TU ELECTRIC COMPANY
COMANCHE PEAK STEAM ELECTRIC S[illegible]ON
UNIT 1 CYCLE 1

STARTUP REPORT

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1.0 - INTRODUCTION

This report describes the required testing at Comanche Peak Steam Electric Station, Unit 1, from the preparations for loading the first fuel assembly into the reactor until the plant was placed in commercial operation. It satisfies the requirement of the Comanche Peak Technical Specifications that a Startup Report be submitted to the NRC after completion of the Startup Testing Program.

Comanche Peak Steam Electric Station, located in North Central Texas, utilizes a four loop Westinghouse Pressurized Water Reactor as the Nuclear Steam Supply System. Westinghouse Electric Corporation, Stone & Webster Engineering Corp., Gibbs & Hill, Inc. Impell Corp., Ebasco, Brown & Root, Inc. and the TU Electric Company jointly participated in the design and construction of Comanche Peak. The plant is operated by the TU Electric Company.

The Nuclear Steam Supply System is designed for a thermal power output of 3425 MWth (3411 MWth reactor power). The equivalent warranted gross electrical output is 1163 MWe. Cooling for the plant is provided by the Squaw Creek Reservoir, a 135,062 acre-foot man-made lake. Post design basis accident cooling is provided by a separate 367 acre-foot Safe Shutdown Impoundment.

Table 1.0-1 provides a cross reference between the test summaries in the Final Safety Analysis Report and the sections of this report.

Table 1.0-1

Cross Reference of FSAR Table 14.2-3
and Unit 1 Cycle 1 Startup Report

FSAR Table 14.2-3 SHEET NUMBER	TITLE	Startup Report Section
2	Reactor Coolant System Flow Test	3.2.8, 3.2.14
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23	Unit Load Transients	3.4.2, 3.4.3, 3.4.5
25	Remote Shutdown	3.4.4

Table 1.0-1

Cross Reference of FSAR Table 14.2-3
and Unit 1 Cycle 1 Startup Report (Continued)

FSAR Table 14.2-3 SHEET NUMBER	TITLE	Startup Report Section
28	Turbine Trip/Generator Load Rejection	3.4.3
29	Reactor Coolant Leak Test	3.2.10
31	Rod Control System Test	3.2.12
33	Automatic Control System Test	3.5.10
34	Incore Nuclear Instrumentation	3.2.4

2.0 - DISCUSSION OF THE INITIAL STARTUP PROGRAM

The Comanche Peak Unit 1 initial startup testing program consisted of single and multi-system tests that were performed commencing with initial fuel loading and continuing through full power operation. The intent of these tests is to assure that tests deferred from the preoperational test program are performed; that the plant is safely brought to rated capacity; that plant performance is satisfactory in terms of established design criteria; and to demonstrate, where practical, that the plant is capable of withstanding anticipated transients and postulated accidents. These tests demonstrated overall plant performance and included such activities as precritical testing, low power tests, and power ascension tests. Testing sequence documents were utilized for each plateau to coordinate the sequence of testing activities at that plateau.

In the subsections that follow, a description of the testing at each plateau is provided. The descriptions include additional details concerning special license conditions and commitments made to the Nuclear Regulatory Commission prior to completion of the startup testing program, where applicable. Also included as a part of Section 2.0 are tables and figures showing major milestones for Comanche Peak Unit 1 which occurred during the initial startup program and a list of operational modes as defined by the Technical Specifications.

TABLE 2.0 - 1

COMANCHE PEAK UNIT 1 MAJOR MILESTONES

MAJOR MILESTONES	DATE
5% Power License Received	2/08/90
Fuel Load Started	2/09/90
Fuel Load Completed	2/14/90
Initial Criticality	4/03/90
5% License (Low Power) Tests Completed	4/06/90
Full Power License Received	4/17/90
Entered Mode 1	4/19/90
Initial Synchronization to Grid	4/24/90
30% Power Reached	4/30/90
50% Power Reached	5/04/90
75% Power Reached	6/27/90
100% Power Reached	7/13/90
Test Review Group Approves Startup Test Program	7/30/90
V. P. Nuclear Operations Declares Completion of Startup Test Program and Commencement of Commercial Operation	8/13/90

TABLE 2.0 - 2

OPERATIONAL MODES

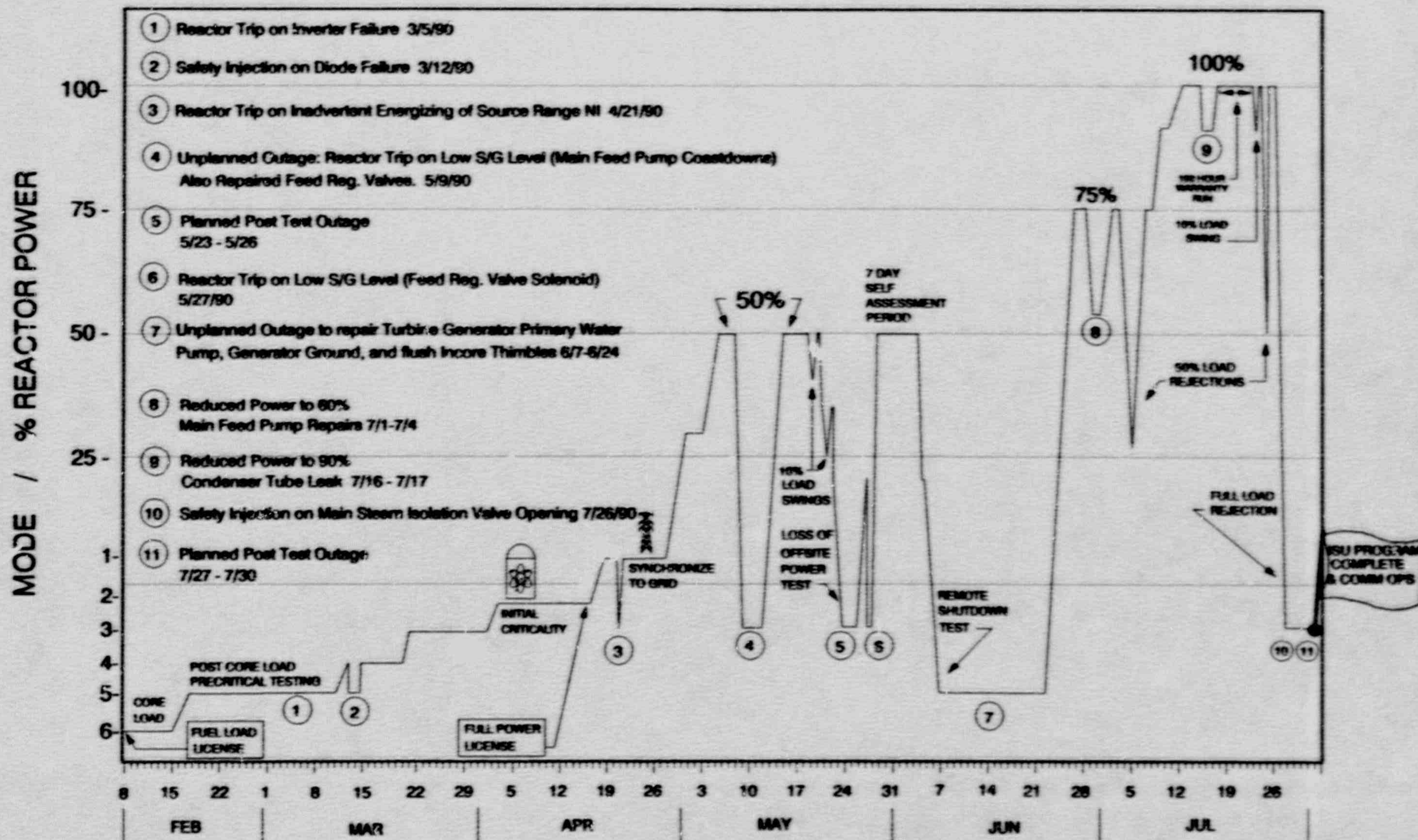
MODE		REACTIVITY CONDITION, K_{eff}	%RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1.	POWER OPERATION	≥ 0.99	$> 5\%$	$\geq 350°F$
2.	STARTUP	≥ 0.99	$\leq 5\%$	$\geq 350°F$
3.	HOT STANDBY	< 0.99	0	$\geq 350°F$
4.	HOT SHUTDOWN	< 0.99	0	$350°F > T_{avg} > 200°F$
5.	COLD SHUTDOWN	< 0.99	0	$\leq 200°F$
6.	REFUELING**	≤ 0.95	0	$\leq 140°F$

*Excluding decay heat.

**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

ISU PROGRAM SUMMARY

Figure 2.0-1



2.1 - INITIAL FUEL LOAD SEQUENCE - ISU-001A

OBJECTIVE

The Initial Load Sequence document defines the sequence of testing and other operations to prepare for and perform initial core loading. This test partially satisfies activities described in FSAR Section 14.2.10.1.

TEST METHODOLOGY

The Fuel Load Sequence Document is used to coordinate the sequence of operations associated with the initial core loading program. This sequence includes scheduling of the individual startup tests and selected key permanent plant procedures associated with core loading. This document specifies as prerequisites which testing had to be completed prior to commencement of core loading, the required status of the plant systems necessary to support core loading, and the reactor vessel status. A log is also included in the sequence document as Form ISU-001A-1 to verify Technical Specification compliance prior to and throughout core loading. This document also provides the criteria for stopping core loading, the criteria for emergency boration, and the actions to be followed prior to the resumption of core loading in the event loading was stopped prior to completion.

SUMMARY OF RESULTS

Initial core loading of 193 fuel assemblies took 121 hours. Prior to the start of core loading, the condition of the reactor vessel and associated components, the reactor coolant system, instrumentation, and administrative controls were verified to be acceptable. The sequence procedure verified that reactor coolant system chemistry was properly established and maintained and verified timely nuclear instrumentation neutron response checks. The procedure also ensured that a final general fuel assembly visual inspection was performed. Fuel loading operations were performed using permanent plant procedures. Results of individual tests completed during the core loading sequence are discussed in Section 3.1 of this report. Upon completion of core loading, plant systems were aligned as directed by the Shift Supervisor.

2.2 - POST CORE LOAD PRECRITICAL TEST SEQUENCE (PCLPC) - ISU-010A

OBJECTIVE

The PCLPC Sequence Document defines the sequence of tests and operations to be performed between completion of initial core loading and prior to initial criticality. This testing is performed in Technical Specification Modes 5, 4 and 3.

TEST METHODOLOGY

This document ensures that core load testing had been successfully completed and results approved prior to continuation of the testing program. This document schedules the performance of precritical tests to ensure the necessary testing was completed prior to initial criticality. This procedure governs the sequence of testing through Modes 5,4 and 3. Plant operating procedures are utilized where appropriate to establish necessary plant conditions.

SUMMARY OF RESULTS

Results of individual tests completed during the post core load precritical testing phase are discussed primarily in Sections 3.2 and 3.6 of this report. A daily log of RCS and pressurizer boron concentration was kept to ensure adequate shutdown margin during testing. Boron concentration varied between 2022 ppm and 2124 ppm. This insured that the boron concentration was greater than the 2000 ppm refueling concentration at all times. Upon completion of this testing phase, plant systems were aligned as directed by the Shift Supervisor.

2.3 - INITIAL CRITICALITY & LOW POWER TEST SEQUENCE (IC & LPT) - ISU-101A

OBJECTIVE

The IC & LPT Sequence Document defines the sequence of tests and operations, beginning with initial criticality, which constitute the low power physics testing program. This program of low power physics testing verifies the design of the reactor by performing a series of selected measurements including core flux distributions, control bank worths and moderator temperature coefficient. This test sequence partially satisfies activities described in FSAR Sections 14.2.10.2 and 14.2.10.3.

TEST METHODOLOGY

This document ensures that post core loading precritical testing has been completed and results approved prior to continuation of the testing program. Prior to commencement of dilution to initial criticality, source range nuclear instrumentation channels are verified to have a signal to noise ratio greater than 2 and power range high level trip setpoints are conservatively set to ≤ 20% of full power. This procedure sequences the low power physics testing into an efficient order and ensures that all required testing is performed. Surveillance Requirements for Technical Specification 3.10.3 usage are also controlled by this test. This Technical Specification Special Test Exception permits physics testing in non-normal operating reactor controls configurations.

A reactivity computer is set up using a power range NIS channel detector output to monitor core flux. This device is an analog computer that calculates the amount of reactivity present in the core based on the time dependence of core flux. This device is used in core physics testing to make measurements of control rod, boron concentration and moderator temperature worths.

A low power flux mapping system is used to augment the installed flux mapping system. This low power system contains very low noise bias voltage supplies and sensitive signal detection instrumentation to permit flux mapping below the point of adding nuclear heat, thus avoiding xenon and power stability effects on flux map results.

Plant operating procedures are utilized where appropriate to establish and maintain plant conditions.

2.3 - INITIAL CRITICALITY & LOW POWER TEST SEQUENCE (IC & LPT) - ISU-101A (Continued)

SUMMARY OF RESULTS

This sequence document obtained a full core flux map at the Hot Zero Power, Xenon-free, All Rods Out Condition. Refer to Table 2.3-1 for a tabulation of the flux map results obtained. The low power flux mapping system was successfully used to obtain this map. These map results were of sufficient quality to eliminate the need to perform a 30% power flux map. Results of individual tests completed during the initial criticality and low power test sequence are discussed primarily in Section 3.3 of this report. A tabulation of key physics measurement results is also included in Table 2.3-1. All required tests were performed. Initial criticality was achieved without incident on 4/3/90. A low power physics testing power range was determined and the reactivity computer was verified to be operating properly. Boron endpoint concentration measurements were performed and data was taken for later comparison with 100% power data to verify proper power defect. The Moderator Temperature Coefficient was then measured and found to be positive. This was not unexpected based on information in the reactor Nuclear Design Report (WCAP-9806). In response to this positive coefficient, rod withdrawal limits were imposed using a permanent plant procedure, NUC-116. Control rod worths were verified using the bank exchange method (rod swap method). All required testing was completed.

Upon completion of this testing phase, the plant was aligned as directed by the Shift Supervisor.

TABLE 2.3-1
HZP PHYSICS TESTING RESULTS

FLUX MAP RESULTS	Actual	Maximum Limit
Reaction Rate Error	8.62%	10%
FDHN	1.605	1.643
FQ(Z)	2.5203	3.314 x K(Z)
Quadrant Power Tilt Ratios	0.9828 | 1.0109 0.9828 | 1.0235	1.04

MISC. PHYSICS TESTING RESULTS	Actual	Allowed Range
All Rods Out Critical Boron(ppm)	1162.1	1096 to 1196
Reference Bank In Critical Boron(ppm)	1087.0	1012 to 11[illegible]2
Isothermal Temperature Coefficient (pcm/°F)	-0.995	-4.4 to +1.6
Moderator Temperature Coefficient(pcm/°F)	+0.835	<0 (unless rod withdrawal limits are set)
Reactivity Computer Error	0.81%	$\leq \pm 4\%$
Source Range/Intermediate Range NIS Overlap (decades)	1.6	$\geq$ 1.5
Reference Rod Bank Worth Error	-0.4%	$\leq \pm 10\%$
All Other Banks Worth Error (max.)	-13.4%	$\leq \pm 15\%$
Total Rod Bank Worth Error	+ 1.8%	-10% to +7%
Differential Boron Worth (pcm/ppm)	-11.63*	-9.40 to -11[illegible]8

* Refer to Test Summary 3.3.7, NUC-120, for discussion of this out of range value.

NOTE: pcm means percent millirho, equivalent to a reactivity value of $10^{-5} \Delta K/K$

2.4 - 50% REACTOR POWER TEST SEQUENCE - ISU-240A

OBJECTIVE

The 50% Reactor Power Test Sequence document defines the activities which constitute the startup testing program between 0% and 50% power and at approximately 50% of rated thermal power. This test partially satisfies activities described in FSAR Table 14.2-3, Sheet 22 and Section 14.2.10.4.

TEST METHODOLOGY

This document ensures that the low power physics testing has been completed and the results approved prior to increasing power. Prior to increasing power for this test sequence, power range high level trip setpoints are conservatively set to ≤ 70% power and reactor core flux map results from a 0% power baseline map are verified acceptable. The flux map results are also extrapolated to 70% power to ensure parameters indicative of DNBR and linear heat rate are acceptable for power ascension to the 50% testing plateau.

Plant operating procedures are utilized where appropriate to establish plant conditions and to change reactor power. During this testing sequence following completion of 50% power testing, power is stabilized near the 30-35% and 20-25% levels to accommodate testing at those power levels.

SUMMARY OF RESULTS

Results of individual tests completed up to and while at the 50% power plateau are discussed primarily in Sections 3.2, 3.4 and 3.5 of this report. Administrative hold points on continued testing were observed at 10%, 20% and 30% power during the initial power ascension to 50% power. A flux map was taken at 47.55% power with satisfactory results as summarized in Table 2.4-1. All required testing was completed.

Upon completion of this testing phase, the plant was aligned as directed by the Shift Supervisor.

TABLE 2.4-1

50% POWER FLUX MAP RESULTS

	Actual	Maximum Limit
Reaction Rate E	8.85%	10%
FDHN	1.4994	1.7126
FQ(Z	2.1131	4.64 x K(Z)
Fxy - unrodded	1.6098	1.7126
Quadrant Power Tilt Ratios	1.0005 1.0100 0.9840 1.0055	1.02

2.5 - 75% REACTOR POWER TEST SEQUENCE - ISU-260A

OBJECTIVE

The 75% Reactor Power Test Sequence document defines the activities which constitute the startup testing program during escalation from 50% to 75% power and at approximately 75% of rated thermal power. This test partially satisfies activities described by FSAR Table 14.2-3, Sheet 22 and Section 14.2.10.4.

TEST METHODOLOGY

This document ensures that the 50% Reactor Power Test Sequence has been completed and the results approved prior to increasing power above the 50% testing plateau. Prior to increasing power for this test sequence, power range high level trip setpoints are conservatively set to ≤ 95% power and reactor core flux map results from a 50% power baseline map are verified acceptable. The flux map results are also extrapolated to 95% power to ensure parameters indicative of DNBR and linear heat rate are acceptable for power ascension to the 75% testing plateau.

Plant operating procedures are utilized where appropriate to establish plant conditions and to change reactor power.

SUMMARY OF RESULTS

Results of individual tests completed while at the 75% plateau are discussed primarily in Sections 3.2, 3.4 and 3.5 of this report. The extrapolation of the 50% power plateau flux map results to 95% power indicated that the Fxy peaking factor limit would be exceeded at 95% power. The Fxy extrapolation was acceptable for power levels up to 94.5%. Reactor Engineering, the TU Electric reactor core design group, evaluated this item and concluded that adequate FQ(Z) margin existed because while Fxy is used as a Technical Specification Surveillance parameter to ensure adequate FQ(z) margin, this use of Fxy assumes a certain operationally varying axial power distribution, F(z). Because the plant was in a power ascension program instead of a load follow operating regime, F(z) values are not as large as are assumed in determination of Fxy limitations. Additional justifications for accepting the flux map extrapolated results as sufficient for ensuring safe operation were that the measured Fxy is typically observed to decrease with power increase and based on Fxy limit satisfaction up to 94.5% power which was judged to be sufficiently close to 95%. Another flux map was taken at approximately 67% power to further confirm peaking factor behavior. These results were satisfactory.

SUMMARY OF RESULTS (Continued)

Heater Drain system and Moisture Separator Reheater 1-B Main Steam Sample flows, temperatures and pressures were also verified acceptable to close a testing item carried over from the preoperational test program. This testing was non-safety related and was not a deferred preoperational test requirement.

A flux map was taken at 77.43% power with satisfactory results as summarized in Table 2.5-1. These results were of sufficient quality such that a 90% power flux map was not required. All required testing was completed.

Upon completion of this testing phase, the plant was aligned as directed by the Shift Supervisor.

TABLE 2.5-1

75% POWER FLUX MAP RESULTS

	Actual	Maximum Limit
Reaction Rate Error	5.98%	10%
FDHN	1.4383	1.6200
FQ(Z)	2.0659	2.9963 x K(Z)
Fxy - unrodded	1.5266	1.6200
Quadrant Power Tilt Ratios	0.9987 | 1.0043 0.9904 | 1.0066	1.02

2.6 - 100% REACTOR POWER TEST SEQUENCE - ISU-280A

OBJECTIVE

The 100% Reactor Power Test Sequence document defines the activities which constitute the startup testing program during escalation from 75% to 100% power and at close to, but not more than, 100% of rated thermal power. This test partially satisfies activities described in FSAR Table 14.2-3, Sheet 22 and Section 14.2.10.4.

TEST METHODOLOGY

This document ensures that the 75% Reactor Power Test sequence has been completed and the results approved prior to increasing power above the 75% testing plateau. Prior to increasing power above 75% for this test sequence, reactor core flux map results from a 75% power baseline map are verified acceptable and the power range high level trip setpoints are set to $\leq$ 109%, their normal Technical Specification values. The flux map results are also extrapolated to 100% power to ensure parameters indicative of DNBR and linear heat rate are acceptable for power ascension to the 100% testing plateau.

Plant operating procedures are utilized where appropriate to establish plant conditions and to change reactor power. During ascension to the 100% plateau, power is stabilized near the 90% and 98% levels to accommodate testing at those power levels.

SUMMARY OF RESULTS

Results of individual tests completed during this power ascension and while at the 100% plateau are discussed primarily in Sections 3.2, 3.4 and 3.5 of this report.

A flux map was taken at 99.03% power with satisfactory results as summarized in Table 2.6-1. All required testing was completed.

Upon completion of this testing phase, the plant was aligned as directed by the Shift Supervisor.

TABLE 2.6-1

100% POWER FLUX MAP RESULTS

	Actual	Maximum Limit
Reaction Rate Error	6.81%	10%
FDHN	1.4504	1.553
FQ(Z)	2.0449	2.34 x K(Z)
Fxy - unrodded	1.5475	1.553
Quadrant Power Tilt Ratios	1.0030 | 1.0067 0.9887 | 1.0016	1.02

3.0 DISCUSSION OF THE INITIAL STARTUP TESTS

TABLE 3.0-1
List of Test Summaries

3.1 CORE LOADING

3.1.1	Development and Implementation of the Reload Fuel Shuffle Sequence Plan, RFO-[illegible]6
3.1.2	Core Loading Instrumentation and Neutron Source Checks, ISU-003A
3.1.3	Inverse Count Rate Ratio Monitoring (Core Load Portion), NUC-111
3.1.4	RCS and Secondary Coolant Chemistry (Core Load Portion), ISU-006A
3.1.5	Verification of Core Loading Pattern, RFO-204

3.2 SYSTEM TESTING AFTER CORE LOAD AND AT VARIOUS POWER LEVELS

3.2.1	Piping Vibration Monitoring, ISU-212A
3.2.2	Steam Generator Level Control Test, ISU-207A
3.2.3	Thermal Expansion, Power Ascension Phase, ISU-308A
3.2.4	Incore Moveable Detector System Alignment, ISU-016A
3.2.5	RCS and Secondary Coolant Chemistry (Post Core Load), ISU-006A
3.2.6	Radiation Survey Tests, ISU-208A
3.2.7	Process and Effluent Radiation Monitoring Performance Test, ISU-210A
3.2.8	Reactor Coolant Flow Measurement, ISU-023A
3.2.9	Reactor Coolant System Flow Coastdown Test, ISU-024A
3.2.10	Reactor Coolant System Leakage Rate Test, ISU-022A
3.2.11	Cold Control Rod Operability Testing, ISU-026A
3.2.12	Hot Control Rod Operability Testing, ISU-027A
3.2.13	Reactor Trip System Tests, ISU-015A
3.2.14	Pressurizer Spray and Heater Capability, ISU-021A
3.2.15	Miscellaneous Balance of Plant Testing

3.3 PHYSICS TESTING

3.3.1	Inverse Count Rate Ratio Monitoring (Initial Criticality Portion), NUC-111
3.3.2	Initial Criticality, NUC-106
3.3.3	Determination of Core Power Range for Physics Testing, NUC-109
3.3.4	Reactivity Computer Checkout, NUC-108
3.3.5	Core Reactivity Balance, NUC-205
3.3.6	Surveillance of Core Power Distribution Factors, NUC-201
3.3.7	Zero Power Isothermal and Moderator Temperature Coefficient Measurements, NUC-207
3.3.8	Determination of Operating Limits to Ensure a Negative MTC, NUC-116

TABLE 3.0-1 (Continued)

3.3.9 Rod Swap Measurements, NUC-120
3.3.10 Boron Endpoint Determination and Differential Boron Worth, NUC-104

3.4 TRANSIENT TESTING

3.4.1 Turbine Generator Trip With Coincident Loss of Offsite Power, ISU-222A
3.4.2 Design Load Swing Tests, ISU-231A
3.4.3 Dynamic Response to Full Load Rejection and Turbine Trip, ISU-284A
3.4.4 Remote Shutdown Capability Test, ISU-223A
3.4.5 Large Load Reduction Tests, ISU-263A

3.5 INSTRUMENTATION AND CALIBRATION TESTING

3.5.1 Calibration of Feedwater and Steam Flow Instrumentation at Power, ISU-202A
3.5.2 Thermal Power Measurement and Statepoint Data Collection, ISU-224A
3.5.3 Operational Alignment of Process Temperature and N16 Instrumentation, ISU-226A
3.5.4 Operational Alignment of Nuclear Instrumentation, ISU-204A
3.5.5 Incore/Excore Detector Calibration, NUC-203
3.5.6 Loose Parts Monitoring Baseline Data, ISU-211A
3.5.7 Startup Adjustments of Reactor Control Systems, ISU-020A
3.5.8 Full Power Performance Test, ISU-281A
3.5.9 P2500 Process Computer Software Verification, ISU-019A
3.5.10 Automatic Reactor Control System Test, ISU-203A

3.6 DEFERRED PREOPERATIONAL TESTING

3.6.1 Process Sampling System, ISU-028A
3.6.2 In-place Atmospheric Cleanup Filter Test - Primary Plant - ESF, EGT-751X
3.6.3 Containment & Penetration Rooms Temperature Survey, ISU-282A
3.6.4 Turbine Driven Auxiliary Feedwater Pump Actuation and Response Time Tests, EGT-768A and EGT-769A
3.6.5 MSIV Isolation Response Time Tests, EGT-764A and EGT-765A
3.6.6 Reactor Coolant System Pressure Isolation Valve Leakage Testing, EGT-712A
3.6.7 Condensate Reject Valve Test, EGT-TP-90A-002

3.1 CORE LOADING

3.1.1 - Development and Implementation of the Reload Fuel Shuffle Sequence Plan, RFO-106

OBJECTIVES

This permanent plant procedure is performed to ensure that the nuclear fuel assemblies are loaded in a safe and cautious manner. This procedure partially satisfies activities described in FSAR Section 14.2.10.1.

TEST METHODOLOGY

The procedure is performed prior to the start of core loading to develop the detailed core loading sequence sheets. Field use of the procedure begins following loading of the temporary core loading instrumentation into its initial position and determination of background count rates for all source range and temporary nuclear instrumentation channels. The four primary source bearing assemblies and six additional assemblies, comprising the "source nucleus", are loaded. Audible indication of neutron population changes from one of the two installed source range plant channels is required to be maintained in both the control room and containment for the duration of the core loading process. After the source nucleus assemblies are loaded, count rate data is taken for the nuclear channels used in the core loading process (two source range and three temporary channels). The first reference value, for use in inverse count rate ratio monitoring, is determined from these counts after the appropriate background values have been subtracted. Subsequent reference values are calculated whenever core loading is suspended for eight hours or longer, a temporary detector is moved, or a primary source bearing fuel assembly is moved to a different core location.

Prior to fuel load, predictions were made for comparison to actual nuclear instrumentation response, to verify that the reactor would remain shutdown throughout the loading process. Inverse count rate ratio monitoring is used following each fuel assembly move to ensure that the reactor is not approaching criticality. To ensure reliability in the monitoring, a minimum of two of the five nuclear instrumentation channels are required to be responding to source neutron population changes throughout core loading. Data obtained during inverse count rate ratio monitoring is trended and extrapolated forward to permit evaluation of any indicated criticality approach. Plant procedure NUC-111 is used to perform the inverse count rate ratio measurements and extrapolations.

3.1.1 - Development and Implementation of the Reload Fuel Shuffle Sequence Plan, RFO-106 (Continued)

SUMMARY OF RESULTS

Core loading was completed in a safe and cautious manner as required by the acceptance criteria of the core loading procedure. Problems encountered during the test were primarily associated with readjustment of the source range NIS high flux at shutdown alarm bistables and actuation of the high flux at shutdown alarms. The alarm actuated several times, due primarily to the presence of 4 Californium primary neutron sources and associated stronger source to detector couplings. Due to a several year delay in actual core loading, the original two primary sources that had been received were augmented with two fresh sources to ensure a sufficiently high neutron count rate for core loading and initial criticality nuclear monitoring. The high flux at shutdown alarm also actuated once in response to source range spiking caused by high voltage switching in the main switchyard.

