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Docket No. 50-397

April 17, 1985

Mr. John B. Martin, Administrator
Region V Office of Inspection and Enforcement
US Nuclear Regulatory Commission
1450 Maria Lane
Walnut Creek, California 94596

Subject: WASHINGTON NUCLEAR PLANT - UNIT 2
FINAL STARTUP REPORT

References:
1) Plant Technical Specification 6.9.1.1
2) Interim Startup Report, Dated October 18, 1985
3) Interim Startup Report, Dated January 18, 1985

Reference 1) requires a Startup Report of Plant startup and power ascension testing to be submitted nine (9) months following initial reactor criticality. The first criticality of WNP-2 occurred on January 19, 1984 and reports were submitted on October 18, 1984 and January 18, 1985 which addressed testing through Test Condition 3. Subsequent reports are required to be submitted every three months until all testing leading to commencement of commercial operation has been reported.

The purpose of this correspondence is to provide you with the final test reports for those tests which FSAR Table 14.2-4 specified to be performed during the Power Ascension Test Program and special tests specific to WNP-2. WNP-2 has completed the Power Ascension Test Program and has met all Level 1 acceptance criteria. This report is being submitted as the final report and summarizes the entire testing program.

IE26

WASHINGTON NUCLEAR PLANT - UNIT 2
FINAL STARTUP REPORT

If there are any questions regarding this submittal, please do not hesitate to contact me.

CM Powers

C.M. Powers (M/D 927M)
WNP-2 Plant Manager

CMP:MRW:mm

Enclosure:
Report (2 copies)

cc: Director
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U.S. Nuclear Regulatory Commission
Washington, DC 20555
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DOCKET NO(S). 50-397

Mr. G. C. Sorensen, Manager
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SUBJECT:

WPPSS Nuclear Project No. 2

The following documents concerning our review of the subject facility are transmitted for your information.

- ☐ Notice of Receipt of Application, dated ____________.
- ☐ Draft/Final Environmental Statment, dated ____________.
- ☐ Notice of Availability of Draft/Final Environmental Statement, dated ____________.
- ☐ Safety Evaluation Report, or Supplement No. ______, dated ____________.
- ☐ Notice of Hearing on Application for Construction Permit, dated ____________.
- ☐ Notice of Consideration of Issuance of Facility Operating License, dated ____________.
- ☒ Monthly Notice; Applications and Amendments to Operating Licenses Involving no Significant Hazards Considerations, dated March 1985.
- ☐ Application and Safety Analysis Report, Volume ____________.
- ☐ Amendment No. ______ to Application/SAR dated ____________.
- ☐ Construction Permit No. CPPR- ______, Amendment No. ______ dated ____________.
- ☐ Facility Operating License No. ______, Amendment No. ______, dated ____________.
- ☐ Order Extending Construction Completion Date, dated ____________.
- ☐ Other (Specify) ____________

Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: See next page

OFFICE➤	LB#2/DL						
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DATE➤	04/1/85						

NRC FORM 318 (1/84) NRCM 0240

FINAL STARTUP REPORT

TABLE OF CONTENTS

		Page
1.0	INTRODUCTION	1
	1.1 Purpose	1
	1.2 Plant Description	1
	1.3 Startup Test Program Description	1
	1.4 Power Ascension Test Program Data Recording Methods	5
2.0	SUMMARY OF THE STARTUP TEST PROGRAM	9
	2.1 Startup Test Program Chronology of Significant Event	9
	2.2 Startup Test Program Scram History	9
	2.3 Power Ascension Test Completion Dates	9
	2.4 Test Results Documentation	9
3.0	SUMMARY OF TEST RESULTS	28
	3.1 General Power Ascension Test Description	28
	3.2 Test Number 1 Chemical and Radiochemical	28
	3.3 Test Number 2 Radiation Measurements	33
	3.4 Test Number 3 Fuel Loading	34
	3.5 Test Number 4 Full Core Shutdown Margin	37
	3.6 Test Number 5 Control Rod Drive System	39
	3.7 Test Number 6 SRM Performance and Control Rod Sequence	42
	3.8 Test Number 10 IRM Performance	43
	3.9 Test Number 11 LPRM Calibration	45
	3.10 Test Number 12 APRM Calibration	46
	3.11 Test Number 13 Process Computer	47
	3.12 Test Number 14 RCIC System	50

TABLE OF CONTENTS (Contd)

		Page
3.13 Test Number 16A	Reactor Vessel Temperature	54
3.14 Test Number 16B	Water Level Measurements	57
3.15 Test Number 17	System Expansion	59
3.16 Test Number 18	Core Power Distribution	64
3.17 Test Number 19	Core Performance	65
3.18 Test Number 20	Steam Production	67
3.19 Test Number 21	Core Power - Void Mode Response	70
3.20 Test Number 22	Pressure Regulator	72
3.21 Test Number 23A	Water Level Setpoint Changes	77
3.22 Test Number 23B	Loss of Feedwater Heating	83
3.23 Test Number 23C	Feedwater Pump Trip	84
3.24 Test Number 23D	Maximum Runout Capability	85
3.25 Test Number 24	Turbine Valves Surveillance	87
3.26 Test Number 25A	MSIV Functional Test	90
3.27 Test Number 25B	Reactor Full Isolation Test	96
3.28 Test Number 26	Relief Valves	97
3.29 Test Number 27	Turbine Trip and Generator Load Rejection	102
3.30 Test Number 28	Shutdown from Outside the Control Room	106
3.31 Test Number 29A	Flow Control - Valve Position Control	107
3.32 Test Number 29B	Recirculation Flow Loop Control	113
3.33 Test Number 30A	Recirculation System - One Pump Trip	122
3.34 Test Number 30B	Recirculation System - RPT Two Pump Trip	125
3.35 Test Number 30C	Recirculation System Performance	129
3.36 Test Number 30D	Recirculation Runback	133
3.37 Test Number 30E	Recirculation System - Non-Cavitation Verification	134

TABLE OF CONTENTS (Contd)

			Page
3.38	Test Number 31	Loss of Turbine Generator and Offsite Power	135
3.39	Test Number 33	Drywell Piping Vibration	139
3.40	Test Number 34	Reactor Internal Vibration	145
3.41	Test Number 35	Recirculation System Flow Calibration	146
3.42	Test Number 70	Reactor Water Cleanup System	149
3.43	Test Number 71	Residual Heat Removal System	152
3.44	Test Number 72	Drywell Atmosphere Cooling System	154
3.45	Test Number 73	Cooling Water System	156
3.46	Test Number 74	Offgas System	159
4.0	SPECIAL TESTS		163
4.1	Moderator Temperature Coefficient Measurement		163
4.2	In-Plant SRV Load Test		163
4.3	Sacrificial Shield Wall Verification		164
4.4	Loose Parts Monitoring Baseline		164

TABLE OF CONTENTS (Contd)

Tables

		Page
1-1	WNP-2 Plant Specification	2
1-2	Test Condition Region Definition	6
2-1	WNP-2 Significant Event of Startup Test Program	10
2-2	WNP-2 Startup Test Schedule	11
2-3	Maintenance Outage During WNP-2 Startup Testing	12
2-4	WNP-2 Scram Summary	14
2-5	Power Ascension Test Performance Dates	18
3-1	Chemical and Radiochemical Test Results	30
3-2	WNP-2 No Cleanup Test TC-6 Summary	31
3-3	Offgas Activity Data at 100% Power	32
3-4	WNP-2 Fuel Loading Problem Summary	35
3-5	CRD Scram Time Measurement Summary	41
3-6	CRD Scram Times from Power	41
3-7	SRM Performance	42
3-8	IRM Performance Data	44
3-9	Process Computer Program Verification Result	49
3-10	Summary of RCIC Cold Quick Start Test	52
3-11	Summary of RCIC System Control Settings	53
3-12	Summary of Selected Process Temperature Measurement	56
3-13	Summary of Water Level Measurement	58
3-14	Main Steam Thermal Expansion Displacement	61
3-15	Recirc. Loop Thermal Expansion Displacement	63
3-16	Summary of TIP Uncertainty Analysis Results	64
3-17	Core Performance Summary	66

TABLE OF CONTENTS (Contd)

Tables

		Page
3-18	Steam Production Data	68
3-19	Core Power - Void Mode Response Data	71
3-20	Final Pressure Control System Setting	74
3-21	Pressure Regulator Test Results	75
3-22	Feedwater Level Control System Settings	79
3-23	Feedwater Level Control Valve - Open Loop Flow Response Summary	79
3-24	Feedwater Pump Turbine - Open Loop Flow Response Summary	80
3-25	Feedwater Maximum Runout Capability Summary	86
3-26	MSIV Closure Times	91
3-27	SRV Performance Data	100
3-28	Summary of Turbine Trip and Generator Load Rejection Test Results	105
3-29	Valve Position Control Data 'A' Loop - TC3	109
3-30	Valve Position Control Data 'B' Loop - TC3	110
3-31	Valve Position Control Loop 'A' Response Summary - TC6	111
3-32	Valve Position Control Loop 'B' Response Summary - TC6	112
3-33	Recirc. Flow Control System Final Settings	116
3-34	Flow Control Loop Response Summary	117
3-35	Neutron Flux Control Loop Response Summary	118
3-36	Scram Margin Verification (100% L.L)	119
3-37	Recirculation One Pump Trip Results	124
3-38	Hanford RPT Coastdown Requirement	128
3-39	Recirculation System Performance	132
3-40	Loss of A-C Test Results	137

TABLE OF CONTENTS (Contd)

Tables

		Page
3-41	Chronology of Significant Events During the Loss of A-C Test	138
3-42	Steady State Drywell Piping Vibration Data - Main Steam Lines	141
3-43	Steady State Drywell Piping Vibration Data - Recirculation Loops	142
3-44	Transient Drywell Piping Vibration Data - Main Steam Lines	143
3-45	Transient Drywell Piping Vibration Data - Recirculation Loop	144
3-46	Recirculation Flow Instrumentation Adjustments	147
3-47	RHR System Performance Data	153
3-48	RHR Heat Exchanger Data	153
3-49	Local Drywell Air Temperatures	155
3-50	Offgas System Design Parameters and Results	160

TABLE OF CONTENTS (Contd)

Illustrations

Figure		Page
1-1	Power - Flow Map	7
1-2	Plant Computer System	8
2-1	Power Histogram of WNP-2 Startup Test Program	22
3-1	Cycle 1 Cold Shutdown Margin vs Core Average Exposure	38
3-2	Rated Steam Output Curve	69
3-3	Pressure Regulator Static Data	76
3-4	Feedwater Turbine 'A' Control Gain Curve	81
3-5	Feedwater Turbine 'B' Control Gain Curve	82
3-6	Turbine Valve Surveillance Results	89
3-7	MSIV Surveillance: Peak Reactor Dome Pressure	92
3-8	MSIV Surveillance: Peak Neutron Flux	93
3-9	MSIV Surveillance: Peak Heat Flux	94
3-10	MSIV Surveillance: Peak Steamline Flow	95
3-11	Bypass Valve Calibration Curve	101
3-12	Recirculation Flow Control Linearization - Loop A	120
3-13	Recirculation Flow Control Linearization - Loop B	121
3-14	Hanford RPT Coastdown Requirement Compared with Test Data From Loop A	126
3-15	Hanford RPT Coastdown Requirement Compared with Test Data From Loop B	127
3-16	Total Core Flow vs Total Loop Flow	130
3-17	Core dP vs Total Core Flow	131
3-18	Jet Pump Flow Distribution	148
3-19	RWCU Bottom Head Flow Indication	151

1.0 INTRODUCTION

1.1 Purpose

The purpose of this report is to provide a concise summary of the Power Ascension Test Program conducted on WNP-2. Included in this report are sections which cover general plant and power ascension test program descriptions and specific test results.

1.2 Plant Description

Washington Public Power Supply System Nuclear Project No. 2 (WNP-2) is located within the Hanford Reservation of the Department of Energy (DOE), Benton County, Washington, approximately 12 miles north of the city of Richland. The station utilizes a Direct-Cycle Forced Circulation Boiling Water Reactor (BWR) provided by General Electric (GE). WNP-2 is rated at 3323 MWt with design power of 3468 MWt. The gross electrical power output is rated approximately at 1150 MWe with design output of 1205 MWe.

The Nuclear Steam Supply System designed and supplied by General Electric Co. is designated as a BWR/5 product line with a 251-inch inside-diameter reactor pressure vessel and 764 fuel assemblies in the reactor core. Significant plant parameters are specified for informational purpose in Table 1-1.

1.3 Startup Test Program Description

After the Preoperational Test Phase has been completed, the Power Ascension Test Phase begins. The Power Ascension Test Phase begins with fuel loading and extends to commercial operation. This phase is subdivided into the following four parts:

a. Open Vessel Testing (Fuel loading and low power physics tests)

b. Initial heatup to rated pressure and temperature

c. Power Ascension tests

d. Warranty demonstration.

The tests conducted during the Power Ascension Phase consist of Major Plant Transients, Stability Tests, and a remainder of tests which are directed towards demonstrating correct performance of the nuclear boiler and numerous auxiliary plant systems while at power. Certain tests may be identified with more than one class of tests.

The general objectives of the Power Ascension Test Phase are as follows:

a. to achieve an orderly safe initial core loading;

b. to accomplish all testing and measurements necessary to determine that the approach to initial criticality and subsequent power ascension is safe and orderly;

TABLE 1-1

WNP-2 PLANT SPECIFICATION

PARAMETER	VALUE
Rated Core Thermal Power (MWt)	3323
Rated Total Core Flow (MLB/HR)	108.5
Reactor Dome Pressure (PSIA)	1020
Rated Feedwater Temperature (°F)	420
Total Steam Flow (MLB/HR)	14.296
Vessel Diameter (IN)	251
Total Number of Jet Pumps	20
Number of Nozzles per Jet Pump	5
Jet Pump Design M-Ratio	2.33
Rated Recirculation Pump Flow (GPM)	47200
Number of Control Rods	185
Number of Fuel Bundles	764
Fuel Type	8x8PR
Core Active Fuel Length (IN)	150
Cladding Thickness (IN)	0.032
Channel Thickness (IN)	0.100
MCPR Operating Limit	1.24
MCPR Safety Limit -- Normal	1.06
-- Single Loop Operation	1.07
MLHGR (KW/ft)	13.4
Turbine Governor Valve Mode	Partial ARC
Turbine Bypass Valve Capacity (% NBR)	25
Safety Relief Valve Capacity (% NBR)	111.5
Number of SRV's	18
Recirculation Flow Control Mode	Flow Control Valves

c. to conduct low power physics tests sufficient to ensure that test criteria have been met;

d. to conduct initial heatup and hot functional testing so that hot integrated operation of all systems is shown to meet test acceptance criteria;

e. to conduct an orderly and safe power ascension program, with requisite physics and systems testing, to ensure that the plant operating at power meets test acceptance criteria; and

f. to conduct a successful warranty demonstration program.

The overall program was comprised of open vessel testing and six test conditions which are for the most part characterized by differences in plant and power/core flow conditions. The power-flow map illustrated in Figure 1-1 displays the test conditions established during power ascension testing.

To assist in the evaluation of proper plant performance from the test results obtained during the Startup Test Program, a set of criteria for each test has been established. These criteria are a result of a combination of factors such as safety analysis assumptions, engineering judgements or expectations, and contractural commitments. Safety concerns are considered Level 1 while other considerations are typically Level 2 or Level 3. Satisfactory compliance with these criteria assure that the plant meets the stated purpose of the Startup Test Program. Portions of various tests are conducted to provide baseline data and as such do not employ an acceptance criteria. Definition of these Level 1, Level 2 and Level 3 criteria and required action in the event of a violation are defined as follows:

Level 1 Criteria

The values of process variables assigned in the design of the plant and equipment are included in this category. If a Level 1 criterion is not satisfied, the plant will be placed in a hold condition which is satisfactory and safe based upon prior testing. A Nonconformance Report (NCR) will be made to document the situation. Plant operating or test procedures or the Technical Specifications may guide the decision or the direction to be taken. Startup testing compatible with this hold condition may be continued. Resolution of the problems must immediately be pursued by appropriate equipment adjustment or through offsite engineering support if needed. Following resolution, the applicable test portion must be repeated to verify that Level 1 requirement is satisfied. A description of the problem resolution must be included in the NCR documenting the successful test.

Level 2 Criteria

The limits considered in this category are associated with expectations in regard to the performance of the system. If a Level 2 criterion is not satisfied, plant operating and startup testing plans would not necessarily be altered. An investigation of the related adjustments, as well as the measurement and analysis method would be initiated. If all Level 2 requirements in a test are ultimately met, there is no need to document a temporary failure in the test report, unless there is an educational benefit involved. Following resolution, the applicable test portion must be repeated to verify that the Level 2 requirement is satisfied. If a certain controller-related Level 2 criterion is not satisfied after a reasonable effort, then the control engineers may choose to document the result with a full explanation of their recommendations. All Level 2 criteria violations shall be documented on a Plant Deficiency Report (PDR). This report must discuss alternatives of action, as well as the concluding recommendation, so that it can be evaluated by all related parties.

The conduct of the Startup Test Program was governed by Volume 8 Section 2.0 "Power Ascension Test Program Administration", of WNP-2 Plant Procedure Manual (PPM). This procedure outlines the relationships between the respective startup ogranizations, test execution, as well as documentation review and approval of procedures. Documents such as USNRC Regulatory Guides 1.68, Rev. 2, 1978, Operating License, Technical Specifications, Plant Procedure Manual, Chapter 14 of the WNP-2 Final Safety Analysis Report and Startup Test Specifications formed integral parts of the Controlling Power Ascension Test Program Administration Procedure. The Startup Test Specification (STS) provides the recommended test program required to demonstrate safe efficient plant operation.

The Power Ascension Test procedures were prepared by WNP-2 Technical Staff with the technical assistance of General Electric site personnel. The procedures specified the required initial conditions, procedural steps, analytical techniques, and supporting information for the performance of each test. General Electric site personnel, as the Nuclear Steam Supply System vendor representative, has reviewed the Power Ascension Test procedures to further verify their adequacy and to confirm that the General Electric Test Specification requirements were fully met. For those procedures dealing with the BOP, Burns and Roe Inc. as the A/E, reviewed and concurred with the test procedure content.

Within this framework of procedures and controlling documents, the Startup Test Program was satisfactorily conducted demonstrating safe operation, proper system performance as well as adequate plant operation using the normal operating procedures and personnel.

A training program was conducted during the PAT in compliance with the position stated in the WNP-2 FSAR, Appendix B, TMI Issue I.G.1. The program provided 1) instruction on the content and goals of the PATP, 2) for selection of special tests designed to demonstrate abnormal scenarios, and 3) for utilization of the experience gained during the test program.

1.4 Power Ascension Test Program Data Recording Methods

The primary mechanism for recording data for tests which provided control systems tuning/optimization, system performance demonstrations, and plant response to major transients was the Transient Data Acquisition System (TDAS). This system receives approximately 950 analog and digital plant signals. The signals are hard-wired from either the control room or remote locations, such as drywell piping displacement measurement signals, to TDAS remote modules. The signal output from the remote modules is transmitted through fiber-optic cables to the TDAS Central Control Unit (CCU). The TDAS Central Control Unit (CCU) controls the monitor, record, and data transfer functions. Data is transfered from the TDAS CCU to a minicomputer for data reduction, analysis and display. The hardware configuration of the plant computer system is shown in Figure 1-2.

Other secondary means of data acquisition included plant process computer, magnet tape recorder, and manual data taking. All of these methods provide documentation of the successful completion of WNP-2 Power Ascension Test Program.

TABLE 1-2

TEST CONDITION REGION DEFINITIONS

Test Condition	Power Flow Map Region and Notes
1	Before main generator synchronization from 5 to 20 percent thermal power and operating on recirculation pump low frequency power supply (25% pump speed).
2	After main generator synchronization from 50 to 75 percent control rod lines, at or below the analytical lower limit of Master Flow Control mode and with the lower power corner within bypass valve capacity.
3	From 50 to 75 percent control rod lines above 80 percent core flow, and within maximum allowed recirculation control valve position.
4	On the natural circulation core flow line within $\pm$ 5 percent of the intersection with the 100 percent power rod line.
5	From the 100 percent loadline to 5 percent below the 100 percent loadline and between minimum flow at rated recirculation pump speed (minimum valve position) to 5 percent above the analytical lower limit of the automatic flow control range.
6	Within 0 to -5 percent of rated 100 percent thermal power, and within $\pm$ 5 percent of rated 100 percent core flow rate.
Increased Core Flow Region	Power between cavitation interlock and 100% power, core flow between 100% and the constant flow control valve position that gives 110.5% core flow at 100% power.

POWER FLOW MAP

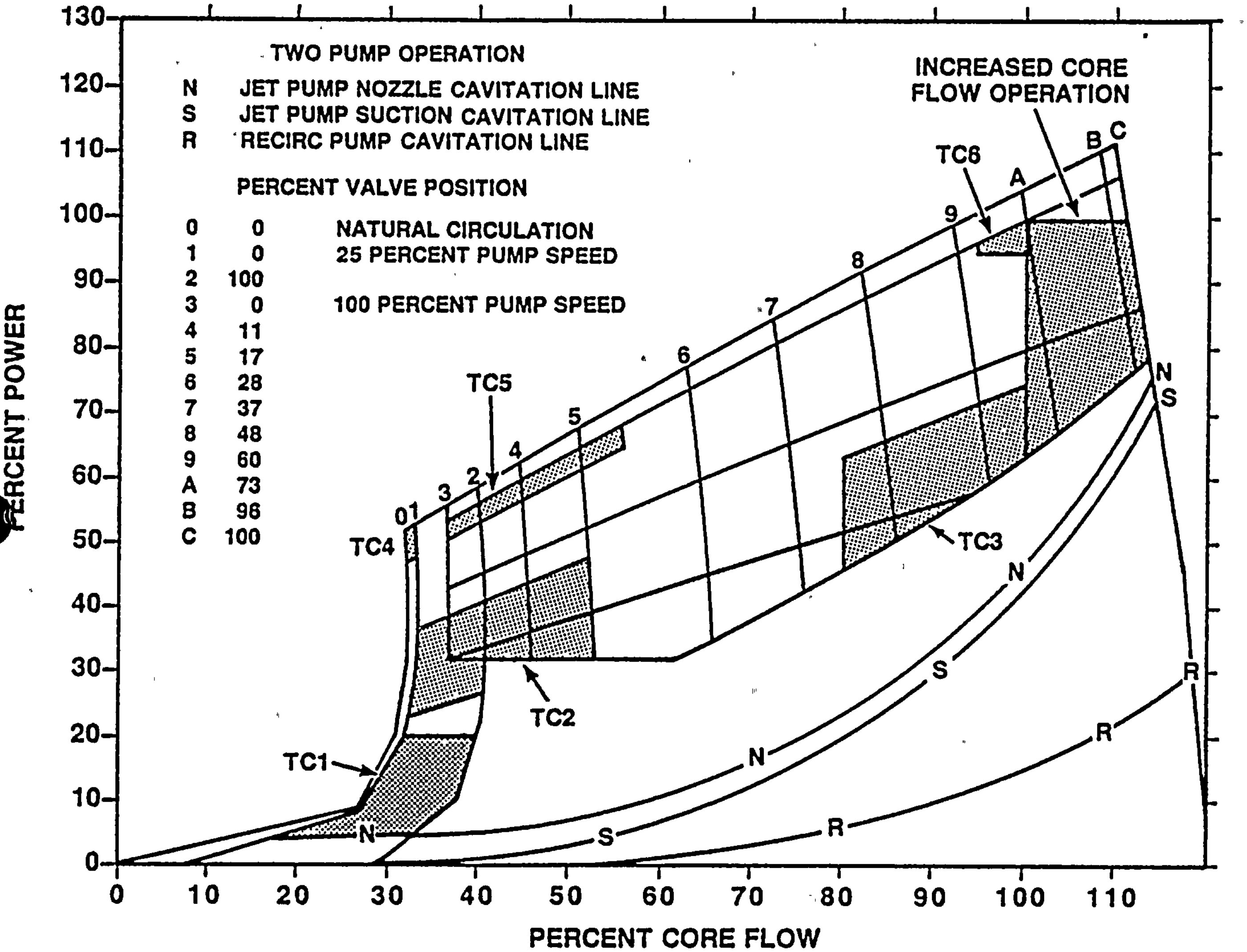

Figure 1-1

PLANT COMPUTER SYSTEM

PRESENT COMPUTER CONFIGURATION

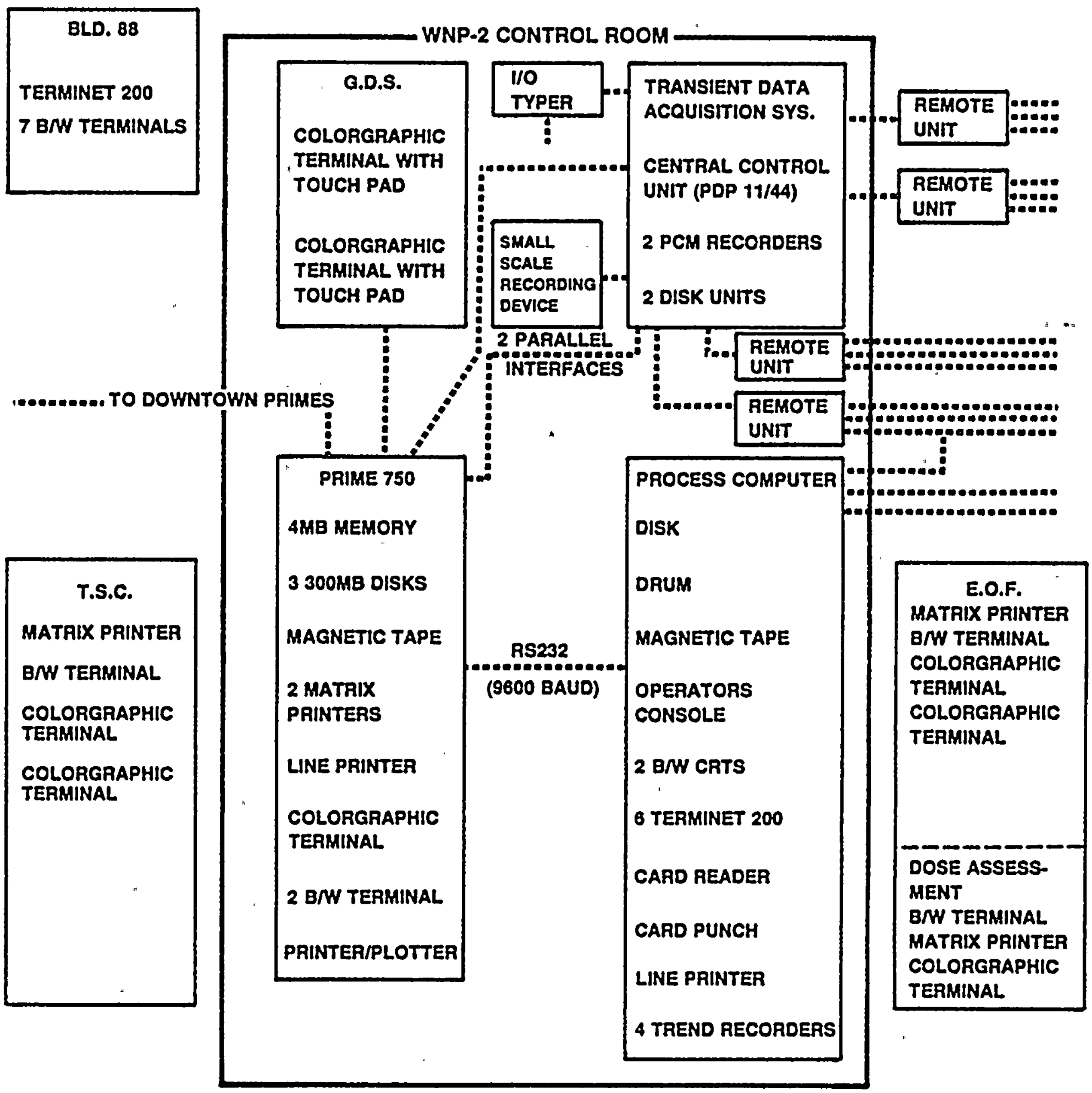

Figure 1-2

2.0 SUMMARY OF THE STARTUP TEST PROGRAM

2.1 Startup Test Program Chronology of Significant Events

The WNP-2 Startup Test Program began on December 25, 1983 with commencement of Fuel Loading and concluded on December 12, 1984 with the completion of the Warranty Run. A chronology of significant events of the Startup Test Program is presented in Table 2-1. A total of 354 days, slightly shorter than the industrial average of 388 days, were required to complete all the testing originally scheduled for 161 days. Table 2-2 presents the comparision of the actual testing days with the scheduled testing days at each test condition. Several major problems which resulted in significant delays are summarized in Table 2-3. A Power Histogram of the whole power ascension testing is displayed in Figure 2-1.

2.2 Startup Test Program SCRAM History

Twenty-eight scrams occurred during the Startup Test Program. Table 2-4 contains a brief description of each scram including date, cause and whether it was planned or unplanned. In all, 22 unexpected scrams occurred while 6 were a normal course of the Startup Test Program.

2.3 Power Ascension Test Completion Dates

As an aid to illustrate Startup Test Program progress, Table 2-5 contains Power Ascension Test completion dates as a function of test condition. Comparison of the dates in this table with the tests specified in the Table 14.2-3 of WNP-2 Final Safety Analysis Report indicates that all required tests were completed.

2.4 Test Results Documentation

During the course of the Power Ascension Test Program an Apparent Test Results form was generated immediately following the completion of a test at each test condition to document the acceptance of the test results. These ATR's were reviewed by Plant Operation Committee (POC) and approved by the Plant Manager prior to the POC giving authority to proceed to the next test condition.

A Nonconformance Report (NCR) was prepared for any Level 1 criteria failure and all Level 2 criteria violations were documented on a Plant Deficiency Report (PDR). These NCRs and PDRs were reviewed and resolved in accordance with the requirements of Section 1.3.12 "Plant Problems" of the Plant Procedure Manual.

In addition, a large data file, Transient Traces and Procedure File, which were generated during the test program, were retained as per PPM 1.6.0 "Plant Record Control". All these data are summarized in the following sections of this report.

TABLE 2-1

WNP-2 SIGNIFICANT EVENT OF STARTUP TEST PROGRAM

Commence Fuel Loading	12-25-83
Fuel Load Complete	1-12-84
Initial Criticality	1-19-84
Initial Heatup to Rated Pressure	4-19-84
Begin Test Condition 1 Testing	5-07-84
Initial Main Turbine Roll	5-08-84
Begin Test Condition 2 Testing	5-26-84
Initial Generator Synchronization	5-27-84
Begin Test Condition 3 Testing	8-10-84
Begin Test Condition 5 Testing	10-17-84
Begin Test Condition 4 Testing	10-20-84
Being Test Condition 6 Testing	10-30-84
Begin Warranty Run	12-08-84
Commercial Operation	12-13-84

TABLE 2-2

WNP-2 STARTUP TEST SCHEDULE

Test Condition	Total Days	Maintenance Outage	Plant Recovery	Actual Test Days	Schedule Days
Open Vessel	36			36	30
Preparation To Heatup	72	72			
Heatup	27	3	3	21	7
TC 1	19	8	4	7	6
TC 2	74	39	13	22	22
TC 3	69	28	13	28	12
TC 4	8	3	4	1	3
TC 5	5		2	3	2
TC 6	39	11	17	11	16
Warranty Run	5			5	7
Total	354	164	56	134	105*

*Exclude the scheduled maintenance outage (15 days) and scheduled heatup preparation (41 days).

