foot increments of elevator travel from top to bottom of elevator shaft.

General area surveys were made in accessible areas of the reactor auxiliary building zoned for less than high radiation areas. Measurements were made at preselected locations with slow scans between locations.

Surveys for gamma radiation were made around penetrations through walls, floors, and ceilings from high to lower radiation zones in the reactor auxiliary building.

Gamma radiation surveys were made at 2, 4 and 6 foot elevations above the floor adjacent to walls bordering high radiation areas.

2.12.3 RESULTS AND EVALUATION

Results of the background survey measurements revealed a maximum of 0.05 mrem/hr. of gamma radiation when measured with a Geiger-Mueller instrument 0 mrem/hr. of gamma when measured with ionization chamber instruments, and 0 mrem/hr of neutron radiation when measured with a BF_3 instrument.

The general area survey inside the reactor building revealed that elevation 336 ft. (except at the elevator), the majority of elevation 357 ft., and elevation 424 ft. (except at the elevator) exceeded the designed zoning of less than 100 mrem/hr. and were high radiation areas.

Surveys of penetrations inside the reactor building showed that radiation streaming through a number of penetrations exceeded 100 mrem/hr. and others approached 100 mrem/hr.

The survey of enclosed areas adjacent to the reactor building determined that some areas contained piping which exceeds the zoning limits on contact. In general the area readings were still within the design limits. However, as activation and corrosion products increase localized areas may exceed their design limits.

Streaming through the penetrations of the reactor building walls was low or negligable except in the north piping penetration room on elevation 354 ft. None of the streaming paths resulted in radiation in excess of the zoning design of 100 mrem/hr. as the three highest readings were 54, 41, and 11.1 mrem/hr. while the other readings were 2.7 mrem/hr. or less.

In unenclosed areas all radiation levels attributed to plant operation were 0.2 mrem/hr. or less.

All readings in the reactor auxiliary building elevator were 0.1 mrem/hr. or less.

With the exception of the valve and piping area outside of the makeup and purification demineralizers the general area survey revealed radiation levels within the designed zone limits. The valve and piping area showed radiation levels of 15 to 40 mrem/hr. versus the design value of 15 mrem/hr. or less. As activation and corrosion products increase these levels are expected to increase; however, since the original design most valves in this area have been equipped with remote operators.

Surveys of the reactor auxiliary building at 2, 4, and 6 ft. above the floor show that the diesel generator room is the only location where design values were exceeded. In this location at one point about 4 inches in diameter the radiation level is 1.9 mrem/hr. on contact with the wall while the general area survey showed levels less than 1.0 mrem/hr. for which the area is zoned.

2.12.4 CONCLUSIONS

Areas exceeding acceptance criteria are undergoing further design evaluation. All high radiation areas have been posted and personnel are being protected against unnecessary exposure to radiation.

2.13 EFFLUENT MONITORING SYSTEM CHECK AT POWER

2.13.1 PURPOSE

The purpose of this test was to verify the calibration of the effluent radiation monitors against an effluent sample of which the concentration was determined by laboratory analysis. An acceptance criteria of ± 20% maximum allowable deviation of the value determined by the radiation monitor from that determined by the laboratory analysis was established.

In addition this check was performed to verify the automatic closing of the waste discharge control valves on signal from their respective high radiation alarms.

2.13.2 METHOD

The following process radiation monitors were checked:

RE-4642 - Liquid Waste Radiation Monitor
RE-3618 - Discharge Flume Radiation Monitor
RE-3632 - Main Condensor Air Discharge Radiation Monitor
RE-4830 - Gaseous Waste Radiation Monitor
RE-7400 - Stack Radiation Monitor

At power levels of approximately 0, 15, 40, 75 and 100% of full power samples were collected from each of the systems above and analyzed by laboratory methods for their activity level. Simultaneous readings were taken from the control room display of the process radiation monitors, converted to activity levels by use of the manufacturers calibration curves, and then compared with the values determined by analysis.

At each of the power levels specified above the interlock between high radiation alarm and discharge control valves for the Liquid and the Gaseous Waste Radiation Monitors was checked for proper operation.

2.13.3 RESULTS AND EVALUATION

Results of the activity levels determined by laboratory analysis and process monitor indications are summarized in Table 2.13-1.

The interlock between high radiation alarms and respective discharge control valves functioned properly to close off the discharge of waste upon indication of high radiation levels.

2.13.4 CONCLUSIONS

The test results failed to conform to acceptance criteria of ± 20% agreement between lab analysis and process monitor values by a large margin in some cases. However, this was because the activity levels being measured by the process monitors were below the lower end of their detection capability of approximately 1×10^{-6} μCi/ml and thus highly inaccurate. The maximum deviation occurred with a lab analysis of 1.55×10^{-8} μCi/ml which was below the detection capability of the process monitor by a factor of about 100.

As stated in the results the interlock between the high radiation alarms and respective discharge control valves functioned properly.

With the exception noted above the acceptance criteria of this test were met.

TABLE 2.13-1

SUMMARY OF ACTIVITY LEVELS DETERMINED BY LAB ANALYSIS AND PROCESS MONITOR INDICATIONS

Power Level (% Full Power)	Activity Determined by Lab Analysis and Process Monitor Indication ($\mu Ci/ml$) (Lab Analysis in parenthesis)				
	RE-4642	RE-3618	RE-3632	RE-4830	RE-7400
0	(1.31×10^{-6}) 2.1×10^{-5}	(6.57×10^{-7}) 2.3×10^{-6}	(9.86×10^{-7}) 1.2×10^{-6}	(4.93×10^{-6}) 2.3×10^{-6}	(5.91×10^{-6}) 1.6×10^{-6}
15	(4.68×10^{-8}) 2.3×10^{-6}	(1.55×10^{-8}) 2.8×10^{-6}	(1.36×10^{-6}) 1.6×10^{-6}	(2.91×10^{-4}) 8.0×10^{-5}	(7.90×10^{-6}) 1.5×10^{-6}
40	(9.4×10^{-5}) 1.6×10^{-4}	(2.0×10^{-7}) 2.7×10^{-5}	(2.4×10^{-7}) 1.8×10^{-6}	(9.33×10^{-5}) 1.3 [illegible]$^{-5}$	(5.1×10^{-7}) 1.4×10^{-6}
75	(5.9×10^{-5}) 9.0×10^{-5}	(5.4×10^{-8}) 2.3×10^{-6}	(1.8×10^{-7}) 1.3×10^{-6}	(8.9×10^{-2}) 4.0×10^{-2}	(1.8×10^{-7}) 1.1×10^{-6}
100	(5.16×10^{-3}) 2.05×10^{-3}	(2.92×10^{-8}) 2.1×10^{-6}	(3.22×10^{-7}) 1.2×10^{-7}	(4.64×10^{-2}) 2.5×10^{-2}	(3.22×10^{-7}) 1.0×10^{-6}

2.14 INITIAL RADIOCHEMISTRY TEST

2.14.1 PURPOSE

The purpose for conducting the Initial Radiochemistry Test was three fold. The first purpose was to monitor the activity buildup in the reactor coolant during initial fuel loading, reactor startup, and power escalation. The second purpose was to establish base activity levels so that rapid determination of failed fuel or primary to secondary leakage is possible. The third purpose was to monitor for radionuclide leakage from fuel pins to reactor coolant, from reactor coolant to steam generator, or reactor coolant to component cooling water.

2.14.2 TEST METHOD

The test methods utilized during this test used the lab test procedures and the laboratory test equipment normally utilized during plant operation. Laboratory methods used to conduct this test include gross alpha analysis, gross beta analysis, tritium, gross gamma, and gamma isotopic analysis. One or more of the above analysis were performed on each sample from each system tested. By periodic sampling, trends at various conditions were established.

2.14.3 RESULTS AND EVALUATION

The Initial Radiochemistry Test verified sampling techniques and laboratory analysis methods. Baseline activity levels for failed fuel determination were accomplished as a result of the test.

The changes made in sampling and analysis techniques have resulted in the establishment of a more uniform sampling and analysis program that will allow better trending of data and more uniform test results.

2.14.4 CONCLUSIONS

The activity buildup in the reactor coolant water was monitored through power escalation to 100% power. No abnormal results were noted.

The monitoring of leakage from primary to secondary system and primary coolant to component cooling water test results were negative.

No deficiencies are noted on this test.

2.15 INTEGRATED CONTROL SYSTEM TUNING AT POWER

2.15.1 PURPOSE

To provide direction in tuning and testing of the Integrated Control System for optimum performance for various modes of operation.

2.15.2 TEST METHOD

A. General Test Method:

Basic plant operational data was recorded, analyzed and, when required, adjustments were made to modules in the ICS so that optimum plant performance and control was obtained by means of the Integrated Control System.

B. Specific Test Methods:

Seven major divisions were accomplished by this test -

(1) Verification of hand (manual) control using Hand/Auto Stations.

(2) Verification of control station limiters and setpoint performance.

Station	Description
ICS10-MS	Turbine Header Setpoint
ICS3-MS	Low Load Limit
ICS2-MS	Max. Load Limit
ICS4-MS	Rate of Load Change
ICS30-MCS	Steam Generator Delta Tc Setpoint
ICS20-MCS	Reactor Coolant TAVE Setpoint

(3) Operational Verification of Basic ICS Loops and Operating Modes -

a. Steam Generator Low Level Setpoint
b. Turbine Header Pressure Control
c. Feedwater Valve Delta "P" Control
d. Reactor Demand Station Control
e. Feedwater Demand Station Control
f. T-AVE Control by Reactor Demand

g. T-AVE Control by Feedwater Demand
h. Delta Tc Control
i. Turbine Header Pressure Control by Turbine Throttle Valve
j. Reactor/Steam Generator Demand Station Control (Turbine Following)
k. EHC Control Station Verification (Reactor/Steam Generator Following)
l. Unit Load Demand Station Control (Integrated Mode)

(4) Verification of Control System Signal Relationship - Steady State

a. Megawatts Electric vs. Neutron Power
b. Megawatts Electric vs. Feedwater Flow
c. Turbine Bypass Valve Position vs. Megawatts Electric
d. Feedwater Temperature vs. Feedwater Flow
e. Feedwater Temperature vs. Megawatts Thermal
f. Feedwater Flow vs. Feedwater Pump Speed
g. Feedwater Startup Valve Position vs. Feedwater Flow
h. Feedwater Low Load Valve Position vs. Feedwater Flow
i. Steam Generator Startup Level vs. Megawatts Thermal
j. Steam Generator Operating Level vs. Megawatts Thermal
k. Reactor Coolant Inlet Temperature vs. Megawatts Thermal
l. Reactor Coolant Outlet Temperature vs. Megawatts Thermal

(5) Verification of Control System Operation During Controlled Transients -

a. Integrated Mode
b. Turbine Following Mode
c. Reactor/Steam Generator Following Mode
d. Reactor Demand Station Control Mode
e. Feedwater Demand Stations Control Mode

Modes b through e were performed during four (4) RCP operation only. Mode a was performed during four (4), three (3), and two (2) pump operation. All modes a through e were performed at various power levels.

(6) Verification of Control System Operation Under Various Contingencies -

a. Reactor Trip
b. Turbine Trip
c. Loss of Reactor Coolant Pump
d. Assymetric Rod

(7) Verification of Control Signal Relationship During Reactor Coolant Pump Restart

a. Reactor Coolant Flow Imbalance vs. Feedwater Flow Ratio

C. The data obtained in running the above tests was accumulated and utilized to prepare and implement a final calibration of the Integrated Control System.

2.15.3 RESULTS AND EVALUATION

Appendix 1 is a summary, referenced by procedure step numbers and by control mode description, of test results. Table 2.15-1 gives a summary of test results vs. acceptance criteria, once again referenced by procedure step numbers. Figures 2.15-1 through 2.15-12 show results of steady state tests at various power levels.

2.15.4 CONCLUSIONS

Through use of data obtained during this test, the Integrated Control System was tuned for optimum performance for transient and steady state control. All transients were performed without exceeding Technical Specification limits or causing Reactor Protection System actuation.

CONTROL STATION VERIFICATION SUMMARY

ICS TUNING AT POWER

2.15

APPENDIX 1

TEST PROCEDURE STEP NOS:

7.4.01 OTSG Low Level Setpoint Verification

Adjusted low level setpoint for OTSG "A" at 23.7" and OTSG "B" at 27.5".

7.4.02 Turbine Header Pressure Control Verification

Turbine tripped and controlling turbine header pressure with turbine bypass valves.

7.4.03 (Part 1) FW Valve Delta Pressure Control Verification

Changed ΔP control setpoint from 35 psi to 70 psi. Verified that each FW pump speed control would maintain ΔP across valve train within acceptance criteria.

7.4.03 (Part 2) FW Valve ΔP Control Verification

Verified that both loops would maintain ΔP across valve trains by selecting low ΔP.

7.4.04 Reactor Demand Control Station Verification

Verified that with Diamond rod control station in auto the ICS system would control reactor power at steady state conditions.

Also verified that with Diamond rod control station in auto and reactor demand station in manual, the ICS will control reactor power during a transient caused by reactor demand.

7.4.05 FW Demand Station(s) Control Verification

Verified that the feedwater demand stations will control FW flow above low level limit with the FW demand stations in manual.

7.4.06 T ave Control by Reactor Demand Verification

Verified that with FW demand stations in manual, the reactor demand in auto will maintain T Ave within acceptance criteria.

7.4.07 T Ave Control By FW Demand Verification

Verified that with FW demand in Auto and reactor demand station in manual that T Ave is controlled by FW within the acceptance criteria.

7.4.08 ΔTC Control Verification

Verified that ΔTC control station will maintain (in auto) ΔTC at set point by ratioing FW flow within the acceptance criteria.

7.4.09 Turbine Header Pressure Control by Turbine Throttle Valve Verification

Verified that the main turbine control system in auto will maintain turbine header pressure within acceptance criteria.

7.4.10 Reactor Steam Generator Demand Control Station Verification

Turbine following mode.

Verified that reactor steam generator demand station in manual with all other stations in auto, including EHC station will control unit within acceptance criteria. Reactor power between 20% and 25%.

7.4.11 EHC Control Station Verification

(Reactor/Steam Generator Following)

Verified that with EHC in manual and all other stations in auto, unit will control within acceptance criteria during a change in load by EHC system.

7.4.12 Unit Load Demand Station Control Verification

Verified that ICS in integrated mode will control unit within acceptance criteria during a 5% power change at 1% per minute.

7.6 Verification of Control System Signal Relationship - Steady State

A. Set up data acquisition.
B. Verification of conditions necessary to continue testing.
C. Verification of steady state conditions.
D. Analyze data to verify acceptance criteria.
E. Plot steady state relationships.
F. Analyze plotted data to determine if signal relationships are within acceptable limits at each power level test.
G. Document data.
H. Test to determine maximum neutron power limits under the following conditions:

1) 3 RCP Operation
2) 2 RCP Operation
3) Asymetric Rod
4) Loss of 1 FWP or 2 Condensate Pumps

I. Calculation of final curves and adjustments of appropriate modules for ranging signals.

NOTE: USING THE ABOVE OUTLINE,DATA WAS TAKEN AT THE FOLLOWING POWER LEVELS:

0%
15%
25%
40%
60.2%
73.8%
84.9%
96.3%
99.2%

7.7 Verification of Control System During Controlled Transients

A. (Integrated Mode)

Verified acceptance criteria met during the following transients:

1) ULD from 34.5 to 27% @ 5%/Min.
ULD from 27.0 to 35% @ 5%/Min.
ULD from 66.8 to 58.4% @ 3%/Min.
ULD from 58.4 to 66.8% @ 3%/Min.
ULD from 68.0 to 24.0% @ 5%/Min.
ULD from 86.5 to 78.0% @ 2.5%/Min.
ULD from 78.0 to 86% @ 2.5%/Min.
ULD from 87.0 to 47% @ 10%/Min.
ULD from 47.0 to 68% @ 10%/Min.

B. Turbine Following Mode

Verified acceptable control with steam generator reactor demand station in manual during the following transients:

1) Reactor Power 39 to 28% @ 5%/Min.
2) Reactor Power 28 to 38% @ 3.3%/Min.
3) Reactor Power 75.1 to 66.3% @ 2%/Min.
4) Reactor Power 66.3 to 76.2% @ 3%/Min.
5) Reactor Power 99 to 90% @ 5%/Min.
6) Reactor Power 90 to 99.1% @ 5%/Min.

C. Reactor/Steam Generator following verified acceptable control with turbine EHC in operator auto during the following transients:

1) Unit Load Demand 34.5 to 23.5% @ 3%/Min.
2) Unit Load Demand 23.5 to 35% @ 3.4%/Min.
3) Unit Load Demand 68 to 59.6% @ 3%/Min.
4) Unit Load Demand 59.6 to 68% @ 5%/Min.
5) Unit Load Demand 86.5 to 78% @ 5%/Min.
6) Unit Load Demand 78 to 86% @ 5%/Min.

D. Reactor demand station control

(All stations in auto except unit master and reactor demand) Verified acceptable control with reactor demand station in manual during the following transients:

1) Reactor Power 40.3 to 30.0% @ 4.5%/Min.
2) Reactor Power 29.2 to 39.9% @ 3.8%/Min.
3) Reactor Power 76.7 to 68.5% @ 3%/Min.
4) Reactor Power 68.5 to 76.7% @ 3%/Min.
5) Reactor Power 99.0 to 90% @ 5%/Min.
6) Reactor Power 90.0 to 98.0% @ 5%/Min.

E. FW Demand Station (s) Control

(All stations in auto except Loop A and Loop B FW Demand and unit master)

Verified acceptable control with both FW demand stations in manual during the following transients:

1) FW Demand 34 5o 22% @ 3.4%/Min.
2) FW Demand 22 to 34% @ 4%/Min.
3) FW Demand 66 to 56% @ 2.7%/Min.
4) FW Demand 56 to 66% @ 2.7%/Min.
5) FW Demand 87 to 79% @ 5%/Min.
6) FW Demand 79 to 87% @ 5%/Min.

7.8 Verification of Control System Operation Under Various Contingencies

B. Reactor Trip
C. Turbine Trip
D. Loss of Two (2) RCP
E. Asymmetric Rod

7.9 Verification of Control System Signal Relationship During 1 RCP Restart

Verified acceptable control during restart of fourth RCP and verified acceptable steady state conditions after restart.

TABLE 2.15-1

SUMMARY OF ICS TUNING RESULTS

SECTION	DATE	PARAMETER	ACCEPTANCE CRITERIA		ACTUAL DEV.	PWR LVL.
			CONDITION	LIMITS		
7.4.01	8-16-74	OTSG Level A	Steady State	± 6"	< ± 2"@ 23.7"	15%
		OTSG Level B	Steady State	± 6"	< ± 2" @ 27.5"	15%
7.4.02	8-16-74	TUR. HDR. PRESS A	Steady State	±9 PSI	↑ 3.6 PSI	NA
		TUR. HDR PRESS B	Steady State	±9 PSI	↑ 3.6 PSI	NA
Part 1 7.4.03	9-8-74	Loop AΔP	Steady State	±3 PSI	±3 PSI	45%
		Loop BΔP	Steady State	± 3 PSI	±3 PSI	45%
Part 2 7.4.03	10-5-74	Loop AΔP	Steady State	±3 PSI	±3 PSI	
		Loop BΔP	Steady State	±3 PSI	±3 PSI	
7.4.04	8-17-74	Neutron Error	Steady State	±1%	±1%	12-15%
7.4.05	9-4-74	Loop A FW Flow Error	Steady State	±120 K #/hr	±25K #/hr	
		Loop B FW Flow Error	Steady State	±120 K #/hr	±25K #/hr	
7.4.06	9-7-74	T AVE	Steady State	± 2°F	±.5°	
		T AVE	Step Change	± 2°F	±.75°	
7.4.07	9-7-74	FW Flow Error A	Steady State	±120 K #/hr	±60K #/hr	
		FW Flow Error B	Steady State	±120 K #/hr	±60K #/hr	
		T AVE	Ramp Change	± 5°F	± .2°	
		T AVE	Steady State	± 2°F	± .2°	

TABLE 2.15-1

SECTION	DATE	PARAMETER	ACCEPTANCE CRITERIA CONDITION	LIMITS	ACTUAL DEV.	PWR LVL.
7.4.08	9-7-74	ΔTC Temp	Steady State	± 1°F	±.6° F	
		ΔTC Temp	Transient	± 5°F	NA	
7.4.09	8-18-74	TUR. HDR. PRESS A	Steady State	± 9 PSI	±5 PSI	
		TUR. HDR. PRESS B	Steady State	± 9 PSI	±5 PSI	
		TUR. HDR. PRESS A	Transient	± 50 PSI	↓ 12 ↑ 7.2	
		TUR. HDR. PRESS B	Transient	± 50 PSI	↓ 12 ↑ 7.2	
7.4.10	9-10-74	T AVE	Steady State	±2°F	±.4° F	
		T AVE	Ramp Change	±5°F	±.4° F	
		ΔTC TEMP	Steady State	±1°F	±.32° F	
		ΔTC TEMP	Transient	±5°F	±.32° F	
		TUR. HDR. PRESS A	Steady State	±9 PSI	±3 PSI	
		TUR. HDR PRESS B	Steady State	±9 PSI	±3 PSI	
		TUR. HDR. PRESS A	Transient	±50 PSI	↑ 5 PSI	
		TUR. HDR. PRESS B	Transient	±50 PSI	↑ 5 PSI	
		MW Error	Steady State	±10 MW	±1 MW	
7.4.11	9-10-74	T AVE	Steady State	±2°F	±.6° F	
		ΔTC	Steady State	±1°F	±.3° F	
		TUR. HDR. PRESS A	Steady State	±9 PSI	±3 PSI	
		TUR. HDR PRESS B	Steady State	±9 PSI	±3 PSI	
		T AVE	Ramp Change	±5°F	±.7° F	
		ΔTC	Transient	±5°F	+.4° to -.2° F	
		TUR. HDR PRESS A	Transient	±50 PSI	±7.0 PSI	
		TUR. HDR PRESS B	Transient	±50 PSI	±7.0 PSI	

TABLE 2.15-1

SECTION	DATE	PARAMETER	ACCEPTANCE CRITERIA		ACTUAL DEV.		PWR LVL.
			CONDITION	LIMITS			
7.4.12	9-10-74	T AVE	Steady State	±2°F	±.3°F		
		T AVE	Ramp Change	±5°F	±.5°F		
		ΔTC	Steady State	±1°F	±.2°F		
		ΔTC	Transient	±5°F	+.2 to -.25°F		
		MW Error	Steady State	±10 MW	±3 MW		
7.6 D	9-23-74	OTSG LEVEL A-B	Steady State	±6"	A ±1.2"	B ±1.2"	15%
		TUR. HDR. PRESS A-B	Steady State	±9 PSI	A ±3.3	B ±3.3	15%
		FWΔP A-B	Steady State	±3PSI	A ±3	B ±3	15%
		Neutron Error	Steady State	±1%	±1%		15%
		FW Flow Error A-B	Steady State	±120K#/hr	A ±60K	B ±60K	15%
		T AVE	Steady State	±2°F	±.4° F		15%
		ΔTC	Steady State	±1°F	±.24°		15%
		MEGAWATT ERROR	Steady State	±10 MW	±1.5 MW		15%
	10-3-74	OTSG LEVEL A-B	Steady State	±6"	A ±2.5"	B ±2"	27.99%
		TUR. HDR. PRESS A-B	Steady State	±9 PSI	A ±3.5	B ±3.5	27.99%
		FW ΔP A-B	Steady State	±3 PSI	A ± 3	B ±3	27.99%
		Neutron Error	Steady State	± 1%	±1%		27.99%
		FW Flow ERROR A-B	Steady State	±120K#/hr	A ±36K	B ±36K	27.99%
		T AVE	Steady State	±2°F	±.4°F		27.99%
		ΔTC	Steady State	±1°F	±.32° F		27.99%
		MW ERROR	Steady State	±10 MW	±2 MW		27.99%

SECTION	DATE	PARAMETER	ACCEPTANCE CRITERIA CONDITION	ACCEPTANCE CRITERIA LIMITS	ACTUAL DEV. A	ACTUAL DEV. B	PWR LVL.
7.6D Cont.	9-24-74	OTSG LEVEL A-B	Steady State	±6"	A ±4"	B ±2.5"	41.55%
		TUR.HDR. PRESS A-B	Steady State	±9 PSI	A ±6.0	B ±6.0	41.55%
		FWΔP A-B	Steady State	±3 PSI	A ± 3	B ±3	41.55%
		Neutron Error	Steady State	± 1%	±1%		41.55%
		FW FLOW ERROR A-B	Steady State	±120K#/hr	A ±60K	B ±60K	41.55%
		T AVE	Steady State	±2"F	±.4"		41.55%
		ΔTC	Steady State	±1°F	±.32° F		41.55%
		MW ERROR	Steady State	±10 MW	± 2.5 MW		41.55%
	11-18-74	OTSG LEVEL A-B	Steady State	±6"	A ± 6"	B ±6"	60.2%
		TUR.HDR. PRESS A-B	Steady State	±9 PSI	A ± 7	B ±7	60.2%
		FWΔP A-B	Steady State	±3 PSI	A NA	B NA	60.2%
		NEUTRON ERROR	Steady State	± 1%	±1%		60.2%
		FW FLOW ERROR A-B	Steady State	±120K #/hr	A ±120K	B ±120K	60.2%
		T AVE	Steady State	± 2°F	± .2°F		60.2%
		ΔTC	Steady State	± 1°F	±.32°F		60.2%
		MW ERROR	Steady State	±10 MW	± 5MW		60.2%
	11-19-74	OTSG LEVEL A-B	Steady State	±6"	A ±6"	B ±6"	73.8%
		TUR. HDR PRESS A-B	Steady State	±9PSI	A ±6.3	B ±6.9	73.8%
		FWΔP A-B	Steady State	±3PSI	A NA	B NA	73.8%
		Neutron Error	Steady State	±1%	± 1%		73.8%
		FW FLOW Error A-B	Steady State	±120 K#/hr	A ±90K	B ±90K	73.8%
		T AVE	Steady State	±2°F	±.3°F		73.8%
		ΔTC	Steady State	±1°F	±.4°F		73.8%
		MW Error	Steady State	±10MW	±4MW		73.8%

TABLE 2.15-1

SECTION	DATE	PARAMETER	ACCEPTANCE CRITERIA CONDITION	ACCEPTANCE CRITERIA LIMITS	ACTUAL DEV. A	ACTUAL DEV. B	PWR LVL.
7.6D Cont	12-5-74	OTSG LEVEL A-B	Steady State	±6"	±2.5"	±2"	84.93%
		TUR.HDR PRESS A-B	Steady State	±9 PSI	±3	±3	84.93%
		FWΔP A-B	Steady State	±3 PSI	NA	NA	84.93%
		Neutron Error	Steady State	± 1%	± 1%		84.93%
		FW Flow Error A-B	Steady State	±120K#/hr	±120K	±120K	84.93%
		T AVE	Steady State	±2°F	579.15 to 578.7		84.93%
		Δ TC	Steady State	±1°F	± .40		84.93%
		MW Error	Steady State	±10 MW	± 5 MW		84.93%
	12-8-74	OTSG Level A-B	Steady State	±6"	±5.5"	±4"	95%
		TUR. HDR Press A-B	Steady State	±9 PSI	±2PSI	±2PSI	95%
		FWΔP A-B	Steady State	±3PSI	NA	NA	95%
		Neutron Error	Steady State	±1%	±1%		95%
		FW Flow Error A-B	Steady State	±120K#/hr	±60K	±60K	95%
		T AVE	Steady State	±2°F	± .075°F		95%
		ΔTC	Steady State	±1°F	±.4°F		95%
		MW Error	Steady State	±10 MW	± 3 MW		95%
	12-9-74	OTSG Level A-B	Steady State	± 6"	±6"	±6"	100%
		TUR.HDR. Press A-B	Steady State	±9PSI	-3.6to+2.3	-1.2to+3	100%
		FWΔP A-B	Steady State	±3PSI	NA	NA	100%
		Neutron Error	Steady State	± 1%	± 1%		100%
		FW Flow Error A-B	Steady State	±120 K#/hr	±60K	±42K	100%
		T AVE	Steady State	± 2°F	±.075°F		100%
		ΔTC	Steady State	± 1°F	±.4°F		100%
		MW Error	Steady State	±10MW	+5.5 to -4.0 MW		100%

TABLE 2.15-1

SECTION	DATE	PARAMETER	ACCEPTANCE CRITERIA CONDITION	ACCEPTANCE CRITERIA LIMITS	ACTUAL DEV.	PWR LVL.
7.7A	10-1-74	TAVE	Ramp Change	± 5°F	↓ to 578.6	ULD 34.5to27
		Δ TC	Transient	± 5°F	↓.6 ↑.32	ULD 34.5to27
717A	10-1-74	T AVE	Ramp Change	±5°F	↑ 580.1 ↓ 578.6	U.L.D. 27.0 to3
		ΔTC	Transient	±5°F	+.96 to -.4	U.L.D. 27.0 to3
7.7A	12-3-74	T AVE	Ramp Change	±5°F	↓ .9° ↑ .2°	U.L.D. 66.8to58
		ΔTC	Transient	±5°F	±.4°	U.L.D. 66.8to58
7.7A	12-3-74	T AVE	Ramp Change	±5°F	↑ .8° ↓ .4°	U.L.D. 58.4to66
		ΔTC	Transient	±5°F	+.6 -.4°	U.L.D. 58.4to66
7.7A	12-4-74	T AVE	Ramp Change	±5°F	579 to 574.8	U.L.D. 68 to 24
		ΔTC	Transient	±5°F	↑.8° ↓ 1.2°	U.L.D. 68 to 24
7.7A	12-10-74	T AVE	Ramp Change	± 5°F	579.75 to 579.0	U.L.D. 86.5 to7
		ΔTC	Transient	± 5°F	-.4 to +.4	U.L.D. 86.5 to7
7.7A	12-10-74	T AVE	Ramp Change	± 5°F	578.55 to 579.0	U.L.D. 78 to 86
		ΔTC	Transient	± 5°F	+.2 to -.6	U.L.D. 78 to 86
7.7A	12-10-74	T AVE	Ramp Change	±5°F	576.75to579.6	U.L.D. 87 to 47
		ΔTC	Transient	± 5°F	-3.4 to 4.6	U.L.D. 87 to 47
					Main Block Valves Hung in Intermediate Pos.	
7.7A	12-10-74	T AVE	Ramp Change	± 5°F	580.2 to 578.55	U.L.D 47 to 68
		ΔTC	Transient	± 5°F	-.6 to +.6	U.L.D. 47 to 68

TABLE 2.15-1

SECTION	DATE	PARAMETER	ACCEPTANCE CRITERIA CONDITION	LIMITS	ACTUAL DEV.		TRANSIENT
7.7B	10-1-74	T AVE	Ramp Change	± 5°F	↓1.2°	↑.2°	Rx Pwr 39 to 28
		Δ TC	Transient	± 5°F	-1.44 to +.6		Rx Pwr 39 to 28
		TUR.HDR Press A-B	Transient	± 50PSI	A ↓5.4	B ↓5.4	Rx Pwr 39 to 28
7.7B	10-1-74	T AVE	Ramp Change	± 5°F	↑1.2°	↓.2°	Rx Pwr 28 to 38
		Δ TC	Transient	± 5°F	+1.2 to -.4		Rx Pwr 28 to 38
		TUR.HDR Press A-B	Transient	± 50PSI	A ↑6↓3.6	B ↑6 ↓3.6	Rx Pwr 28 to 38%
7.7B	12-4-74	T AVE	Ramp Change	± 5°F	577.5 to 578.7		Rx Pwr 75.1to66
		Δ TC	Transient	± 5°F	-.4 to +.2		Rx Pwr 75.1to66
		TUR HDR Press A-B	Transient	±50PSI	A ↓ 14.4	B ↓14.4	Rx Pwr 75.1to66
7.7B	12-4-74	T AVE	Ramp Change	±5°F	576.6 to 578.7		Rx Pwr 66.3to76
		Δ TC	Transient	±5°F	+.8 to -.8		Rx Pwr 66.3to76
		TUR HDR Press A-B	Transient	±50PSI	A ↑10.8	B ↑10.8	Rx Pwr 66.3to76
7.7B	12-10-74	T AVE	Ramp Change	±5°F	578.9 to 577.9		Rx Pwr 99 to 90
		Δ TC	Transient	±5°F	± .5°F		Rx Pwr 99 to 90
		Tur.Hdr Press A-B	Transient	±50PSI	A ↑7.2↓12	B ↑7.2 ↓12	Rx Pwr 99 to 90
7.7B	12-10-74	T AVE	Ramp Change	±5°F	578.7 to 579.3		Rx Pwr 90 to 99
		Δ TC	Transient	±5°F	± .5°		Rx Pwr 90 to 99
		TUR HDR Press A-B	Transient	±50PSI	A ↓14.4↑4.8	B ↓14.4↑4.8	Rx.Pwr 90to99.

