ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 1

MAY 23, 1975 - FINAL SUMMARY REPORT, UNIT 2 STARTUP
BROWNS FERRY NUCLEAR PLANT

MAY 23 1975

MAY 29 1975
ENGINEERING DESIGN
DIRECTOR'S OFFICE

Mr. Benard C. Rusche, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20545

BFN-100
BFN-63B

Dear Mr. Rusche:

STARTUP REPORT - BROWNS FERRY NUCLEAR PLANT UNIT 2 - DOCKET NO. 50-260 - OPERATING LICENSE DPR-52

In accordance with Browns Ferry Nuclear Plant Technical Specifications 6.7.1.a.(1), we are submitting herewith the Final Summary Report for Browns Ferry Unit 2 Startup Test Program. We are sending you 39 copies of this report under separate cover.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

J. E. Gilleland
Assistant Manager of Power

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TENNESSEE VALLEY AUTHORITY

Division of Power Production

FINAL SUMMARY REPORT

UNIT 2 STARTUP

BROWNS FERRY NUCLEAR PLANT

Submitted by

Plant Superintendent

Approved by

Chief, Nuclear Generation Branch

Browns Ferry Nuclear Plant
Decatur, Alabama

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STARTUP TEST RESULTS

FINAL REPORT

BROWNS FERRY NUCLEAR PLANT UNIT 2

Abstract

The final report of the startup test program performed at Browns Ferry Nuclear Plant Unit 2 is presented in three parts: (1) Introduction, (2) Summary, and (3) Results. Res . from core physics, thermal-hydraulics and system performance tests are presented such that the actual empirical values obtained are compared against expected or design values. Where deviations were noted, resolutions or corrective actions are also described.

1.0 Introduction

1.1 Purpose

The purpose of this report is to ,resent a concise summary and pertinent detailed results obtained in the performance of startup tests at Browns Ferry Nuclear Plant Unit 2. The startup test program embraced core physics, thermal-hydraulic, electromechanical and overall system dynamic performance.

1.2 Plant Description

Browns Ferry Nuclear Plant Unit 2 is a single-cycle boiling water reactor designed by General Electric Company (GE) for the Tennessee Valley Authority (TVA) and is the second of a three-unit site to be placed in service. The plant is located on the Tennessee River in Northern Alabama. The design gross electrical output is 1098 MWe, derived from a core thermal power of 3293 MWt.

1.3 Startup Test Program

Near the time of completion of plant construction, the preoperational test program begins. This period is designated as Phase I of the test program, during which testing of components, subsystems and combined systems are performed. These tests are not covered in this report.

The startup test program begins with the loading of nuclear fuel and continues through the completion of 100% power testing and the warranty run. It is composed of Phases II through V, as follows:

- Phase II - Open Vessel and Cold Testing
- Phase III - Initial Heatup
- Phase IV - Power Tests
- Phase V - Warranty Tests

1.3 Startup Test Program (Continued)

During this period the plant is taken to its designed full-power operating condition in a safe, controlled, gradual fashion. Extensive testing is performed under selected, controlled operating conditions to demonstrate safe, efficient performance of plant components.

The startup test program began with fuel loading on July 2, 1974, and continued through completion of the warranty run and 100% power testing. Commercial operation began on March 1, 1975.

1.4 Startup Test Description

Documents such as the Operating License (DPR 52), Technical Specifications, Plant Operating Procedures, and equipment manuals, control operations during the plant startup test program. Two documents are supplied by GE-NED for implementation of the startup testing of the equipment it supplies; the startup test specification and the startup test instruction (STI).

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

The Startup Test Instruction is a document written for use in the control room by qualified GE personnel and for trained TVA personnel working with GE technical direction. It contains sufficient pertinent information to permit such personnel to properly perform and evaluate each startup test.

TVA Division of Engineering Design (DED); Division of Power Production, Plant Engineering Branch; and Browns Ferry engineers reviewed the GE Startup Test Specification and Startup Test Instructions; and with appropriate revisions, specific Browns Ferry Master Hot Functional Test Instruction (MHFTI), Master Startup Test Instruction (MSTI), and Startup Test Instructions (STI's) were issued.

The MHFTI and MSTI coordinated and documented all test activities from initial fuel loading to the completion of all startup tests. These instructions provided guidance for sequence of events, and control points for satisfactory test completion and review before power ascension.

The GE-supplied STI's were revised for clarity, to reference plant instructions, and to include specific instrument numbers on data sheets. These STI's were finally reviewed by the Plant Operations Review Committee (PORC) and approved by the TVA plant superintendent and GE site operations manager.

FINAL SUMMARY REPORT - BFNP UNIT 2

1.5 Startup Test Acceptance Criteria

The Startup Test Instruction for each startup test contains 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 the 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 since there are no planned operations of testing at such levels.

By meeting the criteria, startup test results demonstrate agreement with design specifications and predictions. Startup test results were reviewed and approved by PORC and the plant superintendent and are undergoing a final review and evaluation by TVA DED.

2.0 Summary of Test Results

2.1 Chronology of Startup Testing

This section presents in tabular form the significant dates of the startup test program. Table 2-1 gives the dates of major events in the unit 2 startup. Table 2-2 gives the dates by which each test or major part thereof was completed. Table 2-3 shows a power flow map and the various test conditions.

2.2 Phase II - Open Vessel and Cold Testing

2.2.1 STI-1, Chemical and Radiochemical

Chemical tests of the primary coolant were made prior to heatup and yielded the following results:

Conductivity (umho/cm @ 25° C.)	0.28
Chloride (ppb)	<50
Turbidity (FTU)	0.06
Boron (ppb)	<50
Silica (ppb)	15

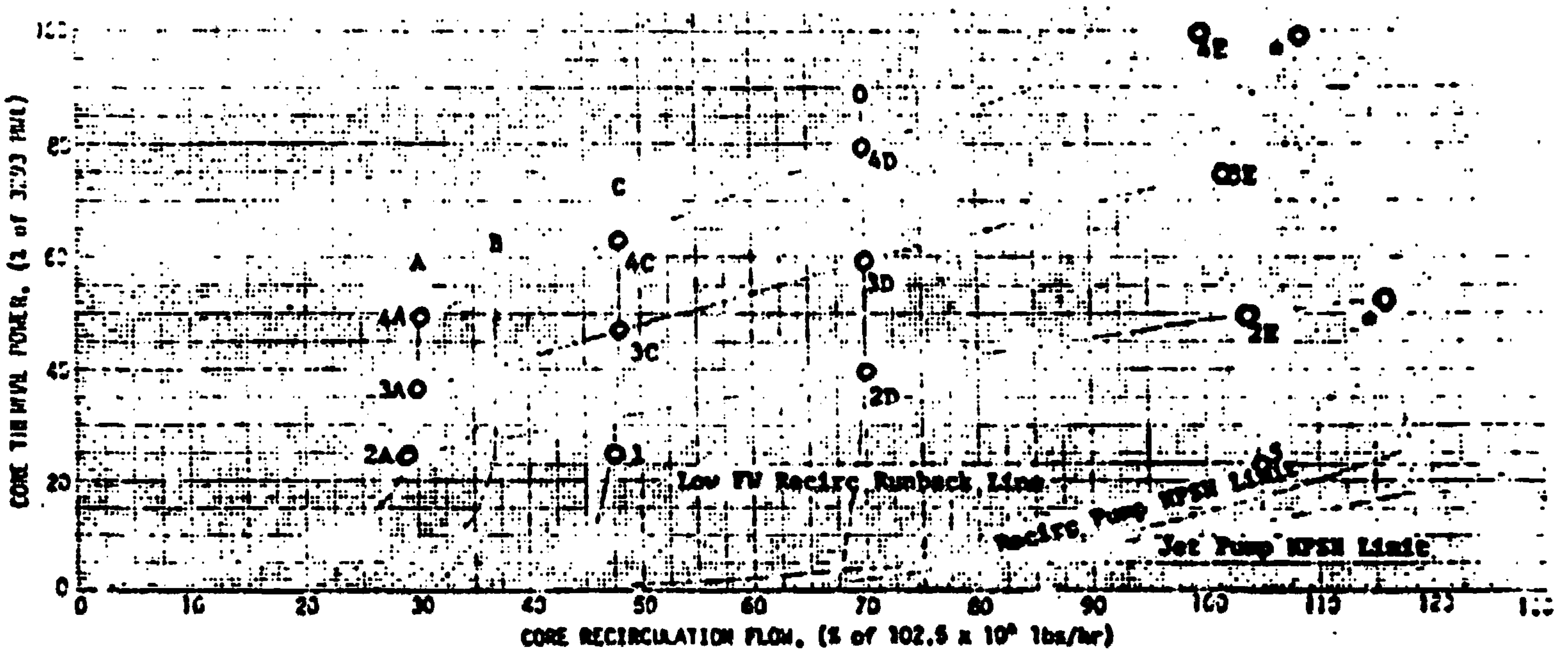
All criteria were satisfied.

Table 2-1

Major Events of Unit 2 Startup Test Program

Date	Event
July 2, 1974 (2125 hours)	First fuel assembly loaded.
July 14, 1974 (2030 hours)	Core fully loaded to 764 fuel assemblies
July 20, 1974	Initial critical during STI-4, Shutdown Margin Demonstration. Also initial in-sequence critical same day.
August 2, 1974	Full Power License receive by TVA for BF-2
Autust 3, 1974	Begin initial nuclear heatup
August 9, 1974	Reached rated temperature and pressure
August 29, 1974	Initial generator synchronization
August 31, 1974	Completion of Heatup Test Phase
September 16, 1974	Completion of 25% testing
October 6-25, 1974	Transformer Outage
November 11, 1974	Completion of 50% testing
December 3, 1974	Completion of 75% testing
February 11, 1975	Completion of 100% testing
February 14, 1975	UT inspection of piping welds in drywell
March 1, 1975	COMMERCIAL OPERATION
March 9, 1975	Completion of 100-hour warranty demonstration

	TEST CONDITIONS	Heating	Cold	Cold	<10%	1	2B	2[illegible]	2A	3[illegible]	3[illegible]	3[illegible]	3[illegible]	4[illegible]	4[illegible]	
[illegible]	[illegible]ical and Radiochemical	7/16			8/8	9/1		9/30						11/23		
[illegible]	[illegible]iation Measurements	7/16			8/11	9/1		9/30				11/16				
[illegible]	[illegible] Loading	7/19				[illegible]										
[illegible]	[illegible] Shutdown Margin	7/[illegible]														
[illegible]	[illegible]	[illegible]		[illegible]	[illegible]/20	[illegible]										
[illegible]	[illegible] and Control Rod Sequence	7/21	8/3		[illegible]/12	9/16										
	[illegible] Level Measurement				8/12	9/3										1/[illegible]
	[illegible] Performance	[illegible]/21	8/3			9/16										
[illegible]	[illegible] Calibration					9/16		10/24				11/26				
[illegible]	[illegible] Calibration		8/4			[illegible]		10/5				[illegible]/26				
[illegible]	[illegible] Computer				[illegible]/16			10/4								
[illegible]	[illegible]		8/6	[illegible]	[illegible]/30	9/3										
[illegible]	[illegible]		8/6	8/13	8/[illegible]			10/7								
	[illegible] Process Temperature				8/11				10/20				11/10			
[illegible]	[illegible] Expansion	10/15	8/6		8/13	9/[illegible]										
[illegible]	[illegible] Power Distribution					9/9		10/20				11/24				
[illegible]	[illegible] Performance					9/15	1[illegible]/[illegible]	10/4	10/26	1[illegible]/23	11/10	11/26	[illegible]1/10	[illegible]/23	1[illegible]/15	
[illegible]	[illegible]															[illegible]
[illegible]	[illegible] Response to Rods					9/1		9/30				11/17			[illegible]	
[illegible]	[illegible] Regulator Setpoint Changes					9/2		9/30				[illegible]		12/23	[illegible]	
[illegible]	Backup Regulator					9/2		9/30				11/17				[illegible]
[illegible]	[illegible] System FW Pump Trip															
[illegible]	[illegible] Level Setpoint Change					9/2		10/6				11/17		12/23	[illegible]	
[illegible]	[illegible] Valves					9/2		1[illegible]/26	[illegible]			11/17		12/23	12/19	
[illegible]	Steam Isolation Valves				8/20			11/6							1/[illegible]	
	[illegible] Valve		8/6			9/9										
[illegible]	[illegible] Trip and															
	[illegible]enerator Load Rejection					9/9										
	Flow Control					9/3		10/6		11/10	11/10	11/12	See [illegible]			
[illegible]	[illegible]irculation System							10/20				11/24				[illegible]
11	[illegible] of T-G and Offsite Power					9/20										
12	[illegible] M-G Set Speed Control						11/11	10/9		11/10	11/10	11/17		12/23	12/19	[illegible]
13	Turbine Stop Valve Surveillance Test					9/3		9/30				11/17				[illegible]
14	[illegible]ration Measurement							10/26	10/26	10/27	11/10	11/18	11/10	12/31	12/31	
15	[illegible]irculation System Flow Calibration							10/3				11/16				[illegible]
20	[illegible] Water Cleanup System				8/26											
21	[illegible] Heat Removal System					9/9										
22	Drywell Atmospheric Cooling System				8/26											[illegible]
23	Cooling Water System				8/26											[illegible]
24	[illegible]-Gas System															



TEST CONDITION NO.	1	2A	2D	2E	[illegible]	3A	3C	3D	3E	4A	4C	4D	4E	[illegible]	5
Flow Line	44%	50%				75%				100%					
% PUMP SPEED	~41	0*	~68	E	[illegible]	0*	~41	~68	E	0*	~41	~68	E	[illegible]	E*
% POWER	15-35*	~25	30-50*	40-60*	*	~37	37-57*	50-70*	65-85*	~50	55-75*	70-90*	95-100*	[illegible]	[illegible]
% CORE FLOW	~47*	NC	~70*	~104*	[illegible]	NC	~48*	~70*	~102*	NC	~46*	~70*	~103*	[illegible]	[illegible]

* Asterisked values are set as initial test conditions; non-asterisked values are estimates.
NC Natural Circulation
V Varies

<u>CONSTANT PUMP SPEED LINES</u>

A Natural Circulation
B 20% Pump Speed
C Analytical lower limit of Master Flow Control (~41% speed)
D Contractual lower limit of Flow Control (~68% speed)
E Pump speed for rated flow at rated power
F Nominal curve of max allowable pump speed (equipment limits other than core)
G Analytical flow corresponding to max allowable steady state fuel channel ΔP

APPROXIMATE POWER FLOW MAP SHOWING STARTUP TEST CONDITIONS
Table 2-3

2.2 Phase II - Open Vessel and Cold Testing (Continued)

2.2.2 STI-2, Radiation Measurement

A complete plant survey was taken with the core fully loaded and all control rods fully inserted. All radiation levels were below instrument minimum detectable limits, so all criteria were met.

2.2.3 STI-3, Fuel Loading

Fuel loading began on June 29, 1974 with the loading of the operational sources, and was successfully completed on July 19, 1974. At that time all seven operational sources were installed, all four SRM's were connected and functional, all 764 fuel assemblies were installed and the core verification completed. Partial core shutdown margin tests were performed periodically during fuel loading, satisfying the criteria.

2.2.4 STI-4, Core Shutdown Margin

After the functional test of the SRM's (STI-6) the shutdown margin test was conducted. The analytically strongest rod, 26-07, was fully withdrawn, and then the adjacent rod, 22-03, was notched to position 14. Subcriticality was verified by the SRM's and it was demonstrated that a reactivity margin >0.38% ΔK/K existed.

The clump critical test demonstrated that the core had an "all rods in" k_{eff} of .933.

All test criteria were satisfied.

2.2.5 STI-5, Control Rod Drive System

All control rods met the criteria of the tests performed. CRD 10-23 failed to meet the 90% scram time limit during initial scram testing, but was retested satisfactorily. All the required tests were performed twice on each CRD, during and following fuel loading.

2.2.6 STI-6, SRM Performance

The SRM's were functionally tested before and after the initial criticality. The signal to noise ratios of the fully inserted SRM's were greater than 2, and the minimum count rate was greater than 3 cps. The RSCS was demonstrated to be operable, and all test criteria were satisfied.

2.2.7 STI-10, IRM Performance

Overlap between the IRM's and SRM's was verified for all IRM's. All IRM's showed response to changes in the neutron flux. Overlap between the IRM's and APRM's remained to be performed at higher power levels.

2.2 Phase II - Open Vessel and Cold Testing (Continued)

2.2.8 STI-13, Process Computer

Checkouts of various signals and programs were performed on a continuing basis. However, most pertinent testing occurs later when significant power levels of greater than 15% are obtained. The criteria are not applicable to open vessel testing.

2.2.9 STI-17, System Expansion

During this phase of testing, base hanger and hydraulic shock and sway arrestor measurements were made. Instrumentation was installed and calibrated in preparation for subsequent heatup. Also, extensive visual inspections were made to detect and correct potential interferences. Criteria applicable to this cold condition were all met.

2.2.10 STI-35, Recirculation System Flow Calibration

The jet pump ΔP transmitters were calibrated as a loop using known pressures. Inputs and outputs of the electronics were observed, and adjustments were made as necessary to give proper response. Criteria are not applicable to this test.

2.3 Phase III - Initial Heatup

2.3.1 STI-1, Chemical and Radiochemical

Chemical tests of the primary coolant were made during the initial heatup. The results were:

Conductivity (umho/cm @ 25° C.)	0.53
Turbidity (FTU)	1.8
Chloride (ppb)	<50
Boron (ppb)	70
Silica (ppb)	102

Reactor water conductivity was within the 10 umho/cm maximum technical specification limit throughout initial heatup testing.

All test criteria were satisfied.

2.3.2 STI-2, Radiation Measurements

A complete plant survey was taken at hot standby and all criteria were met.

2.3 Phase III - Initial Heatup (Continued)

2.3.3 STI-5, Control Rod Drive System

Testing was performed at 600, 800, and 1,000 psig reactor pressures. All the control rod drives met the criteria of the tests performed on them during heatup testing.

2.3.4 STI-6, SRM Performance and Control Rod Sequence

As the reactor was heated to rated temperature, the rod pattern, SRM/IRM readings, moderator temperature, bypass valve positions, and approximate rate of change of the moderator temperature were recorded for each RSCS rod group withdrawn. Applicable test criteria satisfied.

2.3.5 STI-10, IRM Performance

The IRM preamplifiers were adjusted for continuity between ranges 6 and 7. Using the APRM readings, the IRM gains were adjusted so that 120/125 of scale on range 10 equals 24% power. Subsequent overlap with SRM's was verified. All criteria were met.

2.3.6 STI-12, APRM Calibration

The low power calibration of the APRM's was successfully completed and the APRM's were set to read greater than or equal to the actual core power as determined by the low power heat balance equations. All applicable test criteria were met.

2.3.7 STI-13, Process Computer

All computer signals were verified from the TIP system and OD-1 was operated to verify the software. Power levels were insufficient during heatup to perform pertinent testing. The criteria are not applicable at this power level.

2.3.8 STI-14, RCIC

Tests were performed during initial reactor pressurizations to 150, 800, and 1000 psig. All tests were performed with RCIC taking suction from and discharging to the condensate storage tank. All test criteria were satisfied with the exception of the level 2 criteria for high steam flow isolation setpoints. Excessive pressure drop across the elbow taps gives a higher than expected signal to the steam flow instrument switches. Therefore, these switches could not be set at the calculated 300% rated steam flow due to limited instrument range. The switches remain set at the present technical specification limit of <450 inches of water, pending resolution by TVA-DED, and GE. Controller settings were satisfactory for all Phase III testing.

2.3 Phase III - Initial Heatup (Continued)

2.3.9 STI-15, HPCI

Tests were performed during initial reactor pressurizations to 150, 800, and 1000 psig. All tests were performed with HPCI taking suction from and discharging to the condensate storage tank. All test criteria were met and controller settings were satisfactory for Phase III testing.

2.3.10 STI-16, Selected Process Temperature

Data were obtained that showed the drain line thermocouple adequately monitors bottom drain line temperature. Criteria for temperature differences between the upper and lower regions were met on pump starts.

2.3.11 STI-17, System Expansion

Linear voltage differential transmitters and recorders were installed to determine the movements of the main steam lines, recirculation lines, and feedwater lines, in order to verify the freedom for expansion of the various pipes and associated suspension components. These measurements were made during heatup of the unit to rated temperature and pressure conditions and also during cooldown. Movement was recorded in X and Z directions on the main steam and feedwater lines and in the X, Y, and Z directions on the recirculation lines. These recordings were compared with predicted movements for the various pipe lines. These records were coupled with a visual inspection of the systems and readings of selected hanger and hydraulic shock and sway arrestor positions during heatup.

All restrictions of movement for the various pipings were resolved.

2.3.12 STI-25, Main Steam Isolation Valves

The performance of this test at conditions typical of the heatup phase is merely to demonstrate the operability of the MSIV's. The only applicable criteria at this test condition is that all MSIV's close within the 3-5 second limit. All eight valves satisfactorily met the closure time criteria.

2.3.13 STI-26, Relief Valve Actuation

Manual actuation of all valves was performed at a reactor pressure of 250 psig. Valves functioned as expected with 2 exceptions. Valve 1-18 had a failed tail pipe thermocouple and 1-23 tail pipe failed by 33° F. to return to within 10° F. of its initial temperature. When

2.3 Phase III - Initial Heatup (Continued)

2.3.13 (Continued)

retested, valve 1-18 passed and valve 1-23 failed by 3° F. to return to within 10° F. of its initial temperature. Valve 1-23 successfully passed reseating criteria during phase IV testing.

2.3.14 STI-70, Reactor Water Cleanup System

Three tests were performed to demonstrate the heat capacities of the regenerative and non-regenerative heat exchangers. The first test was conducted in the "Hot Standby" mode in which all cleanup flow was returned to the reactor with no bypass flow. With a cleanup flow of 0.132×10^6 lb/hr., a heat removal rate of 16.8×10^6 Btu/hr. was obtained. This compares well with the design figures of 0.14×10^6 lb/hr and 15.8×10^6 Btu/hr.

The second test was run in the "Normal" mode in which all cleanup flow was returned to the reactor with no bypass flow. With a cleanup flow of 0.13×10^6 lbs/hr., a heat removal rate of 17.2×10^6 Btu/hr was obtained. This compares well with the design figures of 0.14×10^6 lb/hr and 15.8×10^6 Btu/hr.

The third test was run in the "Blowdown" mode in which all cleanup flow was discharged to radwaste or the condenser. With a cleanup flow rate of 0.052×10^6 lb/hr., a heat removal rate of 21.9×10^6 Btu/hr. was obtained. This compares well with the design figures of 0.053×10^6 lb/hr and 22.5×10^6 Btu/hr.

During all three tests, the cleanup filter inlet temperature was held below 130° F., the NPSH was determined to be 37 feet and the inlet and outlet temperature of the cooling water supplied to the NRHX's was held within limits, thus satisfying all criteria.

2.3.15 STI-71, Residual Heat Removal System

The residual heat removal system was operated in the "suppression pool cooling" mode and shown to have sufficient heat removal capacity to satisfy design conditions. All four of the RHR heat exchangers met all criteria (107×10^6 Btu/hr.).

2.3.16 STI-72, Drywell Atmospheric Cooling System

Drywell temperatures were monitored at each plateau during initial heatup. All conditions in the drywell met criteria with the exception of two points at the top of the sacrificial shield. The heat removal capability of the drywell coolers met criteria at rated temperature and pressure.

2.3 Phase III - Initial Heatup (Continued)

2.3.17 STI-73, Cooling Water System

The reactor building closed cooling water system was balanced to near design conditions. The heat load on the main RBCCW heat exchangers was within the maximum design specifications. All criteria associated with this test were met.

2.4 Phase IV - Power Operation of 10%-100% Rated Output

2.4.1 STI-1, Chemical and Radiochemical

Throughout the startup test program, chemical and radiochemical sampling and analyses were performed on a routine and special test basis. Routine surveillance of the reactor water, condensate and feedwater, embraced the measurement of conductivity, chloride content, turbidity and boron content. From the point at which sufficiently high steaming rates were achieved, sample testing was done in order to assess the radiolytic gas content in steam, gaseous activity leaving the air ejectors and the performance of the off-gas system.

Testing of steam separator and dryer performance at Browns Ferry 2 consisted of two (@ 50% and 100% power plateaus) injections of sodium sulphate into the reactor water to increase the sensitivity of the Na-24 carryover measurements with the reactor cleanup system out of service. Reactor water conductivity exceeded 2.0 umho/cm @ 25° C. for 6 hours on December 4, 1974, at 60% testing plateau due to the reactor cleanup system being bypassed during the performance of the "no cleanup" test.

The levels of iodines, silica, insolubles and boron were within established limits during the startup testing. Gamma scans of primary water disclosed the expected corrosion and activation products.

All test criteria were satisfied.

2.4.2 STI-2, Radiation Measurement

At 25% and 50% power "Complete Surveys" were conducted with all locations but one within the criteria. The location which caused concern is marked accordingly to prevent excessive exposures while shielding and access control measures are being completed. A "Limited Survey" was performed at test condition 3E (81% power) with all locations except the one previously mentioned meeting the test acceptance criteria.

2.4 Phase IV - Power Operation of 10%-100% Rated Output (Continued)

2.4.2 STI-2, Radiation Measurement (Continued)

Several areas of higher than permissible background dose rate were defined and marked accordingly at this and the previous test condition (50% power). The "Complete Survey" was conducted at the test condition 4E (96% power) with all locations meeting the acceptance criteria except two. One of these was the previously mentioned location and the new one being a similar previously unnoticed location. The dose rates for these two locations were 550 mrem/hr. and 520 mrem/hr. Typical dose rates for this condition were less than 1 mrem/hr. gamma with a few locations exceeding this rate. One other location which approached the test criteria level was placed under frequent surveillance.

Currently, special precautions are in effect at the problem location to prevent inadvertent personnel overexposure. A proposed amendment to the technical specifications is being considered by the NRC to alleviate the existing barrier prob. as required to meet 10CFR20 requirements for the relatively inaccessible locations mentioned which did not meet the test criteria.

2.4.3 STI-5, Control Rod Drive System

Scram times of the four slowest in-sequence rods were measured during planned turbogenerator trips at 100% power. All four in-sequence rods performed in accordance with the applicable acceptance criteria.

2.4.4 STI-6, SRM Performance and Control Rod Sequence

Power was increased to 25% rated in sequence "A". The RSCS was tested at 10%, 20%, and 25% by attempting to select and move out-of-sequence groups, and was shown to be operating properly.

Later the RSCS sequence "B" was utilized to increase power, and the RSCS was found to be operable at the same powers as above.

The RSCS is not required to be operable above 20% power. At approximately 30% thermal power pressure switches at the 1st stage turbine will automatically bypass the RSCS logic.

During the startups in both sequences, the operation of the core was closely observed for irregularities or reactivity anomolies as a result of the rod sequences. Both performed satisfactorily.

STI-6 demonstrated that power could be raised with rod withdrawal in a safe and orderly fashion. All test criteria were met.

2.4 Phase IV - Power Operation of 10%-100% Rated Output (Continued)

2.4.5 STI-9, Water Level Measurements

Calibrations of the Yarway and GEMAC water level instrumentations were verified to asceratin accurate reactor water level indications at all times. Data were also recorded at the 50% and 100% test conditions as reactor water level was varied in 6-inch increments between the high and low level trip points, to obtain knowledge of the tracking performance of those level systems. Adjustments to calibrations were made as seen necessary. There are no criteria associated with this test.

2.4.6 STI-10, IRM Performance

At 16% power the IRM's were adjusted such that a reading of 120/125 of full scale on range 10 was equal to or less than 30% power as indicated by the APRM's. A second calibration was required at 19.5% power to calibrate IRM's G and H which were inoperative during the previous calibration. SRM/IRM overlap was verified on a subsequent startup. All criteria were satisfactorily met.

2.4.7 STI-11, LPRM Calibration

Using the process computer, calibration of the LPRM system was performed at the 25, 60, 80, and 100% power levels. All operable LPRM's were adjusted to read proportional to the neutron flux in the narrow-narrow water gap at the height of the chamber which satisfied required test criteria.

2.4.8 STI-12, APRM Calibration

At each major test condition, the APRM's were calibrated to read equal to or greater than the core thermal power. The calibration was repeated after each LPRM calibration. The ability of the APRM's to maintain sufficient accuracy over large power changes was also verified. A power scram clamp was set 20% over the highest load line in each test condition before ascending to that condition. All test criteria were satisfied.

2.4.9 STI-13, Process Computer

The process computer and oscillary equipment performed well during the startup phase of the test program. Some minor problems were encountered in both the SLA and NSSS programs. The dynamic system test case (DSTC) was performed and all system programs were checked out. All

2.4 Phase IV - Power Operation of 10%-100% Rated Output (Continued)

2.4.9 STI-13, Process Computer (Continued)

test criteria were met. DSTC results have been sent to GE in San Jose for a detailed evaluation. OD-1 and P1 programs showed good agreement with off-line computer computations.

2.4.10 STI-14, RCIC System

The reactor core isolation cooling system was tested in the injection mode at 22% power and 48% flow. Transient response was satisfactory and test criteria were met except for the high steam flow isolation setpoints, which remain conservatively set.

2.4.11 STI-15, HPCI System

Prior to this test the proportional band on the HPCI controller was increased from 600% to 2000%. R/M was left at 100. Full flow was reached in 23.5 seconds. The HPCI turbine did not trip during the test and the turbine gland seal condenser was capable of preventing steam leakage to the atmosphere, thus satisfying all criteria.

2.4.12 STI-16, Selected Process Temperatures

The observation of select process temperatures was conducted at three operating conditions, i.e., at the lower end of the 50, 75, and 100% load lines, respectively. All test criteria were adequately met at each test conditon of interest.

2.4.13 STI-17, System Expansion

Feedwater lines were continuously monitored with LVDT instrumentation to determine if the thermal movement was satisfactory. These systems satisfied all applicable test criteria.

2.4.14 STI-18, Core Power Distribution

Two TIP reproducibility tests were performed on all machines, one at test condition 1, the other at test condition 2R. The results of both tests were within the established criteria and TIP reproducibility was satisfactorily verified.

The core power distribution was determined at several power levels using TIP data and the off-line computer.

FINAL SUMMARY REPORT - BFNP UNIT 2

2.4 Phase IV - Power Operation of 10%-100% Rated Output (Continued)

2.4.15 STI-19, Core Performance

The significant core performance parameters such as maximum fraction or limiting power density (MFLPD), minimum critical heat flux ratio (MCHFR), core thermal power, minimum bundle critical power ratio (MCPR), maximum average planar linear heat generation rate (MAPLHGR) and maximum linear heat generation rate (kW/ft) were monitored throughout the test program at each of the operating plateaus.

Process computer calculations were in close agreement with manual and off-line computer calculations.

At each test condition, the reactor response to rod movements was stable and well damped. All test criteria were met.

2.4.16 STI-20, Electrical Output and Preliminary Heat Rate

The 300-hour gross electrical output warranty demonstration was conducted over the 318-hour interval from '700 hours on February 23, until 0100 hours on March 9, 1975. Data from a 17.83-hour period of reduced power operation was excluded from the test analysis. In all, 302 readings at one-hour intervals were collected from the plant process computer or by direct observation of plant and special test instruments. Except during one inconsequential power transient, all test criteria were satisfied.

Generator Output

Generator terminal output was determined from the unit kWh meter and corrected to rated conditions of condenser back-pressure, generator losses, and generator power factor. Precision test instruments were used in all cases (with one exception: the generator kWh meter was accepted as the test standard) to provide correction factors for station instruments actually used throughout the test.

Core Thermal Output

Core thermal output was determined by process computer calculations (OD-3).

2.4.17 STI-21, Flux Response to Rods

The stability of the core local reactivity feedback mechanism was verified for small perturbations in reactivity due to rod movements at several points during the startup test program. All test criteria were met.

2.4 Phase IV - Power Operation of 10%-100% Rated Output (Continued)

2.4.18 STI-22, Pressure Regulator Setpoint Changes

The following tests were performed on the EHC system to verify the pressure regulator performance with the recirculation flow controllers set in manual mode at each of the test conditions presented in Table 2.2. At each test condition a transient record was made of pertinent reactor process variables.

1) Positive and negative 10 psi set point changes using first one, then the other pressure regulator as the primary regulator controlling pressure in the following manner:

a) With load limiter set high enough so the entire transient was handled by control valves.

b) With the load limiter set so that both the control valves and bypass valves acted during the transient.

c) With the load limiter set low enough that the entire transient was handled by the bypass valves.

2) The regulator acting as primary was "failed" to allow the back-up regulator to take over control. This was performed using first one, then the other as the back-up regulator.

As a result of information obtained from unit 1 testing of STI-22, the work on unit 2 was greatly simplified. A notch filter was added to the EHC circuitry prior to startup which enabled pressure regulator optimization during initial testing. All test criteria were satisfactorily met at all test conditions.

2.4.19 STI-23, Feedwater System

Two types of tests were performed on the feedwater system:

1) Level setpoint changes of ±3 to ±5 inches were made in both 3-element and single-element control in the Master Manual flow control mode at various test conditions.

2) A feedwater pump trip was performed at test condition 42.

For each test a transient record of relevant process variables was made.

The level setpoint changes resulted in minor transients with a slight oscillatory behavior noted in only a few cases. No reactor scrams could be directly attributed to feedwater control system transient response. The level setpoint changes for each test condition satisfied

2.4 Phase IV - Power Operation of 10%-100% Rated Output (Continued)

2.4.19 STI-23, Feedwater System (Continued)

the applicable test criteria.

The feewater pump trip from test condition 4X produced a smooth transient with excellent control of reactor water level. The trip resulted in a recirculation pump runback which controlled reactor power such that the two remaining feedwater pumps could maintain the proper water level thus preventing a low water level scram.

2.4.20 STI-24, Bypass Valves

Testing has been completed on the bypass valves through 100%. The test results establish without question the bypass valves can be tested at any power level in the Master Manual flow control mode. Flux spikes were less than 2% and pressure spikes less than 3 psi at all test conditions. The bypass valves performed as designed. All level 1 and 2 criteria have been met for all test conditions.

2.4.21 STI-25, Main Steam Isolation Valves

All MSIV's were tested by individual full closure and the closing time measured. All valves were either within the criteria or adjusted to meet the criteria. Pressure transients during single valve closures were small.

During functional testing, each valve was closed 10% (90% open) to check operation. Transients were not detectable during this test phase.

An MSIV full isolation occurred at 98% power and test criteria were met.

2.4.22 STI-26, Relief Valves

All valves met timing, capacity and reseating criteria for this test. The total measured capacity for all valves was 8.7×10^6 lb/hr. The slowest delay time was 0.32 seconds and all tailpipe temperatures returned to within 10° F. of their initial temperature.

2.4.23 STI-27, Turbine Stop and Control Valve Trips

Fast closure of the main turbine stop valves was demonstrated at 100% of rated reactor conditions. Fast closure of the main turbine control valves was demonstrated at 25% and 100% of rated reactor conditions. The level 2 criteria which requires that the feedwater level

2.4 Phase IV - Power Operation of 10%-100% Rated Output (Continued)

2.4.23 STI-27, Turbine Stop and Control Valve Trips (Continued)

controller prevent a low water level isolation was not met for either of the 100% power trips. All other test criteria were met for each test.

2.4.24 STI-30, Recirculation System

Pump Trips

Recirculation pump trips were performed at 50%, 75%, and 100% of rated reactor conditions, including both one and two pump trips. Transient MCPR calculation resulted in values which met the criteria satisfactorily. In addition, transient minimum critical power ratio (MCPR) calculations were done for pump trips that required a transient analysis. All trips showed adequate margin to the 1.05 MCPR limit. The two pump trip from 100% power was the most limiting of all tests.

Recirculation System Performance

Performance data was taken at various power levels during this phase of testing. The recirculation system performed satisfactorily at all levels of power and flow.

Non-Cavitation Search

Verification that cavitation did not occur in the recirculation system was performed by inserting control rods until the recirculation pump runback was encountered. Initially the runback was encountered at approximately 26% power. The runback setpoint was reset and the test was again performed. During this test the runback was encountered at approximately 22.5% power. No indications of cavitation were observed during either test.

2.4.25 STI-31, Loss of Turbine-Generator and Off-Site Power

This test was performed with unit two operating at 31% power. Except for having the unit 2 auxiliaries aligned so that unit 1 and the plant common electrical supplies could not feed them, the plant was in normal operation.

The sequence of major operational events is as follows:

Time	Event
0.00 sec	Turbine-generator trip manually initiated
0.10 sec	Control valves closing

2.4 Phase IV - Power Operation of 10%-100% Rated Output (Continued)

2.4.25 STI-31, Loss of Turbine-Generator and Off-Site Power (Continued)

Time	Event
0.10 sec	Stop valves closing Reactor automatic scram 500kV main breakers trip 4-kV unit board breakers trip
5.30 sec	4-kV shutdown board C feeder breaker trip
7.10 sec	Diesel generator C supplying power to unit 2 shutdown auxiliaries
31.15 sec	Full reactor isolation
6 minutes	RCIC manually initiated
8 minutes	Automatic lifting of 3 relief valves

All reactor parameters remained within their expected limits during the transient and all automatic electrical switching was normal.

2.4.26 STI-33, Main Turbine Stop Valve Surveillance Test

Turbine stop valves were individually closed at 25%, 60%, 75%, and 99% power levels. No significant perturbation in the observed operating parameters were noted as a result of the valve closures. All test criteria were met.

2.4.27 STI-34, Vibration Measurements

Vibration data were obtained in conjunction with the recirculation pump trips (STI-30) at 50%, 75%, and 100% power levels. The requirements for data with the equalizer valve open was deleted due to the removal of the cross-tie bypass line.

Final evaluation by a qualified specialist will be made at a later date.

2.4.28 STI-72, Drywell Atmospheric Cooling System

Drywell temperatures were monitored at 100% power. All temperatures in the drywell met test criteria with the exception of 2 points at the top of the sacrificial shield. The drywell heat load was close to the design value. Although criteria were met, with the two exceptions, the RBCCW inlet temperature was lower than its rated maximum value of 105° F. DED will determine if further testing is required and/or if the temperatures are acceptable. Initial evaluation indicated that the temperatures were acceptable.

2.4 Phase IV - Power Operation of 10%-100% Rated Output (Continued)

2.4.29 STI-73, Cooling Water Systems

The heat load on the RBCCW heat exchangers was within 2% of design maximum heat load at approximately 100% reactor power. Due to cold river water, the raw cooling water flow was extremely low and it was not possible to evaluate RBCCW heat exchanger performance at rated maximum flow and temperature. EED will determine if the results are satisfactory.

3.0 Results

3.1 Phase II - Open Vessel and Cold Testing

3.1.1 STI-3, Fuel Loading

Purpose

The purpose of this test is to load fuel safely and efficiently to the full core size.

Criteria

Level 1

The partially loaded core must be subcritical by at least 0.38% ΔK with the geometrically strongest rod fully withdrawn.

Analysis

Fuel loading began on June 29, 1974, with the loading of the operational sources and was successfully completed on July 19, 1974. At that time all 7 operational sources were installed, all 4 SRM's were connected and functional, all 764 fuel assemblies were installed and the core verification completed. Partial core shutdown margins at designated steps in the loading procedure were successfully demonstrated during the loading, satisfying the criteria.

Preparation for fuel loading began by placing the fuel loading chambers in dunking chambers, which were then installed in dummy blade guides. The midpoint of all detectors was at 2/3 core height so that the chambers would be near the axial peak flux. These FLC's were hooked to the plant SRM electronics and the signal to noise ratio was checked. The setpoints for the rod block alarm and scram actuation of the FLC's were determined by establishing their saturation count rate and setting the scram point at 67% of the saturation level; thus the requirement that the instrument electronics do not saturate at a level not less than 150% of the trip point was satisfactorily met. In addition, the rod block alarm was set at 1/2 decade below the scram point. When the SRM's were placed in service, the scram setpoints were not re-established since no method existed to saturate the SRM's, but rather they were set at 5×10^4 cps for scram actuation and 1×10^4 cps rod block alarm as described in the startup test instructions.

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.1 Phase II - Open Vessel and Cold Testing (Continued)

3.1.1 STI-3, Fuel Loading (Continued)

Analysis (Continued)

Fuel loading for unit 2 was performed with the Sb-Be operational sources located in the core. With nearly 9000 curies total source strength at initiation of fuel loading, the FLC's were positioned a significant distance from the first fuel bundle location in order to avoid a scram. As the geometry of the loading pattern encompassed the FLC's, they were moved to appropriate positions again to reduce the possibility of a scram. Fuel loading proceeded from a symmetrical pattern about the center source through a spiral configuration, forming a "pin-wheel" cluster centered around the central control rod. The FLC's were employed through 420 assemblies loaded, then the SRM in-core detectors were utilized to the completion of loading.

Safe loading was accomplished by making subcritical and functional checks before and after loading the control cells (2 x 2 fuel assembly size). In addition, frequent shutdown margin checks made at various core sizes demonstrated that the core was subcritical at all times by at least 0.38% ΔK with the geometrically strongest rod fully withdrawn. This was done by fully withdrawing the strongest rod and withdrawing an adjacent control rod to notch 14 and verifying subcriticality. Inverse multiplication plots were maintained from FLC readings taken with all rods inserted to predict subcriticality before loading additional fuel assemblies. In certain cases, such as where a fuel assembly was loaded nearby an operational source or an FLC, because of geometric effects, special interpretation of these plots were required to predict safe loading of the next fuel assembly. These geometric effects were expected.

The fully loaded core was verified on July 19, 1974, for proper seating and orientation of fuel assemblies and for fuel bundle serial numbers and core locations (see figure STI 3-1). Serial numbers were checked for proper selection of low enrichment and high enrichment fuel per figure STI 3-2.

All steps in STI-3, Fuel Loading were satisfactorily completed and the test program proceeded to the full core shutdown margin test (STI-4) as scheduled. All criteria were met.

STI-1 SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.1 Phase II - Open Vessel and Cold Testing (Continued)

3.1.1 STI-3, Fuel Loading (Continued)

Analysis (Continued)

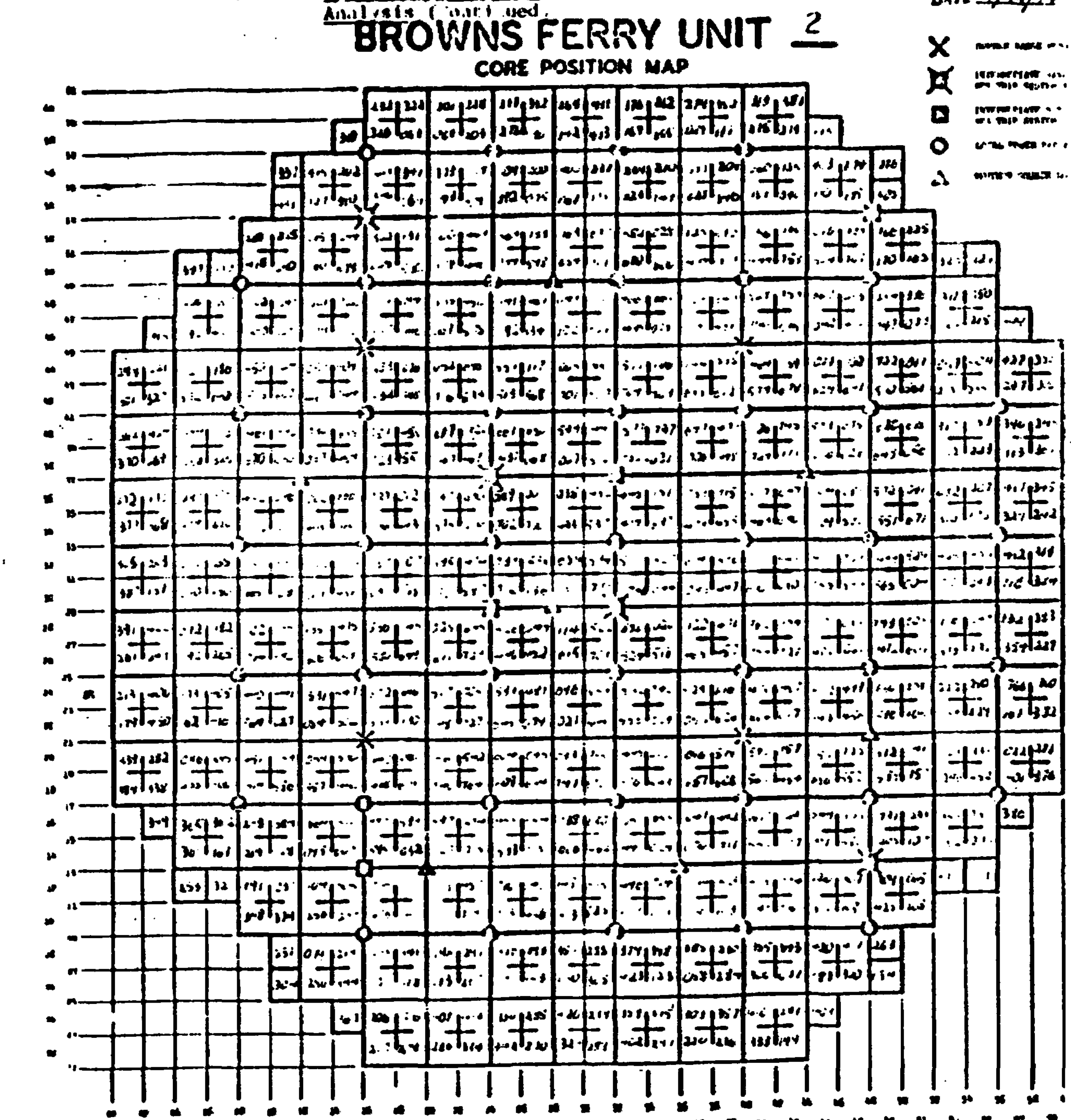


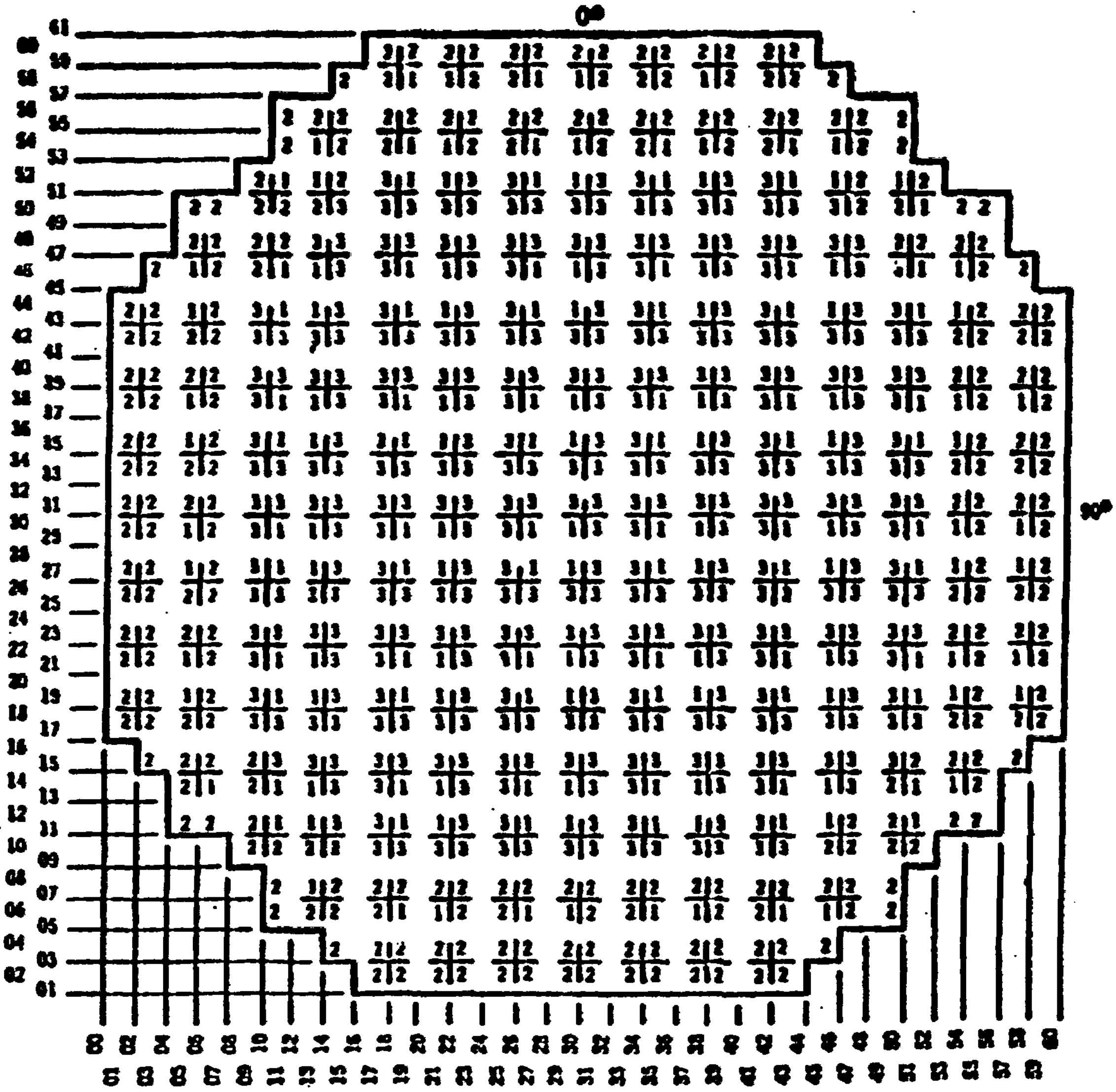
Figure STI 3-1

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.1 Phase II - Open Vessel and Cold Testing (Continued)

3.1.1 STI-3, Fuel Loading (Continued)



NUMBER OF FUEL ASSEMBLIES – 764
NUMBER OF CONTROL RODS – 185
1 LOW (1.1) ENRICHED ASSEMBLIES – 168
2 HIGH (2.5) ENRICHED ASSEMBLIES ON PERIPHERY – 263 (4 Gd_2O_3 RODS)
3 HIGH (2.5) ENRICHED ASSEMBLIES – 333 (5 Gd_2O_3 RODS)

Figure STI 3-2

Fuel Assembly Locations

3.0 Results (Continued)

3.1 Phase II - Open Vessel and Cold Testing (Continued)

3.1.2 STI-4, Core Shutdown Margin

Purpose

The purpose of this test is to demonstrate that the reactor will be subcritical throughout the first fuel cycle with any single control rod fully withdrawn.

