ENCLOSURE 5

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

MAY 9, 1977 - FINAL SUMMARY REPORT, UNIT 3 STARTUP
BROWNS FERRY NUCLEAR PLANT

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May 9, 1977

Mr. Norman C. Moseley, Director
U.S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II
230 Peachtree Street, NW., Suite 1217
Atlanta, Georgia 30303

Dear Mr. Moseley:

FINAL SUMMARY REPORT - UNIT 3 STARTUP - BROWNS FERRY NUCLEAR PLANT - DOCKET NO. 50-296 - OPERATING LICENSE DPR-68

In accordance with Browns Ferry Technical Specifications 6.7.1.a, we are submitting the "Final Summary Report - Unit 3 Startup - Browns Ferry Nuclear Plant."

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 3 STARTUP

BROWNS FERRY NUCLEAR PLANT

Submitted by

H J Green

Plant Superintendent

Approved by

J E Calhoun

Chief, Nuclear Generation Branch

5/5/77

Browns Ferry Nuclear Plant
Decatur, Alabama

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

STARTUP TEST RESULTS

FINAL REPORT

BROWNS FERRY NUCLEAR PLANT UNIT 3

Abstract

The final report of the startup test program performed at Browns Ferry Nuclear Plant Unit 3 is presented in three parts: (1) Introduction, (2) Summary, and (3) Results. Results 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 present a concise summary and pertinent detailed results obtained in the performance of startup tests at Browns Ferry Nuclear Plant Unit 3. The startup test program embraced core physics, thermal-hydraulic, electromechanical and overall system dynamic performance.

1.2 Plant Description

Browns Ferry Nuclear Plant Unit 3 is a single-cycle boiling water reactor designed by General Electric Company (GE) for the Tennessee Valley Authority (TVA) and is the third 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 3, 1976, and continued through completion of the warranty run and 100% power testing. Commercial operation began on March 1, 1977.

1.4 Startup Test Description

Documents such as the Operating License, 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 and TVA personnel. 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, specified 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, as necessary, by TVA engineers. These STI's were reviewed by the Plant Operations Review Committee (PORC) and approved by the TVA Plant Superintendent and GE Site Operations Manager.

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.

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 Program

2.1 Chronology of Test Program

Table 2.1 presents the dates for significant events in the unit 3 startup test program.

2.2 Startup Test Completion Dates

Table 2.2 presents a summary of the dates of completion for all startup tests at each test condition

2.3 Power Flow Map

Figure 2.1 presents a power flow map for Browns Ferry unit 3, showing flow control lines and the nominal positions of test conditions for the startup test program.

Table 2-1

Major Events of Unit 3 Startup Test Program

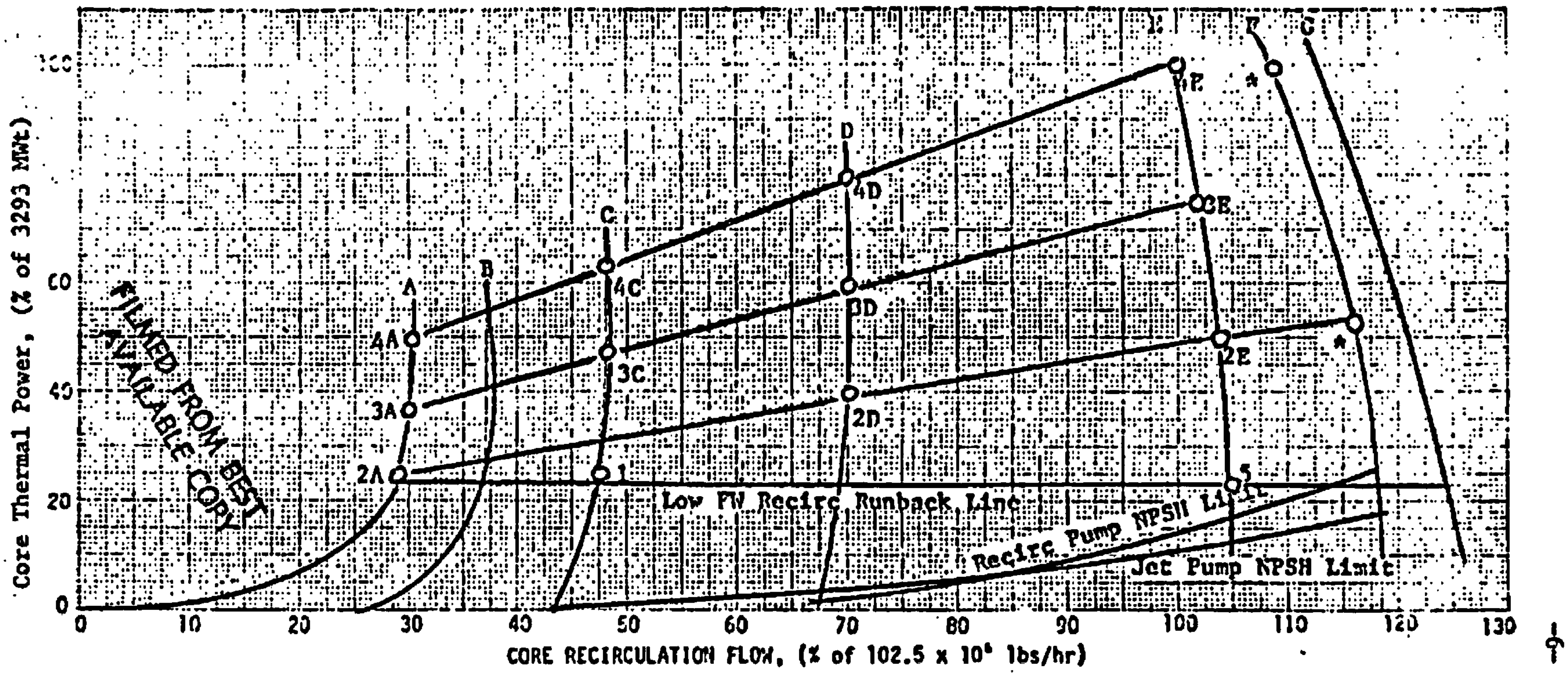
Date	Event
July 3, 1976	First fuel assembly loaded.
July 22, 1976	Core fully loaded to 764 fuel assemblies
August 8,1976	Initial critical during STI-4, Shutdown Margin Demonstration. Also, initial in-sequence critical same day.
August 18, 1976	Full Power license received
August 19, 1976	Begin initial nuclear heatup
August 24, 1976	Reached rated temperature and pressure
September 9, 1976	Initial generator synchronization
September 12, 1976	Completion of Heatup Test Phase
October 6, 1976	Completion of 25% testing
October 29, 1976	Completion of 50% testing
November 12, 1976	Completion of 75% testing
November 20, 1976	100% power first attained
December 24, 1976	Completion of 100% testing
December 26, 1976	Began 300-hour warranty demonstration
January 7, 1977 (1400 hours)	Completion of 300-hour warranty demonstration
March 1, 1977	COMMERCIAL OPERATION

Table 2-2

STI No.	Power, % (1) / Power, % (2) / TEST CONDITIONS	Open Vessel: After Fuel Loading	Heatup: 150-250 psig	Heatup: 600-800 psig	Heatup: Rated T & D <10%	Power: 15-35 / ~47 / 1	50% Flow Cont. Line: 30-50 / ~70 / 2D	50% Flow Cont. Line: 40-60 / ~104 / 2E	50% Flow Cont. Line: ~25 / NC / 2A	75% Flow Cont. Line: 37-57 / ~48 / 3C	75% Flow Cont. Line: 50-70 / ~70 / 3D	75% Flow Cont. Line: 65-85 / ~102 / 3E	75% Flow Cont. Line: ~32 / NC / 3A	100% Flow Cont. Line: 55-75 / ~48 / 4C	100% Flow Cont. Line: 70-90 / ~70 / 4D	100% Flow Cont. Line: 85-102 / ~100 / 4E	100% Flow Cont. Line: ~50 / NC / 4A
1	Chemical and Radiochemical	8/12	8/20		8/24	~~9/15~~ 10/4		10/18				11/5				11/20-22	
2	Radiation Measurements	~~7/17~~ 7/23			8/26	9/17		10/18				11/3				11/22-23	
3	Fuel Loading	7/22															
4	Full Core Shutdown Margin	8/8															
5	CRD	8/8		8/24	8/29	9/13,17						11/9				12/3-16	
6	SRM Perf. and Control Rod Sequence	8/8	8/17			9/17,19											
9	Water Level Measurement				8/29	9/16										11/21	
10	IRM Performance	8/8	8/20			9/20											
11	LPRM Calibration					9/17,18		10/13-15				11/5-7				12/12	
12	APRM Calibration		8/19		8/27	9/15,18		10/14,15				11/5-7				12/14	
13	Process Computer	7/23			8/25	9/18		10/12-17								11/21,12/2	
14	RCIC		8/21	8/23	8/26	9/22											
15	HPCI		8/21	8/24	8/29			10/17,25									
16	Selected Process Temperature				8/28			10/8	10/28								11/26
17	System Expansion	8/13	8/20		8/26	9/27											
18	Core Power Distribution					9/19		10/15,16				11/5-7				12/1	
19	Core Performance					9/17,19	10/7	10/15	10/28	10/30	11/2	11/7		11/27	11/28	12/2	11/26
20	Steam Production															12/26 - 1/8/77	
21	Flux Response to Rods					9/20		10/8				11/3				11/21	11/26
22	Press. Regulator Setpoint Changes					10/5		10/17				11/3,6		11/27	11/28	11/22	
	Backup Regulator					10/5		10/17				11/3,4				11/22	11/26
23	FW System: FW Pump Trip															11/25	
	Water Level STPT Change					9/22		10/21,22				11/3		11/27	11/28	11/22	
24	Bypass Valves					9/24		10/11	10/28			11/3		11/27	11/28	11/23	11/26
25	Main Steam Isolation Valves				8/27			10/17							11/28	12/3	
26	Relief Valve		8/21		8/27	9/22,23						11/9					
27	Turbine Trip and											11/9					
	Generator Load Rejection					9/23										12/16	
29	Flow Control													——See	STI 32——		
30	Recirculation System							10/28				11/4				11/25	11/26
31	Loss of T-G and Offsite Power					9/27											
32	Recirc MG Set Speed Control					9/22	10/7	10/24		10/30	11/2	11/3		11/28	11/28	11/24	
33	Turbine Stop Valve Surveillance Test					9/19		10/8				11/3				11/23	
34	Vibration Measurement	8/16-17				9/27	10/7	10/25	10/28	10/30	11/1	11/2		11/25	11/25	11/23	11/26
35	Recirculation System Flow Calibration	7/25-8/16						10/8				11/3				11/3 12/1	
70	Reactor Water Cleanup System				8/26												
71	Residual Heat Removal System					12/16-18											
72	Drywell Atmospheric Cooling System		8/22	8/24	8/24											12/2	
73	Cooling Water System				8/24											12/2	
74	Off-Gas System					9/26,10/5		10/8-11								11/21,22	
75	Rx Scram Outside C. R.					10/1											

(1) Percent of rated power, 3293 MWt.

(2) Percent of rated flow, 102.5×10^6 lbs/hr.



Test Condition No.	1	2A	2D	2E	*	3A	3C	3D	3E	4A	4C	4D	4E	*	5
Rod Pattern	V	a*	a	a	*	b*	b	b	b	c*	c	c	c	*	V
% Pump Speed	~41	0*	~68	E	*	0*	~41	~68	E	0*	~41	~68	E	*	E*
% Power	15-35*	~25	30-50*	40-60*	*	~37	37-57*	50-70*	65-86*	~50	55-75*	70-90*	95-100*	*	≥20*
% Core Flow	~47	NC	~70*	~104*	*	NC	~48*	~70*	~102*	NC	~48*	~70*	~100*	*	~105

a Rod Pattern Obtained at Test Condition No. 2E
b Rod Pattern Obtained at Test Condition No. 3E
c Rod Pattern Obtained at Test Condition No. 4E
* Asterisked values are set as initial test conditions; non-asterisked values are estimates
NC Natural Circulation
V Varies

A Natural Circulation
B 20% Pump Speed
C Analytical low 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

FIGURE 2.1 APPROXIMATE POWER FLOW MAP SHOWING STARTUP TEST CONDITIONS

3.0 Results

3.1 STI-1, Chemical and Radiochemical

3.1.1 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.

3.1.2 Criteria

Level 1

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

The activity of gaseous and liquid effluents must conform to the license limitations.

Level 2

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

3.1.3 Analysis

STI-1 testing was conducted at open vessel, heatup, test conditions 1, 2E, 3E, and 4E, as defined on the power flow map in section 2.3.

3.0 Results (Continued)

3.1 STI-1, Chemical and Radiochemical (Continued)

3.1.3 Analysis (Continued)

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

Conductivity (μmho/cm @ 25°)	0.32
Chloride (ppb)	< 50
Turbidity (FTU)	0.15
Boron (ppb)	< 50
Silica (ppb)	10

All level 2 criteria were satisfied with the exception of chloride concentrations in the condensate storage and demineralizer water storage tanks. Plant analytical procedures have a minimum chloride sensitivity of 50 ppb. GE limit for chlorides in the storage tanks is 10 ppb. GE field disposition request FDDR ER3-446, dated 8/26/76, permits the acceptance of <50 ppb chloride concentration. Reported data for chloride concentration comply with this limit. No further action is required.

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

Conductivity (μmho/cm @ 25°)	0.32
Turbidity (FTU)	0.46
Chloride (ppb)	< 50
Boron (ppb)	90
Silica (ppb)	540

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.

Testing of steam separator and dryer performance at Browns Ferry 3 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 μmho/cm @ 25° for 33 hours from September 15 to September 19, 1976, @ 25% testing plateau due to placing feedwater heaters in service.

3.0 Results (Continued)

3.1 STI-1, Chemical and Radiochemical (Continued)

3.1.3 Analysis (Continued)

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

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.

All criteria were satisfied with the exception of condensate oxygen concentration at all power testing levels. GE fuel warranty document (22A4367), Browns Ferry 3, sheet 9, changes the limit from 14 ppb to $\leq$ 2000 ppb. All oxygen values met this limit; therefore, disposition of this exception is complete. No further action is required.

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

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

3.1 STI-1, Chemical and Radiochemical (Continued)

3.1.3 Analysis (Continued)

Table STI 1-1

Sample Source and Test	Date	15-35% Power 9/15/76	40-60% Power 10/11/76	65-85% Power 11/3/76	95-100% Power 11/21/76
	MWt	780	1970	2531	3291
Reactor Water	MWe / Limit	193	612	847	1096
Conductivity, μmho/cm	1.0	0.80	0.59	0.55	0.38
Chloride, ppm	0.2	<0.05	<0.05	<0.05	<0.05
Turbidity or insolubles, JTU	10 ppm	0.55	<0.075	0.13	<0.10
Iodine-131, μCi/ml		6.55 E-07	<1.47 E-06	1.24 E-05	2.15 E-05
Iodine-133, μCi/ml		6.52 D-06	3.52 E-05	7.37 E-05	9.87 E-05
Gross Activity					
-filtrate, cpm/ml, 2 hrs.		2716	9852	29834	24084
-crud, cpm/ml, 2 hrs.		3416	6124	3086	2374
Gross Activity					
-filtrate, cpm/ml, 7d		57	112	217	529
-crud, cpm/ml, 7d		5	161	42.9	80
Silica, ppb	5.0 ppm	0.314	0.341	0.28	0.38
Boron, ppb	50 ppm	<0.05	<0.05	<0.05	<0.05

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

3.1 STI-1, Chemical and Radiochemical (Continued)

3.1.3 Analysis (Continued)

Table STI 1-1

Sample Source and Test		15-35% Power	40-60% Power	65-85% Power	95-100% Power
	Date	9/15/76	10/16/76	11/3/76	11/22/76
	MWt	780	1770	2531	3256
Reactor Water (Continued)	MWe / Limit	193	542	847	1070
Chemical Analysis on filtrate, ppb					
-iron		XX	XX	XX	0.167
-copper		XX	XX	XX	19.74
-nickel		XX	XX	XX	< 0.001
-chromium		XX	XX	XX	3.79
Chemical Analysis on Crud, ppb					
-iron		8.95	7.1	12	4.60
-copper		XX	XX	XX	< 0.001
-nickel		XX	XX	XX	0.775
-chromium		XX	XX	XX	< 0.001
Spectral Analysis on major nuclides at 24 hours					
Filtrate		Mo-99 Tc-99m Na-24 As-76	Cr-51 Cu-64 Na-24	Mo-99 Tc-99m Cr-51 Zn-69m W-187 Co-58 As-76 Na-24 Zn-65	Mo-99 Tc-99m Cr-51 W-187 Co-58 Zn-65 Cu-64 As-76 Nb-95 Na-24
NOTE: XX symbol signifies data not required by the test instruction.					

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

3.1 STI-1, Chemical and Radiochemical (Continued)

3.1.3 Analysis (Continued)

Table STI 1-1 (Continued)

Sample Source and Test		15-35% Power	40-60% Power	65-85% Power	95-100% Power
	Date	9/15/76	10/16/76	11/3/76	11/22/76
	MWt	780	1 70	2531	3256
	MWe / Limit	193	542	847	1070
Crud		W-187 Cr-51 Zn-69m Cu-64 Na-24 Zn-65 As-76 Cs-137 Mn-54 Mn-56 Fe-59 Ba-140 La-140 Co-58	Cr-51 Co-58 Mn-54 Fe-59 Co-60 Cs-134 Na-24 Zr-95 Zn-75 Ce-141	W-187 Mo-99 Mo-99m Sb-125 Fe-59 Cr-51 Zn-69m I-135 As-76 Sb-124 Zr-95 Zr-97 Mb-95 Co-58 Mn-54 Mn-56 Zn-65 Co-60 Cu-64 N-924	W-187 Mo-99 Tc-99m Fe-59 Cr-51 Zn-69m Co-58 Zn-65 Cu-64 As-76 Sb-124 Mn-54 Co-60
Condensate Demin. Influent					
Conductivity, µmho/cm		0.34	0.13	0.094	0.076
Chloride, ppm		<0.05	<0.05	<0.05	<0.05
Insoluble iron, ppb		25	<10	<25	10
Condensate Demin. Effluent					
Conductivity, µmho/cm	0.1	0.25(1)	0.072	0.083	0.057

(1) Heater drain problems

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

3.1 STI-1, Chemical and Radiochemical (Continued)

3.1.3 Analysis (Continued)

Table STI 1-1 (Continued)

Sample Source and Test		15-35% Power	40-60% Power	65-85% Power	95-100% Power
	Date	9/15/76	10/12/76	11/3/76	11/22/76
	MWt	780	1770	2531	3256
	MWe / Limit	193	542	847	1070
Condensate Demin. Effluent (Cont'd.)					
Insoluble iron, ppb	20	<10	<10	<10	<10
Oxygen, ppb	14(2)	Lab 150 Anal.	Lab 100 Anal.	Lab 80 Anal.	Lab 100 Anal.
Feedwater					
Conductivity, μmho/cm	0.10	0.46(1)	0.093	0.085	0.072
Iron - insoluble, ppb		10	<10	10	17.64
-soluble, ppb		XX	4.13	16	4.15
Nickel - insoluble, ppb		XX	XX	XX	0.463
-soluble, ppb		XX	XX	XX	0.588
Copper - insoluble, ppb		XX	XX	XX	0.663
- soluble, ppb		XX	XX	XX	<0.001
Chromium - soluble, ppb		XX	XX	XX Crud XX Sol	XX
Off-Gas					
Activity @ SJAE, μCi/sec. (Σ6 gases)		< 0.11	< 61.6	< 98	79.9
N-13 @ SJAE, μCi/sec.		1190	1450	1685	1684
Flow rate, cfm (FR-66-111)		160.6	38	35	38

XX Symbol signifies data not required by the test instruction.

(1) Heaters placed into service.

(2) Limits changed to ≤ 200 ppb in GE fuel warranty document (22A4367), table I, sheet 9.

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

3.1 STI-1, Chemical and Radiochemical (Continued)

3.1.3 Analysis (Continued)

Table STI 1-1 (Continued)

Sample Source and Test			15-35% Power	40-60% Power	65-85% Power	95-100% Power
		Date	9/15/76	10/11/76	11/3/76	11/21/76
		MWt	780	1970	2531	3291
		File / Limit	193	612	847	1096
Off-Gas (Continued)						
Composition - air, cfm			140	38	35	38
Radiolytic - $(H_2 + O_2)$			0	0	0	0
Delay time, min.			XX	XX	XX	186.6
Activity release at stack µCi/sec.			72.5 (1)	128 (1)	155 (1)	128.7 (1)
Activity Pattern			Recoil.	Recoil.	Recoil.	Recoil.
Off-Gas Monitor Reading, mr/hr	A		7	10	18	16
	XX		XX	XX	XX	XX
Stack gas monitor Reading, cps	A		10	12	18	12
	B		10	16	18	16

XX Symbol signifies data not required by the test instruction.
(1) Combined activity from units 1, 2, and 3.

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

3.1 STI-1, Chemical and Radiochemical (Continued)

3.1.3 Analysis (Continued)

Fuel Cladding Integrity

Table STI 1-2 shows representative iodine data data obtained during the startup.

Table STI 1-2

Date	Time	MWt	Estimated I-133(1) Carryover (%)	μCi/ml I-131	μCi/ml I-132	μCi/ml I-133	μCi/ml I-134	μCi/ml I-135
10/11/76	0700	1970	---	<1.47 E-06	2.14 E-05	3.52 E-05	5.36 E-03	6.12 E-05
10/14/76	2000	1693	0.3(2)	---	---	---	---	---
10/25/76	0800	884	---	6.31 E-07	9.5 E-07	3.34 E-06	4.31 E-06	6.00 E-06
11/15/76	0700	2882	---	4.96 D-06	7.0 E-05	6.43 E-05	4.02 E-04	1.11 E-04
11/21/76	1800	3275	0.22(3)	---	---	---	---	---
11/29/76	0800	2075	---	8.75 E-06	1.00 E-04	9.81 E-05	2.95 E-04	1.81 E-04
12/3/76	0800	3178	---	5.48 E-06	1.14 E-04	6.32 E-05	2.30 E-04	1.32 E-04

(1)I-131 activity concentration insufficient.
(2)50% power - no cleanup test
(3)100% power - no cleanup test

3.0 Results (Continued)

3.1 STI-1, Chemical and Radiochemical (Continued)

3.1.3 Analysis (Continued)

Condensate

The condensate pump discharge and condensate demineralizer effluent conductivities were only slightly high during the initial heatup through the 15-35% test conditions, however, they were within established limits throughout the remainder of startup testing. The following table, STI 1-3, shows the plant conductivity history during the startup testing.

Table STI 1-3

Browns Ferry 3 Startup Conductivities (umho/cm)

Date	Power (Thermal)	Condensate Pump Discharge	Condensate Demineralizer Combined Effluent	Reactor Water
8/7/76	0%, No Heat	0.50	0.20	0.32
8/24/76	1%, Heatup	0.15	0.10	0.3 - 0.7(3)
9/15/76	15-35%	0.34	0.185	0.30-2.20(3)
10/15/76	50%	0.11	0.07	0.50-2.40(2)
10/29/76	40-60%	0.088	0.078	0.59
11/3/76	70%	0.094	0.083	0.55
11/21/76	99% (approx.)	0.076	0.057	0.3 - 1.6(2)

(2) No cleanup test
(3) Range of Reactor H_2O conductivity during test period.

3.0 Results (Continued)

3.1 STI-1, Chemical and Radiochemical (Continued)

3.1.3 Analysis (Continued)

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.

Radwaste

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

Condensate and Cleanup Demineralizers

The condensate demineralizers were initially placed into service in late 1975 and were subsequently used to clean water during construction and preoperational testing.

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

3.0 Results

3.2 STI-2, Radiation Measurements

3.2.1 Purpose

The purposes of this test are to:

1. Determine the background radiation levels in the plant environs prior to operation for base data on activity buildup.

2. Monitor radiation at selected power levels to assure the protection of personnel during plant operation.

3.2.2 Criteria

Level 1

The radiation doses of plant origin and the occupancy times of personnel in radiation zones shall be controlled consistent with the guidelines of the standards for protection against radiation as outlined in TVA Radiological Control Instruction.

Level 2

There are no level 2 criteria.

3.2.3 Analysis

STI-2 was performed at the following unit No. 3 conditions.

Table STI 2-1
Survey Conditions

I.	Prefuel Loading	May 12, 1976
II.	Core loaded, Open vessel	July 23, 1976
III.	Plant at 6% power	August 26, 1976
IV.	Plant at 25% power	September 17, 1976 (limited survey)
V.	Plant at 58% power	October 8, 1976
VI.	Plant at 76% power	November 3, 1976 (limited survey)
VII.	Plant at 100% power	November 22, 1976
VIII.	Plant at 100% power-warranty run	December 28, 1976 (limited survey)

3.0 Results (Continued)

3.2 STI-2, Radiation Measurements (Continued)

3.2.3 Analysis (Continued)

At each point gamma and neutron measurements were made as required by the type of survey. "Limited" surveys involved a selected part of the complete surveys, with only those points of normal occupancy being measured. Exceptions to each survey were as follows:

Table STI 2-2
Exceptions to Surveys

Plant Condition (See Table 1)	Test Point	Exception
I	RB-3-38	Neutron survey not made. Inaccessible due to shield plugs not in place.
	RB-3-44	Neutron survey not made due to inaccessibility. (15' above floor)
II	RB-3-38	Same as above
III	NO EXCEPTIONS	
IV	NO EXCEPTIONS	
V	NO EXCEPTIONS	
VI	NO EXCEPTIONS	
VII	RB-3-44	Test point RB-3-44 required rezoning as per RCI-1.
VIII	NO EXCEPTIONS	

As noted in table STI 2-2, only test point RB-3-44 required rezoning to meet criteria level 1. This test point is a blank drywell penetration located in the SE quadrant at the 593' elevation in unit 3 reactor building. It is located in a normally inaccessible location 15 feet above the floor. As a result of the survey, a cage was placed around the area and proper zone posting made. This brought the zone into compliance with RCI-1, thus fulfilling STI-2 requirements.

3.0 Results (Continued)

3.3 STI-3, Fuel Loading

3.3.1 Purpose

The purpose of STI-3 is to load fuel safely and efficiently to the full core size.

3.3.2 Criteria

Level 1

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

Level 2

Not applicable.

3.3.3 Analysis

Fuel loading began with the loading of the first fuel assembly at 1646 hours on July 3, 1976, and was succesfully completed at 0136 hours on July 22, 1976. At that time all 764 fuel assemblies were installed, the seven operational sources were in place, and the four source range monitors (SRM's) were electronically connected and functional. Partial core shutdown margins were verified at designated points during the loading process and met all criteria.

Prior to loading the first fuel assembly, the four fuel loading chambers (FLC) were installed in dummy blade guides at approximately 2/3 core height and were connected to the plant SRM electronics. The signal-to-noise ratio was verified to be >2:1 and the FLC count rate was >3.0 cps. The rod block and scram setpoints were set at $1x10^5$ cps and $5x10^5$ cps, respectively. The shorting links were removed from the circuitry, placing the FLC/SRM and IRM's electronics in the non-coincidence scram mode.

The Sb-Be operational sources were installed prior to fuel loading and used throughout fuel loading to establish neutron flux. The source strength was 686 curies on the initial load date and 552 curies at completion of fuel loading.

After completion of the loading of each control cell (2 x 2 fuel assembly array) functional and subcriticality checks were made by withdrawing the associated control rod. In addition, partial core subcriticality checks were made after the loading of 16, 64, and 144 fuel assemblies to verify that

3.0 Results (Continued)

3.3 (Continued)

3.3.3 Analysis (Continued)

the partially loaded core is subcritical by at least 0.38% $\Delta K/K$ with the analytically strongest rod fully withdrawn. As an added assurance that fuel was being loaded safely, inverse multiplication (1/M) plots were maintained of the FLC/SRM count rates. In certain cases special interpretation of these plots was required of the nuclear engineer because of geometric effects. These geometric effects were caused by loading a fuel assembly near an operational source or FLC and were expected. The FLC's were moved as necessary to maintain the count rate >3 cps and $<1x10^5$ cps. (See figure STI 3-1.) The FLC's were removed after 360 fuel assemblies were loaded and all four SRM's were then operational.

The fully loaded core was verified for fuel assembly orientation, serial number, and proper location of fuel types by lowering the water level in the reactor vessel to allow visual verification. A video-tape was also made for a permanent record. Fuel assembly locations are shown in figure STI-3-2.

