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BRUNSWICK UNIT 1 STARTUP TEST RESULTS FINAL SUMMARY REPORT

I. D. POPPEL

GENERAL ELECTRIC

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STARTUP TEST RESULTS
— FINAL SUMMARY REPORT

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TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

LIST OF ILLUSTRATIONS

LIST OF TABLES

LIST OF TABLES (Continued)

ABSTRACT

This report consists of a summary of the startup test program performed at Unit 1 of the Brunswick Steam Electric Plant. It includes results of static and dynamic reactor performance tests of the reactor and related systems within the General Electric scope of supply.

1. INTRODUCTION

1.1 PURPOSE

The purpose of this report is to present a brief summary of the results of all of the General Electric related startup tests performed at Unit 1 of the Brunswick Steam Electric Plant. The startup test program included tests of the static and dynamic performance of the reactor and related systems.

1.2 PLANT DESCRIPTION

The Brunswick Steam Electric Plant Unit 1 reactor is a single cycle boiling water reactor designed by General Electric for the Carolina Power and Light Company. The plant is located near Southport, North Carolina near where the Cape Fear River enters the Atlantic Ocean. It is warranted for 2436 MWt (at which power the generator is warranted for 849 MWe).

1.3 STARTUP TEST PROGRAM

The startup test program began with fuel loading on September 15, 1976 and finished with the MSIV full closure on September 30, 1977. The plant testing consisted of the following successive phases:

Phase 1: Pre-operational testing (not covered in this report)

Phase 2: Fuel loading and open vessel testing

Phase 3: Initial heatup to rated temperature and pressure

Phase 4: Power tests

Phase 5: Warranty tests

During Phase 4 testing the plant was taken to its designed full power operating condition in a safe controlled fashion. Extensive testing was performed at previously specified operating conditions to demonstrate the safe and efficient performance of plant components. The Phase 5 warranty test began with the beginning of the warranty run and successfully concluded 100 hours later; at Brunswick Unit 1, the warranty run was conducted before the Phase 4 power tests were completed.

1.4 STARTUP TEST DESCRIPTION

Documents such as the operating license, technical specifications, plant operating procedures, and equipment manuals control reactor operations during the startup test program. Two documents are supplied by GE-NEBG for implementation of the startup testing of the equipment it supplies: The Startup Test Specification (22A2212) and The Startup Test Instruction (22A2229AC).

The Startup Test Specification is a document issued for review and approval by GE management and is used for planning and scheduling tests. The chosen tests are required either to demonstrate that it is safe to proceed, to demonstrate performance, or to obtain engineering data. This document defines the minimum test program needed for a safe, efficient startup. The purpose, description, and criteria are given for each test together with a description of each test condition.

The Startup Test Instruction is a document written for use in the control room by qualified GE personnel and for trained customer personnel to properly perform and evaluate each startup test. These instructions may be expanded to include additional testing, or they may be reworded to address plant operating procedures, subject to the review and written approval of designated GE and customer personnel.

1.5 STARTUP TEST ACCEPTANCE CRITERIA

The Startup Test Specification and The Startup Test Instruction contain criteria for acceptance of results of that test. There are two levels of criteria identified, where applicable, as Level 1 and Level 2.

The Level 1 criteria include the values of process variables assigned in the design of the plant and equipment. If a Level 1 criterion is not satisfied, the plant is placed in a satisfactory HOLD condition until a resolution is made. Tests compatible with this HOLD condition may be continued. Following resolution, applicable tests must be repeated to verify that requirements of the Level 1 criterion are satisfied.

The Level 2 criteria are associated with expectations in regard to performance of the system. If a Level 2 criterion is not satisfied, operating and testing plans would not necessarily be altered. Investigations of the measurements and of the analytical techniques used for the predictions would be started.

Safety limits, as set forth in plant technical specifications, are not included because there are no planned operations of testing at such levels. By meeting the criteria, startup test results demonstrate agreement with design specifications and predictions.

2. SUMMARY OF THE TEST PROGRAM

2.1 CHRONOLOGY OF STARTUP TESTING

Some dates that were significant during the test program are listed in Table 2.1.1. The completion dates of testing and the startup test report numbers at the specified test conditions are shown in Table 2.2.1. Figures 2.1.1 through 2.1.6 display a histogram (power/pressure versus time) of the test program beginning with reactor startup in November 1976.

2.2 TEST COMPLETION DATES FOR STARTUP TESTS

Refer to Table 2.2.1 for the test completion dates for the complete startup test program for Brunswick Unit 1.

2.3 POWER/FLOW MAP WITH STARTUP TEST CONDITIONS

Refer to Figure 2.3.1 for a plot of the power and flow conditions which specify each test condition.

Table 2.1.1
SIGNIFICANT DATES OF THE STARTUP TEST PROGRAM

Began Fuel Loading	September 15, 1976
Fuel Loading Complete	September 25, 1976
Initial Criticality Shutdown Margin Demonstration	October 8, 1976
First Heatup	November 17, 1976
First Turbine Roll	December 2, 1976
First Generator Synchronization	December 4, 1976
Nominal Finish of Heatup Testing STARTREC Calibration Complete	December 16, 1976
Nominal Finish of Test Condition 1 Testing	December 29, 1976
Nominal Finish of Test Condition 2 Testing	January 6, 1977
Finish of Test Condition 7 Testing	January 31, 1977
Nominal Finish of Test Condition 3 Testing	March 1, 1977
Finish of Test Condition 4 Testing	April 8, 1977
Began Warranty Run	April 16, 1977
Finished Warranty Run	April 20, 1977
Breakdown of Main Generator	April 27, 1977
Generator Outage Ends	June 28, 1977
Resumption of Startup Testing	July 8, 1977
100% Power First Reached After Outage	August 20, 1977
First Brunswick Rod Sequence Exchange at Power	August 27, 1977
MSIV Full Closure End of Test Condition 6 Testing End of Brunswick Unit 1 Test Program	September 30, 1977

Table 2.2.1
STARTUP TEST COMPLETION DATES AND TEST REPORT NUMBERS

STI	Open Vessel	Heatup	TC1	TC2	TC3	TC4	TC5	TC6	TC7
Chemical and Radiochemical	←	12/15/76 1-1	→	← 1/24/77 1-2	1/31/77 →	—	3/3/77 1-3	3/9/77 | 3/12/77 1-4	—
Radiation Measurement	9/7/76 2-1 9/29/76 2-2	12/2/76 2-3	12/16/76 2-4	—	1/12/77 2-5	—	—	3/9/77 2-6	
Fuel Loading	9/15/76 | 9/25/76 3-1	—	—	—	—	—	—	—	—
Shutdown Margin	10/8/76 4-1	—	—	—	—	—	—	—	—
CRD System	9/14/76 — 9/29/76 5-1	11/26/76 — 12/13/76 5-2	12/15/76 — 12/30/76 5-3	—	1/12/77 — 4/6/77 5-4	—	—	—	—
SRM-CRD Sequence	10/7/76 ← | 10/8/76	12/28/76 — → 6-1	—	—	—	—	—	—	—
RWCU System	—	12/2/76 | 12/3/76 7-1	—	—	—	—	—	—	—
RHR System	—	—	—	1/3/77 8-1	—	—	—	4/28/77 8-2	—
Water Level	—	← 12/12/76	12/16/76 → 9-1	—	—	4/8/77 9-2	—	3/9/77 9-4	1/31/77 9-3
IRM	—	10/8/76 11/17/76 12/30/76 10-1	—	—	—	—	—	—	—
LPRM	—	—	12/21/76 11-1	—	1/20/77 11-2	—	—	3/9/77 11-3	—

Table 2.2.1
STARTUP TEST COMPLETION DATES AND TEST REPORT NUMBERS (Continued)

STI	Open Vessel	Heatup	TC1	TC2	TC3	TC4	TC5	TC6	TC7
APRM	—	11/18/76 12-1	12/21/76 12-2	1/2/77 12-3	1/27/77 12-4	—	2/23/77 12-5	3/23/77 12-6 4/17/77 12-7	—
Process Computer	—	11/17/76 13-1	12/1/76 – 12/24/76 13-2	1/26/77 13-3	—	—	—	3/9/77 13-4	—
RCIC	—	11/25/76 12/30/76 14-1	12/24/76 14-2	1/1/77 14-3	—	—	—	—	—
HPCI	—	11/23/76 15-1	—	4/8/77 15-2	4/9/77 15-3	—	—	—	—
Selected Process Temperatures	—	12/30/76 16-1	—	—	1/13/77 2/1/77 16-2	—	—	4/23/77 – 4/24/77 16-3	—
System Expansion	—	12/1/76 – 12/2/76 17-1	—	—	—	—	—	1/2/77 — 4/24/77 17-2	—
Power Distribution	—	—	—	1/3/77 18-1	1/25/77 18-2	—	—	3/9/77 18-3 18-4	—
Core Performance	—	—	12/23/76 19-1	1/3/77 19-2	1/20/77 19-3	4/8/77 19-6	3/3/77 19-4	3/9/77 19-5	—
Steam Production	—	—	—	—	—	—	—	4/16/77 – 4/20/77 20-1	—
Flux Response	—	—	12/15/76 21-1	1/1/77 21-2	—	4/8/77 21-3	—	—	—
Pressure Regulator	—	—	12/15/76 22-1 (TC1–TC2)	1/3/77	3/1/77 22-2	4/8/77 22-3	3/1/77 – 3/3/77 22-4	3/18/77 4/22/77 22-5	—

Table 2.2.1
STARTUP TEST COMPLETION DATES AND TEST REPORT NUMBERS (Continued)

STI	Open Vessel	Heatup	TC1	TC2	TC3	TC4	TC5	TC6	TC7
Feedwater System									
Setpoint Change	—	—	12/16/76 23-1 7/8/77 23-7	← 1/29/77 23-2 7/9/77 23-7	→ 7/9/77 23-7	—	← 3/2/77 23-3 4/24/77 23-5	3/17/77 → 8/26/77 23-7	—
Pump Trip	—	—	—	—	—	—	—	3/12/77 23-4	—
Loss of Feedwater Heating	—	—	—	—	—	—	—	3/13/77 23-6	—
Turbine Valve Surveillance									
Bypass Valves	—	—	12/16/76 24-1	1/3/77 24-2	1/13/77 24-4	—	3/2/77 24-5	3/10/77 24-6	—
Control Valves	—	—	← 12/17/76 24-3	1/3/77 →	—	—	← 3/3/77 24-7	4/19/77 →	—
MSIV Each Valve	—	← 12/2/76 25-1	12/16/76 →	1/2/77 25-2	1/13/77 25-3	—	← 3/3/77 25-4	→	—
Full Isolation	—	—	—	—	—	—	—	9/30/77 25-5	—
Relief Valves	—	11/24/76 26-1	← 12/24/76 26-2	1/27/77 →	—	—	—	—	—
Load Rejection	—	—	—	1/4/77 27-1	—	—	—	2/28/77 27-3	—
Turbine Trip	—	—	—	—	2/3/77 2/21/77 27-2	—	—	—	—
Shutdown Outside The Control Room	—	—	12/28/76 28-1	—	—	—	—	—	—
Flow Control	—	—	← 1/31/77 29-1		2/28/77 →	—	← 3/12/77 29-2	4/23/77 →	—

Table 2.2.1
STARTUP TEST COMPLETION DATES AND TEST REPORT NUMBERS (Continued)

STI	Open Vessel	Heatup	TC1	TC2	TC3	TC4	TC5	TC6	TC7
Recirc System									
Motor Breaker Trip	—	—	—	—	2/1/77 30-2	—	—	—	—
Field Breaker Trip	—	—	—	—	—	—	—	4/24/77 30-3	—
Two Pump Trip	—	—	—	—	—	—	—	4/23/77 30-4	—
Data	—	—	12/22/76 30-1	—	1/25/77 30-1	4/8/77 30-6	—	—	1/31/77 30-5
Loss of Turbine-Generator and Offsite Power	—	—	—	1/4/77 31-1	—	—	—	—	—
MG Set Speed Control	—	—	12/23/76 32-1	1/29/77 32-3	1/15/77 32-2	—	—	4/21/77 32-4	—
Stop Valve Surveillance	—	—	—	—	1/14/77 33-1	—	3/3/77-3/8/77 33-2		—
Vibration Measurements	—	11/23/76 1/10/77 34-1	—	2/1/77 34-2		—	4/23/77 4/24/77 34-4		1/31/77 34-3
Recirc Flow Calibration	—	—	—	—	1/25/77 35-1	—	—	3/22/77 35-2	—