Source range NIS channels also lost power twice during initial core load. These losses of power were unrelated and not coincident. The N31 channel power loss was caused by an inverter breaker failure. The inverter problem resulted in a delay of greater than 8 hours in core loading, so all neutron monitors were again response tested. The N32 power loss was caused by an error in the switching of the Solid State Protection System. Power was immediately restored.

The fuel handling equipment performed very well, with only one failure. The manipulator crane (refueling machine) gripper jammed once and was mechanically freed with vendor assistance. It jammed in the unlatched position and not while it was gripping a fuel assembly.

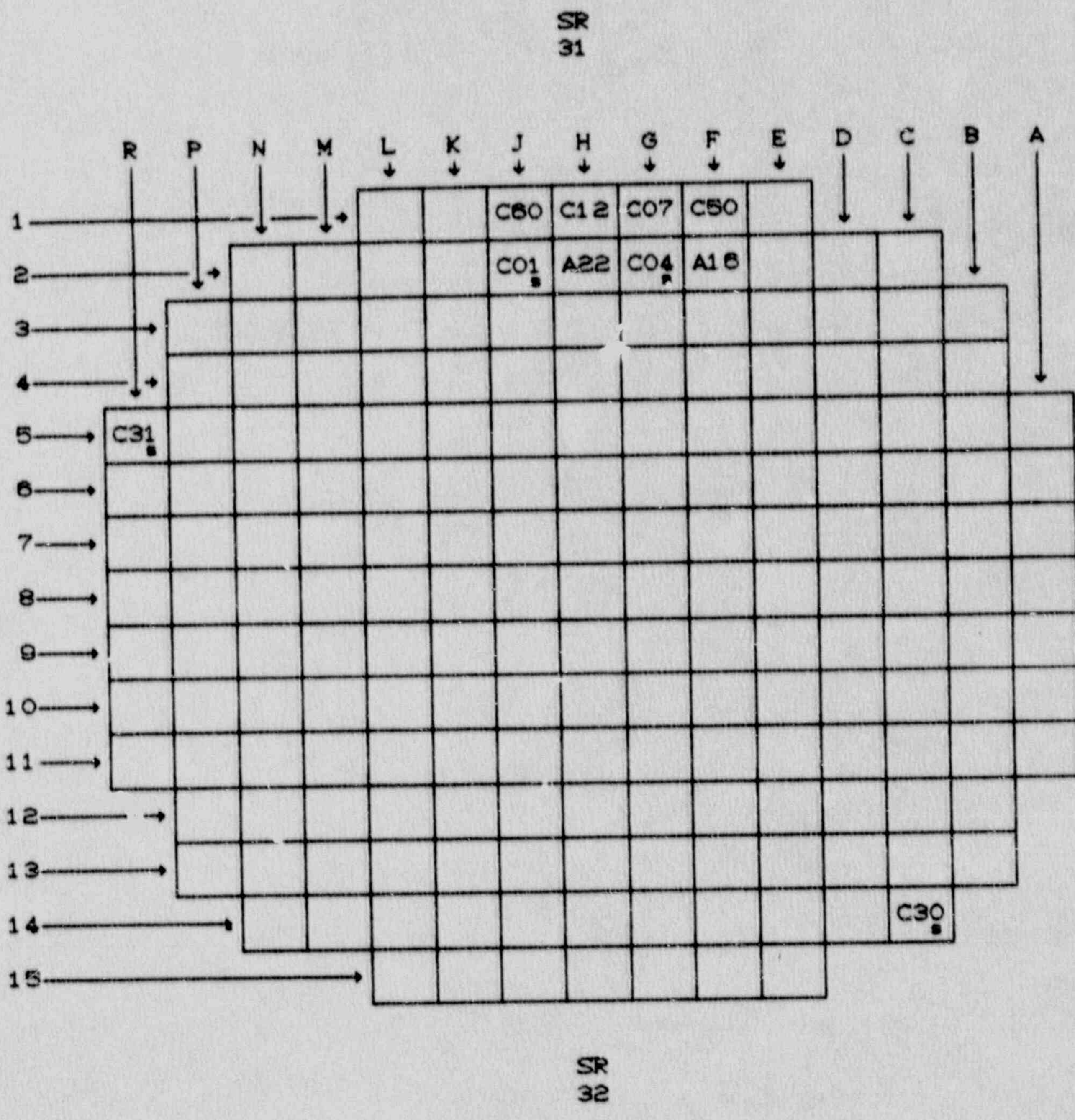
All 193 fuel assemblies were loaded in the core without incident. Fuel assembly A21, however, brushed against a new fuel storage vault lid when withdrawn from its storage rack location. The slight scratch on the fuel assembly bottom nozzle was inspected, blended, and evaluated as acceptable with vendor assistance. There was no damage to any of the fuel pins.

Refer to Figures 3.1.1-1 through 3.1.1-6 for a graphical description of how core loading progressed. Refer to Figures 3.1.1-7 and 3.1.1-8 for information on the locations of control rods and burnable poison assemblies.

Figure 3.1.1-1

UNIT 1 CORE LOADING PATTERN

INITIAL NUCLEUS OF ASSEMBLIES



A - REGION 1 (1.6 w/o)

B - REGION 2 (2.4 w/o)

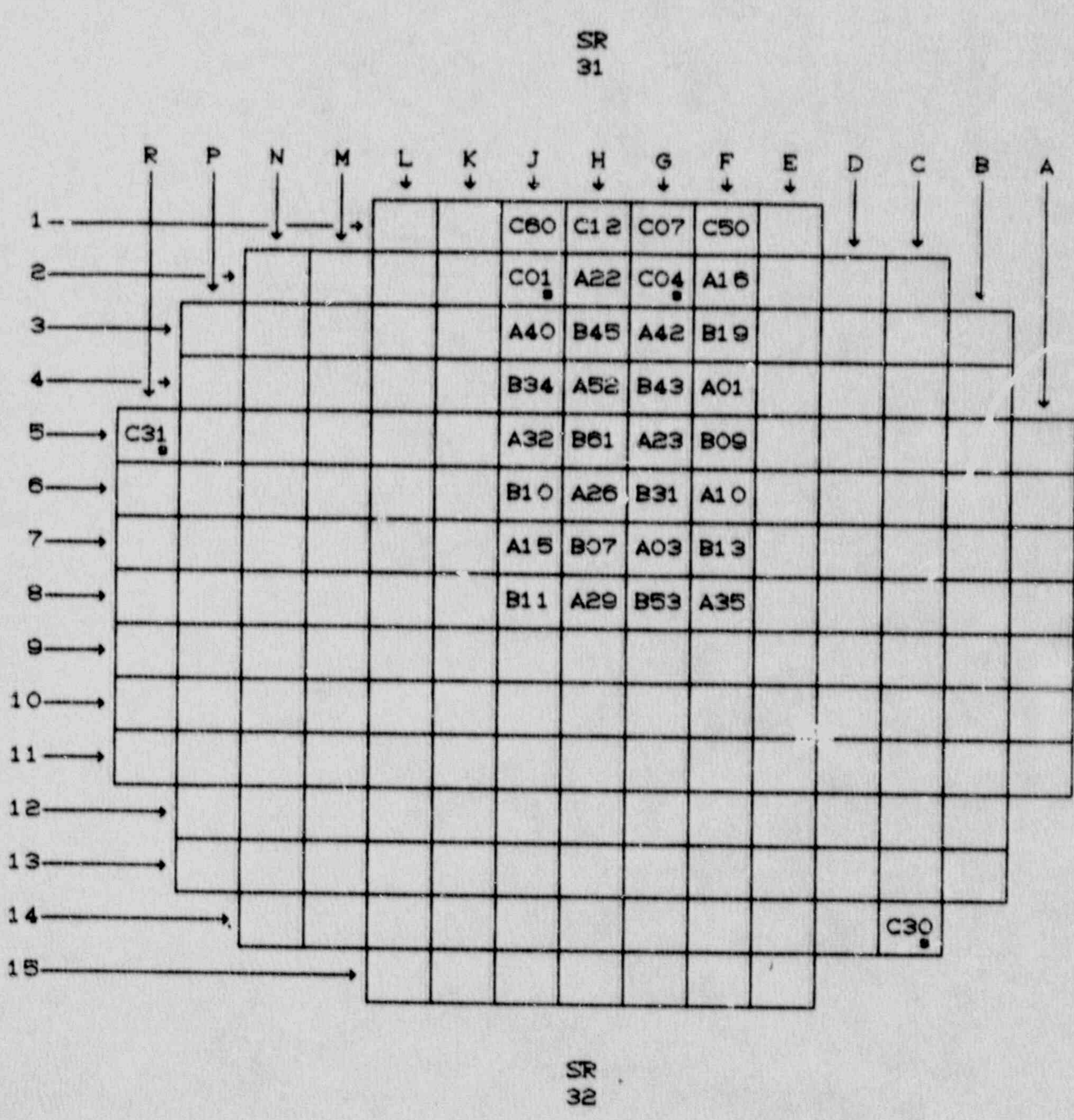
C - REGION 3 (3.1 w/o)

s - SOURCE ROD

Figure 3.1.1-2

UNIT 1 CORE LOADING PATTERN

PARTIAL BRIDGE ACROSS CORE



A - REGION 1 (1.6 w/o)

B - REGION 2 (2.4 w/o)

C - REGION 3 (3.1 w/o)

s - SOURCE ROD

Figure 3.1.1-3

UNIT 1 CORE LOADING PATTERN

COMPLETED BRIDGE ACROSS CORE

SR
31

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							C60	C12	C07	C50					
2							C01 s	A22	C04 s	A16					
3							A40	B45	A42	B19					
4							B34	A52	B43	A01					
5							A32	B61	A23	B09					
6							B10	A26	B31	A10					
7							A15	B07	A03	B13					
8							B11	A29	B53	A35					
9							A47	B17	A44	B02					
10							B[illegible]3	A11	B01	A55					
11							A27	B58	A20	B52					
12							B30	A12	B60	A07					
13							A57	B44	A45	B25					
14							C31 s	A34	C30 s	A19					
15							C52	C10	C37	C58					

SR
32

A - REGION 1 (1.6 w/o)

B - REGION 2 (2.4 w/o)

C - REGION 3 (3.1 w/o)

s - SOURCE ROD

Figure 3.1.1-4

UNIT 1 CORE LOADING PATTERN

PARTIAL COMPLETION OF CORE

SR
31

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							C60	C12	C07	C50	C26				
2							C01 s	A22	C04 s	A16	C28				
3							A40	B45	A42	B19	A33				
4							B34	A52	B43	A01	B49				
5							A32	B61	A23	B09	A14				
6							B10	A26	B31	A10	B63				
7							A15	B07	A03	B13	A63	B06	[illegible]61	C55	C40
8							B11	A29	B53	A35	B32	A21	B57	A24	
9							A47	B17	A44	B02	A56	B64	A38	C11	C24
10							B23	A11	B01	A55	B26	A04	B05	A39	C39
11							A27	B58	A20	B52	A59	340	A65	C64	C49
12							B30	A12	B60	A07	B21	B27	B55	C18	
13							A57	B44	A45	B25	A54	B08	C33	C46	
14							C31 s	A34	C30 s	A19	C62	C57	C48		
15							C52	C10	C37	C58	C29				

SR
32

A - REGION 1 (1.6 w/o)

B - REGION 2 (2.4 w/o)

C - REGION 3 (3.1 w/o)

s - SOURCE ROD

Figure 3.1.1-5

UNIT 1 CORE LOADING PATTERN

PARTIAL COMPLETION OF CORE

SR
31

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					C27	C43	C60	C12	C07	C50	C26				
2					C54	A38	C01 s	A22	C04 s	A16	C28	C02	C63		
3					A3[illegible]	B03	A40	B45	A42	B19	A33	B47	C15	C17	
4					B48	A43	B34	A52	B43	A01	B49	B59	B39	C05	
5					A51	B20	A32	B61	A23	B09	A14	B22	A28	C25	C45
6					B62	A41	B10	A26	B31	A10	B63	A46	B14	A62	[illegible]34
7	C41	C03	A13	B12	A53	B15	A15	B07	A03	B13	A63	B06	A61	C55	C40
8		A37	B18	A36	B38	A60	B11	A29	B53	A35	B32	A21	B57	A24	
9	C20	C61	A58	B50	A64	B33	A47	B17	A44	B02	A56	B64	A38	C11	C24
10					B35	A18	B23	A11	B01	A55	B26	A04	B05	A39	C39
11					A50	B42	A27	B58	A20	B52	A59	[illegible]	A65	C64	C49
12					B51	A31	B30	A12	B60	A07	B21	B27	B55	C18	
13					A08	B41	A57	B44	A45	B25	A54	B08	C33	C46	
14					C21	A09	C31 s	A34	C30 s	A19	C62	C57	C48		
15						C14	C52	C10	C37	C58	C29				

SR
32

A - REGION 1 (1.6 w/o)

B - REGION 2 (2.4 w/o)

C - REGION 3 (3.1 w/o)

s - SOURCE ROD

Figure 3.1.1-6

UNIT 1 CORE LOADING PATTERN

FINAL CONFIGURATION

SR
31

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					C27	C43	C60	C12	C07	C50	C26				
2			C36	C32	C54	A38	C01	A22	C04	A16	C28	C02	C63		
3		C56	C51	B46	A30	B03	A40	B45	A42	B19	A33	B47	C15	C17	
4		C47	B28	B24	B48	A43	B34	A52	B43	A01	B49	B59	B39	C05	
5	C06	C42	A49	B16	A51	B20	A32	B61	A23	B09	A14	B22	A28	C25	C45
6	C38	A06	B37	A05	B62	A41	B10	A26	B31	A10	B63	A46	B14	A62	C34
7	C41	C03	A13	B12	A53	B15	A15	B07	A03	B13	A63	B06	A61	C55	C40
8	C22	A37	B18	A36	B38	A60	B11	A29	B53	A35	B32	A21	B57	A24	C13
9	C20	C61	A58	B50	A64	B33	A47	B17	A44	B02	A56	B64	A38	C11	C24
10	C09	A02	B56	A17	B35	A18	B23	A11	B01	A55	B26	A04	B05	A39	C39
11	C53	C08	A25	B04	A50	B42	A27	B58	A20	B52	A59	B40	A65	C64	C49
12		C19	B54	B36	B51	A31	B30	A12	B60	A07	B21	B27	B55	C18	
13		C35	C59	B29	A08	B41	A57	B44	A45	B25	A54	B08	C33	C46	
14			C16	C23	C21	A09	C31	A34	C30	A19	C62	C57	C48		
15					C44	C14	C52	C10	C37	C58	C29				

SR
32

A - REGION 1 (1.6 $^w/_o$)

B - REGION 2 (2.4 $^w/_o$)

C - REGION 3 (3.1 $^w/_o$)

Figure 3.1.1-7

SHUTDOWN AND CONTROL ROD LOCATIONS

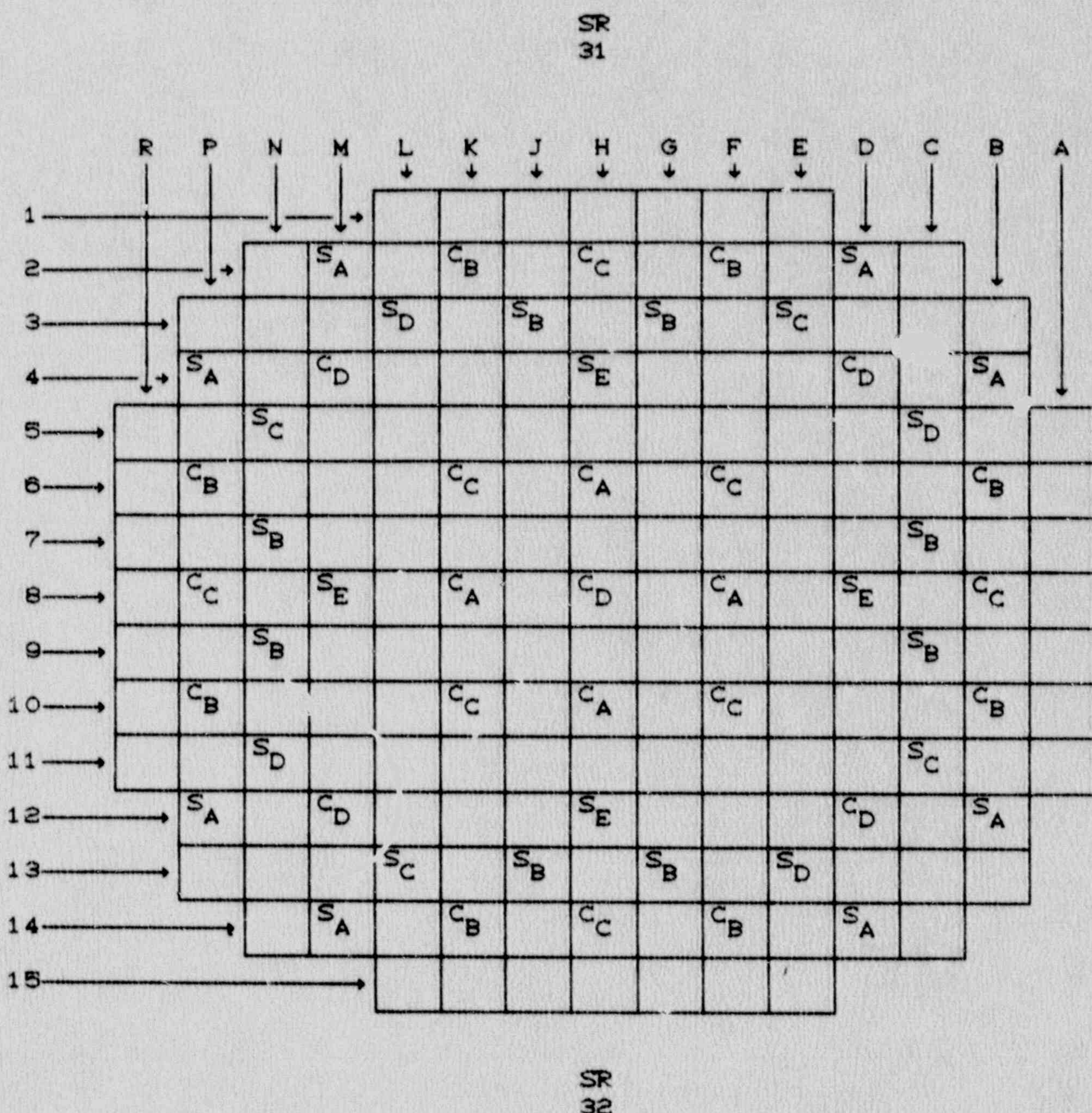


Figure 3.1.1-8

BURNABLE POISON ROD LOCATIONS

SR
31

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1						4		4		4					
2					9		P 23		P* 23		9				
3			3	24		4		8		4		24	3		
4			24		12	S 4	8		8		12		24		
5		9		12		8		8		8		12		9	
6	4		4		8		8		8		8		4		4
7		24		8		8		12		8		8		24	
8	4		8		8		12		12		8		8		4
9		24		8		8		12		8		8		24	
10	4		4		8		8		8		8		4		4
11		9		12		8		8		8		12		9	
12			24		12		8		8	S 4	12		24		
13			3	24		4		8		4		24	3		
14					9		P 23		P* 23		9				
15						4		4		4					

SR
32

944 BURNABLE POISON RODS - 12.5 w/o B_2O_3

NUMBER INDICATES THE NUMBER OF BURNABLE POISON RODS

S INDICATES A SECONDARY SOURCE ROD

P INDICATES A PRIMARY SOURCE ROD

P* INDICATES A DEPLETED PRIMARY SOURCE ROD

3.1.2 - CORE LOADING INSTRUMENTATION AND NEUTRON SOURCE CHECKS - ISU-003A

OBJECTIVE

The core loading instrumentation test is performed prior to core loading to determine the proper operating and discriminator voltage settings for the temporary core loading instrumentation and to verify that both the temporary and permanent nuclear monitoring instrument channels respond properly to a neutron source. The test is also performed to verify that both the temporary and permanent nuclear monitoring instrument channels respond properly to neutrons prior to resuming core loading following any eight hour or longer delay in loading. This test satisfies testing described by FSAR Table 14.2-3, Sheet 8 and Section 14.2.10.1.

TEST METHODOLOGY

Following the initial installation of the equipment, the temporary detectors are positioned near a neutron source. Using the neutron source, an optimum operating voltage is selected for each of the three detectors to ensure that minor fluctuations in detector power supply voltages would not adversely affect detector output. With the individual detector operating voltages selected, discriminator bias voltages are determined based on detector characteristic curves.

Prior to core loading, all five channels (two installed source range and three temporary core loading channels) are neutron response checked by moving a portable neutron source toward and away from each detector to verify detector response.

In the event of a delay in core loading of 8 hours or greater, this test reverifies proper detector neutron response by one of three methods. One method uses a portable neutron source moved toward and then away from a detector to verify detector response. The second method is to use movement of an installed fuel assembly to alter neutron flux at a detector by altering source to detector neutronic coupling. The third method uses an evaluation of counting statistics applied to detector output when in proximity to a fixed neutron source. Nuclear decay is a random process and if the detector output exhibits statistical behavior (standard deviation, etc.) characteristic of a random process, then the detector is judged to be responding to neutrons instead of 60 Hz or other noise.

3.1.2 - CORE LOADING INSTRUMENTATION AND NEUTRON SOURCE CHECKS - ISU-003A (Continued)

SUMMARY OF RESULTS

Upon completion of the procedure, operating voltages were determined to be 2000 volts for all three temporary nuclear instrumentation channels with discriminator bias voltages set at 3.5 volts for all three channels. Seven additional detector tubes were also tested for use as spares, as necessary. All seven spare tubes also had operating voltages of 2000 volts and 3.5 volt discriminator bias voltages. Only one spare tube was actually used. One detector tube exhibited erratic behavior prior to the start of core loading and was replaced with a spare.

Prior to core loading, all five channels (2 installed source range and 3 temporary core loading channels) were neutron response checked using a portable neutron source. The channel count rates were observed to increase by factors of between 30 and 50,000 when the source was placed nearby the various detectors.

The 8 hour delay portion of the test procedure was executed three times during the core loading activity. During the first performance, it was observed that as the neutron source approached each installed source range detector, the channel's count rate increased accordingly, indicating that the detector was responding to neutrons. Also during this same performance, the three temporary channels were verified using the statistical method. The final two performances used only the statistical method for all channels. No fuel assemblies were moved to verify detector neutron responses.

3.1.3 - INVERSE COUNT RATE RATIO MONITORING (Core Load Portion) NUC-111

OBJECTIVE

This permanent plant procedure is performed to obtain and evaluate nuclear monitoring data during core loading to ensure that core loading is done in a cautious and controlled manner. This procedure satisfies activities described in FSAR Section 14.2.10.1.

TEST METHODOLOGY

Neutron count rate data from both installed source range NIS channels and three temporary core load instrument channels is taken following each fuel assembly addition. The sources of the core neutron flux are the four installed Californium primary neutron sources with associated subcritical multiplication due to the loaded fuel lattice. As fuel is loaded, the core neutron flux changes due to changes in fuel lattice geometry and the addition of uranium to the core.

To determine the effect of a single fuel assembly addition on core reactivity, count rate data after each fuel assembly is loaded is compared to a reference value to evaluate the effect of the additional fuel assembly. This comparison is performed as a ratio of the count rates to evaluate the fractional change. If this ratio were to be very large, it would indicate that this fuel assembly addition brought the loaded fuel lattice significantly closer to criticality. For convenience, the procedure evaluates the inverse of the count rate ratios (ICRR) such that an approach to zero would indicate an approach to criticality. Additionally, this procedure trends the inverse count rate ratios and extrapolates the trends to evaluate whether or not additional fuel assembly loadings would be expected to result in an approach to criticality.

Prior to the start of core loading, background counts are taken to allow the elimination of general background radiation from the calculations of ICRR values. Reference count rate data is taken initially after the first ten fuel assemblies are loaded. Eight of these fuel assemblies are loaded together to constitute a "source nucleus", providing a subcritical multiplied flux capable of being used as a basis for meaningful comparisons. Reference values are redetermined if a neutron source bearing fuel assembly is moved or if a temporary detector is moved, both which cause a change in source to fuel to detector geometry. New reference values are also obtained if neutron counting channel equipment or electronics settings are changed, to ensure that a valid reference value for count rate comparison is used. As a conservative measure, new reference count rate values are determined if core loading is delayed by 8 hours or more to ensure that any count rate changes

TEST METHODOLOGY (Continued)

over time are accounted for. Inverse count rate ratio data taking, calculations, plotting, trend evaluation and extrapolation are also repeated hourly during any core loading delay for general core monitoring and to aid in detection of any inadvertent RCS dilutions.

At all times a minimum of two selected channels of instrumentation are designated as "responding channels". This designation is based on source to fuel to detector geometry considerations so as to avoid large local effects that may not be indicative of total core behavior.

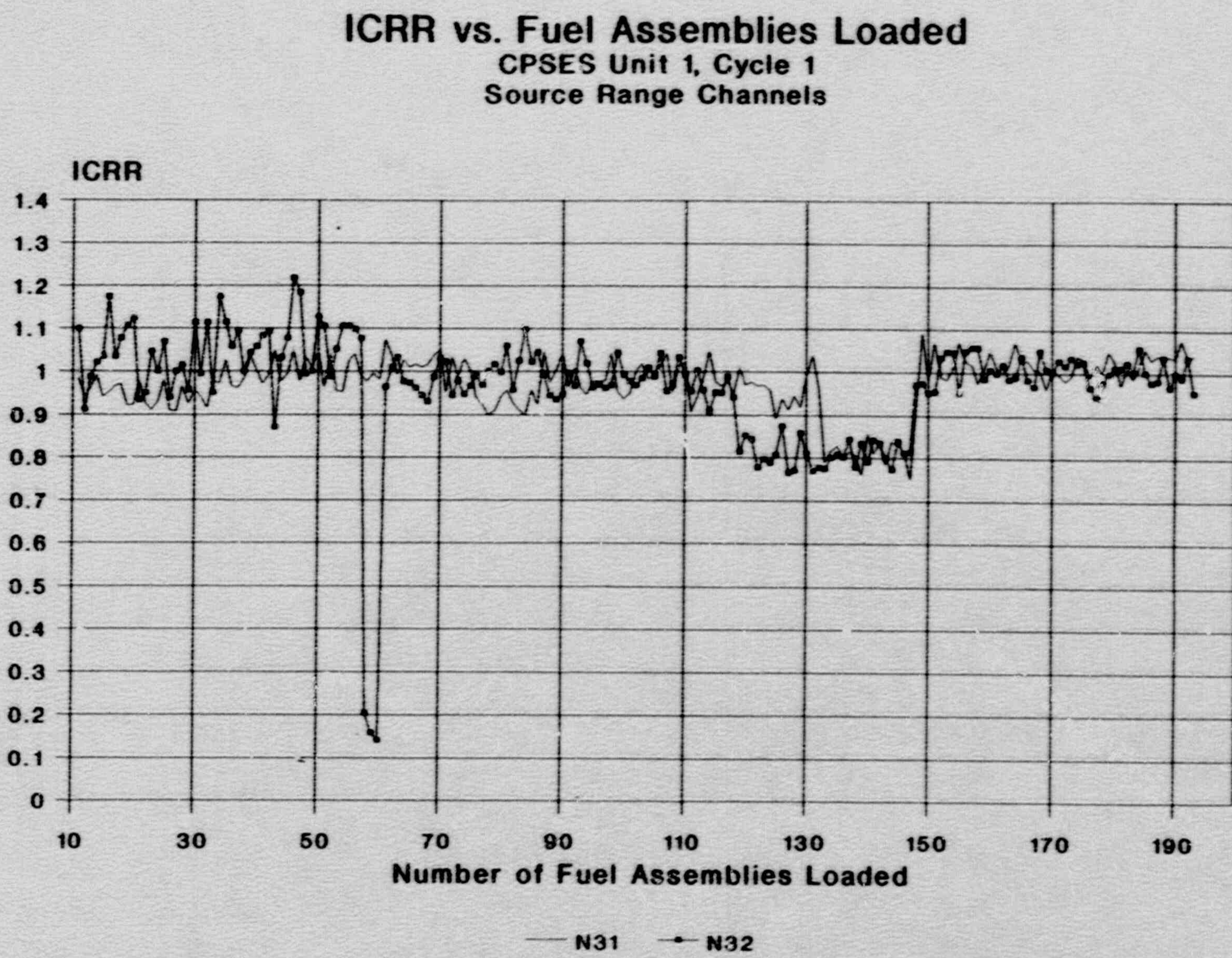
Final reference count rate data is taken following the completion of core loading for use as baseline data to help verify source range NIS signal to noise ratio.

SUMMARY OF RESULTS

All count rate data was properly recorded and ICRRs were calculated, plotted, trended and extrapolated. Refer to Figures 3.1.3-1 and 3.1.3-2 for a graphic display of procedure results during core loading. The inverse count rate ratio shows that core loading was performed in a cautious and controlled manner with no indicated unexpected approaches toward criticality. At no time did the extrapolated data from a responding channel indicate that criticality would be expected to occur with the loading of the next fuel assembly. Large changes in the inverse count rate ratios from one core loading step to the next step are due primarily to local geometric effects when a neutron source was moved near a detector or when fuel was loaded between a source and a detector resulting in a large local countrate increase due to enhanced neutronic source to detector coupling via subcritical multiplication. These were local effects observed on only one or two channels at a time and is the reason five monitoring channels are used.