TABLE 2.15-1

SECTION	DATE	PARAMETER	ACCEPTANCE CRITERIA CONDITION	ACCEPTANCE CRITERIA LIMITS	ACTUAL DEV.		TRANSIENT
7.7C	10-1-74	T AVE	Ramp Change	±5°F	± .2°F		U.L.D. 34.5to23.5
		ΔTC	Transient	±5°F	↓1.28°	↑.32	U.L.D. 34.5to23.5
		TUR HDR Press A-B	Transient	±50PSI	A ↑37.2	B ↑37.2	U.L.D. 34.5to23.
7.7C	10-1-74	T AVE	Ramp Change	±5°F	± 1.5°		U.L.D. 23.5 to 35
		ΔTC	Transient	±5°F	+1.48 to -.32° F		U.L.D. 23.5 to 35
		TUR HDR Press A-B	Transient	±50PSI	A ↓34.8	B ↓34.8	U.L.D. 23.5 to 35
7.7C	12-4-74	T AVE	Ramp Change	±5°F	577.8 to 579.0°F		U.L.D. 68 to 59.6
		ΔTC	Transient	±5°F	±.3		U.L.D. 68 to 59.6
		TUR HDR Press A-B	Transient	±50PSI	A ↑25.2↓14.4	B ↑28.8↓14.4	U.L.D. 68to59.
7.7C	12-4-74	T AVE	Ramp Change	±5°F	579.75 to 578.5°F		U.L.D. 59.6 to 68
		ΔTC	Transient	±5°F	+.5 to -.3		U.L.D. 59.6 to 68
		TUR HDR Press A-B	Transient	±50 PSI	A ↓30.6 ↑10.8	B ↓30.6 ↑10.8	U.L.D. 59.6to68
7.7C	12-10-74	T AVE	Ramp Change	±5°F	578.4 to 579.075		U.L.D. 86.5 to 78
		ΔTC	Transient	±5°F	-.2 to +.4°F		U.L.D. 86.5 to 78
		TUR.HDR Press A-B	Transient	±50PSI	A ↑21.6	B ↑21.6	U.L.D. 86.5 to 78
7.7C	12-10-74	T AVE	Ramp Change	±5°F	578.7 to 579.0°F		U.L.D. 78 to 86%
		ΔTC	Transient	±5°F	+.2 to -.2°F		U.L.D. 78 to 86%
		TUR HDR Press A-B	Transient	±50PSI	A ↑3.6↓14.4	B ↑3.6↓14.4	U.L.D. 78 to 86%

TABLE 2.15-1

SECTION	DATE	PARAMETER	ACCEPTANCE CRITERIA CONDITION	LIMITS	ACTUAL DEV.	TRANSIENT
7.7D	10-2-74	T AVE	Ramp Change	± 5°F	↓3° F ↑1°F	Rx.Pwr 40.3to30%
		Δ TC	Transient	± 5°F	↑1.52° ↓.36	Rx.Pwr 40.3 to 3
		TUR HDR Press A-B	Transient	± 50 PSI	A ↑5.5↓4 | B ↑5.5↓4	Rx.Power 40.3 to 3
7.7D	10-2-74	T AVE	Ramp Change	± 5°F	↑2.5°F ↓2°F	Rx. Pwr 29.2%to 3
		ΔTC	Transient	± 5°F	↓1.2°F ↑3.2°F	Rx.Pwr 29.2% to3
		TUR HDR Press A-B	Transient	± 50PSI	A ↓8 ↑3 | B ↓8 ↑3	Rx.Power 29.2%to39.
7.7D	12-4-74	T AVE	Ramp Change	± 5°F	579.6 to 575.1	Rx.Pwr 76.7%to 6
		ΔTC	Transient	± 5°F	-.7 to +.4°F	Rx.Pwr 76.7to68.
		TUR HDR Press A-B	Transient	± 50PSI	A ↓14.4 ↑10.8	Rx.Pwr 76.7to68.
					B ↓16.2 ↑10.8	
7.7D	12-4-74	T AVE	Ramp Change	±5°F	581.4 to 577.2°F	Rx Pwr 68.5to76.
		ΔTC	Transient	±5°F	-.4 to +.7°F	Rx Pwr 68.5to76.
		TUR HDR Press A-B	Transient	±50PSI	A ↑27 ↓18 | B ↑27 ↓18	Rx.Pwr 68.5to76.
7.7D	12-10-74	T AVE	Ramp Change	±5°F	-1.2° to +.3°F	Rx Pwr 99 to 90%
		ΔTC	Transient	±5°F	±.4°F	RX Pwr 99 to 90%
		TUR HDR Press A-B	Transient	±50 PSI	A -14.4PSI | B -10.8PSI	Rx.Pwr 99 to 90
7.7D	12-10-74	T AVE	Ramp Change	±5°F	+2.4° to -.9°F	Rx Pwr 90 to 98%
		ΔTC	Transient	±5°F	-.4° to +.5°F	Rx. Pwr. 90 to 98%
		TUR HDR Press A-B	Transient	±50 PSI	A +14.4to-3	Rx.Pwr 90 to 98%
					B +14.4 to-4.8	

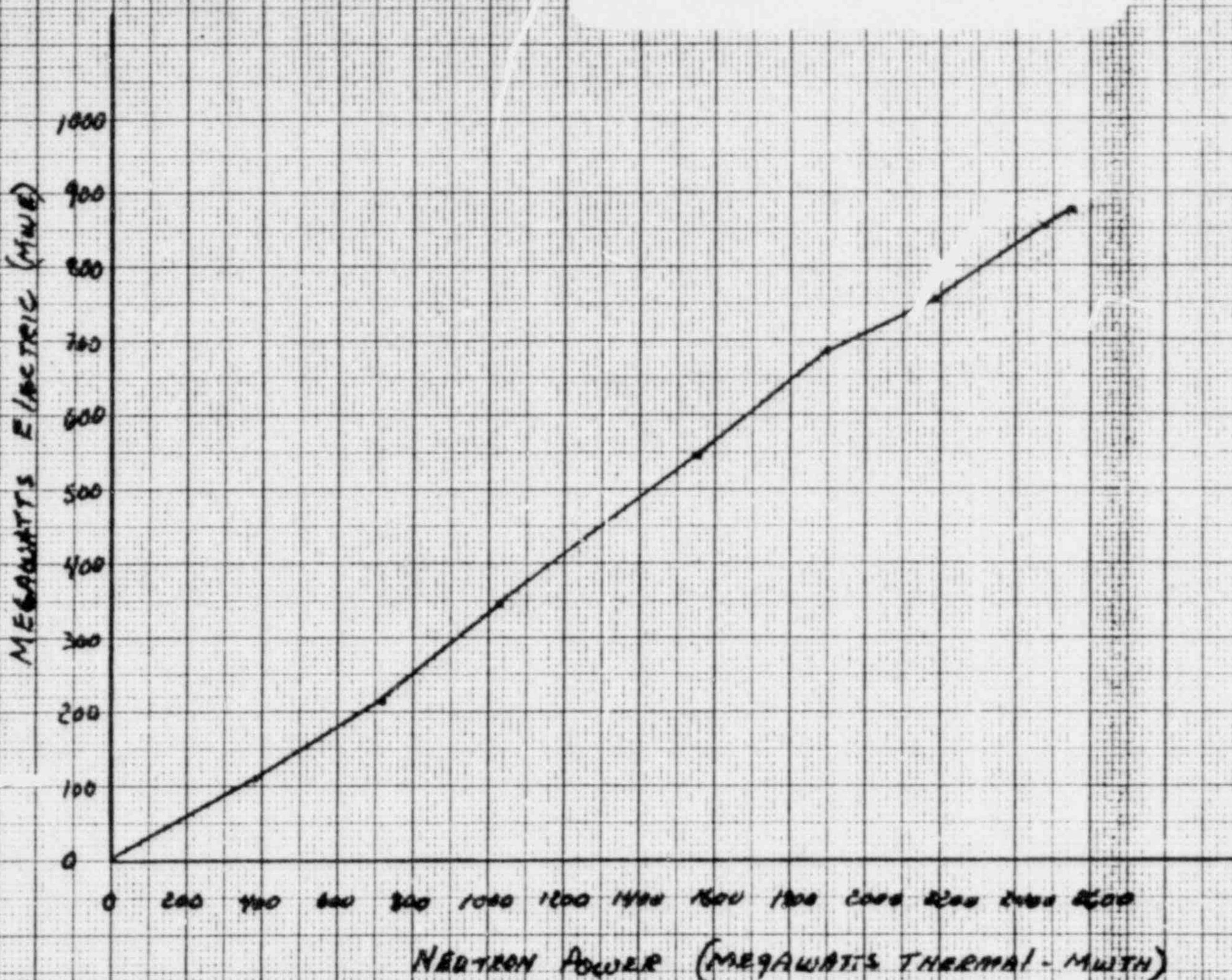
TABLE 2.15-1

SECTION	DATE	PARAMETER	ACCEPTANCE CRITERIA CONDITION	ACCEPTANCE CRITERIA LIMITS	ACTUAL DEV. A	ACTUAL DEV. B	TRANSIENT
7.7F	10-2-74	T AVE	Ramp Change	± 5°F	↓ 2°F		FW Demand 34 to 22%
		TUR HDR Press A-B	Transient	± 50 PSI	↓6 ↑3	↓6 ↑3	FW Demand 34 to 22%
7.7F	10-2-74	T AVE	Ramp Change	± 5°F	↓2° F		FW Demand 22 to 34%
		TUR HDR Press A-B	Transient	± 50PSI	↓4 ↑8	↓4 ↑8	FW Demand 22 to 34%
7.7F	12-4-74	T AVE	Ramp Change	± 5°F	580.2 to 577.5		FW Demand 66 to 56%
		TUR HDR Press A-B	Transient	± 50PSI	↓15.6	↓15.6	FW Demand 66 to 56%
7.7F	12-4-74	T AVE	Ramp Change	± 5°F	576.6 to 579.0		FW Demand 56 to 66%
		TUR HDR Press A-B	Transient	± 50 PSI	↑ 11.4	↑11.4	FW Demand 56 to 66%
7.7F	12-10-74	T AVE	Ramp Change	±5°F	578.4 to 579.45		FW Demand 87 to 79%
		TUR HDR Press A-B	Transient	±50 PSI	↓10.8	↓10.8	FW Demand 87 to 79%
7.7F	12-10-74	T AVE	Ramp Change	±5°F	578.1 to 579.3		FW Demand 79 to 87%
		TUR HDR Press A-B	Transient	±50 PSI	↑6 ↓8.4	↑6 ↓8.4	FW Demand 79 to 87%
7.8B	10-3-74	T AVE	Steady State	±2°F			Rx Trip
		ΔTC	Steady State	±1°F			Rx Trip
		MW Error	Steady State	±10 MW			RxTrip
7.8C	12-11-74	T AVE	Steady State	± 2°F	±.08° F		Turbine Trip
		ΔTC	Steady State	± 1°F	±.4° F		Turb.Trip
		MW Error	Steady State	±10MW	± 5MW		Turb.Trip

TABLE 2.15-1

SECTION	DATE	PARAMETER	ACCEPTANCE CRITERIA CONDITION	LIMITS	ACTUAL DEV.	
7.8D	11-24-74	T AVE	Steady State	±2°F	±.2°F	Loss RCP
		ΔTC	Steady State	±1°F	±.4°F	Loss RCP
		MW Error	Steady State	±10MW	±4MW	Loss RCP
7.8E	10-2-74	T AVE	Steady State	±2°F	±.6°F	Asy Rod
		ΔTC	Steady State	±1°F	±1°F	Asy Rod
		MW Error	Steady State	±10MW	±2MW	Asy Rod
7.9	12-14-74	T AVE	Steady State	±2°F	-.8° to +.4°F	RCP Resta
		ΔTC	Steady State	±1°F	±.2°F	RCP Resta
		MW Error	Steady State	±10 MW	+5 to -10 MW	RCP Resta
		T AVE	Transient	±5°F	-2.7° to +1.2°F	RCP Restar
		ΔTC	Transient	±5°F	+4.8°F to -1.8°F	RCP Resta

FIGURE 2.15-1



% POWER	MWTH	MWE
0	0	0
15	385.2	110
27.99	719.04	206
40	1027.2	347
60.2	1546	546.
73.8	1895	688
84.93	2181	752
96.3	2472.9	857
99.17	2547	877

FIGURE 2.15-2

% Power	Gen MW	Total FW Flow	
0	0	.0068	x10^6
15	110	1.654	″
27.99	206	2.67	″
40	347	4.232	″
60.2	546	6.3	″
73.8	688	7.79	″
84.93	752	8.94	″
96.3	857	10.28	″
99.17	877	10.6	″

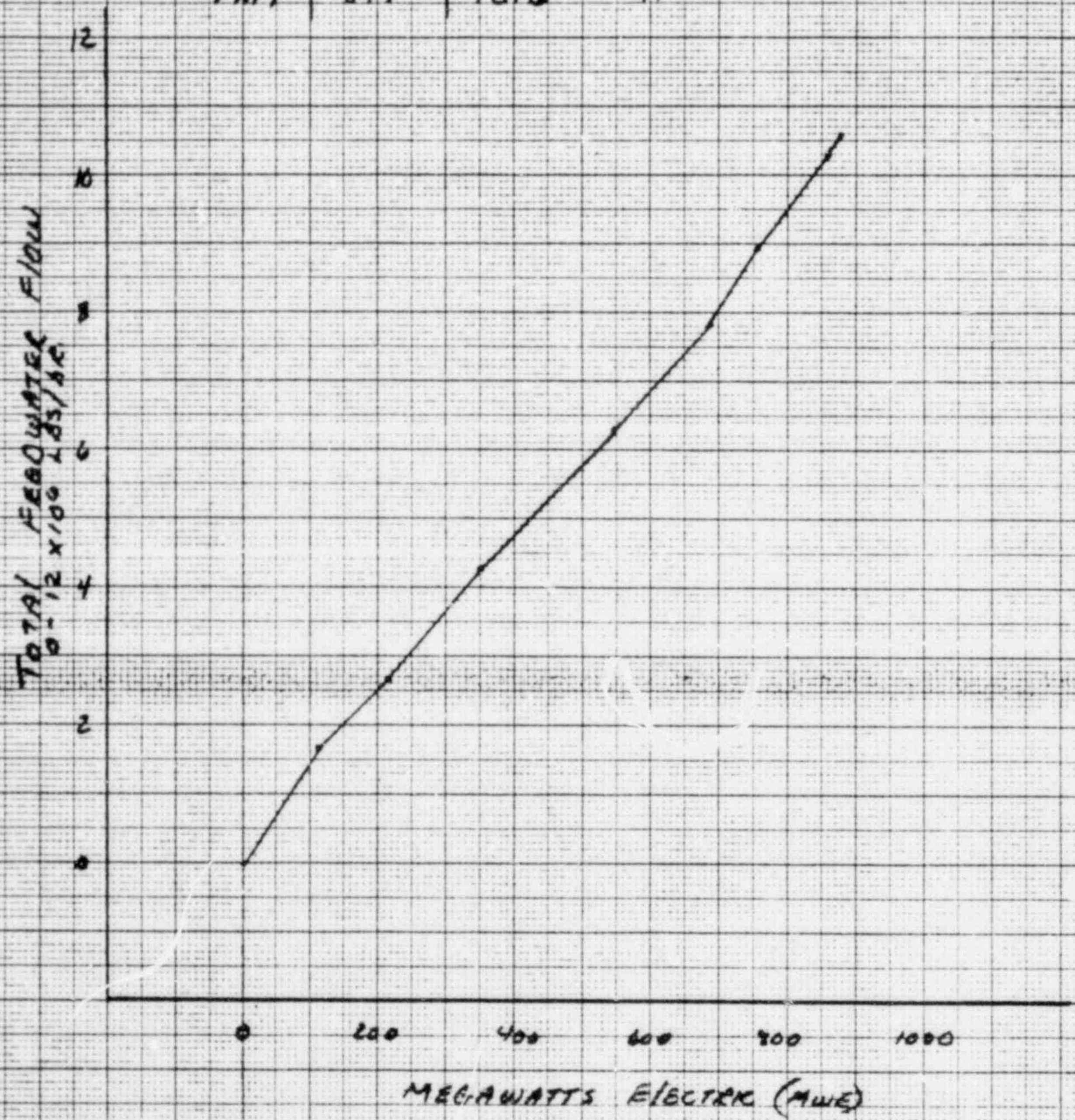
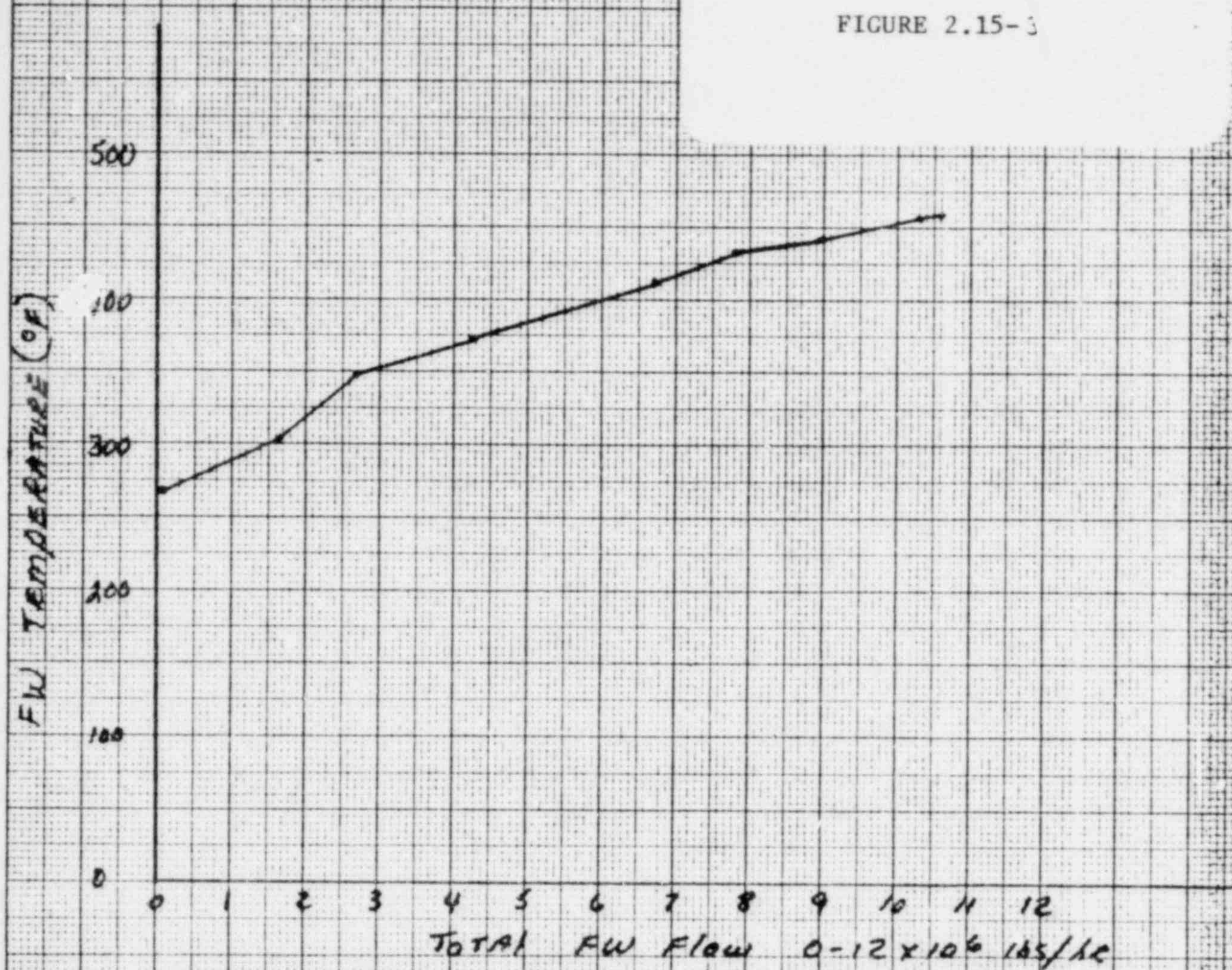
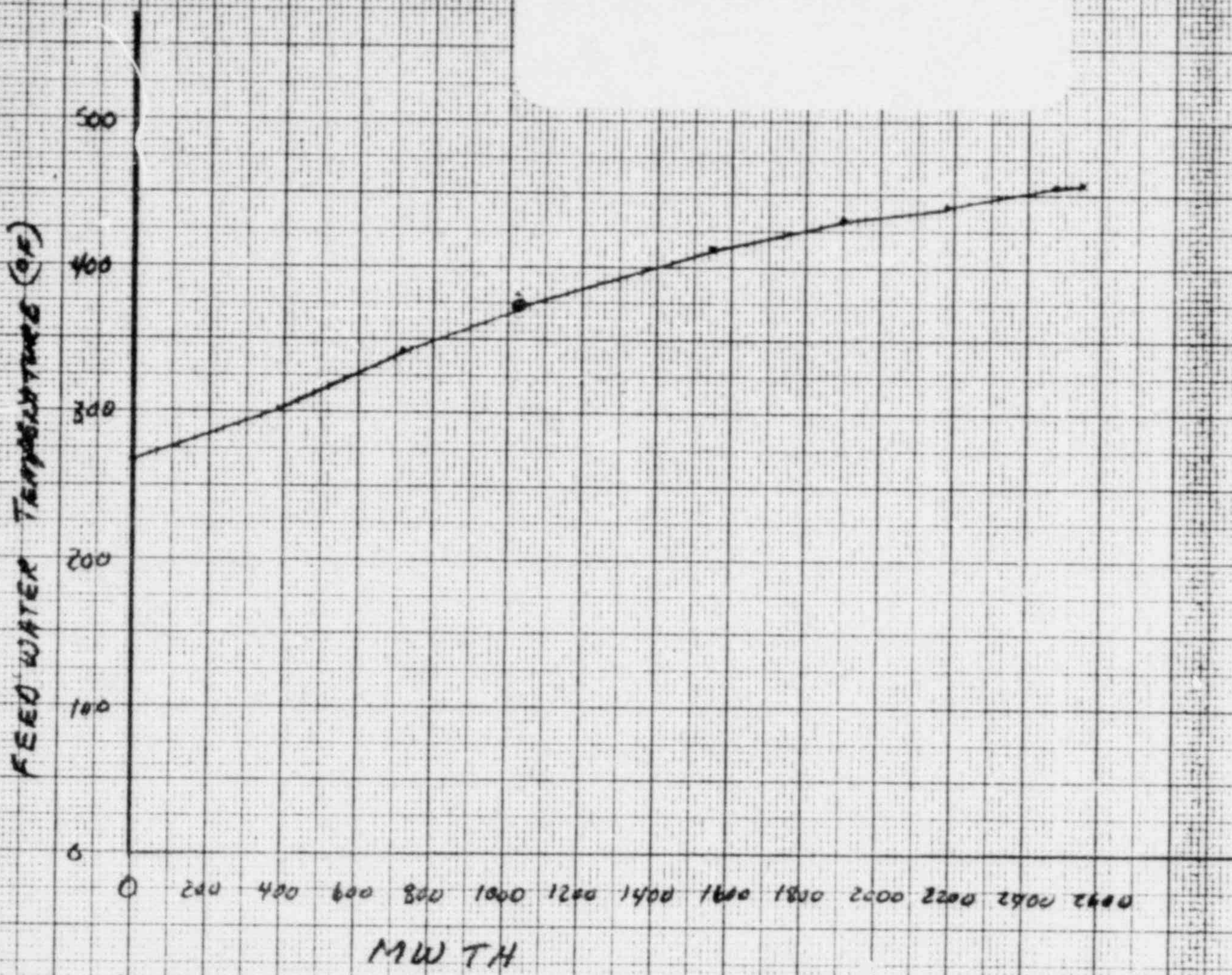


FIGURE 2.15-3



% Power	AVE FW Temp	Total FW Flow
0	26[illegible]	.0068 x10^6 lb/hr
15	302.2	[illegible] "
27.99	342.8	2.67 "
40	373.8	4.23 "
60.2	410.9	6.7 "
73.8	431.2	7.79 "
84.93	442.1	8.94 "
96.3	455.8	10.28 "
99.17	458.5	10.6 "

FIGURE 2.15- 4



% POWER	AVG FW TEMP	MW TH
0	268.9	0
15	302.18	385.2
27.99	342.8	719.04
40	[illegible]73.8	1027.2
60.2	[illegible]10.9	1546
73.8	431.2	1895
84.93	442.1	2181
96.3	455.8	2472.9
99.1	[illegible]5	2547

FIGURE 2.15-5

% Power	MWTH	Total FW Flow	
0	0	.0068	X10⁶ #/hr.
15	385.2	1.654	"
27.99	719.04	2.67	"
40	1027.2	4.232	"
60.2	1546	6.3	"
73.8	1895	7.79	"
85.93	2181	8.94	"
96.3	2472.9	10.28	"
99.17	2547	10.6	"

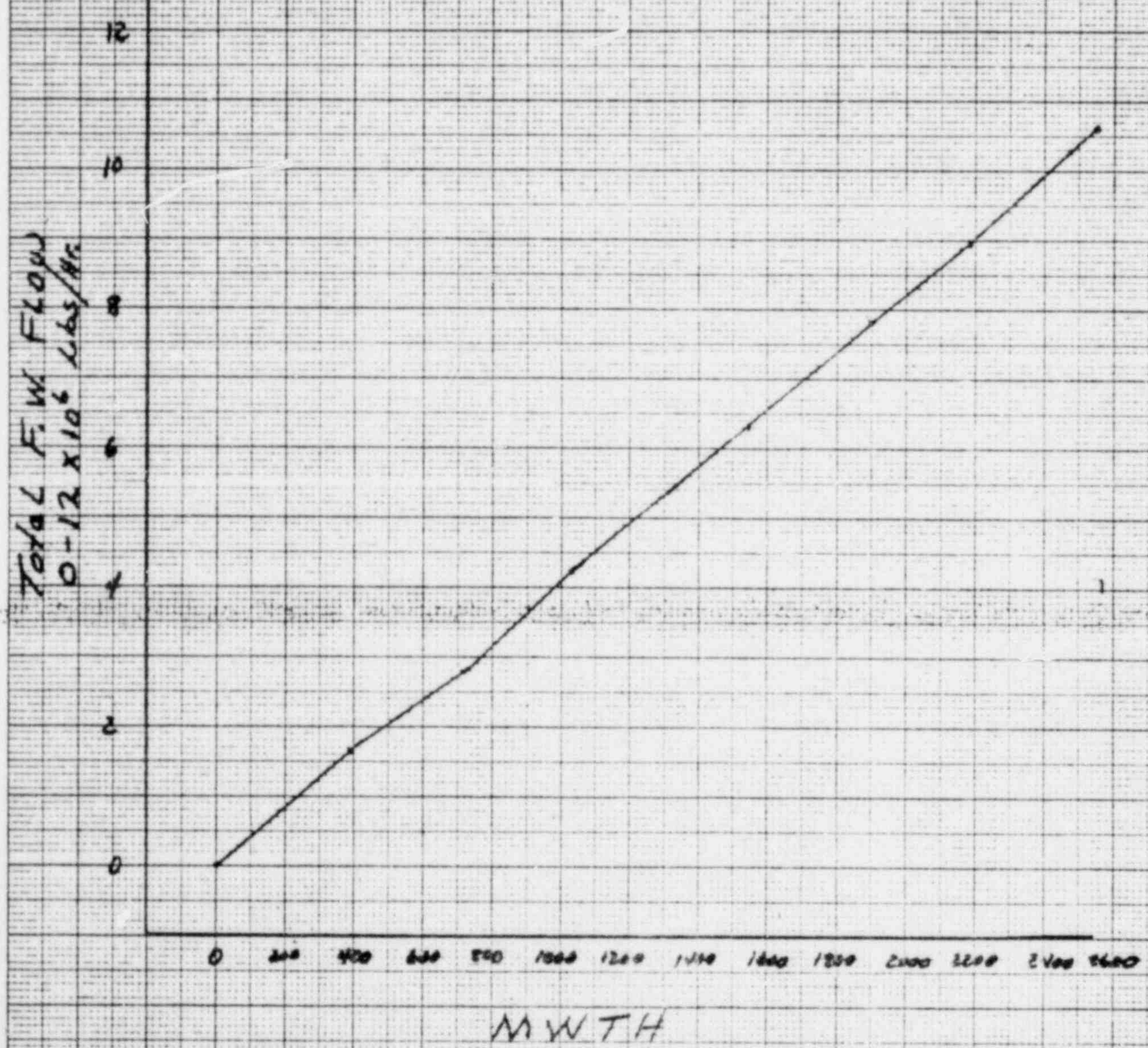
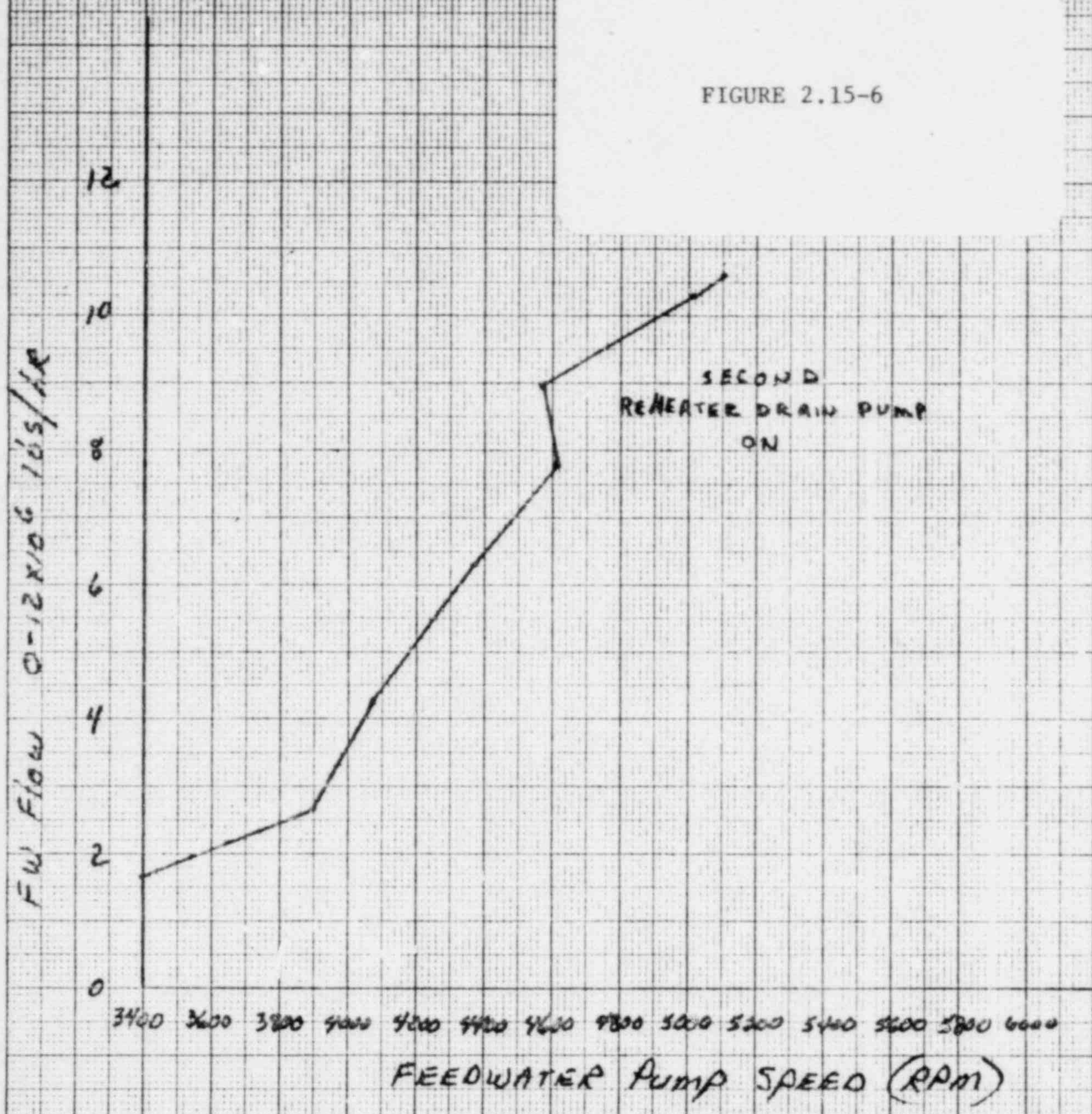
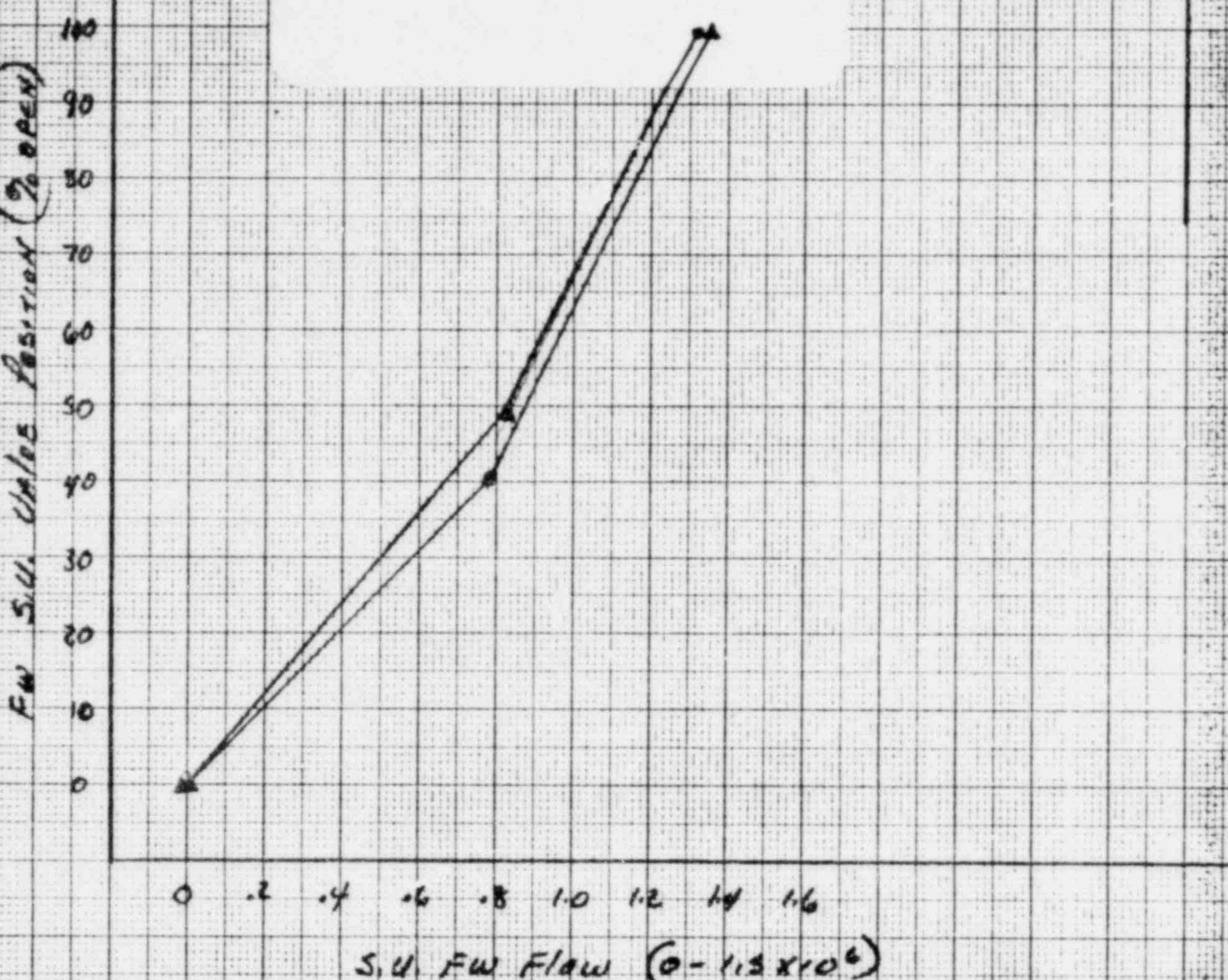