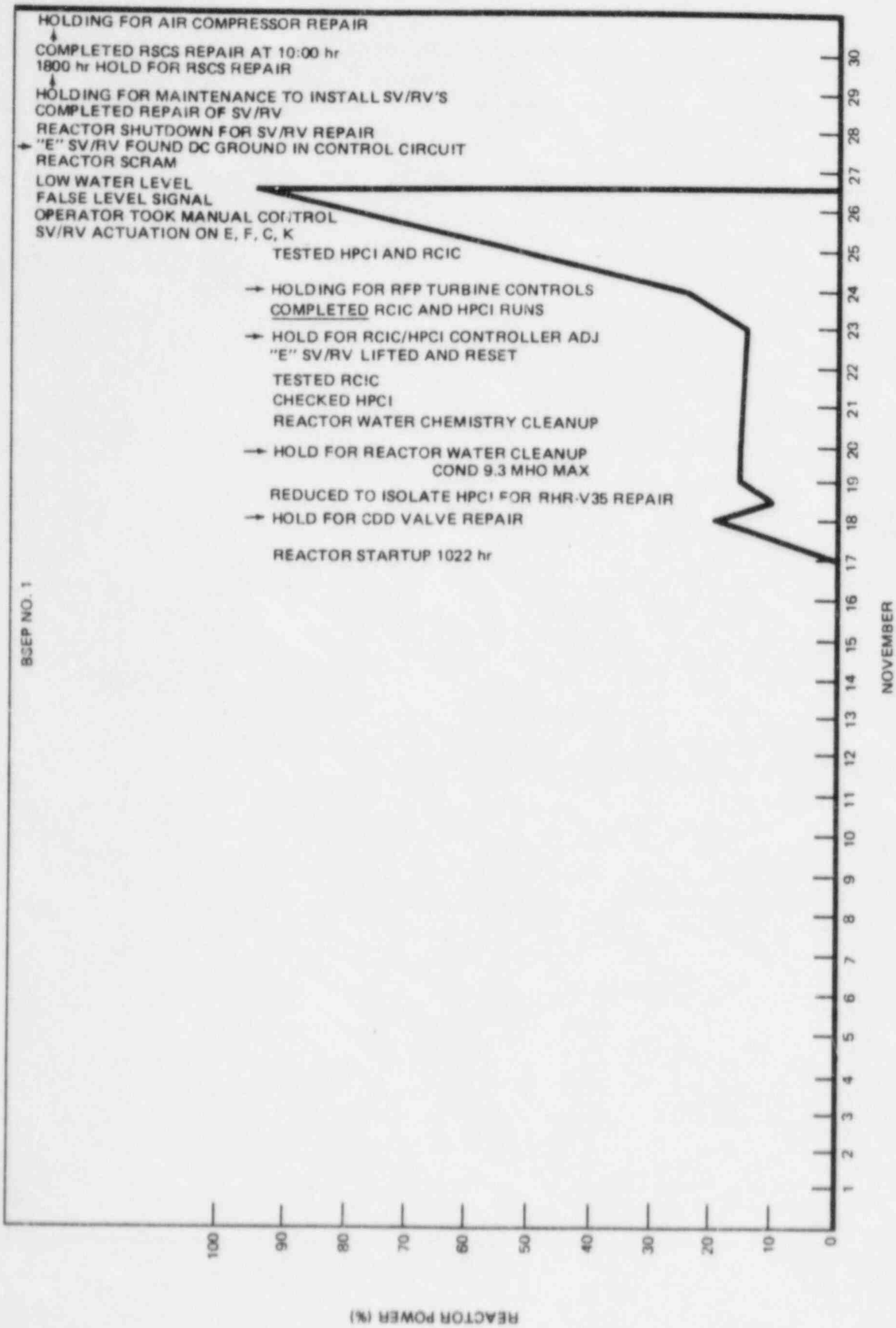


Figure 2.1.1. *Histogram of the Startup Test Program — November.*

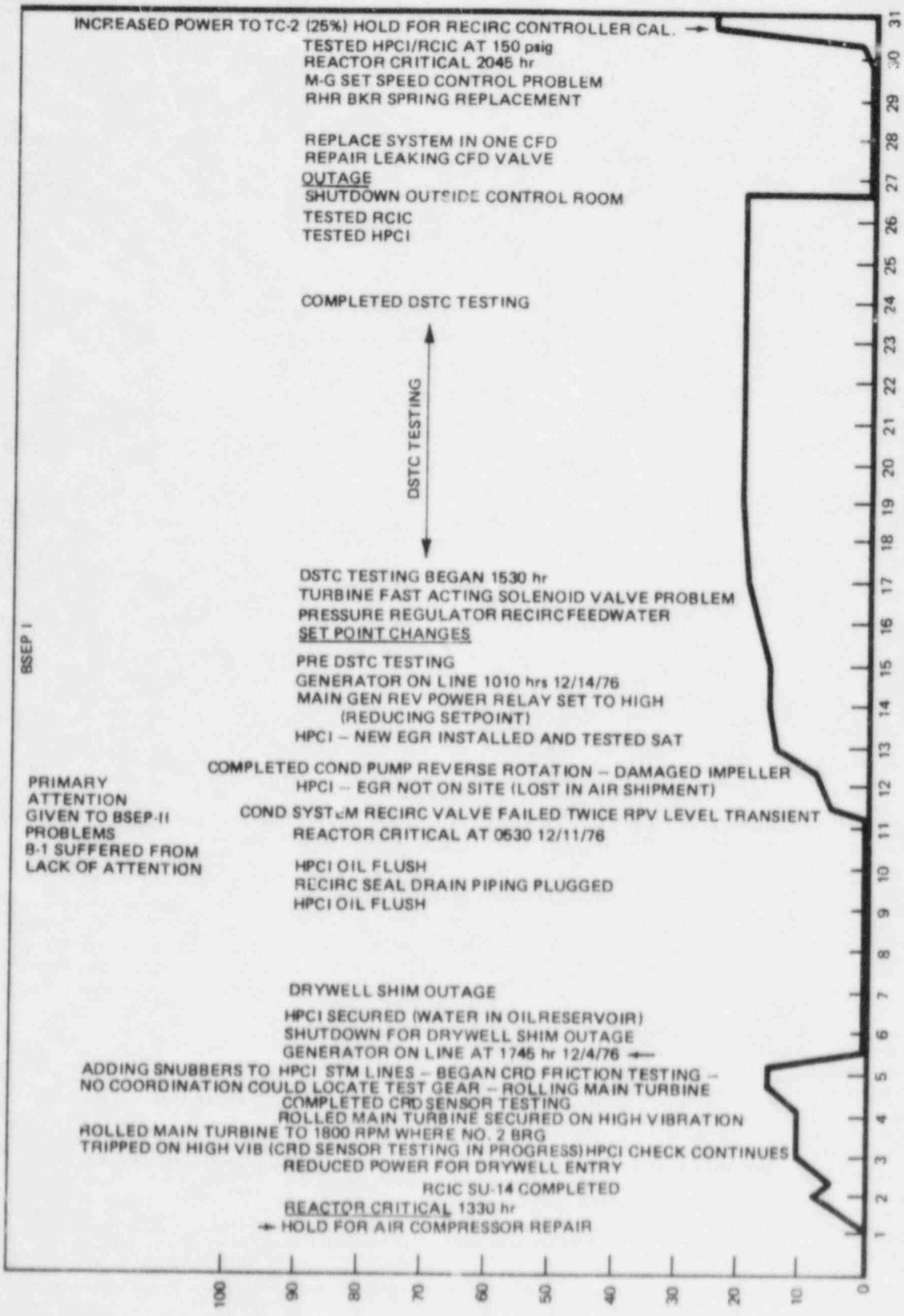


Figure 2.1.2. Histogram of the Startup Test Program — December

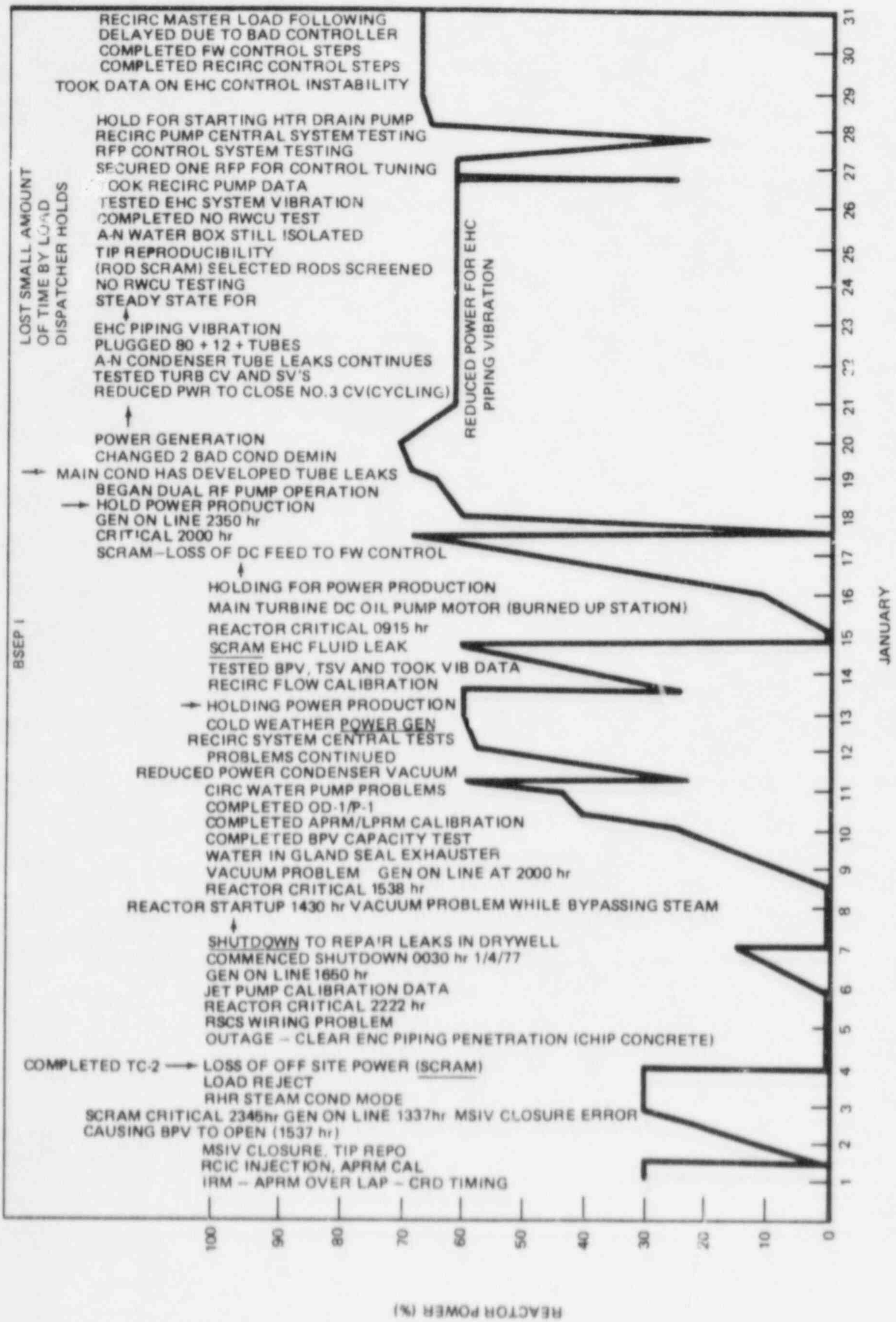
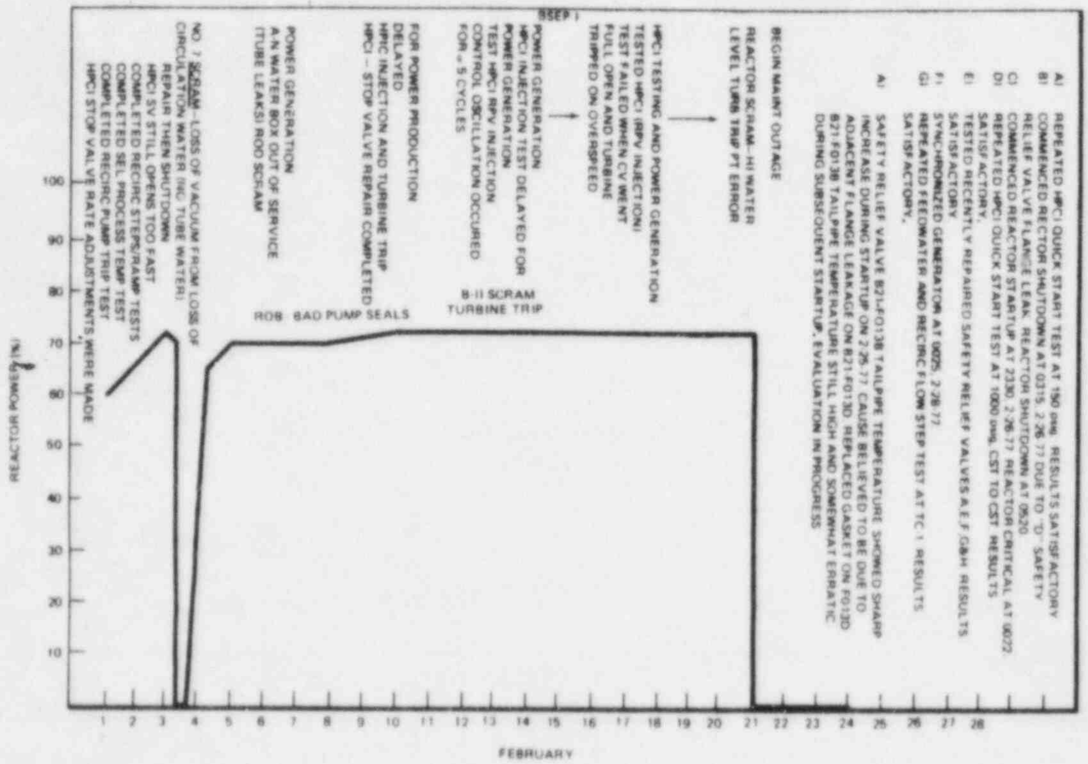
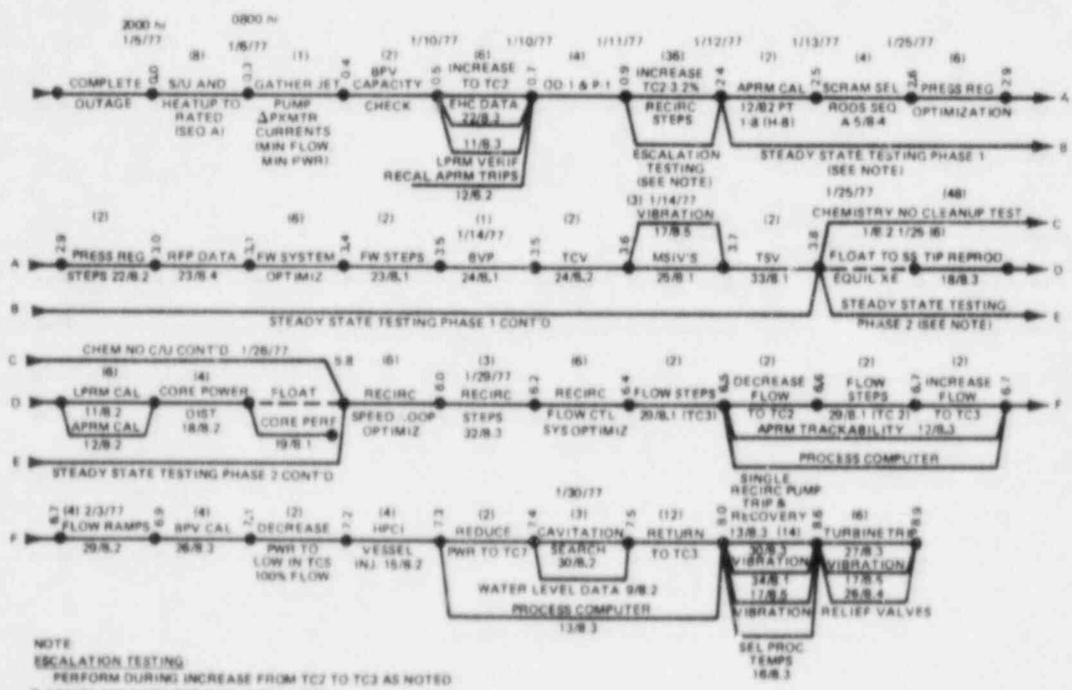


Figure 2.1.3. Histogram of the Startup Test Program — January.





NOTE

ESCALATION TESTING

PERFORM DURING INCREASE FROM TC2 TO TC3 AS NOTED

- RECIRC SYS DATA 32/8.2 (2% SPEED STEPS)
- LPRM IMPACT DATA STP 77-1 (10% CORE FLOW STEPS)
- EHC DATA 22/8.3 (5% POWER STEPS)
- CORE FLOW CALC 35/8.1 (AS 100% APPROACHED)
- TCV 24/8.2 (@50% POWER)

STEADY STATE TESTING PHASE 1

- CHEMISTRY 1/8.1, 1/8.3
- RADIATION MEASUREMENT 2/8.2, 2/8.3

STEADY STATE TESTING PHASE 2

- SEL PROC TEMPS 16/8.2
- CORE FLOW CALC 35/8.1
- VIBRATION 34/8.1 (50-60% PWR)
- RECIRC SYSTEM DATA 30/8.1

Figure 2.1.4. Histogram of the Startup Test Program — February.

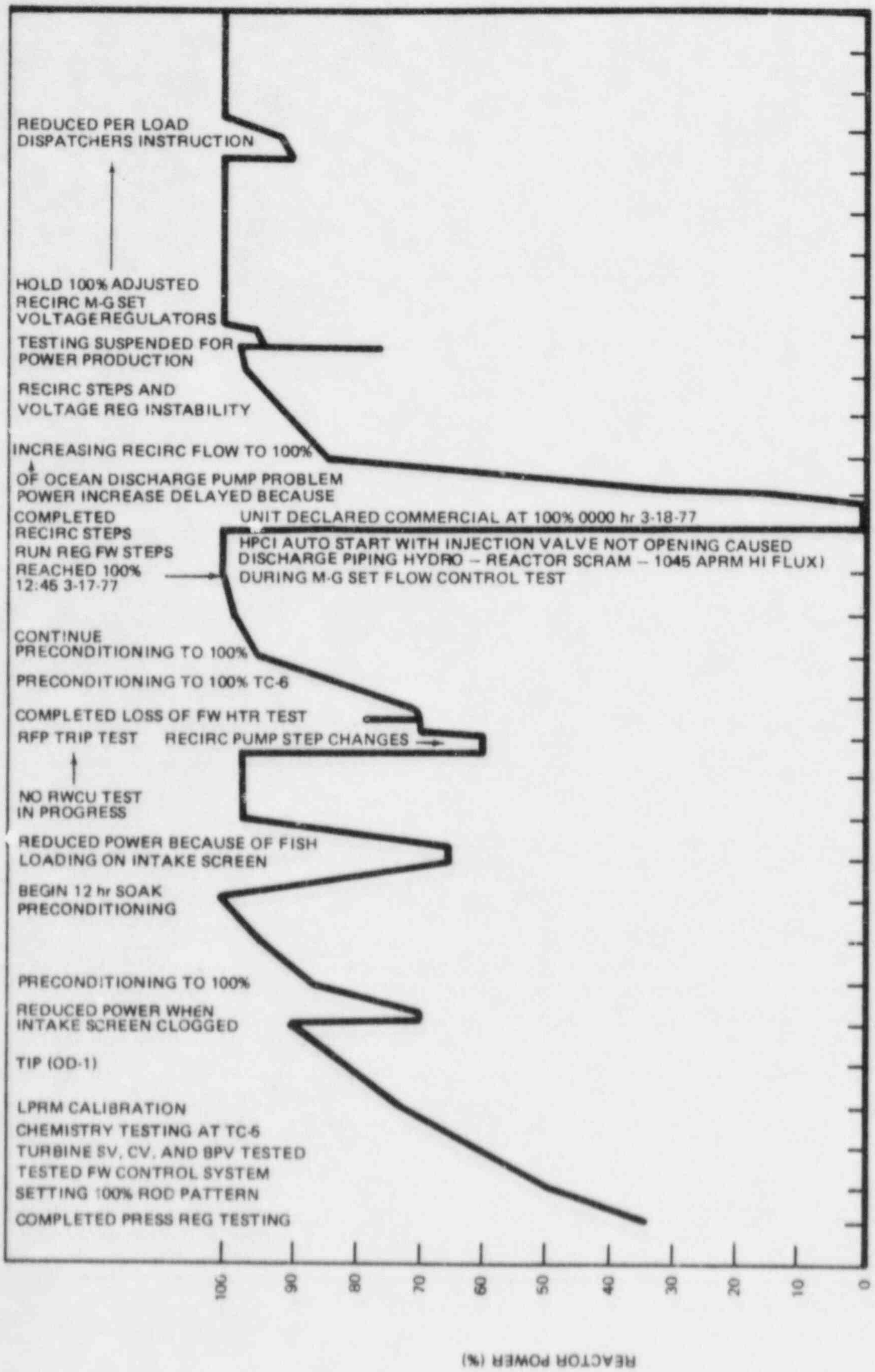


Figure 2.1.5. Histogram of the Startup Test Program — March.

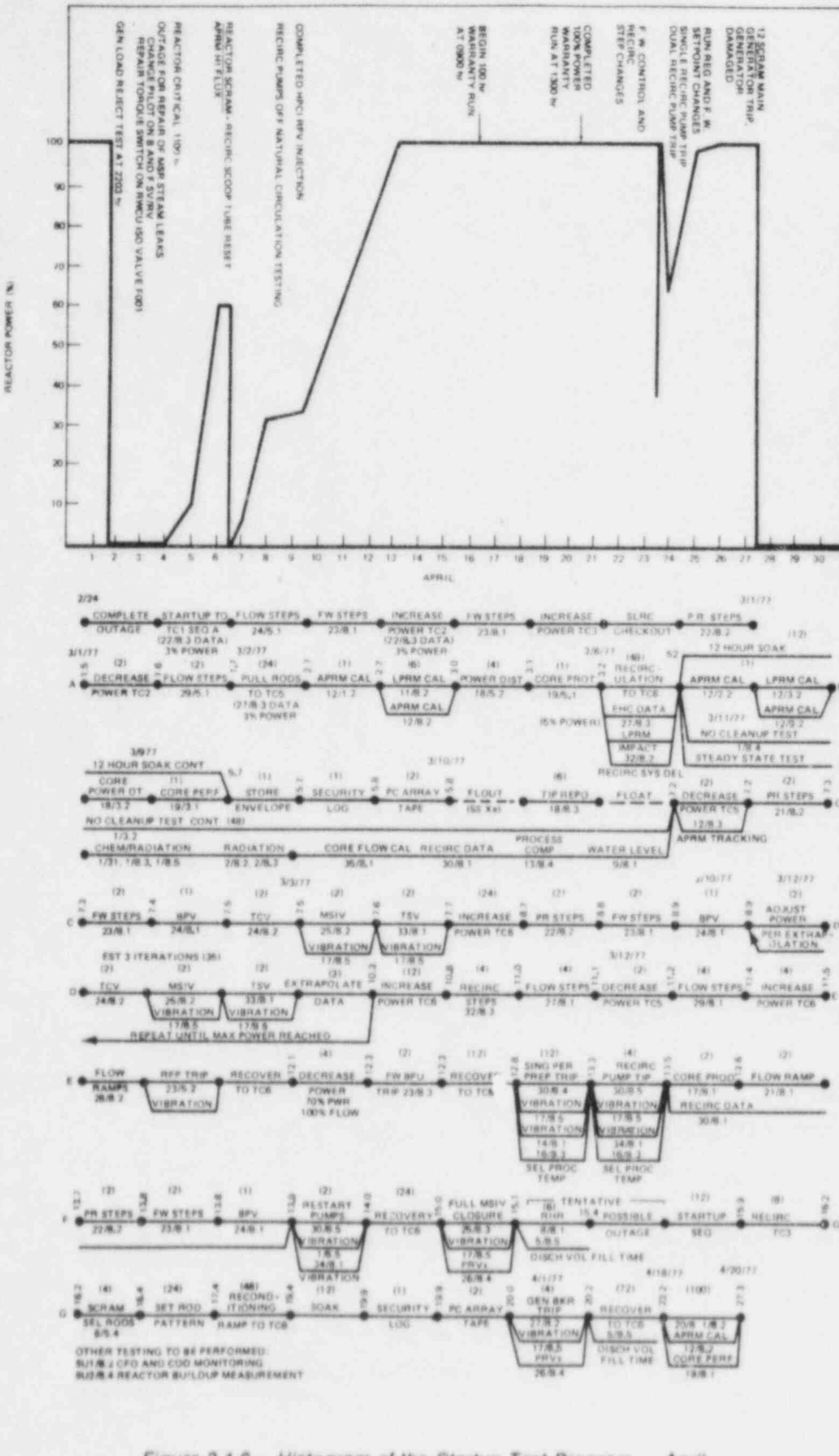
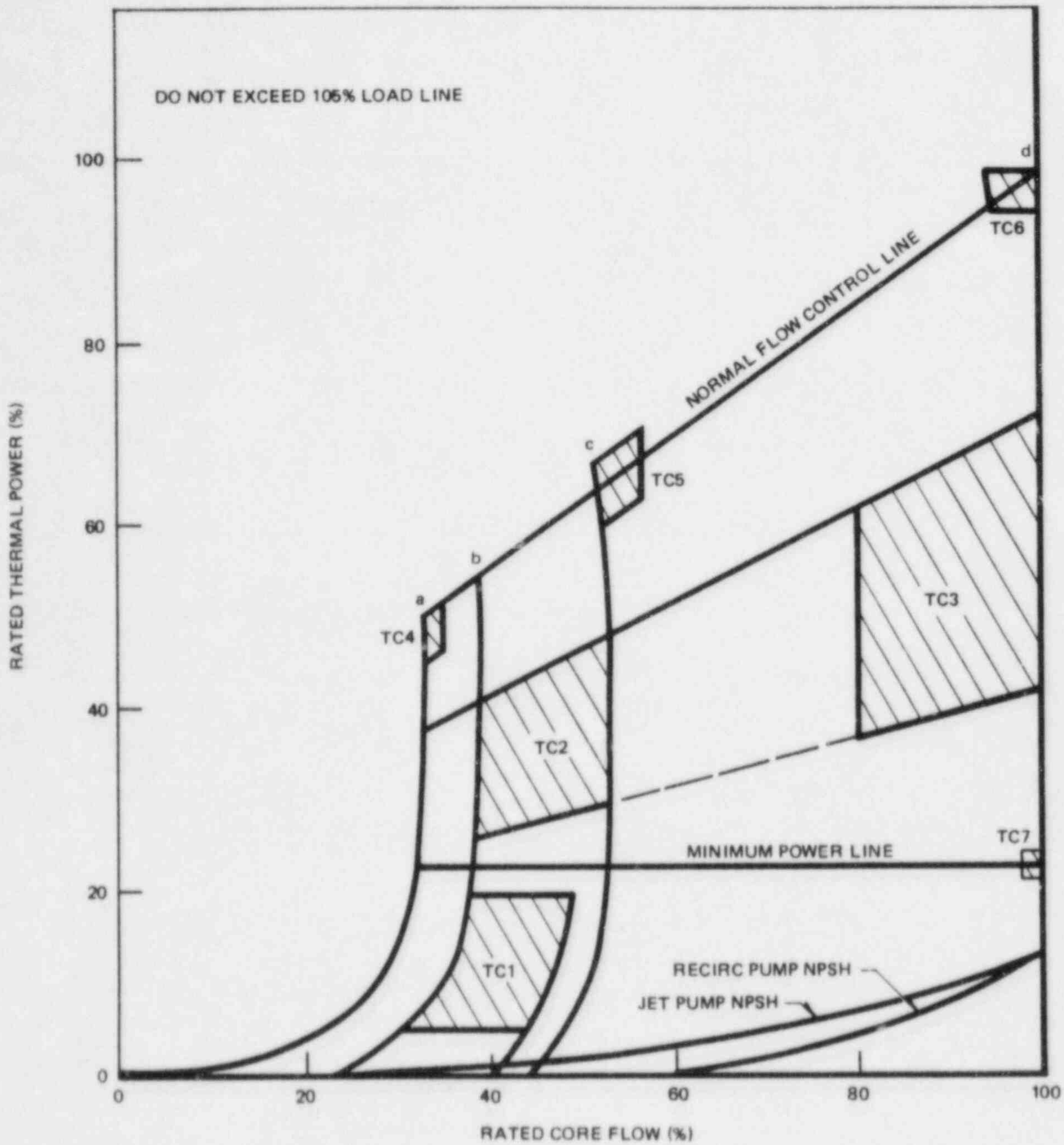


Figure 2.1.6. Histogram of the Startup Test Program — April.



POWER IN PERCENT OF RATED THERMAL POWER, 2436 MWt

CORE FLOW IN PERCENT OF RATED CORE RECIRCULATION FLOW, 77.0 × 10^6 lb/h

CONSTANT PUMP SPEED LINES

a) NATURAL CIRCULATION
b) 20% PUMP SPEED
c) CONTRACTUAL LOWER LIMIT OF MASTER FLOW CONTROL
d) PUMP SPEED FOR FULL FLOW AT FULL POWER

Figure 2.3.1. Test Program Power/Flow Map Startup Test Conditions.

3. SUMMARY OF TEST RESULTS

3.1 STI 1 CHEMICAL AND RADIOCHEMICAL*

Brunswick 1 is a heater drain (forward pumping) equipped plant; because this is a recent development in BWR's, several new procedures were developed. The above referenced report will cover these in detail, but two items are important enough to mention specifically. First, the standpipe in the deaerator tanks was removed to prevent suspended solids from settling out; and second, the plant regularly recirculates (to the condenser and through the condensate cleanup system) the heater drain water during startups. Only when no further improvement in quality is apparent does the plant begin forward pumping. This maximizes the piping surface area that is flushed before vessel injection and minimizes water quality "spiking" to unacceptable levels during startup each time the heater drain pumps are turned on.

The more prosaic aspects of this test such as conformance to water quality, technical specifications, and release limits were continually satisfied throughout the test program.

3.2 STI 2 RADIATION MEASUREMENTS

3.2.1 Level 1 Criteria

Radiation doses of plant origin and the occupancy times of personnel in radiation zones was controlled in accordance with the guidelines of the standards for protection against radiation as outlined in 10CFR20, USNRC General Design Criteria.

3.2.2 Discussion

Radiation surveys were made prior to fuel loading, prior to initial critical, at rated temperature and pressure, Test Condition 1, Test Condition 3 and Test Condition 6. The surveys included the plant general environs, all permanently mounted radiation detection instruments, and surveys of the turbine, reactor, and radwaste buildings. All radiation levels were as expected, and all radiation zones were properly delineated.

3.3 STI 3 FUEL LOADING

3.3.1 Level 1 Criteria

Throughout fuel loading the core remained subcritical by at least 0.38% $\Delta k/k$ with the analytically strongest rod fully withdrawn.