Monitoring data was properly taken and evaluated during core loading delays and reference count rates were properly recalculated. The background count rates were sufficiently low so as to be nearly negligible, also an indication of low neutron detector channel noise. Final reference count data was taken for both source range channels.

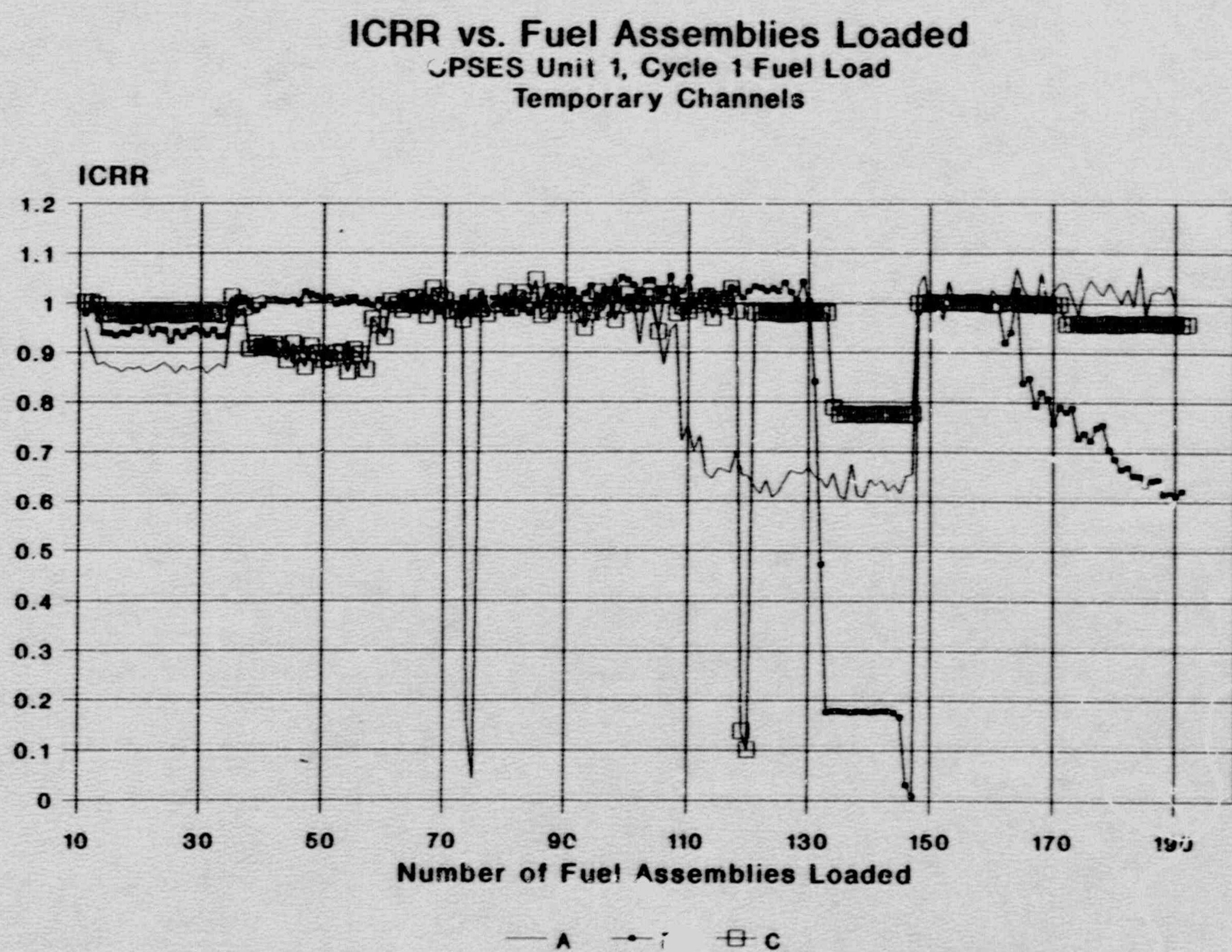
Figure 3.1.3-1



ICRR vs. Fuel Assemblies Loaded
CPSES Unit 1, Cycle 1
Source Range Channels
ICRR
1.4
1.3
1.2
1.1
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
10
30
50
70
90
110
130
150
170
190
Number of Fuel Assemblies Loaded
N31
N32

Figure 3.1.3-2

ICRR vs. Fuel Assemblies Loaded



3.1.4 - RCS AND SECONDARY COOLANT CHEMISTRY (Core Load Portion) - ISU-006A

OBJECTIVE

This test is performed to verify correct and uniform boron concentrations in portions of the reactor coolant system (RCS) and the directly connected portions of fluid systems as required for core loading. This test is also designed to help ensure that the possibility of an inadvertent dilution of the RCS during core loading is minimized. This test satisfies activities described in FSAR Section 14.2.10.1.

TEST METHODOLOGY

Prior to the commencement of core loading, the RCS is sampled and verified to meet specified water chemistry criteria. As a prerequisite to RCS chemistry sampling, the borated water source, the RCS loops, Chemical and Volume Control System piping, Safety Injection System piping, and Containment Spray System piping were verified to have boron concentrations which would preclude inadvertent RCS dilutions.

Each of the RCS crossover legs, the Residual Heat Removal (RHR) system, the reactor vessel, the Volume Control Tank, the Safety Injection System accumulators, the boric acid tanks, and the Refueling Water Storage Tank are sampled, and that water is verified to contain specified boron concentrations.

Following the initial verification of the chemistry in the reactor coolant system, four samples are taken from the reactor vessel at equidistant depths along with a sample from the operating residual heat removal train. These samples are then analyzed for boron to verify a uniform boron concentration between the RCS and the RHR system (within a 30 ppm range). After the RCS and RHR is verified to be at a uniform concentration, the operating residual heat removal train is sampled and analyzed for boron to verify that the water remains at $\geq$ 2000 ppm boron. Sampling continues every 12 hours until the start of core loading. With the start of core loading sampling continues on the operating RHR train every four hours throughout the core loading process. The 12 hour and 4 hour samples also include measurement of RHR inlet temperature for use in monitoring reactor coolant system temperature for compliance with Technical Specifications and to ensure temperature changes do not adversely influence inverse count rate ratio monitoring. The spent fuel pool is dry during initial core load and the fuel transfer system canal portions are not required to be borated.

3.1.4 - RCS AND SECONDARY COOLANT CHEMISTRY (Core Load Portion) - ISU-006A (Continued)

TEST METHODOLOGY - (Continued)

The criterion for the 4 hour samples is to ensure a minimum of 2000 ppm for shutdown margin and a maximum of 2150 ppm to not overly attenuate the neutron detector signals during core loading. Also, consecutive samples are not to differ by more than 20 ppm as a way of detecting any inadvertent dilution.

SUMMARY OF RESULTS

During the execution of this test which started before and lasted throughout the core loading process, all acceptance criteria were met for each system that was sampled. No corrective actions in the core loading process were needed to meet the acceptance criteria of this test. Detailed results obtained prior to core load are tabulated below:

Location	Specified Range (ppm)	Actual Value (ppm)
Volume Control Tank	2000 - 2150	2040
RHR Train A	2000 - 2150	2036
RHR Train B	2000 - 2150	2020
Refueling Water Storage Tank	2000 - 2200	2028
Boric Acid Tank #1	≥ 7000	7376
Boric Acid Tank #2	≥ 7000	7010
Safety Injection Accumulator 1	1900 - 2100	2053
Safety Injection Accumulator 2	1900 - 2100	2064
Safety Injection Accumulator 3	1900 - 2100	2051
Safety Injection Accumulator 4	1900 - 2100	2057
RCS Loop 1 Crossover Leg	> 2000	2038
RCS Loop 2 Crossover Leg	> 2000	2038
RCS Loop 3 Crossover Leg	> 2000	2033
RCS Loop 4 Crossover Leg	> 2000	2039
Reactor Vessel Surface	Within a	2041
Reactor Vessel 1/3 down	30 ppm	2041
Reactor Vessel 2/3 down	range	2045
Reactor Vessel Bottom		2045
RHR Train A		2048

RCS/RHR uniformity values were within a 7 ppm range, well within the 30 ppm limit.

The RHR samples prior to and during core loading varied within a range of 2037 ppm to 2057 ppm, well within the 2000 - 2150 ppm range limits. No two consecutive samples deviated by more than 10 ppm. This satisfied the <20 ppm difference limit. All samples were from RHR Train A. RHR inlet temperature was very steady, increasing from 110°F before core loading to 115°F at the end of core loading, more than 100 hours later.

3.1.5 - VERIFICATION OF CORE LOADING PATTERN - RFO-204

OBJECTIVE

This permanent plant procedure is performed to confirm that the loaded core matches the design loading pattern and to provide a videotape record of the as-loaded core.

TEST METHODOLOGY

Using the manipulator crane (refueling machine) television camera mast, an underwater TV camera is slowly traversed over the entire loaded core allowing fuel assembly and insert numbers, positions and orientations to be observed on a TV monitor. This information is recorded and compared against the core loading pattern design information from the fuel vendor. The TV signal is also sent to a video recorder so a tape record of the as-loaded core pattern is made. The use of the camera within the reactor vessel constitutes a core alteration, so all required Mode 6 core alteration related Technical Specifications are also verified by this procedure to have been satisfied.

SUMMARY OF RESULTS

The entire core was mapped using the underwater TV camera. All fuel assemblies and inserts were found to be in their proper locations and orientations. A videotape record was made and reviewed to ensure that it was legible.

3.2 SYSTEM TESTING AFTER CORE LOAD AND AT VARIOUS POWER LEVELS

3.2.1 - PIPING VIBRATION MONITORING - ISU-212A

OBJECTIVE

This test demonstrates that steady state flow induced piping vibrations and transient response piping vibrations are within allowable design limits. The scope of the test is limited to portions of the Main Steam and Feedwater systems for transient response and the Main Steam, Feedwater and Condensate systems for steady state. These are systems which could not be fully tested during the Preoperational Test Program due to plant conditions. This test partially satisfies the testing described by FSAR Table 14.2-2, Sheet 57 and Sections 3.9B.2.1.2 through 3.9B.2.1.4.

TEST METHODOLOGY

The Main Steam, Feedwater and Condensate systems are operated under normal, steady state conditions during which visual inspections of the piping are conducted. The walkdowns divide the systems into smaller piping subsystems between restraints in order to use the simple beam analogy to determine deflection limits. Portable vibration analyzers are also used to obtain numerical values for selected vibration levels and comparisons are made between the vibration velocities or displacements and the appropriate limits. Selected locations were also instrumented for remote vibration monitoring for safety, accessibility and ALARA considerations. Based on the outcome, vibration levels less than the allowable limit would satisfy the Acceptance Criteria. The steady state testing of various subsystems is performed between 3-6% power, at 15% power and at 100% power.

The transient response portion of the test also combines data taken from selected remotely instrumented portions of the Main Steam and Feedwater systems during the imposed transient with concurrent visual observations of accessible piping system portions. Portions of the Main Steam system are tested in response to a full power main turbine trip with portions of the Feedwater system tested in response to Main Feedwater Pump trip.

SUMMARY OF RESULTS

The steady state portion of the test generated only three calculations where the levels of vibration exceeded the limits specified in the test. These three items were all on Feedwater system recirculation piping to the condenser. All three items were evaluated and dispositioned by Engineering as acceptable based on the small magnitudes of the actual displacements and because the lines are not in continuous service. These miniflow lines have flow through them only intermittently.

SUMMARY OF RESULTS (Continued)

Location	Pipe Line Number	Allowed Velocity (inches/sec)	Measured Velocity (inches/sec)
1A Miniflow	12FW-1-21-2002G	≤0.5	2.3
1A Miniflow	12FW-1-26-2002G	≤0.5	0.95
1B Miniflow (2" Drain Line)	2FW-1-2[illegible]-2002G	≤0.5	18

Location	Allowed Displacement Ratio	Actual Displacement Ratio	Actual Displacement (inches)
1A Miniflow	≤1.0	2.48	0.10
1A Miniflow	≤1.0	1.46	0.04
1B Miniflow (2" Drain Line)	≤1.0	1.13	0.28

For the transient response portions of the test, there were no discrepancies noted with regard to the Main Feedwater Pump trip transient. There were two items noted in connection with the Full Power Turbine Trip transient. One instrumented snubber, MS-1-002-009-C72K (location TR-1-MS-25), exceeded its allowed loading criterion. Engineering evaluated and dispositioned this as acceptable because while the expected loading of 9253 lbs. was exceeded by 16% (10720 lbs), the support had available design margin. Even though the actual transient loading exceeded the expected loading by 16%, when the transient loading is combined with predicted seismic loading the total load change is only an increase of 317 lbs, 0.6%. This is well within the 15% of total load snubber design margin available. A separate calculation indicated that the piping stresses in this area were ≤16750psi which lies well within the ≤21000psi allowable range. Additionally, two remote sensors, at locations TR-1-MS-02 and TR-1-MS-03, failed to function during the transient. Engineering evaluated and dispositioned this missing data as acceptable based on the data obtained from 17 other, functioning main steam system sensors.

Other than the above noted items, the remaining piping system portions all had vibration levels within the acceptable range.

3.2.2 - STEAM GENERATOR LEVEL CONTROL TEST - ISU-207A

OBJECTIVE

This test is performed to demonstrate steam generator level control stability throughout power ascension. Changing feedwater flow configurations and major power changes necessitate the need for multiple performances of this test. Level control stability of the four steam generators is demonstrated while operating on the feedwater bypass control valves and the main feedwater control valves.

TEST METHODOLOGY

In order to verify level control stability while operating on the bypass or main feedwater control valves, a 5% level deviation is manually established in each steam generator. The control system is then transferred to the automatic control position. Steam generators are tested sequentially, one at a time, not simultaneously. The actual steam generator level is monitored to determine overshoot, undershoot and whether or not level returns to and remains within the allowed band of 66.5% $\pm$2% of narrow range level within a specified time frame of 3 times the appropriate reset time constant. The bypass valves are tested at approximately 5% power. The main feedwater control valves are tested at approximately 50% power.

In order to verify level control stability while transferring between the feedwater bypass control valves and the main feedwater control valves, steam generator levels are monitored while performing this transfer at approximately 20% power.

At approximately 50%, 75% and 100% power, data is taken to verify expected main feedwater control valve positions, to verify proper feedwater pump speed control operation on its sliding p program and to verify non-excessive feedwater header pressure oscillations. At 75% and 100% power, data is also taken to verify proper steady state level control operation.

SUMMARY OF RESULTS

Refer to Table 3.2.2-1 for detailed test results.

When given a 5% level deviation (high or low), the bypass control valves returned steam generator level to and remained within the programmed level, $\pm$2%, within 36 minutes (30 minutes for steam generator #2) with less than 4% overshoot or undershoot, as expected. This was done at approximately 5% power.

3.2.2 - STEAM GENERATOR LEVEL CONTROL TEST - ISU-207A (Continued)

SUMMARY OF RESULTS (Continued)

When given a 5% level deviation (high or low), the main feedwater control valves returned the steam generator level to and remained within the programmed level ±2% within 83.5 minutes with less than 4% overshoots or undershoots as expected. This was done at approximately 48% power.

After transferring from the feedwater bypass control valves to the main control valves, steam generator level deviations were to return to and remain within ±2.0% of the programmed level within 10 minutes. This was not satisfied initially. A misinstalled jumper on a circuit board for the Steam Generator #2 level controls was corrected and all steam generators were retested satisfactorily at approximately 20% power.

The feedwater header pressure oscillations were less than 3% of operating pressure range at approximately 50%, 75% and 100% power.

At approximately 75% and 100% power, all steam generator steady state levels were verified to remain within the expected 66.5% ±2% operating band.

At approximately 50%, 75% and 100% power the sliding Δp program value, used to control main feedwater pump speed, was verified to be within ±25psig of the actual Δp value

At approximately 50% power, all feedwater control valves were verified to be within ±10% of their predicted positions. However, at approximately 75% and 100% power, only 3 of the 4 valves satisfied the ±10% band limit. Feedwater Control Valve 1-FCV-510 indicated more than 10% below the predicted position. Valve operator clearances were changed and the valve was verified to be open the proper amount.

Two other plant problems were noted and corrected as a result of performance of this test during power ascension. The sliding Δp program value used for controlling main feedwater pump speed was adjusted during power ascension to reduce the Δp across the feedwater control valves to optimize performance of the feedwater system. Also, high frequency valve motion of Feedwater Control Valve 1-FCV-540, approximately 1/4 inch in displacement, was identified. An instrumentation scaling change minimized the oscillations and further design modifications are to be implemented during a subsequent outage.

TABLE 3.2.2-1

STEAM GENERATOR LEVEL CONTROL SUMMARY

BYPASS CONTROL VALVE LEVEL CONTROL RESPONSE PERFORMED AT APPROXIMATELY 5% POWER

STEAM GENERATOR	LEVEL DEVIATION	ACCEPTANCE CRITERION IN MINUTES	ACTUAL TIME RESPONSE IN MINUTES	OVERSHOOT/ UNDERSHOOT LIMIT IN PERCENT	MAXIMUM OVERSHOOT PERCENT	MAXIMUM UNDERSHOOT PERCENT
1	5% up	≤36	23.3	<4.0	1.5	0
	5% down	≤36	32.6	<4.0	0	2.0
2	5% up	≤30	26.8	<4.0	2.0	0
	5% down	≤30	26.6	<4.0	0	2.0
3	5% up	≤36	17.0	<4.0	2.0	0
	5% down	≤36	26.5	<4.0	1.0	1.0
4	5% up	≤36	15.5	<4.0	1.5	0
	5% down	≤36	22.5	<4.0	0	2.0

LEVEL CONTROL RESPONSE AFTER BYPASS TO MAIN FEEDWATER CONTROL VALVE TRANSFER AT APPROXIMATELY 20% POWER

STEAM GENERATOR	ACCEPTANCE CRITERION IN MINUTES	ACTUAL TIME RESPONSE IN MINUTES
1	<10	0
2	<10	4.4
3	<10	0
4	<10	0

FW HEADER PRESSURE OSCILLATIONS

Power Plateau	Maximum Pressure Oscillation (psig/%)
Allowed Limit	<45/3.0
50%/Pump A	3/.2
50%/Pump B	13/.9
75%	12/.8
100%	13/.9

TABLE 3.2.2-1 (CONTINUED)

FEEDWATER ΔP PROGRAM COMPARISON

Power Plateau	Maximum ΔP Deviation (psi)
Allowed Limit	<25
50%/Pump A	15.0
50%/Pump B	22.9
75%	5.9
100%	0.8

MAIN FEEDWATER CONTROL VALVE LEVEL CONTROL RESPONSE AT APPROXIMATELY 48% POWER

STEAM GENERATOR	LEVEL DEVI-ATION	ACCEPTANCE CRITERION IN MINUTES	ACTUAL TIME RESPONSE IN MINUTES	OVERSHOOT/ UNDERSHOOT LIMIT IN PERCENT	MAX-IMUM OVER-SHOOT PERCENT	MAX-IMUM UNDER-SHOOT PERCENT
1	5% up	≤83.5	15.6	<4.0	0	0
	5% down	≤83.5	33.8	<4.0	0	2.5
2	5% up	≤83.5	14.9	<4.0	0.5	0
	5% down	≤83.5	12.8	<4.0	0	0.5
3	5% up	≤83.5	18.1	<4.0	1	0
	5% down	≤83.5	17.3	<4.0	0	1.5
4	5% up	≤83.5	4.7	<4.0	0.5	0
	5% down	≤83.5	2.3	<4.0	0	0

TABLE 3.2.2-1 (CONTINUED)

FEEDWATER CONTROL VALVE POSITIONS AT VARIOUS POWER LEVELS ACTUAL VALVE POSITIONS IN %

Power Level	1-FCV-510 /	Predicted Range in %	1-FCV-520 /	Predicted Range in %
48/Pump A	50	39-59	56.3	39-59
48/Pump B	50	37-57	56.3	39-59
73	51	54-74	66.0	54-74
100	66.7	68-88	87.5	70-90

Power Level	1-FCV-530 /	Predicted Range in %	1-FCV-540 /	Predicted Range in %
48/Pump A	50	40-60	54.2	39-59
48	50	39-59	54.2	38-58
73	66.0	54-74	66.0	54-74
100	87.5	68-88	83.3	70-90

Following rework of 1-FCV-510:

Power Level	Actual Position	Predicted Range (%)
75%	68.75	53-73
100%	75	68-88

3.2.3 - THERMAL EXPANSION - POWER ASCENSION PHASE - ISU-308A

OBJECTIVE

Thermal expansion testing of plant systems is conducted to verify that components and piping can expand without restriction of movement upon system heatup. It is also conducted to confirm the correct functioning of component supports, piping supports and restraints. This test covered portions of the plant that could not be tested during the Preoperational Test Program due to plant conditions. This test satisfies activities described in FSAR Table 14.2-2, Sheets 52 and 52a and in FSAR Sections 3.9B.2.1.1 and 3.9B.2.1.4.

TEST METHODOLOGY

At ambient and hot conditions, system walkdowns are performed. Both the NSSS and selected secondary plant systems are evaluated. Piping and components are visually examined and specific snubber positions recorded. Pipe whip restraints are verified not to interfere with the piping and variable (spring) hanger movements are recorded. Interferences are identified and dispositioned by the design engineers. When necessary, system walkdowns are again conducted following the resolution of interferences. All piping movements are evaluated by the design engineers. Selected locations are remotely instrumented to measure piping movements for ALARA, safety and accessibility reasons. The walkdowns and remote data collection are performed at NSSS temperatures of approximately 70°F, 350°F and 557°F and at approximately 30%, 50%, 75% and 100% power.

SUMMARY OF RESULTS

The piping and components were not to be constrained from expanding and actual thermal expansion movements could not vary from predicted thermal movements by more than 25% or ± 1/4 inch, whichever was greater, or reconciled by Engineering. Also, spring hanger movements were to remain within their working range and snubbers were not to become fully extended or retracted. During the course of system walkdowns, several minor interferences were observed and determined to be acceptable-as-is, or specific corrective actions were recommended. All recommended corrective actions were initiated. Some portions of the piping systems were again examined and measured following the removal of interferences. Movement of components not within the ±25% or ± 1/4 inch criterion were evaluated by the design engineers on a case-by-case basis. All thermal expansion movements were determined to be acceptable for continued plant operation. Remote movement data was also collected during plant transient testing.

SUMMARY OF RESULTS (Continued)

During the NSSS temperature increase between 70°F and 557°F three pipe whip restraints were evaluated and removed and three others were adjusted or modified to allow free thermal movement of the pipe. These preliminary results were also evaluated by Engineering and used as the basis for revision of selected predicted pipe movements prior to power ascension.

In Mode 2, eight Extraction Steam (EX) system drip pot drain lines were discovered to have crushed insulation. The insulation was removed and the system was refloated. In Mode 1, one EX drip pot drain line was found in contact with the floor. A small amount of floor concrete was chipped out to provide clearance for this line.

At 30% power, the following types of items were noted and evaluated by Engineering:

- o Thermal expansion movements in excess of ±1/4 inch or ±25%
- o Contact between pipe insulation and feedwater pipe whip restraints
- o Heater Drain system piping in contact with building structural steel
- o One bent strut on a Steam Generator Blowdown system line

At 50% power, the following types of items were noted and evaluated by Engineering:

- o Thermal expansion movements in excess of ±1/4 inch or ±25%
- o Contact between pipe insulation and feedwater pipe whip restraints
- o Heater Drain system piping in contact with building structural steel
- o Snubber angularity discrepancies
- o Two EX drip pot drain lines in contact with the floor
- o Higher than expected temperature detected on a feedwater line upstream of a check valve.

At 75% power, the following types of items were noted and evaluated by Engineering:

- o Thermal expansion movements in excess of ±1/4 inch or ±25%
- o Contact between pipe insulation and feedwater pipe whip restraints
- o EX drip pot drain lines in contact with the floor
- o One bent strut on the Heater Drain system
- o Same bent strut on the Steam Generator Blowdown system

3.2.3 - THERMAL EXPANSION - POWER ASCENSION PHASE - ISU-308A (Continued)

SUMMARY OF RESULTS (Continued)

At 100% power, the following types of items were noted and evaluated by Engineering:

- o Thermal expansion movements in excess of ±1/4 inch or ±25%
- o Contact between pipe insulation and feedwater pipe whip restraints
- o Same bent strut on the Heater Drain system
- o Same bent strut on the Steam Generator Blowdown system
- o Heater Drain system piping contacts
- o Heater Drain system support base plates pulled away from columns
- o Main Feedwater system piping in contact with support steel

All of the above items were evaluated as acceptable by Engineering or have had corrective actions initiated via design modifications or Work Orders.

3.2.4 - INCORE MOVEABLE DETECTOR SYSTEM ALIGNMENT - ISU-016A

OBJECTIVE

The purpose of this procedure is to demonstrate the proper operation of the flux mapping system, including the leak detection system. In addition, top and bottom of core limits are set and the actual drive cables and detectors are installed and verified to function properly. This test satisfies activities described by FSAR Table 14.2-3, Sheets 34 and 35.

TEST METHODOLOGY

Using a dummy drive cable, the top and bottom of core limits are established for normal, emergency, calibrate and common modes and for the storage mode endpoint and insert limits by slowly driving the dummy detector to the top of the core (or storage position) where clutch slippage is observed. The position was then recorded from the encoder display. The top limit is obtained by subtracting two inches from the recorded position and the bottom limit is obtained by subtracting 170 inches from the top limit. Storage mode insert limit is the endpoint minus 36 inches. Drive speed is measured by timing cable motion over a given distance to verify the design speed of 144 $\pm$2 inches/minute. The leak detection system is tested by filling the drain header with demineralized water and allowing the leak detection level switch to actuate, thereby draining the water and alarming. The CO_2 purge system is verified to function properly to inject CO_2 at a rate of less than 10 ft^3/hr following detector withdrawal. The withdraw and safety limit switches are verified to prevent the detector from being taken up onto the reel. All push-to-test lights are verified. Simulated signal transmissions to the process computer and from the incore system are made to verify proper computer data logging from the incore system.

SUMMARY OF RESULTS

Figure 3.2.4-1 displays the Moveable Incore Detector Path Locations. Proper operation of all indicating lights were verified along with the proper operation of the leak detection system and alarm as described in the previous section. One position indicating lamp failed to illuminate initially. The wire to the lamp was repaired and lamp operation successfully retested. The CO_2 purge operated properly at an 8 ft^3/hr flowrate. The dummy detector was successfully inserted into all 58 core locations with proper drive speeds verified. All top and bottom limits were properly established. The limit switches were demonstrated operable. The simulated data transmissions verified the ability of the process computer to receive signals from the incore flux mapping system and the ability of the incore system to supply proper signals to the computer. As a final step, the actual

3.2.4 - INCORE MOVEABLE DETECTOR SYSTEM ALIGNMENT - ISU-016A (Continued)

SUMMARY OF RESULTS (Continued)

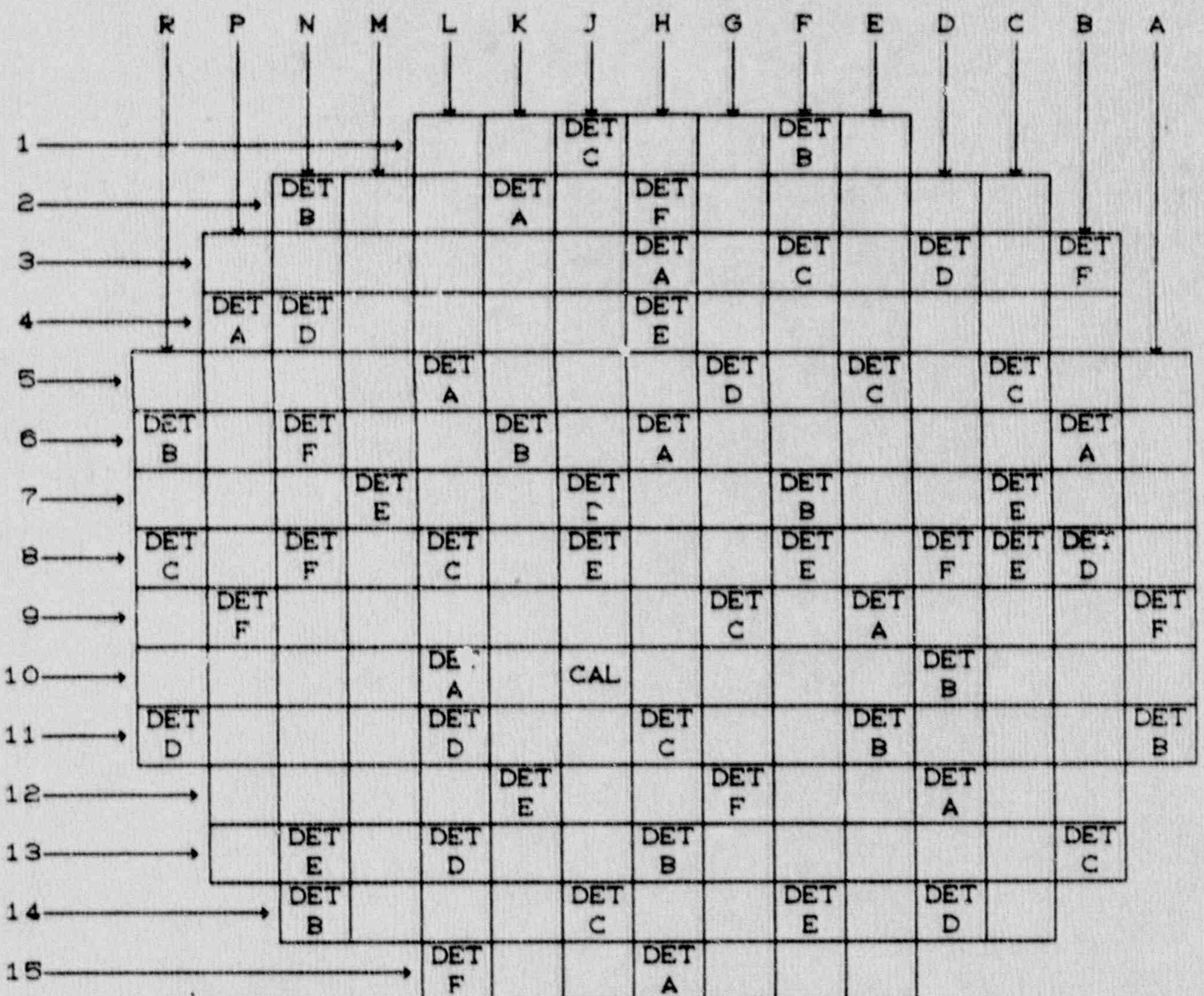
detector cables were installed on the drive units and a demonstration full core flux map was taken, even though no usable neutron flux had yet existed in the core. The detector cable could not access core location B-13, even though the dummy cable had successfully been driven through this core location in the first portion of the test. With this path blocked, the system still satisfied the Technical Requirements Manual minimum number of thimbles limit of 44. The path was accessible during the first portion of the test, but the detector apparently hung up at the seal table fitting when attempting the demonstration flux map using real detectors and detector cables. Repairs are planned for a subsequent outage.