TABLE 2-3

MAINTENANCE OUTAGES DURING WNP-2 STARTUP TESTING

Date	T.C.	Days	Description
4/23/84 - 4/26/84	H/U	3	o D/W cooling modification
5/13/84 - 5/15/84	TC-1	2	o DEH hydraulic modification
5/20/84 - 5/26/84	TC-1	6	o Main turbine GV flange leak
6/4/84 - 6/11/84	TC-2	7	o Main condenser baffle plate modification
			o RHR-V-41B testable check valve repair
6/20/84 - 7/1/84	TC-2	11	o Bypass valve #3 repair
			o RHR pump 'B' repair
7/10/84 - 7/30/84	TC-2	20	o Div. I and II DG bearing replacement
8/7/84 - 8/8/84	TC-2	1	o MSR drain tank handhole flange leakage
8/12/84 - 8/14/84	TC-3	2	o Main condenser tube leak
8/16/84 - 8/28/84	TC-3	12	o Main condenser tube leak
			o Repair FW piping hanger
9/10/84 - 9/17/84	TC-3	7	o Repair FW piping hanger
			o RHR-V-41B testable check valve repair
			o Main turbine GV LVDT repair
10/1/84 - 10/6/84	TC-3	5	o Turbine bypass valve hydraulic system troubleshoot
			o Main steam relief valve vacuum breakers repair
			o RPS MG set 'A' high vibration
10/8/84 - 10/10/84	TC-3	2	o Main turbine bypass valves hydraulic modification

TABLE 2-3 (Contd)

MAINTENANCE OUTAGES DURING WNP-2 STARTUP TESTING

Date	T.C.	Days	Description
10/20/84 - 10/23/84	TC-4	3	o Repair RFW-FCV-15
			o Repair MSR drain line hanger
11/10/84 - 11/19/84	TC-6	9	o Drywell pipe whip restraints inspection
			o FW heater level controller modification
			o Main condenser expansion joint modification
			o Main condenser tube leak check
11/27/84 - 11/29/84	TC-6	2	o FW heater 4C tube leak repair
	TOTAL	92	

TABLE 2-4

WNP-2 SCRAM SUMMARY

SCRAM	DATE	CONDITION	CAUSE	DESCRIPTION
84-01	4-23-84	Heatup	IRM F Hi-Hi and CRD scram disch. volume high level	Neutron flux spike caused by cold FW injection during FW startup level control valve RFW-FCV-10 tuning in conjunction with existing 1/2 scram on channel 'A' due to scram discharge volume Channel Functional Test. (Unplanned)
84-02	5-02-84	Heatup	Manual	Manual scram to measure control rod B-2 sequence scram times. (Planned)
84-03	5-13-84	TC-1	Reactor High Pressure	Initial turbine roll at 1640 RPM, No. 2 bypass valve failed open and other three bypass valves over compensated to cause the reactor high pressure. (Unplanned)
84-04	5-17-84	TC-1	Manual	Loss of reactor feedwater 'A' speed resulted in rapid decreasing reactor water level. (Unplanned)
84-05	5-18-84	TC-1	Reactor High Pressure	Turbine bypass valve failed closed on low DEH hydraulic oil pressure when the main turbine is manually tripped. (Unplanned)
84-06	5-19-84	TC-1	Reactor Low Water Level	Surveillance test of high drywell pressure switches caused actuation of the load shedding logic. The control power to the reactor feedwater turbine control power was lost causing feedwater pump turbine runback to minimum speed which caused a low level scram. (Unplanned)

TABLE 2-4 (Contd)

SCRAM	DATE	CONDITION	CAUSE	DESCRIPTION
84-07	5-20-84	TC-1	Manual	Perform PPM 8.2.28 shutdown from outside the control room. (Planned)
84-08	5-28-84	TC-2	Reactor Low Water Level	Loss of condensate booster pumps and feedwater pumps on low suction pressure during the condensate filter demineralizer differential pressure controller repair. (Unplanned)
84-09	5-29-84	TC-2	Turbine Control Valve Fast Closure	During the main turbine overspeed protection (OPC) test at 20% power, the turbine first stage pressure switches actuated and removed the less than 30% power RPS trip bypass. (Unplanned)
84-10	6-01-84	TC-2	Reactor High Pressure	Turbine bypass valves failed closed due to DEH electronic circuit card failure. (Unplanned)
84-11	6-13-84	TC-2	Reactor Low Water Level	Loss of condensate booster pumps and reactor feedwater pump on low suction pressure when RFW-FCV-15 (condensate cleanup return to condenser) failed open. (Unplanned)
84-12	8-01-84	TC-2	Turbine Trip and Throttle Valve Closure	Turbine first stage pressure switch actuated during the main turbine control transfer at 1650 RPM and less than 30% power RPS trip bypass was removed. (Unplanned)
84-13	8-07-84	TC-2	Turbine Control Valve Fast Closure	Perform PPM 8.2.31, Loss of Turbine Generator and Offsite Power. (Planned)

TABLE 2-4 (Contd)

SCRAM	DATE	CONDITION	CAUSE	DESCRIPTION
84-14	8-09-84	TC-3	Reactor Low Water Level	Loss or reactor feedwater control power due to under-voltage trip of SM-7 during circulating water pump start. (Unplanned)
84-15	8-12-84	TC-3	Manual	Reactor was manually scrammed due to high reactor water conductivity. (Unplanned)
84-16	8-16-84	TC-3	Main Steam Line High Rad (Channel B2) and APRM upscale (Channel A)	Removal of APRM Flow Unit 'A' power supply during the surveillance caused APRMs A,C,E upscale trip in conjunction with existing 1/2 scram on main steam line high radiation surveillance test. (Unplanned)
84-17	9-10-84	TC-3	RPS Power Fuses blown during execution of planned test	RPS fuses F14A,B,C,D were blown following incorrect installation of a test box to be used to simulate RRC transfer to 15 Hz. (Unplanned)
84-18	10-01-84	TC-3	Turbine Trip and Throttle Valve Closure	Perform PPM 8.2.27, Turbine Generator Trip. (Planned)
84-19	10-04-84	TC-3	Reactor Low Water Level	Turbine bypass valves failed open then closed to cause the reactor low water level during DEH transducer troubleshooting. (Unplanned)
84-20	10-06-84	TC-3	MSIV's Closure	Mode switch was placed in RUN with main steam line pressure less than 831 psig. (Unplanned)
84-21	10-13-84	TC-3	Manual	Manual scram due to loss of DEH control of main turbine valves. (Unplanned)

TABLE 2-4 (Contd)

SCRAM	DATE	CONDITION	CAUSE	DESCRIPTION
84-22	10-20-84	TC-5	MSIV's Closure	Main steam line pressure decreased to below 831 psig during the pressure regulator testing. (Unplanned)
84-23	10-23-84	TC-5	IRM Hi-Hi	Cold feedwater injection due to FW startup level control valve failed open. (Unplanned)
84-24	10-28-84	TC-6	Reactor Low Water Level	Loss of condensate booster pumps and feedwater pumps on low suction pressure due to a condensate system flow transient. (Unplanned)
84-25	11-10-84	TC-6	MSIV's	Perform PPM 8.2.25 MSIV's Closure. (Planned)
84-26	11-27-84	TC-6	Turbine Control Valve Fast Closure	Reduction of condenser vacuum caused a turbine trip. (Unplanned)
84-27	12-02-84	TC-6	Turbine Control Valve Fast Closure	Perform PPM 8.2.27 Turbine Generator Trip. (Planned)
84-28	12-03-84	TC-6	Reactor Low Water Level	Momentary loss of level control during transfer from FW startup level control valve RFW-FCV-10 to FW turbine speed control. (Unplanned)

TABLE 2-5

POWER ASCENSION TEST PERFORMANCE DATES

TEST NO.	TEST NAME	OPEN VESSEL	HEATUP	TC-1	TC-2	TC-3	TC-4	TC-5	TC-6	WARRANTY RUN
1.	Chemical & Radiochemical	12/22/84	4/23/84	5/23/84	7/04/84	8/31/84			11/3/84	
2.	Radiation Measurement	12/23, 12/24/84	4/21, 4/26/84	5/8, 5/9/84	6/16/84	8/30, 9/01/84			11/4-11/5/84	
3.	Fuel Loading	12/25/83-1/12/84								
4.	Full Core Shutdown Margin	1/21/84								
5.	Control Rod Drive System	12/25/83-1/16/84	4/11-5/07/84		8/07/84	10/1/84			11/10/84, 12/02/84	
6.	SRM Performance & Control Rod Sequence	1/19-1/21/84	4/10-4/13/84	5/07/84	6/15/84			10/18/84	11/30/84	
10.	IRM Performance	1/21/84	4/26/84	5/16, 5/26/84	7/31, 8/1, 8/5/84					
11.	LPRM Calibration		4/22-5/04/84	5/08, 5/09/84	7/05/84				11/03/84	
12.	APRM Calibration		4/11/84	5/09/84	7/05/84	9/02, 9/19/84		10/18/84	11/03/84	12/11/84
13.	Process Computer	3/28/84	5/1, 5/04/84	5/9, 5/11/84		9/19, 9/20 9/26/84			11/3-12/1/84	
14.	RCIC		4/29, 5/1, 5/03/84	5/2, 5/27, 5/16, 6/02/84	7/1, 7/5, 7/09/84					
16A.	Selected Process Temperatures		5/02/84	5/11/84	7/06/84	9/10, 9/25, 9/26/84	10/20, 10/25/84		12/06/84	

TABLE 2-5 (Contd.)

POWER ASCENSION TEST PERFORMANCE DATES

TEST NO.	TEST NAME	OPEN VESSEL	HEATUP	TC-1	TC-2	TC-3	TC-4	TC-5	TC-6	WARRANTY RUN
16B.	Water Level Ref. Leg Temperature		5/04/84	5/09, 5/11/84	7/05, 8/05/84	9/30/84	10/20/84	10/18/84	12/03/84	
17.	System Expansion	1/19, 1/23/84	4/11, 4/23/84	5/11/84	6/18, 6/19, 7/06, 8/06/84	9/19, 9/25, 9/26, 10/01/84			11/24/84	
18.	Core Power Distribution					9/02, 9/03/84			11/03, 11/05/84	
19.	Core Performance			5/09/84	6/06/84	8/30/84	10/20/84	10/18/84	11/08/84	12/10/84
20.	Steam Production									12/08-12/13/84
21.	Core Power Void Mode Response						10/25/84	10/20/84		
22.	Pressure Regulator	2/28-3/19/84		5/11/84	6/18-6/19/84	10/16/84	10/25/84	10/19/84	11/10/84	
23.	Feedwater System									
	A. Water Level Setpoint Change		5/05/84	5/09/84	7/02, 8/05/84	9/08, 10/15 10/16/84	10/20/84	10/18/84	11/20, 11/26/84	
	B. Heater Loss								12/05/84	
	C. Feedwater Pump Trip								12/08/84	
	D. Maximum Runout Capability			5/09/84				10/19/84	12/04/84	
24.	Turbine Valve Surveillance								11/08, 11/09/84	

TABLE 2-5 (Contd.)

POWER ASCENSION TEST PERFORMANCE DATES

TEST NO.	TEST NAME	OPEN VESSEL	HEATUP	TC-1	TC-2	TC-3	TC-4	TC-5	TC-6	WARRANTY RUN
25.	Main Steam Isolation Valves									
	A. Each Valve/One Valve		4/20/84/ 4/30/84		6/18/84			10/20/84	11/09/84	
	B. Full Isolation								11/10/84	
26.	Relief Valves									
	A. Flow Demonstration					9/26/84				
	B. Operational		4/14/84			9/26/84				
27.	Turbine Trip/Generator Load Reject				6/19/84	10/01/84			12/02/84	
28.	Shutdown From Outside the Control Room		5/20/84						11/27/84	
29.	Recirculation Flow Control System			5/10-5/11/84		9/28-10/12/84			11/23-11/25/84	
30.	Recirculation System									
	A. One-Pump Trip					9/25/84			12/07/84	
	B. RPT Trip of Two Pumps					9/26/84				
	C. System Performance				8/04/84	9/17-9/27/84	10/20-10/25/84		11/08-12/07/84	
	D. Recirc. Pump Runback					9/09/84				
	E. Recirculation System Cavitation				8/06/84	9/26/84				

TABLE 2-5 (Contd.)

POWER ASCENSION TEST PERFORMANCE DATES

TEST NO.	TEST NAME	OPEN VESSEL	HEATUP	TC-1	TC-2	TC-3	TC-4	TC-5	TC-6	WARRANTY RUN
31.	Loss of Turbine-Generator and Offsite Power				8/08/84					
33.	Drywell Piping Vibration					8/06, 9/19, 9/25, 9/26, 10/01/84			11/10, 11/27 12/02/84	
34.	Reactor Internal Vibration									
	A. Vibration Measurement					9/10, 9/25, 9/26, 9/27/84			12/07, 12/08/84	
35.	Recirculation Flow Calibration					9/08/84			11/04, 11/08/84	
70.	Reactor Water Cleanup System		4/30-5/01/84							
71.	Residual Heat Removal System								12/11/84	
72.	Drywell Atmosphere Cooling		4/30/84		6/16, 7/31/84	8/31/84			11/3, 11/24/84	
73.	Cooling Water System		5/01/84						12/12/84	
74.	Offgas System	3/02-4/05/84	5/03, 5/04/84	5/08/84		9/04/84			11/04/84	

POWER HISTOGRAM OF WNP-2 STARTUP TEST PROGRAM (1 OF 6)

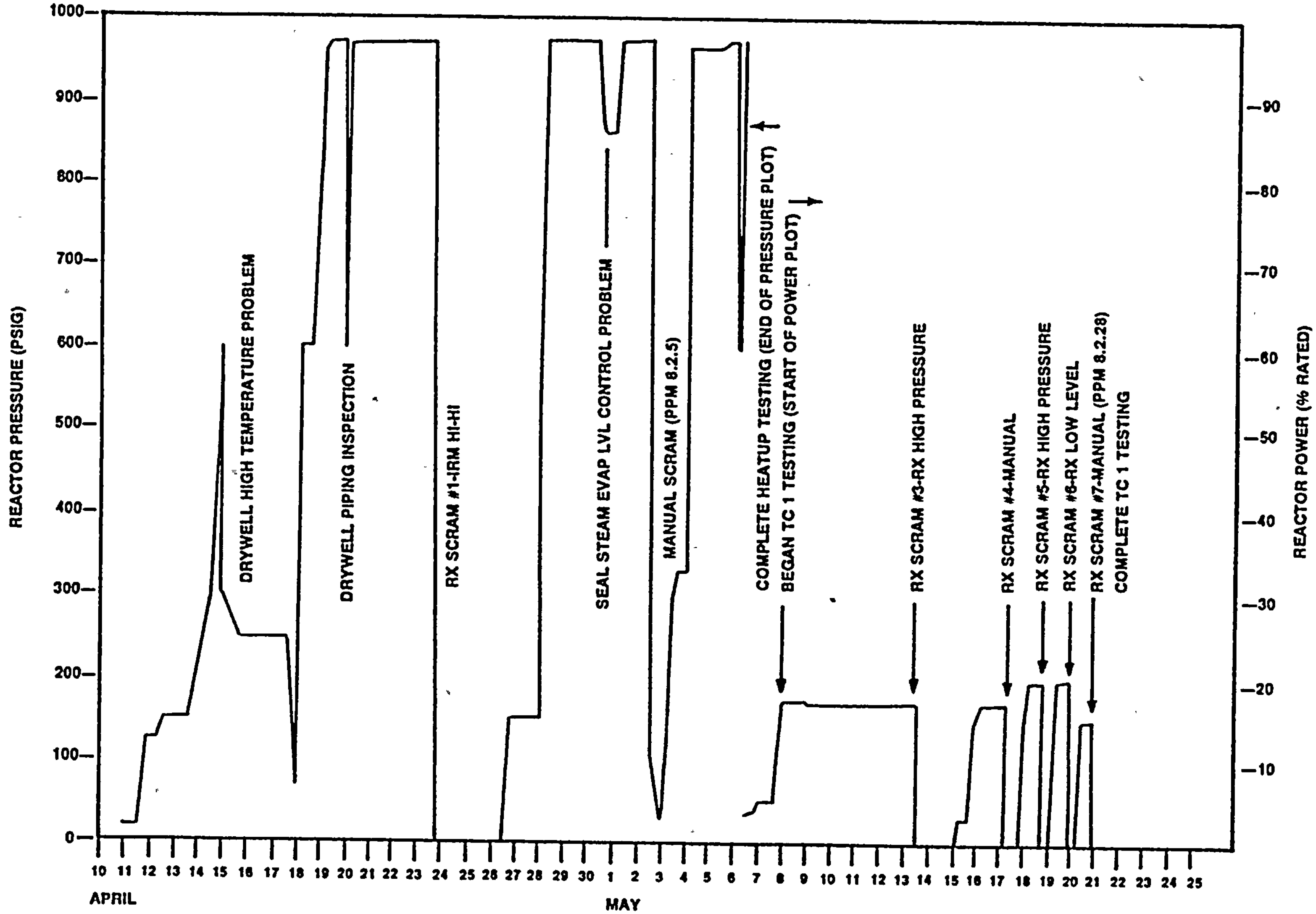

Figure 2-1

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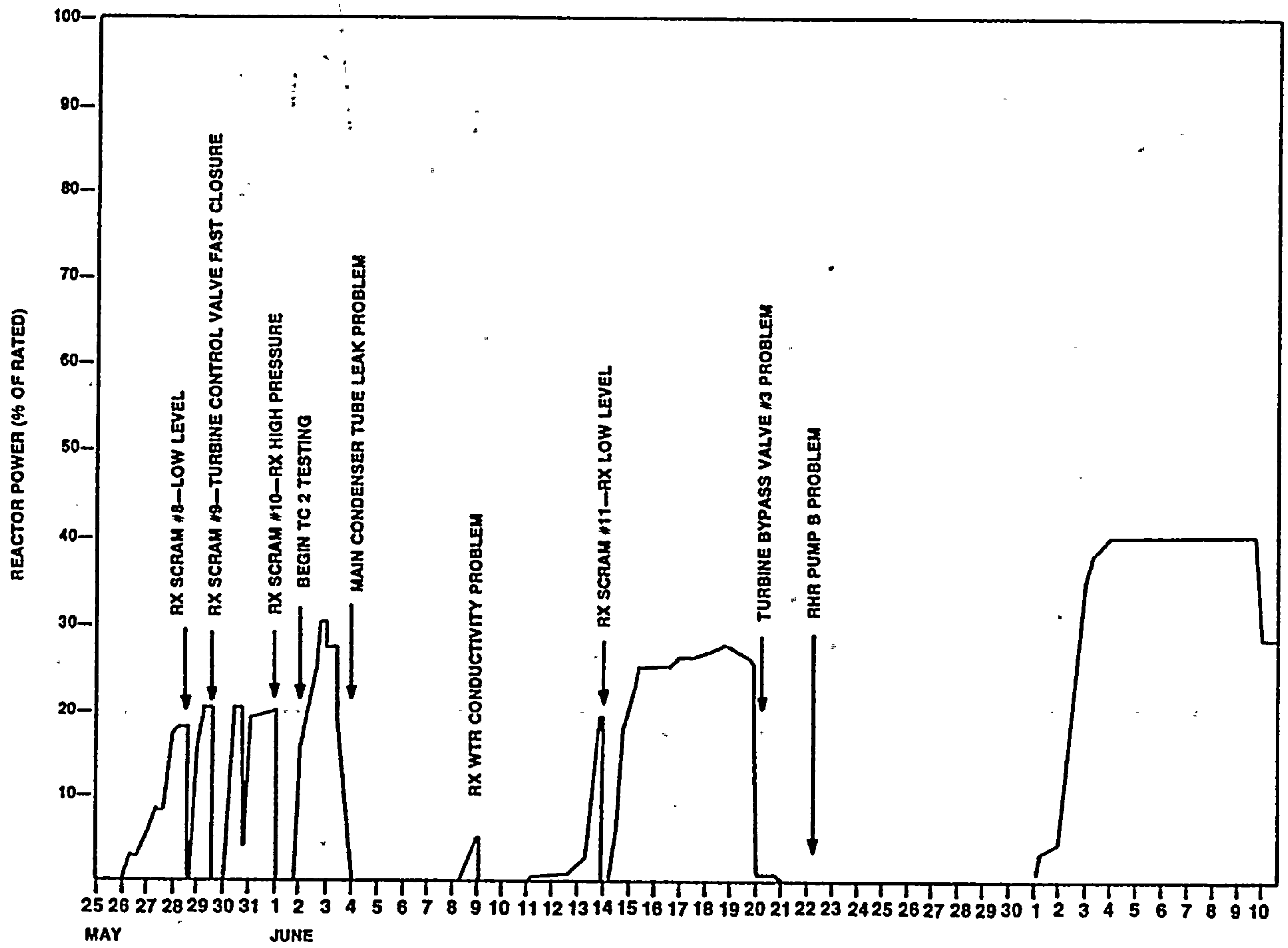
POWER HISTOGRAM OF WNP-2 STARTUP TEST PROGRAM (2 OF 6)



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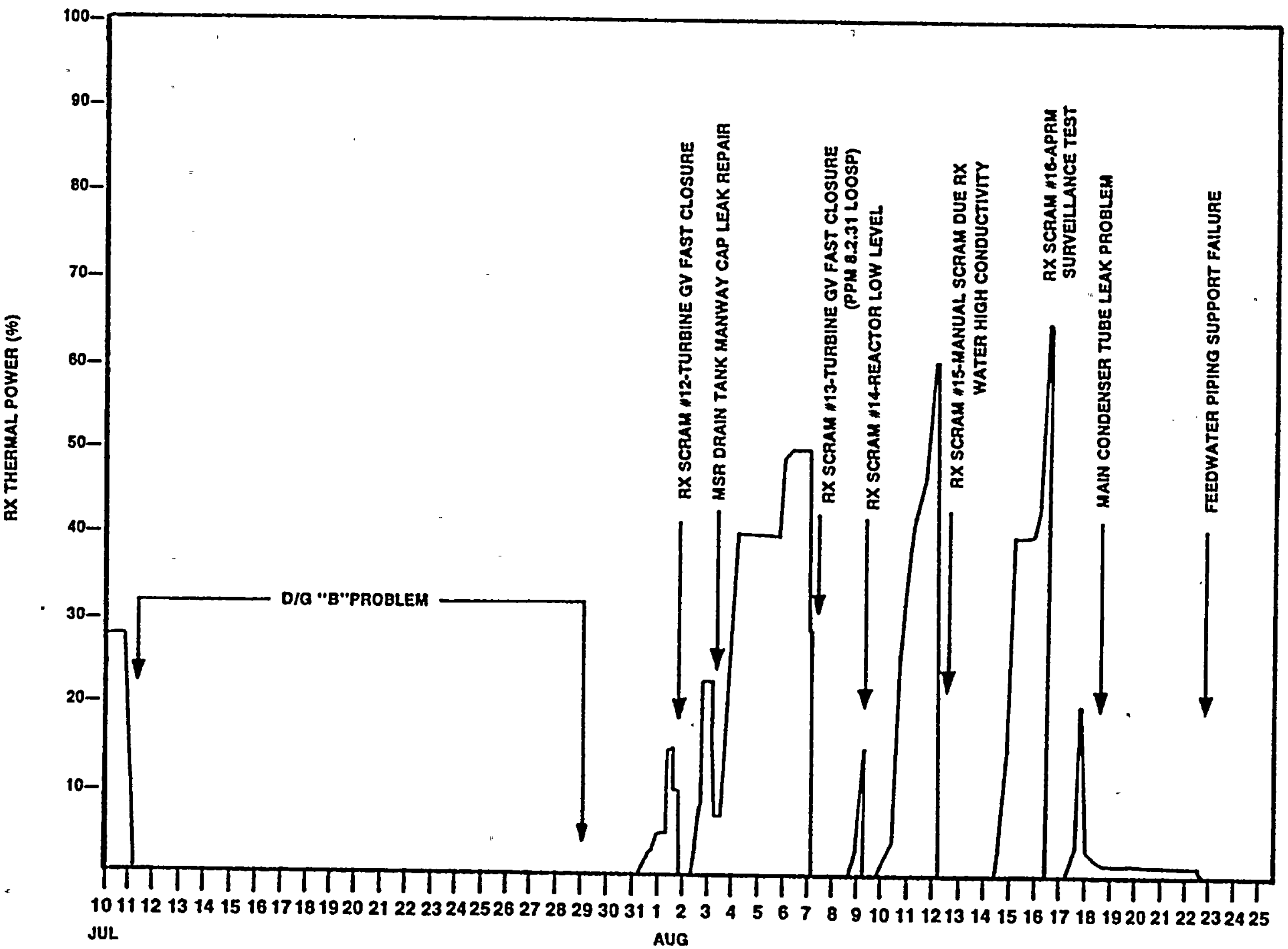
POWER HISTOGRAM OF WNP-2 STARTUP TEST PROGRAM (3 OF 6)



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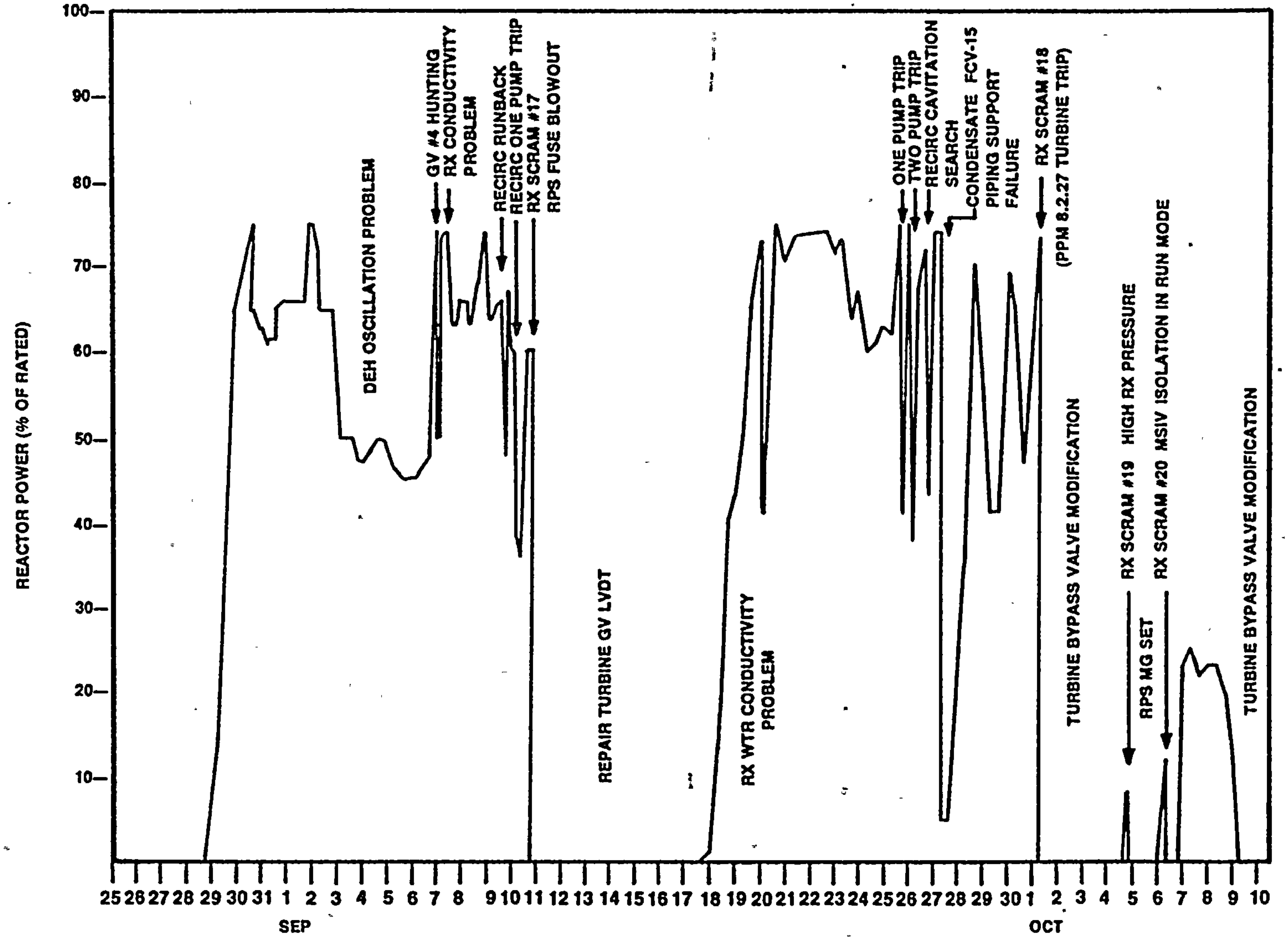
POWER HISTOGRAM OF WNP-2 STARTUP TEST PROGRAM (4 OF 6)



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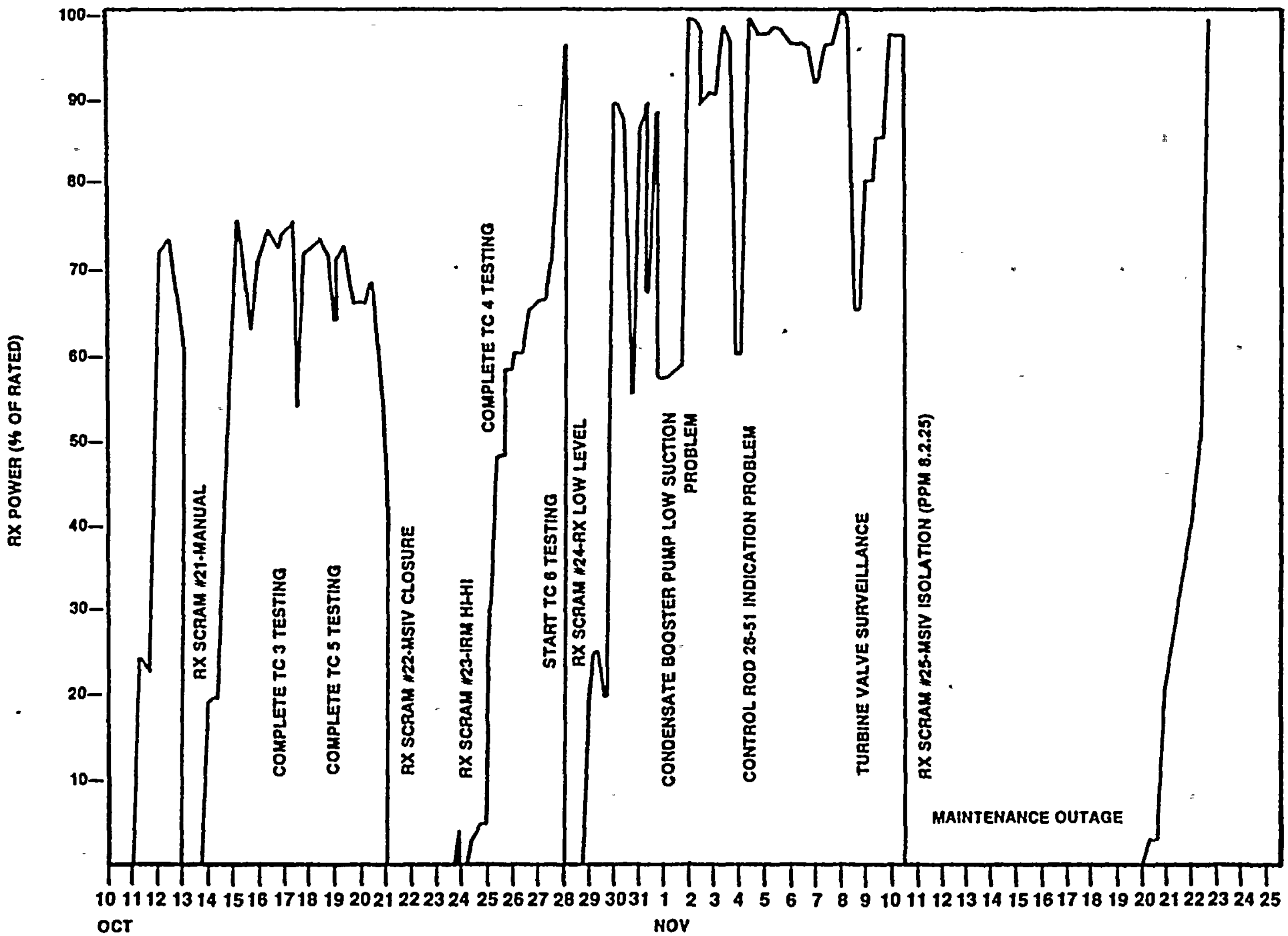
POWER HISTOGRAM OF WNP-2 STARTUP TEST PROGRAM (5 OF 6)



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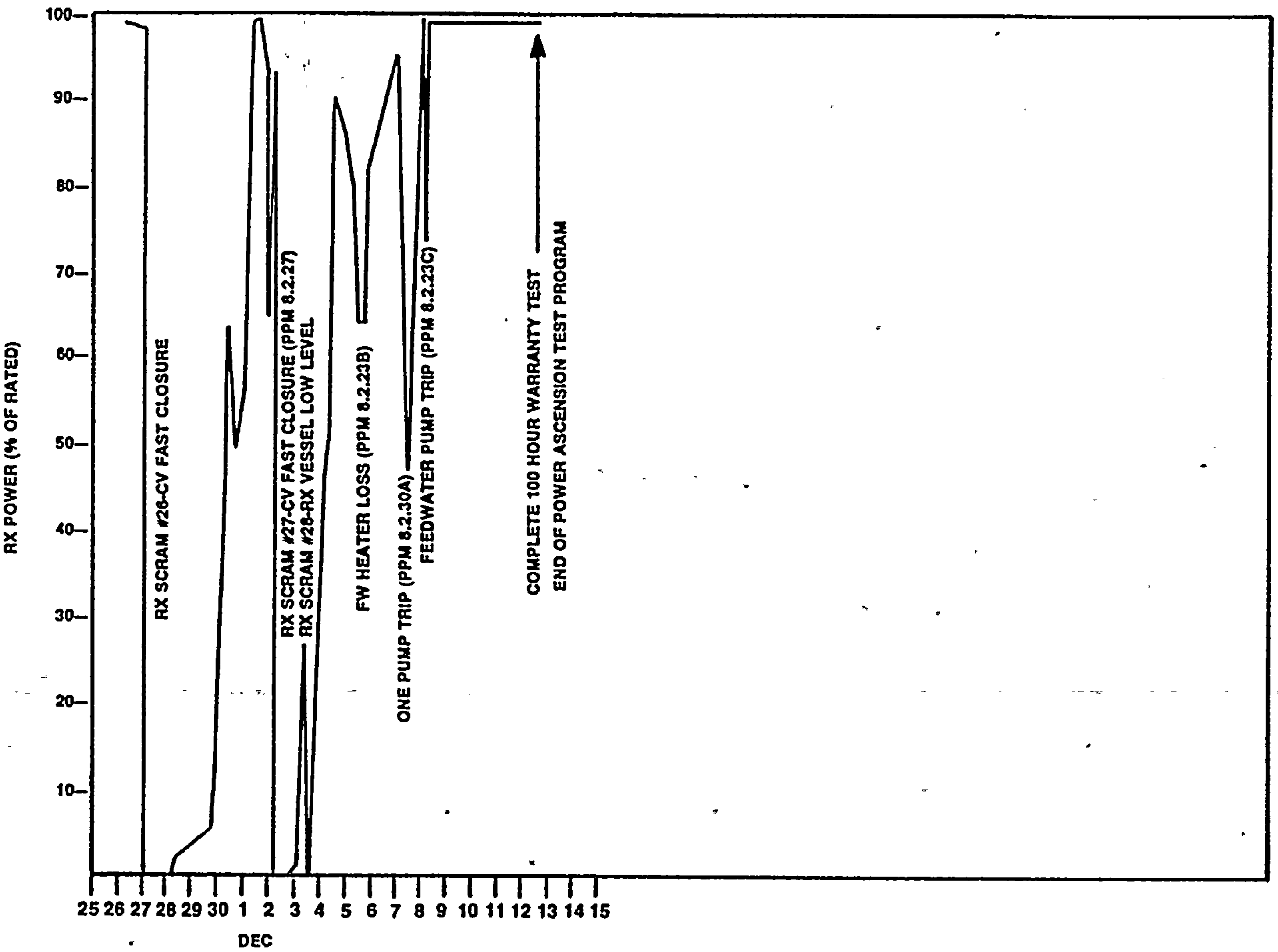
POWER HISTOGRAM OF WNP-2 STARTUP TEST PROGRAM (6 OF 6)



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3.0 SUMMARY OF TEST RESULTS

3.1 General Power Ascension Test Description

In the following sections a summary of test purpose and the associated acceptance criteria, test results and discussion of each power ascension test performed are presented.

3.2 Test Number 1 - Chemical and Radiochemical

3.2.1 Purpose

The principal objectives of this test are a) to secure information on the chemistry and radiochemistry of the reactor coolant, and b) to determine that the sampling equipment, procedures and analytic techniques are adequate to supply the data required to demonstrate that the chemistry of all parts of the reactor system meet specifications and process requirements.

Specific objectives of the test program include evaluation of fuel performance, evaluations of demineralizer operations by direct and indirect methods, measurements 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 specific chemical tests.

3.2.1.1 Level 1 Criteria

Chemical factors defined in the Technical Specifications must be maintained within the limits specified.

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

Water quality must be known at all times and should remain within the guidelines of the Water Quality Specifications.

3.2.1.2 Level 2 Criteria

Not applicable.

3.2.2 Results

A summary of the reactor water, condensate, and feedwater quality throughout the Startup Test Program is presented in Table 3-1.

Steam separator-dryer performance was determined during the no-cleanup test performed at TC-6. The result is summarized in Table 3-2.

Offgas activity data is tabulated in Table 3-3 and indicates acceptable fuel performance.

3.2.3 Discussion

Throughout the Startup Test Program, water quality exceeded recommended limits in the reactor and conductivity was measured at values exceeding the limits of 1.0 umho/cm during TC-2 and TC-3. Reactor power was also limited in several cases due conductivity approaching to the limiting value.

The conductivity problem that occurred in the early stage of Startup Test was caused by main condenser tube leakage and inadequate performance of the condensate filter demineralizers. Remedial actions were taken and reactor water conductivity was reduced and maintained during subsequent test conditions.

The Reactor Water No-Cleanup Test scheduled at TC-2 was not completed due to a RWCU pump failure while the test was in progress, preventing the RWCU system from being returned to service. This test was repeated and completed successfully at TC-6. The steam separator-dryer performance was evaluated during the test. The $7.3 \times 10^{-3}\%$ moisture carryover in the main steam compares favorably with the contract warranty value of 0.3%.

Data collected during operation of the offgas system did not provide a means of correlating activity flow rates to the indicated exposure rate on the offgas rad. monitors (both post and pretreatment). The activity levels in the offgas system was too low to provide sufficient indication on the rad. monitors. The HP/Chemistry Department has provided a means to continue this monitoring process for future correlation efforts.