3.3.2 Discussion

Fuel loading began 0100 September 15, 1976. Advantage was taken of the required subcriticality checks to monitor stroke time, indicator function, and flow checks of each control rod. Previous to fuel loading, all rods had been scram timed. The loading procedure had to be modified several times because of high FLC count rates (rod blocks) (requiring FLC moves before scheduled), and low SRM count rates (requiring the FLC's to remain in core longer than scheduled).

Shutdown margin checks were done after 16, 64, and 144 fuel bundles were loaded, corresponding to September 15th, September 18th and September 19th, respectively. Five hundred and eight bundles had to be loaded before the final FLC was removed. The fully loaded core, completed 0351 September 25th, was given a "quick" shutdown margin check; a more accurate check was done as part of STI 4.

A visual verification of fuel bundle orientation and serial numbers was completed at 1630 September 25th; a video tape of the core was also made. Of the 10 days of fuel loading, approximately 3 days were lost to seconday containment, the refueling bridge (1-1/2 days) and minor mechanical problems (1/2 day). One bundle did not seat properly the first try but was successfully seated after an inspection and interference check.

* See also NEDE-21644 Chemical and Radiochemical Testing of the Brunswick Steam Elecric Plant Unit 1 Startup Test Program.

3.4 STI 4 FULL CORE SHUTDOWN MARGIN

3.4.1 Level 1 Criteria

The full core shutdown margin was successfully demonstrated to be 3.095% Δk/k with the analytically strongest rod fully withdrawn.

3.4.2 Level 2 Criteria

Criticality occurred within 0.594% Δk/k of the predicted critical rod configuration.

3.4.3 Discussion

Control rods were withdrawn in the normal B sequence until criticality occurred with the 38th control rod (22-23) at notch position 12; moderated temperature was 122°F. After the initial criticality, rod 22-23 was pulled to notch 16 causing a positive period of 61 seconds. From this data and appropriate rod worth curves and temperature corrections, a strongest rod out shutdown margin of 3.22% Δk/k was calculated, with a projected reactivity increase of 0.125% Δk/k during the initial cycle, adequate shutdown margin exists with the analytically strongest rod withdrawn at any time during the first cycle. The predicted critical was within ~0.6% Δk/k of the actual criticality (corrected for temperature and reactor period).

3.5 STI 5 CONTROL ROD DRIVE SYSTEM

3.5.1 Level 1 Criteria

The withdrawal speed of each CRD was determined and adjusted where necessary to not exceed 3.6 inches per second as indicated by a full 12-foot stroke in a minimum of 40 seconds.

The mean scram time of all operable CRD's is summarized in Tables 3.5.1.1 and 3.5.1.2.

Table 3.5.1.1
AVERAGE SCRAM TIMES (COLD)

% Inserted	Time	Limit
5	0.272	0.475
20	0.505	1.100
50	0.953	2.000
90	1.667	5.000

Table 3.5.1.2
AVERAGE SCRAM TIMES (RATED)

% Inserted	Time	Limit
5	0.317	0.375
20	0.738	0.900
50	1.513	2.000
90	2.642	3.500

As indicated by the above tables the mean scram times of all rods (and the mean scram time of the 3 fastest CRD's in any 2x2 array) did not exceed the specified values.

No control rod exceeded the 7-second limit for 90% insertion.

3.5.2 Level 2 Criteria

The insertion and withdrawal speed of each CRD was determined and adjusted where necessary to be 3.0 ± 0.6 inches per second as indicated by a full 12-foot stroke in 40 to 60 seconds.

All control rod drives were within the required limit of a 15 psid variation in differential pressure during continuous insertion.

Scram times with normal accumulator charge fell within the time limits indicated in Figure 5.3.1 of the Startup Test Instructions.

3.5.3 Discussion

All CRD's were demonstrated to be satisfactorily coupled and properly indicating their position. Fourteen CRD's required in/out timing adjustments. Five CRD's were selected from each sequence for scramming at various reactor and accumulator pressures; they were, respectively, 26-07, 22-19, 10-23, 18-15, and 26-15 — A sequence rods and 06-23, 18-11, 18-27, 30-31 and 14-31 — B sequence rods. Tables 3.5.3.1 through 3.5.3.3 itemize some of the results from these tests.

Table 3.5.3.1
SELECTED CRD SCRAM TIMES
OPEN VESSEL INSERTION TIMES (sec) (MINIMUM ACCUMULATOR PRESS.)

Rod	5%	20%	50%	90%
10-23	0.283	0.540	1.041	1.839
06-23	0.273	0.520	1.000	1.767
14-31	0.280	0.528	1.008	1.784
26-07	0.284	0.540	1.036	1.831
22-19	0.272	0.529	1.024	1.820
18-15	0.273	0.527	1.012	1.800
26-15	0.288	0.543	1.037	1.827
18-11	0.284	0.541	1.035	1.836
18-27	0.293	0.551	1.048	1.845
30-31	0.283	0.562	1.079	1.920

Similarly, the selected rod's friction tests and stroke time testing was satisfactory at the appropriate conditions.

All CRD's were scram timed at rated reactor conditions and the results are given in Tables 3.5.1.1 and 3.5.1.2. Finally, the CRD system was divided into nominal A and B sequence rods and the average scram times determined for each sequence at 5%, 20%, 50%, and 90% insertion; these data were later used in the analyses of the major transients.

Because Brunswick does not have an automatic control rod scram time recorder, it is impossible to recover rod scram timing data after planned scrams without using a complicated auxiliary instrumentation setup. Thus, the selected rod scram timing data could not readily be obtained from Test Condition 6 (because of thermal limits, PCIOMR's, etc.). The data in Table 3.5.3.1 reflects information obtained at Test Condition 3, the highest powered test condition possible without encroaching on any core limits. Since scram times are to a greater extent a function of pressure rather than power, it was felt that the above exception would not adversely affect the scram time data.

Table 3.5.3.2
ZERO ACCUMULATOR PRESSURE AND REACTOR AT RATED PRESSURE
INSERTION TIMES (sec)

Rod	5%	20%	50%	90%
10-23	0.328	0.778	1.605	2.778
18-15	0.311	0.733	1.528	2.661
26-15	0.350	0.811	1.639	2.806
26-07	0.317	0.733	1.522	2.650
22-19	0.311	0.728	1.500	2.611
30-31	0.322	0.767	1.583	2.755
18-11	0.378	0.784	1.672	2.917
14-31	0.322	0.756	1.561	2.711
18-27	0.328	0.778	1.594	2.722
06-23	0.322	0.750	1.595	2.750

Table 3.5.3.3
NORMAL ACCUMULATOR PRESSURE AND RATED REACTOR PRESSURE
INSERTION TIME (sec)

Rod	5%	20%	50%	90%
26-07	0.294	0.683	1.399	2.550
10-23	0.317	0.745	1.546	2.711
18-15	0.311	0.700	1.444	2.589
26-15	0.333	0.755	1.567	2.828
22-19	0.300	0.672	1.406	2.511
06-23	0.300	0.672	1.422	2.528
14-31	0.300	0.694	1.406	2.533
18-27	0.334	0.745	1.472	2.622
18-11	0.306	0.700	1.495	2.700
30-31	0.306	0.700	1.461	2.622

3.6 STI 6 SRM PERFORMANCE AND CONTROL ROD SEQUENCE

3.6.1 Level 1 Criteria

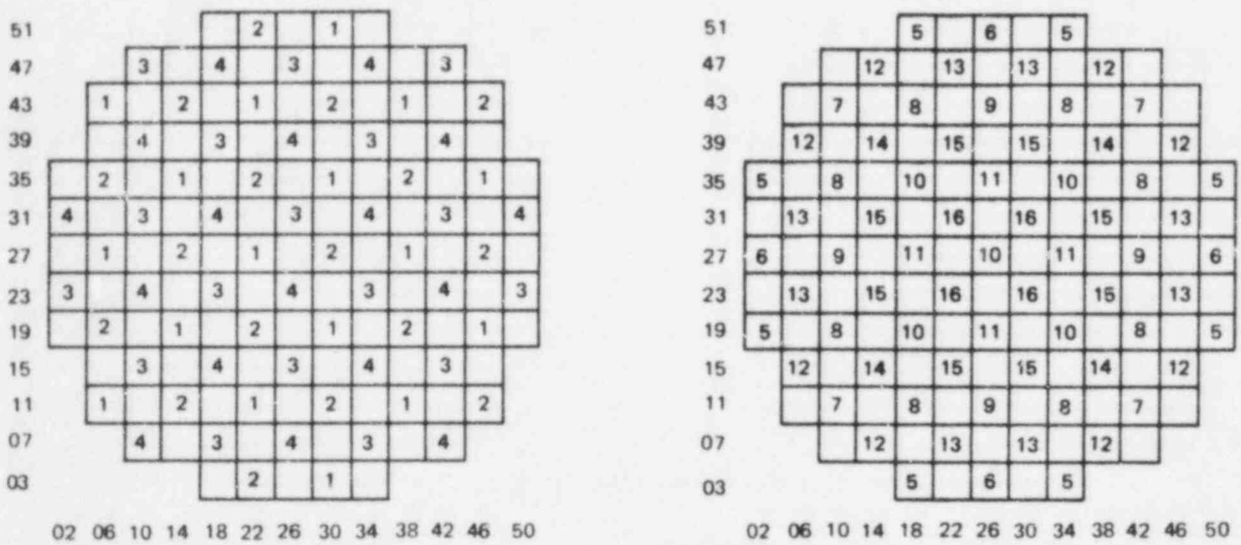
The operational sources and the SRM system were properly set up and matched so as to provide a signal to noise ratio of at least 2 to 1 and a minimum count rate of 3 cps.

The IRM's were properly adjusted so as to be on scale before the SRM's exceeded the rod block set point.

3.6.2 Discussion

After the IRM's had demonstrated suitable overlap with the APRM's the final and satisfactory overlap of the SRM/IRM systems was demonstrated.

The current versions of the A and B control rod withdrawal sequences are included as Figures 3.6.2.1 and 3.6.2.2. It has not yet proven necessary to change the sequences to compensate for core burnup and/or gadolinia burnout. Many startups have proven the ability of the SRM system to efficiently and safely monitor the ascension to criticality and rated power in either of these sequences; similarly the sequences themselves have proven to be safe and efficient.



NOTES:

1. CORE FLOW IS DUE TO 20% PUMP SPEED UNTIL THE THERMAL POWER IS ABOVE 20%
2. COMPLETE WITHDRAWALS IN EACH COLUMN BEFORE GOING TO THE NEXT
3. WITHDRAW RODS IN EACH GROUP INDIVIDUALLY. FOLLOW THE ORDER SHOWN BELOW FOR GROUPS 1, 2, 3, AND 4

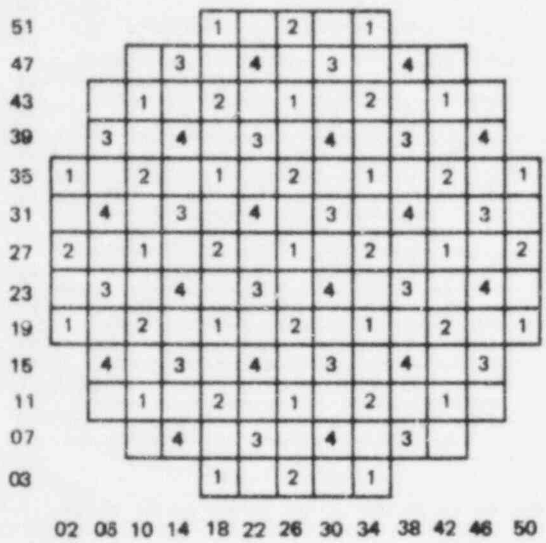
ROD WITHDRAWAL ORDER FOR GROUPS 1, 2, 3, AND 4

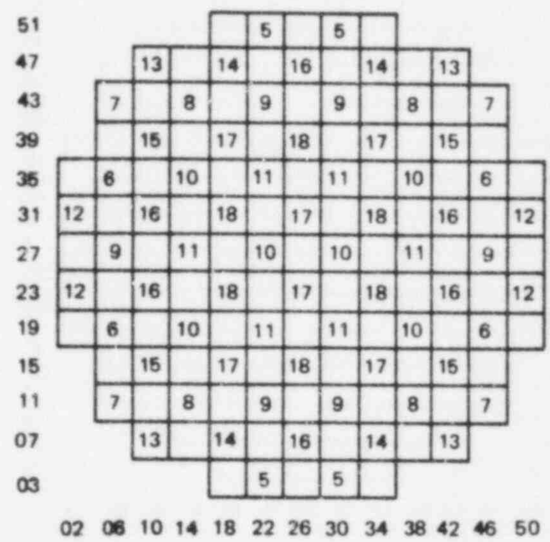
GROUP 1	GROUP 2	GROUP 3	GROUP 4
(22,27)	(30,27)	(26,31)	(26,23)
(30,19)	(22,35)	(18,23)	(34,31)
(38,27)	(14,27)	(26,15)	(26,39)
(30,35)	(22,19)	(34,23)	(18,31)
(22,43)	(30,11)	(42,31)	(10,23)
(14,35)	(38,19)	(34,39)	(18,15)
(06,27)	(46,27)	(26,47)	(26,07)
(14,19)	(38,35)	(18,39)	(34,15)
(22,11)	(30,43)	(10,31)	(42,23)
(38,11)	(14,43)	(10,15)	(42,39)
(46,19)	(06,35)	(18,07)	(34,47)
(46,35)	(06,19)	(34,07)	(18,47)
(38,43)	(14,11)	(42,15)	(10,39)
(30,51)	(22,03)	(50,23)	(02,31)
(06,43)	(46,11)	(42,47)	(10,07)
(06,11)	(46,43)	(10,47)	(42,07)
(30,03)	(22,51)	(02,23)	(50,31)

CONTROL ROD SEQUENCE

ROD GROUPS	STEPS FOR SEQUENCE A								
	1	2	3	4	5	6	7	8	9
1	48								
2	48								
3	48								
4	48								
5	48								
6	48								
7	48								
8	48								
9	48								
10	24	48							
11		10		48					
12	18				48				
13		4			22	40			
14		4						10	16
15		6	8			10	30	40	
16		2				6		10	14
POWER	25%		50%		75%		100%		
FLOW	37%		107%		104%		100%		

Figure 3.6.2.1. A Control Rod Sequence.





NOTES:

1. CORE FLOW IS DUE TO 20% PUMP SPEED UNTIL THE THERMAL POWER IS ABOVE 20%
2. COMPLETE WITHDRAWALS IN EACH COLUMN BEFORE GOING TO THE NEXT
3. WITHDRAW RODS IN EACH GROUP INDIVIDUALLY. FOLLOW THE ORDER SHOWN BELOW FOR GROUPS 1, 2, 3, AND 4

ROD WITHDRAWAL ORDER FOR GROUPS 1, 2, 3, AND 4

GROUP 1	GROUP 2	GROUP 3	GROUP 4
(26,27)			
(18,35)			
(10,27)			
(18,19)			
(26,11)			
(34,19)	(18,27)	(22,23)	(22,31)
(42,27)	(26,19)	(30,15)	(14,23)
(34,35)	(34,27)	(38,23)	(22,15)
(26,43)	(26,35)	(30,31)	(30,23)
(10,43)	(18,43)	(22,39)	(38,31)
(02,35)	(10,35)	(14,31)	(30,39)
(02,19)	(02,27)	(06,23)	(22,47)
(10,11)	(10,19)	(14,15)	(14,39)
(18,03)	(18,11)	(22,07)	(06,31)
(34,03)	(26,03)	(38,07)	(06,15)
(42,11)	(34,11)	(46,15)	(14,07)
(50,19)	(42,19)	(46,31)	(30,07)
(50,35)	(50,27)	(38,39)	(38,15)
(42,43)	(42,35)	(30,47)	(46,23)
(34,51)	(34,43)	(14,47)	(46,39)
(18,51)	(26,51)	(06,39)	(38,47)

CONTROL ROD SEQUENCE

	ROD GROUPS	STEPS FOR SEQUENCE B								
		1	2	3	4	5	6	7	8	9
	1	48								
	2	48								
	3	48								
	4	48								
	5	48								
	6	48								
	7	48								
	8	48								
	9	48								
	10	48								
	11	6	12	30	48					
	12	8	18				48			
	13	8	18				48			
	14		8			26	48			
	15		6			8	12	28	40	48
	16		6		8	18	40			
	17		4		6		8	40		
	18		4							12
POWER		25%		50%		75%		100%		
FLOW		37%		107%		104%		100%		

Figure 3.6.2.2. *B Control Rod Sequence.*

3.7 STI 7 REACTOR WATER CLEANUP SYSTEM

3.7.1 Level 2 Criteria

The temperatures at the tube side outlet of the nonregenerative heat exchangers did not exceed 140°F in any mode of RWCU operation.

The main cleanup pump available NPSH was determined to be greater than 10 feet during the hot standby mode of RWCU operation as defined in the process diagram.

The cooling water supplied to the nonregenerative heat exchangers was found to be within the flow and outlet temperature limits indicated in the process diagrams for the "normal" and "blowdown" mode of RWCU operation.

3.7.2 Discussion

During the first heatups, the cleanup system was tested in the normal, blowdown and hot standby modes of operation. Temperatures at the outlet side of the nonregenerative heat exchangers never exceeded 130°F. All flows and temperatures were within the limits specified in the process diagrams. NPSH calculated during the hot standby mode was determined to be 69.6 ft.

Although not specifically connected with STI 7 testing, the cleanup pumps at Brunswick have experienced alignment problems because of the large temperature range over which they must operate. This had necessitated realignment of the pumps each time the plant passed from "cold" to "hot" operation and vice versa. Presently, one of the pumps is testing a flexible coupling and results so far indicate that this is likely to become the permanent fix.

3.8 STI 8 RESIDUAL HEAT REMOVAL SYSTEM

3.8.1 Level 2 Criteria

The RHR system was capable of operating in the steam condensing mode and shutdown cooling mode (with either one or two heat exchangers operational) at the flow rates indicated on the process diagrams.

The heat removal capability of each RHR heat exchanger was demonstrated to be at least 151 x 10^6 Btu/hr when the inlet flows and temperatures were as indicated on the process diagrams.

The process system variables were visually shown to have a decay ratio of less than 0.25 throughout each (level, pressure and differential pressure) controller's expected operating range.

3.8.2 Discussion

Both RHR heat exchangers were placed in the steam condensing mode of operation. While in this mode of operation the differential pressure, pressure, and level controllers of both heat exchangers were exercised by introducing step changes in set point; acceptable stability was demonstrated. While in the steam condensing mode the very low (42°F) service water temperatures required that the service water flows be throttled to less than rated process diagram flows; however, rated flows were successfully demonstrated in preoperational testing. Shell side (reactor water) flow rates were determined to be 118,000 lb/hr and 160,000 lb/hr versus required flow rates of 99,000 lb/hr and 138,000 lb/hr (one and two heat exchangers operating, respectively).

It was demonstrated several times that the RHR system could be successfully placed in the shutdown cooling mode of operation. If, however, the process diagram flow rates were attempted the low delay heat load of the new core and the low service water temperatures caused the allowed cooldown rate of 100°F/hr to be exceeded. As a result, the service water flow rates had to be throttled to half of the process diagram flow rates of 8000 gpm.

Measured heat exchanger flows and temperatures were used to calculate a heat removal capacity corrected to rated conditions. The calculated value was 177 MBtu/hr (per heat exchanger) versus a required 151 MBtu/hr.

3.9 STI 9 WATER LEVEL MEASUREMENT

3.9.1 Level 2 Criteria

The narrow range level system (GEMAC) readings were adjusted to agree with each other within ±1.5 inches.

The wide range level indicators were adjusted to agree with each other within ±6 inches.

3.9.2 Discussion

The reactor water level measurement systems were checked at heatup and Test Conditions 1, 4, 7, and 6. These tests covered the widest possible range of reactor power and flow conditions; 0 to 100% power and minimum to 100% core flow.

The various GEMAC and Yarway column temperatures were measured at several axial positions (spare analog thermocouple inputs to the process computer were used for this purpose) and the actual Yarway reference leg temperatures and Yarway ranges calculated. In each case the actual values compared quite closely with the assumed values; no recalibration was necessary.

At all test conditions the instruments satisfied or were made to satisfy the Level 2 criteria. The deviation of the average Yarway readings from the (assumed accurate) average GEMAC readings were plotted versus power and core flow. Results were as expected; core flow had much more influence on Yarway readings than did power. Both accuracy and conservatism were served by using hot standby (initial) data to calibrate the Yarways.

Brunswick, like other BWR 4's had a noticeable increase in GEMAC sensed level indication oscillation at high (100%) core flows and moderate (~50%) reactor powers.

3.10 STI 10 IRM PERFORMANCE

3.10.1 Level 1 Criteria

All IRM channels had adequate overlap with the SRM's and the APRM's. The IRM scrams occurred before 96% of full scale.

3.10.2 Discussion

After the initial criticality the IRM's were shown to overlap with the SRM's by at least two decades. During the first heatup all of the IRM's were adjusted to produce adequate continuity between ranges 6 and 7. Finally after the initial APRM calibration, APRM/IRM overlap was found to be one decade. The IRM/SRM check did not have to be repeated since no adjustments were made to obtain IRM/APRM overlap. Any IRM channels inoperative during the first heatups were later readjusted to the above characteristics.

3.11 STI 11 LPRM CALIBRATION

3.11.1 Level 1 Criteria

The local power range monitor (LPRM) gains were adjusted so that their meter readings were proportional to the neutron flux in the narrow-narrow water gap at the chamber elevation.

3.11.2 Discussion

The location, continuity, and correct hookup of the LPRM to its respective meter and computer readout was verified by observance of the appropriate signal response during initial control rod movements in both sequences. The LPRM's were calibrated (to watts per square centimeter) at Test Conditions 1, 3 and 6 using approved procedures. At Test Condition 1 the LPRM GAF's were calculated from BUCLE "TIPNEWRP" and "P1NEWRP" options, the TIP and LPRM data were manually entered from TIP traces and meter readings. At the other test conditions, the process computer hardware and software had

been verified to be correct; the LPRM GAF's were calculated by the process computer OD-1 and P1 programs. Using either method, no problems were encountered in calibrating the TIPs or LPRMs to the given heat flux.