Detector Drive Speeds

Drive	Distance (inches)	Time (seconds)	Actual Speed (inches/min)	Allowed Range
A	200.5	83.54	144.0	142-146
B	200.9	83.52	144.3	142-146
C	200.5	83.49	144.1	142-146
D	200.5	83.48	144.1	142-146
E	200.5	83.49	144.1	142-146
F	200.6	83.49	144.2	142-146

Figure 3.2.4-1

MOVABLE INCORE DETECTOR PATH LOCATIONS



3.2.5 - RCS AND SECONDARY COOLANT CHEMISTRY (Post Core Load) - ISU-006A

OBJECTIVE

This test is performed to verify that the water quality within the reactor coolant system and the steam generators meets the appropriate chemistry requirements. The test is performed at Cold Shutdown (Mode 5), Heat-up Prior to Criticality (Mode 3), at Criticality (Mode 2), and at approximately 30%, 50%, 75%, and 100% Power. This test satisfies activities described by FSAR Table 14.2-3, sheet 11.

TEST METHODOLOGY

The testing is performed by obtaining samples of the reactor coolant system and steam generators from the appropriate sample panels. Each sample is then chemically analyzed. The results of these analyses are tabulated and compared to the chemistry requirements.

SUMMARY OF RESULTS

During the executions of this test, all required Acceptance Criteria were adequately met for each system that was sampled. No corrective actions in plant operation were needed to meet the Acceptance Criteria. On occasions, one of the samples had to be reanalyzed or retaken because a result was not consistent with the others. Upon reanalysis the sample was shown to be within specifications.

Tables 3.2.5-1 and 3.2.5-2 contain a summary of the results for each system sampled along with the Acceptance Criteria or guidelines stated within the test.

While not required by the test, Pressurizer samples were also evaluated from Mode 2 through 100% power and were found to be satisfactory when compared to the RCS criteria. They are not tabulated because no limits are specified by the test.

TABLE 3.2.5-1

RCS CHEMISTRY SUMMARY

CHEMISTRY PARAMETER	CRITERION	MODE 5	MODE 3	MODE 2	29%	48%	76%	100%
Chloride	<150 ppb	3	7	3	6	1	<1	4
Fluoride	<150 ppb	2	5	<1	4	3	2	6
Dissolved Oxygen**	<100 ppb	N/A	1	2	<5	<1	<1	3
Lithium	***	N/A	N/A	N/A	2.1	1.9	2.0	2.0
Hydrogen ****	25-50 cc/kg H_2O	N/A	N/A	N/A	26.4	27.2	32.9	28
Boron*	≥2000 ppm	2063	N/A	N/A	N/A	N/A	N/A	N/A
Gross Activity	<100/$\bar{E}$ Ci/ml	N/A	<limit of detection	4.6 E-5	3.46 E-2	1.07 E-1	1.51 E-1	1.92 E-1
Dose Equivalent I-131	<1.0 Ci/ml	N/A	5.3 E-8	<5.9 E-7	3.33 E-4	1.13 E-3	1.22 E-3	1.80 E-3

* Mode 5 test sampled RHR instead of the RCS, due to system pressure, as allowed by the test procedure.

** When Tave >250°F

*** In accordance with Lithium vs. Boron Curve above 1MW thermal (see Figure 3.2.5-1). RCS boron concentration was between 400ppm and 1200 ppm for all at-power test conditions (>1 MW thermal)

**** When RCS >1MW thermal Reactor Power

N/A = Not applicable as no criterion is specified for this plant condition

TABLE 3.2.5-2

STEAM GENERATOR CHEMISTRY SUMMARY

CHEMISTRY PARAMETER	CRITERION	MODE 5	MODE 3	MODE 2	29%	48%	76%	100%
Cation Conductivity*	≤0.8 mho/cm	N/A	1.5	0.37	0.56	0.63	0.80	0.71
pH**	≥8.8	10.0	9.2	9.0	9.2	9.2	9.0	8.9
Sodium***	≤20ppb	14	51	<1	14	17	12	6
Chloride***	≤20ppb	6	25	2	7	11	8	5
Sulfate***	≤20ppb	14	29	<2	9	7	18	3
Silica	≤300ppb	N/A	N/A	N/A	250	220	270	160
Hydrazine	≥75ppm	80	N/A	N/A	N/A	N/A	N/A	N/A

NOTE: The recorded value is the value from all 4 steam generators having the minimal margin to each criterion. The silica criterion is not applicable in Modes 2, 3 & 5.

* No limit in Mode 5, Limit is ≤2.0 in Modes 2 & 3

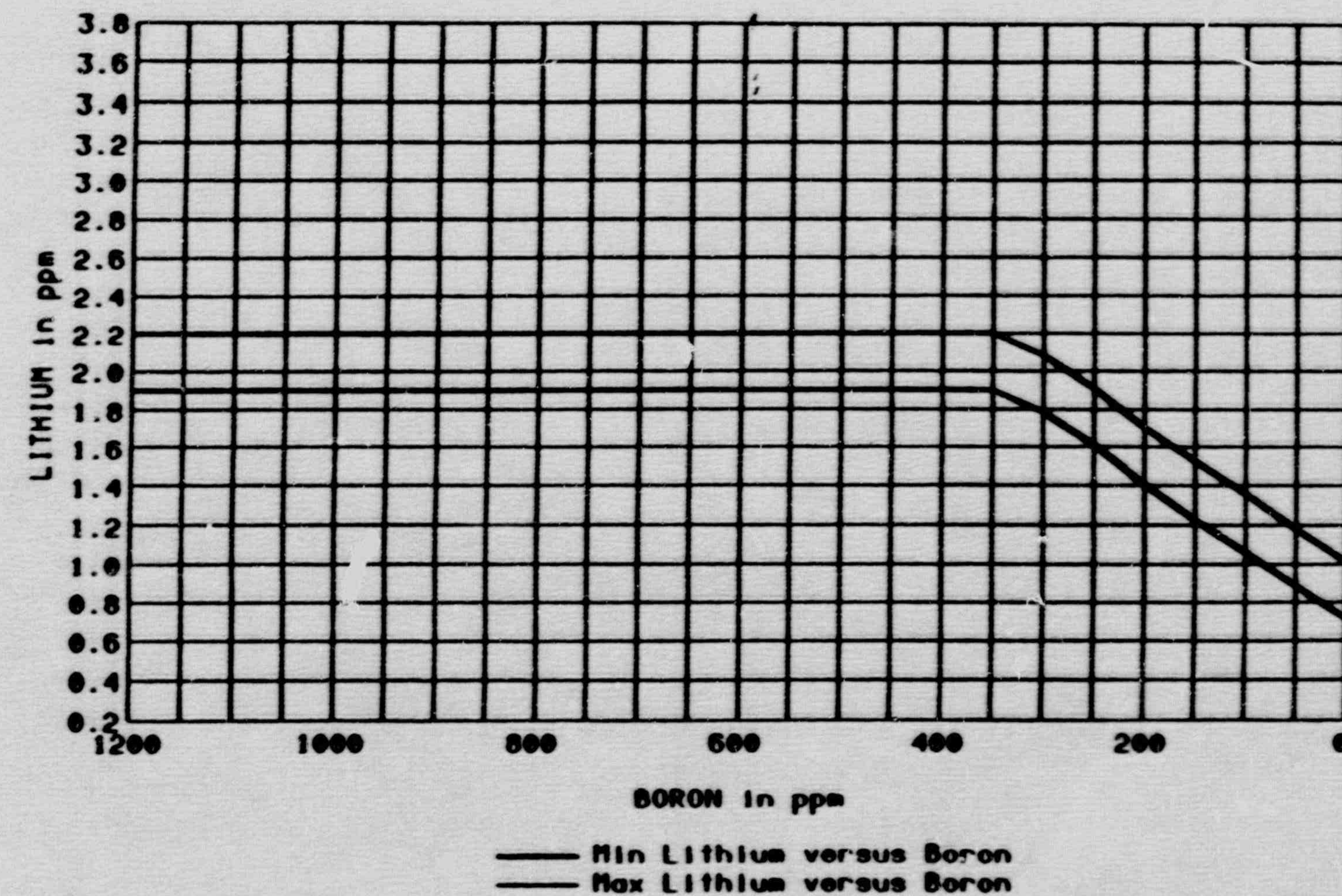
** Limit is ≥9.8 in Mode 5, ≥9.0 in Modes 2 & 3

*** Limit is ≤1000 in Mode 5, ≤100 in Modes 2 & 3

N/A = Not applicable as no criterion is specified for this plant condition

Figure 3.2.5-1

<u>**Lithium vs. Boron Curve**</u>



3.2.6 - RADIATION SURVEY TESTS - ISU-208A

OBJECTIVE

The radiation survey test is performed to determine dose levels at specified points throughout the plant, to verify the effectiveness of radiation shielding, to identify any areas of streaming through shield walls and to verify proper posting of radiation areas. This test satisfies activities described by FSAR Table 14.2-3, sheet 12.

TEST METHODOLOGY

Gamma and neutron radiation dose rate values are established by surveying with portable survey instrumentation in the Safeguards, Radwaste, Fuel, and Auxiliary Buildings, the Unit 1 Containment and penetration areas and the plant outside perimeter. Neutron radiation dose rate values are established in the Unit 1 Containment and certain penetration areas. Surveys are performed precritical, critical at 0-5%, 40-50% and 90-100% power. The key results come from the 100% power execution. The lower power data is used to verify background radiation values and to identify potential problem areas prior to reaching full power.

SUMMARY OF RESULTS

The effectiveness of gamma shielding and the general determination of dose levels were found to be adequate during performances of the test. At nominally full power, gamma dose rates were predominantly <0.1 mR/hr with 21 of 93 locations exceeding 1.0 mR/hr. Of these 21, only 2 exceeded 10 mR/hr.; one at 12 mR/hr and one at 25 mR/hr. Both of these were at the entrances to RCS loop compartments. While several points marginally exceeded their expected values, no dose rate exceeded the maximum allowed limit for that particular location.

At nominal full power, neutron dose rates were predominantly <0.5 mrem/hr with only 18 of 92 locations exceeding 1 mrem/hr. Of these 18, only 5 exceeded 10 mrem/hr; one each at 35, 40 and 50 mrem/hr and two points at 100 mrem/hr. At each of these five points above 10 mrem/hr, the dose rate was less than the estimated maximum for each point. While 23 other locations did exceed the estimated neutron dose rates, these limits were not absolute requirements and were not exceeded by more than 4 mrem/hr at any point. Evaluation by Engineering of the measured dose rate values and comparison with results from five other similar 4 loop Westinghouse PWR plants concluded that these results were acceptable. None of the dose rates was judged to pose extraordinary or undue limitations on personnel access to plant areas during operation.

3.2.6 - RADIATION SURVEY TESTS - ISU 208A (Continued)

SUMMARY OF RESULTS (Continued)

One of the originally selected radiation base points on the outside of the containment dome was found to be inaccessible and was deleted based on availability of symmetrically located points. Nine additional radiation base points were deleted based on ALARA concerns during the at-power measurements. Three radiation base points were relocated due to proximity to area radiation monitor check sources. The relocations were to nearby areas having identical expected dose rates.

Containment penetration survey results were all within allowed limits, indicating that no containment neutron or gamma streaming problems are evident. One penetration indicated a gamma dose rate of 40 mR/hr. This high dose rate was on the chemical and volume control system letdown line from the reactor coolant system and is indicative of the relatively high activity level of the fluid in this line and is not unexpected.

All measured dose rates have been evaluated as acceptable for plant operation.

3.2.7 - PROCESS AND EFFLUENT RADIATION MONITORING PERFORMANCE TEST-ISU-210A

OBJECTIVE

This test is performed to verify proper responses of all process and effluent monitors and the failed fuel monitor to existing sources of radiation (actual process or effluent fluid). This test satisfies activities described by FSAR Table 14.2-3, sheet 13.

TEST METHODOLOGY

Batch liquid monitors are tested in either Modes 1 or 2 when sufficient liquid inventory has accumulated to process. A liquid sample is taken and the radiochemical analysis of this sample is compared to the radiation monitor indication. They are expected to agree to within a factor of 2 of each other.

Continuous process liquid and gaseous monitors have samples drawn from adjacent sample ports and the radiochemical analyses of these samples are compared to the radiation monitor indication. They are also expected to agree within a factor of 2. Some monitors do not have associated sample ports. Because these monitors have no comparison made, the monitor indication is recorded as a baseline value only. If the radiochemical result is less than the minimum detectable activity or the monitor indication is less than the monitor's operational range, then the factor of 2 criterion doesn't apply and the monitor is verified to be indicating a proper background radiation level.

SUMMARY OF RESULTS

With the following listed exceptions, all batch liquid, process and effluent radiation monitors satisfied the factor of 2 comparison criterion or had their appropriate baseline readings recorded. Several monitors failed to satisfy the criterion during the lower power executions of this test. The verification of proper performance for these monitors was deferred to the full power test. The full power test is the best indicator of system ability to monitor process stream and effluent radiation under normal operating conditions. The low power tests are primarily performed to verify monitor backgrounds and to establish system operability prior to ascending to full power.

- o Spent fuel pool monitors XRE-4180, XRE-4181, XRE-4863 and XRE-4864 were not tested because the Spent Fuel Pools were empty. The Spent Fuel Pools are to be filled following completion of associated piping support work and these monitors will then be tested.

3.2.7 - PROCESS AND EFFLUENT RADIATION MONITORING PERFORMANCE TEST- ISU-210A (Continued)

SUMMARY OF RESULTS (Continued)

- o Monitor XRE-3230 was not satisfactorily tested due to inadequate monitor sample flow. The insufficient head available at the Auxiliary Steam Drain Tank loop seal did not produce sufficient flow through the monitor and associated in-line sample cooler to clear the low flow alarm and permit monitor operation. This monitor is not safety related and is to be operationally verified following resolution of the low flow problem.

- o Monitor 1RE-5179 was not satisfactorily tested due to inadequate monitor sample flow. Insufficient pressure existed at the monitor's location in the Steam Generator Blowdown system to provide sufficient flow through the monitor to clear the low flow alarm and permit monitor operation. This monitor is also to be operationally verified following resolution of the low flow problem.

- o Monitor XRE-5698, on the Safeguards Building Ventilation System, had a failed detector that cannot be replaced until the Primary Plant HVAC system can be isolated. Primary Plant HVAC cannot be isolated for this work until the plant is in either Mode 5 or Mode 6. This monitor is to be operationally verified following detector replacement. This monitor had no adjacent sample port and the monitor indication was only to be recorded as a baseline value.

- o Monitor 1RE-2959, on the Condenser Off Gas System, had a flooded detector chamber which prevented its being tested at 100% power. The detector and monitor functioned properly at 50% power and satisfied all criteria during that test portion. The monitor is to be operationally verified following detector replacement. The monitor was accepted as having passed this test based on the 50% power results.

- o Monitors XRE-5250, XRE-5253, XRE-5380, XRE-5567A, XRE-5567B, XRE-5570A and XRE-5570B failed the factor of 2 criterion at 100% power. These results were evaluated as acceptable by Engineering based on the design basis monitor ranges and actual discriminator settings. Monitor XRE-5380 had a lower level discriminator setting of 125 KeV that screened out the 81 KeV gamma from Xe-133 that was included in the radiochemical sample results. The remaining monitors were evaluated as acceptable because the individual radiochemical sample results were below the design basis operating range of the particular monitor.

3.2.8 - REACTOR COOLANT FLOW MEASUREMENT - ISU-023A

OBJECTIVE

The Reactor Coolant Flow Measurement test is performed to determine the Reactor Coolant System (RCS) flowrates for each of the 4 RCS loops, the total RCS flowrate, and to verify proper RCS flow indications. This test is performed prior to initial criticality (Mode 3) and during power ascension at approximately 50%, 75% and 100% power. This test partially satisfies activities described by FSAR Table 14.2-3, sheets 2 and 2a and Technical Specification 3/4.2.5.

TEST METHODOLOGY

Prior to criticality, data is obtained from the installed elbow tap differential pressure (d/p) instrumentation and used to calculate the RCS loop flowrates. Average values for pressurizer pressure, RCS narrow range cold leg temperature and RCS d/p transmitter output voltages are determined concurrently. The temperature and pressure readings are used to obtain cold leg specific volumes using Steam Tables. The d/p transmitter output voltage readings are converted to inches of H_2O using the known individual d/p transmitter scaling. Each loop has three flow transmitters from which a d/p measurement is taken. The d/p readings are used to determine three flowrate values for each loop using an equation for Reactor Coolant Cold Leg Volumetric Flow Rate as a function of Elbow Tap d/p and specific volume. These three flowrates are averaged to obtain the loop average flowrate. The average flowrates from all four loops are summed to obtain the total RCS flowrate.

The flow transmitters are verified to be aligned and calibrated by review of the appropriate completed Instrumentation & Controls work documents. RCS flow indications, processed from the elbow tap d/p transmitters, are read from the P2500 process computer and verified to indicate 100% $\pm$ a specified error tolerance of 1.53%.

With the plant at approximately 50%, 75% and 100% power, data is taken to determine the RCS loop flowrate. This data is a combination of a precision secondary plant calorimetric, cold leg RCS temperature values and N-16 Transit Time Flow Meter (TTFM) outputs. The TTFM is a direct flow measuring device using gamma detectors mounted on the outside of the RCS hot legs. RCS water flowing through the reactor has a portion of the Oxygen-16 nuclei present in the H_2O molecules activated to Nitrogen-16 by the neutron absorption-proton emission reaction. This N-16 leaving the reactor has a half-life of 7.10 seconds and emits gamma rays of 6.129 and 7.115 MeV. These gamma rays penetrate the RCS loop piping and are sensed by the N-16 gamma detectors. The detectors

TEST METHODOLOGY (Continued)

are located transversely to RCS loop flow and are collimated to observe fluctuations in the N-16 gamma activity as flow passes the detector. To measure RCS loop flow, the TTFM uses two pairs of gamma detectors located approximately 2 1/2 feet apart, 2 detectors upstream and 2 downstream of each other. Loop volumetric flow is calculated by multiplying the piping inside cross-sectional area by the fluid velocity. The fluid velocity is the known detector upstream-downstream spacing divided by the fluid transit time between them. A statistical cross-correlation of the N-16 gamma signal between upstream-downstream detector pairs results in this transit time. All possible upstream-downstream detector combinations are used to calculate transit times, then combined to form a mean transit time. The cross-correlation data collection, analysis and calculation of volumetric flowrate is performed by the TTFM that is connected to the N-16 detector outputs for the given loop under test. The TTFM is moved from loop to loop sequentially and does not measure all 4 RCS loop flows simultaneously. The RCS hot leg volumetric flowrates from the TTFM are converted to RCS cold leg flows using measured RCS cold leg temperature combined with a RCS hot leg temperature that is calculated from calorimetric power, cold leg temperature and hot leg volumetric flowrate. These temperatures are used to calculate hot and cold leg specific volumes and the ratio of specific volumes is used to convert hot leg volumetric flow to cold leg volumetric flow.

SUMMARY OF RESULTS

All values are in gallons/minute

% POWER	LOOP 1	LOOP 2	LOOP 3	LOOP 4	TOTAL RCS FLOWRATE	REQUIRED TOTAL FLOWRATE
MODE 3	101,827	108,093	103,067	107,265	420,252	>344,520
50%	103,494	104,184	104,093	102,257	414,028	>389,700
75%	103,300	104,300	104,000	102,650	414,250	>389,700
100%	103,331	103,928	103,932	101,948	413,139	>389,700 (and also <420,000)

The total RCS flowrate must be equal to or greater than 344,520 gpm (90% of the Thermal Design Flow) as determined by elbow tap d/p instruments prior to criticality. This was satisfied in Mode 3 at 0% power.

3.2.8 - REACTOR COOLANT FLOW MEASUREMENT - ISU-023A (Continued)

SUMMARY OF RESULTS (Continued)

At 50% and 100% power, the total RCS flowrate must be greater than or equal to 389,700 gpm (101.8% of the Thermal Design Flow) as determined by the TTFM. This was satisfied. The 50% power results also satisfied the requirements of Technical Specification 4.2.5.4 to have a flowrate of greater than or equal to 389,700 gpm, as determined by the TTFM, prior to exceeding 75% power. At 100% power the flow was also verified to be less than 420,000 gpm so as not to exceed vendor recommended NSSS mechanical design flow limits.

The RCS flow elbow tap d/p transmitters were verified to have been aligned for both zero and 100% flow prior to Mode 3 testing.

The indicated percent RCS flows at normal RCS operating conditions in Mode 3 ranged from 99.9 to 100.8% which satisfied the specified 100% ± 1.53% flow range.

3.2.9 - REACTOR COOLANT SYSTEM FLOW COASTDOWN - ISU-024A

OBJECTIVE

The Reactor Coolant System Flow Coastdown test is performed with the unit in Hot Standby (Mode 3) to verify that the measured core flow during Reactor Coolant Pump (RCP) coastdown exceeds the flow decay assumed in the accident analysis during flow decay. In addition, the low flow trip time delay is verified to be within acceptable limits. This test satisfies activities described by FSAR Table 14.2-3, sheet 3.

TEST METHODOLOGY

Strip chart recorders are connected to the RCS elbow tap d/p transmitter outputs and the Solid State Protection System (SSPS) to monitor Reactor Coolant System flow characteristics and Reactor Trip Breaker positions as a function of time. A P-8 permissive is simulated (>48% power) to ensure that a single loop loss of flow results in generation of a reactor trip signal. All 4 Reactor Coolant Pumps are tripped by manual actuation of the RCP Underfrequency Trip relay. Flow and SSPS data are taken while the RCS flow decays. All 4 Reactor Coolant pumps are verified to trip within 0.100 seconds of each other to ensure that the flow decay data corresponds to essentially a simultaneous loss of all forced RCS flow. Data from the strip charts is then statistically evaluated to verify acceptability of the measured flow values and related time delays.

SUMMARY OF RESULTS

The required Flow Coastdown Time Constant was required to be greater than or equal to 11.64 seconds. The measured value was 13.81 seconds. The Low Flow trip time delay was required to be less than or equal to 1.0 seconds. The measured value was 0.976 seconds. The Reactor Coolant Pumps were also verified to trip within 0.055 seconds of each other, which was well within the 0.100 second limit.

3.2.10 - REACTOR COOLANT SYSTEM LEAKAGE RATE - ISU-022A

OBJECTIVE

The purpose of this procedure is to verify the Reactor Coolant System (RCS) leak tightness after the system has been closed. This test satisfies activities described by FSAR Table 14.2-3, Sheets 29 and 30.

TEST METHODOLOGY

With the plant in Hot Standby (Mode 3) conditions, prior to initial criticality, the reactor coolant system is tested to verify leak tightness. After RCS pressure is stabilized, a visual leak test is conducted with the reactor pressure vessel, pressurizer and all four reactor coolant loops verified to be leak tight. Also, the unidentified, identified, and controlled leakage rates are determined using normal operating Technical Specification surveillance techniques or results from OPT-303A and OPT-110A. Pressure isolation valve leakage is also verified, based on normal Technical Specification surveillance results from EGT-712A. Primary to secondary leakage is determined by measuring boron concentration of the steam generator liquid. This calculation is based on RCS boron concentration, steam generator boron concentration, steam generator blowdown flowrate and time. This primary to secondary leakrate is measured in gpd, gallons per day. Under normal conditions the minimum detectable boron concentration of 0.2 ppm would result in a calculated leakage rate of 5.76 gpd.

SUMMARY OF RESULTS

During the visual inspection, no pressure boundary leakage was observed nor was any leakage past the Reactor Vessel flange seal observed from the flange seal leakoff. No boron was detected in the steam generators, so the conservative 0.2 ppm value was assumed. Leakage rate results are tabulated below:

Leakage Rate Type	Acceptance Criterion (gpm)	Test Results (gpm)
Controlled	≤ 40	39.8
Identified	≤ 10	0.027
Unidentified	≤ 1	0.[illegible]5
Pressure Isolation Valve	≤ 5	3.13
Primary to Secondary	≤ 500 gpd/steam generator	<5.76 gpd/steam generator

SUMMARY OF RESULTS (Continued)

The technique used to measure Controlled Leakage with the Chemical and Volume Control System flow control valve (1FCV-121) fully open failed to satisfy the $\leq$ 40 gpm criterion the first three times it was attempted. This technique was identical to that contained in surveillance OPT-110A. OPT-110A was revised based on information from the NSSS vendor and from other, similar 4 loop Westinghouse PWR plants. The revised technique satisfied the $\leq$40 gpm criterion and the results from the executed OPT-110A were used to satisfy this test requirement.

This test verified acceptable leak tightness of the reactor coolant system.

3.2.11 - COLD CONTROL ROD OPERABILITY TESTING - ISU-026A

OBJECTIVE

The purpose of this test is to verify coil polarities, proper Digital Rod Position Indication (DRPI) system operation, rod drop timing, alarm functions, DC Hold Cabinet operation, and proper slave cycler timing and to perform an operational check of each control rod drive mechanism (CRDM) with a rod cluster control assembly (RCCA) attached prior to initial use of the mechanism. This test partially satisfies activities described by FSAR Table 14.2-3, Sheets 4 and 5 and Technical Specifications 3/4.1.3.3 and 3/4.10.5.

TEST METHODOLOGY

This test is performed under two plant conditions: Mode 5 - cold, no flow and Modes 4 and 3, full flow. Proper operation of coil polarities, DRPI operation, rod drop timing, CRDM operation and slave cycler timing are verified under cold, no flow conditions. The rod bottom, rod deviation, urgent and non-urgent failure alarms and the DC Hold Cabinet are tested in Modes 3, 4 and 5.

Coil polarities are verified to preclude individual magnetic fields from the stationary gripper, movable gripper and lift coils from interfering with one another. This test uses a battery to inject low voltage pulses into the moveable gripper coil and observes the direction of current flow induced in the other two coils. Then the voltage is injected into the stationary gripper coil and the direction of induced current flows in the other two coils is again verified. Each of the 53 CRDM coil stacks is individually tested in this manner.

Slave cycler timing and CRDM operational checks are performed starting with all RCCAs positioned at the core bottom. A selected single bank is withdrawn 50 steps to ensure the RCCAs are above the dashpot region. Each RCCA in the withdrawn bank is then individually withdrawn 5 steps and reinserted 5 steps. When all RCCAs in a bank have been tested, the entire bank is reinserted to the bottom of the core. This is then repeated for each bank. While the individual RCCAs are being withdrawn and inserted 5 steps, a visicorder is used to monitor lift coil, stationary gripper coil and movable gripper coil currents. An optional signal from a microphone attached to the CRDM housing is used to help relate actual mechanical events to the coil currents.

These visicorder traces are evaluated to verify that the coil currents were of the proper shape, the currents were of the proper magnitudes at the proper times, and to verify that the events associated with mechanism movements occur in the proper order. The

TEST METHODOLOGY (Continued)

traces for each RCCA are compared against vendor supplied criteria and model traces or against the first actual trace taken, that for the RCCA at core location B-12. Rod speeds are calculated from the period of successive rod steps as being the inverse of stepping frequency.