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

3.3 STI-3, Fuel Loading

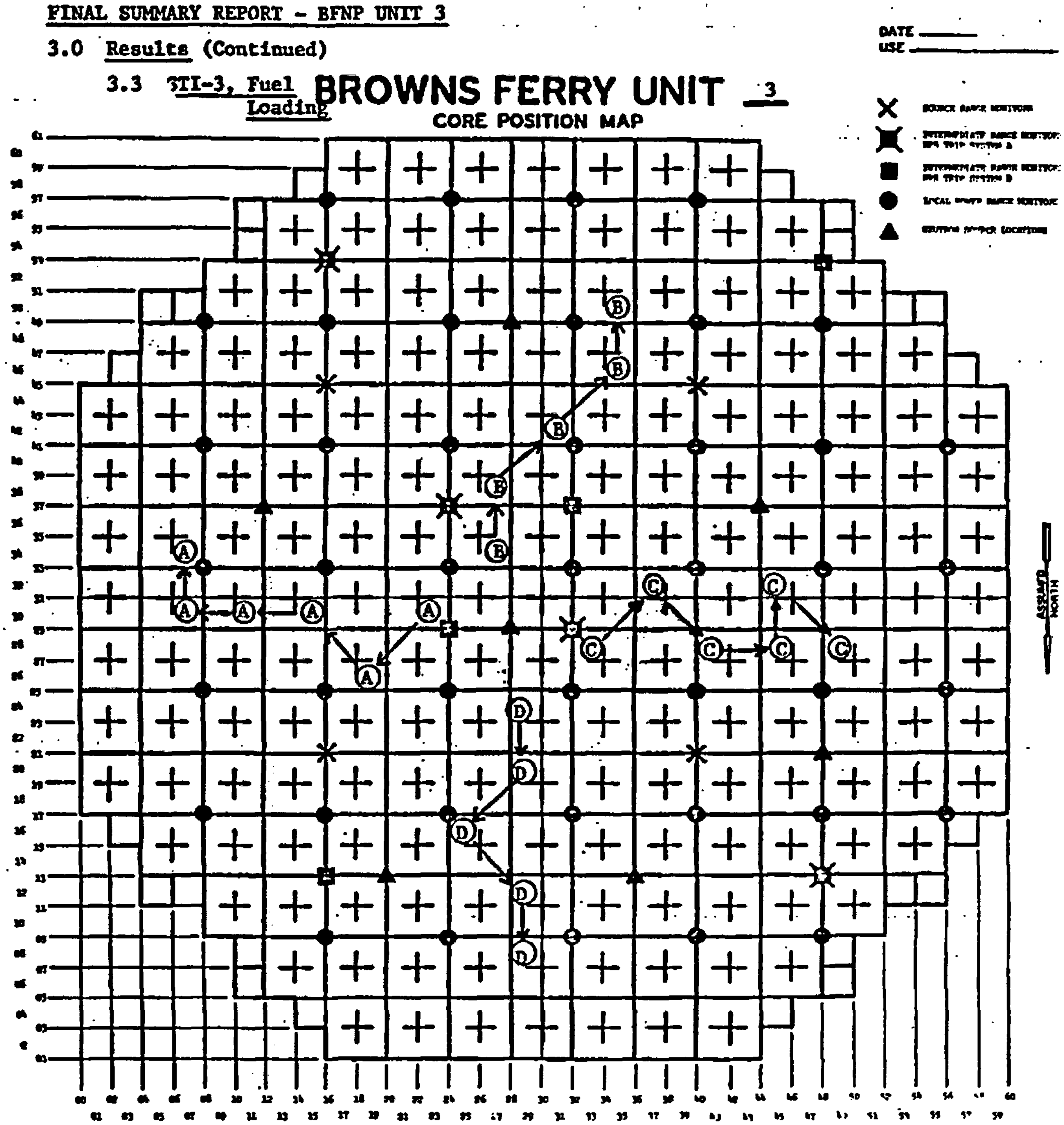


Figure STI 3-1 - FLC Movements

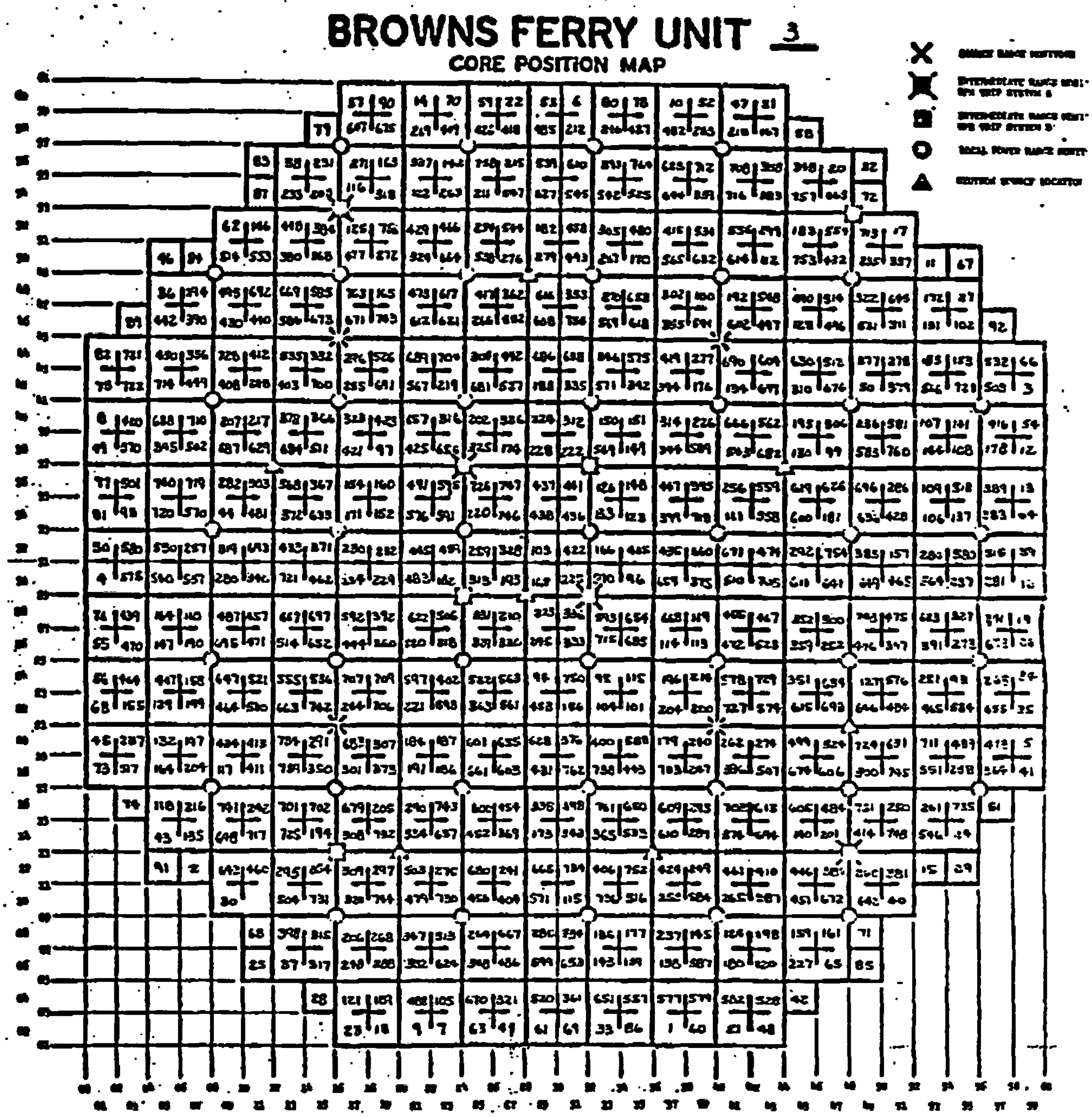


Figure 3-2
Final Core Configuration

3.0 Results

3.4 STI-4, Full Core Shutdown Margin

3.4.1 Purpose

The purpose of STI-4 is to demonstrate that the reactor will be subcritical throughout the first fuel cycle with any single control rod fully withdrawn.

3.4.2 Criteria

Level 1

The shutdown margin of the fully loaded core with the analytically strongest rod withdrawn must be at least R + 0.38% ΔK/K. (Refer to analysis section for R.)

Level 2

Criticality should occur, within ± 1.0% ΔK/K, for the configuration described in table 4.8-1 and figure 4.8-3 of STI-4 (See attachments A and B).

3.4.3 Analysis

Control rods were withdrawn in the order specified in STI-6 for "B" sequence until criticality was achieved when the 28th control rod (46-23) was pulled to notch 28 on August 8, 1976, during open vessel testing. The reactor period was estimated to be approximately 238 seconds. Sufficient SRM/IRM overlap data was obtained and the reactor was taken subcritical by inserting the 27th and 28th control rods in order to remove the shorting links. The reactor was brought critical for a second time by withdrawing the 27th and 28th control rods in order to obtain accurate period measurements for the Keff calculation. The reactor was critical on a 132-second period on the 18th notch of the 29th control rod (38-15) with a moderator temperature of 92° F. (See figure STI 4-1.)

A temperature correction was made using the 7.5×10^{-5} ΔK/K° F temperature coefficient and a period correction using table 4.8-2 of STI-4. This results in a corrected Keff of 1.0023514. Subtracting the ΔK for the rods pulled gives a Keff for all-rods-in of .95484. Subtracting the sum of the Keff for all-rods-in and the worth (from 1.000) of the strongest rod fully withdrawn, yields an actual shutdown margin of 2.586% ΔK/K.

3.0 Results (Continued)

3.4 STI-4, Full Core Shutdown Margin

3.4.3 Analysis (Continued)

The fully loaded core is required to be shutdown with the strongest rod withdrawn by at least R + 0.38% $\Delta K/K$. From figure 4.8-1 of STI-4, R=b-a = .0115 $\Delta K/K$ = 1.15% $\Delta K/K$. Therefore, the required shutdown margin is 1.53% $\Delta K/K$. Level 1 criteria have been met.

The reactor was critical with an actual Keff of 1.0023514. The calculated Keff of the core with 28 rods and 18 notches on the 29th rod withdrawn was .9992. The difference between these two values, .315% $\Delta K/K$, satisfies level II criteria that criticality occurs within $\pm$ 1.0% $\Delta K/K$ of the actual and theoretical Keff values.

3.0 Results

3.4 STI-4, Full Core Shutdown Margin (Continued)

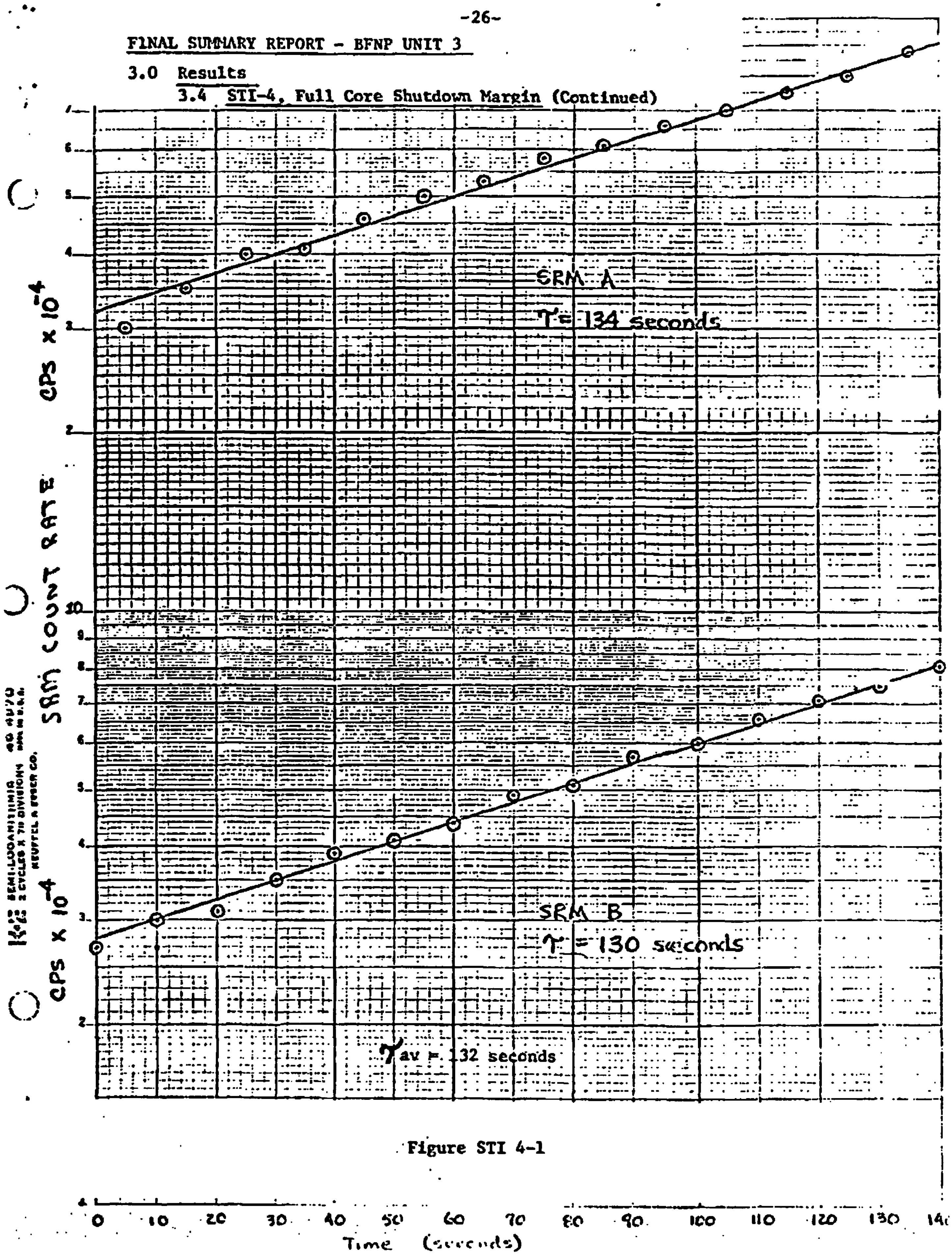
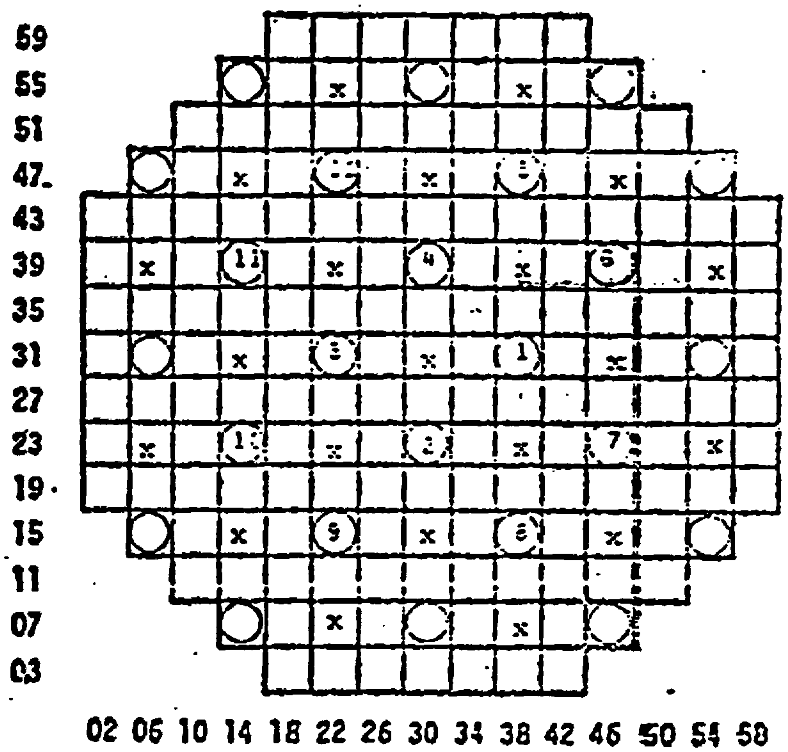


Figure STI 4-1

ATTACHMENT A

TITLE: CONTROL ROD WORTHS

TABLE: 4.8-1



[x] RWM group 1 out

[O] RWM group 2 out

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Control Rod Configuration Sequence B	K_{eff}
All rods inserted	0.9517
RWM group 1 and rods 1-6 of group 2 withdrawn	0.9981
RWM group 1 and rods 1-9 of group 2 withdrawn	0.9996
RWM group 1 and rods 1-12 of group 2 withdrawn	1.0007
RWM groups 1 and 2 withdrawn	1.0019

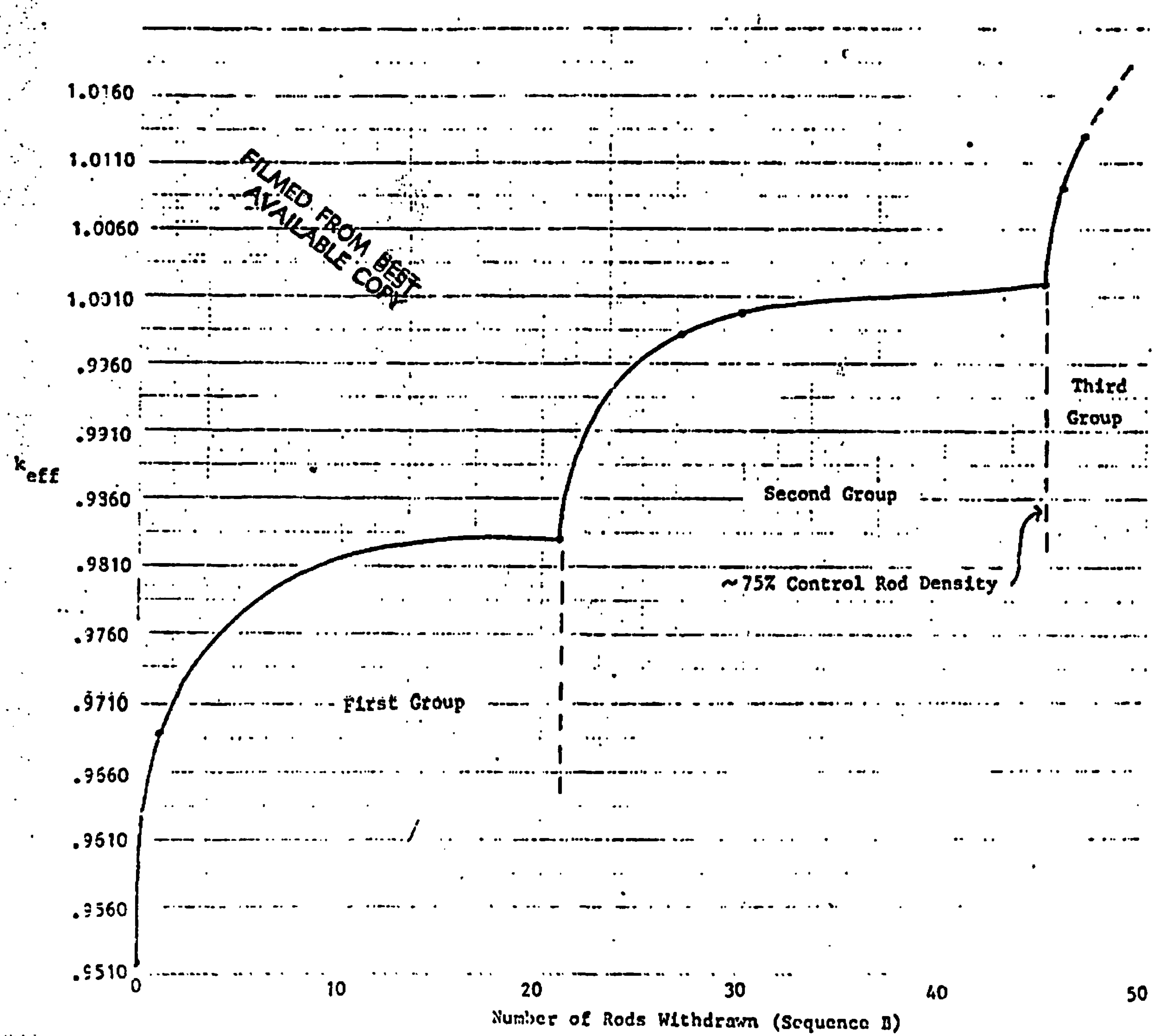
* Strongest rod is 30-31

*Revision

ATTACHMENT B

Figure 4.8-3

Full Core Shutdown Margin Calculations



3.0 Results

3.5 STI-5, Control Rod Drive System

3.5.1 Purpose

The purposes of the control rod drive system test are:

1. 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.

2. To determine the initial operating characteristics of the entire CRD system.

3.5.2 Criteria

Level 1

Each CRD must have a normal withdraw speed less than or equal to 3.6 inches per second (9.14 cm/sec), indicated by a full 12-foot stroke in greater than or equal to 40 seconds.

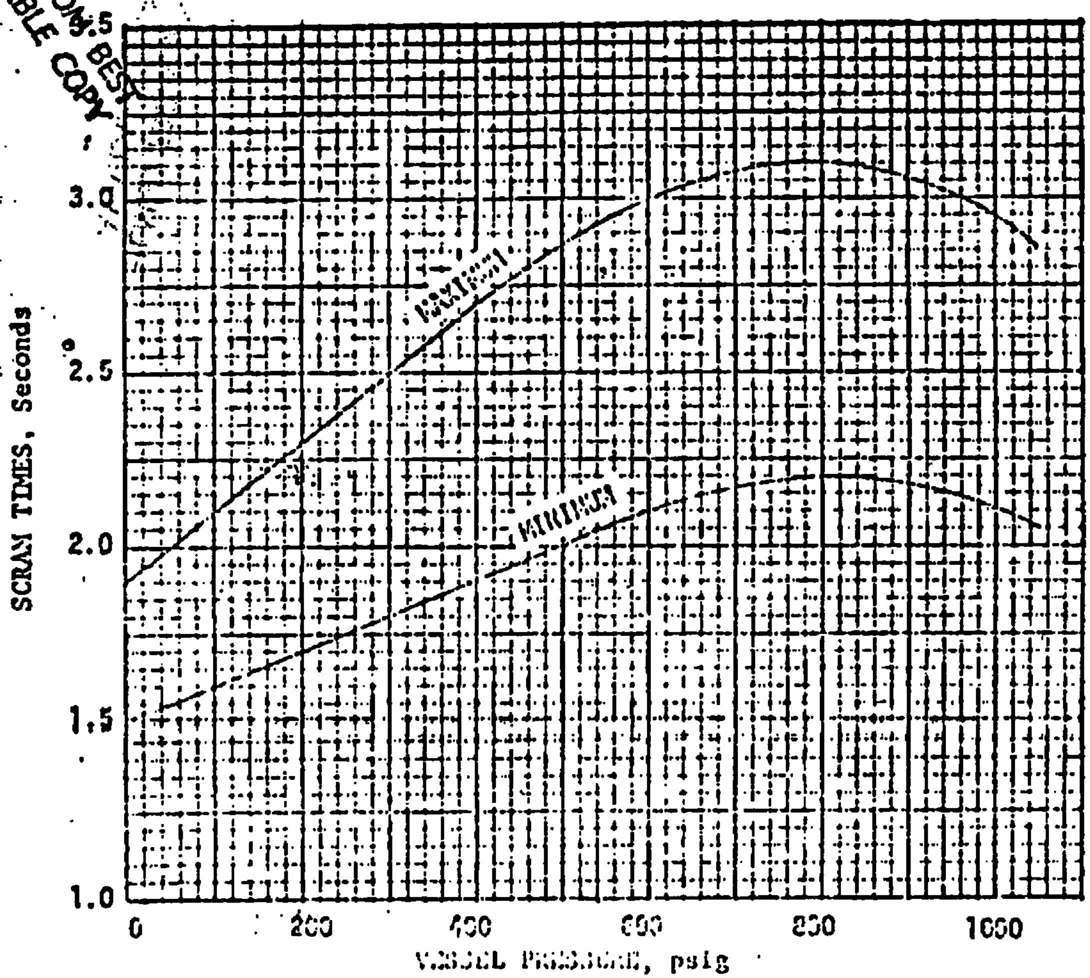
The control rod scram insertion times must be within the limiting conditions for operation specified in technical specification 3.3.C.

Level 2

Each CRD must have a normal insert or withdraw of 3.0 $\pm$ 0.6 inches per second (7.62 $\pm$ 1.52 cm/sec), indicated by a full 12-foot stroke in 40 to 60 seconds.

With respect to the control rod drive friction tests, if the differential pressure variation exceeds 15 psid (1 kg/cm^2) 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 (2.1 kg/cm^2) nor should it vary by more than 10 psid (0.7 kg/cm^2) over a full stroke.

Scram times with normal accumulator charge should fall within the time limits indicated in figure STI 5-1.



SCRAM PERFORMANCE CURVE FOR MODEL
7RDB144A-2 and 7RDB144B-1 CRD's

Figure STI 5-1

SYSTEM OPERATING CONDITIONS:

1. Accumulator precharge 565/585 psig at 70° F. (39,9/41.2 kg/cm^2 at 20° C.)
2. Accumulator water side 1510 psig, (106.3 kg/cm^2) max. 1390 psig, (97.7 kg/cm^2) min.
3. Scram valve air pressure 70/75 psig. (4.9/5.30kg/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").

3.0 Results (Continued)

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

3.5.3 Analysis

STI-5 testing was conducted at open vessel, heat up, and test conditions 1, 3E, and 4E, as defined on the power flow map in section 2.3.

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 on each CRD.

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.

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.

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

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

3.5.3 Analysis (Continued)

Scram Testing

During open vessel testing all control rods were scram tested. The average scram times fell well within technical specifications and criteria requirements. (See table STI 5-4.) Initially all rods met the level 2 criteria for individual scram times, except control rod drives 18-07 and 26-15, which had 90% scram insertion times of 1.914 and 1.913, respectively. The scram tests were repeated for CRD's 18-07 and 26-15 with normal accumulator pressure. The 90% scram insertion times were measured to be 1.712 and 1.752, respectively; thus satisfying the level 2 criteria.

From this data the four slowest control rod drives were chosen to be scrammed three times each with minimum accumulator pressure. All level 1 and 2 criteria were met for testing during the open vessel phase. Table STI 5-1 summarizes testing of the four slowest drives.

Table STI 5-1
Four Slowest Control Rod Drives
At Zero Reactor Pressure And
Minimum** And Normal Accumulator Pressure

Rod Location	Mean* 90% Scram Time (sec) Min. Accu. Press.	90% Scram Time (sec) Norm. Accu. Press.
30-27	1.908	1.825
18-07	2.047	1.712
26-15	1.974	1.752
14-19	1.854	1.825

*Mean of three scrams
**970 psig

3.0 Results (Continued)

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

3.5.3 Analysis (Continued)

Scram Testing (Continued)

During the initial heatup, the four slowest in-sequence CRD's were selected for scram testing at 600 and 800 psig. Results are summarized in table STI 5-2. Level 2 criteria were not met by CRD's 34-43 and 30-07 at 600 psig and by CRD 22-55 at 800 psig. All technical specifications and level 1 criteria were met for all testing at 600 psig and 800 psig.

At rated reactor pressure scram times were measured for all in-sequence CRD's with normal accumulator pressure. The selected four in-sequence CRD's were scrammed three times each with zero accumulator pressure. The results for rated pressure scram testing are summarized in table STI 5-2. The four selected CRD's were friction tested and timed at rated pressure. All level 1 and 2 criteria were met for testing at rated pressure.

Table STI 5-2
Four Slowest In-Sequence Rod Scram Tests

Drive Location	Test Number	90% Insertion Scram Time		
		Rx Press. 600 psig	Rx Press. 800 psig	Rx Press. 1000 psig
22-55	1	2.87	2.77	2.34
	2	2.94	3.14	2.72
	3	2.84	3.06	2.80
	Mean	2.88	2.99	2.62
26-27	1	2.77	2.95	2.85
	2	2.81	2.84	2.76
	3	2.97	2.82	2.61
	Mean	2.85	2.87	2.76
30-07	1	2.89	2.86	2.72
	2	3.02	2.83	2.79
	3	2.81	2.82	2.77
	Mean	2.91	2.84	2.76
	4	2.90		
34-43	1	2.95	2.93	2.79
	2	2.94	2.90	2.66
	3	3.19	2.83	2.64
	Mean	3.03	2.88	2.70
	4	2.99		
	5	3.01		
	6	3.03		

3.0 Results (Continued)

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

3.5.3 Analysis (Continued)

Scram Testing (Continued)

A reactor scram from hot-standby permitted a subsequent startup in "A" control rod sequence. This permitted "A" in-sequence CRD's to be scram timed at hot-standby instead of after the rod sequence control system interlocks were cleared during startup to test condition 1 as had been projected by the Master Startup Test Instruction (MSTI). The average scram times for all 185 CRD's at rated reactor pressure are summarized in table STI 5-4. Individual rod scram times are listed in table STI 5-3.