3.12 STI 12 APRM CALIBRATION

3.12.1 Level 1 Criteria

The APRM channels were calibrated to read equal to or greater than actual core thermal power at all test conditions.

In the startup mode all APRM channels produced a scram at less than or equal to 15% of rated power.

Technical specification and fuel warranty limits on APRM scram and rod blocks were not exceeded.

3.12.2 Level 2 Criteria

The APRM channels were considered to be tracking core thermal power accurately when they agreed with the heat balance to within ±6% of rated power.

3.12.3 Discussion

The APRM system was calibrated after each change in the gain of the LPRM system and whenever changes in the rod pattern had a significant effect on the APRM readings. Actual core thermal power was calculated by a heat balance done by the process computer (after OD-3 was proven accurate) or manually. During the first heatup, a special heat balance using the estimated heat capacity of the vessel and coolant was used. No problems were encountered in adjusting the APRM system to read reactor power or higher.

The tracking ability of the APRM system was verified by monitoring all APRM readings through a 38% change (by flow) in rated power from Test Condition 6 and through a 24% change in rated power from Test Condition 3. All APRMs were shown to track within the ±6% criteria. Data from the Test Condition 6 trial are presented in Table 3.12.3.1 and Figure 3.12.3.1.

Table 3.12.3.1
APRM TRACKING

% Core Flow	Actual % Power	Highest APRM Reading	Lowest APRM Reading
98.3	96.0	96.7	94.81
77.5	83.9	85.2	80.4
54.9	70.3	71.8	70.7
49.9	66.6	68.2	67.6
36.8	58.3	61.4	60.5

3.13 STI 13 PROCESS COMPUTER

3.13.1 Level 2 Criteria

The process computer dynamic and static systems test cases were successfully completed; all the computer programs are operational.

Programs OD-1 and P-1 were found to calculate MCPR's and LPRM GAF's to an accuracy of better than 2% relative to an independent method (BUCLE).

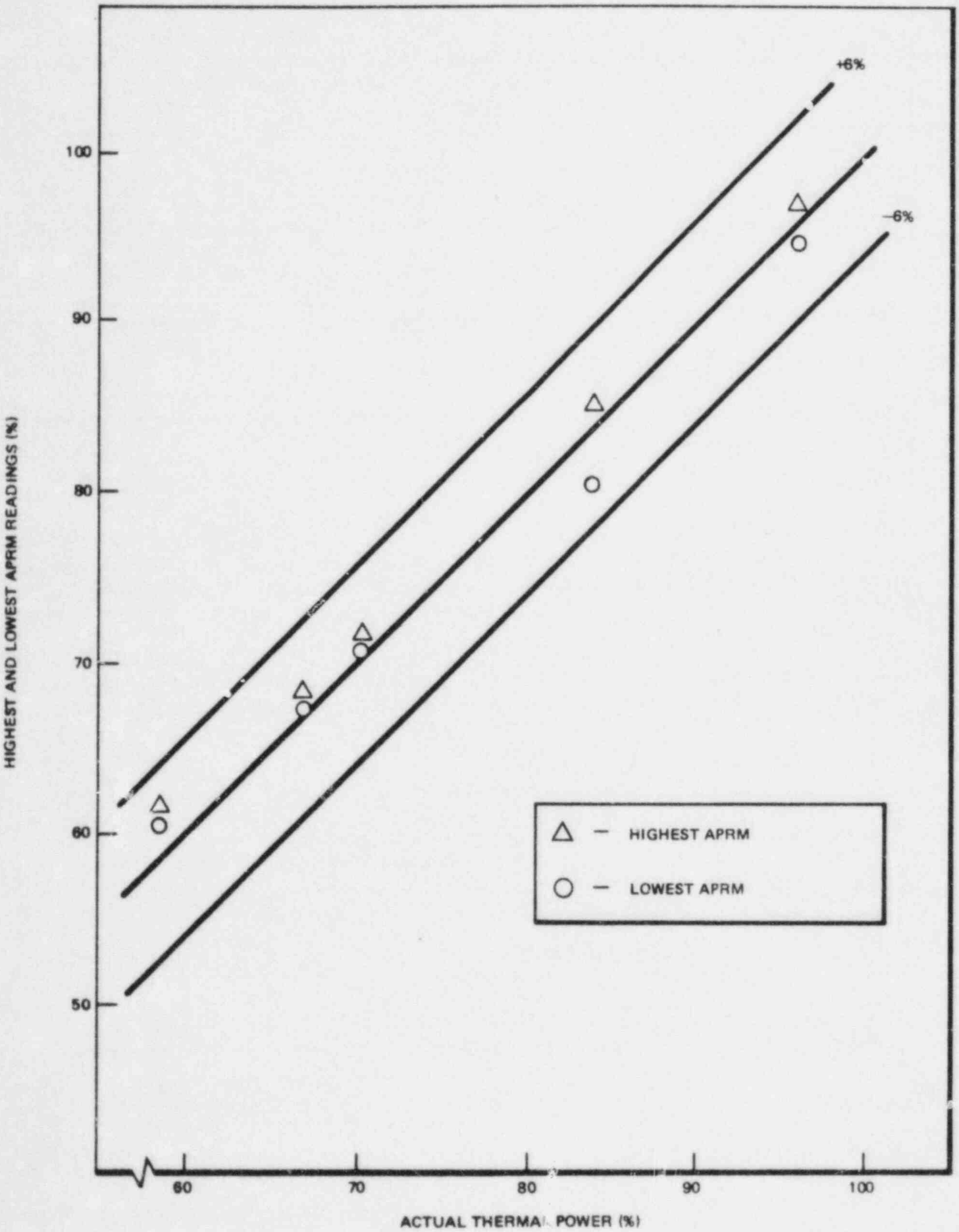


Figure 3.12.3.1. APRM Tracking.

3.13.2 Discussion

The dynamic system test case was completed before Test Condition 2 of the startup test program was completed. No significant problems were encountered with the process computer hardware or software, but the TIP machine/process computer interface did delay the checkout of the major nuclear programs. All of the NSSS programs were checked out and proven operational. The process computer is capable of accurately determining plant power, thermal limits, and accumulating core exposure.

At several test conditions after the DSTC was completed the process computer's OD-1 and P-1 programs were checked by comparison with the off line program BUCLE. The level 2 criteria was always satisfied. In most cases the comparisons of CPR, core maximum peaking factor, maximum linear heat generation rate, maximum average planar heat generation rate, and LPRM GAF's were within 0.5% in value and agreed as to location.

During the startup test program several minor problems were encountered and corrected. The computer analog signal taps were rearranged in the plant cabinets to measure only the voltage drop across the sensing resistors (and not the sensing resistor plus wiring). This made a difference of ~0.5% in the sensed values. A new core flow/drive flow correlation consistent with final calibration results was entered. (See STI-35.) Finally, the data bank was changed (after mechanically limiting core flow to 102%) to allow the process computer to calculate CPR using the most "liberal" K_f limit curve. This corresponds to the 102.5% core flow limit rather than the master auto limit used throughout the test program.

3.14 STI 14 RCIC SYSTEM

3.14.1 Level 1 Criteria

The reactor core isolation cooling system (RCIC) was able to deliver rated flow, 400 gpm, in less than 30 seconds against any reactor pressure between 150 psig and rated (including line losses).

The RCIC turbine did not trip on overspeed during manual or auto starts. The peak turbine speed in either mode exhibited satisfactory rpm margin to the overspeed trip.

3.14.2 Level 2 Criteria

The RCIC turbine gland seal condenser system was capable of preventing steam leakage to the atmosphere.

The RCIC steam supply line high flow isolation switches have been adjusted to actuate at 300% of the maximum required steady state flow (with reactor pressure near rated).

The RCIC system controllers and mechanical components have been adjusted to provide RCIC system decay ratios of 0.25 or less, during small speed or flow command changes.

The RCIC turbine margin to the overspeed trip (for manual and auto starts) is at least 10% of the trip value.

3.14.3 Discussion

The RCIC system was tested by injection from the condensate storage tank (CST) to the CST at 150 psig, 550 psig and rated reactor pressures while simulating a pump discharge pressure 100 psi above reactor pressure. A special 1220 psig pump discharge pressure test was also performed. Additionally, an actual vessel injection was done at Test Condition 2.

Each of these tests were monitored for peak turbine rpm, time to rated flow, and decay ratio response to small step changes in speed and flow. At appropriate times RCIC steam supply line flow data were taken. All Level 1 and 2 criteria were satisfied; the turbine was stable at all reactor conditions.

RCIC pump performance was satisfactory and as predicted by the vendor.

The steam supply high flow switch set points were calculated by San Jose; the trip set points are 422 inches of water and have been transmitted to the appropriate CP&L personnel.

Final setpoint data and measured values are summarized in Table 3.14.3.1.

Table 3.14.3.1
RCIC SYSTEM DATA

400 gpm Line Losses to Vessel	40 psi
Time to Rated Flow — Vessel Injection	26.6 sec
Peak Turbine Speed — Vessel Injection	4560 rpm
Flow Controller	
Proportional Band	600%
Resets/Minute	100
Ramp Generator Ramp Time	22 sec
Idle Voltage	−2.0 volts
Woodward Controller	
Gain	5
Stability	5

3.15 STI 15 HPCI SYSTEM

3.15.1 Level 1 Criteria

The average HPCI pump discharge flow was equal to or greater than 4250 gpm after 25 seconds had elapsed from initiation on auto starts at any reactor pressure between 150 psig and rated (except the 1220 psig HPCI pump discharge pressure test).

With HPCI pump discharge pressure at any value between 150 psig and 1120 psig, the required flow is 4250 gpm (the limit of 1120 psig includes a conservatively high value of 100 psi for line losses. The measured value may be used if available).

The HPCI turbine did not trip on overspeed during auto or manual starts. The peak turbine speed, in any auto or manual start, exhibited a satisfactory rpm margin to avoid a possible high speed trip.

3.15.2 Level 2 Criteria

The turbine gland seal condenser system was capable of preventing leakage to the atmosphere.

The differential pressure switches for the HPCI steam supply line high flow isolation trip were adjusted to actuate at 300% of the maximum required steady state flow (with the reactor pressure near rated).

For small speed or flow command changes in either manual or automatic mode, the decay ratio of each recorded HPCI system variable was less than 0.25 in order to demonstrate acceptable stability (except the 1220 psig HPCI pump discharge pressure test).

The margins to avoid the overspeed trip were at least 10% of the trip value.

3.15.3 Discussion

The HPCI system was tested by injection from the condensate storage tank (CST) to the CST at 150 psig, 550 psig, and rated reactor pressures while simulating a pump discharge pressure 100 psi above reactor pressure. A special 1220 psig pump discharge pressure test was also performed. Additionally, an actual vessel injection was done at Test Condition 3.

Each of these tests were monitored for peak turbine rpm, time to rated flow, and decay ratio response to small step changes in speed and flow. At appropriate times HPCI steam supply line flow data were taken. All Level 1 and Level 2 criteria

were satisfied; the turbine was stable at all reactor conditions. HPCI pump performance was satisfactory and as predicted by the vendor.

Several problems were encountered in setting up the HPCI turbine. A constant speed offset in the relationship between flow controller output and turbine rpm was introduced by a Woodward EGR actuator with an incorrect "null" (offset) voltage. Some confusion also existed in setting the turbine high speed stop to be compatible with the pacific pump. Additionally, four HPCI injections were unsuccessfully run before the proper compromises were made between turbine speed overshoot, injection time, and stability. The Woodward components on the HPCI turbine were completely changed out and many adjustments of the turbine's hydraulic system were required.

The steam supply high flow switch set points were calculated by San Jose; the trip set points are 230 inches of water and have been transmitted to the appropriate CP&L personnel.

Final set point data and measured values are summarized in Table 3.15.3.1.

Table 3.15.3.1
HPCI SYSTEM DATA

4250 gpm Line Losses to Vessel	87 psi
Time to Rated Flow — Vessel Injection	22.3 sec
Peak Turbine Speed — Vessel Injection	4350 rpm
Flow Controller	
Proportional Band	400%
Resets/Minute	20
Ramp Generator	
Ramp Time	12 seconds
Idle Voltage	−0.5 volt
Woodward Controller	
Gain	5
Stability	5

3.16 STI 16 SELECTED PROCESS TEMPERATURES

3.16.1 Level 1 Criteria

The reactor recirculation pumps were not started nor was a natural circulation startup attempted unless the coolant temperature difference between the steam dome and vessel bottom head drain was less than 145°F.

The recirculation pump in the idle loop was not started unless the temperature of the coolant within the idle and operating recirculation loops was within 50°F of each other.

3.16.2 Level 2 Criteria

During operation of both recirculation pumps at rated core flow, the bottom head coolant temperature as measured by the bottom drain line thermocouple was within 30°F of the recirculation loop temperatures.

3.16.3 Discussion

During heatup testing the reactor recirculation pump low speed stops were determined. One of the criteria for a permissible low speed set point is that temperature stratification of the vessel be avoided. At 20% recirculation pump speed, the following data was recorded:

20% RECIRCULATION PUMP SPEED
TEMPERATURE DATA

Average Recirculation Pump Inlet Temperature	528.8°F	276°C
Saturation Temperature	533.0°F	278.3°C
Bottom Drainline Temperature	519.0°F	270.6°C

All criteria were easily satisfied and the low speed electrical stops were set to 20% speed and the mechanical stops were set to 19% speed.

At Test Conditions 3 and 6 the recirculation pumps were tripped as part of startup testing; a two pump trip was also performed at Test Condition 6. Temperature data for these trips and 100% core flow operation is presented in Table 3.16.3.1. All temperature criteria were met, and it should be specifically noted that even in natural circulation only a 55° difference existed between the bottom drain and saturation temperatures.

Table 3.16.3.1
SELECTED PROCESS TEMPERATURES

	Test Condition 3		Test Condition 6	
	Both Pumps Running	**"B" Pump Tripped**	**"A" Pump Tripped**	**Both Pumps Tripped**
Saturation Temperature	548.8	539	543	540
Bottom Drain Temperature	545	500	500	485
Δ Loop Temperature	—	7.7	2	—
Δ (Saturation — Bottom Drain)	3.8	39	43	55
Δ (Loop — Bottom Drain)	19.5	—	—	—

All Temperatures in °F

3.17 STI 17 SYSTEM EXPANSION

3.17.1 Level 1 Criteria

There was no evidence of blocking of the displacement of any system component caused by thermal expansion.

Electrical cables were not fully stretched. Flow induced or continuous (steady state) vibration range displacement measurements for the recirculation system did not exceed 0.035 inch and for the main steam system did not exceed 0.065 inch (mean to peak).

The measured range of displacement for vibration of the recirculation system due to recirculation pump trip(s) did not exceed 0.06 inch (mean to peak).

The measured range of displacements in the main steam lines for relief valve operation were less than the calculated displacements.

3.17.2 Level 2 Criteria

At the steady state condition the displacements of instrumented (with displacement measuring devices) points did not vary from the calculated values by more than ±50% or ±0.5 inch (except as described below) i.e., this criterion was satisfied if the resultant of the displacement along the three mutually perpendicular axes meets either the ±50% or the ±0.5 inch difference. Displacements of less than 0.25 inch could be neglected since 50% of this value is bordering upon the accuracy of the measurement. If the measured displacements did not meet these criteria, the piping design engineer was contacted to reanalyze the data with regard to design stresses.

During the heatup cycle, the trace of the instrumented points fell within a range of 150% of the calculated value from the initial cold position in the direction of the calculated value and 50% of the calculated value from the initial position in the opposite direction of the calculated value.

Hangars were always within their operating range (between hot and cold settings).

Vibration did not reach 80% of the applicable Level 1 criteria.

The acceleration at any measured location did not exceed 5 *g*'s, based on a sinusoidal vibration mode.

3.17.3 Discussion

STI 17 testing was conducted during plant heatups, transients, and major trips. Thermal expansion data were obtained by lanyard potentiometers and actual observation at rated and intermediate temperatures. Before the first heatup, the drywell was inspected and all possible expansion problems were corrected. Investigation of actual heatup data and visual inspection indicated that, in general, the drywell piping moved in the correct direction and returned to its base setting after cooldown.

Several instrumented points fell slightly short (in displacement) of meeting Level 2 criteria. These points were specifically identified in the heatup data returned to San Jose. Further San Jose examination of these points verified that the resulting stresses were acceptable.

Two hangars were found to be outside their operating range and were readjusted to comply with the Level 2 criteria. Thus, system thermal expansion met or was resolved to meet all Level 1 and 2 criteria. Later heatups confirmed that the system expansion data were repeated and that thermal motion was thus unimpeded.

System expansion vibration data were taken for all the transient tests indicated in Table 3.17.3.1. All data were returned to San Jose for analysis and found acceptable; all Level 1 and 2 criteria were met. In general, discernible vibration levels were seen only on relief valve lifts and major trips (that resulted in scrams). MSIV full closure and Test Condition 3 turbine trip data taking were waived because of previous relief valve testing or because a more severe test would be performed later.

Table 3.17.3.1
SYSTEM EXPANSION VIBRATION TESTING

Test Condition	Test Being Performed	Pipe(s) to be Monitored	Power %	Flow %	Date
2	MSIV	Steam Lines	41	35	1-2-77
3	MSIV	Steam Lines	63	99	1-13-77
6	Load Reject	Steam & Recirc. Lines	100	96.5	4-1-77
3	1 Recirc. Pump Trip	Recirc. Lines	68	100	2-1-77
5	MSIV	Steam Lines	80 87	70 79	3-10-77 3-11-77
6	Feedpump Trip	Feedwater Lines	98	96	3-12-77
6	Full Isol. (MSIV)	Steam Lines		Waived	
5	Stop Valve Test	Steam Lines	64	57	3-3-77
6	2 Recirc. Pump Trip	Recirc. Lines	97	99	4-23-77
7	Recirc. Cavitation Test	Recirc. Lines	20	94	1-31-77
6	1 Recirc. Pump Trip	Recirc. Lines	96	100	4-24-77
Rated Temp. & Pres.	Relief Valve Test	Steam Lines	9	29	2-27-77
3	Turbine Trip	Steam and Recirc. Lines	69	100	2-12-77

3.18 STI 18 CORE POWER DISTRIBUTION

3.18.1 Level 2 Criteria

The results of the TIP reproducibility test indicate that the overall standard deviation of the segment averaged TIP values (BASE distribution from the process computer) of the central nodes (nodes 5-22) was less than 7.8%.

3.18.2 Discussion

Complete sets of TIP scans and appropriate process computer output were obtained at Test Conditions 2, 3, and 6. An offline computer program was written to analyze the random, geometrical and total noise of the TIP system.

The random noise data were obtained by the analysis of multiple traces (and BASE output edits) of the same (common channel) location for each TIP machine. The data from the OD-2 outputs were corrected for machine normalization constants (LPRM effects).

The total noise data were obtained by the analysis of diagonally symmetric TIP locations (and BASE output edits) from a complete OD-1. All data were taken while the reactor was operating in a diagonally and rotationally symmetric rod pattern; non-central nodes were not considered. All Level 2 criteria were satisfied.

Table 3.18.3.1 summarizes the results of the above investigation.

Table 3.18.3.1
TIP REPRODUCIBILITY DATA

Test Condition	Uncertainty		
	Random	Geometric	Total
2	1.99%	4.68%	5.086%
3	1.33%	2.905%	3.19%
6	1.52%	2.05%	2.55%

The generally lower values of uncertainty at the higher power levels reflect the increasingly better setup of the TIP machines with time and the higher neutron flux levels. After Test Condition 3, the TIP machine flux amplifiers were calibrated (in W/cm^2) to the average heat flux in the appropriate narrow-narrow channels.

Both symmetric TIP location traces (after TIP machine calibration) and process computer bundle power edits were investigated over a wide range of reactor powers. The results confirm that the reactor core does operate with an appropriately symmetrical (rotational or mirror) power distribution given a symmetric rod pattern.

3.19 STI 19 CORE PERFORMANCE

3.19.1 Level 1 Criteria

The maximum linear heat generation rate (MLHGR) during steady state conditions did not exceed the allowable heat flux as specified in the plant technical specifications.

The steady state minimum critical power ratio (MCPR) was maintained greater than or equal to the allowable MCPR as specified in the plant technical specifications.

The maximum average planar linear heat generation rate (MAPLHGR) did not exceed the limits given in the plant technical specifications.

Steady state reactor power was limited to full rated maximum values on or below the design flow control .

3.19.2 Discussion

Core performance evaluations were performed at each (powered) test condition and whenever it was judged necessary during routine operations. Process computer calculations, which were periodically checked by off-line BUCLE calculations, were used throughout the test program. The resulting core parameters at the various test conditions are summarized in Table 3.19.2.1.

Table 3.19.2.1
SUMMARY OF CORE PERFORMANCE PARAMETERS

Rated Power = 2436 MWt			Rated Core Flow = 77 Mlb/hr			
Test Condition	Power	Flow	MCPR	CMPF	LHGR	MALHGR
Limit (At Rated Power)	100%	100%	(1.28)	(2.43)	13.4	—
1	18.9	19.4	3.574	2.306	2.412	2.02
2	36.7	34.7	2.547	2.032	4.154	3.48
3	63.0	95.4	2.013	2.477	8.616	7.25
4	36.5	23.3	2.113	2.389	4.18	4.00
5	63.4	55.0	1.665	2.412	8.442	7.18
6	98.1	99.6	1.329	2.374	12.85	10.90

No problems were encountered in keaping the plant within its licensed thermal limits; in fact, even PCIOMR limitations did not evidence themselves until 75%-85% reactor power (depending upon Xenon conditions). Later in the test program, the Bailey mechanical and electrical stops were adjusted and the master limiter set to 100% core flow. Consistent with this, the process computer data bank was changed to allow the CPR calculation to take advantage of the 102.5% core flow K_f CPR limit curve.