DRPI system operability is verified by monitoring DRPI Light Emitting Diode (LED) indications on the control board during bank withdrawal and comparing these indications against other indications of RCCA position; the P2500 process computer, the demand step counters, and the pulse to analog converter. A selected bank of RCCAs is withdrawn to 12 steps and slowly reinserted to verify when the RB (rod bottom) LED illuminates for each RCCA. The bank is then withdrawn to 231 steps, the mechanical RCCA limit of motion. Shutdown bank withdrawals are stopped at 18, 210, and 228 steps to record the various position indications listed above. Control bank withdrawals are stopped every 24 steps and at 228 steps to record this data. These periodic indication verifications are also used to demonstrate Technical Specification operability of the DRPI system per Surveillance Requirement 4.1.3.3.

With a selected rod bank fully withdrawn to 231 steps, the DRPI data cabinets in the containment building are de-energized. A dedicated, personal computer based, Data Acquisition System (DAS) is hooked up to the DRPI data cabinets and the reactor trip breakers are then opened. As the RCCAs drop, the slightly magnetized, individual CRDM drive shafts, which are connected to the RCCAs, also drop through the deenergized DRPI sensing coils and induce a current in these coils which is proportional to drop velocity. As the RCCA enters the dashpot region of the fuel assembly guide tubes, it is hydraulically braked, which also shows up as a significant velocity change in the induced current signal. The DAS records the induced current signals as a function of time from all RCCAs in the selected bank, a signal proportional to stationary gripper current, and an event mark for the opening of the reactor trip breakers. From this information the rod drop time to dashpot entry can be evaluated. This rod drop timing testing is performed only one bank at a time, but is performed simultaneously for all RCCAs within a bank. The Surveillance Requirements for Technical Specification Special Test Exception 3.10.5 are satisfied within this test to allow the DRPI system to be de-energized for rod drop timing measurements.

Optionally, the drop times from all RCCAs are statistically evaluated and any RCCAs with drop times deviating more than two standard deviations from the mean drop time for all RCCAs may be redropped to confirm their actual performance.

3.2.11 - COLD CONTROL ROD OPERABILITY TESTING - ISU-026A (Continued)

TEST METHODOLOGY (Continued)

The rod deviation and dropped rod alarms are verified by withdrawing all shutdown RCCA banks, withdrawing Control Bank A to 18 steps, and then moving individual RCCAs as necessary to activate the particular alarm being tested. The rod deviation alarm is verified by deviating two RCCAs in Control Bank A by 12 steps or more and also by partially inserting a Shutdown Bank C RCCA from its fully withdrawn position. The dropped rod alarms are verified by inserting one RCCA from Control Bank A to near full insertion and then by inserting a second RCCA for the $\geq$2 rods at bottom alarm. This alarm circuitry is such that a successful test using any RCCA verifies the alarm for all other associated RCCAs.

The non-urgent failure alarm is tested by removing the input power fuses for one power supply in each of the five rod drive system's power cabinets and the logic cabinet. The cabinets are tested sequentially, not simultaneously. The urgent failure alarm is tested by interrupting the lift coil firing circuit to all RCCAs powered by the rod drive system power cabinet under test. When the RCCAs associated with the cabinet under test are ordered to withdraw, the urgent failure alarm actuates in response to the missing lift coil current. The cabinets are tested sequentially, not simultaneously, using the permanently installed lift coil disconnect switches. The logic cabinet urgent failure alarm is tested by removing a preselected circuit board.

The DC Hold Cabinet serves as an alternate power source to hold a single group of RCCAs in a withdrawn position to allow for maintenance on the stationary gripper power circuitry for that group. A group of RCCAs consists of 2,3, or 4 RCCAs. The DC Hold Cabinet is tested by switching it to hold a group of 4 withdrawn RCCAs, de-energizing the normal power circuitry for that group and verifying the RCCAs remain withdrawn.

SUMMARY OF RESULTS

Coil polarities were all verified to be correct when tested in Mode 5.

The current and sound traces from all 53 RCCAs were all verified proper when tested in Mode 5. The traces were all of the proper shape with no notable anomalies. The timing of events and current magnitudes were all verified to be acceptable and concurred with by

SUMMARY OF RESULTS (Continued)

vendor representatives. Actual rod speeds from evaluation of the inverse of the period of successive rod steps were as follows:

RCCA Bank Type	Expected Speed(steps/min)	Actual Speed(steps/min)
Control Bank	48	45.5
Shutdown Bank A or B	64	62.82
Shutdown Bank C,D or E	64	63.5

There were no tolerances on the expected speed values because these rod speed values are to be remeasured at hot (Mode 3) RCS conditions. These cold values are expected to differ from the expected speed values due to mechanical (thermal expansion) conditions associated with the low RCS temperature. One problem was noted with respect to rod speeds. While taking trace data for Shutdown Bank C, the time between rod steps was observed to be significantly smaller than expected. Evaluation of the traces resulted in a measured rod speed of 75 steps/minute. An adjustment was made to the rod speed circuitry for that cabinet and the final value following this adjustment was recorded above.

The DRPI LED indications on the main control board and P2500 computer were verified to be within ±4 steps of the rod drive system group step counter indications for all 53 RCCAs. The actual deviation was 0 steps. The pulse to analog converter indications were verified to be within ±1 step of the group step counter indications for all 4 control banks. The actual agreement was also exact, with a deviation of 0 steps. The DRPI LED indication for rod bottom (RB) indicated at or prior to reaching zero steps, as indicated by the group step counter, during RCCA insertion. The RB LEDs all illuminated at an indicated 3 steps. This DRPI testing was performed in Mode 5.

The rod deviation alarms functioned properly for an actual rod vs. rod deviation of 12 steps and a shutdown rod at 210 steps withdrawn. The rod bottom and ≥2 rods at bottom alarms functioned properly in response to actual RCCA insertions. These alarms were tested in Modes 3 and 4.

The urgent and non-urgent alarms functioned properly in response to failed power supplies, missing lift coil currents and the missing circuit board. These alarms were tested in Mode 3.

The DC Hold Cabinet was verified to hold a group of 4 RCCAs in a withdrawn position for 10 minutes. This verification was performed in Mode 3.

SUMMARY OF RESULTS (Continued)

All rod drop times were verified to be less than the Technical Specification limit of 2.4 seconds. That limit does not actually apply to this test performance because the limit is for a hot, full RCS flow test and these rod drops were done cold in Mode 5. The average drop time was 1.603 seconds. No two standard deviation redrops were performed. They were not required at this plant condition. The rod drop data was taken for baseline purposes and to verify rod drop DAS operation only. Additionally, evaluation of the rod drop DRPI coil current traces verified proper operation of the dashpot decelerating devices.

The test procedure directed the performance of the normal Instrumentation and Controls calibration procedure which was performed in Mode 3 to further confirm proper operation of all portions of the DRPI system.

Three miscellaneous problems were noted with respect to rod drive system operation during performance of this test:

- o Blown fuses within both rod drive system motor-generator sets' generator voltage control circuitry resulted in improper phase voltages when the generator field was flashed. The fuses were replaced and the motor-generator sets operated properly thereafter.
- o Irregularities were noted with the operation of the main control board switch that closes the reactor trip breakers. Closure of the breakers is dependent on how long the switch is held in the closed position and sometimes also requires two switch actuations to close the breakers. Opening (tripping) of the reactor trip breakers is unaffected by this closing problem. Operator awareness of this condition eliminated further problems.
- o CRDM testing in Mode 4 resulted in occasional rod misstepping due to dissimilar thermal expansions between the CRDM and the Control Rod Drive Shaft (CRDS) which resulted in mechanical misalignments of the CRDM gripper teeth and CRDS grooves. The rod motions were proper in both Modes 5 and 3. In both Modes 5 and 3 the tooth and groove alignments were correct but the temperature regime in Mode 4 is such that the alignments were not quite correct. The mechanical design is such that it allows for proper engagement when cold or hot, but not necessarily when in between. Normally, rod motion is demanded only when hot or cold, it is not customary to move rods in Mode 4. The testing was deferred to Mode 3 and was satisfactorily performed there.

3.2.12 - HOT CONTROL ROD OPERABILITY TESTING - ISU-027A

OBJECTIVE

The purpose of this test is to verify proper DRPI system operation, rod drop timing, rod speed and direction, overlap operation, manual operation and to perform an operational check of each CRDM with a RCCA attached in Mode 3 prior to initial criticality. The actual mechanical RCCA withdrawal limit is also verified. This test partially satisfies activities described by FSAR Table 14.2-3, Sheets 4, 19, 31 and 32 and Technical Specifications 3/4.1.3.4 and 3/4.10.5.

TEST METHODOLOGY

Slave cycler timing, Control Rod Drive Mechanism (CRDM) operational checks, measurement of the mechanical withdrawal limit, DRPI system checks and rod drop timing is performed in an integrated fashion, on a sequential bank by bank basis. The selected bank is withdrawn to 228 steps, with the operation of every DRPI LED verified during this withdrawal with respect to the group step counter indications. There is an LED for every 6 steps of RCCA motion. Shutdown banks have no LEDs to represent position between 18 and 210 steps, only a transition region (TR) LED. Each individual RCCA in the withdrawn bank is inserted 5 steps and then withdrawn 10 steps, ending at an indicated position of 233 steps on the group step counters. During these 5 and 10 step movements, visicorder trace data is taken as was done in ISU-026A. Refer to Test Summary 3.2.11. This trace data is also evaluated as was done in ISU-026A with respect to rod speeds and timing and magnitudes of coil current changes. Sound traces are taken for only one CRDM per rod drive power cabinet due to microphone integrity concerns while at normal RCS operating temperature. The trace data is also evaluated to verify the mechanical RCCA withdrawal limit. When the CRDM drive shaft reaches its mechanical limit of travel there are no more grooves on the Control Rod Drive Shaft available for the CRDM grippers to latch into. This shows up on the trace as a gripper current anomaly. The traces are evaluated near the top of travel, above 228 steps, with respect to where this anomaly occurs. This mechanical limit is typically 231 steps but may vary from plant to plant. Once the mechanical withdrawal limits have been determined for all RCCAs in a bank, that bank is dropped to measure rod drop times as was done in ISU-026A. This set of rod drop times satisfies Surveillance Requirements for Technical Specification 3.1.3.4 and is performed at RCS full flow conditions. The Surveillance Requirements for Technical Specification Special Test Exception 3.10.5 are also satisfied within this test. Any RCCA having a drop time deviating from the mean drop time by more than two standard deviations is redropped an additional three times to confirm its actual performance.

TEST METHODOLOGY (Continued)

Rod speed and direction indications on the main control board are verified while withdrawing and inserting various RCCA banks. Control bank overlap is verified by withdrawing the control banks in the manual overlap mode instead of in the individual bank select mode of operation. As a prerequisite to this test portion, the overlap switch settings are changed to lower, yet sequential, values. This allows verification of overlap without the need for complete withdrawal of the control banks. As the control banks are withdrawn in manual overlap, data is recorded each time a bank starts or stops motion. This data is compared to the switch settings.

The ability of an urgent failure alarm to block RCCA motion is tested by creating an actual urgent failure alarm, by interrupting lift coil signals using permanently installed disconnect switches, and then attempting to move the RCCAs. The urgent failure alarm is then cleared and RCCA motion is verified to have been restored.

SUMMARY OF RESULTS

The current traces from all 53 RCCAs were verified to be proper. The traces were of the proper shape with no notable anomalies. The timing of events and current magnitudes were all verified to be acceptable and were concurred with by vendor representatives. Sound trace data from each power cabinet was also verified to be proper.

Actual rod speeds were verified as follows:

RCCA Bank Type	Expected Speed(steps/min)	Actual Speed(steps/min)
Control Bank	48 ±2	46.2
Shutdown Bank A or B	64 ±2	62.9
Shutdown Bank C,D or E	64 ±2	63.2

The DRPI system indications were typically within 1 or 2 steps of the group step counter indication with only one group of 4 RCCAs off by 3 steps at one rod position, Shutdown Bank D at 207 vs. 210 steps. This satisfied the ±4 step agreement criterion.

The mechanical withdrawal limit was established to be 231 steps by inspection of visicorder trace data above 228 steps. This trace data was repeated for Shutdown Bank A due to legibility problems with that portion of the original trace data for that bank. This trace data was also repeated for Control Bank A due to a visicorder paper jam.

SUMMARY OF RESULTS (Continued)

Rod drop timing measurements were made for all 53 RCCAs from the 231 step full mechanical withdrawal position. All times were less than 2.4 seconds from decay of stationary gripper voltage to dashpot entry. The fastest RCCA took 1.30 seconds. The slowest RCCA took 1.47 seconds. The average RCCA drop time was 1.40 seconds with a standard deviation of 0.03 seconds. Only two RCCAs were outside of the two standard deviation limits. They were redropped 3 times each with the following results:

Drop Times (seconds)

Drop Type	RCCA F-6	RCCA H-14
Original	1.30	1.47
Redrop #1	1.33	1.43
Redrop #2	1.33	1.42
Redrop #3	1.32	1.43

These redrops also satisfied the criterion that for each redropped RCCA, the three redrop times shall all be within a 0.02 seconds band. This rod drop timing test satisfied the Surveillance Requirements for Technical Specification 3.1.3.4 and Special Test Exception 3.10.5. One minor problem occurred while taking rod drop timing data. The RCCA at Core Location H-12 initially generated a bad trace due to poor DRPI system cabinet test probe contact. A new test probe was used and the RCCA was successfully retested.

The rod speed and direction indications on the main control board were verified to be correct. The speed indications of either 48 or 64 steps/minute were correct. Control bank overlap was verified to occur exactly at the overlap switch settings with no deviation. This satisfied the allowed ±1 step deviation criterion.

The urgent failure was generated and was verified to inhibit manual RCCA motion. RCCA motion was restored following clearing of the alarm.

3.2.13 - REACTOR TRIP SYSTEM TESTS - ISU-015A

OBJECTIVE

The purpose of this test is to verify proper operation of the automatic and manual reactor trip breaker circuitry and to verify proper operation of the reactor trip breakers prior to initial criticality. This procedure also tests reactor trip bypass breaker functions and verifies proper unlatching of the control rods following opening of the reactor trip breakers. This test satisfies activities described by FSAR Table 14.2-3, Sheets 6 and 7.

TEST METHODOLOGY

The Solid State Protection System (SSPS) general warning interlocks associated with the trip breakers are tested by closing both reactor trip breakers (RTBs) and one of the trip bypass breakers (TBBs). The SSPS train opposite to the TBB that is closed is placed into test and it is verified that all three breakers then open automatically. This sequence is repeated for the other TBB and SSPS train.

TBB interlocks are tested by closing one TBB and verifying that an attempt to close the second TBB results in the automatic opening of both TBBs. This sequence is repeated with the other TBB starting in the closed position.

Functional testing of RTB and TBB operation is performed by closing both RTBs and one TBB. A trip signal is then simulated on the SSPS train associated with the closed TBB. It is verified that the RTB corresponding to the tripped SSPS train opens and the other two breakers remain closed. This sequence is repeated for the other TBB and SSPS train.

Manual trip function is demonstrated by closing both RTBs and one TBB and generating a manual trip signal from a control board reactor trip switch. All three breakers are verified to open and the remaining TBB is closed and verified to open in response to a second actuation of the reactor trip switch. This sequence is repeated for the second main control board reactor trip switch.

Verification of actual control rod release following RTB opening is tested by withdrawing all 53 control rods to 12 steps and manually initiating a trip signal using a main control board reactor trip switch. All control rods are verified to return to their fully inserted positions using the Digital Rod Position Indication System.

3.2.13 - REACTOR TRIP SYSTEM TEST - ISU-015A (Continued)

SUMMARY OF RESULTS

SSPS general warning interlocks were verified to properly result in the opening of the RTBs and TBBs. TBB interlocks were verified to properly prevent simultaneous closure of both TBBs. Proper function of RTB and TBB operation was verified, demonstrating that the TBBs permit individual RTB trip testing without resulting in an actual reactor trip. The manual reactor trip switch trip function was properly demonstrated for both main control board reactor trip switches. All 53 control rods were verified to unlatch and fall from the 12 step position to the fully inserted position following opening of the RTBs.

Only one problem occurred during test performance. As discussed previously in the Summary of Results for ISU-026A, the main control board control switch often required multiple actuation to close the reactor trip breakers. This had no adverse impact on these test results because the trip function of this switch did not require multiple actuation.

3.2.14 - PRESSURIZER SPRAY AND HEATER CAPABILITY - ISU-021A

OBJECTIVE

This test is performed to verify pressurizer spray and heater effectiveness. In addition, the spray line bypass valves are adjusted to maintain spray line temperature above 540°F. This test partially satisfies activities described by FSAR Table 14.2-3, Sheets 2 and 2a.

TEST METHODOLOGY

In order to set the spray line bypass flows, the spray valves and spray bypass valves are closed and the line temperatures allowed to stabilize. The valves are then opened in 1/16 turn, or greater, increments until a satisfactory temperature reading is achieved. The spray line low temperature alarm is verified to actuate at 540 ±4 °F.

To verify spray effectiveness, the heaters are manually isolated and both spray valves are placed into the full open position. Pressurizer parameters are monitored via strip chart recorders. These parameters are then analyzed and plotted to verify the pressure transient falls within the allowable limits. Data is also taken for the response to a single spray valve opening.

To verify heater effectiveness, the spray valves are manually isolated and the heaters are placed to the full on position. Pressurizer parameters are monitored via strip chart recorders. These parameters are then analyzed and plotted to verify the pressure transient falls within the allowable limits.

To verify stable pressurizer pressure control ability, the spray valves and heaters are manually operated to adjust pressurizer pressure to approximately 2200 psig. The controls are placed in automatic and pressurizer pressure is verified to stabilize within the normal operating band of 2235 ±30 psig. A similar test is also performed starting at approximately 2300 psig.

SUMMARY OF RESULTS

The pressurizer spray bypass valves were properly set to ensure that adequate spray line temperatures exist when the spray valves are closed. This prevents excessive spray line cooldown which can cause potentially deleterious thermal effects on piping and components when sprays are activated. Valve 1RC-8051 was set to 4 turns open and valve 1RC-8052 to 1 1/2 turns open. These settings result in spray line temperatures of 543°F, which also allows for sufficient margin above the 540°F low temperature alarm setpoint.

Testing also determined that these settings are the minimum valve positions that can maintain line temperatures adequately above

SUMMARY OF RESULTS (Continued)

540°F and that, at these valve settings, pressurizer control heater bank C could not maintain pressurizer pressure by itself, without periodic backup heater bank actuation. Periodic backup heater operation is not a safety or operability concern, only one of efficiency.

Initial testing of spray bypass valve 1RC-8051 could not achieve the originally expected spray line temperature of 548°F at any valve position, full open yielded 546.6°F. The plot of spray line temperature vs. valve position did indicate a significant slope change, a plateau, at 545°F. This plateau region is the area at which changes in valve position have minimal influence on spray line temperature. Investigation into the basis of the original test requirement of a 548°F minimum spray line temperature resulted in the change of this value to 540°F. This permits operation at 543°F with 3°F of margin to the 540°F low temperature alarm and still allows 10°F margin from the alarm setpoint to the 530°F minimum spray line temperature basis.

The spray line low temperature alarms were verified to be set at 539.38°F and 539.56°F which satisfied the 540 ± 4°F criterion.The pressure transient resulting from the spray effectiveness test fell within the required band, refer to Figure 3.2.14-1. The opening of a single spray valve was verified to result in an average pressure decay rate of 75.4 psi/min. The opening of both spray valves resulted in an average rate of 88.3 psi/min. The spray effectiveness plot demonstrated that the spray valves capacity was such that the pressurizer pressure response was sufficient to respond to design plant transients but not so large as to cause excessive rates of change of pressurizer pressure and temperature.

Two points of the pressure transient resulting from the heater effectiveness tests fell outside the required band, refer to Figure 3.2.14-2. The heater response was mostly in the low range of the allowed band indicating that the effectiveness of the heaters was marginally sufficient to support optimal response to design transients. However, sufficient heater capacity was judged to be available to adequately support design transients. The Technical Specification heater capacity requirements are more than adequately satisfied. Engineering and the NSSS vendor evaluated the data and determined the test results to be acceptable to support subsequent plant operations.

The pressurizer pressure was verified to stabilize at 2235 ±30 psig when controls were placed in automatic from starting points at approximately 2200 and 2300 psig. No sustained or diverging oscillations were noted.

3.2.14 - PRESSURIZER SPRAY AND HEATER CAPABILITY - ISU-021A (Continued)

SUMMARY OF RESULTS (Continued)

Following the completion of this test, the spray line low temperature alarm was reduced to 525 ±3°F by a plant design modification.

Figure 3.2.14-1

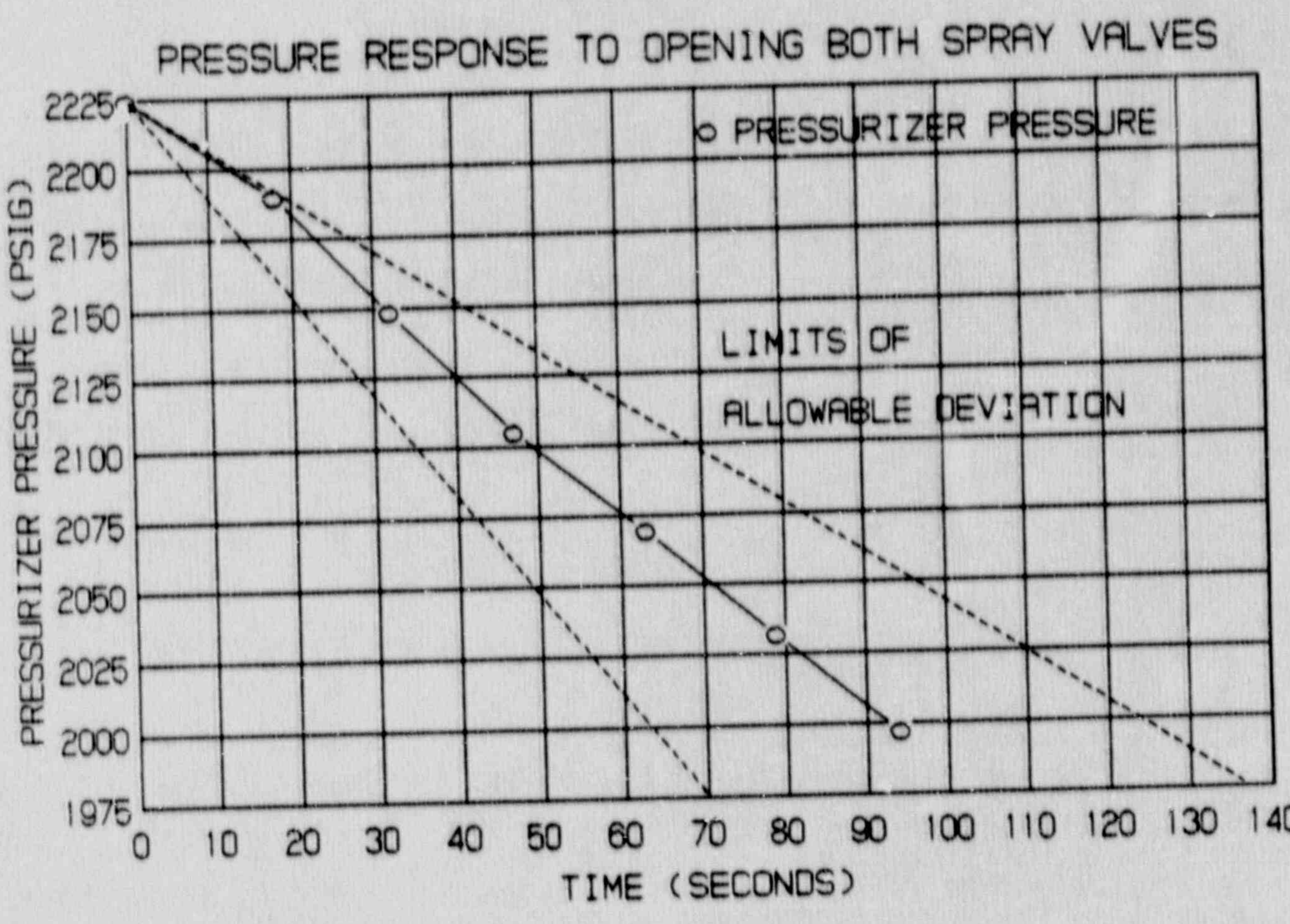
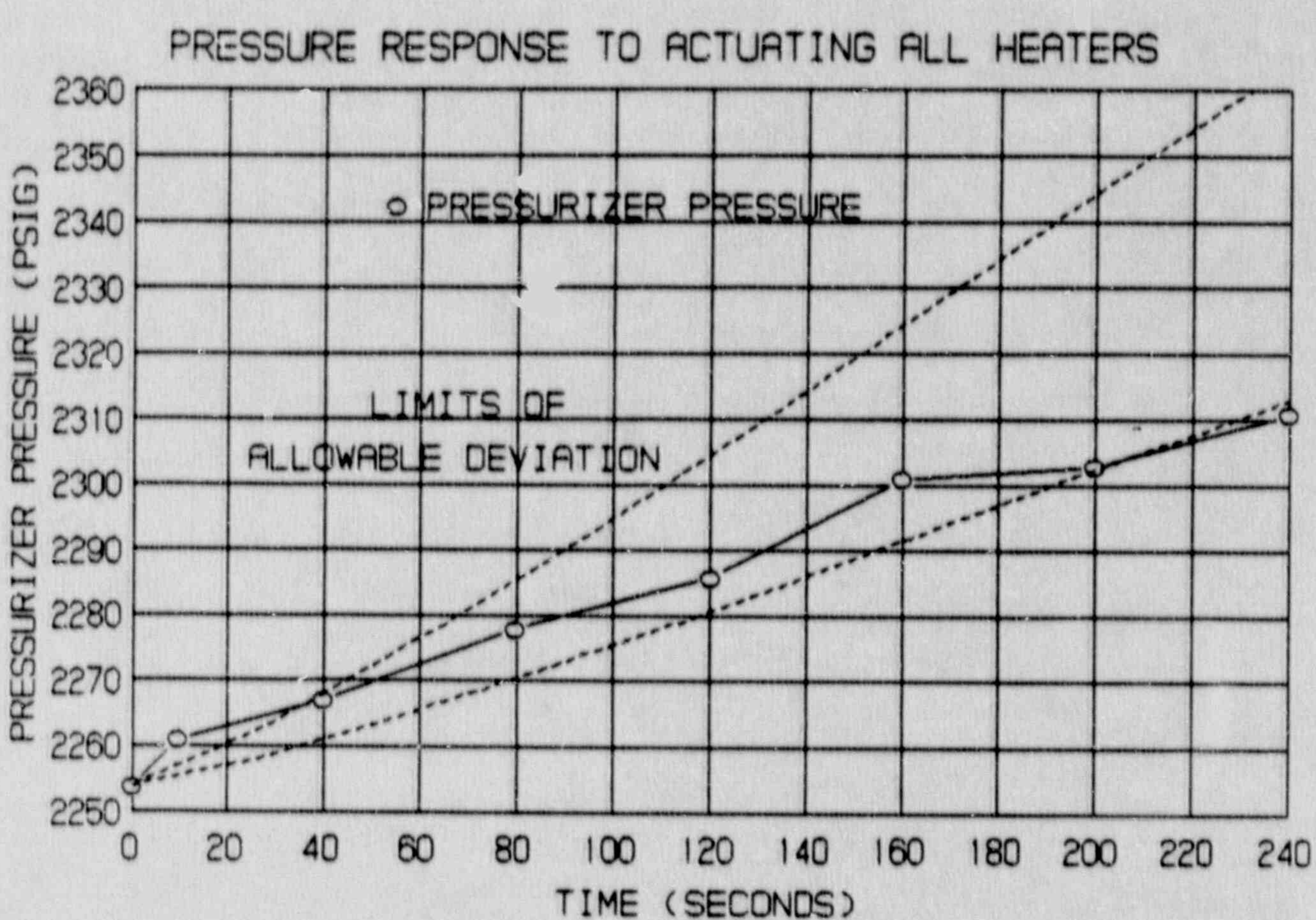


Figure 3.2.14-2



3.2.15 - MISCELLANEOUS BALANCE OF PLANT TESTING

OBJECTIVE

These tests are performed to verify proper performance of various balance of plant systems and components which cannot be fully tested prior to actual power operations. The systems to be tested include Main Feedwater, Steam Dump Valves, Steam Generator Atmospheric Relief Valves, and the Main Turbine and Generator. These tests were performed to identify any significant problems in the secondary plant and to perform tuning of controls to optimize plant performance. This testing is not described in the Final Safety Analysis Report and is included for information only.

TEST METHODOLOGY

These tests are performed at the earliest time possible (lowest practical power level) during the initial startup program. Some tests are also repeated at other power plateaus, as appropriate.