Criteria

Level 1

The shutdown margin of the fully loaded core must be at least 0.38% ΔK/K at the most reactive time in the fuel cycle.

Level 2

The reactor shall have an "all rods in" k_{eff} = 0.933 ± .010

Analysis

The shutdown margin test was conducted for the fully loaded core. The analytically strongest rod in the core, 26-07, was fully withdrawn. Next, 22-03 was notched to position 14. Subcriticality at this point was sufficient to guarantee a shutdown margin of at least 0.38% ΔK/K.

The clump critical was performed by pulling rods in a prescribed sequence. Table STI 4-1 shows the sequence and the analytic worth of each rod. With the moderator at 90° F., the reactor went critical on the 14th notch of the 11th rod. In order to obtain a better priod measurement (see figures STI 4-1 and STI 4-2), it was notched one position farther. The total worth of the withdrawn rods was 6.7522% ΔK/K. Using the period measurement from the critical, it was determined that the core had an "all rods in" k_{eff} of 0.933 ± .001.

All test criteria for STI-4 were satisfied.

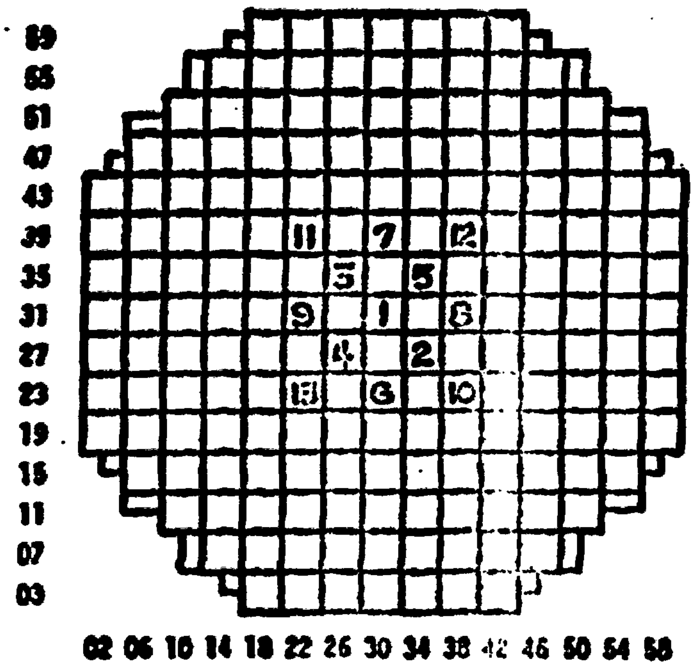


Table STI 4-1

Differential Reactivity (%Δk/k) of Core Configuration From All Rods In to Rod 13 (22-23) Withdrawn

Rods Out	%Δk/k	Rods Out	%Δk/k
1	1.76	8	6.30
2	3.25	9	6.63
3	3.41	10	6.74
4	4.32	11	6.85
5	5.23	12	6.95
6	5.60	13	7.04
7	5.96		

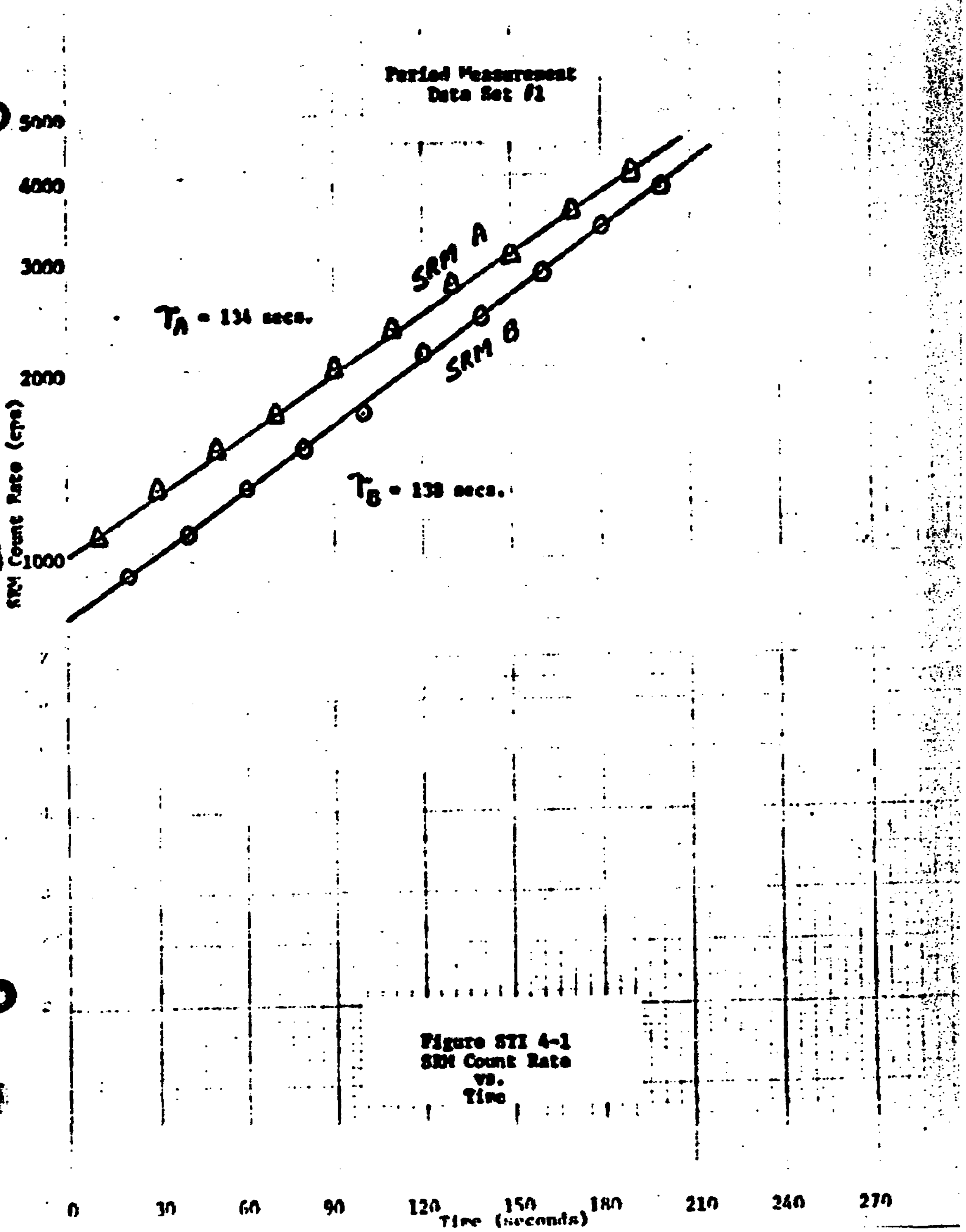


Figure STI 4-1
SRM Count Rate
vs.
Time

FINAL SUMMARY REPORT - RFSP UNIT 2

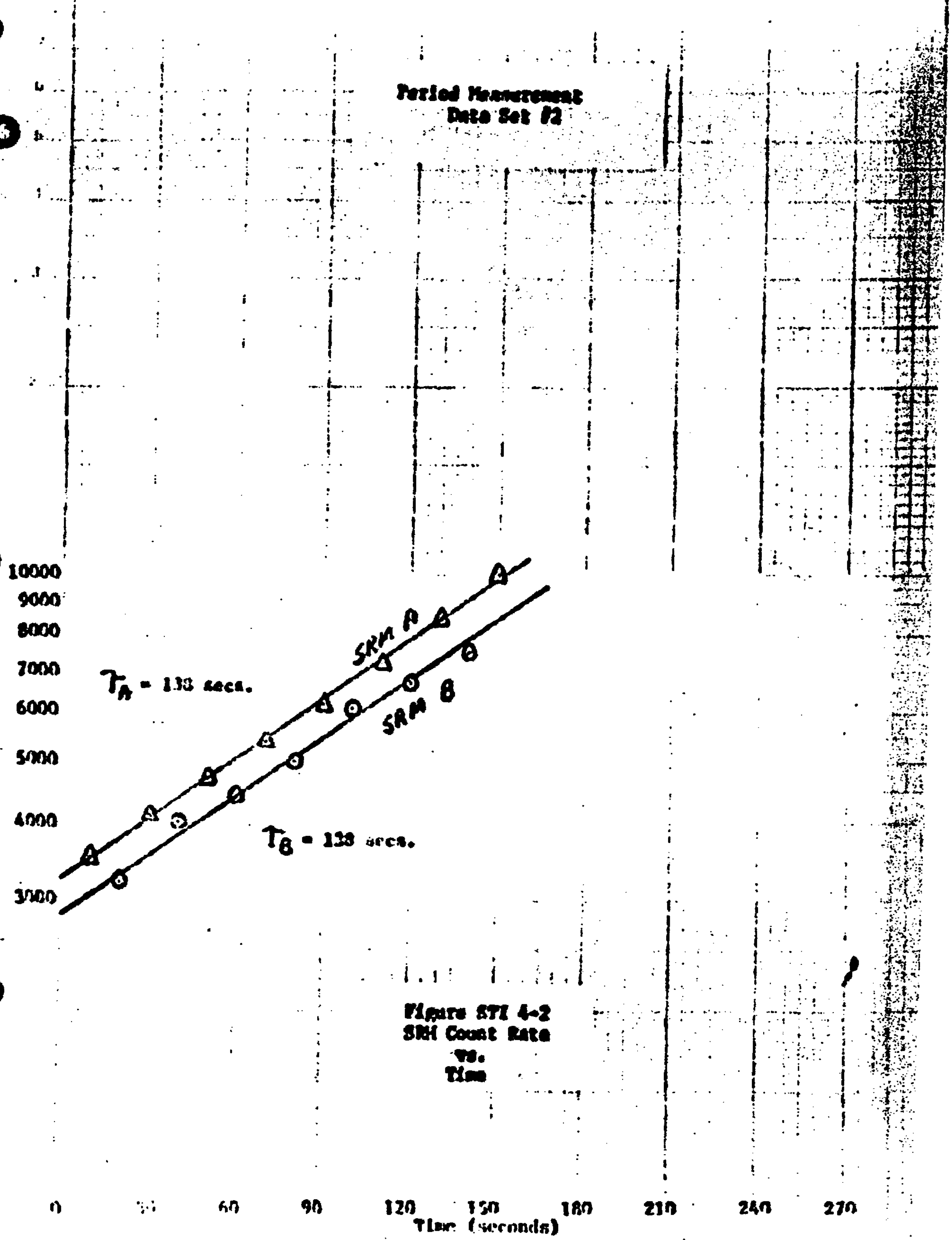


Figure STI 4-2
SRM Count Rate
vs.
Time

3.0 Results (Continued)

3.1 Phase II - Open Vessel and Cold Testing (Continued)

3.1.3 STI-5, Control Rod Drives

Purpose

The purposes of the Control Rod Drive System test are: (a) to demonstrate that the Control Rod Drive (CRD) system operates properly over the full range of primary coolant temperatures and pressures from ambient to operating, and particularly that thermal expansion of core components does not bind or significantly slow control rod movements; and (b) to determine the initial operating characteristics of the entire CRD system.

Criteria

Level 1

(a) Each drive speed in either direction (insert or withdraw) must be 3.0 ± 0.6 in per sec., indicated by a full 12-ft. stroke in 40 to 60 secs.

(b) The average scram insertion time of all operable control rods, based on the deenergization of the scram pilot valve solenoids as time zero, shall be no greater than:

% Inserted from Fully Withdrawn	Average Scram Insertion Times (sec.)
5	0.375
20	0.90
50	2.0
90	5.0

(c) The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two-by-two array shall be no greater than:

% Inserted from Fully Withdrawn	Average Scram Insertion Times (sec.)
5	0.398
20	0.954
50	2.120
90	5.3

3.0 Results (Continued)

3.1 Phase II - Open Vessel and Cold Testing (Continued)

3.1.3 STI-5, Control Rod Drives (Continued)

Criteria (Continued)

Level 1 (Continued)

(d) The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

Level 2

(a) With respect to the control rod drive friction tests, if the differential pressure variation exceeds 15 psid for a continuous drive-in, a settling test must be performed, in which case, the differential settling pressure should not be less than 30 psid, nor should it vary by more than 10 psid over a full stroke. Lower differential pressures in the settling tests are indicative of excessive friction.

(b) Scram times with normal accumulator charge should fall within prescribed time limits.

Analysis

All the control rods met the requirements of the tests performed on them during zero-reactor-pressure testing. Position indications, rod timing, stall flows, coupling checks, and friction tests were performed twice on each CRD: during and following fuel loading. The results reported here are those of the latter testing period.

Position-Indicating Check

The rod position information system was extensively checked and was operating properly.

Rod Timing and Stall Flows

The normal rod withdrawal and insert times, together with the stall flows were measured. Some of the drives were adjusted so that their times were within the above criteria.

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.1 Phase II - Open Vessel and Cold Testing (Continued)

3.1.3 STI-5, Control Rod Drives (Continued)

Analysis (Continued)

Coupling Check

This check was performed during fuel loading whenever a rod was fully withdrawn to position 48. All rods were coupled to their drives.

Friction Testing

All of the CRD's were friction tested by continuously inserting them from position 48 to position 0 and photographing the insertion pressure throughout the insert process.

The friction test data were acquired using a strain gauge differential pressure cell and a storage oscilloscope. Polaroid photographs of the oscilloscope traces were taken to record the data.

All control rods passed the continuous insertion $\Delta P_{max.}$ $-\Delta P_{min.}$ criteria.

Scram Testing

During open vessel testing all control rods were individually scram tested. The average scram times fell well within the level requirements. (See table STI 5-8)

From these data the four slowest in sequence control rod drives were chosen to be scrammed three times each with minimum accumulator pressure. The mean scram times to 90% insertion were found to fall within the limits set by Figure STI 5-1. Table STI 5-1 gives the scram times for the slowest drives with normal and minimum scram accumulator pressures.

3.0 Results (Continued)

3.1 Phase II - Open Vessel and Cold Testing (Continued)

3.1.3 STI-5, Control Rod Drives (Continued)

Analysis (Continued)

Scram Testing (Continued)

Table STI 5-1

Four Slowest Insequence Control Rod Drives at Zero Reactor Pressure and Minimum** and Normal Accumulator Pressure

Rod Location	Mean* 90% Scram Time Min (Sec)	90% Scram Time Norm (Sec)
34-07	1.894	1.840
10-23	1.977	1.914
46-11	1.811	1.776
30-27	1.879	1.720

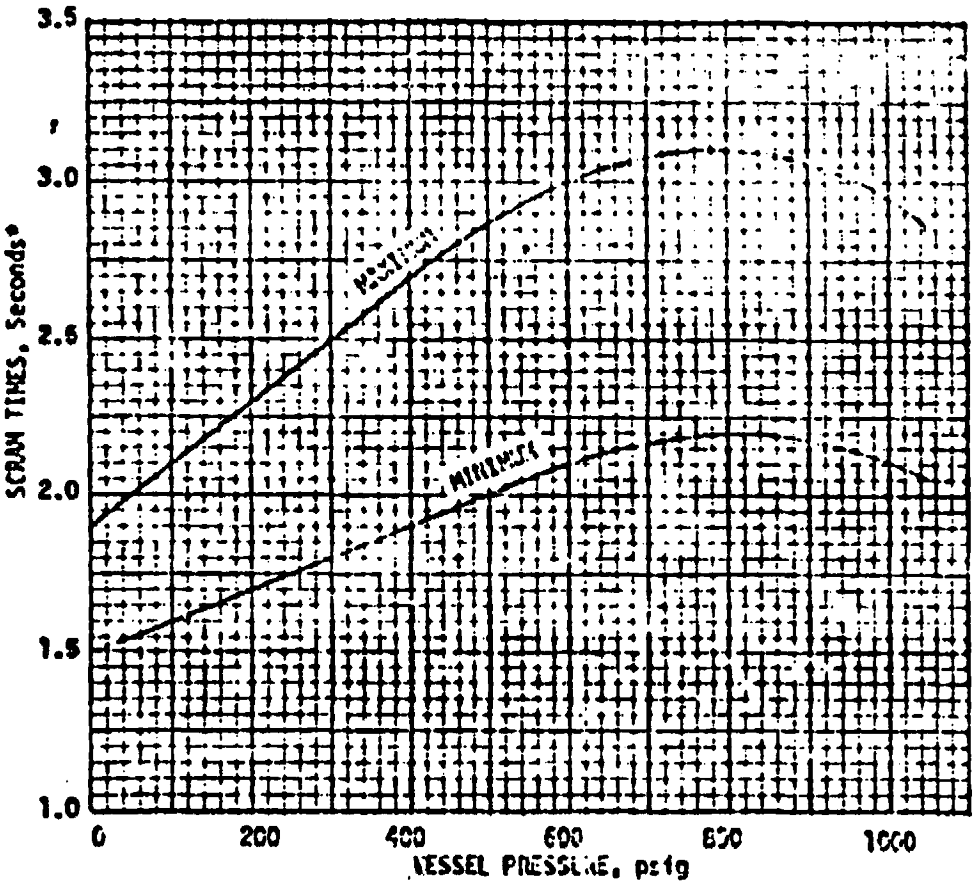
*Mean of three scrams
**Accumulator pressure at which low pressure alarm initiates.

All scram times were well within the criteria. The rod times were taken at the scram test panel in the auxiliary instrument room using a two channel Brush recorder. On one channel the recorder sensed the pickup and dropout of the reed switches in the RPIS (rod position information system) probe. The other channel recorded the deenergization of the scram pilot valve solenoids. The data were analyzed using a program written for a Wang programmable calculator.

3.0 Results (Continued)

3.1 Phase II - Open Vessel and Cold Testing (Continued)

3.1.3 STI-5, Control Rod Drives (Continued)



SYSTEM OPERATING CONDITIONS:

1. Accumulator precharge 555/535 psig at 70°F ($39.9/41.2\ kg/cm^2$ At 20°C)
2. Accumulator water side 1510psig, ($106.3\ kg/cm^2$) max. 1390psig, ($97.7\ kg/cm^2$) min.
3. Scram valve air pressure 70/75 psig. ($4.9/5.30\ kg/cm^2$)

Data applicable to single CRD scrams with charging valve closed (V-113) or full reactor scram with charging valve open.

* Scram time is the time from loss of voltage to scram air pilot valves to 90% insertion (pickup of "04").

SCRAM PERFORMANCE CURVE FOR MODEL 7RDB144A2 and 7RDB144B1 CRDs

Figure STI 5-1

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.1 Phase II - Open Vessel and Cold Testing (Continued)

3.1.4 STI-6, SRM Performance and Control Rod Sequence

Purpose

The purpose of this test is to demonstrate that the operational sources, SRM instrumentation, and rod withdrawal sequences provide adequate information to achieve criticality and to increase power in a safe and efficient manner. The effect of typical rod movements on reactor power will be determined.

Criteria

Level 1

(a) There must be a neutron signal-to-noise ratio of at least 2:1 on the required operable SRM's or fuel loading chambers.

(b) There must be a minimum count rate of 3 cps on the required operable SRM's or fuel loading chambers.

(c) The IRM's must be on scale before the SRM's exceed the rod block set point.

(d) The RSCS shall be operable as specified in the technical specifications.

Analysis

The operational sources were loaded in a manner consistent with STI-3, Fuel Loading. Source locations are shown in figure STI 6-1.

Before the SRM's were inserted into the core, their count rates were observed to determine their background readings. After fully driving the SRM's into the core, their count rates were again recorded to insure that the signal-to-noise criterion was met. This data is contained in table STI 6-1, and the discriminator and high voltage settings for the SRM units are in table STI 6-2.

The RSCS was demonstrated to operate correctly by the inability to select out-of-sequence rods.

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.1 Phase II - Open Vessel and Cold Testing (Continued)

3.1.4 STI-6, SRM Performance and Control Rod Sequence (Continued)

Analysis (Continued)

The reactor was brought to criticality in sequence "A" on the 10th notch of the 53rd rod. The moderator temperature was 92° F.

After SRM/IRM overlap was verified by STI-10, the SRM's and IRM's were removed from the non-coincident scram mode, and the SRM high level blocks set at their normal point of 1×10^5 cps. It was also shown that the SRM's were capable of monitoring 7.5×10^5 cps without saturating.

All test criteria were satisfied.

Table STI 6-1

	SRM Count Rate (cps)			
	A	B	C	D
SRM Fully Inserted	5	4	15.0	8
SRM Full Retracted	<0.1	0.1	0.2	0.1
Signal-to-Noise Ratio	49	39	74.9	79

Table STI 6-2

Parameter	SRM A	SRM B	SRM C	SRM D
Hi Hi Trip	5×10^5 cps	5×10^5 cps	5×10^5 cps	5×10^5 cps
Hi Alarm	1×10^5 cps	1×10^5 cps	1×10^5 cps	1×10^5 cps
Inop. Voltage	360 vdc	375 vdc	325 vdc	355 vdc
High Voltage	386 vdc	403 vdc	350 vdc	379 vdc
Discriminator	7 turns	7 turns	5 1/2 turns	5 1/2 turns

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.1 Phase II - Open Vessel and Cold Testing (Continued)

3.1.4 STI-6, SRM Performance and Control Rod Sequence (Continued)

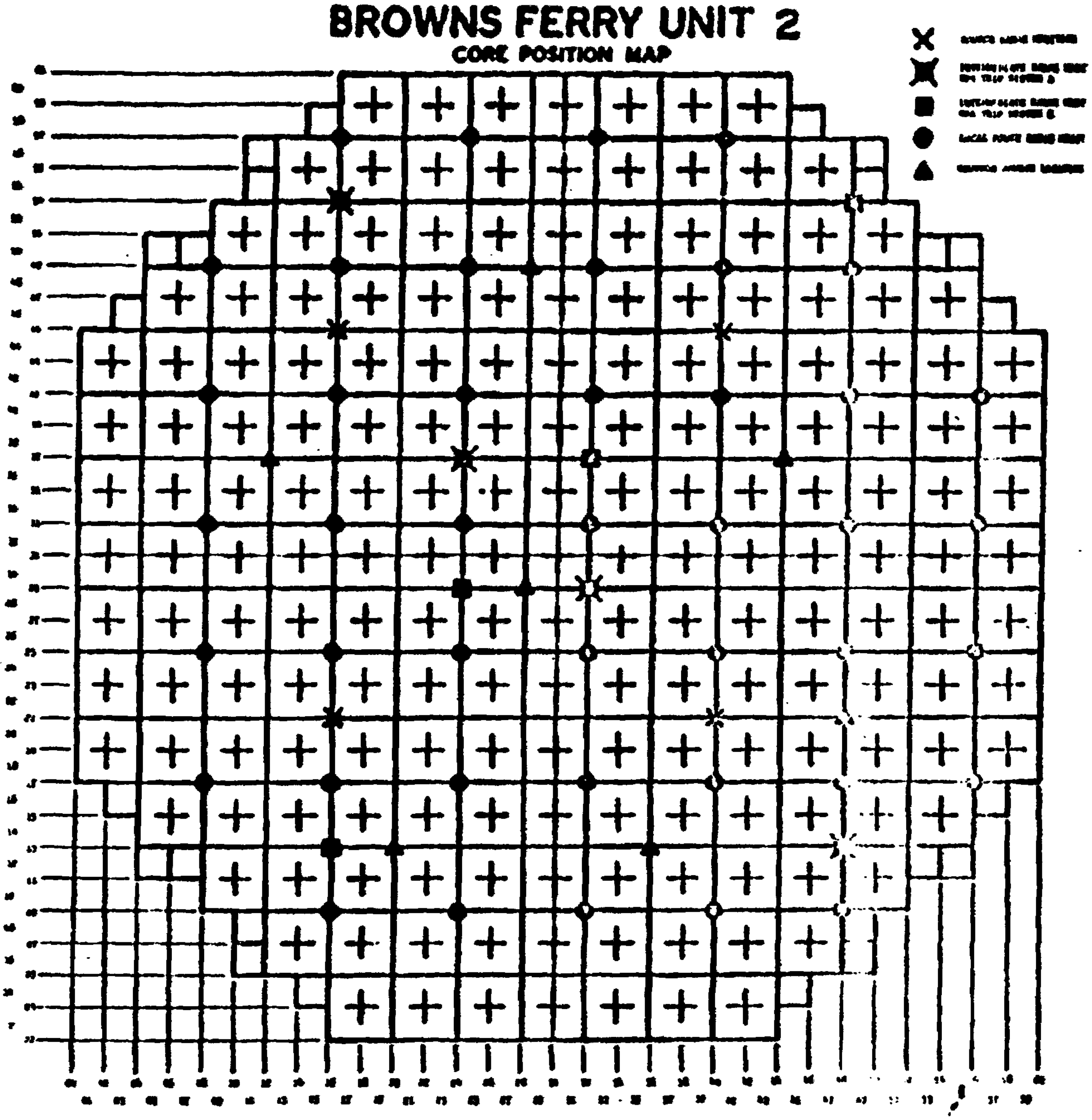


Figure STI 6-1

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.1 Phase II - Open Vessel and Cold Testing (Continued)

3.1.5 STI-10, IRM Performance

Purpose

The purpose of this test is to adjust the intermediate range monitor system to obtain an optimum overlap with the SRM and APRM systems.

Criteria

Level 1

Each IRM channel must be adjusted so that overlap with the SRM's and APRM's is assured.

The IRM's must produce a scram at 120/125 of full scale.

The IRM reading 120/125 of full scale on range 10 will be set equal to or less than 30% of rated power.

Analysis

The IRM gains were initially set to maximum gain. The IRM scram setpoints were checked during preoperational testing and are maintained through plant surveillance testing at intervals of three months. The IRM's had been placed in a non-coincidence scram mode prior to fuel loading. At the time the initial overlap data were taken, the SRM scram settings were at 5×10^5 cps. SRM's were therfore withdrawn during power ascensions to keep readings below 10^4 cps. Readings were normalized to the full-in values.

Rods were withdrawn in sequence "A" to bring the reactor critical. All the IRM's were on scale before the normalized SRM readings reached the operational limit of 3×10^5 cps. All the IRM's responded to changes in the neutron flux.

After the IRM response and IRM/SRM overlap were verified, the SRM's and IRM's were taken out of non-coincidence scram mode.

All criteria applicable to the open vessel test phase were met.

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.1 Phase II - Open Vessel and Cold Testing (Continued)

3.1.6 STI-13, Process Computer

Checkout of various computer signals and programs was performed on a continuing basis. The TIP system was thoroughly checked out and some problems with the Veeder root counters were experienced. The criteria are not applicable at this power level.

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup

3.2.1 STI-5, Control Rod Drives

Purpose

The purposes of the control rod drive system test are: (a) to demonstrate that the control rod drive (CRD) system operates properly over the full range of primary coolant temperatures and pressures from ambient to operating, and particularly that thermal expansion of core components does not bind or significantly slow control rod movements; and (b) to determine the initial operating characteristics of the entire CRD system.

Criteria

Level 1

Each drive speed in either direction (insert or withdraw) must be 3.0 ± 0.6 in. per sec. indicated by a full 12-ft. stroke in 40 to 60 sec.

The average scram insertion time of all operable control rods, based on the deenergization of the scram pilot valve solenoids as time zero, shall be no greater than:

Table STI 5-2

% Inserted from Fully Withdrawn	Average Scram Insertion Times (sec.)
5	0.375
20	0.90
50	2.0
90	5.0

The average of the scram insertion times for the thre fastest control rods of all groups of four control rods in a two-by-two array shall be no greater than:

Table STI 5-3

% Inserted from Fully Withdrawn	Average Scram Insertion Times (sec.)
5	0.398
20	0.954
50	2.120
90	5.3

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Restup (Continued)

3.2.1 STI-5, Control Rod Drives (Continued)

Criteria (Continued)

Level 1 (Continued)

The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

Level 2

With respect to the control rod drive friction tests, if the differential pressure variation exceeds 15 psid for a continuous drive-in, a settling test must be performed, in which case, the differential settling pressure should not be less than 30 psid nor should it vary by more than 10 psid over a full stroke. Lower differential pressures in the settling tests are indicative of excessive friction.

Scram times with normal accumulator charge should fall within prescribed time limits.

Analysis

Normal Insertion and Withdrawal Times

The four slowest insequence control rods were timed at rated temperature and pressure and were satisfactory.

Friction Testing

The four slowest insequence rods (control rods 10-23, 34-07, 46-11, and 30-27) were friction tested at 1000 psig reactor dome pressure. None of the rods had pressure variations on a continuous insertion exceeding 15 psid.

Scram Testing

The four slow rods were scrammed three times at 600 and 800 psig reactor pressure. See tables STI 5-4 and 5-5.

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.1 STI-5, Control Rod Drives (Continued)

Analysis (Continued)

Scram Testing (Continued)

At 1000 psig, four rods were scrammed with zero accumulator pressure. Table STI 5-6 gives the results. Table STI 5-7 lists the individual rod scram times for all rods at rated reactor pressure. All scram times were well within the criteria. Table STI 5-8 summarizes all scram test results.

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.1 STI-5, Control Rod Drives (Continued)

Table STI 5-4

600 psig Scram Tests - Four Slowest Rods

Drive Location	Test Number	Reactor Pressure, psig	Accumulator Pressure, psig	Scram Insertion Time, Sec.			
				5%	20%	50%	70%
34-07	1	600	1098	0.314	0.666	1.400	2.732
	2	600	1098	0.342	0.669	1.440	2.956
	3	600	1098	0.289	0.610	1.364	2.750
	Mean			0.315	0.648	1.401	2.819
10-23	1	600	1100	0.294	0.629	1.416	2.842
	2	600	1100	0.299	0.626	1.388	2.794
	3	600	1100	0.286	0.612	1.380	2.820
	Mean			0.293	0.622	1.395	2.820
30-27	1	600	1098	0.312	0.617	1.332	2.633
	2	600	1098	0.331	0.637	1.364	2.658
	3	600	1098	0.321	0.624	1.336	2.642
	Mean			0.321	0.626	1.344	2.644
46-11	1	600	1098	0.321	0.677	1.508	2.999
	2	600	1098	0.337	0.687	1.516	2.974
	3	600	1098	0.332	0.687	1.492	2.947
	Mean			0.330	0.683	1.505	2.913

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.1 STI-5, Control Rod Drives (Continued)

Table STI 5-5

800 psig Scram Tests - Four Slowest Rods

Drive Location	Test Number	Reactor Pressure, psig	Accumulator Pressure, psig	Scram Insertion Time, Sec.			
				5%	20%	50%	70%
30-27	1	800	1100	0.342	0.686	1.644	2.893
	2	800	1100	0.[illegible]29	0.695	1.560	2.718
	3	800	1100	0.340	0.698	1.564	2.748
	Mean			0.337	0.693	1.589	2.786
10-23	1	800	1100	0.322	0.732	1.596	2.757
	2	800	1100	0.316	0.688	1.632	2.788
	3	800	1100	0.318	0.716	1.556	2.716
	Mean			0.318	0.712	1.595	2.754
46-11	1	800	1100	0.332	0.742	1.612	2.894
	2	800	1100	0.340	0.772	1.620	2.854
	3	800	1100	0.346	0.769	1.672	3.029
	Mean			0.339	0.761	1.635	2.926
34-07	1	800	1100	0.340	0.748	1.624	2.836
	2	800	1100	0.351	0.756	1.612	2.796
	3	800	1100	0.341	0.746	1.588	2.756
	Mean			0.344	0.750	1.608	2.796

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.1 STI-5, Control Rod Drives (Continued)

Table STI 5-6

1000 psig Scram Tests - 0 Accumulator Pressure
Four Slowest Rods

Drive Location	Test Number	Reactor Pressure, psig	Accumulator Pressure, psig	Scram Insertion Time, Sec.			
				5%	20%	50%	70%
46-11	1	1000	0	0.362	0.748	1.55	2.60
	2	1000	0	0.346	0.748	1.56	2.67
	3	1000	0	0.370	0.772	1.50	2.65
	Mean			0.359	0.756	1.56	2.64
30-27	1	1000	0	0.354	0.754	1.59	2.71
	2	1000	0	0.332	0.711	1.44	2.40
	3	1000	0	0.376	0.716	1.52	2.57
	Mean			0.337	0.727	1.52	2.56
34-07	1	1000	0	0.335	0.742	1.60	2.76
	2	1000	0	0.330		1.60	2.76
	3	1000	0	0.315	0.752	1.60	2.72
	Mean			0.327	0.747	1.60	2.75
10-23	1	1000	0	0.340	0.734	1.57	2.69
	2	1000	0	0.324	0.732	1.56	2.68
	3	1000	0	0.335	0.756	1.63	2.79
	Mean			0.333	0.741	1.59	2.72

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3.0 Results (Continued)

3.3 Phase III - Initial Heatup (Continued)

3.3.3 STI-?, Control Rod Drives (Continued)

Table 3-?
Sequence A Rod Scram Tests - Reactor Power 1?

Drive Location	Reactor Pressure psig	Scram Insertion Times, Sec. 5%	20%	50%	90%
18-55	900	0.323	0.671	1.38	2.47
26-55		0.322	0.681	1.39	2.38
34-55		0.328	0.740	1.31	2.24
42-55		0.326	0.695	1.45	2.47
10-39		0.305	0.704	1.50	2.09
34-31		0.354	0.724	1.48	2.50
58-23		0.478	0.714	1.41	2.30
18-39		0.331	0.662	1.47	2.38
42-31	1000	0.323	0.705	1.49	2.37
10-31		0.318	0.668	1.37	2.35
26-39		0.328	0.682	1.44	2.30
50-31		0.318	0.671	1.38	2.30
18-15		0.313	0.646	1.39	2.42
34-39		0.332	0.668	1.49	2.42
50-31		0.318	0.663	1.37	2.41
26-15		0.328	0.649	1.47	2.34
10-47		0.32[illegible]	0.675	1.49	2.41
42-39		0.328	0.695	1.41	2.44
52-23		0.318	0.658	1.36	2.32
34-15		0.318	0.695	1.47	2.39
18-47		0.342	0.708	1.49	2.44
50-39		0.316	0.705	1.55	2.81
10-23		0.335	0.687	1.42	2.41
42-15		0.318	0.697	1.42	2.44
26-47		0.328	0.695	1.45	2.35
58-39		0.318	0.636	1.37	2.36
18-23		0.312	0.671	1.46	2.41
50-15		0.302	0.700	1.50	2.45
34-47		0.312	0.643	1.39	2.43
58-31		0.304	0.636	1.27	2.20
26-23		0.317	0.703	1.43	2.46
18-07		0.318	0.679	1.38	2.39
42-47		0.312	0.657	1.44	2.42
10-31		0.320	0.674	1.42	2.46
34-23		0.365	0.695	1.45	2.50

Drive Location	Reactor Pressure psig	Scram Insertion Times, Sec. 5%	20%	50%	90%
26-07	1000	0.312	0.663	1.35	2.18
30-47		0.314	0.661	1.30	2.43
10-31		0.326	0.695	1.42	2.44
42-23		0.309	0.695	1.48	2.30
34-07		0.329	0.670	1.41	2.30
02-39		0.304	0.694	1.21	2.18
26-31		0.314	0.663	1.38	2.45
50-23		0.324	0.679	1.44	2.53
42-07		0.326	0.695	1.41	2.49
34-23		0.310	0.687	1.46	2.33
27-59		0.332	0.683	1.43	2.47
30-43		0.331	0.709	1.44	2.50
06-27		0.334	0.708	1.41	2.45
38-19		0.334	0.675	1.41	2.33
39-19		0.303	0.653	1.35	2.36
38-43		0.300	0.639	1.30	2.30
14-27		0.328	0.687	1.42	2.43
46-19		0.300	0.666	1.36	2.55
18-59		0.321	0.643	1.32	2.36
14-51		0.326	0.653	1.32	2.35
27-51		0.330	0.701	1.42	2.45
30-51		0.326	0.671	1.37	2.36
38-51		0.319	0.629	1.39	2.35
46-51		0.316	0.703	1.47	2.30
06-43		0.313	0.663	1.41	2.46
14-43		0.321	0.670	1.49	2.45
22-43		0.337	0.709	1.45	2.47
46-43		0.308	0.679	1.42	2.44
54-43		0.326	0.679	1.43	2.47
06-35		0.329	0.674	1.36	2.38
14-35		0.326	0.669	1.40	2.40
22-35		0.306	0.661	1.42	2.48
30-35		0.329	0.677	1.40	2.42
38-35		0.333	0.673	1.40	2.42
46-35		0.330	0.690	1.41	2.41

3.0 Results (Continued)

3.2 Phase III - Initial System (Continued)

3.2.1 CRD's Control Rod Drives (Continued)

Table 3-7 (Continued)
Sequence A Rod Scram Tests - Reactor Power IX

Drive Location	Reactor Pressure (psig)	Scram Insertion Times, Sec. 5%	20%	50%	90%
30-35	1000	0.335	0.661	1.46	2.45
22-29		0.318	0.670	1.44	2.52
30-27		0.303	0.647	1.31	2.30
38-27		0.305	0.663	1.30	2.43
46-27		0.351	0.687	1.47	2.46
14-19		0.306	0.633	1.36	2.38
06-19		0.308	0.633	1.38	2.38
14-15		0.343	0.703	1.43	2.43
22-19		0.329	0.677	1.43	2.44
30-19		0.324	0.700	1.49	2.37
34-19		0.309	0.609	1.37	2.36
14-11		0.310	0.679	1.39	2.41
22-11		0.318	0.667	1.44	2.52
30-11		0.313	0.671	1.49	2.48
38-11		0.308	0.643	1.30	2.44
46-11		0.398	0.703	1.47	2.32
22-03		0.309	0.669	1.37	2.33
46-19		0.300	0.633	1.30	2.40
30-03		0.300	0.670	1.43	2.38
38-03	↓	0.308	0.630	1.30	2.37

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3.0 Results (Continued)

3.2 Phase III - Initial Startup (Continued)

3.2.1 STI-5, Control Rod Drives (Continued)

Table 3-7
Sequence A and Scram Tests - Reactor Power [illegible]

Drive Location	Reactor Pressure psig	Scram Insertion Time, Sec. 5%	20%	50%	90%
38-39	925	0.313	0.643	1.46	2.43
42-35		0.332	0.761	1.69	2.79
46-39		0.304	0.706	1.52	2.56
50-39		0.338	0.730	1.56	2.64
36-31		0.332	0.719	1.46	2.48
10-19		0.341	[illegible]	1.50	2.35
14-19		[illegible]	0.698	1.56	2.48
38-15		0.314	0.666	1.37	2.42
02-27		0.318	[illegible]	1.52	2.49
06-31		0.356	[illegible]	1.32	2.41
10-27		0.322	[illegible]	1.50	2.46
46-47		0.349	[illegible]	1.38	2.08
[illegible]		[illegible]	[illegible]	[illegible]	[illegible]
32-35		[illegible]	[illegible]	1.61	2.36
34-47		[illegible]	[illegible]	1.70	2.41
22-47		[illegible]	[illegible]	1.61	2.42
26-43		0.313	[illegible]	1.44	2.47
30-47		0.338	0.732	1.56	2.72
14-43		0.313	0.647	1.45	2.45
42-43		0.296	[illegible]	1.39	2.43
42-50		0.322	[illegible]	1.37	2.38
14-55		0.320	0.695	1.44	2.52
19-51		0.332	0.693	1.43	2.50
38-55		0.324	0.710	1.55	2.72
42-51		0.334	[illegible]	1.44	2.46
46-55		0.320	0.677	1.38	2.35
50-51		0.296	0.650	1.36	2.37
46-47		0.319	0.650	1.36	2.31
42-53		0.318	0.661	1.40	2.47
18-43		0.367	0.730	1.60	2.56
38-47		0.340	0.716	1.53	2.61
06-39		0.326	0.695	1.48	2.55
26-59		0.319	0.674	1.35	2.22
18-59		0.313	0.679	1.40	2.39
34-59		[illegible]	[illegible]	1.46	2.54

Drive Location	Reactor Pressure psig	Scram Insertion Time, Sec. 5%	20%	50%	90%
38-11	925	0.326	0.695	1.49	2.38
22-15		0.340	0.717	1.65	2.52
10-11		0.345	0.703	1.43	2.49
14-15		0.340	0.744	1.57	2.72
30-11		0.332	0.703	1.46	2.53
06-15		0.310	0.633	1.38	2.40
34-23		0.326	0.712	1.56	2.76
54-19		0.326	0.687	1.44	2.56
50-19		0.316	0.693	1.48	2.60
06-23		0.326	0.718	1.54	2.58
[illegible]		0.326	0.648	1.38	2.42
58-27		0.313	0.661	1.37	2.36
54-31		0.326	0.712	1.56	2.78
50-27		0.310	0.695	1.44	2.32
[illegible]		[illegible]	[illegible]	1.49	[illegible]
62-27		[illegible]	0.677	1.43	2.48
38-31		0.354	0.764	1.57	2.74
34-27		0.313	0.679	1.48	2.61
26-27		0.329	0.693	1.44	2.51
22-31		0.334	0.836	1.53	2.64
14-31		0.350	0.764	1.65	2.80
30-35		0.319	0.717	1.52	2.63
26-51		0.326	0.703	1.48	2.46
18-51		0.316	0.677	1.37	2.33
14-19		0.326	0.687	1.48	2.46
38-23		0.336	0.737	1.60	2.80
42-19		0.326	0.687	1.42	2.48
46-23		0.324	0.710	1.56	2.68
30-15		0.310	0.707	1.42	2.67
10-35		0.330	0.748	1.52	2.53
18-35		0.308	0.676	1.46	2.53
22-35		0.326	0.669	1.48	2.52
26-35		0.330	0.781	1.56	2.69
30-39		0.332	0.736	1.52	2.62
34-39		0.312	0.709	1.40	2.59

[illegible] - [illegible]

3.0 [illegible] (Continued)

3.3 [illegible] (Continued)

3.3.1 [illegible] (Continued)

Table 3-[illegible] (Continued)
[illegible]

[illegible]	[illegible] psig	[illegible] Sec.			
		[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	0.3[illegible]	0.6[illegible]	1.4[illegible]	2.[illegible]
[illegible]		0.3[illegible]	0.[illegible]	1.45	2.[illegible]
[illegible]		0.3[illegible]	0.7[illegible]	1.4[illegible]	2.[illegible]
[illegible]		0.3[illegible]	0.[illegible]	1.[illegible]	2.45
[illegible]		0.3[illegible]	0.[illegible]	1.4[illegible]	2.[illegible]
[illegible]		0.3[illegible]	0.[illegible]	1.4[illegible]	2.[illegible]
[illegible]		0.3[illegible]	0.[illegible]	1.[illegible]	2.[illegible]
[illegible]-47		0.3[illegible]	0.7[illegible]	1.64	2.[illegible]
[illegible]-15		0.3[illegible]	0.7[illegible]	1.[illegible]	2.[illegible]
[illegible]-07		0.3[illegible]	0.7[illegible]	1.[illegible]	2.[illegible]
[illegible]-03		0.3[illegible]	0.[illegible]	1.46	2.[illegible]
[illegible]-03		0.3[illegible]	0.[illegible]	1.44	2.[illegible]
[illegible]-07		0.3[illegible]	0.[illegible]	1.44	2.[illegible]
[illegible]-11		0.3[illegible]	0.[illegible]	1.4[illegible]	2.[illegible]
[illegible]-15		0.3[illegible]	0.7[illegible]	1.[illegible]	2.[illegible]
[illegible]-11		0.3[illegible]	0.[illegible]	1.4[illegible]	2.[illegible]
[illegible]-11		0.3[illegible]	0.[illegible]	1.4[illegible]	2.[illegible]
[illegible]-15		0.3[illegible]	0.[illegible]	1.44	2.[illegible]
[illegible]-15		0.3[illegible]	0.[illegible]	1.4[illegible]	2.[illegible]
[illegible]-07		0.3[illegible]	0.7[illegible]	1.4[illegible]	2.[illegible]
[illegible]-07		0.3[illegible]	0.7[illegible]	1.46	2.[illegible]
[illegible]-03		0.3[illegible]	0.7[illegible]	1.[illegible]	2.[illegible]
[illegible]-03		0.3[illegible]	0.7[illegible]	1.4[illegible]	2.47
[illegible]-07		0.3[illegible]	0.[illegible]	1.4[illegible]	2.43
[illegible]		0.3[illegible]	0.7[illegible]	1.[illegible]	2.43
[illegible]		0.3[illegible]	0.7[illegible]	1.4[illegible]	2.[illegible]
[illegible]-03		0.3[illegible]	0.[illegible]	1.[illegible]	2.[illegible]
[illegible]		0.3[illegible]	0.[illegible]	1.44	2.4[illegible]

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3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.1 STI-5, Control Rod Drives (Continued)

Table STI 5-8

Summary of Scram Test Results

Reactor Pressure	Accumulator Pressure	Number Of Rods	Mean Insertion Times (Sec.) 5%	20%	50%	90%
0	Normal	185	0.288	0.503	0.966	1.642
0	Minimum	4*	0.316	0.569	1.101	1.890
600	Normal	4*	0.315	0.645	1.411	2.814
800	Normal	4*	0.335	0.729	1.007	2.816
1000	Zero	4*	0.339	0.743	1.568	2.668
1000	Normal	185	0.326	0.692	1.29	2.484

* Four slowest insequence rods

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.2 STI-6, SRM Performance and Control Rod Sequence

Purpose

The purpose of this test is to demonstrate that the operational sources, SRM instrumentation, and rod withdrawal sequences provide adequate information to achieve criticality and to increase power in a safe and efficient manner. The effect of typical rod movements on reactor power will be determined.

Criteria

Level 1

(a) There must be a neutron signal-to-noise ratio of at least 2:1 on the required operable SRM's or fuel loading chambers.

(b) There must be a minimum count rate of 3 cps on the required operable SRM's or fuel loading chambers.