FIGURE 2.15-6



% Power	FW Pump	FW Flow
0	3400 RPM	.0068 $\times 10^6$ lb/hr
15	3400 "	1.654 "
27.99	3900 "	2.67 "
40.0	4073 "	4.232 "
60.2	4350 "	6.3 "
73.8	4613 "	7.79 "
84.93	4568 "	8.94 "
96.3	5013 "	10.28 "
99.17	5100 "	10.6 "

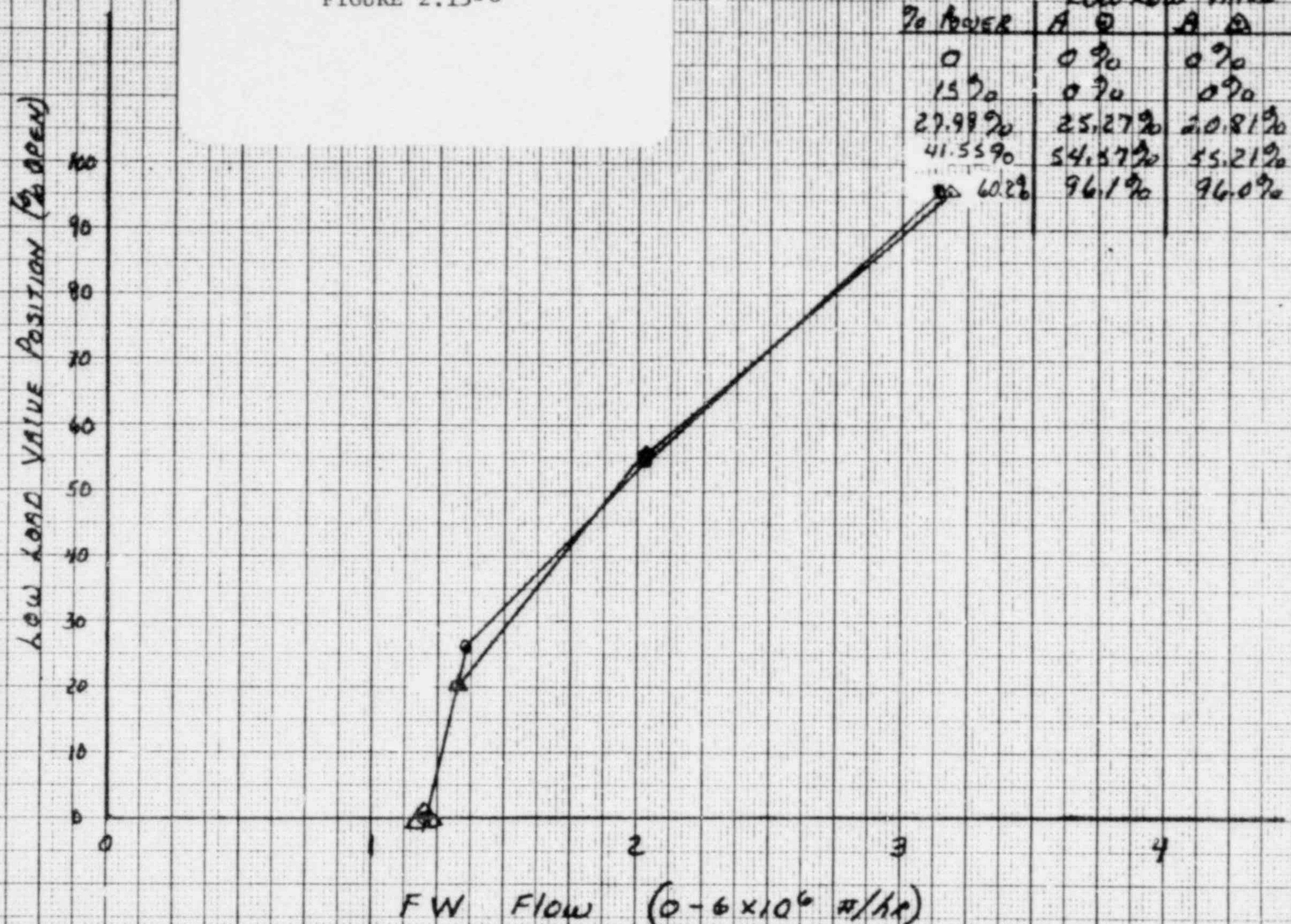
FIGURE 2.15-7



% Power	S.U. Valve A %	Position B %	S.U. FW A ⊙	Flow B △
0	0	0	0	0
15	41.21 %	48.75 %	.78	.82
27.99	100 %	100 %	1.347	1.318 *

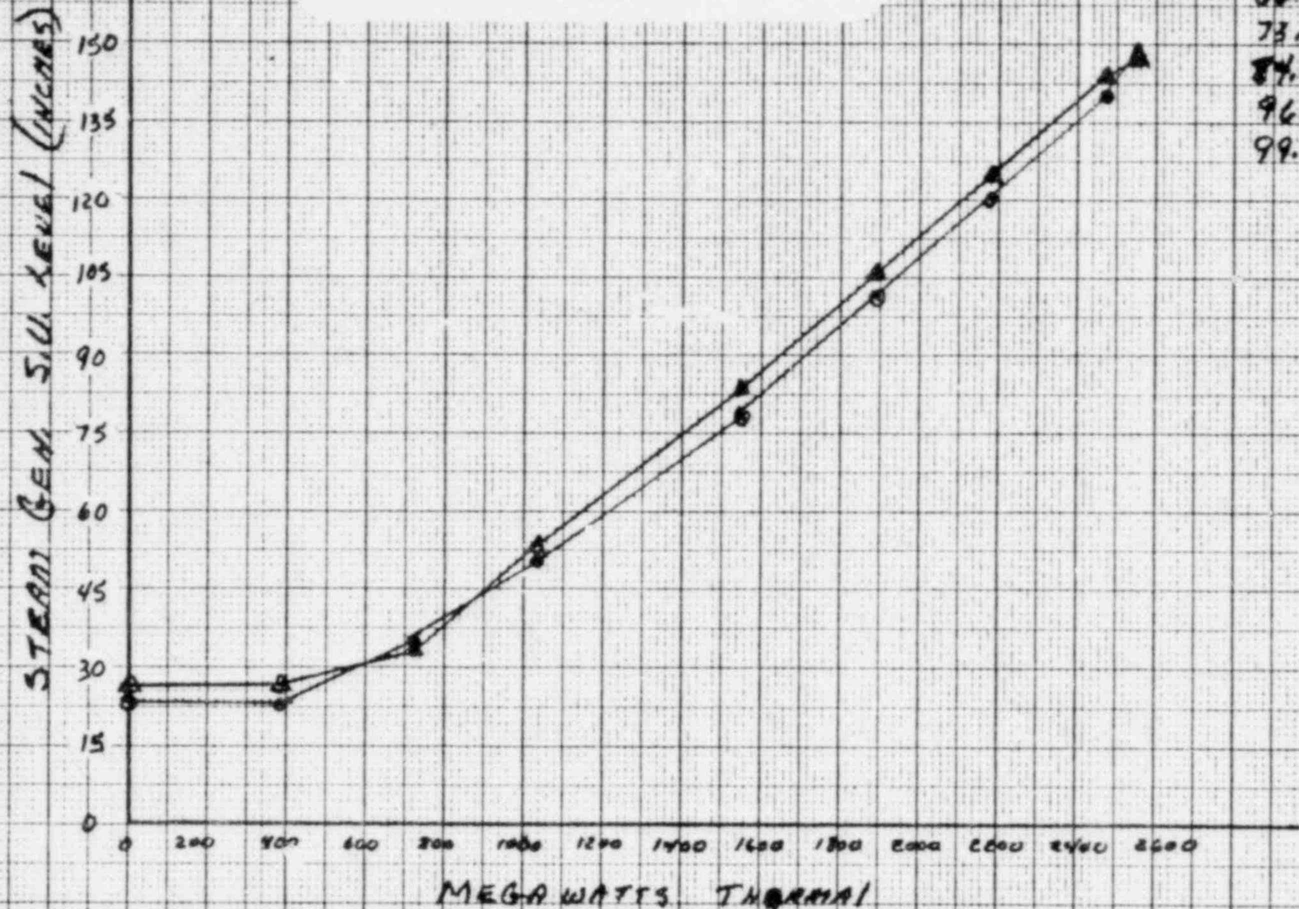
* Not S.U. Flow

FIGURE 2.15-8



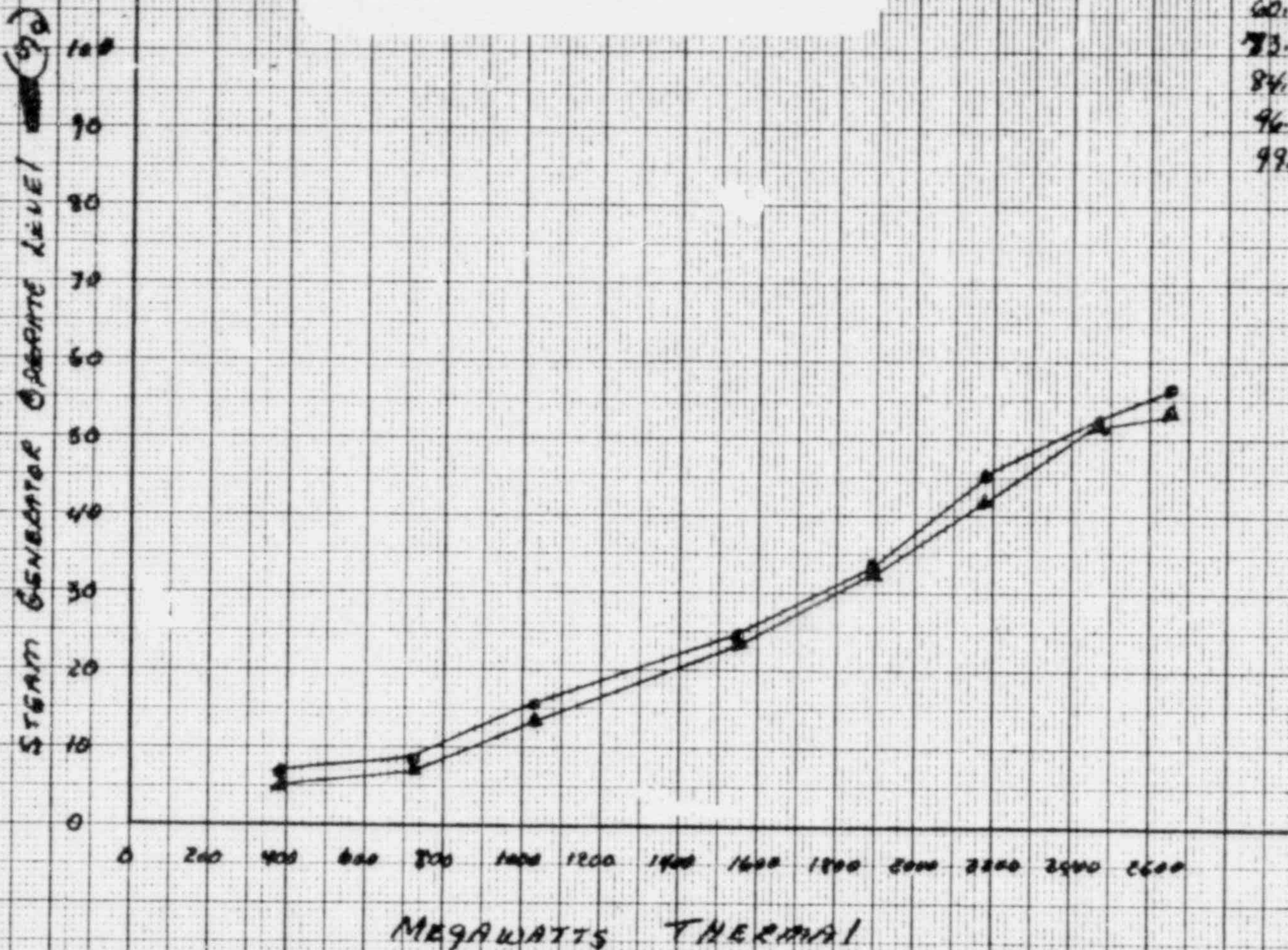
% Power	Low Low Valve A	B	FW Flow A	B
0	0 %	0 %	.0068	.0068
15 %	0 %	0 %	.781	.816
27.99 %	25.27 %	20.81 %	1.347	1.319
41.55 %	54.57 %	55.21 %	2.029	2.035
60.2 %	96.1 %	96.0 %	3.130	3.170

FIGURE 2.15- 9



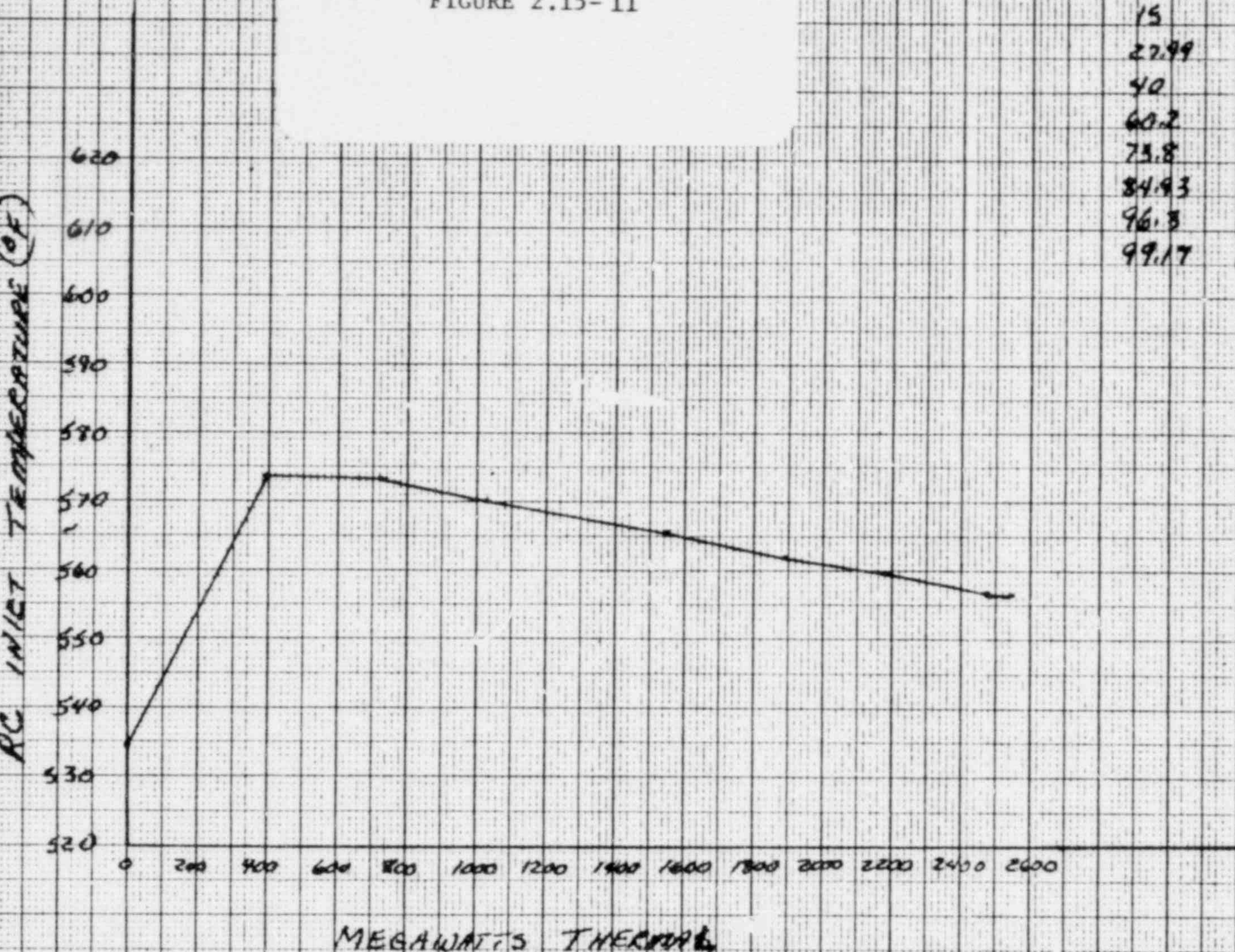
		STEAM GENERATOR LEVEL	
% POWER	MWTH	S.G. A ⊙	S.G. B Δ
0	0	23.24"	26.75"
15	385.2	23.3	26.75"
27.99	719.04	35"	33.6"
40	1022.2	50.3"	53.9"
60.2	1546	77.5"	83.4"
73.8	1895	101.2"	106.1"
84.93	2181	120.5"	125"
96.3	2472.9	139.8	143.9
99.17	2547	142.6	147.6

FIGURE 2.15-10



		STM. GEN. % OPER. LEVEL	
% POWER	MWTH	A ●	B △
0	0	0	0
15	383.2	6.74	5.06
27.94	719.04	8.3	7.6
40	1027.2	15.4	13.2
60.2	1546	24.6	24.3
73.8	1895	33.6	33.4
84.93	2181	45	41.3
96.3	2472.9	52.6	52.2
99.17	2547	56.5	53.6

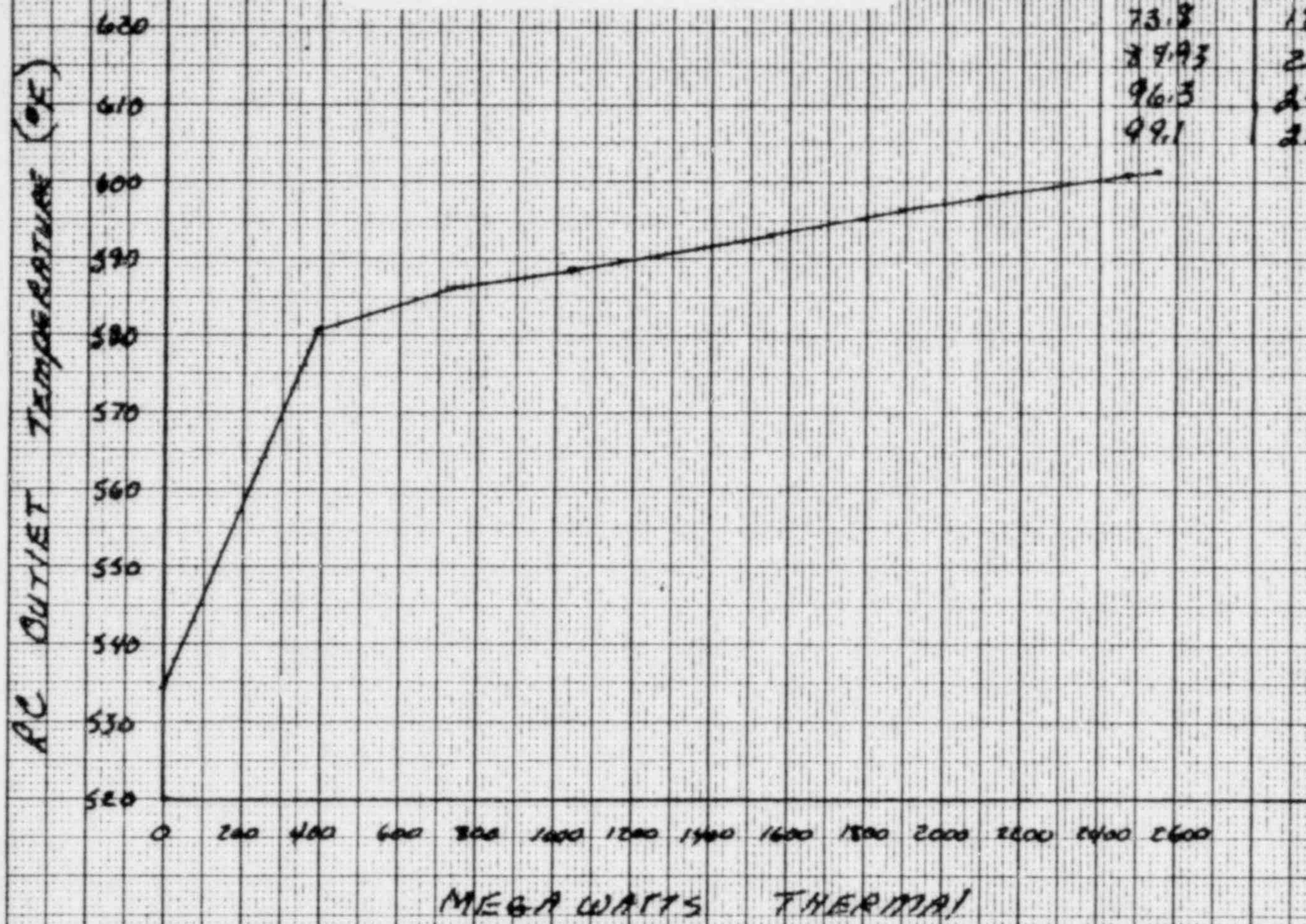
FIGURE 2.15- 11



% Power	MWTH	RC INLET TEMP A	B	AVE
0	0	534.6	534.6	534.6
15	385.2	572.5	574.45	573.5
27.99	719.04	572.2	574.1	573.2
40	1027.2	569.6	570.4	570
60.2	1546	565.2	565.1	565.1
73.8	1895	552.1	562	562.0
84.93	2181	559.6	559.7	559.6
96.3	2472.9	556.9	557.0	556.9
99.17	2547	555.7	556.9	556.3

FIGURE 2.15-12

% POWER	MWTH	RC OUTLET TEMP A	RC OUTLET TEMP B	RC OUTLET TEMP AVE
0	0	534.2	534.9	534.5
15	385.2	580.7	580.8	580.8
27.99	719.04	585.6	586.2	585.9
40	1027.2	588.5	538.5	588.5
60.2	1546	593.5	593.1	593.3
73.8	1895	596.5	595.8	596.2
84.93	2181	598.3	597.7	598.0
96.3	2472.9	600.8	600.9	600.9
99.1	2547	601	601.3	601.2



2.16 UNIT LOAD STEADY STATE TEST

2.16.1 PURPOSE

A. The purpose of the Unit Load Steady State Test was:

1) To measure the Reactor Coolant System and Steam Generator steady state parameters as a function of reactor power, and

2) To verify the ability of the Integrated Control System (ICS) to control the Nuclear Steam Supply under unit load steady state conditions.

B. Specific purposes of this test were:

1) Measurements of Reactor Coolant System and OTSG parameters with two, three, and four reactor coolant pumps operating between 0 and 100% reactor power.

2) Determination as to whether adjustments are required for the ICS between 0 and 100% reactor power.

3) Measurement of primary and secondary system operating parameters for comparison with future performance data.

C. The acceptance criteria specified by the Unit Load Steady State test are:

1) The unit operates at power with no unstable situations or oscillations of the controlling parameters,

2) Specific steady state parameters, as given in Figures 2.16-1 through 2.16-7, are within their respective minimum and maximum limits as a function of power.

2.16.2 TEST METHOD

A. The data parameters were recorded during steady state conditions at power levels required by the power escalation controlling procedure. The power levels used were 0, 15, 25, 40, 60, 75, 85, 95, and 100% of reactor power.

Steady state conditions were defined as follows:

Tave equal to 579 ± 2 degrees

Reactor Coolant System pressure equal to 2155 ± 50 psig.

Reactor power level stable within ± 2 percent.

B. The data taken was averaged and plotted for comparison with the design limits required by the acceptance criteria (See Figures 2.16-1 through 2.16-5). Parameter stability during steady state operation was measured in terms of the deviation from the average value of the parameter during the test.

C. Steady state data was taken with two and three reactor coolant pumps running and the units ability to maintain reactor coolant average temperature versus reactor power level was checked (See Figures 2.16-6 and 2.16-7).

2.16.3 RESULTS AND EVALUATION

A. The measured operating parameters of the primary and secondary systems were compared to the design values by plotting the averaged values against the theoretical values versus reactor power levels on Figures 2.16-1 through 2.16-7. As can be seen in Figure 2.16-5, the total feedwater flow was higher than expected at the 25% and 40% power levels. It has been concluded that this is a characteristic of the plant and that no changes are warranted since the plant will not normally be operated in this power range. Figure 2.16-1 shows that the Steam Generator outlet pressure is being controlled at approximately 885 psig. This is due to the controlling pressure indicator being located adjacent to the OTSG outlet instead of at the turbine header. Thus the line losses between the OTSG and turbine header are not compensated for. All other parameters plotted were within specified limits as can be seen in the remaining figures.

B. The total feedwater flow plotted on Figure 2.16-5, was acceptable since it was within the limits previously experienced at their related power levels with four pumps running. All other parameters plotted were within specified limits as can be seen in the remaining figures.

C. The unit stability was measured by determining the deviation of selected parameters from their respective average value. Table 2.16-1 is a listing of the absolute deviation of these variables at various power levels. From this analysis, it was concluded that flows, temperatures, and pressures were stable within the following limits:

RC Temperatures ± 1°F of the average value
Feedwater Temperatures ± 4°F of the average value
Coolant Pressure ± 50 psig of the average value
Steam Pressure ± 5 psig of the average value
RCS & Feedwater Flow ± 1% of the average value

2.16.4 CONCLUSIONS

The average of the measured unit parameters were within specified limits with the exception of the steam generator outlet pressure and the feedwater flow at the lower power levels. Analysis of the unit parameters indicates that all are relatively stable over the entire power range especially at the 100% reactor power level.

TABLE 2.16-1

UNIT PARAMETER STABILITY FOR ANO UNIT 1

PARAMETER	15% AVG.VAL.	15% DEV.	40% AVG	40% DEV	75% AVG	75% DEV	100% AVG	100% DEV
T-Avg, ° F	578.94	.38	579.16	.37	579.03	.77	578.55	.08
RC Inlet Temp Loop A, ° F	574.74	.35	569.1	.5	561.5	.2	556.7	.4
RC Inlet Temp Loop B, ° F	576.55	.44	570.7	.5	561.6	.1	555.7	.2
RC Outlet Temp Loop A, ° F	583.06	.44	588.9	.3	595.2	.4	601.3	.2
RC Outlet Temp Loop B, ° F	583.37	.43	588.56	.36	593.8	.2	601.6	.2
OTSG "A" Outlet Psr, PSIG	884.14	.86	896.14	4.14	887.8	2.2	890.7	2.0
OTSG "B" Outlet Psr, PSIG	884.0	1.00	884.57	4.57	890.04	1.6	895.0	1.0
Total Feedwater Flow, 10^6 Lb/Hr	----	---	4.335	.047	8.091	.05[illegible]	10.746	.049
Pressurizer Level, "H_2O	190.78	1.67	193.34	.86	197.9	.1	193.7	.5
RC Pressure, PSIG	2155	44.5	2155	39.2	2155	17.1	2155	22.7
OTSG "A" Op. Level, %	7.9	.1	14.72	.28	36.1	.3	62.4	.5
OTSG "B" Op. Level, %	4.54	.06	13.94	.33	33.7	.1	52.9	.5
OTSG "A" Startup Level, "H_2O	22.69	.21	52.44	1.36	107.2	1.2	148.7	.6
OTSG "B" Startup Level, "H_2O	21.31	.19	55.76	.94	105.8	3.7	149.2	1.3
Turbine Steam Inlet Psr.,PSIG	875.29	.71	873.57	2.57	877.5	2.5	869.3	1.3
OTSG "A" Outlet Steam Temp, Degrees F	583.69	.39	588.78	.52	596.4	.1	594.4	.1
OTSG "B" Outlet Steam Temp, Degrees F	582.83	.43	589.11	.49	597.4	.1	593.7	.1
Feedwater Temp Loop A Degrees F	289.46	3.64	381.01	.39	433.6	.3	459.1	.5
Feedwater Temp Loop B Degrees F	289.70	3.40	381.2	.20	434.3	1.0	459.7	.4
RP Channel A RC Flow Loop A, MLB/Hr.	65.29	.25	66.47	.27	67.28	.42	68.33	.33
RP Channel A RC Flow Loop B, MLB/Hr.	64.87	.26	67.47	.61	67.62	.31	68.85	.26

FIGURES 2.16-1-7

LEGEND

□	Loop A
○	Loop B
△	Combined Loops A and B
----	Max. and Min. Limits
——	Expected

FIGURE 2.16-1

STEAM GENERATOR OUTLET PRESSURE

Vs.

NUCLEAR STEAM SUPPLY SYSTEM POWER

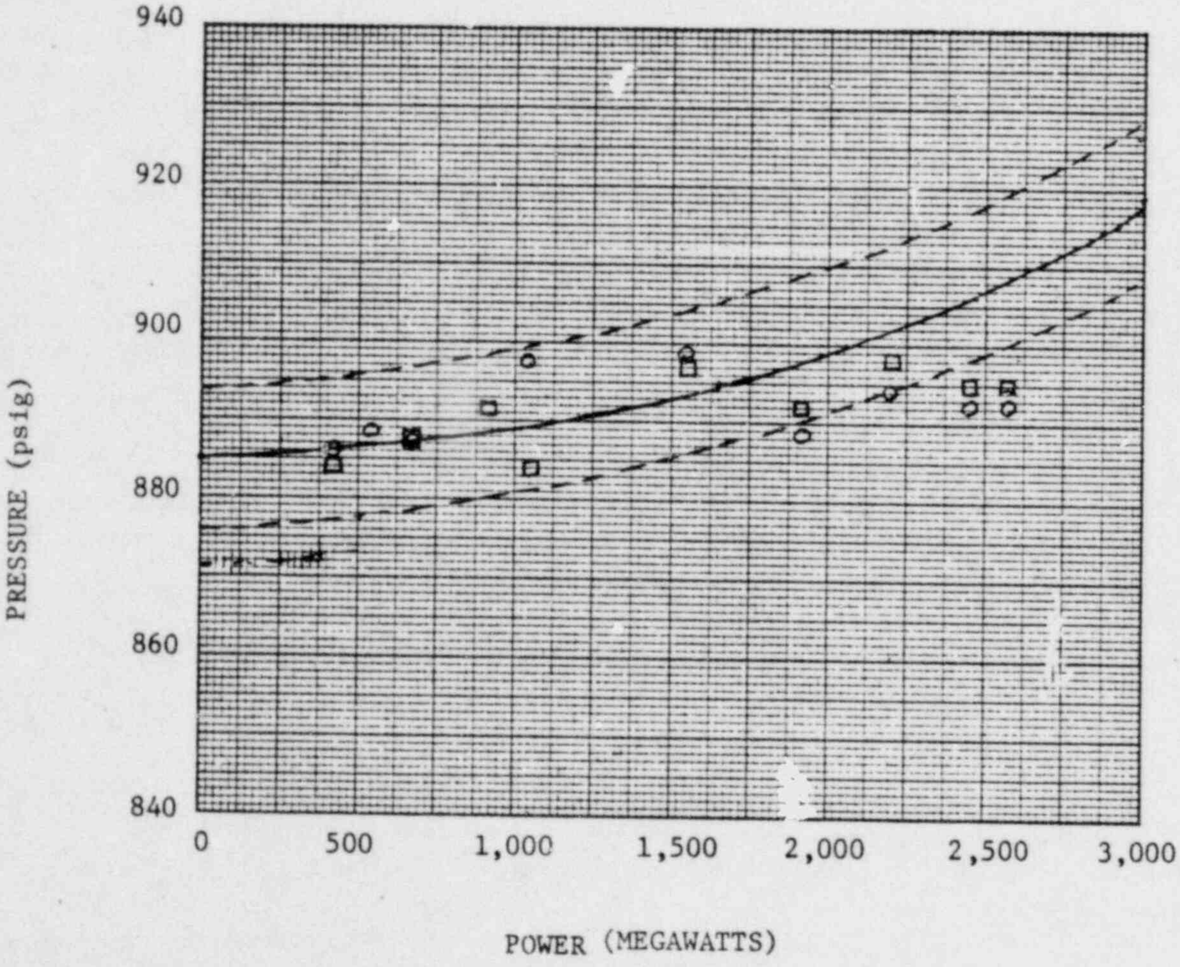


FIGURE 2.16-2

STEAM GENERATOR STARTUP RANGE LEVEL PRESSURE DROP

Vs.