TABLE 3-1

CHEMICAL AND RADIOCHEMICAL TEST RESULTS

Test Condition	Open Vessel	Heatup	TC 1	TC 2	TC 3	TC 6
Reactor Thermal Power	0	2%	18%	45%	65%	100%
Reactor Water						
Conductivity, umho/cm	0.80	0.24	0.21	1.48	0.79	0.47
Chloride, ppm	0.015	0.015	0.015	0.015	0.018	0.015
Turbidity or Insolubles, ppm	0.20	0.009	0.02	0.75	0.20	0.017
Silica, ppm	0.009	0.52	0.14	0.280	0.77	0.354
Boron, ppm	5E-3	0.012	0.056	0.010	0.010	0.010
Crud - Iron, ppb	-	9.7	9.0	220	115	17.2
Gamma Isotopic & Gross Activity						
Iodine -131 (uCi/ml)	-	1.2E-7	3.8E-7	5E-6	1.0E-5	4.6E-6
Iodine -133 (uCi/ml)	-	5.5E-6	2.6E-6	1E-4	1.6E-4	7.6E-5
Filtrate - 2 Hr, cpm/ml	-	1.1E3	1.4E3	8.4E3	8.0E3	2.7E4
Crud - 2 Hr, cpm/ml	-	7.8E1	2.0E3	2.8E3	1.8E3	7.1E3
Condensate						
Conductivity, umho/cm	1.89	0.24	0.16	0.28	0.21	0.084
Chloride, ppm	0.015	0.015	0.015	0.015	0.015	0.015
Isoluble Iron, ppb	0.0875	22	0.81	20	844	9.7
Condensate Demin Effluent						
Conductivity, umho/cm	0.67	0.096	0.092	0.083	0.07	0.062
Iron (Insoluble), ppm	-	1.0	1.5	4.1	28.5	1.2
Oxygen, ppb	0.080	150	20	50	45	60
Feedwater						
Conducitivty, umho/cm	0.79	0.073	0.06	0.087	0.121	0.079
Iron (Insoluble), ppb	1.68	2.3	0.81	12.2	6.8	2.1
Iron (Soluble), ppb	-	-	-	-	-	0.31
Dissolved O_2, ppb	-	100	20	40	45	80

TABLE 3-2

HANFORD-2 NO-CLEANUP TEST AT TC-6: DATA SUMMARY

Reactor Data:

Power:	3090 $\pm$ 120 megawatt thermal (100% = 3323 MwtT)
Core Flow:	1.00 $\pm$ 0.02 x 10^8 lb/hr
Feedwater:	1.33 $\pm$ 0.02 x 10^7 lb/hr
Reactor Pressure:	994 $\pm$ 3 PSIG
RWCU Flow:	8.42 x 10^4 lb/hr (rated at 1.34 x 10^5 lb/hr)
Reactor Water Level:	20.0 $\pm$ 1.5 inches above reactor
Reactor Temperature:	520 $\pm$ 3°F

Results (Summary):

RWCU Half Life:	4.75 hr (predicted 4.9 hours, rated 3.1 hr)
Clean-up:	0.146 hr^{-1} (predicted 0.141 hr^{-1}, rated, 0.224 hr^{-1})
Dynamic Reactor Volume:	5.77 x 10^5 lbs (predicted 5.97 x 10^5)
Conductivity Increase:	0.056 umho/hr/cm
Background Conductivity:	0.15 umho/cm
Equilibrium Conducitvity:	0.480 umho/cm
Clean-up Rate with RWCU Off:	0.032 hr^{-1}
Na^+ Input Rate:	0.96 ppb/hr
Cl^- Input Rate:	2.1 x 10^{-2} ppb/hr
Na^+ Concentration In Reactor:	6.5 ppb
Cl^- Concentration In Reactor:	0.15 ppb
Na^+ Concentration In Feedwater:	4.3 x 10^{-2} ppb
Cl^- Concentration In Feedwater:	9.4 x 10^{-4} ppb
I-133 Transport:	0.25%
I-131 Transport:	0.23%
% Carryover:	7.3 x 10^{-3}%
Neutron Flux:	3.35 x 10^{12} neutrons/cm^2/sec
Steam Quality:	99.993%

TABLE 3-3

OFFGAS ACTIVITY DATA AT 100% POWER

Plant Parameters	Data
Activity at SJAE (uCi/sec)	101
Activity at Post Filter (uCi/sec)	Background
Offgas Flow (SCFM)	74
Pre-Treatment Radiation Monitor	6 mR/hr
Post-Treatment Radiation Monitor	'A' 85 cpm 'B' 33 cpm (60 to 90 cpm is background)

3.3 Test Number 2 - Radiation Measurements

3.3.1 Purpose

The principle objectives of the radiation measurement test are as follows:

1. To determine the background radiation levels in the plant environs prior to operation for use as baseline data for activity build-up.

2. To monitor radiation at selected plant locations during initial power ascension and subsequent power operation to assure the protection of personnel.

3. To provide sufficient data on exposure rate and dose equivalent rates to allow comparison of the actual dose rates with design dose rates throughout the plant.

3.3.1.1 Level 1 Criteria

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 10CFR20, NRC General Design Criteria.

3.3.1.2 Level 2 Criteria

Not applicable.

3.3.2 Test Results

The maximum observed radiation measurements occurred at 100% power. The points were R5-15 (Main Reactor Water Cleanup Suction Isolation, above RWCU pump room), and T2-2 (Column F6 & E7, 471' ele.) in which the readings were 1000 mR/hr and 800 mR/hr, respectively.

3.3.3 Discussion

A complete standard survey of all designated locations was performed in the open vessel condition to determine the background radiation levels in the plant environs prior to startup. During the initial heatup and at 20, 50, 75 and 100% power, gamma dose rate, and where appropriate, neutron dose rate measurements were performed. This established contamination buildup trends and identified areas of unexpected high radiation. Several areas in the reactor building and one point in the turbine building had higher than desired gamma fields. Administrative action was taken to limit access to those areas as defined in 10CFR20.

3.4 Test Number 3 - Fuel Loading

3.4.1 Purpose

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

3.4.1.1 Level 1 Criteria

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

3.4.1.2 Level 2 Criteria

Non applicable.

3.4.2 Results

Partial core shutdown margin was demonstrated with 144 bundles loaded. Minimum shutdown margin was 4.09% delta k/k with the analytically strongest rod fully withdrawn.

3.4.3 Discussion

Fuel loading commenced with the loading of the first bundle at 0656 on 12/25/83. Loading was completed 19 days later with the seating of the last of the 764 fuel bundles at 1700 on 1/12/84. After the first 144 bundles had been loaded in a 12 x 12 array at the center of the core a partial core shutdown margin demonstration was successfully performed. The test also verified that the shutdown margin will be met throughout the loading process.

Control rod functional and subcriticality check were performed on all 185 control rods as each fuel cell was loaded. After the core was fully loaded the seating, orientation and location of all bundles was verified to be correct.

Overall, the fuel load went smoothly taking a total of 446 hours compared to the schedule of 367 hours, based on 50 bundles per day. Table 3-4 lists the problems encountered that became critical path, adding up to a total of 115 hours. Most of the problems were of a mechanical or electrical nature involving either the refueling bridge or the neutron monitoring system.

TABLE 3-4

WNP-2 FUEL LOADING PROBLEM SUMMARY

DATE	# OF HRS DELAY	PROBLEM DESCRIPTION	RESOLUTION
12-25-83	40	Loss of refueling grapple "loaded" interlock. Surveillance Test Procedure inadequacy.	Recalibrated the load cell to account for the weight of the mast while submerged. Revised Surveillance Test Procedure.
12-27-83	6	SRM "C" high and period alarm indications on H13-P603 were inoperable.	Wiring change/correction required.
12-28-83	12	Refueling bridge interlock precluded bridge from moving over core.	Replaced damaged cable which lost continuity when over-stretched.
12-30-83	3	Lost IRM "D" indication on P603 panel. Problem was traced to a loose grounding wire in the drawer	Restored IRM "D" by securing the grounded wire.
12-31-83	1	RWM became inoperable and haulted the partial shutdown margin demonstration.	Reset by re-initiating the process computer.
12-31-83	1	FLC "C" cable hose clamp turned loose.	Secured the hose clamp.
01-01-84	3	Bundle LJT 422 had nick indication on the nose piece.	Further investigation revealed that it was acceptable.
01-02-84	7	The flexible drive shaft for the grapple was disassembled.	Examination discovered that a lock pin was missing. A pin was installed.
01-03-84	6	The newly installed pin on the flexible drive shaft was inspected and found to be moving out of coupling hold.	Restored the pin with a clamp to hold the pin in place.
01-04-84	4	After 43% of core loaded, attempted to transfer FLC to SRM detector, but without success.	The neutron flux level was too low for SRM's to register acceptable count rate. Decided to continue loading using with FLC's.

WNP-2 FUEL LOADING PROBLEM SUMMARY

DATE	# OF HRS DELAY	PROBLEM DESCRIPTION	RESOLUTION
01-09-84	7	Refueling bridge power failed.	Power was restored.
01-10-84	5	Refueling grapple slack cable interlock setpoint drifted.	Reset load switch setpoint and performed interlock check.
01-11-84	11	SRM "A" failed to respond to neutron flux in the core.	Replaced the detector and put SRM "A" in service.
01-12-84	9	Bundle LJT 795 would not go in. Suspected fuel support casting was misaligned.	Inspected the core bottom with underwater camera to clarify any suspected misalignment. Successfully loaded the bundle by changing the cell loading sequence.

3.5 Test Number 4 - Full Core Shutdown Margin

3.5.1 Purpose

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

3.5.1.1 Level 1 Criteria

The shutdown margin of the fully loaded, cold (68°F or 20°C), xenon-free core occurring at the most reactive time during the cycle must be at least 0.38% delta k/k with the analytically strongest rod (or its reactivity equivalent) withdrawn. If the shutdown margin is measured at some time during the cycle other than the most reactive time, compliance with the above criterion is shown by demonstrating that the shutdown margin is 0.38% delta k/k plus an exposure-dependent correction factor which corrects the shutdown margin at that time to the minimum shutdown margin.

3.5.1.2 Level 2 Criteria

Criticality should occur within ± 1% delta k/k of the predicted critical.

3.5.2 Results

The full core shutdown margin with the analytically strongest rod withdrawn was determined to be 2.716% delta k/k. A 0.015% delta k/k difference was observed between the actual and theoretical critical eigenvalues. All test criteria were satisfied.

3.5.3 Discussion

The fully loaded core was made initially critical on 1/19/84 by withdrawing control rods in the B sequence. A 750 second period was observed with Group 3 Rod 26-11 withdrawn to notch position 04. The SRM/IRM overlap was verified and the shorting links installed to remove the reactor protection system out of the noncoincidence scram mode. Later, the reactor was again brought supercritical on 1/21/84 by withdrawing control rod 26-27 to position 08. SRM measurements were taken to determine the reactor period. The average period was found to be 96 seconds with moderator temperature at 85°F. Actual core keff was then determined with the correction terms for period and moderator temperature.

Figure 3-1 illustrates core reactivity with the analytically strongest rod fully withdrawn as a function of core average exposure. The figure indicates an additional margin of 0.26% delta k/k should be added to the 0.38% delta k/k criteria to obtain the required shutdown margin at the time of most reactive core. However, the measured shutdown margin has adequately satisfied the acceptance criteria.

CYCLE EXPOSURE (GWD/ST) CYCLE 1 COLD SHUTDOWN MARGIN VS. CORE AVERAGE EXPOSURE

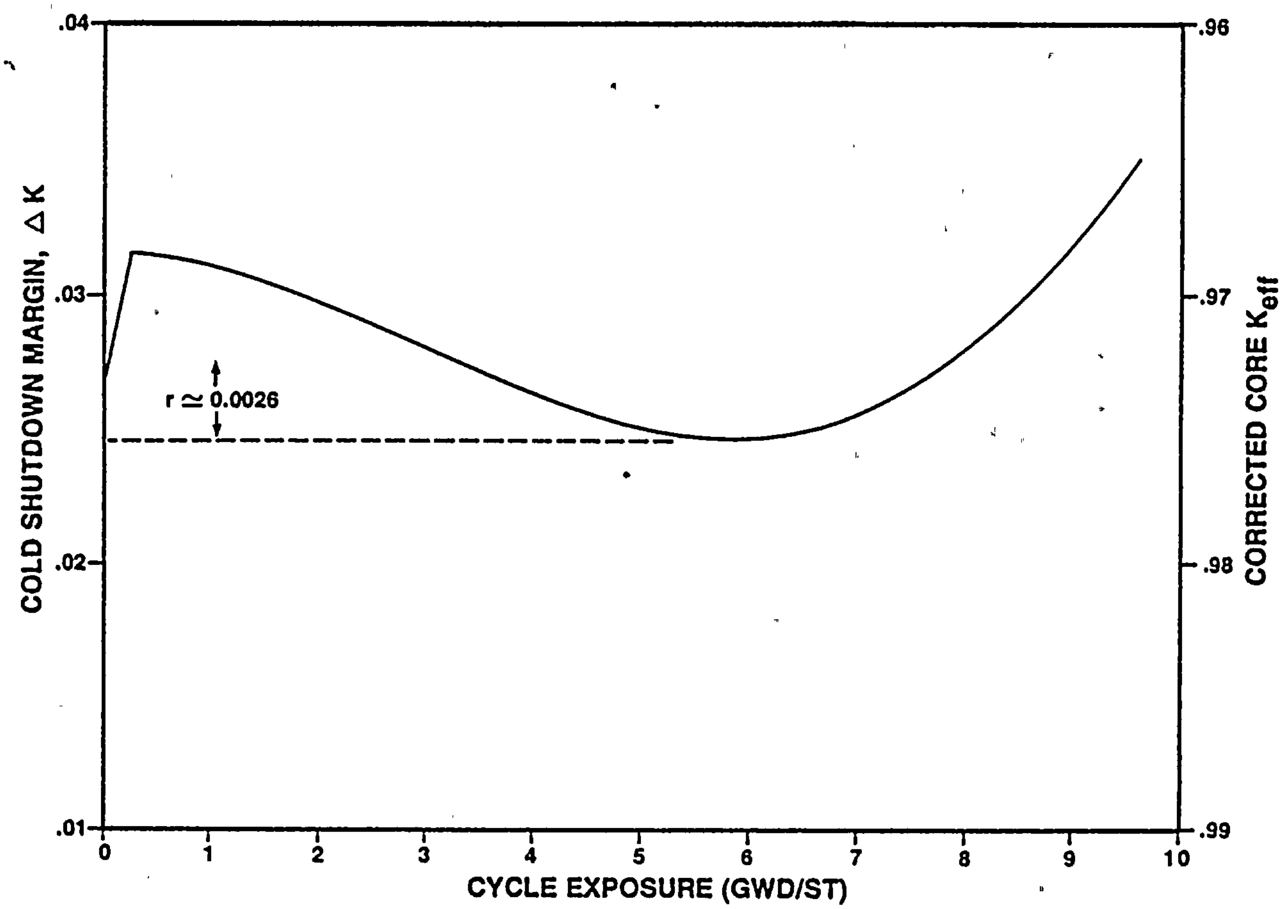

Figure 3-1

3.6 Test Number 5 - Control Rod Drive System

3.6.1 Purpose

The primary objectives of the control rod drive system tests are as follows:

1. to demonstrate that the control rod drive system operates properly over the full range of primary coolant temperatures and pressures from ambient to operating; and

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

3.6.1.1 Level 1 Criteria

Each CRD must have a normal withdrawal speed less than or equal to 3.6 inches per second indicated by a fully 12 foot stroke in greater than or equal to 40 seconds.

The mean scram time of all operable CRDs with functioning accumulators must not exceed the following times: (Scram time is measured from the time the pilot scram valve solenoids are de-energized).

Position Inserted From Fully Withdrawn	Scram Time (Seconds)
45	0.430
39	0.868
25	1.936
05	3.497

The mean scram time of the three fastest CRDs in a two by two array must not exceed the following times: (Scram time is measured from the time the pilot scram valve solenoids are de-energized).

Position Inserted From Fully Withdrawn	Scram Time (Seconds)
45	0.455
39	0.920
25	2.052
05	3.706

The scram insertion time of each control rod from full out to position 5, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

3.6.1.2 Level 2 Criteria

Each CRD must have a normal insertion or withdrawal speed of 3.0 ± 0.6 inches per second indicated by a full 12-foot stroke in 40 to 60 seconds.

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

3.6.2 Test Results

A summary of the single rod scram tests conducted during open vessel and heatup test phases is presented in Table 3-5.

Table 3-6 presents a summary of measured scram times of the four CRDs selected during heatup testing to be monitored in conjunction with planned reactor scrams.

3.6.3 Discussion

During and after fuel loading all 185 CRDs were functionally and scram tested. Normal withdraw/insert times were found to range from 40 to 57 seconds. Each CRD was friction tested with the maximum variation observed for the entire stroke length equal to 13 psid.

During heatup CRDs selected on the basis of previously demonstrated slow scram or functional speeds were scram tested to confirm that no significant binding from thermal expansion of the core components occurred. In addition, the scram speeds of selected CRDs were monitored for measurable changes during the course of the Startup Test Program. No significant change in operating characteristics of the CRD system was observed indicating proper system performance.

Table 3-5

CRD SCRAM TIME MEASUREMENT SUMMARY

Reactor Pressure / Accumulator Pressure	0.0 psig / 1050 psig	600 psig / 1050 psig	800 psig / 1050 psig	960 psig / 1050 psig	960 psig / 0
Mean Time to Notch Position:					
45	0.26 sec	0.287 sec	0.294 sec	0.276 sec	0.292 sec
39	0.44 sec	0.537 sec	0.653 sec	0.608 sec	0.613 sec
25	0.90 sec	1.239 sec	1.530 sec	1.413 sec	1.367 sec
05	1.64 sec	2.586 sec	2.627 sec	2.545 sec	2.600 sec
Number of Rods Tested	185	8	8	185	8

Table 3-6

CRD SCRAM TIMES FROM POWER

Reactor Power (MWt)%	957(29%)	2226(67%)	3227(97%)	3183(96%)
Reactor Pressure (psig)	909	963	980	1001
Date Performed	9/7/84	10/1/84	11/10/84	12/2/84
Mean Time to Notch Position (Sec)				
45	0.290	0.304	0.230	0.323
39	0.612	0.604	0.518	0.623
25	1.347	1.322	1.210	1.373
05	2.503	2.421	2.277	2.490
Number of Rods Tested	4	4	4	4
Initiating Event	STI-27GLR	STI-27TT	STI-25MSIV	STI-27GRL

GRL = Generator Load Rejection
TT = Main Turbine Trip

3.7 Test Number 6 - Source Range Monitor Performance and Control Rod Withdrawal Sequence

3.7.1 Purpose

The major objective of this test is to demonstrate that the operational sources, SRM instrumentation, and rod withdrawal sequences provide adequate information to achieve criticality and increase power in a safe and efficient manner.

3.7.1.1 Level 1 Criteria

There must be a neutron signal count-to-noise count ratio of at least 2:1 on the required operable SRMs.

There must be a minimum count of 1/2 counts/second on the required operable SRMs.

The IRMs must be on scale before the SRMs exceed their rod block setpints.

3.7.1.2 Level 2 Criteria

Not applicable

3.7.2 Test Results

Source range monitor performance during fuel loading is summarized in Table 3-7. Initial SRM trip points were 1×10^4 and 5×10^4 cps for the rod block and scram functions. Following a nonsaturation demonstration above 7.5×10^5 cps, the setpoints were raised to 1×10^5 cps and 2×10^5 cps, respectively.

Table 3-7

SRM PERFORMANCE

Parameter	SRM A	SRM B	SRM C	SRM D
Pre-Amp Gain:	1	1	1	1
Discriminator Setting(V):	7.5	8	7.5	8
High Voltage Setting(V):	350	375	375	350
Count Rate in Core (CPS):	25	33	70	20
Signal-to-Noise Ratio:	249	164	466	104

See Test Number 10 for results to verify adequate SRM to IRM overlap.

3.7.3 Discussion

Proper SRM performance was confirmed during fuel loading through the initial criticality. During the initial criticality SRM D was found to be failed. Examination of the SRM revealed a failure in the connection between the detector cable and the detector mechanism caused by excessive vibration during the insertion and withdrawal of the detector. Necessary adjustments of the drive mechanism were made to preclude recurrence.

The control rod withdrawal sequences were monitored through generator synchronization to about 45% rated power. The withdrawal sequences utilized the Banked Position Withdrawal (BPW) method in order to minimize individual control rod worth as well as notch worth. Adherence to BPW during control rod withdrawal assures compliance with rod drop accident constraints.

For the data obtained through 45% power, some minor modifications to the withdrawal sequences were made. This was done to minimize core total peaking factor during power ascension.

3.8 Test Number 10 - IRM Performance

3.8.1 Purpose

The purpose of this test is to adjust the Intermediate Range Monitor System to obtain an optimum overlap with the SRM and APRM systems.

3.8.1.1 Level 1 Criteria

Each IRM channel must be on scale before the SRMs exceed their rod block setpoint. Each APRM must be on scale before the IRMs exceed their rod block setpoint.

3.8.1.2 Level 2 Criteria

Each IRM channel must be adjusted so that a half decade overlap with the SRMs and one decade overlap with the APRMs are assured.

3.8.2 Test Results

Test data on IRM-SRM and IRM-APRM overlap is presented in Table 3-8. The IRMs were verified to indicate approximately 100 on range 10 at approximately 40% thermal power.

3.8.3 Discussion

The IRM system was adjusted to provide adequate overlap with both the SRM and APRM systems. Range 6-7 correlation was established to verify proper adjustment of detector preamplifier gain during the initial heatup.

A verification of the IRM indications with reactor power was provided so that the operator could estimate reactor power when operating in the intermediate neutron range. The calibration is strongly dependent upon rod pattern and power distribution but is sufficiently accurate for this purpose.

Table 3-8

IRM PERFORMANCE DATA

IRM/SRM OVERLAP VERIFICATION

IRM Detector	A	B	C	D	E	F	G	H
Reading.	40	40	65	50	55	65	50	55
Range	2	2	2	2	2	2	2	2

SRM Detector	A	B	C	D
Reading (CPS)	$6x10^4$	$7x10^4$	$2x10^4$	$3x10^4$

IRM/APRM OVERLAP VERIFICATION

IRM Detector	A	B	C	D	E	F	G	H
Reading	17.5	12	22	22	22	24	17	19.5
Range	10	9	10	10	10	10	9	9

APRM	A	B	C	D	E	F
Reading (%)	7.0	5.2	7.0	4.0	5.2	5.2

IRM RANGE CORRELATION VERIFICATION

IRM Detector	A	B	C	D	E	F	G	H
Reading-Range 6	50	52	90	66	65	99	62	65
Reading-Range 7	5	5	9.3	6.5	6.5	10	6.2	6.7

IRM/THERMAL POWER CALIBRATION

IRM Detector	A	B	C	D	E	F	G	H
Reading	104	93	100	102	100	107	100	100
Range	10	10	10	10	10	10	10	10

Rx Thermal Power (%) 39.35

3.9 Test Number 11 - LPRM Calibration

3.9.1 Purpose

The major objectives of this test are as follows:

1. Verify proper response of the Local Range Power Monitoring (LPRM) System to local changes in the reactor power level.

2. Calibrate the LPRM system.

3.9.1.1 Level 1 Criteria

Not applicable

3.9.1.2 Level 2 Criteria

Each LPRM reading will be within ± 10% of its calculated value, as determined by a process computer or offline calculation from Traversing Incore Probe power distribution data.

3.9.2 Test Results

Initial LPRM calibration currents were established at 400 uA. Following the first core power distribution calcualtion, LPRM gain adjustment factors (GAFs) ranged from 0.34 to 0.76. With a complete LPRM gain adjustment and subsequent recalibration the LPRM GAFs ranged from 0.79 to 3.02. Further attempts at LPRM calibration were postponed until higher reactor power levels were reached, at which time all LPRM GAFs were brought within the desired 0.90-1.10 range.

3.9.3 Discussion

LPRM operability and correct location response were confirmed during heatup by observing individual indications as adjacent control rods were moved.

Prior to the verification of the process computer calculations, the GE Mark III computer program, BUCLE, was used to perform the LPRM calibration. At a reactor power of 42% rated, all LPRM GAFs were adjusted to between 0.90 and 1.10.

3.10 Test Number 12 - APRM Calibration

3.10.1 Purpose

The purpose of this test is to calibrate the Average Power Range Monitor System.

3.10.1.1 Level 1 Criteria

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

Technical Specification 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.

3.10.1.2 Level 2 Criteria

If the above criteria are satisfied then the APRM channels will be considered to be reading accurately if they agree with the heat balance or the minimum value required based on peaking factor MLHGR, and fraction of rated power to within (+7, -0)% of rated power.

3.10.2 Results

During the initial heatup, with a constant heatup rate of 55°F/hr and a reactor heat capacity of 0.35 MWt Hr/°F the preliminary core thermal power was calculated to be 24.85 MWt, 0.75% of rated power. All APRM's were adjusted to read between 1.8 and 2.3 times higher than real power.

When power level was increased to a point where an accurate manual heat balance could be performed, the APRM's were recalibrated to provide accurate thermal power indications.

After the process computer was verified to be functional, the APRM's were adjusted to indicate the actual thermal power as determined by the process computer heat balance program OD-3.

3.10.3 Discussion

Throughout the Startup Test Program technical specification compliance for APRM, SCRAM and Rod Block setpoints was obtained by adjustment to the APRM outputs, so that the gain adjustment factor (GAF) was always equal to or less than the 'T' factor. The 'T' factor is a ratio of fraction of rated thermal power (FRTP) divided by the maximum fraction of limiting power density (MFLPD). In addition, the APRM scram clamp was maintained at nominally 20% above the test plateau power level. These efforts ensured safe plant operation during the test program.

3.11 Test Number 13 - Process Computer

3.11.1 Purpose

The major objective of this test is to verify the performance of the process computer under plant operating conditions.

3.11.1.1 Level 1 Criteria

Not applicable

3.11.1.2 Level 2 Criteria

Programs OD-1, P1 and OD-5 will be considered operational when:

The MCPR calculated by Back Up Core Limits Evaluation (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 (2) methods shall agree with 2%.

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

1. Are in the same fuel assembly and do not differ in value 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 by the two (2) methods shall agree within 2%.

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

The LPRM gain adjustment factors calculated by BUCLE and the process computer agree to within two percent (2%).

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

3.11.2 Test Results

Initial TIP alignment and computer interface parameters were adjusted such that the difference between the physical top of the TIP detector and the top of the TIP tubing was 2 inches (I Turn), core bottom was established 141 inches below core top and the interface parameter (I Tube) was set to 3 inches. During heatup, proper alignment was confirmed by observing correspondence of spacer-induced flux dips with correct physical dimensions. Proper operation of the TIP-process computer interface was confirmed by comparing TIP X-Y recorder traces with OD-1 edits.

The process computer dynamic system test case was performed at 42% rated power and verified power distribution, exposure accounting, and thermal limits calculations. Subsidiary programs were verified throughout the remainder of the startup. A comparison between the process computer and BUCLE calculations of important parameters is presented in Table 3-9.

3.11.3 Discussion

Detailed verification of all process computer functions was performed to ensure proper operation of the system. Power distribution, exposure accounting, and thermal limits calcualtions were confirmed by comparison with both manual calculations and BUCLE results using identical inputs. These checks displayed excellent agreement. At the conclusion of the Startup Test Program the process computer was completely verified and operational.

Table 3-9:

PROCESS COMPUTER

	40% Power				98% Power			
Parameter	Location	P-1 Value	BUCLE Value	% Deviation[a]	Location	P-1 Value	BUCLE Value	% Deviation
MCPR	25-42	2.960	2.961	-0.03	47-16	1.578	1.572	+0.38
MLHGR (kW/Ft)	24-42-4	5.72	5.72	0.0	47-16-5	12.15	12.20	-0.41
MAPLHGR (Kw/Ft)	25-42-4	5.00	4.94	+0.20	47-16-5	10.26	10.71	-0.37
PBUN (Mw)	25-42	2.44	2.44	0.0	47-16	5.306	5.324	-0.34
W (MLb/Hr	25-42	0.0689	0.0688	+0.15	47-16	0.1282	0.1284	-0.016
LPRM GAF	24-41-D	0.98	0.98	0.0	48-17-D	1.68	1.70	+1.2

a $\frac{\text{P.C. - BUCLE}}{\text{P.C.}} \times 100\%$

3.12 Test Number 14 - RCIC System

3.12.1 Purpose

The purpose of this test is to verify the proper operation of the Reactor Core Isolation Cooling (RCIC) System over its expected operating pressure range.

3.12.1.1 Level 1 Criteria

The average pump discharge flow must be equal to or greater than 600 gpm after thirty seconds have elapsed from automatic initiation at any reactor pressure between 150 psig and rated.

The RCIC turbine must not trip off or isolate during auto or manual start tests.

If any Level 1 criteria are not met, the reactor operation will be restricted to the power level defined by WNP-2 FSAR Figure 14.2-5. This restriction is in addition to any restrictions defined by the Technical Specification.

3.12.1.2 Level 2 Criteria

The Turbine Gland Seal Condenser System shall be capable of preventing steam leakage to the atmosphere.

The differential pressure switch for the RCIC steam supply line high flow isolation trip shall be adjusted to actuate at the value specified in Plant Technical Specification (about 300%).

The speed and flow control loops shall be adjusted so that the decay ratio of any RCIC system related variable is not greater than 0.25.

In order to provide an overspeed trip avoidance margin, the transient start first and subsequent speed peaks shall not exceed 5% above the rated RCIC turbine speed.

3.12.2 Test Results

The RCIC system performance of five (5) cold quick start tests is summarized in Table 3-10. The final controller settings which were acceptable at all test pressures are shown in Table 3-11. The RCIC Turbine Gland Seal Condenser System capability of preventing steam leakage to the atmosphere was verified by observation and detection of the leak detection system in the RCIC equipment room. Extended operation of up to 30 minutes at rated flow conditions was demonstrated to ensure the adequacy of the turbine oil cooling system to maintain a stabilized oil temperature during such operation. Automatic transfer of pump suction from condensate storage tank to suppression pool upon CST low level was also demonstrated at rated pump discharge flow. The RCIC steam supply line high flow isolation trip setpoints were determned to be 225.05 inch of water, equivalent to 300% of the maximum steady state steam flow. However, the actual setpoints were adjusted to be less than 290% steam flow to include the allowance for instrumentation drift and calibration accuracy.

3.12.3 Discussion

The problems encountered during the RCIC system testing involved several turbine trips on overspeed and high exhaust pressure. The overspeed trip problem was found to be caused by a ground fault of a lead connection at the EGR actuator. The grounding of the EGR coil, resulting in a demand input voltage drop from 9 volts to 7.5 volts, slowed down the response of the turbine governor valve and caused the turbine overspeed trip. This problem was resolved with proper wiring of the EGR coil lead. The high turbine exhaust pressure was caused by the malfunctioning of two drain lines. The exhaust line drain was found plugged upstream of the trap by debris. The other condensate drain upstream of steam admission valve was not functioning due to a collapsed float in the steam supply drain pot. It was discovered that an incorrect level switch had been installed. A 1200 psi rated Magnetrol level switch was installed for the drain pot. The consequence of accumulation of condensate in the steam supply line without proper drainage resulted in excessive exhaust pressure causing the trip. Upon fixing both drain line problems, cold and hot quick start tests of the RCIC system were successfully performed.

The RCIC pump discharge flow exhibited a peak-to-peak limiting cycle of 200 GPM during low flow operation (below 400 GPM pump discharge flow) with remote shutdown panel flow controller (C51-R600) in control. Since the controller has been tuned for an optimal performance at rated flow condition where the RCIC system is normally operated, this deviation will not adversely affect the actual RCIC system operation and is considered to be acceptable.

The RCIC pump performance at rated flow was verified under all required operating conditions as indicated by the data in Table 3-10.

Ta[illegible] 3-10

SUMMARY OF RCIC COLD QUICK START TEST

COLD START NO.	1	2	3	4	5
TEST MODE	CST to RPV	CST to RPV (RSP)	CST to CST	CST to CST	CST to CST
Date Performed	05/16/84	06/02/84	07/01/84	07/05/84	07/09/84
Reactor Pressure (PSIG)	930	950	155	928	928
Time to Rated Flow (SEC) (Limit Less Than or equal to 30 sec)	20	22	20	18.5	19
Turbine Speed (RPM) Peak	4333	4458	2400	4500	4650
Turbine Speed (RPM) Steady	4250	4250	2400	4300	4580
Pump Discharge Pressure (PSIG)	1030	1043	250	1050	1200
Pump Suction Pressure (PSIG)	17	20	17	20	18
Pump Discharge Flow (GPM) (Limit Greater Than or equal to 600 GPM)	617	600	655	610	600

TABLE 3-11

SUMMARY OF RCIC SYSTEM CONTROL SETTINGS

		DIAL	ACTUAL
RCIC Flow Controller E51-R600 (control room)			
	Gain	.08	.083
	Resets/min	40	27.9
RCIC Flow Controller C61-R601 (RSP)			
	Gain	.083	.083
	Resets/min	30	27.9
Woodward EGM Controller			
	Gain	10	10
	Stability	9	9
	Idle Voltage	-0.9	-0.9
EGR Actuator			
	Needle Valve (turns)	3/4	3/4

3.13 Test Number 16A - Selected Process Temperatures

3.13.1 Purpose

The purposes of this test are to 1) assure that the measured bottom head drain temperature corresponds to bottom head coolant temperature during normal operations, 2) identify any reactor operating modes that cause temperature stratification, 3) determine the proper setting of the low flow control limiter for the recirculation pumps to avoid coolant temperature stratification in the reactor pressure vessel bottom head region, 4) familiarize the plant personnel with the temperature differential limitations of the reactor system.

3.13.1.1 Level 1 Criteria

The reactor recirculation pumps shall not be started nor flow increased unless the coolant temperatures between the steam dome and bottom head drain are within 145°F.

The recirculation pump in an idle loop must not be started, active loop flow must not be raised and power must not be increased unless the idle loop suction temperature is within 50°F of the active loop suction temperature. If two pumps are idle, the loop suction temperature must be within 50°F of the steam dome temperature before pump startup.

3.13.1.2 Level 2 Criteria

During two pump operation at rated core flow, the bottom head temperature as measured by the bottom drain line thermocouple should be within 30°F (17°C) of the recirculation loop temperatures.

3.13.2 Test Results

At rated pressure and temperature the test was performed for both two-pump and single RRC pump operation. First the FCV's were decreased to their minimum positions resulting in a maximum temperature differential between steam dome and bottom head drain of 48°F for two-pump operation and 59°F for single loop operation. Next the RWCU flow was decreased from 270 GPM to 93 GPM for two-pump operation and 270 GPM to 115 GPM for single pump operation, resulting in a maximum temperature differential of 82°F and 92°F respectively. The CRD flow was then increased from 64 GPM to 76 GPM for two-pump operation and 60 GPM to 75 GPM for single-pump operation, resulting in a maximum temperature differential of 100°F and 104°F respectively.

Table 3-12 presents the selected temperature measurement results during the steady state operation throughout the test program and following one-pump and two-pump trips at various power levels.

3.12.3 Discussion

The vessel temperature measurement data indicated that no anticipated operating condition such as subcooling change or pump trip exists in which Technical Specification limits on reactor temperature are likely to be exceeded. All data confirmed that the vessel temperature differentials are within acceptable limits.

TABLE 3-12

SUMMARY OF SELECTED PROCESS TEMPERATURE MEASUREMENT

Test Condition	Heatup	1	2	3	3	3	4	6	6
Test Mode	Steady State	Steady State	Steady State	Steady State	One Pump Trip	Two Trip Trip	Steady State	Steady State	One Pump Trip
Recirculation Pump Speed (Hz)									
Pump A	15	15	60	60	60	15	0	60	0
Pump B	15	15	60	60	0	15	0	60	60
Recirculation Loop Suction Temperature (°F)									
Loop A	530	520	520	530	520	518	509	530	517
Loop B	535	520	519	530	520	518	508	530	515
Reactor Bottom Head Drain Temperature	515	500	504	515	510	501	593	518	507
Reactor Steam Dome Saturation Temperature	540	538	538	542	540	539	537	544	541
T, Saturated Steam - Bottom Head Drain	25	38	34	27	30	38	44	26	34
T, Recirculation Loop Suction - Bottom Head Drain	20	18	18	12	10	17	16	12	10

3.14 Test Number 16B - Water Level Reference Leg Temperature Measurement

3.14.1 Purpose

The purpose of this test is to measure the reference leg temperature and recalibrate the affected level instruments if the measured temperature is different than the value assumed during the initial calibration.

3.14.1.1 Level 1 Criteria

Not applicable

3.14.1.2 Level 2 Criteria

The indicator readings on the narrow range level system should agree within $\pm$ 1.5 inches of the average readings or the reading calculated from the correct reference leg temperatures.

The wide and upset range level system indicators should agree within $\pm$ 6 inches of the average readings or the readings calculated from the correct reference leg temperatures.

3.14.2 Test Results

The Reference Leg Temperature Measurement data indicated that the actual temperatures of drywell and reactor building were in good agreement with the initial calibration assumptions, thus no adjustment of calibration settings is necessary.

Table 3-13 summarized the water level measurements throughout the test program.