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

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

3.5.3 Analysis (Continued)

Scram Testing (Continued)

Table STI 5-3
Individual Rod Scram Times
Sequence A Rods

Drive Location	Reactor Press. Psig	Scram Insertion Times, Sec. 5%	20%	50%	90%
30-03	956	0.324	0.687	1.428	2.475
22-03	956	0.332	0.709	1.468	2.571
14-11	956	0.313	0.671	1.380	2.403
06-19	956	0.302	0.671	1.424	2.474
02-31	956	0.292	0.639	1.352	2.354
02-39	956	0.294	0.645	1.356	2.339
30-59	956	0.308	0.671	1.412	2.427
18-55	956	0.334	0.703	1.468	2.539
26-55	956	0.326	0.703	1.444	2.508
26-07	956	0.294	0.661	1.416	2.507
18-07	956	0.308	0.685	1.456	2.523
30-11	956	0.316	0.716	1.464	2.578
22-11	956	0.313	0.671	1.420	2.459
02-23	956	0.294	0.653	1.364	2.402
06-27	956	0.318	0.695	1.464	2.572
10-31	956	0.316	0.604	1.492	2.787
06-35	956	0.300	0.671	1.388	2.419
06-43	958	0.318	0.682	1.416	2.491
22-59	958	0.302	0.645	1.328	2.290
14-51	958	0.313	0.687	1.472	2.555
22-51	958	0.292	0.637	1.308	2.274
30-51	958	0.310	0.661	1.388	2.370
30-19	958	0.324	0.714	1.492	2.547
26-15	958	0.310	0.645	1.372	2.419
22-19	958	0.310	0.679	1.444	2.499
18-15	958	0.324	0.695	1.440	2.499
14-19	958	0.332	0.730	1.532	2.644
10-15	958	0.310	0.671	1.400	2.426
30-35	958	0.308	0.687	1.488	2.555
22-35	958	0.294	0.674	1.472	2.546
14-35	958	0.322	0.703	1.468	2.555
10-39	960	0.326	0.685	1.440	2.522
10-47	955	0.308	0.692	1.512	2.586
18-47	960	0.386	0.767	1.652	2.786
26-47	960	0.318	0.687	1.456	2.474

Location	Reactor Press. Psig	Scram Insertion [illegible] 5%	20%	50%	
26-23	960	0.332	0.722	1.504	2.555
30-27	960	0.332	0.714	1.520	2.611
18-23	960	0.318	0.685	1.440	2.564
22-27	955	0.318	0.690	1.464	2.563
10-23	955	0.324	0.730	1.512	2.554
14-27	955	0.362	0.762	1.496	2.555
18-31	955	0.326	0.703	1.444	2.523
26-31	955	0.297	0.629	1.376	2.419
26-39	955	0.310	0.671	1.388	2.411
18-39	955	0.338	0.738	1.508	2.636
14-43	955	0.294	0.682	1.500	2.619
22-43	955	0.310	0.669	1.360	2.355
30-43	955	0.327	0.722	1.496	2.571
38-27	955	0.302	0.671	1.400	2.410
34-23	955	0.332	0.706	1.492	2.546
46-27	955	0.310	0.690	1.460	2.466
42-23	955	0.341	0.751	1.504	2.587
54-27	955	0.322	0.730	1.572	2.764
50-23	955	0.305	0.650	1.368	2.378
50-31	955	0.318	0.716	1.516	2.503
42-31	955	0.332	0.716	1.536	2.594
34-39	955	0.422	0.866	1.712	2.820
46-35	955	0.297	0.647	1.396	2.458
54-43	955	0.318	0.679	1.408	2.394
46-43	955	0.286	0.655	1.412	2.530
38-43	955	0.414	0.850	1.664	2.763
34-15	955	0.318	0.661	1.400	2.443
38-19	955	0.308	0.698	1.524	2.683
42-15	955	0.294	0.684	1.456	2.530
46-19	955	0.310	0.695	1.472	2.555
54-19	955	0.310	0.671	1.412	2.475
34-31	955	0.300	0.682	1.480	2.572
38-35	955	0.292	0.674	1.500	2.530
42-39	955	0.308	0.666	1.460	2.496
50-47	955	0.302	0.677	1.412	2.450

3.0 Results (Continued)

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

3.5.3 Analysis (Continued)

Scram Testing (Continued)

Table STI 5-3 (Continued)
Individual Rod Scram Times
Sequence A Rods (Continued)

Location	Reactor Press. Psig	Scram Insertion Times, Sec.			
		5%	20%	50%	90%
42-47	955	0.265	0.637	1.364	2.411
34-47	955	0.419	0.850	1.620	2.755
38-03	955	0.302	0.653	1.388	2.420
38-11	955	0.365	0.757	1.480	2.523
50-15	955	0.314	0.738	1.563	2.683
50-39	955	0.314	0.719	1.492	2.643
46-51	955	0.310	0.663	1.388	2.451
38-51	955	0.318	0.695	1.464	2.539
34-07	955	0.294	0.653	1.360	2.338
42-07	955	0.310	0.637	1.344	2.322
46-11	955	0.302	0.653	1.356	2.354
58-23	955	0.305	0.626	1.288	2.242
58-31	955	0.350	0.709	1.228	2.435
54-35	955	0.316	0.695	1.464	2.466
58-39	955	0.318	0.677	1.380	2.395
38-59	955	0.308	0.679	1.412	2.434
42-55	955	0.308	0.679	1.384	2.378
34-55	955	0.324	0.703	1.484	2.612

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

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

3.5.3 Analysis (Continued)

Scram Testing (Continued)

Table STI 5-3 (Continued)
Individual Rod Scram Times
Sequence B Rods

Drive Location	Reactor Press. Psig	Scram Insertion Times, Sec.				
		5%	20%	50%	90%	
54-47	1000	0.329	0.687	1.41	2.44	
34-59	1000	0.329	0.669	1.38	2.39	
50-51	1000	0.324	0.669	1.38	2.41	
42-51	1000	0.326	0.687	1.42	2.47	
34-51	1000	0.381	0.143	1.51	2.60	
30-07	1000	0.340	0.709	1.52	2.64	
38-07	1000	0.324	0.685	1.40	2.41	
34-11	1000	0.313	0.679	1.46	2.51	
50-11	1000	0.329	0.669	1.34	2.33	
54-15	1000	0.331	0.666	1.34	2.31	
54-31	1000	0.342	0.685	1.39	2.38	
50-35	1000	0.366	0.746	1.52	2.57	
58-35	1000	0.321	0.656	1.32	2.27	
42-59	1000	0.313	0.642	1.31	2.27	
46-55	1000	0.353	0.719	1.42	2.41	
38-55	1000	0.332	0.711	1.51	2.63	
30-55	1000	0.326	0.703	1.44	2.51	
26-03	1000	0.324	0.671	1.40	2.45	
18-03	1000	0.342	0.682	1.39	2.41	
18-11	1000	0.342	0.690	1.41	2.46	
10-11	1000	0.338	0.724	1.60	2.75	
02-14	1000	0.342	0.703	1.47	2.56	
06-31	1000	0.342	0.693	1.42	2.45	
02-35	1000	0.329	0.669	1.36	2.37	
02-43	1000	0.350	0.669	1.36	2.32	
18-59	1000	0.337	0.687	1.46	2.55	
26-59	1000	0.337	0.695	1.44	2.47	
14-55	1000	0.362	0.738	1.40	2.56	
22-55	1000	0.324	0.703	1.50	2.57	
22-07	1000	0.313	0.658	1.40	2.45	
14-07	1000	0.340	0.709	1.46	2.57	
26-11	1000	0.397	0.757	1.49	2.60	
02-27	1000	0.326	0.679	1.44	2.45	
10-35	1000	0.356	0.741	1.46	2.52	
06-39	1000	0.313	0.682	1.46	2.53	

Location	Reactor Press. Psig	Scram Insertion			
		5%	20%	50%	
34-27	1000	0.348	0.703	1.44	2.48
42-27	1000	0.326	0.682	1.40	2.42
38-23	1000	0.321	0.714	1.59	2.68
50-27	1000	0.313	0.663	1.41	2.43
46-23	1000	0.315	0.674	1.44	2.52
38-31	1000	0.334	0.703	1.52	2.67
30-39	1000	0.443	0.837	1.55	2.62
38-39	1000	0.313	0.653	1.42	2.52
50-43	1000	0.346	0.732	1.50	2.63
42-43	1000	0.342	0.701	1.47	2.55
34-43	1000	0.342	0.711	1.50	2.59
30-23	1000	0.334	0.706	1.52	2.62
34-19	1000	0.326	0.671	1.42	2.45
38-15	1000	0.320	0.722	1.52	2.60
42-19	1000	0.313	0.679	1.46	2.51
46-15	1000	0.324	0.687	1.41	2.43
50-19	1000	0.321	0.677	1.41	2.55
46-31	1000	0.316	0.695	1.49	2.61
30-31	1000	0.430	0.831	1.56	2.62
34-35	1000	0.476	0.898	1.62	2.67
42-35	1000	0.332	0.703	1.51	2.65
46-47	1000	0.310	0.653	1.37	2.36
38-47	1000	0.324	0.698	1.48	2.54
30-47	1000	0.330	0.711	1.49	2.57
46-07	1000	0.348	0.730	1.58	2.78
34-03	1000	0.324	0.703	1.49	2.56
42-03	1000	0.318	0.671	1.41	2.43
30-15	1000	0.321	0.706	1.54	2.64
42-11	1000	0.313	0.677	1.44	2.56
58-19	1000	0.321	0.653	1.32	2.29
54-23	1000	0.313	0.653	1.35	2.34
58-27	1000	0.313	0.653	1.348	2.307
46-39	1000	0.302	0.645	1.38	2.44
54-39	1000	0.313	0.669	1.38	2.40
58-43	1000	0.318	0.677	1.39	2.41

3.0 Results (Continued)

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

3.5.3 Analysis (Continued)

Scram Testing (Continued)

Table STI 5-3 (Continued)
Individual Rod Scram Times
Sequence B Rods (Continued)

Drive Location	Reactor Press. Psig	Scram Insertion Times, Sec.				
		5%	20%	50%	90%	
10-51	1000	0.313	0.642	1.35	2.37	
18-51	1000	0.318	0.693	1.44	2.45	
26-51	1000	0.367	0.754	1.53	2.64	
26-19	1000	0.342	0.716	1.52	2.60	
22-15	1000	0.326	0.711	1.56	2.75	
18-19	1000	0.362	0.756	1.54	2.60	
14-15	1000	0.338	0.732	1.58	2.75	
06-51	1000	0.342	0.706	1.43	2.49	
10-19	1000	0.353	0.698	1.47	2.50	
14-31	1000	0.334	0.711	1.50	2.57	
26-35	1000	0.326	0.716	1.63	2.77	
18-35	1000	0.326	0.679	1.48	2.55	
06-47	1000	0.329	0.671	1.36	2.36	
14-47	1000	0.313	0.661	1.39	2.45	
22-47	1000	0.342	0.687	1.41	2.40	
22-23	1000	0.350	0.722	1.48	2.54	
26-27	1000	0.358	0.733	1.49	2.56	
14-23	1000	0.318	0.700	1.52	2.67	
18-27	1000	0.318	0.714	1.528	2.595	
06-23	1000	0.313	0.653	1.36	2.35	
10-27	1000	0.329	0.693	1.40	2.35	
22-31	1000	0.340	0.701	1.52	2.61	
22-39	1000	0.329	0.687	1.44	2.47	
14-39	1000	0.337	0.666	1.38	2.43	
10-43	1000	0.329	0.701	1.42	2.43	
18-43	1000	0.342	0.714	1.56	2.66	
26-43	1000	0.345	0.671	1.44	2.51	

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

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

3.5.3 Analysis (Continued)

Scram Testing (Continued)

Table STI 5-4
Summary of Scram Test Results

Reactor Pressure	Accumulator Pressure	Number Of Rods	Mean Insertion Times (Sec.) 5%	20%	50%	90%
Tech Spec			0.375	0.90	2.0	3.5
0	Normal	185	0.286	0.511	1.007	1.664
0	Minimum	4*	0.317	0.578	1.13	1.95
600	Normal	4*	0.321	0.661	1.46	2.92
800	Normal	4*	0.350	0.768	1.66	2.90
1000	Zero	4*	0.355	0.763	1.60	2.71
1000	Normal	185	0.327	0.695	1.45	2.51

* Four slowest in-sequence rods.

The scram insertion times of the four selected in-sequence CRD's were measured in conjunction with full-core scrams per STI-75, Reactor Scram From Outside The Control Room, STI-27, Turbine Trip, STI-25, Main Steam Isolation Valve Full Isolation, and STI-27, Generator Load Rejection. All applicable criteria were met. The results are summarized in table STI 5-5.

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

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

3.5.3 Analysis (Continued)

Scram Testing (Continued)

Table STI 5-5
Four In-Sequence Rods Scram Tests

Reactor Scram	Reactor Power (%)	CRD	Scram Insertion Times (sec) 5%	20%	50%	90%
Tech. Spec. Limit			0.375	0.90	2.0	3.5
STI-75 Rx Scram From Outside Control Room	10%	30-27	.343	.770	1.572	2.717
		18-07	.340	.756	1.620	2.642
		26-15	.332	.732	1.584	2.732
		14-19	.338	.780	1.624	2.805
STI-27 Turbine Trip	75%	30-27	.265	.553	1.18	2.66
		18-07	.265	.571	1.22	2.16
		14-19	.265	.579	1.28	2.25
		26-15	.265	.581	1.67	2.11
STI-25 MSIV Full Isolation	86%	30-27	.324	.677	1.42	2.55
		18-07	.316	.685	1.48	2.64
		26-15	.324	.729	1.56	2.74
		14-19	.324	.727	1.56	2.74
STI-27 Generator Load Rejection	98.5%	18-07	.336	.679	1.484	2.603
		26-15	.313	.669	1.432	2.564
		14-15	.313	.722	1.556	2.758
		46-07	.289	.655	1.468	2.547

3.0 Results

3.6 STI-6, SRM Performance and Control Rod Sequence

3.6.1 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.

3.6.2 Criteria

Level 1

There must be a neutron signal-to-noise ratio of at least 2:1 on the required operable SRM's or fuel loading chamber prior to pulling rods.

There must be a minimum count rate of 3 cps on the required operable SRM's or fuel loading chambers prior to pulling rods.

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

The RSCS shall be operable as specified in the technical specification 3.3.B.

3.6.3 Analysis

STI-6 testing was performed during the open vessel, initial critical and heatup phases, and at test condition 1 as defined on the power flow map in section 2.3.

The operational sources were loaded in a manner consistant with STI-3 fuel loading as shown in figure STI 6-1.

Prior to pulling rods the SRM's were demonstrated to have a count rate greater than 3 cps and a signal-to-noise ratio greater than 2:1 by taking count rate data with the detector fully withdrawn and fully inserted. This data is contained in table STI 6-1. The SRM Hi Hi trips were initially set to 5×10^5 cps.

3.0 Results (Continued)

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

3.6.3 Analysis (Continued)

Prior to pulling rods for the initial critical Rod Sequence Control System (RSCS) was demonstrated to be operable by the performance of surveillance test SI 4.3.B.3-2. This surveillance performs a system diagnostic test and demonstrates that the RSCS will not allow selection of out-of-sequence rods, thereby assuring compliance with technical specification 3.3.B.

The reactor was brought critical in rod sequence B on the 18th notch of the 29th rod (38-15) with a moderator temperature of 92° F. The period was determined to be 132 seconds.

The IRM's were shown to be functional, and to overlap with the SRM's. The non-coincident scram circuitry was removed from the SRM's and they were subsequently shown not to saturate at a count rate of 7.5×10^5 cps.

The reactor was heated up from atmospheric to rated pressure by pulling control rods in sequence B. Neutron instrumentation was monitored to insure a safe heat-up rate. The RSCS prevented out-of-sequence rod movement, thus minimizing the worth of individual rods. No anomalies were noticed and control rod sequence B performed acceptably.

The reactor was heated up and brought to approximately 30% of rated power in sequence A. Performance of control rod sequence A was acceptable. The RSCS was verified to perform properly at 22% and 27% of rated thermal power as evidenced by the inability to select out-of-sequence rods. The RSCS enforcement interlock cleared at 27.9% of rated thermal power.

Level 1 criteria were met for all phases of STI-6 testing. No level 2 criteria apply.

3.0 Results (Continued)

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

3.6.3 Analysis (Continued)

Table STI 6-1

SRM Channel	SRM Count Rate (cps)			
	A	B	C	D
SRM Fully Inserted	45	45	110	32
SRM Fully Withdrawn	.1	.1	.1	1.
Signal-to-Noise Ratio	449	449	1099	31

3.0 Results

3.7 STI-9, Water Level Measurements

3.7.1 Purpose

The purposes of this test are:

1. To check the calibration of the various narrow and wide range indicators.
2. To measure the reference leg temperature and recalibrate the wide range instruments if the measured temperature is different than the value assumed during the initial calibration.
3. Collect plant data which can be used to investigate the effects of core flow, carryunder and subcooling on indicated wide range level.

3.7.2 Criteria

Level 1

Not applicable

Level 2

The GEMAC indicator readings on the narrow range level system should agree within ± 1.5 inches of the average reading.

The wide range level indicators should agree within ± 6 inches of the average reading.

3.7.3 Analysis

STI-9 testing was conducted at heatup, test conditions 1 and 4E, as defined on the power flow map in section 2.3. Calibrations of the GEMAC and Yarway water level instrumentation were verified to give accurate reactor water level indication at all times. Graphs of indicated water level versus power (flow constant) and indicated water level versus flow (power constant) were plotted from data accumulated during the startup test program to obtain knowledge of the tracking performance of these level systems (refer to figures STI 9-1 and STI 9-2). Note that at high flows, the Yarway level was approximately 13 inches lower than the GEMAC readings due to flow velocity effects on the Yarway vessel taps.

3.0 Results (Continued)

3.7 STI-9, Water Level Measurements (Continued)

3.7.3 Analysis (Continued)

At test condition 4E the average Yarway reference column temperatures were 265°F and 256°F for columns A and B, respectively. This indicates excellent agreement with the assumed cold water calibration reference leg temperature of 264°F.

The GEMAC water level indicators read within ± 1.5 inches of their average reading of 33.5 inches. All wide range level indicators agreed within ± 6 inches of the average reading except for 4 indicators which were one to two inches outside criteria. These 4 indicators were recalibrated and verified to meet criteria.

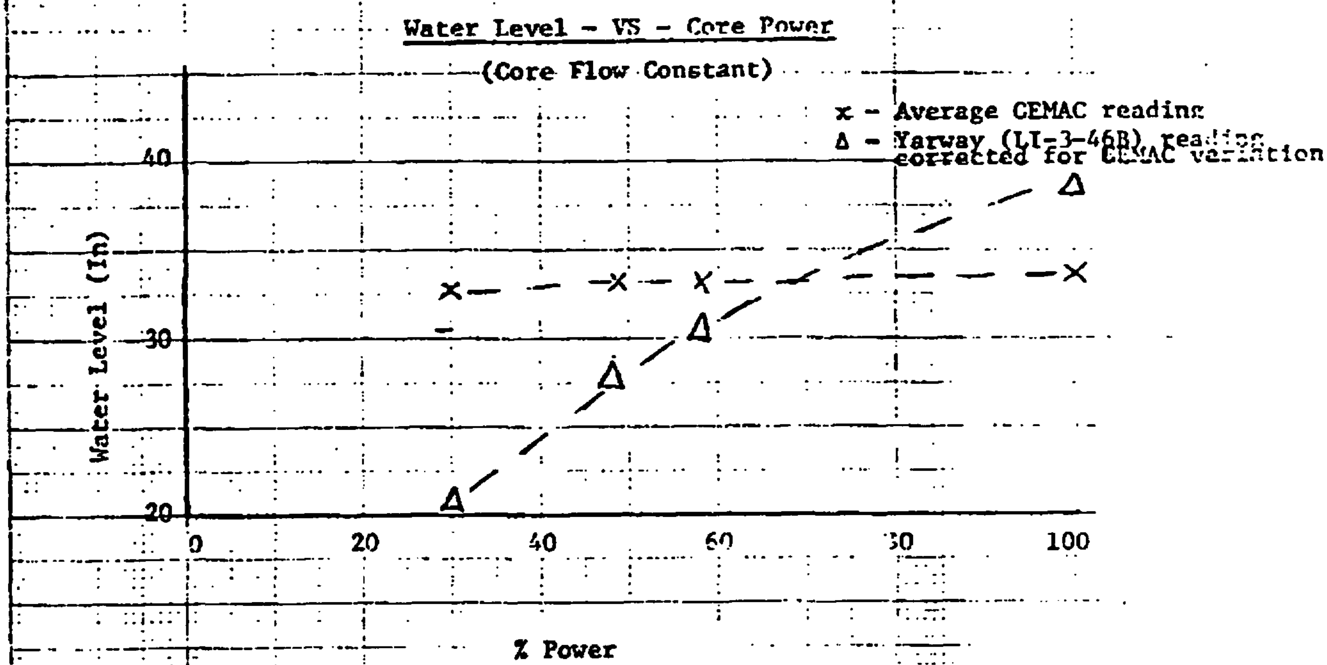


Figure STI 9-1

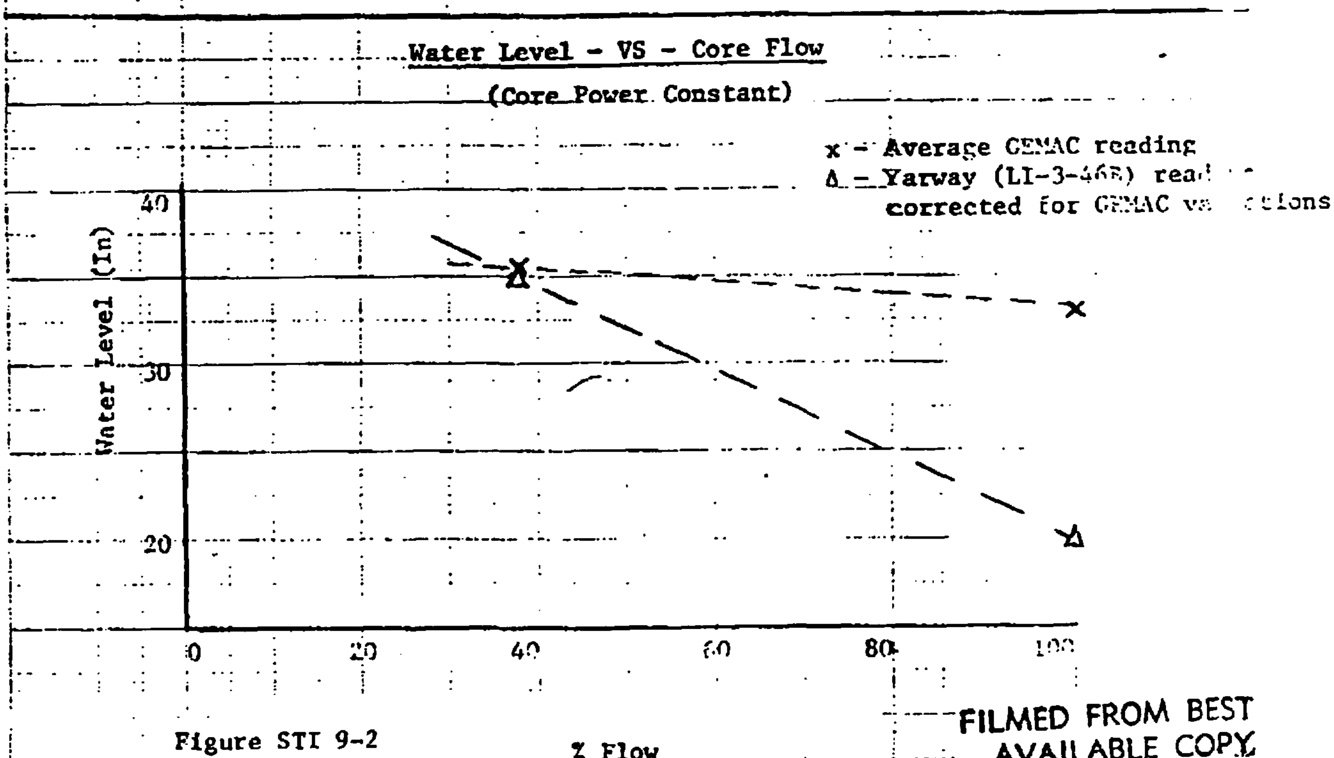


Figure STI 9-2

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FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results

3.8 STI-10, IRM Performance

3.8.1 Purpose

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

3.8.2 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 (96%) of full scale.

Level 2

Not applicable.

3.8.3 Analysis

STI-10 testing was conducted at open vessel, initial heatup, and test condition 1 levels as defined on the power flow map in section 2.3.

Prior to pulling rods for the initial critical the IRM's were fully inserted and adjusted to give a scram at 120/125 of full scale per surveillance test SI 4.2.C-3.

Rods were withdrawn in rod sequence B to bring the reactor critical. All the IRM's were on scale before any of the normalized SRM readings reached the operational limit of 2.0×10^7 cps. All IRM's responded to changes in neutron flux.

The reactor was taken subcritical and the non-coincidence scram shorting links were removed. All applicable criteria were met.

During the initial heatup, the IRM's were adjusted to correspond to the reactor power level as measured by the calibrated APRM's. This verifies the IRM/APRM overlap. Following this adjustment the IRM/SRM overlap was reverified, and surveillance test SI 4.2.C-3 was performed to verify that the IRM's will provide a scram signal at 120/125 of full scale.

3.0 Results (Continued)

3.8 STI-10, IRM Performance (Continued)

3.8.3 Analysis (Continued)

With the reactor at test condition 1 (approximately 30%) the IRM's were adjusted in accordance with surveillance test SI 4.1.B-1 to read consistent with the APRM's. All IRM's read equal to or greater than the APRM's. During a subsequent reactor startup satisfactory IRM/SRM overlap was verified.

All STI-10 criteria were satisfied.

3.0 Results

3.9 STI-11, LPRM Calibration

3.9.1 Purpose

The purpose of STI-11 is to calibrate the Local Power Range Monitor (LPRM) system.

3.9.2 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

Not applicable.

3.9.3 Analysis

STI-11 testing was conducted at heatup, test conditions 1, 2E, 3E, and 4E levels as defined on the power flow map in section 2.3.

With the reactor at hot standby, LPRM hookup and response was checked in conjunction with STI-5, control rod drive scram testing. Detector 32-49C could not be verified because of upscale failure. All other LPRM's responded satisfactorily to flux changes. During operation at test condition 3E it was discovered that LPRM's 56-33A and B had their leads reversed. These two LPRM's were bypassed until their leads were correctly connected during the next outage.

The operable LPRM's were calibrated at test conditions 1, 2E, 3E, and 4E. This corresponds to power levels of 21%, 52%, 76%, and 96% of rated power, respectively. The Traversing Incore Probe (TIP) system interface with the unit 3 process computer was not operational for the initial LPRM calibration at test condition 1. A full set of tip traces were taken and the data digitized for manual input into the BUCLE offline computer program. The gain adjustment factors (GAF) were calculated by BUCLE and used to calibrate the LPRM's to read proportional to the neutron flux according to surveillance test SI 4.1.B-3. A second TIP set was run and the data digitized and loaded

3.0 Results (Continued)

3.9 STI-11, LPRM Calibration (Continued)

3.9.3 Analysis (Continued)

into BUCLE. The GAF's calculated showed 148 of the 169 operable LPRM's reading properly with 21 needing recalibration. Twenty-three LPRM's were recalibrated according to the GAF's calculated by BUCLE. Following this calibration the TIP interface with the process computer was available. Therefore, a full tip set was loaded into the process computer. GAF's calculated by the process computer and BUCLE agreed within ± 10%.

For LPRM calibrations at test conditions 2E, 3E, and 4E the process computer was used to calculate the GAF's. The calculations at test condition 2E were verified by the offline computer program BUCLE. Agreement was within ± 1%. The calibrations were performed according to surveillance test SI 4.1.B-3.

At all times there were more than 14 operable LPRM's per APRM channel. This is the minimum number required for an APRM channel to be operable. There were 2, 6, and 3 LPRM's inoperable at test conditions 2E, 3E, and 4E, respectively.

The LPRM's were adjusted to read proportional to the neutron flux in the narrow-narrow water gap, thereby satisfying all criteria.

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results

3.10 STI-12, APRM Calibration

3.10.1 Purpose

The purpose of STI-12 is to calibrate the Average Power Range Monitor (APRM) System.

3.10.2 Criteria

Level 1

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

Technical Specification and fuel warranty limits on APRM scram and rod block shall not be exceeded.

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

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 do not read greater than the actual core thermal power by more than 7% of rated power.

3.10.3 Analysis

STI-12 testing was performed at heat up, test conditions 1, 2E, 3E, and 4E levels as defined on the power flow map in section 2.3

Prior to pulling rods for the initial startup the APRM's were set to scram at $\leq$ 15% and to give a control rod withdrawal block at $\leq$ 12% by the performance of surveillance test SI 4.2.C-1.

Initially the APRM's were calibrated based on the low power heat balance calculated using the heat-up rate. The heat-up rate was measured to be approximately 70° F/hr. Gain adjustment factors were calculated for each APRM, and

3.0 Results (Continued)

3.10 STI-12, APRM Calibration (Continued)

3.10.3 Analysis (Continued)

the APRM's were then adjusted to read 4.7% of rated thermal. This value was determined, based upon the highest APRM reading with a 0.3% margin for calculation inaccuracies.

At test conditions 1, 2E, 3E, and 4E, the APRM's were calibrated to read equal to or greater than the actual core thermal power. The core thermal power was obtained from the process computer heat balance program (OD-3). The program was verified by the offline heat balance (CORPWR) and by a detailed manual heat balance. The APRM's were recalibrated following each LPRM calibration. All calibrations were performed according to surveillance test SI 4.1.B-2. For each test condition a scram clamp was set at 20% above the nominal load line of that plateau.

Immediately after an APRM calibration at test condition 4E, power was reduced to approximately 40% using core flow and control rods and returned to the initial power level (approximately 95%). During this power ramp process computer heat balances (OD-3) were run to monitor the ability of the APRM's to track the core power level. Gain adjustment factor's for each APRM remained less than 1.0 throughout the power ramp.

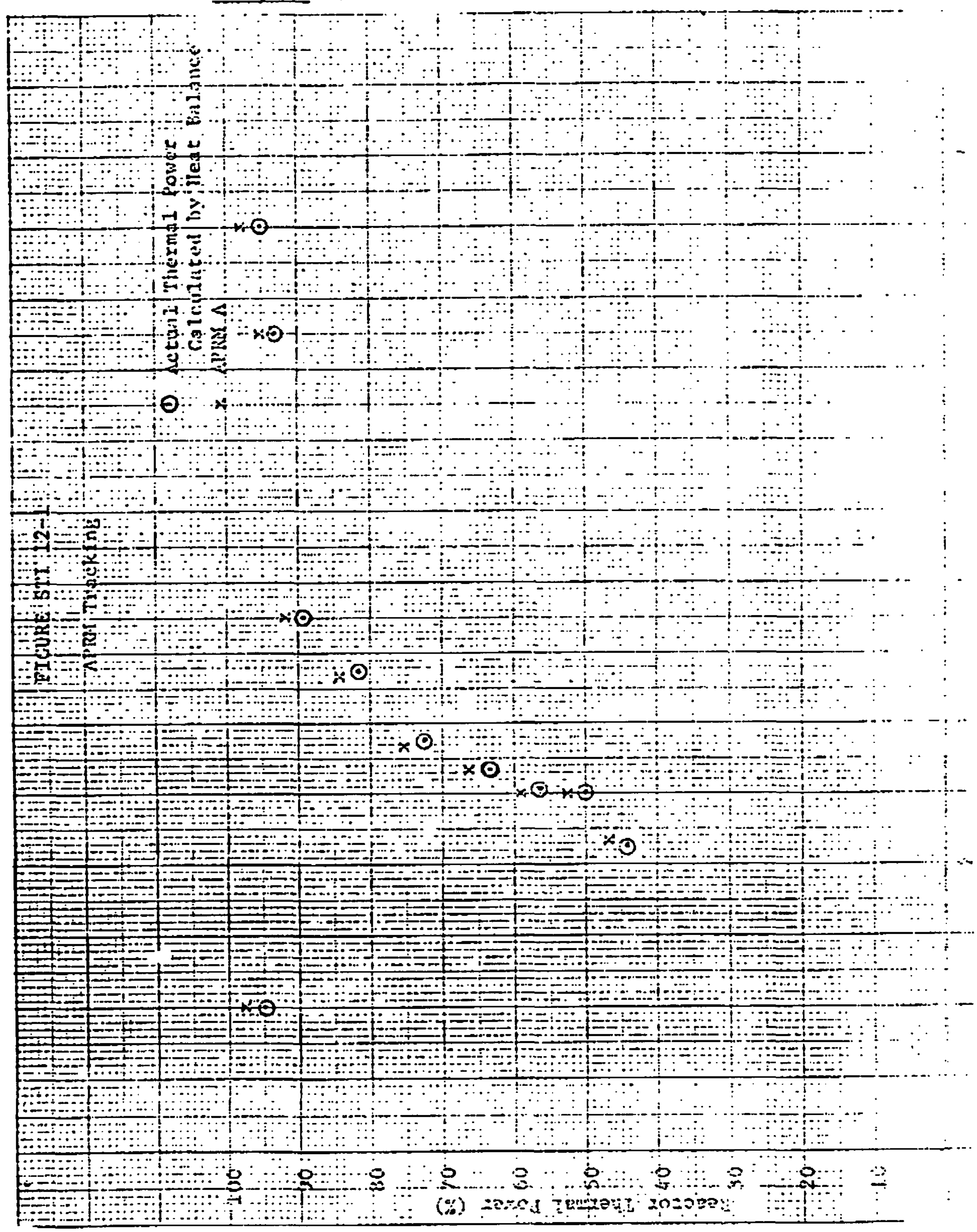
All applicable criteria for STI-12 have been satisfied at each test condition. Typical results of this APRM tracking test are shown on figure STI 12-1.