NUCLEAR STEAM SUPPLY SYSTEM POWER

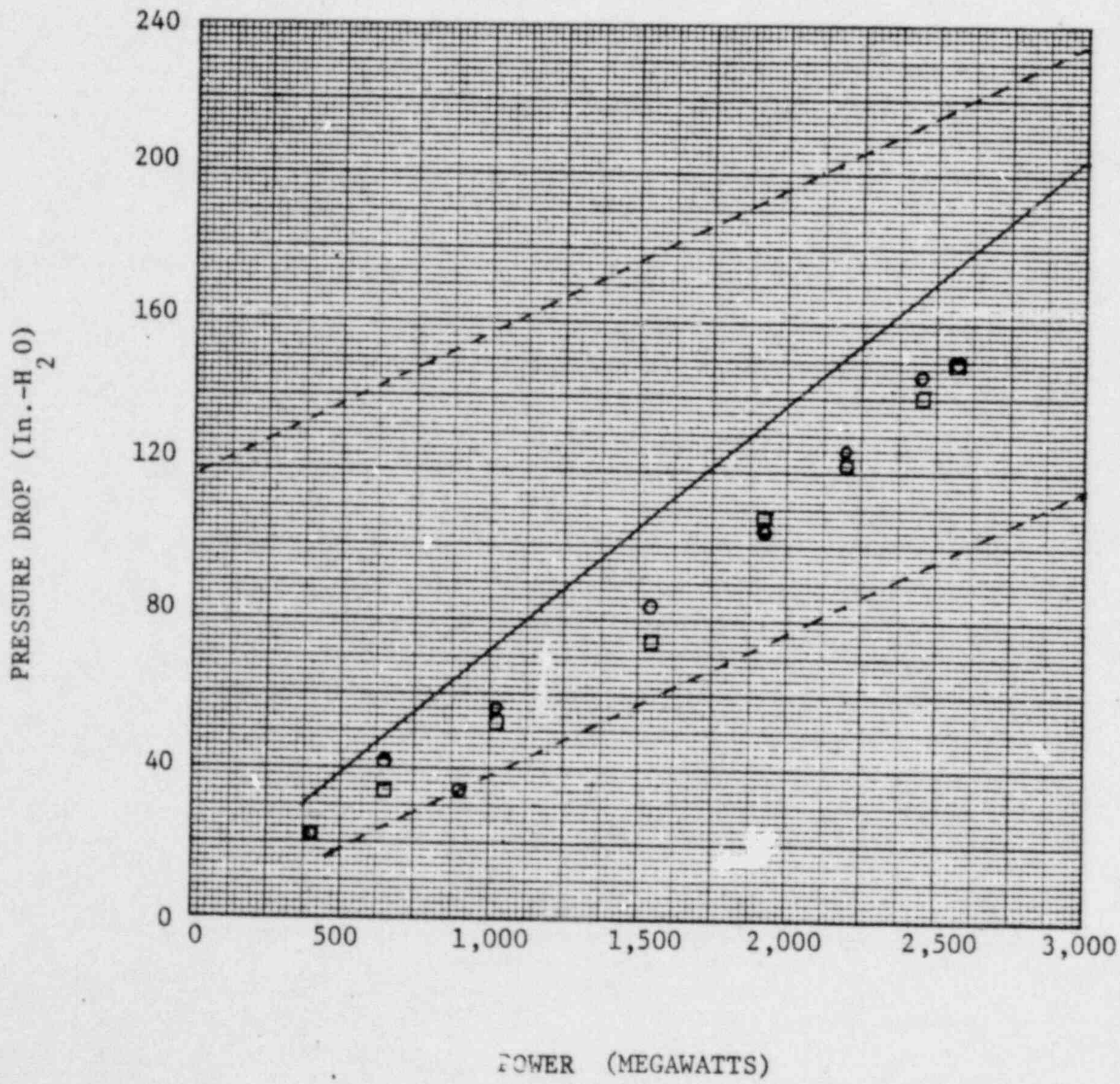


FIGURE 2.16-3

TOTAL FEEDWATER FLOW RATE

Vs.

NUCLEAR STEAM SUPPLY SYSTEM POWER

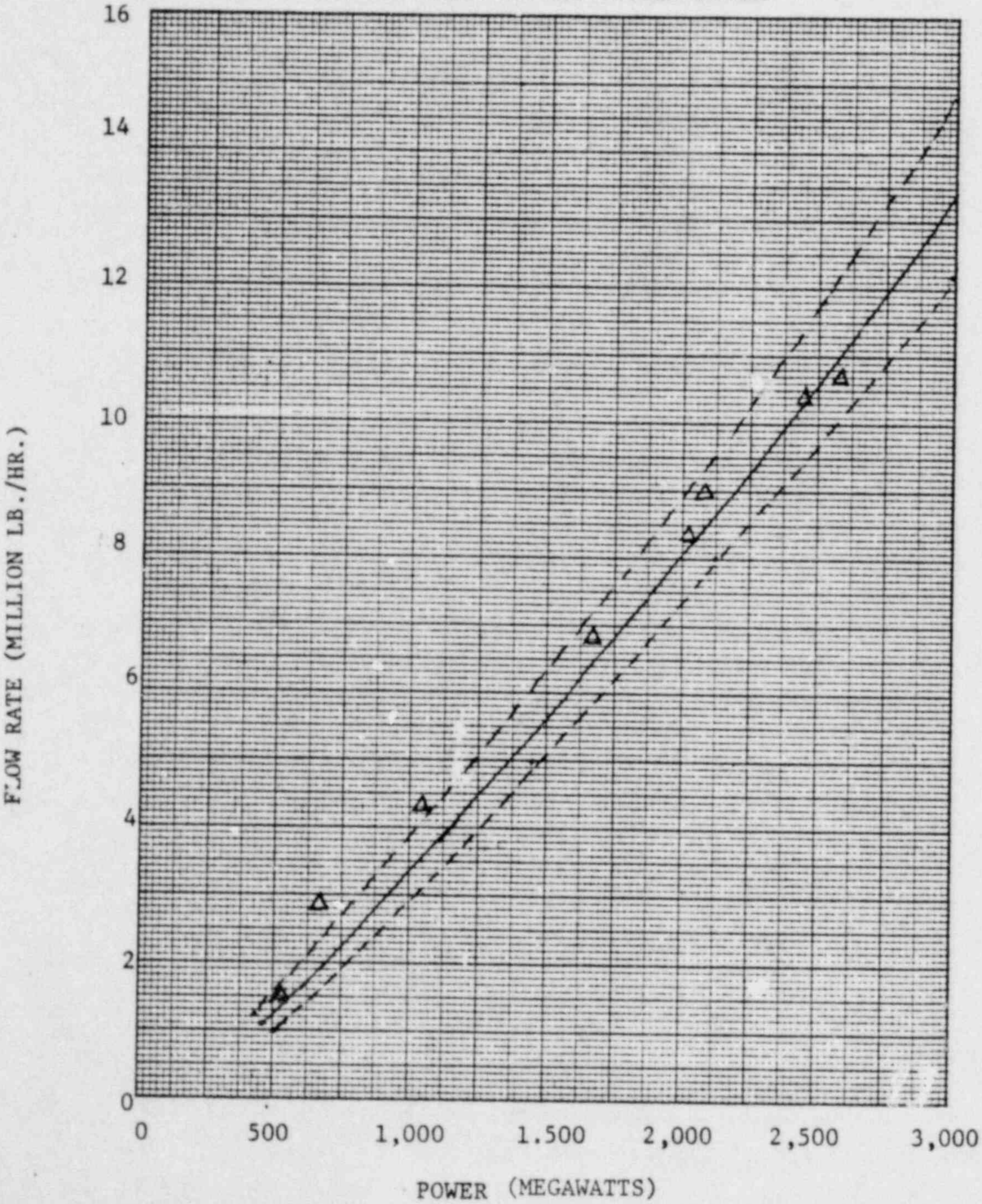


FIGURE 2.16-4

FEEDWATER TEMPERATURE

Vs.

NUCLEAR STEAM SUPPLY SYSTEM POWER

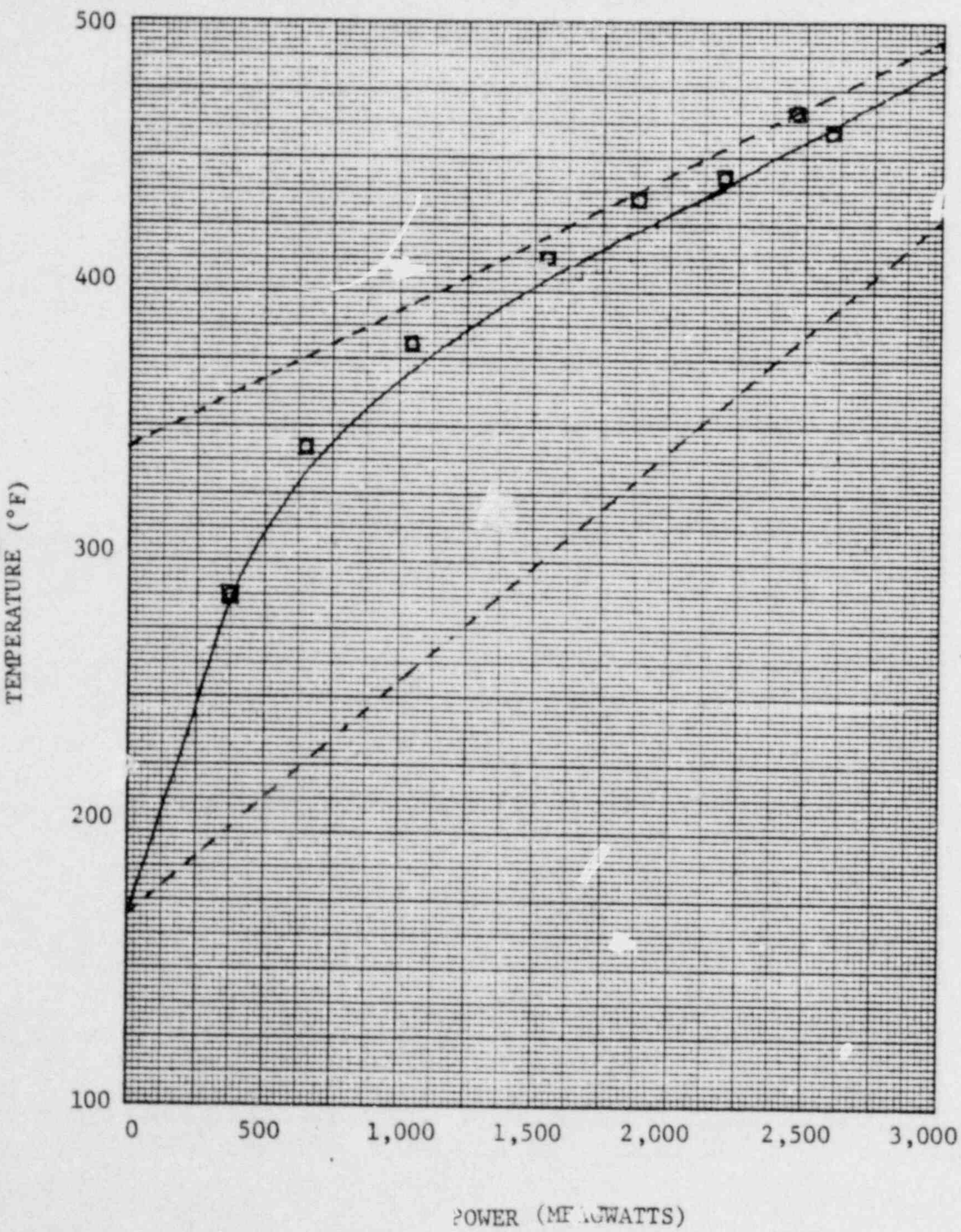


FIGURE 2.16-5

REACTOR COOLANT SYSTEM TEMPERATURE

Vs.

NUCLEAR STEAM SUPPLY SYSTEM POWER

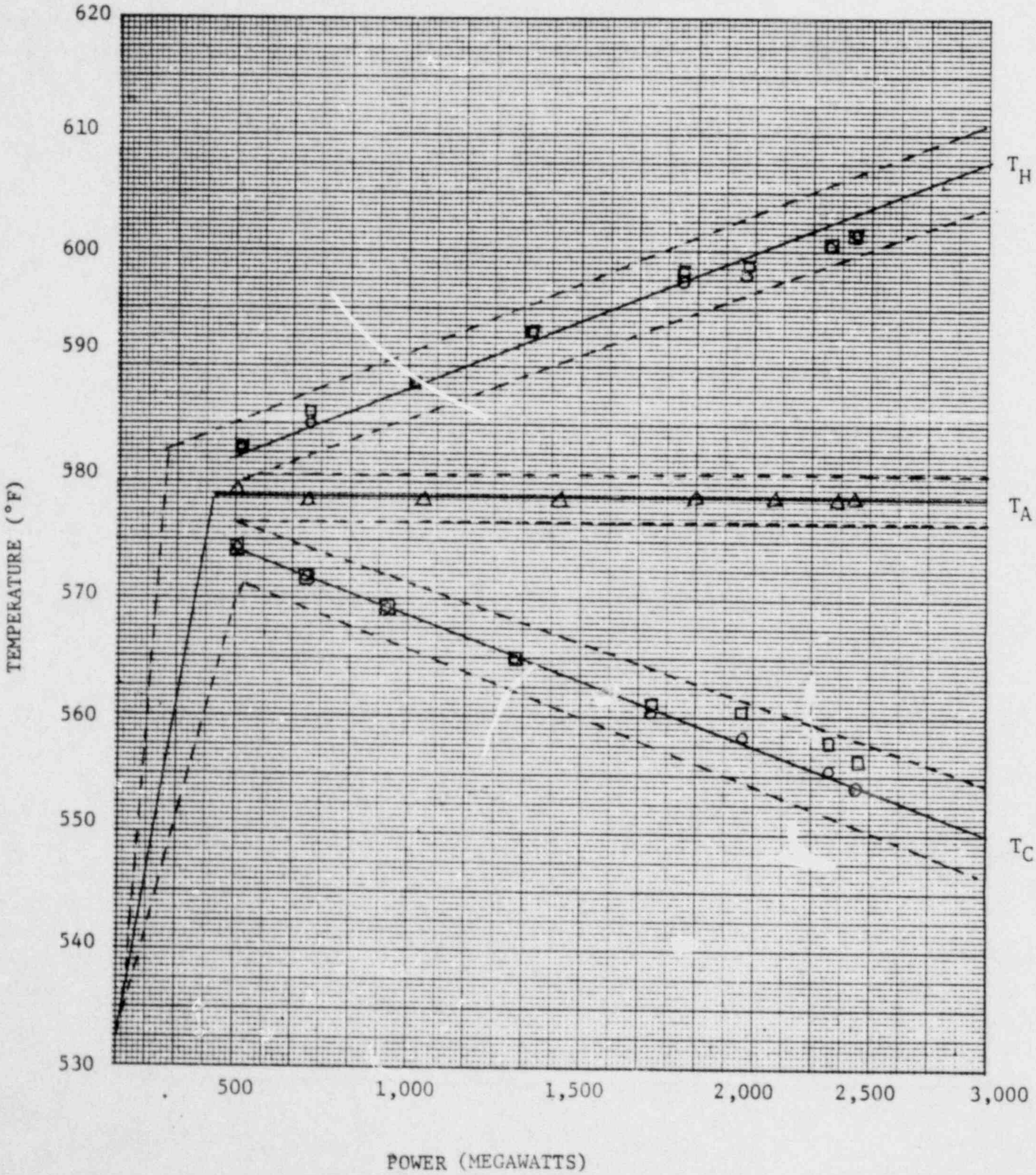


FIGURE 2.16-6

REACTOR COOLANT SYSTEM TEMPERATURE

Vs.

NUCLEAR STEAM SUPPLY SYSTEM POWER

(TWO PUMP OPERATION)

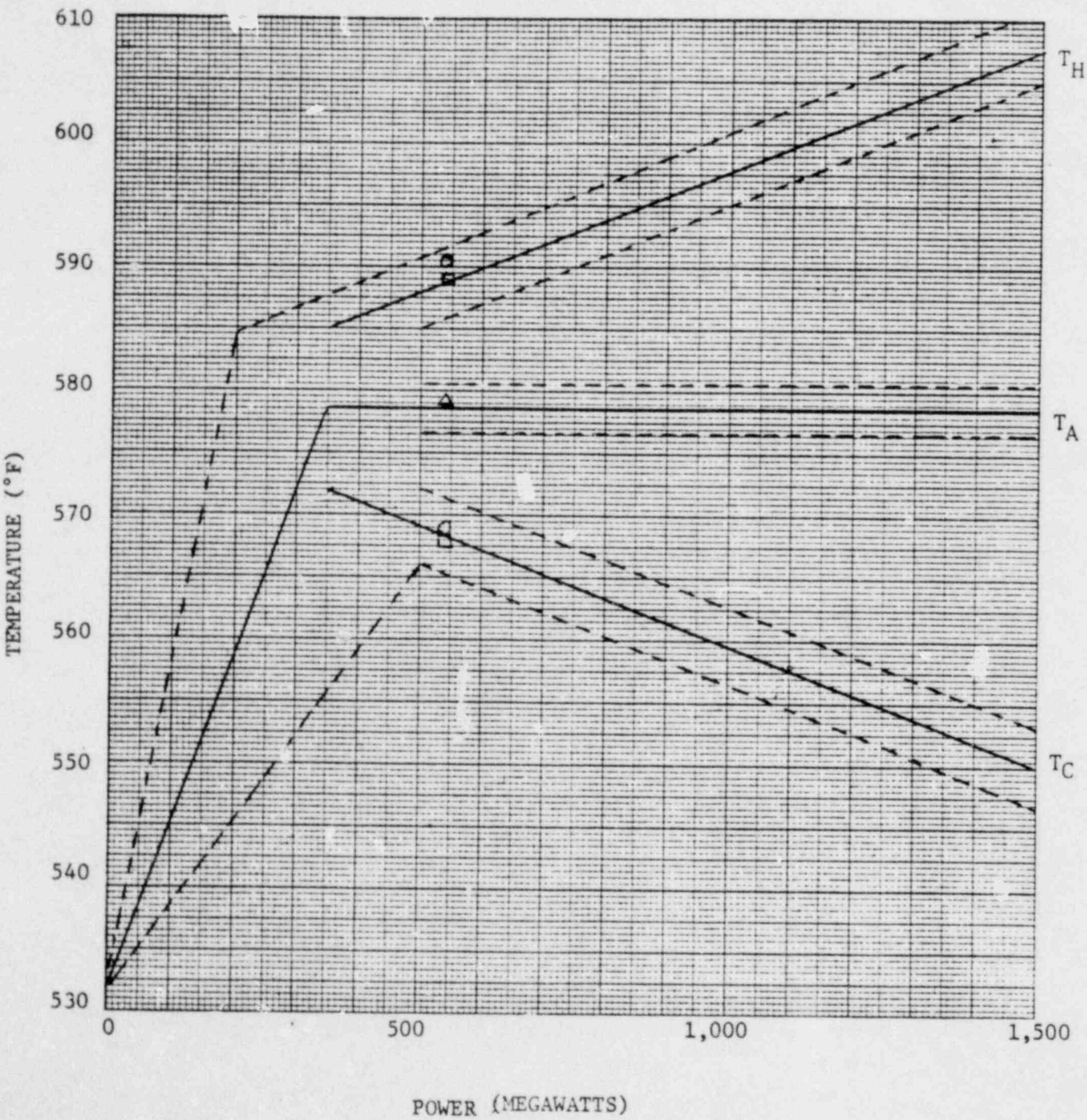


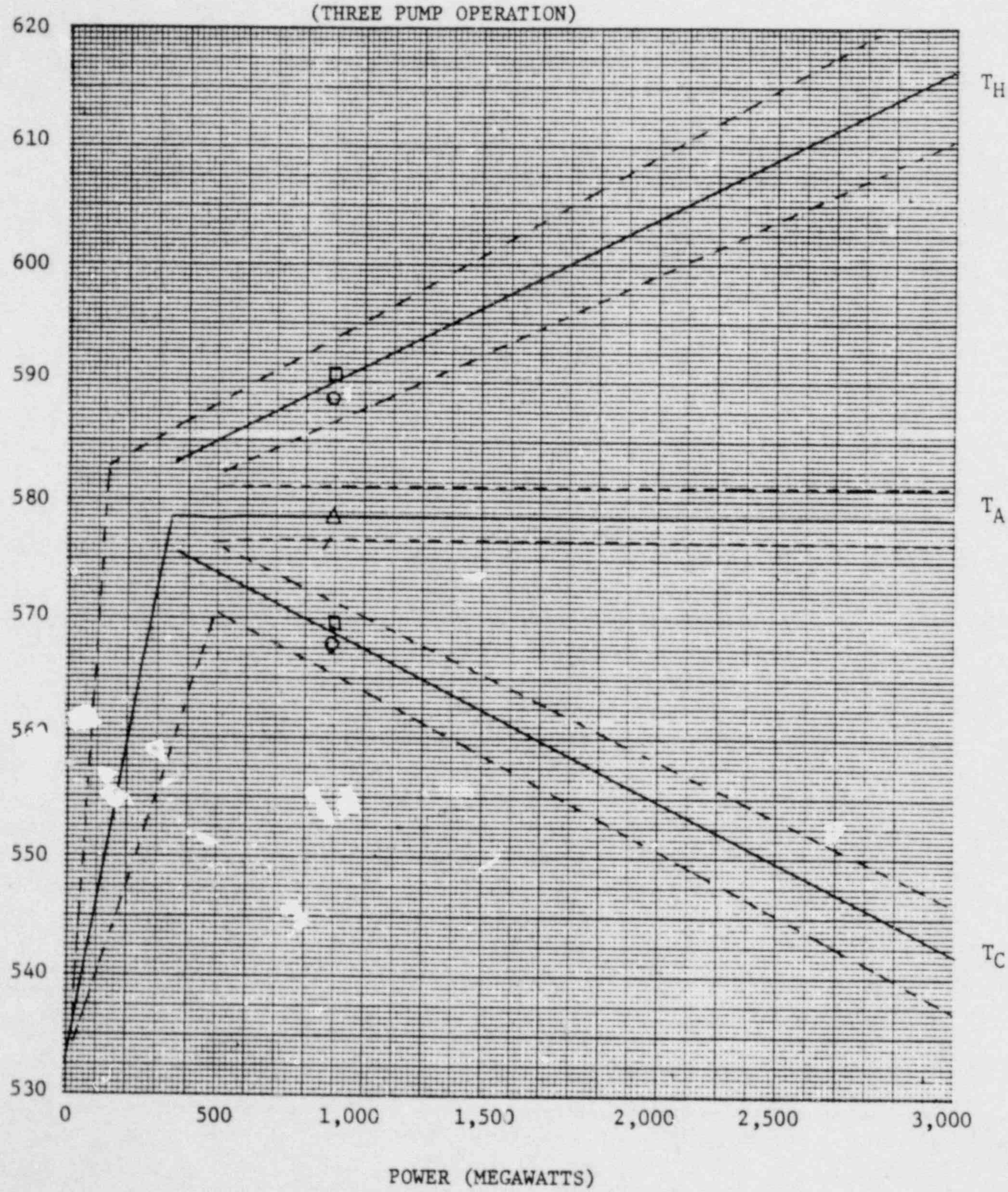
FIGURE 2.16-7

REACTOR COOLANT SYSTEM TEMPERATURE

Vs.

NUCLEAR STEAM SUPPLY SYSTEM POWER

(THREE PUMP OPERATION)



2.17 UNIT LOAD TRANSIENT TEST

2.17.1 PURPOSE

The purposes of the Unit Load Transient Test are listed below:

(a) To observe and record certain primary and secondary system parameters during transient conditions imposed on the unit.

(b) To demonstrate the ability of the Integrated Control System to maintain control of the unit during transients at different transient rates and in different modes of control.

(c) To aid in the tuning of the Integrated Control System.

The acceptance criteria for this test included:

(a) Each transient must be completed without exceeding the limits of the Unit 1 Technical Specifications or Plant Limits and Precautions.

(b) Each transient must be completed without causing the Reactor Protective System to actuate.

2.17.2 TEST METHOD

During power escalation testing the Unit Load Transient Test was performed at several different power plateaus: 40, 75 and 100%FP. Table 2.17-1 gives a general summary of all transients required by the Unit Load Transient Test. The transients were performed with the Integrated Control System in five separate modes of control:

(a) The fully integrated control mode
(b) The turbine-following control mode
(c) The reactor/steam generator - following control mode
(d) EHC operator/auto
(e) Feedwater demand stations in manual

Transients in each mode of Integrated Control System control were performed twice at the 40% FP plateau; at a ramp rate less than 5 percent per minute and at a 5 percent per minute ramp rate. At the 75% FP plateau, an initial ramp of 3 percent per minute was used during the integrated mode. All other transients at 75% FP were at a rate of 5 percent per minute. One additional transient was performed at the 75 percent full power plateau--a decrease to 30% FP at a rate of 5 percent per minute in the fully Integrated Mode. Testing at the 100% FP level consisted of 10% FP transients (increasing and decreasing) in each mode of control. An Integrated mode transient from 850 MWE down to approximately 450 MWE was performed at a rate of 10 percent per minute. An increase back to 82% FP was then performed at the same rate. Two and three Reactor Coolant Pump testing was performed at various power levels, utilizing the Integrated mode of control.

Throughout this test, Integrated Control System tuning was performed when selected unit parameters indicated that tuning was necessary in order to optimize the transient response of the Integrated Control System. During all transients, the variation of pertinent primary and secondary system parameters during negative and positive power ramps were monitored and recorded.

The technique of inducing each transient, in most cases, was to decrease reactor power from the test plateau to a predetermined lower power level, then after establishing steady-state conditions, reactor power was increased to the test plateau.

2.17.3 RESULTS AND EVALUATION

A. Integrated Control System Transient Test at 40% FP

When the 40% FP level had been established, the Unit Load Transient Test was conducted to evaluate the ability of the Integrated Control System to accomplish smooth 10% FP negative and positive changes in power level.

The data taken during each transient were analyzed and the behavior of the unit during negative and positive ramps in power level is presented. All acceptance criteria were met.

B. Integrated Control System Transient Test at 75% FP

The Unit Load Transient Test was conducted at the 75% FP level to evaluate the capability of the ICS to properly control 10% transients and a 45% transient.

C. Integrated Control System Transient Test at 100% FP

The Unit Load Transient Test was conducted at the 100% FP level to evaluate the capability of the Integrated Control System to handle small transients to and from 100% FP. A transient test was also run at 10% per minute down to 450 MWE and back up to 82% FP using the Integrated Mode. All tests at 100% met applicable acceptance criteria.

D. Two and Three Reactor Coolant Pump Testing at Power

Several tests were accomplished with less than four Reactor Coolant Pumps running. Transients caused by tripping of pump(s) and restarting of pumps were recorded and evaluated. Escalations to 75% with three pumps on and to 49% with two pumps on were accomplished. All acceptance criteria were verified.

E. Table 2.17-1 is a summary of all transients showing the deviation in Tave and Turbine Header Pressure during the power transient.

2.17.4 CONCLUSIONS

From analysis of all test data taken to date, the following conclusions may be made from all sections of the Unit Load Transient Test.

(a) All transients were performed without exceeding the limits of the Technical Specifications or Plant Limits and Precautions.

(b) All transients were completed without causing the Reactor Protective System to actuate.

(c) The Integrated Control System has been tuned satisfactorily for all modes of plant operation.

TABLE 2.

UNIT LOAD TRANSIENT TEST SUMMARY

% FP Based on 902 MWe — Set Point 579° F — Set Point 885psig

TRANSIENT NUMBER	ICS MODE OF OPERATION	% FP CHANGE ___% TO ___%	RATE, %/MIN. AVERAGE	MAXIMUM T AVG. DEVIATION °F		MAXIMUM THP DEVIATION		REMARKS
				ABOVE	BELOW	ABOVE	BELOW	
1	Integrated Mode	38.47% - 29.4%	2.57	.41	1.46	31.7	6	10-01-74 08:52
2	Integrated Mode	29.5% - 38.8%	2.3	.93	1.93	4.2	34	10-01-74 09:58
3	SG/Rx In Manual	38.4% - 24.6%	3.08	.43	1.09	0	9.9	10-01-74 10:43
4	SG/Rx In Manual	24.2% - 36.6%	2.61	1.11	.31	1.4	6.2	10-01-74 11:33
5	Turbine In Manual	38.77%- 26.0%	2.22	.56	.15	43.3	5.7	10-01-74 14:48
6	Turbine In Manual	26.0% - 39.55%	1.93	.14	.94	6.3	5.2	10-01-74 15:53
7	Rx Demand In Manual	39.5% - 24.0%	1.88	1.71	1.71	1.2	7.0	10-01-74 16:33
8	Rx Demand In Manual	26.8% - 37.6%	2.67	3.19	1.74	3.3	6.4	10-01-74 18:53
9	Rx Demand In Manual	37.48%-26.73%	4.0	.29	2.77	1.5	8.0	10-01-74 20:03
10	Rx Demand In Manual	25.16%-39.62%	3.40	3.01	1.64	4.7	11.4	10-01-74 20:44
11	Rx Demand In Manual	38.81%- 26.1%	2.42	.89	2.37	1.3	4.7	10-02-74 07:37
12	Rx Demand In Manual	25.94%-38.33%	2.48	2.3	1.52	2.8	5.7	10-02-74 08:43
13	Integrated Mode	20.76%- 22.0%	0	.36	1.93	30.3	0	11-23-74 22:01
14	Integrated Mode	23.36%- 38.34%	1.362	.25	.75	23.1	0	11-24-74 05:13
15	Integrated Mode	37.44%-45.06%	1.3	.29	.5	17.5	3.2	11-24-74 06:44

TABLE 2.17-1

UNIT LOAD TRANSIENT TEST SUMMARY

% FP Based on 902 MWe

TRANSIENT NUMBER	ICS MODE OF OPERATION	% FP CHANGE ___% TO ___%	RATE, %/MIN. AVERAGE	Set Point 579° F MAXIMUM T AVG. DEVIATION °F ABOVE	BELOW	Set Point 885psig MAXIMUM THP DEVIATION ABOVE	BELOW	REMARKS
16	Integrated Mode	44.01%- 44.9%	0	0	.62	22.5	1.1	11-24-74 07:00
17	Integrated Mode	44.67%-44.64%	0	0	.52	14.7	.7	11-24-74 11:50
18	Integrated Mode	45.23%-45.23%	0	0	.59	13.1	.4	11-24-74 14:15
19	Integrated Mode	45.01%-24.8%	(.8067)	3.06	2.93	16.3	0	11-24-74 15:15
20	Integrated Mode	23.56%-23.57%	0	.86	1.26	18.2	1.5	11-25-74 02:31
21	Stop RCP "D"	22% -22 %	0	.99	1.69	0	19	11-29-74 0113-0123
22	3 RCP	22.67%-32.25%	1.42	.29	.63	0	18.2	11-29-74 0128-0140
23	Integrated Mode	69.91%-25.03%	.561	.49	1.91	0	62.4	11-29-74 1535-1722
24	Integrated Mode	73.2 %-63.9%	.93	0	1.59	19.7	4	12-3-74 2213-2232

TABLE 2.17-1

UNIT LOAD TRANSIENT TEST SUMMARY

% FP 902 MWe

TRANSIENT NUMBER	ICS MODE OF OPERATION	% FP CHANGE ___% TO ___%	RATE, %/MIN. AVERAGE	MAXIMUM T AVG. DEVIATION °F (Set Point 579° F) ABOVE	BELOW	MAXIMUM THP DEVIATION (Set Point 885psig) ABOVE	BELOW	REMARKS
25	Integrated Mode	75.74%-27.31%	4.5	0	4.5	37.6	5.9	12-04-74 04:38
26	Integrated Mode	96.56%-87.95%	1.43	.78	.04	16.4	0	12-10-74 18:03
27	Integrated Mode	87.42%-95.03%	1.69	.85	.57	9	0	12-10-74 18:16
28	Turbine Following	96.17%-87.87%	2.21	0	1.08	12.3	2.4	12-10-74 18:33
29	Turbine Following	87.61%-95.82%	1.56	.35	.96	14.7	0	12-10-74 18:48
30	Operation Auto	95.46%-86.53%	1.05	.16	.63	25.3	.9	12-10-74 19:08
31	Operation Auto	86.94%-96.24%	1.86	.06	.56	12.5	10.7	12-10-74 19:27
32	Rx Demand Manual	95.33%-85.96%	1.409	.37	2.57	6.5	3.9	12-10-74 19:53
33	Rx Demand Manual	85.1% -94.58%	2.23	2.38	.87	15.5	0	12-10-74 20:13
34	Rx Demand Manual	95.14%-87.36%	1.11	.53	.29	11.4	1.5	12-10-74 20:28
35	FW Demand Man.	87.97%-95.38%	2.69	.31	.74	13.8	0	12-10-74 20:53
36	Integrated Mode	97.37%-52.14%	6.23	.33	1.42	62.7	2.3	12-10-74 23:13
37	Integrated Mode	52.67%-75.66%	7.66	1.34	.59	24.0	23.0	12-10-74 23:38
38	Integrated Mode	17.67%-17.77%	0	1.1	2.12	4.2	14.9	12-14-74 12:19
39	Integrated Mode	15.44%-15.46%	0	1.48	2.85	2.6	9.9	12-14-74 13:35

2.18 VIBRATION & LOOSE PARTS MONITOR - BASELINE DATA AT POWER

2.18.1 PURPOSE

The purpose of the V&LPM Baseline Data Test was to accumulate data during power escalation that will be used for future analysis of any questionable vibration or loose parts in the Reactor Coolant System.

The baseline data taken for the test is deemed acceptable if no peaks greater than 26 db or no system trips, either vibration or loose parts, occur.

2.18.2 TEST METHOD

Baseline data for the V&LPM was taken at 0,15,40 and 75% reactor power. Due to a monitor power supply problem, baseline data at 100% FP has not been taken as of this writing. Locations monitored are the upper reactor vessel, lower reactor vessel and the steam generators.

With the Reactor Building clear of all personnel and the reactor at a specified power level, averages X-Y plots were taken of the background noises at the frequencies of 200 Hz, 2 KHz, and 20 KHz for all channels. This signature analysis constitutes the baseline data.

2.18.3 RESULTS AND EVALUATION

All signatures taken were acceptable and were stored for future analysis. The 100% reactor power baseline data will be taken when the monitor becomes available.

2.18.4 CONCLUSION

Baseline data for the Reactor Coolant System is now available for analysis and comparison with future signatures of questionable origin.

No unacceptable peaks were noted during the testing either due to vibration or loose parts.

2.19 CORE POWER DISTRIBUTION

2.19.1 PURPOSE

The objective of the core power distribution test was to measure the power distribution of the Reactor core at the major power plateaus during the initial power escalation in order to verify that the DNBR, linear heat rate, quadrant power tilt and power peaking factors did not exceed allowable limits.