(c) The IRM's must be on scale before the SRM's exceed the rod block set point.

(d) The RSCS shall be operable as specified in the technical specifications.

Analysis

Unit 2 was heated to rated temperature in sequence "A". Neutron instrumentation was carefully monitored to insure safe heatup rate and power ascension.

The RWM software and the hard wired RSCS prevented out-of-sequence rod movement, thus minimizing the worth of individual rods.

In sequence "A", the RSCS is composed of four major rod groups; A12, A34, B12, and B34. Figure STI 6-2 shows the A12, A34 rods, and figures STI 6-3 and STI 6-4 contain the B12, B34 as numbered subgroups.

In heating up and raising power, the procedure for rod withdrawal is as follows for sequence "A". Any rod in

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.2 STI-6, SRM Performance and Control Rod Sequence (Continued)

Analysis (Continued)

RSCS A12 may be selected and continuously withdrawn. (The specific sequence is prescribed by the RWM.) All the A12 rods must be full out before any other RSCS group can be moved. Next the A34 may be selected and rods moved continuously to the 50% control density, known as "checkerboard" pattern. Only after both A12 and A34 are full out may any "B" RSCS group be moved. The "B" RSCS groups are moved in the group notch mode, i.e., all the rods within any given RSCS group must be within one notch position of each other, or further rod moves are prohibited in that group. When the reactor power is below 20%, all rod moves must be consistent with the programmed RWM sequence as well. The rod groups and operation in sequence "B" are analogous. All test criteria were met.

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3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.2 STI-6, SRM Performance and Control Rod Sequence (Continued)

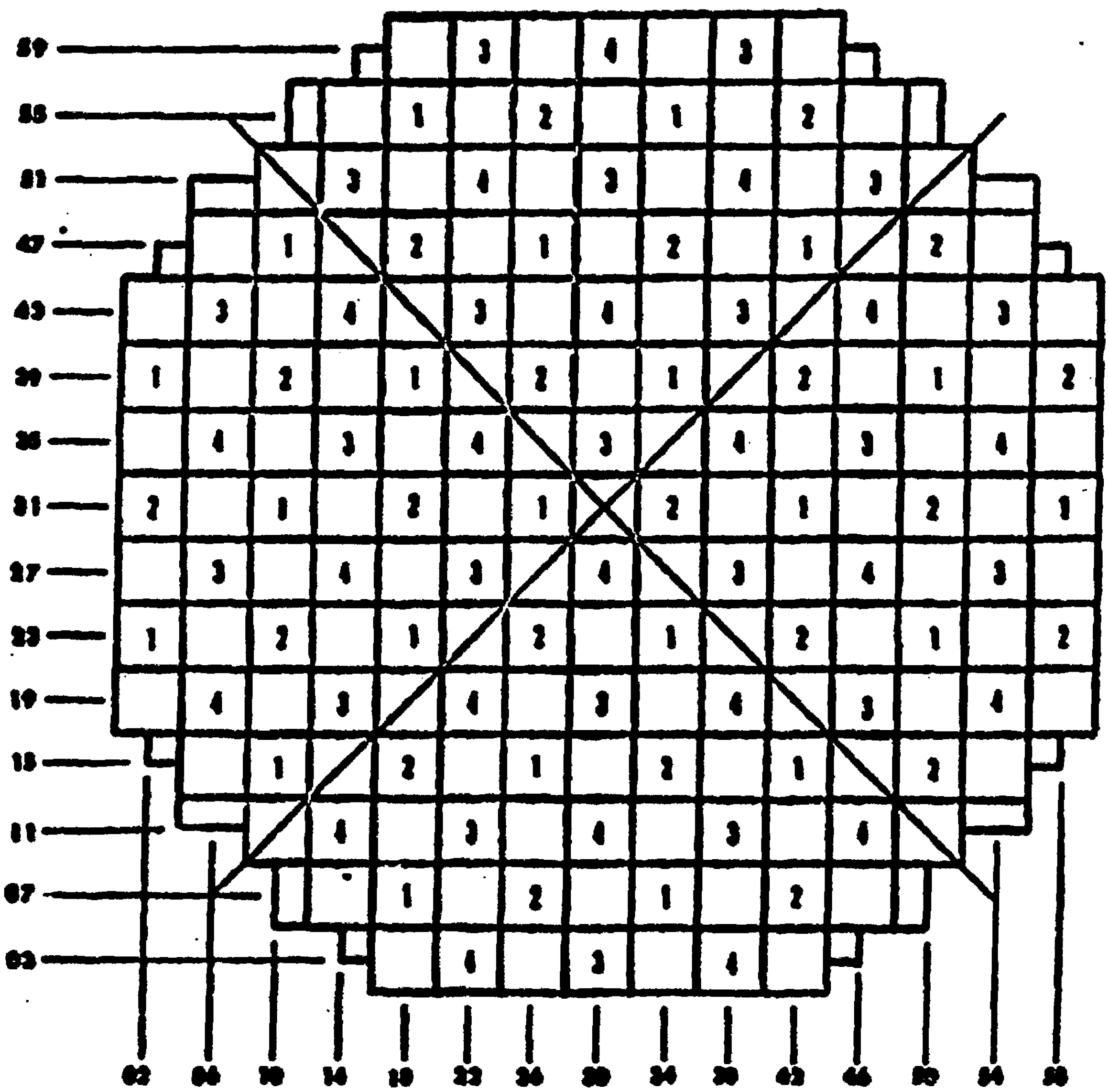

Figure STI 6-2
RSCS Rod Groups A12 and A34

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.2 STI-6, SRM Performance and Control Rod Sequence

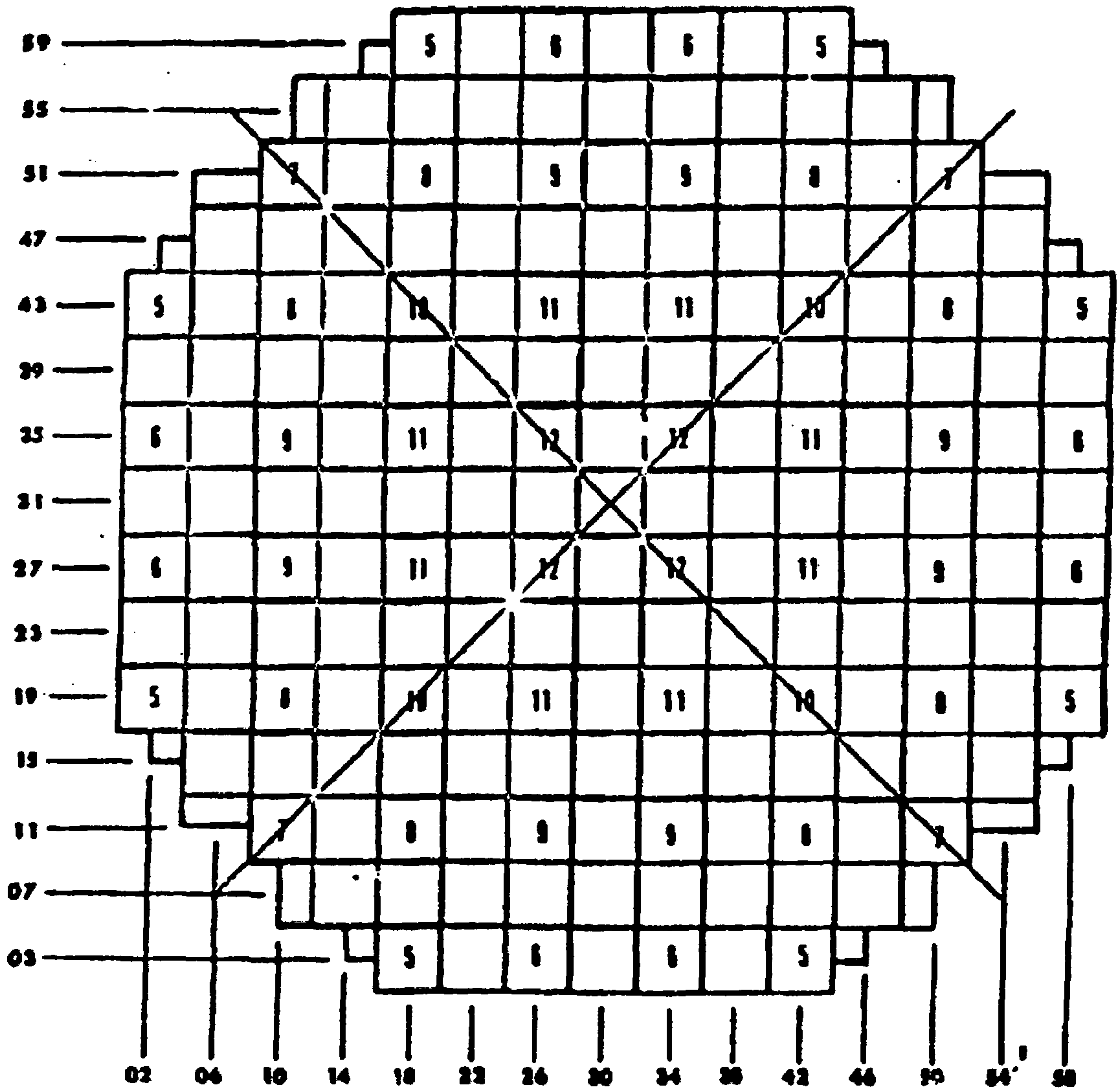

Figure STI 6-3
RSCS Rod Group B12

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3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.2 STI-6, SRM Performance and Control Rod Sequence

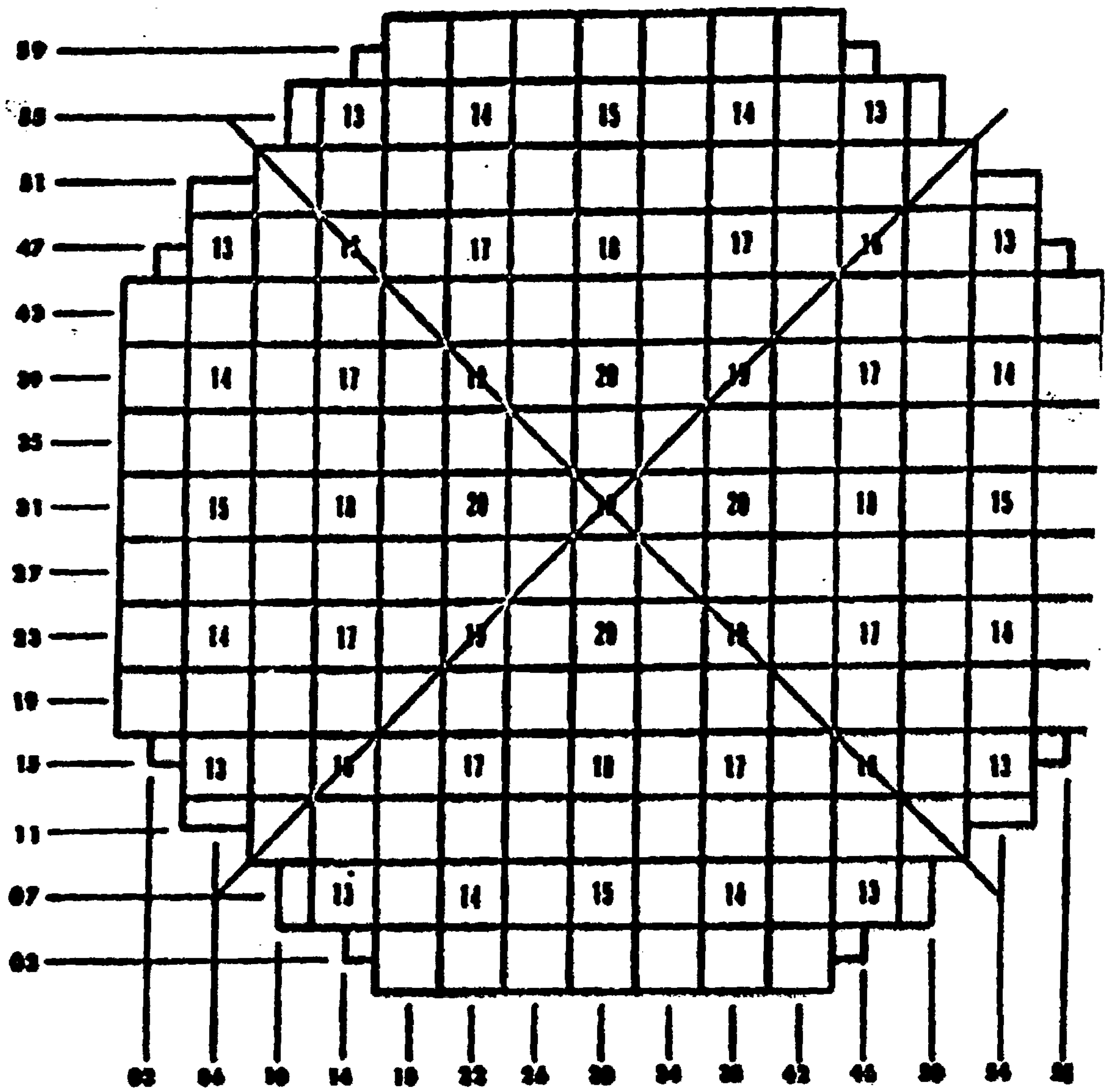

Figure STI 6-4
RSCS Rod Group B34

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3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.3 STI-10, IRM Performance

Purpose

The purpose of this test is to adjust the intermediate range monitor system to obtain an optimum overlap with the SRM and APRM systems.

Criteria

Level 1

(a) Each IRM channel must be adjusted so that overlap with the SRM's and APRM's is assured.

(b) The IRM's must produce a scram at 120/125 of full scale.

(c) The IRM reading 120/125 of full scale on range 10 will be set equal to or less than 30% of rated power.

Analysis

The IRM preamplifiers were adjusted for continuity between ranges six and seven during the initial heatup. Following the calibration of the APRM's by heatup rate testing, the IRM's were adjusted to match the APRM's based on 120 divisions of range 10 equaling 25% power. This calibration was performed per SI 4.1.B-1 with the APRM's readings averaging 8.7% power.

The calibration was performed at this power level to lower the high IRM gains which had been set to the maximum.

Proper SRM and APRM overlap with the IRM's was reverified after this calibration. The IRM's were set to scram at 120/125 of full scale per SI 4.2.C-3B. All criteria were satisfied during heatup.

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.4 STI-12, APRM Calibration

Purpose

The purpose of this test is to calibrate the average power range monitoring system.

Criteria

Level 1

(a) The APRM channels must be calibrated to read equal to or greater than the actual core thermal power.

(b) Technical specification and fuel warranty limits on APRM scram and rod block shall not be exceeded.

(c) In the startup mode, all APRM channels must produce a scram at less than or equal to 15% of rated thermal power.

(d) Recalibration of the APRM system will not be necessary from safety considerations if at least two APRM channels per RPS trip circuit have readings greater than or equal to core power.

Level 2

(a) If the above criteria are satisfied then the APRM channels will be considered to be reading accurately if they agree with the heat balance to within 7% of rated power.

Analysis

The APRM's were calibrated using the low power heat balance based on the heatup rate. After the heatup rate had stabilized at 60° F. per hour, the APRM's were set to read .67% thermal power. This calibration was used until a more accurate heat balance could be performed at a higher power level. All applicable test criteria were satisfied.

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.5 STI-14, RCIC System

Purpose

The purpose of this test is to verify the proper operation of the reactor core isolation cooling system over its required operating pressure range.

Criteria

Level 1

(a) The time from actuating signal to required flow must be less than 30 seconds at any reactor pressure between 150 psig and rated (1020 psig).

(b) With pump discharge at any pressure between 150 psig and 1220 psig, the required flow is 600 gpm. (The limit of 1220 psig includes a nominally high value of 100 psi for line losses. The measured value of 50 psig may also be used.)

(c) The RCIC turbine must not trip off during startup.

(d) If either of the first two level 1 criteria is not met, the reactor will only be allowed to operate at a restricted power level.

Level 2

(a) The turbine gland seal condenser system shall be capable of preventing steam leakage to the atmosphere.

(b) The ΔP switch for the RCIC steam supply line high flow isolation trip shall be adjusted to actuate at 300% of the maximum required steady state steam flow.

Analysis

All testing was conducted during this phase with RCIC taking suction from and discharging to the condensate storage tank.

At 150 psig nominal vessel pressure, the RCIC test was accomplished with a discharge pressure of 270 psig.

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3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.5 STI-14, RCIC System (Continued)

At 800 psig nominal vessel pressure, the RCIC test was accomplished with a discharge pressure of 920 psig.

At 1000 psig nominal vessel pressure, the RCIC test was successful against discharge pressures of 1120 psig and 1220 psig.

Throughout phase III testing, all controller settings were considered satisfactory. The pertinent data from these test is presented in table STI 14-1. Transient response is shown in figures STI 14-1 through STI 14-4.

All test criteria were satisfied with the exception of the level 2 criteria for high steam flow isolation setpoints. Excessive pressure drop across the elbow taps gives a higher than expected signal to the steam flow instrument switches. Therefore these switches could not be set at the calculated 300% rated steam flow due to limited instrument range. The switches remain set at the present technical specification limit of <450 inches of H_2O pending resolution by TVA DED and GE.

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3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.5 STI 14. RCIC System (Continued)

Table STI 14-1

Date	Time	Reactor Pressure (psig)	Discharge Pressure (psig)	Turbine Speed (rpm)	Controller Settings PB	r/m	Time to Rated Flow (sec.)	Figure Number
8/6/74	0220	150	270	2150	600	100	12.2	STI 14-1
8/10/74	1455	810	920	3850	600	100	5.5	STI 14-2
8/23/74	1910	1000	1120	4200	600	100	13.0	STI 14-3
8/26/74	1245	1000	1220	4450	600	100	19.0	STI 14-4

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.3 STI-14, RCIC (Continued)

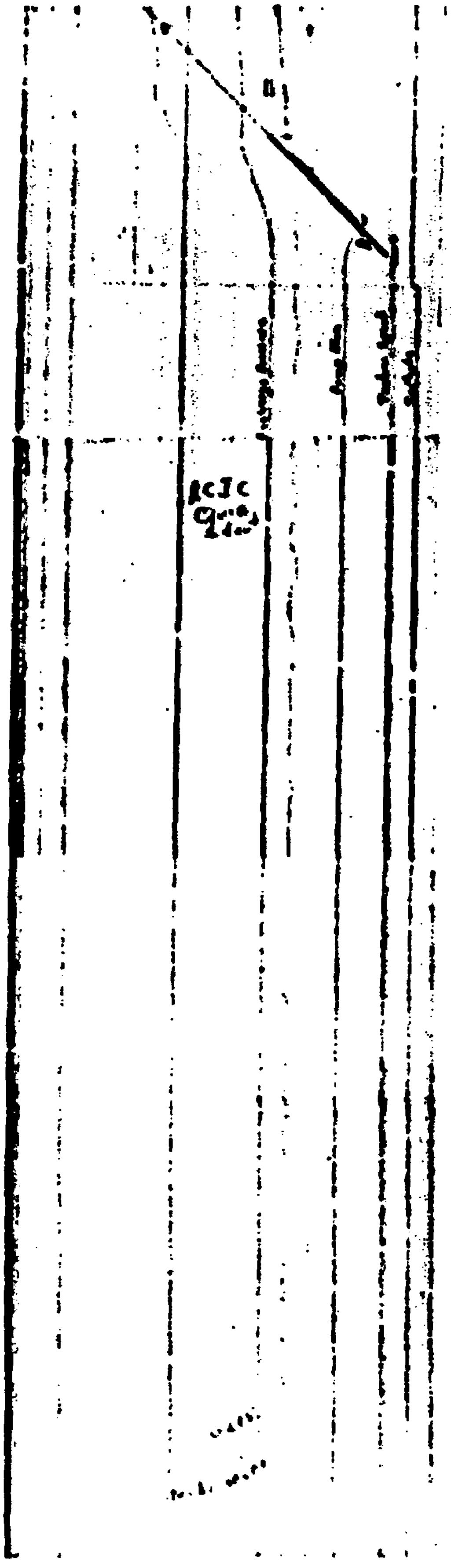

Figure STI 14-1A
Auto Testing at 150 psig
(Case #1)
1 second per division

3.0 <u>Results</u> (Continued)

3.2 <u>Phase III</u> - Initial Heatup (Continued)

3.2.5 <u>STI-14, RCIC</u> (Continued)

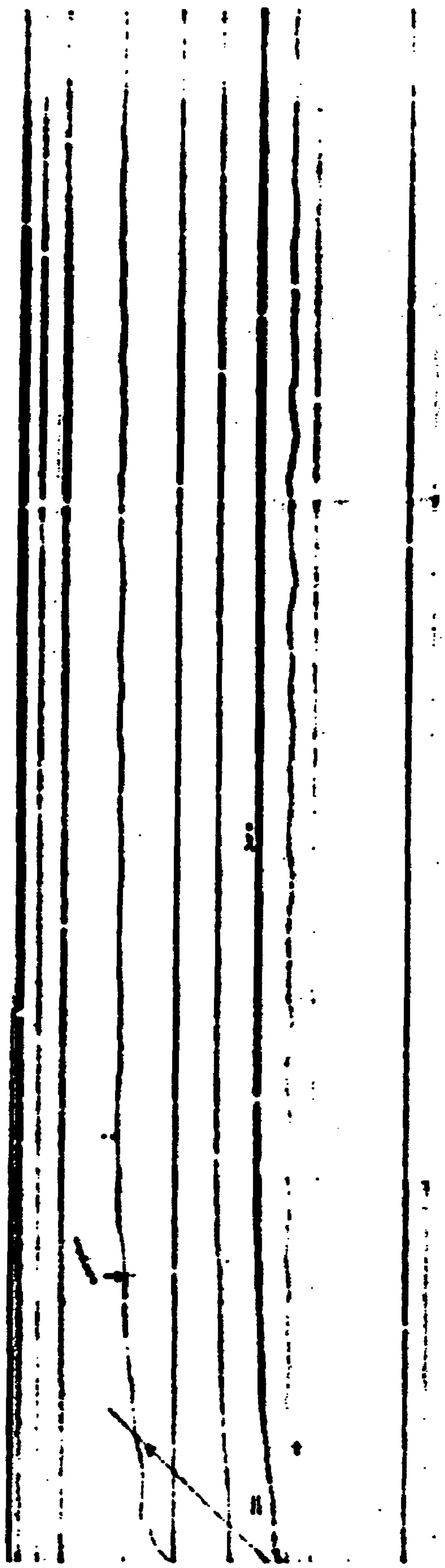

Figure STI 14-13
RCIC Testing at 150 psig
(Trace #1).
<u>1 second per division</u>

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.5 STI 14, RCIC (Continued)

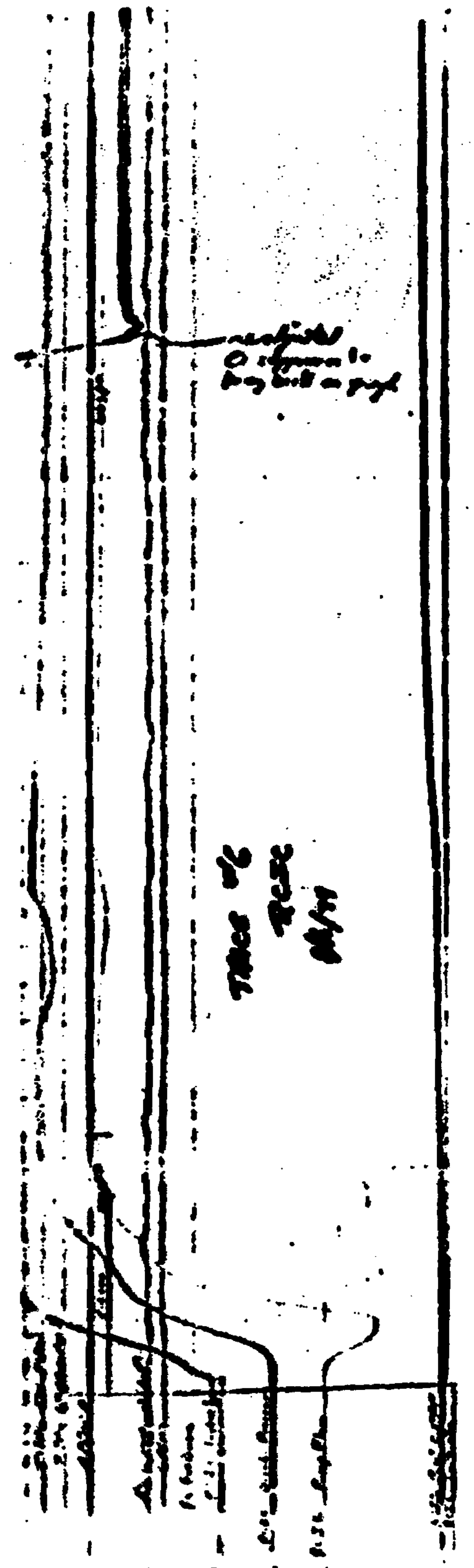

Figure STI 14-2
RCIC Testing at 800 psig
(Trace #6)
1 second per division

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.5 STI 1A, RCIC (Continued)

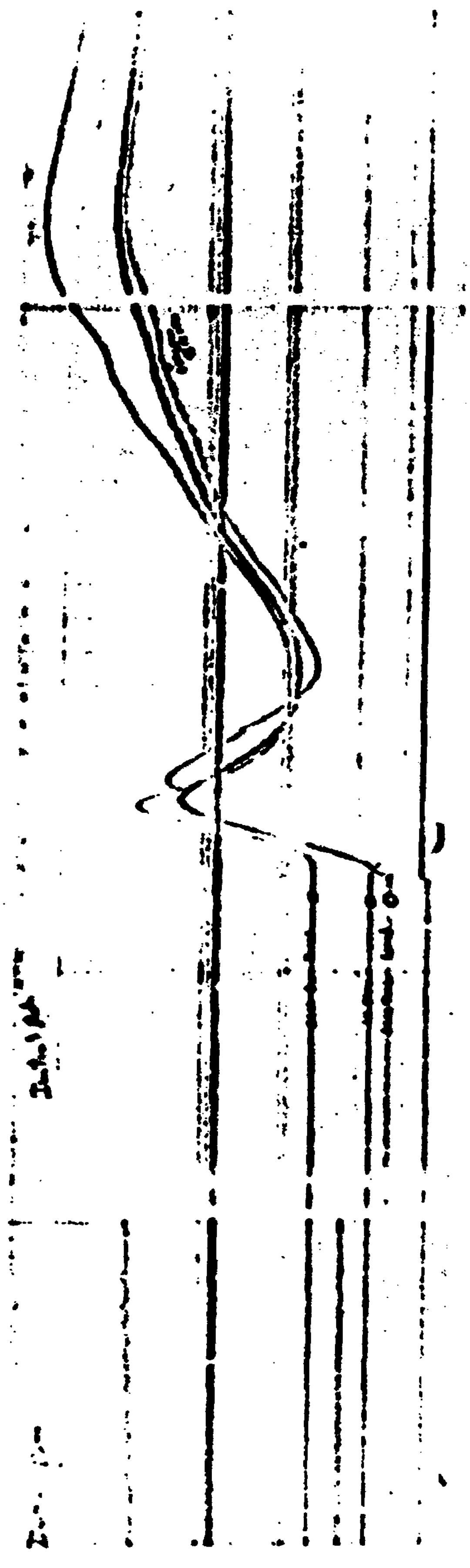

Figure III 1A-3A
RCIC Testing at Rated Press
(Trace #9)
1 second per division

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.5 STI 14, RCIC (Continued)

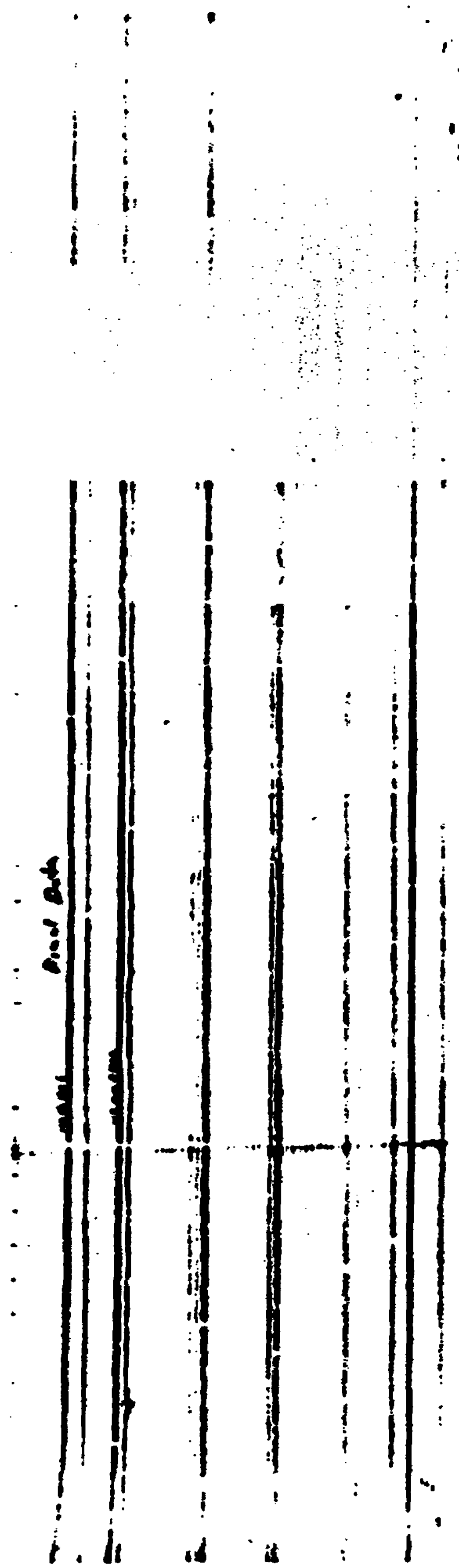

Figure STI 14-28
RCIC Testing at Rated Press
(Trace #3)
1 second per division

3.0 <u>Results</u> (Continued)

3.2 <u>Phase III</u> - Initial Heatup (Continued)

3.2.5 <u>STI-14, RCIC</u> (Continued)

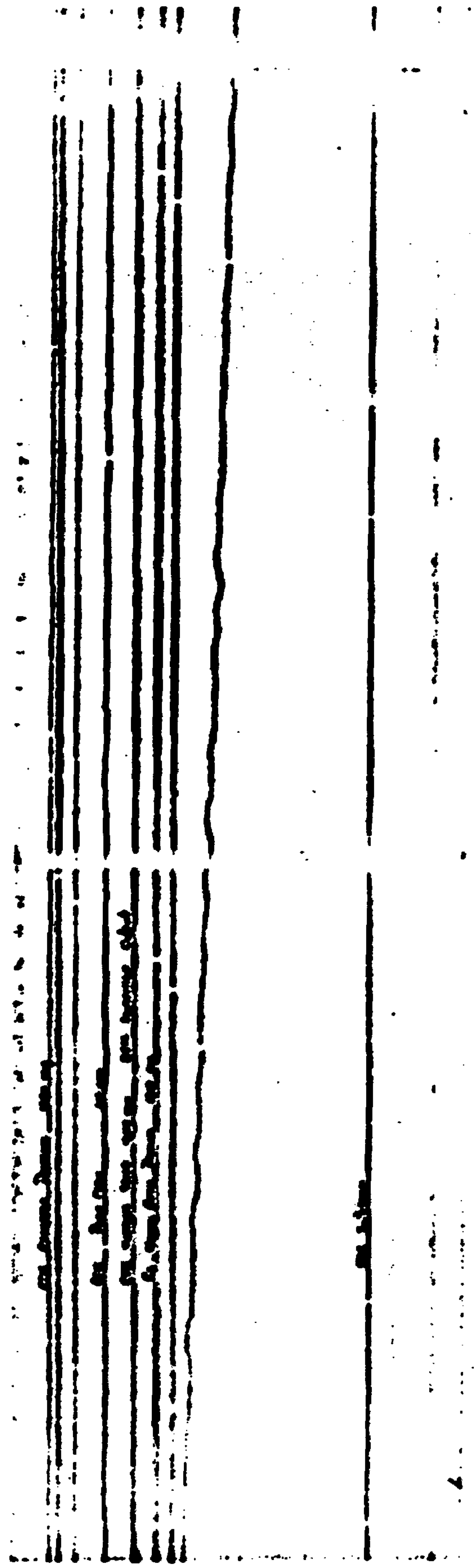

Figure STI 14-4A
RCIC Testing at Maximum Pressure
(Trace #10)
<u>1 second per division</u>

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.5 STI-1A, RCIC (Continued)

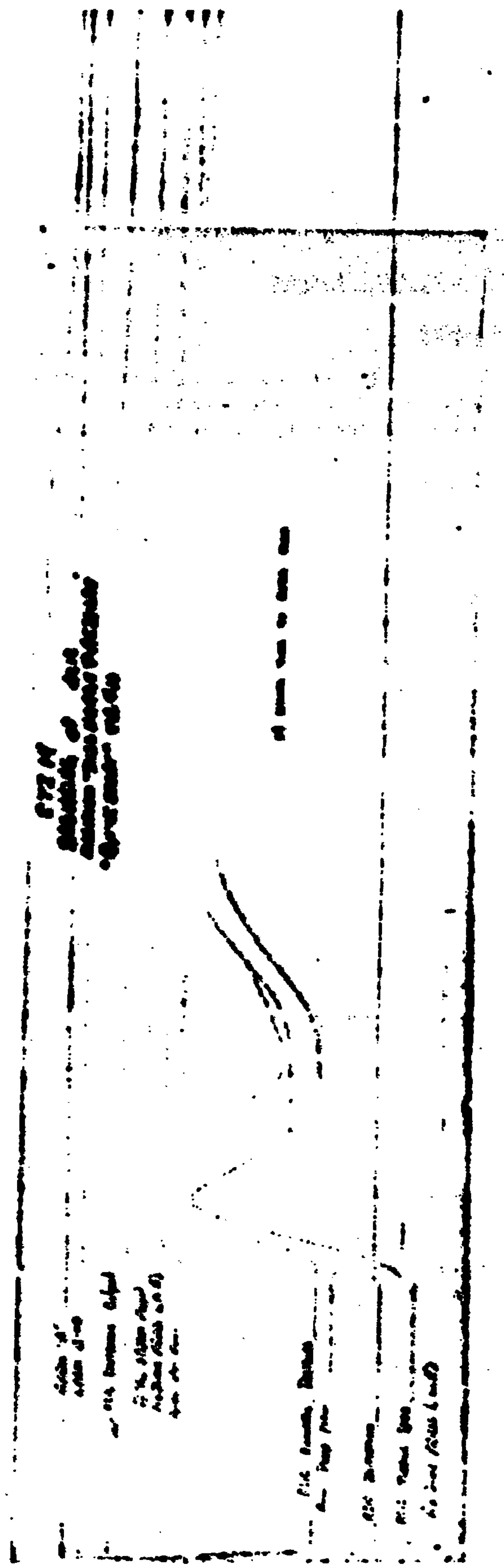

Figure STI 1A-43
RCIC Testing at Maximum Pressure
(Trace #10)
1 second per division

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3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.6 STI-15, HPCI System

Purpose

The purpose of this test is to verify the proper operation of the high pressure coolant injection system throughout the range of reactor pressure conditions.

Criteria

Level 1

(a) The time from actuating signal to required flow must be less than 25 seconds at any reactor pressure between 150 psig and rated (1020 psig).

(b) With pump discharge at any pressure between 150 psig and 1220 psig, the flow should be at least 5000 gpm. (The limit of 1220 psig includes a nominally high value of 100 psi for line losses. The measured value may also be used, if available.

(c) The HPCI turbine must not trip off during startup.

Level 2

(a) The turbine gland seal condenser system shall be capable of preventing steam leakage to the atmosphere.

(b) The ΔP switch for the HPCI steam supply line high flow isolation trip shall be adjusted to actuate at 225% of the maximum required steady-state steam flow.

Analysis

All testing was conducted with HPCI taking suction from and discharging to the condensate storage tank.

At 150 psig and 800 psig nominal reactor vessel pressures, HPCI was tested successfully with discharge pressures of 320 psig and 920 psig respectively.

With a 1000 psig nominal vessel pressure HPCI was tested successfully with discharge pressures of 1140 psig and 1300 psig.

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.6 STI-15, HPCI System (Continued)

Analysis (Continued)

Throughout phase III testing all controller settings were considered satisfactory. The pertinent data from these tests is presented in table STI 15-1. Transient response is shown in figures STI 15-1 through STI 15-4. All test criteria were satisfied.

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3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.6 STI-15, HPCI System (Continued)

Table STI 15-1

Date	Time	Reactor Pressure (psig)	Discharge Pressure (psig)	Turbine Speed (rpm)	Controller Settings PB	r/m	Time to Rated Flow (sec.)	Figure Number
8/7/74	1220	150	320	2300	600	100	20.3	STI 15-1
8/10/74	0801	822	920	3400	600	100	17.3	STI 15-2
8/26/74	1505	1000	1140	3700	600	100	22.3	STI 15-3
8/26/74	1600	1000	1300	3800	600	100	24.0	STI 15-4

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Cont.)

3.2.6 STI-13, HPCI System (Cont.)

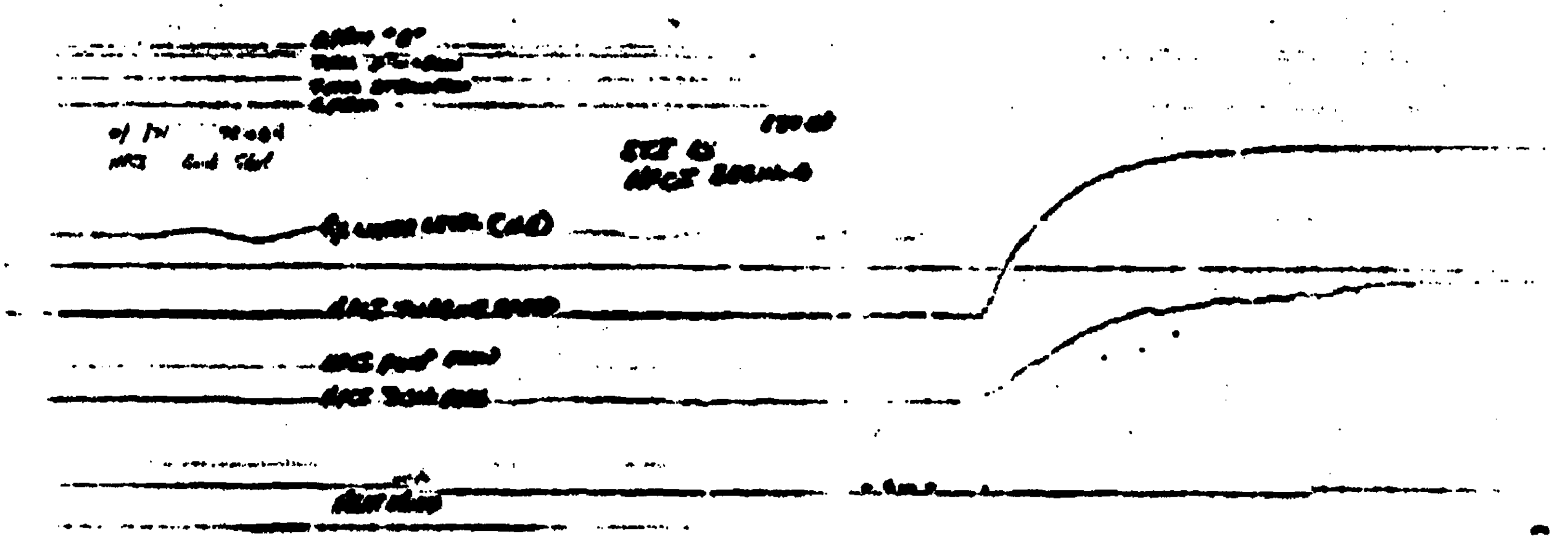

Figure STI 13-1
HPCI Testing of 150 psig
Trace #4
1 second per division

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Cont.)

3.2.6 STI-15, HPCI Sys. (Cont.)

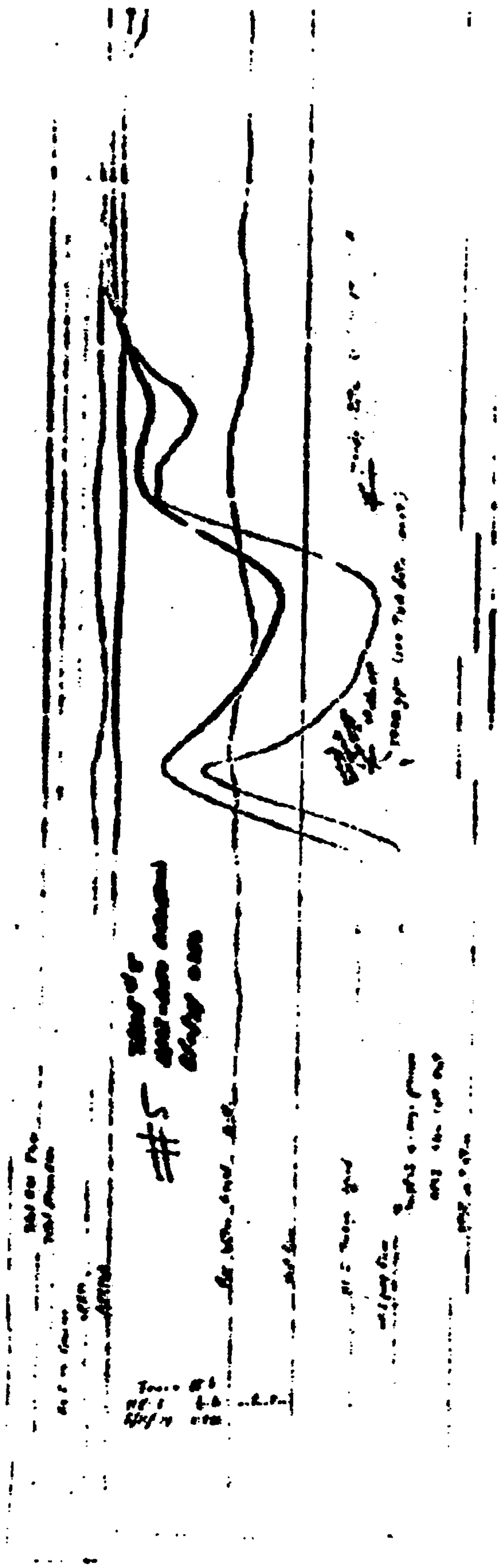

Figure STI 15-2A
HPCI Testing at 800 psig
(Trace #5)
1 second per division

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.6 STI-15, HPCI System (Cont.)

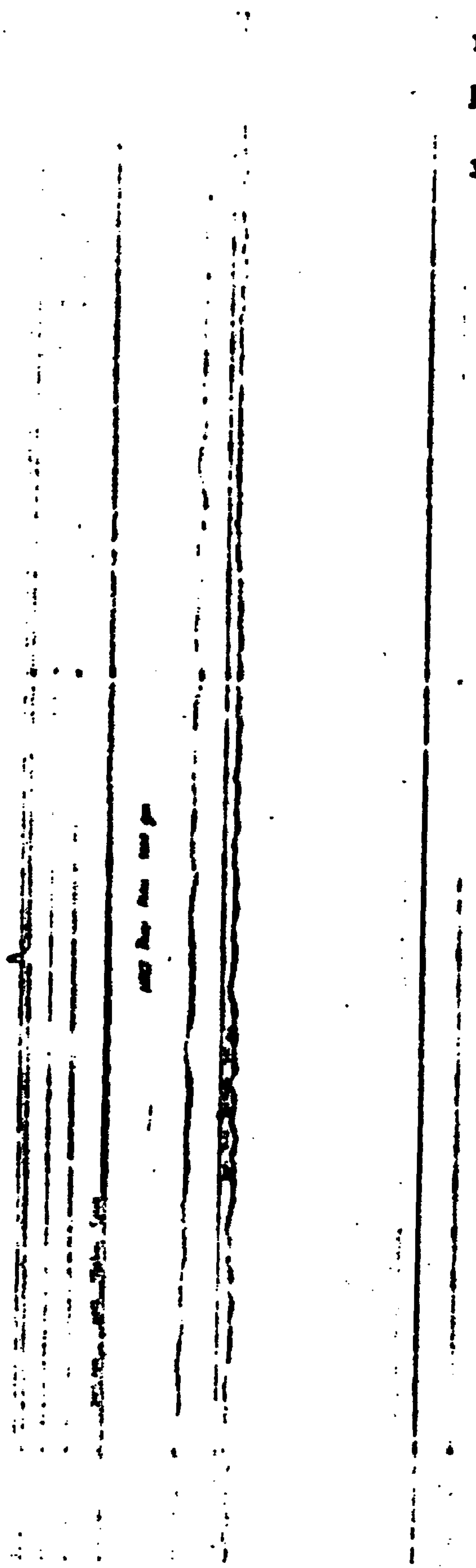

Figure STI 15-23
HPCI Testing of 800 psig
(Trace #5)
1 second per division

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.6 STI-15 HPCI System (Cont)

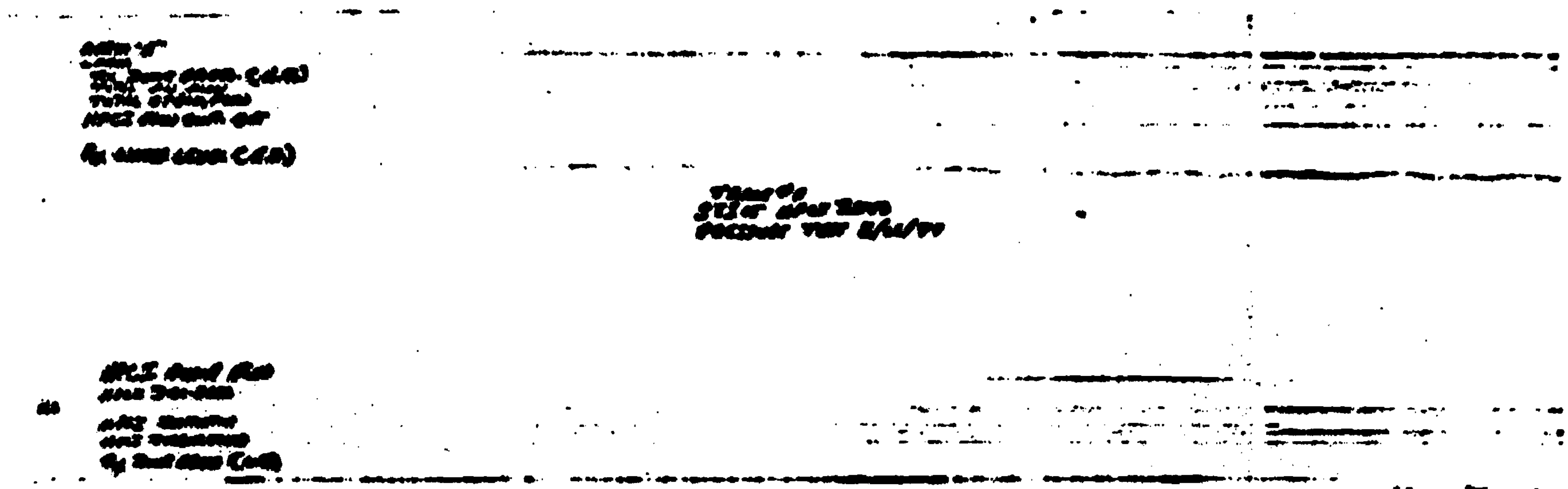

Figure 15-3A
HPCI Testing at Rated Pressure
Trace #11
1 second per division

3.0 Results (Continued)

3.2 Phase III - Initia
Heatup (Cont.)

3.2.6 STI-15, HPCI
System (Cont

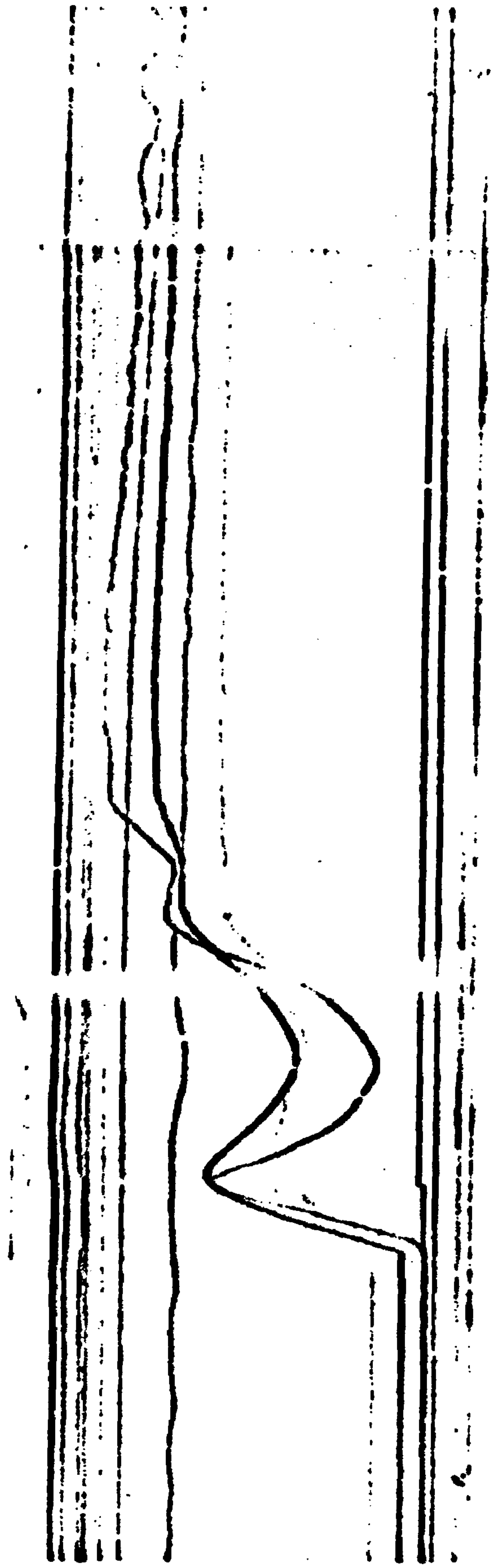

Figure 15-38
HPCI Tracing at Rated Pressure
Trace #11
1 second per division

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.6 STI-15, HPCI System (Cont.)