3.14.3 Discussion

At rated temperature and pressure the reference leg temperature of the wide range and narrow range instruments was verified to be consistent with the value assumed for initial calibration. Level indications were recorded during steady state operation at each test condition throughout the test program. Minor adjustment of the level instruments to their calibration settings was required to ensure proper indication for compliance with the acceptance criteria.

The water level measurement data indicates that the variation between narrow range and wide range level was a function of core flow and reactor power. The wide range level indication deviation from the narrow range level indication increases with increasing core flow on a constant rod line as well as increasing power on a constant flow line. These phenomena were well explained as being caused by the jet pump velocity effect and the subcooling change in the vicinity of the instrument tap for the wide range level instrumentation and is typical of similar BWR's.

At 100% power and core flow, the wide range water level indication was 16 inches below that indicated by the narrow range level instrument.

TABLE 3-13

SUMMARY OF WATER LEVEL MEASUREMENT

Test Condition	Heatup	1	2	3	4	5	6
Reactor Power (%)	4.6	16	40	65	40	72	99
Core Flow (%)	26	33	41	87	28	55	98
Reactor Pressure (PSIG)	965	934	922	950	947	966	978
Feedwater Temperature (°F)	78	90	332	373	330	380	410
Core Subcooling (BTU/LB)	5	18	27	16	38	36.5	18.7
Average Drywell Temperature (°F)	99.7	110.7	120	126	121	121.5	131.7
Average Reactor Building Temperature (°F)	72.5	74	90	88.9	74	76	74
Average Wide Range Level (In.)	33.1	34.1	31.8	22.1	37.4	30.6	19.3
Average Narrow Range Level (In.)	35.6	35.9	34.9	35.7	36.5	36.1	35.3

3.15 Test Number 17 - System Expansion

3.15.1 Purpose

The purpose of this test is to 1) verify that piping systems and components that are unrestrained with respect to thermal expansion, 2) verify that suspension components are functioning in the specified manner, 3) provide confirmatory data for the calculated stress levels in nozzles and weldments, 4) perform in inspection to satisfy ASME Section XI, IWF-220 post heatup (shakedown) inspection requirements, and 5) satisfy the inspection requirements for the condensate and feedwater systems per Regulation Guide 1.68.1.

3.15.1.1 Level 1 Criteria

Thermally induced displacement of system components shall be unrestrained, with no evidence of binding or impairment.

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

Snubbers shall not reach the limits of their travel. The displacements at the established transducer locations used to measure pipe deflections shall not exceed the allowable values. The allowable values of displacement shall be based on not exceeding ASME Section III Code stress allowables.

3.15.1.2 Level 2 Criteria

Spring hangers will be in their operating range (between the hot and cold settings).

Snubber settings must be within their expected operating range.

The displacements at the established transducer locations shall not exceed the expected values.

3.15.2 Test Results

The visual inspection of the drywell piping of NSSS and the auxiliary system was conducted prior to heatup, at an intermediate temperature of 250°F and rated temperature during initial heatup, and following a reactor shutdown after a minimum of 3 thermal cycles. These walkdowns have verified that the selected drywell piping systems and components are unrestrained with respect to thermal expansion. Each pipe whip restraint was verified to have sufficient clearance and final torqueing of the holddown bolts was completed following clearance verification. In addition, the RPV stabilizer clearances were properly adjusted for the relative growth between the vessel bracket and the stabilizer yoke.

During the initial heatup and the following thermal cycle, displacement of the instrumented piping systems were monitored to confirm that the pipe suspension is working as designed and that the pipe is free of obstructions. These movements are summarized in Table 3-14 and 3-15.

3.15.3 Discussion

The drywell inspection performed during the initial heatup revealed piping interferences on HPCS and feedwater piping. Heatup testing was held until the identified interference problems were rectified. No further evidence of blockage or binding of the piping was observed during the subsequent inspections. The settings on the pipe supports (both snubber and hanger) were recorded at cold and rated conditions to ensure the proper operation of the pipe suspension systems.

In some instances piping displacements from instrumented locations exceeded the Level 2 limits. Data of the criteria failures were reviewed and evaluated to be acceptable by GE piping engineering and Supply System engineering. Piping movements after several thermal cycles were found to be consistent in both magnitude and direction with the movement as measured during the initial heatup. Only a modest number of Level 2 violations were recorded through all phases of the thermal expansion tests and no Level 1 violations were found. As defined by PPM 8.2.17, a Level 2 violation is a low level deviation from anticipated test criteria which requires the test engineer's investigation and rationalization. A Level 2 violation does not involve an overstress condition, whereas a Level 1 violation may involve exceedance of ASME allowable stress limits. In the case of thermal expansion, a Level 1 violation typically results from support system binding or other interference to system expansion. As an example, Level 2 thermal expansion deviations were typically resolved by comparison of the actual system temperatures versus the conservative upper bound temperatures assumed by the A/E for his piping design calculations. In all cases, an acceptable physical rationalization of the Level 2 violations has been provided. In summary, it is concluded that the PAT program has demonstrated that all visually inspected and remotely monitored piping systems are responding in an acceptable manner to thermal loads as predicted by the A/E and GE.

Visual inspections, consisting of pipe walkdowns by VT-3&4 qualified inspectors, were performed on the hot piping systems identified outside the drywell to ensure unrestrained thermal growth. The systems inspected are as follows:

1. Main Steam Including MSLC
2. Feedwater and Portions of Condensate
3. RWCU
4. RHR Shutdown Cooling Supply and Return
5. RCIC Steam Supply

The inspections were successfully performed providing acceptable results. Copies of all the visual inspections performed have been sent to the NRC as part of the PSI program report.

TABLE 3-14
DRYWELL PIPING THERMAL EXPANSION DATA SHEET

System	Data Point	Baseline (ambient)	Int. (200 to 300°F)	Int. (400 to 500°F)	At 100% Power Values	1000°F & Rated+ (545°F)
Reactor Recirc Loop A	1RA X	-.007	-.052	-.180	-.2	-.225
	1RA Y	-.111	-.242	-1.044	-.49	-1.208
	1RA Z	+.015	+.022	+.001	-.01	-.004
(Reference only to check RPV expansion)	2RA X	-.022	+.137	+.369	+.32	+.448
	2RA Y	+.002	-.111	-.294	-.2	-.353
	2RA Z	-.019	+.001	-.051	-.4	-.054
	3RA X(E-W)	+.009	+.148	+.416	+.38	+.508
	3RA Y	-.004	-.045	-.100	-.09	-.100
	3RA Z(N-S)	+.001	+.021	+.061	-.05	+.085
	4RA X	-.006	-.171	-.523	-.46	-.629
	4RA Y	-.010	-.007	-.002	0	+.004
	4RA Z	-.010	-.021	-.069	-.400	-.096
Reactor Recirc Loop B	1RB X	-.001	+.071	+.157	+.18	+.181
	1RB Y	+.006	-.186	-.761	INOP	-.894
	1RB Z	+.010	-.033	-.041	-.02	-.038
	2RB X	-.005	-.178	-.522	-.42	-.612
	2RB Y	-.010	-.277	-.818	-.45	-.932
	2RB Z	+.012	-.014	+.028	-.04	+.066
	3RB X(E-W)	-.006	-.168	-.462	-.43	-.548
	3RB Y	-.003	-.042	-.094	-.18	-.091
	3RB Z(N-S)	-.006	-.058	-.154	-.15	-.185
	4RB X(E-W)	+.005	+.037	+.113	+.03	+.145
	4RB Y	-.001	.000	+.015	-.01	+.039
	4RB Z(N-S)	-.009	+.171	+.530	+.47	+.665
Main Steam Loop A (For Reference only to check RPV expansion)	1MSA X(E-W)	-.002	+.040	+.195	N/A	+.217
	1MSA Y	-.041	+.480	+1.412	N/A	+1.741
	1MSA X(N-S)	+.001	-.254	-.570	N/A	-.609
	2MSA X	-.002	-.084	-.174	N/A	-.189
	2MSA Y	-.020	+.194	+.401	N/A	+.372
	2MSA Z	+.017	-.114	-.258	N/A	-.321

NOTE: Displacements shown are in inches.

TABLE 3-14 (Contd)

System	Data Point	Baseline (ambient)	Int. (200 to 300°F)	Int. (400 to 500°F)	At 100% Power Values	1000°F & Rated+ (545°F)
Main Steam Loop B	1MSB X	-.001	-.104	-.367	N/A	-.461
	1MSB Y	+.004	+.043	+.253	N/A	+.331
	1MSB X	-.001	+.201	+.372	N/A	+.443
	2MSB X	+.002	+.120	+.355	N/A	+.411
	2MSB Y	+.012	+.366	+.886	N/A	+1.034
	2MSB Z	+.007	-.090	-.167	N/A	-.143
Main Steam Loop C	1MSC X	-.016	-.299	-.707	N/A	-.806
	1MSC Y	-.005	+.069	+.317	N/A	+.419
	1MSC X	-.003	-.238	-.599	N/A	-.721
	2MSC X(E-W)	-.023	+.001	-.009	N/A	+.005
	2MSC Y	-.007	+.298	+.734	N/A	+.844
	2MSC Z(N-S)	-.007	-.012	-.030	N/A	-.036
Main Steam Loop D	1MSD X	-.131	-.310	-.705	N/A	-.814
	1MSD Y	-.001	+.081	+.433	N/A	+.555
	1MSD X	+.001	-.022	-.192	N/A	-.254
	2MSD X	-.001	-.113	-.206	N/A	-.255
	2MSD Y	-.002	+.203	+.555	N/A	+.640
	2MSD Z	-.032	+.027	+.003	N/A	-.019

NOTE: Displacements shown are in inches.

Table 3-15
DRYWELL PIPING THERMAL EXPANSION DATA SHEET

System	Data Point	Baseline (ambient)	Intermediate 25% Power)	Rated (100% Power)
Feedwater Line A	1FWA X(E-W)	+.003	+.240	+.41
	1FWA Y	+.004	+.965	+.62
	1FWA Z(N-S)	-.011	-.040	-.42
	2FWA X	+.029	+.290	+.58
	2FWA Y	+.306	+.850	-.52
	2FWA Z	+.020	+.360	+.97
H. Feedwater Line A	1FWB X(E-W)	-.036	-.345	-.90
	1FWB Y	-.017	+.825	-.70
	1FWB Z(N-S)	+.006	+.050	+.43
	2FWB X(E-W)	-.019	-.020	-.54
	2FWB Y	-.016	+.475	-.75
	2FWB Z(N-S)	+.015	-.028	-.02

Intermediate data F/W Temp. 300°F
Rated data F/W Temp. 413°F

NOTE: Feedwater piping will reach rated temperature only when at rated reactor power. The intermediate temperature levels will be at 25% power testing during T/C 2 and rated at 100% power during T/C 6.

System	Data Point	Baseline (ambient)	Intermediate (200 to 300°F)	Rated (545°F)
I. Shortest SRV Discharge Pipe (1B)	1SRV X(E-W)	+.003	-.059	-.266
	1SRV Y	+.023	+.036	+.044
	1SRV Z(N-S)	+.001	-.053	-.219
Longest SRV Discharge Pipe (3C)	2SRV X(E-W)	-.013	-.053	+.006
	2SRV Y	+.020	+.110	+.300
	2SRV Z(N-S)	-.014	-.073	-.203
K. RCIC Steam Supply (150 psig/366°F for Immediate Data)	1RCIC X	+.001	-.025	-.045
	1RCIC Y	-.007	+.049	+.003
	1RCIC Z	-.003	-.052	-.241
L. Reactor Water Cleanup	RWCU X(E-W)	+.003	+.050	+.305
	RWCU Y	-.003	-.011	-.022
	RWCU Z(N-S)	+.015	+.028	+.133
M. RHR Shutdown Cooling System (Suction)	1RHR X(E-W)	+.014	+.038	+.193
	1RHR Y	-.012	-.110	-.386
	1RHR Z(N-S)	+.005	-.045	-.189
N. RHR Return (A Loop)	2RHR X(E-W)	+.007	+.108	+.293
	2RHR Y	-.035	-.176	-.600
	2RHR Z(N-S)	+.036	+.021	-.137

3.16 Test Number 18 - Core Power Distribution

3.16.1 Purpose

The major objectives of this test are as follows:

1. To confirm the reproducibility of the Traversing In-Core Probe (TIP) system readings.

2. To determine the core power distribution in three dimensions.

3.16.1.1 Level 1 Criteria

Not applicable

3.16.1.2 Level 2 Criteria

The total TIP uncertainty (including random noise and geometric uncertainties) obtained by averaging the uncertainties for all data sets must be less than 6%.

NOTE: A minimum of two and a maximum of six data sets may be used to meet the above criteria.

3.16.2 Test Results

At both 72% and 98% power the core power distributions in sequence A were symmetrical. Bundle powers from symmetrical locations were comparable.

A summary of the TIP uncertainty analysis performed at 72% and 98% power is presented in Table 3-16.

TABLE 3-16

SUMMARY OF TIP UNCERTAINTY ANALYSIS RESULTS

Parameter	RESULTS 72%	RESULTS 98%	RESULTS Avg.	Level 2 Criteria
Total TIP Uncertainty	3.20%	2.30%	2.75%	6.0%
Random Noise Component	1.42%	1.43%	1.43%	-
Geometric Component	2.87%	1.80%	2.34%	-

3.16.3 Discussion

The random noise component of total TIP uncertainty was determined by calculating the standard deviation of six sets of common channel traverses on each of five TIP machines at 72% power and five sets of common channel traverses on each of five TIP machines at 98% power. The total TIP uncertainty was calculated by computing the standard deviation of 19 sets of symmetric pair ratios from a complete core scan. The geometrical component was determined from the measured data given that total TIP uncertainty is the statistical sum of the random noise and geometric components.

3.17 Test Number 19 - Core Performance

3.17.1 Purpose

The major objectives of this test are as follows:

1. to evaluate core thermal power and flow rate; and

2. to evaluate the following core performance parameters:

MLHGR-Maximum Linear Heat Generation Rate
MCPR-Minimum Critical Power Ratio
MAPLHGR-Maximum Average Planar Linear Heat Generation Rate

3.17.1.1 Level 1 Criteria

The Maximum Linear Heat Generation Rate (MLHGR) of any rod during steady state conditions shall not exceed 13.4 Kw/ft.

The steady state Minimum Critical Power Ratio (MCPR) shall be equal to or greater than the MCPR limit times the k_f factor determined from Figure 3.2.3-1 in the WNP-2 Technical Specifications with MCPR limit equal to 1.24.

All Average Planar Linear Heat Generation Rates (APLHGRs) for each type of fuel as a function of Average Planar Exposure shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2 and 3.2-1.3 in the WNP-2 Technical Specifications.

Steady state reactor power shall be limited to 3323 MWt and values on or below the analyzed flow control line.

3.17.1.2 Level 2 Criteria

Not applicable

3.17.2 Test Results

A summary of core performance parameters from throughout the startup test program is presented in Table 3-17.

TABLE 3-17

CORE PERFORMANCE SUMMARY

Parameter	TEST RESULTS TC1	TC2	TC3	TC4	TC5	TC6	Warranty
Core Thermal Power (MWt)	575	974	2412	1348	2346	3221	3277
Core Flow (MLb/hr)	36.0	38.4	93.6	30.0	59.5	106.5	106.8
Core Inlet Subcooling (Btu/Lbm)	22.3	22.6	17.6	38.3	29.1	18.4	18.6
MCPR	5.087	3.021	1.970	2.298	1.826	1.554	1.404
MLHGR (Kw/Ft)	3.50	5.24	8.31	5.85	8.93	12.14	11.97
MAPLHGR (Kw/Ft)	3.03	4.46	7.36	5.23	7.94	10.67	10.74
Total Peaking Factor	3.70	--	2.05	2.58	2.27	2.24	2.17

3.17.3 Discussion

Prior to completion of the process computer, DSTC, core thermal limits evaluation was performed using BUCLE. Test Condition 1 and 2 results reported were generated in this manner. Subsequent results are from process computer edits.

3.18 Test Number 20 - Steam Production

3.18.1 Purpose

The major objective of this test is to demonstrate that the Nuclear Steam Supply System for WNP-2 is providing steam in sufficient quantity and quality to satisfy all appropriate warranties as defined in the contract GE and WPPSS.

3.18.1.1 Level 1 Criteria

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

The NSSS will be capable of supplying steam in an amount and quality corresponding to the final feedwater temperature and other conditions shown on the Rated Steam Output Curve (Figure 3-2). The Rated Steam Output Curve provides the warrantable reactor vessel steam output as a function of feedwater temperature, as well as warrantable steam conditions at the outboard main steam isolation valves.

3.18.1.2 Level 2 Criteria

Not applicable

3.18.2 Test Results

During the 100 hr warranty demonstration it was determined that the Nuclear Steam Supply System delivered an average of 14.050 Mlb/hr steam flow at a quality of 99.86% with a steam line pressure at the second MSIV of 989.1 psia. Feedwater temperature and enthalpy were 412.25°F and 389.27 Btu/Lbm, respectively. Average inputs from three 2-hour test runs within the 100-hour run are listed in Table 3-18.

3.18.3 Discussion

The 100-hr warranty demonstration was performed at an average power level of 3302.6 MWt (99.4%). Steam flow was extrapolated at 100% power and adjusted to account for differences in system parameters between the rated heat balance (Figure 1.1-1 in the WNP-2 FSAR) and actual test conditions.

TABLE 3-18

STEAM PRODUCTION DATA

Parameter	Data Run 1	Run 2	Run 3
Feedwater Flow (Mlb/hr)	13.8968	14.0761	14.0789
Feedwater Temperature (°F)	412.00	412.75	412.02
Feedwater Pressure (psia)	1094.7	1099.7	1099.7
Feedwater Enthalpy (btu/lbm)	388.98	389.81	389.02
CRD Flow (Mbl/hr)	0.0325	0.0325	0.0325
CRD Temperature (°F)	107.17	109.90	109.06
CRD Pressure (psia)	1269.7	1279.35	1287.00
CRD Enthalpy (btu/lbm)	78.46	81.20	80.44
Cleanup System Flow (Mlb/hr)	0.0945	.1029	.13226
Cleanup System Inlet Temperature (°F)	527.4	528.3	529.2
Cleanup System Outlet Temperature (°F)	441.7	440.9	438.7
Cleanup System Inlet Enthalpy (btu/lbm)	520.90	522.01	523.11
Cleanup System Outlet Enthalpy (btu/lbm)	421.40	420.40	418.59
Recirculation Pump Power (Mw)			
A	6.23	6.23	6.21
B	6.14	6.14	6.11
Steam Flow (Mlb/hr)	13.9293	14.1086	14.1114
Reactor Pressure (Psia)	1006.7	1014.2	1015.2
Steam Enthalpy (btu/lbm)	1192.6	1192.3	1192.4
Main Steam Line Pressure (psia)	983.7	990.7	992.9

RATED STEAM OUTPUT CURVE

Rated Core Thermal Power: 3323 MW_t

Steam Output Equation: $Wpv = \frac{11317.7 \times 10^6}{1191.5 - h_{fw}} + 39{,}000$

Steam Conditions at Exit of Second Isolation Valve:

- Moisture 0.3%
- Pressure 985 psia
- Enthalpy 1191.5 Btu/lb

(Consistent with 1967 ASME Steam Tables)

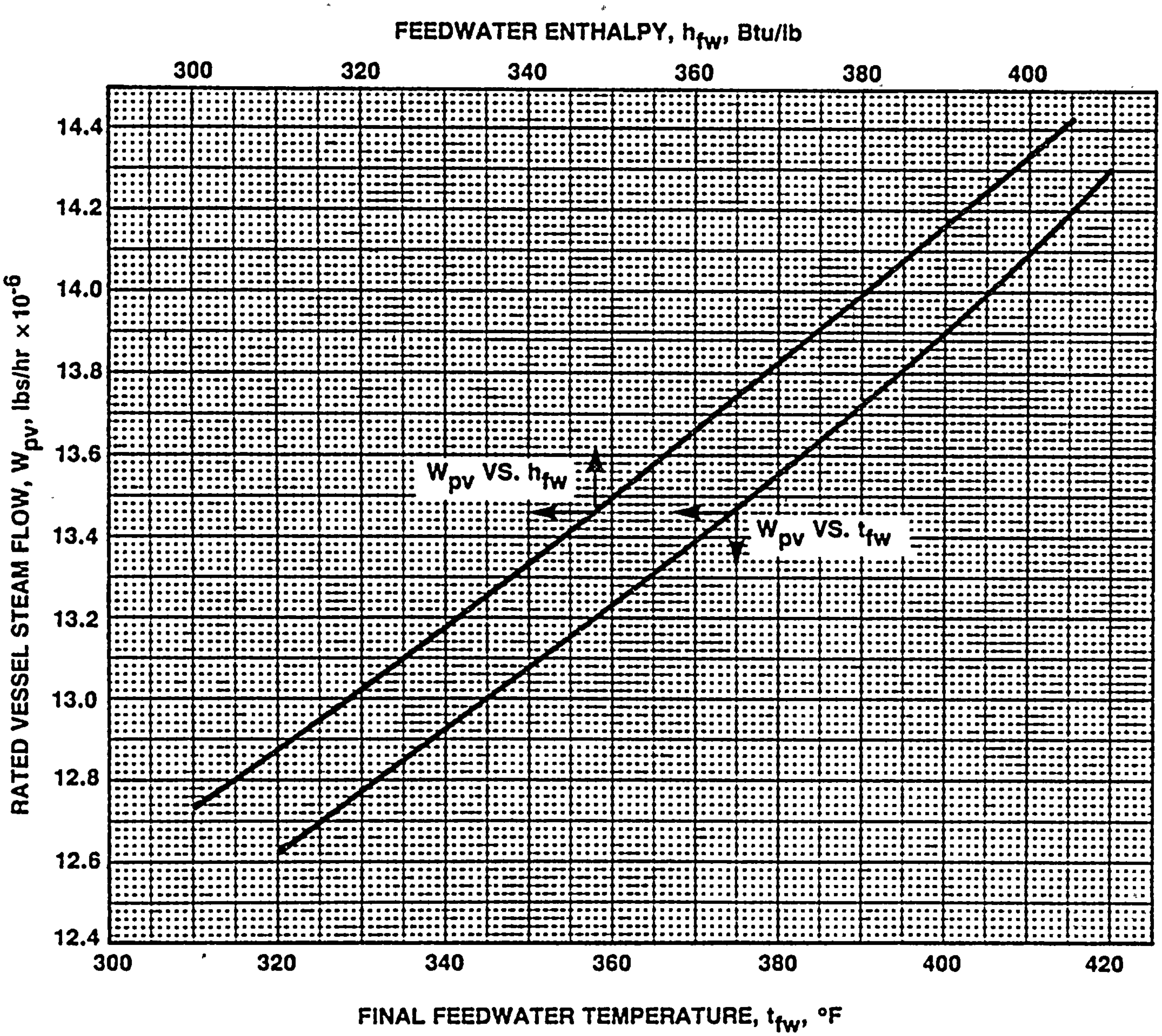

Figure 3-3-

3.19 Test Number 21 - Core Power-Void Mode Response

3.19.1 Purpose

The major objective of this test is to measure the stability of the core power-void mode dynamic response and to demonstrate that its behavior is within specified limits. The core power void response, that results from a combination of the neutron kinetics and core thermal hydraulic dynamics, is least stable near the natural circulation end of the rated 100 percent power rod line. A fast change in the reactivity balance is obtained by a pressure regulator step change and by moving a very high worth rod only 1 or 2 notches. Both local flux and total core response were evaluated by monitoring selected LPRM's during the transient.

3.19.1.1 Level 1 Criteria

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

3.19.1.2 Level 2 Criteria

The decay ratio must be less than or equal to 0.5 for each total core process variable and local hydrodynamic channel (LPRM) that exhibits oscillatory response.

3.19.2 Test Results

The results of the dynamic response measurements from the natural circulation and high power/low flow test conditions are summarized in Table 3-19. The test data indicates that the core response to local reactivity perturbations is well damped while the response to whole core reactivity perturbations is within acceptable limits.

3.19.3 Discussion

Test condition 4 results were obtained in the natural circulation condition at 39.4% thermal power and 27.1% core flow. Test condition 5 results were obtained along the 100% load line.

TABLE 3-19

CORE POWER-VOID MODE RESPONSE DATA

Test Condition	4			5		
Perturbution	LPRM Change	APRM Change	Decay Ratio	LPRM Change	APRM Change	Decay Ratio
CR 34-43 Insertion 36-28	10.0%	0.0	0.0			
Withdrawal 28-36	10.0%	0.0	0.0			
Pressure Regulator Failure	10.5%	12.0%	0.42			
CR 46-39 Insertion 10-0				7.0%	0.0	0.0
Withdrawal 0-10				7.0%	0.0	0.0
Pressure Regulator Failure				1.9%	3.0%	0.27

3.20 Test Number 22 - Pressure Regulator

3.20.1 Purpose

The major objectives of this test are as follows:

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 regulator.

2. To demonstrate the takeover capability of the backup pressure regulator via simulated failure of the controlling pressure regulator and to set the regulating pressure difference between the two regulators to an appropriate value.

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

3.20.1.1 Level 1 Criteria

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

3.20.1.2 Level 2 Criteria

Pressure control system related variable may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25.

The turbine inlet pressure response time from initiation of pressure setpoint change to the inlet pressure peak shall be less than or equal to 10 seconds.

Pressure control system deadband, delay, etc., shall be small enough that steady state limit cycles (if any) shall produce steam flow variations no larger than $\pm$ 0.5% of rated steam flow.

For all pressure regulator transients the peak neutron flux and/or peak vessel pressure shall remain below the scram setting by 7.5% and 10 psi respectively (maintain a plot of power versus the peak variable values along the 100% load line).

The variation in incremental regulation (ratio of the maximum to the minimum value of the quantity, "incremental change in pressure control signal/incremental change in steam flow" for each flow range) shall meet the following:

% of Steam Flow Obtained With Valves Wide Open	Maximum Variation
0 to 90%	Less than or equal to 4:1
90 to 97%	Less than or equal to 2:1
90 to 99%	Less than or equal to 5:1

3.20.2 Test Results

The final pressure control system settings are presented in Table 3-20. The pressure regulator settings were not adjusted from the original values determined before initial reactor heatup. The results of the demonstration tests performed to confirm the adequacy of these settings are presented in Table 3-21.

At 100% power steady state limit cycles were less than ± 0.5% of rated steam flow. Based upon regulator failure testing, a setpoint differential between the primary and secondary regulators of 3 psi was selected. Figure 3-3 presents the relationships between main steam flow, control valve opening, and generator output to control valve demand (pressure regulator output) at a constant throttle pressure.

3.20.3 Discussion

The pressure control system settings were determined based on suggested initial system setup in the Hanford-2 Nuclear Power Station Control Systems Design Report (GEZ-6894) and testing carried out during open vessel testing. Based on extensive testing of the pressure control system during the power ascension test program, no adjustment to the original pressure regulator settings were made.

Upon reaching the 60-75% power region for the first time an instability was discovered in the pressure control system. A "knee" in the pressure regulator gain curve due to break points in the governor valve function generator curves caused limit cycle behavior in the pressure control system. Generator output swings as high as 100 MWe peak to peak were observed. Adjustment of the function generator curves for governor valves 1 and 4 reduced the gain curve knee to allow acceptable pressure control system operation.

The incremental regulation performance is shown graphically in Figure 3-3. This performance was judged to be satisfactory.

TABLE 3-20

FINAL PRESSURE CONTROL SYSTEM SETTING

		PRESSURE REGULATOR A	PRESSURE REGULATOR B
Gain (% psi)	Dial	31.6	31.9
	Actual	3.33	3.33
Lead (T_1) second	Dial	Tu = 1.93	Tu = 1.93
	Actual	3.0	3.0
Lag (T_2) second	Dial	B = 2. x 0.21	B = 2 x 0.21
	Actual	7.4	7.4
Steam Line Compensator			
T_0		0.113	0.113
T_3		0.32	0.28
2		1.5	1.5
1/2		0.2	0.2

TABLE 3-21

PRESSURE REGULATOR TEST RESULTS

Parameter	TEST RESULTS							
Test Condition	1	2	3		4	5	6	
Recirc Flow Control Mode	POS	POS	POS	FLX	POS	POS	POS	FLX
Power (%)	17.2	27.0	63.5	63.5	38.6	66.7	97.2	97.2
Core Flow (%)	32.3	30.4	79.8	79.8	27.6	53.5	95.9	95.9
Reactor Pressure (psig)	937	920	964	964	929	963	972	972
Test Mode:								
GV, Setpoint Change								
Peak Dome Pressure (psig)	--	930	963	954	955	970	1000	991
Peak APRM (%)	--	43.5	94	79	39	78	104	105
Time to Peak Press. (sec)	--	3.3	4.8	3.0	3.4	2.8	2.7	3.2
Maximum Decay Ratio	--	0.44*	0.0	0.0	0.41*	0.0	0.0	0.0
GV+BPV, Setpoint Change								
Peak Dome Pressure (psig)	--	925	--	--	937	965	990	994
Peak APRM (%)	--	34.2	--	--	42	70	100	103
Time to Peak Press. (Sec)	--	5.0	--	--	5.0	5.2	7.0	7.0
Maximum Decay Ratio	--	0.39	--	--	0.0	0.0	0.0	0.0
BPV, Setpoint Change								
Peak Dome Pressure (psig)	949	934	--	--	944	971	996	997
Peak APRM (%)	25.6	35.5	--	--	42	71	104	105
Time to Peak Press. (Sec)	5.0	5.8	--	--	6.8	6.4	7.0	4.0
Maximum Decay Ratio	0.65*	0.24	--	--	0.23	0.0	0.0	0.0
GV, Regulator Failure								
Peak Dome Pressure (psig)	--	937	966	958	946	976	995	997
Peak APRM (%)	--	40.7	82	72	53	80	103	105
Time to Peak Press. (Sec)	--	--	--	--	2.2	1.0	1.2	1.2
Maximum Decay Ratio	--	0.22	0.0	0.0	0.34*	0.0	0.0	0.0
GV+BPV, Regulator Failure								
Peak Dome Pressure (psig)	--	937	--	--	944	--	997	1002
Peak APRM (%)	--	31.2	--	--	38	--	101	102
Time to Peak Press. (Sec)	--	--	--	--	15.4	--	10.0	10.0
Maximum Decay Ratio	--	0.0	--	--	0.0	--	0.0	0.0
BPV, Regulator Failure								
Peak Dome Pressure (psig)	956	935	--	--	946	--	997	1001
Peak APRM (%)	25.6	32.6	--	--	42	--	100	104
Time to Peak Press. (Sec)	3.9	--	--	--	4.2	--	4.6	2.9
Maximum Decay Ratio	0.41*	0.0	--	--	0.2	--	0.0	0.21

* Each decay ratio 7.25 was evaluated as acceptable.

PRESSURE REGULATOR STATIC DATA

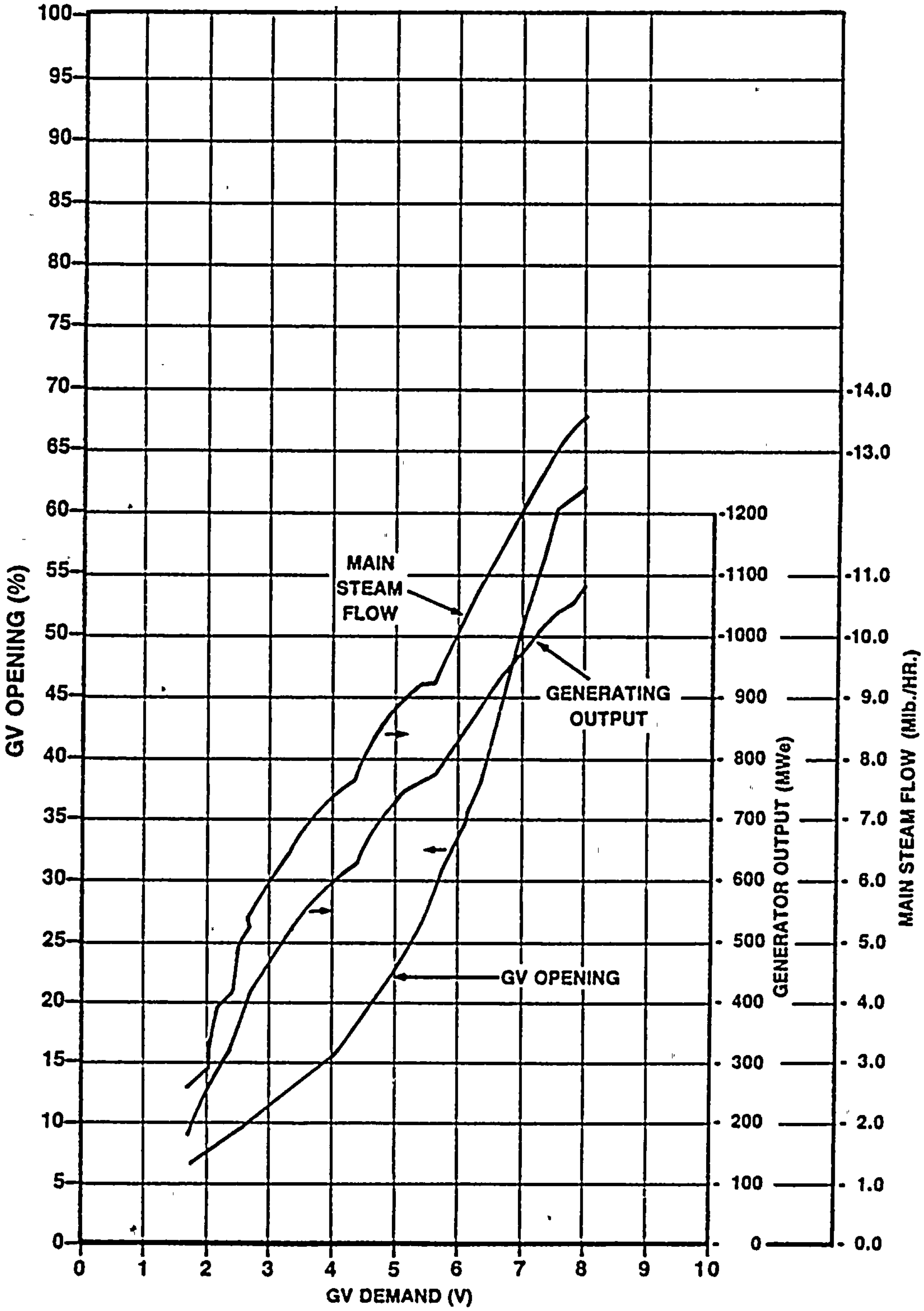

Figure 3-4

3.21 Test Number 23A - Water Level Setpoint and Manual Flow Changes

3.21.1 Purpose

The purpose of this test is to verify that the feedwater system has been adjusted to provide acceptable reactor water level control.

3.21.1.1 Level 1 Criteria

The transient response of any level control system-related variable to any test must not diverge.

3.21.1.2 Level 2 Criteria

Level control system-related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25.

The open loop dynamic flow response of each feedwater actuator (turbine or valve) to small (less than 10%) step disturbances shall be:

1.	Maximum time to 10% of step disturbance	1.1 sec
2.	Maximum time from 10% to 90% of a step disturbance	1.9 sec
3.	Peak overshoot (% of step disturbance)	15%
4.	Settling time, 100%, ±5%	14 sec

The average rate or response of the feedwater actuator to large (20% of pump flow) step disturbances shall be between 10% and 25% rated feedwater flow/second. This average response rate will be assessed by determining the time required to pass linearly through the 10% and 90% response points.

At steady-state generation for the 3/1 element systems, the input scaling to be mismatch gain should be adjusted such that level error due to biased mismatch gain output should be within ±1 inch.

The dynamic response of each individual level or flow sensor shall be as fast as possible. Band width must be at least 4.0 radians/second (faster than 0.25 second equivalent time constant), except for the steam flow sensors which must have band width of at least 1.0 radian/second (faster than 1.0 second equivalent time constant).