3.20 STI 20 STEAM PRODUCTION

3.20.1 Level 1 Criteria

The NSSS parameters determined by normal operating procedures were within the appropriate license restrictions.

The nuclear steam supply system was determined to be capable of supplying steam of better than 99.7% quality at a pressure of 985 psia at the second isolation valve in an amount consistent with the final feedwater temperature and control rod drive flow as given by the formula:

$$W_{STEAM} = \frac{8284}{1190.4 - H_{fw}} + W_{CRD} \quad \text{(Mlb/hr)}$$

3.20.2 Discussion

The plant was brought to an indicated power of 100% and allowed to stabilize for one day. The 100 hour warranty run was divided into two 50 hour periods, and a special 2 hour data taking session was run during each period.

The averaged readings of each parameter from each period were combined into a single set of parameters to be used in determining reactor power and steam flow. Carryover was determined from STI 1 Test Condition 6 testing.

The results of these calculations were:

Reactor Power	100.34%
Steam Flow	10.492 Mlb/hr
Carryover	0.0023%
Outboard Isolation Valve Pressure	952.4 psig

Recently calibrated plant instrumentation was used to measure most parameters making up the power calculation; feedwater nozzle differential pressure and MSIV pressure were determined using specially installed instruments.

Because of several measurement problems and accuracy considerations, an error analysis was performed on the steam flow and outboard MSIV pressure calculations. Additionally, the calculated values had to be corrected to rated (100% power) conditions. When the above corrections were taken into account, the results indicate that all the criteria are satisfied. All of the licensed thermal limits applicable to Brunswick Unit 1 were maintained throughout the warranty run.

3.21 STI 21 FLUX RESPONSE TO RODS (CORE POWER-VOID MODE RESPONSE)

3.21.1 Level 1 Criteria

The decay ratio was less than 1.0 for each process variable that exhibited an oscillatory response to control rod motion.

3.21.2 Level 2 Criteria

The decay ratio was less than 0.25 for each total core process variable that exhibited an oscillatory response to control rod motion when operating above the lower limit of the master flow controller.

The decay ratio was less than 0.5 for each localized process variable (LPRM) that exhibited oscillatory response to control rod motion when operating above the lower limit of the master flow controller.

3.21.3 Discussion

At Test Conditions 1, 2, and 4 (natural circulation) a control rod and an adjacent LPRM were selected for movement and monitoring. The test involved moving the control rod one or more notches past the LPRM while simultaneously monitoring the response of the selected LPRM and other process variables. Over the ranges of reactor power and flow given by the above test conditions, the Level 1 and 2 criteria were satisfied. In fact, all of the tests the LPRM signal stabilized without any observable oscillatory motion. Additionally, the gross core signals (pressure, core flow, APRM, etc.) did not display any oscillatory behavior that could be attributed to rod motion.

It should be noted that STI 21 testing scheduled for Test Condition 6 was not performed because of utility test scheduling problems and the loss of STARTREC after the April 1977 generator breakdown.

The loss of data from this test condition was not considered serious because of the excellent plant response to the same test in natural circulation, a potentially more unstable position on the power/flow map. Also, many informal observations of LPRM and core response to control rod motion at high power (including Test Condition 6) have demonstrated the plant's ability to satisfy the Level 1 and 2 criteria of Startup Test 21.

3.22 STI 22 PRESSURE REGULATOR

3.22.1 Level 1 Criteria

The decay ratio was less than 1.0 for each process variable that exhibited oscillatory response to pressure regulator changes.

3.22.2 Level 2 Criteria

The decay ratio of any oscillatory variable was ≤ 0.25 when operating above the minimum speed for the master manual recirculation system mode. Below this speed, the decay ratio was ≤ 0.50 with the recommendation that each control system be adjusted to meet ≤ 0.25 unless there is an identifiable performance loss at higher power levels.

Pressure control system deadband, delay, etc., was small enough that steady state limit cycles, if any, produced turbine steam flow variations no larger than $\pm 0.5\%$ of rated flow (except as discussed below).

The response time from set point input until pressure peak was within 20 seconds in the recirculation system manual mode.

The normal difference between regulator set points was small enough that the peak neutron and thermal flux and/or peak vessel pressure remained below the scram settings by 7.5% and 10 psi, respectively.

3.22.3 Discussion

The pressure regulators were tested by the introduction of ± 10 psi step changes at Test Conditions 1 through 6. Various conditions of load limiting (bypass valves, control valves or both in control of the transient) and recirculation system control modes were used. Additionally, the backup capability of each pressure regulator was demonstrated via simulated failure of the controlling regulator. Actual plant test data indicated that a 5 psi bias between regulator set points was necessary for maintenance of adequate scram margins should a regulator fail.

After reviewing Test Condition 3 results, the settings of both regulators were changed. Additionally, a second steam line resonance compensator card (in each regulator) was found to be necessary to reduce a 5 Hz oscillation (thought to be from the bypass piping — Brunswick 1 has a different bypass configuration than Brunswick 2). The new, final settings are described in Table 3.22.3.1 and were used for the remainder of the test program. These settings allowed the regulators to successfully meet all criteria for the remainder of the test program and for the retesting of the lower power test conditions. Times to pressure peaks were typically 5 seconds for set point changes and 7 seconds for simulated failures, but in all cases always less than 10 seconds. Decay ratios were less than 0.25 at all test conditions and scram avoidance margins were generous. Table 3.22.3.2 summarizes several test conditions of the simulated regulator failure data.

During many of the ascensions to 100% power EHC data were taken to determine the steam flow to steam flow demand linearity and to determine how well the control valves were following their partial arc programming. The data from one power ascension is presented in Figures 3.22.3.2.6 through 3.22.3.2.10; linearity is within limits and the control valves were found to be accurately following their programming.

The pressure regulator (possibly) has one remaining unresolved problem. Oscillations have appeared at approximately 735 MWe corresponding to the 30% open position of the number 3 control valve; their magnitude is about 15 MWe peak-to-peak and the cause has not been identified. The utility is investigating the phenomenon and LSTG and NED have been notified.

3.23 STI 23 FEEDWATER SYSTEM

3.23.1 Level 1 Criteria

The decay ratio was less than 1.0 for each process variable that exhibited oscillatory response to feedwater system changes.

Table 3.22.3.1
FINAL PRESSURE REGULATOR SETTINGS

Pressure Regulator

Pressure Regulation		3.35
A Regulator		
	Lag (R6)	2.3
	Lead (R5)	3.5
B Regulator		
	Lag (R4)	2.6
	Lead (R3)	5.6

Steam Line Resonance Compensator Cards

Card Location	A42	A46	A38	A85
Notch Center	5.23	5.23	0.99	1.02
Notch Depth	2.0	2.0	2.0	2.02
Notch Width	1.655	1.655	1.77	1.77
Small Lag	2.12	2.12	14.21	14.32

	Bias Potentiometer
A Regulator In Control	5.0
B Regulator In Control	7.1
(5 psi Difference Between Regulators)	

All units given are potentiometer turns

Table 3.22.3.2
PRESSURE REGULATOR BACKUP PERFORMANCE

	Test Condition								
	1	2	3	5	5	5	6	6	
Recirc Mode	**LM**	**LM**	**MM**	**MM**	**MA PB=500**	**MA PB=1200**	**MM**	**MA PB=1200**	**Failure Type**
Peak Pressure (Limit 1025 psig)	940	947.3	963	970	970	971	1010.7	1008.5	B→A
	945	942.4	967	968	969	968.5	1011	1013	A→B
Peak Neutron Flux %	26.5	49.2	74.2	76.7	77.7	77.7	101.8	100.0	B→A
	26.5	56.7	79.1	73.8	74.8	75.7	102.2	105.0	A→B
Peak Heat Flux %	20.7	40.6	64.4	66.0	68.9	67.0	97.9	97.0	B→A
	21.2	42.0	64.9	65.5	68.0	65.5	97.3	98.0	A→B
Heat Flux Limit	51.5	54.4	103.5	80.1	85.0	79.1	108.2	109.5	B→A
	51.5	54.5	103.5	80.1	82.1	82.1	108.2	109.5	A→B
Time to Peak Maximum (Limit 20 sec)	<5	<5	<5	4	8	4	6	5	

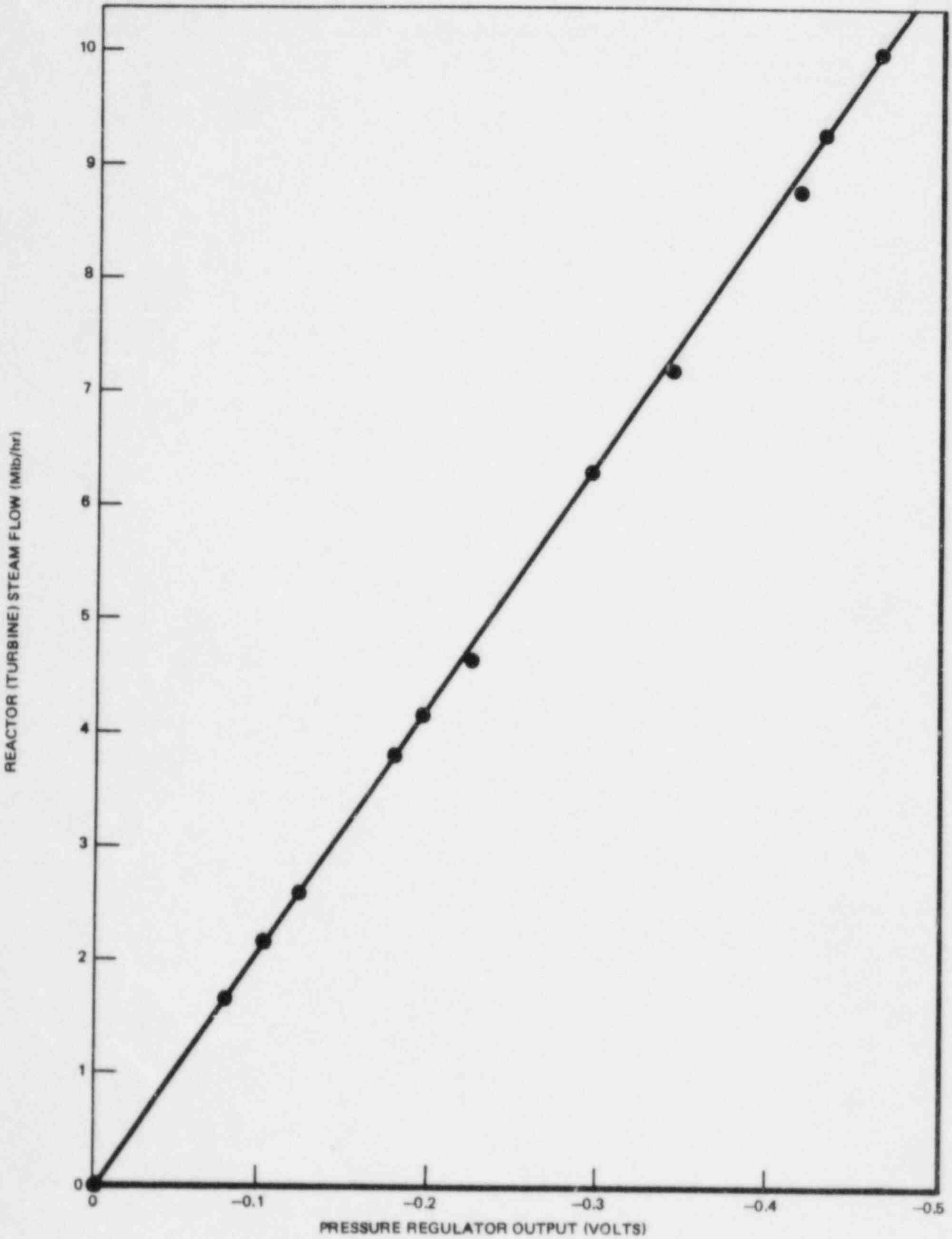


Figure 3.22.3.2.6. Steam Flow versus Pressure Regulator Output

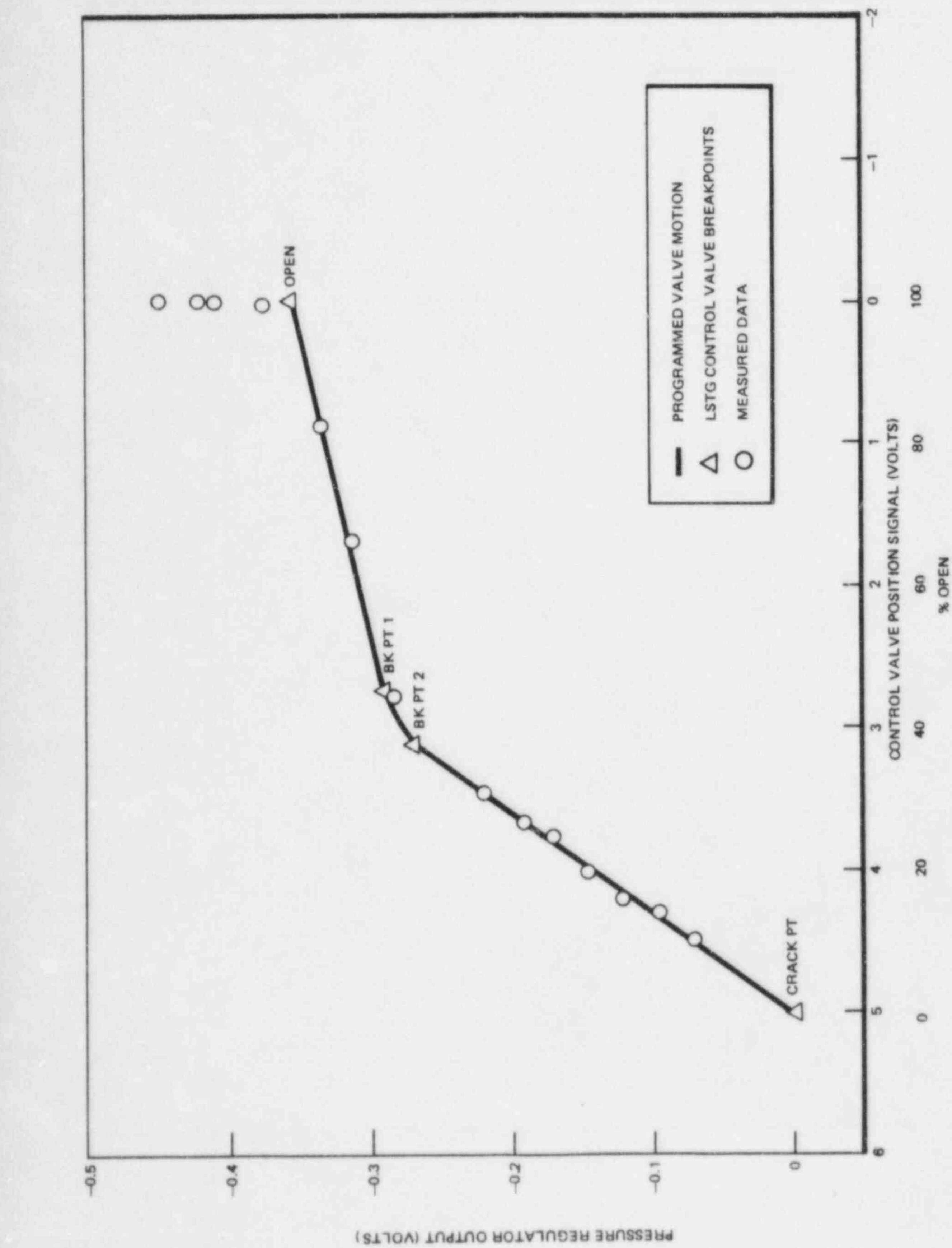


Figure 3.2.3.2.7. Control Valve No. 1 Programming.

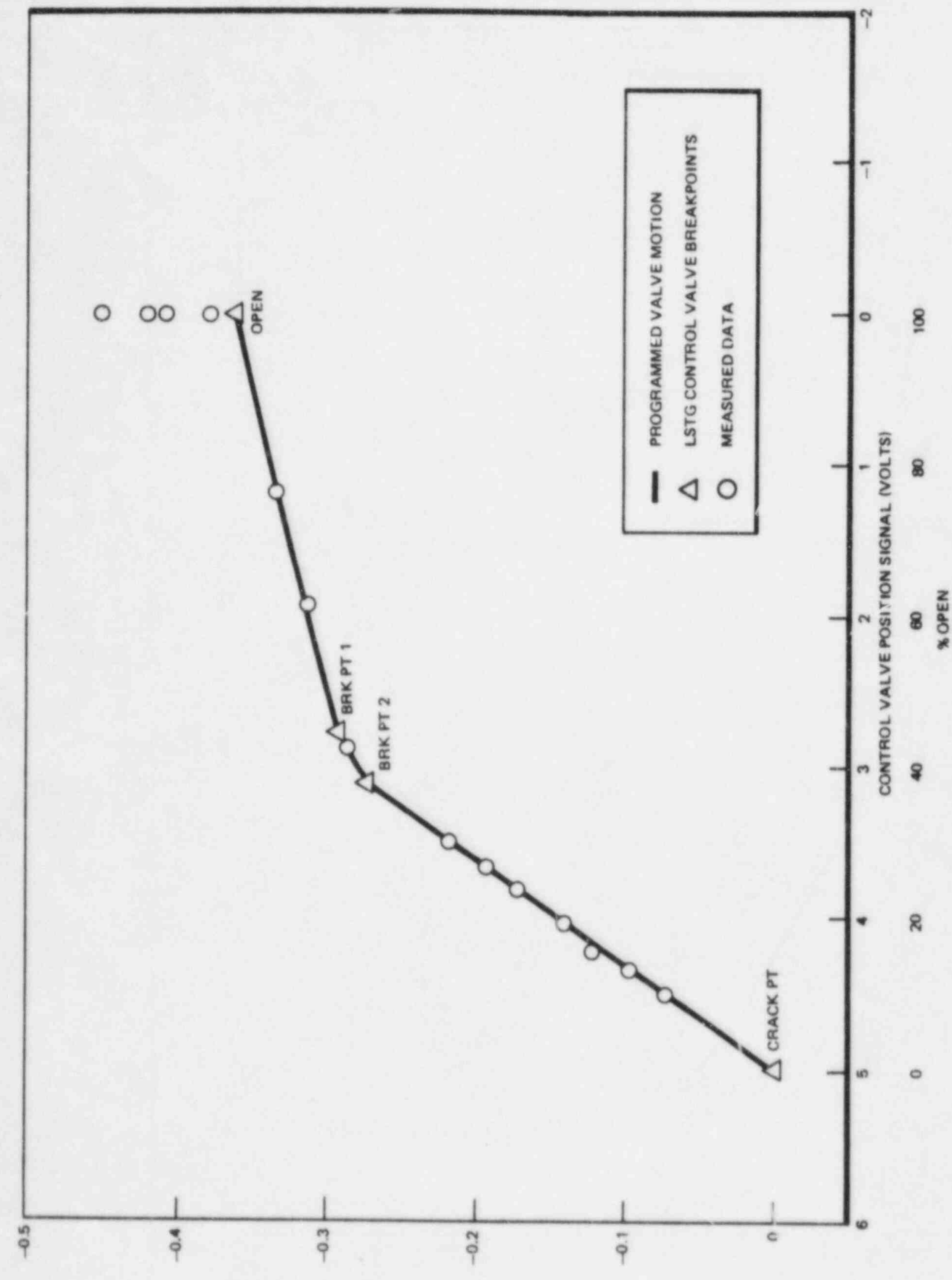


Figure 3.22.3.2.8. Control Valve No. 2 Programming.

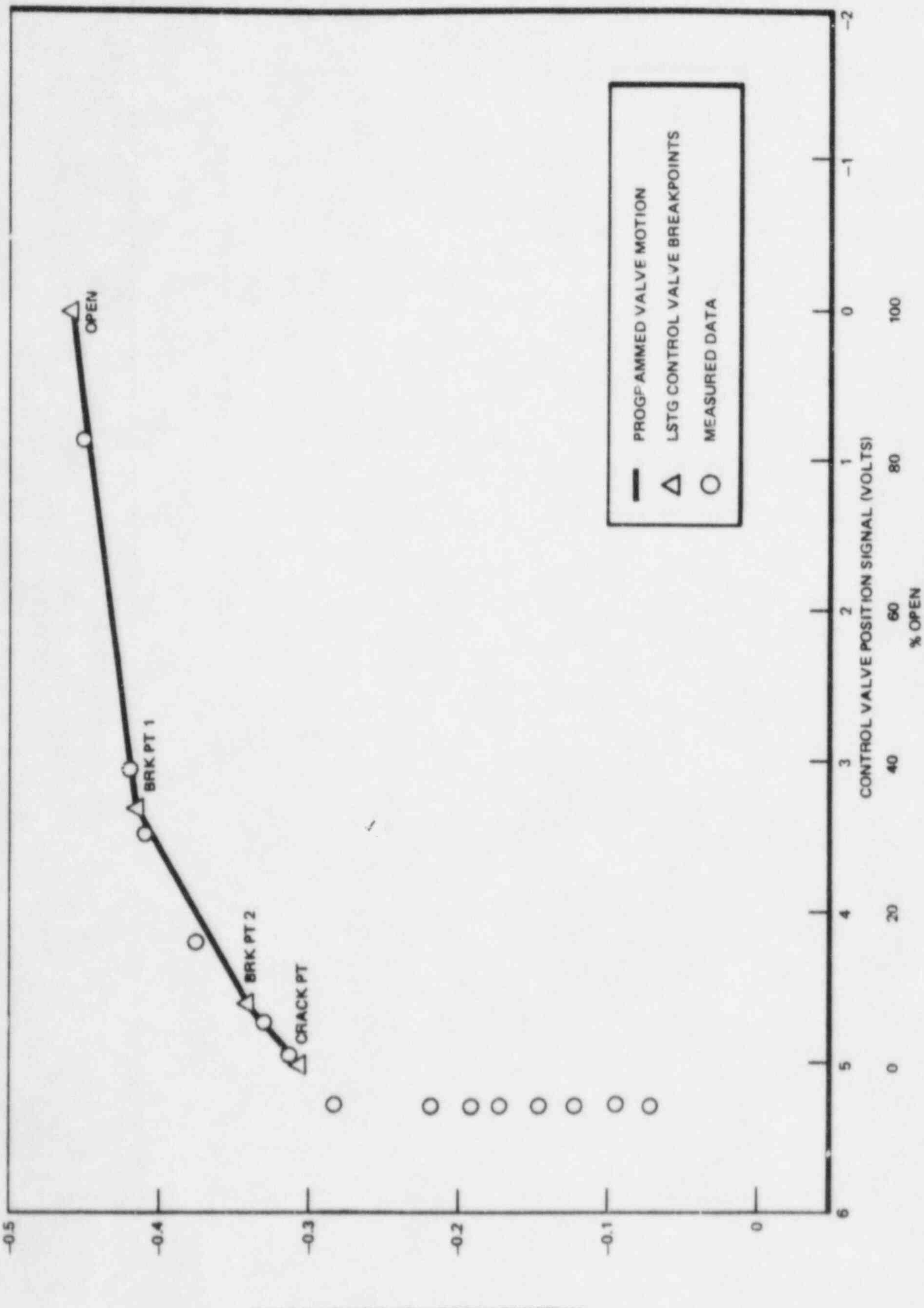


Figure 3.22.3.2.9. Control Valve No. 3 Programming.