The Main Feedwater System Test determines if there is any significant leakage past the Feedwater Isolation and Feedwater Isolation Bypass Valves by monitoring downstream piping temperature before and after starting a feedwater pump. Prior to opening a Feedwater Isolation Valve, the purge flow through the Feedwater Isolation Bypass Valve is measured with an Ultrasonic Flowmeter. At approximately 50% power, the Auxiliary Feedwater check valve backleakages are verified to be insignificant by monitoring temperature profile of the upstream piping. At approximately 75%, 90%, and 100% power, the feedwater split flow to the steam generator upper nozzle (steam generator auxiliary feedwater nozzle) is measured and verified to satisfy the limitations provided by the NSSS vendor.

The Feedwater Pressure Oscillation test monitors the magnitude of the continuous pressure oscillations in the feedwater piping at the inlet to the steam generator main nozzles. These pressure oscillation measurements are used to verify the fatigue capability of the steam generator preheat section structure. Piezoelectric pressure transducers installed in the feedwater line near the main inlet nozzles provide signals to a test data acquisition system. This data is evaluated with a spectrum analyzer and a plot of the peak-to-peak amplitude versus frequency is compared to the allowable pressure oscillations provided by the NSSS vendor. This data is collected for the following plant conditions:

- With only auxiliary feedwater being supplied to the steam generators
- Following transfer of Feedwater flow from the upper nozzle to the main nozzle with one main feedwater pump in service
- While bringing the second main feedwater pump into service
- At steady state conditions of 90% and 100% power operation
- Following a 50% step load reduction.

3.2.15 - MISCELLANEOUS BALANCE OF PLANT TESTING (Continued)

TEST METHODOLOGY (Continued)

The Main Feedwater Pump Performance Test verifies the pump and pump turbine trip functions, the pump hydraulic performance in the recirculation mode, and the response of the recirculation valves.

The Steam Generator Atmospheric Relief Valve Capacity Test is performed to demonstrate the operability of the relief valves under hot, steaming conditions. With the plant at approximately 25% reactor power, each relief valve is manually opened and the increase in feedwater flow is used to calculate the flow capacity of each valve.

The Turbine Generator Initial Synchronization and Overspeed Test performs an overspeed trip test, including an actual turbine trip, synchronizes the generator to the grid and verifies no excessive vibrations occur on the hydraulic control lines to the turbine. The turbine speed is brought up to 1800 rpm and then overspeed using the Trip Testing Lever until the Hydraulic Governor Stop Setting causes a trip. After the turbine has slowed to turning gear speed, the turbine speed is raised back to 1800 rpm and the generator is synchronized to the grid. The Electro-Hydraulic Control (EHC) hydraulic lines are monitored for vibration during the test.

The Dynamic Automatic Steam Dump Control test demonstrates the capability of the Steam Dump system Plant Trip, Load Rejection, and Steam Pressure controllers to control either Tavg or Steam Pressure. With the steam dumps in Steam Pressure mode, reactor power is increased from 0% to approximately 4% and decreased back to 0% by control rod motion. Steam pressure is verified to remain stable within the control band of 1092 ±20 psig. With the reactor power at approximately 1%, a simulated trip signal (P-4) injected and Tavg elevated to approximately 560°F, the steam dump controller is placed into automatic. While reactor power is increased from 1% to 5% and then reduced to 3%, the reactor temperature and steam dump valve response is monitored to verify that the Plant Trip controller maintains temperature correctly at approximately 557°F. Then, reactor power is increased to approximately 5% and "Turbine in Operation" and "Loss of Load" condition signals are simulated to cause the steam dump controller to be in Load Rejection mode. The reactor coolant average temperature and steam dump valve responses are monitored to verify the Load Rejection controller maintains temperature correctly.

The Steam Dump Performance and Timing Test verifies the time response and stroke length of each of the steam dump valves. With the steam dump isolation valves closed, the steam dumps are modulated open with the pressure controller. The fully closed to fully open stroke lengths are measured. The valves are timed as

TEST METHODOLOGY (Continued)

they trip closed by deenergizing each solenoid. The valves are also timed as they modulate closed using the pressure controller. Finally, the valves are timed as they trip open in response to the Tavg controller.

The Steam Dump Valves Capacity Test determines the steam flow capacity of each bank of three steam dump valves. Each bank of steam dump valves are individually opened while maintaining turbine load constant. The resultant increase in power is verified to be approximately 10% for each bank. This increase is determined by measuring changes in feedwater flow which is more precise than nuclear instrumentation at these plant conditions.

The Balance-of-Plant Data Collection test is performed to gather data for the PEPSE program to calculate the Main Turbine and Secondary Plant component performances and plant heat rate and for the recording of baseline secondary systems performance data. With the plant operating at steady state conditions, data is collected by data acquisition systems, the process computer and manually from plant instruments for a period of one to two hours. The data is entered into the PEPSE program for calculations and analysis. The PEPSE program calculates Plant Heat Rate, Main Turbine Efficiency, Moisture Separator performance, Reheater performance, Main and Auxiliary Condenser performance, Feedwater, Condensate and Heater Drains pumps performance, and Feedwater Heater performance.

SUMMARY OF RESULTS

The NSSS Vendor evaluated the Main Feedwater System Test temperature data from the feedwater piping before and after a feedwater pump start and determined the Feedwater Isolation and Feedwater Isolation Bypass Valves of all four loops were acceptably leaktight. The purge flows through the Feedwater Isolation Bypass Valves were determined to satisfy the review criterion.

Loop	Purge Flow	Review Criterion
1	88,378 lbm/hr	60,000 to 120,000 lbm/hr
2	93,810 lbm/hr	60,000 to 120,000 lbm/hr
3	110,914 lbm/hr	60,000 to 120,000 lbm/hr
4	86,981 lbm/hr	60,000 to 120,000 lbm/hr

Prior to performance of this test, problems with the Auxiliary Feedwater check valves leakage were well documented and operational controls were in effect to monitor and minimize this leakage.
The NSSS vendor evaluated the temperature data recorded during the test from the auxiliary feedwater piping and this indicated that

3.2.15 - MISCELLANEOUS BALANCE OF PLANT TESTING (Continued)

SUMMARY OF RESULTS
(Continued)

none of the eight check valves experienced significant leakage at that time. The NSSS vendor recognized the ongoing evaluations of this issue. As discussed in TXX-90188, dated May 18, 1990, TU Electric is planning to order check valves of a different design for this Auxiliary Feedwater application to cover the contingency that the replacement of the present valves becomes appropriate.

The results of the split flow measurements are as follows (all values are in 10^6lbm/hr):

Loop	Flow to Upper Nozzle at 75% Power	Criterion for 75% Power	Flow to Main Nozzle at 100% Power	Criterion for 100% Power
1	0.171580	0.0379 - 0.3785	3.308	$\leq$ 3.39
2	0.198823	0.0379 - 0.3785	3.428	$\leq$ 3.39
3	0.[illegible]72559	0.0379 - 0.3785	3.417	$\leq$ 3.39
4	0.190818	0.0379 - 0.3785	3.274	$\leq$ 3.39

The 100% power measured flows to the #2 and #3 steam generator main nozzles exceeded the limitation provided by Westinghouse. Westinghouse has recommended that operation above the specified limit may continue for the remainder of Cycle 1 and that the steam generator preheater tubes be inspected at the first planned outage of the steam generators. Westinghouse has recommended raising the high flow alarm setpoint to 3.55×10^6 lbm/hr which is 93.9% of full flow. Nominally 10% of full flow is expected to bypass the main nozzle and flow through the upper nozzle. Evaluation of this condition is ongoing.

The Feedwater Pressure Oscillation test started in Mode 3 and finished at the 100% power plateau. The peak to peak amplitude versus frequency plots were well within the allowable limit curves provided by Westinghouse.

The Main Feedwater Pump Performance Test verified all the pump and pump turbine trip functions were satisfactory. The pump and turbine auxiliary equipment operated correctly and the recirculation valves operated satisfactory. During the 1B feedwater pump turbine performance test, the pump inboard bearing overheated. Following realignment of the pump, the two-hour performance run was completed successfully. All other tests results were satisfactory.

SUMMARY OF RESULTS (Continued)

The Steam Generator Atmospheric Relief Valve Capacity Test, Revision 0, verified that each valve fully opened and closed under hot, steaming conditions. The measured valve flow capacities were not consistent with each other or with the specified test criterion. Valves 1-PV-2325 and 1-PV-2328, loops #1 and #4, were retested in Revision 1 of the test procedure and still had insufficient flow capacity. Revision 2 of the test procedure retested all four valves and again the valves' capacities were calculated to be too low, with the capacity of valve 1-PV-2326 significantly lower than the other three valves. Based on this data and a re-evaluation of the design basis for the steam generator atmospheric relief valves, a design modification was made to the atmospheric relief valves to increase their stroke lengths from 1 3/8 to 1 7/16 (+1/16,-0) inches. Valve 1-PV-2326 was also found to need a control loop recalibration, which was performed. It was determined that the method of measuring steam flow capacity used in the test was not accurate enough and that measuring the valve stroke length was a more accurate measure of capacity. The valves' stroke lengths were verified to be acceptable and the valves were declared operable.

The Turbine Generator Initial Synchronization and Overspeed Test was satisfactorily performed. The turbine overspeed trip occurred at 1984.5 rpm. The acceptance criterion was to trip between 1980 and 1998 rpm. The generator was successfully synchronized to the grid at 1530 hours on 4/24/90. The EHC lines showed no excessive vibrations. During the initial attempt to overspeed the turbine, the SPEED REFERENCE signal to the control room was found to be incorrect and the SPEED REFERENCE card was recalibrated.

The Dynamic Automatic Steam Dump Control test was performed prior to power ascension above ten percent reactor power. The Steam Pressure controller properly maintained steam pressure between 1072 and 1112 psig. During the Plant Trip controller and the Load Rejection controller portions of this test, Tavg was not maintained at the temperature anticipated by the procedure. However, the steam dump valve response was evaluated and still determined to be acceptable for both modes of the controller. The error was caused by the conservative gains set in the nuclear instruments, thus indicating higher reactor power than actually existed. Note: Upon reaching 30% reactor power, a secondary calorimetric was performed to correct the gain settings on the nuclear instruments. Other problems encountered in this test included the need to add a jumper to simulate a trip condition and the need to remove the lead function of the steam dump control card to improve controller response.

3.2.15 - MISCELLANEOUS BALANCE OF PLANT TESTING (Continued)

SUMMARY OF RESULTS (Continued)

The Steam Dump Performance and Timing Test results were as follows:

Valve	Bank	Stroke Length (in.)	Solenoid Trip Time Closed(sec)	Modulate Closed Time(sec.)	Trip Open Time(sec.)
Acceptance Criteria		2 3/4 ±1/8	<5	≤20	<3
1-PV-2369A	I	2 3/4	2.8	17	2.2
1-PV-2369B	I	2 3/4	3.2	25	1.8
1-PV-2369C	I	2 3/4	2.6	18	2.5
1-TV-2370A	II	2 3/4	2.6	15	3.2
1-TV-2370B	II	2 3/4	2.8	24	2.1
1-TV-2370C	II	2 3/4	2.8	15	2.5
1-TV-2370D	III	2 3/4	2.6	13	2.4
1-TV-2370E	III	2 3/4	2.6	13	2.6
1-TV-2370F	III	2 11/16	2.4	12	2.0
1-TV-2370G	IV	2 3/4	2.4	10	2.2
1-TV-2370H	IV	2 11/16	3.2	10	2.8
1-TV-2370J	IV	2 3/4	2.6	10	2.6

The stroke lengths for valves 1-TV-2370F and 1-TV-2370J were initially too short and were readjusted to satisfy the criterion. The modulated close times for valves 1-PV-2369B and 1-TV-2370B were too long to meet the criterion. The engineering evaluation of this data determined these times were within the manufacturer's tolerance, and the overall response of all the valves compensated for the slightly longer times on these two valves. These two valves are not in the same bank. These two valves were readjusted to reduce their closure times following the completion of this test. The trip open time of valve 1-TV-2370A was too long to satisfy the criterion. The engineering evaluation of this data determined that this time was also within the manufacturer's tolerance and the results of this test are acceptable.

The indicated Steam Dump Valves Capacity Test results were as follows:

Bank I = 23.055% of rated power
Bank II = 21.656% of rated power
Bank III = 21.859% of rated power
Bank IV = 26.962% of rated power

3.2.15 - MISCELLANEOUS BALANCE OF PLANT TESTING (Continued)

SUMMARY OF RESULTS (Continued)

The expected results were that each bank would be worth 10% ±2% of rated reactor power. During the initial testing of Bank IV, Valve 1-TV-2370H did not open. It was readjusted and stroked and the test repeated for Bank IV. An engineering evaluation of these results determined that the change in power caused by opening the steam dump valves resulted in changes to the steam flows in too many other flow paths for the increased feedwater flow to be used to accurately measure the flow equivalent of power change. It was concluded that this test provides a qualitative information for the flow passing capacity of the steam dump valves but not a true quantitative value and that the test results are acceptable for the purposes here. The operational valve capacities were later verified in the plant transient tests.

The Balance-of-Plant Data Collection test started with the plant initially at 45% power and was performed periodically throughout the remainder of the initial startup. This permanent plant procedure will also continue to be used throughout the life of the plant. Initially, the test was used as a means to identify any performance problems and excessive heat losses. Numerous steam leaks were identified and corrected. Tuning, recalibration, and modification of the design of secondary system instruments, controls and processes were identified and performed as a result of this test. Additional potential improv ients in performance were identified for further evaluation and fu.ure implementation. The following set of data is an example of the results obtained:

Parameter	Design Value	Actual Value on 7/24/90
Gross Output (MWe)	1163	1130
Turbine Power (%)	100	97.16
Condenser Vacuum ("Hg)	3.38	3.37
Feedwater Flow (lbm/hr)	15,140,015	14,951,403
Feedwater Pressure (psia)	1172	1136.9
Steam Pressure (psia)	975	1007.98
Heat Rate (BTU/KW-HR)	10,048	10,268.94
Heater Drain Flow (lbm/hr)	5,289,632	5,588,051
Feedwater Temperature (°F)	440	437.3
HP Turbine Inlet Pressure (psia)	880.28	889.09
LP Turbine Inlet Pressure (psia)	152.2	128.3
LP Turbine Inlet Temperature (°F)	510.8	524.0
S/G Blowdown Flow (GPM)	0.0	329.1
Nuclear Power (%)	100.0	98.29

3.3 PHYSICS TESTING

3.3.1 - INVERSE COUNT RATE RATIO MONITORING, (Initial Criticality Portion) - NUC-111

OBJECTIVE

This permanent plant procedure is performed to obtain and evaluate nuclear monitoring data during the approach to criticality to ensure that the approach is done in a cautious and controlled manner. This procedure satisfies activities described in FSAR Section 14.2.10.2.

TEST METHODOLOGY

Neutron count rate data, as an indicator of core nuclear flux, from both installed source range NIS channels is taken periodically during core reactivity additions. The sources of the core neutron flux are the four installed Californium primary neutron sources with associated subcritical multiplication due to the loaded fuel lattice. As control rods are withdrawn and boron is removed from the RCS water, the core neutron flux and source range count rates increase due to the reduction of these neutron absorbers in the core. If the count rates were to become very large, this would indicate that the reactor was approaching criticality. To determine the effect of a given change on core reactivity, count rate data taken after the neutron absorber decrease is compared to a reference value to evaluate the effect of the neutron absorber decrease. This comparison is performed as a ratio of the count rates to evaluate the fractional change. If this ratio were to be very large, it would indicate that this neutron absorber decrease brought the reactor significantly closer to criticality. For convenience, the procedure evaluates the inverse of the count rate ratios (ICRR) such that an approach to zero would indicate an approach to criticality. Additionally, this procedure trends the inverse count rate ratios and extrapolates the trends to evaluate what additional neutron absorber decrease would be expected to result in criticality. Prior to the start of the approach to criticality, background counts are taken to allow the verification of adequate source range channel signal to noise ratios. This data taken at nominally 557°F in Mode 3 is compared against similar data taken cold, at approximately 115°F, in Mode 6 at the end of core loading. This preliminary cold data was taken in the earlier performance of NUC-111. An equation relating the cold and hot count rates to the signal to noise ratio is supplied by the core designers. This equation is used to verify that a signal to noise ratio of at least two exists. Reference values are redetermined just prior to the start of the dilution with control rod banks withdrawn and also to renormalize the ICRRs. When an ICRR value falls below 0.3, it is renormalized by using the latest average count rate as the new reference value. This effectively resets the

3.3.1 - INVERSE COUNT RATE RATIO MONITORING, (Initial Criticality Portion) - NUC-111 (Continued)

TEST METHODOLOGY (Continued)

ICRR plot to a value of 1.0. Renormalization improves the resolution of the plot and may also be done at the discretion of the test engineer.

Count data is taken periodically during the approach to criticality. Data is taken, the ICRR calculated, plotted, trended and extrapolated as a function of bank withdrawal following each incremental control rod bank withdrawal, as specified by NUC-106. This is every 116 steps for shutdown banks and approximately every 50 steps for the control banks as withdrawn in normal overlap. Count rate data is also taken, and the ICRR calculated, plotted, trended and extrapolated during the dilution to initial criticality. The ICRR values are plotted, extrapolated and evaluated as both a function of elapsed time during dilution and mixings and also as a function of the quantity of reactor makeup water added. The plot versus water added is a good indicator of core condition change as a function of the quantity of dilution (neutron absorber removed) but contains discontinuities at points where the dilution is stopped and the RCS is allowed to mix. The plot versus time does not have mixing discontinuities. For constant rates of dilution, this plot would approximate the ICRR versus quantity of water added plot. A Chi-Squared statistical analysis of the count data is performed to verify data quality. If the Chi-Squared value is unsatisfactory for the three data values taken, then additional data is taken to calculate an average value. Eventually, as the count rate increases when approaching criticality, only one data value is taken and no Chi-Squared analysis is performed.

SUMMARY OF RESULTS

Refer to Figures 3.3.2-1 through 3.3.2-3 for ICRR curves during the approach to initial criticality.

All count rate data was properly recorded and ICRRs were calculated, plotted, trended and extrapolated. The ICRRs show that the approach to criticality was performed in a cautious and controlled manner with no indicated unexpected approaches toward criticality. Large changes in the ICRR plots are due to renormalization.

Monitoring data was properly taken and evaluated during RCS mixing. Renormalizations and reference count rates were properly recalculated. The signal to noise ratios were calculated to be 52.9 for Source Range Channel N31 and 76.9 for Channel N32. These both satisfied the $\geq$2.0 criterion by wide margins.

3.3.2 - INITIAL CRITICALITY - NUC-106

OBJECTIVE

This permanent plant procedure provides a method by which initial criticality is attained in a deliberate and controlled manner. This procedure is used to enter Mode 2 for the first time. The sequence, frequency, and conditions for collection of nuclear data is specified as well as the method of analysis of this data. Criteria for suspending the approach to criticality and for emergency boration are also specified. This procedure satisfies activities described in FSAR Section 14.2.10.2 and Technical Specification 3/4.10.3 and 3/4.1.1.1.

TEST METHODOLOGY

Initial conditions are established with the RCS at an average temperature of approximately 557°F, RCS pressure at approximately 2235 psig, RCS boron concentration greater than 2000 ppm, and all control rod banks fully inserted.

Procedures are initiated to monitor neutron flux, boron concentration and various other plant parameters for the duration of the test.

Reference counts are determined for each source range channel per NUC-111 . These values are used in the ICRR (Inverse Count Rate Ratio) calculations performed following reactivity additions.

Physics Testing is declared to be in progress to permit usage of Technical Specification Special Test Exception 3.10.3 with respect to Mode 2 testing at off normal conditions and also based on the predicted positive MTC.

Shutdown banks are then withdrawn in their normal, alphabetical order. The withdrawals are made in increments of 116 steps or less and the value of the ICRR is determined after each withdrawal, prior to subsequent withdrawals. These ICRR values are plotted against the cumulative shutdown bank position to trend and predict by extrapolation any unexpected approach to criticality. Each bank is withdrawn to an indicated 232 steps, the rod drive step counters are reset to the actual mechanical withdrawal limit of 231 steps, and the bank is reinserted to 228 steps. This sets the control rods to their proper full out heights for monthly control rod repositioning to reduce localized control rod cladding wear.

The Mode 2 entry checklist is verified to be completed and the control banks are then manually withdrawn in their normal overlap configuration, in nominally 50 step increments. Mode 2 is entered with the initial withdrawal of Control Bank A. Control bank withdrawal is completed when Control Bank D is positioned at 160

TEST METHODOLOGY (Continued)

steps. During control bank withdrawal, proper bank overlap and rod insertion limit alarm functions are verified. ICRR monitoring, plotting and extrapolation is also performed as was done for the shutdown banks previously.

The remaining reactivity insertion required to achieve criticality is made by diluting the RCS boron concentration by addition of reactor makeup water to the RCS. Periodic ICRR monitoring during the dilution is performed using NUC-111 to plot, trend, and extrapolate predictions of expected time and quantity of water added for initial criticality. The dilution rate is initially approximately 60 gpm until the ICRR value falls below 0.3, where the dilution is then terminated and the RCS allowed to mix. The ICRR is also renormalized at this point. A dilution at approximately 30 gpm is then started and maintained until the renormalized ICRR value again falls below 0.3 at which time the RCS dilution is again terminated to allow for mixing. Criticality is achieved during this mixing time period, by Control Bank D motion or by small batch water additions. The core flux level is then increased to and stabilized at approximately 10^{-8} amps using Control Bank D motion.

SUMMARY OF RESULTS

The Acceptance Criteria was met in that criticality was achieved within the range of the predicted boron concentration, and the neutron flux level was established within specified bounds on the Intermediate Range NIS channels. The predicted critical boron concentration was 1139 $\pm$50 ppm and the measured value was 1153 ppm. The neutron flux was increased and stabilized at 9×10^{-9} amps and 9.5×10^{-9} amps on the Intermediate Range channels. This was approximately 10^{-8} amps, as required. Shutdown bank withdrawals began at 1803 hrs on 4-2-90 and were completed at 2000 hrs. Mode 2 was entered at 2101 hours on 4-2-90 with Technical Specification Special Test Exception 3.10.3 invoked at 2109 hrs. The 60 gpm RCS dilution was started at 0209 hrs on 4-3-90. This initial dilution was terminated at 1448 hrs on 4-3-90 and, after mixing, a 30 gpm dilution was started at 1616 hrs. The ICRR plot at that time indicated that the addition of 2200 gallons of water was necessary to achieve criticality. 1000 gallons of water were added by 1650 hrs and allowed to mix. Following a 37 minute mixing time, the addition of another 1200 gallons was started at 1727 hrs. Initial criticality was achieved at 1742 hrs on 4-3-90. The final 1200 gallon dilution was terminated early, with only approximately 450 gallons actually added. The flux was stabilized at approximately 10^{-8} amps at 1806 hrs on 4-3-90.

3.3.2 - INITIAL CRITICALITY - NUC-106 (Continued)

SUMMARY OF RESULTS (Continued)

Three minor problems occurred during the approach to criticality, none of which involved any unexpected core reactivity responses.

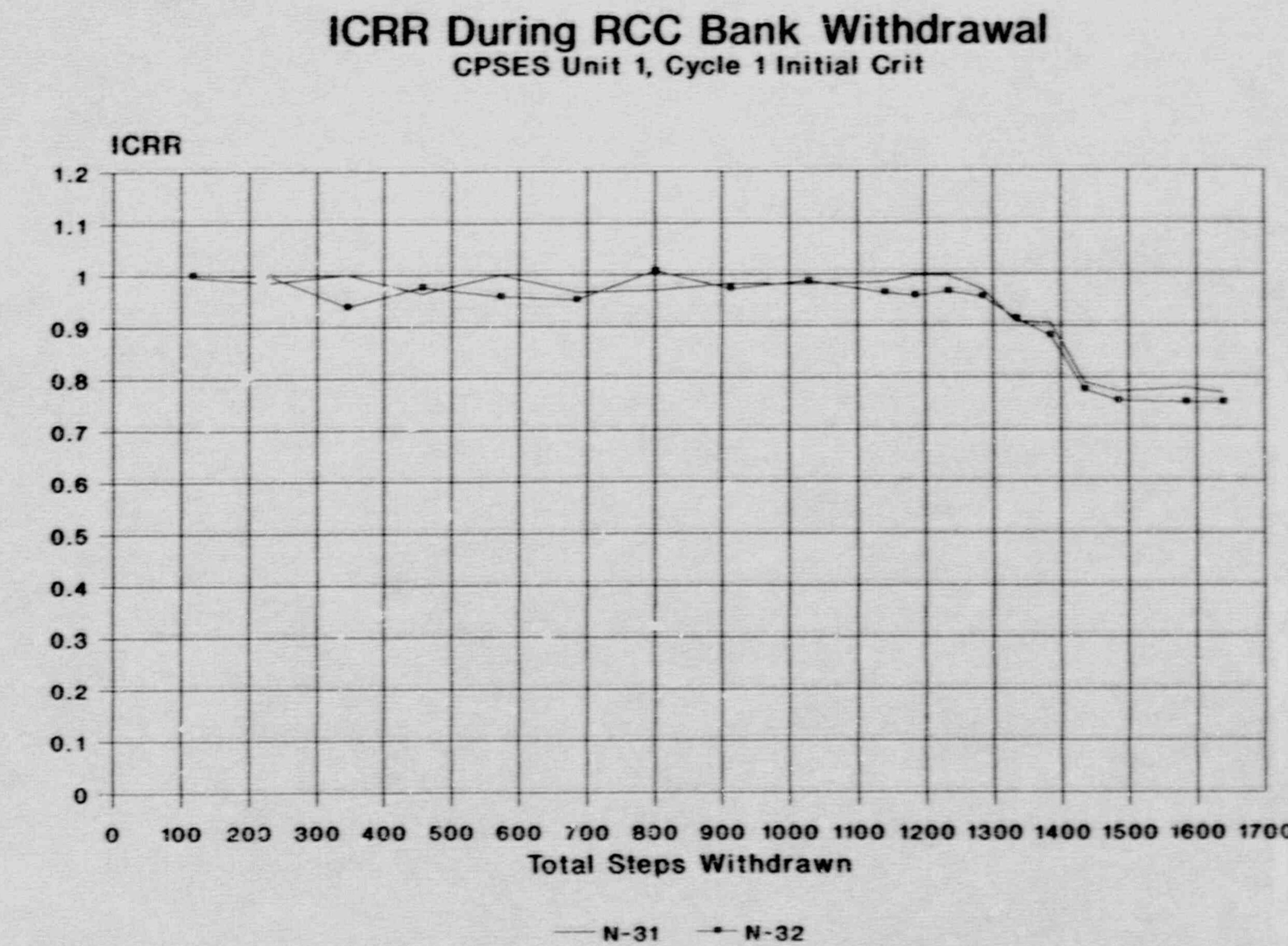
Noise in the Power Range channel N44 signal used by the reactivity computer was traced to the N-16 circuitry Power Range Module. The noise amplitude was low enough to be insignificant with respect to this circuitry's function during normal operations but was large enough to interfere with this signal application for physics test measurements. The Power Range Module was de-energized and the noise was eliminated.

The Rod Insertion Limit alarms did not clear when expected for Control Banks C and D. These alarms were declared inoperable and were later recalibrated by Instrumentation and Control. The alarms cleared at higher rod bank positions than expected which was conservative. The alarms were not needed for this criticality approach because the rod bank positions established prior to the dilution were well above these limits.

During the dilution, RCS and pressurizer boron samples deviated by more than the ±50 ppm difference criterion originally used. This was caused by mixing time delay between the RCS and pressurizer combined with sampling purge delays. The criterion was changed to + 200, - 50 ppm to account for the pressurizer lagging the RCS during dilution. This did not present a safety concern because the pressurizer boron concentration was always higher than the RCS boron concentration, as would be expected.

Refer to Figures 3.3.2-1 through 3.3.2-3 for plots of ICRR versus control and shutdown bank withdrawals, reactor makeup water addition and time.