The specific limits placed on the measured parameters were as follows:

a) The minimum DNBR must be greater than 1.55.
b) The maximum linear heat rate must be less than 17.2 KW/ft.
c) The quadrant power tilt must not exceed 4 percent.
d) Above 40% FP the three largest measured radial power peaking factors should be no greater than 1.05 times the predicted values.
e) Above 40% FP the three largest measured total power peaking factors should be no greater than 1.075 times the predicted values.

2.19.2 TEST METHOD

Equilibrium conditions were established at each test power plateau ensuring that xenon was in three-dimensional equilibrium with no APSR motion and minimal power fluctuations and/or controlling rod group motion. The incore monitoring system and the plant computer were used for data collection. Plant computer calculated core power distributions were checked by hand calculations in all cases.

2.19.3 RESULTS AND EVALUATION

A summary of the test results is given in Table 2.19-1. This table indicates that the minimum DNBR at rated power was 3.31 which is well above the lower limit of 1.55. The worst case linear heat rate at 100% FP was 14.5 KW/ft which is significantly less than the LOCA limit of 17.2 KW/ft. The measured radial and total power peaking factors were all within the acceptance criteria.

The maximum measured quadrant tilt from the incore monitors was found to be less than 1% above 40% FP. The axial imbalances were obtained intentionally to match the conditions for which the predicted values of power peaking factors were calculated.

The minimum DNBR is plotted as a function of power level in Figure 2.19-1. This figure shows how the measured DNBR was extrapolated to the next power plateau during power escalation to assure that no DNBR limit would be exceeded. All extrapolated DNBRs were no smaller than 3.28 at the 105.5% FP overpower trip.

Figure 2.19-2 shows the core grid/SPND string correlation for the core locations used to measure the radial and total power peaking factors on a symmetric 1/8 core basis. The results of the power distribution measurements are tabulated in Figures 2.19-3 & 4 for the 40% FP plateau, 2.19-5 & 6 for the 75% FP plateau and in figures 2.19-7 & 8 for the 100% FP plateau. These figures indicate that the predicted power peaking factors are in good agreement with measured values. All measured peaking factors were well within acceptance limits. No comparisons were made at 15% due to the low flux signals and high statistical errors.

2.19.4 CONCLUSIONS

Measured DNBR's, Linear Heat Rates and Quadrant Tilts verified that the core can be operated at rated power with no danger of exceeding Technical Specification or ECCS criteria.

All DNBRs were greater than the 1.55 minimum; all linear heat rates were less than the 17.2 KW/ft LOCA limit and all quadrant tilts were well below the 4% Technical Specification limit.

The measured power distributions verified the predicted distributions and the three largest radial and total peaking factors were within the limitation of being within 5% and 7.5% of predicted respectively.

The measured DNBR and Linear Heat Rates verified that the Reactor Protection System is sufficient to protect the core against exceeding DNBR or maximum linear heat rate limits.

Table 2.19-1

SUMMARY OF RESULTS

DATE TIME	6/20/74 1930	9/26/74 2000	11/21/74 0500	12/9/74 2200
POWER LEVEL (%)	15.2	41.2	75	100
GROUP 1-5 %W/D)	100	100	100	100
GROUP 6 (%W/D)	78	74	71.1	94.0
GROUP 7 (%W/D)	0	0	0	17.4
GROUP 8 (%W/D)	23.5	30.0	30.7	13.1
CORE BURNUP (EFPD)	0.67	3.97	12.2	20.2
BORON CONCENTRATION (ppmb)	1290	1144	1074	1079
AXIAL IMBALANCE (%FP)	-1.6	-5.8	-14.4	-8.2
MAX QUADRANT PWR TILT (%)	-2.09	+.55	-.67	-.75
DNBR*	24.9	9.52	4.80	3.31
LHR*	2.593	6.518	12.1	14.45
MAX RADIAL PWR PEAK	1.39	1.38	1.41	1.39
MAX TOTAL PWR PEAK	1.77	1.82	1.88	1.82
MAX PEAK AT CORE GRID/LEVEL	B-8/4	H-5/3	H-5/3	M-10/5
WORST CASE LHR (HAND CALC)	2.11	5.876	11.45	14.5
WORST CASE LHR (EXTRAPOLATED**)	14.35	14.98	15.57	14.7
WORST CASE DNBR (EXTRAPOLATED***)	3.30	3.95	3.85	3.28
EQUILIBRIUM XENON	No	Yes,3-D	Yes,3-D	Yes,3-D

* Computer measured DNBR is reduced by 0.68 to correct for calculational uncertainties. Computer measured LHR is multiplied by 1.417 to correct for calculational uncertainties.

** Extrapolated to 102% (LOCA Limit).

*** Extrapolated to 105.5% (trip setpoint).

FIGURE 2.19-1

HOT CHANNEL MINIMUM DNBR VS. CORE POWER LEVEL

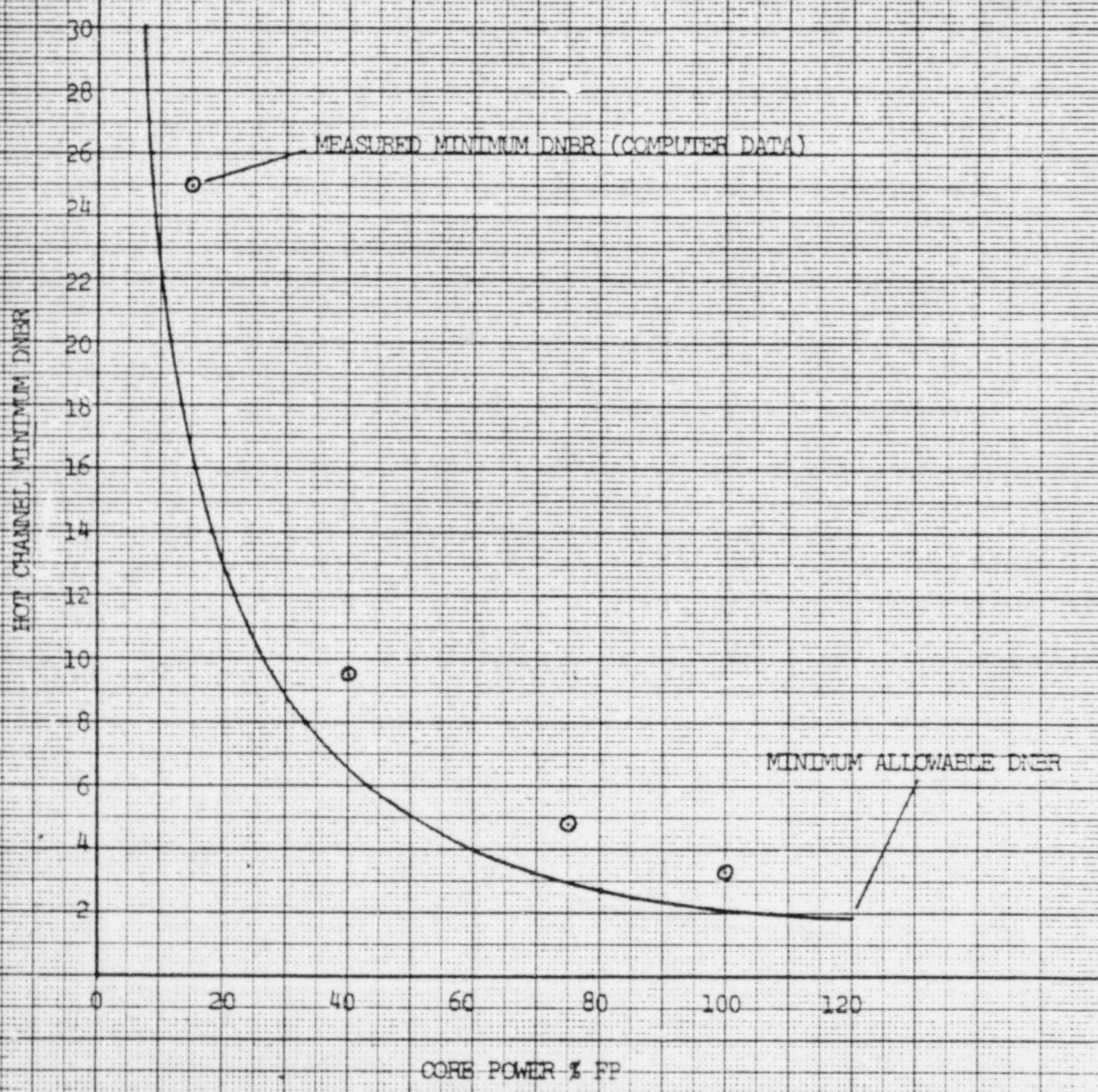


FIGURE 2.19-2

SYMMETRIC 1/8 CORE SPND STRING/
CORE GRID CROSS-REFERENCE

1 H-8	2 H-9	4 F-8	10 H-5	14 N-8	21 H-13	30 B-8	37 H-1
	3 G-9	6 F-7	5 E-9	20 K-12	29 C-9	31 B-7	45 R-7
		12 L-6	17 M-10	27 D-10	28 C-10	44 P-6	46 R-10
			26 E-11	33 D-5	42 O-5	49 M-14	
				41 N-4	48 O-12	51 D-14	
					52 C-13		

1 — SPND STRING NUMBER
H-8 — CORE GRID

FIGURE 2.19-3

COMPARISON OF PREDICTED AND CALCULATED STEADY STATE RELATIVE RADIAL POWER DISTRIBUTION AT 40 % FP, EQUILIBRIUM XENON

Measurement Conditions

Control Rod Group Positions
- Gps 1-4 100 % wd
- Gp 5 100 % wd
- Gp 6 74 % wd
- Gp 7 0 % wd
- Gp 8 30 % wd

Core Power Level 41.2 % FP
Boron Concentration 1154 ppm
Core Burnup 3.97 EFPD
Axial Imbalance -5.8 % FP
Max Quadrant Tilt +0.55 %

Core Centerlines

0.98 1.02	1.29 1.28	1.26 1.27	1.43 1.38	1.12 1.22	1.28 1.25	1.35 1.36	0.96 0.96
	1.21 1.24	1.47 1.36	1.21 1.23	1.27 1.26	1.00 1.04	0.97 1.04	0.78 0.75
		1.24 1.29	1.32 1.31	0.95 0.98	0.94 0.98	0.64 0.72	0.48 0.46
			1.02 1.07	1.11 1.07	0.81 0.86	0.70 0.70	
				0.90 0.91	0.90 0.86	0.57 0.51	
					0.66 0.53		

Quadrant Centerline

X.XX X.XX	Predicted Values Measured Values

FIGURE 2.19-4

COMPARISON OF PREDICTED AND CALCULATED STEADY STATE TOTAL PEAK POWER DISTRIBUTION AT 40 % FP, EQUILIBRIUM XENON

Measurement Conditions

Control Rod Group Positions

Gps 1-4	100	% wd
Gp 5	100	% wd
Gp 6	74	% wd
Gp 7	0	% wd
Gp 8	30	% wd

Core Power Level	41.2	% FP
Boron Concentration	1154	ppm
Core Burnup	3.97	EFPD
Axial Imbalance	-5.8	% FP
Max Quadrant Tilt	+0.55	%

Core Centerlines

1.21 1.33	1.60 1.60	1.60 1.61	1.88 1.82	1.54 1.59	1.68 1.61	1.72 1.75	1.21 1.21
	1.51 1.60	1.86 1.76	1.59 1.63	1.74 1.66	1.32 1.37	1.23 1.27	0.99 0.98
		1.61 1.69	1.81 1.75	1.41 1.48	1.28 1.30	0.82 0.92	0.60 0.56
			1.48 1.56	1.55 1.45	1.08 1.17	0.90 0.89	
				1.19 1.21	1.16 1.08	0.73 0.64	
					0.85 0.69		

Quadrant Centerline

X.XX X.XX	Predicted Values Measured Values

FIGURE 2.19-5

COMPARISON OF PREDICTED AND CALCULATED STEADY STATE RELATIVE RADIAL POWER DISTRIBUTION AT 75 % FP, EQUILIBRIUM XENON

Measurement Conditions

Control Rod Group Positions

Gps 1-4	100	% wd
Gp 5	100	% wd
Gp 6	71.1	% wd
Gp 7	0	% wd
Gp 8	30.7	% wd

Core Power Level	75	% FP
Boron Concentration	1074	ppm
Core Burnup	12.2	EFPD
Axial Imbalance	-14.4	% FP
Max Quadrant Tilt	-0.67	%

Core Centerlines

1.00 1.04	1.32 1.31	1.30 1.28	1.46 1.41	1.13 1.18	1.25 1.23	1.28 1.31	0.90 0.91
	1.25 1.26	1.51 1.37	1.24 1.25	1.27 1.27	0.99 1.04	0.92 1.01	0.74 0.74
		1.27 1.31	1.34 1.34	0.96 0.98	0.94 0.99	0.62 0.69	0.46 0.45
			1.05 1.09	1.12 1.06	0.82 0.88	0.69 0.70	
				0.91 0.93	0.91 0.85	0.58 0.51	
					0.68 0.55		

Quadrant Centerline

X.XX	Predicted Values
X.XX	Measured Values

FIGURE 2.19-6

COMPARISON OF PREDICTED AND CALCULATED STEADY STATE TOTAL PEAK POWER DISTRIBUTION AT 75 % FP, EQUILIBRIUM XENON

Measurement Conditions

Control Rod Group Positions

Gps 1-4	100	% wd
Gp 5	100	% wd
Gp 6	71.1	% wd
Gp 7	0	% wd
Gp 8	30.7	% wd

Core Power Level	75	% FP
Boron Concentration	1074	ppm
Core Burnup	12.2	EFPD
Axial Imbalance	-14.4	% FP
Max Quadrant Tilt	-0.67	%

Core Centerlines

1.29 1.37	1.72 1.73	1.74 1.68	2.04 1.88	1.67 1.65	1.76 1.67	1.72 1.76	1.19 1.17
	1.64 1.65	2.03 1.87	1.75 1.75	1.88 1.79	1.39 1.42	1.24 1.33	0.98 0.98
		1.77 1.77	1.99 1.85	1.53 1.54	1.36 1.36	0.84 0.91	0.60 0.59
			1.62 1.64	1.68 1.53	1.16 1.21	0.94 0.91	
				1.29 1.26	1.24 1.12	0.78 0.66	
					0.91 0.72		

Quadrant Centerline

X.XX Predicted Values
X.XX Measured Values

FIGURE 2.19-7

COMPARISON OF PREDICTED AND CALCULATED STEADY STATE RELATIVE RADIAL POWER DISTRIBUTION AT 100 % FP, EQUILIBRIUM XENON

Measurement Conditions

Control Rod Group Positions

Gps 1-4	100	% wd
Gp 5	100	% wd
Gp 6	94	% wd
Gp 7	17.4	% wd
Gp 8	13.1	% wd

Core Power Level	100	% FP
Boron Concentration	1079	ppm
Core Burnup	20.2	EFPD
Axial Imbalance	-8.2	% FP
Max Quadrant Tilt	-0.75	%

Core Centerlines

1.02 1.09	1.28 1.29	1.27 1.24	1.43 1.39	1.16 1.19	1.24 1.23	1.24 1.29	0.88 0.88
	1.22 1.22	1.45 1.33	1.23 1.23	1.27 1.27	1.00 1.05	0.92 1.01	0.74 0.73
		1.25 1.28	1.33 1.34	0.98 0.99	0.95 1.01	0.66 0.75	0.48 0.47
			1.08 1.13	1.13 1.07	0.83 0.88	0.72 0.72	
				0.92 0.91	0.90 0.84	0.58 0.50	
					0.67 0.54		

Quadrant Centerline

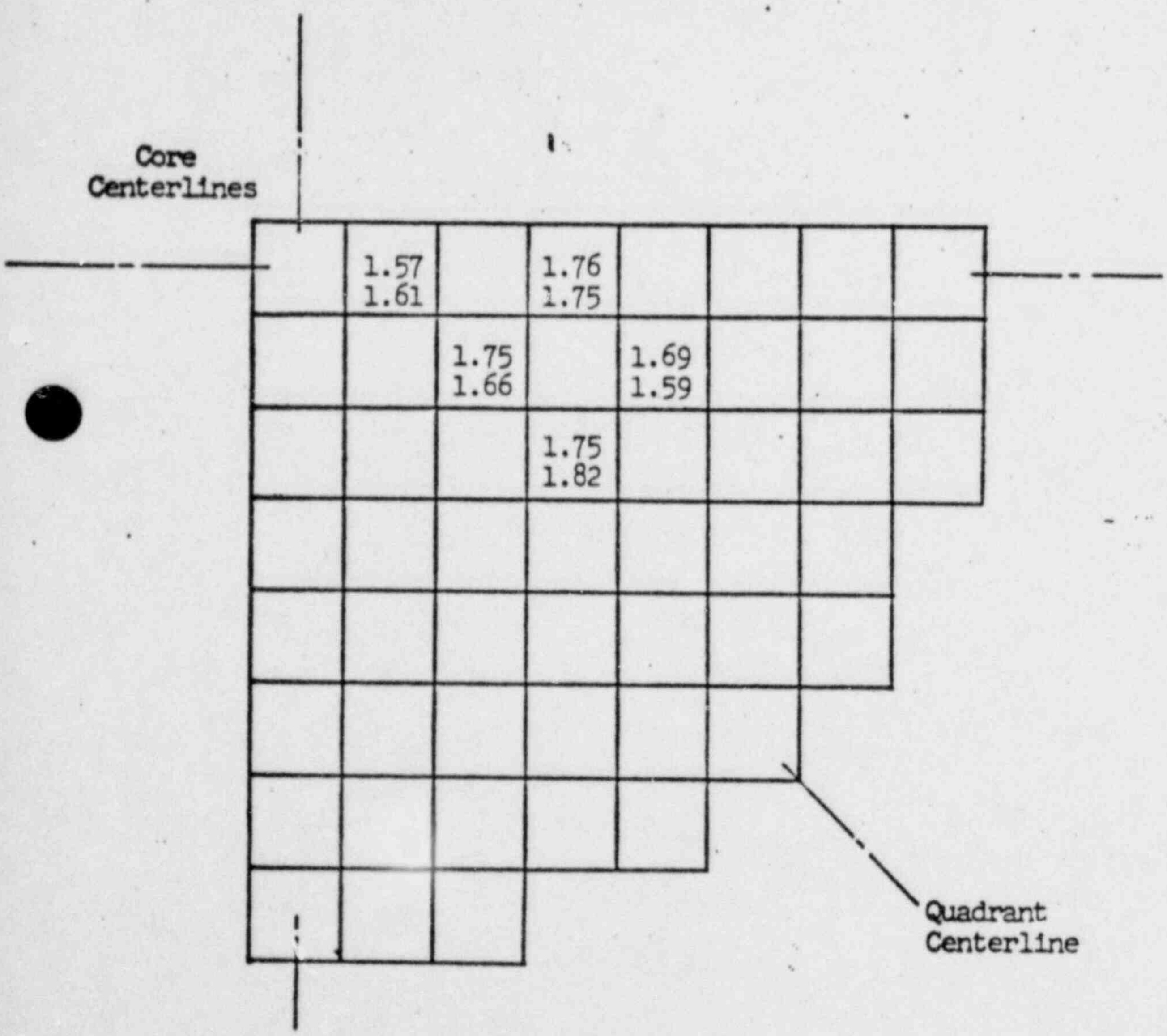
X.XX	Predicted Values
X.XX	Measured Values

FIGURE 2.19-8

COMPARISON OF PREDICTED AND CALCULATED STEADY STATE TOTAL PEAK POWER DISTRIBUTION AT 100 % FP, EQUILIBRIUM XENON

Measurement Conditions

Control Rod Group Positions			Core Power Level	100 % FP
Gps 1-4	100	% wd	Boron Concentration	1079 ppm
Gp 5	100	% wd	Core Burnup	20.2 EFPD
Gp 6	94	% wd	Axial Imbalance	-8.2 % FP
Gp 7	17.4	% wd	Max Quadrant Tilt	-0.75 %
Gp 8	13.1	% wd		



X.XX	Predicted* Values
X.XX	Calculated Values

*Only points computed for comparison

2.20 REACTIVITY COEFFICIENTS AT POWER

2.20.1 PURPOSE

The purpose of this test was to measure the temperature coefficient (α_T) and the power doppler coefficient (α_{PD}) at 40, 75, and 100% of full power; to determine the doppler coefficient (α_D) and the moderator coefficient (α_M) from the measured values; and to compare the results with predicted values.

Acceptance criteria for this test were that the power doppler coefficient be more negative than -0.55×10^{-4} $(\Delta k/k)/\%FP$ at 100% of full power, and that the moderator coefficient must be non-positive when extrapolated to 95% of full power.

2.20.2 TEST METHOD

At 40, 75 and 100% of full power the temperature coefficient and power doppler coefficient were measured. This was done by measuring the differential worth of the controlling rod group by the fast insert and fast withdrawal technique; varying the reactor coolant system average temperature by 5°F and measuring the resulting temperature change and controlling rod group motion required to maintain the same reactor power; and by reducing the reactor power level by 5% of full power and measuring the resulting power change and controlling rod group motion required to maintain criticality. The rod motion due to temperature change or power change was then related to reactivity from the differential rod worth and the temperature and power doppler coefficients were then calcualated.

The doppler coefficient was then calculated from the power doppler coefficient as follows using the inverse of the predicted rate of fuel temperature rise per increase in power level:

$$\alpha_D = \alpha_{PD} \times \frac{\%FP}{°F}$$

The moderator coefficient was determined from the temperature and doppler coefficients as follows:

$$\alpha_M = \alpha_T - \alpha_D$$

2.20.3 RESULTS AND EVALUATION

The results of the reactivity coefficient test are summarized in table 2.20-1.

The measured temperature and doppler coefficients agreed very well with predicted values. The power doppler coefficient was less negative than predicted and well below the maximum acceptable value. The moderator coefficient was more negative than predicted and well below the non-positive limit. The extrapolation of the moderator coefficient to 95% of full power indicated it would be negative for all Boron concentration and power level.

2.20.4 CONCLUSION

The measured values of all reactivity coefficients were well within the acceptable limits.

The acceptance criteria of this test were met in full without deficiencies.

TABLE 2.20-1

SUMMARY OF MEASURED AND PREDICTED REACTIVITY COEFFICIENTS

Parameter		Reactor Power Level (%full power)		
		40	75	100
Control Rod Assembly Group (% withdrawn)	6	74	73	90
	7	0	0	17
	8	30	12	7
Boron Concentration (ppm)		1151	1079	1060
Differential Rod Worth $\frac{\Delta k/k}{^\circ F}$		0.0107	0.0113	0.0092
Temperature Coefficient $\frac{\Delta k/k}{^\circ F}$	Measured	-0.23×10^{-4}	-0.31×10^{-4}	-0.41×10^{-4}
	Predicted	-0.04×10^{-4}	-0.22×10^{-4}	-0.30×10^{-4}
Power Doppler Coefficient $\frac{\Delta k/k}{\%\ \text{Full power}}$	Measured	-0.97×10^{-4}	-0.92×10^{-4}	-0.84×10^{-4}
	Predicted	-1.37×10^{-4}	-1.28×10^{-4}	-1.16×10^{-4}
Doppler Coefficient $\frac{\Delta k/k}{^\circ F}$	Measured	-0.14×10^{-4}	-0.13×10^{-4}	-0.12×10^{-4}
	Predicted	-0.19×10^{-4}	-0.20×10^{-4}	-0.16×10^{-4}
Moderator Coefficient $\frac{\Delta k/k}{^\circ F}$	Measured	-0.09×10^{-4}	-0.18×10^{-4}	-0.29×10^{-4}
	Predicted	$+0.15\times10^{-4}$	-0.04×10^{-4}	-0.14×10^{-4}

2.21 INCORE DETECTOR TEST

2.21.1 PURPOSE

The power distribution within the core is measured at 364 locations (seven axial positions in 52 fuel assemblies) by incore self-powered neutron detectors. The purpose of the incore detector test was to verify:

(1) The proper operation of each incore detector and the Incore Monitoring System

(2) That the measured core power distribution is consistent with the design calculations

(3) The functional and operational requirements of the Incore Instrumentation System as described in the Technical Specifications.

2.21.2 TEST METHOD

The incore detector test was performed at the 40% FP test plateau. After 3-D equilibrium Xenon was established, the uncorrected and corrected SPND signals for the seven levels of all detectors along with a 3-D power map, performance data output, and heat balance were obtained form the plant computer.

The computer-generated correction factors for depletion and length were then verified by hand calculations.

After verification of the correction factors, the best estimate heat balance was used to calculate the average core segment power Dividing the incore segment power (3-D power map) by the average core segment power yielded the relative power for each incore segment.

The 52 SPND strings are divided into 29 groups on the basis of core symmetry. The relative power axial distributions were plotted for comparison with the calculated power distributions and SPND strings of the same group.

2.21.3 EVALUATION OF TEST RESULTS

The detector outputs were found to be consistent and reasonable. The hand calculations for length and depletion correction agreed well with the computer values.

Comparison of the relative power axial distribution plots to the predicted axial distributions in the Physics Test Manual indicated only a very general a reement. Good agreement could not be expected since the predi ted conditions (Group 8 @ 37.5% WD and 0% Xenon) were either not in effect or not attainable. Subsequent analysis of data obtained in other test procedures supports the conclusion that the relative power plots are correct and acceptable.

2.21.4 CONCLUSIONS

The acceptance criteria of the test were met. Specifically, the required backup detectors were operable and the hand corrected signals agree with the computer corrected values.

Generally, the detector outputs are consistent and reasonable, the testing confirmed the satisfactory operation of the incore detector system, and the incore detectors have performed according to design during the startup phase.

2.22 ROD REACTIVITY WORTH MEASUREMENT

2.22.1 OBJECTIVE

The rod reactivity worth measurement procedure was used to measure control rod group differential and integral worths during the Zero Power Physics test by making soluble poison changes while compensating with control rod motion and for measuring control rod group differential worths at power using the fast insertion/fast withdrawal technique. These measurements were used to verify calculated rod worths, for use in other reactivity measurements and to verify the acceptance criteria of the test which were that control rod differential worths must be less than .0303 %ΔK/K/%wd to limit the maximum reactivity insertion rate yet greater than .000835 %ΔK/K/%wd to allow adequate reactivity for control.

2.22.2 METHOD

2.22.2.1 Soluble Poison Change - Rod Swap Method

During the Zero Power Physics Test, the control rod group differential and integral worths were measured by deborating and inserting control rods in the core in discrete steps while measuring rod motion and reactivity changes using the Babcock & Wilcox Reactimeter. The rod motion was related to reactivity changes from which control rod group worths (differential and integral) were determined.

2.22.2.2 Fast Insertion/Fast Withdrawal Method

During power escalation testing at the 40% FP, 75% FP and 100% FP plateaus, controlling rod group differential worths were measured in order to verify acceptance criteria and to facilitate measurement of reactivity coefficients. The method used was to establish steady state equilibrium conditions and measure the reactivity change associated with a six second rod insertion followed by a six second withdrawal. The reactivity change was related to control rod movement to determine the controlling rod differential worth in % ΔK/K/%wd.

2.22.3 RESULTS AND EVALUATION

2.22.3.1 Integral Control Rod Worths (Soluble Poison/Rod Swap)

The results of the integral control rod worth measurements by the soluble poison/rod swap method used during zero power testing are tabulated in Table 2.22-1. These results show very good agreement between measured and predicted values.

2.22.3.2 Differential Control Rod Worths (Fast Insertion/Fast Withdrawal)

The results of the differential control rod worth measurements by the fast insertion/fast withdrawal during power testing are tabulated in Table 2.22-2. These results are in excellent agreement with predicted values and are well within acceptance criteria.

2.22.4 CONCLUSIONS

Both methods used to determine control rod worths yielded acceptable results. All measured values compared well with the predicted control rod worths.

The acceptance criteria for maximum and minimum differential control rod worths were met.

Table 2.22-1

INTEGRAL CONTROL ROD WORTHS MEASURED BY THE SOLUBLE POISON/ROD SWAP METHOD DURING ZERO POWER PHYSICS TESTING

Group	No. Rods.	Predicted Worth, %ΔK/K	Measured Worth, %ΔK/K
5	12	-1.07	-1.09
6	8	-1.22	-1.14
7	9	-1.20	-1.05
8	8	-0.38	-0.37

Table 2.22-2

DIFFERENTIAL CONTROL ROD GROUP WORTHS BY THE FAST INSERT/FAST WITHDRAWAL METHOD

Reactor Power	Controlling Group and Position	Differential Worth %ΔK/K/%wd	
		Measured	Predicted
40	6 @ 72.6% wd	.01067	.0111
75	6 @ 68.6% wd	.01131	.0117
100	6 @ 89.4% wd	.0092	.0092

2.23 POWER IMBALANCE DETECTOR CORRELATION

2.23.1 PURPOSE

The Power Imbalance Detector Correlation Test had two objectives - to determine the relationship between out-of-core and incore imbalance and to verify the adequacy of the imbalance system trip setpoints.

2.23.2 TEST METHOD

Imbalance measurements were made to determine the acceptability of the out-of-core detectors to detect imbalance and to establish a basis for verifying that DNBR and LHR limits would not be exceeded while operating within the flux/delta flux/flow envelope set in the Reactor Protective System. These imbalance measurements were made at different part length rod positions with two different gain factors applied to the multiplier on the delta flux amplifier output.

In performing the test, part length rod position was varied and reactivity compensation made by rod groups 6 and/or 7. At each required incore imbalance, core power distribution and other data were taken. From these data, plots of incore offset versus out-of-core offset were maintained. In addition, minimum DNBR, and maximum LHR were also obtained to assure safe conduct of the test. Offset = (Imbalance/%FP) 100%.

The relationship between incore offset and out-of-core offset has been determined to be a linear equation of the form below:

$$OCO = M \times GF \times ICO + B$$

Where: OCO = Out-of-Core Offset (percent)
ICO = Incore Offset (percent)
M = Slope of Relationship
B = Intercept at Zero ICO
GF = Gain Factor

From the 40% FP plot of incore offset versus out-of-core offset (GF = 3.7), the initial slope was determined. With this value, it was then possible to determine the gain factor necessary to meet the acceptance criteria by using the equation below:

$$GF = [M_2/M_1] \times 3.7$$

Where: GF = Gain Factor
M_1 = Slope derived from offset relationship with gain factor equal to 3.7
M_2 = Desired Slope

2.23.3 EVALUATION OF TEST RESULTS

The measurement of the relationship between the incore and out-of-core offset at 40% FP resulted in a linear equation with a conservatively calculated slope of 1.14. Using this value and the slope of the acceptance criteria of 0.920, the required gain factor was calculated to be 3.0. The gain factor was, therefore, adjusted to 3.0 prior to the 75% FP measurements. From these measurements, a conservatively estimated slope of 1.0 was obtained. All data points fell well within the acceptance criteria. The slope obtained is greater and, therefore, more conservative than the acceptance criteria. The incore vs. out-of-core correlation was also found to be acceptable for the minimum 23 incore monitors as read out on the backup incore recorder.

During testing, the computer calculated minimum DNBR and maximum LHR were recorded against incore offset. Extrapolation to the maximum operational power envelope from the 75% FP data results in a minimum DNBR and a maximum LHR of 3.5 and 16.9 kw/ft, respectively, which are within the acceptance criteria of 1.55 and 17.2 kw/ft.

In similar fashion, extrapolation to the over-power trip setpoint resulted in a minimum DNBR and a maximum LHR of 3.4 and 17.6 kw/ft, respectively. Acceptance criteria being 1.55 and 20.1 kw/ft.

2.23.4 CONCLUSIONS

Upon completion of power imbalance detector correlation testing at 40 and 75% FP, the following conclusions were drawn:

(a) The relationship between the incore and out-of-core offset is linear with a constant slope.

(b) The imbalance trip envelope as set in the Reactor Protective System will protect the reactor core from exceeding LHR and DNBR limits when a gain factor of 3.0 is set in the circuit.

2.24 DROPPED CONTROL ROD TEST

2.24.1 PURPOSE

There were four major objectives in conducting the dropped control rod test:

A. To verify that minimum DNBR and maximum linear heat rate values did not exceed acceptable limits with the worst case control rod dropped in the core while operating at power.

B. To demonstrate that the asymmetric alarm lamp and asymmetric fault lamp are each activated by an asymmetric control rod at the appropriate setpoints.

C. To show that reactor power is automatically reduced by the integrated control system when a control rod asymmetry fault exists and the same control rod groups in-limit is activated.

D. To demonstrate that automatic control rod withdrawal is inhibited beyond a predetermined power level when a control rod asymmetry fault exists and the in-limit for that control rod group is activated.

2.24.2 TEST METHOD

The methods used to test that the four objectives in the purpose were met are as follows:

A. The control rod in core location H-12 (& symmetric locations) was calculated to cause the most adverse thermal conditions by a computer simulation of core operation at power. The control rod in this core location (group 6, rod 3) was inserted in small steps while operating at 40% FP. The plant computer was used to measure the power distribution and thermal conditions. The reactivity inserted was compensated by withdrawal of the other rods in group 6 to maintain 40% FP. The differential worth of rod 6-3 was measured at each increment of insertion to determine the integral reactivity worth of the dropped rod.

B. Since an inadvertently dropped control rod must be detected by means of the asymmetric control rod detection system, this system was verified to indicate an asymmetric rod at the appropriate setpoint as rod 6-3 was inserted for the test.

C. The dropped rod test was conducted at 40% FP; thus it was necessary to reset the automatic runback from 60% to 30% FP for the conduct of this test to verify a runback would occur. An asymmetric rod and an in-limit were simultaneously simulated for the same rod and the runback feature was tested.

D. An attempt was made to withdraw control rods following the simulation in C in order to test the automatic control rod withdrawal inhibit.

2.24.3 RESULTS AND EVALUATION

The results of the portion of the test which was used to check the core power distribution are summarized in table 2.24-1. The resultant core power distribution is reported in figures 2.24-2 & 3. Integration of the differential control rod worths which were taken during the insertion of rod 6-3 shows that the inserted rod worth was 0.088%ΔK/K.