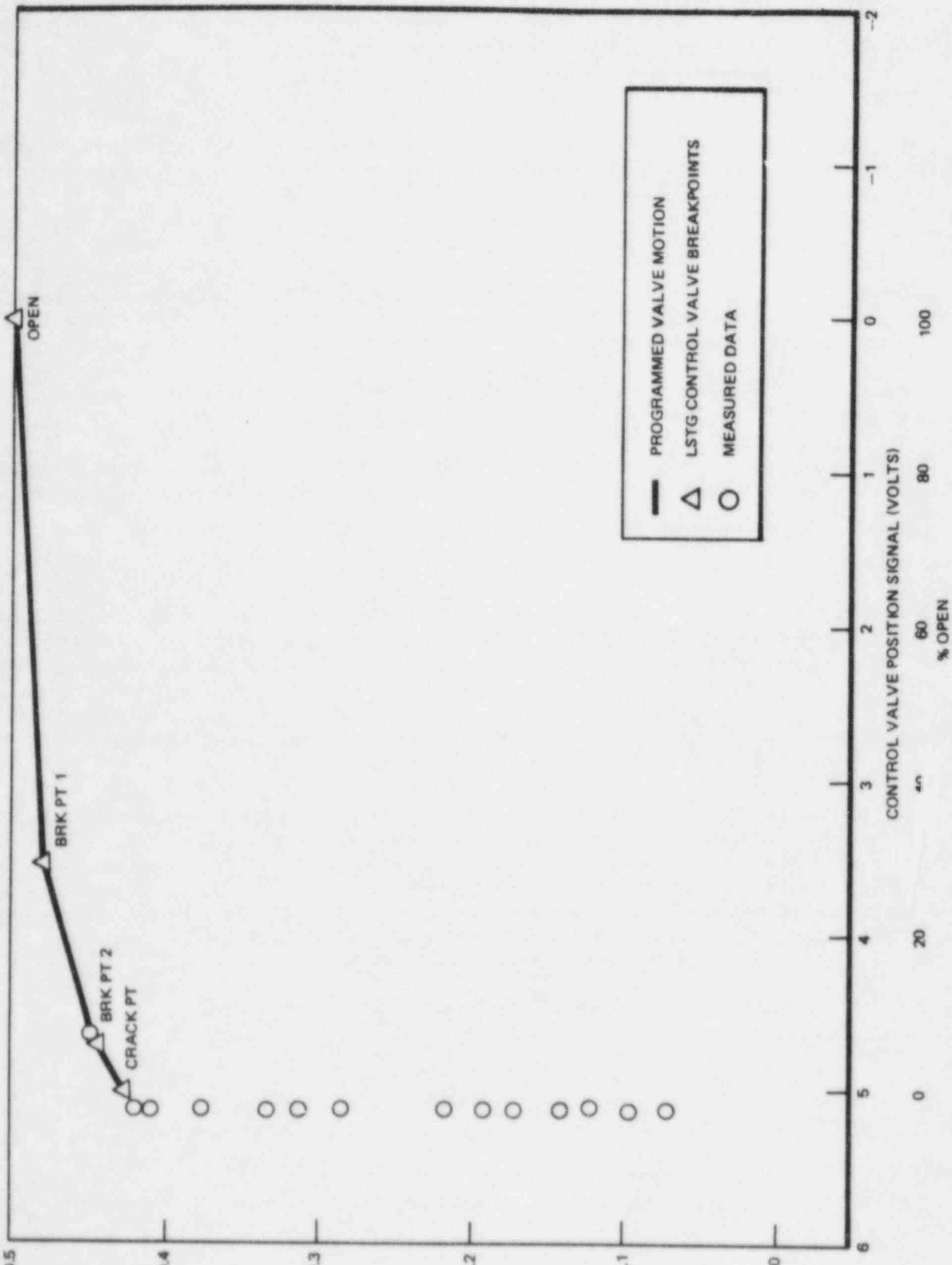


Figure 3.22.2.2.10. Control Valve No. 4 Programming.

Adequate fuel thermal limit margins were maintained following the trip of one feedwater heater; i.e., all applicable reactor and fuel license requirements were met throughout the transient.

The maximum feedwater temperature decrease due to a single failure was ≤100°F.

The increase in simulated heat flux did not exceed the predicted Level 2 criteria by more than 2%. The predicted value was based on the actual test values of feedwater temperature change and power level. In addition, the peak heat flux did not rise above the thermal power monitor scram set point.

3.23.2 Level 2 Criteria

The decay ratio of any oscillatory variable was ≤0.25 when operating above the minimum speed for the master manual recirculation system mode. Below this speed, the decay ratio was ≤0.50, with the recommendation that each control system be adjusted to meet ≤0.25 unless there is an identifiable performance loss at higher power levels.

A scram did not occur because of the trip of one of the operating feedwater pump turbines. There was greater than a 3 inch water level margin to avoid a scram for a feedwater pump turbine trip initiated from 100% reactor power.

The increase in simulated heat flux did not exceed the predicted value referenced to the actual feedwater temperature change and power level. Instrument uncertainty may be applied to the observed heat flux, but this uncertainty cannot exceed the design value for the standard plant instrumentation.

3.23.3 Discussion

During Test Conditions 1 through 6 of the startup test program a ±5-inch step change in level set point was imposed upon the water level control system. Single element mode, three element modes and master manual and master auto recirc system modes were tested. At low powers (Test Conditions 1 and 2) no attempt was made to optimize the feedwater system with only 1 pump operating. The initial setup produced a very stable (but slow) system. After both pumps were operated for the first time, the system was optimized concentrating on stable two pump operation at low powers. Later the low power test conditions were retested. Optimization of the level control system included the testing and setup of each feedwater turbine speed control system (signal converter).

At higher reactor powers, Test Conditions 5 and 6, level set point change results indicated that the feedwater system was not as responsive as it could be. These observations were accompanied by operator complaints of "sluggishness." Accordingly the level control system was optimized a second time, sacrificing some stability at lower powers (but still maintaining decay ratios of ≤0.25) but gaining "responsiveness" (shortening the time to level peaks). Before the reoptimization, reactor level would drop with large, fast increasing power changes; afterwards, this problem had been greatly mitigated. After these final adjustments level set point change testing was repeated at all affected test conditions (to maintain a consistent set of test results at all power levels). Final level control system settings were as follows:

FINAL FEEDWATER CONTROL SYSTEM SETTINGS

Proportional Band	82.5%
Resets Per Minute	1
Mismatch Gain	48 Inches
Lag Time Constant	1 Second
Lead Time Constant	5 Seconds

Table 3.23.3.1 summarizes the results of level set point change testing with the final feedwater control system settings.

On March 12, 1977 while operating at 97% power and flow in the master manual recirc system mode and 3 element feedwater control system mode, one of the two operating feedwater pump turbines ("A" Pump) was tripped. The recirc system runback began 7 seconds after the pump trip and reduced reactor power to 60% in approximately 33 seconds. By 24

Table 3.23.3.1
LEVEL SETPOINT CHANGE TESTING DATA

Test Condition	Recirc Mode	Control Mode	Step Direction	Decay Ratios	Time to Level Peak (Seconds)
1	LM	1 Element	Up	<0.25	31
			Down	<0.25	34
		3 Element	Up	<0.25	30
			Down	<0.25	31
2	LM	1 Element	Up	<0.25	34
			Down	<0.25	30
		3 Element	Up	<0.25	34
			Down	<0.25	34
3	MM	1 Element	Up	<0.25	21
			Down	<0.25	26
		3 Element	Up	<0.25	30
			Down	<0.25	28
5	MM	1 Element	Up	<0.25	10
			Down	<0.25	15
		3 Element	Up	<0.25	19
			Down	<0.25	23
5	MA (PB=500)	1 Element	Up	<0.25	12
			Down	<0.25	16
		3 Element	Up	<0.25	25
			Down	<0.25	25
6	MM	1 Element	Up	<0.25	18
			Down	<0.25	20
		3 Element	Up	<0.25	31
			Down	<0.25	30
6	MA (PB=1200)	1 Element	Up	<0.25	20
			Down	<0.25	19
		3 Element	Up	<0.25	29
			Down	<0.25	31
6	MA (PB=500)	1 Element	Up	<0.25	21
			Down	<0.25	20
		3 Element	Up	<0.25	31
			Down	<0.25	34

NOTES: LM = Local Manual
MM = Master Manual
MA = Master Automatic

seconds into the transient, the water level had dropped from its normal level of 36 inches to a minimum of 21.5 inches (maintaining a 6.5 inch scram margin).

Reactor level recovery and later power changes from subcooling were smooth and well behaved. Control system response was nonoscillatory, and the overall plant response was excellent.

Brunswick Unit 1 was the first plant to perform the "new" loss of feedwater heating test that was modified in response to new NRC concerns about feedwater temperature changes and plant modelling. For the Brunswick plant the worst single failure in the feedwater system (that can produce colder feedwater) was not the tripping of the last stages of feedwater heaters but rather the open failure of the valve in the line which bypasses both strings of fourth and fifth stage feedwater heating. The former failure produced feedwater temperature changes of ~30°F; the latter failure was predicted to produce a change of ~65°F. Both failures are slow, taking place over 3 to 5 minutes.

The NED transient group produced predicted power changes versus initial power and varying feedwater temperature changes. Utility (CP&L) heat balance computer codes and actual plant data was used to predict the temperature change that would result from bypassing the heater strings; i.e., how would the feedwater flow split between the bypass line and still-operating feedwater heaters. All of the data and predictions were used to determine an acceptable initial power from which to begin the test without violating PCIOMR's or plant safety limits.

On March 13, 1977 the feedwater heater bypass valve (1-F10-V120) was opened with the reactor at 73% power and 99% flow. The valve was completely open in 2 minutes and steady state conditions reached in 9 minutes. The transient was smooth and well behaved, and all thermal limits on the core were continuously satisfied. The increase in simulated heat flux was 111% (of initial) instead of a predicted value of 113%; all criteria were satisfied. Feedwater temperature dropped 63°F.

Before and after the transient computer program P1 was run; during the transient reactor power, core flow, subcooling, pressure and LPRM data were taken at approximately 1 minute intervals. The initial and final data were used to "benchmark" the off line computer program "BUCLE." Transient data were used as inputs to the BUCLE program "P1NEWRP." It was thus possible to obtain initial and final core thermal limits (which agreed closely with the process computer) as well as transient thermal limits. As expected the CPR smoothly decreased between its initial and final value; future plants should not find it necessary to obtain transient data. Before and after transient data are included in Table 3.23.3.2. Data generated during the transient are included in Table 3.23.3.3. The reactor power and CPR transient are graphed in Figure 3.23.3.

Table 3.23.3.2
THERMAL PARAMETERS DURING LOSS OF FEEDWATER HEATING TEST FROM 1651 TO 1659 3-13-77

Parameter	1651 Before	1659 Peak	%Change
CMWt	1774.0	1951	7.26
DHS (Btu/lbm)	15.40	22.0	30.1
MCHFR	4.19	3.64	13.1
MFLPD	0.676	0.777	13.0
CMPF	2.255	2.355	4.2
TFW (°F)	392.0	329.0	16.1
CPR	1.753	1.631	6.9 (Δ CPR=0.122)
MAPLHGR	7.75	8.90	12.9
Pressure	990.65	992.20	0.15

The above data were generated by the Off-Line Process Computer Program BUCLE using inputs from the Plant Process Computer Programs OD-8 (LPRM Console Readings) and OD-3 (reactor heat balance).

Table 3.23.3.3
TRANSIENT THERMAL PARAMETERS

LOSS OF FEEDWATER HEATING TEST

	Pre-Transient Bench Mark 1539 pm		Transient BUCLE Cases					Post-Transient Bench Mark 1659 pm	
	P1	BUCLE	1651	1652	1654	1656	1658	BUCLE	P1
Power (MWt)	1778	1778	1774	1851	1914	1940	1946	1951	1951
Subcooling	15.46	15.46	15.40	18.71	21.54	21.85	22.07	22.04	22.04
Core Flow	76.38	76.38	76.18	76.21	76.43	76.85	76.29	76.57	76.57
Core Peaking	2.210	2.207	2.255	2.262	2.339	2.331	2.355	2.355	2.353
CMFLPD	0.664	0.663	0.676	0.708	0.757	0.764	0.775	0.777	0.776
Location	27-8-4	27-8-4	27-8-4	27-8-4	27-8-4	29-8-4	27-8-4	27-8-4	29-8-4
CPR	1.7514	1.753	1.756	1.703	1.672	1.648	1.633	1.631	1.6342
Location	9-28	9-28	9-28	9-28	9-26	9-28	9-26	9-26	9-28
MAPLHGR	7.61	7.60	7.75	8.11	8.67	8.76	8.88	8.90	8.89
Location	7-26-4	7-26-4	7-26-4	7-26-4	27-8-4	7-24-4	27-8-4	27-8-4	7-24-4

3.24 STI 24 TURBINE VALVE SURVEILLANCE

3.24.1 Level 1 Criteria

The decay ratio was less than 1.0 for each process variable that exhibited oscillatory response to bypass valve changes.

3.24.2 Level 2 Criteria

The reactor did not scram because of the test. Peak neutron flux was at least 7.5% below the scram trip setting (flow biased thermal power monitor and 120% scram clamp). Peak vessel pressure remained at least 10 psi below the high pressure scram.

The reactor did not isolate because of the test. Peak steam flow in each line remained 10% below the high flow isolation trip setting. Additionally vessel pressure remained at least 25 psi above the steam line low pressure isolation.

3.24.3 Discussion

The turbine bypass valves were tested at Test Conditions 1, 2, 3, 5 and 6. The successfully completed bypass valve test program demonstrated that the EHC system had adequate capability to respond to abrupt changes in steam flow. Because of the slow opening and closing times of the bypass valves (5 to 10 seconds), little or no heat flux or neutron flux spiking was observed even at Test Condition 6. Scram avoidance margins were easily maintained and, as would be expected from a slow transient, all of the observed oscillatory responses had decay ratios less than 0.25.

The four turbine control valves were tested at Test Conditons 1, 2, 5 and 6. The latter two test conditions were actually part of a series of tests along the 100% load line wherein the observed heat flux, neutron flux, and pressure spikes were extrapolated to the next higher powered test. The highest power at which the valves were actually tested was 95%; however, the extrapolations of the above measured parameters indicate that (not considering PCIOMR's) it is possible to safely test all four control valves at 100% power. Taking PCIOMR's into account, assuming a valid 100% power envelope and considering the magnitude of the observed heat flux spikes, a power level of 95% should be considered an operational maximum until preconditioning restrictions are removed.

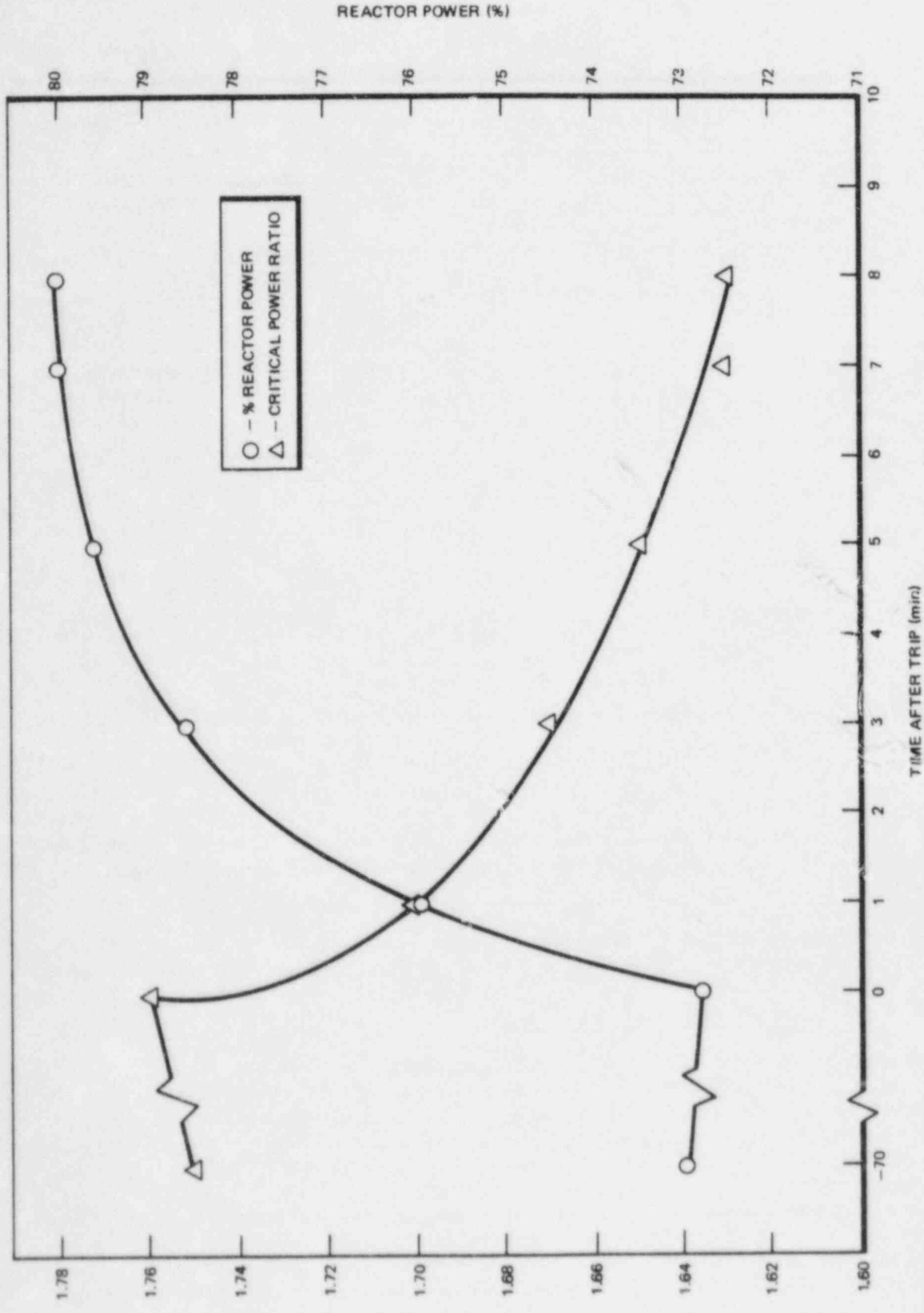


Figure 3.23.3.2.1. CPR and Reactor Power versus Time — Loss of Feedwater Heating Transient.

Before the startup program began, the control valve test circuitry was adjusted to maximize valve closing time; as a result valve reopening (which cannot be controlled) produced a greater transient than closing — although in a conservative direction. Additionally, Brunswick 1 added circuitry to the EHC system that minimized control valve number 3 testing transients (by biasing valve number 4 less full closed). It was recommended to CP&L that these modifications be made permanent. In all cases of valve testing, decay ratios were less than 0.25 and the regulator behaved in a smooth, stable manner; differential steam line flow and pressure was not a problem in control valve testing.

3.25 STI 25 MAIN STEAM ISOLATION VALVES

3.25.1 Level 1 Criteria

The MSIV closure times were found to be within the limits provided in the technical specifications.

For the full MSIV closure from full power, assuming no equipment failures and applying appropriate parametric corrections, predicted analytical results based on beginning-of-cycle design basis analysis were used as the basis to which the actual transient was compared. The following table specifies the upper limits of these criteria during the first 30 seconds following initiation at the indicated conditions.

Initial Conditions		Criteria	
Power (%)	Dome Pressure (psia)	Increase in Heat Flux (%)	Increase in Dome Pressure (psi)
100	1020	2	145

The feedwater system settings prevented flooding of the steam lines.

A reactor scram limited the severity of the neutron flux and simulated fuel surface heat flux transients within thermal limits during the MSIV full closure.

3.25.2 Level 2 Criteria

For the full MSIV closure from full power, assuming no equipment failures and applying appropriate parametric corrections, predicted analytical results based on beginning-of-cycle design basis analysis were used as the basis to which the actual transient was compared. The following table specifies the upper limits of these criteria during the first 30 seconds following initiation at the indicated conditions.