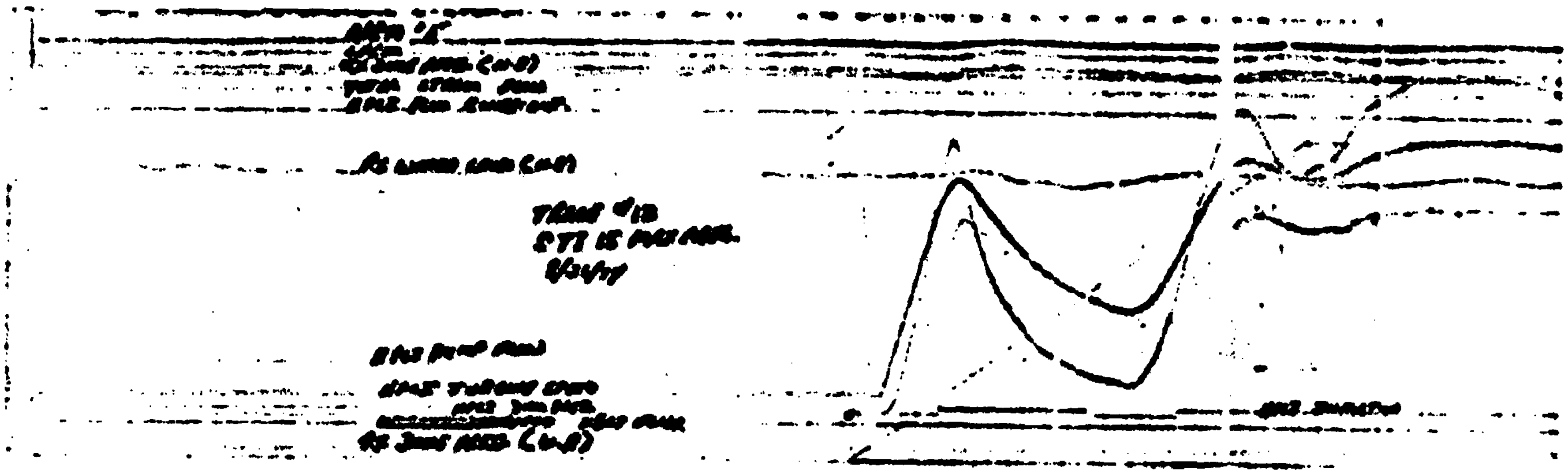

Figure STI 15-4A
HPCI Testing at Maximum Pressure
(Trace #12)
1 second per division

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.6 STI-15, HPCI System (Cont.)

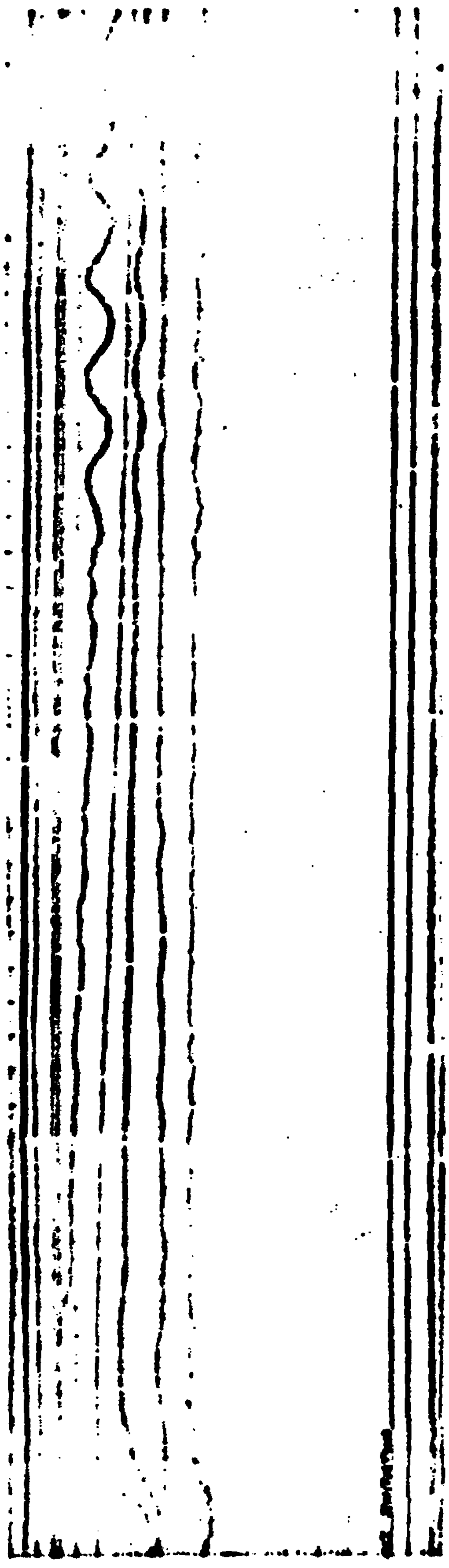

Figure 15-43
HPCI Testing at Maximum Pressure
(Trace #12)
1 second per division

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.7 STI-16, Selected Process Temperatures

Purpose

The purposes are:

1. To establish the proper setting for the low speed limiter for the recirculation pumps.

2. To provide assurance that the measured bottom head drain temperature corresponds to bottom head coolant temperature during normal operations.

Criteria

Level 1

The reactor recirculation pump shall not be operated unless the coolant temperatures in the upper and lower regions of the vessel are within 145° F. of each other.

Level 2

The bottom head coolant temperature as measured by the bottom drain line thermocouple should be within 50° F. of reactor coolant saturation temperature.

Analysis

Data was taken to verify the adequacy of the bottom drain thermocouple to monitor the bottom head temperature. The test data in table STI 16-1 shows that test criteria was met at minimum pump speed.

Table STI 16-1

Selected Process Temperatures

Date	Time	Recirculation Pump Discharge Temperature		Saturation Temperature	Reactor Bottom Drain Temperature
		A	B		
8-31-74	0500	530° F.	520° F.	535° F.	520° F.
	0510	530° F.	520° F.	535° F.	520° F.
	0520	530° F.	520° F.	535° F.	519° F.
	0530	530° F.	520° F.	535° F.	519° F.
	0540	530° F.	520° F.	534° F.	518° F.

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3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI 17, System Expansion

Purpose

The purpose of this test is to verify that the reactor drywell piping systems identified below and shown in the attached piping system isometric diagrams are free and unrestrained in regard to thermal expansion and that suspension components are functioning in the specified manner. The test also provides data for calculation of stress levels in nozzles and weldments.

Criteria

Level 1

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

Hangers shall not be bottomed out or have the spring fully stretched.

Hydraulic shock and sway arrestors shall be set to within ± 1" of the defined setting.

Level 2

Displacements of instrumented points with special recording devices shall not vary from the calculated values by more than ± 50% or ± 0.5 inches, whichever is smaller. Displacements of less than 0.25 inch can be neglected, since 50% of this value is bordering on the accuracy of measurement. If measured displacements do not meet these criteria, the system designer must be contacted to analyze the data with regard to design stresses.

The trace of the instrumented points during the heatup cycle shall be full 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.

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI 17, System Expansion

Criteria (Continued)

Level 2 (Continued)

Hangers shall be in their operating range (between the hot and cold settings).

Hydraulic shock and sway arrestors shall be within their operating range (between the hot and cold setting ± 1").

Conduit connections shall remain flexible (no tight linear or axial junctions).

Analysis

Initial Readings

Prior to initial nuclear heatup, hanger and hydraulic shock and sway arrestor readings were recorded for the drywell piping listed below. These readings will be compared with readings taken during a future shutdown to assure that the piping returns to its basic position.

Recirculation	- Reactor Water Cleanup
Steam	- HPCI
Feedwater	- RCIC
Core Spray	- RHR
CRD Hydraulic System Return Lines-	

The drywell piping in general was inspected for restrictions to free and unrestrained motion during thermal expansion. Minor restrictions were observed and corrected.

Drywell Piping Thermal Movement

Eighteen linear voltage differential transmitters (LVDT's) were installed in the drywell to record thermal movement of selected piping. The LVDT's provide a continuous, remote (outside drywell) readout of movements due to thermal expansion. Table STI 17-1 illustrates representative data from these special instruments.

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI 17, System Expansion

Criteria (Continued)

Level 2 (Continued)

Hangers shall be in their operating range (between the hot and cold settings).

Hydraulic shock and sway arrestors shall be within their operating range (between the hot and cold setting ± 1").

Conduit connections shall remain flexible (no tight linear or axial junctions).

Analysis

Initial Readings

Prior to initial nuclear heatup, hanger and hydraulic shock and sway arrestor readings were recorded for the drywell piping listed below. These readings will be compared with readings taken during a future shutdown to assure that the piping returns to its basic position.

Recirculation	- Reactor Water Cleanup
Steam	- HPCI
Feedwater	- RCIC
Core Spray	- RHR
CRD Hydraulic System Return Lines-	

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FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI 17, System Expansion

Analysis (Continued)

Drywell Piping Thermal Movement (Continued)

In general, the drywell piping moved in the correct direction during heatup and returned to its base setting after cooldown. However, the recirculation system loops both experienced interference problems and did not meet test criteria. Several drywell entries were required to locate and remove these problems and ascertain free and unrestrained movement capacity. Even after removing all visible interferences, both recirculation loops experienced thermal expansion not meeting the level II criterion. After ascertaining the systems to be free and unrestrained, the LVDT measurements were sent to the system designers for analysis of the data for the possibility of undue stresses on the systems. Their response was positive and the systems were declared "operable". These interferences along with those found on other systems and the corrective actions taken are described in table STI 17-2.

During the first two heatup-cooldown cycles, all major drywell piping hangers and hydraulic shock and sway arrestors visually inspected were found to return to their cold settings, indicating no deformation of the major drywell piping systems. For a summary of the hanger and hydraulic shock and sway arrestor deflection data for the first two heatup cycles, see tables STI 17-3 and STI 17-4. Drywell entry inspections during heatup indicated the selected hangers and hydraulic shock and sway arrestors were within their operating ranges. See table STI 17-5 and table STI 17-6 for a summary of this data. All levels I and II test criteria were satisfactorily met with regard to hanger and hydraulic shock and sway arrestor readings.

Drywell Inspection at Various Recirc. Temps.

During the first heatup, several drywell inspections were made (i.e., ambient, 300° F., rated coolant temperature) to perform the following:

1. Visually inspect selected hangers and hydraulic shock and sway arrestors to verify expected performance.

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3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI 17, System Expansion

Analysis (Continued)

Drywell Inspection at Various Recirc. Temps. (Cont'd)

2. Visually verify there are no unexpected constraints of system components.

3. Record selected hanger positions.

4. Record selected hydraulic shock and sway arrestor positions.

5. Inspect each LVDT displacement recording device to verify it is functioning properly.

6. Record the approximate X and Z displacements from those at ambient conditions.

7. Check the flexible conduit connection to the components to assure that system heat and expansion has not placed any linear or perpendicular strain on the flexible conduits.

Following any major plant cooldown to cold shutdown, the LVDT's were inspected and the cold settings recorded and above steps 1-4 were repeated.

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI 17, System Expansion (Continued)

Table STI 17-1

LVDT Data Summary

Monitored Location		Calculated Displacements	Measured Displacements	Absolute Difference	% Difference	Comments
Recirc Loop "A" (154)	X	-0.012	-0.069	0.057	-	X-Direction less than 1/4" displacement See Discussion
	Y	-1.743	-1.434	0.309	17.7	
	Z	-0.927	-0.382	0.545	58.8	
Recirc Loop "B" (322)	X	-1.635	-1.016	0.619	37.9	See Discussion
	Y	-1.701	-1.165	0.536	31.5	
	Z	+0.597	+0.108	0.489	81.9	
Main Steam "A" (23)	X	+2.520	+2.542	0.022	0.9	
	Z	+1.074	+0.871	0.203	18.9	
Main Steam "B" (72)	X	+2.258	+2.057	0.201	8.9	
	Z	+1.167	41.198	0.031	2.7	
Main Steam "C" ([illegible]60)	X	+1.704	+1.574	0.130	7.6	
	Z	-1.097	-1.203	0.106	9.7	
Main Steam "D" (34)	X	-1.891	+1.624	0.267	14.1	
	Z	-0.803	-0.523	0.280	34.9	
Feedwater "A" ([illegible]17)	X	+1.202	+1.069	0.133	11.1	System Temperature was 320° F.
	Z	+0.548	+0.321	0.227	41.4	
Feedwater "B" (36)	X	+0.728	+0.459	0.269	37.0	System Temperature was 320° F.
	Z	-0.279	-0.248	0.031	11.1	

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI 17, System Expansion (Continued)

Table STI 17-2
Interferences and Corrective Actions

Interference	Corrective Action
Main Steam Lines	**Main Steam Lines**
1. "C" line had insulation and air duct interference on relief valve PCV 1-41	1. Air duct modified
2. "A" line had insulation and relief valve tail pipe interference.	2. Insulation modified
Feedwater Lines	**Feedwater Lines**
"A" line had insulation and air duct interference	Modified insulation
Miscellaneous	**Miscellaneous**
1. Reactor Water Cleanup System had insulation and grating interference	1. Grating modified
2. Yarway column had mechanical shock suppressors locked up.	2. Seven new mechanical shock suppressors installed.
Recirculation System	**Recirculation System**
A. General 1. Seismic cables around the pumps too tight	A. General 1. Lengthened cables
2. Insulation and spacer bearing plates under both suction valves	2. Modified spacer bearing plate
B. "A" Line 1. Valve on bypass line for discharge valve had interference between insulation and catwalk grating.	B. "A" Line 1. Modified insulation
2. Bottom seismic ring on discharge riser	2. Removed choker
C. "B" Line 1. Discharge valve had interference between insulation and catwalk	C. "B" Line 1. Modified catwalk

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI 17, System Expansion (Continued)

Table STI 17-2 (Continued)
Interferences and Corrective Actions

Interference	Corrective Action
2. SS7 shock suppressor on suction line leaking oil	2. Repaired shock suppressor

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI 17, System Expansion (Continued)

Table STI 17-3
Hanger Deflection Data

System	Hanger Number	First Cold Reading 7/18/74	Cycle 1 8/19/74	Cycle 2 10/13/74	Cold Setting
Recirc. "A"	H-3	2 1/4	3 1/2	3	2 1/4
Recirc. "B"	H-3	2 1/4	3	2 3/4	2 1/4
Main Steam "A"	HA-2	1 5/8	1 11/16	1 3/4	1 3/4
Main Steam "B"	HB 2-2	1 3/4	1 3/4	1 3/4	1 7/8
Main Steam "C"	HC 4-2	1"	1"	7/8	7/8
Main Steam "D"	HD 3-2	1"	15/16	1	7/8
F. W. "A"	H-5	1 5/8	1 7/16	1 1/2	1 5/8
F. W. "B"	H-11	1 5/8	1 1/4	1 3/8	1 5/8
RHR "A"	H-2	1/2	1/2	1/2	11/16
RHR "B"	H-10	5/8	1/2	5/8	1/2
RHR H.S.	H-11	1/2	1/2	9/16	11/16
HPCI	H-48-2	2 1/4	2 3/8	2 1/4	*
RCIC	H-50	2 1/2	2 1/2	2 1/2	*
C. S. "A"	H-2	1 1/4	1"	1 1/4	1 3/8
C. S. "B"	H-5	1 1/8	1 5/16	1 1/8	*
RWCU	H-2	1 3/4	1 11/16	1 3/4	1 3/4
CRD	H-2	5	3	*	*

*Data not taken

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI 17, System Expansion (Continued)

Table STI 17-4
Hydraulic Shock and Sway Arrestor Data

System	Arrestor Number	1st Cold Reading 7/18/74	Cycle 1 8/19/74	Cycle 2 10/15/74	Cold Setpoint
Recirc. "A"	SS3	2 11/16	2 5/8	2 3/4	2 11/16
Recirc. "B"	SS3	2 11/16	2 5/8	2 3/4	2 11/16
M.S. "A"	SSA1	2 5/8	2 5/8	3	2 5/8
M.S. "B"	SSB2	4 3/4	4 5/8	4 3/4	4 3/4
M.S. "C"	SSC1	3	3 1/8	3	3
M.S. "D"	SSD2	3 1/4	3 3/4	3 3/4	3 3/4
F.W. "A"	SSA3	3 1/2	3 1/2	3 1/2	3 1/2
F.W. "B"	SSB3	3 1/2	4	3 7/8	3 1/2
RHR H.S.	R72	3 3/4	3 3/4	3 3/4	3 3/4
C.S. A	R-2	4	4 1/16	4	4
C.S. B	R-8	3 3/4	3 3/4	3 3/4	3 3/4
CRD	R-1	2	2 1/2	2 3/4	2
HPCI	R-6	5	5	5 1/4	5
Recirc. "A"	SS8	2 13/16	2 5/8	2 7/8	2 13/16
Recirc. "B"	SS7	2	2 1/8	2 1/8	2

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI 17, System Expansion (Continued)

Table STI 17-5
Drywell Entry Hanger Data Summary

System	Hanger Number	Cold Setpoint	Cold Data	Interm. Data	Hot St'by Data	Hot Setpoint
Recirc. "A"	H-3	2 1/4	2 1/4	4 1/2	7 1/4	7 3/4
Recirc. "B"	H-3	2 1/4	2 1/4	4 1/4	6 1/2	7 3/4
Main Steam "A"	HA-2	1 3/4	1 5/8	1 1/2	1 3/8	1 3/8
Main Steam "B"	HB-2-2	1 7/8	1 3/4	1 3/4	1 1/2	1 5/8
Main Steam "C"	HC-4-2	7/8	1	7/8	3/8	3/8
Main Steam "D"	HD-3-2	7/8	1	3/4	3/8	3/8
Feedwater "A"	H-5	1 5/8	1 5/8	1 1/8	3/4	9/16
Feedwater "B"	H-11	1 5/8	1 5/8	1 1/4	3/8	9/16
RHR Return "A"	H-2	11/16	5/8	3/4	15/16	15/16
RHR Return "B"	H-10	1/2	1/2	1/2	3/4	3/4
HPCI Steam Supply	H-48-2	*	2 1/4	2 1/4	2 3/4	*
RCIC Steam Supply	H-50	*	2 1/2	2 3/4	+	*
RHR Head Spray	H-11	11/16	1/2	1/2	3/8	5/16
Core Spray "A"	H-2	1 3/8	1 1/4	7/8	1/2	5/8
Core Spray "B"	H-5	*	1 1/8	3/4	3/8	11/16
Cleanup	H-2	1 3/4	1 3/4	1 1/2	1 3/8	1 1/8
CRD	H-2	*	5	4 1/2	2 1/2	*

+Impossible to read during hot operation
*No hot and/or cold setpoints visible on scale

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI 17, System Expansion (Continued)

Table STI 17-6
Drywell Entry-Shock and Sway Arrestor Data Summary

* System	Arrestor Number	Cold Setpoint	Cold Data	300° F.	500° F.	Set
Recirc. "A"	SS3	2 11/16	2 11/16	2 1/2	2	2 5
Recirc. "A"	SS8	2 13/16	2 13/16	3	3 1/4	3 1
Recirc. "B"	SS3	2 11/16	2 11/16	2 1/2	2 1/4	2 5
Recirc. "B"	SS7	2	2	2 3/4	3	4 1
M.S. "A"	SSA1	2 5/8	2 5/8	3	4 3/4	4 3
M.S. "B"	SSB2	4 3/4	4 3/4	4 3/4	3 1/2	2 3
M.S. "C"	SSC1	3	3	3 1/8	4 3/4	4 1
M.S. "D"	SSD1	2 5/8	2 5/8	4	4 3/4	4 1
F.W. "A"	SSA3	3 1/2	3 1/2	3 7/8	3	4
F.W. "B"	SSB3	3 1/2	3 1/2	4	4 1/4	4
RHR M.S.	R-72	3 3/4	3 3/4	3 3/4	4	3
C.S. A	R-2	3 3/4	4	4 1/8	4 3/4	
C.S. B	R-8	3 3/4	3 3/4	3 3/4	4	
CRD	R-1	2	2	2 1/4	2 1/2	3
HPCI	R-6	5	5	5 1/8	5	

*Data Not Taken

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI 17, System Expansion (Continued)

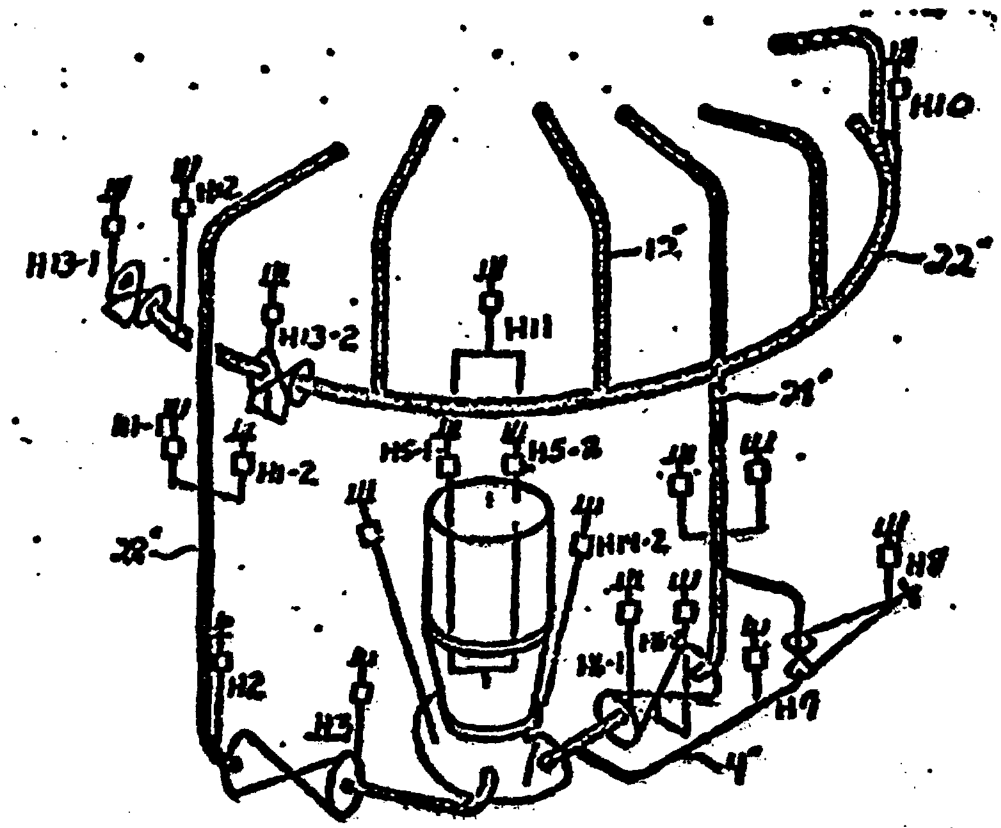


Figure STI 17-1

Recirculation Loop A

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI-17, System Expansion (Continued)

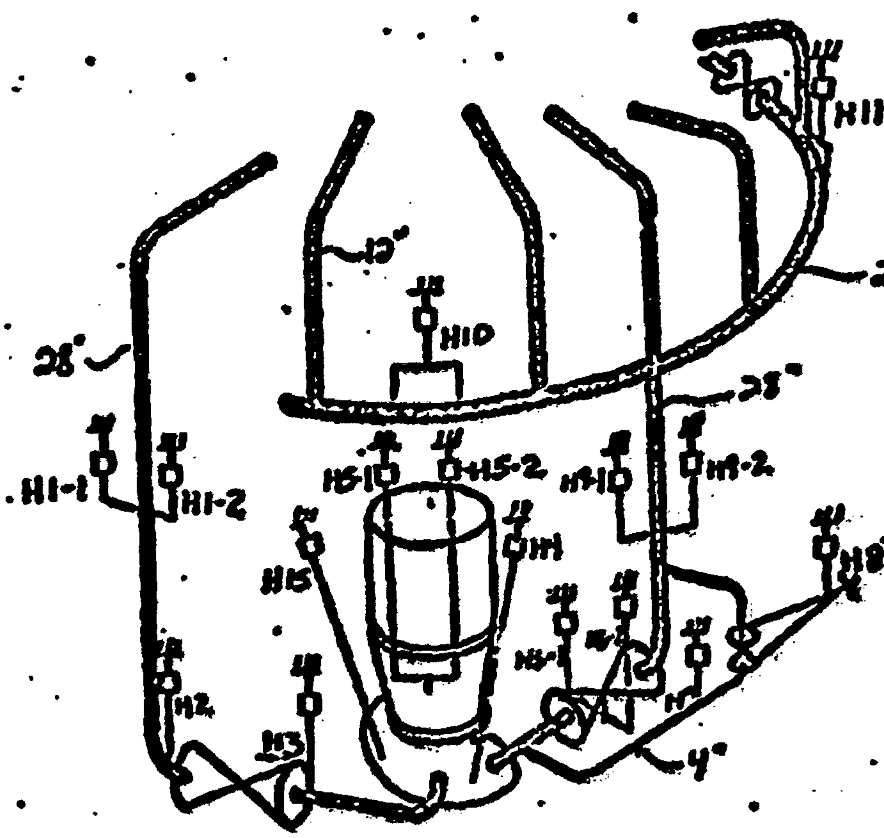


Figure STI 17-2

Recirculation Loop B

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI-17, System Expansion (Continued)

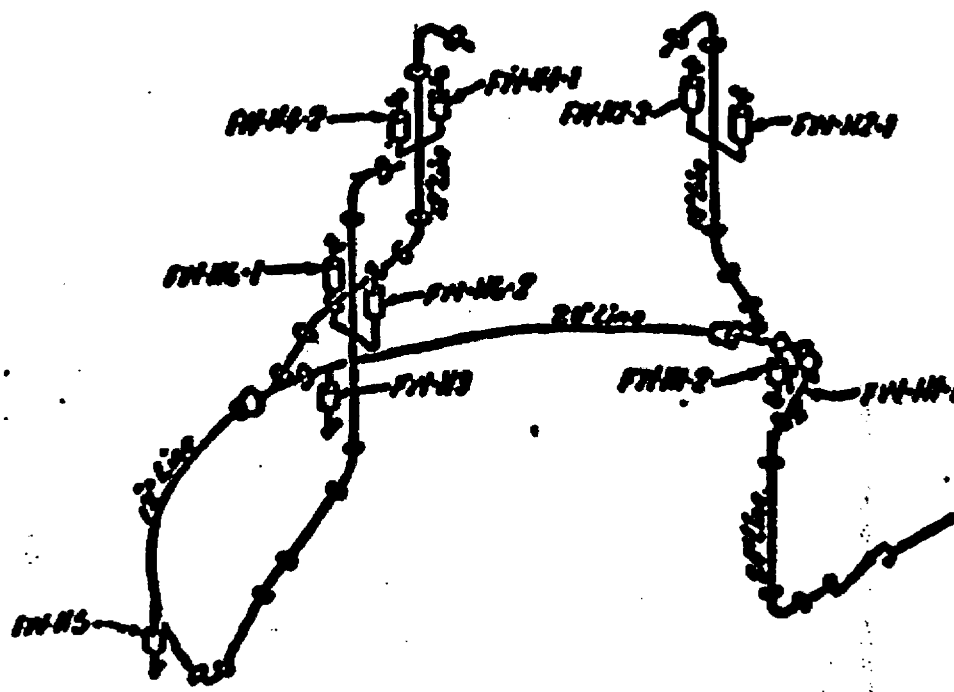


Figure STI 17-3

Feedwater Line A

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI-17, System Expansion (Continued)

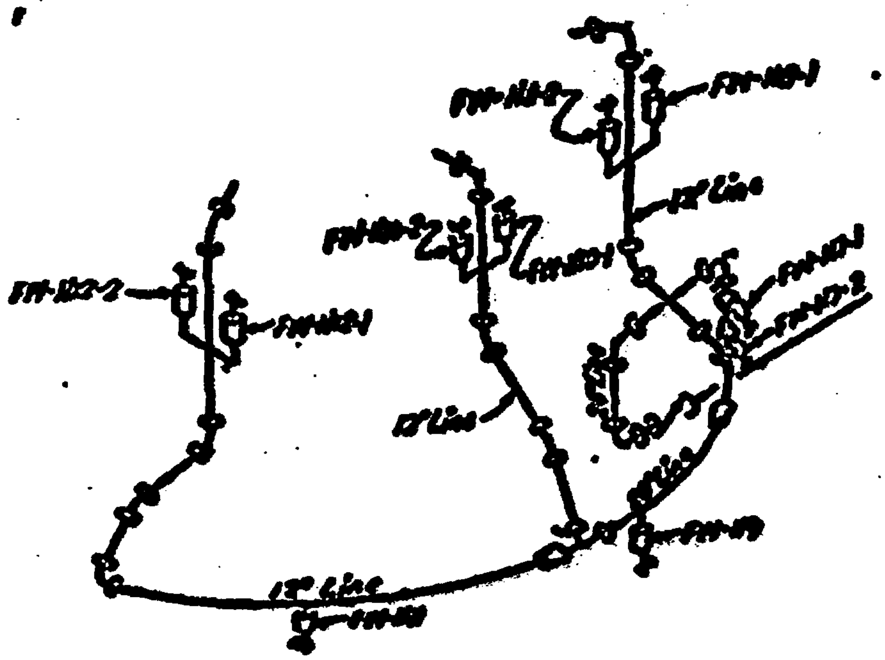


Figure STI 17-4

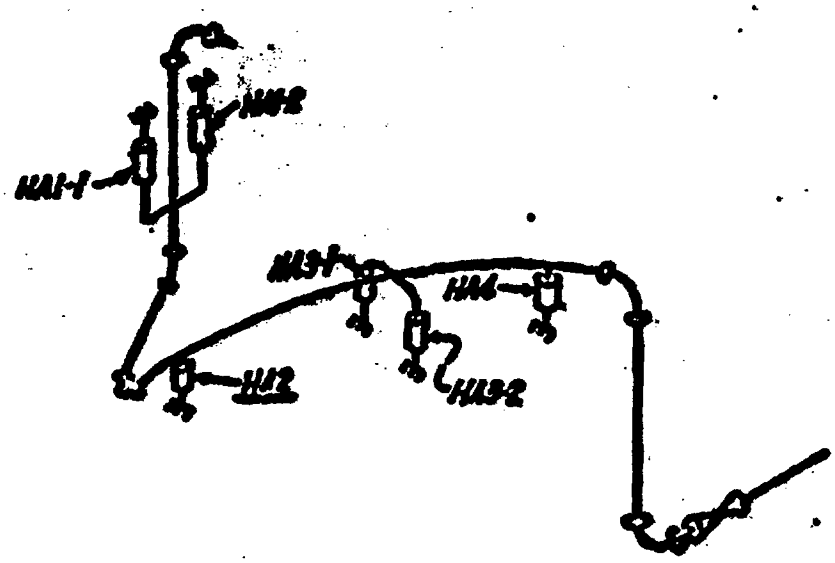
Feedwater Line B

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI-17, System Expansion (Continued)



LINE ROUTING - SKETCH

Figure STI 17-5

Main Steam Line A

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI-17, System Expansion (Continued)

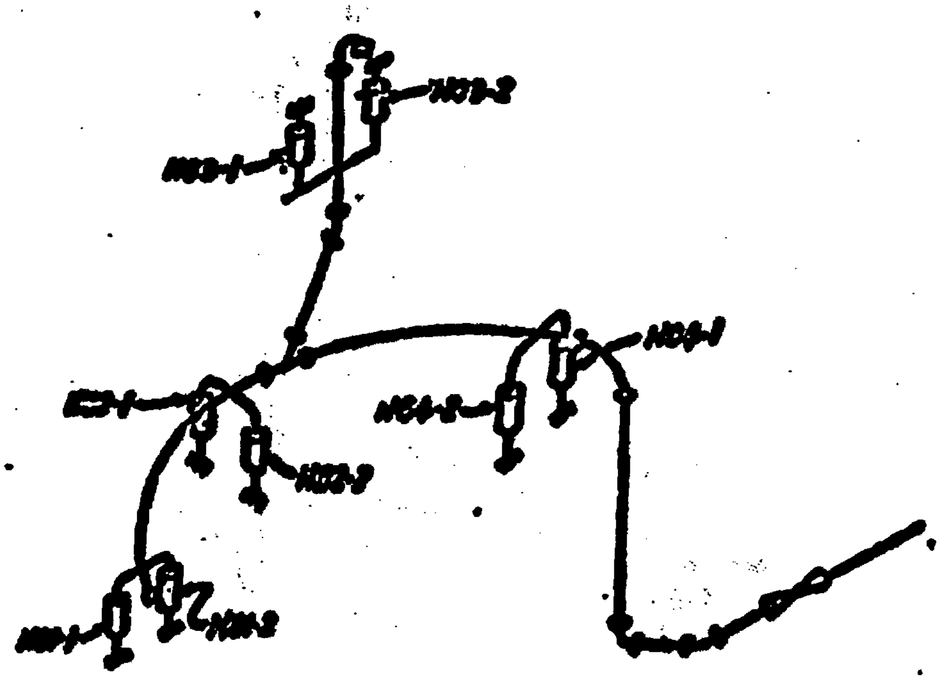


Figure STI 17-6

Main Steam Line B

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Startup (Continued)

3.2.8 STI-17, System Expansion (Continued)

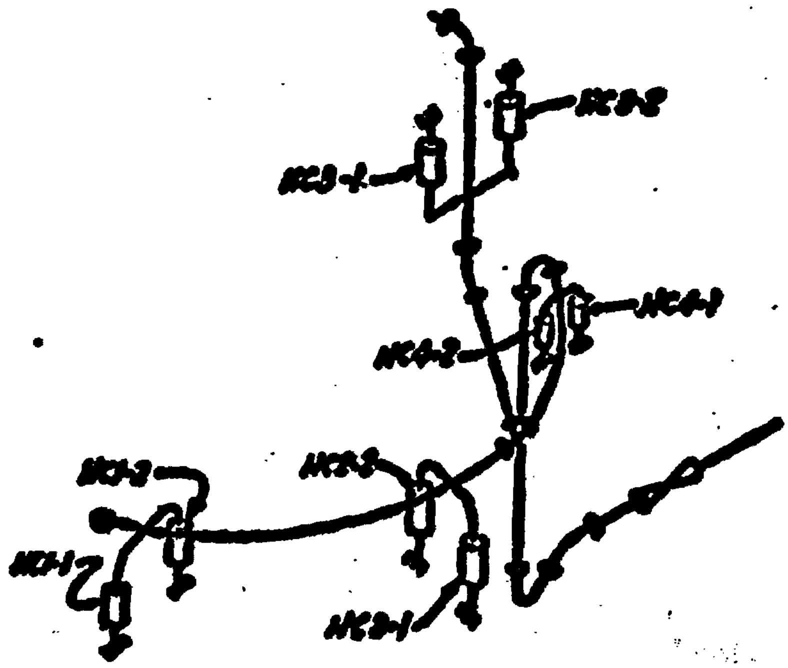


Figure STI 17-7

Main Steam Line C

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI-17, System Expansion (Continued)

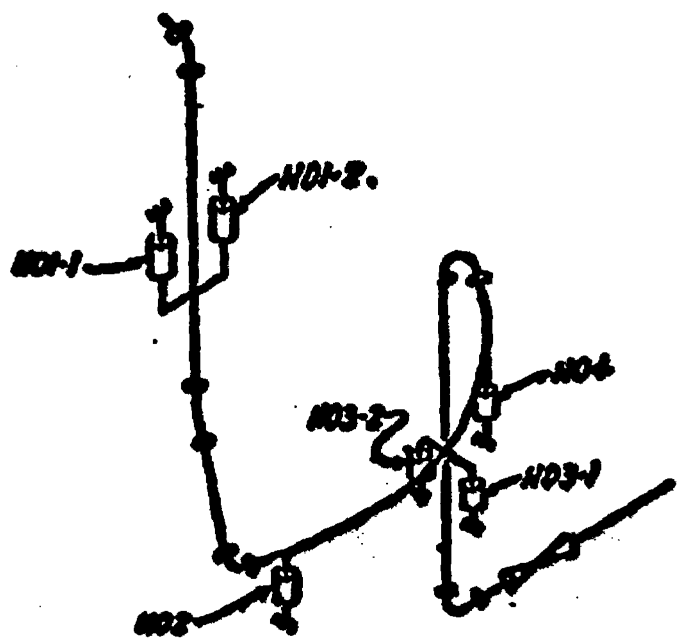


Figure STI 17-8

Main Steam Line D

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI-17, System Expansion (Continued)

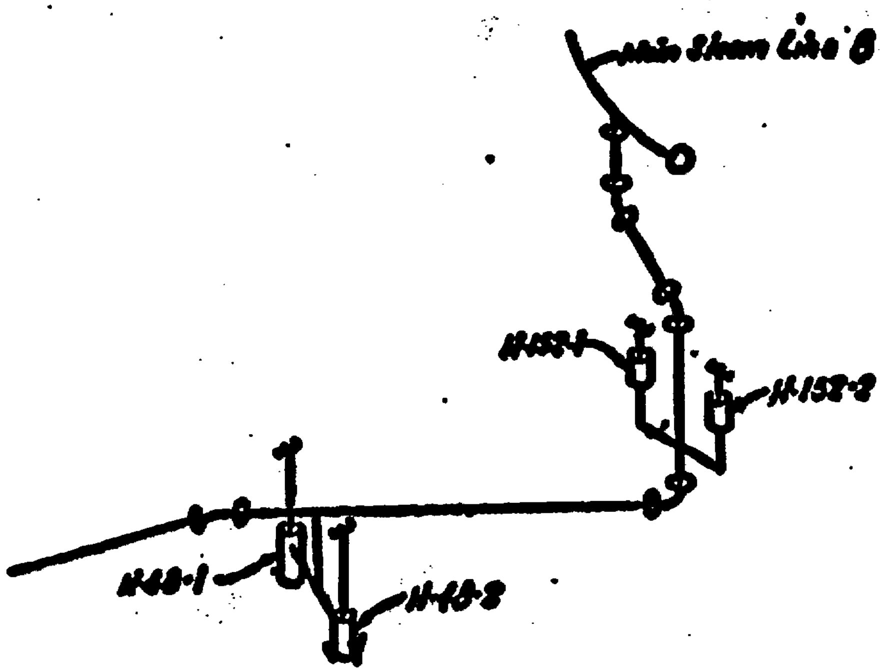


Figure STI 17-9

HPCI Steam Supply

FINAL SUMMARY REPORT - WNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI-17, System Expansion (Continued)

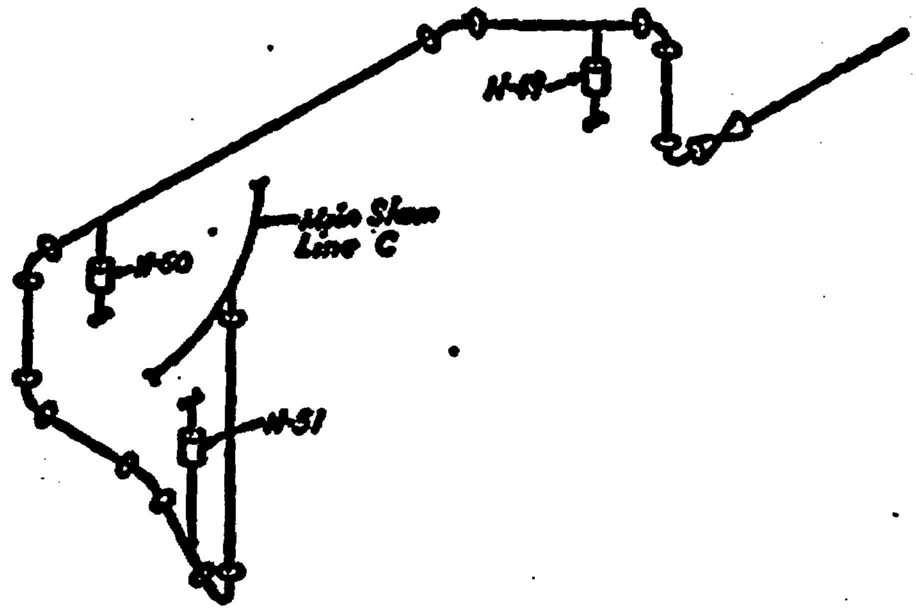


Figure STI 17-10

RCIC Steam Supply

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI-17, System Expansion (Continued)

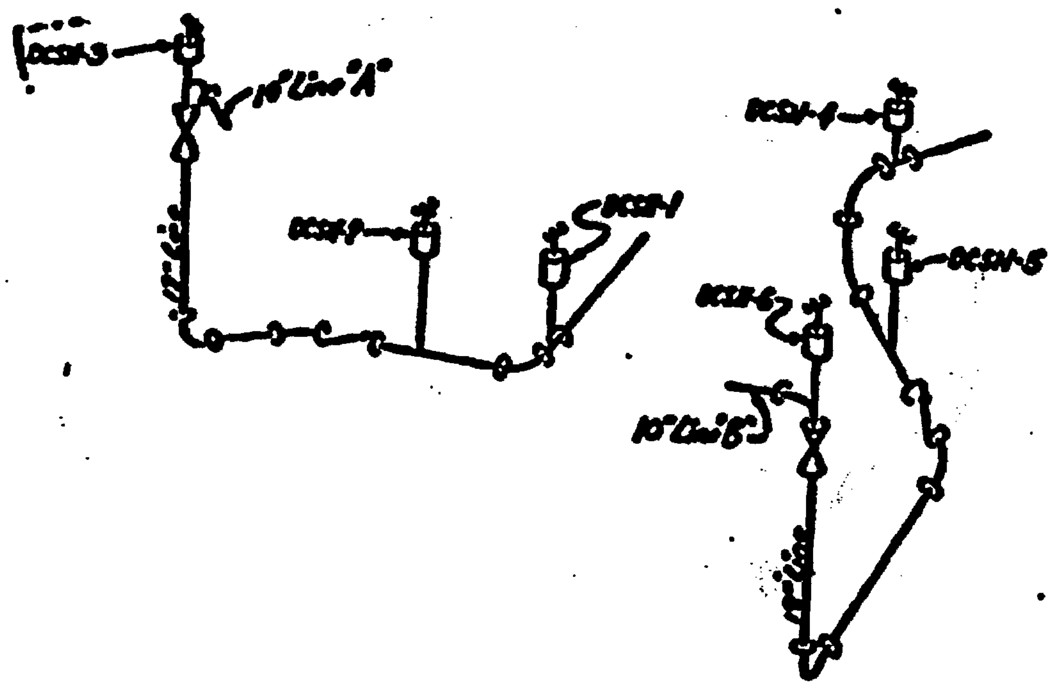


Figure STI 17-11

Core Spray

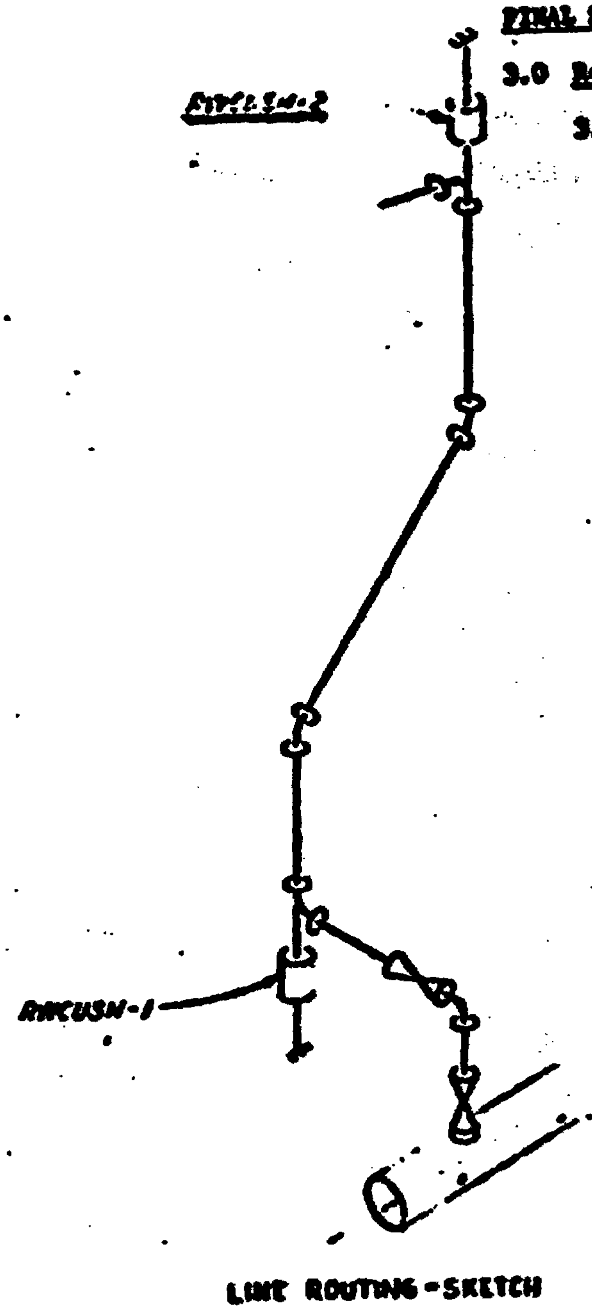


Figure STI 17-12

RHCU System Supply

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI-17, System Expansion (Continued)

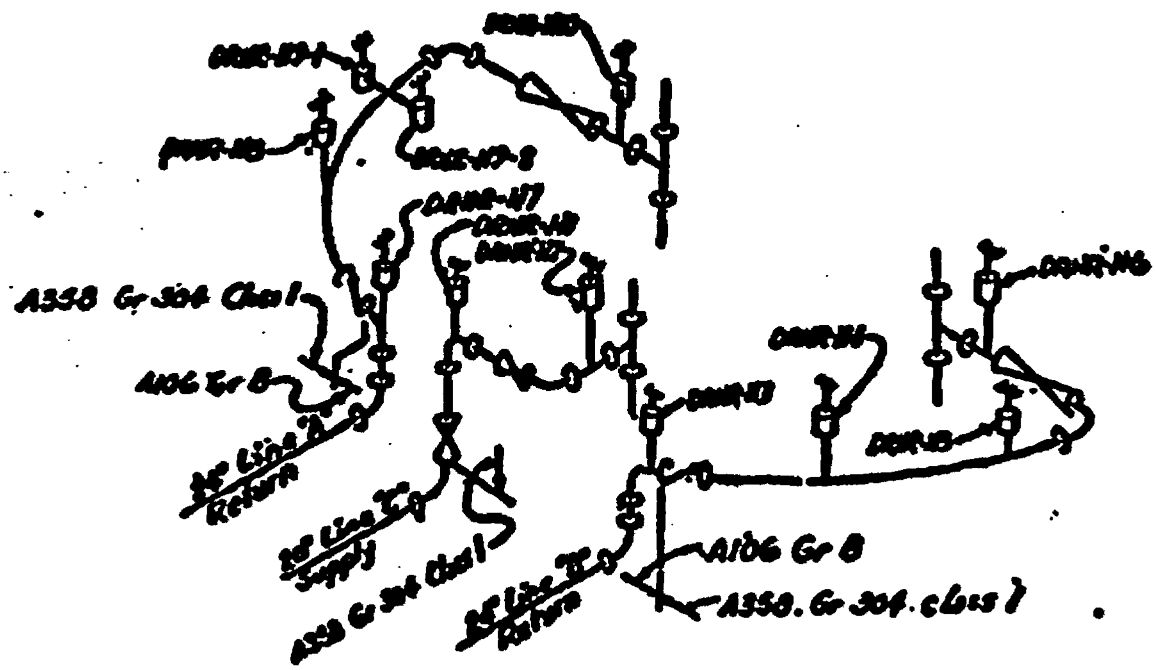


Figure STI 17-13

RHR Supply and Return

FINAL SUMMARY REPORT - WNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI-17, System Expansion (Continued)

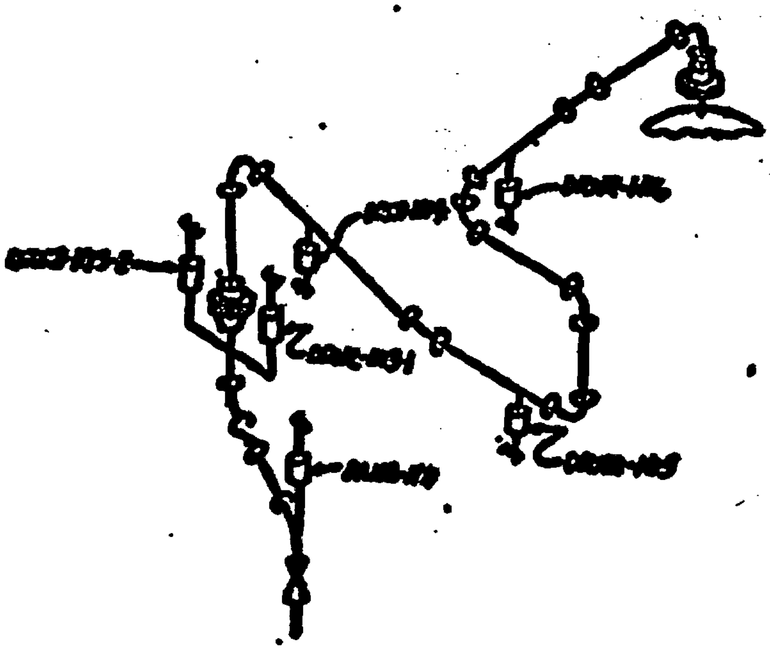


Figure STI 17-14

RHR Head Spray

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.8 STI-17, System Expansion (Continued)

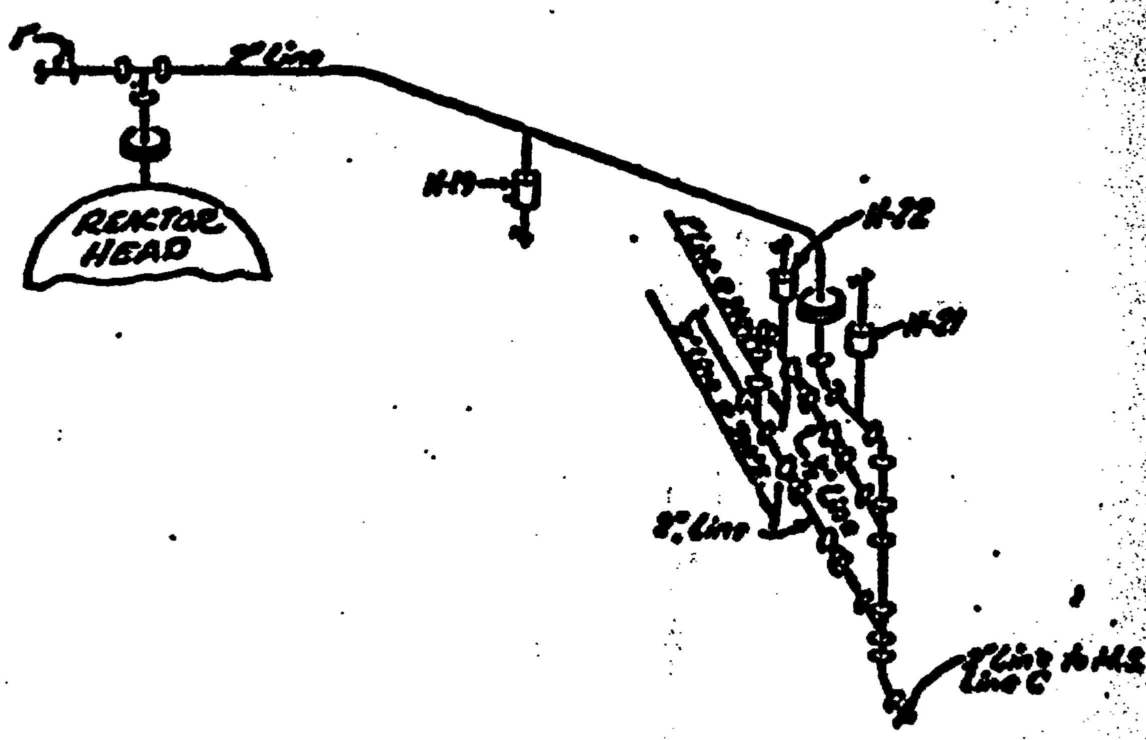


Figure STI 17-15

RHR Head Vent

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.9 STI-25, Main Steam Isolation Valves

Purpose

The purposes of this test are to: (a) functionally check the main steamline isolation valves (MSIV) for proper operation at selected power levels; (b) determine reactor transient behavior during and following simultaneous full closure of all MSIV and following closure of one valve; and (c) determine isolation valve closure time.