The test showed that the incore monitors are capable of detecting a dropped control rod and the power peaks were such that the vicinity of the dropped rod was distinguishable.

A core thermal analysis utilizing the incore monitors and the plant computer indicated a minimum DNBR of 8.17 and a maximum linear heat rate of 7.01 kw/ft. When these values are extrapolated to the overpower trip setpoint of 105.5%FP, the DNBR is 3.4 and the maximum linear heat rate is 18.5 kw/ft. These are well within the Technical Specification limits of 1.3 and 20.1 kw/ft respectively.

The asymmetric rod indications, dropped rod automatic power runback and automatic control rod withdrawal inhibit portions of the test were completed successfully.

2.24.4 CONCLUSIONS

All acceptance criteria were met for the Dropped Control Rod Test. The indicators and alarms detected the asymmetric control rod, the power was run back and automatic control rod withdrawal was inhibited when the control rod was asymmetric and the "in" limit detector was activated.

The minimum DNBR and maximum Linear Heat Rate extrapolated to 105.5% of full power were respectively greater than 1.35 and less than 20.1 kw/ft. The incore monitoring system was shown to be an effective indicator of a dropped control rod.

TABLE 2.24-1

CORE POWER DISTRIBUTION AND THERMAL HYDRAULICS DATA

TAKEN DURING PERFORMANCE OF DROPPED CONTROL ROD TEST

ROD POSITION %WD	INCORE IMBALANCE	QUADRANT TILT %				MAXIMUM LINEAR HEAT RATE (kw/ft)[1]	MINIMUM DNBR[2]	EXTRAPOLATED LINEAR HEAT RATE @ 105.5 %FULL POWER	EXTRAPOLATED DNBR @105.5% FULL POWER	MAXIMUM POWER PEAK
		WX	XY	YZ	ZW					
77.5	-5.10	+ .171	+ .567	- .145	- .593	5.86	10.38	15.45	5.6	1.76
50.2	.23	+ 5.25	- 4.42	- 5.20	+ 4.40	6.24	9.36	16.45	4.6	1.78
0	-3.11	+13.12	-12.36	-12.93	+12.17	7.01	8.17	18.48	3.4	2.02

[1]Maximum Linear Heat Rate has Been Multiplied By 1.417 to Allow for Possible Uncertainties.

[2]Minimum DNBR Has Been Reduced By 0.68 to Allow for Possible Uncertainties.

40% FP DROPPED ROD TEST - CALCULATED
RADIAL PEAKING FACTORS X
(HAND CALCULATED)

FIGURE 2.24-2
FUEL TRANSFER
CANAL →

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A															
B							1.038 1.064 1.124	1.356 1.388 1.454							
C						0.991 1.039 1.114			1.045 1.077 1.104	0.990 1.003 1.002			0.535 0.515 0.982		
D					1.077 1.159 1.254					1.001 0.997 0.976				0.510 0.475 0.422	
E				1.098 1.186 1.295			1.222 1.286 1.355		1.221 1.248 1.248		1.094 1.10[illegible] 1.078				
F			0.997 1.058 1.172				1.334 1.386 1.460	1.238 1.266 1.299				1.012 0.929 0.786	1.006 0.895 0.736		
G		0.998 1.046 1.169			1.223 1.304 1.422	1.337 1.408 1.516			1.208 1.190 1.159		1.233 1.110 0.936		1.079 0.907 0.680		
H	W 0.953 0.985 1.103				1.375 1.474 1.604			0.982 0.997 1.015	1.241 1.216 1.174			Rod 6-3	1.251 1.018 0.723		
K					1.223 1.306 1.425						1.231 1.107 0.932	1.260 1.071 0.814			
L		0.714 0.749 0.833	0.952 1.010 1.121			1.227 1.351 1.452					1.313 1.253 1.118		0.996 0.889 0.731		
M			0.915 0.960 1.064				1.256 1.320 1.391		1.203 1.230 1.232	1.311 1.320 1.272				0.703 0.638 0.552	
N				0.920 0.974 1.069				1.229 1.307 1.346	1.259 1.305 1.319						
O					0.866 0.907 0.985	0.967 1.012 1.089				0.911 1.000 0.998		0.859 0.835 0.797			
P						0.711 0.731 0.783									
R							0.749 0.761 0.809			0.445 0.447 0.462					

Z

X.XX	=	6-3 @ 77.5% wd
Y.YY	=	6-3 @ 50 % wd
Z.ZZ	=	6-3 @ 0 % wd

40% DROPPED ROD TEST - CALCULATED
TOTAL PEAKING FACTORS X
(HAND CALCULATED)

FIGURE 2.24-3
Fuel Transfer
Canal →

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A															
B							1.238 1.310 1.447	1.710 1.677 1.897							
C						1.267 1.314 1.472			1.335 1.282 1.399	1.276 1.205 1.303			0.674 0.666 0.623		
D					1.426 1.431 1.625					1.450 1.269 1.348				0.631 0.551 0.558	
E				1.455 1.473 1.680			1.533 1.541 1.732		1.571 1.448 1.646		1.530 1.336 1.323				
F			1.301 1.306 1.498				1.694 1.683 1.871	1.530 1.537 1.682				1.441 1.247 1.099	1.271 1.097 1.027		
G		1.246 1.264 1.460			1.607 1.592 1.814	1.673 1.687 1.905			1.514 1.400 1.487		1.575 1.384 1.200		1.339 1.155 0.968		
H	W 1.181 1.186 1.367				1.761 1.775 2.020			1.231 1.154 1.234	1.497 1.415 1.498			Rod 6-3	1.573 1.356 0.934		
--					1.568 1.585 1.807						1.555 1.365 1.203	1.631 1.415 2.099			
L		0.872 0.952 1.094	1.255 1.222 1.413			1.651 1.678 1.898					1.648 1.444 1.489		1.293 1.113 0.997		
M			1.135 1.193 1.368				1.549 1.716 1.914		1.533 1.435 1.555	1.702 1.565 1.659				0.870 0.755 0.735	
N				1.182 1.196 1.368				1.546 1.639 1.798	1.639 1.582 1.714						
O					1.151 1.081 1.229	1.280 1.243 1.400				1.293 1.128 1.367		1.061 1.012 1.053			
P						0.909 0.862 0.970									
R							0.964 0.905 1.008			0.554 0.542 0.588					

Z

X.XX	= 6-3 @ 77.5% wd
Y.YY	= 6-3 @ 50 % wd
Z.ZZ	= 6-3 @ 0 % wd

2.25 PSEUDO CONTROL ROD EJECTION TEST

2.25.1 PURPOSE

The purpose of the pseudo control rod ejection test was to measure the reactivity worth of the most reactive control rod if ejected from the reactor core while operating at power and the subsequent core power distribution. These measurements were used to verify that the reactivity worth assumed in the Safety Analysis Report is conservative.

2.25.2 TEST METHOD

Control rod assembly 7-1 was withdrawn from the core in 20% increments while operating at 40% FP and compensating for reactivity increases with CRA Group 6 insertion to maintain the reactor critical. At each 20% step, the differential reactivity worth of Group 6 was determined and a worst case core thermal conditions calculation was made.

At the point of 100% withdrawal of CRA 7-1 a complete core power distribution measurement from the incore monitors was made to determine the magnitude of power peaking factors.

The integral worth of CRA 7-1 was obtained by integration of the Group 6 differential reactivity worth over the length of travel of Group 6 rods. The values calculated for DNBR and Maximum Linear Heat Rate were compared to acceptance limits at each 20% increment of CRA 7-1's movement.

2.25.3 RESULTS AND EVALUATION

The test results are tabulated in Table 2.25-1. Radial and total power distributions for the worst case (CRA 7-1 @ 100% w.d.) are reported in figures 2.25-1 and 2.25-2. Since the ejected rod was located in the center of the core, the power distribution is reported using 1/8 core symmetry.

The reactivity worth of CRA 7-1 as measured by integration of the Group 6 differential worth over the Group 6 travel is 0.20% $\Delta K/K$. Using the calculated integral control rod worth curves provided by Babcock & Wilcox Company in the ANO-1 Physics Test Manual to relate the reactivity to the Group 6 travel, a value of 0.47% $\Delta K/K$ is obtained. Both results are lower than the acceptance criteria limit of 0.65% $\Delta K/K$. The measured value agrees reasonably well with the predicted value of 0.49% $\Delta K/K$.

The power distribution data taken at the worst case conditions (CRA 701 @ 100% w.d.) indicated a maximum linear heat rate of 12.72 KW/ft (A conservative multiplicative correction factor of 1.417 has been applied to the computer measured value due to measurement uncertainties) which was below the 17.2 KW/ft LOCA limit and the minimum DNBR was 4.53 (the measured DNBR has been reduced by a subtractive factor of 0.68 to correct for measurement uncertainties) which was well above the acceptable minimum of 1.55.

2.25.4 CONCLUSIONS

The measured worth of CRA 7-1 when ejected from the core at power was found to be between 0.20 and 0.47% $\Delta K/K$. This is well below the maximum allowable worth of 0.65% $\Delta K/K$ specified in the Technical Specifications.

Table 2.25-1

Summary of Results of the Pseudo Control Rod Ejection Test

CRA 7-1 %wd	CRA Gp 6 % wd	Gp 6 %ΔK/K/% wd	Incore Imbalance	Outcore Imbalance	Min.* DNBR	Max** KW/ft
0	68.2	.0127	- 4.48	- 5.05	---	---
20.5	65.4	.0114	- 7.32	- 7.78	---	---
41.1	60.9	.0100	-11.02	-11.34	4.90	11.59
63.0	58.2	.0100	-14.44	-14.81	4.56	12.26
81.0	49.0	.0105	-13.79	-14.43	4.53	12.54
100.0	47.9	.0097	-13.32	-13.87	4.57	12.72

* Computer calculated DNBR reduced by 0.68

** Computer calculated Max LHR multiplied by 1.417

NOTE: These corrections are made to correct for measurement and calculational uncertainties.

FIGURE 2.25-1

PSEUDO CONTROL ROD EJECTION TEST

Total Local Power Peaking Factors Pmax/$\bar{P}$ core

	8	9	10	11	12	13	14	15
H	1.33 3.48	1.60 3.02	1.61 2.37	1.82 2.29	1.59 1.85	1.61 1.79	1.75 1.91	1.21 1.26
K		1.60 2.53	1.76 2.48	1.63 2.03	1.66 1.86	1.37 1.49	1.27 1.37	0.98 1.01
L			1.69 2.16	1.75 2.01	1.48 1.40	1.30 1.32	0.92 0.75	0.56 0.61
M				1.56 1.70	1.45 1.48	1.17 1.20	0.89 0.87	
N					1.21 1.25	1.08 1.13	0.64 0.65	
O						0.69 0.72		
P								
R								

Reactor Power = 40% FP

X.XX	CRA7-1 @ 0% w.d.
X.XX	CRA7-1 @ 100% w.d.

FIGURE 2.25-2

PSEUDO CONTROL ROD EJECTION TEST (40% FP)

Radial Power Peaking Factors

Fuel Assembly to Average Fuel Assembly $P/\bar{P}$ core

	8	9	10	11	12	13	14	15
H	1.02 2.258	1.28 2.008	1.27 1.493	1.38 1.480	1.22 1.098	1.25 1.127	1.36 1.223	0.96 0.86
K		1.24 1.748	1.36 1.636	1.23 1.296	1.26 1.193	1.04 0.948	1.04 0.942	0.75 0.672
L			1.29 1.417	1.31 1.302	0.98 1.092	0.98 0.893	0.72 0.571	0.46 0.400
M				1.07 0.932	1.07 0.952	0.86 0.770	0.70 0.628	
N					0.91 0.794	0.86 0.764	0.51 0.452	
O						0.53 0.470		
P								
R								

Reactor Power - 40% FP

X.XX	CRA 7-1 @ 0% w.d.
X.XX	CRA 7-1 @ 100% w.d.

2.26 TURBINE/REACTOR TRIP TEST

2.26.1 PURPOSE

The purpose of the Turbine/Reactor Trip Test was to measure the plant response during and after a deliberate turbine or reactor trip from power. Specifically, the trip tested the control of: Pressurizer Level, eactor oolant Pressure, Reactor Coolant Temperature, Feedwater Flow, OTSG Level, and Main Steam Pressure during a turbine or reactor trip.

The Turbine/Reactor Trip Test provides data that can be used to optimize the performance of the Integrated Control System and provide baseline data for comparison with future performance data.

The acceptance criteria specified by the test are:

A. The Reactor Coolant System must remain within its safety limits.

B. The OTSG outlet steam pressure, pressurizer level, OTSG level, feedwater level, and turbine bypass system remain within their respective limits.

2.26.2 TEST METHOD

A. The Reactor Trip Test was performed at 40% FP by manually tripping the reactor from the control room console. Data parameters were recorded during the transient. Specific data was plotted for comparison with the design limits required by the acceptance criteria. (See figures 2.26-1 through 2.26-5)

B. The Turbine Trip Test was performed at 100% FP by manually tripping the turbine from the control room console. Data parameters were recorded and plotted in the same manner as for the reactor trip. (See Figures 2.26-6 through 2.26-10)

2.26.3 RESULTS AND EVALUATION

The measured parameters were compared to the design values by plotting the recorded values versus time as shown in figures 2.26-1 through 2.26-10.

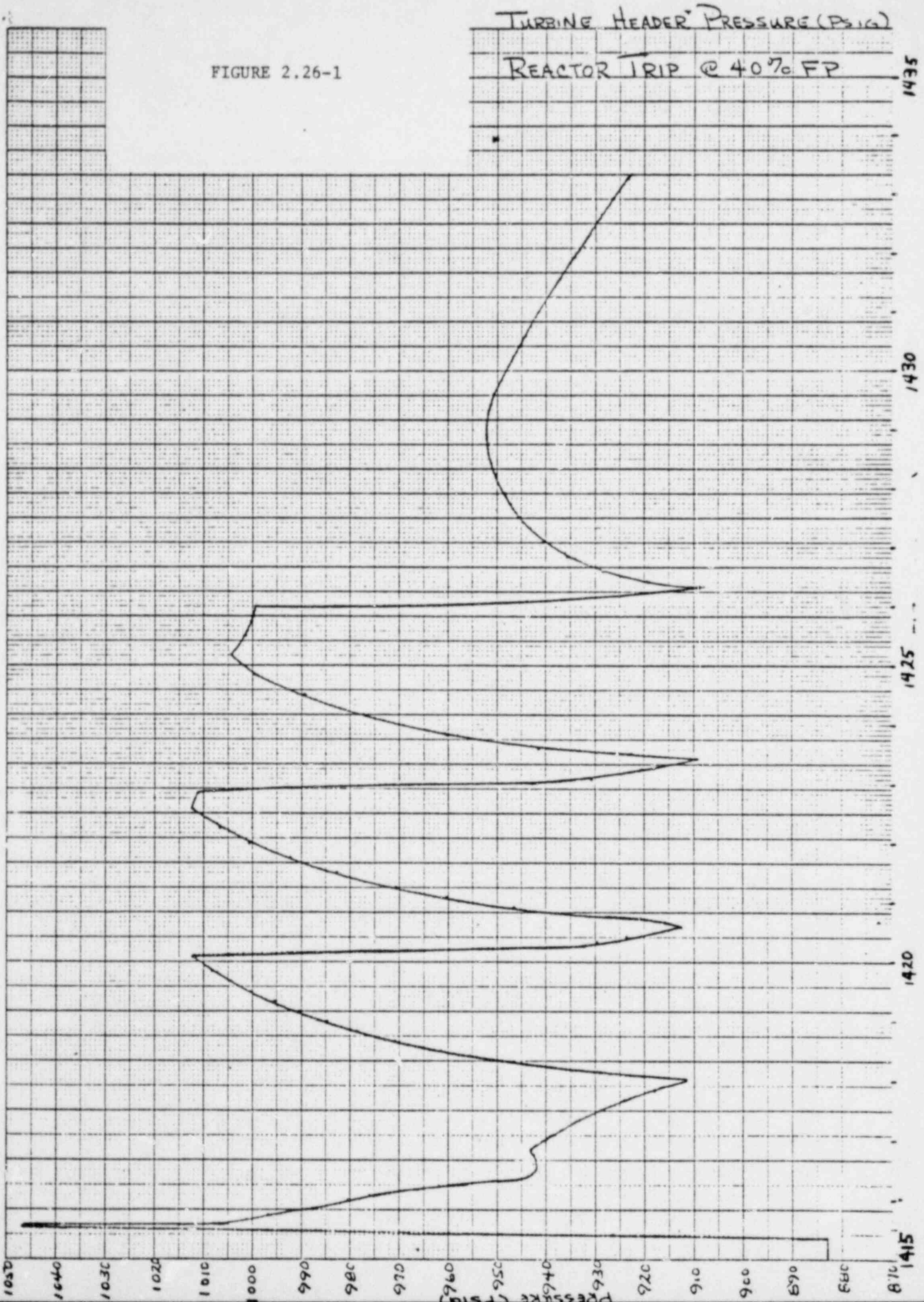
During the Reactor Trip Test at 40% FP Figure 2.26-3 shows that pressurizer level exceeded its design low level limit of 40 inches by 9 inches. This was due to a more rapid than normal cooldown rate when a code safety valve blew back excessively following the trip. Further evaluation has shown that maintaining indication of pressurizer level is adequate to show that RCS volume exists to cover the reactor core at all times. All other data plotted were within acceptable limits as can be seen in the remaining figures.

During the turbine trip from 100% FP, the reactor tripped due to pump power monitoring relay trips during the bus transfer from the unit auxiliary transformer to the Startup transformer #1 approximately 270 mSec after the Turbine trip. It is expected that the reactor would have tripped due to high RC pressure had the transfer not caused the reactor trip. This was demonstrated to be true by a later non-test trip and by testing at other similar B&W units. Since no reactor safety limits were exceeded, this trip was acceptable. All other data plotted were within acceptable limits as shown in Figures 2.26-6 through 2.26-10.

2.26.4 CONCLUSIONS

The data parameters recorded were within acceptable limits. No reactor safety limits were exceeded and the Integrated Control System performed as designed.

FIGURE 2.26-1



TURBINE HEADER PRESSURE (PSIG)
REACTOR TRIP @ 40% FP
1435
1430
1425
1420
1415
1050
1040
1030
1020
1010
1000
990
980
970
960
950
940
930
920
910
900
890
880
870
PRESSURE (PSIG)

FIGURE 2.26-2

OTSG Outlet Steam Pressure
Reactor Trip @ 40% FP

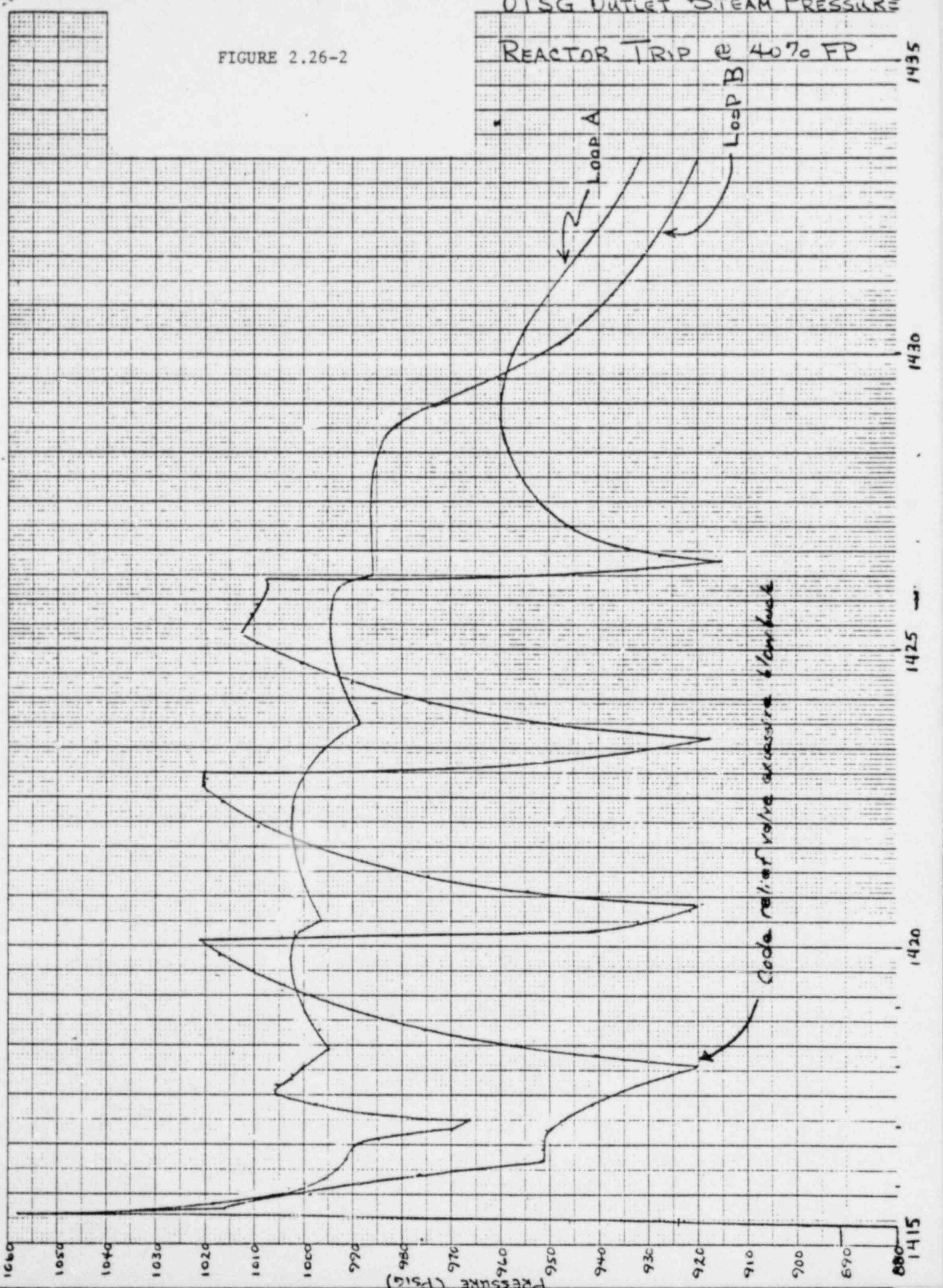


FIGURE 2.26-3

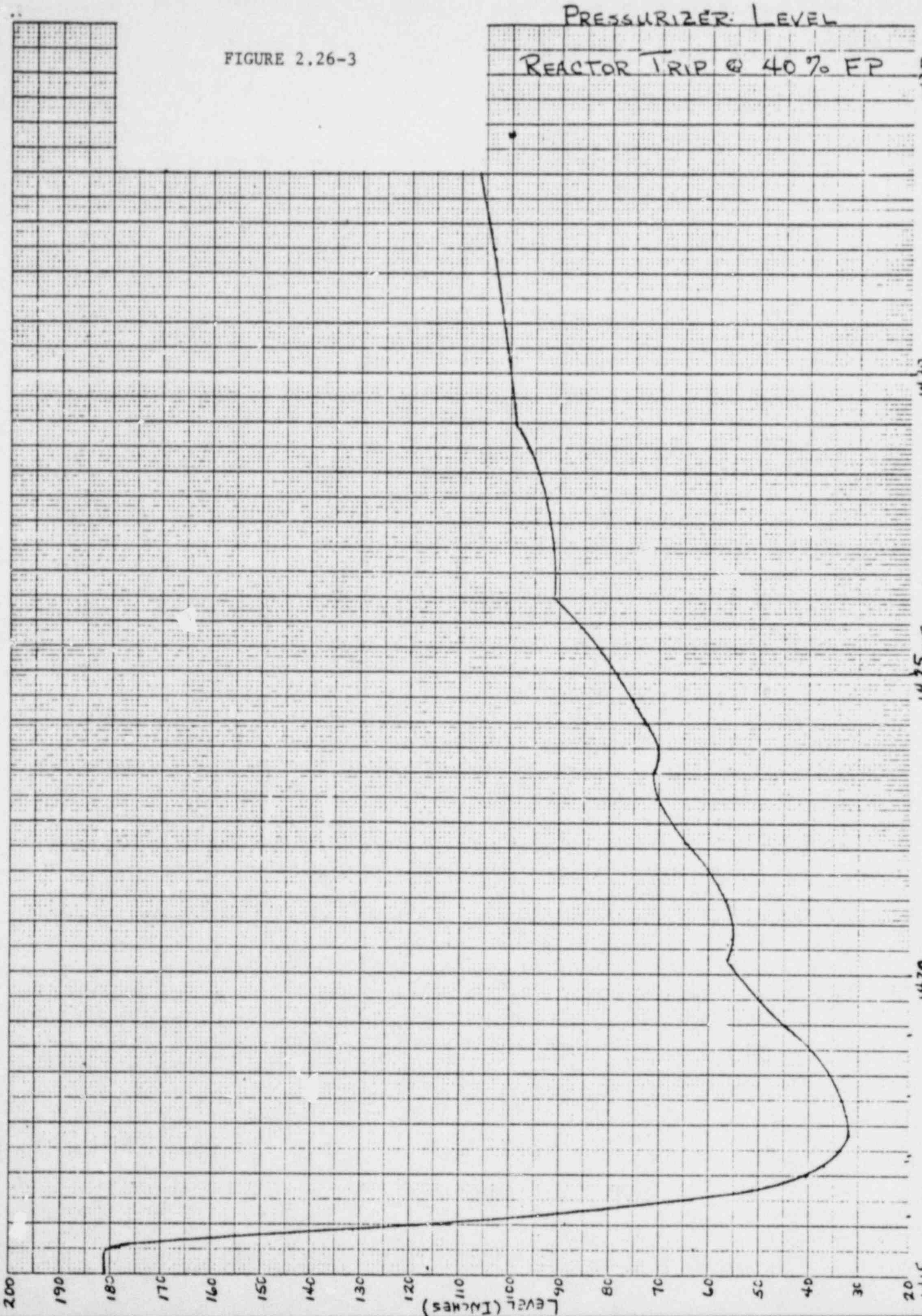
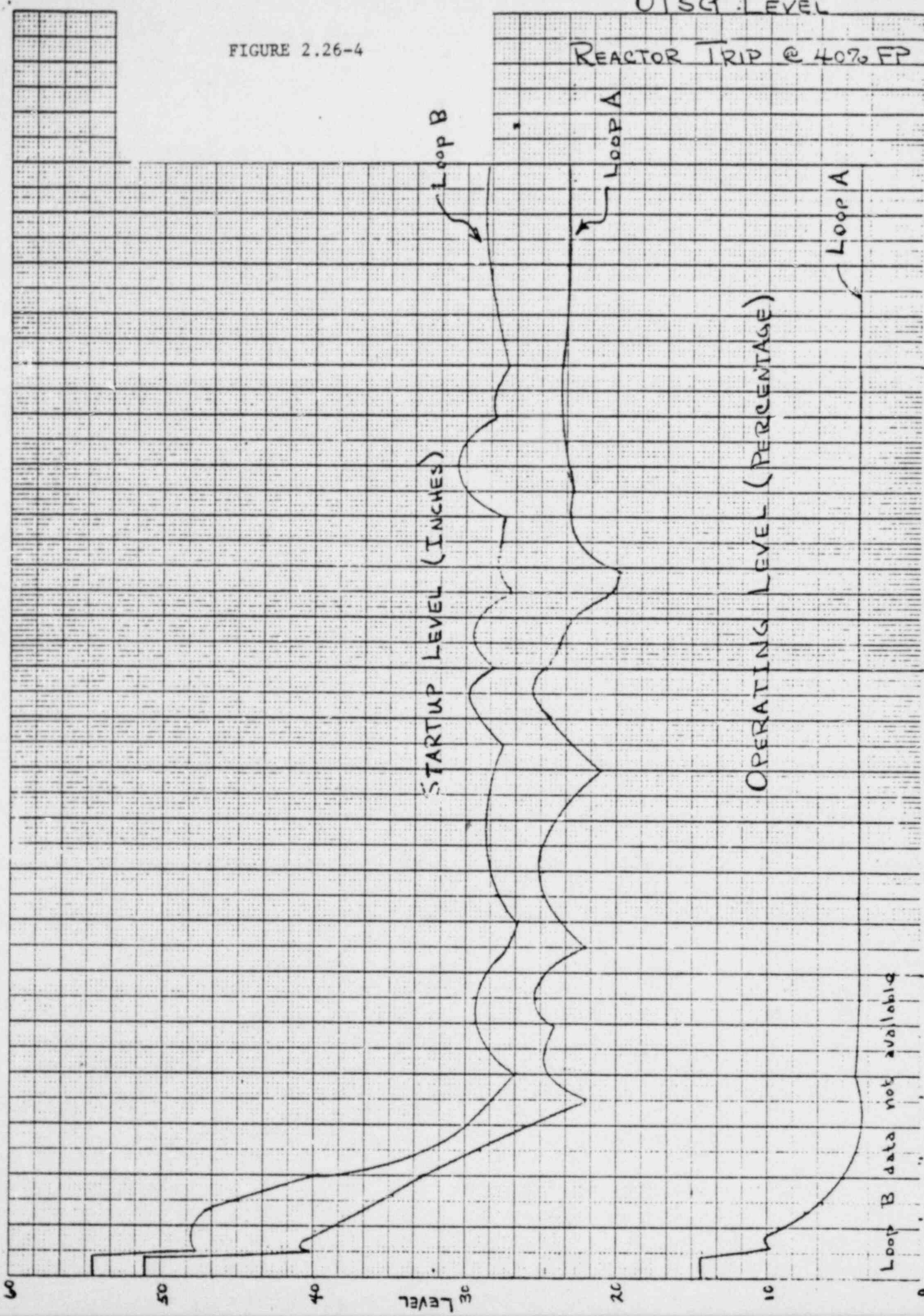
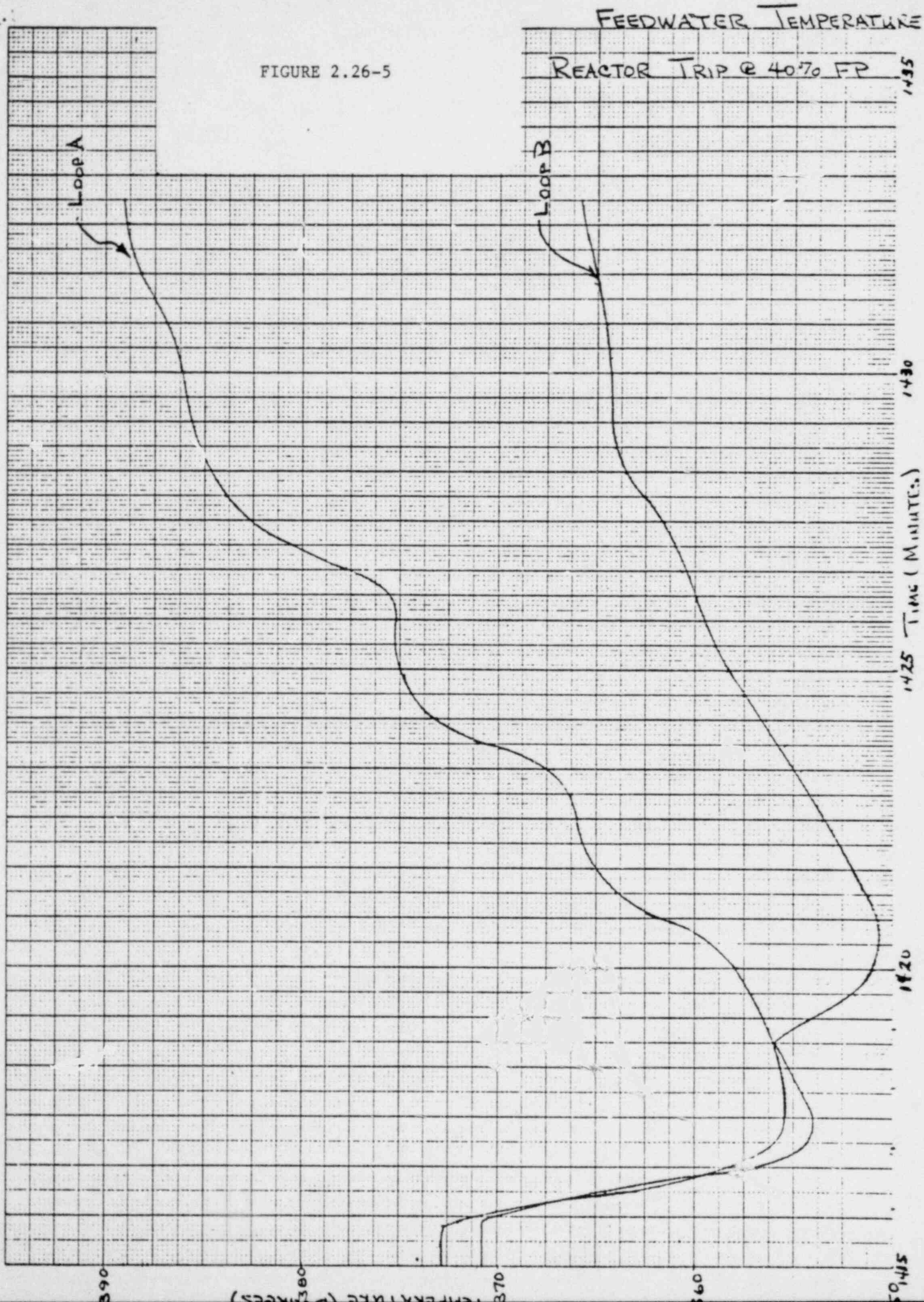

PRESSURIZER LEVEL
REACTOR TRIP @ 40% FP
1435
1430
1425 TIME (MINUTES)
1420
1415
200
190
180
170
160
150
140
130
120
110
100
90
80
70
60
50
40
30
20
LEVEL (INCHES)

FIGURE 2.26-4



OTSG LEVEL
REACTOR TRIP @ 40% FP
Loop B
Loop A
Loop A
STARTUP LEVEL (INCHES)
OPERATING LEVEL (PERCENTAGE)
Loop B data not available
LEVEL
0
10
20
30
40
50
60
1415
1420
1425
1430
1435