Figure 3.3.2-1



ARO Pos=228 steps, Overlap=115 steps

Figure 3.3.2-2

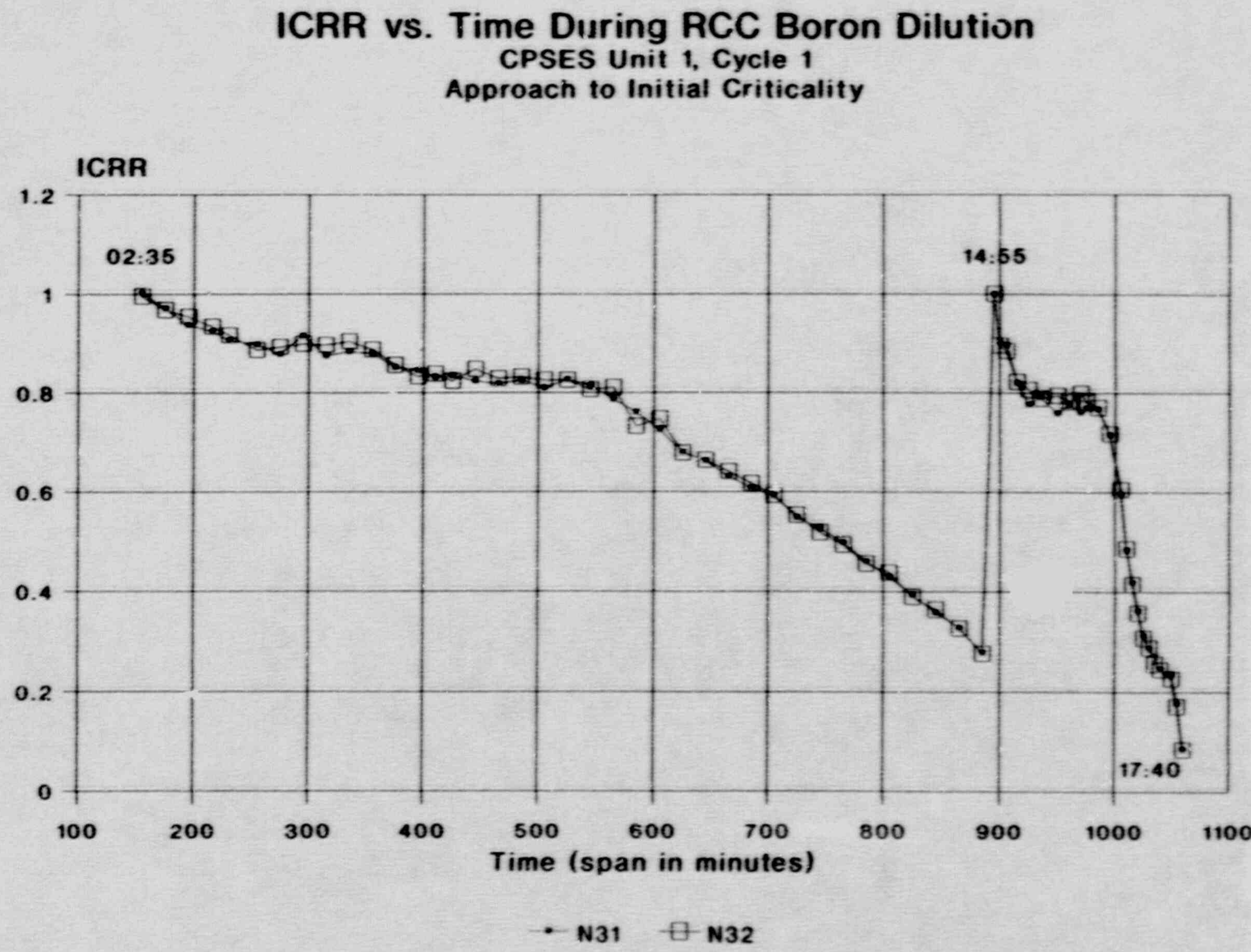
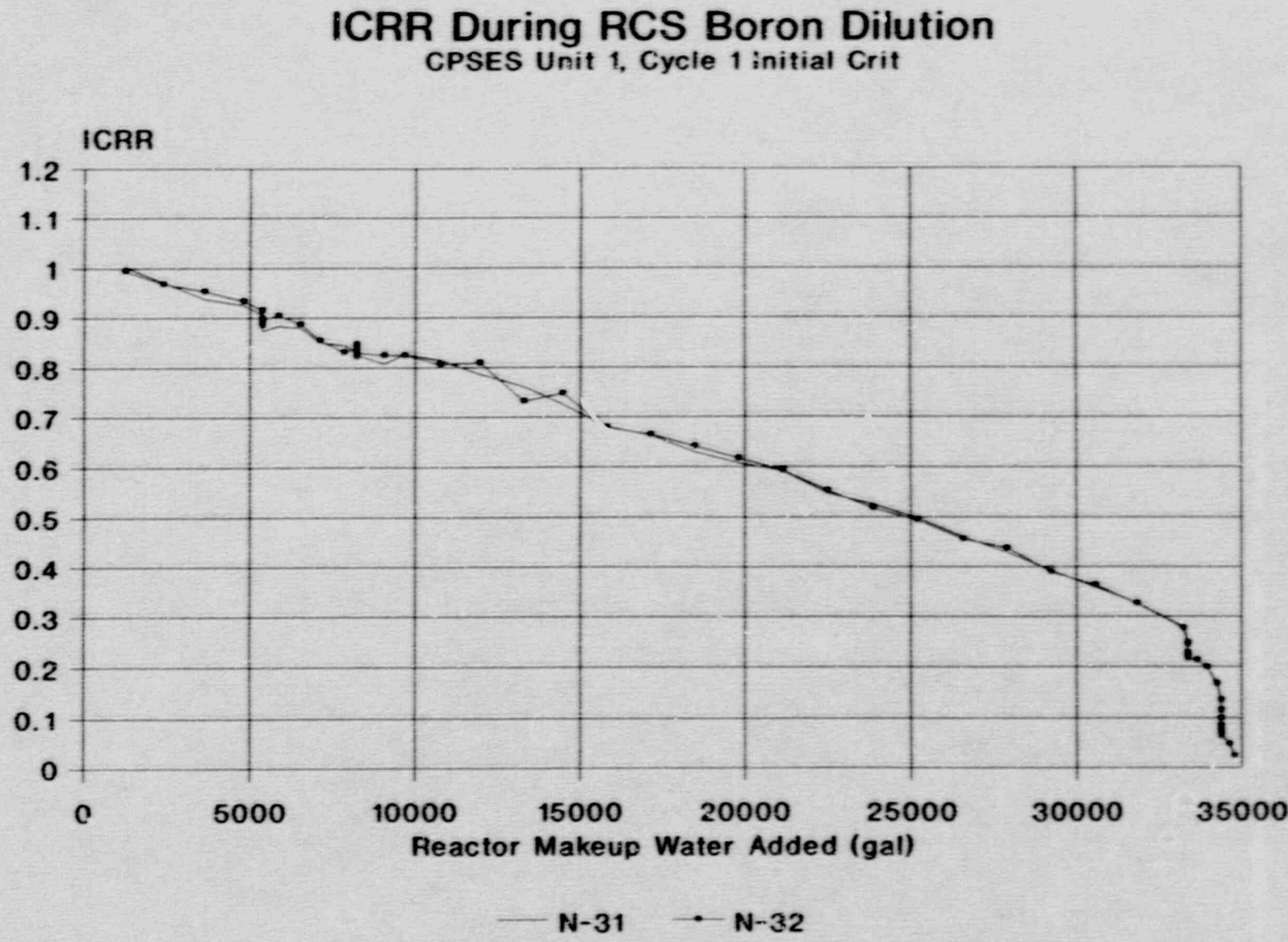

ICRR vs. Time During RCC Boron Dilution
CPSES Unit 1, Cycle 1
Approach to Initial Criticality
ICRR
1.2
1
0.8
0.6
0.4
0.2
0
02:35
14:55
17:40
100
200
300
400
500
600
700
800
900
1000
1100
Time (span in minutes)
N31
N32

Figure 3.3.2-3


ICRR During RCS Boron Dilution
CPSES Unit 1, Cycle 1 Initial Crit
ICRR
1.2
1.1
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
0
5000
10000
15000
20000
25000
30000
35000
Reactor Makeup Water Added (gal)
N-31
N-32

3.3.3 - DETERMINATION OF CORE POWER RANGE FOR PHYSICS TESTING, NUC-109

OBJECTIVE

This permanent plant procedure is used to determine the power level (neutron flux level) at which detectable reactivity feedback effects from nuclear fuel heating occur and to establish the range of neutron flux in which zero power reactivity measurements are performed to avoid interference with these feedback effects.

TEST METHODOLOGY

Initial conditions are established with the RCS at an average temperature of approximately 557°F, RCS pressure at approximately 2235 psig and the reactor critical with flux at approximately 10^{-8} amps on both Intermediate Range channels. Control Bank D is positioned such that approximately 40 pcm of worth remains available to increase core reactivity.

Initially, the reactivity computer is set up using the power range channel N-44 detector, which was taken out of service. Reactivity computer outputs of reactivity and flux along with RCS cold leg temperature are displayed on strip chart recorders. The temperature input is from the process instrumentation racks.

The determination of the power range for physics testing is made by withdrawing Control Bank D to achieve a positive reactivity addition of 30 ± 10 pcm. Reactivity and flux level are then observed to determine the point of adding nuclear heat as indicated by negative reactivity addition from the Doppler fuel temperature coefficient. RCS temperatures are also monitored for an increase as an indication of nuclear heating. The flux is then reduced back to approximately 10^{-8} amps and the measurement is repeated, at least once, to confirm the value.

SUMMARY OF RESULTS

Two measurements were performed. The reactor power level at which detectable reactivity feedback effects from nuclear heating occurred was determined to be 1 X 10^{-6} amps on both Intermediate Range (IR) NIS channels and 1.3 X 10^{-6} amps on the reactivity computer picoammeter. These values are from the second measurement which was more refined than the first. During the first measurement, the point of adding heat is completely unknown. It is common to actually overshoot the value on this first run. However, even if overshot, the first run does yield an approximate value of the point of adding heat such that the second measurement can begin with a good prediction of where the point of adding heat lies. The

SUMMARY OF RESULTS (Continued)

first measurement resulted in values of 1.1 X 10^{-6} amps (IR Channel N35), 1.3 X 10^{-6} amps (IR Channel N36) and 1.65 X 10^{-6} amps (reactivity computer picoammeter) for the point of nuclear heat addition.

The neutron flux level range at which zero power reactivity measurements were to be performed was determined to be 1 X 10^{-8} to 1 X 10^{-7} amps as indicated on the reactivity computer. The range of neutron flux levels allowed for physics testing that was actually trended and used applied to the reactivity computer picoammeter.

3.3.4 - REACTIVITY COMPUTER CHECKOUT - NUC-108

OBJECTIVE

This permanent plant procedure is performed to demonstrate proper operation of the reactivity computer through dynamic testing using actual neutron flux signals and core reactivity changes. This ensures that the reactivity computer is operating properly before it is used to measure reactor physics parameters.

TEST METHODOLOGY

A reactivity increase of approximately 25 pcm, as shown on the reactivity computer strip chart, is initiated by withdrawal of Control Bank D. A stopwatch is used to measure the reactor period time. This period, P, is the time interval over which indicated core flux increases by a factor of e, with flux increasing on a stable period, after the dampening of initial transient effects. The period comes from the following equation in terms of the initial flux, ϕ_i , final flux, ϕ_f , and measured time interval, t:

$$P = t/\ln \frac{\phi_f}{\phi_i}.$$

This measured period is used to determine the theoretical reactivity increase using core design report predictions of reactivity as a function of reactor period. This prediction is given by the inhour equation using core physics constants from the core design report.

The predicted reactivity increase is compared to the reactivity indicated on the reactivity computer strip chart. This measurement may be repeated for reactivity increases of up to approximately +50 pcm. A negative reactivity insertion of up to -20 pcm may also be optionally performed.

SUMMARY OF RESULTS

Two runs were made, one each at approximately +25 and + 50 pcm. No negative reactivity insertion runs were made. The acceptance criterion for this measurement is that the average of the absolute

3.3.4 - REACTIVITY COMPUTER CHECKOUT - NUC-108 (Continued)

SUMMARY OF RESULTS (Continued)

values of the reactivity differences be less than ±4%. The results were as follows:

Approximate Reactivity Insertion (pcm)	Measured Reactor Period (seconds)	Predicted Reactivity Based on Period (pcm)	Indicated Reactivity from Computer (pcm)	Absolute Value of %Difference
25	298.65	25.2	25.0	0.80
50	141.5	48.3	48.7	0.82
		Average % Difference		0.81

The average difference of +0.81% satisfied the < ±4% criterion.

3.3.5 - CORE REACTIVITY BALANCE - NUC-205

OBJECTIVE

The purpose of this permanent plant procedure is to verify the design predictions of core reactivity during the power ascension startup testing sequence. This procedure satisfies activities described by FSAR Table 14.2-3, Sheet 17.

TEST METHODOLOGY

This core reactivity verification is performed by comparing reactor criticality parameters at zero power with those at full power. Parameters measured include control bank positions, RCS temperature, RCS boron concentration, power level and core burnup. After compensating for differences in control bank position, boron concentration, reactor power and Xenon and Samarium buildup, with respect to predicted core conditions, the actual critical boron concentration present is compared to the design prediction. This verifies the accuracy of the design predictions of core reactivity.

SUMMARY OF RESULTS

The Hot Zero Power, All Rods Out, Xenon and Samarium free critical boron concentration was 1162.1 ppm. The Hot Full Power, All Rods Out, equilibrium Xenon and Samarium critical boron concentration was 754.1 ppm at an average RCS temperature of 589 °F and 999 MWD/MTU burnup. This represents a decrease of 408 ppm in boron concentration to get from Hot Zero Power, All Rods Out and no fission product poisons to just critical at Hot Full Power, All Rods Out and equilibrium Xenon and Samarium. For a 1000 MWD/MTU burnup, essentially the same as the 999 MWD/MTU burnup as tested, the predicted Full power value was 743 ppm. Using the predicted just critical Hot Zero Power, All Rods Out, no fission product poison value of 1146 ppm yields a predicted difference of 1146 - 743 = 403 ppm. There is no specific criterion for this agreement, but 5 ppm indicates that the core reactivity change was very close to the design predictions. This indicated that the design predictions of Xenon and Samarium buildup and power defect were valid.

3.3.6 - SURVEILLANCE OF CORE POWER DISTRIBUTION FACTORS - NUC-201

OBJECTIVE

The purpose of this permanent plant procedure is to evaluate reactor core power distribution factors based on incore flux map results. This procedure partially satisfies activities described in FSAR Table 14.2-3, Sheets 20-22, Section 14.2.10.4 and Technical Specifications 3/4.2.2 and 3/4.2.3.

TEST METHODOLOGY

The results from an incore flux map are processed using the CONFORM core physics code. The axially varying heat flux hot channel factor, FQ(Z), is evaluated by inspection of the CONFORM code results for the maximum FQ(Z). Values from portions of the core known to have relatively uncertain results are eliminated and the remaining FQ(Z) value closest to the FQ(Z) limit is then multiplied by 1.05 and 1.03 to account for manufacturing tolerances and measurement uncertainties. The portions of the core eliminated are the top and bottom 15% of the core, regions within $\pm 2\%$ of grid straps and the region within $\pm 2\%$ of the Control Bank D RCCA tips. This maximum FQ(Z) result is compared against the appropriate limit, $\leq 2.32/P$ for $P>0.5$ and ≤ 4.64 for $P \leq 0.5$, multiplied by the factor K(Z) from Technical Specification Figure 3.2-2 evaluated at the same Z core height as the FQ(Z) value. P is the fractional thermal power.

The radial peaking factor, Fxy, is evaluated by inspection of the CONFORM code results for the maximum Fxy, eliminating values from the same portions of the core as above for FQ(Z) and again by multiplying by 1.05 and 1.03. This maximum Fxy result is compared against a power varying limit of $\leq$Fxy RTP (1 + 0.2(1-P)), where P is fractional thermal power and Fxy RTP is defined in the Radial Peaking Factor Limit Report required by Technical Specification 6.9.1.6. For this initial startup, Fxy RTP was 1.55 for unrodded core portions and 1.71 for rodded core portions.

The evaluations of FQ(Z) and Fxy are performed in conjunction with the appropriate sequence document and may also be used to satisfy the Surveillance Requirements for Technical Specification 3/4.2.2.

The nuclear enthalpy rise hot channel factor, FDHN, is evaluated by inspection of the CONFORM code results for the maximum FDHN value and multiplying it by 1.04 for measurement uncertainty. This maximum FDHN result is compared against the limit of $\leq$1.55 (1 + 0.2(1-P)), where P is fractional thermal power. This FDHN evaluation is done in conjunction with the appropriate sequence document and may also be used to satisfy the surveillance requirements for Technical Specification 3/4.2.3.

3.3.6 - SURVEILLANCE OF CORE POWER DISTRIBUTION FACTORS - NUC-201 (Continued)

SUMMARY OF RESULTS

This procedure was performed as necessary to support sequence document performances throughout power ascension. The required power plateau maps were those flux maps taken at approximately 0%, 50%, 75% and 100% power. The results were recorded in the test summaries in Section 2 of this report. All performances were satisfactory. Some performances were also used to satisfy Technical Specification requirements as well as criteria contained in the sequence documents.

3.3.7 - ZERO POWER ISOTHERMAL AND MODERATOR TEMPERATURE COEFFICIENT MEASUREMENTS - NUC-207

OBJECTIVE

This permanent plant procedure is performed to determine the Isothermal Temperature Coefficient (ITC) of reactivity and to derive, from this, the Moderator Temperature Coefficient (MTC) of reactivity at the beginning of core life. This procedure satisfies activities described by FSAR Table 14.2-3, Sheet 14 and Technical Specification 3/4.1.1.3.

TEST METHODOLOGY

The ITC is determined by measuring the change in reactivity induced by changing the temperature of the moderator, cladding and fuel and dividing by the temperature change. The MTC is obtained by analytically removing a precalculated Doppler broadening coefficient factor from the ITC value to eliminate the fuel temperature change portion of the ITC.

A voltage signal proportional to core reactivity is obtained from the reactivity computer output and a voltage signal proportional to RCS cold leg temperature is obtained from the process instrumentation racks. These signals are input to an X-Y plotter such that the slope of the X-Y plot corresponds to the ITC, change in core reactivity per unit change in RCS temperature. RCS temperature is slowly changed by manipulation of the rate of heat removal from the RCS by the secondary plant. The resulting reactivity as a function of the varying temperature is plotted and evaluated. By changing the RCS temperature slowly, the fuel, cladding and moderator temperatures all change at the same rate, nearly isothermal, with minimal temperature gradients. This measurement result must be adjusted to eliminate the effect of Doppler resonance peak broadening in the fuel to yield the effect of the moderator alone. The effect of the cladding is negligible in this analysis. The slow RCS temperature change permits the fuel temperature to change uniformly, isothermal, without the heat transfer that would result in a non-linear temperature profile across the fuel pellets. Because of this, the fuel temperature at a given time is essentially the same as the RCS temperature. This allows a fuel type and enrichment specific calculation of the Doppler broadening effect to be performed for the temperature regime at which the test is executed. Variable fuel temperature distributions would render analysis of the isothermal temperature coefficient impossible. The Doppler broadening coefficient of -1.83 pcm/°F is subtracted from the ITC value to result in the MTC. The measured ITC values are evaluated to verify they are within ±1 pcm/°F of each other, to demonstrate data consistency, and the

3.3.7 - ZERO POWER ISOTHERMAL AND MODERATOR TEMPERATURE COEFFICIENT MEASUREMENTS - NUC-207 (Continued)

TEST METHODOLOGY (Continued)

average ITC value is verified to be within ±3 pcm/°F of the predicted ITC value of -1.4 pcm/°F. The MTC is verified to be ≤0 pcm/°F or, if not, Rod Withdrawal Limits, using NUC-116, are imposed to ensure that the MTC is maintained ≤0 by operational controls.

This test is performed from Hot Zero Power conditions, nominally 557°F, starting with a Reactor Coolant System (RCS) cooldown of approximately 3°F at a rate of approximately 10°F/hr. After a stabilization period at this lower temperature, an RCS heatup is then initiated for an approximate 3°F increase, also at a rate of approximately 10°F/hr. A plot of reactivity vs. temperature is made for both the cooldown and heatup portions of the test. The cooldown and heatup are performed at the All Rods Out (Control Bank D > 200 steps) control rod configuration. Multiple cooldown and heatup cycles may be performed, if required for data consistency.

SUMMARY OF RESULTS

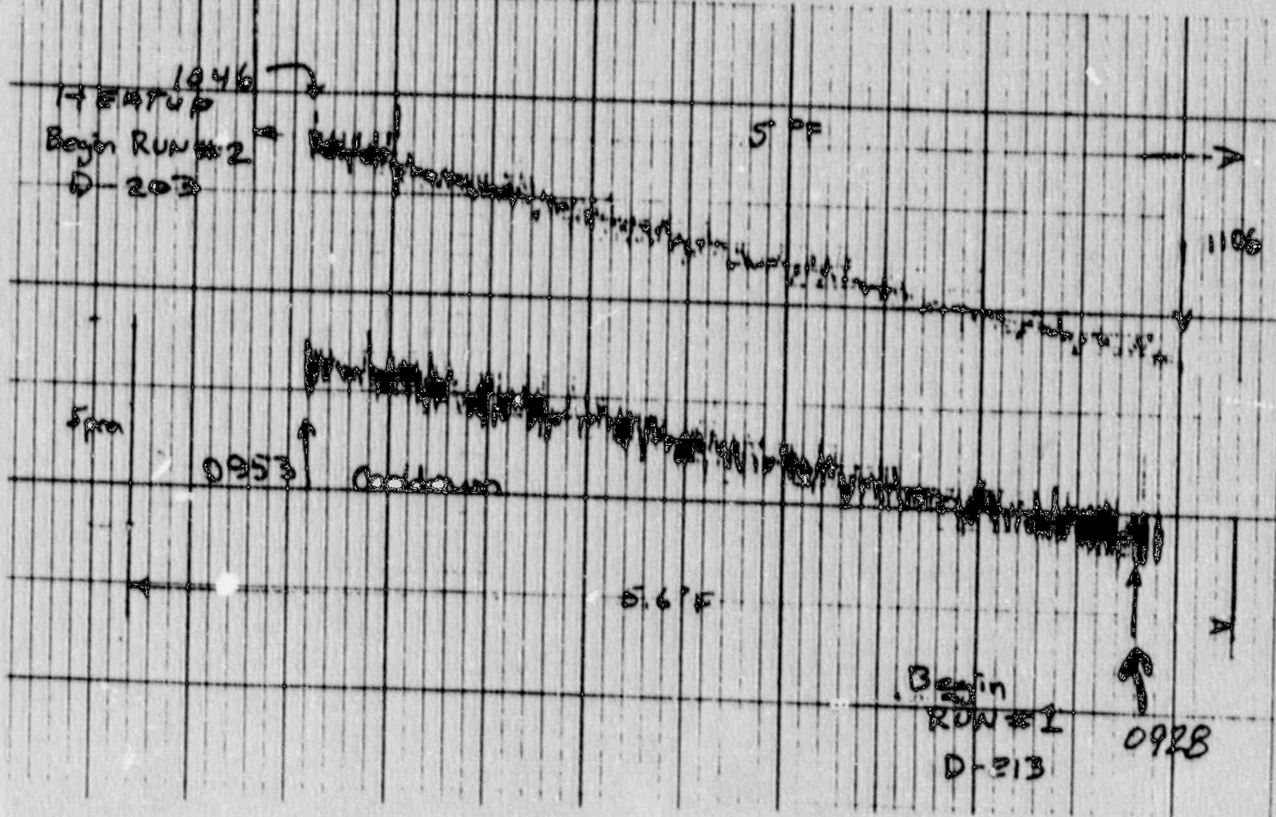
The cooldown resulted in an ITC of -0.89 pcm/°F. The heatup resulted in an ITC of -1.10pcm/°F. Only one cooldown and heatup cycle was performed. The average ITC was -0.995 pcm/°F. The ±1 pcm/°F criterion between the cooldown and heatup values was satisfied as they differed by only 0.21 pcm/°F. The ±3 pcm/°F criterion between average measured ITC and the prediction of -1.4 pcm/°F was satisfied as they differed by only 0.405 pcm/°F. The calculated MTC value of -0.995 -(-1.83) = +0.835 pcm/°F did not satisfy the ≤ 0 pcm/°F criterion so Rod Withdrawal Limits were calculated and imposed using NUC-116. The calculated MTC value was within the measurement tolerance of the design value and the positive value was not unexpected.

Control Bank D was at >200 steps during this test performance with all other control rods fully withdrawn. Core neutron flux was maintained below the point of nuclear heat addition to preclude nuclear heating feedback effects from invalidating test results. Refer to Figure 3.3.7-1 for the plot of reactivity vs. temperature.

Figure 3.3.7-1
Reactivity vs. Temperature

Reactivity

pcm



Temperature

°F

3.3.8 - DETERMINATION OF OPERATING LIMITS TO ENSURE A NEGATIVE MTC - NUC-116

OBJECTIVE

This permanent plant procedure is performed to establish operating limits, also called Rod Withdrawal Limits, to ensure that the Moderator Temperature Coefficient (MTC) remains negative. It is only performed if the All Rods Out MTC value measured in NUC-207 is found to be positive. This procedure satisfies activities described by FSAR Table 14.2-3, Sheet 14.

TEST METHODOLOGY

Boron, a neutron absorber, is dissolved in the RCS water, which also serves as the reactor moderator. As moderator temperature increases, the moderator becomes less dense and is less efficient at slowing down fission neutrons. The effect is to add negative reactivity and cause the reactor neutron flux to decrease. But, as the moderator density decreases so does the density of the soluble boron. This boron density decrease reduces the number of parasitic neutron captures by the boron and effectively adds positive reactivity which causes the reactor neutron flux to increase. These conflicting effects can either cancel each other out or the balance will shift one way or the other. As boron concentration increases, the net effect is a positive reactivity addition with an increase in moderator temperature.

For reactor stability considerations, the MTC is restricted to non-positive values only, such that an increase in moderator temperature results in addition of negative reactivity which tends to shut the reactor down. If the measured MTC is positive, measures must be taken to limit it to negative values.

This is done by limiting the maximum moderator boron concentration to values which ensure a negative MTC. This can be achieved by operating control rods partially inserted (i.e. limiting their withdrawal) such that the boron concentration is reduced to maintain criticality. This procedure generates a family of curves which relate permitted Control Bank D positions as functions of reactor power level and boron concentration. The curves are generated using reactor core designer supplied methodology and are based on the actual measured MTC, actual All Rods Out endpoint boron concentration and design predictions of control bank worth, boron concentrations and MTCs.

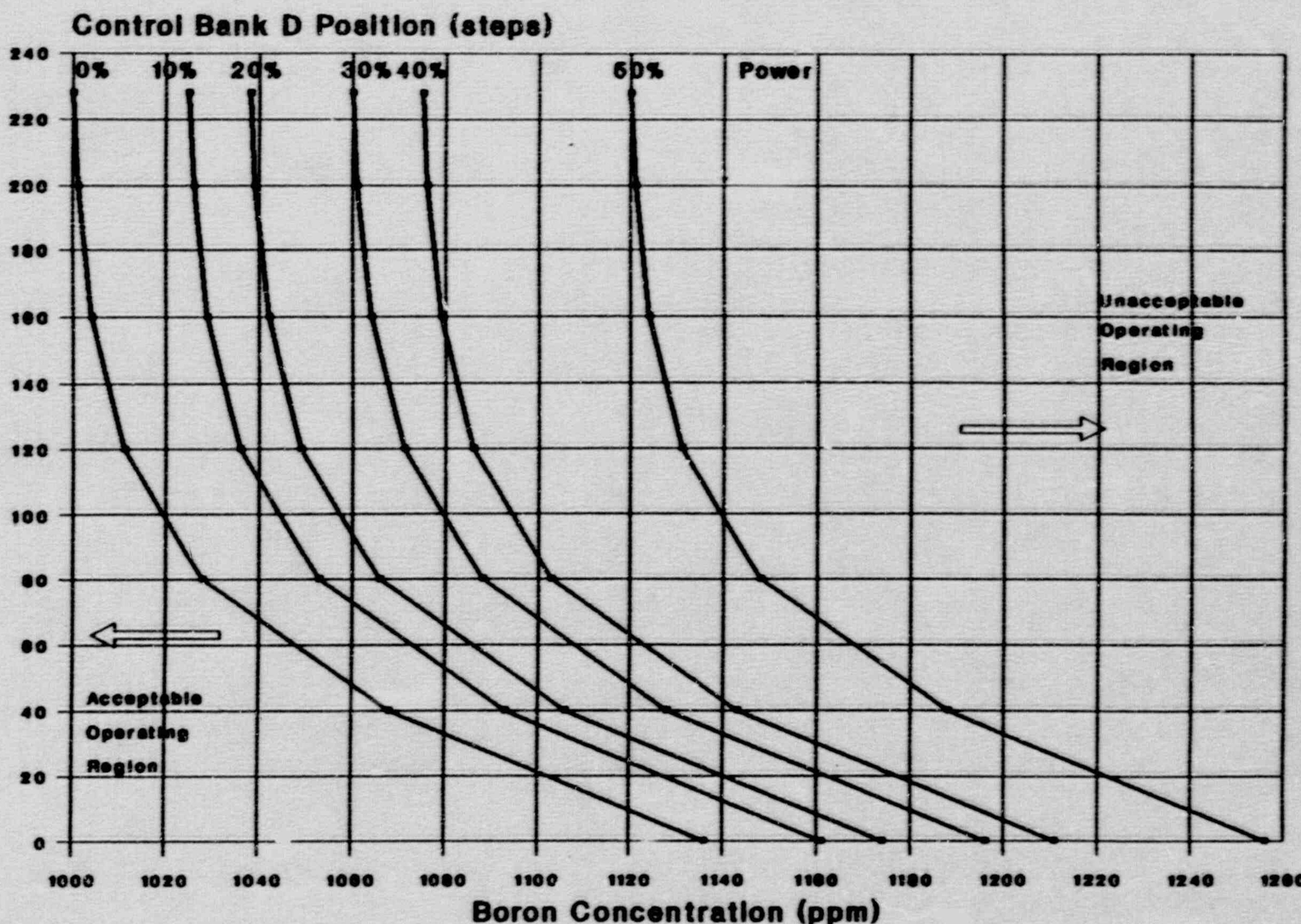
3.3.8 - DETERMINATION OF OPERATING LIMITS TO ENSURE A NEGATIVE MTC - NUC-116 (Continued)

SUMMARY OF RESULTS

The MTC in NUC-207 was + 0.835 pcm/°F so performance of this procedure was required. Rod Withdrawal Limit curves were properly generated and implemented, refer to Figure 3.3.9-1. Note that Control Bank D is allowed to be withdrawn further as power is increased. This is due to the normal reduction of the boron concentration to compensate for power defect as power is increased.