Criteria

Level 1

Closure time must be greater than 3 and less than 5 sec. Reactor pressure shall be maintained below 1230 psig (the setpoint of the first safety valve) during the transient following closure of all valves.

Level 2

The maximum reactor pressure should be 1190 psig, 40 psi below the first safety valve setpoint following closure of all valves. This is a margin of safety for safety valve weeping. During full closure of individual valves, pressure must be 20 psi below scram, neutron flux must be 10% below scram, and steam flow in individual lines must be below the trip point.

Analysis

The performance of this test at conditions typical of the heatup phase is merely to demonstrate the operability of the MSIV's. The only applicable criteria at this test condition is that all MSIV's close within the 3-5 second limit. Table STI 25-1 indicates the operational history of the MSIV's during the initial heatup phase. Each of the valves satsifactorily met the restriction as evidenced by the data in table STI 25-2.

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.9 STI-25, Main Steam Isolation Valves (Continued)

Table STI 25-1

MSIV Operational History During Initial Heatup

Date	MSIV	Remarks
8/12/74	FCV 1-38	Too Slow (5.2 sec.)
	FCV 1-51	Too Slow (6.0 sec.)
8/13/74	FCV 1-38	Closing Time Good
	FCV 1-51	Too Slow (6.1 sec.)
	FCV 1-37	Too Slow (5.1 sec.)
8/14/74	FCV 1-37	Closing Time Good
	FCV 1-51	Closing Time Good
	FCV 1-52	Would Not Open
	FCV 1-38	Too Fast (2.4 sec.)
8/21/74	FCV 1-52	Repaired
	FCV 1-27	Would Not Open
8/24/74	FCV 1-27	Repaired
8/26/74	All MSIV's	Closure Time Good

Table STI 25-2

MSIV Closure Time

MSIV Valve Number	Closure Time
FCV-1-14 (1A)	4.4
FCV-1-15 (2A)	4.1
FCV-1-26 (1B)	4.1
FCV-1-27 (2B)	4.4
FCV-1-37 (1C)	4.1
FCV-1-38 (2C)	4.1
FCV-1-51 (1D)	4.2
FCV-1-52 (2D)	3.9

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.10 STI-26, Relief Valves

Purpose

The purpose of this test is to verify that all relief valves may be manually opened and that they function correctly.

Criteria

Level 1

None

Level 2

Relief valve leakage must be low enough so that the temperature measured by the thermocouples in the discharge side of the valves returns to within 10° F. (5.6° C.) of the temperature recorded before the valve was opened.

NOTE: Other criteria not applicable for phase III.

Analysis

All main steam relief valves were manually actuated from the main control room and each ADS valve was actuated from the backup panel (25-32). Each valve functioned properly. Valve PCV 1-23 failed, by 33° F., to return to the "initial tailpipe temperature + 10° F." and PCV 1-18 had a failed thermocouple. Retest of both valves resulted in satisfactory performance of PCV 1-18 and a final tailpipe temperature on PCV 1-23 of 3° F. over the "initial temperature + 10° F." During phase IV testing of PCV 1-23, the final tailpipe temperature returned to within 10° F. of the initial temperature. Even though there was a minor deviation from the level 2 test criteria for PCV 1-23, proper valve operation was demonstrated

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.11 STI-70, Reactor Water Cleanup

Purpose

The purpose of this test is to demonstrate specific aspects of the mechanical operability of the reactor water cleanup system at or near rated reactor temperature and pressure.

Criteria

Level 1

None

Level 2

(a) The temperature at the tube side outlet of the non-regenerative heat exchangers shall not exceed 130° F. in any mode.

(b) The pump available NPSH will be 13 feet or greater during the hot standby mode defined in the process diagrams.

(c) The cooling water supplied to the non-regenerative heat exchangers shall be within the flow and outlet temperature limits indicated in the process diagrams. (This is applicable to "normal" and "blowdown" modes.)

Analysis

Three tests were performed to demonstrate the heat capacities of the regenerative and non-regenerative heat exchangers. The first test was conducted in the "hot standby" mode in which all cleanup flow was returned to the reactor with no bypass flow. With a cleanup flow of 0.132×10^6 lb./hr., a heat removal rate of 16.8×10^6 Btu/hr. was obtained. This compares well with the design figures of 0.14×10^6 lb./hr. and 15.8×10^6 Btu/hr.

The second test was run in the "normal" mode in which all cleanup flow was returned to the reactor with no bypass flow. With a cleanup flow of 0.13×10^6, lb/hr, a heat removal rate of 17.2×10^6 Btu/hr. was obtained. This compares well with the design figures of 0.14×10^6 lb./hr. and 15.8×10^6 Btu/hr.

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.11 STI-70, Reactor Water Cleanup (Continued)

The third test was run in the "blowdown" mode in which all cleanup flow was discharged to radwaste or the condenser. With a cleanup flow rate of 0.052×10^6 lb./hr., a heat removal rate of 21.9×10^6 Btu/hr. was obtained. This compares well with the design figures of 0.053×10^6 lb./hr. and 22.5×10^6 Btu/hr.

Since the measured value for pump inlet temperature is so far below the process diagram, the process diagram value of 545° F. was used in the calculation of NPSH in order to be conservative.

$$NPSH = P_s - P_v + \frac{V^2}{2g} \qquad \left[\frac{V^2}{2g} \text{ is negligible and therefore ignored}\right]$$

$$P_s = 1019 \quad P_v = 1003$$

$$NPSH = (1019\text{-}1003)\ \text{lb./in.}^2 \left(\frac{144\ \text{in.}^2}{\text{ft}^2}\right)\left(\frac{0.0216\ \text{ft}^3}{\text{lb.}}\right) =$$

$$50\ \text{ft.} @ 545^\circ \text{F.} = 37\ \text{ft.} @ 68^\circ \text{F.}$$

During all three tests, the cleanup filter inlet temperature was held below 130° F., the NPSH was determined to be 37 feet and the inlet and outlet temperature of the cooling water supplied to the NRHX's was held within limits, thus satisfying level 2 criteria.

The cleanup system was tested near rated temperature and pressure. It was found that the regenerative heat exchanger capacity was 29.1×10^6 Btu/hr. at a flow of 1.3×10^5 lb./hr.

The non-regenerative heat exchanger capacity was 17.2×10^6 Btu/hr. and had a maximum exit temperature of 145° F.

All values were close to the process diagram values and are satisfactory with the normal and blowdown test values being marginal.

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.11 STI-70, Reactor Water Cleanup (Continued)

Plant Conditions

Test Condition: Heatup - All Tests
Date Performed: 8/26/74 - All Tests
Reactor Power: 3-3-70
Core Flow: 39 MPPH
Reactor Pressure: Hot Standby 996.5 psig
Normal 1002.0 psig
Blowdown 1002.0 psig

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.12 STI-71, RHR System

Purpose

The purpose of this test is to demonstrate the ability of the residual heat removal (RHR) system to: (a) remove residual and decay heat from the nuclear system so that refueling and nuclear system servicing can be performed, and (b) remove heat from the suppression pool water.

Criteria

Level 1

The heat removal capability of each RHR heat exchanger in the "Shutdown Cooling" mode or the "Suppression Pool Cooling" mode shall be ~18.7 x 10^7 Btu/hr. (~4.69 x 10^7 Kcal/hr.) or greater.

Analysis

Several fruitless attempts were made to demonstrate the capacity of the RHR heat exchangers using the "Shutdown Cooling Mode." Each attempt failed to produce sufficient data to quantitatively demonstrate the capacity of the heat exchangers due to the vessel cooldown rate limit of 100° F./hr., or because of the time required by the operational procedure to flush the RHR system lines following cooling of the reactor system to less that 122 psig. (i.e., ~352° F.).

The "Suppression Pool Cooling" mode of operation was finally used to obtain the data necessary to compute the heat exchanger capacities. These values, when corrected to the design operating conditions are shown in table STI 71-1. All criteria were met.

Table STI 71-1
RHR Heat Exchanger Capacities

Heat Exchanger	Capacity (MBtu/hr.)	Level 1 Criteria
A	358.7	187 MBtu/hr.
B	405	187 MBtu/hr.
C	218	187 MBtu/hr.
D	293.5	187 MBtu/hr.

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3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.13 STI-72, Drywell Atmosphere Cooling System

Purpose

The purpose of this test is to verify the ability of the drywell atmosphere cooling system to maintain design conditions in the drywell during operating conditions.

Criteria

Level 1

None

Level 2

(a) The heat removal capability of the drywell coolers shall be approximately 3.19×10^6 Btu/hr.

(b) The drywell cooling system shall have a standby capability of $\geq$15% of the design heat removal capability.

(c) The drywell cooling system shall maintain temperatures in the drywell below the following design values during normal operation.

During normal reactor operation:

135° F. (57° C.) average throughout drywell

50 percent relative humidity

128° F. (53.4° C.) maximum around the recirculating pump motors

150° F. (65.5° C.) maximum for all other areas

200° F. maximum above the bulkhead

Ten hours after shutdown:

within 15° F. (8.3° C.) of closed cooling water inlet temperature in all areas beneath the vessel-to-drywell bulkhead

Cooling water supply:

100° F. maximum

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3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.13 STI-72 Drywell Atmosphere Cooling System (Continued)

Criteria (Continued)

Level 2 (Continued)

(g) Uniform circumferential temperature at which the refueling bellows/bulkhead assembly must be maintained:

within 25° F. maximum point-to-point variation

Analysis

Data recorded during heatup indicated that all temperatures in the drywell met criteria, except for TE 80-13 and TE 80-14. Since no equipment is located near these thermocouples, BED approved proceding to the next test plateau.

The drywell heat load was within design specifications at this test level.

3.2.14 STI-73 Cooling Water Systems

Purpose

The purpose of this test is to verify that the performance of the reactor building closed cooling water (RBCCW) system is adequate.

Criteria

Level 1

None

Level 2

(a) Verification that the system performance meets the cooling requirements constitutes satisfactory completion of this test.

(b) The RBCCW was designed to transfer a maximum heat load of 31.3 x 10^6 Btu/hr. in order to limit equipment inlet water temperature to 100° F. assuming a service (raw cooling) water inlet temperature of 90° F.

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.2 Phase III - Initial Heatup (Continued)

3.2.14 STI-73 Cooling Water Systems (Continued)

Analysis

The heat load on the RBCCW system was within the design heat load of 31.3 x 10^6 Btu/hr. at hot standby. The only problem encountered at this test condition were tube rattles in the RBCCW heat exchangers. Raising the raw cooling water flow rates eliminated the problem of tube rattle.

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3.0 Results (Continued)

3.3 Phase IV - Power Testing

3.3.1 STI-1, Chemical and Radiochemical

Purpose

The principal objectives of this test are:

1. To secure information on the chemistry and radiochemistry of the reactor coolant.

2. To determine that the sampling equipment, procedures, and analytical techniques are adequate to supply the data required to demonstrate that the chemistry of all parts of the entire reactor system meet specifications and process requirements.

3. Specific objectives of the test program include evaluation of fuel performance, evaluation of demineralizer operations by direct and indirect methods, measurement of filter performance, confirmation of condenser integrity, demonstration of proper steam separator-dryer operation, measurement and calibration of the off-gas system, and calibration of certain process instrumentation. Data for these purposes is secured from a variety of sources: plant operating records, regular routine coolant analysis, radiochemical measurements of specific nuclides, and special chemical tests.

Criteria

Level 1

(a) Chemical factors defined in the technical specifications must be maintained within the limits specified.

(b) The activity of gaseous and liquid effluents must conform to license limitations.

Level 2

Water quality must be known and should remain within the guidelines of GE water quality specifications.

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3.0 Results (Continued)

3.3 Phase IV - Power Test (Continued)

Analysis

Table STI 1-1 summarizes the results of the chemical and radiochemical testing performed during startup.

Table STI 1-1

Sample Source and Test		15-35% Power	40-60% Power	65-85% Power	95-100% Power
	Date	9/1/74	9/30/74	11/22/74	12/19/74
	MWt	780	1953	2325	3050
Reactor Water	Limit \ MWe	206	599	755	1026
Conductivity, umho/cm	1.0	.90	0.5	0.20	0.23
Chloride, ppm	0.2	<.050	<.050	<.050	<.050
Turbidity or insolubles, ppm	10	0.25 JTU	0.11 JTU	2.0E-03JTU	0.001 JTU
Iodine-131, Ci/ml		1.4 E-06	1.7 E-06	2.25 E-06	1.53 D-06
Iodine-133, Ci/ml		1.25E-05	2.8 E-05	3.06 E-05	4.9 E-05
Gross Activity					
-filtrate, cpm/ml, 2 hrs.		9.61E+03	1.93E+04	1.43 E+04	2.14 E+04
-crud, cpm/ml, 2 hrs.		4.26E+02	7.4 E+02	3.46 E+02	1.34 E+03
Gross Activity					
-filtrate, cpm/ml, 7d		6.5 E+01	1.95E+02	2.58 E+02	3.65 E+02
-crud, cpm/mg Fe, 7d		3.0 E+00	2.7 E+01	2.1 E+01	1.18 E+02
Silica, ppb	5.0 ppm	4.08E+02	2.24E+02	3.42 E+02	3.31 E+02
Boron, ppb	50 ppm	1.85E+02	1.49E+02	8.8 E+01	2.05 E+02

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3.0 Results (Continued)

3.3 Phase IV - Power Tests (Continued)

Table STI 1-1 (Continued)

Sample Source and Test		15-35% Power	40-60% Power	65-85% Power	95-100% Power
	Date	9/1/74	9/30/74	11/22/74	12/19/74
	MWt	780	1553	2325	3050
Reactor Water (Continued)	Limit / MWe	206	599	755	1026
Chemical Analysis on filtrate, ppb					
-iron		XX	XX	XX	16.2
-copper		XX	XX	XX	13.0
-nickel		XX	XX	XX	<0.01
-chromium		XX	XX	XX	12.4
Chemical Analysis on crud, ppb					
-iron		3.95	7.0	2.22	3.9
-copper		XX	XX	XX	<0.27
-nickel		XX	XX	XX	<0.46
-chromium		XX	XX	XX	1.7
Spectral Analysis on major nuclides at 24 hours					
Filtrate		Co-58 Cr-51 Cu-64 Mo-99 W-187 Mn-54 Mn-56 Na-24 Zn-69m As-76 Sb-122 Nb-97 Zr-97	As-76 Cr-51 Sb-122 W-187 Cu-64 Nb-97	Co-60 Na-24 Cr-51 Mn-54 Co-58 Cu-74 Zn-65 Tc-99m W-187	Co-60 Na-24 Cr-51 Mn-54 Cu-64 I-131 As-76 W-187 Mo-99 Co-58

XX symbol signifies data not required by the test instruction.

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.3 Phase IV - Power Tests (Continued)

Table STI 1-1 (Continued)

Sample Source and Test		15-35% Power	40-60% Power	65-85% Power	95-100% Power
	Date	9/1/74	9/30/74	11/22/74	12/19/74
	MWt	780	1953	2325	3030
	Limit / MWe	206	599	755	1026
Crud		Mn-56 Cu-64 Sb-122 W-187 As-76 Zr-97 Nb-95 Mn-54	Cr-51 Co-58 Co-60 Mn-54 W-187 As-76 Zr-97	Cr-51 Mn-54 Co-58 Co-60 Cu-64 Zr,Nb-95 Zr,Nb-97 W-187 As-76 Sb-124 Fe-59	Cr-51 Mn-54 Co-58 Co-60 Zn-65 Zr,Nb-95 Zr,Nb-97 W-187 Cu-64 As-76 Sb-124
Condensate Demin. Influent					
Conductivity, umho/cm		0.26	0.12	0.084	0.060
Chloride, ppm		<.050	<.050	<.050	<.050
Insoluble iron, ppb		6.2	163	27.49	130.4
Condensate Demin. Effluent					
Conductivity, umho/cm	0.1	0.11	0.08	0.065	0.058
Insoluble iron, ppb	20	1.48	0.73	3.3	2.48
Oxygen, ppb	14	Lab 15 Anal.	Lab 300 Anal.	Lab 10 Anal.	Lab 10 Anal.
Feedwater					
Conductivity, umho/cm	0.10	0.09	0.078	0.070	0.058
Iron - insoluble, ppb		4.13	2.01	8.01	1.21
- soluble, ppb		XX	0.53	2.21	1.21
Nickel - insoluble, ppb		XX	XX	XX	0.01
- soluble, ppb		XX	XX	XX	0.29

XX Symbol signifies data not required by the test instruction.

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.3 Phase IV - Power Tests (Continued)

Table STI 1-1 (Continued)

Sample Source and Test			15-35% Power	40-60% Power	65-85% Power	95-100% Power
		Date	9/1/74	9/30/74	11/22/74	12/19/74
		MWt	780	1953	2325	2050
Feedwater (Continued)		Limit \ MWe	206	599	755	1026
Copper - insoluble, ppb			XX	XX	XX	0.01
- soluble, ppb			XX	XX	XX	0.08
Chromium - soluble, ppb					Crud Sol	XX XX
Off-Gas						
Activity @ SJAE, µCi/sec. (Σ6 gases)			9.36E-04	44	<87.4	<128.4
N-13 @ SJAE, µCi/sec.			133	240	1450	3227
Flow rate, cfm (FR-66-20)			60	116	80	150
Composition - air, cfm			40	86	12.4	16.2
$-(H_2 + O_2)$			20	30	67.6	133.8
Delay time, min.			133.3	70.2	100	53.3
Activity release at stack µCi/sec.			108 (1)	10	13.3	64.6
Activity Pattern			Recoil.	Recoil.	Recoil.	Recoil.
Off-Gas Monitor Reading, mr/hr	A		3.5	4.0	4	3.5
	B		3.0	3.0	5.0	2.8
Stack gas monitor Reading, cps	A		18	9	9	28
	B		20	9	8	30

(1) Combined activity from units 1 and 2.

XX symbol signifies data not required by the test instruction.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.1 STI-1, Chemical and Radiochemical (Continued)

Analysis

Reactor Water

Reactor water conductivity was within the 10 umho/cm at 25° C., technical specification maximum limit, throughout the startup. The conductivity exceeded the operational technical specification limit of 2.0 umho/cm @ 25° C. for 6 hours from 10/3/74 until 10/4/74 because of sodium injection at 60% power.

Reactor water chloride concentration was within the 1 ppm technical specification maximum limit throughout the startup. The chloride concentration was within the operational technical specification limit of 0.2 ppm throughout the startup.

Fuel Cladding Integrity

Tables STI 1-2 and -3 show representative off-gas and iodine data obtained during the startup. Since the off-gas release was low and no evidence of spiking was evident during startup, a great deal of effort was not expended in determining fission gas release distributions.

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3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.1 STI-1, Chemical and Radiochemical (Continued)

Table STI 1-2
Browns Ferry 2 Off-Gas Summary

Date	Time	MWt	cfm	Xe-138 (2)	Kr-87 (2)	Kr-88 (2)	Kr-85m (2)	Xe-135 (2)	Xe-133 (2)	SJAE (2) Σ6	N-13 (2)	Ar-41 (2)	Stack Decay Minutes	Stack Σ22 (2)
11/4/74	0800	1445	95	68.2	18.2	5.7	1.4	1.3	0.8	96	—	—	84	24
11/11/74	1130	2094	100	47.0	21.4	4.7	1.2	2.7	0.9	78	—	24.6	80	23
11/18/74	0719	1960	130	89.6	20.9	5.5	1.6	1.2	1.1	120	—	18.8	62	42
11/25/74	0800	2325	105	14.7	16.3	4.6	1.3	2.0	0.9	40	—	15.4	76	28
12/2/74	0708	1991	100	53.9	14.9	5.4	1.4	1.0	0.8	77	—	22.9	80	23
12/9/74	0730	389	100	12.9	15.6	3.0	1.2	1.4	0.8	35	—	14.7	80	6
12/16/74	1055	3110	153	75.8	17.6	7.0	2.0	4.0	1.3	108	—	—	52	41
12/23/74	0706	1926	105	74.8	15.1	5.0	1.6	2.4	0.9	100	—	34.5	76	32
12/30/74	0716	3271	160	44.5	23.3	2.4	1.8	4.4	1.1	77	—	22.2	50	30
1/6/75	0818	3267	130	44.9	18.9	5.9	1.6	1.9	1.4	75	—	—	62	44
1/13/75	0706	2189	140	51.7	18.2	5.6	1.8	2.4	1.3	81	—	—	57	59
1/15/75	0922	2783	135	74.6	21.3	6.3	1.7	2.5	1.0	107	—	16.7	60	78
1/17/75	0843	0	100	15.6	7.8	3.1	0.7	0.6	1.0	29	—	—	80	19
1/20/75	1701	0	65	5.5	6.5	2.1	0.7	0.4	0.5	16	—	—	123	4
1/24/75	0705	575	275	33.5	27.6	10.3	3.0	1.9	1.7	78	—	—	29	65
1/27/75	0705	2119	130	65.7	17.2	5.9	1.7	3.1	1.0	95	—	—	61	34
1/29/75	0724	1433	110	26.8	10.9	5.4	1.7	1.5	1.1	47	—	—	73	15

(2) Units µCi/sec.

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3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.1 STI-1, Chemical and Radiochemical (Continued)

Table STI 1-3
Browns Ferry 2 Reactor Water Iodine Summary

Date	Time	MWt	Estimated I-131 Carryover (%)	μCi/sec. I-131	μCi/sec. I-132	μCi/sec. I-133	μCi/sec. I-134	μCi/sec. I-135
10/3/74	0950	1980	—	0.02	0.603	0.371	3.70	1.32
10/4/74	2130	1980	3.0 (1)	—	—	—	—	—
10/31/74	0715	1051	—	0.017	0.727	0.357	7.0	1.12
12/2/74	0720	0	—	0.0140	1.70	0.05	3.4	0.28
12/29/74	1500	3267	0.2 (2)	—	—	—	—	—
1/2/75	1400	2742	—	0.0702	3.38	1.01	19.5	3.97
2/3/75	0615	2854	—	0.051	2.30	0.743	12.13	1.55

(1) 50% Power - No cleanup test
(2) 100% Power - No cleanup test

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3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.1 STI-1, Chemical and Radiochemical (Continued)

Analysis (Continued)

Condensate

The condensate pump discharge and condensate demineralizer effluent conductivities were only slightly high during the initial heatup and startup conditions, however, they were within established limits throughout the remainder of startup testing. The following table, STI 1-4, shows the plant conductivity history during the startup testing.

Table STI 1-4
Browns Ferry 2 Startup Conductivities (umho/cm)

Date	Power (Thermal)	Condensate Pump Discharge	Condensate Demineralizer Combined Effluent	Reactor Water
7/14 - 7/19/74	0%, No Heat	0.74 (1)	0.12	0.28
8/5 - 8/9/74	1%, Heatup	0.15	0.07	0.15 - 1.40 (3)
9/1 - 9/2/74	15 - 35%	0.26	0.11	0.90
10/4 -10/5/74	50%	0.10	0.06	0.3 - 2.6 (2)
9/30 - 9/31/74	40 - 60%	0.12	0.08	0.50
11/22 -11/23/74	70%	0.084	0.065	0.20
12/18 -12/19/74	~93%	0.06	0.058	0.23
12/28--12/29/74	~99%	~0.06	~0.08	0.15 - 1.7 (2)

(1) No vacuum on condenser
(2) No cleanup test
(3) Range of Rx H_2O conductivity during August 1974

Sampling System

Prior to startup, a root valve verification program was conducted to ensure that the origin and approximate length of sampling lines was known.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.1 STI-1, Chemical and Radiochemical (Continued)

Analysis (Continued)

Radwaste

Both the liquid and solid radwaste systems performed satisfactorily during the startup period even though inputs to the liquid system exceeded design values.

Condensate and Cleanup Demineralizers

The condensate demineralizers were initially placed into service in late 1973 and were subsequently used to clean water during construction and preoperational testing.

Both the condensate and cleanup demineralizers performed satisfactorily during the startup period.

3.3.2 STI-5, Control Rod Drive System

Purpose

The purposes of the Control Rod Drive System test are: (a) to demonstrate that the control rod drive (CRD) system operates properly over the full range of primary coolant temperatures and pressures from ambient to operating, and particularly that thermal expansion of core components does not bind or significantly slow control rod movements; and (b) to determine the initial operating characteristics of the entire CRD system.

Criteria

Level 1

(a) Each drive speed in either direction (insert or withdraw) must be 3.0 $\pm$ 0.6 in. per sec., indicated by a full 12-ft. stroke in 40 to 60 secs.

(b) The average scram insertion time of all operable control rods, based on the deenergization of the scram pilot valve solenoids as time zero, shall be no greater than:

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.2 STI-5, Control Rod Drive System (Continued)

Criteria (Continued)

Level 1 (Continued)

% Inserted from Fully Withdrawn	Average Scram Insertion Times (Sec.)
5	0.375
20	0.90
50	2.0
90	5.0

(c) The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two-by-two array shall be no greater than:

% Inserted from Fully Withdrawn	Average Scram Insertion Times (Sec.)
5	0.398
20	0.954
50	2.120
90	5.3

(d) The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

Level 2

(a) With respect to the control rod drive friction tests, if the differential pressure variation exceeds 15 psid for a continuous drive-in, a settling test must be performed, in which case the differential settling pressure should not be less than 30 psid nor should it vary by more than 10 psid over a full stroke. Lower differential pressures in the settling tests are indicative of excessive friction.

(b) Scram times with normal accumulator charge should fall within prescribed time limits.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.2 STI-5, Control Rod Drive System (Continued)

Analysis

Scram times of the four slowest in-sequence rods were measured in conjunction with the scrams caused by the turbogenerator trips on January 7, 1975, and February 11, 1975. Table STI 5-8 summarizes the results of these scrams.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.2 STI-5, Control Rod Drive System (Continued)

Table STI 5-8
Control Rod Scram Data

Date	Reactor Scram	Initial Reactor Power (% Rated)	Initial Reactor Pressure (psig)	CRD	Scram Insertion Times (Sec.) 5%	20%	50%	90%
1/7/75	STI-27 Turbine Trip from stop valve closure	100%	980	30-27	0.302	0.663	1.400	2.435
				10-23	0.300	0.666	1.488	2.588
				34-07	0.322	0.692	1.552	2.676
				46-11	0.300	0.679	1.488	2.556
2/11/75	STI-27 Generator trip from control valve closure	100%	1000	10-23	0.294	0.666	1.51	2.63
				34-07	0.300	0.676	1.51	2.64
				46-11	0.303	0.684	1.51	2.65
				42-35	0.302	0.666	1.50	2.57

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3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.3 STI-10, IRM Performance

Purpose

The purpose of this test is to adjust the intermediate range monitor system to obtain an optimum overlap with the SRM and APRM systems.

Criteria

Level 1

(a) Each IRM channel must be adjusted so that overlap with the SRM's and APRM's is assured.

(b) The IRM's must produce a scram at 120/125 of full scale.

(c) The IRM reading 120/125 of full scale on range 10 will be set equal to or less than 30% of rated power.

Analysis

The IRM's were adjusted such that a reading of 120/125 on range 10 was equal to or less than 30% power. The APRM's had been calibrated using a power range heat balance. The results are summarized in table STI 10-1.

Table STI 10-1

APRM	A	B	C	D	E	F	Avg	
READING	16.0	16.0	16.0	16.0	16.0	16.0	16.0	
IRM	A	B	C	D	E	F	G	H
Initial Reading	102	103	113	86	92	120	INOP	INOP
Range	10	10	10	10	10	10	10	10
GAF	.784	.777	.708	.930	.870	.667	—	—
Final Reading	80.8	75.3	78.6	80.9	76.5	74.7	—	—

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.3 STI-10, IRM Performance (Continued)

Analysis (Continued)

IRM G would not drive and IRM H had a suspected low gain. These IRM's were repaired and successfully calibrated in accordance with STI 10 at 19.5% power. The SRM/IRM overlap was verified on a subsequent startup. The results are summarized in table STI 10-2. The scram setpoints are checked every three months by normal plant surveillance test. All test criteria were met.

Table STI 10-2
IRM/SRM Final Overlap

IRM Channel	A	B	C	D	E	*F	G	H
Reading	75	10	75	23	75	4	10	75
Range Switch Position	2	1	2	1	1	1	1	2

SRM Channel	A	B	C	D
Reading	3.5×10^5	2.5×10^4	2.0×10^4	4.0×10^4
Reading*	6.0×10^4	2×10^4	5×10^4	9.5×10^4

*IRM F data taken separately

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.4 STI-11, LPRM Calibration

Purpose

The purpose of this test is to calibrate the local power range monitor system.

Criteria

Level 1

The meter readings of each LPRM chamber will be proportional to the neutron flux in the narrow-narrow water gap at the height of the chamber.

Level 2

None

Analysis

The LPRM's were calibrated at power levels of approximately 25, 60, 80, and 100%. Table STI 11-1 summarizes the calibrations.

Table STI 11-1

Test Condition	Power Level	Maximum Power Variations During TIP Set	Number of Inoperable LPRM's
1	25	1.7%	6
2E	60	1.2%	5
3E	80	1.7%	3
4E	100	1.0%	4

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3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.4 STI-11, LPRM Calibration (Continued)

Analysis

The largest number of inoperable LPRM's that were encountered during any of the calibrations was six and at all times there were more than 14 operable LPRM's per APRM channel which is the minimum number required for any APRM channel to remain operable.

The process computer programs which are used to determine the gain adjustment factor (GAF) were verified prior to using the computer calculated GAF's. This was done using an off-line computer and manual methods. The GAF's from the process computer were compared with the GAF's calculated by the off-line computer program (BUCLE) at the 25, 60, and 80% test levels and these calculations were in very close agreement (within approximately 1%).

Additional TIP sets were required at the 25 and 60% levels due to minor problems with the process computer. At the 25% level it was found that the process computer was not seeing the proper reading for LPRM 32-49D. This was corrected and an additional TIP set was run. Problems were found with the TIP scan program software and corrected, requiring a fourth TIP set at this level. A third TIP set was run at the 60% level when a correction to a bad feedwater temperature caused the fraction of rated power (FRP) to change on OD-1 during the second TIP set.

Changes to the TIP system between the first and second TIP sets at the 60 and 100% levels caused the GAF's calculated after the calibration for the associated LPRM's to be slightly off from 1.0. The core top limit was changed on some channels at the 60% level and the drive speed was changed at the 100% level. Both of these changes caused small shifts of the TIP trace, thus affecting the GAF's.

In the few cases where the LPRM GAF varied from 1.0 after the calibration, the process computer corrects each LPRM reading using its corresponding GAF. Therefore, the core calculations are still valid and the core monitoring is not affected.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.4 STI-11, LPRM Calibration (Continued)

Analysis (Continued)

The calibration was performed according to the plant surveillance instruction, SI 4.1.B-3. This involves adjusting the meter readings of each LPRM chamber by the appropriate gain adjustment factor (GAF), thereby setting the LPRM to read proportional to the neutron flux in the narrow-narrow water gap at the height of the chamber. This satisfied all criteria.

3.3.5 STI-12, APRM Calibration

Purpose

The purpose of this test is to calibrate the average power range monitor system (APRM).

Criteria

Level 1

(a) The APRM channels must be calibrated to read equal to or greater than the actual core thermal power.

(b) Technical specification and fuel warranty limits on APRM scram and rod block shall not be exceeded.

(c) In the startup mode, all APRM channels must produce a scram at less than or equal to 15% of rated thermal power.

(d) Recalibration of the APRM system will not be necessary from safety considerations if at least two APRM channels per RPS trip circuit have readings greater than or equal to core power.

Level 2

If the above criteria are satisfied, then the APRM channels will be considered to be reading accurately if they agree with the heat balance to within 7% of rated power.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.5 STI-12, APRM Calibration (Continued)

Analysis

At test conditions 1, 2F, 3E, and 4F, the APRM's were calibrated to read equal to or greater than core thermal power. The core thermal power was obtained from the process computer heat balance (OD-3), which had been verified to be accurate previously by a detailed manual heat balance. The APRM's were also recalibrated after each LPRM calibration.

In the startup mode the APRM scram setpoint was set at ≤ 15% thermal power, and the rod block at ≤ 12%. A scram clamp was set for each test condition 20% above the test load line.

To verify the accuracy of the APRM's over large power changes, the gain adjustment factors of each APRM channel were recorded during the "2" recirculation pump trip of STI 30 and the subsequent recovery to test condition 4E. Typical results of this APRM tracking test are shown on figure STI 12-1 for a specific APRM (F). The worst error observed for all the channels was 4.5%, much of which is attributed to an approximate steady state background error of ± 1-1/2% present in the APRM system. All test criteria were satisfied for STI 12.

3.0 Results (Continued)

3.3 Phase IV-Power Testing (Cont'd.)

3.3.5 STI-12, APRM Calibration (Continued)

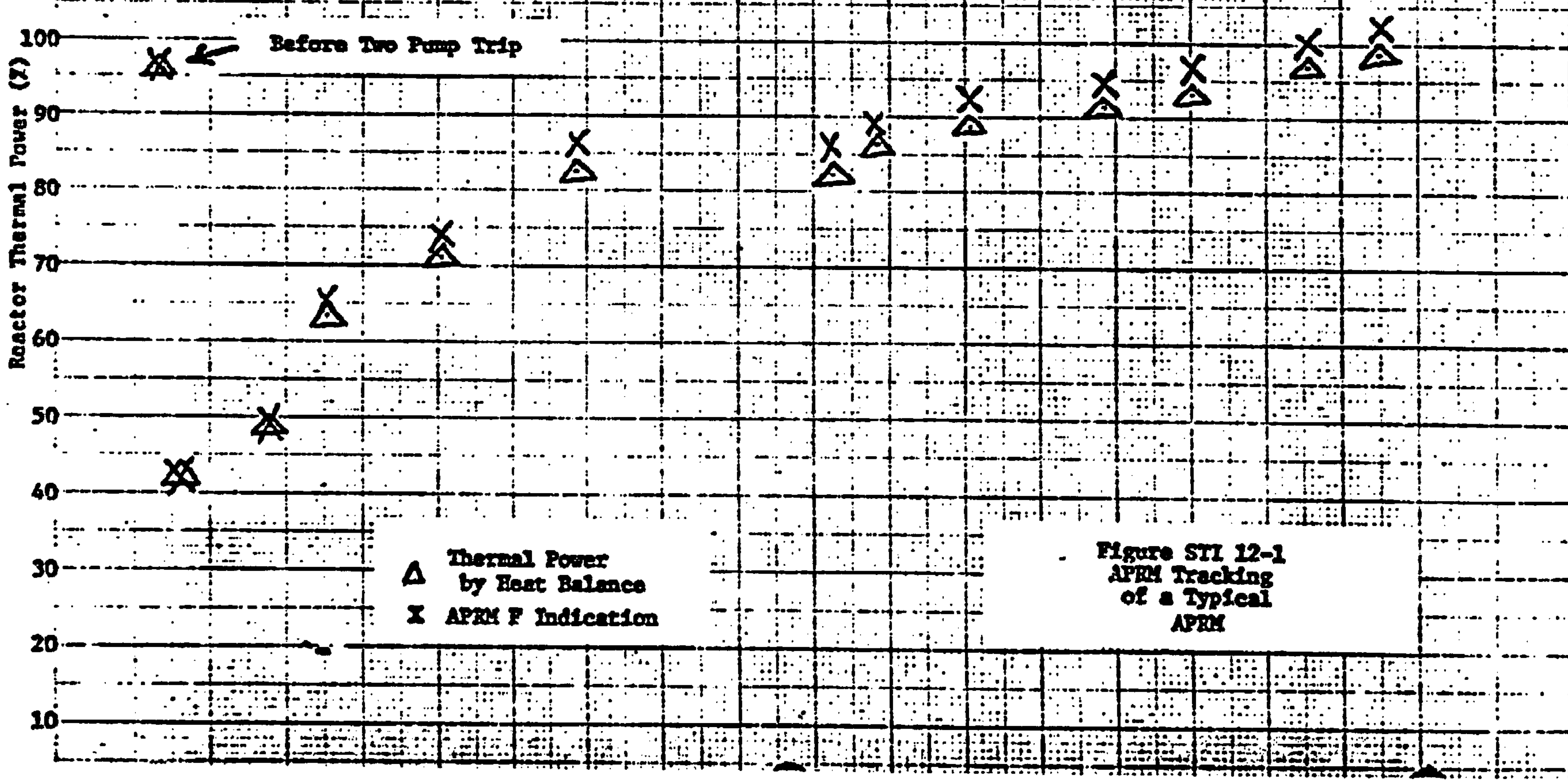

Figure STI 12-1
APRM Tracking
of a Typical
APRM

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3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.6 STI-14, RCIC System

Purpose

The purpose of this test is to verify the operation of the reactor core isolation cooling system in the injection mode.

Criteria

Level 1

The time from actuating signal to required flow must be less than 30 seconds at any reactor pressure between 150 psig and rated (1020 psig).

With pump discharge at any pressure between 150 psig and 1220 psig, the required flow is 600 gpm. (The limit of 1220 psig includes a nominally high value of 100 psi for line losses).

The RCIC turbine must not trip off during startup.

If either of the first two Level 1 criteria is not met, the reactor will only be allowed to operate at a restricted power level.

Level 2

The turbine gland seal condenser system shall be capable of preventing steam leakage to the atmosphere.

The ΔP switch for the RCIC steam supply line high flow isolation trip shall be adjusted to actuate at 300% of the maximum required steady state steam flow.

Analysis

A cold quick start was conducted on the RCIC system with the pump aligned to take suction from the condensate storage tank and discharge to the vessel. On the initial vessel injection attempt, erratic oscillations developed. The flow controller was reset which resulted in satisfactory operation. Pertinent data from the test is contained in table STI 14-2. Transient response is presented on figure STI 14-5. All criteria were satisfied with the exception of

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.6 STI-14, RCIC System (Continued)

Analysis (Continued)

the high steam flow isolation setpoints discussed in phase III testing.

Table STI 14-2 RCIC Injection Data	
Date:	September 3, 1974
Time:	1215 Hours
Reactor Pressure:	930 psig
Discharge Pressure:	1000 psig
Turbine Speed:	4000 rpm
Discharge Flow:	625 gpm
Suction Pressure:	30 psig
Controller Settings:	
PB = 2000 R/M = 50	
Time to Rated Flow	19 sec.

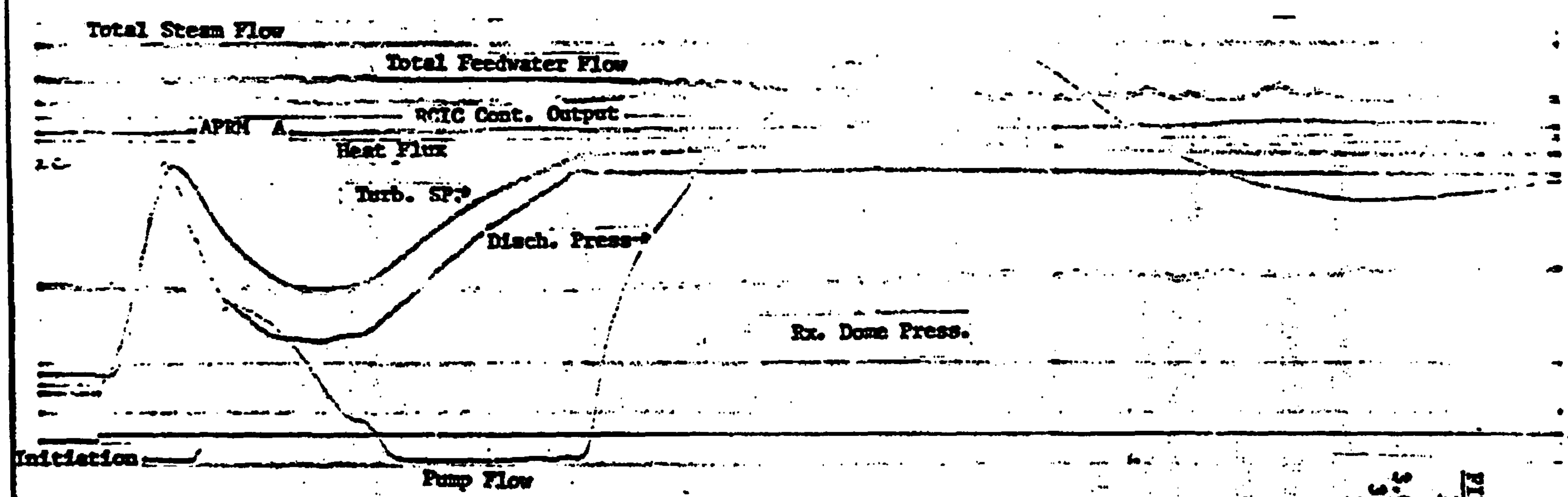

Figure STI 14-5
RCIC Vessel Injection
(Trace #22C)
1 second per division

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.7 STI-15, HPCI System

Purpose

The purpose of this test is to verify the proper operation of the high pressure coolant injection system in the injection mode.

Criteria

Level 1

The time from actuating signal to required flow must be less than 25 seconds at any reactor pressure between 150 psig and rated (1020 psig).

With pump discharge at any pressure between 150 psig and 1220 psig, the flow should be at least 5000 gpm. (The limit of 1220 psig includes a nominally high value of 100 psi for line losses).

The HPCI turbine must not trip off during startup.

Level 2

The turbine gland seal condenser system shall be capable of preventing steam leakage to the atmosphere.

The ΔP switch for the HPCI steam supply line high flow isolation trip shall be adjusted to actuate at 225% of the maximum required steady-state steam flow.

Analysis

Prior to this test, an increase in the proportional band setting (600 to 2000%) was made to the flow controller.

A cold quick start was conducted with the pump aligned to take suction from the condensate storage tank and discharge to the vessel. Pertinent data from the test is contained in table STI 15-2. The transient response is presented in figure STI 15-5. Specified criteria were satisfied.