FIGURE 2.26-5



FEEDWATER TEMPERATURE
REACTOR TRIP @ 40% FP
LOOP A
LOOP B
1435
1430
1425
1420
1415
TIME (MINUTES)
390
380
370
360

FIGURE 2.26-6

TURBINE HEADER PRESSURE

TURBINE TRIP @ 100% FP

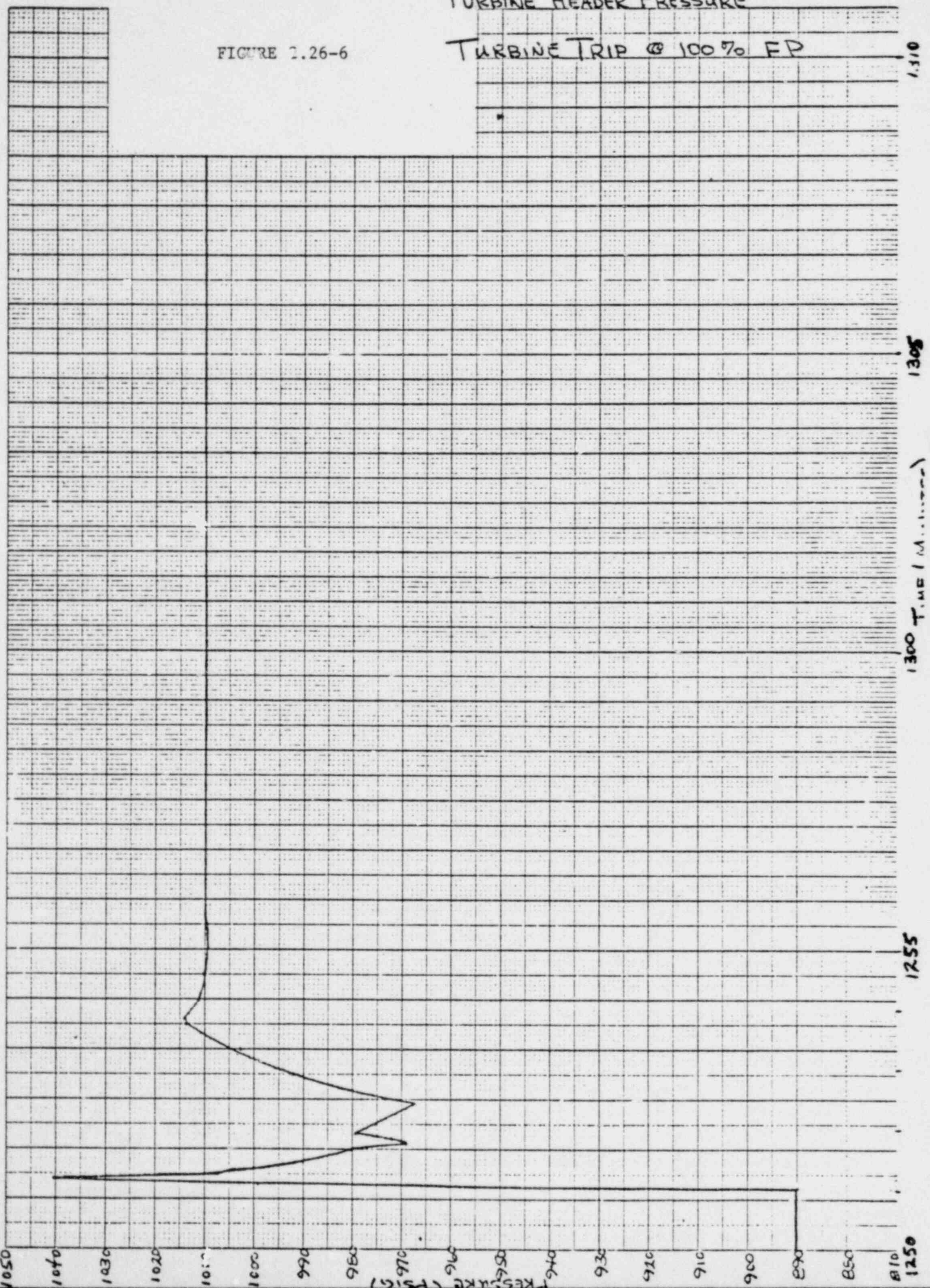


FIGURE 2.26-7

OTSG OUTLET STEAM PRESSURE

TURBINE TRIP @ 100% FP

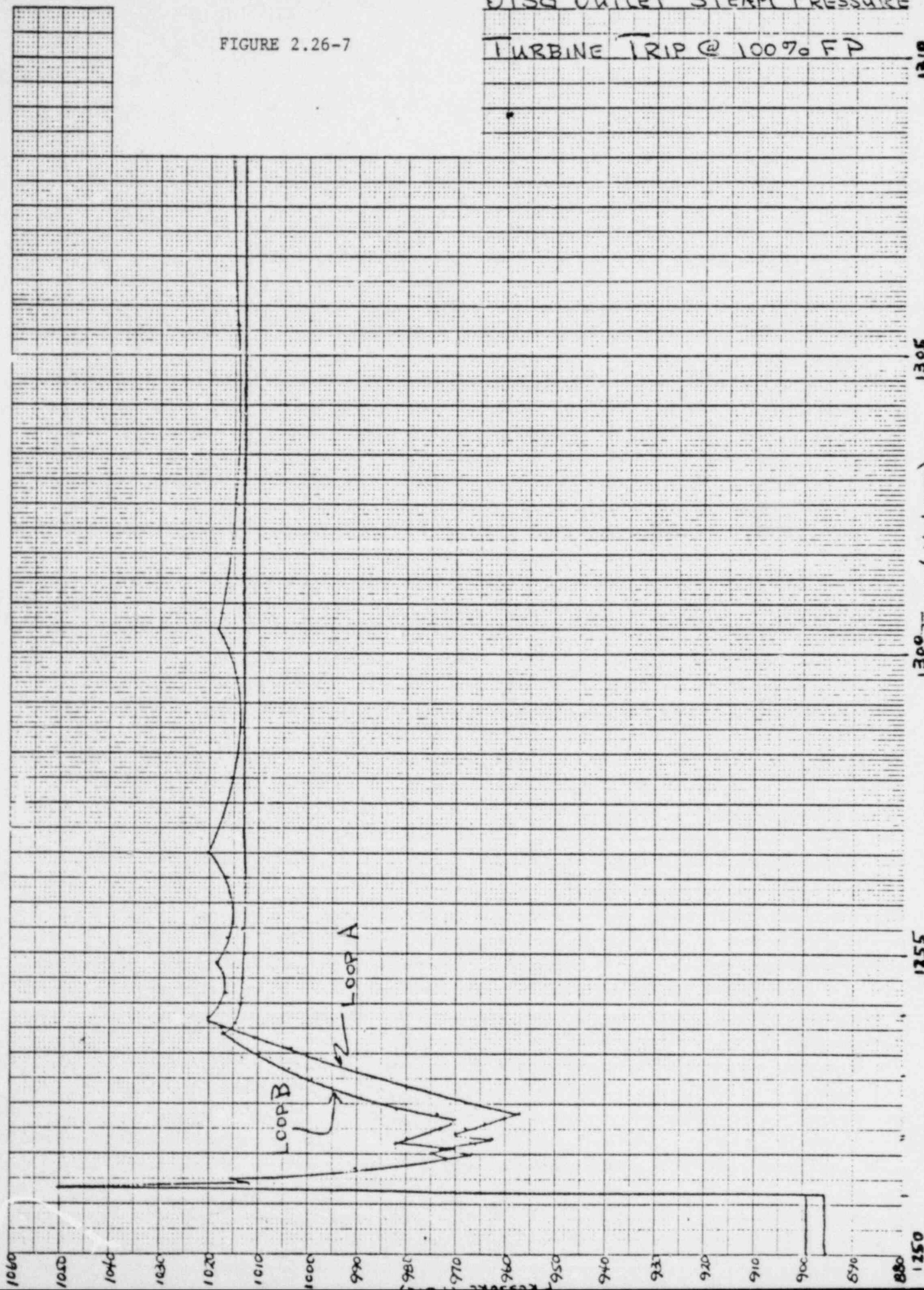


FIGURE 2.26-8

PRESSURIZER LEVEL

TURBINE TRIP @ 100% FP

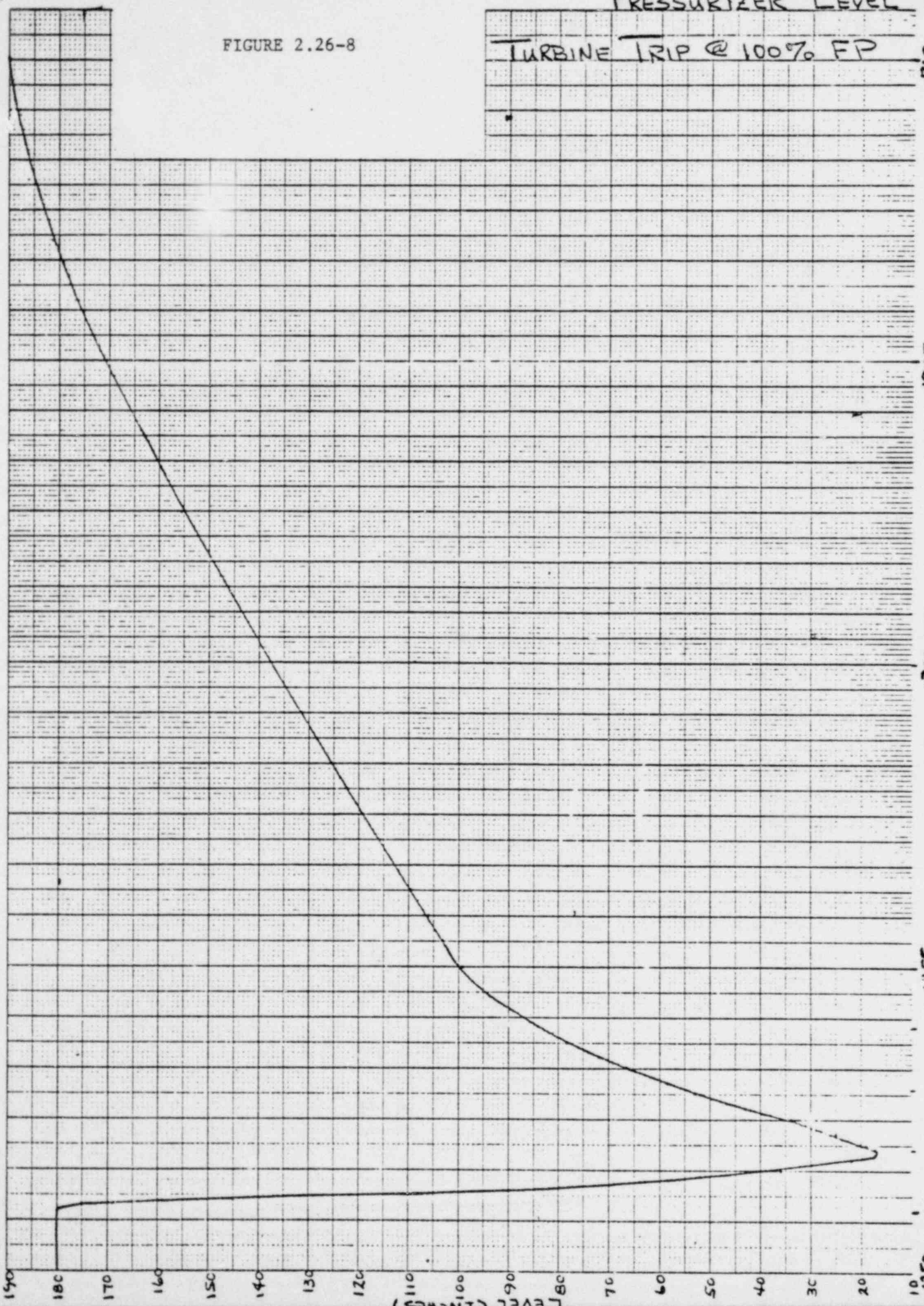
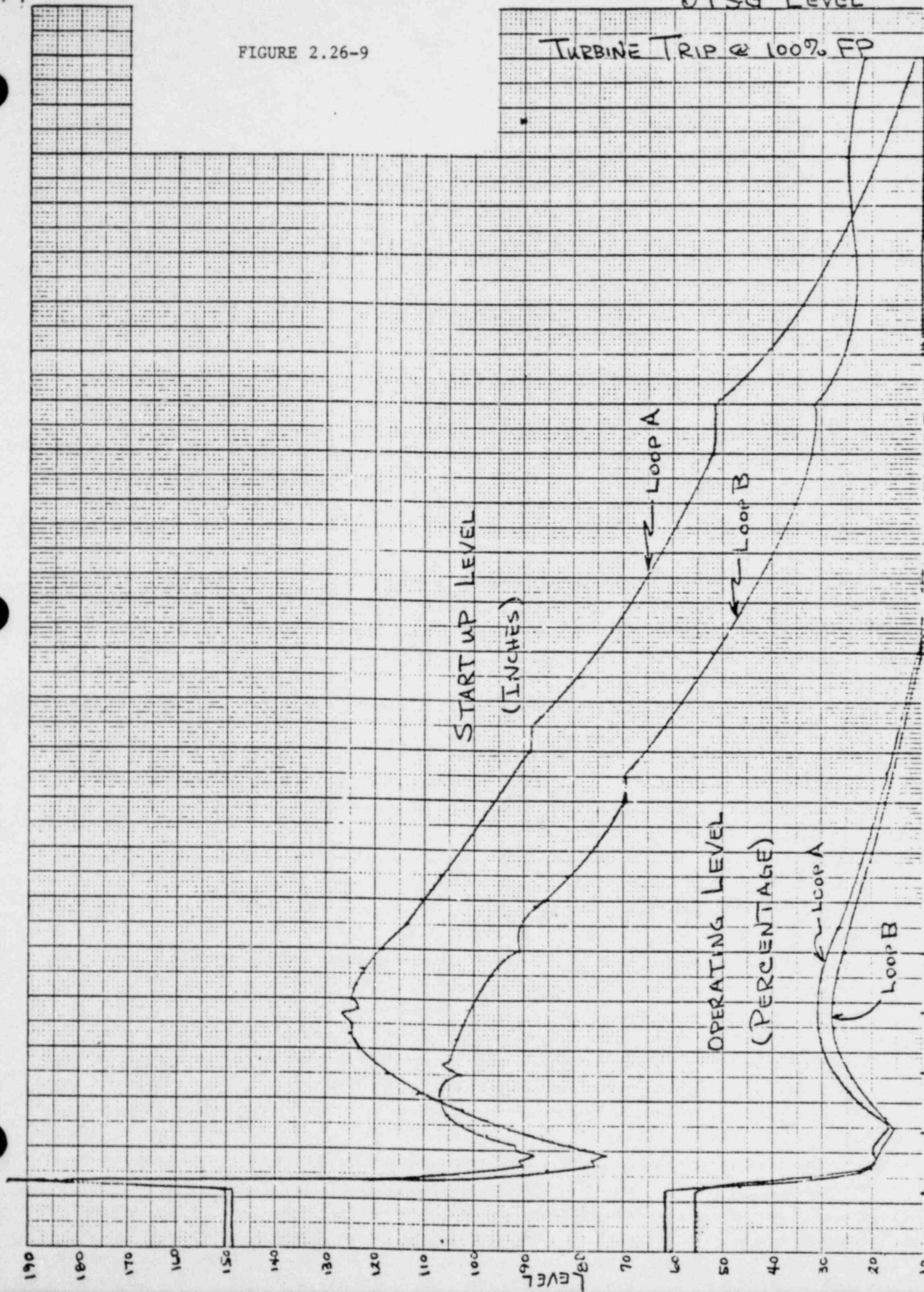


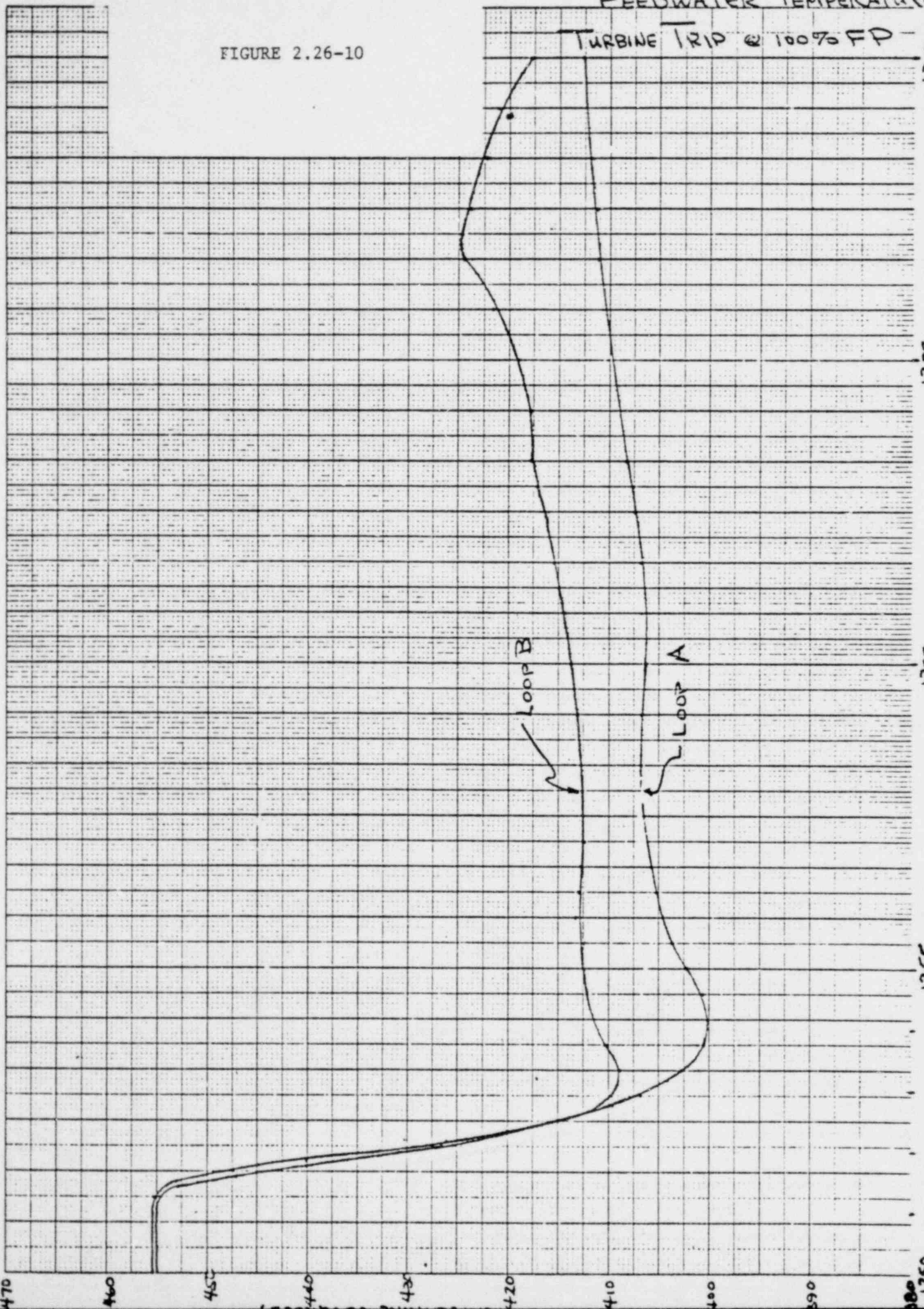
FIGURE 2.26-9



OTSG LEVEL
TURBINE TRIP @ 100% FP
START UP LEVEL
(INCHES)
OPERATING LEVEL
(PERCENTAGE)
Loop A
Loop B
Loop A
Loop B
LEVEL
190
180
170
160
150
140
130
120
110
100
90
80
70
60
50
40
30
20
10
1250
1255
1300
1305
1310

FEEDWATER TEMPERATURE
TURBINE TRIP @ 100% FP

FIGURE 2.26-10



2.27 PIPING SHOCK AND VIBRATION TEST

SUPPLEMENTARY REPORT

TO BE SUBMITTED LATER

2.28 GENERATOR SEPARATION TEST

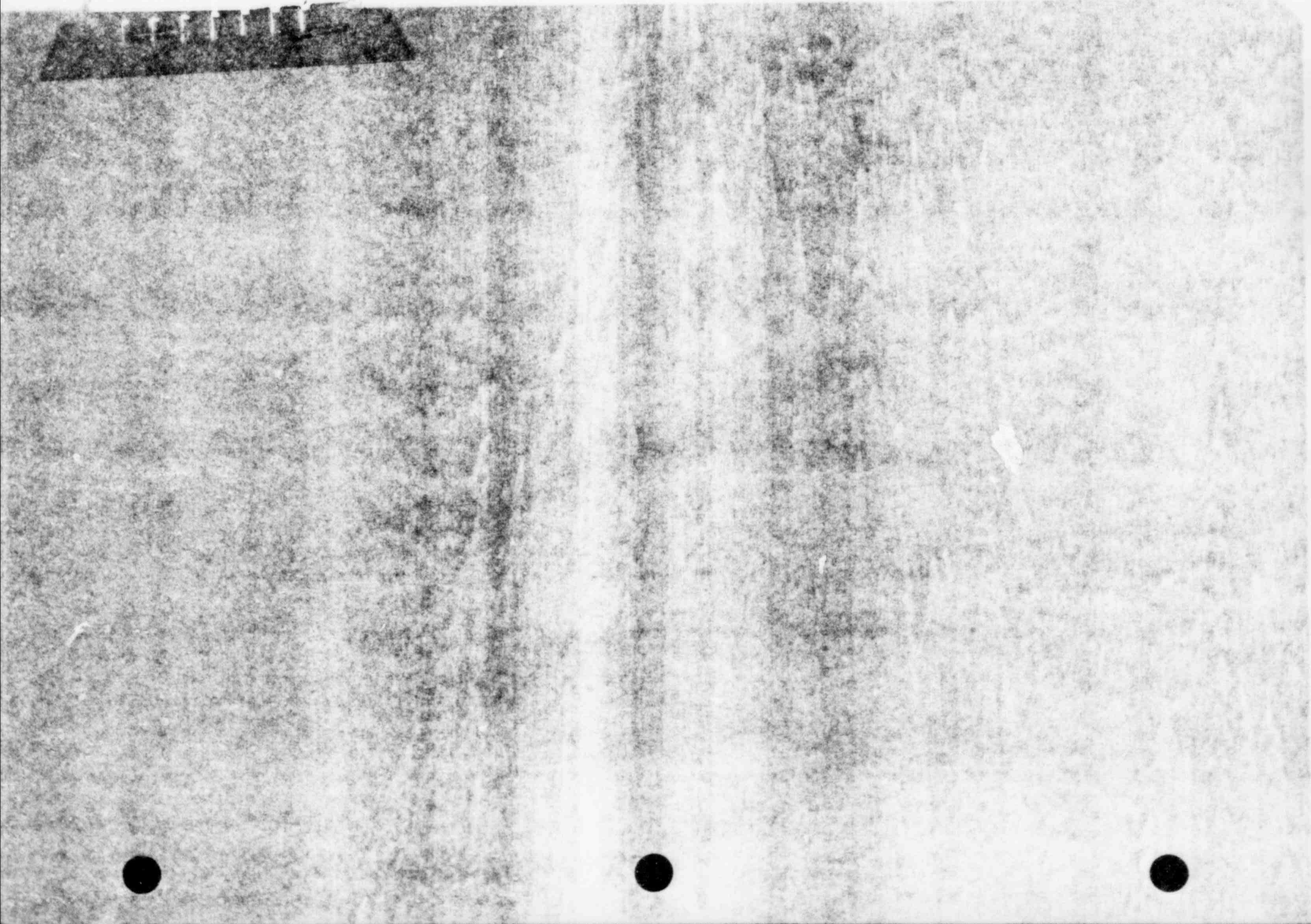
Per conversations with the Nuclear Regulatory Staff, the generator separation test will be performed and results submitted at a later date.

3.0 CONCLUSION

The results and conclusions summarized in the body of this report demonstrate that Arkansas Nuclear One Unit 1 has been properly designed and constructed and operates in a manner that will not endanger the health and safety of the public.

With this report, the preoperational and initial startup test program for ANO-Unit 1 as described in the ANO FSAR is complete.

Cycle 2 Startup
For Period Ending 4-11-77



ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE

STEAM ELECTRIC STATION

UNIT ONE

CYCLE 2

STARTUP REPORT

TO THE

U.S. NUCLEAR REGULATORY COMMISSION

LICENSE NUMBER DPR-51

DOCKET NUMBER 50-313

FOR THE

PERIOD ENDING 11 APRIL 1977

TABLE OF CONTENTS

1.0 INTRODUCTION

On January 27, 1977, the first refueling outage of ANO Unit 1 began and was completed with successful achievement of criticality on April 2, 1977.

Zero Power Physics Testing, which commenced on April 2, 1977, was successfully completed on April 5, 1977. This program was conducted at a reactor coolant temperature of 532°F and below the nuclear heat power level to eliminate any temperature feedback effects.

Power escalation was begun on April 5, 1977. A testing program was carried out at three major power plateaus during the power escalation:

Power Level (%FP)	Date
40	April 5, 1977
75	April 6, 1977
100	April 11, 1977

The startup and power escalation testing sequence was completed on April 11, 1977.

2.0 CONTROL ROD DRIVE TRIP TIME TEST

2.1 Purpose

The purpose of the Control Rod Drive Trip Time Test was to verify the integrated, functional trip capability of the Control Rod Drive System and to determine for each control rod assembly, the total elapsed drop time from the initiation of the trip signal until the control rod assembly was three-fourths inserted.

2.2 Test Method

Initial RCS conditions were established at a temperature of approximately 532^{o}F, at a pressure of 2155 $\pm$ 30 psig, all four (4) reactor coolant pumps running, with Boron at a concentration of 2121 ppmB. Control Rod Groups 1 thru 7 were fully withdrawn and Group 8 (APSR's) was withdrawn approximately 25%. The Control Rod Drive Mechanisms (CRDM) were then tripped via the manual trip button. The insertion times for each CRDM from its initial position to its 3/4 insertion point were measured by the plant computer Rod Drop Timer program. The printout of this program includes trip initiation time, initial position and trip insertion time for each CRDM (excluding Group 8).

2.3 Results and Evaluation

An analysis of the drop times indicate that rods 4-6 and 2-7 were fastest at 1.133 $\pm$ 0.017 seconds and rod 5-4 was the slowest at 1.200 $\pm$ 0.017 seconds.

2.4 Conclusions

The rod drop times were well below the acceptance criteria stated in Section 4.7 of the Technical Specifications of 1.46 seconds at full flow conditions.

3.0 ZERO POWER PHYSICS TEST

3.0.1 Purpose

The purpose of the Zero Power Physics Test was to verify the nuclear design parameters used in the safety analysis, the Technical Specification limits, and the basis for the operational parameters. This test was conducted after refueling and before power escalation. Testing was performed at 532°F and 2155 psig. The test included the following measurements:

A. Critical Boron Concentration for the "all rods out" and "rods inserted" conditions.

B. Moderator temperature coefficients for "all rods out" and "rods inserted" conditions.

C. Control Rod Assembly (CRA) Regulating Group Worth.

D. Ejected Rod Worth.

3.0.2 Test Method

Criticality was achieved by control rod withdrawal and Boron dilution of the Reactor Coolant System after system conditions had been established at 532°F and 2155 psig. During the approach to criticality, a plot of inverse neutron count rate ratio versus Boron concentration and time was maintained by using NI-1 and NI-2 of the nuclear instrumentation. After achieving criticality, nuclear power was increased and the source and intermediate range nuclear instrumentation overlap was verified to be in excess of one decade. During this same increase in power, the sensible heating point was determined to be 10^{-7} amps, and the upper power limit for Zero Power Physics testing was established at 5×10^{-8} amps.

Physics testing was then conducted which included the following measurements:

A. The "all rods out" Critical Boron Concentration.

B. Moderator Temperature Coefficient of Reactivity for "all rods out".

C. Differential and Integral Rod Worth of Control Rod Groups 7,6 and 5 by the rod versus Boron swap technique and the integral worth of Group 8 by the rod swap technique.

D. Critical Boron Concentration at a "rods inserted" condition.

E. Moderator Temperature Coefficient of Reactivity at "rods inserted" condition.

F. Ejected Rod Worth by the Boron swap technique.

3.1 DETERMINATION OF CRITICAL BORON CONCENTRATION

3.1.1 Purpose

The purpose of this test was to determine the Boron concentration required to maintain criticality at Hot Zero Power (approximately 10^{-9} amps) with all control rods withdrawn and no Xenon and to check the Critical Boron Concentration at a known critical control rod configuration. The resultant values are used to verify the predicted fuel depletion curves used in OP 1103.15, Reactivity Balance Calculation, and to verify the predicted "all rods out" Boron concentration in the Physics Test Manual.

3.1.2 Test Method

The "all rods out" Critical Boron Concentration Measurement was made after criticality was achieved, nuclear instrumentation overlap was verified, and point of sensible heating was established. The following initial conditions were established:

A. Equilibrium Boron concentration attained,

B. Power leveled off at 10^{-9} amps, and

C. Control Rod Groups 1-6 and Group 8 at 100% withdrawn and Group 7 at approximately 85% withdrawn.

The remaining reactivity held in the inserted portion of Group 7 was measured by withdrawal of Group 7 to its out limit and concurrent doubling time measurements. The doubling time was converted to reactivity and the reactivity to equivalent Boron concentration change using the Boron differential worth. The "all rods out" Boron concentration is the sum of the measured Boron concentration and the equivalent Boron from the reactivity measurement.

The Critical Boron Concentration at a condition other than "all rods out" was determined after the Control Rod Reactivity Worth Measurements had been made. The predicted Critical Boron Concentration was determined for the case of Hot Zero Power, "all rods out", and Group 8 at 15% withdrawn. The difference in control rod group worth and APSR worth between the predicted conditions and the measured conditions was then determined. This reactivity worth was then converted to an equivalent Boron concentration change. The Critical Borc Concentration is the sum of the measured Boron concentration and the equivalent Boron concentration change.

3.1.3 Results and Evaluation

The results of the predicted Critical Boron Concentration and measured Critical Boron Concentration for "all rods out" and "rods inserted" conditions are listed in Table 3-1. Both measured Critical Boron Concentrations compared favorably with the predicted values and were within the acceptance criterion of $\pm$ 100 ppmB.

3.2 DETERMINATION OF MODERATOR TEMPERATURE COEFFICIENT

3.2.1 Purpose

The purpose of this test was to determine the moderator temperature coefficient of reactivity at Hot Zero Power. The values measured are used to verify that the moderator temperature coefficient is within Technical Specification limits, that the moderator temperature coefficient agrees within specified limits of predicted values in the Physics Test Manual and to provide data for the curves in OP 1103.15, Reactivity Balance Calculation.

3.2.2 Test Method

The moderator temperature coefficient at Hot Zero Power was measured by two methods. In both methods the first step was to achieve steady state critical conditions at approximately 10^{-9} amps on the intermediate range detectors. The first method used a Reactivity Calculator to measure reactivity changes as T_{ave} was varied by adjusting the turbine header pressure setpoint. The second method used control rod worth curves measured per Control Rod Reactivity Worth Measurements, to determine the reactivity changes as T_{ave} was varied while just critical conditions were maintained at 10^{-9} amps by control rod motion.

The moderator temperature coefficient at hot conditions with all rods withdrawn was measured using the Reactivity Calculator method. The moderator temperature coefficient of reactivity at hot conditions with regulating control rod assembly groups inserted was measured after the Control Rod Reactivity Worth Measurements were made and utilized both Reactivity Calculator and Rod Swap Methods.

3.2.3 Results and Evaluation

The measured and predicted values of moderator temperature coefficient for the conditions of "all rods out" and "rods inserted" are listed in Table 3-2. The measured values of moderator temperature coefficient compared favorably with the predicted values and all are within the acceptance criteria of $\pm$ 0.4 X 10^{-4} $\Delta K/K/^{o}F$ of the predicted values and less than + 0.5 X 10^{-4} $\Delta K/K/^{o}F$ at Hot Zero Power conditions. The extrapolation of the moderator temperature coefficient to 95% of full power indicated it would be negative for all expected Boron concentrations and allowable control rod configurations.

3.3 CONTROL ROD REACTIVITY WORTH MEASUREMENT

3.3.1 Purpose

The purpose of this test was to determine the integral worth of the regulating control rods and Axial Power Shaping Rods at Hot Zero Power ($\approx 10^{-9}$ amps) for the purpose of updating the integral control rod worth curves in the Reactivity Balance Calculation and for comparison with the predicted worths. It also was to determine the adequacy of the shutdown margin analysis for the reload core.

3.3.2 Test Method

The initial Boron concentration of the RCS was first determined by sampling. Then, using the predicted control rod worths from the Physics Test Manual, the amount of Boron dilution required to bring the control rods from the "all rods out" condition to the normal operating configuration, Group 6 at 88% withdrawn and Group 8 at its maximum worth, was determined. Deboration was initiated and the reactor was maintained critical at approximately 10^{-9} amps by insertion of Group 8 until it was approximately at its maximum worth, then by Group 7/6 insertion in sequence, while making concurrent reactivity change measurements, until deboration was complete. Frequent sampling of the RCS and MU Tank during deboration was required to insure that the Boron concentration versus time would be known. The control rod insertion stopped at approximately the normal full power regulating group configuration and APSR integral worth curve was determined by rod swap with the controlling group(s).

The Reactor was then deborated by the amount required to obtain criticality at approximately 10^{-9} amps with CRA Group 5 at approximately 0% wd. Then, using both reactivity measurements and differential Boron worths and position of the CRA groups versus Boron concentration, the reactivity worth versus CRA Group position was determined. The APSR group worth was determined by comparing APSR motion with controlling group position change during the APSR versus controlling group swap.