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

3.10 STI-12, APRM Calibration (Continued)

3.10.3 Analysis (Continued)



FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results

3.11 STI-13, Process Computer

3.11.1 Purpose

The purpose of STI-13 is to verify the performance of the process computer under plant operating conditions.

3.11.2 Criteria

Level 1

Not applicable

Level 2

Program OD1 and P1 will be considered operational when:

a. The MCPR calculated by BUCLE and the process computer either:

   1) Are in the same fuel assembly and do not differ in value by more than 2%, or

   2) For the case in which the MCPR calculated by the process computer is in a different assembly than that calculated by BUCLE, for each assembly, the MCPR and CPR calculated by the two methods shall agree within 2%.

b. The maximum LHGR calculated by BUCLE and the process computer either:

   1) Are in the same fuel assembly and do not differ in value by more than 2%, or

   2) For the case in which the maximum LHGR calculated by the process computer is in a different assembly than that calculated by BUCLE, for each assembly, the maximum LHGR and LHGR calculated the two methods shall agree within 2%.

c. The MAPLHGR calculated by BUCLE and the process computer either:

   1) Are in the same fuel assembly and do not differ in value by more than 2%, or

   2) For the case in which the MAPLHGR calculated by the process computer is in a different assembly

3.0 Results (Continued)

3.11 STI-13, Process Computer (Continued)

3.11.2 Criteria (Continued)

Level 2 (Continued)

c. (Continued)

2) than that calculated by BUCLE, for each assembly, the MAPLHGR and APLHGR calculated by the two methods shall agree within 2%.

d. The LPRM calibration factors calculated by the independent method and the process computer agree to within 2%.

e. The remaining programs will be considered operational upon successful completion of static and dynamic testing.

3.11.3 Analysis

Process computer testing was conducted during open vessel, heatup, and test conditions 1 and 4E. The system was re-initialized at 1830 on October 12, 1976, for the beginning of the dynamic testing.

The dynamic system test case was completed at 51.1% power and 102.7% flow with the exception of minor testing on subsidiary programs. The manually calculated heat balance agreed to within 0.7% of the OD-3 calculated heat balance. The offline program BUCLE and P1 were compared and all the thermal limits agreed to within 0.2%. Core thermal hydraulic calculations, exposure calculations, and exposure updating were verified as being correct by comparing with manual calculations or BUCLE. LPRM calibration factors as calculated by the process computer and BUCLE agreed within 1%. See table STI 13-1 for comparison of process computer and BUCLE results.

3.0 Results (Continued)

3.11 STI-13, Process Computer (Continued)

3.11.3 Analysis (Continued)

Table STI 13-1

Comparison of Process Computer and BUCLE Results

Variable	Symbol	Process Computer	BUCLE	% Difference
Critical Power Ratio	MCPR	2.431	2.431	0%
Linear Heat Generation Rate	MLHGR	6.003	6.017	0.23%
Average Planar Heat Generation Rate	MAPLHGR	5.05	5.06	0.2%

NOTE: The core locations of MCPR, MLHGR, and MAPLHGR limits were the same as calculated by the process computer and BUCLE.

3.0 Results

3.12 STI-14, Reactor Core Isolation Cooling System

3.12.1 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.

3.12.2 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 measured value may be used if available.)

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

For small speed or flow demand changes while injecting into the reactor vessel in either manual or automatic mode, the decay ratio of each recorded RCIC system variable must be less than 0.25, in order to demonstrate acceptable stability.

The maximum RCIC turbine speed during quick starts shall be at least 10% below the overspeed trip setting.

3.12.3 Analysis

STI-14 testing was conducted at heatup and test condition 1 as defined by the power flow map in section 2.3 The RCIC system demonstrated under all test conditions the ability to reach rated flow in less than 30 seconds. After

3.0 Results (Continued)

3.12 STI-14, Reactor Core Isolation Cooling System (Continued)

3.12.3 Analysis (Continued)

running the rated pressure test, the system response was improved by lengthening the control system ramp stroke time from 14 to 19 seconds. After the adjustment the three test points were repeated. The results of these three tests during heatup and the cold quick start reactor vessel injection test are presented in table STI 14-1.

Required system flow of 600 gpm was reached at all test conditions and the RCIC turbine did not trip. The turbine gland seal condenser system prevented steam leakage. The high steam flow isolation switch trip was conservatively set to actuate at $\leq$ 450 inches of water per the technical specifications. All process variables exhibited a decay ratio of less than .25. The maximum RCIC turbine speed during the quick start test was 4375 rpm which is more than 10% below the overspeed trip setting.

During each test condition it was noted that the barometric condenser did not develop a sufficient vacuum. Repair work to the vacuum pump is pending arrival of parts to improve vacuum pump performance. The RCIC high steam flow switches were found to have a required setpoint (calculated via field data) greater than the installed instrument range of 500 inches of water. G.E. Design Engineering evaluated the data and calculated the setpoint to be 1064 inches of water. Final resolution to the problem is pending TVA's review.

Experience has shown that after extended periods of idleness, the margin to the RCIC turbine overspeed setpoint may be reduced on a cold quick start. The reason for this is that the Woodward actuator receives its oil supply from a separate sump, resulting in a starved oil supply actuator. A modification to the auxiliary oil sump in the oil supply line to the Woodward EG-R hydraulic actuator has been specified.

Based on the observed system operation and the transient recordings, it was concluded that RCIC was fully operational. The final RCIC controller settings are as follows:

Proportional Band:	600
Resets per Minute:	100
Ramp Time:	19 seconds
Ramp Idle:	-0.5 volt
EGR Needle Valve:	1/2 turn ccw

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

3.12 STI-14, Reactor Core Isolation Cooling System (Continued)

3.12.3 Analysis (Continued)

Table STI 14-1

Results of RCIC Tests

Test Condition	Measured		Required		Reactor Pressure psig	Pump Discharge Pressure		Turbine Speed		Controller	
	Flow gpm	Time (sec)	Flow gpm	Time		Measured psig	Required psig	S.S. rpm	Peak rpm	R/M	F.S.
Heatup	600	9.75	600	30	140	230	240	2000	2000	[illegible]	[illegible]
Heatup	612.5	16.5	600	30	590	710	690	3300	3875	100	600
Heatup	612.5	18.75	600	30	980	1120	1080	4010	4375	100	600
Heatup	618.0	19.75	600	30	980	1220	1220	4035	4375	100	600
1	610	20	600	30	954	1010	N/A	3900	4125	100	600

RCIC electrical turbine trip setpoint: 4950 rpm

3.0 Results

3.13 STI-15, High Pressure Coolant Injection System

3.13.1 Purpose

The purpose of this test is to verify the proper operation of the high pressure coolant injection system over its required operating pressure range.

3.13.2 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.

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 be used, if available.)

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.

For small speed or flow demand changes while injecting into the reactor vessel in either manual or automatic mode, the decay ratio for each recorder HPCI system variable must be less than 0.25, in order to demonstrate acceptable stability.

The maximum HPCI turbine speed during quick starts shall be at least 10% below the overspeed trip setting.

3.13.3 Analysis

STI-15 testing was conducted at heatup and test condition 2E as defined on the power flow map in section 2.3. During the heatup testing phase, the High Pressure Coolant Injection system (HPCI) took suction from and discharged to the condensate storage tank. The first test at 150 psig

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

3.13 STI-15, High Pressure Coolant Injection System (Continued)

3.13.3 Analysis (Continued)

was repeated because the test valve (73-35) was not de-energized. Since the valve was shut, the discharge pressure continued to climb to 1200 psig after reaching rated flow when the valve was opened. The system performed satisfactorily during the second test at 150 psig.

The 1100 psig pump discharge pressure test was repeated due to a slow opening time on the HPCI stop valve. To increase the opening time, the ramp generator stroke time was changed from 14 to 12 seconds, and the test was repeated successfully. Observed pump performance was within the tolerance of the vendor pump performance results. The final controller settings on HPCI were as follows:

Proportional Band:	600%
Reset per Minute:	100%
Ramp Generator Stroke Time:	12 seconds
Ramp Idle:	-0.5 volts
EGR Needle Valve:	1/2 turn CCW

The maximum time required to reach 5000 gpm at any reactor pressure between 150 psig and rated was $\leq$ 24 seconds; the HPCI system flow was $\geq$ 5000 gpm at all pressures between 150 psig and 1220 psig; and the turbine did not trip off during testing. This satisfied all level 1 criteria.

The turbine gland seal condenser system prevented steam leakage to the atmosphere. The decay ratio for each recorded HPCI system parameter was $\leq$ .25 for a 5% flow step change while injecting to the vessel.

Using the steady-state steam line ΔP indicator readings, the calculated steam line high flow trip settings were greater than the maximum instrument range (100 psig) and greater than allowed by technical specifications. GE Engineering Design has evaluated the data and determined that the differential pressure setpoint should be 114 psi. Final resolution to the problem is pending TVA DED review.

The HPCI turbine speed peaked at 4700 rpm during the vessel injection test due to an air pocket formed beneath the stop valve hydraulic oil piston. The result was that the stop valve initially spiked open and then returned to its normal opening ramp. An ECN to correct this problem by rerouting the oil line to the stop valve hydraulic actuator was approved and awaits receipt of the necessary materials.

3.0 Results (Continued)

3.13 STI-15, High Pressure Coolant Injection System (Continued)

3.13.3 Analysis (Continued)

The final results of testing performed at each test condition is presented in table STI 15-1.

Table STI 15-1

Final Results of HPCI Testing

Test Condition: Date	Measured		Required		Reactor Pressure (psig)	Pump Discharge Press.		Turbine Speed	
	Flow (gpm)	Time (sec)	Flow (gpm)	Time (sec)		Actual (psig)	Required (psig)	Maximum (rpm)	Stable (rpm)
Heatup 8/24/76	5062	14.5	5000	≤ 25	800	890	900	4190	3830
Heatup 8/28/76	5050	17.3	5000	≤ 25	161	300	250	2560	2375
Heatup 8/29/76	5060	23.5	5000	≤ 25	1000	1100	1100	4470	3844
Heatup 8/29/76	5125	23.75	5000	≤ 25	1000	1200	1200	4500	4000
T.C. 2E 10/17/76	5000	24	5000	≤ 25	950	1050	1050	4700	3800
T.C. 2E 10/25/76	5000	24	5000	≤ 25	930	1030	1030	4650	3750

3.0 Results

3.14 STI-16, Selected Process Temperatures

3.14.1 Purpose

The purposes of STI-16 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.

3.14.2 Criteria

Level 1

The reactor recirculation pumps shall not be operated unless the coolant temperatures between the upper and lower regions of the reactor vessel are within 145°F (80°C).

Level 2

The bottom head coolant temperature as measured by the bottom drain line thermocouple should be within 50°F (28°C) of reactor coolant saturation temperature.

3.14.3 Analysis

STI-16 testing was conducted at heatup and test conditions 2A, 2E, and 4A as defined on the power flow map in section 2.3. The results for selected process temperatures for all the test conditions are presented in table STI 16-1. Note that in natural circulation the flow is insufficient to maintain the bottom drain line temperature and reactor coolant saturation temperature within 50°F. Since steady state operation without forced recirculation is not permitted by the technical specifications, except during the startup testing, this criteria does not apply to natural circulation.

The difference between the bottom head drain line temperature and the reactor coolant saturation temperature was 79°F during single recirculation pump trips at test condition 4E. This does not meet the level 2 criteria and the problem will be resolved during the first refueling outage.

3.0 Results (Continued)

3.14 STI-16, Selected Process Temperatures (Continued)

3.14.3 Analysis (Continued)

Table STI 16-1

Summary of Temperature Behavior (^{o}F)

Test Condition	Heatup	2A	2E	4A	4E "A" Tripped	4E "B" Tripped
Pump Discharge Temp. A	530	513	528	505	500	524
Pump Discharge Temp. B	530	513	529	505	511	513
Saturation Temp.	544	539.6	540	538	539	539
Rx. Bottom Head Drain Temp.	500	478	501	460	461	460
ΔT (Disch. - Bottom Drain)	14	35	27	45	39, 50	64, 53
ΔT (Sat. - Bottom Drain)	44	61.6*	39	78*	78	79

*Level 2 criteria not applicable in natural circulation.

3.0 Results

3.15 STI-17, System Expansion

3.15.1 Purpose

The purposes of STI-17 are to:

1. Verify that the reactor drywell piping system is free and unrestrained in regard to thermal expansion.

2. Verify that suspension components are functioning in the specified manner.

3. Provide data for calculation of stress levels in nozzles and weldments.

3.15.2 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 $\pm$ 1 inch of the defined setting.

Electrical cables shall not be fully stretched.

Level 2

Displacements of instrumented points with special recording devices shall not vary from the calculated values by more than $\pm$ 50 percent or $\pm$ 0.5 inch, whichever is smaller. Displacements of less than 0.25 inch can be neglected, since 50 percent 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 fall within a range of 150 percent of the calculated value from the initial cold position in the direction of the calculated value, and 50 percent of the calculated value from the initial position in the opposite direction of the calculated value.

3.0 Results (Continued)

3.15 STI-17, System Expansion (Continued)

3.15.2 Criteria (Continued)

Level 2 (Continued)

Hangers shall be in their operating range (between the hot and cold settings ± 10 percent).

Hydraulic shock and sway arrestors shall be within their operating range. If the operating range is not available, verify that there is a minimum of 1" stroke left for the piston.

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

3.15.3 Analysis

STI-17 testing was conducted during open vessel, heatup, and test conditions 1 and 4E as defined via the power flow map in section 2.3. Thermal expansion data for the reactor drywell piping system was obtained by actual observations and by lanyard potentiometers. In general, the drywell piping moved in the correct direction during heatup and returned to its base setting after cooldown.

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

There were no preselected hangers found to have their springs bottomed-out or fully stretched at any temperature level.

At ambient and 300°F all hydraulic shock and sway arrestors were found to be within ±1 inch of the defined setting; however, in all three heatups, some of the feedwater pipe movements did not satisfy level 1 criteria. A more extensive compilation of feedwater expansion data was sent to TVA's engineering design for review and the expansion was judged to be acceptable (refer to attachment number 1). The hydraulic shock and sway arrestors on all other systems fell within ± 1 inch of their designed setting during the three above mentioned heatups.

No electrical cables were found to be fully stretched.

3.0 Results (Continued)

3.15 STI-17, System Expansion (Continued)

3.15.3 Analysis (Continued)

Displacements of instrumented points with special recording devices did not vary from the calculated values by more than ± 50% or ± 0.5 inches, whichever was smaller. Exceptions to the criteria were resolved at the heatup test plateau (refer to attachment number 2).

The traces of the instrumented points during the heatup cycle fell within 150% of the calculated value from the initial cold position in the direction of the calculated value and within 50% of the calculated value in the opposite direction. Exceptions to this criteria were specific points on the recirculation lines and the "A" and "B" feedwater lines; however, the recirculation exceptions were eventually resolved and the feedwater exceptions were cleared as the feedwater system reached rated temperature (378° F).

All hangers were found to be between their hot and cold settings ± 10 percent with the exception of one feedwater hanger. This hanger was deemed acceptable after exhibiting correct movement at upper feedwater temperatures.

All hydraulic shock and sway arrestors were within their operating range.

All conduit connections remained fully flexible.

Three complete heatup cycles were completed on 8/2/77, 11/16/77, and 12/27/77. The comparison of these three cycles indicated that the pipe movements were approximately the same for all three cycles. Movements that deviated slightly from calculated were deemed acceptable by piping design. Table STI 17-1 summarizes the results of the displacements at rated temperature for the three cycles. Attachment 3 shows the location of the instruments monitored during the heatups.

All Level I and Level II criteria have been met for STI-17 testing.

Table STI 17-1

Displacements at Rated Temperature

		Cycle 1	Cycle 2	Cycle 3
Recirc. A Suction	X	.408	.379	.347
	Y	.054	.033	.082
	Z	-.410	-.390	-.350
Recirc. A Discharge	X	-.670	-.539	-.653
	Y	-.848	-.646	-.863
	Z	-.430	-.500	-.360
Recirc. B Suction	X	.096	-.019	.090
	Y	-.560	-.297	-.539
	Z	-1.520	-1.190	-1.450
Recirc. B Suction	X	-.907	-.869	-.900
	Y	-.190	.065	-.159
	Z	-.290	-.380	-.220
Recirc. B Discharge	X	.124	.140	.627
	Y	1.030	.789	.807
	Z	-.440	-.100	-.380
Recirc. B Pump	X	-.954	-.789	-.807
	Y	.599	.596	.627
	Z	-1.480	-1.440	-1.470
Feedwater A	X	.912**	1.253	1.559
	Z	.512**	.742	.901
Feedwater B	X	.654**	.975	1.179
	Z	-.092**	-.518	-.657

** Data taken at 268° F.

Table STI 17-1 (Continued)

Displacements at Rated Temperature

		Cycle 1	Cycle 2	Cycle 3
Main Steam A Lower	X	.536	.593	.738
	Y	.102	.125	.157
	Z	.850	.710	.910
Main Steam A Upper	X	1.674	1.335	1.975
	Y	1.192	.936	1.381
	Z	.200	.020	.050
Main Steam B Lower	X	.855	.943	1.023
	Y	.565	.518	.668
	Z	.680	.570	.500
Main Steam B Upper	X	1.431	1.541	1.656
	Y	.966	.903	1.060
	Z	.260	.100	.070
Main Steam C Lower	X	1.233	1.310	1.418
	Y	-.806	-.866	-.877
	Z	.510	.520	.250
Main Steam C Upper	X	1.790	1.561	1.953
	Y	-1.464	-1.419	-1.462
	Z	.290	.120	-.020
Main Steam D Lower	X	1.578	1.630	1.631
	Y	.033	-.062	.089
	Z	.750	.640	.740
Main Steam D Upper	X	1.929	2.229	*
	Y	-1.066	-1.041	-1.206
	Z	.140	.340	.090

* Failed potentiometer

...D STATES GOVERNMENT

Attachment 1

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : J. R. Calhoun, Chief, Nuclear Generation Branch, 702 EB-C

FROM : D. R. Patterson, Chief, Mechanical Engineering Branch, W10C126 C-K

DATE : December 22, 1976

SUBJECT: BROWNS FERRY NUCLEAR PLANT - STI-17 - SYSTEM EXPANSION - UNIT 3 - BFN-633

Reference: Your memorandum to me dated December 15, 1976, subject as above.

We have reviewed the expansion movements of the feedwater lines A and B for the 50%, 75%, and 95% power levels. Although the movements do not agree exactly with the theoretical values, we do not believe the deviations are larger than what might be expected. All of the recorded movements are in the same direction as the theoretical movements and are linear as predicted. This indicates that the feedwater lines are free to expand and are not restrained. Based upon this review, it is EN DES's judgment that it is acceptable to continue to proceed with the startup and operation of unit 3.

D. R. Patterson

JDJ:CRM:GYL
CC: E. G. Beasley, W9C165 C-K
B. S. Montgomery, 5100 MIB-K
H. C. Russell, W5C126 C-K
MEDS, E4B37 C-K

5/12/79

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[illegible]
San Jose, California

FINAL SUMMARY REPORT - BFNP UNIT 3

ATTACHMENT 2

August 27, 1976

. English
Mail Code 690

SUBJECT: TVA 3 MAIN STEAM AND RECIRCULATION PIPING THERMAL EXPANSION TEST, (STI-17)

REFERENCE: Telecopy - Thermal Data, (attached)

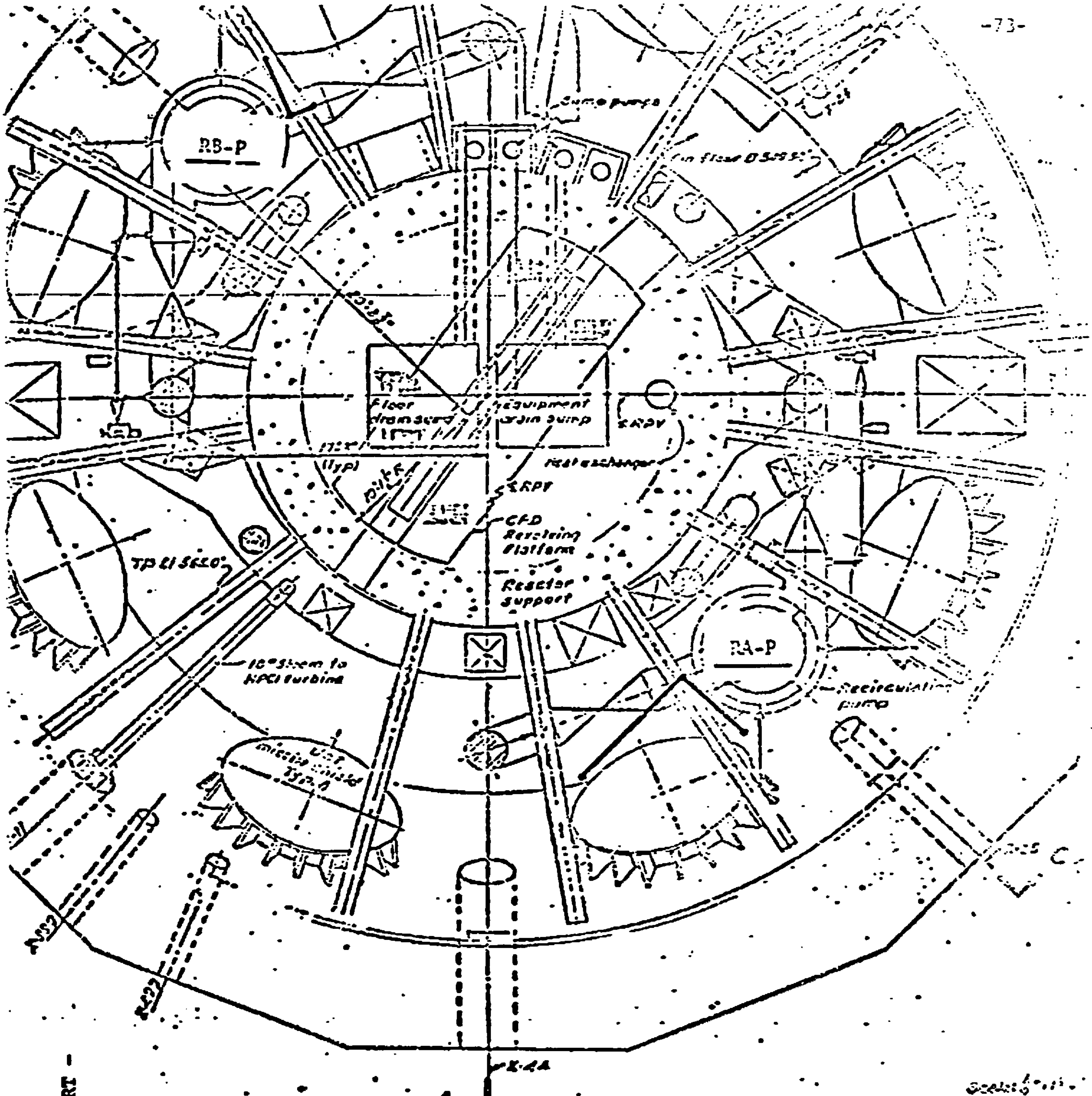
We have reviewed the field test data on the above subject, and concluded that all pipe movements seems to be going in the same expected direction from computer analysis, and it would be safe to continue heating up of the plant.

K. A. Knuzaic
Requisition Plant Piping Design
Mail Code 760 - Ext. 1785

H. M. Srivastava
Special Analysis & Analytical Procedures
Mail Code 760 - Ext. 1798

KAK/HMS/cm

ATTN. J. MILLER



Lanyard Potentiometer Locations - El. 550

Attachment 3

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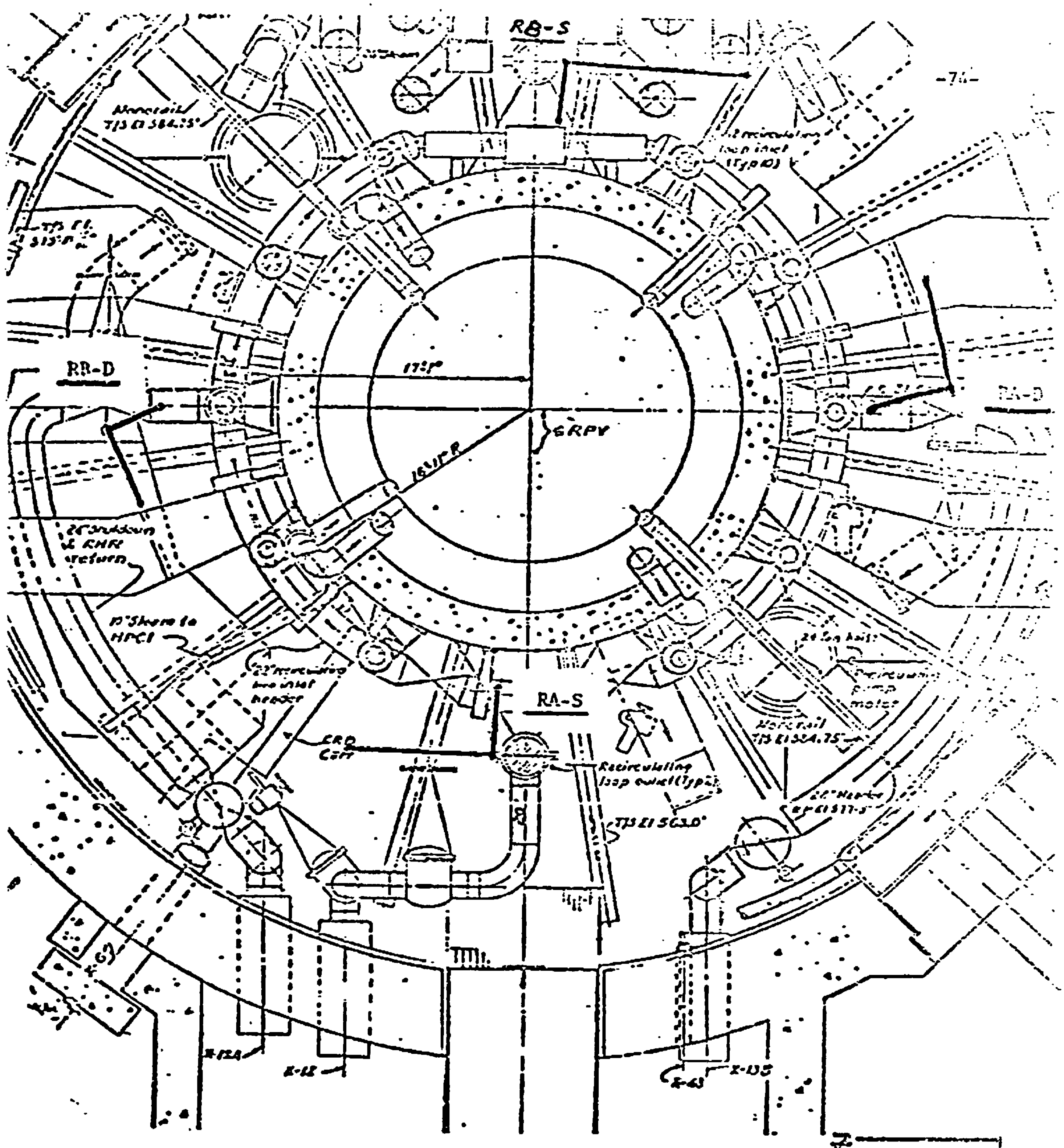
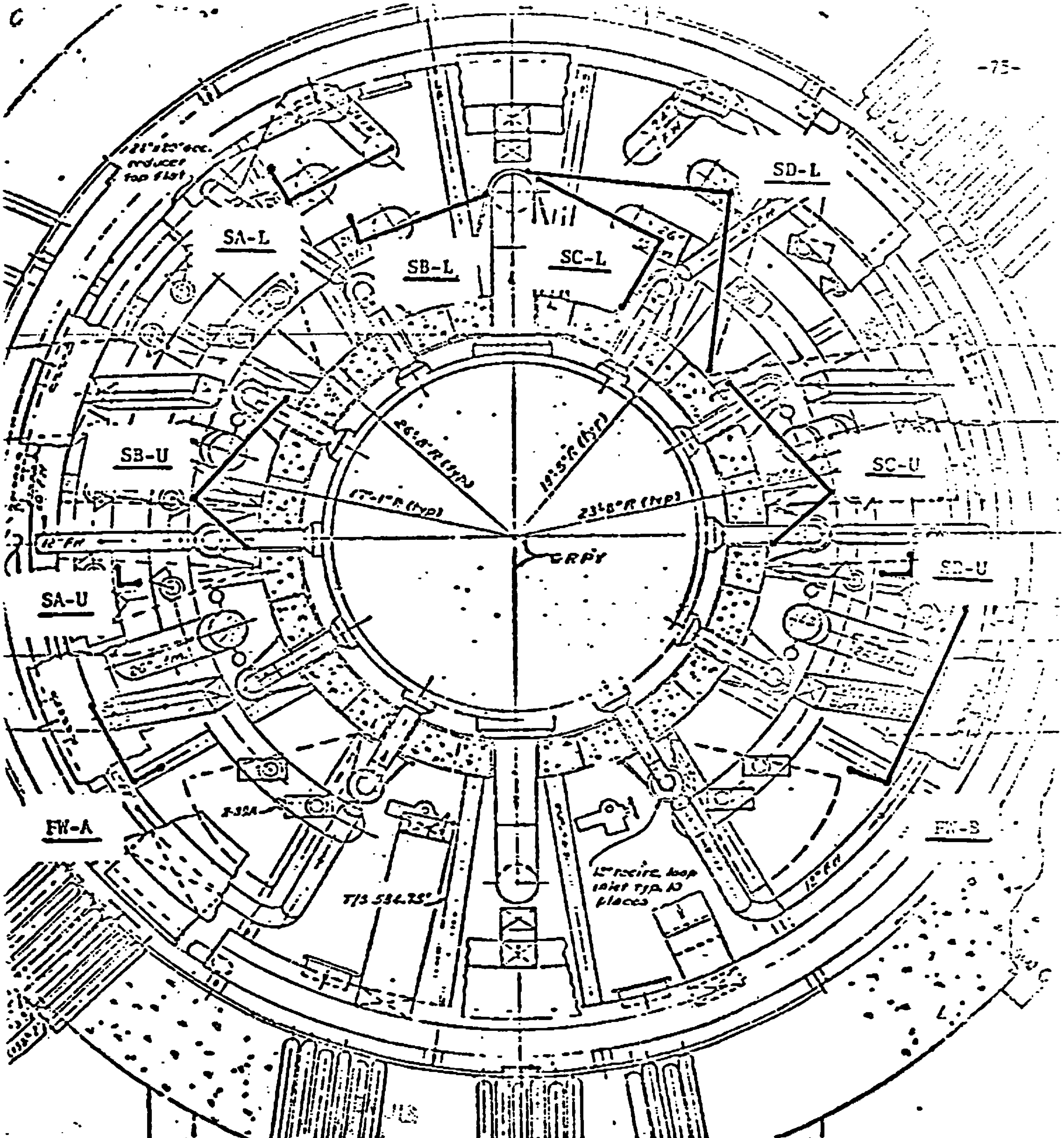


Lanyard Potentiometer Locations - El. 563

Attachment 3



Lanyard Potentiometer Locations - El. 585

<u>Attachment 3</u> (Cont'd)

3.0 Results

3.16 STI-18, Core Power Distribution

3.16.1 Purpose

The purposes of STI-18 are to:

1. Confirm the reproducibility of the TIP system readings.
2. Determine the core power distribution in three dimensions.
3. To determine core power symmetry.