3.21.2 Test Results

All transient response of the level control system related variables to the step change did not diverge. The decay ratio of the transient response of the level controller at the final settings as shown in Table 3-22, was less than 0.25 or evaluated as acceptable.

The open loop dynamic flow response of the startup level control valve was determined to be acceptable even with the exception of slightly excessive overshoot. The result is indicated in Table 3-23.

The open loop dynamic flow response of the feedwater turbine speed control indicates that the acceptance criteria were met except that excessive overshoot was observed on the flow response of feedwater pump 'A'. Table 3-24 summarizes the test results.

The average rate of response of the feedwater pump to large (less than 20%) step disturbance were 15% and 14% rated feedwater flow/second for pump 'A' and 'B' respectively.

The steam flow/feedwater flow mismatch gain was verified to be acceptable by measuring the level error due to biased mismatch gain. The level error was found to be 0.7 inch at steady state rated conditions.

The feedwater system and the level control system gain data was obtained during the power escalation to rated power. The gain curves, as shown in Figures 3-4 and 3-5, produced a linear relationship between the FW loop flow and the level controller output over the operating range.

3.21.3 Discussion

During an unplanned scram transient, the feedwater turbines were tripped on L-8. Investigation revealed that the turbine minimum speed (2500 RPM) was too high. It was necessary to lower the minimum speed to 2000 RPM in order to prevent the feedwater pump from continuously filling the vessel to the L-8 trip following the reactor scram. The feedwater turbine control (Woodward Governor) was respanned to a range of 2000 RPM to 5000 RPM for a 4-20 MA control output signal. The speed loop gain change had negligible effect on the flow loop response. The open loop test performed following the gain change indicated that the flow response was within the same order of magnitude as previously tested. The system is desirably set for as fast a response as possible. The slightly excessive overshoot was considered acceptable as the reactor water level was adequately controlled.

TABLE 3-22

FEEDWATER LEVEL CONTROL SYSTEM SETTINGS

	Actual	Dial	Control System Report
Master Controller (C34-K633)			
Proportional Gain (Kp)	1.03	1.9 x 1	3%/in.
Reset Gain (R/M)	0.1 x 10	.09 x 10	1.5
Startup Level Controller (C34-K645)			
Proportional Gain	.95	1.4 x 1	3%/in.
Reset Gain (R/M)	.35	.08 x 10	1.5

TABLE 3-23

STARTUP LEVEL CONTROL VALVE
OPEN PUMP FLOW RESPONSE SUMMARY

	Measured	Acceptance Criteria
Time to 10% of A Step, second	0.5	1.1
Time from 10% to 90% of A Step, second	0.75	1.9
Peak Overshoot, % of Step	25	15
Settling Time, 100% ± 5%, second	8	14

TABLE 3-24

FEEDWATER PUMP TURBINE
OPEN LOOP FLOW RESPONE SUMMARY

Feedwater Pump	Initial Flow (GPM)	Step Size (%)	Delay Time (Sec)	Response Time (Sec)	Overshoot (%)	Settling Time (Sec)
A	7×10^3	+12	0.6	1.0	25.4	2.4
		-10	0.6	0.2	58.3	2.4
	10×10^3	+ 9	0.25	0.4	24.7	2.0
		-11	0.2	0.6	34.0	2.0
	13×10^3	+11	0.5	0.5	23	1.6
		-10	0.5	0.3	42	1.5
	16×10^3	+10	0.5	0.5	11	2.1
		- 9	0.4	0.4	36	2.0
B	7×10^3	+10	0.6	0.42	12.5	1.64
		- 9	0.6	0.9	9.1	1.8
	10×10^3	+ 7	0.8	1.4	13.6	1.6
		- 8	0.7	1.4	11.1	1.7
	13×10^3	+ 7	0.8	1.0	0	2.4
		- 7	0.6	1.0	0	3.1
	16×10^3	+ 7	0.5	1.0	12	3.05
		- 7	0.6	1.0	0	2.7
Criteria			1.1	1.9	15%	14 sec

FEEDWATER TURBINE CONTROL GAIN CURVE LOOP "A"

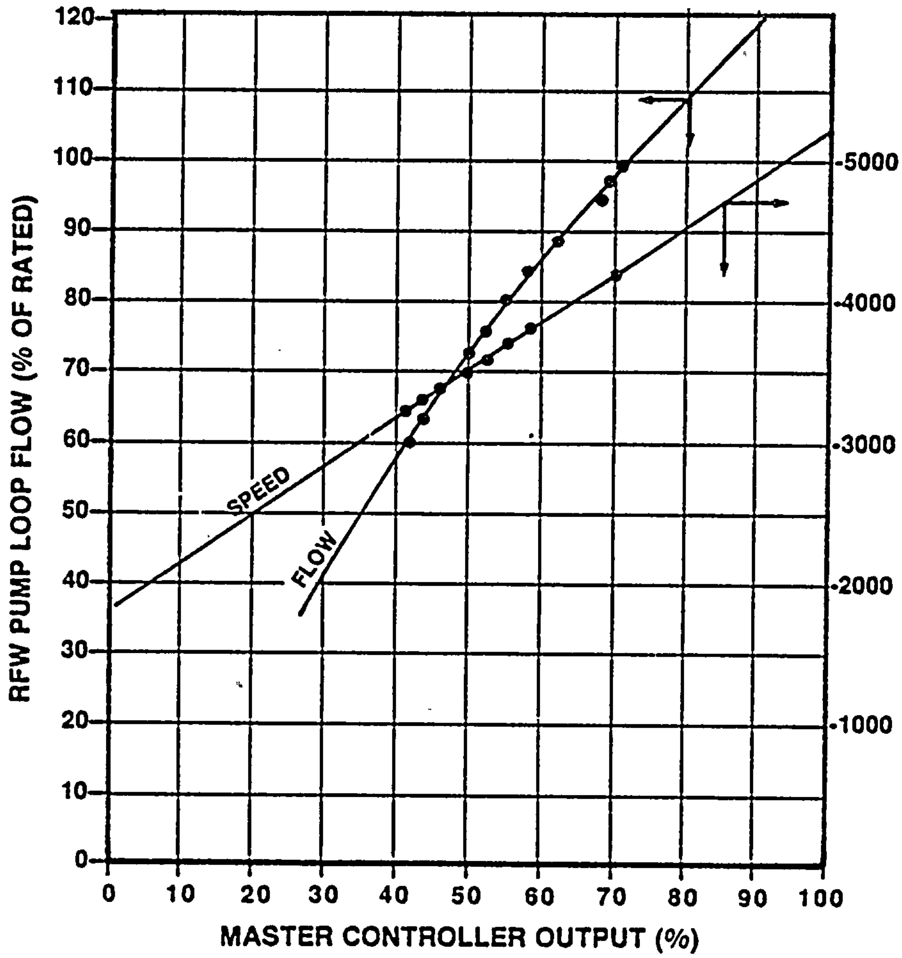

Figure 3-5

FEEDWATER TURBINE CONTROL GAIN CURVE LOOP "B"

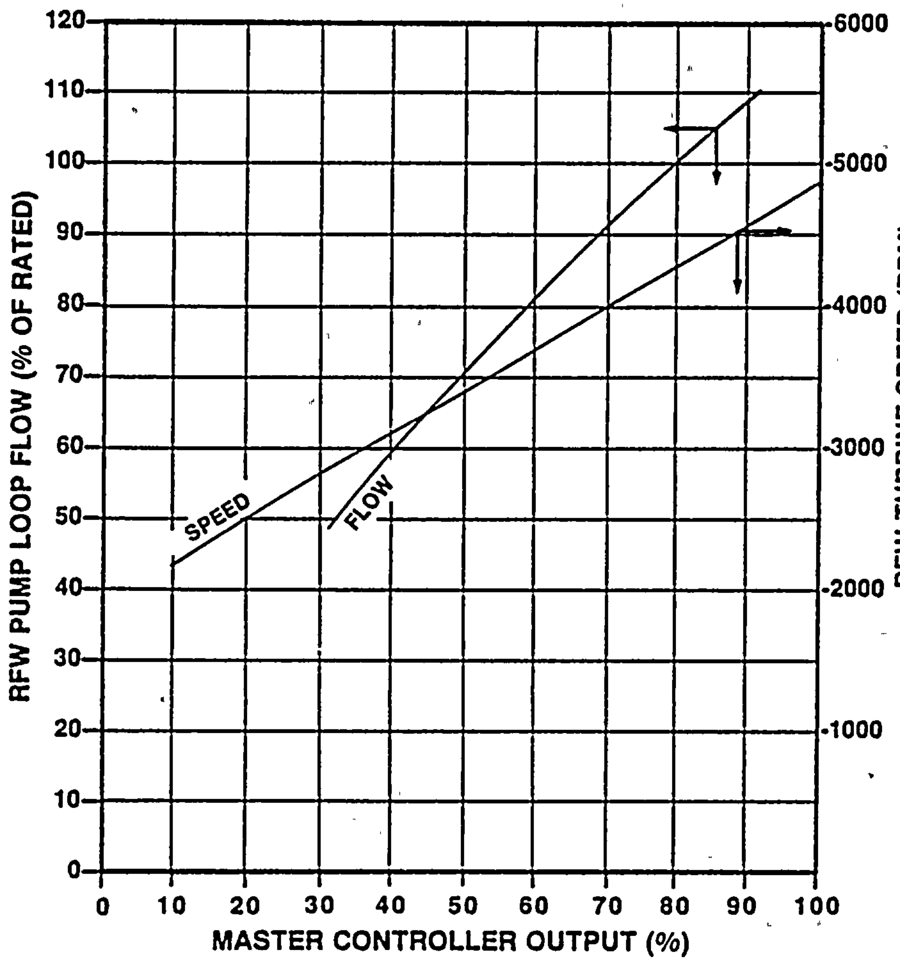

Figure 3-6

3.22 Test Number 23B - Loss of Feedwater Heating

3.22.1 Purpose

The major objective of this test is to demonstrate adequate plant response to a feedwater temperature loss.

3.22.1.1 Level 1 Criteria

For the feedwater heater loss test, the maximum feedwater temperature decrease due to a single failure case must be less than or equal to 100°F. The resultant MCPR must be greater than the fuel thermal safety limit of 1.06.

The increase in simulated heat flux cannot exceed the predicted Level 2 value by more than 2%. The predicted value will be based on the actual test values of feedwater temperature change and power level.

3.22.1.2 Level 2 Criteria

The increase in simulated heat flux cannot exceed the predicted value referenced to the actual feedwater temperature change and power level in the Transient Safety Analysis Design Report.

3.22.2 Test Results

The maximum feedwater temperature decrease due to the worst single failure case (loss of extraction steam to both high pressure heaters) was 34.11°F. Approximately ten minutes elapsed between the isolation of the high pressure heater's extraction steam and the stabilization of the feedwater temperature at its new, lower, value. The minimum critical power rate (MCPR) decreased from a pre-transient value of 1.732 to 1.675, leaving adequate margin to the fuel thermal safety limit of 1.06. The actual increase in simulated heat flux was 4.47% of rated, 0.83% of rated less than the predicted value.

3.22.3 Discussion

The loss of feedwater heating test transient was initiated by simultaneously opening extraction steam dump valves BS-DV-6A&B and closing extraction steam valves BS-V-6A&B. The transient response was within the predicted response and all criteria were satisfied.

3.23 Test Number 23C - Feedwater Pump Trip

3.23.1 Purpose

The major objective of this test is to demonstrate the capability of the automatic core flow runback feature to prevent low water level scram following the trip of one feedwater pump.

3.23.1.1 Level 1 Criteria

Not applicable

3.23.1.2 Level 2 Criteria

The reactor shall avoid low water level scram by three inches margin from an initial water level halfway between the high and low level alarm setpoints.

3.23.2 Test Results

The partial loss of feedwater transient was initiated by tripping the "A" turbine driven reactor feed pump while the reactor was operating at 98.7% rated power. The recirculation flow runback occurred when water level dropped to 31.5 inches (level 4) coincident with less than two feedpumps running. The minimum water level reached during the transient was 24.7 inches. This corresponds to a 11.55 inch level margin above the low level scram setpoint (13 inches) at 100% rated power.

3.23.3 Discussion

The feed pump trip was initiated from steady state conditions with the feedwater control system in 3-element and the recirculation control system in master manual. The resulting transient generally followed the predicted response.

3.24 Test Number 23D - Maximum Runout Capability

3.24.1 Purpose

This test calibrates the feedwater flow and determines if the maximum feedwater runout capability is compatible with the licensing basis.

3.24.1.1 Level 1 Criteria

Maximum speed attained shall not exceed the speeds which will give the following flows with the normal complement of pumps operating.

1. 135% NBR at 1075 psia

2. [135% + .2 (1075-P)] % NBR at P rated, psig

3.24.1.2 Level 2 Criteria

The maximum speed must be greater than the calculated speeds required to supply:

1. With rated complement of pumps -115% NBR at 1,075 psi.

2. One feedwater pump tripped condition -68% NBR at 1,025 psia.

3.24.2 Test Results

The feedwater turbine high speed limit would be acceptable 5450 RPM but was set conservatively at 5075 RPM. The feedwater pump maximum runout capability was calculated and summarized in Table 3-25.

3.23.3 Discussion

During power ascension to 100% power feedwater turbine and pump data was taken every 2% power increment. A system demand curve was generated by fitting a curve through the test data and extrapolating to maximum speed. This curve was then adjusted for the reactor pressure specified in the acceptance criteria and superimposed on a family of pump head curves at different pump speeds. Intersection of these curves determined pump runout flows. The high speed limit of 5450 RPM was determined to be acceptable. A very conservative limit of 5075 RPM was implemented by WNP-2. Both of these limits would allow the feedwater system to meet all the acceptance criteria and still allow acceptable plant maneuverability.

TABLE 3-25

FEEDWATER MAXIMUM RUNOUT CAPABILITY SUMMARY

Reactor Pressure	Feedwater Pump	5075 RPM Calculated	5450 RPM Calculated	Required Level 1	Required Level 2
1075 PSIA	A + B	119%	128%	135%	115%
1020 PSIA	A + B	122%	131%	146%	NA
1025 PSIA	A	73%	82%*	NA	68%
1025 PSIA	B	73%	82%*	NA	68%

* From best estimate of demand curve, 3 condensate pumps, 3 booster pumps, 1025 PSIA.

3.25 Test Number 24 - Turbine Valve Surveillance

3.25.1 Purpose

The major objectives of this test are as follows:

1. to determine the maximum power at which periodic surveillance testing of the main turbine control valves can be performed without causing a reactor scram;

2. to determine the valve testing procedure which may be used as a guidance for preparing the surveillance procedure;

3. to establish baseline data for evaluating test condition proximity to the PCIOMR envelope during future testing activity.

3.25.1.1 Level 1 Criteria

The decay ratio of any oscillatory response must be less than 1.0.

3.25.1.2 Level 2 Criteria

Peak neutron flux; less than 7.5% below scram trip setting (118%).

Peak heat flux; less than or equal to 5% below scram trip setting (.66 Wrt + 51%), where Wrt is the percent of recirculation flow.

Peak vessel pressure; less than or equal to 10 psi below scram trip setting (1037 psig).

Peak steam flow; less than or equal to 10% below the high flow isolation trip setting (104 psid).

Decay ratio of any oscillatory response must be less than .25.

NOTE: The data collected relative to the PCIOMR limits will be used for determining proximity to the preconditioned or threshold power level during furture surveillance testing.

3.25.2 Test Results

Results of governor valve and throttle valve closure tests along the 100% load line are displayed in Figure 3-6. Bypass valve and interceptor/reheat stop valve testing results are not included as the data showed very little reactor system response during these transients. The maximum power level for valve survellance was determined to be 80% with the DEH in mode 4.

3.25.3 Discussion

Turbine valve surveillance tests were successfully performed at 65% and 80% power along the 100% load line. While attempting to perform the governor valve test at 85% power, limit cycles developed in most pressure control system related process variables. Valve closure testing was halted and 80% power was set as the maximum power level at which turbine valve surveillances are to be performed. It is believed that the limit cycles occurred at 85% power due to the closure of GV 2 of 3 concurrent with GV 1 and 4 passing through a break point in their gain curves. No limit cycles or oscillations occur during the turbine valve surveillance test when conducted at 80% power. In addition, data evaluation determined that an increase in power level to greater than 85% to preclude the limit cyclying may not have provided sufficient margins as defined by the Level 2 criteria. 80% power was thereby selected as the required power level for the technical specification turbine valve surveillance testing and the appropriate procedures were revised.

TURBINE VALVE SURVEILLANCE RESULTS

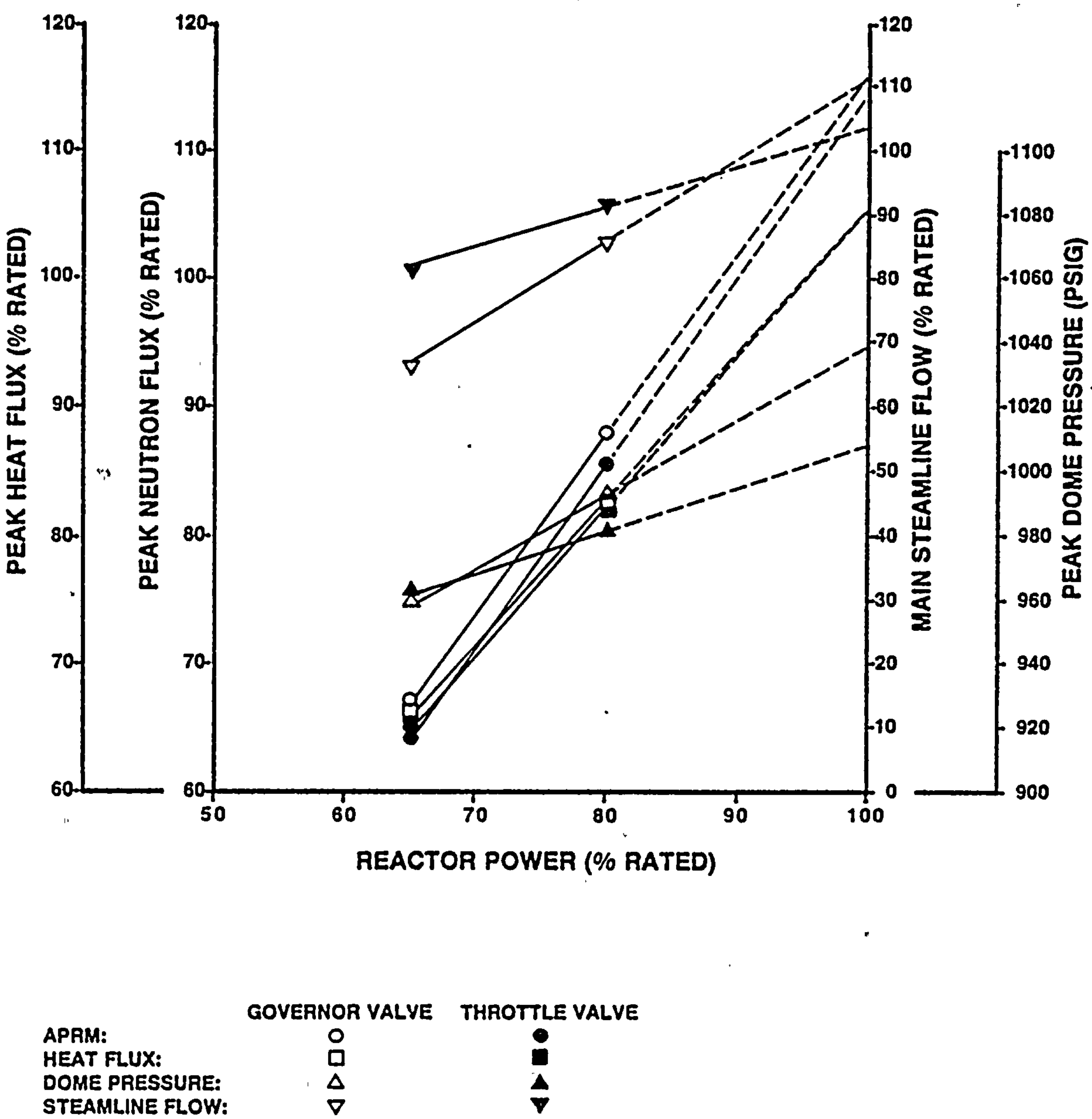

Figure 3-7

850207.15A

3.26 Test Number 25A - MSIV Functional Test

3.26.1 Purpose

The major objectives of this test are as follows:

1. to functionally check the Main Steam Isolation valves for proper operation at selected power levels;
2. to determine MSIV closure time;
3. to determine the maximum power at which full closure of a single MSIV can be performed without a scram.

3.26.1.1 Level 1 Criteria

The MSIV stroke time (ts) shall be no faster than 3.0 seconds (average of the fastest valve in each steam line) and for any individual valve ts shall be between 2.5 and 5.0 seconds. Total effective closure time for any individual MSIV shall be t_{sol} plus the maximum instrumentation delay time as determined by pre-operational test and shall be less than or equal to 5.5 seconds.

3.26.1.2 Level 2 Criteria

The reactor shal not scram or isolate.

During full closure of individual valves peak reactor vessel pressure must be 10 psi (0.7 kg/cm^2) below the scram setpoint, peak neutron flux must be 7.5% below the scram setpoint, and steam flow in individual line must be 10% below the isolation trip setting. The peak heat flux must be 5% less than its trip point.

3.26.2 Test Results

Valve closure times of all eight MSIV's from three test points are presented in Table 3-26. During these tests both RPS and position indicating limit switches were observed for indication of proper valve and logic operation.

Results of the single valve fast closure tests indicate the MSIV single valve full closure surveillances can be performed at below 85% power.

3.26.3 Discussion

Figures 3-7, 3-8, 3-9, and 3-10 depict the results of data extrapolation to determine the maximum power level where MSIV single valve full closure surveillance can safely be performed. Presented are peak reactor pressure, peak neutron flux, peak simulated heat flux, each as a function of power level. The extrapolation indicates that at 85% power and above the required margins to scram are not satisfied for the steam line flow parameter.

TABLE 3-26

MSIV CLOSURE TIMES

Valve Description	TEST RESULTS Heatup (Seconds)	28% Power (Seconds)	97% Power (Seconds)
Inboard A, F022A	4.031	3.817	3.655
Outboard A, F028A	3.753	3.599	3.841
Inboard B, F022B	4.031	4.115	3.446
Outboard B, F028B	4.309	4.441	4.213
Inboard C, F022C	3.763	4.251	3.910
Outboard C, F028C	3.614	3.617	3.564
Inboard D, F022D	4.031	4.061	3.711
Outboard D, F028D	3.700	3.617	3.601

MSIV SURVEILLANCE:
PEAK REACTOR DOME PRESSURE

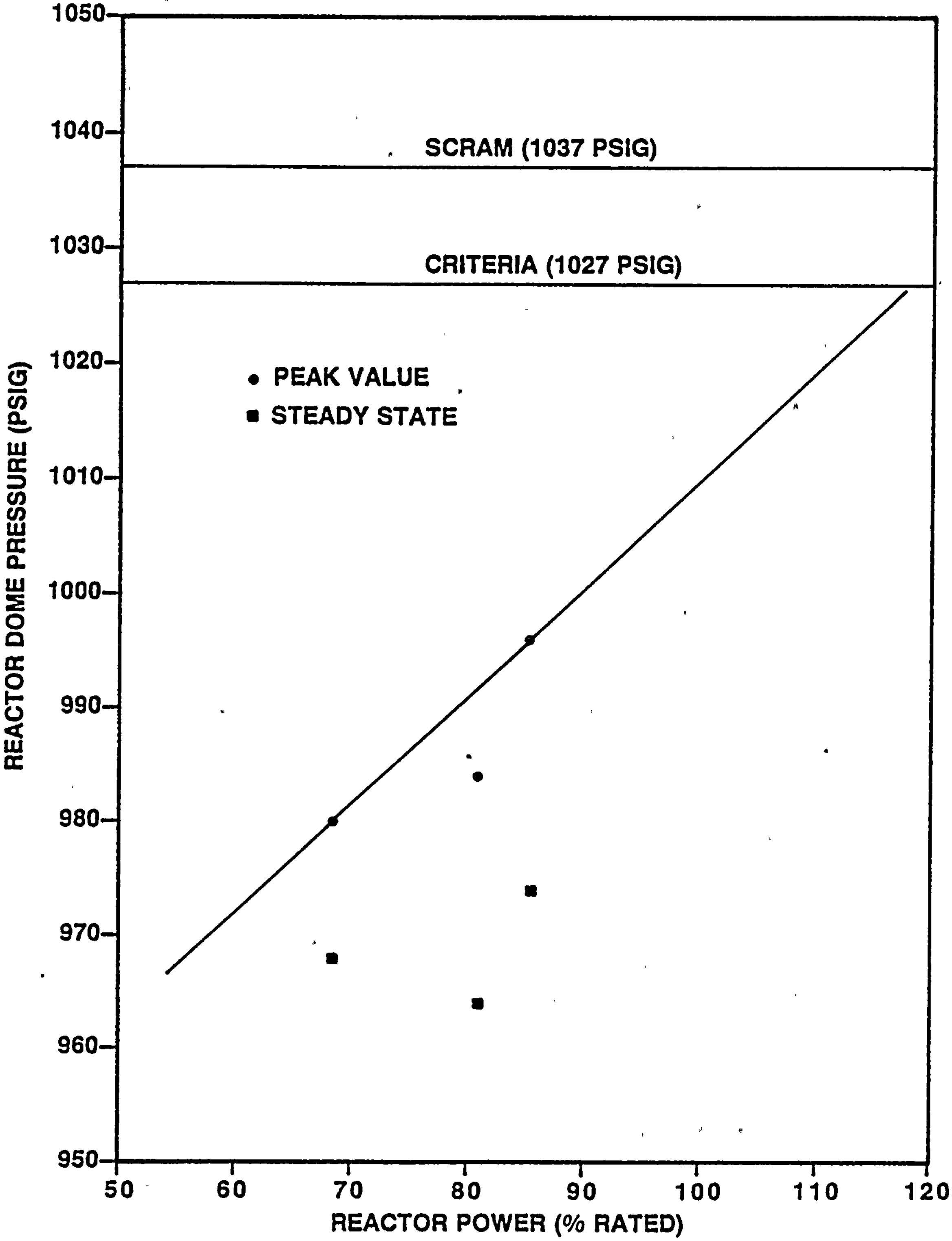

Figure 3-8

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MSIV SURVEILLANCE:
PEAK NEUTRON FLUX

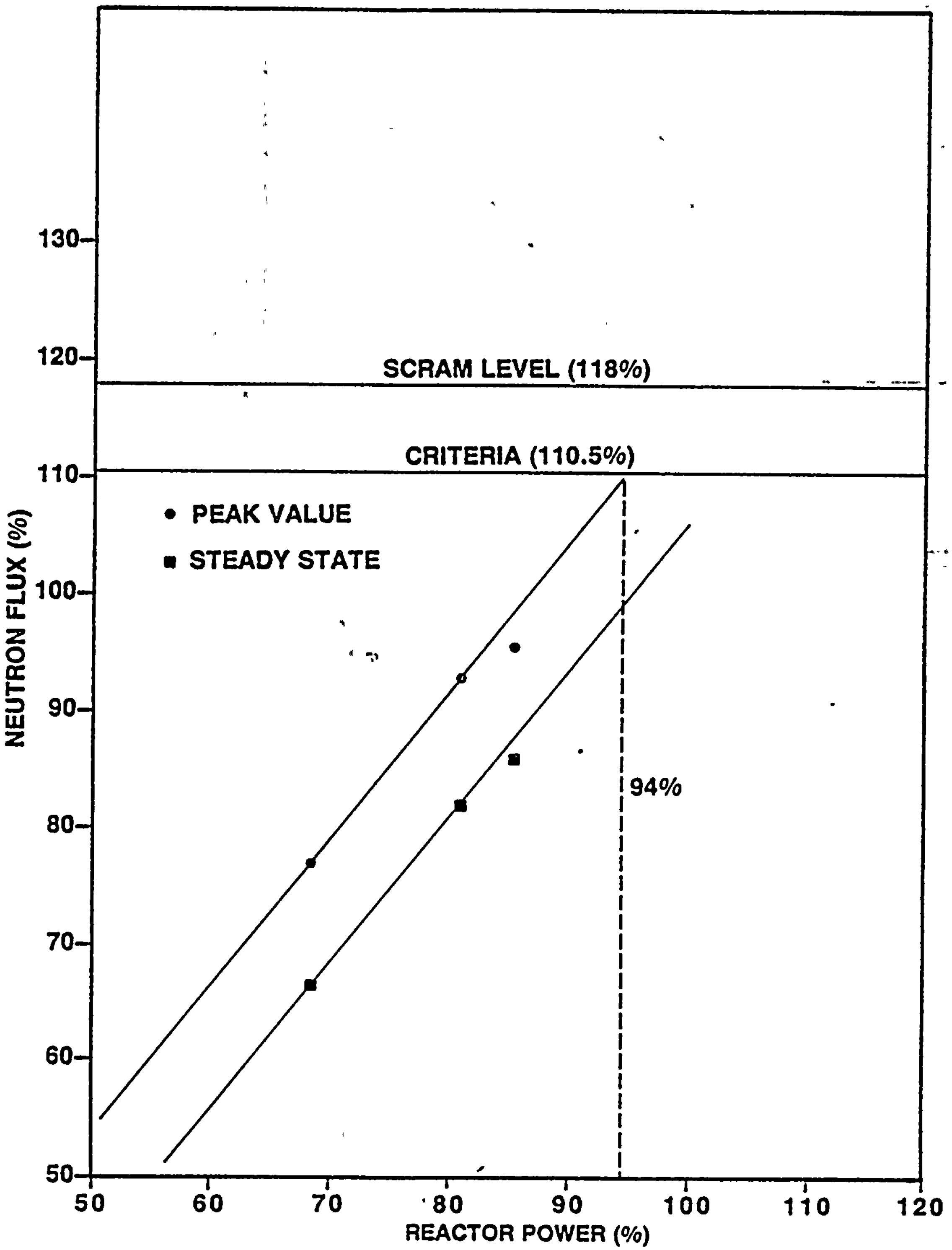

Figure 3-9

MSIV SURVEILLANCE: PEAK HEAT FLUX

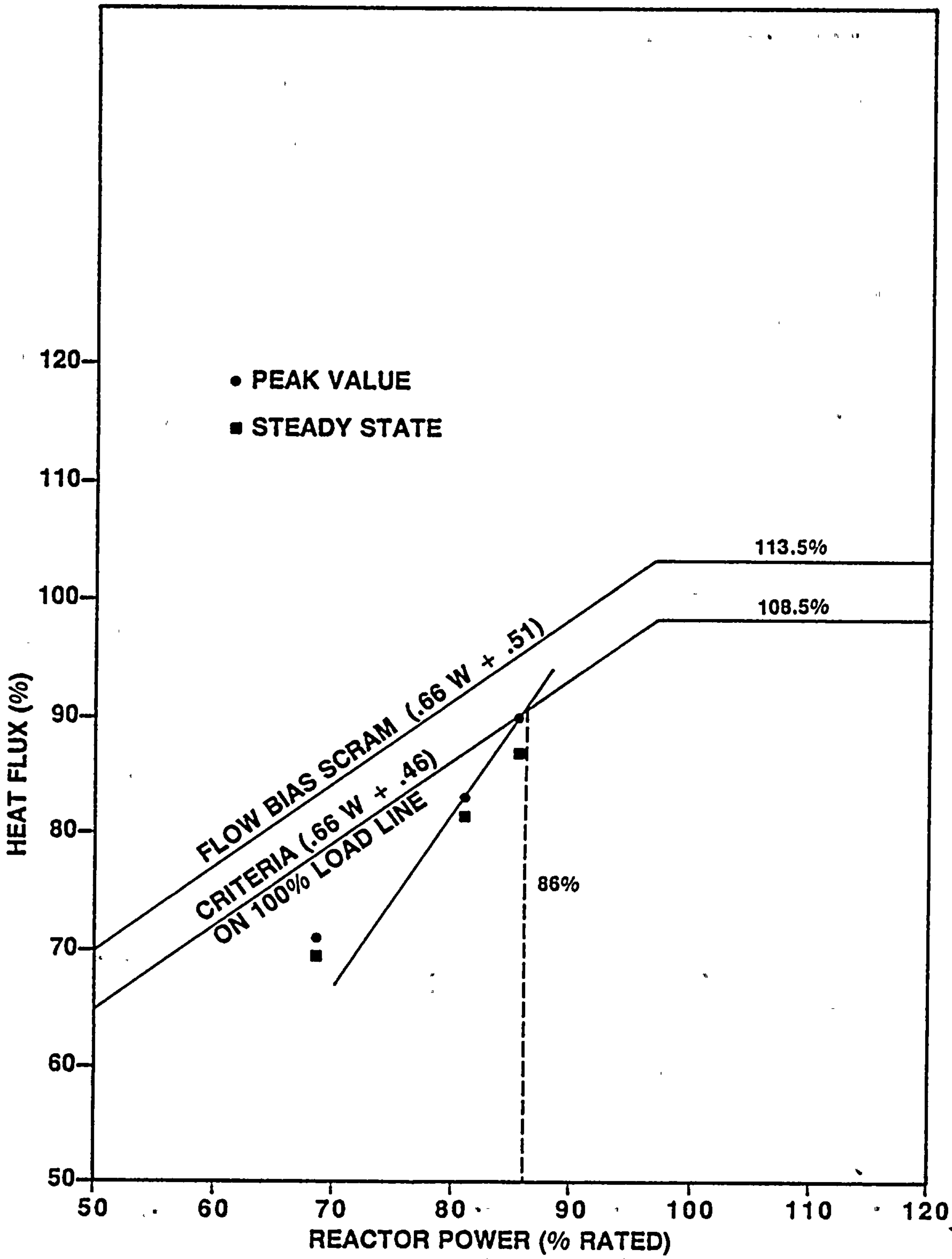

Figure 3-10

MSIV SURVEILLANCE: PEAK STEAMLINE FLOW

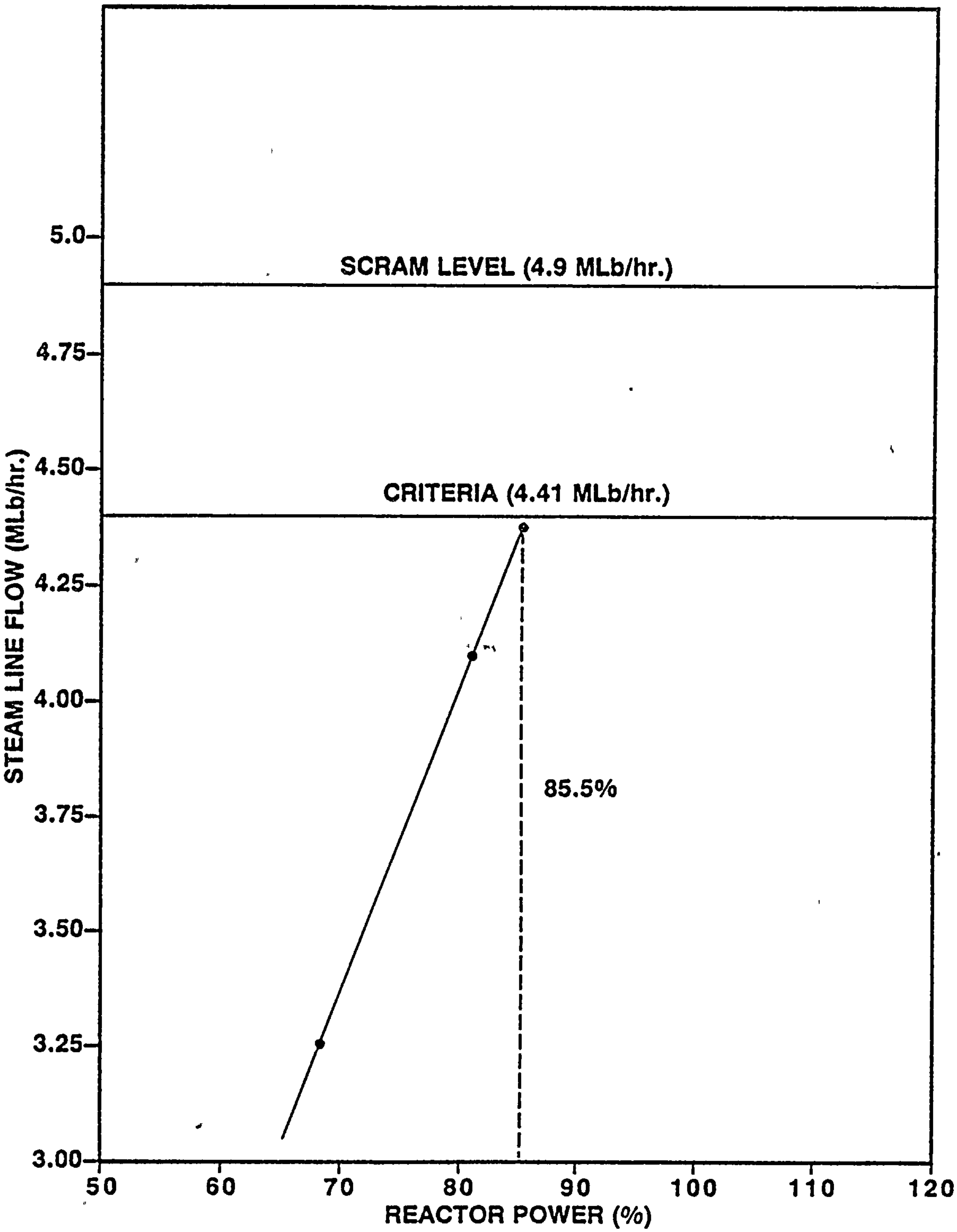

Figure 3-11

3.27 Test Number 25B - Reactor Full Isolation Test

3.27.1 Purpose

The major objective of this test is to determine the reactor transient behavior resulting from the simultaneous full closure of all MSIV's.