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3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.7 STI-15, HPCI System (Continued)

Table STI 15-2
HPCI Injection Data

Date:	October 6, 1974
Time:	2110 Hours
Reactor Pressure:	940 psig
Discharge Pressure:	1030 psig
Turbine Speed:	3500 rpm
Discharge Flow:	5000 gpm
Suction Pressure:	32 psig
Controller Settings:	
PB = 2000% R/M= 100	
Time to Rated Flow:	23.5 sec.

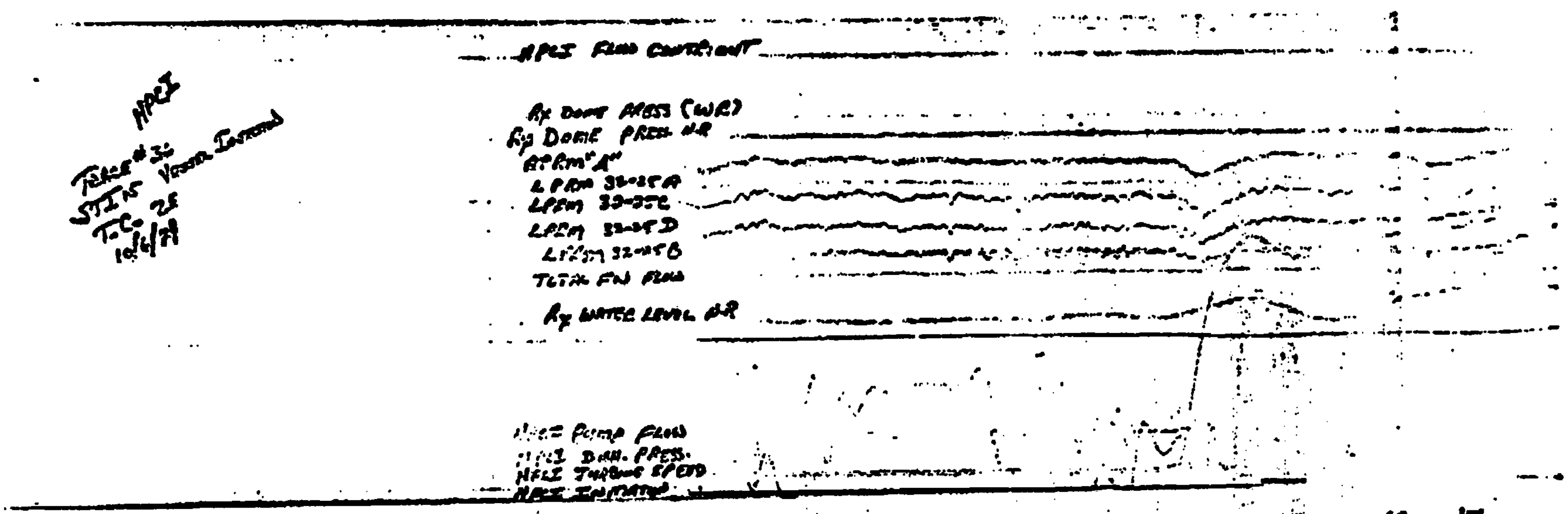

Figure STI 15-5
HPCI Vessel Injection
(Trace #36)
1 Second per Division

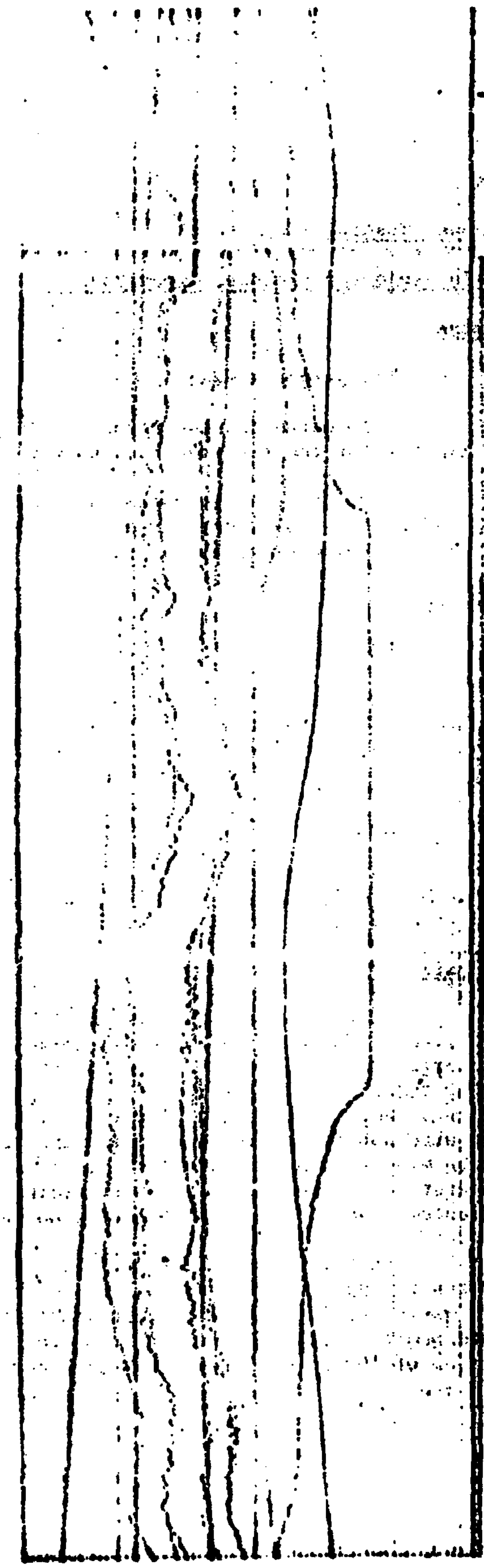

Figure STI 15-5 (Continued)
HPCI Vessel Injection
(Trace #36)
1 Second per Division

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.8 STI-16, Selected Process Temperatures

Purpose

The purposes are:

1. To establish the proper setting for the low speed limiter for the recirculation pumps.

2. To provide assurance that the measured bottom head drain temperature corresponds to bottom head coolant temperature during normal operations.

Criteria

Level 1

The reactor recirculation pump shall not be operated unless the coolant temperatures in the upper and lower regions of the vessel are within 145° F. of each other.

Level 2

The bottom head coolant temperature as measured by the bottom drain line thermocouple should be within 50° F. of reactor coolant saturation temperature.

Analysis

Test data was taken on the 50, 75, and 100 percent load lines. The data in table STI 16-2 shows the pump discharge temperatures and bottom drain temperature measured during these tests. The 50, 75, and 100 percent load line data was taken during natural circulation. The data shows that with natural circulation only, the temperature in the bottom head remained within 145° F. of the saturation temperature as specified in the criteria. The maximum difference was 45° F. which occurred on the 100% flow line.

At the 75% and 100% load lines, recirculation pump "A" was tripped and "B" was allowed to continue running. This was to verify that the temperature within an idle loop would remain stable within 50° F. of the active loop. A maximum of 10° F. difference was also observed on the 75% load line.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.8 STI-16, Selected Process Temperatures

Analysis (Continued)

Also at the 75% and 100% load lines, recirculation pump "B" was tripped and "A" was allowed to continue running. A maximum of 10° F. difference was also observed on the 75% load line.

Table STI 16-2

Selected Process Temperatures

Date	Time	Test Condition Load Line	Recirc. Pump Discharge Temp. A	Recirc. Pump Discharge Temp. B	Saturation Temp.	Bottom Drain Temp.
10-26-74	2200	50%	525° F.	520° F.	532° F.	508° F.
11-18-74	0415	75%	510° F.	505° F.	535° F.	536° F.
1-1-75	0625	100%	510° F.	510° F.	535° F.	490° F.
11-18-74	0017	75%*	510° F.	520° F.	539° F.	520° F.
11-18-74	0210	75%**	520° F.	510° F.	539° F.	519° F.
12-31-74	0440	100%*	513° F.	512° F.	537° F.	500° F.
12-31-74	0650	100%**	513° F.	517° F.	537° F.	499° F.

*Pump "A" tripped

**Pump "B" tripped

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3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.9 STI-18, Core Power Distribution

Purpose

The purposes of this test are: (a) to confirm the reproducibility of the TIP system readings, (b) to determine the core power distribution in three dimension, (c) to determine core power symmetry.

Criteria

Level 1

Not applicable.

Level 2

In the TIP reproducibility test, the TIP traces should be reproducible within ±3.5% relative error or ±0.15 inch (3.8mm) absolute error at each axial position, whichever is greater.

Analysis

TIP Reproducibility

The results of the TIP reproducibility are summarized in table STI 18-1. The test criteria was satisfied at both test conditions TC1 and TC2B.

Table STI 18-1

Summary of TIP Reproducibility Data

	Max. Relative Error/Machine	Maximum Absolute Error/Machine	Test Criteria Relative	Test Criteria Absolute
Test Condition 1 23% Power 46.6% Flow	8.5%/B	2.5mm/B	3.5%	3.8mm
Test Condition 2B 60% Power 104% Flow	4.4%/B	1.8mm/D	3.5%	3.8mm

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.9 STI-18, Core Power Distribution (Continued)

Analysis (Continued)

Core Power Distribution

The core power distribution was calculated at each appropriate test condition following a complete set of TIP traces. The results at 95% power are shown in figure STI 18-1 and table 18-2. Figure STI 18-1 shows the radial power distribution (bundle powers in MWt) for one quadrant of the core. Table STI 18-2 shows an axial (Z) distribution for each of eight radial (R) rings, the core average axial distribution, figure STI 18-2 shows the locations of the eight radial rings.

Table STI 18-2

95% R-Z Power Distribution

	NRG LVL	1	2	3	4	5	6	7	8	AVE
Core Top	12	0.671	0.604	0.569	0.547	0.508	0.433	0.324	0.192	0.302
	11	1.290	1.156	1.123	1.112	1.021	0.882	0.673	0.402	0.706
	10	1.530	1.371	1.343	1.341	1.243	1.001	0.907	0.562	1.003
	9	1.553	1.389	1.380	1.401	1.321	1.196	1.030	0.662	1.095
	8	0.872	1.281	1.320	1.331	1.227	1.236	1.126	0.740	1.106
	7	0.698	1.061	1.074	1.061	1.109	1.182	1.148	0.769	1.034
	6	0.618	0.937	0.939	0.948	1.053	1.175	1.166	0.787	1.006
	5	0.662	0.995	1.001	1.012	1.137	1.269	1.245	0.834	1.076
	4	0.800	1.213	1.231	1.219	1.353	1.471	1.408	0.942	1.252
	3	0.947	1.447	1.465	1.418	1.415	1.542	1.475	0.957	1.340
	2	0.876	1.368	1.195	1.318	1.167	1.360	1.164	0.778	1.128
Core Bottom	1	0.610	0.992	0.853	0.948	0.823	0.957	0.769	0.482	0.772
	AVE	0.927	1.151	1.125	1.138	1.115	1.150	1.037	0.676	1.000

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3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.9 STI-18, Core Power Distribution (Continued)

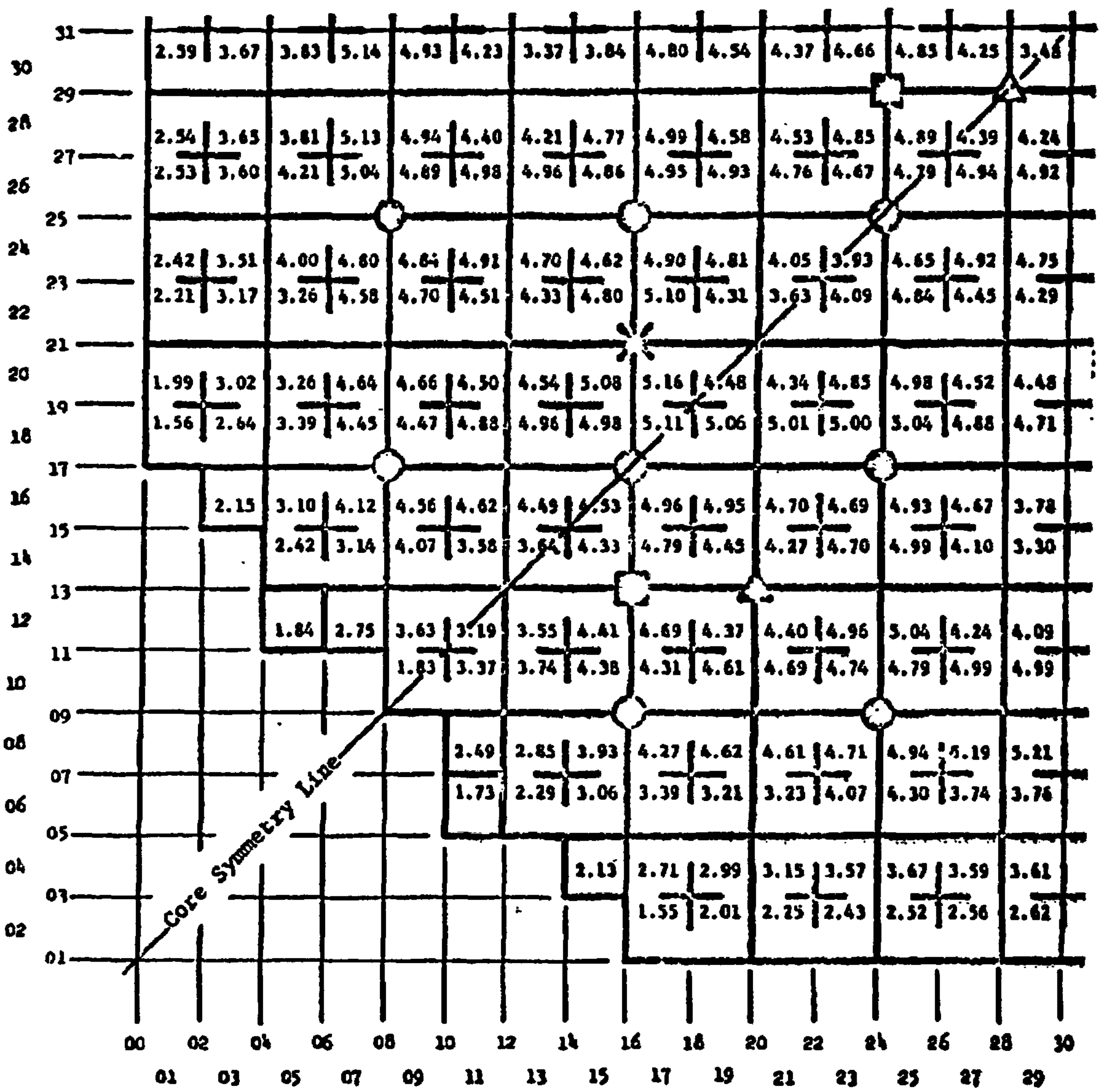

Figure STI 18-1

Bundle Power (MWt) Map at 95% Power

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3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.9 STI-18, Core Power Distribution (Continued)

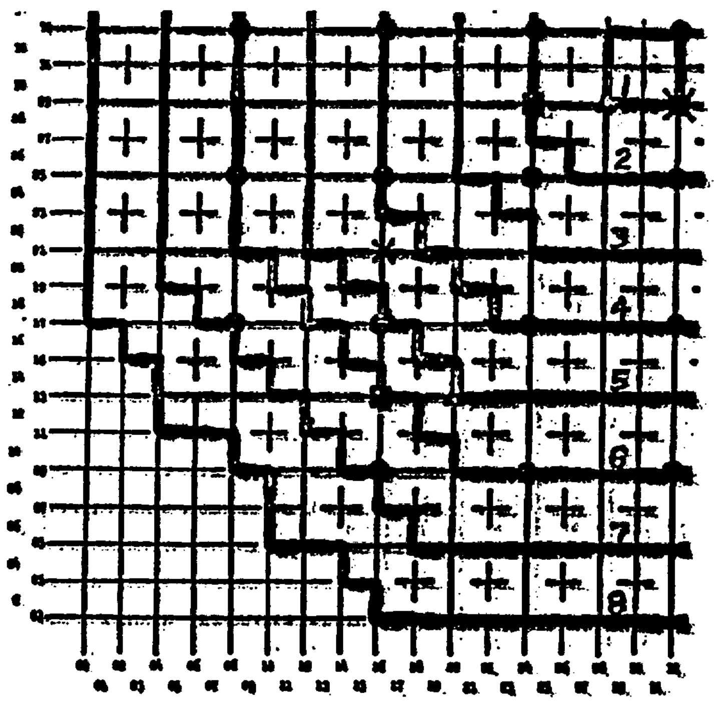

Figure STI 18-2

Ring (NRG) Map

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.10 STI 19, Core Performance Evaluation

Purpose

The purpose of this test is to evaluate the core performance parameters of core flow rate, core thermal power level, core minimum critical heat flux ratio (MCHFR), the core minimum critical power ratio (MCPR), the maximum average planar linear heat generation rate (MAPLHGR), and the maximum linear heat generation rate (LHGR) of any rod in any fuel assembly.

Criteria

Level 1

(a) The maximum linear heat generation rate of any rod during steady-state conditions shall not exceed the limit specified by section 3.5.J of the technical specifications.

(b) MCHFR shall be maintained at or above the flow dependent minimum fuel warranty MCHFR limit (line "B", figure 19.3-2).

(c) Steady-state reactor power shall be limited to 3293 MWt and values on or below the design flow control line (defined as 3440 MWt with core flow of at least 102.5 x 10^6 lb./hr.).

(d) The minimum critical power ratio (MCPR) shall be maintained greater than or equal to 1.20 times kf.

(e) The maximum average planar linear heat generation rate (MAPLHGR) shall not exceed the limits of the plant technical specification 3.5.I.

Analysis

The significant core performance parameters such as maximum fraction of limiting power density (MFLPD), minimum critical heat flux ratio (MCHFR), minimum critical power ratio (MCPR), core thermal power, maximum average planar linear heat generation rate (MAPLHGR), and maximum linear heat generation rate (KW/ft.) were monitored throughout the test program at each of the test plateaus. Table STI 19-1 contains a summary of the behavior of these

3.0 Results (Continued)

3.3 Phr'e IV - Power Testing (Continued)

3.3.10 STI-19, Core Performance Evaluation (Continued)

Analysis (Continued)

parameters. In addition, the test criteria for acceptable performance at each test condition is presented. Figures STI 19-1 and STI 19-2 show the MCHFR and MCPR data plotted, compared to the limit lines. The test data shown on these figures is from the test condition on the 100% load line.

Extensive effort was expended to verify the process computer calculations. Heat balances were verified by both hand calculations and off line computer calculations. The off line computer program BUCLE was used to calculate the core performance parameters which were compared with the process computer calculations. Close agreement was seen between the process computer and these independant calculations.

As indicated by table STI 19-1, all test criteria were satisfied.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.10 STI-19, Core Performance Evaluation (Continued)

Table STI 19-1

Summary of the Behavior of Core Performance Parameters

Test Condition	PWR %	Flow %	MCHFR Value	MCHFR Limit	MCPR Value	MCPR Limit	MAPLHGR Value kw/ft	MAPLHGR Limit kw/ft	MFLPD	LHGR Corresp. kw/ft	LHGR Limit kw/ft
1	31	47	8.63	2.82	2.575	1.362	3.85	14.51	.266	4.92	18.00
			*8.65		*2.484	1.362	*3.91	*14.51	*.274	*5.07	*18.00
2E	60	104	4.41	1.90	1.855	1.20	7.89	15.05	.535	9.90	18.33
2A	23	~28	9.73	4.10	2.352	1.446	3.55	15.05	.244	4.51	17.95
2D	42	75	*5.03	2.17	2.172	1.218	*6.90	*14.49	*.470	*8.70	*17.96
3C	48.7	50	4.73	2.74	1.763	1.344	7.45	15.07	.500	9.25	18.35
3D	64	67.5	3.45	2.30	1.515	1.257	9.31	15.08	.622	11.51	18.33
3E	82.2	103	3.14	1.90	1.480	1.20	10.75	15.10	.743	13.75	18.35
			*3.13		1.480	1.20	*10.87	*14.46	*.745	*13.78	*18.35
3A	39	33.7	5.47	3.30	1.510	1.430	6.34	15.10	.432	7.99	18.33
4C	67	49.5	3.41	2.75	1.499	1.341	9.93	15.48	.658	12.17	18.35
4D	78.7	70	2.89	2.75	1.374	1.248	11.25	15.49	.746	13.80	18.35
4E	95.4	105	2.69	1.90	1.324	1.20	12.37	15.17	.830	15.36	18.35
4A	41.6	33.6	*5.34	3.30	1.809	1.430	*6.71	*14.43	*.443	*8.20	*18.33

* BUCLE Results

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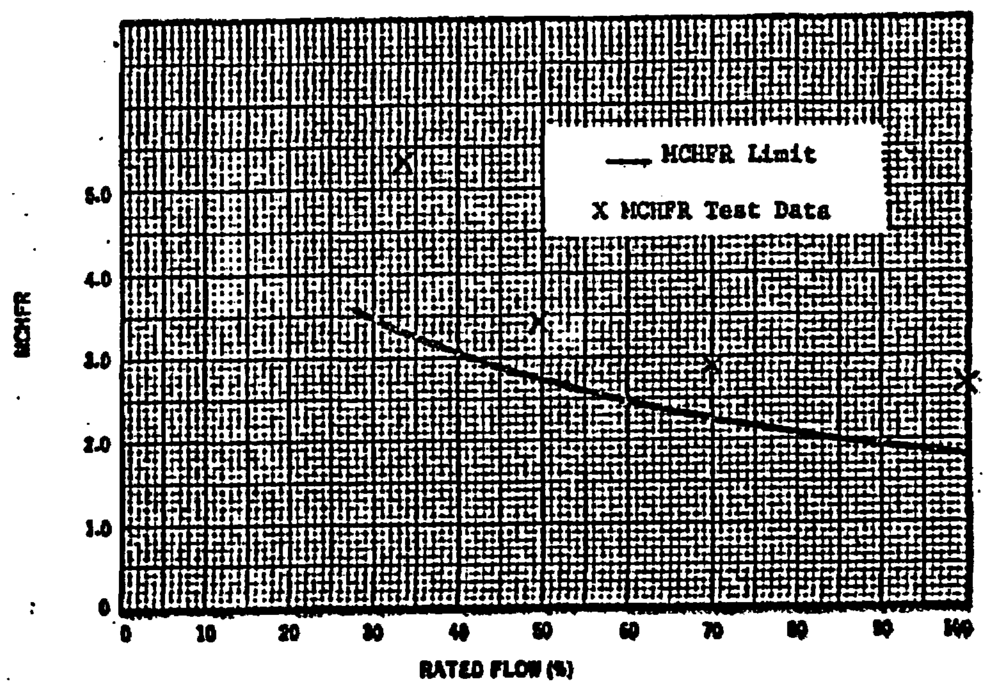
3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.10 STI-19, Core Performance Evaluation (Continued)



MCHFR Curve for Fuel Warranty Determination

Figure STI 19-1

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.10 STI 19, Core Performance Evaluation (Continued)

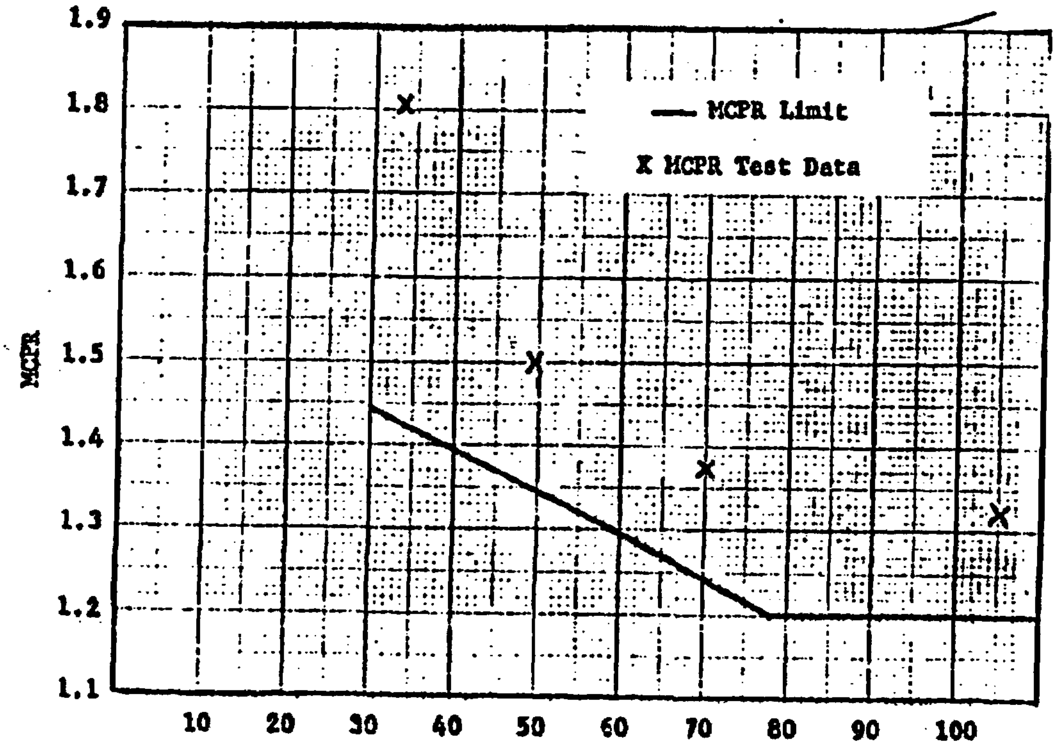


Figure STI 19-2

MCPR Test Data

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3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.11 STI-21, Flux Response to Rods

Purpose

The purpose of this test is to demonstrate the stability of the core local power-reactivity feedback mechanism with regard to small perturbations in reactivity caused by rod movement.

Criteria

Level 1

The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to control rod movement.

Level 2

The decay ratio must be less than or equal to 0.25 for each process variable that exhibits oscillatory response to control rod movement when the plant is operating above the lower limit of the master flow controller.

Analysis

The stability of the local core power-reactivity feedback mechanism was proved by observing the local and macroscopic effects induced by control rod movements.

Generally the rod was moved near a location with the most limiting thermal conditions. The local power deviation was monitored using the nearest LPRM. Other gross core parameters as control valve position, dome pressure, water level, core flow, steam flow, and APRM indication were also recorded. Table STI 21-1 summarizes the test results. Except for the LPRM flux, no discernible oscillations or instabilities were noted for the observed process variables. The LPRM flux, upon rod movement, moved promptly to a new steady value, showing only a minor well-damped oscillation immediately following the rod displacement. The test criteria were met.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.11 STI-21, Flux Response to Rods (Continued)

Table STI 21-1

Summary of the Flux Response to Rods Data

Test Condition	Rod Moved	Rod Displacement	LPRM Monitored	Local Power Change	Highest[1] Decay Ratio of Measured Variables
1	22-47	12 ⟶ 10 10 ⟶ 12	24-49D	-15% + 5%	< .25
2E	30-55	42 ⟶ 36 36 ⟶ 42	32-49A	- 7% + 7%	< .25
3E	22-11	40 ⟶ 36 36 ⟶ 42	24-09A	-12% +12%	< .25
4E	06-23	40 ⟶ 36 36 ⟶ 42	08-25A	-10% +20%	< .25

1 - LPRM flux was only variable showing any oscillation

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.12 STI-22, Pressure Regulator

Purpose

The purposes of this test are:

1. To demonstrate the optimum settings for the pressure control loop by analysis of the transients induced in the reactor pressure control system by means of the pressure regulators.

2. To demonstrate the take-over capability of the back-up pressure regulator upon failure of the controlling pressure regulator and to set spacing between the setpoints at an appropriate value.

3. To demonstrate smooth pressure control transition between control valves and bypass valves when reactor steam generation exceeds steam used by the turbine.

Criteria

Level 1

The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to pressure regulator changes.

Level 2

(a) In all tests except the simulated failure of the operating pressure regulator, the decay ratio is expected to be 0.25 for each process variable that exhibits oscillatory response to pressure regulator changes when the plant is operating above the lower limit setting of the master flow controller.

(b) Pressure control deadband, delay, etc., if any shall produce variations in steam flow to the turbine no larger than the values of rated flow specified in the following table, as measured by gross generated electrical power:

Percent of Full Power	Percent of Rated Flow
90-100%	±0.5%
70-90	±1.5 to ±0.5
70 and below	±1.5

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.12 STI-22, Pressure Regulator (Continued)

Criteria (Continued)

(c) Optimum gain values for the pressure control loop shall be determined in order to give the fastest return from the transient condition to the steady-state condition within the limit of the above criteria.

(d) During the simulated failure of the primary controlling pressure regulator, if the setpoint of the back-up pressure regulator is optimmly set, the back-up regulator shall control the transient such that the reactor does not scram.

(e) Following a ±10 psi pressure setpoint change, the time between the setpoint change and the occurrence of the pressure peak shall be 10 seconds or less.

Analysis

Table STI 22-1 summarizes the pertinent data concerning each of the pressure regulator tests. Figure STI 22-1A/1M shows the transient recorder traces for test condition 4E, 95-100% rated power, 100% flow. Using built-in test switches located on the EHC pressure regulator cards, positive and negative 10 psi setpoint changes were made on each pressure regulator. A test switch which simulates a pressure regulator failure is also provided to test backup takeover.

As a result of the information obtained from STI 22 on unit 1, the unit 2 testing was greatly simplified. A notch filter was added to the EHC circuitry prior to the beginning of startup testing which allowed the initial adjustment of the pressure regulator to provide the fast response to step changes observed only after the 50% load line testing on unit 1. The pressure regulator settings were initially set at approximately the same point as unit 1 and did not require further adjustment at any test condition. The EHC pressure controller setings were left as follows:

psi regulator = 3.1 turns (3.3% valve travel/psi)
lag pot = 2.0 turns (6.0 sec.)
lead pot = 1.4 turns (1.0 sec.)

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.12 STI-22, Pressure Regulator (Continued)

Table STI 22-1

Pressure Regulator Test Data

TC	Reactor Power (%)	Recirc. Flow (%)	Date Performed	10 psi Steps	Backup Takeover	Transient Trace No.	Core Thermal Power (MWt)	Core Flow (mlb/hr.)	Reactor Pressure (psig)	FW Flow (mlb/hr.)	Elec Load (MWe)
1	25	43	9/2/74	x	x	16	823	45	963	2.5	188
2E	60	104	9/30/74	x	x	29	1976	106	945	7.5	630
3E	75	97	11/17/74	x	x	48	2470	98	970	9.8	805
4C	59	47	12/23/74	x	NR	65	1937	48	938	7.4	621
4D	74	70.5	12/19/74	x	NR	56	2411	72.3	960	9.4	794
4E	96	101	12/20/74	x	x	61	3165	103.8	983	12.8	1065

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.12 STI-22, Pressure Regulator (Continued)

Analysis (Continued)

At test condition 1, a fast response to step changes was observed which was greatly improved as compared to similar testing on unit 1. Slight oscillatory behavior was observed in the reactor water level and APRM power transient, but the required 0.25 damping criteria was met in all cases.

Test condition 2E and 3E netted similar results as compared to test condtion 1 except that the response time was slightly longer, as expected for higher power levels. Reactor water level and APRM power transients continued to be the parameters with the highest decay ratio, although at all times the 0.25 damping criteria was adequately met.

Pressure regulator testing on the 100% load line was smooth and performed as expected. The parameter observed to have the largest decay ratio was in all cases, total feed-water flow, however, the decay ratio of this parameter remained much less than 0.25. Steady-state limit cycles produced an observed variation of ±4MWe (±0.003%) at test condition 4E however this remained well below the ±0.5% criteria for 90-100% of rated power operation. Limit cycles at all other test conditions if any were not measurable by ordinary means thus are considered non-existant.

As a result of the installation of a notch filter, the pressure regulator was capable of being optimized to produce a fast response to setpoint changes. The time between pressure setpoint change and the occurrance of the first pressure peak was measured to be between 3 and 7 seconds on the 100% load line which was a satisfactory response when compared to the acceptance criteria value of 10 seconds.

Pressure regulator testing on the 100% load line satisfied all test acceptance criteria. A summary of the worst transient cases for test condition 4C, 4D, and 4E is contained in table STI 22-2.

As a result of testing the pressure regulator at all test conditions listed, it was possible to plot the relationship of control valve position versus total steam flow. This plot is found in figure STI 22-2. From the graph it can be seen that at 100% steam flow (13.38 mlbs./hr.), control valve position is 45%.

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.12 STI-22, Pressure Regulator (Continued)

Table STI 22-2

Transient Summary, 100% Test Plateau

Test Condition	4C			4D			4E		
Trace No.	65			56			61		
Step Input	-10 psi	+10 psi	+10 psi	-10 psi	-10 psi	-10 psi	-10 psi	-10 psi	+10 psi
Regulator A/B	A	A	B	A	A	B	A	B	B
Recirc. Mode	Master Manual	Master Manual	Master Manual	Master Manual	Master Manual	Master Manual	Master Manual	Master Manual	Master Manual
Valves CV/BPV	BPV Incipient	BPV	CV	CV	CV	BPV	BPV Incipient	CV	BPV
Initial Dome Pressure	938 psig	928 psig	926 psig	957 psig	944 psig	944 psig	982.5 psig	983 psig	973.5 psig
Final Dome Pressure	928 psig	938 psig	936 psig	944 psig	955 psig	955 psig	973 psig	973 psig	983.5 psig
Time to First (1) Pressure Peak	3.5 sec.	3.0 sec.	3.5 sec.	5.0 sec.	3.5 sec.	3.5 sec.	7.0 sec.	5.0 sec.	6.0 sec.
Parameter of Highest (2) Decay Ratio (ratio)	Total FW Flow (<.25)	Total FW Flow (<.25)	Total FW Flow (<.25)	Total FW Flow (<.25)	Total FW Flow (<.25)	Total FW Flow (<.25)	Total FW Flow (<.25)	Total FW Flow (<.25)	Total FW Flow (<.25)
Steady-State Limit Cycle (3)	±0%	±0%	±0%	±0%	±0%	±0%	±0.003 (±4MWe)	±0.003 (±4MWe)	±0.003 (±4 MWe)

Notes:

(1) Level 2 criteria limit is 10 sec.
(2) Level 2 criteria limit is 0.25
(3) Level 2 criteria limit is ±0.5%
(Unit cycle is measured by indicated total steam flow at steady-state conditions)

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.12 STI-22, Pressure Regulator (Continued)

Parameter Name	Input Signal No.	Initial* Value	Value Per Division
APRM A	1	97	1%
APRM B	2	91	1%
Core dP	3	20.5	2.0 psi
Total FW Flow	5	12.9	0.4 Mlb/hr.
Total Steam Flow	6	13.9	0.4 Mlb/hr.
Rx Water Level NR	7	33.5	0.5 inch
Total Core Flow	8	104.0	1.0 Mlb/hr.
Control Valve Position, All	9	47	1%
Recirc. Drive Flow A	11	50.5	1.0 kgpm
EHC Output	12	—	2%
Rx Dome Pressure NR	13	983	0.5 psi
Recirc. Drive Flow B	14	49.5	1.0 kgpm
Bypass Valve Ø1 Position	18	25	2.0%

*Measured with "A" regulator in BPV control

Legend for Figure STI 22-1

Recorder Calibration Data

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.12 STI-22 Pressure Regulator (Continued)

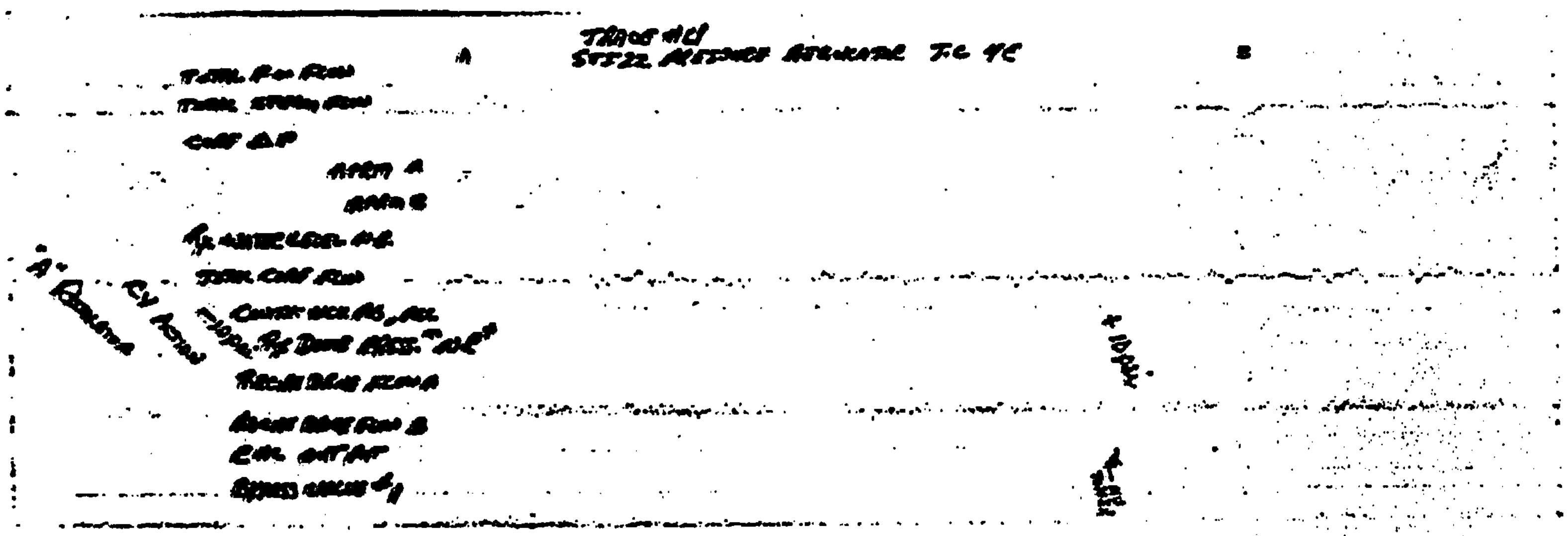

Figure STI 22-1
Trace 61A and B
±10 psi step, CV action, "A" regulator

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.12 STI-22, Pressure Regulator (Continued)

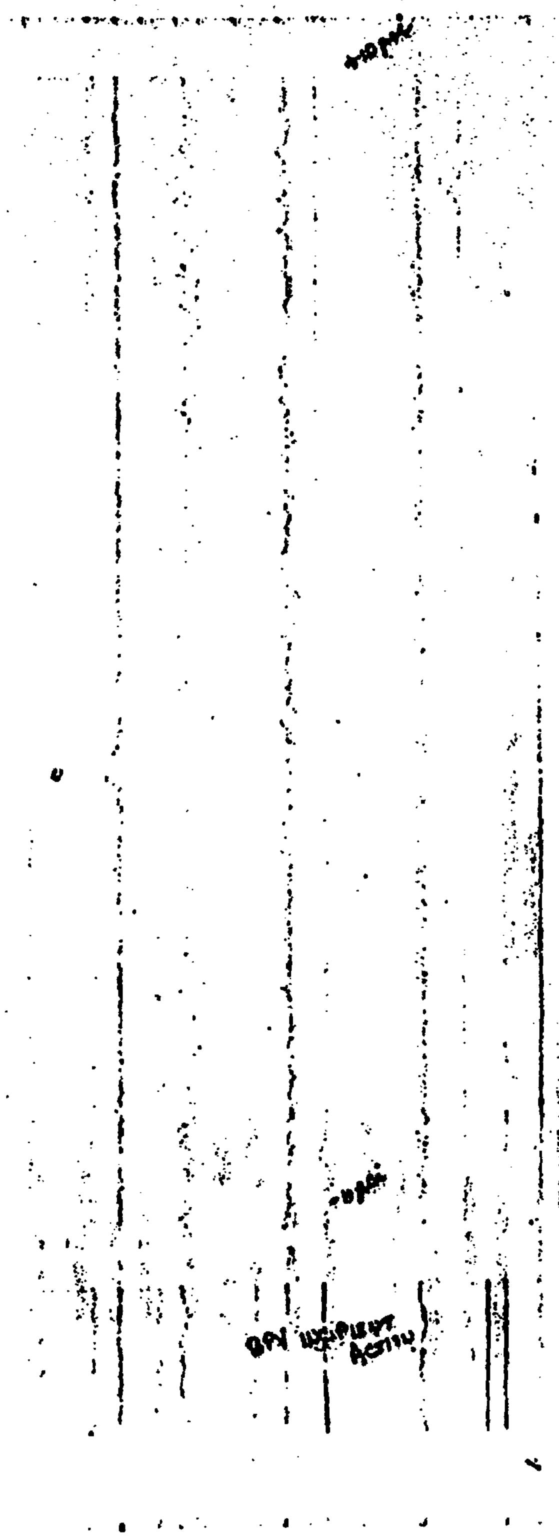

Figure STI 22-1 (Continued)
Trace 61-C
-10 psi step, BPV Analpient, "A" Regulator

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continu...

3.3.12 STI-22, Pressure Regulator (Con...

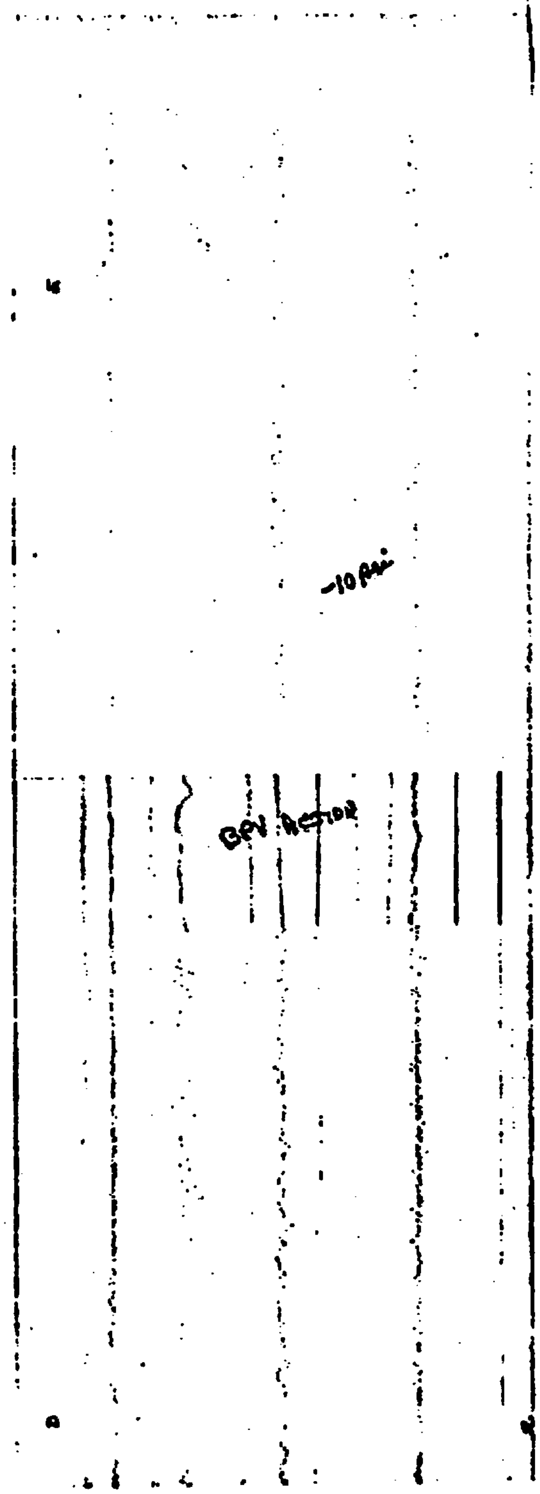

Figure STI 22-1 (Continued)
Trace 61-D: +10 psi step, BPV incipient, "A" regulator
Trace 61-E: -10 psi step, BPV action, "A" regulator

3.0 Results (Continued)

3.3 Phase IV - Power
Testing (Continu

3.3.12 STI-22, Press
sure Regula
(Continue

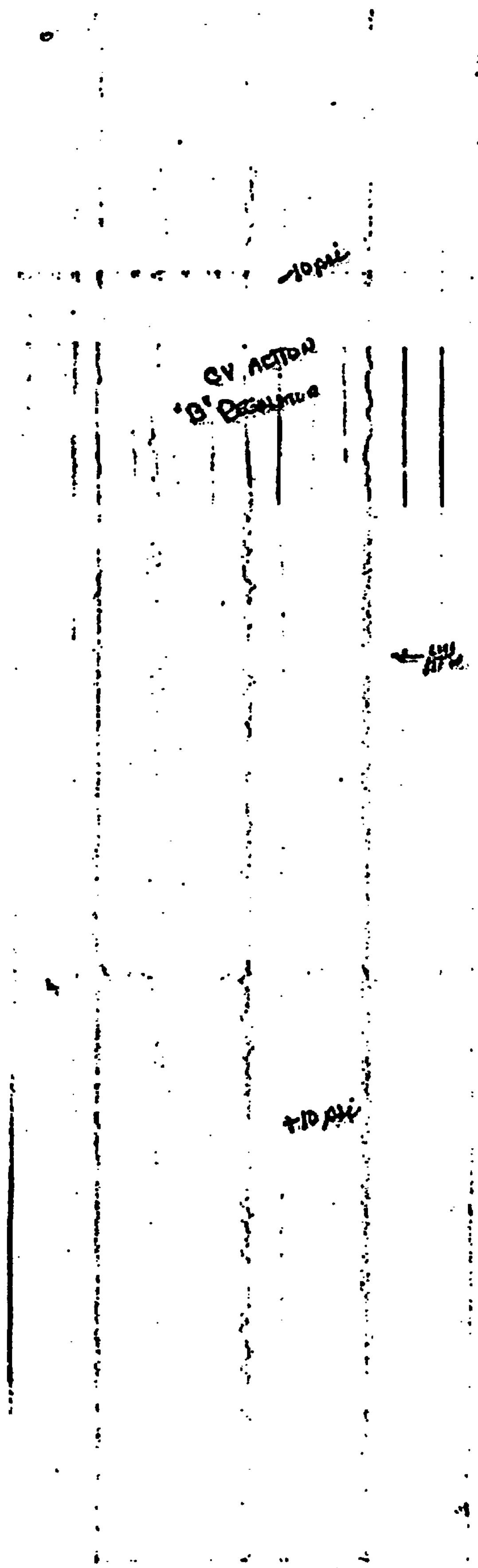

Figure STI 22-1 (Continued)
Trace 61-F: +10 psi step, RPV action, "A" regulator

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Cont.)

3.3.12 STI-22; Pressure Regulator (Cont.)

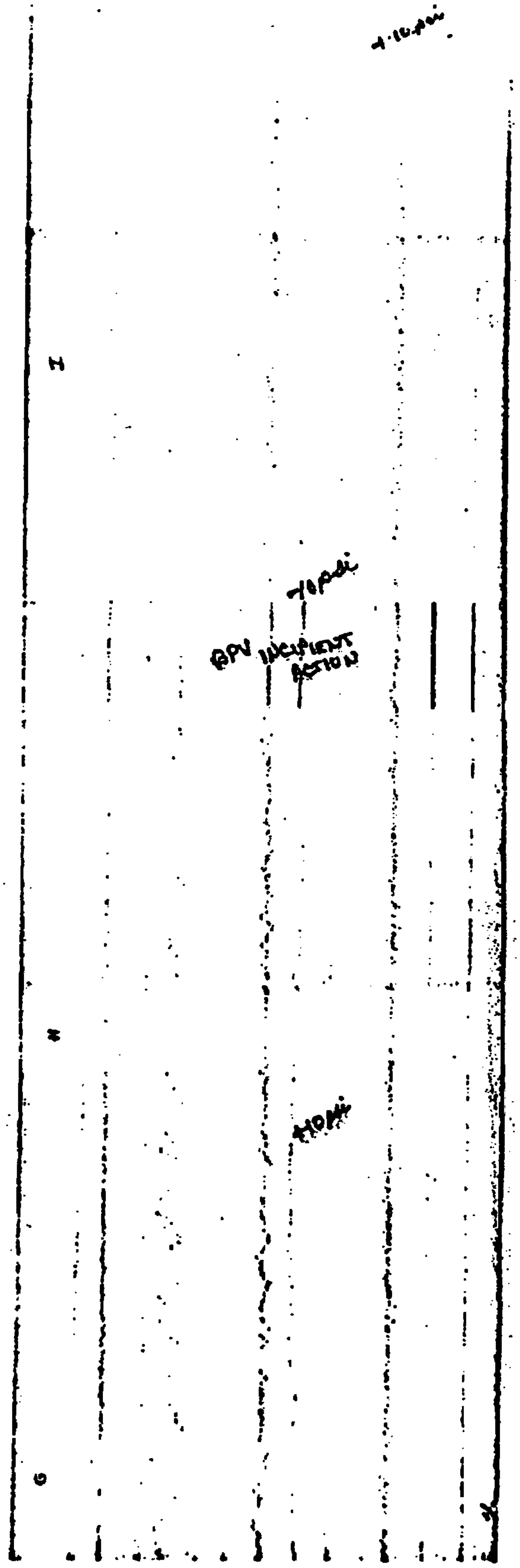

Figure STI 22-1 (Continued)

Trace 61-H: +10 psi step, CV action, "B" regulator
Trace 61-I: -10 psi step, BPV incipient, "B" regulator
Trace 61-G: -10 psi step, CV action, "B" regulator

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Cont.)