3.16.2 Criteria

Level 1

The total TIP uncertainty (including random noise and geometrical uncertainties) shall be less than 7.8%. This total TIP uncertainty will be obtained by averaging the total uncertainty for all data sets obtained. A minimum of two data sets is sufficient for the determination of total TIP uncertainty. However, if the first two data sets do not meet the above criteria, testing may be continued and up to 6 data sets obtained and compared with the criteria. If the 7.8% total TIP uncertainty criteria has not been met by the 6 sets of data, testing may continue and additional data sets be obtained provided (a) the MCPR limit is adjusted to reflect the TIP uncertainty determined by the 6 data sets, (b) the NRC is informed of the adjusted MCPR limit, (c) the data generated from the 6 sets of data is transmitted to the NRC, and (d) TVA's intentions for continuing to test and expand the data base is provided to NRC. If the total TIP uncertainty is reduced by taking additional sets of data to expand the data base, the MCPR limit will be adjusted accordingly until the 7.8% total TIP uncertainty is met. At this time, the MCPR limit will be returned to its original value.

Level 2

Not applicable

3.16.3 Analysis

TIPs sets were run at test conditions 1, 2E, 3E, and 4E to provide the process computer with proper base LPRM data, and to analyze the core power symmetry. Table STI 18-1 shows an axial (Z) distribution for each of eight radial (R) rings. The core bundle power maps were inspected, and no

3.0 Results (Continued)

3.16 STI-18, Core Power Distribution (Continued)

3.16.3 Analysis (Continued)

analyzing 20 TIP traces in the common TIP channel, and the geometric uncertainty found from the analysis of TIP traces from symmetric TIP locations in accordance with the methods outlined in section 7.0 of the startup test instruction. The program "TIPTWO" was written to handle the calculations.

The results of the test are outline in table STI 18-2. The total noise uncertainty (σtotal) was below the allowable 7.8% at both test conditions, easily satisfying the test criteria.

Table STI 18-2

Uncertainty \ Test Condition	2E	3E	Limit
σ (total)	2.61%	3.99%	≤ 7.80%
σ (random)	1.26%	.595%	N.A.
σ (geometric)	2.28%	2.76%	N.A.

3.0 Results (Continued)

3.16 STI-18, Core Power Distribution (Continued)

3.16.3 Analysis (Continued)

anomalies were found. Figure STI 18-1 shows the radial power distribution (bundle powers in MWt) for one quadrant of the core.

Table STI 18-1
95% R - Z Power Distribution

	NRG. LVL.	1	2	3	4	5	6	7	8	AVG.
Core Top	12	0.215	0.282	0.267	0.252	0.252	0.263	0.232	0.156	0.228
	11	0.384	0.543	0.513	0.474	0.409	0.515	0.467	0.307	0.434
	10	0.546	0.778	0.731	0.666	0.571	0.737	0.681	0.449	0.622
	9	0.675	0.982	0.920	0.836	0.719	0.937	0.869	0.571	0.789
	8	0.834	1.224	1.147	1.041	0.901	1.181	1.104	0.726	0.993
	7	0.969	1.419	1.332	1.212	1.052	1.382	1.296	0.859	1.164
	6	1.106	1.643	1.542	1.400	1.217	1.599	1.519	1.007	1.354
	5	1.139	1.681	1.581	1.417	1.238	1.636	1.564	1.027	1.384
	4	1.132	1.653	1.572	1.401	1.222	1.612	1.545	1.002	1.363
	3	1.182	1.737	1.540	1.513	1.328	1.669	1.516	1.006	1.397
	2	1.148	1.736	1.427	1.565	1.388	1.654	1.404	0.940	1.361
Core Bottom	1	0.665	1.190	0.982	1.084	0.949	1.142	0.943	0.568	0.912
	AVG.	0.833	1.239	1.129	1.072	0.937	1.194	1.095	0.718	1.000

At test conditions 2E and 3E, additional TIP traces were run to verify that the TIP signal uncertainty was below the allowable criteria. The random noise (σ_{rn}) was found by

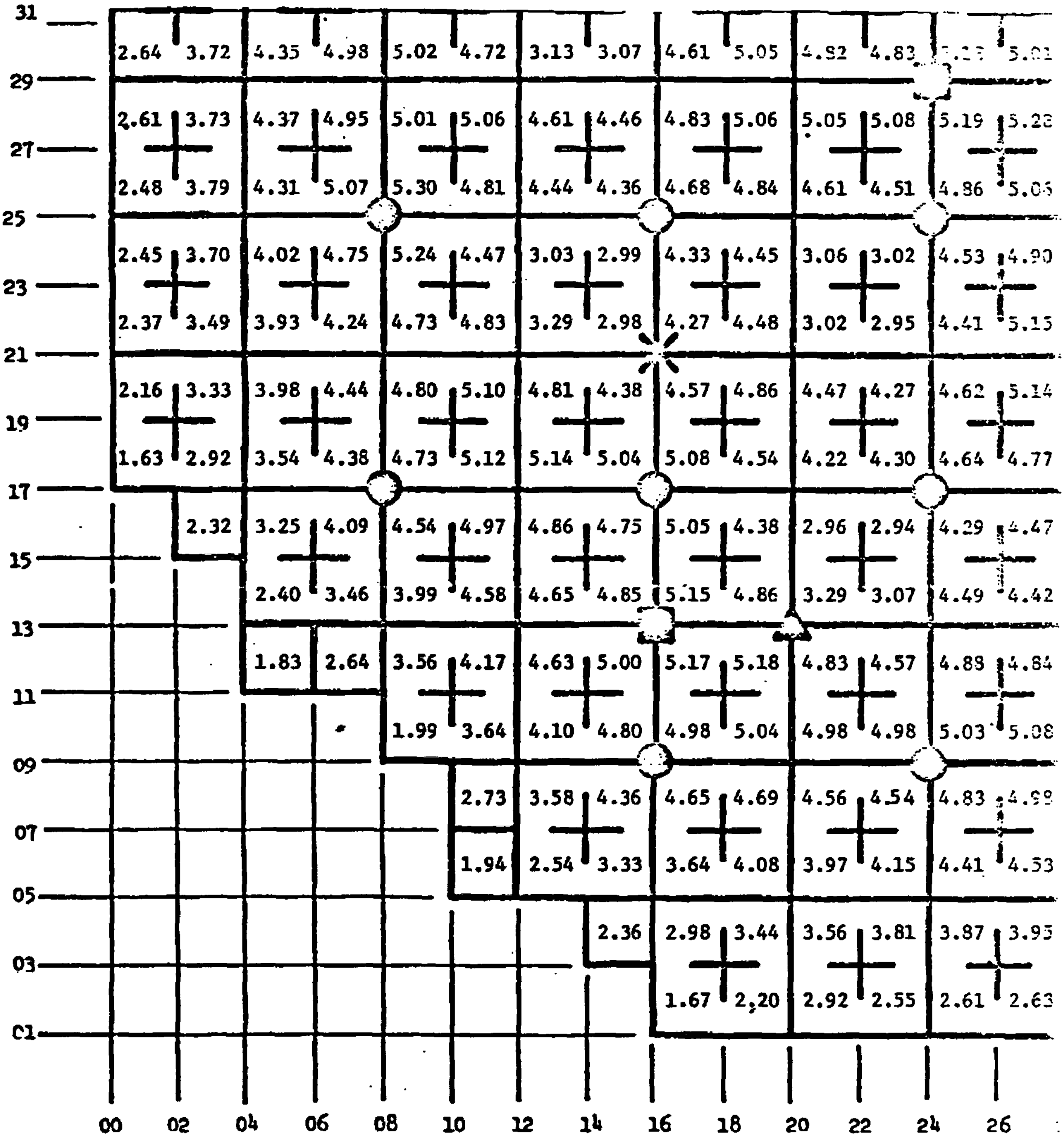


Figure STI 18-1

Bundle Power (MWt) Map at 95% Power

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results

3.17 STI-19, Core Performance

3.17.1 Purpose

The purposes of STI-19 are:

1. To evaluate the core thermal power.
2. To evaluate the following core performance parameters:

   Maximum Linear Heat Generation Rate (MLHGR)
   Minimum Critical Power Ratio (MCPR)
   Maximum Average Planar Linear Heat Generation Rate (MAPLHGR).

3.17.2 Criteria

Level 1

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

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.)

The minimum critical power ratio (MCPR) shall not exceed the limits specified by the technical specifications.

The maximum average planar linear heat generation rate (MAPLHGR) shall not exceed the limits of the technical specifications.

Level 2

Not applicable.

3.17.3 Analysis

STI-19 testing was performed at test conditions 1, 2A, 2D, 3E, 3C, 3D, 3E, 4A, 4C, 4D, and 4E as defined on the power flow map as shown in section 2.3.

The core performance parameters; linear heat generation rate (LHGR), core thermal power (CTP), minimum

3.0 Results (Continued)

3.17 STI-19, Core Performance (Continued)

3.17.3 Analysis (Continued)

critical power ratio (MCPR), and maximum average planar linear heat generation rate (MAPLHGR), were monitored at each test plateau of the startup test program. Table STI 19-1 contains a summary of these core parameters compared to the criteria limit.

All calculations were performed using the plant process computer. Core thermal power calculation of the process computer was verified using an offline computer program (CORPWR), and a detailed manual heat balance. Core performance parameters (LHGR, MCPR, MAPLHGR) calculated by the process computer were verified by the offline program BUCLE. All calculations agreed within the required 2%.

All test criteria have been satisfied.

Table STI 19-1
Core Performance Parameters

Test Condition	Core Power (MWt)	LHGR Value	LHGR Limit	MCPR Value	MCPR Limit	MAPLHGR Value	MAPLHGR Limit
1	768	3.82	<13.3	3.493	>1.514	3.20	<11.1
2A	783	4.275	<13.36	3.133	>1.572	3.53	<11.15
2D	1544	5.963	<13.26	2.486	>1.328	4.94	<11.11
2E	1689	6.19	<13.27	2.428	>1.27	5.20	<11.13
3C	1536	6.981	<13.36	2.178	>1.445	5.80	<11.14
3D	2136	8.92	<13.35	1.805	>1.315	7.49	<11.15
3E	2502	9.56	<13.275	1.659	>1.270	8.03	<11.15
4A	1329	5.427	<13.36	1.9605	>1.566	4.50	<11.20
4C	1902	7.625	<13.24	1.6789	>1.454	6.40	<11.19
4D	2309	10.35	<13.35	1.665	>1.311	8.75	<11.21
4E	3173	12.26	<13.35	1.4259	>1.270	10.36	<11.22

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results

3.18 STI-20, Steam Production

3.18.1 Purpose

The purpose of STI-20 is to demonstrate that the Nuclear Steam Supply System (NSSS) is providing sufficient steam to satisfy all appropriate warranties.

3.18.2 Criteria

Level 1

The NSSS parameters as determined by using normal operating procedures shall be within the appropriated license restrictions.

The appropriate warranty requirements, as outlined here, shall be satisfied.

The nuclear steam supply system shall be capable of supplying steam, of not less than 99.7% quality at a pressure of 985 psia at the second isolation valve. The system shall supply a maximum continuous steam flow output of 13,422,000 pounds per hour contingent upon the feedwater flow being 13, 372,000 pounds per hours at 378° F., and CRD flow being 50,000 pounds per hour at 80° F.

Level 2

Not applicable.

3.18.3 Analysis

Warranted plant conditions were attained on December 26, 1976, and the start of the warranty demonstration was officially declared at 2230 hours. The warranty demonstration was officially declared completed on January 8, 1977, at 1400 hours after 303.5 hours of operation. The 300-hour warranty run was interrupted twice for routine weekly control valve surveillance testing for a total of 3.5 hours. This time was not included in the 300-hour accumulation.

Reactor power was raised as close as possible to its rated value of 3293 MWt, such that during the warranty demonstration the average reactor power was 99.51%. Hence, for the two 2-hour runs it was necessary to extrapolate the plant conditions to the conditions of the contract. During

3.0 Results (Continued)

3.18 STI-20, Steam Production (Continued)

3.18.3 Analysis (Continued)

the 4-hour precision test runs the average main steam flow, adjusted to contract conditions, was 13.4155 x 10^6 lb/hr. Uncertainty calculations determined that the uncertainty in measured feedwater flow (parameter which mainly affects steam flow) was $\pm$ 0.02745 x 10^6 lb/hr. This made the uncertainty in steam flow calculations to be 13.4155 $\pm$ .02745 x 10^6 lb/hr and the contract specification of 13.422 x 10^6 lb/hr was satisfied.

All core performance parameters were within limits throughout the 300 hours. The following table is a summary of the two hour precision test runs and the average of the process computer data accumulated for the 300-hour duration.

Table STI 20-1

Parameter	Rated	Run 1	Run 2	300 hr. Ave.
Main Steam Flow	13.422 Mlb/hr	13.234	13.266	13.286
Feedwater Flow	13.372 Mlb/hr	13.195	13.228	13.254
CRD Flow	.050 Mlb/hr	.039	.038	.036
Recirc Pump PWR	10.52 MW	8.803	8.24	10.04
Rx Water Cleanup Loss	4.3 MW	2.061	0.0	2.53
Fixed Loss	0.6 MW	1.0	1.0	1.0
Reactor Thermal PWR	3293 MWt	3271	3281	3277
Feedwater Temperature	378° F	372.5	371.4	373.15
Reactor Dome Pressure	1020 PSIA	**1019	**1019	*1032
Steam Quality @ 2nd MSIV	99.7% DRY	99.84	99.86	N/A
Steam Pressure @ 2nd MSIV	985 PSIA	995	995.8	N/A
Steam Flow @ Contract Conditions	13.422 Mlb/hr	13.411	13.420	N/A

*Station Instrument
**Test Dead-Weight Gauge

3.0 Results

3.19 STI-21, Flux Response to Rods

3.19.1 Purpose

The purpose of STI-21 is to demonstrate the stability of the core local power-reactivity feedback mechanism with regard to small perturbations in reactivity caused by rod movement.

3.19.2 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.

3.19.3 Analysis

STI-21 testing was conducted at test conditions 1, 2E, 3E, 4A, and 4E as defined on the power flow map in section 2.3.

At each test condition the stability of the core power-reactivity feedback mechanism was tested by checking the local and macroscopic effects of control rod movement. The selected rod was moved near a location of limiting core thermal conditions. A nearby LPRM was used to monitor local power changes. Overall plant and core conditions were monitored by STAR TREC. Only local power as monitored by the LPRM and local heat flux responded to the control rod movement. The LPRM reading and local heat flux moved promptly to a new reading following the control rod movement and exhibited negligible oscillatory characteristics. Table STI 21-1 summarizes the results. All test criteria were met.

3.0 Results (Continued)

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

3.19.3 Analysis (Continued)

Table STI 21-1

Response To Control Rod Movement Summary

Test Condition	Rod Moved	Rod Movement	LPRM Monitored	Peak LPRM Change	Peak Heat Flux Change	Highest Decay Ratio
1	50-35	48 → 40	48 - 33A	6.4%	6.4%	<.25
		40 → 48		6.4%	6.4%	<.25
2E	42-43	48 → 44	40 - 41A	9%	4%	<.25
		44 → 48		7%	4%	<.25
3E	50-19	48 → 44	48 - 17A	5%	4%	<.25
		44 → 48		5%	4%	<.25
4A	26-15	48 → 40	24 - 17A	9.6%	7.2%	<.25
		40 → 48		10.0%	7.2%	<.25
4E	50-15	48 → 40	48 - 17A	19.2%	13.9%	<.25
		40 → 48		16.8%	13.9%	<.25

3.0 Results

3.20 STI-22, Pressure Regulator

3.20.1 Purpose

The purposes of STI-22 are:

1. To determine 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 set points 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.

3.20.2 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

In all tests except the simulate 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.

Pressure control system deadband, delay, etc., shall be small enough that steady-state limit cycles, if any, shall produce turbine steam flow variations no larger than ± 0.5% of rated steam flow.

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 limits of the above criteria.

3.0 Results (Continued)

3.20 STI-22, Pressure Regulator (Continued)

3.20.2 Criteria (Continued)

Level 2 (Continued)

During the simulated failure of the controlling pressure regulator, if the setpoint of the backup pressure regulator is optimumly set, the backup regulator shall control the transient such that the peak neutron flux and/or peak vessel pressure remain below the scram settings by 7.5% and 10 psi respectively. Maintain a plot of the peak variable values versus power.

Following a ± 10 psi (0.7 kg/cm^2) pressure setpoint change, the time between the setpoint change and the occurrence of the pressure peak shall be 10 seconds or less.

3.20.3 Analysis

STI-22 testing was conducted at test conditions 1, 2E, 3E, 4A, 4C, 4D, and 4E as defined on the power flow map in section 2.3 The Electrohydraulic Control (EHC) system controller setting was adjusted to provide for stability of the pressure control loop. The backup capability of each pressure regulator was demonstrated via simulated failure of the controlling regulator. Final adjustments of the EHC system was completed at test condition 3E with implementation of the following settings:

The EHC system pressure regulator settings were:

"A" Lag Pot (R5)	2.4 turns (τ = 5 seconds)
"A" Lead Pot (R6)	4.6 turns (τ = 2 seconds)
"B" Lag Pot (R3)	2.4 turns (τ = 5 seconds)
"B" Lead Pot (R4)	4.0 turns (τ = 2 seconds)

The EHC system steam line resonance compensator settings were:

"A" Notch Center	3.63 turns
"A" Notch Depth	2.00 turns
"A" Notch Width	1.67 turns
"A" Small Lag	1.47 turns
"B" Notch Center	3.63 turns
"B" Notch Depth	2.00 turns
"B" Notch Width	1.67 turns
"B" Small Lag	1.47 turns

3.0 Results (Continued)

3.20 STI-22, Pressure Regulator (Continued)

3.20.3 Analysis (Continued)

EHC bias adjustments:

Regulator Potentiometer	3.33 turns (30 psi)
Pressure setpoint bias (3 psi separation)	4.24 turns
Speed regulator	7.39 turns
Intercept valve bias	10.00 turns
Bypass valve opening bias	1.20 turns
Recirculation flow signal limit	7.72 turns

3 psi separation between the regulators was established for normal operation

Table STI 22-1 summarizes the results of the pressure regulator setpoint changes. A smooth pressure transition between control valves and bypass valves was demonstrated during the setpoint changes.

The simulated failure test of the pressure regulator was conducted with a 2 to 4 psi bias between regulators in order to minimize the neutron flux margin to scram to $\leq$ 7.5%. A 5 psi differential had been generally recommended in the past before the plugged bottom core plate caused greater sensed neutron flux peaking. In order to minimize the neutron flux peaking during the backup regulator event, a setpoint differential of 1 to 4 psi has been recommended by General Electric and accepted by TVA, Division of Engineering Design. The current operating setpoint differential is 3. psi. With this setpoint pressure regulator testing satisfied all level 1 and 2 criteria.

Table STI 22-1

Pressure Regulator Response Summary
(Recirculation in Master Manual Mode)

Test Condition	1				2E				3E				4E			
Step Input	-10%	-10%	+10%	-10%	-10%	+10%	-10%	+10%	-10%	+10%	-10%	+10%	-10%	+10%	-10%	+10%
Regulator (A/B)	A	A	B	B	A	A	B	B	A	A	B	B	A	A	B	B
Valves (CV/BPV)	C.V.	BPV Incpnt.	C.V.	BPV 50%	C.V.	BPV 50%	C.V.	BPV Incpnt.	C.V.	BPV 50%	C.V.	BPV 50%	C.V.	BPV 50%	BPV Incpnt.	BPV 50%
Initial Dome Press.	950	957	945	957	951	941	947	938	954	944	952	940	990	980	996	984
Final Dome Press.	941	948	955	947	940	950	938	946	943	957	943	951	980	996	983	998
Time to First Press. Peak (1)	2.0	2.7	2.0	2.8	5.0	3.5	3.0	2.8	4.0	6.0	2.5	7.0	4.0	7.0	5.8	8.0
Highest Decay Ratio Parameter (2)	<.25 APRM	<.25 APRM	<.25 APRM	<.25 APRM	<.25 APRM	<.25 APRM	<.25 APRM	<.25 APRM	<.25 APRM	<.25 APRM	<.25 APRM	<.25 APRM	<.25 APRM	<.25 APRM	<.25 APRM	<.25 APRM

(1) Level 2 criteria limit is 10 seconds.
(2) Level 2 criteria is 0.25.

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results

3.21 STI-23, Feedwater System

3.21.1 Purpose

The purposes of STI-23 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 low water level scram following the trip of one feedwater pump.

3.21.2 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

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.

Following a 3-inch (7.5 cm) level set-point step adjustment in three-element control, the time from set-point 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.)

The automatic recirc-flow runback feature shall prevent a scram from low water level following a trip of one of the operating feedwater pumps. The water level margin to scram should be greater than 3 inches for a pump trip from the 100% power condition.

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

3.0 Results (Continued)

3.21 STI-23, Feedwater System (Continued)

3.21.3 Analysis

STI-23 testing was conducted at test conditions 1, 2E, 3E, 4C, 4D, and 4E as defined on the power flow map in section 2.3.

Step changes of $\pm$ 3 inches were made at each test condition listed above with the feedwater system in both the single and three element mode of control. Response of the feedwater system during the transients is summarized in table STI 23-1.

At test condition 1 the time from initiation of the setpoint change to reaching the level peak was greater than the criterion of 35 seconds. No attempt was made to optimize system response at that power level, because only one feed pump was in operation. During all subsequent testing with three feed pumps in operation the level peak was reached within the required 35 seconds, thus satisfying the criterion.

During level setpoint change testing at all levels, the decay ratio was less than 0.25 for all process variables exhibiting response to the changes. Therefore, all criteria applicable to level setpoint change testing were met.

During testing at test condition 2E, all three feed pumps were in operation. Final system optimization was, therefore, performed at this level. The final settings on the level controller were: Proportional Band = 200% Reset = 1 repeat/minute. The mismatch gain was set for a 36-inch corrected level for 100% mismatch of rated feedwater flow and steam flow. The lead-lag unit was set for a lag time constant of 5 seconds, and a lead time constant of 1 second.

From test condition 4E, with all three feedwater pumps operating and the feedwater controller in the 3-element mode, one feedwater pump was tripped to test the automatic recirculation pump run back feature. The time from pump trip until the minimum reactor water level was reached was 27 seconds. The minimum reactor water level reached was 22.5 inches, which is well above the scram setpoint of 11 inches. The feedwater and recirculation systems responded satisfactorily to the feedwater pump trip, and all criteria were satisfied.

3.0 Results (Continued)

3.21 STI-23, Feedwater System

3.21.3 Analysis (Continued)

Table STI 23-1
LEVEL SETPOINT CHANGES

Test Condition	Controlling Level	Control Mode	Setpoint Change (in.)	Time To Peak Level (sec.)	Max. Decay Ratio
1		1 element	+ 3"	66	<.25
1		1 element	- 3"	76	<.25
1		3 element	+ 3"	64	<.25
1		3 element	- 3"	71.25	<.25
2E		3 element	+ 3"	30.	<.25
2E		3 element	- 3"	30.	<.25
3E	A	3 element	+ 3"	30.5	<.25
3E	A	3 element	- 3"	31	<.25
3E	B	3 element	+ 3"	31	<.25
3E	B	3 element	- 3"	28.5	<.25
3E	A	1 element	+ 3"	25.	<.25
3E	A	1 element	- 3"	28	<.25
3E	B	1 element	+ 3"	26.5	<.25
3E	B	1 element	- 3"	25.5	<.25
4C	B	3 element	- 3"	34.5	<.25
4C	B	3 element	+ 3"	34.	<.25
4C	A	3 element	- 3"	34.	<.25
4C	A	3 element	+ 3"	33	<.25
4C	A	1 element	- 3"	44.	<.25
4C	A	1 element	+ 3"	35	<.25
4C	B	1 element	- 3"	32	<.25
4C	B	1 element	+ 3"	42	<.25
4D	B	3 element	- 3"	32	<.25
4D	B	3 element	+ 3"	32	<.25
4D	A	3 element	- 3"	31	<.25
4D	A	3 element	+ 3"	34.5	<.25
4D	A	1 element	- 3"	21	<.25
4D	A	1 element	+ 3"	30	<.25
4D	B	1 element	- 3"	30	<.25
4D	B	1 element	+ 3"	31	<.25

3.0 Results (Continued)

3.21 STI-23, Feedwater System (Continued)

3.21.3 Analysis (Continued)

Table STI 23-1
LEVEL SETPOINT CHANGES (Continued)

Test Condition	Controlling Level	Control Mode	Setpoint Change (in.)	Time To Peak Level (sec.)	Max. Decay Ratio
4E	A	3 element	- 3"	30	<.25
4E	A	3 element	+ 3"	32	<.25
4E	B	3 element	- 3"	31	<.25
4E	B	3 element	+ 3"	32	<.25
4E	B	1 element	- 3"	18	<.25
4E	B	1 element	+ 3"	21	<.25
4E	A	1 element	- 3"	21	<.25
4E	A	1 element	+ 3"	31	<.25

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results

3.22 STI-24, Bypass Valves

3.22.1 Purpose

The purposes of STI-24 are:

1. To demonstrate the ability of the pressure regulator to minimize the reactor pressure disturbance 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.

3.22.2 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

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.

To avoid approaching steam line low pressure isolation, the maximum pressure decrease at the turbine inlet during valve opening shall not exceed 50 psi (3.5 kg/cm^2).

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

The regulator shall limit the pressure disturbance during valve reclosure so that a margin of at least 7.5% shall be maintained below flux scram.

3.22.3 Analysis

Bypass valve testing was conducted at test conditions 1, 2A, 2E, 3E, 4A, 4C, 4D, and 4E as defined in the power flow map in section 2.3. The successfully completed bypass valve test program demonstrated that the EHC system had adequate capability to respond to abrupt changes in steam flow.

3.0 Results (Continued)

3.22 STI-24, Bypass Valves (Continued)

3.22.3 Analysis (Continued)

For test purposes, the bypass valve opening time was adjusted so that the valve would open in as short a time as possible. Since it is not possible to have both fast opening and closing times, the valves were adjusted for a fast opening time of approximately 3.0 seconds and a slower closing time of approximately 16 seconds.

Table STI 24-1 contains a summary of the bypass valve test transient data from all test conditions. Bypass valve testing at all test conditions listed in the table satisfied all test acceptance criteria.

Throughout the startup test program, data were 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 24-1. Each test netted results which showed this margin to be approximately 18.3% of rated power, which satisfies the level 2 criteria.