Initial Conditions		Criteria	
Power (%)	Dome Pressure (psia)	Increase in Heat Flux (%)	Increase in Dome Pressure (psi)
100	1020	0	120

The relief valves reclosed properly (without leakage) following the MSIV full closure.

During the full MSIV closure, RCIC started without manual assistance and operated without isolating when initiated from an automatic initiation signal, except as discussed below.

The reactor did not scram from individual MSIV closure; the peak neutron flux was at least 7.5% below the trip setting. The peak heat flux was at least 5% below the trip setting. The peak vessel pressure remained at least 10 psi below the high pressure scram setting.

The reactor did not isolate from an individual MSIV closure; the peak steam flow on each line remained 10% below the high steam flow isolation trip setting.

3.25.3 Discussion

All 8 MSIV's were tested for technical specification closing times at heatup, Test Conditions 1, 2, 3, 5 and during the MSIV full closure. The measured times are presented as Table 3.25.3.1; all closing times were found or adjusted to be within criteria. The ability of each valve to be functionally tested by a partial (100% to 90% open) closure was also demonstrated.

Table 3.25.3.1
MSIV CLOSING TIMES
(sec)

		Test Condition					
MSIV		**Heatup**	**1**	**2**	**3**	**5**	**MSIV Full Closure**
F022	A	4.40	4.4	4.63	4.4	4.63	4.35
	B	4.30	4.4	4.53	4.53	4.63	4.12
	C	4.53	4.7	4.98	4.64	4.54	4.28
	D	4.19	4.3	4.13	4.20	4.41	4.16
F028	A	4.19	4.2	3.96	3.97	4.09	4.08
	B	4.13	4.2	4.09	4.09	4.44	4.35
	C	4.53	4.5	4.31	4.55	4.31	4.37
	D	4.19	4.1	3.86	3.86	4.20	4.21

Additionally individual MSIV closure data were taken along the 100% load line to determine the maximum reactor power at which the test can safely be performed. For each test the observed steam line flow, reactor pressure, heat flux and neutron flux spikes were extrapolated to the next higher power. The highest power at which an MSIV was actually closed was 87%; however, extrapolations of the above measured parameters indicate that it is possible to safely shut an MSIV up to 90% power (pressure spike margin to scram becomes limiting).

The above extrapolations do not take PCIOMR's into account, but assuming a valid 100% power envelope they would be less limiting than the pressure margin to scram (but more limiting than the heat flux margin to scram). With the pressure scram raised, PCIOMR's would limit the power at which an MSIV could be shut to about 93%; the other parameters would begin limiting at about 96% power. In any case, Brunswick Unit 1's technical specifications require this test to be run at no higher than 75% power.

On September 30, 1977 Brunswick 1 performed the MSIV full closure test from 100% power/97% flow. Brunswick 1 was the first plant to attempt verification of transient modelling methods as well as demonstrating the reactor's response to a very severe transient. The reactor responses of interest (for modelling) were the pressure rise and heat flux increase that could possibly result from simultaneous closure of the MSIV's during the first 30 seconds of the transient. In advance of the actual test, pressure and heat flux rises were calculated for about 30 simulated cases which varied (one at a time) one of 9 independent variables. The resulting data were used to determine the sensitivity of the transient pressure and heat flux rises to the input variables. The 9 independent variables, their range (for the sensitivity study) and their nominal values are presented in Table 3.25.3.2. A 110 psi pressure and 0% heat flux rise was calculated for the base case using nominal nuclear characteristics; 120 psi is calculated for the same case using conservative nuclear characteristics.

The test was initiated by simulating high steam line radiation; a reactor scram (caused by MSIV's less than 90% open) occurred seven seconds later but before any pressure increase. The scram, in fact, allowed pressure to decrease ~10 psi by 3.6 seconds into the transient, but full closure of the MSIV's (4.3 seconds) rapidly started pressure increasing. An initial

Table 3.25.3.2
MSIV FULL CLOSURE SENSITIVITY STUDY

Nominal Value	Range	Variable
100%	(85% to 100%)	Initial Power
1020 psia	(1010 to 1030)	Initial Dome Pressure
1095 psia	(1085 to 1106)	Relief Valve Setpoint Pressure
85% NBR	(68% to 102%)	Relief Valve Capacity
0.4 sec	(0.2 to 0.6)	Relief Valve Opening Delay
0.1 sec	(0.05 to 0.2)	Relief Valve Opening Time Constant
67B	(0% to 25%)	Scram Speed
270 msec	(100 to 270)	Scram Delay
3 sec	(3 to 5)	MSIV Closure Speed

This results in a 110 psi pressure and 0% heat flux rise using "nominal" nuclear characteristics.

pressure peak of 1074 psig was reached by 6.6 seconds, almost simultaneously with the water level negative peak (caused by collapsing voids). Auto initiation of the HPCI and RCIC turbines, coupled with the still very large feedwater flow, caused reactor pressure to level out for about 6 seconds. A second reactor pressure peak of 1096 psig was reached (by 19 seconds) and relief values A and C lifted to reduce pressure to ~1043 psig.

The reactor feedpumps were not tripped until well over a minute into the transient (they remained running on the stored energy of the steam in the crossaround/reheater systems), but the feedwater control system responded correctly in recovering water level. Also, as sensed level crossed its set point feedwater flow was reduced to zero. The relief valves successfully reclosed and vessel level was normal at 26 seconds but increasing rapidly because of the successful HPCI injection. The cold HPCI water and running turbine held pressure almost level until the HPCI turbine was tripped (at 87 seconds). The relief valves did not lift again until over 2 minutes into the transient; no problems were encountered in equalizing and opening the MSIV's and getting back on the pressure regulator for a normal shutdown. A sequence of events is included as Table 3.25.3.3.

After using the sensitivity studies and inputting actual measured or calculated plant parameters existing at the time of the trip, a corrected pressure and heat flux rise of 96.4 psi and 0% was calculated for the Brunswick 1 trip. The corresponding measured values were 93 psi and 0% satisfying all Level 1 and 2 criteria. The reactor scram reduced power long before the −38-inch vessel level tripped the recirculation pumps, thus maintaining the reactor core within thermal limits.

During the transient the RCIC turbine auto initiated on level but tripped before it could begin significant pumping. The trip was unexplained but it was not a high steam flow isolation; it was felt that the probable cause was the RCIC system's recent flooding. CP&L operations was notified and the RCIC turbine was successfully run at a later time, indicating the tripping problem was of a temporary nature.

3.26 STI 26 RELIEF VALVES

3.26.1 Level 1 Criteria

The sum total of the capacity measurements of all 11 relief valves was greater than 9.181 Mlb/hr corrected to an inlet pressure of 1112 psig.

3.26.2 Level 2 Criteria

Each relief valve had an individual measured capacity of greater than 0.7512 Mlb/hr corrected to an inlet pressure of 1112 psig.

Table 3.25.3.3
MSIV FULL CLOSURE SEQUENCE OF EVENTS

Time (sec)	Events
−15	Fuse A71B-F2B Pulled (Channel B Trip)
0	Fuse A71B-F2A Pulled (Total Isolation Initiation)
0.694	Reactor Scram (MSIV's at 90% Open)
~0.8	APRM A & Heat Flux Start Decrease
~1.5	Steam Flow & Reactor Pressure Start Decrease
~3.0	Vessel Level Scram (12 inches)
3.6	Reactor Pressure Negative Peak 995 psig
~4.3	All MSIV's Shut Steam, Flow ~0.0
5.68	RCIC Initiation
5.77	HPCI Initiation
~6	Recirc Pump Trip
6.6	1st Pressure Peak (1073.6 psig)
~7	Vessel Level Negative Peak (FW Flow Still 8.8 Mlb/hr)
8.1	RCIC Turbine Start
~12	Reactor Pressure Starts to Increase
13.3	RCIC Turbine Speed Zero (Turbine Tripped)

Table 3.25.3.3
MSIV FULL CLOSURE SEQUENCE OF EVENTS (Continued)

Time (sec)	Events	
18.8	2nd Reactor Pressure Peak (1096 psig) (FW Flow 8.6 Mlb/hr) (Relief Valve Temperature Recorders Indicate Valves A and C Open)	
19.1	HPCI Turbine Start	
21.2	HPCI at 3600 rpm	
~24	Vessel Level at 25 inches	
26.1	Vessel Negative Pressure Peak (1043 psig) (Vessel Level 36 inches) (FW Flow 9.7 Mlb/hr)	
27.6	HPCI at 4700 rpm (FW Flow 8.6 Mlb/hr)	
32.2	HPCI Stable at 4400 rpm (FW Flow 1.6 Mlb/hr) (Vessel Pressure 1052 psig)	
~42	Vessel Level at 49 in. (Vessel Pressure 1052 psig)	
60	HPCI at 4400 rpm (FW Flow 0 Mlb/hr) (Vessel Pressure 1042 psig)	
~70.3	Isolation Reset	
76.3	A FW Pump Turbine Trip	
Initial	Power	99.9%
	Dome Pressure	1003 psig
	FW Flow	10.48 Mlb/hr
	Water Level	38 inches

The pressure regulator satisfactorily controlled the reactor transient by closing the turbine control or bypass valves by an amount equivalent to the relief valve steam flow. The characteristic response signatures for each valve did not show any significant differences.

The relief valve leakage was low enough that temperature measured by the thermocouples in the discharge side of the valves fell to within 10°F of the temperature recorded before the valve was opened. All thermocouples operated properly.

3.26.3 Discussion

The relief valves were functionally tested at 250 psig reactor pressure during heatup and at rated (950 psig) reactor pressure at Test Condition 1. All the valves were individually opened for about 10 seconds and successfully shut while recording the plant response (specifically pressure) on STARTREC. Discharge thermocouple response and pressure regulator response was acceptable.

A review of the reactor pressure response to valve opening indicated similar relief valve timing and flow behavior.

Two of the relief valves had been instrumented with ultrasonic probes to determine timing response, but due to instrumentation problems and the limited number of relief valve cycles the amount of data obtained did not fairly represent valve response. Therefore, the manufacturer's data were deemed an acceptable substitute.

Relief valve capacity was determined by recording bypass valve position response during the functional testing and subsequently calibrating the bypass valve positions versus an accurately measured steam flow (feedwater flow plus CRD flow). The calculated relief valve flows were corrected for reactor power and pressure, also measured during bypass and relief valve opening, and finally corrected to 1112 psig inlet pressure. A summary of relief valve capacities appears in Table 3.26.3.1.

Table 3.26.3.1
RELIEF VALVE CAPACITIES

Relief Valve	(Mlb/hr Flow at 1112 psig)
A	1.02
B	1.01
C	1.04
D	1.00
E	1.04
F	1.03
G	1.04
H	1.03
J	1.03
K	1.00
L	1.00
Total	11.24 Mlb/hr

3.27 STI 27 TURBINE TRIP AND GENERATOR LOAD REJECTION

3.27.1 Level 1 Criteria

High Power Trips:

Predicted analytical results based on beginning of cycle design basis analysis, assuming no equipment failure and applying appropriate parametric corrections, were used as the bases to which actual transient results are compared. The

following table specifies the appropriate upper limits for these criteria during the first 30 seconds following initiation at the indicated conditions.

Initial Conditions				Criteria	
Transient	Test Condition	Power (%)	Dome Pressure (psia)	Increase in Heat Flux (%)	Increase in Dome Pressure (psi)
Turbine Trip	3	75	988	2	105
Generator Breaker Trip	6	100	1020	2	131

The feedwater system settings prevented flooding of the steam lines.

3.27.2 Level 2 Criteria

Predicted analytical results based on beginning of cycle design basis analysis assuming no equipment failure and applying appropriate parametric corrections, were used as the bases to which actual transient results are compared. The following table specifies the appropriate upper limits for these criteria during the first 30 seconds following initiation at the indicated conditions.

Initial Conditions				Criteria	
Transient	Test Condition	Power (%)	Dome Pressure (psia)	Increase in Heat Flux (%)	Increase in Dome Pressure (psi)
Turbine Trip	3	75	988	0	80
Generator Breaker Trip	6	100	1020	0	106

Pressure regulator settings were such that the bypass valves regained pressure control before the turbine inlet pressure reached the low pressure main steam line isolation set point.

Feedwater control system settings helped prevent low level initiation (−32 inches) of the HPCI system and main steam line isolation (−32 inches) for as long as feedwater flow remained available.

Bypass valve quick opening began by 0.1 second after the start of stop valve closure, and 80% of the bypass valve motion was complete within another 0.2 second.

3.27.3 Discussion

A generator load rejection within bypass valve capacity was performed on January 4, 1977 by opening the main generator output breakers at 20.3% reactor power. The control valves closed from 7.3% open in ~0.7 seconds. The bypass valves began to move 0.1 second after the control valves and reached a final value of 70% in about 3 seconds. No increase in heat flux or reactor pressure was observed, the APRM spikes were about 1%. The maximum turbine speed was 1920 rpm and within 36 seconds it returned to normal.

The reactor handled the transient very smoothly with no danger of an isolation or a scram. The level and pressure control systems were quite stable and maintained their process parameters within their normal range.

Brunswick Unit 1 experienced two inadvertent turbine trips; one from 72.5% power/100% flow on February 21, 1977 and the other from 70.0% power/100% flow on Feruary 3, 1977. Both turbine trips were acceptable for use as the planned startup testing transient and, where applicable, both trips met all Level 1 and 2 criteria. Brunswick Unit 1 was the first plant to attempt verification of transient modelling methods as well as demonstrating the reactor's response to turbine trips and load rejections (discussed later). The reactor responses of interest (for modelling) were the pressure rise and heat flux increase that could possibly result during the first 30 seconds of the transient. In advance of the actual test, pressure and heat flux rises were calculated or about 20 simulated cases which varied 6 independent variables. The resulting data were used to determine the sensitivity of the transient pressure and heat flux rises to the input variables. The 6 independent variables, their range (for the sensitivity study) and their nominal values are presented as Table 3.27.3.1. A 75 psi pressure and 0% heat flux rise was calculated for the base case using nominal nuclear characteristics; 80 psi is calculated for the same case using conservative nuclear characteristics.

Table 3.27.3.1
TURBINE TRIP SENSITIVITY STUDY

Nominal Value	Range	Variable
75%	(65% to 85%)	Initial Power
988 psia	(988 to 1000)	Initial Dome Pressure
26%	(20% to 35%)	Bypass Valve Capacity
0.1 sec	(0.05 to 0.2)	Bypass Valve Delay Time
67B	(0 to 25%)	Scram Speed
270 msec	(100 to 270)	Scram Delay

This results in a 75 psi pressure and 0% heat flux rise using "nominal" nuclear characteristics.

The February 21, 1977 turbine trip was accidentally initiated from a false high water level (58 inches) signal from a mistakenly drained reference leg. The resulting turbine trip was in all respects identical to a planned trip except that the feedpump, HPCI, and RCIC turbines were tripped with the main turbine; a sequence of significant events is included as Table 3.27.3.2.

The bypass valves were quick enough to satisfy the time to first motion and opening time criteria; however, because the reactor isolated on low level (because of the feedpump turbine trip) the feedwater and pressure control systems could not be fairly tested. HPCI and RCIC did successfully auto initiate and inject.

Using the sensitivity studies and inputting actual measured plant parameters existing at the time of the trip, a corrected pressure and heat flux rise of 58.4 psi and 0% was calculated for this Brunswick 1 turbine trip. The corresponding measured values were 51.1 psi and (essentially) 0%.

The February 3, 1977 turbine trip was a legitimate trip caused by low condenser vacuum, caused in turn by the loss of the circulating water pumps. Since the feedpump turbines did not (immediately) trip, the resulting transient was closer to the intent of the planned turbine trip. The sequence of events is included as Table 3.27.3.3.

The bypass valves were again quick enough to satisfy their time to first motion and speed criteria, thus enabling them to regain pressure control before a low pressure isolation was reached. The feedwater control system prevented steam line flooding (by reducing flow to zero) and low water level (by increasing flow to maximum). The lowest level reached was approximately −8 inches, the reactor did not isolate nor did HPCI or RCIC initiate.

By using the sensitivity studies a corrected pressure and heat flux rise of 54.3 psi and 0% was calculated for this trip. The corresponding measured values were 49.6 psi and (essentially) 0%. Both turbine trips met all Level 1 and 2 criteria; neither trip opened a relief valve and no neutron flux spikes were observed.

Table 3.27.3.2
FEBRUARY 21, 1977 TURBINE TRIP
SEQUENCE OF EVENTS

Time (sec)	Event	
0	Turbine, Feedpump, HPCI & RCIC Trip	
0.100	Stop Valve No. 2 1st Motion, Heat Flux Peak	
0.160	Bypass Valve 1st Motion, Reactor Scram	
0.200	Stop Valve No. 2 Shut	
0.260	Control Valve 1st Motion	
0.340	Main Generator CB's Open, Bypass Valve No. 1 Full Open, Total Bypass Valve Position 83.5% Open Control Valve Fast Closure Signal	
0.720	Control Valves Shut	
1.320	Turbine Speed Peak (101.6% Speed)	
~3.0	0-inch Water Level	
3.460	Reactor Dome Pressure Peak (1026.69 psig)	
6.020	HPCI and RCIC Auto Initiate on Level	
6.920	Total Isolation Signal	
~7	APRM's 0%, Heat Flux 49%	
12.40	Bypass Valves in Pressure Control (Valves Closing)	
Initial	Power	72.5%
	Dome Pressure	975 psig
	FW Flow	7.25 Mlb/hr
	Water Level	37 inches

On February 28, 1977 a generator load rejection was performed from 100% power and 97% flow by opening the main generator output breakers. The sequence of events is included in Table 3.27.3.4. As for the turbine trips a parametric study was done in advance of the transient wherein about 30 simulated cases — which varied (one at a time) one of 10 independent variables — were modelled. The resulting data were used to determine the sensitivity of the transient pressure and heat flux rises to the input variables. The 10 independent variables, their range (for the sensitivity study) and their nominal values are presented as Table 3.27.3.5. A 95 psi pressure and 0% heat flux rise was calculated for the base case using nominal nuclear characteristics; 106 psi is calculated for the same case using conservative nuclear characteristics.

Table 3.27.3.3
FEBRUARY 3, 1977 TURBINE TRIP
SEQUENCE OF EVENTS

Time (sec)	Events	
0	Turbine Trip (from Low Vacuum)	
0.100	Stop Valve No. 2 1st Motion	
0.160	Bypass Valve 1st Motion, Reactor Scram	
0.200	Stop Valve No. 2 Shut, Control Valve 1st Motion	
0.340	Control Valve Fast Closure Signal	
0.720	Control Valves Shut	
0.820	Turbine Speed Peak (100.8% Speed)	
~3.0	APRM's 3%, Heat Flux 57%	
3.480	Reactor Dome Pressure Peak (1013.69 psig)	
4.25	Negative Level Peak (−8 inches)	
15.660	Bypass Valves in Pressure Control	
Initial	Power	70.0%
	Dome Pressure	964 psig
	FW Flow	6.62 Mlb/hr
	Water Level	35 inches

Because the trip was above 30% power (and not within the capacity of the bypass system) the reactor scrammed at 35 milliseconds on control valve low oil pressure. The bypass valves were quick enough to satisfy their first motion and speed criteria and to show that the pressure regulator had regained pressure control before the low pressure isolation was reached. Feedwater flow was available throughout the transient, and the control system acted to prevent a low level isolation and steam line flooding. The lowest water level reached was −30 inches, and HPCI and RCIC did not auto initiate.

The turbine speed peak was 104% and was reached 1 second into the transient. Reactor pressure peaked at 1094 psig, 2.9 seconds into the transient, and the three relief valves (C, D and E) which opened to relieve pressure successfully reclosed.

Using the sensitivity studies a corrected pressure and heat flux rise of 90.0 psi and 0% respectively, was calculated for this load rejection. The corresponding measured values were 86.3 psi and 0%; additionally no APRM spikes were recorded. The load reject test satisfied all Level 1 and 2 criteria.

Table 3.27.3.4
FEBRUARY 28, 1977 LOAD REJECTION
SEQUENCE OF EVENTS

Time (sec)	Event	
0.0	Main Generator Breaker Trip, CV Fast Closure Initiated	
0.035	CV First Motion, Scram	
0.050	BPV First Motion	
0.075	SV No. 2 First Motion	
0.165	SV No. 2 Full Shut	
0.235	BPV 80% Open	
0.305	BPV 90% Open	
0.395	CV Full Shut	
0.505	BPV Full Open	
0.950	Turbine Speed Peak (1879.58 rpm)	
2.0	APRM 12% Heat Flux 84%	
2.885	Peak Dome Pressure (1093.97 psig)	
3.25	Vessel Level 0 inches	
6.25	Vessel Level Peak −30 inches APRM 0% Heat Flux 56%	
21.5	Dome Pressure Negative Peak (912 psig) Water Level 20 inches	
22.5	Bypass Valves Shut	
Initial	Power	100%
	Dome Pressure	1007.6 psig
	FW Flow	10.48 Mlb/hr
	Water Level	38 inches

Table 3.27.3.5
GENERATOR LOAD REJECT SENSITIVITY STUDY

Nominal Value	Range	Variable
100%	(85% to 100%)	Initial Power
1020 psia	(1010 to 1030)	Initial Dome Pressure
85%	(68% to 102%)	Relief Valve Capacity
0.1 sec	(0.05 to 0.2)	Relief Valve Time Constant
1095 psia	(1085 to 1106)	Relief Valve Setpoint
0.4 sec	(0.2 to 0.6)	Relief Valve Delay Time
26%	(20% to 35%)	Bypass Valve Capacity
0.1 sec	(0.05 to 0.2)	Bypass Valve Delay
67B	(0% to 25%)	Scram Speed
270 msec	(100 to 270)	Scram Delay

This results in a 95 psi pressure and 0% heat flux rise using "nominal" nuclear characteristics

3.28 STI 28 SHUTDOWN FROM OUTSIDE THE CONTROL ROOM

3.28.1 Level 2 Criteria

During the simulated control room evaluation, the reactor was brought to the point where cooldown was initiated and under control, the reactor vessel pressure and water level were controlled using equipment and controls outside the control room.