3.3.12 STI-22, Pressure Regulator (Cont.)

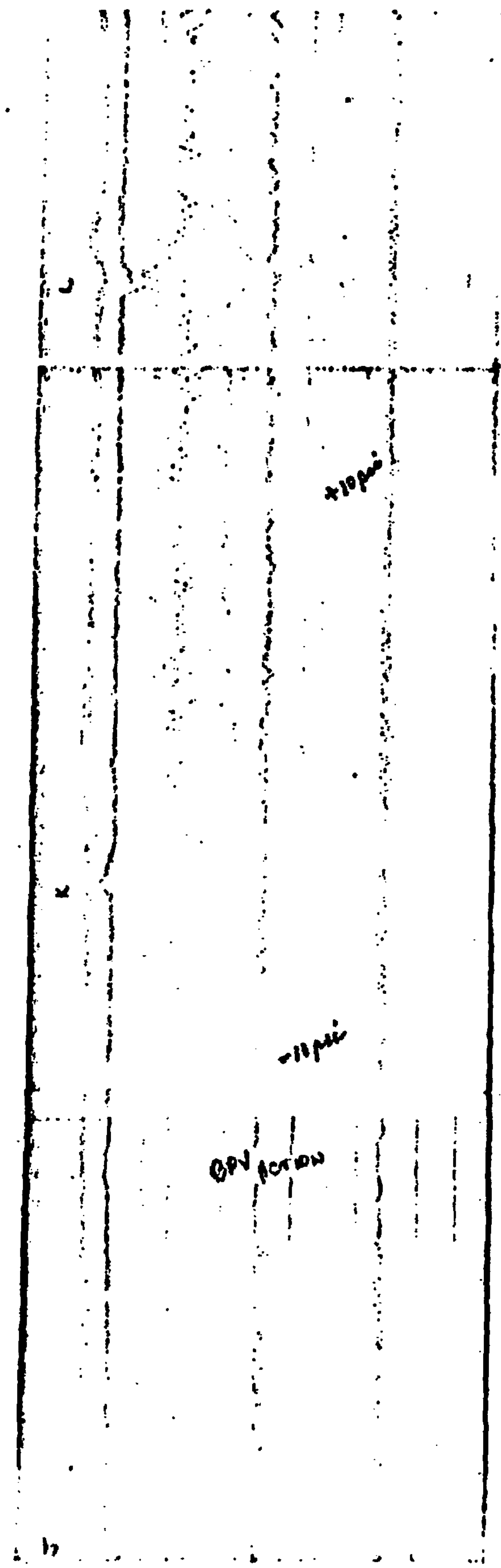

Figure STI 22-1 (Continued)
Trace 63-J: +10 psi step, BPV incipient, "B" regulator
Trace 63-K: -10 psi step, BPV incipient, "B" regulator
Trace 63-L: +10 psi step, BPV action, "B" regulator

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Cont.)

3.3.12 STI-22, Pressure Regulator (Cont.)

Figure STI 22-1 (Continued)

Trace 61-M Fail "B" regulator

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.12 STI-22, Pressure Regulator (Continued)

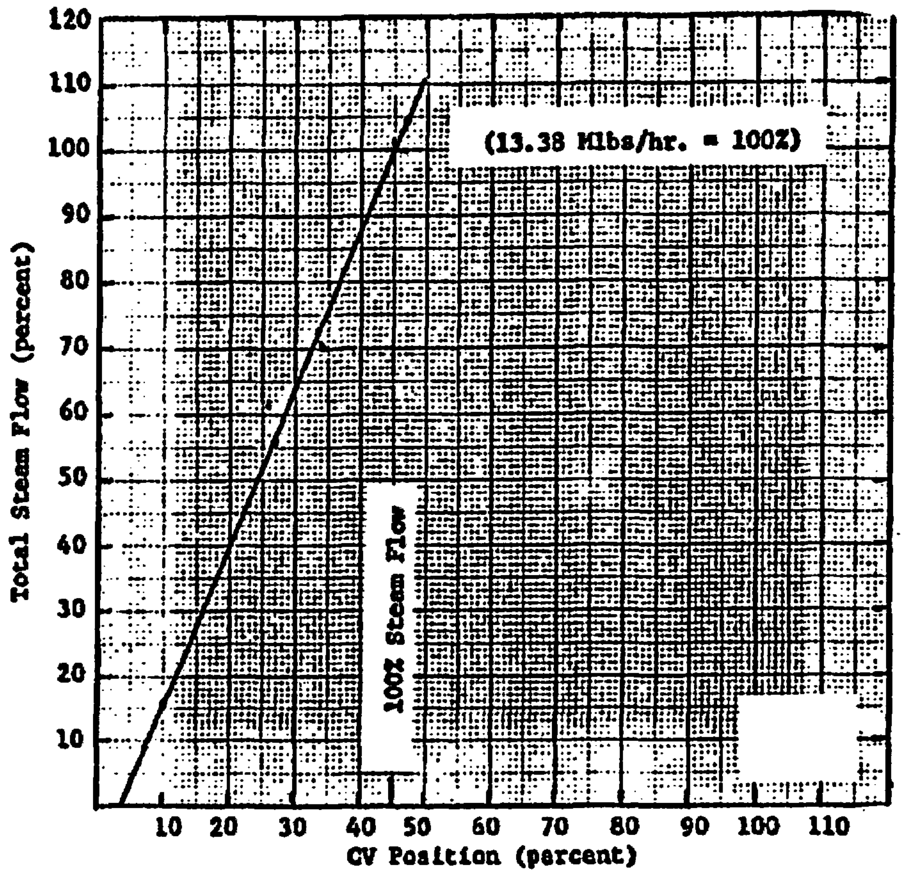

Figure STI 22-2

CV Position vs Total Steam Flow

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.13 STI-23, Feedwater System

Purpose

The purposes of this test are:

1. To adjust the feedwater control system for acceptable reactor water level control.

2. To demonstrate stable reactor response to subcooling changes.

3. To demonstrate the capability of the automatic core flow runback feature to prevent a low water level scram following the trip of one feedwater pump.

Criteria

Level 1

The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to feedwater system changes.

Level 2

(a) The decay ratio is expected to be less than or equal to 0.25 for each process variable that exhibits oscillatory response to feedwater system changes when the plant is operating above the lower limit of the master flow controller.

(b) Following a 3-inch level setpoint step adjustment in three-element control, the time from the setpoint step change until the water level peak occurs shall be less than 35 seconds without excessive feedwater swings (changes in feedwater flow greater than 25% of rated flow).

(c) The automatic recirc flow runback feature shall prevent a scram from low water level following a trip of one of the operating feedwater pumps.

(d) With the condensate system operating normally, the maximum turbine speed limit shall prevent pump damage due to cavitation.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.13 STI-23, Feedwater System (Continued)

Analysis

The capability of the feedwater system to satisfactorily control reactor water level was demonstrated at the test conditions summarized in table STI 23-1. Transient recorder traces were taken of the specified maneuvers while in alternative modes of system operation.

Due to problems associated with maintaining operation within the preconditioning interim operating requirements (PCIOMR) the automatic mode of the recirculation flow control could not be used. As a result testing of the feedwater system in the auto mode had been deleted midway into the startup test program on unit 2. Plant operating procedures were written requiring that the recirculation flow control system be operated in the master manual (MM) or local manual (LM) modes only, therefore the auto mode was not tested. In the event the PCIOMR limitations are relaxed such that the auto mode of operation becomes possible, special testing will be performed at that time, upon approval of the NSRB, to demonstrate the operability of that control mode.

Water level setpoint step changes were performed at various test conditions, and a one feedwater pumptrip was performed at test condition 4E on the feedwater system.

Level Setpoint Changes

Experience from unit 1 startup testing of the feedwater control system provided valuable data which could be used to minimize the "tune-up" procedure on unit 2. Approximate controller settings could easily be determined which provided acceptable system response at test condition 1. The intial controller settings were as follows:

Proportional Band = 360%
Resets = 0.4 R/min.

During testing at TC-1, the controller proportional band was reduced to 300%. Plus and minus 3-inch setpoint changes were introduced in the 3-element mode, and ±5 inch changes in single element. The time to respond to the setpoint change and the time to the new steady-state level were shortened with the increased gain (decreased proportional band).

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.13 STI-23, Feedwater System (Continued)

Table STI 23-1

Feedwater System Test Data

TC	Reactor Power (%)	Recirc. Flow (%)	Date Performed	Transient Trace No.	Core Thermal Power (MWt)	Core Flow (Mlb/hr.)	Reactor Pressure (psig)	FW Flow (Mlb/hr.)	As-left Proportional Band	As-left reset	Remarks
1	25	44	9/2/74	19	790	46	919	2.5	300	0.04	
2E	60	104	10/6/74	32	1950	106.8	942	7.6	300	0.04	Recirc in Manual
	60	104	10/6/74	33	1950	107.8	945	7.6	300	0.04	Recirc in Auto
3E	75	97	11/17/74	47	2466	99.1	969	9.8	300	0.04	
4C	59	46.8	12/23/74	67	1938	48.0	938	7.4	250	0.04	
4D	71.7	70.5	12/19/74	58	2360	72.3	955	9.2	260	0.04	
4E	96.6	101	12/20/74	63	3181	103.8	984	12.8	250	0.04	
	98.2	98.7	12/30/74	71	3233	101.2	985	13.1	250	0.04	One pump trip

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.13 STI-23, Feedwater System (Continued)

Analysis (Continued)

Level Setpoint Changes (Continued)

Test condition 2 was the only time the feedwater system was tested in the auto recirculation flow control mode, since the deletion of this mode had not yet been finalized. Transient response in the auto mode, when subjected to a +5 inch level change produces an unexplained minor excursion of about 6 kgpm in recirculation drive flow "A" (caused by pump "A" speeding up for several seconds). The cause of this anomaly cannot be further investigated since future testing in the auto mode has been prohibited by the Nuclear Safety Review Board (NSRB). Testing of the feedwater system in the master manual flow control mode was satisfactory and did not produce the transient response witnessed in the auto mode. Plus and minus 3-inch setpoint changes were made in the 3-element mode using both the "A" and "B" water level reference column, and ±3 inch changes were made in the single element mode again using the "A" and "B" column. No further controller adjustments were necessary to improve the response of the feedwater system.

At test condition 3E, ±3-inch setpoint changes were made in the 3-element mode, and ±5-inch changes in the single element mode in both cases using the "A" water level reference column. No controller adjustments were made. All test criteria had been satisfied for the tests performed.

On the 100% load line, testing was performed at the test conditions summarized in table STI 23-2. When ±3-inch changes were made in the 3-element mode, a fast response was noted and the water level attained its new setpoint value without overshoot. Only a minor overshoot was experienced in the single element tests. All test criteria for level setpoint changes have been satisfied on the 100% load line tests. All process variables which exhibit oscillatory response had decay ratios of <0.25. The time to water level peak following a 3-inch level setpoint change was <35 seconds.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.13 STI-23, Feedwater System (Continued)

Table STI 23-2

Test Condition/ Trace No.	Controller Settings		Controller Mode/Column/ Demand	Time to RFP Response, Seconds	Time to New SS Level Seconds	Time to Max. or Min. level, Seconds	Max. FW Change (% of Initial)	Time Criteria P/F	Decay Ratio Criteria P/F
	PB	RM							
4C/67	250	0.04	3el/A/-3"	2.2	32*	32	13.2	P	P
	250	0.04	3el/A/+3"	2.7	29*	29	10.5	P	P
	250	0.04	1el/A/-5"	2.0	49.5	30.5	26.3	P	P
	250	0.04	1el/A/+5"	2.0	46.5	29	18.4	P	P
4D/58	260	0.04	3el/B/+3"	2.5	26*	26	10.0	P	P
	260	0.04	3el/B/-3"	2.5	19*	19	13.0	P	P
	260	0.04	1el/B/+5"	3.0	42	28	19.8	P	P
	260	0.04	1el/B/-5"	2.0	55	24.5	23.0	P	P
4E/63	250	0.04	3el/B/+3"	2.25	19.25*	19.25	7.1	P	P
	250	0.04	3el/B/-3"	2.5	31.5*	31.5	6.3	P	P
	250	0.04	1el/B/+5"	2.0	44.0	34.0	11.8	P	P
	250	0.04	1el/B/-5"	2.0	44.5	30.0	15.7	P	P

*No overshoot experienced

P = Pass
F = Fail

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.13 STI 23, Feedwater System (Continued)

Analysis (Continued)

Level Setpoint Changes (Continued)

Adjustment to the feedwater controller during this series of tests resulted in final controller settings as follows:

Proportional Band = 250%
Resets - 0.4 R/min.

Figure STI 23-1A, B, C, and D show the transient response for tests performed at test condition 4E.

Parameter Name	Trace No.	Initial Value	Value Per Division
APRM A	1	97	1.0%
APRM B	2	90	1.0%
Core dP	3	20.1	2.0 psi
Rx Dome Pressure NR	4	984	4.0 psi
Total FW Flow	5	12.6	0.4 Mlb/hr.
Total Steam Flow	6	14.0	0.4 Mlb/hr.
Rx Water Level NR	7	32	0.5 inch
Total Core Flow	8	105.0	1.0 Mlb/hr.
FW Controller Output	10	80	1.0%
Recirc Drive Flow "A"	11	51	1.0 kgpm
EHC Output	12	-	2.0%
Bypass Valve #1 Position	18	Closed	2.0%

Legend for Figure STI 23-1

Feedwater Level Setpoint Changes

Calibration Data

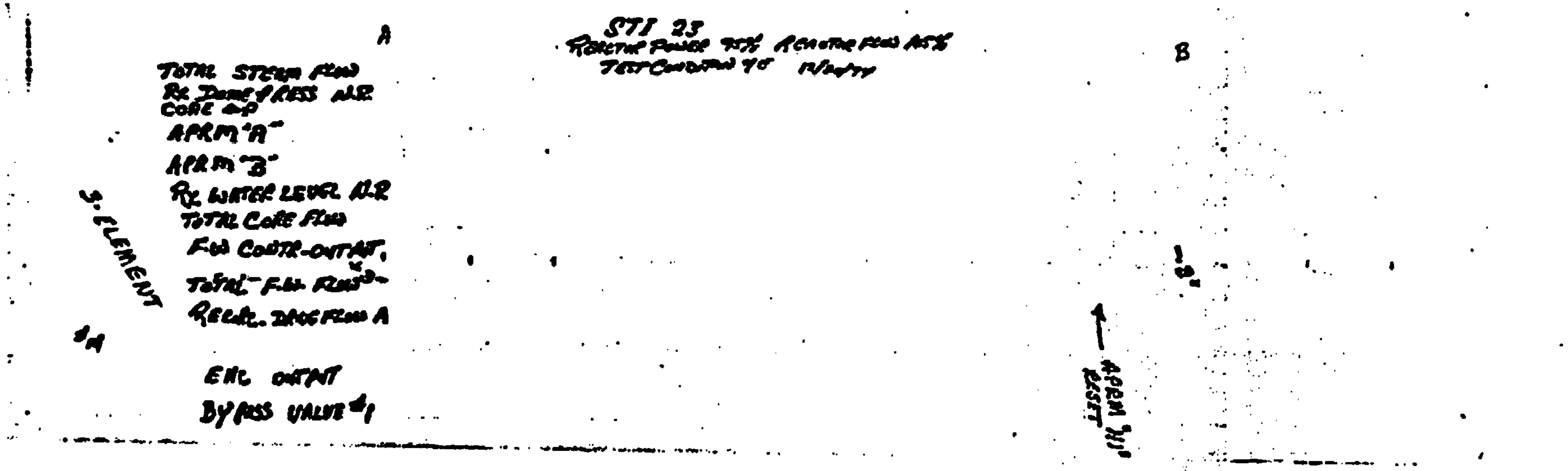

Figure STI 23-1A and B
FW Level Setpoint Changes,
±3" step, 3-element

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Cont.)

3.3.13 STI-23, Feedwater Sys (Cont.)

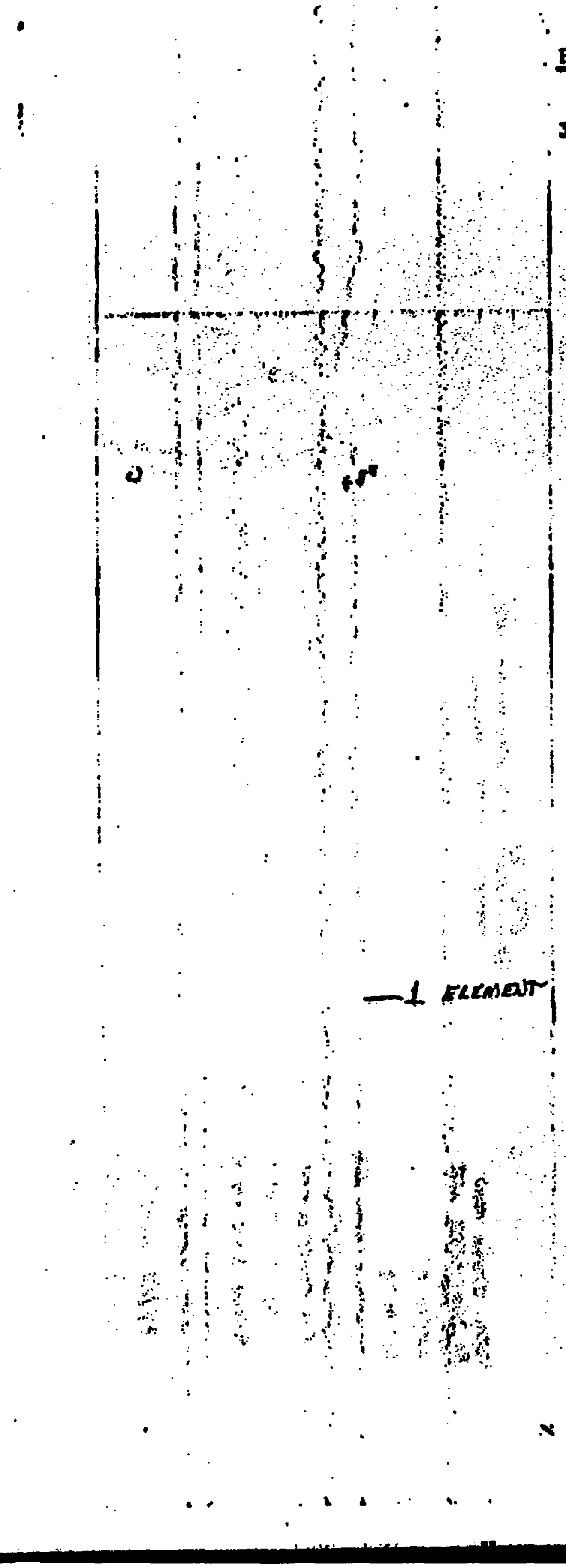

Figure STI 23-1C
FW Level Setpoint Changes,
+5" step, Single Element

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Cont.)

3.3.13 STI-23, Feedwater Sys. (Cont.)

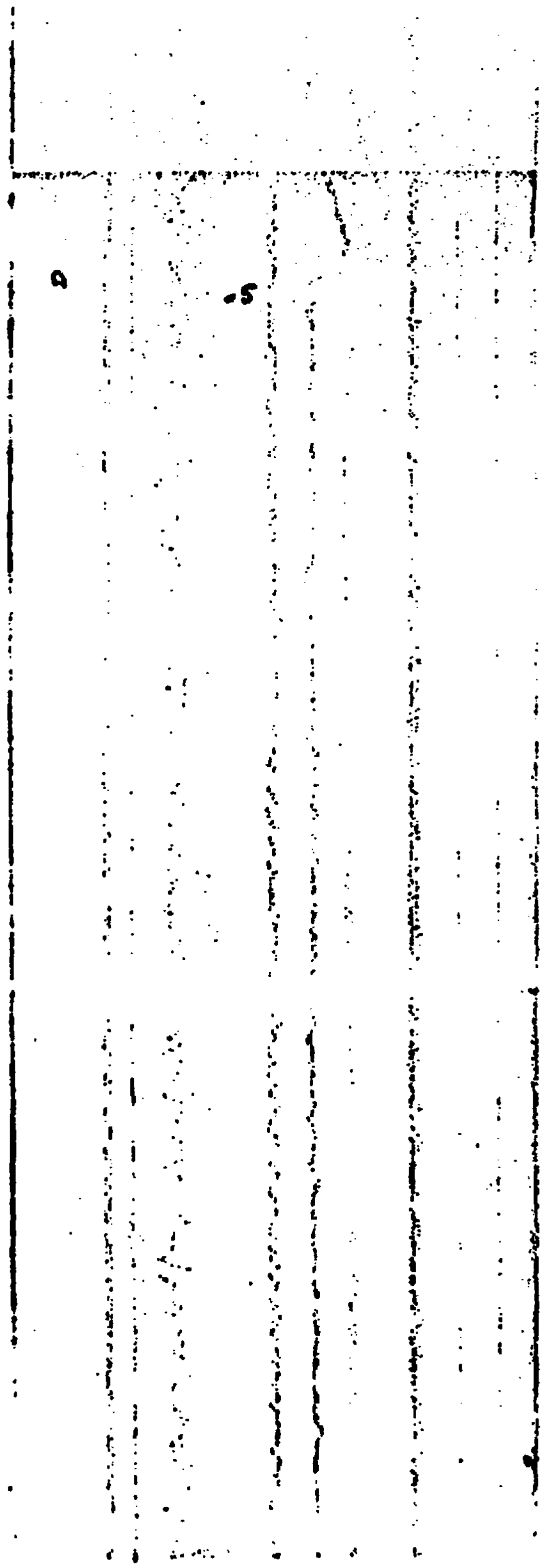

Figure STI 23-1D
FW Level Setpoint Changes,
-5" step, single element

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Cont)

3.3.13 STI-23, Feedwater System (Cont.)

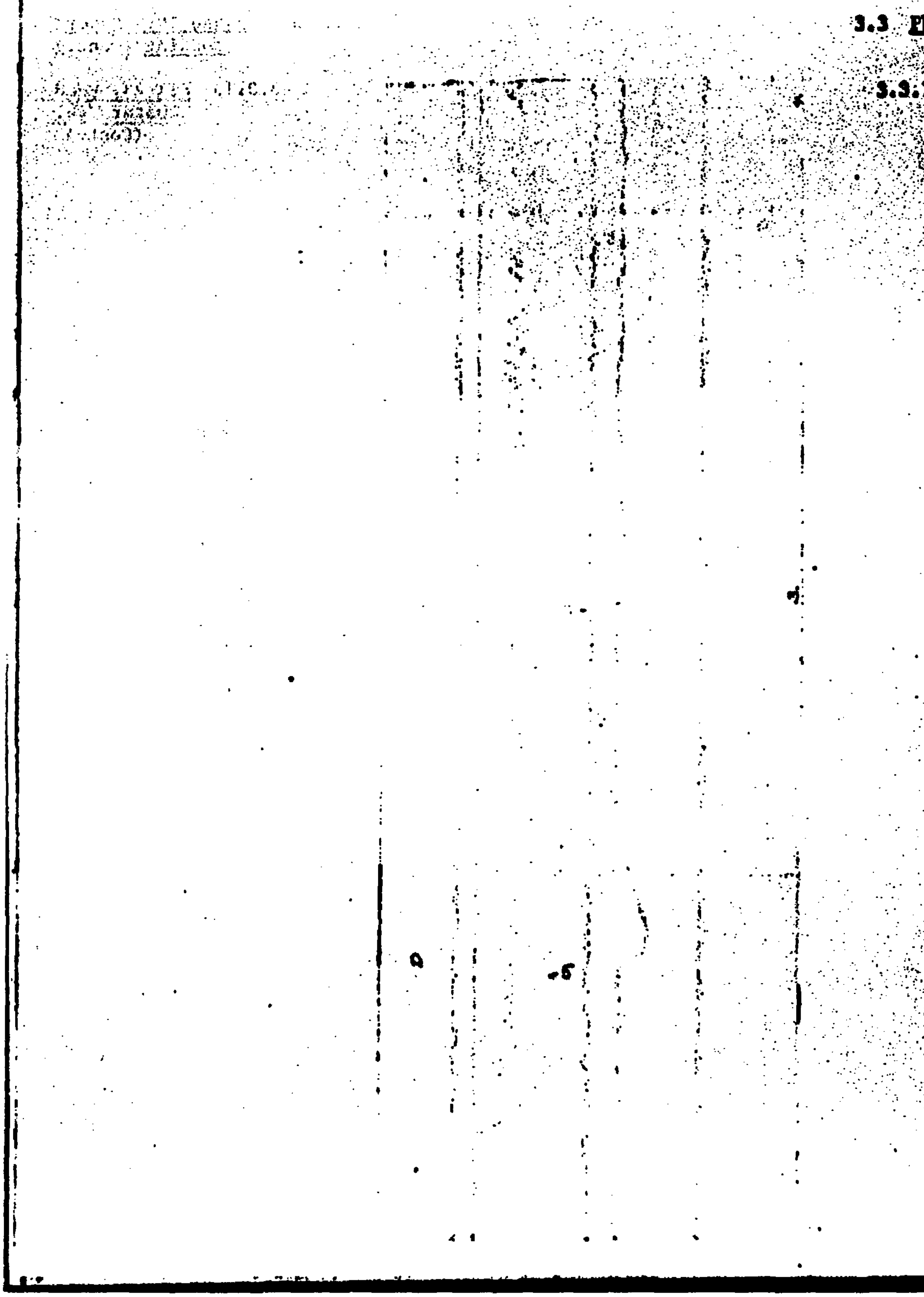

Figure STI 23-1D (Continued)
FW Level Setpoint Changes,
-5" step, single element

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.13 STI-23, Feedwater System (Continued)

Analysis (Continued)

One Pump Trip

The feedwater pump trip from test condition 4E was performed by tripping the "B" feedwater pump from a condition with all three feedwater pumps operating and the feedwater controller in the 3-element mode. Following the pump trip initiation, the sequence of events were as follows:

(1) Feedwater flow began dropping in about one second.

(2) The feedwater controller at the same time began responding to increase the speed of the two remaining pumps.

(3) At about 3.75 seconds following the pump trip, feedwater flow reached a minimum of ~10.1 Mlb/hr., a drop of 3.0 Mlb/hr., and began increasing at 4.0 seconds.

(4) Water level began a steady descent at the time of minimum feedwater flow and decreased to a minimum of 22.5 inches at ~30 seconds, which was a drop of 11.5 inches.

(5) Neutron flux decreased to about 6% at 13 seconds into the transient due to changes in inlet subcooling and continued dropping after a semi-equilibrium condition as the recirc pump runback circuitry took hold and began decreasing recirculation drive flow.

There was only a 2 inch overshoot, and equilibrium conditions were reached in about 5 minutes. The capability of the automatic recirculation flow runback feature was satisfactorily demonstrated to prevent a low water level scram following a one pump trip. Reactor power decreased 19.3 percent of rated to 78.9 percent due to the recirculation flow runback.

The overall transient was smooth and the system response was excellent, and is summarized in table 23-3. Figure 23-2 shows the transient response for the one pump trip test.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.13 STI-23, Feedwater System (Continued)

Analysis (Continued)

One Pump Trip (Continued)

Table 23-3
One Pump Trip Data Summary

Water Level Drop (initial-minimum)	11.5" drop to 22.5"
Level Overshoot	2"
Time FW Flow Start Increasing	3.5 seconds
Time to Level and Flow Steady-State	~5 minutes

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.13 STI-23, Feedwater System (Continued)

Parameter Name	Trace No.	Initial Value	Value per Division
APRM B	1	97.5	1.0%
LPRM 24-25A	2	92	1.0%
FW Pump A Trip	3	—	—
Scram Indication	4	—	—
Total FW Flow	5	13.1	0.4 Mlb/hr.
Total Steam Flow	5	14.2	0.4 Mlb/hr.
Rx Water Level NR	7	34	0.5 inch
Total Core Flow	8	100	1.0 Mlb/hr.
Control Valve Position, All	9	52	1.0%
FW Controller Output	10	—	1.0%
Rx Dome Pressure WR	11	985	10 psi
Rx Water Level WR	12	~30	10 inches
Recirc Drive Flow A	13	49	1.0 kgpm
Recirc Drive Flow B	14	47.5	1.0 kgpm
FW Pump B Trip	15	—	—
Recirc MG Speed A	16	86	2.0%
Recirc MG Speed B	17	86	2.0%
FW Pump C Trip	18	—	—

Langend for Figure STI 23-2

Feedwater Pump Trip Calibration Data

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Cont.)

3.3.13 STI-23, Feedwater Sys. (Cont.)

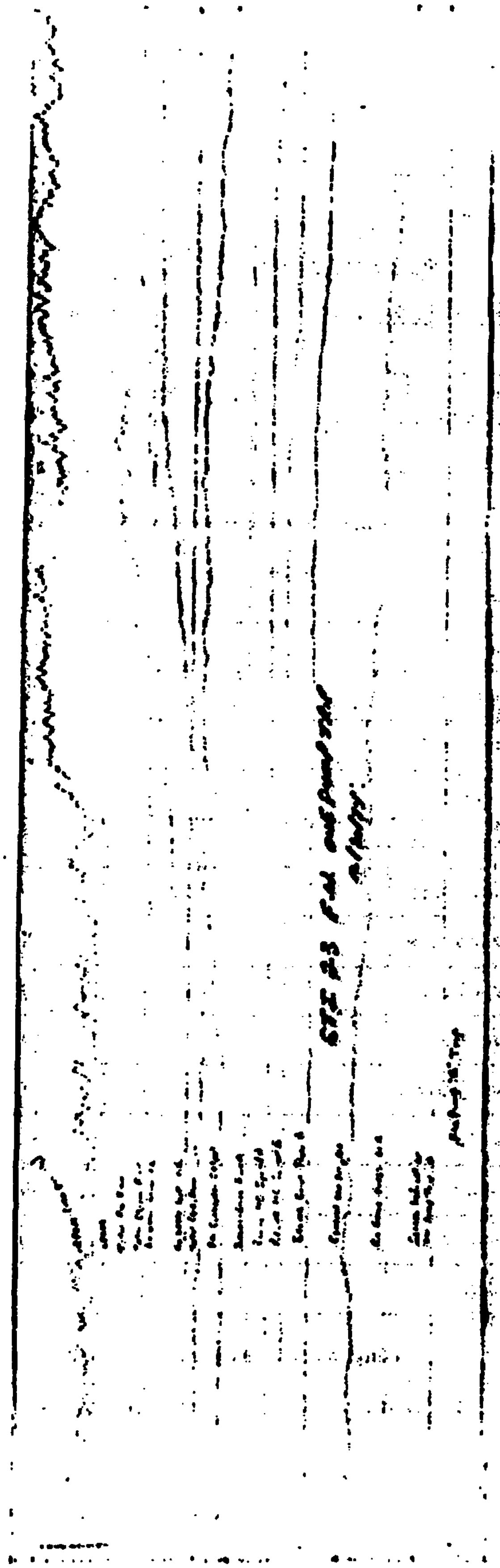

Figure STI 23-2
One Pump Trip
Start of Transient

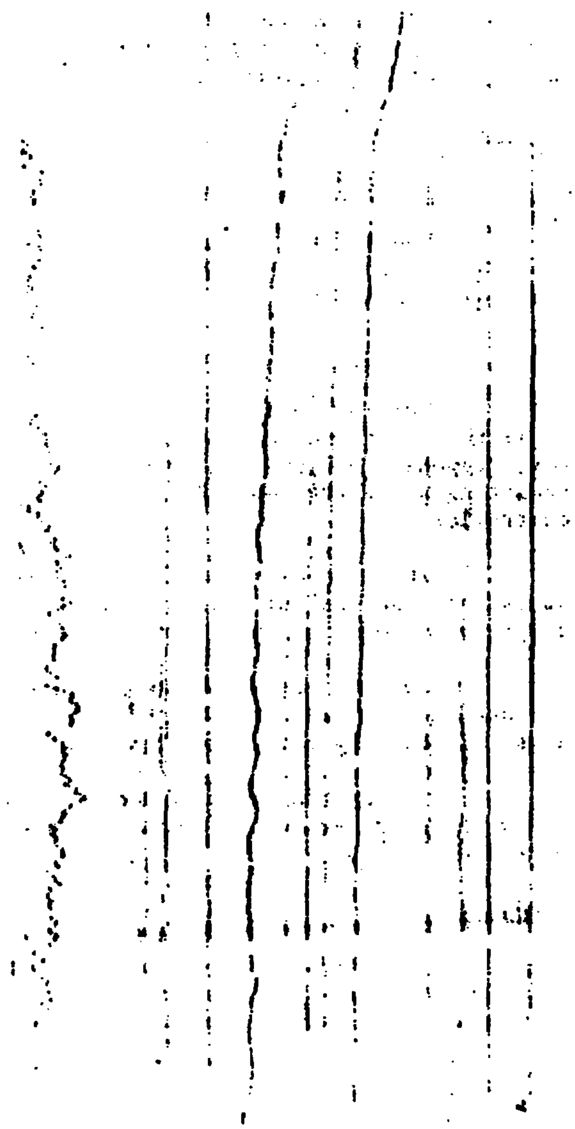

Figure STI 23-2
One Pump Trip (Continued)

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.14 STI-24, Bypass Valves

Purpose

The purposes of this test are:

(1) To demonstrate the ability of the pressure regulator to minimize the reactor pressure disturbances during an abrupt change in reactor steam flow.

(2) To demonstrate that a bypass valve can be tested for proper functioning at rated power without causing a high flux scram.

Criteria

Level 1

The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to bypass valve changes.

Level 2

(a) The decay ratio is expected to be less than or equal to 0.25 for each process variable that exhibits oscillatory response to bypass valve changes when the plant is operating above the lower limit setting of the master flow controller.

(b) To avoid approaching steam line low pressure isolation, the maximum pressure decrease at the turbine inlet during valve opening shall not exceed 50 psi.

(c) The regulator shall limit the pressure disturbance during valve reclosure so that a margin of at least 5% shall be maintained below a high flux scram.

(d) System pressure shall reach a steady-state value within 25 seconds after the bypass valve has been opened or closed.

Analysis

The ability of the pressure regulator to minimize the reactor pressure disturbance during the functional testing of a bypass valve as well as the capability to functionally

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.14 STI-24, Bypass Valves (Continued)

Analysis (Continued)

test a bypass valve at rated power without causing a high flux scram was demonstrated satisfactorily at the test conditions listed in table STI 24-1.

For test purposes, the BPV opening time was adjusted so that the valve would open in as short a time as possible. The measured valve time was as follows:

BPV opening time = 3.0 seconds
BPV closing time = 8.5 seconds

The measured bypass valve capacity in percent rated steam flow was approximately 2.7%.

Table STI 24-2 contains a summary of the bypass valve test transient data from all test conditions. In addition, figure STI 24-1A and B show the transient recorder traces from test condition 4E.

Due to problems associated with maintaining operation within the preconditioning interim operation requirements (PCIOMR), the automatic mode of the recirculation flow control could not be used. As a result, testing of the bypass valves in the auto mode has been deleted from the unit 2 startup test program. Plant operating procedures were written requiring that the recirculation flow control system be operated in the master manual and local manual modes only, therfore the auto mode was not tested. In the event the PCIOMR limitations are relaxed such that the auto mode of operation becomes possible, special testing will be performed at that time, upon approval of the NSRB, to demonstrate the operability of that control mode.

Bypass valve testing at all test conditions listed in table STI 24-1 satisfied all test acceptance criteria.

Throughout the startup test program, data was taken to extrapolate for the minimum flux margin to scram when operating at 100% rated power. The graph containing all points is shown in figure STI 24-2. Each test netted results which showed this margin to be ≥16% of rated power, which satisfies the level 2 criteria.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.14 STI-24, Bypass Valves (Continued)

Table STI 24-1
Bypass Valve Test Data

TC	Reactor Power (%)	Recirc. Flow (%)	Date Performed	Transient Trace No.	Core Thermal Power (MWt)	Core Flow (Mlb/hr.)	Reactor Pressure (psig)	FW Flow (Mlb/hr.)	Remarks
1	25	44	9/2/74	18	750	46	919	2.5	
2A	20	NC	10/26/74	42	659	30	940	1.9	
[illegible]E	60 60 56	104 104 104	9/30/74 10/6/74 10/26/74	30 34 38	1978 1976 1844	106 104.6 106	945 945 920	7.5 7.6 6.4	Recirc in Manual, Bad Transient Trace. Recirc in AUTO. Repeat Recirc in Manual.
3E	75	97	11/17/74	49	2470	98	973	9.8	
4A	44	NC	1/1/75	75	1451	35	930	5.4	
4C	59	47	12/23/74	66	1938	48	938	7.6	
4D	72	70	12/19/74	57	2373	72	955	9.3	
4E	96	100	10/22/74	62	3169	103	984	12.9	

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.14 STI-24, Bypass Valves (Continued)

Parameter Name	Trace No.	Initial Value	Value per Division
APRM A	1	96.5	1.0%
LPRM 48-41A	2	91	1.0%
Core dP	3	20.5	2.0 psi
Rx Dome Pressure NR	4	984	4.0 psi
Total FW Flow	5	12.9	0.4 Mlb/hr.
Total Steam Flow	6	13.9	0.4 Mlb/hr.
Rx Water Level NR	7	33.5	0.5 inch
Total Core Flow	8	103	1.0 Mlb/hr.
Control Valve Position, All	9	48	1.0%
Recirc. Drive Flow A	11	50	1.0 kgpm
EHC Output	12	—	2.0%
Recirc Drive Flow B	14	49	1.0 kgpm
Bypass Valve #1	18	Closed	2.0%

Legend for Figure STI 24-1
Recorder Calibration Data

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Cont.)

3.3.14 STI-24, Bypass Valves (C

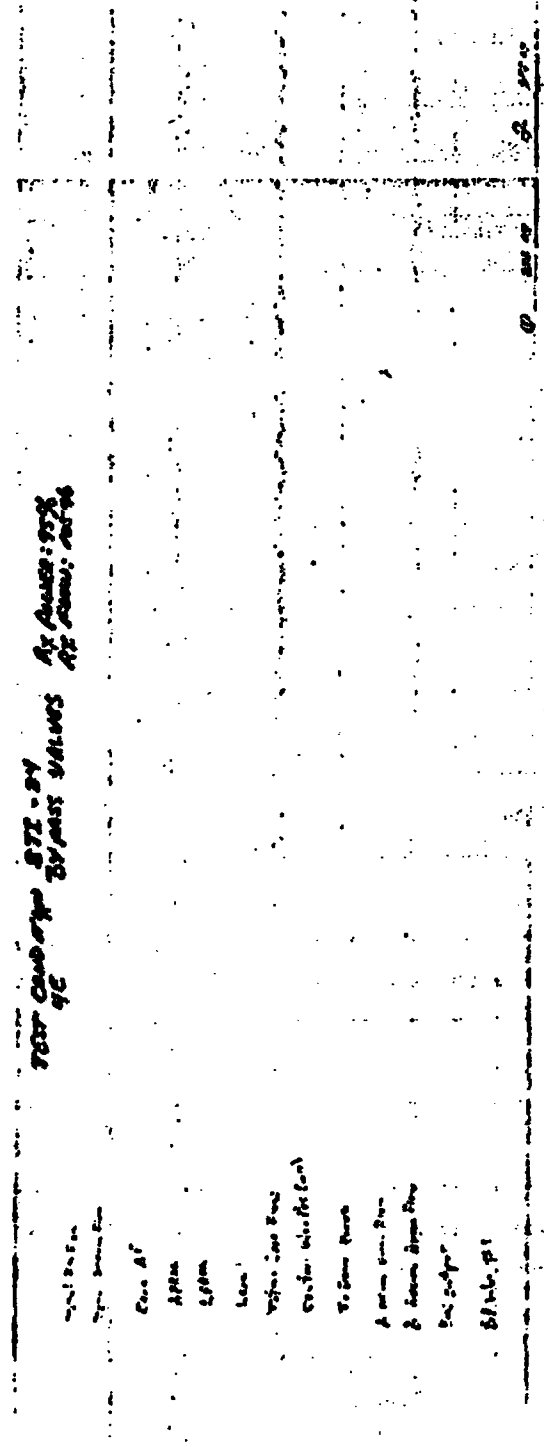

Figure STI 24-1A
HPV Opening

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Cont.)

3.3.16 STI-24, Bypass Valves (Cont.)

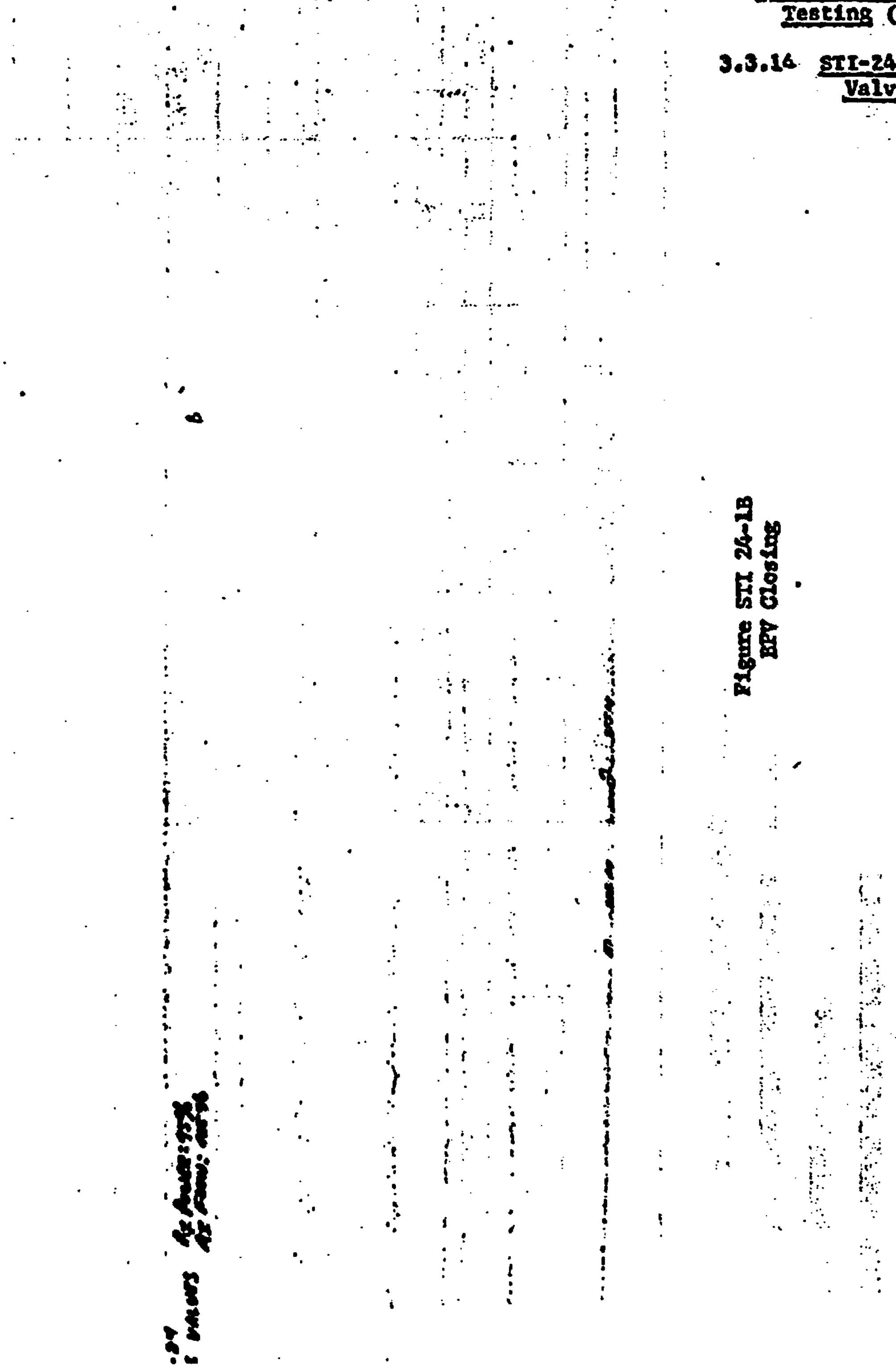

Figure STI 24-1B
HPV Closing

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.14 STI-24, Bypass Valves (Continued)

Table STI 24-2
Bypass Valve Transient Data Summary

Test Condition	Limit	1	2A	2E	3E	4A	4C	4D	4E
Recirc Mode*	---	MM	NC	MM	MM	NC	MM	MM	MM
BPV Number	---	1	1	1	1	1	1	1	1
Inital APRM (%)	---	24.5	20	55	75	50	58	70	96.5
Extrapolated Flux Margin to Scram (%)	>5%	15	16	16	16	16	16	16	16
Initial Dome Pressure (psig)	---	919	940	920	973	930	938	955	984
Dome Pressure Decrease (psi)	On Opening ≤50 psi	~0	1	2	1	2.5	2	0.75	0.8
Dome Pressure Increase (psi)	On Closing <50 psi	~0	1	2	1	1	2	0.75	0.8
Highest Decay Ratio (opening)	<0.25	<0.25 (APRM)	<0.25 (APRM)	<0.25 (APRM)	<0.25 (Rx Press)	<0.25 (Rx Wtr Lvl)	<0.25 (Rx Wtr Lvl)	<0.25 (Rx Wtr Lvl)	<0.25 (Rx Press)
Highest Decay Ratio (closing)	<0.25	<0.25 (APRM)	<0.25 (APRM)	<0.25 (APRM)	<0.25 (Rx Press)	<0.25 (Rx Wtr Lvl)	<0.25 (Rx Wtr Lvl)	<0.25 (Rx Wtr Lvl)	<0.25 (Rx Press)
Settling Time (after open)(sec)	25 sec	6	20	9	8	3	4	2	4
Settling Time (after close)(sec)	25 sec	5	11	7	7	3	2	2	5

*MM = Master
LM = Local Manual
NC = Natural Circulation

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.14 STI-24, Bypass Valves (Continued)

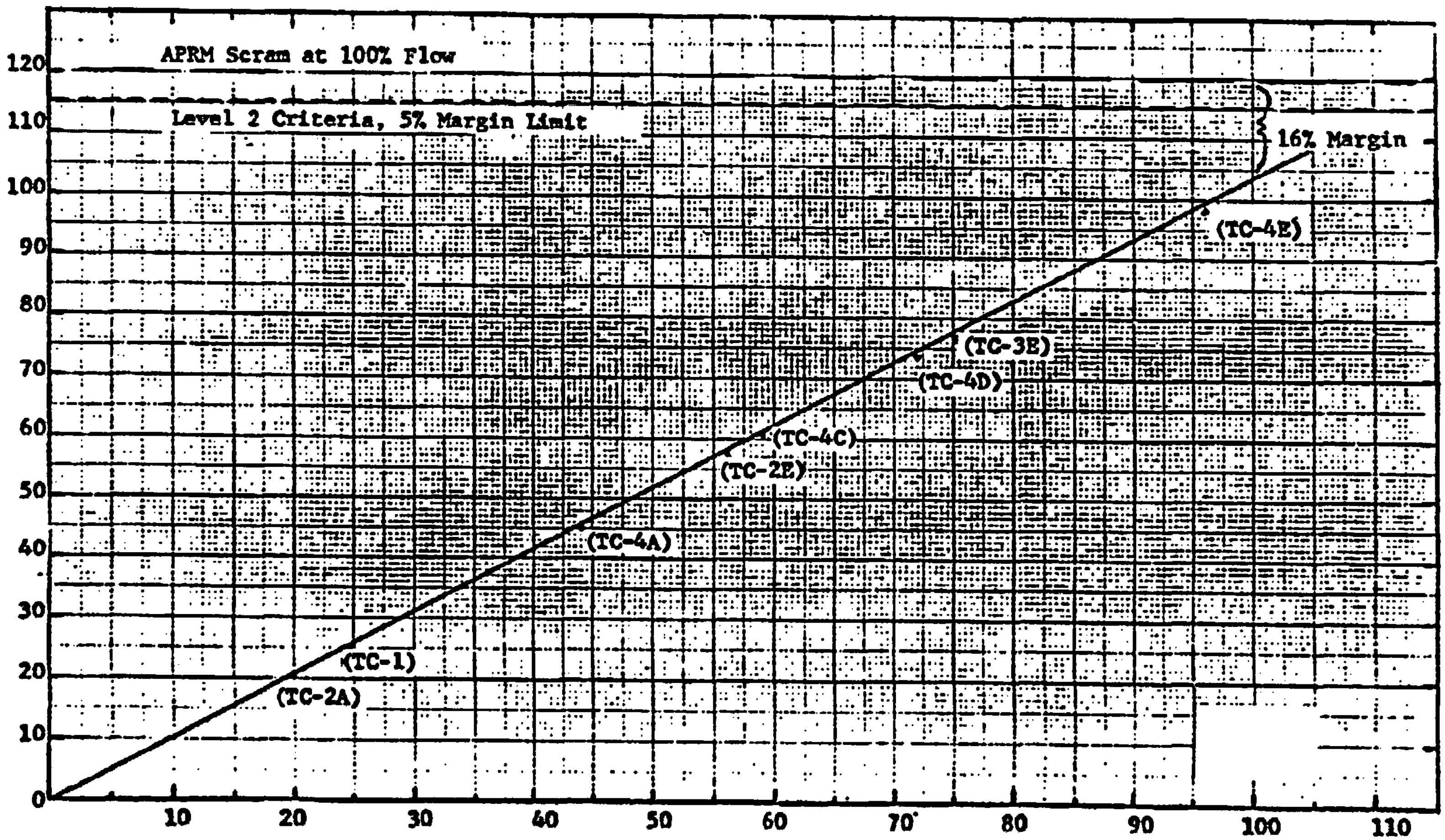

Figure STI 24-2
Bypass Valve, Flux Margin to Scram at 100% Rated Power

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.15 STI-25, Main Steam Line Isolation Valves

Purpose

The purposes of this test are to: (a) functionally check the main steam line isolation valves (MSIV) for proper operation at selected power levels; (b) determine reactor transient behavior during and following simultaneous full closure of all MSIV and following closure of one valve; (c) determine isolation valve closure time; and (d) determine the maximum power at which a single valve may be closed without causing a reactor scram.