3.27.1.1 Level 1 Criteria

Reactor must scram to limit the severity of the neutron flux and simulated fuel surface heat flux transient.

Feedwater control system settings must prevent flooding of the steam line.

The recorded MSIV full closure times must meet the functional test criteria. This criteria is discussed in Test Number 25A of this report.

The positive change in vessel dome pressure occurring within 30 seconds after closure of all MSIV valves must not exceed the level 2 criteria by more than 25 psi. The positive change in simulated heat flux shall not exceed the level 2 criteria by more than 2% of rated value.

3.27.1.2 Level 2 Criteria

The temperature measured by the thermocouples on the discharge side of the safety/relief valves must return to within 10°F of the temperature recorded before the valve was opened.

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

Initial Conditions		Criteria	
Power (%)	Dome Pressure (psia)	Increase In Heat Flux (%)	Increase In Dome Pressure psi
100	1020	0.0	125

Initial action of RCIC and HPCS shall be automatic if low water level (L2) is reached, and system performance shall be within specification.

Recirculation pump trip shall be initiated when reactor water level 2 is reached.

3.27.2 Test Results

During the full isolation test from 97% power the dome pressure rise was 79 psi. Four safety/relief valves opened during the initial transient and five more were opened manually to control pressure later in the test. Minimum water level was -42 inches which caused the RCIC system to initiate. Water level reached a maximum of +93 inches which tripped all sources of water to the vessel and prevented flooding of the main steam lines.

3.27.3 Discussion

The MSIV full closure transient followed predicted analytical results very closely in that no increase in simulated heat flux was observed and peak reactor pressure was within criteria limits. Major systems worked properly with two minor exceptions, RCIC initiation occurred at -42 inches reactor water level instead of -50 inches (level 2) and one safety/relief valve opened at 1069 psig, 7 psig below the design lowest SRV setpoint. Both were later corrected by instrument recalibration.

3.28 Test Number 26 - Relief Valves

3.28.1 Purpose

The purpose of this test is a) to verify the proper operation of the main steam relief valves, b) to verify that the discharge piping is not blocked, c) to verify their proper seating following operation, d) to obtain signature information of relief valve response for subsequent comparisons, and e) to determine their capacities.

3.28.1.1 Level 1 Criteria

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

The sum of capacity measurements from all relief valves shall be equal to or greater than 15.8 x 10^6 lb/hr at an inlet pressure of 103% of 1,205 psig.

The total flow capacity of the safety relief valves used in the Automatic Depressurization system must be equal to or greater than 4.8 x 10^6 lb/hr at 1,125 psig when the valve having the highest measured capacity is assumed to be out of service.

3.28.1.2 Level 2 Criteria

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. The thermocouples are expected to be operating properly.

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.

Each relief valve shall have a capacity between 90% and 122.5% of its expected value corrected to an inlet pressure of 103% of 1,205 psig.

No more than 25% of the relief valves may have an individual corrected flow rate that is between 90 and 100% of their expected flow rates.

The transient recorder signatures for each valve must be analyzed for relative system response comparison.

3.28.2 Test Results

During the initial heatup proper operation of the safety relief valves was verified by demonstrating that relief valve steam was discharged to the suppression pool and that the valves reseated after actuation. This was accomplished by cycling each relief valve and recording tailpipe temperature prior to and after relief valve actuation. In addition, accoustic monitors were tuned to adequately indicate the discharge of steam to the suppression pool and the reseating of the relief valves.

At about 50% power the capacity of each SRV was measured. The results are presented in Table 3-27. Total SRV capacity extrapolated to 1256.3 psia was 18.35 x 10^6 Lb/Hr. The total capacity of SRV's used in ADS was 5.7 x 10^6 Lb/Hr. All individual relief valves except MS-RV-2D have the corrected flow rates within the Level 2 acceptance ranges. The capacity of MS-RV-2D was found to be 0.2 x 10^6 Lb/Hr less than the expected flow rate.

A correlation of steam flow and bypass valve position was first established by measuring the feedwater flow change with varying reactor thermal power level which changed the bypass valve position. The relationship of feedwater flow and bypass valve position is shown in Figure 3-11. The SRV capacity was determined by the steam flow change as measured by the difference between the initial and final bypass valve positions.

The capacity of MS-RV-2D was found to be outside the Level 2 acceptance range. The total SRV capacity was found to be 14% higher than required. In addition, MS-RV-2D is not an ADS valve. Therefore the deviation was considered as acceptable.

The curve depicting BPV position versus changes in feedwater flow (Fig. 3-11) is provided to present data collected in determining total BPV capacity. The data used to determine SRV capacity ranged from 5 to 50% total BPV position The data points are listed below;

Total Bypass Position (%)	Feedwater Flow (Mlb/Hr) Differential
4.6	.18
8.8	.28
14.1	.35
18.7	.46
23.1	.53
28.2	.62
33.5	.83
39.6	1.1
44.9	1.33
49.4	1.55

TABLE 3-27

SRV PERFORMANCE DATA

Relief Valve	BPV Total Response %	SRV Capacity (x 10^6 Lb/Hr
MS-RV-3B	17.62	1.10
MS-RV-4D	17.54	1.04
MS-RV-2B	17.20	1.05
MS-RV-3C	16.90	1.04
MS-RV-2A	16.85	1.04
MS-RV-1A	17.42	1.05
MS-RV-1D	16.42	0.98
MS-RV-1C	17.05	1.05
MS-RV-4C	14.59	0.93
MS-RV-5C	17.19	1.04
MS-RV-3A	16.85	1.04
MS-RV-2D	13.50	0.71
MS-RV-2C	16.64	1.03
MS-RV-1B	17.20	1.05
MS-RV-4B	17.24	1.05
MS-RV-4A	17.54	1.05
MS-RV-5B	17.57	1.06
MS-RV-3D	16.83	1.04

BYPASS VALVE CALIBRATION CURVE

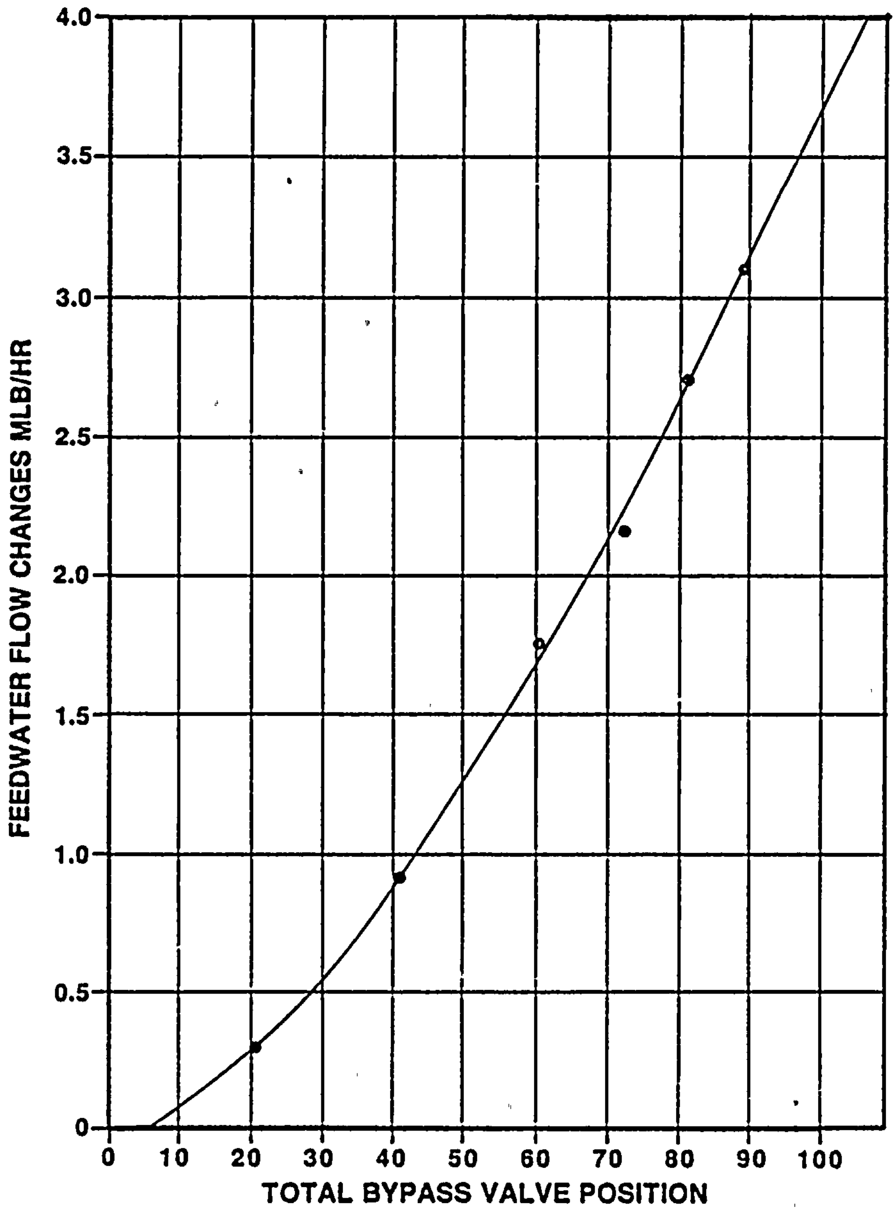

Figure 3-12

850207.20A

3.29 Test Number 27 - Turbine Trip and Generator Load Rejection

3.29.1 Purpose

The purpose of this test is to demonstrate the response of the reactor and its control systems to protective trips in the turbine and the generator.

3.29.1.1 Level 1 Criteria

1. For turbine and generator trips above 50% nuclear boiler rated steam flow, there must be a delay of less than 0.1 seconds following the beginning of control or stop valve closure before the beginning of bypass valve opening. The bypass valves must be opened to a point corresponding to approximately 80 percent of their capacity within an additional 0.2 seconds, or 0.3 seconds total, from the beginning of control or stop valve closure motion.

2. Feedwater system settings must prevent flooding of the steam lines following these transients.

3. The two recirculation pump drive flow coastdown transient during the first six seconds must be bounded by criteria specified in Power Ascension Test 8.2.30, Recirculation System Performance.

4. The positive change in vessel dome pressure occurring within 30 seconds after either generator or turbine trip must not exceed the Level 2 criteria by more than 25 psi.

5. The positive change in simulated heat flux shall not exceed the Level 2 criteria by more than 2% of rated value.

6. The total time delay from start to breakers throttle valve or governor valve motion to the complete suppression of electrical arc between the fully open contacts of the circuit breakers (3A, 3B, 4A, 4B) shall be less than or equal to 190 milliseconds.

3.29.1.2 Level 2 Criteria

1. There shall be not MSIV closure in the first three minutes of the transient and operator action should not be required in the period to avoid an MSIV isolation.

2. The positive change in vessel dome pressure and in simulated heat flux which occurs within the first 30 seconds after the initiation of either generator or turbine trip must not exceed the predicted values from the WNP-2 Transient Safety Analysis Design Report.

   NOTE: Predicted values will be referenced to actual test conditions of initial power level and dome pressure, and will use BOL (Beginning of Life) nuclear data. Worst case design or technical specification values of all hardware performance shall be used in prediction, with the exception of control rod insertion time and the delay from beginning of turbine control valve or stop valve motion to the generation of the scram signal. The predicted pressure and heat flux will be corrected for the actual measured values of these two parameters.

3. Electrical load transfers should occur as designed.

4. The reactor should not scram for initial thermal power at less than or equal to 25% of rated.

5. The measured bypass capacity (in percent of rated power) shall be equal to or greater than used for FSAR safety analyses (3,576,000 lbm/hr).

6. Recirculation LFMG sets shall take over after the initial recirculation pump trips and adequate vessel temperature difference should be maintained.

7. Feedwater level control shall avoid loss of feedwater due to possible high level (L8) trip during the event.

8. Low water level (L2) total recirculation pump trip, HPCS and RCIC should not be initiated.

9. The temperature measured by thermocouples on the discharge side of the safety/relief valves must return to within 10°F of the temperature recorded before the valve was opened. In addition the acoustical monitors should indicate the valve is closed after the transient is complete.

3.29.2 Test Results

During the Generator Load Rejection test within bypass valve capacity, the pressure regulator performed adequately; maintaining pressure at 920 psig with a transient peak of 958 psig. The reactor did not scram and the electrical load transferred properly.

The results of high power turbine trip and generator load rejection are contained in Table 3-28.

3.29.3 Discussion

The two recirculation pump RPT trip showed the drive flow coastdown was faster than the minimum ECCS coastdown criteria during the turbine trip and generator load rejection test. The test exception was analyzed and accepted by General Electric Transient Performance Engineer (refer to test 30B discussion of this report).

A temporary procedure was written to simulate a RFW pump trip and auto level setdown during a scram. This would provide data that would be useful in designing the features necessary to the feedwater control system that precluded the system from tripping following a scram. The indicated level transient immediately following a scram had on past scrams caused a feedwater turbine trip on Level 8.

The temporary feedwater procedure was performed during the generator load rejection test at TC-6. The operating feedwater pump however, eventially tripped on level 8. The reason for level 8 trip of the second feedwater pump is because of the low reactor decay heat and the resulting drop in reactor pressure that enabled feedwater to continually feed the vessel. At higher decay heat levels, the existing control system setting could possibly be sufficient. Additional attention and design changes are currently under investigation.

The load rejection transient was simulated by utilizing the main generator trip pushbutton at TC-6. The turbine overspeed reached 1900 RPM at the time of the turbine trip.

A slow bypass valve opening time condition during testing in Test Condition 3 was caused by the DEH Isolation Amplifier to BPV 2 and 4 causing excessive signal response time delay compared to the signal conditioning circuit for the 1 & 3 valves. This was corrected and the BPV opening time was later demonstrated to be within 300 milliseconds criteria. During the BPV response time troubleshooting process, modifications were made to improve BPV response. The main steam bypass valve individual accumulators were increased from 1 gallon to 2-1/2 gallon rated capacity. The DEH hydraulic oil supply line to the accumulators were also rerouted to eliminate about 150 psig pressure drop from the DEH pumps to the bypass valves.

TABLE 3-28

SUMMARY OF TURBINE TRIP AND GENERATOR LOAD REJECTION TEST RESULTS

CRITERIA	LEVEL 1	LEVEL 2	ACTUAL RESULT TC-3	TC-6
Time between control valve closure and bypass valve opening	.1 (sec)	n/a	0.093 (sec)	0.010 (sec)
Time between stop valve closure and bypass valves at 80% capacity	.3 (sec)	n/a	2.742 Average	0.168 (sec) Average
Minimum water level to prevent steamline flooding and reactor feed pump trip	107 inches	55 inches	52 "	55", L-8 Trip
Recirculation pump coastdown bounded by criteria curves	6 (sec)	n/a	6 sec transient 0 sec ECCS	6 sec transient 0 sec ECCS
Positive change in vessel dome pressure within 30 seconds of turbine trip	96 (psi)TC-3 134.5(psi)TC-6	71 (psi)TC-3 109.5(psi)TC-6	65 (psi)	68 (psi)
Positive change in simulated heat flux	2%	0%	0%	0%
Total delay from start of control valve motion to the complete suppression of electrical arc	.190 sec	n/a	0.175 (sec)	0.150 (sec)
MSIV isolation shall not occur in the first 3 minutes, nor shall operator take action to avoid MSIV trip	n/a	no isolation no operator action	no isolation no operator action	no isolation no operator action
Measured bypass valve capacity shall be assumed value in FSAR	n/a	3.565 Mlb/hr	3.6 mlb/hr	3.6 mlb/hr
Recirc transfer to LFMG	n/a	transfer to LFMG, no temperature stratification	transfer to LFMG, no temperature stratification	transfer to LFMG, no temperature stratification
Low water level recirc pump trip, HPCS and RCIC shall not initiate	n/a	-50 inchs	- 3 inches	+ 1 inch
Discharge side of S/RV must return in 10°F of initial	n/a	10°F	no SRV actuation	MS-RV-3B below 215°F with initial 207°F

3.30 Test Number 28 - Shutdown From Outside the Control Room

3.30.1 Purpose

The major objectives of this test are as follows:

To demonstrate that the reactor can be brought down from a normal initial steady-state power to the point where cooldown is established, and to demonstrate that the reactor vessel pressure and water level can be controlled from outside the control room.

3.30.1.1 Level 1 Criteria

Not applicable

3.30.1.2 Level 2 Criteria

During a simulated main control room evacuation, the reactor must be brought to the point where cooldown can be initiated, and the reactor vessel pressure and water level must be controlled using equipment and controls, outside the main control room.

3.30.2 Test Results

During TC-6, the reactor was scrammed following the turbine load reject test at 100% power. At this point in the test program it was decided to enter into a short outage for various repairs and perform the RHR SDC portion of this test. The testing at TC-6 amounted to successfully entering into RHR shutdown cooling. That activity was completed. During TC #1, the ability to maintain reactor level with the RCIC system and pressure with SRV's at the remote shutdown panel was verified. RHR suppression pool cooling was also placed in service at that time. The combination of each of the tests provided valuable information leading to a revision to the procedure used to evacuate the control room & utilize the remote shutdown equipment. The revision was also successfully demonstrated on the WNP-2 simulator which provided for additional minor procedure changes.

3.30.3 Discussion

During TC-1 the reactor was manually scrammed and the MSIV's were closed prior to evacuating the control room. Operators located at the Remote Shutdown Panel in the Radwaste Building monitored the ensuing reactor transient. No action was required in the main control room to maintain the plant in a safe condition. The relief valves were operated from the Remote Shutdown Panel and RCIC was initiated to demonstrate operability. The RHR system was placed in suppression pool cooling mode of operation from the Remote Shutdown Panel.

Following the reactor scram at TC-6, the RHR B loop was placed into shutdown cooling mode from the Remote Shutdown Panel after reactor pressure was reduced to 75 psig. The minimum required 50°F/hr cooldown rate was maintained which demonstrated the ability to cooldown the reactor from outside the main control room.

3.31 Test Number 29A - Flow Control-Valve Position Control

3.31.1 Purpose

The purpose of this test is to demonstrate the recirculation flow control systems capability while in the valve position (POS) mode.

3.31.1.1 Level 1 Criteria

The transient response of any recircultion system-related variables to any test input must not diverge.

3.31.1.2 Level 2 Criteria

1. Recirculation system related variables may contain oscillatory modes of response. In these cases, the decay ratio of each controlled mode of response must be less than or equal to 0.25.

2. Maximum rate of change of valve position shall be 10 $\pm$ 1%/sec.

   During TC-3 and TC-6 while operating on the high speed (60 Hz) source, gains and limiters shall be set to obtain the following response.

3. Delay time for position demand step shall be:

   For step inputs of 0.5% to 5% less than or equal to 0.15 sec.

4. Response time for position demand step shall be:

   For step inputs of 0.5% to 5% less than or equal to 0.45 sec.

5. Overshoot after a small position demand input (1 to 5%) step shall be less than 10% of magnitude of input.

3.31.2 Test Results

The transient response of any recirculation system-related variable to any step change did not diverge. The decay ratio of the transient response of the valve position loop was always less than 0.25. The maximum stroking rate of the flow control valve in both opening and closing directions was 10.5%/sec. The valve response results during TC-3 and TC-6 are summarized in Tables 3-29 through 3-32. In addition, the position loop deadband was found to be less than 0.2% of full stroke.

3.31.3 Discussion

The delay time and response time of the flow control valve response were not within the Level 2 acceptance limits. However, in many cases the combination of these two has satisfied the intent of the individual criteria. The peak overshoot was satisfied for the larger step (5%). The overshoot for smaller steps (0.2% and 0.5%) was approximately double the criteria. In general the valve position control loops were optimized such that Level 2 criteria were satisfied. The small deviations from the Level 2 criteria are considered to be insignificant and acceptable.

A problem of excessive valve duty cycle was encountered during early stage of testing. In order to rectify the problem an FDDR which was recommended by GE was implemented. In the FDDR the integral gain circuit was removed from the velocity servo controller and the derivative gain from the position controller. As a result of this modification the valve duty cycle was eliminated.

TABLE 3-29

VALVE POSITION CONTROL DATA 'A' LOOP - TC-3

INITIAL VALVE POSITION %	STEP SIZE %	DELAY TIME (sec) *≦0.15	RESPONSE TIME (sec) *≦0.45	% OVER-SHOOT *≦10%	COMB. DELAY & RESPONSE *≦0.6
15	0.5(D)	0.2	0.35	10	0.55
15	0.5(U)	0.2	0.3	10	0.5
15	1(D)	0.2	0.2	13.9	0.4
15	1(U)	0.2	0.2	11	0.4
15	5(D)	0.2	0.5	4	0.7
15	5(U)	0.2	0.5	3	0.7
25	0.5(D)	0.2	0.45	25	0.65
25	0.5(U)	0.2	0.3	7	0.5
25	1(D)	0.15	0.2	13.3	0.35
25	1(U)	0.15	0.2	13.3	0.35
25	5(D)	0.2	0.55	3	0.75
25	5(U)	0.15	0.6	3	0.75
50	0.5(D)	0.2	0.1	20	0.3
50	0.5(U)	0.2	0.2	20	0.4
50	1(D)	0.2	0.4	16.6	0.6
50	1(U)	0.2	0.2	21.7	0.4
50	5(D)	0.2	0.55	5	0.75
--	5(D)	0.2	0.44	4.5	0.64
50	5(U)	0.2	0.6	4	0.8
--	----	0.2	0.44	4.5	0.64
75	0.5(D)	0.2	0.2	20	0.4
75	0.5(U)	0.3	0.3	13	0.6
75	1(D)	0.2	0.2	10	0.4
75	1(U)	0.2	0.2	16	0.4
75	5(D)	0.2	0.55	3	0.75
75	5(U)	0.2	0.6	3	0.8

*Acceptance Criteria
(D) = down; (U) = up
--- indicates data inclusive

TABLE 3-30

VALVE POSITION CONTROL DATA 'B' LOOP - TC-3

INITIAL VALVE POSITION %	STEP SIZE %	DELAY TIME (sec) *≦0.15	RESPONSE TIME (sec) *≦0.45	% OVER-SHOOT *≦10%	COMB. DELAY & RESPONSE *≦0.6
15	0.5(D)	0.4	0.5	13.3	0.9
15	0.5(U)	0.5	0.4	33.3	0.9
15	1(D)	0.25	0.3	17	0.55
15	1(U)	0.25	0.3	20	0.55
15	5(D)	0.2	0.4	5.0	0.6
15	5(U)	0.2	0.4	5.0	0.6
25	0.5(D)	0.5	0.3	10	0.8
25	0.5(U)	0.4	0.3	10	0.7
25	1(D)	0.2	0.3	10	0.5
25	1(U)	0.3	0.2	17	0.5
25	5(D)	0.2	0.4	4	0.6
25	5(U)	0.2	0.4	4	0.6
50	0.5(D)	0.4	0.3	10	0.7
50	0.5(U)	0.4	0.3	10	0.7
50	1(D)	0.3	0.35	10	0.65
50	1(U)	0.3	0.2	17	0.5
50	5(D)	0.2	0.4	4	0.6
50	5(D)	0.2	0.4	4	0.6
75	0.5(D)	0.2	0.4	10	0.6
75	0.5(U)	0.3	0.3	10	0.6
75	1(D)	0.2	0.3	10	0.5
75	1(U)	0.2	0.3	12.5	0.5
75	5(D)	0.2	0.4	2	0.6
75	5(U)	0.2	0.4	2	0.6

*Acceptance Criteria
(D) = down; (U) = up
--- indicates data inclusive

TABLE 3-31

VALVE POSITION CONTROL LOOP A RESPONSE SUMMARY TC-6

Recirc Loop	Initial FCV Position	Initial Core Flow	Step Size	Delay Time (Sec)	Response Time (Sec)	Overshoot (%)	Setting Time (Sec)
A	23%	60%	-0.5%	0.7	0.4	37.5	4.2
A	23%	60%	+0.5%	0.6	0.5	27.5	4.7
A	23%	60%	- 1%	0.2	0.4	22.2	1.5
A	23%	60%	+ 1%	0.3	0.4	16.7	1.2
A	23%	60%	- 5%	0.2	0.4	8.2	0.8
A	23%	60%	+ 5%	0.2	0.5	3.3	0
A	37%	75%	-0.5%	0.5	0.5	10.0	2.5
A	37%	75%	+0.5%	0.5	0.4	6.9	1.4
A	37%	75%	- 1%	0.3	0.5	4.5	0
A	37%	75%	+ 1%	0.2	0.4	10	1.4
A	37%	75%	- 5%	0.2	0.4	1.6	0
A	37%	75%	+ 5%	0.2	0.3	2.4	0
A	64%	95%	-0.5%	0.5	0.5	0	0
A	64%	95%	+0.5%	0.4	0.8	0	0
A	64%	95%	- 1%	0.2	0.5	0	0
A	64%	95%	+ 1%	0.2	0.5	0	0
A	64%	95%	- 5%	0.2	0.5	0	0
A	64%	95%	+ 5%	0.3	0.4	0	0
Acceptance Criteria				$\leq$ 0.15	$\leq$ 0.45	$\leq$ 10%	$\leq$ 0.6

TABLE 3-32

VALVE POSITION CONTROL LOOP B RESPONSE SUMMARY TC-6

Recirc Loop	Initial FCV Position	Initial Core Flow	Step Size	Delay Time (Sec)	Response Time (Sec)	Overshoot (%)	Setting Time (Sec)
B	22%	60%	-0.5%	0.3	0.3	8.3	1.0
B	22%	60%	+0.5%	0.2	0.2	18.3	1.7
B	22%	60%	- 1%	0.2	0.4	13.5	1.0
B	22%	60%	+ 1%	0.2	0.4	15	1.0
B	22%	60%	- 5%	0.2	0.5	5.5	0.7
B	22%	60%	+ 5%	0.2	0.4	4.5	0.3
B	39%	75%	-0.5%	0.4	0.4	18.2	1.4
B	39%	75%	+0.5%	0.4	0.4	40.9	1.3
B	39%	75%	- 1%	0.2	0.4	16	1.4
B	39%	75%	+ 1%	0.2	0.3	18	1.2
B	39%	75%	- 5%	0.2	0.4	4.5	0
B	39%	75%	+ 5%	0.2	0.4	5.3	0.3
B	67%	95%	-0.5%	0.2	0.3	13.8	3.6
B	67%	95%	+0.5%	0.2	0.2	29.3	4.2
B	67%	95%	- 1%	0.2	0.4	9.2	2.9
B	67%	95%	+ 1%	0.2	0.3	17.4	0.5
B	67%	95%	- 5%	0.2	0.4	2.4	0
B	67%	95%	+ 5%	0.2	0.4	5.0	0
Acceptance Criteria				$\leq$ 0.15	$\leq$ 0.45	$\leq$ 10%	$\leq$ 0.6

3.32 Test Number 29B - Recirculation Flow Loop Control

3.32.1 Purpose

The purpose of this test is to a) demonstrate the core flow system's control capability over the entire flow control range, including both core flow neutron flux and load following modes of operation, and b) determine that all electrical compensators and controllers are set for desired system performance and stability.

3.32.1.1 Flow Loop Criteria

Level 1 Criteria

The transient response of any recirculation system-related variable to any test input must not diverge.

Level 2 Criteria

A. The decay ratio of the flow loop response to any test inputs shall be less than or equal to 0.25.

B. The flow loops provide equal flows in the two loops during steady-state operation. Flow loop gains should be set to correct a flow imbalance in less than 25 seconds.

C. The delay time for flow demand step (less than or equal to 5%) shall be 0.4 seconds or less.

D. The response time for flow demand step (less than or equal to 5%) shall be 1.1 seconds or less.

E. The maximum allowable flow overshoot for step demand of less than or equal to 5% of rated shall be 6% of the demand step.

F. The flow demand step settling time shall be less than or equal to 6 sec.

3.32.1.2 Flux Loop Criteria

Level 1

The flux loop response to test inputs shall not diverge.

Level 2

A. Flux overshoot to a flux demand step shall not exceed 2% of rated for a step demand of less than or equal to 20% of rated.

B. The delay time for flux response to a flux demand step shall be less than or equal to 0.8 sec.

C. The response time for flux demand step shall be less than or equal to 2.5 sec.

D. The flux setting time shall be less than or equal to 15 sec. for a flux demand step less than or equal to 20% of rated.

3.32.1.3 SCRAM Avoidance and General Criteria

Level 1

Not applicable

Level 2

For any one of the above loops' test maneuvers, the trip avoidance margins must be at least the following:

A. For APRM 7.5%

B. For simulated heat flux 5.0%

3.32.1.4 Flux Estimator Test Criteria

Level 1

Not applicable

Level 2

A. Switching between estimated and sensed flux should not exceed 5 times/5 minutes at steady-state.

B. During flux step transient there should not be switching to sensed flux or if switching does occur, it should switch back to estimated flux within 20 seconds of the start of the transient.

3.32.1.5 Flow Control Valve Duty Test Criteria

Level 1

Not applicable

Level 2

The flow control valve duty cycle in any operating mode shall not exceed 0.2% Hz. Flow control valve duty cycle is defined as:

$$\frac{\text{Integrated Valve Movement in Percent (\% Hz)}}{\text{2x time span in seconds}}$$

3.32.2 Test Results

The response of all the recirculation flow control system related parameters to any step change in each control mode exhibited stable transient with a decay ratio less than 0.25. Table 3-33 summarizes the final setting of each controller.

The recirculation loop flow response to the flow controller demand step change showed that the Level 2 acceptance criteria were met with the following exceptions; 1) the flow delay time and response time criteria were exceeded and 2) the maximum flow overshoot exceeded the Level 2 criteria. Table 3-34 indicates the results. The function generators were re-verified after the adjustment on the valve actuators. Figures 3-12 and 3-13 indicate gain curves for the function generators exhibit a linear relationship between the function generator input and recirculation loop flow.

With the exception of the slight excessive flux overshoot for flux demand step all the Level 2 criteria for the flux loop were satisfied. The flux estimator was demonstrated to adequately adjust to minimize the valve cycle due to neutron flux noise. Table 3-35 indicates the flux loop test results.

Sufficient scram avoidance margins of neutron and heat flux were demonstrated for operation in flow and flux modes. The minimum scram margin of neutron and heat flux were 13.8% and 11.5% respectively. Table 3-36 summarizes the test results.

3.32.3 Discussion

The recirculation flow control system was initially tuned on the 75% load line and minimal changes on the controller were made since the system adjustments made in TC #3. The controller settings were verified again in TC #6 along the 100% load line. A well-behaved and stable response was demonstrated at these final settings.

The established test criteria is provided to support the Automatic Load Following (ALF) mode of operation. WNP-2 has elected not to use the ALF mode and therefore the acceptance criteria could be significantly relaxed. If and when WNP-2 elects to use the ALF mode, the system will need additional testing and tuning. Summarizing, the system is deemed adequate for the current power operation.