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

3.22 STI-24, Bypass Valves (Continued)

3.22.3 Analysis (Continued)

Table STI 24-1
Bypass Valve Transient Data Summary

Test Condition / Parameter	Limit	1	2A	2E	3E	4A	4C	4D	4E
Thermal Power	NA	1120 MWt 34.0%	823 MWt 25.0%	1811 MWt 55.0%	2637 MWt 80%	1322 MWt 40.2%	1873 MWt 56.9%	2387 MWt 72.5%	3239 MWt 98.3%
Total Core Flow	NA	51.0 Mlb/hr 49.8%	26.7 Mlb/hr 26.0%	106.6 Mlb/hr 104.0%	104.2 Mlb/hr 101.6%	29.1 Mlb/hr 28.4%	47.6 Mlb/hr 46.4%	74.4 Mlb/hr 72.6%	99.0 Mlb/hr 96.6%
Date	NA	10/24/76	10/28/76	10/11/76	11/3/76	11/26/76	11/27/76	11/28/76	11/23/76
Maximum Time to S.S. Pressure (sec)	<25	16.0	11.0	16.0	19.0	11.2	16.0	0.0	18.0
Margin to Flux Scram (%)	>7.5	15.8	10.8	31.1	15.0	13.0	20.26	13.08	18.29
Scram Setpoint (%)	NA	51.8	35.3	86.1	95.0	54.0	80.26	88.28	115.99
Decay Ratio	<.25	0.0	≤.25	0.0	0.0	0.0	0.0	0.0	0.0
Initial Dome Pressure (psig)	NA	988.0	946.0	979.0	970.0	964.3	960.0	975.0	998.0
Change in Dome Pressure (psig)	<50	2	2	2	2	1	2	0	1
Opening Time of Bypass Valve (sec)	≈3.0	≈3.0	≈3.0	≈3.0	≈3.0	3.52	3.70	3.76	3.28

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

3.22 STI-24, Bypass Valves (Continued)

3.22.3 Analysis (Continued)

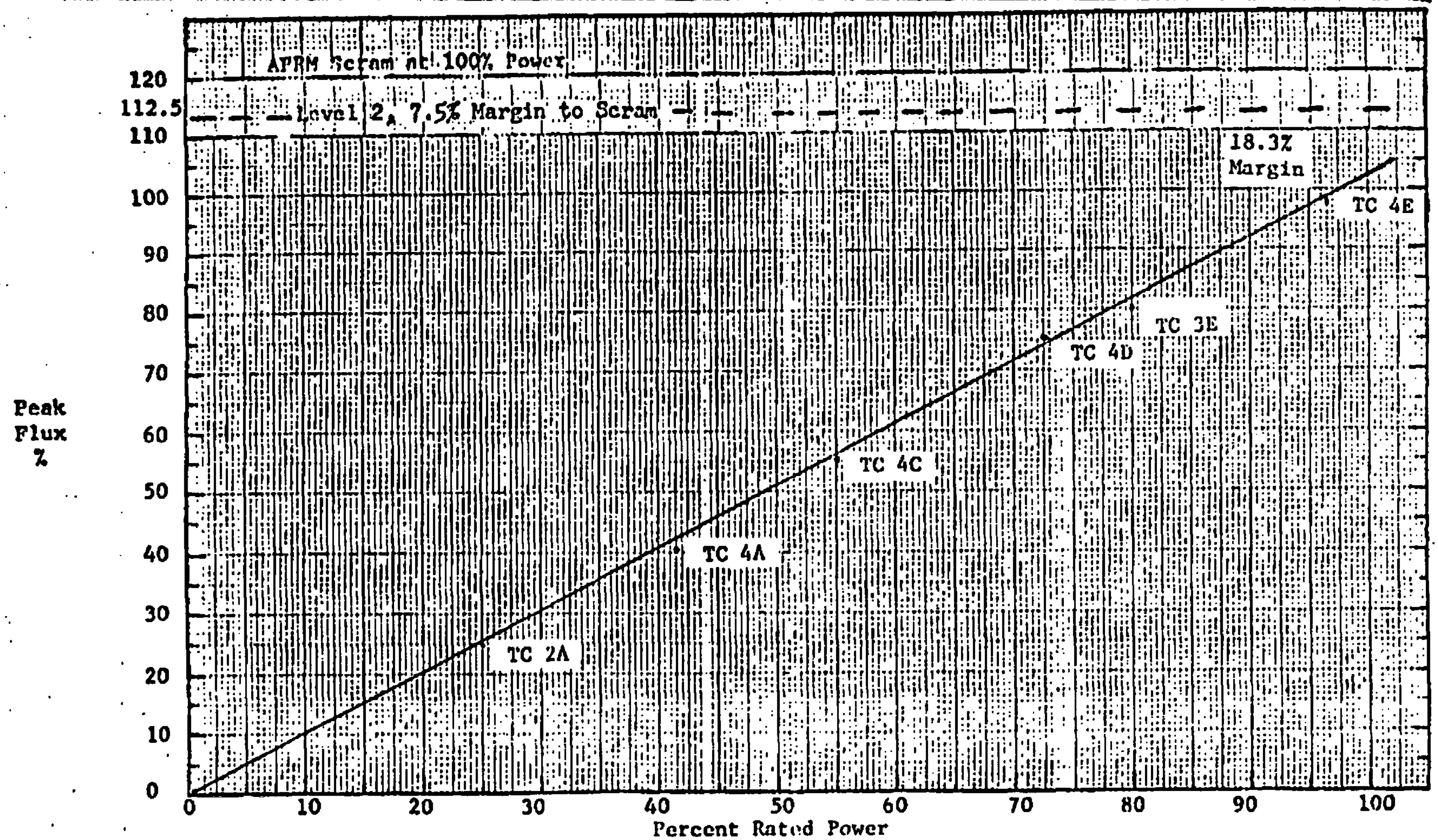


Figure STI 24-1
Bypass Valves

Description: Flux Margin to Scram at 100% Power

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results

3.23 STI-25, Main Steam Line Isolation Valves

3.23.1 Purpose

The purposes of STI-25 are:

1. To functionally check the Main Steam Line Isolation Valves (MSIVs) for proper operation at selected power levels.

2. To determine reactor transient behavior during and following simultaneous full closure of all MSIVs, and following full closure of one valve.

3. To determine isolation valve closure time.

4. To determine the maximum power at which a single valve may be closed without a reactor scram.

3.23.2 Criteria

Level 1

MSIV closure time must be greater than 3 and less than 5 seconds.

The initial transient rise in vessel dome pressure occurring within 20 seconds of the main steam isolation valve trip initiation shall not be greater than 150 psi, and the transient rise in simulated heat flux shall not exceed 10%.

Level 2

The initial transient peak in vessel dome pressure occurring within 20 seconds following initiation of the MSIV closure and the transient peak in simulated surface heat flux shall not be more limiting than the predicted transients in the Transient Analysis Design Report (100 psi and no heat flux increase.)

During full closure of individual valves, pressure must be 20 psi (1.4 kg/cm^2) below scram, neutron flux must be 10% below scram, and steam flow in individual lines must be 10% below the isolation trip setting.

3.23.3 Analysis

STI-25 testing was conducted at heatup, test conditions 2E, 4E, and 4E levels as defined on the power flow map in section 2.3.

3.0 Results (Continued)

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

3.23.3 Analysis (Continued)

Main Steam Isolation Valves (MSIV) were individually closed at heatup, test conditions 2E and 4D. Closing times are summarized in table STI 25-1. Data taken at each plateau was analyzed to ensure that individual closures could be performed at the next plateau of higher power. Closure times at all levels of testing were between the require 3-5 seconds. Slow closure to the 90% open position for each MSIV was satisfactorily performed at heatup and test conditions 2E and 4D. During all MSIV closures transient behavior of significant reactor and plant parameters were monitored by STARTREC. For all parameters performance during the transient met level 1 and 2 criteria. Transient behavior is summarized in table STI 25-2.

On December 3, 1976, a simultaneous full closure of all MSIV's was initiated from 96.5% of rated core thermal power. Reactor transient behavior and MSIV closure times were recorded by STARTREC. Closure times were within the required 3-5 seconds. During the initial 20 seconds after the scram the peak dome pressure rise was 84 psi. No increase in simulated heat flux was measured. All level 1 and 2 criteria were satisfied.

3.0 Results (Continued)

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

3.23.3 Analysis (Continued)

Table STI 25-1
MSIV Closure Times

MSIV Number	Closure Time (sec.)* Heatup	T.C. 2E	T.C. 4D
FCV-1-14 (1A)	3.47	3.39	3.481
FCV-1-15 (2A)	3.09	2.99**	3.069
FCV-1-26 (1B)	3.30	3.70	3.296
FCV-1-27 (2B)	3.50	3.50	3.605
FCV-1-37 (1C)	3.50	3.60	3.605
FCV-1-38 (2C)	4.20	4.60	4.223
FCV-1-51 (1D)	3.40	3.30	3.193
FCV-1-52 (2D)	3.30	3.20	3.193

* Times are for 0 - 97% closure.
** Closure time for 0 - 100% was 3.08 sec.

Table STI 25-2
Transient Behavior During MSIV Closure

Parameter	Heatup	T.C. 2E	T.C. 4D
Dome Pressure (psig)			
Scram Setpoint	1055	1055	1055
Peak Value	No Change	990	1005.5
Margin to Scram		65	49.5
APRM Heat flux (%)			
Scram Setpoint	15%	70%	91.7%
Peak Value	No Change	48%	80.5%
Margin to Scram		22%	11.2%
Individual Steam Line Flow (Mlb/hr)			
Scram Setpoint		4.69	4.69
Peak Value	No Change	2.0	3.20
Margin to Scram		2.69	1.49

3.0 Results

3.24 STI-26, Relief Valves

3.24.1 Purpose

The purposes of this test are:

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

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

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

4. To verify that the discharge piping is not blocked.

3.24.2 Criteria

Level 1

There should be positive indication of steam discharge during the manual actuation of each valve.

The sum total of capacity measurements from the 11 relief valves shall be equal to or greater than 8.83×10^6 lb/hr $\pm$ 2% corrected for an inlet pressure of 1112 psig.

Level 2

Relief valve leakage shall be low enough 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. Each individual relief valve shall have a minimum capacity of 720,000 lb/hr corrected to an inlet pressure of 1112 psig.

The pressure regulator must satisfactorily control the reactor transient and close the control valves or bypass valves by an amount equivalent to the relief valve discharge. The transient recorder signatures for each valve must be analyzed for relative system response comparison.

3.24.3 Analysis

STI-26 testing was conducted at heatup, test conditions 1 and 3E. The bypass valve calibration phase of STI-26 was performed in test condition 1 testing. A least-squares fit was made to the data to relate the bypass valve capacity to the relief valve capacity. During TC 1 relief valve testing, the feedwater flow decreased by approximately

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

3.24 STI-26, Relief Valves (Continued)

3.24.3 Analysis (Continued)

3.9 Mlb/hr, reactor pressure dropped by 6 psig, steam flow decreased by approximately .75 Mlb/hr, and APRM A decreased by 3% when the valve was opened.

Table STI 26-1 represents a summary of all the pertinent data obtained during relief valve testing. All relief valves met steam discharge, capacity, and reseating criteria at all levels of testing. The pressure regulator satisfactorily controlled the pressure transient when the relief valves were opened.

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

3.24 STI-26, Relief Valves (Continued)

3.24.3 Analysis (Continued)

Table STI 26-1

Summary of Relief Valve Data

General Electric Relief Valve No.	TVA Relief Valve No.	Corrected Capacity Mlb/hr		Time For Temp. Return to within 10°F (sec.)	Relief valve Thermocouple Temp. at TC 3E	
		Test Condition 1	Test Condition 3E		Initial °F	Final °F
1-4	A	.8212	.8385	1.25	208	218
1-5	B	.8301	.8734	1.50	220	230
1-18	C	.8301	.8734	1.00	221	229
1-19	D	.8186	.8297	1.00	195	203
1-22	E	.8036	.8122	1.00	174	184
1-23	F	.7965	.8473	1.62	181	190
1-30*	G	.8770	.8821	1.00	222	220
1-31*	H	.8780	.8909	2.20	261	271
1-34	J	.8231	.8647	.75	208	217
1-41	K	.8372	.7598	1.53	225	235
1-42	L	.8328	.7949	1.00	269	276
Total Capacity Mlb/hr		9.1483	9.27			

*Crosby Relief Valves

Capacity Limit

Individual Capacity: .720 Mlb/hr

Total Capacity: 8.83 Mlb/hr

3.0 Results

3.25 STI-27, Turbine Trip and Generator Load Rejection

3.25.1 Purpose

The purpose of STI-27 is to demonstrate the response of the reactor and its control systems to protective trips in the turbine and generator.

3.25.2 Criteria

Level 1

The initial transient rise in vessel dome pressure occurring within 10 seconds of the turbine/generator trip initiation shall not be greater than 150 psi and the transient rise in simulated heat flux shall not exceed 10 percent.

The turbine stop valves must begin to close before the control valves for the turbine trip. The turbine control valves must begin to close before the stop valves during the generator load rejection.

Following fast closure of the turbine stop and control valves, a reactor scram shall occur if the turbine first stage pressure is greater than 154 psig.

Feedwater systems must prevent flooding of the steamline following the transients.

Level 2

The initial transient rise in vessel dome pressure occurring within 10 seconds of the turbine/generator trip initiation and the transient rise in simulated surface heat flux shall not be more limiting than the predicted transient presented in the Transient Analysis Design Report (100 psi and no heat flux increase.)

The pressure regulator must prevent a low pressure reactor isolation.

The wide range level sensing system and the feedwater controller must prevent a low level initiation of the HPCI and MSIV's as long as feedwater flow remains available.

The trip scram function for higher power levels must meet RPS specifications.

3.0 Results (Continued)

3.25 STI-27, Turbine Trip and Generator Load Rejection (Continued)

3.25.2 Criteria (Continued)

Level 2 (Continued)

The load rejection within bypass capacity must not cause a scram.

For the case of turbine trip at 75-percent power, the measured transient parameters will be compared with the predicted values. If any parameter is significantly different from the predicted values the test will be repeated at 100-percent power.

3.25.3 Analysis

STI-27 was performed at test conditions 1, 3E, and 4E as defined on the power flow map in section 2.3.

A generator load rejection within bypass valve capacity was performed by opening the main transformer breakers at 24.5% power. The control valves closed in approximately 0.5 seconds after the main generator breaker was opened. The bypass valves opened to 85% of total capacity, APRM A increased by approximately 1%, the control valves decreased from 14 to 0% open, and feedwater flow decreased by 0.1 Mlb/hr. The wide range level sensing system and the feedwater controller prevented a low level initiation of HPCI and MSIV's.

The turbine trip test was performed at 75.3% power. The reactor immediately scrammed, initiated by the 10% stop valve closure condition. The peak reactor dome pressure was 1044 psig after 4.0 seconds, well below the 1080 psig relief valve setpoint. A low-low water level reactor isolation occurred. As resolution to this problem, the following feedwater controller system changes will be made:

1. The low level isolation setpoint will be lowered.

2. Installation will be made of an automatic level setpoint setdown and a high level feedwater pump trip.

All reactor protection systems functioned as expected. The pressure rise was less than the predicted and the projected 100% power case. The following table summarizes the significant events during the test.

3.0 Results (Continued)

3.25 STI-27, Turbine Trip and Generator Load Rejection (Continued)

3.25.3 Analysis (Continued)

Table STI 27-1

Time (sec.)	Event
0.0	APRM A - 76.5%; Dome pressure - 965 psig; Feedwater flow - 9.8 Mlb/hr; water level - 36 inches; Main turbine trip.
0.2	Stop valves closed.
0.3	Control valves closed; reactor scram.
1.7	APRM A - 17%.
4.0	Feedwater flow - 8.4 Mlb/hr; water level - 0 inch.
4.6	Reactor isolation on low water level; dome press - 1040 psig.
9.0	Feedwater flow - 19.4 Mlb/hr.
12.0	Simulated thermal power - 0%; feedwater flow - 8.0 Mlb/hr.

The generator load rejection test was performed at 98.7% power by opening the main transformer breakers. Due to the failure of the time delay relay in the power/load unbalance circuit, a control valve fast closure did not occur. This resulted in a turbine stop valve trip due to turbine overspeed. The resulting transient on the turbine was more severe than a control valve fast closure transient because the turbine overspeed reached ∿ 113% compared to approximately 105% for a control valve trip. The transient on the reactor is comparable to that resulting from a control valve fast closure. No increase in LPRM's, APRM's, or simulated heat flux were noted after the trip. As noted in the turbine trip test, a low water level isolation occurred. The first pressure peak occurred at 4.43 seconds with a maximum reactor dome pressure of 1085 psig, and the second at 25.63 seconds at 1101 psig, due to

3.0 Results (Continued)

3.25 STI-27, Turbine Trip and Generator Load Rejection (Continued)

3.25.3 Analysis (Continued)

the low reactor water level isolation. Relief valves D and F opened in both cases to reduce the reactor pressure to less than 1075 psig. The feedwater controller system changes discussed previously should enhance the post-scram recoverability and prevent low water level isolations.

The time delay relay that prevented a control valve fast closure was repaired and a special test was performed to demonstrate its operability. The following table summarizes the significant events of the test.

Tabel STI 27-2

Time (sec.)	Event
0.0	APRM A - 98.3%; Dome pressure - 1000 psig; water level - 33 inches; Main transformer breakers opened.
0.020	Initiates control valve fast closure.
0.120	C.V. begin to close as turbine overspeeds.
1.6	Water level - 38.1 inch.
1.63	Turbine stop valve trip; reactor scram.
2.00	Water level - -63 inches; APRM A - 65%.
4.0	APRM A - 0%.
4.43	Dome pressure - 1085 psig; D and F relief valves open.
6.4	Water level - 32 inches; Low water level isolation.
6.63	Dome pressure - 1077 psig; Water level - 20 inches.
25.63	Dome pressure - 1101 psig; D and F relief valves open; Water level 31.1 inches.
29.63	Dome pressure - 1070 psig.

3.0 Results

3.26 STI-30, Recirculation System

3.26.1 Purpose

The purposes of STI-30 are:

1. To verify that the feedwater control system can satisfactorily control water level without a resulting turbine trip/scram, and to obtain actual pump speed/flow coastdown data.

2. To verify recirculation pump startup under pressurized reactor conditions.

3. To obtain recirculation system performance data.

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

5. To provide the opportunity to obtain flow induced vibration data.

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

3.26.2 Criteria

Level 1

Not applicable

Level 2

The power and flow coastdowns are expected to agree with pre-calculated power and flow coastdown rates. The plant shall not scram as a result of a high level turbine trip.

3.26.3 Analysis

STI-30 testing was performed at test conditions 2A, 2E, 3E, 4A, and 4E as defined on the power flow map in section 2.3.

Recirculation system performance data was taken on the 50% flow control line at various combinations of pump speeds as specified by section 6.3 of STI-30, and

3.0 Results (Continued)

3.26 STI-30, Recirculation System (Continued)

3.26.3 Analysis (Continued)

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 from ~ 50% power by inserting control rods in the reverse order of rod sequence "A" until the feedwater flow limit that initiates a recirculation pump runback was reached. The recirculation pump runback circuitry was disconnected during the test to prevent an actual runback from occurring. Power was reduced to 22.3% (736 MWt) of rated, which corresponds to feedwater flow of 2.61 X 10^6 lb/hr. The recirculation pump runback setpoint is set at 2.7 X 10^6 lb/hr. No signs of cavitation were seen in the jet pumps or recirculation pumps at any power level during the test.

A single pump trip was performed at ~ 50% core thermal power and 100% flow by opening the generator field breaker on pump "A". Single pump trips and simultaneous 2 pump trips were performed at 50% and 100% core thermal power and 100% flow by tripping the drive motors. Transient traces were taken by STARTREC of significant plant and recirculation system parameters. Figures STI 30-1 through STI 30-7 compare plant parameters as recorded by STARTREC with predicted behavior for the first 10 seconds of analyzed trips.

Except for "A" recirculation pump drive flow signal, all parameters agreed closely or were conservatively compared to predicted behavior for analyzed transients. "A" pump drive flow did not decay off as expected. Analysis of loop jet pump flow and total core flow indicated that "A" pump was actually performing as predicted, and that "A" and "B" pumps reacted in substantially the same manner during the transients.

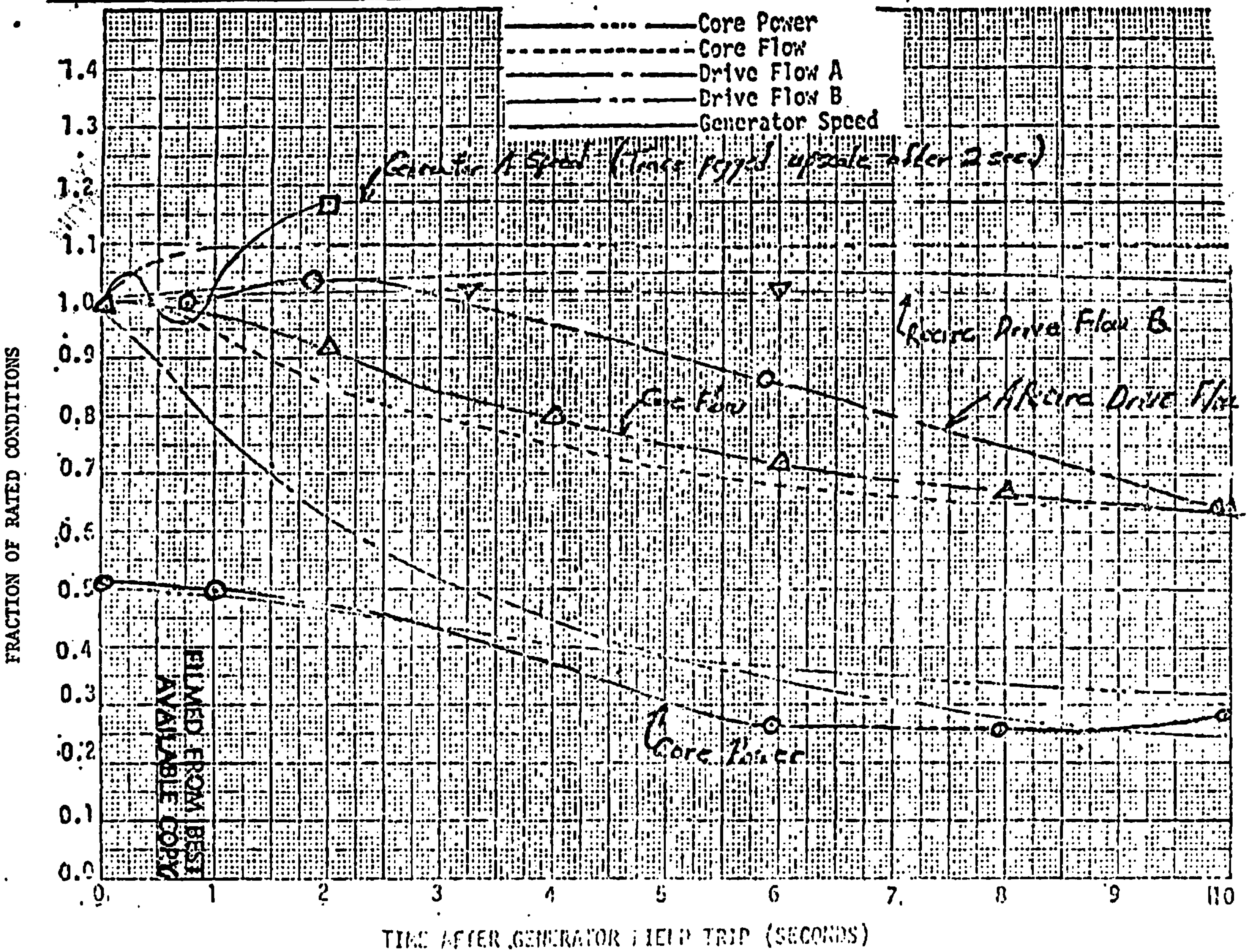
It was therefore felt that the difference in drive flow signals was in the flow measurement circuitry. Circuit repairs have been completed. All level 2 criteria have therefore been met.

3.0 Results (Continued)

3.26 STI-30, Recirculation System (Continued)

3.26.3 Analysis (Continued)

Following pump trips at 50% power testing, each recirculation pump was tested for its ability to restart under pressurized conditions. Significant system parameters were recorded by STARTREC during the restart. No difficulties were encountered and each pump performed as expected.



A Pump Gen. Field Breaker Trip

Figure STI 30-1 [illegible]