Criteria

Level 1

Closure time must be greater than 3 and less than 5 seconds. Reactor pressure shall be maintained below 1230 psig (the setpoint of the first safety valve) during the transient following closure of all valves.

Level 2

The maximum reactor pressure should be about 1190 psig, 40 psi below the first safety valve setpoint following closure of all valves. This is a margin of safety for safety valve weeping. During full closure of individual valves, pressure must be 20 psi below scram, neutron flux must be 10% below scram, and steam flow in individual lines must be below the trip point.

Analysis

MSIV Individual Closures

The MSIV individual closure test is power dependent only, therefore the tests at condition 2E (40-60% power, ~104% flow) were performed at test condition 3C (41% power, 44% flow) to facilitate scheduling. This test was performed on November 6, 1974, and all valves closed within the required 3-5 seconds.

The peak reactor pressure following valve closures was 939 psig. The steady-state reactor pressure was 920 psig. Main steam line pressure was not affected by MSIV closure. Main steam line flow, reactor water level, and turbine inlet pressure were slightly perturbed by MSIV's closing.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.15 STI-25, Main Steam Line Isolation Valves (Continued)

Analysis (Continued)

For the testing at test condition 4, the power was reduced to ~70% to perform the test. At this condition the maximum reactor pressure was 1010 psig, a rise of 50 psi over the pretest condition, and the maximum power spike was ~8%. No isolation trips from high steam line flow occurred. All level 1 and 2 criteria were met for each of the 3 MSIV's.

Table STI 25-3 summarizes main steam isolation valve closing times for all eight MISV's at the three test plateaus of interest. After appropriate adjustments where required, all valves met both level 1 and 2 test criteria.

MSIV Simultaneous Closure

On December 21, 1974, a simultaneous full closure of all MSIV's occurred from 98% power due to vibration of high steam line flow instrumentation. Transient recorder data was obtained for all pertinent parameters and all level 1 and 2 criteria were met.

Table STI 25-3
MSIV Closure Times

MSIV Valve Number	Closure Times - Seconds		
	10% 8-26-74	TC-3C 42% 11-6-74	TC-4D 70% 1-1-75
FCV-1-14 (1A)	4.4	4.2	4.2
FCV-1-15 (2A)	4.1	4.2	4.2
FCV-1-26 (1B)	4.1	4.1	3.5
FCV-1-27 (2B)	4.4	4.3	3.4
FCV-1-37 (1C)	4.1	4.1	3.7
FCV-1-38 (2C)	4.1	4.2	3.7
FCV-1-51 (1D)	4.2	4.5	4.1
FCV-1-52 (2D)	3.9	4.6	3.6

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3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.16 STI-26, Relief Valves

Purpose

The purposes of this test are:

To verify the proper operation of the primary system relief valves.

To determine the capacity and response characteristics of the relief valves.

To verify the proper seating of the relief valves following operation.

Criteria

Level 1

None

Level 2

(a) Relief valve leakage must be low enough so that the temperature measured by the thermocouples in the discharge side of the valves returns to within 10° F. of the temperature recorded before the valve was opened.

(b) Sum total of capacity measurements from the eleven relief valves shall be equal to or greater than 8.2×10^6 lbs./hr., corrected for an inlet pressure of 1112 psig.

(c) Delay times measured with the ultrasonic translator probe during relief valve manual actuations shall be not greater than 0.4 seconds. Delay time is defined as the elapsed time from electrical initiation signal to the time the main disc starts to open. There shall be at least 1 hour elapsed from any earlier actuation of the relief valve of interest for this criteria to apply.

Analysis

All valves met timing, capacity and reseating criteria for this test. Table STI 26-1 summarizes the test data. The total measured capacity was 8.7×10^6 lb./hr. compared to the criteria of $\geq 8.2 \times 10^6$ lb./hr. The slowest

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.16 STI-26, Relief Valves (Continued)

Analysis (Continued)

delay time was 0.32 seconds on valve 1-4 compared to test criteria of <0.4 seconds. All tailpipe temperatures returned to within 10° F. of their initial temperatures.

Table STI 26-1
Relief Valve Manual Actuation Data

Date	Relief Valve No.	Corrected Measured Capacity (10^6 lb./hr.)	Delay Time (sec.)	Initial Temp. (°F.)	Final Temp. (°F.)	Reseating Acceptable (P=pass) (F=fail)
9/8/74	1-4	.732	.32	260	265	P
	1-5	.820	.27	256	250	P
	1-18	.804	.23	240	248	P
	1-19	.811	.26	250	255	P
	1-22	.764	.24	230	230	P
	1-23	.787	.27	223	229	P
	1-30	.796	.25	252	262	P
	1-31	.787	.25	250	250	P
	1-34	.804	.26	230	232	P
	1-41	.777	.28	240	215	P
	1-42	.796	.26	264	272	P
	Total Capacity - 8.678 x 10^6					

(1) Capacity corrected to design pressure of 1112 psig

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.17 STI-27, Turbine Stop and Control Valve Trips

Purpose

The purposes of this test are to (a) determine the response of the reactor system to a turbine stop or control valve trip and (b) evaluate the response at the bypass, relief valve and reactor protection systems. The parametric responses of particular interest are the peak values and the rate of change of both reactor power and reactor steam dome pressure.

Criteria

Level 1

(a) Reactor pressure shall be maintained below 1230 psig, the setpoint of the first safety valve, during the transient following fast closure of the turbine stop and control valves.

(b) Reactor thermal power, as indicated by the simulated heat flux readout, must not exceed the safety limit line defined in section 1.1.A of the technical specifications. The turbine control valves must begin to close before the stop valves during the generator load rejection. The turbine stop valves must begin to close before the control valves during the turbine trip.

(c) Feedwater system settings must prevent flooding of the steam lines following these transients.

Level 2

(a) The maximum reactor pressure should be less than 1190 psig, 40 psi below the first safety valve setpoint, during the transient following fast closure of the turbine stop and control valves. This pressure margin should prevent safety valve weeping.

(b) The measurement of simulated heat flux must not be significantly greater than pre-analysis calculations. The pressure regulator must prevent a low pressure reactor isolation.

(c) The feedwater controller must prevent a low level initiation of the HPCI and MSIV's as long as feedwater flow remains available.

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3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.17 STI-27, Turbine Stop and Control Valve Trips (Continued)

Criteria (Continued)

Level 2 (Continued)

(d) The load rejection within bypass capacity must not cause a scram. The trip scram function for higher power levels must meet RPS specifications.

Analysis

Fast closure of the main turbine stop valves was demonstrated at 100% of rated reactor conditions.

Sequence of events and the times respective to the turbine stop valve trip are presented in table STI 27-1 below. Table STI 27-2 summarizes the principal parameters measured.

Table STI 27-1
Sequence of Events for Main Turbine Stop Valve Closure

Time (sec.)	Event
0	Turbine Trip
.15	Start of SV Closure
.19	Start of CV Closure and BPV Opening
.21	Reactor Scram
.25	Maximum Heat Flux
.39	CV Full Closed
N/A	BPV Full Open
~3.9	Peak Reactor Pressure
~5 sec.	Minimum Reactor Water Level Reactor Isolation
100 sec.	Maximum Reactor Water Level

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.17 STI-27, Turbine Stop and Control Valve Trips (Continued)

Table STI 27-2
Summary of Principal Parameters
Turbine Stop Valve Trip from 100%

Parameter	Expected	Actual
Peak Reactor Pressure	<1190 psi	1085
Maximum Heat Flux		<1% increase
Minimum Reactor Water Level	>Lo-Lo-Trip (-31")	-35 inches
Maximum Reactor Water Level	<steam line (~130")	64 inches
Control Valve Closure Time	0.2 sec.	.2 sec.

Analysis (Continued)

Fast closure of the main turbine control valves was demonstrated at 25% and 100% of rated reactor conditions.

At the 25% reactor condition, the transient was within the capacity of the bypass valve system as these valves open fast enough to virtually eliminate a pressure spike.

A low-low water level insolation occurred during both STI-27 trips from 100% power. It was not within the capability of the feedwater control system to prevent the low water level condition from occurring as required by the level 2 criteria. For the generator load rejection, the test was purposefully started from 7" higher than normal level to prevent the isolation. However, the initial drop was too much and too swift for the feedwater system to compensate. Since a reactor isolation does not constitute a safety problem, the results of this test were accepted as satisfactory. The low level isolation problem is being reviewed by GE and TVA DED.

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.17 STI-27, Turbine Stop and Control Valve Trips (Continued)

Table STI 27-3
Fast Closure of Main Turbine Control Valve Data

Time Seconds	Event
0.0	Initiate Generator Trip (Main Breaker #1)
~0.07	Start of CV Closure and BPV Opening
.184	CV Full Closed
0.53	Reactor Scram
0.37	BPV FUll Open
~3.5	Peak Reactor Pressure
~3.8	Minimum Reactor Water Level Reactor Isolation
~43.5	Maximum Reactor Water Level

3.3.18 STI-30, Recirculation System

Purpose

To evaluate the recirculation flow and power level transients following trips of one or both of the recirculation pumps.

To obtain recirculation system performance data.

To verify that no recirculation system cavitation will occur in the operable region of the power flow map.

Criteria

Level 1

MCHFR shall be greater than 1.0 during the pump-trip transient.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.18 STI-30, Recirculation System (Continued)

Criteria (Continued)

Level 2

For each pump trip test, the minimum transient MCHFR based on operating data divided by the corresponding minimum transient MCHFR evaluated from deisgn values is expected to be equal to or greater than 1.0.

Analysis

Recirculation system performance data was taken on the 50% flow control line at various combinations of pump speeds as well as at each end of the 75% and 100% flow control lines. Performance of the system was satisfactory at all conditions.

A test for cavitation in the recirculation system was performed on the 50% flow control line by inserting control rods in the reverse of the nominal rod sequence "A", while recirculation pumps were maintained at a speed to give approximately 104 Mlb./hr. total core flow. Initially, the recirculation pump runback was encountered at approximately 26% power. The runback setpoint was lowered and the test performed again. In this retest, the runback was encountered at approximately 22.5% power. No indications of cavitation were seen in either test. During the resetting of the runback setpoint, a cavitation search was performed down to approximately 20% power, with the feedwater runback interlock jumpered. No cavitation occurred.

Recirculation pump trips were performed at 50%, 75%, and 100% power with 100% flow, by tripping the pump drive motors. Single pump trips were performed at 50% and 100% power with 100% flow by tripping the drive motors. A single pump trip was performed at 50% power with 100% flow by opening the generator field breaker on pump "B." A transient minimum critical heat flux ratio (MCHFR) analysis was made for the 50% and 100% power two pump trips, and for the 1-pump trip initiated by opening the generator field breaker. The analysis was performed at one second intervals during the transient by using the off line time share computer (BUCLE). Although not required by test criteria, transient minimum critical power ratio (MCPR) analyses were performed for the same tests as were the MCHFR analyses.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.18 STI-30, Recirculation System (Continued)

Analysis (Continued)

MCPR remained well above the 1.05 limit during all analyzed transients. MCHFR remained above 1.0 during all the transients. This satisfies the level 1 criteria. The level two criteria requiring that the ratio of the minimum transient operating MCHFR divided by the minimum transient MCHFR be greater than 1.0 was satisfactorily met for all analyzed transients. At test condition 2E (~50% power) the steady-state MCHFR was 4.10, while the predicted experimental MCHFR was 4.25 for the 1-pump generator field breaker trip. The steady-state MCHFR is a function of the rod pattern prior to the initiation of the pump trip, and does not violate the criteria. MCHFR immediately began to rise in magnitude during the transient and was above the predicted values during the transient, therefore the level 2 criteria were satisfactorily met.

Figures STI 30-1 through STI 30-3 compare plant parameters and MCHFR, as calculated from the transient traces and BUCLE, with the predicted behavior for all analyzed trips. Table STI 30-1 shows MCHFR and MCPR behavior during the two pump trip from 100% power. This test was the most limiting case of the analyzed trips.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.18 STI-30, Recirculation System (Continued)

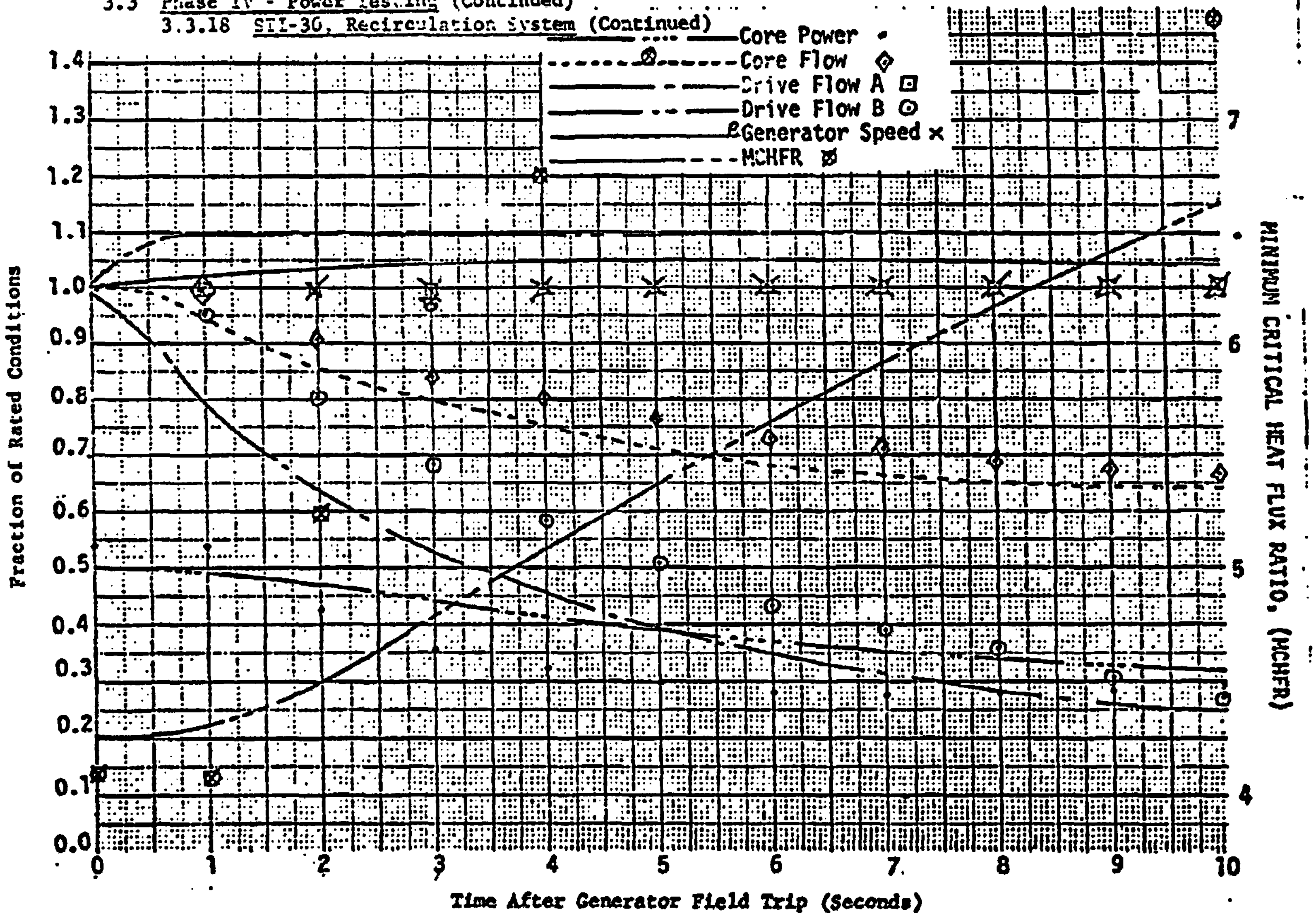

Figure STI 30-1

3293 MWt Plant Recirculation System Performance

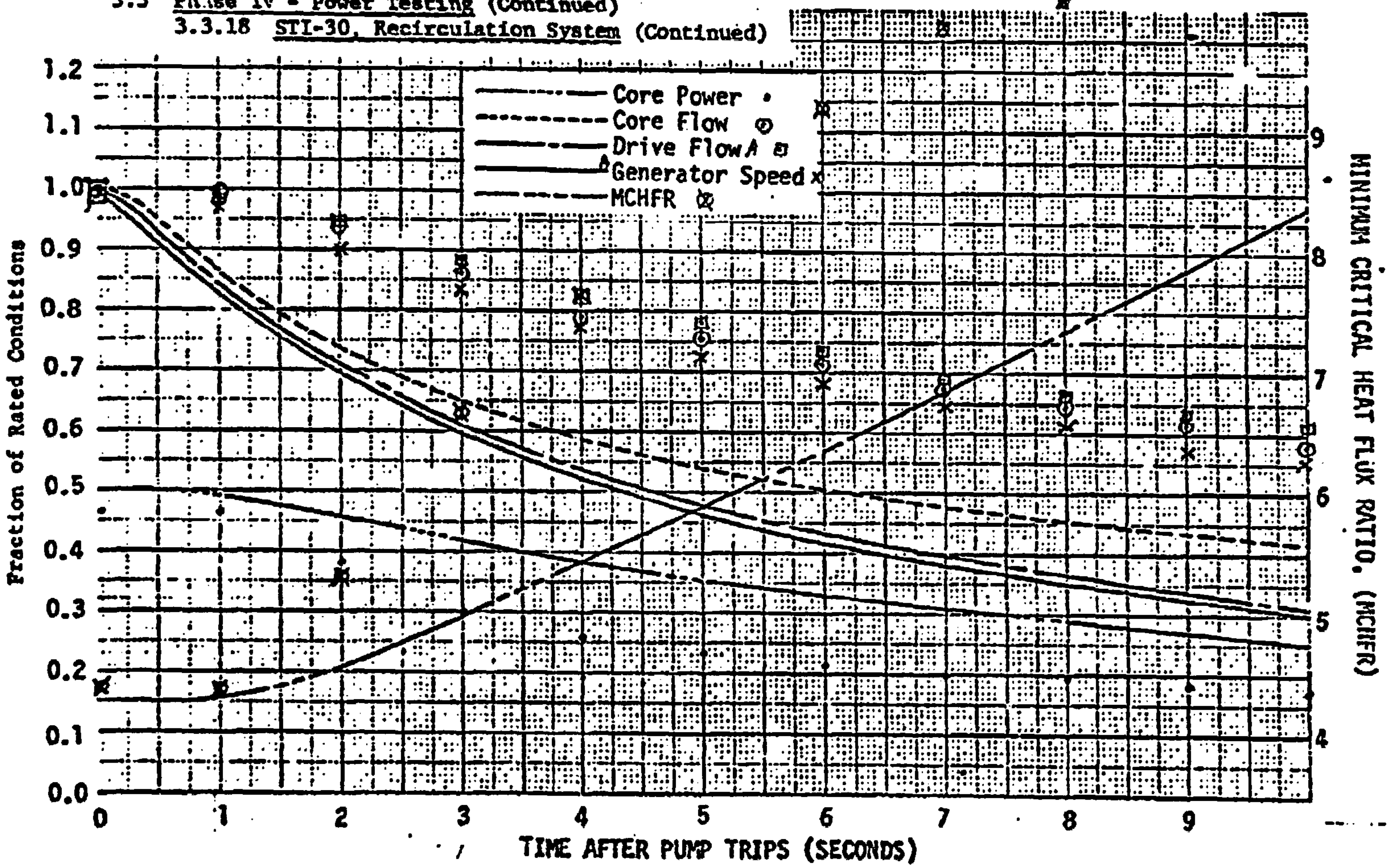

Figure STI 30-2

3293 MWt Plant Recirculation System Performance
Following Trip of Two Drive Motors - 50% Power

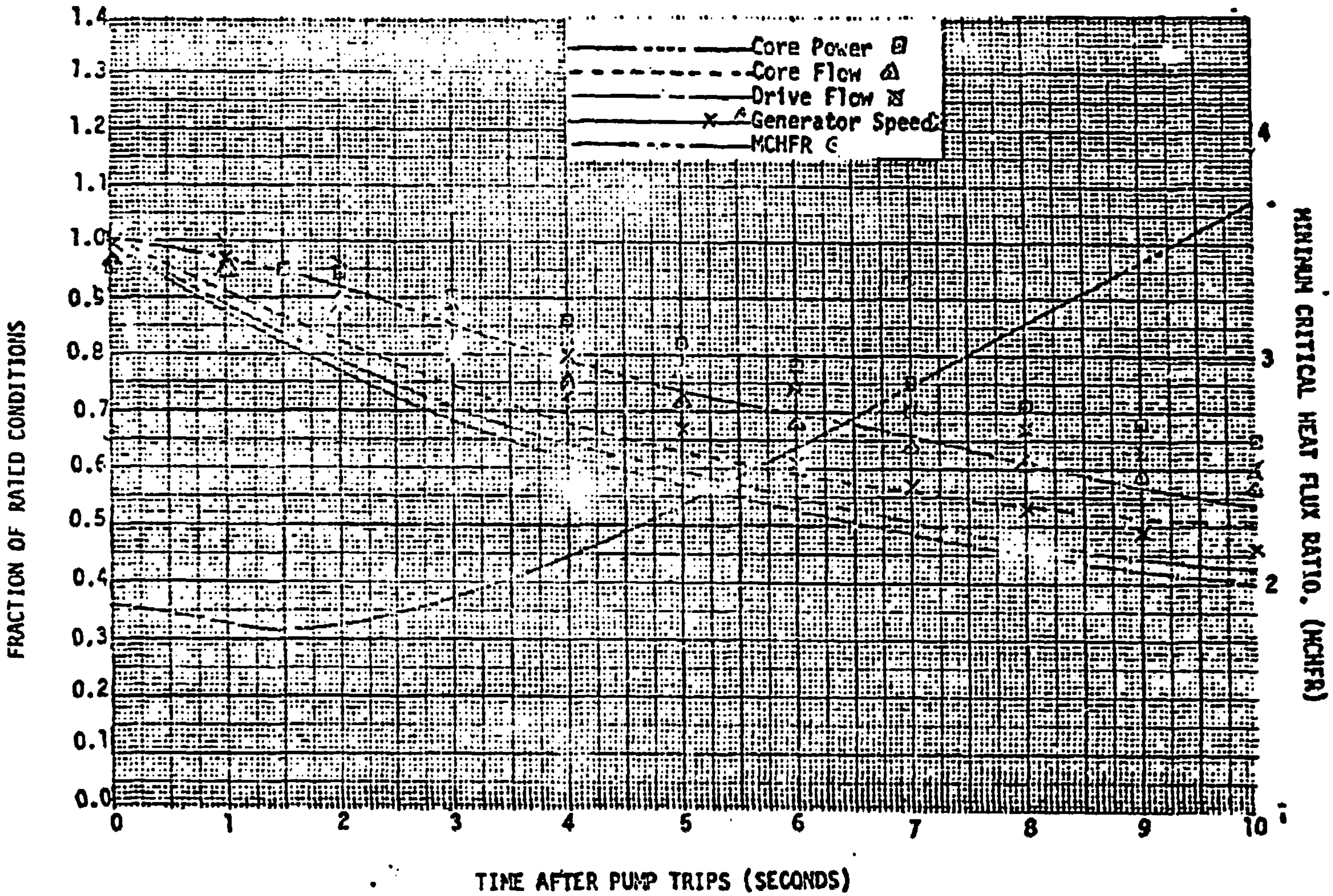

Figure STI 30-3
3293 MWt Plant Recirculation System Performance
Following Trip of Two Drive Motors - 100% Power

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.18 STI-30, Recirculation System (Continued)

Table STI 30-1

Transient MCPR and MCHFR
Two Pump Trip

Time After Trip (Sec.)	MCPR Value	Margin to 1.05 Limit	P/F	MCHFR Value	Margin to 1.0 Value	P/F
0	1.257	0.207	P	2.65	1.65	P
1	1.251	.201	P	2.64	1.64	P
2	1.223	.173	P	2.63	1.63	P
3	1.188	.138	P	2.67	1.67	P
4	1.116	.066	P	2.77	1.77	P
5	1.180	.130	P	2.96	1.96	P
6	1.185	.135	P	3.14	2.14	P
7	1.189	.139	P	3.33	2.33	P
8	1.214	.164	P	3.60	2.60	P
9	1.222	.172	P	3.76	2.76	P
10	1.241	.191	P	3.96	2.96	P

3.3.19 STI-31, Loss of Turbine Generator and Offsite Power

Purpose

To investigate the reactor transient performance during the loss of the main generator and all offsite power.

To demonstrate the acceptable performance of the station electrical supply system during the loss of the main generator and all offsite power.

Criteria

Level 1

(a) Reactor pressure shall be maintained below 1230 psig, the setpoint of the first safety valve, during the transient.

(b) All safety systems, such as the reactor protection system, RCIC, HPCI, and diesel generators, must function properly without manual assistance.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.19 STI-31, Loss of Turbine Generator and Offsite Power (Continued)

Criteria (Continued)

Level 2

(a) The maximum reactor pressure should be less than 1190 psig, 40 psi below the first safety valve setpoint, during the transient. This pressure margin should prevent safety valve weeping.

(b) Normal reactor cooling systems should be able to maintain adequate suppression pool water temperature, maintain adequate drywell cooling, and prevent actuation of the auto-depressurization system.

Analysis

The test was successfully performed on September 8, 1974. The reactor was at ~31% power with the main generator output at 277 MW. Prior to the test, the plant electrical system was aligned so that the only source of power to the unit 2 auxiliaries was the unit 2 station service transformer and also, so that the test would not affect unit 1 operation. The auxiliary electrical shutdown system was aligned so that only 4-kV shutdown board "C" was feeding from unit 2. The unit 2 480 volt shutdown boards and motor operated valve boards were isolated so that their sole source of power was 4-kV shutdown board "C". In this configuration, all unit and shutdown auxiliaries for unit 2 were being fed from the unit 2 main generator and could not be supplied by unit 1 or the plant common electrical system.

The reactor was operating at steady-state conditions with normal feedwater control for one reactor feedwater pump operation, when the turbine generator was tripped by manual operation of the generator negative phase sequence relay.

Reactor Response

As anticipated, the reactor scrammed due to control valve fast closure when the turbine tripped. The reactor feedwater pump tripped immediately due to the loss of NPSH which was caused by the automatic tripping of the condensate booster pumps. The reactor water recirculation pumps also tripped immediately on undervoltage.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.19 STI-31, Loss of Turbine Generator and Offsite Power (Continued)

Analysis (Continued)

Reactor Response (Continued)

Thirty seconds after the reactor scram, the reactor isolated by MSIV closure due to the RPS M-G sets tripping on undervoltage. The reactor water level continued to fall until the RCIC pump was manually started six minutes after the scram. The water level at that time was 27 inches below normal operating level. The level dropped six more inches before the RCIC pump began restoring coolant inventory.

Peak reactor pressure (1084 psig) was reached eight minutes after the scram when three relief valves automatically operated. At that time scram recovery operations were initiated with reactor level and pressure controlled by RCIC and manual operation of relief valves.

During the turbine trip, reactor scram, and reactor isolation, the reactor parameters followed their expected increasing and decreasing patterns. Drywell and suppression pool temperatures and pressures remained essentially unchanged during the transient.

Electrical Response

Following the main generator trip, the unit and reactor recirculation pump board feeder breakers immediately tripped on undervoltage, resulting in a loss of voltage to all unit and shutdown auxiliaries associated with unit 2. 4-kV shutdown board "C" automatically isolated itself 5.4 seconds after the generator tripped and voltage was automatically restored 1.8 seconds later by diesel generator "C". All other automatic tripping and switching operations occurred in the time sequences expected.

3.3.20 STI-32, Recirculation Speed Control and Load Following

Purpose

To determine correct gain for optimum performance of individual recirculation loops.

3.0 Results (Continued)

3.3 Phas IV - Power Testing (Continued)

3.3.20 STI-32, Recirculation Speed Control and Load Following (Cont.)

Purpose (Continued)

To determine that the recirculation loops are correctly set up for desired speed range and for acceptable variations in loop gain.

To demonstrate plant response to changes in recirculation flow.

To determine that the load following loop operates acceptably over the desired range of recirculation flow.

Criteria

Level 1

The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to flow control changes.

Level 2

(a) The decay ratio should be less than 0.25 for any process variable that exhibits oscillatory response to 10% speed change inputs in local or master manual modes.

(b) In automatic mode the flow control range limits will be set to include that portion of the total flow range over which the decay ratio is less than 0.25.

(c) Steady-state limit cycles, if any exist, must not cause turbine steam flow to vary in excess of $\pm 0.5\%$ rated flow as measured by the gross generator electrical power output.

Analysis

The recirculation flow control system was set up during the initial heatup for good stable operation. Existing diode type demodulators in the Bailey positioners were replaced with a transistorized type and the slave driver remained as a part of the system. These units were fine-tuned for maximum tracking ability and their performance was acceptable. The initial settings of the system

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.20 STI-32, Recirculation Speed Control and Load Following (Cont.)

Analysis (Continued)

controllers were as follows:

Loop "A"	1000% P.B.	15 R/M
Loop "B"	1000% P.B.	15 R/M
Master	2000% P.B.	2 R/M

As power was increased to the various test conditions, it was noted that the two loops performed adequately, however some problems were encountered in that the two loops did not track each other as desired. More adjustment and some component replacement on the Bailey positioners improved this condition but never completely relieved it.

On the 100% flow control line at rated conditions, it was found that the settings on the system loop controllers must be changed to 2000% proportional band to maintain the rate of power change to less than 15% per minute on a larger step change in speed command (10%). This meets the restrictions of the PCIOMR. Under these conditions, all of the criteria of the test were met. The system response was stable but heavily damped. The "A" loop did tend to drift a small amount under steady-state conditions but within the tolerance of the criteria. Representative results of test step changes for the initial and final controller settings are shown in table STI 32-1.

The M-G set scoop tube positioner gain curves are shown in figure STI 32-1. It can be seen that the curves are slightly exponential which results in better speed control capability.

The portion of the test which required operation in the automatic mode was not completed and was carried as an exception on the final test report. This part of the test can not be accomplished until the Nuclear Safety Review Board approves the automatic mode of testing.

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3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.21 STI-32, Recirculation M-G Set Speed Control (Continued)

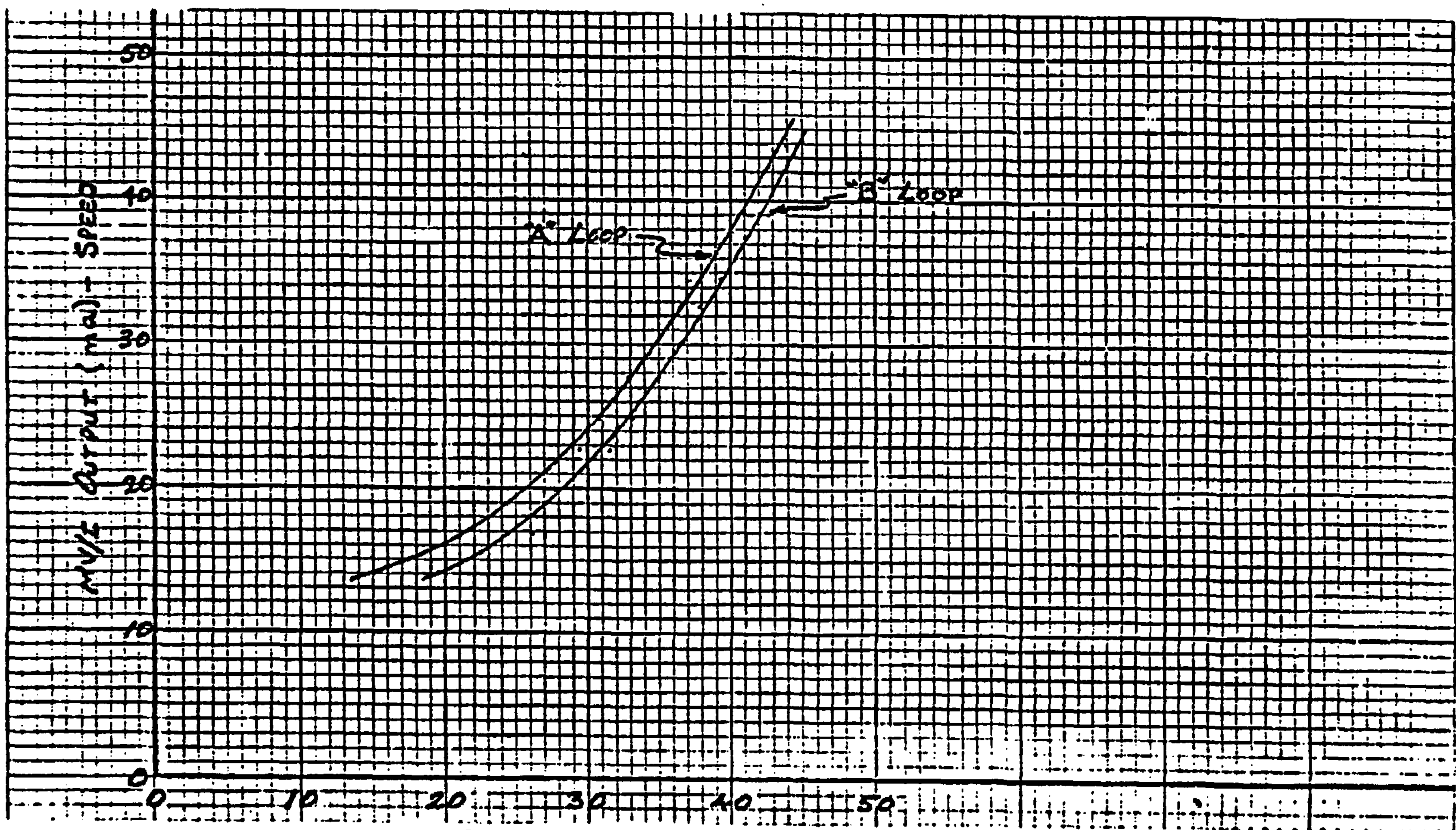
Table STI 32-1
Representative Results of Recirculation Controller System Testing

Test Condition	Flow Mlb/hr	Power MWt	Controller Settings	Total Flow Step, %	ΔP Rx psi	ΔFlux %
2E	104.5	1958	Master PB=2000 R/M=2 A PB=1000 B PB=1000 A & B R/M=15	-10	0	-2.5
3E	99	2470	Same as Test Cond. 2E	-5	-1	-3
4E	100	3262	Master PB=2000 R/M=2 A & B PB=2000 R/M=15	-10	-3	-8

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.20 STI-32, Recirculation Speed Control and Load Following (Continued)



Controller Output (ma)
(Input to Scoop Tube Positioner)

Figure STI 32-1

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.21 STI-33, Main Turbine Stop Valve Surveillance Test

Purpose

The purpose of this test is to demonstrate acceptable procedures for the daily stop valve surveillance testing at a power level as high as possible without producing a reactor scram.

Criteria

Level 1

Not applicable.

Level 2

(a) Peak neutron flux must be at least 5% below the scram trip setting. Peak vessel pressure must remain at least 10 psi below the high-pressure scram setting.

(b) Peak steam flow in the main steam lines must remain 10% below the high flow isolation trip setting.

Analysis

Turbine stop valves were closed individually at selected power levels. The design of the main steam lines provides cross ties upstream of the stop valves, which prevent large perturbations as a result of the stop valve closure. Table STI-33-1 summarizes the test results. Included are the parameters of greatest interest: Peak neutron flux; peak vessel pressure; and peak steam line flow. Little perturbation was observed due to the valve closures, and the peaks listed in table STI 33-1 are not much different than one would observe as background variation at steady state. STI-33 demonstrated that the stop valve surveillance test may be satisfactorily performed at full power. All test criteria were met.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.21 STI 33, Main Turbine Stop Valve Surveillance Test (Continued)

Table STI 33-1
Summary of Turbine Stop Valve Closure Data

Reactor Power	25%	59.5%	74.9%	99.2%
Peak Neutron Flux	26%	60.5%	78%	101%
Margin to Limit	>10%	>10%	>10%	>10%
Peak Vessel Pressure	919 psi	950 psi	970 psi	988 psi
Margin to Limit	12.9%	10%	8.05%	7.3%
Peak Steam Line Flow	1.06 Mlb/Hr	2.25 Mlb/Hr	2.8 Mlb/Hr	3.95 Mlb/Hr
Margin to Limit	75%	48%	40%	15%

Note that data above are peak values at each test level. The margin to limits are defined by the following trip limits:

Neutron Flux - Depends on peaking factor set in
Vessel Pressure - 1035 psig
Peak Steam Flow - 140% rated flow, ~ 4.7 Mlb/hr

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3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.22 STI-34, Vibration Measurements

Purpose

The purpose of this test is to obtain vibration measurements on various reactor components to demonstrate the mechanical integrity of the system to flow induced vibrations and to check the validity and accuracy of the analytical vibration model.

Criteria

The vibration criteria, used to judge the results of the vibration measurements, is the precalculated vibration amplitude at each sensor when the maximum stress in any one of the internal's structures or components equals 10,000 psi including stress concentration factors. This stress represents approximately one half the stress limit given in ASME Code Section III for 40-year life. Because of their complexity the criteria are not presented here but will be evaluated by the GE vibration test engineer conducting the test.

Analysis

Vibration test data was taken in conjunction with the recirculation pump trips, one pump at a time and then both pumps simultaneously.

These tests were conducted at 50, 75, and 100 percent power levels.

The General Electric Vibration Specialist states that the vibration amplitudes are well within criteria limits. A final evaluation will be made at a later date by the TVA central office staff or by an independent consultant.

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3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.23 STI-35, Recirculation System Flow Calibration

Purpose

The purpose of this test is to perform a complete calibration of the installed recirculation system flow instrumentation.

Criteria

Level 1

Not applicable.

Level 2

(a) Jet pump flow instrumentation shall be adjusted such that the jet pump total flow recorder will provide a correct core flow indication at rated conditions.

(b) The APRM/RBM flow-bias instrumentation shall be adjusted to function properly at rated conditions.

Analysis

Recirculation system flow data was taken on the 50%, 75%, and 100% flow control lines. Redundant data was taken at each test condition and calculations were made on each set, utilizing the vendors' computer system (JPUMPS and RPUMPS programs). The results were analyzed at each test condition; however, it was not necessary to make any instrument adjustments until rated conditions were attained. At this point the data analysis revealed that the indicated recirculation flow was approximately 10% higher than calculated values. New flow nozzle coefficients were determined based on the calculated valves of flow using pump characteristics. From the new nozzle coefficients, a new flow transmitter calibration range was determined.

Final calibration of the core flow measuring system was based on the data taken at rated conditions. Comparison of the double tap jet pump flows with the single tap jet pump flows show that they are within approximately 2% of each other. In addition, the M-ratios are within the band of expected theoretical values. The gain adjustment factors and the as-left gains are as follows:

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3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.23 STI-35, Recirculation System Flow Calibration (Continued)

Analysis (Continued)

Loop	Instrument Gain Adjustment Factor	As Left Gains
A	1.041	0.5205
B	1.04	0.52

The APRM/RBM flow bias instrumentation was found to perform its function adequately at rated conditions. However, an unresolved problem was encountered in this area. It was found that a non-linearity exists between the recirculation flow (drive flow) and the core flow which causes premature rod-blocks at low power levels. The flow bias instrumentation is set up at rated conditions. When the flows are reduced, the non-linearity causes the drive flow to decrease at a faster rate than the core flow, creating a rod-block condition. The problem has not been solved at this time; however, the vendor has proposed a solution and is presently undergoing evaluation.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.24 STI-72, Drywell Atmospheric Cooling System

Purpose

The purpose of this test is to verify the ability of the drywell atmospheric cooling system to maintain design conditions in the drywell during operating conditions.

Criteria

Level 1

None.

Level 2

(a) The heat removal capability of the drywell coolers shall be approximately 5.19×10^6 Btu/hr.

(b) The drywell cooling system shall have a standby capability of $\geq$ 25% of the design heat removal capability.

(c) The drywell cooling system shall maintain temperatures in the drywell below the following design values during normal operation.

During Normal Reactor Operation:

•135° F. (57° C.) average throughout drywell

•50% relative humidity

•128° F. (53.4° C.) maximum around the recirc. pump motors

•150° F. (65.5° C.) maximum for all other areas

•200° F. (65.5° C.) maximum above the bulkhead

Ten Hours After Shutdown:

Within 15° F. (8.3° C.) of closed cooling water inlet temperature in all areas beneath the vessel-to-drywell bulkhead.

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3.0 Results (Continued)

3.3 Phase IV, Power Testing (Continued)

3.3.24 STI-72, Drywell Atmospheric Cooling System (Continued)

Criteria (Continued)

Cooling Water Supply:

•100° F. maximum

Uniform Circumferential Temperature at Which the Refueling Bellows/Bulkhead Assembly Must be Maintained:

•Within 25° F. maximum point-to-point variation

Analysis

At 100% power all points within the drywell met the 150° F. maximum temperature criteria with the exception of TE-80-13 and TE-80-14. The average drywell temperature was 124° F. It should be noted that RBCCW temperature into the drywell was 83.4° F. It is expected that the inlet temperature of RBCCW into the drywell will approach 105° F. during the summer months. At these conditions it is doubtful that drywell temperatures would meet design criteria. DED will evaluate and resolve this problem.

The drywell heat load was measured to be 6.11×10^6 Btu/hr at 100% power. This compares favorably with the design heat load at 6.09×10^6 Btu/hr with the drywell sump pump cooler out of service.

Data was taken before and after one of the full power scrams to illustrate the ability of the RBCCW system to handle scram heat loads. Although at ten hours after the scram the average drywell temperature had not decreased to within 15° F. of the RBCCW inlet temperature, only 8 cooling coils and fans were in service during this time. Ten coils and fans were placed in operation and the temperature decreased to within 15° F. of the RBCCW inlet temperature.

3.0 Results (Continued)

3.3 Phase IV - Power Testing (Continued)

3.3.25 STI-73, Cooling Water Systems

Purpose

The purpose of this test is to verify that the performance of the reactor building closed cooling water (RBCCW) system is adequate.

Criteria

Level 1

None

Level 2

(a) Verification that the system performance meets the cooling requirements constitutes satisfactory completion of this test.

(b) RBCCW was designed to transfer maximum heat load of 31.3×10^6 Btu/hr in order to limit equipment inlet water temperature to 100° F., assuming a service (raw cooling) water inlet temperature of 90° F.

Analysis

The heat load as measured on the RBCCW side of the heat exchangers was 21.85×10^6 Btu/hr. Assuming design heat load on the fuel pool heat exchangers, the RBCCW heat exchangers heat transfer rate would be within 2% of the design value of 31.3×10 Btu/hr.

Due to attempts to raise RBCCW temperature into the drywell to 100° F. for more realistic Startup Test 72 evaluation, raw cooling water flow was very low. This, in combination with cold river temperatures, prevented evaluation of the heat exchangers at rated maximum raw cooling water temperature and flow. DED will evaluate and resolve this exception.

FINAL SUMMARY REPORT - BFNP UNIT 2

3.0 Results (Continued)

3.4 Phase V - Warranty Tests

3.4.1 STI-20, Electrical Output and Preliminary Heat Rate

Purpose

The purpose of this test is to demonstrate that the requirements of the gross electrical output warranty are satisfied without exceeding reactor power level warranty of 3292 MWt and to demonstrate the net plant heat rate.

Criteria

Level 1

(a) The demonstrated gross electrical output must be greater than or equal to 1098.4 MWe at rated conditions.

(b) The reactor power level must be equal to or less than 3293 MWt at a gross electrical output of 1098.4 MWe.

(c) The net plant heat rate must be equal to or less than 10359 Btu/kWh at a reactor power level equal to or less than 3293 thermal megawatts.

Analysis

Test results are summarized in table STI 20-1, which shows that Level 1 criteria (b) and (c) were satisfied. Criterion (a) was satisfied except for one brief power reduction as follows. This brief transient was of no consequence to the demonstration of gross electrical generation.

Date	Time Interval	Cause
3/6/75	0310 - 2100	Partial loss of feedwater flow caused by feed pump trip. Power was reduced to 70%.

3.0 Results (Continued)

3.4 Phase V - Warranty Tests

3.4.1 STI-20, Electrical Output and Preliminary Heat Rate

Table STI 20-1 Electrical Output and Heat Rate Test	
Test Interval	300 Hrs (1)
Corrected gross electrical output at generator terminals (MWe)	1114.2 (2)
Assumed auxiliary load (contractural-MWe)	24.4
Net generator output (MWe)	1089.8
Core thermal output (MWt)	3277
Net Plant Heat Rate (Btu/kWh)	10262
Net Plant Efficiency (%)	33.26

(1) *During the 300-hour warranty interval, one inconsequential power reduction occurred.*

Corrections were made for an assumed -0.8% KWh meter error. This was the error measured on the identical unit 1 installation caused by voltage loss in conductors as well as the following deviations from rated turbogenerator conditions :

Warranty Condition	Actual	Rated
Condenser BAckpressure (in Hg abs)	1.47	2.0
Power Factor (unitless)	.9996	0.9
Generator Stator Coolant Pressure (psig)	49.64	55