3.28.2 Discussion

The reactor was scrammed and isolated from ~20% power by tripping both RPS M-G set circuit breakers. Adequate vessel level was maintained by the coasting feedpumps. At no time did vessel water level approach the HPCI/RCIC system's auto initiation level (−38 inches). Reactor pressure did not rise to the relief valves' pressure set point; as water level rose the operator manually opened 2 relief valves (one at a time) for 10 seconds. Since the pressure peak was only 1005 psig, the valve openings were more of a functional check of the shutdown panel rather than a pressure relief function.

At 840 psig and 65-inch reactor pressure and level, a successful cooldown having been started, the test was terminated 19 minutes after initiation.

3.29 STI 29 FLOW CONTROL

3.29.1 Level 1 Criteria

The decay ratio was less than 1.0 for each process variable that exhibited oscillatory response to recirculation flow changes.

3.29.2 Level 2 Criteria

The decay ratio of any oscillatory variable was less than 0.25 when operating above the minimum speed for the master manual recirculation mode, except as discussed below. Below this speed, the decay ratio was less than 0.50 with the recommendation that each control loop be adjusted to meet ≤0.25 unless there is an identifiable performance loss at higher power levels.

The auto load following range along the full power rod line was at least 35% of rated power, except as discussed below.

In both the master manual and master auto modes, along the power flow line on which the load following specification was demonstrated, a flow change which would cause a 65% to 100% power change on the 100% load line was accomplished in 60 seconds.

Plus or minus 10% and ±20% power step changes were performed within 40 seconds.

A reactor scram did not occur due to the flow control maneuvers. The neutron flux margin was ≥7.5% and the heat flux margin ≥5.0%. All the power maneuver rates were extrapolated to those that would occur along the 100% rated rod line to obtain their true effect.

Flow control system deadband, delay, etc., was small enough that steady state limit cycles (if any) produced turbine steam flow variations no larger than ±0.5% of rated steam flow.

3.29.3 Discussion

The recirculation flow control system was tested (after STI-32 optimization) at Test Conditions 1, 2, 3, 5, and 6 in the master automatic (preconditioning and nonpreconditioning modes) and master manual modes of operation (the reactor power and flow were actually somewhat higher at Test Conditions 2 and 5 to allow operation on the master recirculation controller). Small flow and power changes and large ramp changes in power were used to demonstrate acceptable system operation; the data from all power changes were examined for flux scram margins, stability, and responsiveness. Results of system testing were used to determine controller settings for preconditioning and nonpreconditioning modes of system operation. Additional test data were used to set the various limiters in the recirculation system.

Results of recirculation system step flow change testing are presented in Table 3.29.3.1. Test Condition number 1 testing was not included in Table 3.29.3.1 (because it cannot be performed on the master recirculation controller) but the local manual tests passed all criteria. Because the high APRM decay ratios at Test Condition 2 did not repeat at Test Condition 5 (a potentially more unstable position on the power flow map) and because a recirculation system noise problem was thought to have existed during Test Condition 2 testing, the tests were not repeated. Although within criteria, the step change times in Table 3.29.3.1 are somewhat deceptive; times from 10% to 90% of the step were typically 8-10 seconds with the remaining time spent in system settling. The long power change times in the preconditioning mode (PB = 1200%) are, of course, deliberate.

Remaining flow testing consisted of large (~50% flow) flow ramps beginning from 100% core flow at Test Conditions 3 and 6. The ramps from Test Condition 3 (70% load line) had to be used to demonstrate ultimate plant performance along the 100% load line (by extrapolation) because of PCIOMR restrictions. After several trials a proportional band of 500% was chosen for the nonpreconditioning mode of master auto operation; this produced ramp times within criteria, flux margins, and the required stability. Both master auto (PB = 500%) and master manual modes are capable of producing a 35% power change along the 100% load line within one minute. A proportional band of 1200% was chosen for the preconditioning mode of master auto operation. The lower gain provided much slower ramp response, typically, 12% power/minute on the 100% load line, which is well within the PCIOMR required 15%/minute.

The master recirculation controller limiter settings needed to produce the (approximately) 35% power change along the 100% load line corresponded to 41% and 89% recirculation pump speeds. Specifically, 89% recirculation pump speed results in 100% core flow at 100% power; the Bailey scoop tube positioner electrical and mechanical stops were set to pump speeds corresponding to 101% and 102% core flow. Several of the Test Condition 6 large power ramps fell 1% or 2% short of meeting the required 35% power change. Although there was enough decay ratio margin to lower the minimum master recirculation controller speed, this would have required extensive retesting. Because of the small failure margin, the utility's desire not to retest and because the NSSS contractual requirement is **approximately** 35% power, it was decided to accept the test results as satisfactory.

Data from the large power ramp tests are summarized in Table 3.29.3.2. The reactor response to all flow induced power changes was stable, predictable (because of the cam shape), well behaved, nonoscillatory, very responsive and had adequate flux margins to scram. Final controller settings are indicated below; the controller was left in the preconditioning mode.

FINAL MASTER RECIRC CONTROLLER SETTINGS

Proportional Band	
500%	Nonpreconditioning Mode
1200%	Preconditioning Mode
Resets/Minute	
10	
Rate	
0 (Minimum)	
Limiter Settings	
Master Controller	
High	89% Speed (100% Core Flow at 100% Power)
Low	41% Speed
Bailey (High Speed)	
Mechanical	102% Core Flow (at 100% Power)
Electrical	101% Core Flow (at 100% Power)

3.30 STI 30 RECIRCULATION SYSTEM

3.30.1 Level 1 Criteria

Not applicable.

3.30.2 Level 2 Criteria

The single recirculation pump trips did not result in a high water level turbine trip. The level margin was at least 3.0 inches at Test Condition 6.

The reactor did not scram during the recirculation pump restart. The scram avoidance margins were 7.5% for neutron flux and 5% for thermal flux.

3.30.3 Discussion

STI 30 testing consisted of a cavitation search, recirculation pump trips, and data accumulation; they are discussed separately below.

Data from the recirculation pump trips is presented in Table 3.30.3.1. In all cases reactor power, pressure, and core flow decreased as expected. The feedwater system was able to mitigate the resulting level swell and maintain adequate

Table 3.29.3.1
RECIRC SYSTEM STEP FLOW CHANGE TESTING
Test Condition

	2						3					
	Negative			Positive			Negative			Positive		
	MM	MA PB=500	MA PB=1200	MM	MA PM=500	MA PB=1200	MM	MA PB=500	MA PB=1200	MM	MA PB=500	MA PB=1200
Decay Ratios												
Core Flow	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25
A Drive Flow	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25
B Drive Flow	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25
APRM	0.5	0.465	<0.25	0.44	0.418	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25
Trip Margins (%)												
Heat Flux	34	36	33.5	38	34.5	37	45	41.5	39	48	39	41
APRM	70	74	75	58	52	59	57	61.5	53	50	35	45
Steady State Limit Cycle	<0.5%	<0.5%	<0.5%	<0.5%	<0.5%	<0.5%	<0.5%	<0.5%	<0.5%	<0.5%	<0.5%	<0.5%
Step Size (% Power)	9.2	8.6	8.1	9.3	7.9	9.5	6.7	7.1	7.2	7.8	7.4	8.1
Time to Complete Transient (sec)	38	22	22	32	32	18	20	32	66	20	36	82

	5						6					
	Negative			Positive			Negative			Positive		
	MM	MA PB=500	MA PB=1200	MM	MA PM=500	MA PB=1200	MM	MA PB=500	MA PB=1200	MM	MA PB=500	MA PB=1200
Decay Ratios												
Core Flow	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25
A Drive Flow	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25
B Drive Flow	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25
APRM	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25
Trip Margins (%)												
Heat Flux	24	20.7	18.9	19.1	16.8	19.3	20.1	20	20.4	16.5	16.1	19.2
APRM	65	47.7	47.2	40	33.8	46.6	21	19.2	20.4	15.7	13.5	18.6
Steady State Limit Cycle	<0.5%	<0.5%	<0.5%	<0.5%	<0.5%	<0.5%	<0.5%	<0.5%	<0.5%	<0.5%	<0.5%	<0.5%
Step Size (% Power)	8.5	8.5	8.4	9.2	8.7	9.1	10.9	11.5	17.8	10.4	12.5	17.8
Time to Complete Transient (sec)	32	40	48	18	40	68	38	40	120	38	34	54

Table 3.29.3.2
LARGE FLOW RAMP TESTING

	Test Condition											
	3						6					
	Negative			Positive			Negative			Positive		
	MM	MA PB=500	MA PB=1200	MM	MA PB=500	MA PB=1200	MM	MA PB=500	MA PB=1200	MM	MA PB=500	MA PB=1200
Decay Ratios												
Core Flow	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	—*	<0.25
A Drive Flow	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	—	<0.25
B Drive Flow	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	—	<0.25
APRM	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	<0.25	—	<0.25
Trip Margins												
Heat Flux	—	—	—	40	40	42.7	—	—	—	15	—	12
APRM	—	—	—	39	38	41.3	—	—	—	19	—	20
Differential												
Power (%)	23.3	25.4	23.3	25.4	24.1	23.3	34.7	32.0	31.7	35.6	—	30.6
Flow (%)	48.8	49.5	48.1	49.0	49.0	49.0	53.0	53.3	52.4	54.0	—	51.9
Time to Complete Flow Change (sec)	38	54	264	56	44	480	52	46	480	210*	—	156
Time to Complete Power Change (sec)	36	46	264	50	42	480	50	48	480	210*	—	156

*PCIOMR Restrictions

margins to the high level trip (58 inches). Because of the flow biased scram and rod block, reactor power was reduced with control rods shortly after the 2 recirculation pump trip.

Data from the recirculation pump trips is presented in Table 3.30.3.1 in all cases reactor power, pressure, and core flow decreased as expected. The feedwater system was able to mitigate the resulting level swell and maintain adequate margins to the high level trip (58 inches). Because of the flow biased scram and rod block, reactor power was reduced with control rods shortly after the 2 recirculation pump trip.

Operation after the recirculation pump(s) were tripped was stable and well behaved, even in natural circulation. The pump restarts were equally uneventful and all maintained adequate flux margins to scram. Vibration and performance data were taken during the pump trips and later maneuvering necessary to restart them and subsequent return to initial power. All Level 2 criteria were met.

A cavitation search was made from Test Condition 3 and 95% flow by inserting control rods in the inverse order of the operational sequence. The resulting power reduction continued until the feedwater flow limit that initiates a recirculation pump runback was reached (A and B recirc pump "flow limit"); since the scoop tubes had been locked before the test was started an actual runback could not occur.

Because the runback signal was received at 25% power (21% feedwater flow) and the nominal set point is 20% feedflow, power was further reduced to 20% (17% feedflow). No signs of cavitation were observed in the jet pumps or recirc pumps at any power level during the test.

Recirculation system performance data were taken at Test Condition 1 along the 60% and 100% flow control lines and at Test Condition 4 (natural circulation). Performance of the system was satisfactory at all conditions.

Table 3.30.3.1
RECIRCULATION PUMP TRIP DATA

Recirc Pump Tripped	Initial Power/Flow	Type of Trip	Level Swell (inches)/ Time into Transient (sec)	Level Trip Margin (inches)	Neutron/Thermal Flux Margin to Scram (Pump Restart)
B	62%/100%	Motor Breaker	6.5/5.6	19	No Spikes Observed
A	96%/100%	Field Breaker	11/43	14	55%/7%
Both	97%/99%	Motor Breaker	8/50	17.5	No Spikes/12.5% A Pump No Spikes/14.3% B Pump

3.31 STI 31 LOSS OF TURBINE GENERATOR AND OFFSITE POWER

3.31.1 Level 1 Criteria

HPCI and/or RCIC systems were not required to maintain the reactor vessel water level above the initiation level of the LPCI, core spray, and automatic depressurization systems.

All safety systems, including the reactor protection system, RCIC, HPCI, and the diesel generators functioned properly or were made to function properly without manual assistance.

The reactor protection system prevented violation of neutron flux and simulated fuel surface heat flux thermal power limitations.

3.31.2 Level 2 Criteria

Reactor water level, neutron flux, heat flux, and pressure signals were not unexplainably worse than the predictions in the FSAR and plant transient safety analysis report.

The normal reactor cooling systems were able to maintain adequate suppression pool water temperature, maintain adequate drywell cooling, and to prevent actuation of the automatic depressurization system.

3.31.3 Discussion

The test was performed at 32% reactor power by manually operating a (differential) relay to simulate a generator fault. The plant switch gear had been previously arranged to simulate failure of the busses that the automatic switching circuitry would fast transfer to.

The main turbine tripped and the reactor scrammed and isolated as expected. Reactor water level did not drop to the HPCI/RCIC automatic initiation set point. No increase was noted in the heat or neutron flux signals, nor was the transient behavior of the level, flux and pressure signals worse than the transient described in the FSAR. Suppression pool and drywell temperatures were maintained at adequate levels throughout the transient and resulting cooldown.

Five minutes after the isolation, reactor pressure had increased to the point where three relief valves opened; a fourth valve was manually opened to control reactor pressure. The RCIC and HPCI systems were manually started and used to successfully control vessel level and pressure.

As a result of the simulated failure of all a-c power, all diesel generators were expected to start, but only number 1 and number 2 were expected to synch and load into the emergency busses. Diesels number 3 and number 4 started (and were not expected to load) but tripped off after approximately 30 seconds due to low oil pressure. A plant modification was generated, approved, and installed to prevent this condition from repeating. The loss of offsite power was again simulated in the same manner (but with the reactor shutdown); all four diesels performed satisfactorily.

3.32 STI 32 RECIRCULATION M-G SET SPEED CONTROL

3.32.1 Level 1 Criteria

The decay ratios of each process variable that exhibited an oscillatory response to recirculation M-G set speed changes were less than 1.0.

3.32.2 Level 2 Criteria

The decay ratios of each process variable that exhibited an oscillatory response to recirculation M-G set speed changes were less than 0.25 when operating above the lower speed limiter of the master recirculation controller and less than 0.5 when operating below the lower speed limiter.

The recirculation speed control systems were adjusted to meet the requirements of the Level 2 open loop performance criteria of the startup transient test specification.

After a 10% speed demand step from the low end of the speed range, the time from the step demand to the generator peak speed was less than 25 seconds.

Steady state limit cycles were less than 0.5% as measured by the turbine steam flow.

Step inputs between 90% and 100% speed were adjusted so that 10% of the demanded change was reached within 2 seconds, and the time between 10% and 90% of the demanded change was less than 5 seconds, except as discussed below.

3.32.3 Discussion

The recirculation M-G set speed control loops were not really set up until the plant first reached 100% flow at Test Condition 3. Enough data were taken to enable properly shaped cams to be cut for each M-G set; comparison of the speed and speed demand meters over the operating ranges of interest indicate a good shape was obtained. It was also necessary to readjust the Bailey positioner linkage to ensure that an appropriately large portion of the cam's surface was used.

After the above setup was accomplished, the M-G set controllers were adjusted for best response at approximately 5 points evenly spaced in core flow (~20% to 90% speed). Over the speed ranges of most plant operation (30% to 90%), the recirculation system was adjusted to meet all Level 1 and 2 criteria.

Some difficulty was encountered at high speeds (90% +) both because the recirculation M-G set speeds had to be deliberately mismatched to avoid exceeding the 100% core flow limit and because of voltage regulator instability problems. The voltage regulator problems will eventually be resolved by correcting an MV/I converter noise problem and by new voltage regulator controller settings. This checkout also discovered a tachometer calibration problem. The test schedule did not allow time to retest at 100% recirculation M-G set speeds; the data obtained are from ~94% speed.

In any case, this is not an operational problem since the Bailey mechanical and electrical stops have been set at ~92% and ~91% speed (102% and 101% core flow at 100% power). Over the speed ranges of interest, the M-G sets are very stable and responsive and meet the Level 1 and 2 criteria. Final controller settings are:

	A M-G Set	B M-G Set
Proportional Band	200	200
Resets/min	15	16
Rate	0	0

3.33 STI 33 MAIN TURBINE STOP VALVE SURVEILLANCE TEST

3.33.1 Level 2 Criteria

The reactor did not scram because of these tests. The peak neutron flux was maintained at least 7.5% below the scram trip settings. Peak reactor pressure remained at least 10 psi below the high pressure scram setting. Peak heat flux was maintained at least 5% below the scram trip setting.

The reactor did not isolate because of these tests. The peak steam flow in each steam line remained at least 10% below the high flow isolation trip setting.

3.33.2 Discussion

The turbine stop valves were tested at Test Condition 3 (63% power) and later were tested along the 100% load line at increasingly higher powers. Each stop valve was individually closed while monitoring steam line flow, neutron and heat flux and reactor pressure. The highest value of each parameter (for any stop valve closure) was then plotted versus reactor power and extrapolated to the next anticipated power level. If the scram avoidance margins satisfied the Level 2 criteria, the test was repeated at that higher power level.

In this manner the highest allowable power level at which a single stop valve could be closed and maintain prudent scram avoidance margins was determined to be ~93% (limited by reactor pressure). Considering the results of other BWR/4 testing, this is a low value. However, it is perhaps biased by the lower than normal reactor pressure (for the power level) and the 1035 psig (versus 1045 psig) pressure scram set point. Further testing could result in a higher permissable power level, but in any case the observed heat flux spikes would quickly limit the highest allowable power level to ~95%-96% power because of current PCIOMR restrictions. Even this value assumes a valid 100% envelope and a test in the same rod pattern at which the envelope was stored.

3.34 STI 34 VIBRATION MEASUREMENTS

3.34.1 Level 1 Criteria

The peak stress intensity did not exceed 10,000 psi (single amplitude) when any (vessel internals) component was deformed in a manner corresponding to one of its normal or natural modes. This is the low stress limit which is suitable for sustained vibration in the reactor environment for the design life of the reactor components.

3.34.2 Level 2 Criteria

The peak stress intensity did not exceed 80% of the Level 1 criteria.

3.34.3 Discussion

Before fuel loading, RTD's, strain gauges, and accelerometers had been installed in or on selected reactor vessel internals. During heatup, through generator synchronization, and at selected points to 100% power, feedwater sparger temperature data were obtained. Although there are no criteria associated with this test, a subsequent analysis indicated that the spargers' blend radii are well protected against thermal cycling.

During the single motor breaker trip at 70% power and during the subsequent recovery along an intermediate load line vibration data were obtained. Similarly, during the single field breaker trip and two recirculation pump motor breaker trip and subsequent recovery along the 100% load line more data were obtained.

Finally, at selected power levels, feedwater sparger vibration data were obtained. All of the above data were determined to have satisfied the Level 1 and 2 criteria after a quick site analysis, and later, after a more detailed analysis by Component Engineering in San Jose. It is expected that a topical report on Brunswick Unit 1 vibration testing will be issued some time after the startup testing period.

Aside from the normal testing, the vessel internal vibration sensors were used to monitor for the onset of cavitation during STI 30 testing. This is not required by the test program but no cavitation was detected in any case.

3.35 STI 35 RECIRCULATION SYSTEM FLOW CALIBRATION

3.35.1 Level 2 Criteria

The jet pump flow instrumentation was adjusted such that the jet pump total flow recorder provides the correct core flow indication at rated conditions.

The APRM/RBM flow-bias instrumentation was adjusted to function properly at rated conditions.

3.35.2 Discussion

The core and drive flow instrumentation was calibrated at a nominal 100% core flow at Test Conditions 3 and 6. Rated drive flow (that flow which gives 100% core flow at 100% power) was determined to be 43,174 gpm (each loop). This flow was obtained at ~89% recirculation pump speed and 52.16 ft of differential pressure across the drive flow nozzles. The flow converters were adjusted to produce 8 volts (100%) total flow output into the flow biased scram and rod block circuitry with the above mentioned A and B loop inputs.

It should be noted that Brunswick Unit 1's drive flows have been recalibrated to 55 kgpm (from 70 kgpm) full scale flow. The flow converter gains, plant meters and recorders, nozzle ΔP calibrations and process computer calibrations were all adjusted to accommodate this change. The change seems desirable in that Brunswick is not licensed to use its equalizer valve and thus will not have need of the higher measurement capability.

The San Jose time share computer program "JR PUMP" was also used to calculate total core flow. Three sets of single tap and double tap jet pump differential pressures (from 4 minute MAXMIN averaged STARTREC recordings) were used to generate 3 "JR PUMP" runs. The 3 results (loop and core flows) were averaged, and the results used to adjust the loop flow proportional amplifiers and total flow proportional amplifier.

The computer output was also used to determine the double tap jet pump differential pressure transmitter full scale pressures (35.33 psi). The average M ratio was determined to be 1.34 as compared to the design value of 1.26. Jet pump riser and nozzle plugging data were all acceptable. The CP&L I&C shop was notified of those electronic components whose input/output relationships indicated that an adjustment was needed.

In order to use the most advantageous CPR calculation in the process computer, each Bailey positioner mechanical stop was set to a speed corresponding to 102% core flow and each electrical stop set at a speed corresponding to 101% core flow. The master flow controller limiter was set to disallow any pump speed above 100% core flow.

Finally, Table 3.35.2.1 itemizes the final drive flow/core flow correlation that was input to the process computer; this has allowed a WTFLAG = 2 (WTSUB within 5% of WT) to be obtained at any drive flow and with the reactor anywhere between the 70% and 100% load lines.

Table 3.35.2.1
DRIVE FLOW/CORE FLOW CORRELATION

WD (Mlb/hr)	8	12	16	21	27	33
WTSUB (Mlb/hr)	24.96	33.12	40.48	50.40	63.72	77.00
M Ratio	2.12	1.76	1.53	1.4	1.36	1.33

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SUMMARY

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