TABLE 3-33

RECIRC FLOW CONTROL SYSTEM FINAL SETTINGS
(Dial Setting in Turns)

POSITION CONTROLLER	COMPONENT	A	B
Proportional Gain	RV 4	1.0	1.75
Derivative Gain	RV 5	n/a	n/a
VELOCITY SERVO CONTROLLER			
Proportional Gain	RV 10	8.0	8.0
Integral Gain	RV 11	n/a	n/a
FLOW CONTROLLER			
Reset Gain (K_I)	RV 6	5.0	0
FLUX CONTROLLER			
Lead	RV 12	0.5	
Lag	RV 13	2.8	
Integral Gain (K_I)	RV 14	1.0	
Proportional Gain (K_P)	RV 15	2.0	
Gain (K_P)	RV 8	5.95	

TABLE 3-34

FLOW CONTROL LOOP RESPONSE SUMMARY

Recirc Loop	Initial Loop Flow	Initial Core Flow	Step Size	Delay Time (Sec)	Response Time (Sec)	Overshoot (%)	Setting Time (Sec)
A	53%	60%	-5%	0.2	1.0	15.6	0.6
A	53%	60%	+5%	0.6	1.1	0	0
A	71%	75%	-5%	0.0	1.2	0	0
A	71%	75%	+5%	0.4	1.5	0	0
A	95%	95%	-5%	0.6	2.0	0	0
A	95%	95%	+5%	0.2	1.3	2.8	0
B	54%	60%	-5%	0.5	1.4	10	0.4
B	54%	60%	+5%	0.6	4.4	0	0
B	70%	75%	-5%	0.2	1.4	0	0
B	70%	75%	+5%	0.4	1.4	15	0
B	94%	95%	-5%	0.8	1.0	0	0
B	94%	95%	+5%	0.8	1.4	10	0
A/A&B *	53%	60%	-5%	0.6	1.4	4.0	0
A/A&B *	53%	60%	+5%	0.7	6.0	4.0	0
B/A&B *	72%	75%	-5%	0.8	1.9	0	0
B/A&B *	72%	75%	+5%	0.5	1.5	0	0
Acceptance Criteria				$\leq$0.4	$\leq$1.1	$\leq$ 6%	$\leq$6.0

* Maximum of Combined Loops

TABLE 3-35

NEUTRON FLUX CONTROL LOOP RESPONSE SUMMARY

Initial Neutron Flow	Core Flow	Step Size	Delay Time (Sec)	Response Time (Sec)	Overshoot (%)	Setting Time (Sec)
75%	60%	-5%	0.9	0.4	2.8	0
75%	60%	+5%	0.7	0.5	3.2	3.6
85%	75%	-5%	0.6	0.2	2.0	0.7
85%	75%	+5%	0.6	0.3	5.0	0.6
98%	95%	-5%	2.6	2.8	1.2	0.4
98%	95%	+5%	0.8	0.2	2.8	2.0
Acceptance Criteria			≤ 0.8	≤ 2.5	≤ 2.0%	≤ 15

TABLE 3-36

SCRAM AVOIDANCE MARGIN VERIFICATION (100% L.L)

Control Mode	Initial Core Flow	Step Size	Scram Avoidance Margin APRM	Scram Avoidance Margin Heat Flux
FLO	60%	2%	38.4%	-
FLO	60%	5%	24.8%	14.8%
FLO	75%	2%	31%	15.6%
FLO	75%	5%	16%	14.4%
FLO	95%	2%	13.8%	11.5%
FLO	95%	5%	13.9%	-
FLX	60%	2%	39.6%	11.7%
FLX	60%	5%	39.8%	12.1%
FLX	75%	2%	30.4%	14.4%
FLX	75%	5%	27.8%	13.4%
FLX	95%	2%	13.8%	13.4%
FLX	95%	5%	20%	17.0%
Acceptance Criteria			$\geq$ 7.5%	$\geq$ 5%

RECIRCULATION FLOW CONTROL LINEARIZATION LOOP "A"

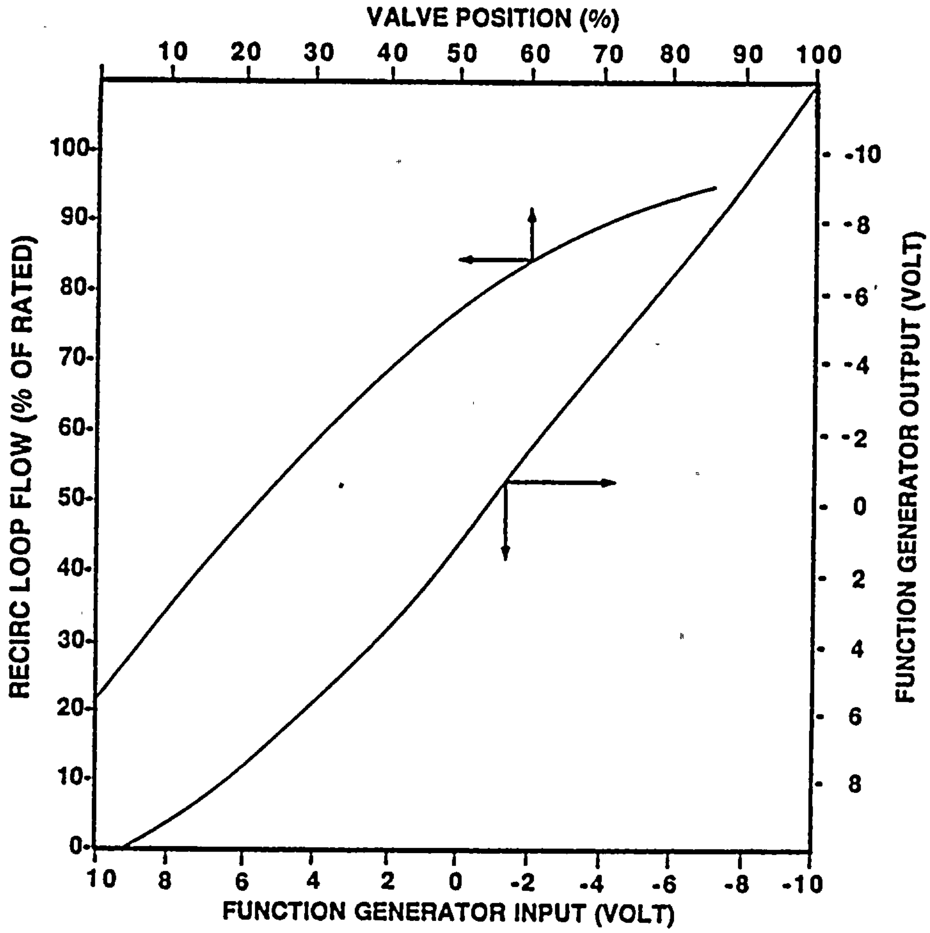

Figure 3-13

850207.21A

RECIRCULATION FLOW CONTROL LINEARIZATION LOOP "B"

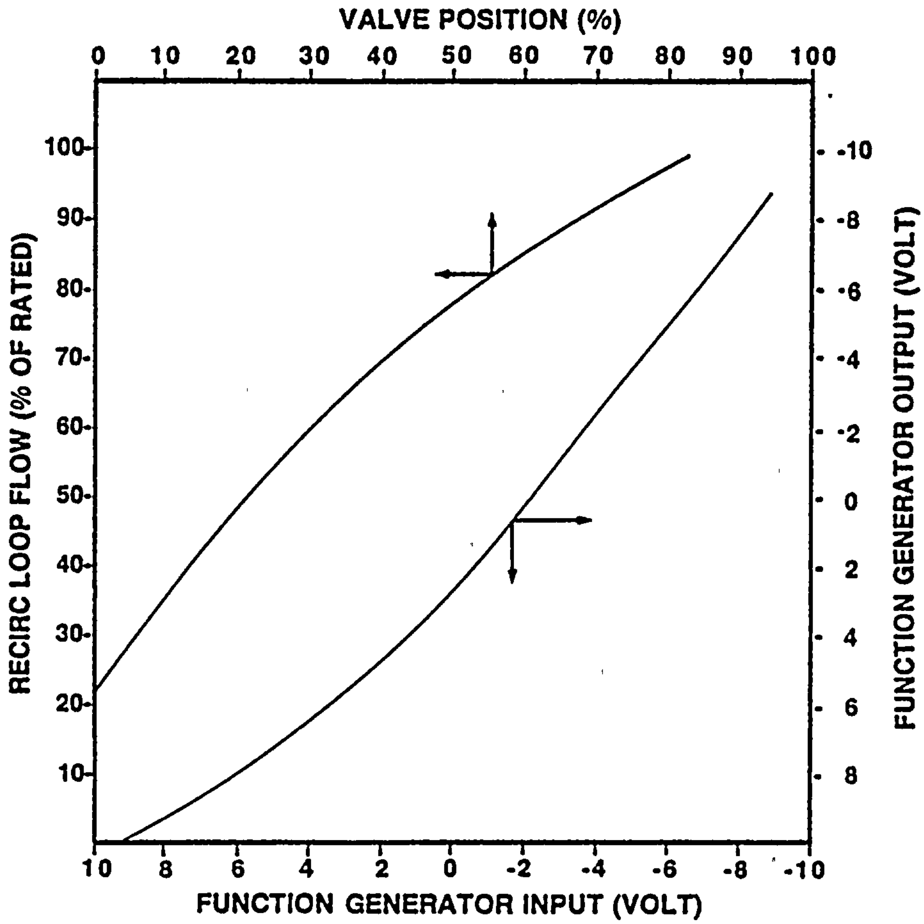

Figure 3-14

850207.22A

3.33 Test Number 30A - Recirculation System-One Pump Trip

3.33.1 Purpose

The major objectives of this test are as follows:

A. To obtain recirculation system performance data during the pump trip, one pump operation, and pump restart;

B. To verify the feedwater control system can satisfactorily control water level without resulting in a turbine trip and/or scram.

3.33.1.1 Level Criteria 1

The reactor shall not scram during the one pump trip recovery.

3.33.1.2 Level 2 Criteria

1. The reactor water level margin to avoid a high level turbine trip shall be greater than or equal to 3 inches during the one pump trip.

2. The simulated heat flux margin to avoid a scram shall be greater than or equal to 5 percent both during the one pump trip and also during the recovery.

3. The APRM margin to avoid a scram shall be greater than or equal to 7.5 percent during the one pump trip recovery.

4. The maximum one pump flow shall not cause excessive reactor internal vibration.

3.33.2 Test Results

No reactor scram occurred during either the pump trip or the recovery of the tripped pump. The feedwater control response was adequate to prevent the Level 8 high level turbine trip.

Reactor internal vibration data was recorded during the pump trip and subsequent single loop operation. A preliminary report provided by a GE Vibration Engineer showed the vibrations were within the level 2 criteria.

Summary of the results are listed in Table 3-37.

3.33.3 Discussion

During the recirculation pump trips and restarts conducted in TC-3 and TC-6, recirculation system performance data was obtained to evaluate heat flux, reactor power, water level, recirculation loop flow and combined jet pump flow response. During single loop operation individual jet pump flow data was recorded at various FCV positions to establish baseline flow patterns required for Technical Specification surveillance testing should extended single loop operation become necessary. FCV position versus loop flow for the inservice loop data was taken for the same purpose. The data collected has been used to prepare the Technical Specification surveillance procedure criteria.

TABLE 3-37

RECIRCULATION ONE PUMP TRIP RESULTS

Parameters	Test Condition TC-3	Test Condition TC-6	Acceptance Criteria
Reactor Water Level Margin	9.1"	6"	3"
Simulated Heat Flux Scram Margin During Trip	14.2%	12%	5%
Simulated Heat Flux Scram Margin During Recovery	23.8%	17%	5%
APRM Scram Margin During Recovery	47.6%	58%	7.5%

3.34 Test Number 30B - Recirculation System-RPT Two Pump Trip

3.34.1 Purpose

The purpose of the test is to record and verify acceptable performance of the recirculation two pump trip circuit system.

3.34.1.1 Level 1 Criteria

The two pump drive flow coastdown transient during the first 6 seconds must be bounded by the limiting curves.

3.34.1.2 Level 2 Criteria

Not applicable

3.34.2 Test Results

Figures 3-14 and 3-15 display the results of the recirculation pumps trip flow coastdown transient and the comparision to the coastdown criteria. Both the A and B loop drive flow coastdowns exceeded the Level 1 ECCS criteria 5 second time constant curve.

3.34.3 Discussion

Table 3-38 contains tabulated values for the criteria curves provided by GE Plant Transient Performance Engineering for the above figures. The test exception was analyzed and accepted by General Electric Transient Performance Engineering. The basis of this conclusion is an ECCS pump coastdown sensitivity study which showed inertial time constants as low as 3 seconds were acceptable. The test data for both loops fell between 6 seconds and 3.5 seconds (3 seconds plus .5 second added conservatism) bounding curves and was therefore deemed to be acceptable. The 3 second inertia time constant resulted in a peak clad temperature increase of less than 10°F. The 10°F peak clad temperature increase does not impact the MAPLHGR limit.

RPT COASTDOWN DATA FOR LOOP "A"

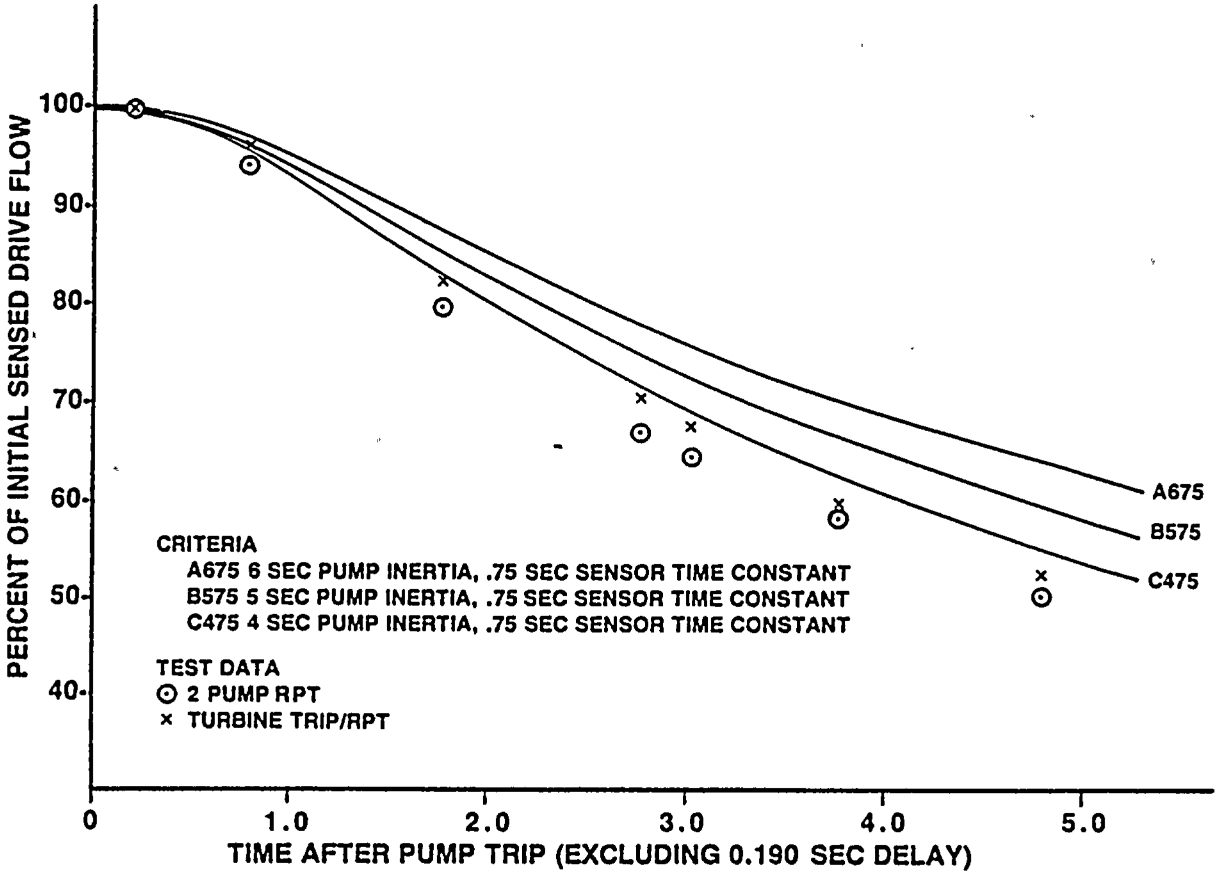

Figure 3-14

RPT COASTDOWN DATA FOR LOOP "B"

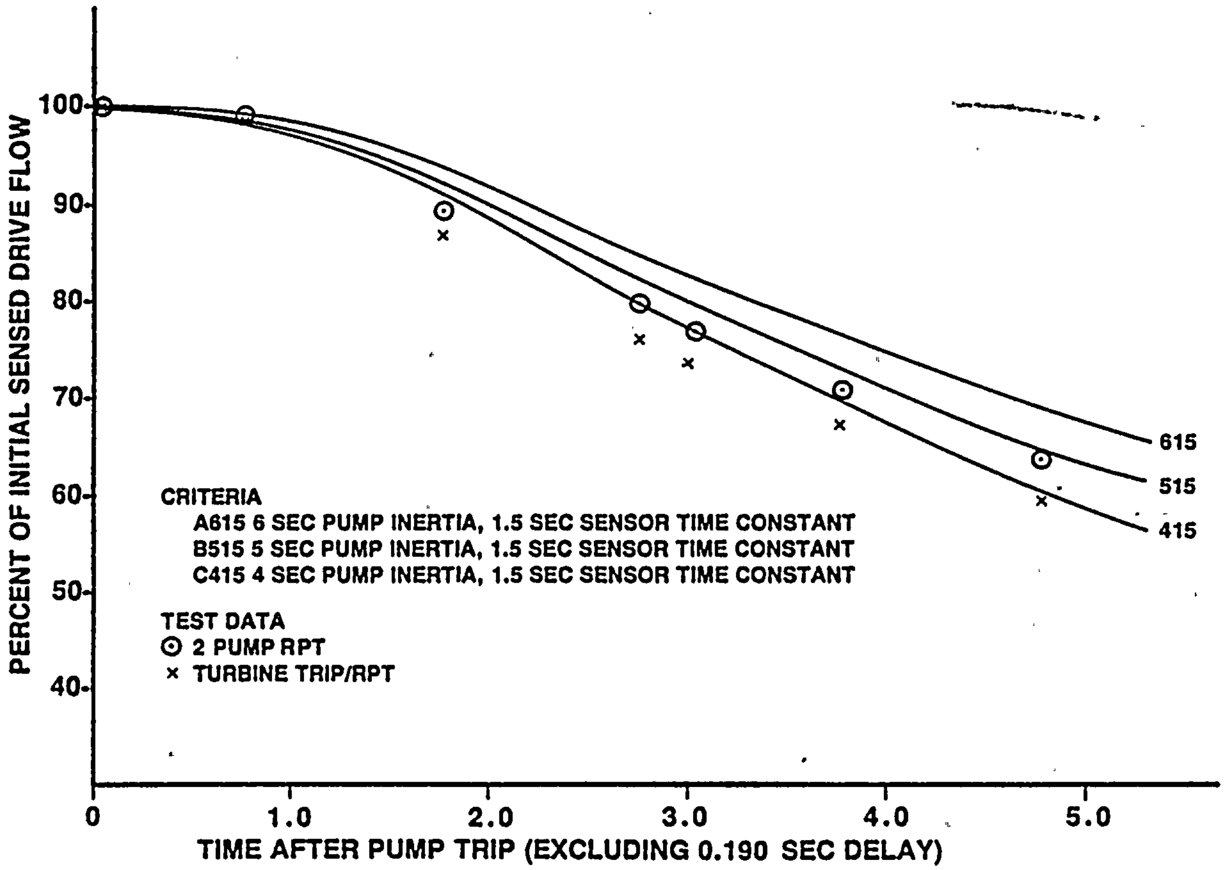

Figure 3-15

TABLE 3-38

HANFORD RPT COASTDOWN REQUIREMENTS

TABULATED VALUES FOR CRITERIA CURVES SHOWN IN FIGURES 3-14 and 3-15

TIME	A615	B515	C415	A675	B575	C475
0.32	100.10	99.90	99.70	99.80	99.60	99.40
0.40	99.90	99.90	99.90	99.45	99.30	99.15
0.56	99.90	99.70	99.50	98.55	98.20	97.85
0.67	99.60	99.50	99.40	97.80	97.25	96.70
0.73	99.40	99.20	99.00	97.30	96.65	96.00
0.81	99.20	98.90	98.60	96.65	95.75	94.85
1.06	98.00	97.80	97.60	94.25	93.15	92.05
1.31	96.70	95.90	95.10	91.90	90.35	88.80
1.56	94.90	94.10	93.30	89.35	87.50	85.65
1.81	92.90	91.60	90.30	86.90	84.70	82.50
2.06	90.90	89.30	87.80	84.50	81.95	79.40
2.31	88.80	86.80	84.80	82.20	79.30	76.40
2.56	86.80	84.40	82.00	79.90	76.90	73.90
2.81	84.40	81.60	78.80	77.80	74.40	71.00
3.31	80.20	76.90	73.60	73.90	70.15	66.40
3.81	76.10	72.60	69.10	70.40	66.45	62.50
4.31	72.70	68.60	64.50	67.30	62.90	58.50
4.81	69.20	64.90	60.60	64.25	59.65	55.05
5.31	66.00	61.40	56.80	61.35	56.75	52.15

NOTE: Curve A615, for example, is for a 6 second pump inertia with a 1.5 second sensor time constant. See Figures 3-14 and 3-15 for description of curves.

3.35 Test Number 30C - Recirculation System Performance

3.35.1 Purpose

The purpose of this test is to obtain recirculation system performance data under different operating conditions to verify design parameters.

3.35.1.1 Level 1 Criteria

Not applicable

3.35.1.2 Level 2 Criteria

1. The core flow shortfall shall not exceed 5% at rated power.

2. The measured core delta P shall not be greater than 0.6 PSI above prediction.

3. The calculated jet pump M ratio shall not be less than 0.2 points below prediction.

4. The drive flow shortfall shall not exceed 5% at rated power.

5. The measured recirculation pump efficiency shall not be greater than 8 percent below the vendor tested efficiency.

6. The maximum nozzle and riser plugging criteria of 12% and 10% respectively, shall not be exceeded.

3.35.2 Test Results

Table 3-39 summarized the recirculation system performance over the operating conditions. Figure 3-16 indicates the relationship between total core flow and total loop flow. The process computer data bank was updated with the established relationship to provide substitute core flow for the OD-3 and P1 programs. Figure 3-17 indicates the core delta P as a function of total core flow. Both design and actual curves agree within the tolerance of the instrument accuracy.

3.35.3 Discussion

Recirculation system performance data was also obtained during single loop operation. The data was evaluated and used to establish operation boundary for single loop operation.

During natural circulation in TC-4, core flow was found to be about 6% less than predicted. No criteria was affected. The results also agreed with LaSalle 1 test data.

The maximum core flow achieved with both recirculation flow control valves fully open was 106% of rated. The safety design basis for the MCPR calculation at 115% maximum core flow was not exceeded. The recirculation flow control limiter was set to limit the maximum core flow to 102.5% of rated.

TOTAL CORE FLOW VS TOTAL LOOP FLOW

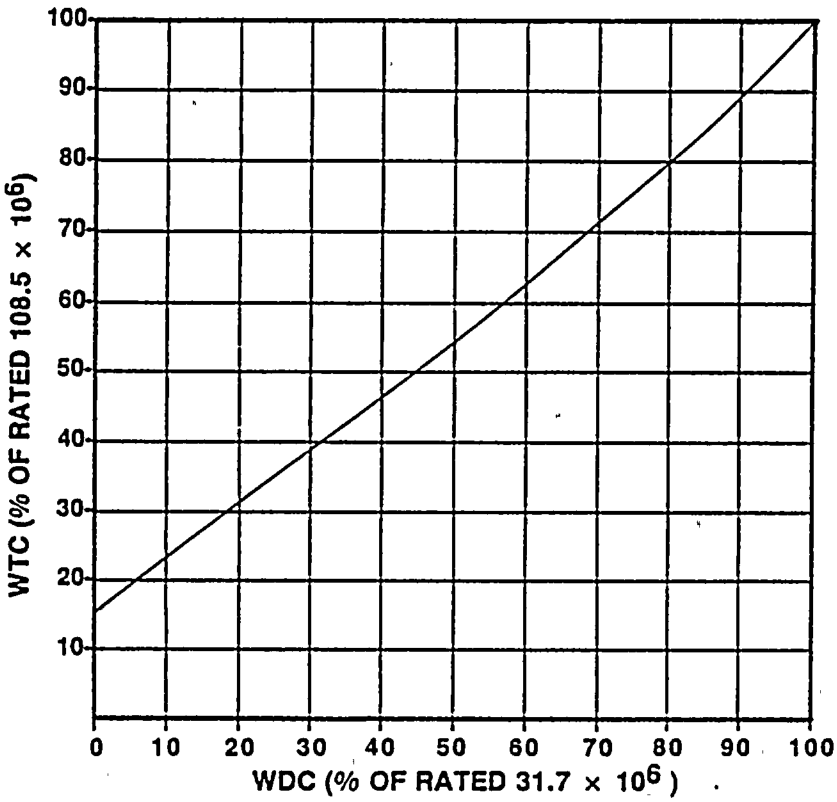

Figure 3-16

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CORE DP VS TOTAL CORE FLOW

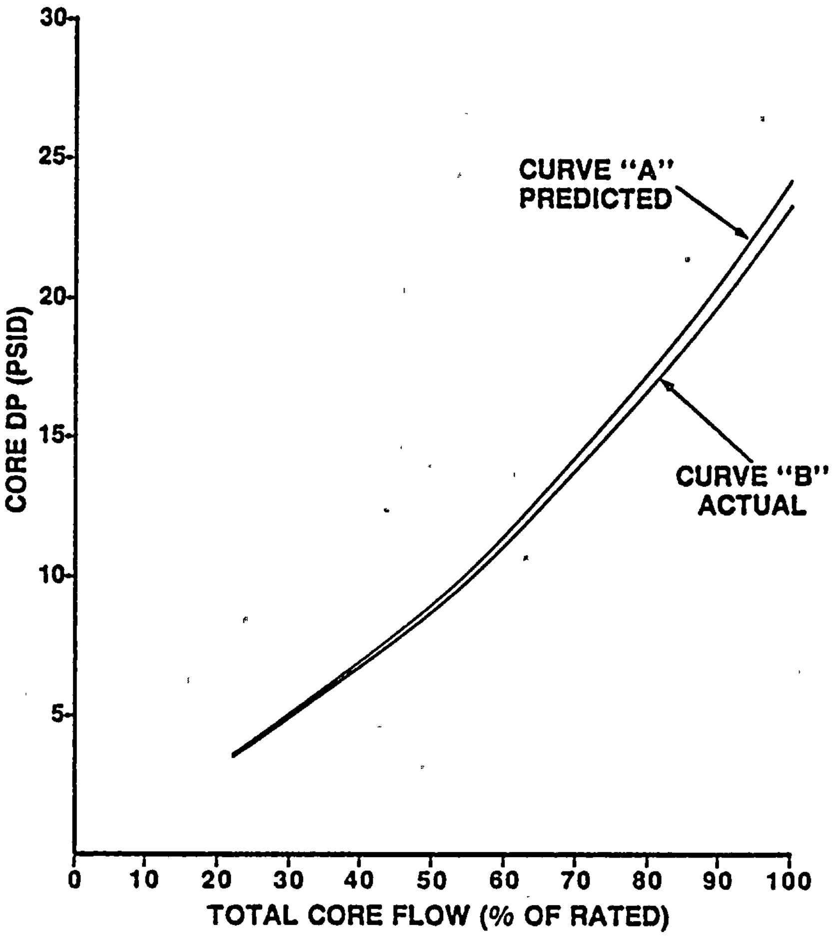

Figure 3-17

TABLE 3-39

RECIRCULATION SYSTEM PERFORMANCE

Parameter	TC-2	TC-3	TC-4	TC-6	Acceptance Criteria
Core flow shortfall (%)	N/A	N/A	N/A	.37	Less than 5% at rated power
Core dP (psid)					
Actual / Predicted	N/A	N/A	N/A	23.29 / 24.74	Less than 0.6 psid above prediction
Calculated M-ratio					
Loop A: Actual / Predicted	N/A	N/A	N/A	2.416 / 2.330	Shall not be less than 0.200 point below prediction
Loop B: Actual / Predicted	N/A	N/A	N/A	2.490 / 2.330	
Drive flow shortfall (%)	N/A	N/A	N/A	3.07	Less than 5% at rated power
Pump efficiency (%)					
Pump A: Actual / Predicted	N/A	80 / 84	N/A	80.81 / 83	Shall not be less than 8% below prediction
Pump B: Actual / Predicted	N/A	83 / 86	N/A	83.76 / 83	
Nozzle/Riser Plugging (%)					
Maximum nozzle plugging	N/A	6.2	N/A	8.5	12%
Maximum Riser plugging	N/A	5.8	N/A	8.3	10%

3.36 Test Number 30D - Recirculation Runback

3.36.1 Purpose

The purpose of this test is to verify the adequacy of the recirculation flow control valve runback to avoid a reactor scram upon the loss of one feedwater pump.

3.36.1.1 Level 1 Criteria

Not applicable

3.36.1.2 Level 2 Criteria

The recircultion flow control valves shall runback upon a trip of the runback circuit.

3.36.2 Test Results

The runback reduced the recirculation flow control valve position properly, core flow was reduced from 95% to 59% on the 75% load line. The extrapolated reactor power from the 100% load line determined that the existing flow control valve runback position limiter setpoint would result in a reactor power equal to 69% of rated.

3.36.3 Discussion

A single feedwater pump trip test performed at 98.7% power during TC-6 on the 100% load line, further demonstrated the adequacy of the recirculation flow control valve runback to prevent a reactor low water level scram.

3.37 Test Number 30E - Recirculation System-Non Cavitation Verification

3.37.1 Purpose

The purpose of this test is to verify that no recirculation system cavitation will occur in the operating region of the power flow map.

3.37.1.1 Level 1 Criteria

Not applicable

3.37.1.2 Level 2 Criteria

Runback logic shall have settings adequate to prevent operation in areas of potential cavitation.

3.37.2 Test Results

The recirculation pump high to low speed transfer logics were verified on low feedwater flow and low differential temperature between reactor dome and recirculation suction during TC-2 and TC-3 respectively. The low feedwater flow setpoint of 27.5% rated (3.93 M lb/hr) and low differential temperature between reactor dome to recirculation suction setpoint at 9.9°F were verified to be adequate to prevent operation in areas of potential cavitation.

3.37.3 Discussion

Both recirculation pumps did not transfer simultaneously in TC-2 due to the relay logic sequencing (relay race condition). The logic was corrected so that any of the low feedwater flow, steam dome to recirculation suction low differential temperature or low level (L-3) signals will cause both pumps to transfer to the LFMG sets at the same time to provide the additional cavitation protection.

3.38 Test Number 31 - Loss of Turbine Generator and Offsite Power

3.38.1 Purpose

The major objectives of this test are as follows:

A. To demonstrate the reactor system transient performance during the loss of the main generator and all off-site power; and

B. To demonstrate acceptable performance of station electrical equipment.

3.38.1.1 Level 1 Criteria

Reactor protection system actions shall prevent violation of fuel thermal limits.

All safety systems, such as the Reactor Protection System, the Diesel Generators, and HPCS must function properly without manual assistance, and HPCS and/or RCIC system action, if necessary, shall keep the reactor water level above the initiation level of the Low Pressure Core Spray, LPCI and ADS systems, and MSIV closure. Diesel generator shall start automatically.

3.38.1.2 Level 2 Criteria

A proper instrument display to the reactor operator shall be demonstrated, including power monitors, pressure water level, control rod position, suppression pool temperature and reactor cooling system status. Displays shall not be dependent on specially installed instrumentation.

If safety/relief valves open, the temperature measured by thermocouples on the discharge side of the safety/relief valves must return to within 10°F of the temperature recorded before the valve was opened.

3.38.2 Test Results

Reactor transient behavior and station electrical supply system performance during the loss of A-C test is summarized in Table 3-40, Loss of A-C Test Results. A chronology of significant events during the transient is outlined in Table 3-41.

The suppression pool temperature increased by 5°F from 75°F to 80°F. The maximum drywell temperature reached was 129°F.

Proper instrument display to the reactor operator for power, pressure, water level, control rod position, suppression pool temperature and reactor cooling system status was maintained with no dependence on specially installed instrumentation.

3.38.3 Discussion

The loss of turbine generator and off-site power was initiated by opening the generator output breakers while preventing transfer of the station electrical loads to the Startup and Backup Startup Transformers. The reactor was scrammed on Turbine Control Valve Fast Closure signal. MSIV isolation occurred 8 seconds into the transient due to under-frequency on the RPS buses. No ECCS initiation nor SRV actuation during the transient. All safety systems and electrical supply system performed properly.

TABLE 3-40

LOSS OF A-C TEST RESULTS

Parameter	Value	Criteria or Setpoint
Diesel Generator Loading Time		
Div. I D/G	8.252 seconds	10.0 seconds
Div. II D/G	7.202 seconds	10.0 seconds
HPCS D/G	11.102 seconds	13.0 seconds
Peak Reactor Pressure	950 psig	1076 psig
Minimum Reactor Level	-10"	----

TABLE 3-41

CHRONOLOGY OF SIGNIFICANT EVENTS DURING THE LOSS OF A-C TEST

Time				Significant Event
Hr.	min.	sec.	msec.	
01	35	36	980	Generator Output Breakers Open
01	35	37	204	TCV Tast Closure RPS B1 Tripped
01	35	37	304	Reactor Scram, TCV Fast Closure
01	35	40	004	Control Rods Fully Inserted
01	35	43	000	RCIC Manually Started
01	35	43	754	RPS MG Set A Underfrequency
01	35	45	304	RPS MG Set B Underfrequency
01	35	45	680	Div. II D/G Loaded to SM-8 Bus
01	35	46	880	Div. I D/G Loaded to SM-7 Bus
01	35	48	654	MSIV's Fully Closed
01	35	49	880	HPCS D/G loaded to SM-4
01	35	57	154	Minimum Water Level (-10")
01	45			Reset Scram

3.39 Test Number 33 - Piping Vibration

3.39.1 Purpose

The purpose of this test is to verify that the design stress levels due to piping vibration are not exceeded and satisfy the inspection requirements for condensate and feedwater systems per Regulation Guide 1.68.1.

3.39.1.1 Level 1 Criteria

The measured vibration amplitude (peak-to-peak) of the systems monitored shall not exceed the maximum allowable displacements.

3.39.1.2 Level 2 Criteria

The measured amplitude (peak-to-peak) of vibration shall not exceed the expected values.

3.39.1.3 Visual Inspection Acceptance Criteria

The vibration levels experienced will be evaluated as acceptable if they are too small to be detected by the naked eye with consideration given to the following facts:

A. Proximity to sensitive equipment (pumps, valves, motor control centers, control panels, etc.).

B. Branch connection behavior

C. Performance of nearby component supports

If an acceptable assessment of the observed deflections cannot be performed and corrective measures are not available, the inspector will then obtain the magnitude and frequency of the vibration using a portable vibration instrument.

3.39.2 Test Results

The steady state vibration amplitudes on the main steam and recirculation piping over the entire flow range are summarized in Table 3-42 and Table 3-43.

The transient vibration amplitudes on the main steam line piping during the relief valve actuation, 75% turbine trip, MSIV full isolation and 100% generator load rejection are summarized in Table 3-44.

The transient vibration amplitudes on the recirculation piping during the pump trip and pump restart are summarized in Table 4-45.

At no time did the measured vibration amplitude for the systems monitored exceed the established maximum allowable (Level 1) or expected displacement (Level 2) criteria.

3.39.3 Discussion

The drywell piping vibration data was obtained from the dual purpose lanyard potentiometers used in conjunction with the system expansion test. All vibration data was recorded on PCM tape using the Transient Data Acquisition System, which is capable of recording data at maximum sample rate of 500 samples per second, and providing a .005 inch resolution and a frequency response higher than 20 Hertz. The PCM tape was then played back for data reduction and analysis. All vibration data is given in respect to the local coordinate system. The acceptance criteria is also presented in the local coordinate system to provide direct comparison capability.

Other drywell piping systems that were monitored for steady-state and transient vibration are the RWCU, RCIC Steam Supply, SRV (2 lines) tail pipes, RHR SDC Supply and return (Loop A) and Reactor Feedwater.

The transients conducted produced no Level 2 violations. The following is a list of all the transients where data was collected for those systems influenced;

1. Generator Load Reject at 25% Power (RRC & MS piping)

2. Turbine Trip at 75% power (RRC & MS piping)

3. Main Steam SRV Testing (MS and SRV piping)

4. Load Reject at 100% Power (MS, RRC, FW and SRV piping)

5. Reactor Feedpump Trip at 100% Power (FW piping)

6. RHR Pump Start and Trip During SDC Initiation (RHR piping)

7. MSIV Full Closure (RRC, MS and SRV piping)

8. RRC Single Pump Standard Trip (RRC piping)

9. RRC Simulated RPT (2-Pump Transfer to 15 Hz) (RRC piping)

Visual inspections were performed on hot piping systems outside the drywell during steady state and transient conditions. The systems inspected for steady state vibration were RHR, Feedwater and Condensate, RCIC Steam Supply and Exhaust, Main Steam and MSLC & RWCU. During the load reject test, personnel were stationed in the turbine building to assess the MS, FW and Condensate systems vibration. The RHR piping was also visually monitored during pump starts and trips. No excessive piping vibration was noted during these tests.