3.3.3 Results and Evaluation

The results of both the predicted control rod group worths and the measured control rod group worths are tabulated in Table 3-3. The measured and predicted values of control rod group worths compared favorably. The CRA group worths measured were within the criterion range of ± 15% of predicted values per the Physics Test Manual and the total Group 5, 6 and 7 worths were well within the ± 10% acceptance limit.

3.4 EJECTED ROD WORTH MEASUREMENT

3.4.1 Purpose

The purpose of this test was to determine the reactivity worth of the worst case ejected control rod as specified by the Cycle 2 Reload Report, to verify its worth is less than 1.0% ΔK/K at Hot Zero Power and that it is in agreement with the predicted value in the Physics Test Manual.

3.4.2 Test Method

Initial conditions were established with the Reactor critical at approximately 10^{-9} amps, Regulating Groups at approximately 0% wd, Group 8 at its maximum worth and Boron concentration at equilibrium. The initial (steady state) Boron concentration of the RCS was determined by sampling. Then, using the predicted ejected rod worth from the Physics Test Manual, the amount of Boron addition required to bring the worst case ejected rod, Control Rod 6-4, to 100% withdrawn was determined. Boration was initiated and the Reactor was maintained critical at 10^{-9} amps by withdrawal of Rod 6-4. Frequent sampling of the Reactor Coolant System (RCS) and Makeup Tank (MU) during boration was required so that Boron concentration versus time would be known. The rod withdrawal stopped at 100% wd and additional reactivity compensation was made by withdrawal of Group 5. When steady state Boron concentration was re-established, Control Rod 6-4 was returned to 0% wd by using Group 5 withdrawal for reactivity compensation. The reactor was maintained critical at 10^{-9} amps during the swap.

Using reactivity measurements, the differential Boron worth, and the position of control rods involved, the worst case ejected rod worth was determined.

3.4.3 Results and Evaluation

The measured worth of the worst case ejected rod, Control Rod 6-4, compared favorably with the predicted value and its worth met the acceptance criterion of <1.0% $\Delta K/K$. The test results are tabulated in Table 3-4.

TABLE 3-1

CRITICAL BORON CONCENTRATION AT HOT ZERO POWER

Condition	Measured Value	Vendor Predicted Value	In-house Predicted Value
All Rods Out	1334 ppmB	1320 ppmB	1286 ppmB
Rods Inserted (Group 8 @ 15% w/d)	1046 ppmB	1064 ppmB	967 ppmB

TABLE 3-2

MODERATOR TEMPERATURE COEFFICIENT AT HOT ZERO POWER

Condition	Measured Value		Vendor Predicted Value	In-house Predicted Value
	Reactivity Calculator	Rod Swap		
All Rods Out	-0.0224×10^{-4} ΔK/K/°F	NA	-0.115×10^{-4} ΔK/K/°F	-0.0269×10^{-5} ΔK/K/°F
Rods Inserted	-0.794×10^{-4} ΔK/K/°F	-0.726×10^{-4} ΔK/K°F	-0.73×10^{-4} ΔK/K/°F	-0.78×10^{-5} ΔK/K/°F

TABLE 3-3

CONTROL ROD REACTIVITY WORTHS

CRA Group	Measured Worth (%ΔK/K)	Vendor Predicted Worth (%ΔK/K)	In-house Predicted Value (%ΔK/K)	% Error Between Measured and Vendor Predicted Values
5	0.650	0.690	0.74	-5.8
6	0.868	0.830	0.89	+4.4
7	0.972	1.010	1.11	-3.8
8	0.493	0.430	0.55	+14.7
Totals 5-7	2.490	2.530	2.74	-1.6

TABLE 3-4

EJECTED CONTROL ROD WORTH

CRA/CORE GRID	Measured Value		Predicted Vendor Value
	Boron Swap	Rod Swap	
6-4/N-12	0.582%ΔK/K	0.511%ΔK/K	0.54%ΔK/K

4.0 CORE POWER DISTRIBUTION TEST

4.1 Purpose

The objective of the Core Power Distribution Test was to measure the power distribution of the reactor core at the major power plateaus of 40%, 75% and 100% full power during power escalation in order to verify that the DNBR, linear heat rate, quadrant power tilt and power peaking factors did not exceed allowable limits.

The limits placed on the measured parameters were as follows:

i) The maximum linear heat rate, LHR, in the core is less than the LOCA limit per Technical Specifications for the axial location of the peak. When testing at a power level below rated power, the maximum LHR when extrapolated to rated power must also meet this criterion.

ii) The minimum DNBR must be greater than 1.30 at rated power conditions and when extrapolated to rated power conditions from a lesser test plateau.

iii) The quadrant power tilt must not exceed the value allowed in the Technical Specifications.

iv) The highest measured radial and total power peaking factors shall not exceed the highest predicted peaks by more than 5% and 7.5% at the 75% and 100% power plateaus, respectively.

These acceptance criteria are established to verify that core nuclear and thermal hydraulic calculational models are conservative with respect to measured conditions thereby verifying the acceptability of data from these models for input to safety analysis. The acceptance criteria also serve to verify acceptable operating conditions at each test plateau and eventually at rated power conditions.

4.2 Test Method

Equilibrium conditions were established at 75% and 100% FP ensuring that Xenon was in three-dimensional equilibrium (equilibrium Xenon was not required for the 40% tests) with no APSR motion and minimal power fluctuations and/or controlling rod group motion. The incore monitoring system and the plant computer were used for data collection.

4.3 Results and Evaluation

A summary of the test results is given in Table 4-1. This table indicates that the minimum, conservative DNBR at rated power was 3.06 which is well above the lower limit of 1.30. The worst case conservative linear heat rate was 11.8 kw/ft which is significantly less than the Technical Specification LOCA limit per Figure 3.5.2.4. The measured and total power peaking factors were all within the acceptance criteria.

Figure 4-1 shows the core grid/Self Powered Neutron Detector (SPND) string correlation for the core locations used to measure the radial and total power peaking factors. The results of the power distribution measurements are tabulated in Figures 4-2 for the 40% FP plateau,

Table 4-1

SUMMARY OF RESULTS

DATE TIME	4/5/77 1238	4/7/77 0740	4/11/77 1145
POWER LEVEL (%)	40	75	100
GROUP 1-5 (%w/d)	100	100	100
GROUP 6 (%w/d)	89.2	88.6	85.3
GROUP 7 (%w/d)	12.0	8.8	8.6
GROUP 8 (%w/d)	47.0	34.1	30.7
CORE BURNUP (EFPD)	0.2	2.1	5.1
BORON CONCENTRATION (ppmB)	970	787	729
AXIAL IMBALANCE (%FP)	-1.7	1.2	-0.9
MAX QUADRANT PWR TILT (%) (Incore Detectors)	-3.95	-2.17	-2.08
DNBR	9.42	4.38	3.06
LHR	5.29	9.11	11.80
MAX RADIAL PWR PEAK	1.417	1.394	1.394
MAX TOTAL PWR PEAK	NA	1.760	1.704
MAX PEAK AT CORE GRID/LEVEL	E-5/3	M-11/6	M-11/6
EQUILIBRIUM XENON	NO	YES, 3-D	YES, 3-D

FIGURE 4-1

CORE SPND STRING/CORE GRID CROSS REFERENCE

H-8 1	H-9 2	F-8 4	H-5 10	N-8 14	H-13 21	B-8 30	H-1 37
	G-9 3	F-7 6	E-9 5	K-12 20	C-9 29	B-7 31	R-7 45
		L-6 12	M-10 17	D-10 27	C-10 28	P-6 44	R-10 46
			E-11 26	D-5 33	0-5 42	M-14 49	
				N-4 41	0-12 48	D-14 51	
					C-13 52		

FIGURE 4-2
COMPARISON OF PREDICTED AND CALCULATED STEADY STATE RELATIVE RADIAL POWER DISTRIBUTION AT 40 % FP, NOT EQUILIBRIUM XENON

Measurement Conditions

Control Rod Group Positions		
Gps 1-4	100	% wd
Gp 5	100	% wd
Gp 6	89.2	% wd
Gp 7	12.0	% wd
Gp 8	47.0	% wd

Core Power Level	39.7	%FP
Boron Concentration	970	ppm
Core Burnup	0.2	EFPD
Axial Imbalance	-1.7	% FP
Max Quadrant Tilt	-3.95	%

X.XXX	Predicted Values
X.XXX	Measured Values

					0.532	0.576	0.629	0.596	0.532					
					0.607	0.656	0.641	0.652	0.601					
			0.729	1.024	1.077	0.707	0.796	0.707	1.077	1.024	0.729			
			0.798	1.134	1.153	0.735	0.716	0.730	1.123	1.122	0.787			
		0.771	1.137	1.298	0.982	0.611	0.835	0.611	0.982	1.298	1.137	0.771		
		0.805	1.280	[illegible]	0.956	0.588	0.865	0.583	0.993	1.368	1.267	0.712		
	0.731	1.139	1.010	1.347	1.156	1.342	1.202	1.342	1.156	1.347	1.010	1.139	0.731	
	0.798	1.280	1.073	1.340	1.239	1.341	1.214	1.336	1.231	1.329	1.061	1.268	0.789	
	1.028	1.302	1.352	1.401	1.070	1.120	1.038	1.120	1.070	1.401	1.352	1.302	1.028	
	1.133	1.356	1.382	1.417	1.061	1.019	1.066	1.102	1.056	1.410	1.375	1.348	1.127	
0.534	1.082	0.989	1.170	1.105	1.054	1.167	0.996	1.167	1.054	1.105	1.170	0.989	1.082	0.534
0.606	1.118	[illegible]	1.218	1.141	[illegible]	1.051	0.917	1.051	[illegible]	1.148	1.217	[illegible]	1.117	0.606
0.599	0.711	0.616	1.357	1.136	1.172	1.186	1.043	1.186	1.172	1.136	1.357	0.616	0.711	0.599
0.653	0.671	0.601	1.372	[illegible]	1.046	1.032	[illegible]	[illegible]	1.051	1.060	1.380	0.607	0.679	0.657
0.632	0.800	0.841	1.214	1.049	1.000	1.044	1.088	1.044	1.000	1.049	1.214	0.841	0.800	0.632
0.634	0.845	0.851	1.197	1.047	0.934	0.961	[illegible]	1.107	0.974	1.063	1.214	0.868	0.801	0.646
0.599	0.711	0.616	1.357	1.136	1.172	1.186	1.043	1.186	1.172	1.136	1.357	0.616	0.711	0.599
0.643	0.677	0.594	1.354	[illegible]	1.030	1.022	0.917	1.030	1.045	1.040	1.377	0.611	0.701	0.661
0.534	1.082	0.989	1.170	1.105	1.054	1.167	0.996	1.167	1.054	1.105	1.170	0.989	1.082	0.534
0.581	1.012	0.965	1.194	1.115	[illegible]	1.024	0.875	1.034	0.961	1.143	1.216	0.980	1.121	0.610
	1.028	1.302	1.352	1.401	1.070	1.120	1.038	1.120	1.070	1.401	1.352	1.302	1.028	
	1.100	1.314	1.356	[illegible]	1.017	0.987	1.034	1.101	1.040	1.402	1.373	1.351	1.131	
	0.731	1.139	1.010	1.347	1.156	1.342	1.202	1.342	1.156	1.347	1.090	1.139	0.731	
	0.765	1.234	1.021	[illegible]	1.183	[illegible]	1.177	1.308	1.213	1.319	1.058	1.271	0.793	
		0.771	1.137	1.298	0.982	0.611	0.835	0.611	0.982	1.298	1.137	0.771		
		0.764	[illegible]	1.326	0.946	[illegible]	[illegible]	0.553	0.970	1.357	1.264	0.715		
			0.729	1.024	1.077	0.707	0.796	0.707	1.077	1.024	0.729			
			0.757	[illegible]	1.084	0.671	0.759	0.702	1.105	1.113	0.783			
					0.532	0.576	0.629	0.516	0.532					
					[illegible]	[illegible]	[illegible]	[illegible]	[illegible]					

FIGURE 4-3

COMPARISON OF PREDICTED AND CALCULATED STEADY STATE RELATIVE RADIAL POWER DISTRIBUTION AT 75 % FP, EQUILIBRIUM XENON

Measurement Conditions

Control Rod Group Positions

- Gps 1-4 100 % wd
- Gp 5 100 % wd
- Gp 6 88.6 % wd
- Gp 7 8.8 % wd
- Gp 8 34.1 % wd

Core Power Level 74.3 %FP
Boron Concentration 787 ppm
Core Burnup 2.1 EFPD
Axial Imbalance 1.2 % FP
Max Quadrant Tilt -2.17 %

X.XXX	Predicted Values
X.XXX	Measured Values

					0.568	0.640	0.674	0.640	0.568					
					0.606	0.654	0.669	0.654	0.606					
			0.747	1.042	1.106	0.737	0.825	0.737	1.106	1.042	0.747			
			0.792	1.141	1.095	0.745	0.811	0.745	1.095	1.141	0.792			
		0.780	1.128	1.280	0.986	0.617	0.843	0.617	0.986	1.280	1.128	0.780		
		0.791	1.258	1.316	0.979	0.586	0.861	0.586	0.984	1.316	1.258	0.791		
	0.749	1.130	1.076	1.313	1.140	1.327	1.188	1.327	1.140	1.313	1.076	1.130	0.749	
	0.792	1.258	0.996	1.310	1.210	1.304	1.190	1.304	1.210	1.310	0.996	1.258	0.792	
	1.046	1.284	1.318	1.359	1.054	1.106	1.029	1.106	1.054	1.359	1.318	1.284	1.046	
	1.141	1.328	1.352	1.394	1.051	1.015	1.037	1.029	1.051	1.394	1.352	1.328	1.141	
0.570	1.111	0.992	1.153	1.088	1.038	1.142	0.982	1.142	1.038	1.088	1.153	0.9[illegible]	1.111	0.570
0.606	1.102	0.991	1.211	1.136	0.970	1.042	0.882	1.042	0.970	1.136	1.211	0.995	1.102	0.606
0.643	0.740	0.622	1.340	1.121	1.147	1.157	1.024	1.157	1.147	1.121	1.340	0.622	0.740	0.643
0.654	0.704	0.610	1.370	1.040	1.047	1.042	0.930	1.042	1.047	1.060	1.370	0.610	0.704	0.654
0.677	0.829	0.849	1.199	1.038	0.986	1.025	1.068	1.025	0.986	1.038	1.199	0.849	0.829	0.677
0.669	0.811	0.861	1.190	1.037	0.882	0.930	0.977	0.930	0.882	1.037	1.190	0.861	0.811	0.669
0.643	0.740	0.622	1.340	1.121	1.147	1.157	1.024	1.157	1.147	1.121	1.340	0.622	0.740	0.643
0.654	0.704	0.610	1.370	1.010	1.047	1.042	0.930	1.042	1.047	1.047	1.370	0.610	0.704	0.654
0.570	1.111	0.992	1.153	1.088	1.038	1.142	0.982	1.142	1.038	1.088	1.153	0.992	1.111	0.570
0.606	1.102	0.966	1.211	1.136	0.970	1.042	0.882	1.042	0.970	1.136	1.211	0.986	1.102	0.606
	1.046	1.284	1.318	1.359	1.054	1.106	1.029	1.106	1.054	1.359	1.318	1.284	1.046	
	1.141	1.328	1.352	1.394	1.051	0.999	1.037	1.028	1.051	1.394	1.352	1.328	1.141	
	0.749	1.130	1.076	1.313	1.140	1.327	1.188	1.327	1.140	1.313	1.076	1.130	0.749	
	0.792	1.258	0.996	1.310	1.210	1.304	1.190	1.304	1.210	1.310	0.996	1.258	0.792	
		0.780	1.128	1.280	0.986	0.617	0.843	0.617	0.986	1.280	1.128	0.780		
		0.791	1.258	1.316	0.951	0.586	0.861	0.586	0.974	1.316	1.258	0.791		
			0.747	1.042	1.106	0.737	0.825	0.737	1.106	1.042	0.747			
			0.792	1.141	1.095	0.745	0.811	0.745	1.095	1.141	0.792			
					0.568	0.640	0.674	0.640	0.568					

FIGURE 4-4

COMPARISON OF PREDICTED AND CALCULATED STEADY STATE TOTAL PEAK POWER DISTRIBUTION AT 75 % FP, EQUILIBRIUM XENON

Measurement Conditions

Control Rod Group Positions

Gps 1-4	100	% wd
Gp 5	100	% wd
Gp 6	88.6	% wd
Gp 7	8.8	% wd
Gp 8	34.1	% wd

Core Power Level	74.3	%FP
Boron Concentration	787	ppm
Core Burnup	2.1	EFPD
Axial Imbalance	1.2	% FP
Max Quadrant Tilt	-2.17	%

X.XXX Predicted Values
X.XXX Measured Values

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						0.679	0.752	0.787	0.752	0.679					
						0.718	0.773	0.775	0.773	0.718					
B				0.899	1.267	1.346	0.838	0.952	0.838	1.346	1.267	0.899			
				0.949	1.353	1.276	0.841	0.963	0.841	1.276	1.353	0.949			
C			0.934	1.351	1.537	1.201	0.835	1.003	0.835	1.201	1.537	1.351	0.934		
			0.952	1.489	1.565	1.135	0.744	1.007	0.744	1.159	1.565	1.489	0.952		
D		0.901	1.353	1.308	1.596	1.537	1.630	1.442	1.630	1.537	1.596	1.308	1.353	0.901	
		0.949	1.489	1.304	1.624	1.582	1.602	1.463	1.602	1.582	1.624	1.304	1.489	0.949	
E		1.272	1.543	1.601	1.644	1.313	1.369	1.263	1.369	1.313	1.644	1.601	1.543	1.272	
		1.353	1.565	1.624	1.760	1.252	1.247	1.271	1.287	1.252	1.760	1.624	1.565	1.353	
F	0.682	1.352	1.209	1.554	1.352	1.282	1.414	1.254	1.414	1.282	1.352	1.554	1.209	1.352	0.68
	0.718	1.276	1.173	1.582	1.252	1.198	1.305	1.193	1.305	1.198	1.252	1.582	1.166	1.276	0.7
G	0.756	0.843	0.842	1.648	1.386	1.420	1.429	1.272	1.429	1.420	1.386	1.648	0.846	0.843	0.75
	0.773	0.841	0.744	1.602	1.299	1.305	1.326	1.204	1.326	1.305	1.313	1.602	0.744	0.841	0.77
H	0.790	0.957	1.009	1.453	1.275	1.259	1.273	1.325	1.273	1.259	1.275	1.453	1.009	0.957	0.79
	0.775	0.963	1.007	1.463	1.271	1.193	1.204	1.270	1.204	1.193	1.271	1.463	1.007	0.963	0.77
K	0.756	0.843	0.842	1.648	1.386	1.420	1.429	1.272	1.429	1.420	1.386	1.648	0.842	0.843	0.75
	0.773	0.841	0.744	1.602	1.266	1.305	1.326	1.204	1.326	1.305	1.301	1.602	0.744	0.841	0.77
L	0.682	1.352	1.209	1.554	1.352	1.282	1.414	1.254	1.414	1.282	1.352	1.554	1.209	1.352	0.68
	0.718	1.276	1.162	1.582	1.252	1.198	1.305	1.193	1.305	1.198	1.252	1.582	1.160	1.276	0.71
M		1.272	1.543	1.601	1.644	1.313	1.369	1.263	1.369	1.313	1.644	1.601	1.543	1.272	
		1.353	1.565	1.624	1.760	1.252	1.247	1.271	1.286	1.252	1.760	1.624	1.565	1.353	
N		0.901	1.353	1.308	1.596	1.537	1.630	1.442	1.630	1.537	1.596	1.308	1.353	0.901	
		0.949	0.489	1.304	1.624	1.582	1.602	1.463	1.602	1.582	1.624	1.304	1.489	0.949	
O			0.934	1.351	1.537	1.201	0.835	1.003	0.835	1.201	1.537	1.351	0.934		
			0.952	1.489	1.565	1.127	0.744	1.007	0.744	1.153	1.565	1.489	0.952		
P				0.899	1.267	1.346	0.838	0.952	0.838	1.346	1.267	0.899			
				0.949	1.353	1.276	0.841	0.963	0.841	1.276	1.353	0.949			
R						0.679	0.752	0.787	0.752	0.679					
						0.718	0.773	0.775	0.773	0.718					

FIGURE 4-5

COMPARISON OF PREDICTED AND CALCULATED STEADY STATE RELATIVE RADIAL POWER DISTRIBUTION AT 100 % FP, EQUILIBRIUM XENON

Measurement Conditions

Control Rod Group Positions		
Gps 1-4	100	% wd
Gp 5	100	% wd
Gp 6	85.3	% wd
Gp 7	8.6	% wd
Gp 8	30.7	% wd

Core Power Level	99.9	%FP
Boron Concentration	729	ppm
Core Burnup	5.1	EFPD
Axial Imbalance	-0.9	% FP
Max Quadrant Tilt	-2.08	%

X.XXX	Predicted Values
X.XXX	Measured Values

					0.580	0.655	0.690	0.655	0.580					
					0.620	0.673	0.691	0.673	0.620					
			0.753	1.044	1.111	0.74	0.836	0.748	1.111	1.044	0.753			
			0.791	1.146	1.105	0.761	0.829	0.761	1.105	1.146	0.791			
		0.785	1.126	1.272	0.987	0.625	0.849	0.625	0.987	1.272	1.126	0.785		
		0.788	1.251	1.307	0.980	0.596	0.871	0.596	0.983	1.307	1.251	0.788		
	0.754	1.127	1.075	1.301	1.131	1.317	1.184	1.317	1.131	1.301	1.075	1.127	0.754	
	0.791	1.251	0.988	1.292	1.180	1.308	1.144	1.308	1.180	1.292	0.988	1.251	0.791	
	1.047	1.275	1.305	1.343	1.049	1.102	1.027	1.102	1.049	1.343	1.305	1.275	1.047	
	1.146	1.332	1.330	1.394	1.052	1.012	1.033	1.027	1.052	1.394	1.330	1.332	1.146	
0.582	1.115	0.993	1.143	1.082	1.034	1.137	0.984	1.137	1.034	1.082	1.143	0.993	1.115	0.582
0.620	1.112	0.990	1.181	1.099	0.976	1.040	0.883	1.040	0.976	1.099	1.181	0.997	1.112	0.620
0.657	0.751	0.630	1.329	1.115	1.142	1.152	1.024	1.152	1.142	1.115	1.329	0.630	0.751	0.657
0.673	0.718	0.621	1.372	1.009	1.044	1.039	0.936	1.039	1.044	1.048	1.372	0.621	0.718	0.673
0.692	0.840	0.854	1.193	1.037	0.988	1.025	1.066	1.025	0.988	1.037	1.193	0.854	0.840	0.692
0.691	0.829	0.871	1.144	1.038	0.883	0.936	0.977	0.936	0.883	1.038	1.144	0.871	0.829	0.691
0.657	0.751	0.630	[illegible]	1.115	1.142	1.152	1.024	1.152	1.142	1.115	1.329	0.630	0.751	0.657
0.673	0.718	0.621	1.372	1.009	1.044	1.039	0.936	1.039	1.044	1.048	1.372	0.621	0.718	0.673
0.582	1.115	0.993	1.143	1.082	1.034	1.137	0.984	1.137	1.034	1.082	1.143	0.993	1.115	0.582
0.620	1.112	0.967	1.181	1.099	0.976	1.040	0.883	1.040	0.976	1.099	1.181	0.992	1.112	0.620
	1.047	1.275	1.305	1.343	1.049	1.102	1.027	1.102	1.049	1.343	1.305	1.275	1.047	
	1.146	1.332	1.330	1.394	1.052	1.003	1.038	1.034	1.054	1.394	1.330	1.332	1.146	
	0.754	1.127	1.075	1.301	1.131	1.317	1.184	1.137	1.131	1.301	1.075	1.127	0.745	
	0.791	1.251	0.988	1.292	1.180	1.308	1.144	1.308	1.180	1.292	0.988	1.251	0.791	
		0.785	1.126	1.272	0.987	0.625	0.849	0.625	0.987	1.272	1.126	0.785		
		0.788	1.251	1.307	0.955	0.596	0.871	0.596	0.978	1.307	1.251	0.788		
			0.753	1.044	1.111	0.748	0.836	0.748	1.111	1.044	0.753			
			0.791	1.146	1.105	0.761	0.829	0.761	1.105	1.146	0.791			
					0.580	0.655	0.690	0.655	0.580					

FIGURE 4-6
COMPARISON OF PREDICTED AND CALCULATED STEADY S[illegible]E TOTAL
PEAK POWER DISTRIBUTION AT 100 % FP, EQUILIBRIUM XENON

Measurement Conditions

Control Rod Group Positions

Gps 1-4	100	% wd
Gp 5	100	% wd
Gp 6	85.3	% wd
Gp 7	8.6	% wd
Gp 8	30.7	% wd

Core Power Level	99.9	%FP
Boron Concentration	729	ppm
Core Burnup	5.1	EFPD
Axial Imbalance	-0.9	% FP
Max Quadrant Tilt	-2.08	%

X.XXX Predicted Values
X.XXX Measured Values

A						0.720	0.802	0.840	0.802	0.720					
						0.704	0.764	0.770	0.764	0.704					
B				0.939	1.318	1.402	0.880	0.976	0.880	1.402	1.318	0.939			
				0.973	1.406	1.289	0.880	0.940	0.880	1.289	1.406	0.973			
C			0.973	1.395	1.574	1.225	0.904	0.978	0.904	1.225	1.574	1.395	0.973		
			0.929	1.491	1.470	1.124	0.812	0.964	0.812	1.124	1.470	1.497	0.929		
D		0.941	1.396	1.335	1.600	1.492	1.645	1.436	1.645	1.492	1.600	1.335	1.396	0.941	
		0.973	1.497	1.211	1.510	1.495	1.620	1.475	1.620	1.495	1.510	1.211	1.497	0.973	
E		1.322	1.579	1.605	1.636	1.268	1.219	1.215	1.219	1.268	1.636	1.605	1.579	1.322	
		1.406	1.470	1.510	1.704	1.234	1.196	1.195	1.232	1.234	1.704	1.510	1.470	1.406	
F	0.723	1.407	1.232	1.509	1.318	1.220	1.322	1.180	1.322	1.220	1.318	1.509	1.232	1.407	0.72
	0.704	1.284	1.183	1.495	1.234	1.145	1.219	1.117	1.219	1.145	1.234	1.495	1.134	1.289	0.70
G	0.805	0.884	0.911	1.661	1.296	1.327	1.346	1.204	1.346	1.327	1.296	1.661	0.911	0.884	0.80
	0.764	0.880	0.812	1.620	1.246	1.219	1.233	1.125	1.233	1.219	1.253	1.620	0.812	0.880	0.76
H	0.843	0.980	0.985	1.44[illegible]	1.227	1.184	1.205	1.268	1.205	1.184	1.227	1.448	0.985	0.980	0.84
	0.770	0.940	0.964	1.475	1.195	1.117	1.125	1.1[illegible]	1.125	1.117	1.195	1.475	0.964	0.940	0.77
K	0.805	0.884	0.911	1.661	1.296	1.327	1.346	1.204	1.346	1.327	1.296	1.661	0.911	0.884	0.8
	0.764	0.880	0.812	1.620	1.209	1.219	1.233	1.125	1.233	1.219	1.247	1.620	0.812	0.880	0.76
L	0.723	1.407	1.232	1.509	1.318	1.220	1.322	1.180	1.322	1.220	1.318	1.509	1.232	1.407	0.72
	0.704	1.289	1.124	1.495	1.234	1.145	1.219	1.117	1.219	1.415	1.234	1.495	1.133	1.289	0.70
M		1.322	1.579	1.605	1.636	1.268	1.219	1.215	1.219	1.268	1.636	1.605	1.579	1.322	
		1.40[illegible]	1.470	1.510	1.704	1.234	1.202	1.195	1.242	1.234	1.704	1.510	1.470	1.406	
N		0.941	1.396	1.335	1.600	1.492	1.645	1.436	1.645	1.492	1.600	1.335	1.396	0.941	
		0.973	1.497	1.211	1.510	1.495	1.620	1.475	1.620	1.495	1.510	1.211	1.491	0.973	
O			0.973	1.3[illegible]5	1.574	1.225	0.904	0.978	0.904	1.225	1.574	1.395	0.973		
			0.929	1.4[illegible]	1.470	1.097	0.812	0.964	0.812	1.121	1.470	1.491	0.929		
P				0.9[illegible]	1.318	1.4[illegible]2	0.880	0.976	0.880	1.402	1.318	0.939			
				0.[illegible]73	1.406	1.2[illegible]9	0.880	0.940	0.880	1.259	1.406	0.973			
R						0.720	0.820	0.540	0.8[illegible]0	0.720					
						0.704	0.764	0.77[illegible]	0.764	0.704					

5.0 POWER IMBALANCE DETECTOR CORRELATION TEST

5.1 Purpose

The Power Imbalance Detector Correlation Test had two objectives - to determine the relationship between out-of-core and incore imbalance and to verify the adequacy of the imbalance system trip setpoints.

5.2 Test Method

Imbalance measurements were made to determine the acceptability of the out-of-core detectors, to detect imbalance, and to establish a basis for verifying that DNBR and LHR limits would not be exceeded while operating within the Flux/Delta Flux/Flow envelope set in the Reactor Protection System. These imbalance measurements were made at different part length rod positions with two different gain factors applied to the multiplier on the Delta Flux amplifier output.

In performing the test, part length rod position was varied and reactivity compensation made by Rod Groups 6 and/or 7. At each required incore imbalance, core power distribution and other data were taken. From these data, plots of incore offset versus out-of-core offset were maintained. In addition, minimum DNBR, and maximum LHR were also obtained to to allow verification of the adequacy of the RCS and LOCA imbalance power limits for LHR and DNBR protection. Offset = (Imbalance/%FP) 100%.

The relationship between incore offset and out-of-core offset has been determined to be a linear equation of the form

$$OCO = M \times GF \times ICO + B$$

Where: OCO = Out-of-Core Offset (percent)

ICO = Incore Offset (percent)

M = Slope of relationship

B = Intercept at Zero ICO

GF = Gain Factor

5.3 Evaluation and Results

Extrapolation, with conservatism, of the maximum Linear Heat Rate and minimum DNBR at the maximum positive and the maximum negative imbalance test points to the overpower trip power yielded the following results:

Imbalance	+7%	-22.6%
Max LHR	12.85 kw/ft	15.42 kw/ft
Min DNBR	3.98	1.73

These are within the acceptable limits on LHR and DNBR of 19.4 kw/ft and 1.30, respectively. The Linear Heat Rates were also extrapolated and compared to the LOCA Imbalance Limits per Tech. Spec. Figure 3.5.2.3-A with the following results which are within the allowable limit per Tech. Spec. Figure 3.5.2-4:

Imbalance	+7%	-22.6%
Max LHR	12.42 kw/ft*	NA**

* Extrapolated to 102% FP

** Test Imbalance is outside the LOCA Imbalance limit, therefore extrapolation is not applicable.

5.4 Conclusions

Upon completion of power imbalance detector correlation testing at 75% FP, the following conclusions were drawn:

A. The relationship between the incore and out-of-core offset is linear with a constant slope, and

B. The imbalance trip envelope as set in the Reactor Protective System will protect the reactor core from exceeding LHR and DNBR limits when a gain factor of 3.20 is set in the circuit. This gain was set and test data was taken to confirm its adequacy.

6.0 DETERMINATION OF REACTIVITY COEFFICIENTS AT POWER

6.1 Purpose

The purpose of this test was to measure the moderator temperature coefficient and power doppler coefficient at 75% and 100% full power and to compare the results with predicted values.

Acceptance criteria for the test were that the power doppler coefficient be more negative than -0.55×10^{-4} $(\Delta K/K)/\%$ FP, and that the moderator temperature coefficient measured at power operating conditions be non-positive above 95% FP.

6.2 Test Method

The moderator temperature coefficient at power operating conditions was measured by varying T_{ave} using the T_{ave} setpoint controller on the Reactor Demand Station and maintaining constant power with the ICS in full automatic. The corresponding control rod motion is related to the reactivity change which is used to determine the moderator temperature coefficient.

The power doppler coefficient is measured by varying Reactor power using the Integrated Control System Unit Load Demand (ICS ULD) station and recording the corresponding control rod motion. The control rod reactivity worth was determined by a differential control rod worth measurement with a reactivity calculator.

6.3 Results and Evaluation

The results of the reactivity coefficients test are summarized in Table 6-1.

The power doppler coefficients at 75% and 100% full power were less negative than predicted but well below the maximum acceptable value. The moderator temperature coefficient was well below the non-positive limit and within the $\pm$ 0.4 X 10^{-4} $\Delta K/K/^{o}F$ acceptance limit.

6.4 Conclusion

The measured values of all reactivity coefficients were well within the acceptable limits. The acceptance criteria of this test were met in full without deficiences.

TABLE 6-1

SUMMARY OF MEASURED AND PREDICTED REACTIVITY COEFFICIENTS AT POWER

Parameter		Reactor Power Level (% Full Power) 75	Reactor Power Level (% Full Power) 100
Control Rod Assembly Group (% Withdrawn)	6	89.2	85.9
	7	12.0	8.6
	8	34.2	30.5
Boron Concentration (ppmB)		787	729
Moderator Temperature Coefficient $\frac{\Delta K/K}{^{o}F}$	Measured	-1.448×10^{-4}	-1.115×10^{-4}
	Vendor Predicted	-1.42×10^{-4}	-1.28×10^{-4}
	In-House Predicted	-1.399×10^{-4}	-1.[illegible]9 x 10^{-4}
Power Doppler Coefficient $\frac{\Delta K/K}{\%\ Full\ Power}$	Measured	-1.493×10^{-4}	-1.114×10^{-4}
	Vendor Predicted	-1.54×10^{-4}	-1.16×10^{-4}
	In-house Predicted	-1.399×10^{-4}	-2.214×10^{-4}

7.0 CONCLUSION

The results and conclusions summarized in the body of this report demonstrate that the Arkansas Nuclear One Unit 1 Cycle 2 reload has been properly designed and the unit can be operated in a manner that will not endanger the health and safety of the public.