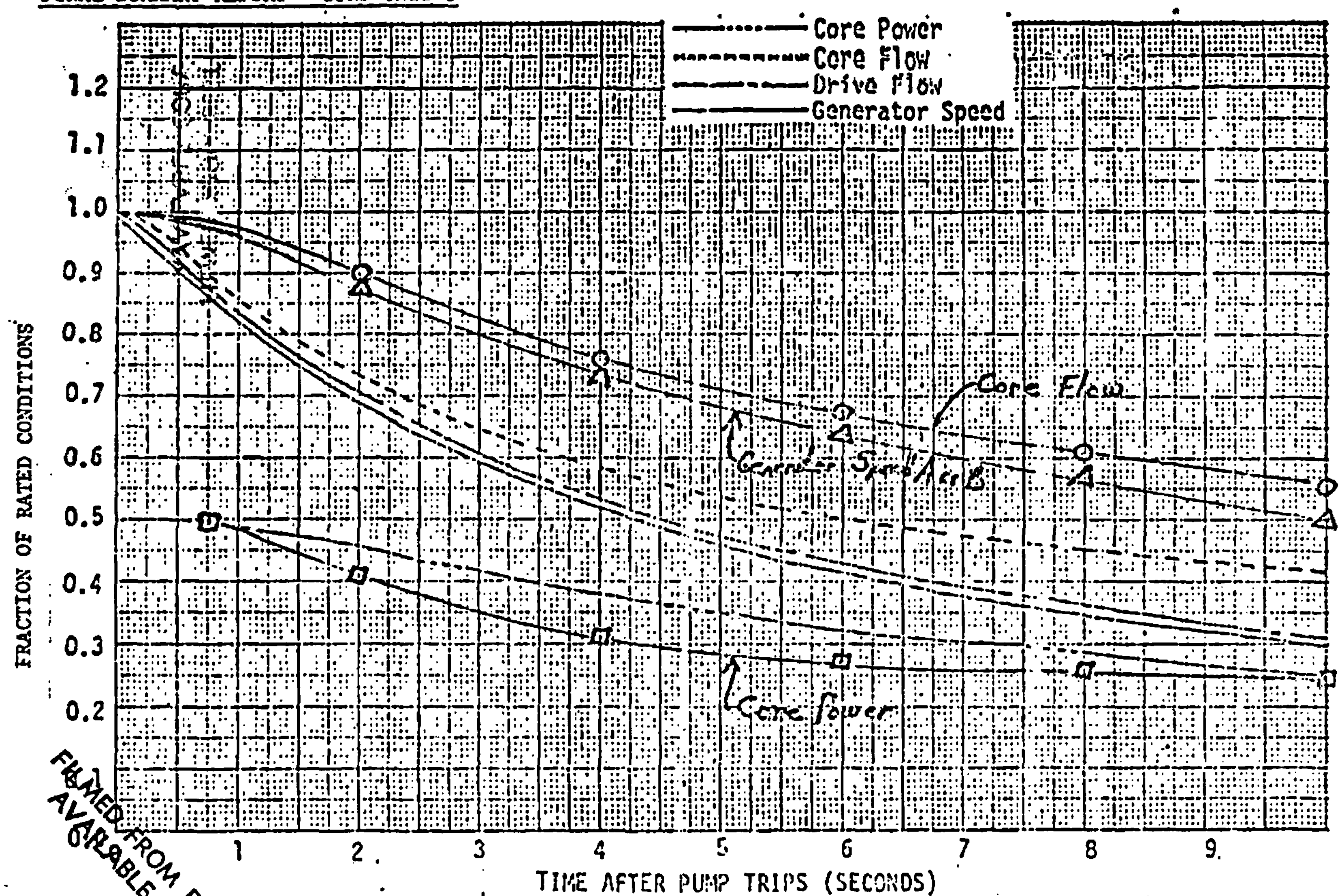


Figure STI 30-2

3293 MWT PLANT RECIRCULATION SYSTEM PERFORMANCE
FOLLOWING TRIP OF TWO DRIVE MOTORS - 50% POWER

Chart 1

Drive Motor Trip of Both Recirc Pumps

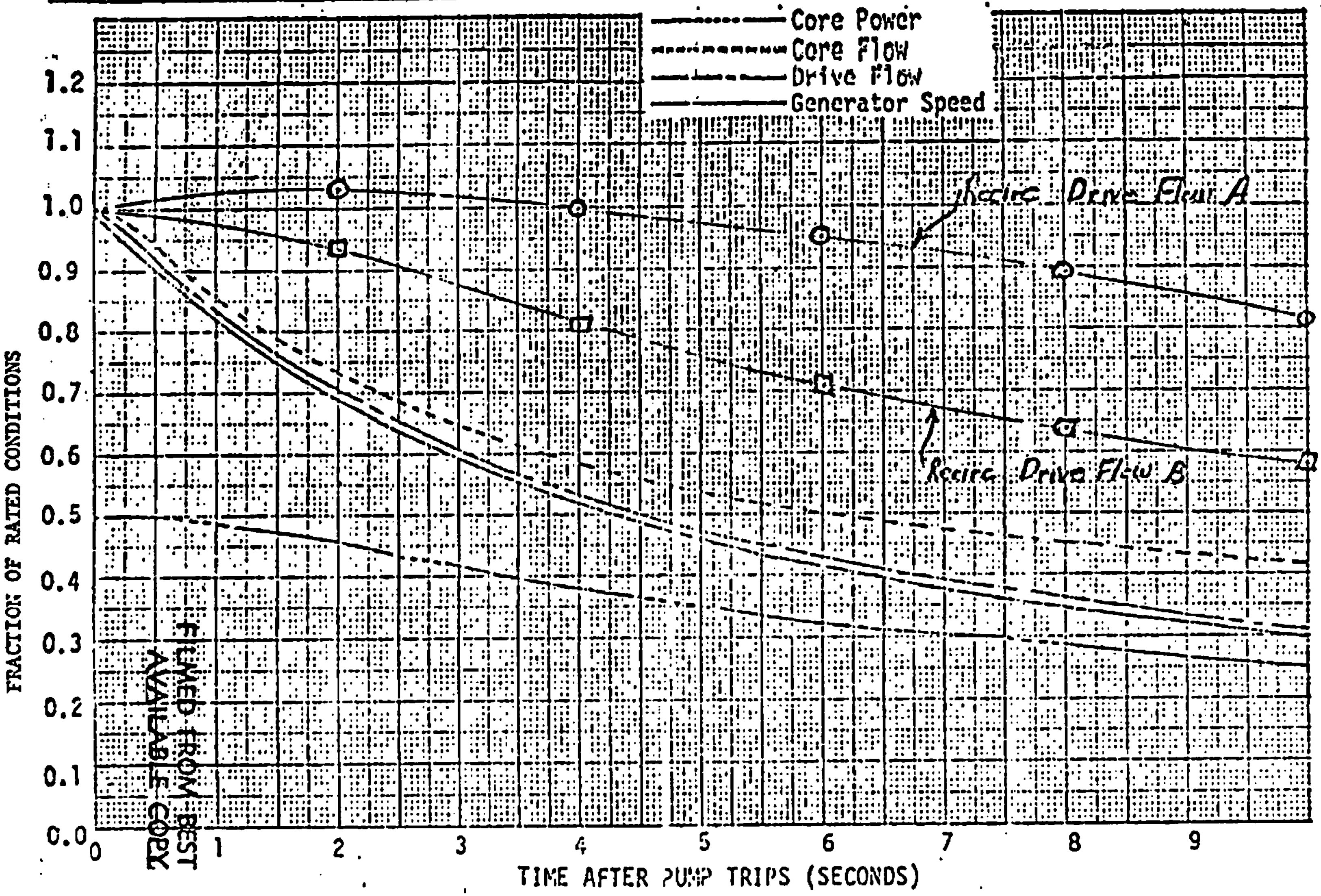


Figure STI 30-3

3293 MWT PLANT RECIRCULATION SYSTEM PERFORMANCE
FOLLOWING TRIP OF TWO DRIVE MOTORS - 50% POWER

Chart 2

Drive Motor Trip of Both Recirc Pumps

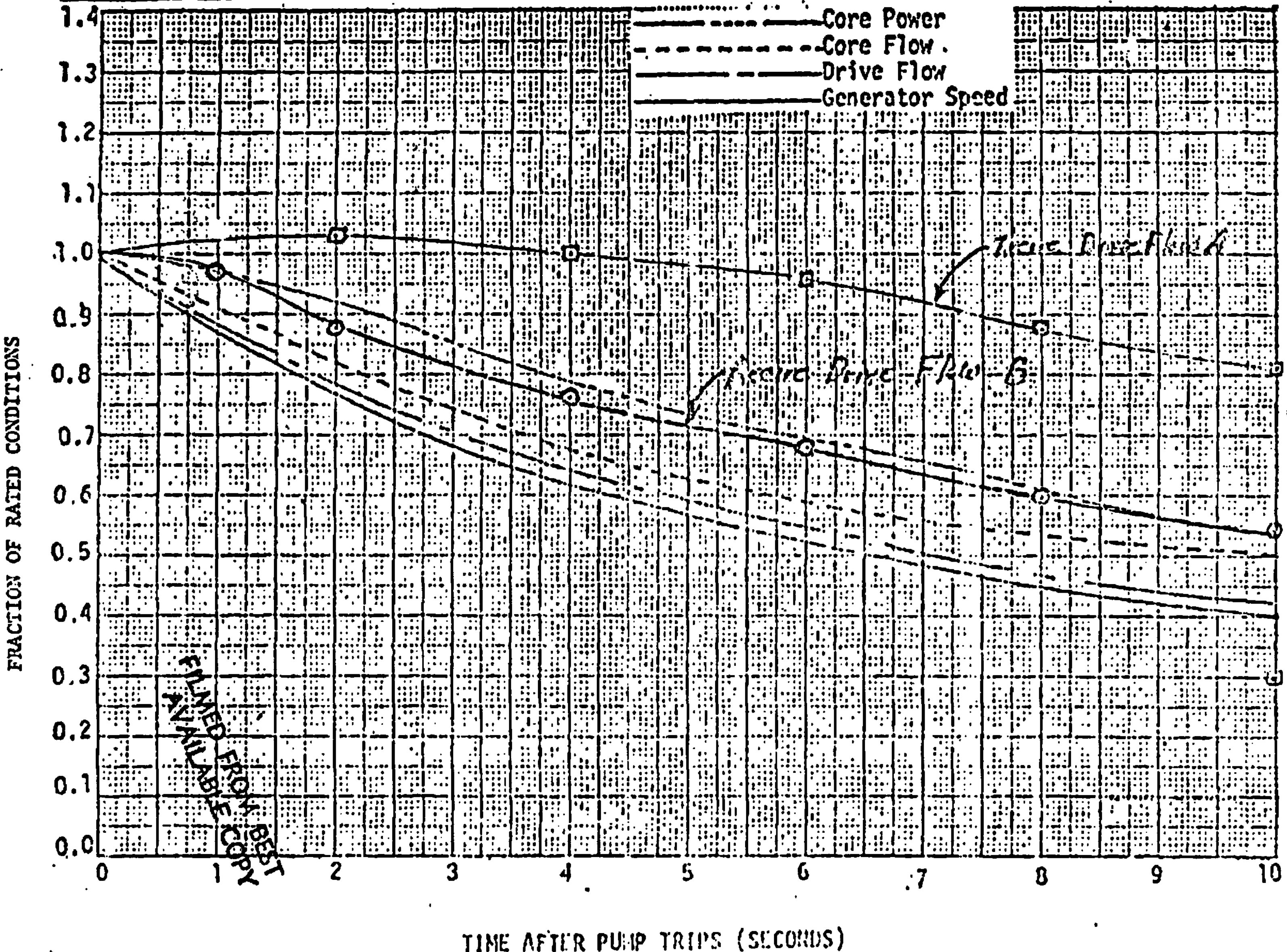


Figure STI 30-4

3293 MWT PLANT R[illegible] SYSTEM PERFORM[illegible]NCE
FOLLOWING TRIP OF [illegible] - 100% P[illegible]ER

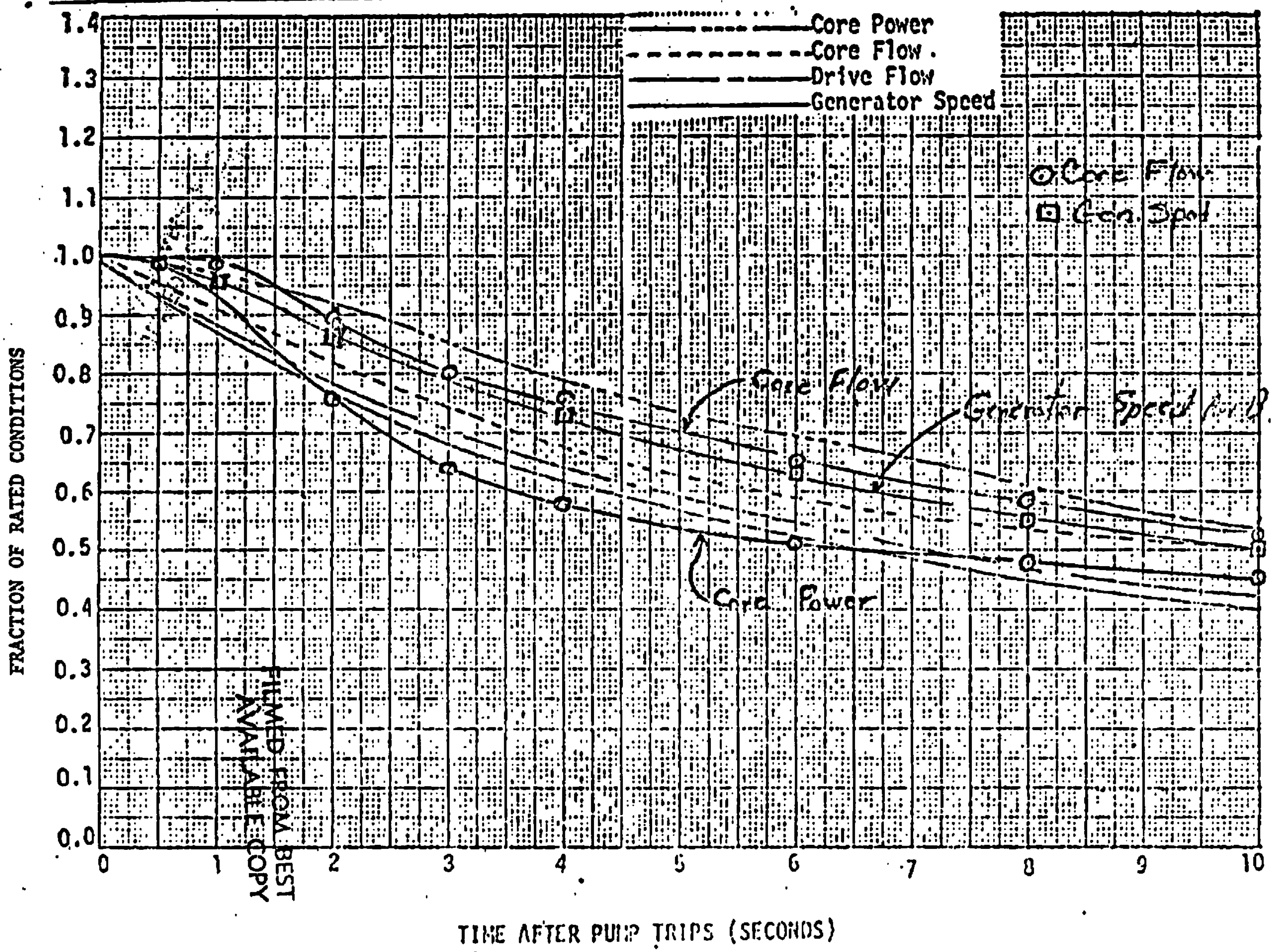


Figure STI 30-5

3293 MWT PLANT RECIRCULATION SYSTEM PERFORMANCE
FOLLOWING TRIP OF TWO DRIVE MOTORS - 100% POWER

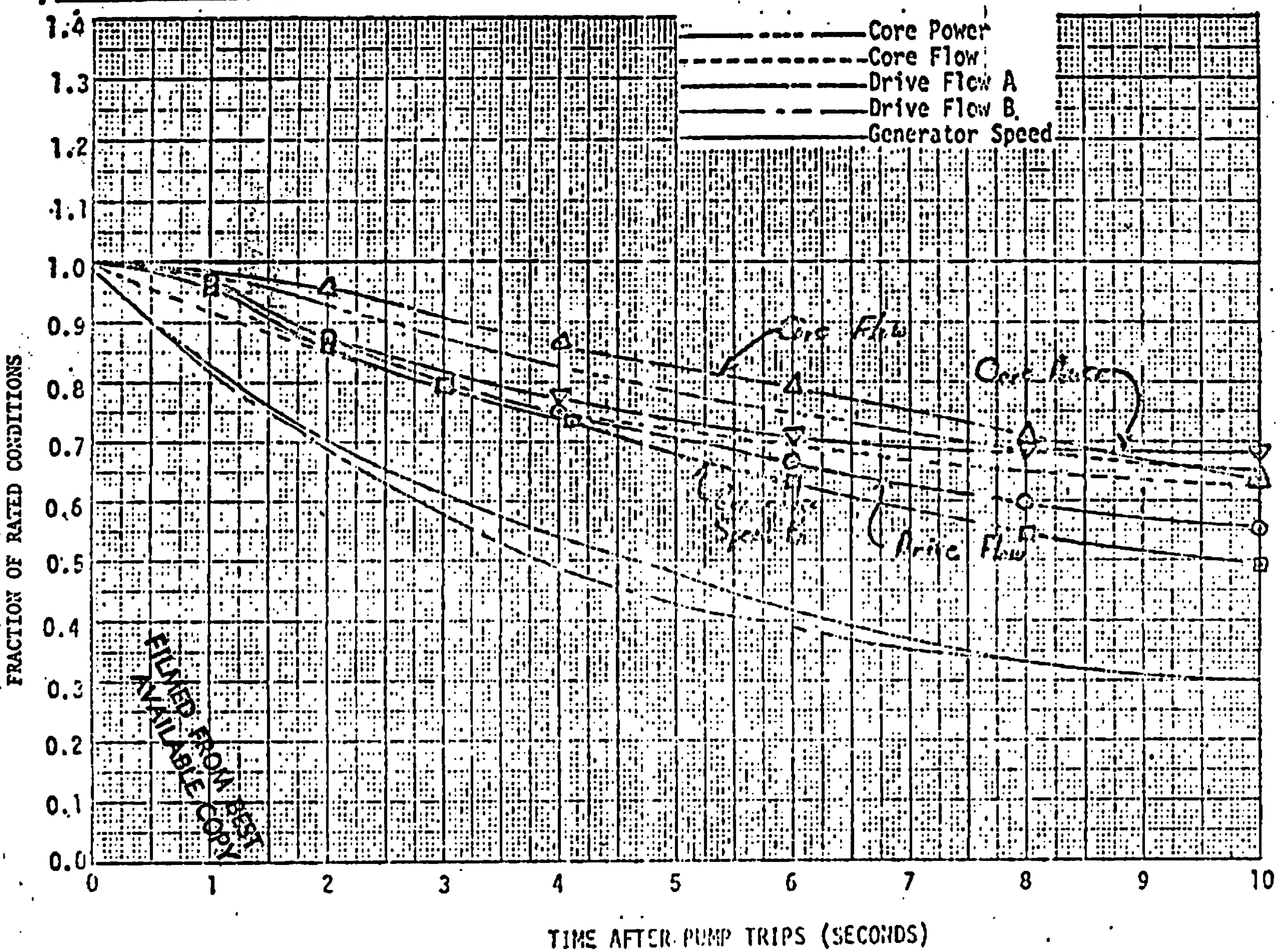


Figure STI 30-6

3293 MWT PLANT RECIRCULATION SYSTEM PERFORMANCE FOLLOWING TRIP OF ONE DRIVE MOTOR - 100% POWER

B Recirc Pump Trip

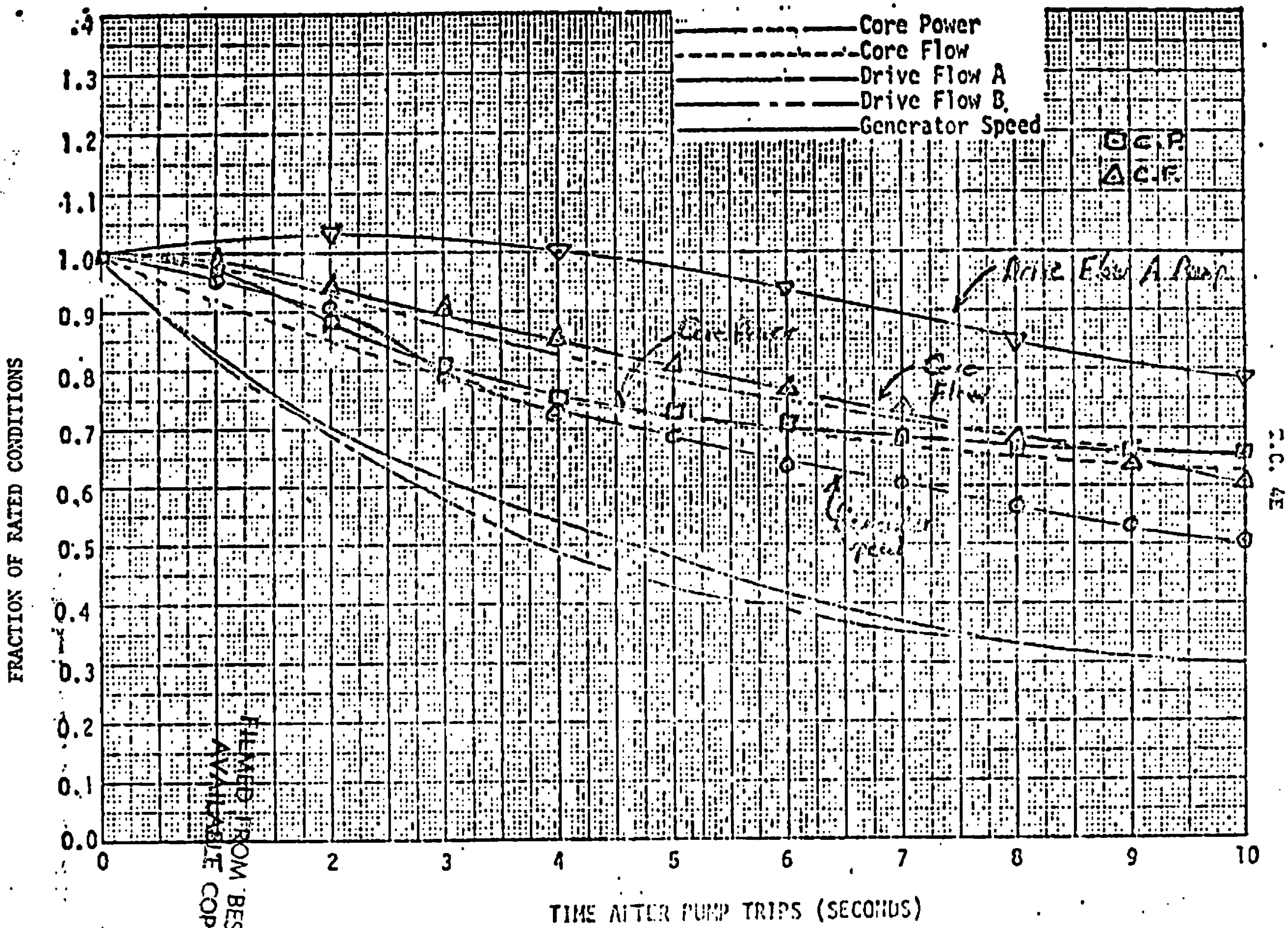


Figure STI 30-7

3293 MWT PLANT RECIRCULATION SYSTEM PERFORMANCE FOLLOWING TRIP OF ONE DRIVE MOTOR - 100% POWER

A Recirc Pump Trip

T.C. 4E

3.0 Results

3.27 STI-31, Loss of T-G and Offsite Power

3.27.1 Purpose

The purpose of STI-31 was to investigate the reactor transient performance during the loss of the main generator and all offsite power and to demonstrate the acceptable performance of the station electrical supply system during the loss of the main generator and all off-site power.

3.27.2 Criteria

Level 1

The initial transient rise in vessel dome pressure occurring within 10 seconds of turbine/generator trip action when initiated simultaneously with loss of offsite power when performed at 25-percent power shall not exceed 150 psi and the simulated heat flux rise shall not exceed 10 percent.

All safety systems, such as the RPS, diesel-generators, and the RCIC and HPCI, must function properly without manual assistance.

Level 2

The initial transient rise in vessel dome pressure occurring within 10 seconds of turbine/generator trip shall not be greater than 75 psi, and there shall be no significant increase in simulated heat flux.

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

3.27.3 Analysis

STI-31 testing was conducted at test condition 1 as defined in the power flow map in section 2.3. Prior to the test, the plant electrical system was aligned so that the only source of power to the unit 3 auxiliaries was the unit 3 station service transformer. The loss of offsite power test was performed by tripping the unit 3 generator negative phase sequence relay 346X and opening breaker 1405 on September 27, 1976. Water level dropped to -9.0 inches below the bottom of the dryer separators. Without intervention, auto initiation of

3.0 Results (Continued)

3.27 STI-31, Loss of T-G and Offsite Power (Continued)

3.27.3 Analysis (Continued)

HPCI and RCIC would have occurred at -31.5 inches. Approximately 5 minutes after the trip, RCIC was manually initiated to demonstrate operability. All diesel-generators came on-line after approximately 6.44 seconds. At approximately 18 seconds the reactor was manually scrammed. The scram function of the RPS was verified to operate properly by indication of AUTO scram at approximately 24 seconds due to low water level.

During the test, RPS MG set A continued running and MG set B's load breaker did not trip. Normally, the MG set motor input contactor will be opened in approximately 3 seconds; then the flywheel will carry the RPS bus loads until the frequency drops to 54.2 hertz at which time the breaker will trip. Investigation of MG set A and MG set B found that the time delay relays were improperly set to trip at 6.5 and 5.2 seconds and the output load breakers were incorrectly set. Both MG sets time delay relays were adjusted to drop out in approximately 3.0 seconds and the load breakers were correctly reset so that they would respond to an underfrequency trip signal.

The initial transient rise in vessel pressure occurring within 10 seconds of the turbine/generator trip was measured to be 3 psi. No rise in simulated heat flux was observed.

Normal cooling water systems maintained satisfactory suppression pool and drywell temperatures and prevented actuation of the auto-depressurization system. After the above corrections were made to the RPS-MG sets, all level 1 and 2 criteria were considered satisfied.

3.0 Results

3.28 STI-32, Recirculation Speed Control and Load Following

3.28.1 Purpose

The purposes of STI-32 are:

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

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

3. To demonstrate plant response to changes in recirculation flow.

3.28.2 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

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.

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

Following a 10% speed demand step from the low end of the master manual flow control range, the time from the step demand until the speed peak occurs shall be less than 25 seconds.

3.28.3 Analysis

STI-32 testing was conducted at test conditions 1, 2D, 2E, 3C, 3D, 3E, 4C, 4D, and 4E, as defined on the power flow map in section 2.3.

Prior to power operation, the recirculation system controllers were set up for stable operation. The initial settings were: proportional band = 500%; resets/min. = 20. At test condition 1 the settings were changed to give a slightly faster response with negligible overshoot. The new settings were: proportional band = 225%; resets/min. = 12.

3.0 Results

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

3.28.3 Analysis (Continued)

Further optimization of system controls resulted in final settings as summarized below:

Controller A: P.B. = 500%, 22 resets/min.
Controller B: P.B. = 200%, 9 resets/min.
Master Controller : P.B. = 80%, .9 resets/min.

To determine system response, $\pm$ 10% speed changes were performed on each pump individually, and with the pumps in the master-manual mode of control. Speed change testing was conducted at each test condition as required by section 6.1 of the test instruction. For all speed changes the decay ratio of all effected parameters was less than 0.25. No steam flow variations caused by steady state limit cycles were observed. For speed changes performed at the lower end of the master manual flow control range, the maximum time from the step demand to the speed peak was 24 seconds. All level 1 and level 2 testing criteria have been met.

Gain curves were obtained for each pump at test condition 2E. The curves were very nearly linear for both pumps; therefore, no cam cutting or linkage adjustment was necessary. The gain curve is shown in figure STI 32-1.

The mechanical stops of the recircupation pumps were set at a point corresponding to 105% core flow at test condition 4E. The electrical stops were set just below this. The load following range limiter was set for 44% pump speed on the low end and 105% core flow on the high end.

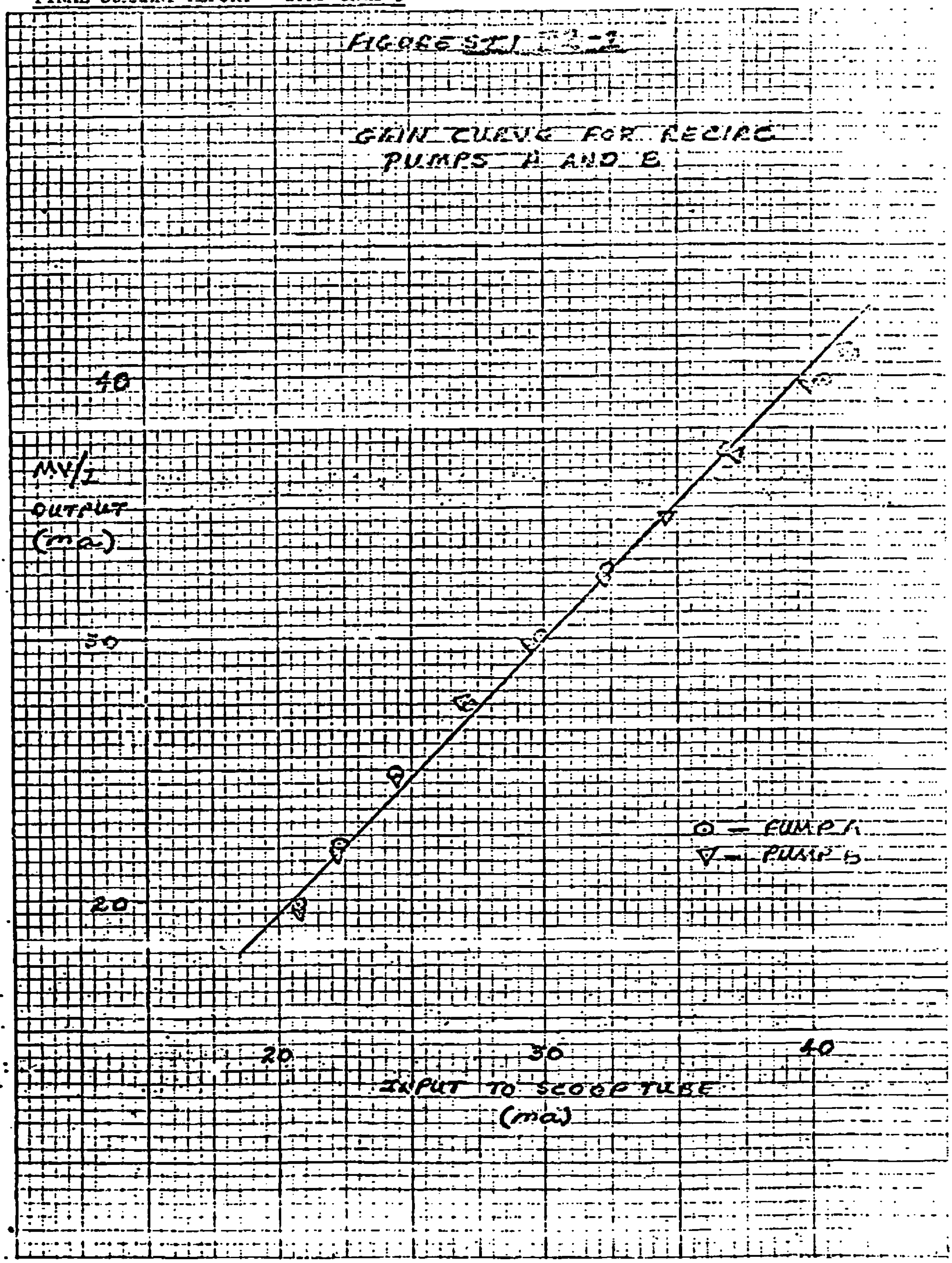


FIGURE STI
GAIN CURVE FOR RECIRC
PUMPS A AND B
40
MV/I
OUTPUT
(ma)
30
20
20
30
40
INPUT TO SCOOP TUBE
(ma)
O — PUMP A
▽ — PUMP B

3.0 Results

3.29 STI-33, Main Turbine Stop Valve Surveillance Test

3.29.1 Purpose

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

3.29.2 Criteria

Level 1

Not applicable

Level 2

Peak neutron flux must be at least 7.5% below the scram trip setting. Peak vessel pressure must remain at least 10 psi below the high pressure scram setting.

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

3.29.3 Analysis

STI-33 testing was successfully conducted at test conditions 1, 2E, 3E, and 4E as per the power flow map in section 2.3. Turbine stop valves were closed individually at selected power levels. Due to the turbine bypass header, most of the pressure peaking effect was dampened, producing negligible perturbations in the reactor. STI-33 demonstrated that the stop valve surveillance test may be satisfactorily performed at full power. The following table summarizes all the pertinent results from the stop valve surveillance test.

All test criteria were met.

3.0 Results (Continued)

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

3.29.3 Analysis (Continued)

Table STI 33-1

Test Condition / Parameter	1	2E	3E	4E	Limit
Date	9/19/76	10/8/76	11/3/76	11/23/76	NA
Reactor Power	750MWt=22.8%	1799MWt=54.6%	2705MWt=82.1%	3214MWt=97%	NA
Reactor Pressure	956 psig	950.6 psig	987 psig	997 psig	NA
Peak Neutron Flux	25.4%	57.2%	84.5%	98%	NA
Margin to Scram	10.5%	12.5%	10.5%	22%	≥ 7.5%
Peak Vessel Press Margin to Limit	98.5 psi	95 psi	90.4 psi	56.2 psi	≥ 10 psi
Peak Steam Line Flow Margin to Limit	110.5%	92.15%	55%	34%	≥ 10%

3.0 Results

3.30 STI-34, Vibration Measurements

3.30.1 Purpose

The purpose of STI-34 is to obtain vibration measurements on various reactor components to demonstrate the mechanical integrity of the system to flow induced vibration and to check the validity and accuracy of the analytical vibration model.

3.30.2 Criteria

Level 1

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 administered on site by the vibration test engineer conducting the test. (See section 8 of the startup test instruction for more detail.)

Level 2

Not applicable

3.30.3 Analysis

STI-34 testing was conducted at heatup and test conditions 1, 2D, 2E, 2A, 3C, 3D, 3E, 4C, 4D, 4E, and 4A as per the power flow map found in section 2.3. Vibration data was taken in conjunction with the recirculation pump trips and with the pumps at different speeds. Review of the data by the General Electric vibration specialist indicates that the vibration amplitudes are well within criteria limits.

3.0 Results

3.31 STI-35, Recirculation System Flow Calibration

3.31.1 Purpose

The purpose of STI-35 is to perform a complete calibration of the installed recirculation system flow instrumentation.

3.31.2 Criteria

Level 1

Not applicable.

Level 2

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.

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

3.31.3 Analysis

STI-35 testing was conducted at the open vessel test plateau and test conditions 2E, 3E, and 4E as defined by the power-flow map in section 2.3. Prior to power testing, the recirculation flow nozzle transmitters were calibrated for a 0 to 29.4 psi span and an off-set of .2 ma on the single tap ΔP transmitters. During test conditions 2E and 3E, the indicated core flow was verified to be within 2% of the calculated values. At these two test conditions, the jet pump flow instrumentation provided an accurate indication of core flows such that adjustments were not necessary. Experience has shown that the accuracy of the core flow calibration increases with power level.