TABLE 3-42

STEADY STATE DRYWELL PIPING VIBRATION DATA

Main Steam Lines

Sensor Identification	Steam Flow % 25%	50%	75%	100%	Criteria Level 1/Level 2 (Inch)
1MSA X	2.1	2.1	2.	2.8	.278/.138
Y	2.8	1	2.	2.8	.038/.020
Z	2.1	1.3	2.	3.3	.266/.132
2MSA X	2.6	1.6	2.	2.8	.170/.084
Y	2.8	1.7	3.	4.2	.160/.080
Z	3.2	1	2.	3.3	.138/.070
1MSB X	3.0	2.8	3.	1.1	.220/.110
Y	3.0	1.2	4.	1	.060/.030
Z	3.6	1.4	4.	3.8	.202/.100
2MSB X	3.7	2.8	3.	8.3	.116/.058
Y	3.3	1.9	3.	4.5	.176/.088
Z	3.4	2.0	5.	4.4	.118/.060
1MSC X	3.3	1.9	4.	3.9	.220/.110
Y	3.8	1.4	4.	4.9	.060/.030
Z	3.6	1.9	4.	5.8	.202/.110
2MSC X	3.7	1.6	4.	3.9	.118/.060
Y	3.8	2.1	3.	4.0	.176/.088
Z	3.5	1.4	3.	3.8	.116/.058
1MSD X	3.1	1.4	4.	3.0	.212/.106
Y	3.5	2.3	3.	4.2	.038/.020
Z	2.8	2.1	3.	3.2	.288/.144
2MSD X	3.3	2.1	3.	4.0	.106/.054
Y	3.0	2.0	5.	8.8	.156/.078
Z	3.5	2.0	3.	5.9	.138/.068

NOTE: The values given are in mils (.001 inch = one mil)

TABLE 3-43

STEADY STATE DRYWELL PIPING VIBRATION DATA

Recirculation Loops

Sensor Identification S.S.		GE's	Recirculation Flow 25%	50%	75%	100%	Criteria Level 1/Level 2 (Inch)
1. 1RA	X	RA2	1.2	1	2.	2.4	.102/.050
	Y		.9	2.1	2.	2.8	.164/.082
	Z		1.3	2.1	8.	5.4	.064/.032
2. 2RA	X	RA3	1.4	2.1	2.	2.6	.232/.116
	Y		1.6	1.4	2.	2.6	.064/.032
	Z		1.1	2.1	5.	3.7	.160/.080
3. 3RA	X	RA4	2.1	1.9	4.	3.7	.118/.060
	Y		1.7	1.4	2.	3.1	.062/.030
	Z		1.4	1.6	3.	2.6	.080/.040
4. 4RA	X	RA1	2.3	2.1	4.	3.7	.104/.052
	Y		1.6	1.4	3.	3.5	.026/.014
	Z		2.3	1.4	3.	3.3	.052/.026
5. 1RB	X	RB2	1.6	1.4	2.	2.3	.100/.050
	Y		5.7	0	1.	Inop	.156/.078
	Z		2.1	4.2	2.	3.3	.064/.032
6. 2RB	X	RB3	1.8	4.0	2.	3.0	.132/.116
	Y		1.4	1.8	3.	3.5	.064/.032
	Z		2.3	1.4	2.	3.0	.160/.080
7. 3RB	X	RB4	1.8	2.5	2.	3.0	.118/.060
	Y		.9	1.6	2.	2.3	.062/.030
	Z		1.9	2.3	4.	5.4	.080/.040
8. 4RB	X	RB1	1.6	1	4.	6.8	.064/.032
	Y		1.6	4.0	3.	4.9	.028/.014
	Z		3.0	4.0	1.	3.1	.104/.052

NOTE: The values given are in mils (.001 inch = 1 mil)

TABLE 3-44

TRANSIENT DRYWELL PIPING VIBRATION DATA

Main Steam Line

Sensor Identification		Rated Pressure Relief Valve Test	75% Turbine Trip	MSIV Full Isolation	100% Generator Load Rejection
1MSA	X	40	1.	45	24
	Y	4	3.	2	0
	Z	14	20.	66	32
2MSA	X	16	19.	30	22
	Y	22	47.	66	52
	Z	10	21.	15	28
1MSB	X	31	3.	2	0
	Y	13	10.	12	8
	Z	39	29.	50	32
2MSB	X	30	55.	35	36
	Y	54	84.	70	68
	Z	34	46.	40	40
1MSC	X	28	59.	8	72
	Y	18	8.	2	12
	Z	43	14.	2	20
2MSC	X	30	35.	2	60
	Y	41	79.	2	90
	Z	28	56.	2	51
1MSD	X	13	5.	30	16
	Y	9	14.	15	18
	Z	33	32.	35	20
2MSD	X	52	42.	55	64
	Y	76	73.	110	90
	Z	30	40.	45	35

NOTE: Values given are in mils (.001 inch = 1 mil)

TABLE 3-45

TRANSIENT DRYWELL PIPING VIBRATION DATA

Recirculation Loops

Sensor Identification		RPT Two Pump Trip	75% One Pump Trip		100% One Pump Trip	
			Pump Trip	Pump Restart	Pump Trip	Pump Restart
1RA	X	2.			2.	3.
	Y	2.			2.	3.
	Z	4.			4.	5.
2RA	X	3.			3.	3.
	Y	2.			2.	3.
	Z	3.			3.	4.
3RA	X	3.			3.	3.
	Y	2.			2.	2.
	Z	2.			2.	3.
4RA	X	2.			3.	4.
	Y	3.			3.	3.
	Z	3.			3.	4.
1RB	X	2.	2.	4.		
	Y	1.	1.	1.		
	Z	3.	3.	6.		
2RB	X	2.	2.	9.		
	Y	2.	3.	3.		
	Z	2.	2.	29.		
3RB	X	2.	2.	7.		
	Y	2.	2.	3.		
	Z	3.	3.	3.		
4RB	X	4.	4.	6.		
	Y	4.	4.	4.		
	Z	3.	4.	10.		

NOTE: Values given are in mils (.001 inch = one mil)

3.40 Test Number 34 - Reactor Internals Vibration

3.40.1 Purpose

The major objectives of this test are as follows:

A. To provide information needed to confirm the similarity between the reactor internals design and the prototype with respect to flow induced vibration;

B. To fulfill the NRC Regulatory Guide 1.20 for a vibration measurement program for nonprototype, Category IV reactor internals.

3.40.1.1 Level 1 Criteria

The peak stress intensity may exceed 10,000 psi (single amplitude) when the component is deformed in a manner corresponding to one of its normal or natural modes but the fatigue usage factor must not exceed 1.0.

3.40.1.2 Level 2 Criteria

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

3.40.2 Test Results

The maximum vibration frequencies and corresponding amplitude for the jet pump riser brace and shroud were measured on the 60%, 75% and 100% load line, during extended recirculation flow, and single and dual recirculation pump trip. The preliminary measurement analysis indicated that all vibrations were well within both Level 1 and Level 2 criteria.

3.40.3 Discussion

The final vibration analysis to determine the actual mode of vibration will be performed by the General Electric at their San Jose office. The results will be presented as a final report following the completion of the detailed data analysis. The current scheduled completion date is June 1, 1985. Following Supply System review, the report will be forwarded to the NRC.

3.41 Test Number 35 - Recirculation System Flow Calibration

3.41.1 Purpose

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

3.41.1.1 Level 1 Criteria

Not applicable

3.41.1.2 Level 2 Criteria

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

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

The flow control system shall be adjusted to limit maximum core flow to 102.5% of rated by limiting the flow control valve opening position.

3.41.2 Test Results

Data in Table 3-46 documents the calibration efforts on the core flow instrumentation and represents the final system configuration.

At 100% power, it requires 41936/41492 GPM recirculation drive flow A/B to achieve 100% core flow. The transmitter inputs into the APRM/RBM flow-bias instrumentation were recalibrated to reflect these 100% power and flow conditions.

The flow control valve opening position limit was adjusted such that 102.5% of rated core flow is the maximum core flow.

3.41.3 Discussion

Prior to performing the calibration data collection, each flow instrumentation was zero checked against the instrument data sheets. At 100% power, 100% core flow conditions the total core flow and individual loop flow recorder/indications were adjusted to comply with calculated values.

The jet pump flow distribution is plotted in Figure 3-18, the center jet pumps are exhibiting higher flows than the average jet pump flows. This is consistent with other BWR-5 five nozzle jet pump plants.

TABLE 3-46

RECIRCULATION FLOW INSTRUMENTATION ADJUSTMENTS

Date	11/4/84	11/8/84	
Time	2125	2350	
Power (MWt, %)	3259.5 (98.1)	3323 (100%)	
Core Flow (M lb/hr, %)	102 (94%)	106.3 (98%)	Average Gain
Loop Flows:			
Loop A: Calc/Ind	49.68/51.08	53.721/53.67	
Gain	0.972	1.001	0.987
Loop B: Calc/Ind	52.16/53.32	53.721/53.95	
Gain	0.978	0.973	0.976
Drive Flows:			
Loop A: Calc/Ind	38454/37557	40655/40099	
Gain	1.024	1.014	1.019
Loop B: Calc/Ind	39513/39208	40419/40329	
Gain	1.008	1.002	1.005
Core Flow:			
Calc/Ind	101.84/106	106.2/105.5	
Gain	0.961	1.007	0.984
Double Tap Flows:			
JP5 = Calc/Ind	5.269/5.3811	5.637/5.685	
Gain	0.979	0.992	0.986
JP10 = Calc/Ind	4.884/4.9218	5.277/5.305	
Gain	0.992	0.995	0.994
JP15 = Calc/Ind	5.307/5.3662	5.353/5.517	
Gain	0.989	0.970	0.979
JP20 = Calc/Ind	5.232/5.2785	5.22/5.363	
Gain	0.991	0.973	0.982

NOTES: Drive Flows are given in GPM
Calc = Flow from JRPMP01 Edit
Ind = Flow from indication
Gain = Calc/Ind

JET PUMP FLOW DISTRIBUTION

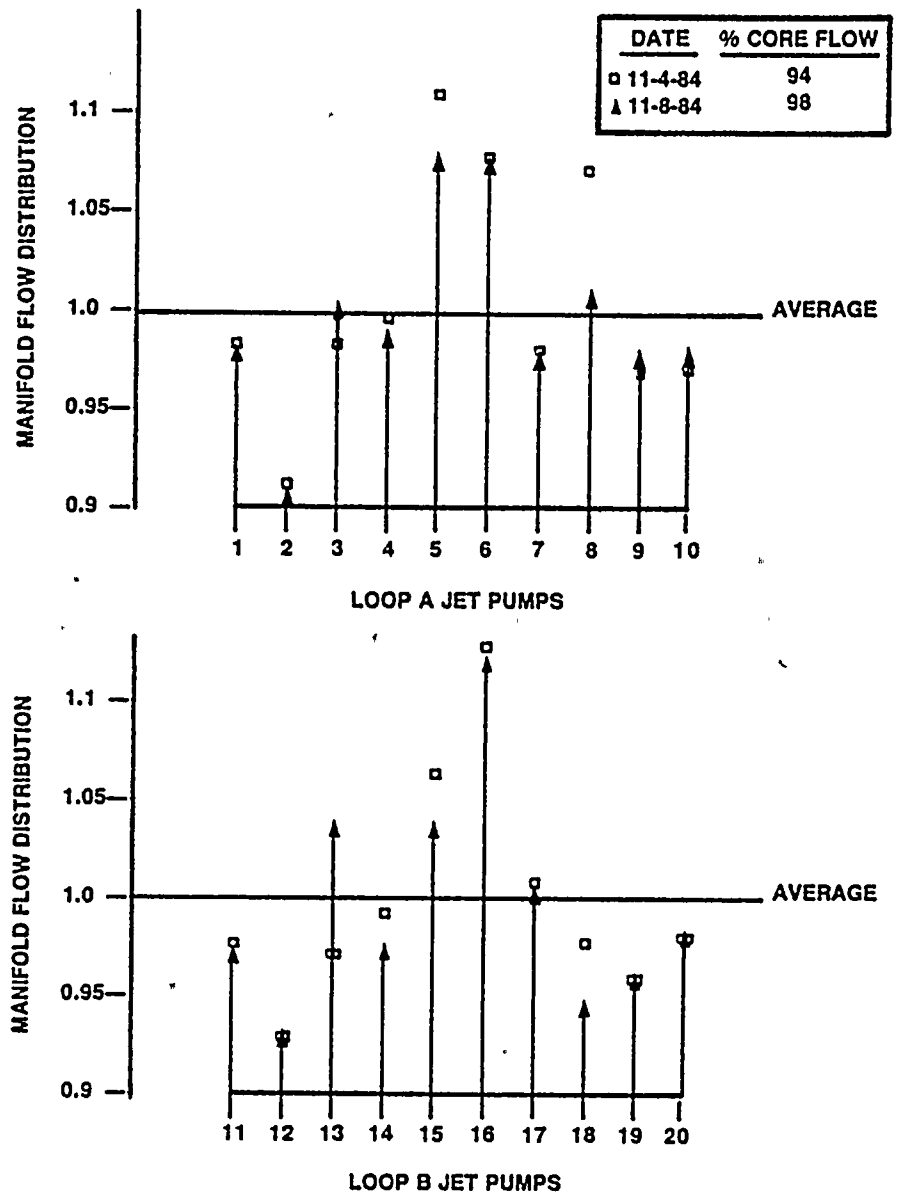

Figure 3-18

3.42 Test Number 70 - Reactor Water Cleanup System

3.42.1 Purpose

The major objective of this test is to demonstrate specific aspects of the mechanical operability of the Reactor Water Cleanup System.

3.42.1.1 Level 1 Criteria

Not applicable

3.42.1.2 Level 2 Criteria

1. The temperature at the tube side outlet of the non-regenerative heat exchangers shall not exceed 130°F in the blowdown mode and shall not exceed 120°F in the normal mode.

2. The pump available NPSH will be 13 feet or greater during the hot shutdown with loss of RPV recirculation pumps mode defined in the process diagram.

3. The cooling water supplied to the non-regenerative heat exchangers shall be less than 6% above the flow corresponding to the heat exchanger capacity and the exisiting temperature differential across the heat exchangers. The outlet temperature shall not exceed 180°F.

4. Recalibrate bottom head flow indicator, RWCU-FI-610 against RWCU flow indicator, RWCU-FI-609, if the deviation is greater than 25 gpm.

5. Pump vibration shall be less than or equal to 2 mils peak-to-peak (in any direction) as measured on the bearing housing and 2 mils peak-to-peak shaft vibration as measured on the coupling end.

3.42.2 Test Results

During normal operation with process diagram flows established, the temperature of the tube side outlet from the non-regenerative heat exchangers was 100°F while during blowdown operation it was 91°F.

The calculated pump available net positive suction head (NPSH) was 523.9 feet in the hot standby mode.

The temperature of the closed cooling water supplied to the non-regenerative heat exchangers was 62°F with the flow at 365 gpm (370 gpm predicted). The non-regenerative heat exchanger outlet temperature was 140°F.

The maximum deviation between the bottom drain flow and RWCU system flow indicator was less than 8 gpm.

The peak-to-peak vibration on RWCU pump A and B were 1.3 and 0.5 mils respectively; the peak-to-peak shaft vibrations were 0.75 mils on both pumps.

3.42.3 Discussion

For the RWCU system tests, appropriate flows and temperatures were established based upon the process flow diagram. On the BWR/5 product line the bottom head drain line flow orifice and associated flow instrumentation have been replaced with a system which relates differential pressure across the bottom head (lower-plenum-to-drain-pressure differential) to the drain line flow. An approximate flow-differential pressure relationship was determined by drawing all the Reactor Water Cleanup Unit (RWCU) system flow through the bottom head drain and comparing the system flow indication with the differential pressure measurements. This relationship is illustrated in Figure 3-19. The differential pressure indication was used to confirm adequate bottom head drain flow for both test and normal operation purposes.

RWCU BOTTOM HEAD FLOW INDICATION

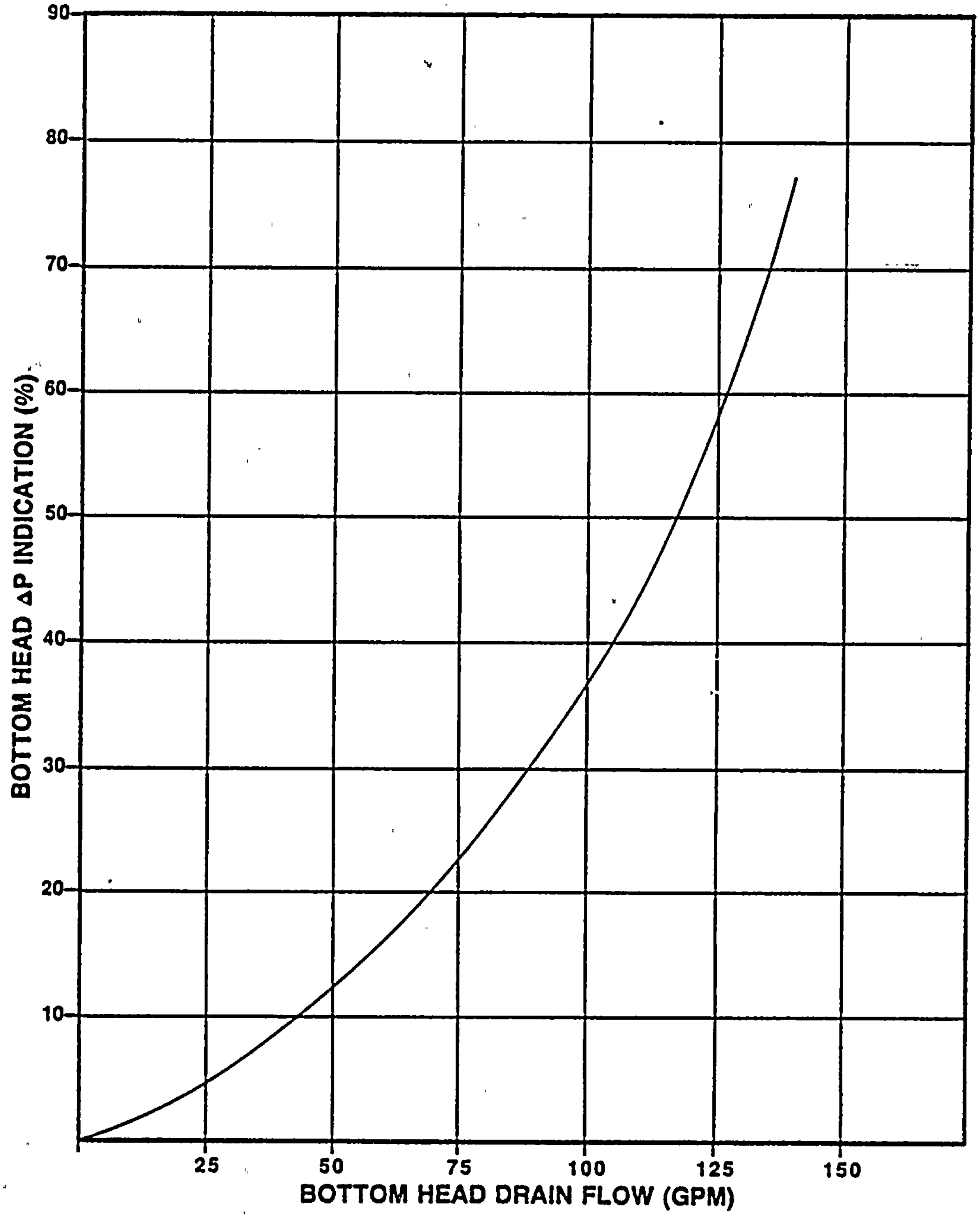

Figure 3-19

3.43 Test Number 71 - Residual Heat Removal System

3.43.1 Purpose

The major objectives of this test are as follows:

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 servicing can be performed.

3.43.1.1 Level 1 Criteria

Not applicable

3.43.1.2 Level 2 Criteria

The RHR system shall be capable of operating in suppression pool cooling and shutdown cooling modes (with each heat exchanger) at the flow rate and temperature differential indicated on the process diagram.

3.43.2 Test Results

A summary of the RHR system data obtained when in suppression pool cooling and shutdown cooling modes of operation is contained in Table 3-47. The system performance levels were acceptable.

TABLE 3-47

RHR SYSTEM PERFORMANCE DATA

Parameter	Test Data Loop A	Test Data Loop B	Process Diagram Value
Suppression Pool Cooling Mode			
Hx cooling water inlet (°F)	59.3	59.6	95
Hx cooling water outlet (°F)	67.8	68.4	103.2
Hx cooling water flow (GPM)	8586	8996	7400
Hx RHR inlet (°F)	91.9	90.1	70-120
Hx RHR outlet (°F)	80.5	75.4	103.1
RHR Flow (GPM)	7364	7348	7450
Shutdown Cooling Mode			
Hx cooling water inlet (°F)	46	54.1	85
Hx cooling water outlet (°F)	100	97.1	125
Hx cooling water flow (GPM)	8963	8859	7400
Hx RHR inlet (°F)	286	302	335
Hx RHR outlet (°F)	120	141	295
RHR Flow (GPM)	2314	2768	7450

3.43.3 Discussion

Following a reactor scram in TC-6, the RHR shutdown cooling mode was demonstrated. RHR system was operated at a reduced rate to allow shutdown cooling operation at time intervals long enough to establish steady state heat exchanger operation without exceeding reactor cooldown limit because of the low reactor decay heat inventory. The heat removal rate data has been corrected to account for the greater than design hx cooling flow rates to provide for direct comparison to the criteria. The RHR Heat Exchanger Duty is summarized as follows:

TABLE 3-48

RHR HEAT EXCHANGER DUTY

Mode	HX Loop	Heat Removal (Rate) Design	Heat Removal Rate (Actual)
Suppression Pool Cooling	A	30.15 MBTU/hr	31.7 MBTU/hr
Suppression Pool Cooling	B	30.15 MBTU/hr	32.6 MBTU/hr
Shutdown Cooling	A	148 MBTU/hr	200 MBTU/hr
Shutdown Cooling	B	148 MBTU/hr	160 MBTU/hr

Due to the excess cooling capacity of the heat exchangers and the capability to adjust flow to meet the process diagram flow levels, the performance data is considered acceptable.

3.44 Test Number 72 - Drywell Atmosphere Cooling

3.44.1 Purpose

The major objective of this test is to verify the ability of the Drywell Atmosphere Cooling System to maintain design conditions in the drywell during operation and post-scram conditions.

3.44.1.1 Level 1 Criteria

Not applicable

3.44.1.2 Level 2 Criteria

The drywell cooling system shall maintain an average ambient air temperature of 135°F or less and an 150°F or less ambient temperature at any single location in containment.

3.44.2 Test Results

The latest survey of selected maximum local drywell temperatures is presented in Table 3-49. This represents the existing drywell cooling system performance.

3.44.3 Discussion

The drywell cooling system has a total of five cooling units of which three are located in the lower drywell area and two service the upper drywell region. The average heat removal capacity per unit exceeded design values considerably. The data presented on Table 3-49 was taken with all the drywell cooling units in operation. The average drywell air temperature (as measured at the air inlet to the cooling units) did not exceed the 135°F limit during the entire test program. The local containment temperatures exceeding 150°F in the lower sacrificial shield was dispositioned acceptable because no safety equipment was located in this area. The drywell cooling system has undergone major ducting modifications during T/C heatup and other plant modifications are being implemented to improve the systems cooling capacity. A significant aspect of the modification reversed cooling air flow in the vessel annulus region causing an increase in what had been previously considered. An evaluation of the reactor vessel skirt and pedestal indicated that temperatures up to 210°F or an average of the three detectors in the skirt region less than 203°F is acceptable. These limits have been included in the operations staff surveillance process.

TABLE 3-49

LOCAL DRYWELL AIR TEMPERATURES

Location	Air Temperature (°F)
Upper Ring Header Return	135
Upper Sacrificial Shield	142
Vessel Head Flange	127
Head Return Duct	130
Safety Relief Valve Area	124
Lower Sacrificial Shield	176
CRD Area	135

3.45 Test Number 73 - Cooling Water Systems

3.45.1 Purpose

The objective is to assess the heat removal performance of the Standby Service Water (SW) system, the Reactor Building Closed Cooling Water (RCCW) System, and the Turbine Building Service Water (TSW) System.

3.45.1.1 Level 1 Criteria

Not Applicable

3.45.1.2 Level 2 Criteria

The system heat transport parameters either meet the requirements of the design specifications, or provide adequate cooling to the components serviced such that they operate satisfactorily.

3.45.1.3 Level 3 Criteria

Not Applicable

3.45.2 Test Results

A survey of selected groups of SW, RCCW, and TSW equipment heat transport performance data is presented in Table 73.1.

3.45.3 Discussion

Each system has been assessed as providing adequate component cooling. Some of the differences between the A/E design parameters selected as the acceptance criteria and the test data is a combination of the following:

1. The design conditions for flow and inlet temperature were not matched due to the current ambient conditions and a problem encountered at WNP-2 with silt accumulation in cooling system regions of low flow. This fouling process was minimized by increasing flows to components with low design flow/velocity and therefore subject to silt fouling. This increased the flow above design levels. The SW controller error data is also influenced by the silting problem.

2. Lower inlet temperatures due to the winter conditions and plant configuration created higher than design heat transport rates.

TABLE 73.1

EQUIPMENT HEAT TRANSPORT PERFORMANCE DATA

SERVICE WATER SYSTEM:

Parameter Parameter	Data	A/E Design
PRA-CC-1A Heat Transport	4.403×10^5 Btu/hr	4.040×10^5 Btu/hr
PRA-CC-1B Heat Transport	4.983×10^5 Btu/hr	4.040×10^5 Btu/hr
DG Room 1A Heat Transport	13.929×10^6 Btu/hr	14.366×10^6 Btu/hr
DG Room 1B Heat Transport	14.679×10^6 Btu/hr	14.366×10^6 Btu/hr
LPCS-P-1 Heat Transport	5.256×10^4 Btu/hr	3.200×10^5 Btu/hr
RHR-P-2A Heat Transport	2.704×10^4 Btu/hr	2.850×10^5 Btu/hr
RHR-P-2B Heat Transport	5.356×10^3 Btu/hr	2.850×10^5 Btu/hr
RHR-P-2C Heat Transport	4.005×10^3 Btu/hr	2.850×10^5 Btu/hr
RRA-CC-11 Heat Transport	5.397×10^4 Btu/hr	$7..128 \times 10^4$ Btu/hr
Control Room Ambient Air Temp.	73°F	78°F
Cable Spreading Room Ambient Air Temperature	73°F	96°F
Radwaste Critical Switchgear Room Ambient Air Temperature	85°F	102°F
RCIC Pump Room Ambient Air Temperature	85°F	103°F
HPCS DG Area Heat Transport	5.43×10^5 Btu/hr	7.366×10^6 Btu/hr
SW-P-1A Discharge Pressure	220 psig (507.5 ft.)	435 ft.
SW-P-1B Discharge Pressure	210 psig (484.4 ft.)	435 ft.
HPCS-P-2 Discharge Pressure	59 psig (128.5 ft.)	85 ft.
CCH-CU-1A Heat Transport	3.858×10^5 Btu/hr	N/A

TABLE 73.1 (Condt)

REACTOR BUILDING CLOSED COOLING WATER SYSTEM:

Parameter	Data	A/E Design Parameter
Total Heat Transported by RCC System	136.54×10^5 Btu/hr	44.39×10^6 Btu/hr
RWCU Non-Regenerative Heat Exchanger Heat Transfer Rate	9.46×10^6 Btu/hr	15.09×10^6 Btu/hr
Drywell Coolers Heat Transport	6.03×10^6 Btu/hr	5.00×10^6 Btu/hr

PLANT SERVICE WATER SYSTEM:

Main Turbine Oil Cooler Controller Error	-11.67%	± 1%
Main Turbine Hydrogen Cooler Controller Error	3.81%	± 2%
Exciter Coolers Controller Error	45.33%	± 2%
Stator Water Coolers Controller Error	-2.15%	± 2%
TO-HX-2A,B Effluent Temperature Controller Error	0%	± 2%
TO-HX-2C,D Effluent Temperature Controller Error	1.82%	± 2%
WMA-AH-53A Effluent Air Temperature Controller Error	14.29%	± 3%
WMA-AH-53B Effluent Air Temperature Controller Error	0%	± 3%

3.46 Test Number 74 - Offgas System

3.46.1 Purpose

The major objectives of this test are as follows:

A. To verify the proper operation of the Offgas System over its expected operating parameters;

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

3.46.1.1 Level 1 Criteria

The release of radioactive gaseous and particulate effluents must not exceed the limit specified in the Technical Specification.

There shall be no loss of flow of dilution steam to the non-condensing stage when the steam jet air ejectors are pumping.

3.46.1.2 Level 2 Criteria

The system flow, pressure, temperature, and relative humidity shall comply with the design specifications.

The catalytic recombiner, the hydrogen analyzer, the activated carbon beds, and the filters shall be operable.

3.46.2 Test Results

The system performance data were taken at steady state conditions during heatup, Test Condition 1, 3 and 6. All applicable Level 1 Criteria were satisfied at each testing level.

Several parameters were initially outside the system design specification. The Offgas System operating results are summarized in Table 3-50. It was concluded that the Offgas System is capable of performing all design functions.

The Krypton-85 retention time prior to initial steam flow to the main condenser was measured equal to 136 cc/gram which satisfied the expected performance level of 105 cc/gram.

3.46.3 Discussion

During the heatup and Test Condition 1, the maximum dilution steam flow to the non-condensable stage of the steam jet air ejector was 6400 lb/hr which was below the minimum required value of 8464 lb/hr. Bypass piping was added around the second stage air ejector steam supply and the proper dilution steam flow was obtained.

The offgas flow was reduced from 140 scfm during TC-3 to 73 scfm at TC-6 but was still higher than maximum desirable 30 scfm. The reduction of condenser air inleakage will be a continuous effort.

TABLE 3-50

OFFGAS SYSTEM DESIGN PARAMETERS AND RESULTS

Parameter	Indicator	Normal Operation Range	Result for Test Condition Heatup	1	3	6
Date			05/04/84	05/08/84	09/04/84	11/04/84
Core Thermal Power (%of 3323 MWt)	OD3, Opt 2	N/A	3.5	15	48.7	99.3
Dilution Steam (lbm/hr)	MS-FI-25A(B)	10100-10600 lb/hr	6300	6400	9800	10700
SJAE Outlet Pressure (psig)	OG-PIS-600A(B)	0.5-5 psig	3	1.5	3	1.5
Preheater Outlet Temperature (°F)	OG-TIS-601A(B)	325-375°F	370	365	350	340
Active Recombiner Temperature (°F)	OG-TRS-602		(Loop B)	(Loop B)	(Loop B)	(Loop B)
Bottom	TE-3A(B)	375-830°F	395	418	480	625
Middle	TE-4A(B)	375-830°F	395	417	480	625
Top	TE-5A(B)	375-830°F	395	420	485	625
Active Recombiner Temperature (°F)	OG-TRS-602		(Loop A)	(Loop A)	(Loop A)	(Loop A)
Bottom	TE-3A(B)	375-830°F	210	235	350	325
Middle	TE-4A(B)	375-830°F	302	290	400	420
Top	TE-5A(B)	375-830°F	340	335	410	460
O.G. Condenser Condensate Outlet (°F)	COND-TI-4	130°F	68	73	97	118
O.G. Condenser Offgas Outlet (°F)	COND-TIS-6	150°F	87	90	97	110

TABLE 3-50

OFFGAS SYSTEM DESIGN PARAMETERS AND RESULTS

Parameter	Indicator	Normal Operation Range	Result for Test Condition			
			Heatup	1	3	6
H_2 Concentration (%)	OG-H2R-605	0-0.1%	Non-Detectable	Non-Detectable	0	0.1
Offgas Flow (scfm)	OG-FR-617	6-30 scfm	150-170	185	140	73
Glycol Pump Disch. Pressure (psig)	GY-PI-631	15-50 psig	49	50	47	30
Glycol Tank Temp (°F)	GY-TRS-630	32.5-35.5°F	34	33	33	34
Moisture Separator Outlet Temp (°F)	OG-TRS-610A(B)	36-45°F	39	42	40	43
Prefilter dP (inch water)	OG-DPIS-611A(B)	1" WC	0.5-2	0	3	0.6
Dryer Out Dewpoint Temp (°F)	OG-TR-641A(B)	-80 - -100°F	-47	-79	-86	-90
Regen Chiller Out Temp (°F)	OG-TIS-641A(B)	36-45°F	64	48	40-60	40-45
Adsorber Train dP (psid)	OG-DPIS-612	2.6 psid	(Note 1) Not in Service	0-1.5	0.6	0.3
Adsorber Vessel Temp	OG-TRS-613					
12A, (12B)	OG-TE-23A(B)	+5 - -5°F	Not in Service	95	-2	0
12A, (12B)	OG-TE-24A(B)	+5 - -5°F	Not in Service	99	2	0
12A, (12B)	OG-TE-25A(B)	+5 - -5°F	Not in Service	98	16	0
13A, (13B)	OG-TE-26A(B)	+5 - -5°F	Not in Service	99	4	0
14A, (14B)	OG-TE-27A(B)	+5 - -5°F	Not in Service	88	4	0
15A, (15B)	OG-TE-28A(B)	+5 - -5°F	Not in Service	88	5	0

TABLE 3-50

OFFGAS SYSTEM DESIGN PARAMETERS AND RESULTS

Parameter	Indicator	Normal Operation Range	Heatup	Result for Test Condition 1	3	6
Adsorber Vault Temp (°F)	OG-TRS-614	+5 - -5°F	Not in Service	90	0-20	0.4
After Filter dP (inch water)	OG-DPIS-619	1" WC	0.5-2	0.5	1	0.5
Outside Air Temp (°F)			51	54	88	40
Outside Air % Relative Humidity			41	47	18	71

Note 1: The charcoal beds were bypassed during the heatup

4.0 SPECIAL TEST RESULTS

4.1 Moderator Temperature Coefficient

4.1.1 Purpose

The moderator temperature coefficient measurement was conducted to provide a benchmark for the SIMULATE core simulator code utilized by the Supply System. This measurement was also performed to supply temperature reactivity correction factors for estimated critical positions and supplement the data provided in the cycle management report.

4.1.2 Test Description

The measurement was performed with the reactor in a just critical condition approximately one decade below the heating range. An insequence control rod was withdrawn, placing the reactor on a positive period. The reactor was allowed to continue on this period until the moderator and doppler effects again returned the reactor to a just critical condition.

4.1.3 Test Results

The temperature coefficient (moderator plus doppler) was measured to be -4.9×10^5 delta K/K/°F at a 210°F moderator temperature.

4.2 In-Plant Safety Relief Valve Test

4.2.1 Purpose

In response to NUREG-0763, "Guidelines for Confirmatory In-Plant Tests of Safety-Related Discharges for BWR Plants", the Supply System committed to perform an in-plant test to measure the differences between local and bulk suppression pool temperatures during a main steam relief valve discharge.

4.2.2 Test Description

In accordance with the NUREG-0783 guidelines, the local temperature was to be measured by a sensor mounted on the containment wall opposite the discharging quencher. Sensor SPTM-TE-12 and the discharge piping from MSRV-3B met this geometric criteria.

The test was conducted in conjunction with the safety relief valve capacity testing performed between test conditions 2 and 3. MSRV-3B was the first valve opened in the series of relief valve capacity measurements and was held open for 4-1/2 minutes. Data was acquired on tape at 500 samples per second and plotted by playback at 20 samples per second

4.2.3 Test Result

At 40 seconds into the transient the largest differences between local and average suppression pool temperature was experienced at 14°F delta T. The average difference during the 4-1/2 minute discharge was approximately 8°F delta T. Acceptable performance was demonstrated by an average temperature differenece of less than 15°F delta T. WNP-2 performance was within acceptable bounds.

4.3 Sacrificial Shield Wall (SSW) Verification

To confirm the adequacy of the WNP-2 sacrificial shield wall, a special test was performed to obtain radiation measurements at specific locations inside the drywell. The test had two objectives:

1. To verify that voids discovered in the sacrificial shield wall were adequately filled to bring the wall to its designed radiation shielding capability.

2. To verify that Class 1E safety related electrical equipment in the Primary Containment between elevations 533' and 557' (reactor core zone) will not receive a reactor lifetime radiation exposure above designed limits.

Based on the results of the radiation measurements it was concluded that all equipment locations had radiation levels below the design criteria and radiation levels near the weld ring gap and the voids that were filled were within the design criteria verifying the sacrificial shield wall was adequately repaired.

4.4 Loose Parts Detection System

The primary purpose of this test was to adjust the trip threshold and sensitivity of the Loose Parts Detection System (LPDS) as primary coolant system noise varied during power ascension, to obtain optimal settings for normal plant operation. Another objective was to record on magnetic tape, baseline noise characteristics for various operating modes to use as comparisons during plant life.

Tests were performed during test conditions heatup, 2, 3 and 6. Channel gains, trip thresholds, and filter roll-offs were adjusted to optimize system sensitivity to noise characteristic of metallic impacts while desensitizing the system to characteristic fluid noises. The LPDS system was activated to record baseline data during specific plant maneuvers.

1. Recirculation pump motor speed changes (slow to fast and fast to slow).

2. Recirculation flow control valve position changes.

3. MSIV position chantes and SRV actuation.

The objectives of this test have successfully been met. The LPDS system is adjusted such that the system is capable of detecting a loose part with a minimum of false alarm. Also a better data base has been created to help locate loose parts and develop system behavioral trends.