Three sets of core flow data were taken at rated conditions. Based on this data, the gains of the jet pump loop and total core flow proportional amplifiers were adjusted to give the correct control room indications of total core flow and jet pump loops A and B flows. Comparison of the total core flow recorder and the process computer core flow data point showed agreement within 0.08%. Subsequently, three additional data sets were taken to confirm the recirculation flow nozzle transmitter spans. Based upon analysis of this data, the flow nozzle transmitters were subsequently spanned to 24.5 psid for Loop A, and 29.8 psid for Loop B. The M-ratios

3.0 Results

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

3.31.3 Analysis (Continued)

calculated via the computer program "JRPUMP", were within the band of expected theoretical values. The gain adjustment factors and as-left gains are as follows:

Loop	Instrument Gain Adjustment Factor	As-Left Gains
A	.99	.495
B	1.01	.505

The APRM/RBM flow bias instrumentation was adjusted and found to perform satisfactorily. In addition, all jet pump riser plugging, nozzle plugging, and loop flow variation criteria were satisfied.

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results

3.32 STI-70, Reactor Water Cleanup System

3.32.1 Purpose

The purpose of STI-70 is to demonstrate specific aspects of the mechanical operability of the Reactor Water Cleanup System. (This test, performed at rated reactor pressure and temperature, is actually the completion of the preoperational testing that could not be done without nuclear heating.)

3.32.2 Criteria

Level 1

Not applicable

Level 2

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

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

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.)

3.32.3 Analysis

STI-70 testing was conducted during heatup as defined on the power flow map in section 2.3. The reactor water cleanup system was successfully tested at rated reactor pressure and temperature in the blowdown, hot standby, and normal mode. It was demonstrated that the service water could remove 24.70×10^6 Btu/hr from the non-regenerative heat exchangers when the cleanup system was in the blowdown mode. The regenerative exchangers were found to have a capacity of 37.95×10^6 Btu/hr when the cleanup system was in the hot standby mode.

The NPSH is strongly dependent on the temperature of the water on leaving the pressure vessel and entering the cleanup system. Because the actual value of the pump inlet temperature was below the process diagram, the process

3.0 Results (Continued)

3.32 STI-70, Reactor Water Cleanup System (Continued)

3.32.3 Analysis (Continued)

diagram value of 545°F was used for conservatism. This temperature resulted in an available NPSH of 37.3 ft at 545° F, considerably larger than the required 13 ft.

Figure STI 70-1 summarizes the results of the reactor water cleanup system test in each mode of operation.

Figure STI 70-1
Summary of RWCU System Test

R.W.C.U. System Mode	Temp Measured @ Outlet of NRHX's (Tube side) °F	Required Temp °F
Normal	109	<130
Hot Standby	108	<130
Blowdown	121	<130

Cooling Water to Non-Regen. Heat Exchangers

Mode	Required Flow	Actual Flow	Required Temp.	Actual Temp.
Normal	618gpm	618 gpm	150° F	136° F
Blowdown	625gpm	625 gpm	180° F	177° F

All test criteria were satisfied.

3.0 Results

3.33 STI-71, Residual Heat Removal System

3.33.1 Purpose

The purpose of STI-71 is to demonstrate the ability of the Residual Heat Removal (RHR) system to remove residual and decay heat from the nuclear system so that refueling and nuclear system servicing can be performed.

3.33.2 Criteria

Level 1

Not applicable

Level 2

The heat removal capability of each RHR heat exchanger in the shutdown cooling mode shall be at least 187 X 10^6 Btu/hr when the inlet flows and temperatures are as indicated on the process diagrams. (See section 8 of this test for summary of flow rates.)

3.33.3 Analysis

STI-71 testing was conducted at test conditions 1 as defined on the power flow map in section 2.3 and at hot shutdown. At test condition 1, the capacity of the RHR heat exchangers from the shutdown cooling mode test could not be demonstrated due to insufficient decay heat. Also, the suppression pool cooling mode method was unsuccessful in determining the RHR heat exchanger capacity because of an insufficient ΔT. Therefore, this test was repeated following the load rejection trip from test condition 4E. The calculated heat removal capacities ranged from 188.7 to 532 MBtu/hr. Additionally, the head spray capacity was verified by obtaining a rated flow of 1000 gpm.

3.0 Results

3.35 STI-72, Drywell Atmosphere Cooling System

3.35.1 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 and post-scram conditions.

3.35.2 Criteria

Level 1

Not applicable

Level 2

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

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

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

During normal reactor operation:

150° F average throughout drywell

50% relative humidity

135° F maximum around the recirculating pump motors

200° F maximum above the bulkhead

180° F maximum for all other areas

Ten hours after shutdown:

Within 15° F of closed cooling water inlet temperature (average throughout the drywell)

Cooling water supply:

100° F maximum

3.0 Results (Continued)

3.35 STI-72, Drywell Atmosphere Cooling System (Continued)

3.35.3 Analysis

STI-72 testing was performed at heatup and test condition 4E levels as defined on the power flow map in section 2.3.

Data recorded at each plateau of heatup indicated a uniform temperature increase as was expected. All temperatures were within design limits for this level of testing. (See table STI 72-1) The estimated heat removal rate of the drywell coolers was 4.4 x 10^6 Btu/hr. Drywell humidity could not be evaluated due to the inoperability of instrument MR-80-36. This item was carried as an exception. It should be noted that the cooling water inlet temperature was 84° F. Extrapolation of data to a design maximum of 100° F inlet temperature indicates that all temperatures will be within design limits.

Data recorded at test condition 4E indicated that all normal operational temperature limits were within design limits. (See table STI 72-1) Extrapolation of data during heatup testing to a design maximum inlet water temperature of 100° F. indicates that all temperatures will be within design limits. The estimated heat removal rate of the drywell coolers was 5.13 x 10^6 Btu/hr. This meets level 2 criteria, that the cooler heat removal rate be approximately 5.19 x 10^6 Btu/hr.

Instrument MR-80-36 was repaired prior to reaching test condition 4E. Channels A and B indicated 36% and 53% relative humidity. This cleared the exception to STI-72 during heatup testing. Level 2 criteria required drywell humidity to be below 50%. Drywell humidity was therefore carried as an exception to STI-72. Following inerting of the unit 3 drywell MR-80-36 indicated 29% and 33% relative humidity on channels A and B, respectively. This cleared the associated exception.

During test condition 4E testing, drywell cooler fans A2 and B2 were inoperative. This prevented testing following a full power scram to determine if level 2 criteria, requiring that the average drywell temperature be within 15° F of the closed cooling water inlet temperature 10 hours after shutdown, can be met. This item is carried as an exception to STI-72. Drywell cooling fans A2 and B2 have been repaired. This test will be performed as soon as plant conditions permit.

3.0 Results (Continued)

3.35 STI-72, Drywell Atmosphere Cooling System (Continued)

3.35.3 Analysis (Continued)

Table STI 72-1

Parameter	Design Limit	Heatup	T.C. 4E
Avg. DW Temp.	150° F	126° F	130.6° F
Recirc. Pump Temp.	135° F	109° F	108° F
Above Bulkhead Temp.	200° F	153° F	157° F
Max. Temp. Other Areas	180° F	150° F	156° F

3.0 Results

3.36 STI-73, Cooling Water Systems

3.36.1 Purpose

The purpose of this test is to verify that the performance of the Reactor Building Closed Cooling Water (RBCCW) system is adequate with the reactor at rated conditions.

3.36.2 Criteria

Level 1

Not applicable

Level 2

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

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

3.36.3 Analysis

STI-73 testing was performed at heatup and test condition 4E levels as defined on the power flow map in section 2.3.

At hot standby the calculated heat load was 18.98×10^6 Btu/hr on the RBCCW side of the heat exchangers and 21.0×10^6 Btu/hr on the RCW side. At test condition 4E the heat load was 24.86×10^6 Btu/hr on the RBCCW side and 21.86 on the RCW side. It should be noted that the RCW flow was extremely low at test condition 4E due to cold river water. Therefore, the RCW side heat balance cannot be considered reliable due to inaccuracies in the flow measurement system at low flow rates.

Data indicates that the RBCCW system component flow and heat exchangers are properly balanced. Significant parameters are summarized in table STI 73-1.

3.0 Results (Continued)

3.36 STI-73, Cooling Water Systems (Continued)

3.36.3 Analysis (Continued)

Due to low RCW flow and temperature it is not possible to extrapolate the data to design rated conditions. Therefore, it cannot be determined if design criteria will be met at rated system heat load and temperatures. All criteria were met for conditions at which testing was conducted. The RBCCW system is adequate for handling system heat loads until the fuel pool heat exchangers approach design heat load. The Division of Engineering Design is evaluating system performance at rated system heat load and temperature. When RCW temperatures approach design values, additional testing will be performed to clear this exception.

Table STI 73-1
RBCCW Operation at T.C. 4E

Parameter	Max. or Design Value	Measured Value
Total RBCCW Flow	3369.5 gpm	3648.5
RBCCW Inlet Temp.		
Ht. x A	118.5° F	96.2
Ht. x B	118.5° F	96.2
RBCCW Outlet Temp.		
Ht. x A	100° F	84.5
Ht. x B	100° F	80.5
RCW Flow		
Ht. x A	2550 gpm	~ 331 gpm
Ht. x B	2550 gpm	~ 689 gpm
RCW Inlet Temp.		
Ht. x A	90° F	44.4° F
Ht. x B	90° F	44.5° F
RCW Outlet Temp.		
Ht. x A	102.3° F	88.8° F
Ht. x B	102.3° F	87.0° F
Heat Removal Rate		
RBCCW Side	—	24.86 x 10^6 Btu/hr.
RCW Side	—	21.9 x 10^6 Btu/hr.

3.0 Results (Continued)

3.37 STI-74, Modified Off-Gas System

3.37.1 Purpose

The purposes of this test are:

1. To verify the proper operation of the off-gas system over its expected operating parameters.

2. To determine the performance of the activated carbon adsorbers.

3.37.2 Criteria

Level 1

The release of radioactive gaseous particulate effluents must not exceed the limits specified in BFNP technical specifications 3.8.B.

There shall be no loss of flow for dilution steam to the noncondensing stages when the steam jet air ejectors are pumping.

Level 2

The system flow, pressure, temperature, and relative humidity shall comply with the design specifications shown in form 74.6-1.

The catalytic recombiner, the hydrogen analyzer, the activated carbon beds, and the filters shall be working as designed.

3.37.3 Analysis

STI-74 testing was performed at test conditions 1, 2E, 3E, and 4E as defined on the power flow map in section 2.3.

Airborne Releases - Airborne releases during testing were documented in surveillance tests SI 4.8.B.1-a and SI 4.8.B.2-6. There were no violations of the BFNP Tech. Specs.' 3.8.B limits at any test condition. Therefore, level 1 criteria were fully satisfied.

3.0 Results (Continued)

3.37 STI-74, Modified Off-Gas System (Continued)

3.37.3 Analysis (Continued)

Dilution Steam Flow - There were no losses of dilution steam flow to the noncondensing stages of the pumping SJAE during any testing. The total dilution steam flows are recorded in table STI 74-1. Level 1 criteria were fully satisfied.

System Parameters - Table STI 74-1 summarizes system operating parameters during startup.

The system temperatures, pressures, flow, and relative humidity complied with design specifications, except for the following:

1) A malfunctioning gauge prevented SJAE outlet pressure from being obtained during test condition 1. However, the gauge was repaired before subsequent test conditions where the pressures were maintained within the normal operating range. This was a level 2 criterion exception.

2) Adsorber bed F temperature anomaly was reported at all test conditions and is believed to be due to a cooling effect of moisture being removed from the bed. In addition, the thermocouple that provides this temperature as recorded on TRS-66-115 seems to be responding properly, but, as outage time permits, will be examined at the adsorber bed inside the vault. This was a level 2 criterion exception.

3) Hydrogen analyzer malfunctions are discussed below.

Table STI 74-1

System Parameters	% Power	15-35	40-60	65-85	[illegible]
	Date	10/4/76	10/11/76	11/3/76	[illegible]
	MWt	820	1937	2531	[illegible]
	Norm Operating Range	TC1	TC2E	TC3E	TC4E
DIL Steam Flow (Total)	9100#/hr	9350	9350	9700	[illegible]
SJAE Outlet Pressure	5-10 psig	Inop. (1)	5	5	[illegible]
OG Preheater T Outlet	275°-360° F	350	350	350	[illegible]
Active Recomb. Temp.					
Bottom	275°-875° F	425	555	605	605
Middle	275°-875° F	420	543	605	605
Top	275°-875° F	405	535	585	[illegible]
Standby Recomb. Temp.					
Bottom	275°-360° F	320	325	320	325
Middle	275°-360° F	305	315	315	335
Top	275°-360° F	295	315	315	[illegible]
OG Cond. Coolant Out	120° F	110	103	109	105
OG Cond. Outlet Temp.	140° F	123	119	117	[illegible]
H_2 Concentration	0-1%	.05	.05	0	(1)
OG Flow	20-40 scfm	35	35	30	[illegible]
Glycol Pump P	20-40 psig	32	31	38	[illegible]
Glycol Tank T	33°-38° F	34	36	36	36
Moist. Sep. T Out	55° F	50	49	55	55
Reheater Dewpoint	48° F	42	42	43	42
Reheater T Out	72°-76° F	74	74	74	74
Prefilter D.P.	0-2" water	.05	.2	0	0
Adsorber D.P.	.5-2.6 psi	2.2	.8	.75	.8
Bypass D.P.	0-2" water	0	0	0	0
Adsorber Vessel T					
Bed A Pt. 1	68°-79° F	70.0	72	69(2)	69
Bed A Pt. 2	68°-79° F	71.0	71	68(2)	69
Bed B Pt. 3	68°-79° F	70.0	67.5	68(2)	69
Bed B Pt. 4	68°-79° F	68.0	70	69(2)	69
Bed C Pt. 5	68°-79° F	68.5	69.5	68(2)	69.5
Bed D Pt. 7	68°-79° F	70.0	75.5	68.5(2)	69.5
Bed F Pt. 6	68°-79° F	62.0	58	64(2)	52.5
Adsorber Vault T	73°-81° F	75.0	73.5	75.5	76.5
After Filter D.P.	0-2" water	.35	.5	0	.4
% Rel. Hum.	40%	32	32	34	32

(1)Data not obtained or was out of operating range and carried as an STI exception .

(2)These readings were taken on 11/4/76 at 2490 MWt and the same test condition.

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

3.37 STI-74, Modified Off-Gas System (Continued)

3.37.3 Analysis (Continued)

Tables STI-74-2A and -2B summarize hydrogen analyzer performance data taken during startup.

Table STI 74-2A

	% Power	15-35	40-60	65-85	95-100
	Date	9/27/76	10/11/76	11/3/76	11/22/76
HYDROGEN ANALYZER PERFORMANCE	MWt	1038	1937	2531	3274
H_2 Analyzer	Normal Operating Range	T.C. 1	T.C. 2E	T.C. 3E	T.C. 4E
Process Reading % H_2	0-1	.08	.05	0	INOP
Sample Flow scfh	3-4	4	4.0	< 2	
Demin. Water flow gph	1-2	2	1.5	1.5	
Vacuum regulator water	10-25	20	17	10-40	
Calibration Standard scfh	3-4	3.5	4.0	< 2	
Calibration Standard % H_2	1.0	1.0	1.0	1.0	
Calibration Gas Results % H_2	1.0	1.0	1.1	1.0	
H_2 Free Standard scfh	3-4	3.5	4.0	< 2	
H_2 Free Standard % H_2	0	0	0	0	
H_2 Free Results % H_2	0	0	0	0	

FINAL SUMMARY REPORT - BFNP UNIT 3

3.0 Results (Continued)

3.37 STI-74, Modified Off-Gas System (Continued)

3.37.3 Analysis (Continued)

Table STI 74-2B

	% Power	15-35	40-60	65-85	95-100
	Date	9/27/76	10/11/76	11/3/76	11/22/76
HYDROGEN ANALYZER PERFORMANCE	MWt	1038	1937	2531	3274
H_2 Analyzer B	Normal Operating Range	T.C. 1	T.C. 2E	T.C. 3E	T.C. 4E
Process Reading % H_2	0-1	.05	.1	INOP	INOP
Sample Flow scfh	3-4	4	2-4		
Demin. Water flow gph	1-2	2	1.5		
Vacuum regulator water	10-25	17	15		
Calibration Standard scfh	3-4	3.8	4.0		
Calibration Standard % H_2	1.0	1	1.0		
Calibration Gas Results % H_2	1.0	1	1.5		
H_2 Free Standard scfh	3-4	3.8	4.0		
H_2 Free Standard % H_2	0	0	0		
H_2 Free Results % H_2	0	0	0		

3.0 Results (Continued)

3.37 STI-74, Modified Off-Gas System (Continued)

3.37.3 Analysis (Continued)

The hydrogen analyzers were not reliable for continuous process use. This was attributed to moisture which, when condensed, caused erratic sample flow and improper sensor response. Engineering Change Notice, ECN 1825 will install the required modifications to the hydrogen analyzers to resolve this problem.

Both hydrogen analyzers failed to perform satisfactorily at test conditions 2E, 3E, and 4E, and, therefore, do not fulfill level 2 criteria. Grab samples taken and analyzed by the radiochemical laboratory insured that the hydrogen concentration was less than 4%.

Catalytic Recombiner - Table STI 74-3 summarizes catalytic recombiner performance during startup.

Table STI 74-3

RECOMBINER PERFORMANCE	Power % Date MWt TC	15-35 9/27/76 1038 T.C. 1	40-60 10/11/76 1937 T.C. 2E	65-85 11/3/76 2531 T.C. 3E	95-100 11/22/76 3274 T.C. 4E
Radiolytic Gas Production Rate, CFM/MWt		.03	.04	.038	.035
Active Recombiner Temp, °F		425	555	605	605
OG Preheater Temp Outlet, °F		350	350	350	340
Δ T Actual, °F		75	250	255	265
Δ T Expected, °F		87	225	288	261

The catalytic recombiners performed satisfactorily during startup. Level 2 criteria was satisfied.

3.0 Results (Continued)

3.37 STI-74, Modified Off-Gas System (Continued)

3.37.3 Analysis (Continued)

Adsorber Beds - Table STI 74-4 summarizes the calculated residence times for four radionuclides and the Xe/Kr ratios across the six charcoal adsorber beds operated in series.

Table STI 74-4

Charcoal Adsorber Performance (Residence Time)	% Power Date MWt T.C.	15-35 9/27/76 1038 T.C. 1	40-60 10/9/76 1890 T.C. 2E	65-85 11/5/76 2555 T.C. 3E	95-100 11/22/76 3274 T.C. 4E
Kr88 (Actual), Hr.		33	7.6	10.4	15
Kr85m (Actual), Hr.		43	7.3	10.1	13
Kr (Expected), Hr.		19.2	11.5	9.7	15
Xe135 (Actual), Day		7.3	7.8	10.1	10
Xe133 (Actual), Day		89.7	68	23.8	16
Xe (Expected), Day		11.5	8.8	7.3	12
Ratio Xe/Kr (Actual)		5/1(1)	25/1(1)	23/1(1)	22:1
Ratio Xe/Kr (Expected)		18/1	15/1	18/1	19:1

(1) Xe133 was not averaged into ratio because it was not in equalibrium. This was the result of the unit 1 offgas flow, heavily laden with Xe133, being routed through unit 3 adsorber beds during unit 1 maintenance. A large Xe133 inventory remained to slowly be eluted from the unit 3 adsorber beds.

Calculated and expected radionuclide delay times through the adsorber beds showed good agreement at all test conditions. In particular, fuel power testing performed after several days of steady reactor operation represented the expected adsorption of the Xe and Kr radionuclides. Level 2 criteria has been satisfied.

3.0 Results (Continued)

3.37 STI-74, Modified Off-Gas System (Continued)

3.37.3 Analysis (Continued)

System HEPA Filters - Table STI 74-5 summarizes the results of radiochemical testing of the offgas system prefilter and after filters.

Table STI 74-5

Hepa Filter(1) Efficiencies	% Power	15-35	40-60	65-85	90-100
	Date	9/26/76	10/8/76	11/3/76	11/22/76
	MWt	1038	1876	2531	3274
	T.C.	T.C. 1	T.C. 2E	T.C. 3E	T.C. 4E
Prefilter A					
Cs138, %		99.9	99.6	>99.6(2)	>99.8(2)
Rb88, %		>98.7(2)	97.9	>99.6	>98.4(2)
Prefilter B					
Cs138, %		99.9	>99.9(2)	>99.6(2)	>99.8(2)
Rb88, %		>98.9(2)	99.5	>97.9(2)	>98.9(2)
Afterfilter A					
Cs138, %		-126(3)	-126(3)	>77.5(2)	>68(2)
Rb88, %		>12.3(2)	-91.5(3)	>91.4(2)	>94.7(2)
Afterfilter B					
Cs138, %		-87.6(3)	-24.5(3)	>58.4(2)	>31(2)
Rb88, %		>33.8(2)	189	>85.7(2)	>92.9(2)

Footnotes on next page

3.0 Results (Continued)

3.37 STI-74, Modified Off-Gas System (Continued)

3.37.3 Analysis (Continued)

Table STI 74-5 (Continued)

(1) Activity levels of Ba140 before and after both the prefilters and afterfilter were too low to detect statistically. Therefore, the calculated efficiencies were meaningless and were omitted from this test.

(2) ">" means that the actual efficiency is some value larger than this value, but because a concentration (or both) used to calculate the efficiency was itself less than the detectable concentration, the actual value could not be determined.

(3) When the afterfilter outlet concentration was decay-corrected to sample time, this effluent appeared to have more activity than the inlet. (The efficiencies were negative.) Actually, both the inlet and outlet had activity levels too low to detect statistically. This was remedied at test conditions 3E and 4E by using a partial prefilter bypass.

Efficiencies of the prefilters were measured and found to be satisfactory.

Laboratory analyses of the afterfilters indicated that they were operating properly. Level 2 criteria were satisfied.

All required startup testing for the modified off-gas system has been satisfactorily completed with those exceptions listed.

3.0 Results

3.37 STI-75, Reactor Scram From Outside Main Control Room

3.37.1 Purpose

The purposes of STI-75 are:

1. To demonstrate that the plant design permits safe reactor shutdown from outside the main control room.
2. To demonstrate that the reactor can be maintained in a safe condition after shutdown from outside the main control room.
3. To demonstrate that the minimum number of personnel required by the tech specs is adequate to perform steps 1.1 and 1.1.1 without affecting the safe continuous operation of the other units.
4. To demonstrate that EOI-34, Control Room Abandonment, is adequate to perform steps 1.1, 1.1.1, and 1.1.2 without affecting unit safety.

3.37.2 Criteria

Level 1

Not applicable.

Level 2

Initiation of reactor scram must occur from outside the main control room.

Reactor water level must be maintained greater than 490" above vessel zero level and less than the high level turbine trip point.

The RHR and RHRSW pumps and control valves shall be operable from the backup controls to initiate suppression pool cooling.

The minimum number of personnel as required by the tech specs can conduct this test.

3.37.3 Analysis

STI-75 was conducted at a power level of 11.5% with the

3.0 Results (Continued)

3.37 STI-75, Reactor Scram From Outside Main Control Room (Continued)

3.37.3 Analysis (Continued)

turbine/generator off-line. Control was transferred from the main control room to the remote panel 25-32 prior to initiating a reactor scram by closure of the MSIV's. Reactor water level on a Yarway initially started at +45" and decreased to +10" after the scram. The reactor core isolation cooling system initiated to maintain level at +10". The minimum water level observed was 538 inches above vessel zero (+10 inches on Yarway A). The maximum water level observed was 566 inches above vessel zero, well below the high level turbine trip setpoint at 582 inches.

There were no unexpected events during the performance of this test and all test criteria were satisfied. Prior to terminating the test (at ≈ 17 minutes), the following plant conditions were observed:

RHR HDR A - 155 psig	EECW Pump A - 3500 gpm
RHR HDR B - 60 psig	EECW Pump B - 0
RHR HDR C - 40 psig	EECW Pump C - 0
RHR HDR D - 70 psig	EECW Pump D - 3500 gpm

RCIC Flow - 520 gpm
Drywell Temp - 85°F
Suppression Chamber Temp - 100°F
Suppression Chamber Level - 2 inch
Reactor Pressure - 660 psig

ENCLOSURE 6

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

EXTENDED POWER UPRATE RS-001 REVISED TEMPLATE SAFETY EVALUATION

The attached pages have been revised. On the affected pages, the revised portions have been highlighted. A line has been drawn through the deleted text and a double underline for new or revised text.

2.8.4.4 Residual Heat Removal System

Regulatory Evaluation

The RHR system is used to cool down the RCS following shutdown. The RHR system is typically a low pressure system which takes over the shutdown cooling function when the RCS temperature is reduced. The NRC staff's review covered the effect of the proposed EPU on the functional capability of the RHR system to cool the RCS following shutdown and provide decay heat removal. The NRC's acceptance criteria are based on (1) draft GDC-40 and 42, insofar as they require that ESFs be protected against dynamic effects; ~~and~~ (2) draft GDC-4, insofar as it requires that reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing; and (3) draft GDC-6, insofar as it requires that decay heat removal systems shall be provided for all expected conditions of normal operation. Specific review criteria are contained in SRP Section 5.4.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the RHR system. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the RHR system will maintain its ability to cool the RCS following shutdown and provide decay heat removal. Based on this, the NRC staff concludes that the RHR system will continue to meet the requirements of draft GDC-4, 6, 40 and 42 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RHR system.

C-01

2.8.5 Accident and Transient Analyses

2.8.5.1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Main Steam Relief or Safety Valve

Regulatory Evaluation

Excessive heat removal causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) postulated initial core and reactor conditions, (2) methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor system components, (5) functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; (~~2~~3) draft GDC-14 and 15, insofar as they require that the core protection system be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; and (~~3~~4) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.1.1-4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a

C-02

clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the excess heat removal events described above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the AFDLs and the RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 9, 14, 15, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

2.8.5.2 Decrease in Heat Removal by the Secondary System

2.8.5.2.1 Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)

Regulatory Evaluation

A number of initiating events may result in unplanned decreases in heat removal by the secondary system. These events result in a sudden reduction in steam flow and, consequently, result in pressurization events. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (~~2~~3) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.2.1-5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the decrease in heat removal events described above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were

C-04

performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the AFDLs and the RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 9, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

2.8.5.2.2 Loss of Nonemergency AC Power to the Station Auxiliaries

Regulatory Evaluation

The loss of nonemergency ac power is assumed to result in the loss of all power to the station auxiliaries and the simultaneous tripping of all reactor coolant circulation pumps. This causes a flow coastdown as well as a decrease in heat removal by the secondary system, a turbine trip, an increase in pressure and temperature of the coolant, and a reactor trip. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (~~2~~3) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.2.6 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the loss of nonemergency ac power to station auxiliaries event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were

performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the AFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 9, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the loss of nonemergency ac power to station auxiliaries event.

2.8.5.2.3 Loss of Normal Feedwater Flow

Regulatory Evaluation

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a LOOP. Loss of feedwater flow results in an increase in reactor coolant temperature and pressure which eventually requires a reactor trip to prevent fuel damage. Decay heat must be transferred from fuel following a loss of normal feedwater flow. Reactor protection and safety systems are actuated to provide this function and mitigate other aspects of the transient. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (~~2~~3) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.2.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the loss of normal feedwater flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the

licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the AFDLs and the RCPB pressure limits will not be exceeded as a result of the loss of normal feedwater flow. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 9, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the loss of normal feedwater flow event.

2.8.5.3 Decrease in Reactor Coolant System Flow

2.8.5.3.1 Loss of Forced Reactor Coolant Flow

Regulatory Evaluation

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. An increase in fuel temperature and accompanying fuel damage could then result if AFDLs are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the postulated initial core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor systems components, (5) the functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (~~2~~3) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.3.1-2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the decrease in reactor coolant flow event and concludes that the

licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the AFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 9, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the decrease in reactor coolant flow event.

2.8.5.4.3 Startup of a Recirculation Loop at an Incorrect Temperature and Flow Controller Malfunction Causing an Increase in Core Flow Rate

Regulatory Evaluation

A startup of an inactive loop transient may result in either an increased core flow or the introduction of cooler water into the core. This event causes an increase in core reactivity due to decreased moderator temperature and core void fraction. The NRC staff's review covered (1) the sequence of events, (2) the analytical model, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; (~~2~~3) draft GDC-14 and 15, insofar as they require that the core protection systems be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; (~~3~~4) draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling; and (~~4~~5) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.4.4-5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the increase in core flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the AFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 9, 14, 15, 27, 28, and 32 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the increase in core flow event.

2.8.5.5 Inadvertent Operation of ECCS or Malfunction that Increases Reactor Coolant Inventory

Regulatory Evaluation

Equipment malfunctions, operator errors, and abnormal occurrences could cause unplanned increases in reactor coolant inventory. Depending on the temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could lead to fuel damage or overpressurization of the RCS. Alternatively, a power level decrease and depressurization may result. Reactor protection and safety systems are actuated to mitigate these events. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (~~2~~3) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.5.1-2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the inadvertent operation of ECCS or malfunction that increases reactor coolant inventory and concludes that the licensee's

analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the AFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 9, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the inadvertent operation of ECCS or malfunction that increases reactor coolant inventory.

2.8.5.6 Decrease in Reactor Coolant Inventory

2.8.5.6.1 Inadvertent Opening of a Pressure Relief Valve

Regulatory Evaluation

The inadvertent opening of a pressure relief valve results in a reactor coolant inventory decrease and a decrease in RCS pressure. The pressure relief valve discharges into the suppression pool. Normally there is no reactor trip. The pressure regulator senses the RCS pressure decrease and partially closes the turbine control valves (TCVs) to stabilize the reactor at a lower pressure. The reactor power settles out at nearly the initial power level. The coolant inventory is maintained by the feedwater control system using water from the condensate storage tank via the condenser hotwell. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (~~2~~3) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.6.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the inadvertent opening of a pressure relief valve event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the AFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 9, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the inadvertent opening of a pressure relief valve event.