Figure 3.3.8-1

Rod Withdrawal Limits



3.3.9 - ROD SWAP MEASUREMENT - NUC-120

OBJECTIVE

This permanent plant procedure is performed to verify that the differential and integral worth of individual control rod banks agree with the design predictions made in the core design report. This procedure also measures differential boron worth. This procedure partially satisfies activities described by FSAR Table 14.2-3, Sheets 15 and 16.

TEST METHODOLOGY

The rod swap method, also called the bank exchange method, of determining bank worth only directly measures one bank, this bank is designated the reference bank, and infers the worth of the other banks based on the reference bank worth. The reactor core designers selected Shutdown Bank B as the reference bank for this initial fuel cycle. Starting with the reactor critical and Control Bank D partially inserted to adjust core neutron flux, Control Bank D is fully withdrawn. The reference bank, Shutdown Bank B, is then immediately inserted to restore the core to the just critical condition. An endpoint measurement is made for the currently inserted worth of the reference bank by fully withdrawing it, measuring the core reactivity change with the reactivity computer, and reinserting the bank to the just critical position. Next, an RCS dilution at a rate of approximately 25 gpm is started. This is roughly equivalent to a 300 pcm/hour reactivity addition rate. This positive reactivity addition is compensated by periodic insertions of the reference bank in -10 to -20 pcm increments in order to maintain the just critical reactivity condition. The individual worth of these incremental insertions is measured with the reactivity computer. The dilution is suspended with the reference bank near fully inserted. Another endpoint measurement is made for the last portion of reference bank worth by inserting the reference bank to core bottom and measuring this worth with the reactivity computer. The sum of the incremental worths and the two endpoint worths is the integral worth of the reference bank. The differential worth is calculated as the incremental worth divided by the number of rod steps moved to result in that reactivity change. Both of these worth values are recorded and plotted.

With the reference bank nearly fully inserted and all other rod banks fully withdrawn, the integral worth of the other rod banks are individually verified by comparing their relative worths with respect to that of the just measured reference bank. A selected test bank is inserted to result in approximately 20 pcm of negative core reactivity. The reference bank is then immediately withdrawn to result in approximately 20 pcm of positive core reactivity. This process is repeated until the test bank is fully inserted and the reference bank is adjusted to a just critical position. The

3.3.9 - ROD SWAP MEASUREMENT - NUC-120 (Continued)

TEST METHODOLOGY (Continued)

worth of the test bank is then inferred as being equal to the fractional portion of the reference bank worth that was withdrawn to compensate for the test bank's insertion. The test and reference banks are returned to their initial positions, reference bank in and the test bank fully withdrawn, in the reverse order of steps used for the measurement. The same process is repeated for each remaining bank until all banks have been exchanged, or swapped, against the reference bank.

Following the exchange of all banks against the reference bank, core conditions are restored in one of two ways. If Rod Withdrawal Limits are not to be imposed as a result of the MTC measurement, an RCS boration is started and the reference bank is withdrawn to compensate for this negative reactivity addition until it is fully withdrawn. If Rod Withdrawal Limits are to be imposed, the reference bank is exchanged against Control Banks D and C until the reference bank is fully withdrawn.

Differential boron worth is obtained by dividing the total worth of the reference bank by the difference of the two endpoint boron concentrations to yield pcm/ppm.

SUMMARY OF RESULTS

The reference bank measured and predicted worth were to differ by no more than ±7%. They differed by only -0.4%. The remaining test banks measured and predicted worths were to differ by no more than ±10% or ±100 pcm, whichever was greater. Three banks, Shutdown Banks A and C and Control Bank A, had differences of greater than 10%. However, the difference did not exceed ±100 pcm for any of these banks. The other banks, which met the ±10% criterion, automatically met the ±100 pcm criterion because no bank had a predicted worth of more than 1000 pcm.

The percent error between measured and predicted total worth was 1.8%. This satisfied the criterion that the measured total worth be within ±7% of the predicted total worth. The best individual bank agreement was found in the measurement of Shutdown Bank B which was within -0.4% of its predicted value. The worst agreement was found in the measurement of Control Bank A which was within -13.4% of its predicted value. A summary of the bank worth measurements and the predicted values appears in Table 3.3.9-1 and Figure 3.3.9-1 is a plot of the integral and differential reference bank worths.

3.3.9 - ROD SWAP MEASUREMENT - NUC-120 (Continued)
SUMMARY OF RESULTS (Continued)

Rod Withdrawal Limits had been imposed by NUC-116, so Shutdown Bank B was swapped against Control Banks C and D to restore core conditions.

No problems were encountered during field performance of this test. However, two items occurred pertaining to the calculations performed. The methodology used in the calculations of inferred bank worths differed from the original Westinghouse methodology. The Westinghouse method uses the average of the initial and final inserted reference bank positions in the inferred worth calculation to account for any net core reactivity drift due to outside influences over the course of the rod swap measurement. NUC-120 used only the initial reference bank position in this calculation. The inferred worths were recalculated using the Westinghouse method and documented in accordance with plant procedures. The changes to the resulting values were so small as to have no adverse impact on test results acceptability. These recalculated values were those tabulated in Table 3.3.9-1.

The other item pertains to the differential boron worth calculation. The calculated value from core physics measurements was -11.63 pcm/ppm. The expected value was -10.44 pcm/ppm ±10%. However, this -10.44 pcm/ppm value was not specifically calculated for this core configuration by the reactor designer. It was calculated based on predictions of the All Rods Out (ARO) and Reference Bank In boron concentrations and the reference bank worth contained in the core design report.

The Shutdown Bank B In boron concentration of 1062 ppm was calculated using the rod swap model which is different from the model used to generate the CPSES Unit 1, Cycle 1 boron endpoints (ARO critical boron concentration of 1146 ppm). The rod swap method gave an ARO critical boron concentration of 1139 ppm. Based upon use of a consistent model, the predicted differential boron worth would be:

Rod Swap Model Predicted Reference Bank In Boron Endpoint	1062	ppm
Rod Swap Model Predicted ARO Boron Endpoint	1139	ppm
Predicted Reference Bank Worth	877.1	pcm

$$\text{Revised Predicted Differential Worth} = \frac{877.1 \text{ pcm}}{1062 \text{ ppm} - 1139 \text{ ppm}} = -11.39 \text{ pcm/ppm}$$

Using this consistent model prediction, the revised predicted differential worth acceptance range would have been -11.39 pcm/ppm ±10%. The measured value differs from this revised prediction by only 2.1%.

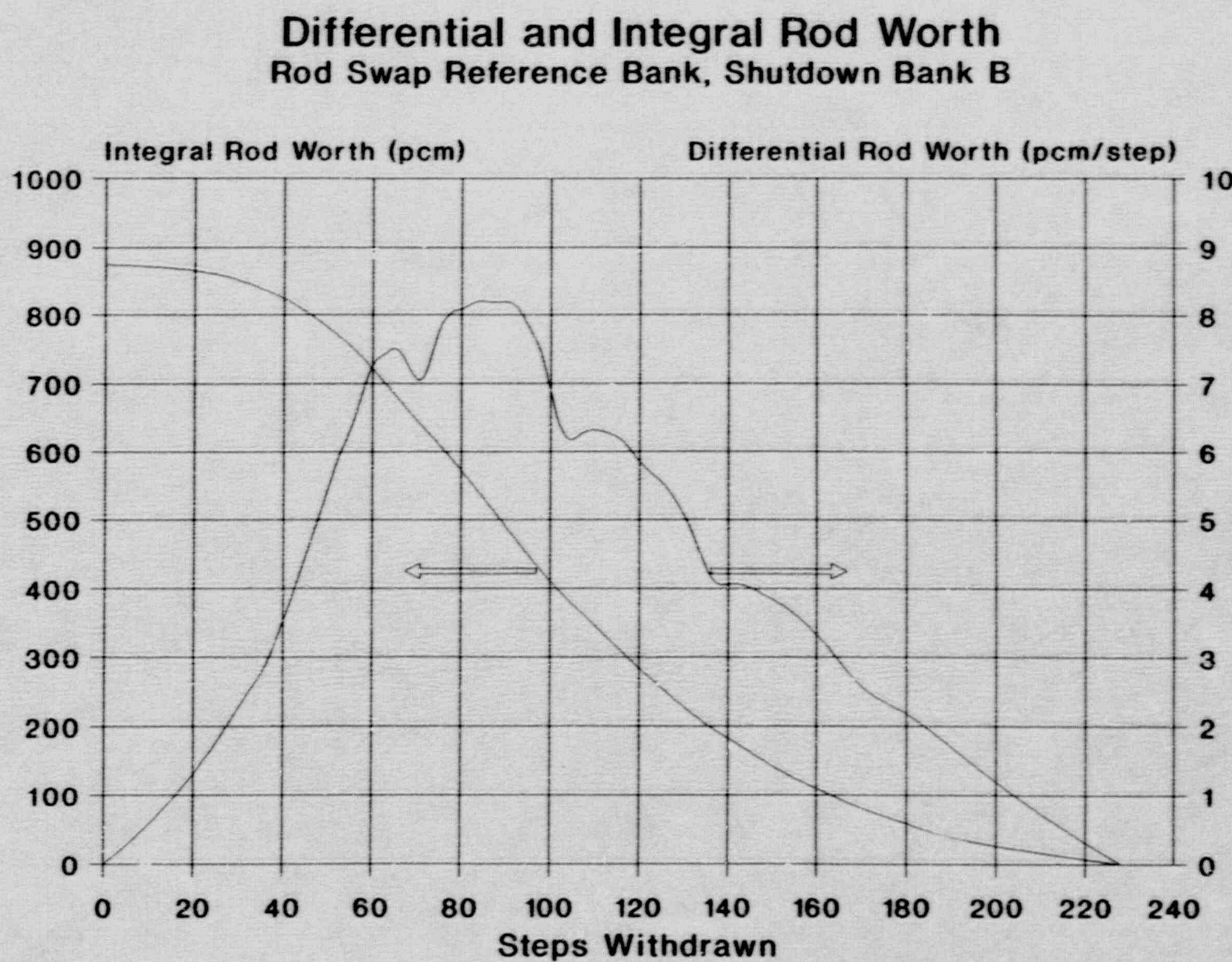
TABLE 3.3.9-1

Measured and Inferred Versus Predicted Control Rod Bank Worth

BANK	MEASURED/ INFERRED WORTH(pcm)	PREDICTED WORTH(pcm)	ABSOLUTE DIFFERENCE(pcm)	% Difference
Shutdown A	591.6	524.8	66.8	12.7
Shutdown B*	873.3	877.1	-3.8	-0.4
Shutdown C	468.4	425.2	43.2	10.2
Shutdown D	461.3	425.2	36.1	8.5
Shutdown E	459.1	487.9	-28.8	-5.9
Control A	301.9	348.5	-46.6	-13.4
Control B	816.4	767.2	49.2	6.4
Control C	824.1	853.1	-29.0	-3.4
Control D	662.7	654.4	8.3	1.3
Total	5458.8	5363.4	+95.4	+1.8

* Shutdown Bank B was the reference bank and has a measured worth. All other banks have inferred worths.

Figure 3.3.9-1


Differential and Integral Rod Worth
Rod Swap Reference Bank, Shutdown Bank B
Integral Rod Worth (pcm)
Differential Rod Worth (pcm/step)
1000
900
800
700
600
500
400
300
200
100
0
10
9
8
7
6
5
4
3
2
1
0
0
20
40
60
80
100
120
140
160
180
200
220
240
Steps Withdrawn

3.3.10 - BORON ENDPOINT DETERMINATION AND DIFFERENTIAL BORON WORTH - NUC-104

OBJECTIVE

This permanent plant procedure is performed to determine rod worth at the extreme ends of rod bank travel, at the near fully withdrawn or near fully inserted positions. In addition, the just critical, All Rods Out (ARO), Reactor Coolant System (RCS) boron concentration is determined. This procedure partially satisfies activities described by FSAR Table 14.2-3, Sheet 16.

TEST METHODOLOGY

The test starts with RCS temperatures and boron concentration verified stable and all control rods withdrawn except for Control Bank D, which is controlling flux at the just critical condition. With no more than approximately 50 pcm of Control Bank D worth inserted, Control Bank D is then fully withdrawn to reach the desired ARO endpoint configuration while neutron flux, reactivity, RCS temperature and pressurizer level are monitored on strip chart recorders. When the reactivity trace stabilizes, Control Bank D is repositioned to re-establish the initial flux level and core reactivity. This process is repeated at least two more times. The endpoint boron concentration is obtained by dividing the measured reactivity change due to Control Bank D withdrawal by the design prediction for differential boron worth at this particular rod bank configuration. This converts the measured pcm of reactivity worth to ppm of equivalent boron worth. This boron worth value is combined with the actual measured boron concentration to yield the boron concentration that would exist with Control Bank D fully withdrawn and the reactor just critical. This is a calculated method which replaces the alternative of actually adding small quantities of boron to the RCS until all of the control rods are fully withdrawn with the core at the just critical condition.

Differential boron worth over a particular bank is obtained by dividing the total integral worth of the selected rod bank by the difference in the endpoint boron concentrations, one for the bank fully withdrawn and one for the bank fully inserted. This results in a value of pcm/ppm and is always negative.

SUMMARY OF RESULTS

The All Rods Out just critical RCS boron concentration was found to be 1162.1 ppm which satisfied the acceptance criterion of 1146 ±50 ppm.

Differential boron worth over the reference bank was calculated in NUC-120 using the above technique.

3.4 - TRANSIENT TESTING

3.4.1 - TURBINE GENERATOR TRIP WITH COINCIDENT LOSS OF OFFSITE POWER - ISU-222A

OBJECTIVE

This test is performed to verify the plant's ability to safely sustain a turbine-generator trip with no offsite power available for at least thirty minutes. This test satisfies activities described by FSAR Table 14.2-3, Sheet 18.

TEST METHODOLOGY

The test is initiated with Unit 1 in Mode 1, at greater than 10% reactor power, with the main generator output at approximately 130 MWe. All normal 6.9 kV electrical buses are initially energized by the Unit Auxiliary Transformer (1UT). Their alternate source, Startup Transformer 1ST, is locked out, preventing a designed automatic bus transfer to this backup supply. Class 1E Safeguards Busses 1EA2 and 1EA1 are initially aligned to Startup Transformer XST2, with their alternate source, Startup Transformer XST1, locked-out. The turbine is manually tripped, the feeder breaker to non-class 1E bus XA1 is opened, and the offsite power feeder breakers to the Class 1E busses are opened. This results in an immediate loss of Class 1E AC power and a loss of Unit 1 non-Class 1E AC power when the main generator trips, approximately 11.5 seconds after the turbine trip. The 11.5 seconds is due to normal protective relaying time delays.

Plant conditions are monitored to ensure that the standby emergency diesel generators start and re-energize the safeguards buses and that plant equipment functions properly to stabilize the Reactor Coolant System (RCS) in a Mode 3, hot standby condition and maintain it in that condition for at least 30 minutes. Also monitored is the ability of the Steam Generator Atmospheric Relief Valves to control steam line pressures below 1185 psig for at least 30 minutes.

SUMMARY OF RESULTS

The alignment of Unit 1 power supply breakers was completed, the turbine was manually tripped and the appropriate feeder breakers were opened. Both standby emergency diesel generators started and powered the safeguards buses. The safeguards sequencers both loaded the required plant equipment onto the safeguards buses at the proper times. Stabilization of and recovery from this event was performed in accordance with the permanent plant emergency operating and abnormal operating procedures.

SUMMARY OF RESULTS (Continued)

The safeguards buses were energized and plant equipment functioned properly to stabilize and maintain the RCS in a safe shutdown condition. This condition was maintained for 31 minutes satisfying the ≥30 minute criterion. The non-Class 1E buses remained de-energized for the duration of this test. RCS hot leg, cold leg and core exit thermocouple readings were verified to stabilize following the initiation of the transient indicating that natural circulation cooling was established. RCS subcooling was verified to be greater than 60°F.

Even though the Main Steam Isolation Valves were closed early in the event, in accordance with the permanent plant procedure used to stabilize the plant, main steam line pressures never exceeded the 1125 psig setpoint where the Steam Generator Atmospheric Relief Valves begin to open. Therefore, the limit of 1185 psig for maximum main steam line pressure was never exceeded. The atmospheric relief valves were opened manually to initiate natural circulation cooling as prescribed by the permanent plant procedure used for plant recovery.

The following is a summary of indicated plant conditions during the event:

Item	Maximum Value	Minimum Value
RCS Cold Leg Temp.	558.5°F	550.7°F
RCS Hot Leg Temp.	569.3°F	561.1°F
Pressurizer Pressure	2296 psig	2202.2 psig
Auctioneered High Core Exit Temp.	573 °F	571 °F
Main Steamline Press.	1119.6 psig	1036.7 psig

During the transient, only one unexpected event occurred. Pressurizer Power Operated Relief Valve (PORV) 1-PCV-455A opened for less than 5 seconds. While actual pressurizer pressure never reached the normal 2335 psig PORV setpoint, this valve actuated in response to a compensated pressure signal which includes an integral signal component. The pressurizer pressure was above the 2235 psig nominal pressure setpoint for a long enough time period to allow the integrated signal to grow large enough to open the PORV. This PORV actuation did not disrupt the RCS cooling by natural circulation.

A number of the Test Data Acquisition System channels originally called to be monitored were unavailable for use during test performance. None of the unavailable channels were required to verify test acceptance criteria.

3.4.2 - DESIGN LOAD SWING TESTS - ISU-231A

OBJECTIVE

This test is performed to demonstrate the dynamic response of the Reactor Coolant System (RCS) and the Rod Control System to automatically bring the plant to steady state conditions following a rapid 10% reduction in turbine load, and then to a rapid 10% increase in turbine load. This test partially satisfies activities described by FSAR Table 14.2-3, Sheets 23 and 24.

TEST METHODOLOGY

With plant conditions stable at approximately 35%, 50% or 100% power, a 10% load decrease is manually initiated from the turbine-generator Electro-Hydraulic Controls (EHC) at a rate of approximately 200% power/minute. Plant parameters are allowed to stabilize, and after stabilization, a 10% load increase is manually initiated. Plant parameters are again allowed to stabilize. The load decrease is performed by manually reducing the turbine-generator load limit setpoint to a value approximately 10% in power below the initial load reference operating power level. The load increase is performed by manually raising the load limit setpoint back above the original load reference operating power level. This allows the load to increase back to its original value at the start of the test. The load limit setpoint adjustment occurs at a rate of approximately 200% power/minute and is performed by main control board manual push button operation of a motor driven potentiometer that is set to move at that rate. These push buttons are permanent plant control features and the related circuitry is closely associated with the built-in turbine-generator runback circuits. In fact, prior to initiation of the 10% load increase, the runback circuits are temporarily bypassed by actuating a switch inside the EHC cabinets. This is done to disable load increase inhibiting circuitry that is activated by the 10% load decrease via the shared runback circuit portions. The 10% power load changes are nominal values and are actually specified to be 10% $\pm$2% in magnitude. The 10% load change may result in reactor power changes of greater than 10% power due to relatively low plant efficiency at lower power levels.

During the course of the test, strip chart recordings and Test Data Acquisition System (TDAS) recordings of key plant parameters are taken so that plant response can be analyzed. The principal parameters monitored included RCS Tavg, Tcold, Tref, pressurizer pressure and level, steam generator pressures and levels, steam and feedwater flows, control rod positions and speed, OTN16 and OPN16 setpoints, reactor power, feedwater pump speed and discharge pressure, N16 power, safety and relief valve positions, and steam dump valve positions.

3.4.2 - DESIGN LOAD SWING TESTS - ISU-231A (Continued)

SUMMARY OF RESULTS

The first test performed was at the 50% power plateau from approximately 47.5% reactor power. The second test execution was from approximately 34% reactor power and the final test execution was from approximately 100% reactor power. All three test executions satisfied the following criteria:

- o The load decreases and increases did not cause the reactor to trip nor the turbine to trip.
- o Safety injection did not initiate.
- o The steam generator safety or atmospheric relief valves and pressurizer safety or power operated relief valves did not lift during any of the load swings.
- o Nuclear power over/undershoot was less than 3%.
- o No manual intervention was required to bring plant conditions to steady state.
- o Plant variables returned to steady state conditions without sustained or diverging oscillations.

The first test performance, from approximately 50% power, resulted in two retests. The magnitude of the first load decrease was only 7% power, which did not adequately approximate a 10% load change. Plant initial conditions were restored and Retest #1 was performed to repeat the load decrease. The Retest #1 load decrease was approximately 12% power in magnitude and equilibrium Tavg was reached approximately 17 minutes following initiation of the load decrease. The subsequent load increase was blocked by the turbine-generator control circuitry after an increase of only 20 MWe. The load was reduced by this 20 MWe, the plant allowed to stabilize, the runback circuits were bypassed and the load increase repeated as Retest #2. The Retest #2 load increase was approximately 11.8% power in magnitude and equilibrium Tavg was reached in approximately 12 minutes. The interference of the runback circuitry with respect to the load increase was unknown at the time the test was written. The procedure was changed to bypass the runback circuits by actuation of a permanently installed switch inside the EHC cabinets. This procedure change was made prior to the performance of Retest #2.

The second test performance, from approximately 35% power, did not result in any retesting. The load decrease was approximately 9.5% power in magnitude and equilibrium Tavg was reached in approximately 15 minutes. The load increase was also approximately 9.5% power in magnitude and equilibrium Tavg was also reached in approximately 15 minutes.

SUMMARY OF RESULTS (Continued)

The final test performance, from approximately 100% power, also did not result in any retesting. The load decrease was approximately 9% power in magnitude and equilibrium Tavg was reached in approximately 8 minutes. The load increase was also approximately 9% power in magnitude and equilibrium Tavg was reached in approximately 9 minutes. Ten minutes of Axial Flux Difference (AFD) penalty time was accrued following the load decrease due to AFD deviating outside of its target band. This short deviation was an expected occurrence, caused by the deep insertion of Control Bank D in response to the load decrease. Normally, power changes are made using RCS boron concentration changes, leaving Control Bank D nearly fully withdrawn. The AFD is permitted by the Technical Specifications to be outside of its target band for up to one hour without subsequent action, when below 90% reactor power.

Numerical acceptance and review criteria are summarized on Table 3.4.2-1. Test performances were generally without incident with the following items noted:

During the tests performed at approximately 35% and 50% power, the first bank of three steam dump, or turbine bypass, valves momentarily opened as a result of the rapid load decrease. The steam dumps are armed by a 10% or greater decrease in turbine impulse chamber pressure. The turbine impulse chamber pressure is proportional to turbine-generator power. The 50% power test had a load decrease of approximately 12% which allowed the steam dumps to arm. While the 35% power test had a load decrease of only approximately 9.5%, the initial impulse pressure change was greater than 10% due to a minor overshoot in turbine governor valve positions in response to the rapid load decrease. This also allowed the steam dumps to arm. The minor overshoot is a normal result of a rapid load change and damps out very quickly. Once armed, the steam dump valves modulate to force average RCS temperature (Tavg) to within 5°F of the load varying reference RCS average temperature (Tref). In both test performances, Tavg was only marginally more than 5°F greater than Tref. This is why only three of the twelve steam dump valves modulated open for a short time, less then one minute. These steam dump actuation did not invalidate the test results because the rod control system properly responded to and stabilized the load decrease transient. The steam dumps aided the rod control system, as they are designed to do, during the initial portion of the transient. These test performances both had reactor power changes in excess of 10%, 11% at 35% power and 14% at 50% power. The rod control system is designed to respond to absorb a nominal 10% power change. The assistance of the steam dumps absorb the excess over 10% was proper.

3.4.2 - DESIGN LOAD SWING TESTS - ISU-231A (Continued)

SUMMARY OF RESULTS (Continued)

During the first test performance, at approximately 50% power, the recorded values of initial, final and transient steam header pressures resulted in an indicated excessive undershoot of 82 psig. Investigation of the data disclosed that because the initial and final values were recorded from a main control board indication and the transient values from temperature data logging instrumentation, the slight bias between the two data sources was enough to induce excessive error into the results. When the results from a single data source, a strip chart recorder, were evaluated, they showed the true undershoot to be only 65 psig, which met the <70 psig criterion. The discrepancy between the two data sources of approximately 10 psig was minimal, representing only 0.77% of the 1300 psig span of the instrument loop, well within normal tolerances.

During the final test performance, at approximately 100% power, it was noted that more time could have been allowed between the load decrease and the subsequent increase to achieve greater stability. However, the actual time was judged to have been acceptable based on examination of the convergence of monitored parameters.

Refer to Tables 3.4.2-2 through 3.4.2-7 for additional detailed data.

TABLE 3.4.2-1

DESIGN LOAD SWING TESTS SUMMARY

Power Plateau(%)	Load Swing	Nuclear Power Over/Under-shoot(%)	Allowed Limit(%)	Pressurizer Pressure Swing(psig)	Allowed Limit(psig)
35	Decrease	2	<3	+0,-5	<±100
	Increase	1.5	<3	+0,-5	<±100
50	Decrease	2	<3	+8,-27	<±100
	Increase	2	<3	+10,-35	<±100
100	Decrease	0	<3	+12,-30	<±100
	Increase	0	<3	+0,-20	<±100

Power Plateau(%)	Load Swing	Steam Gen. Level Swing(%)	Allowed Limit(%)	Steam Header Pressure Over/Undershoot (psig)	Allowed Limit(psig)
35	Decrease	+4,-3	≤ ±10	39	≤70
	Increase	+6,-2	≤ ±10	52	≤70
50	Decrease	+3,-1	≤ ±10	21	≤70
	Increase	+7,-3	≤ ±10	65	≤70
100	Decrease	+7,-3	≤ ±10	42	≤70
	Increase	+4,-1	≤ ±10	13	≤70

Power Plateau(%)	Load Swing	Tavg Over/Undershoot(°F)	Allowed Limit(°F)
35	Decrease	< 1.0	≤2.0
	Increase	< 1.0	≤2.0
50	Decrease	0	≤2.0
	Increase	2.0	≤2.0
100	Decrease	2.0	≤2.0
	Increase	0	≤2.0

TABLE 3.4.2-2

10% LOAD DECREASE AT 35% POWER SUMMARY

	INITIAL CONDITION	FINAL CONDITION
Generator Load (MWe)	315	205
Nuclear Power (%)	34	23
Tavg Auctioneered (°F)	567	562
Tref (°F)	566	562.5
N-16 Power (%)	35	25
OPN16 Setpoint (%)	112	112
OTN16 Setpoint (%)	114	116
Pressurizer Pressure (psig)	2235	2235
Pressurizer Level (%)	36	31
Steam Generator Level Loop 1 (%)	65	65
Steam Generator Level Loop 2 (%)	66	66
Steam Generator Level Loop 3 (%)	65	65
Steam Generator Level Loop 4 (%)	65	65
Steam Header Pressure (psig)	1058	1045
Steam Flow Loop 1 (pounds/hour)	1.0E6	0.7E6
Steam Flow Loop 2 (pounds/hour)	1.0E6	0.7E6
Steam Flow Loop 3 (pounds/hour)	1.0E6	0.75E6
Steam Flow Loop 4 (pounds/hour)	0.9E6	0.6E6
Feedwater Flow Loop 1 (pounds/hour)	1.1E6	0.75E6
Feedwater Flow Loop 2 (pounds/hour)	1.1E6	0.8E6
Feedwater Flow Loop 3 (pounds/hour)	1.1E6	0.8E6
Feedwater Flow Loop 4 (pounds/hour)	1.1E6	0.8E6
Feedwater Temperature Loop 1 (°F)	350	330
Feedwater Temperature Loop 2 (°F)	350	330
Feedwater Temperature Loop 3 (°F)	350	330
Feedwater Temperature Loop 4 (°F)	350	330
Feed Pump Discharge Pressure (psig)	1175	1162
Control Bank D Position (steps)	166	136.5
Control Bank C Position (steps)	227	227
Feedwater Pump 1-A Speed (rpm)	4090	3900

TABLE 3.4.2-3

10% LOAD INCREASE AT 35% POWER SUMMARY

	INITIAL CONDITION	FINAL CONDITION
Generator Load (MWe)	205	315
Nuclear Power (%)	23	34.5
Tavg Auctioneered (°F)	562	565
Tref (°F)	562.5	566
N-16 Power (%)	25	33.5
OPN16 Setpoint (%)	112	112
OTN16 Setpoint (%)	116	116
Pressurizer Pressure (psig)	2235	2230
Pressurizer Level (%)	31	34
Steam Generator Level Loop 1 (%)	65	65
Steam Generator Level Loop 2 (%)	66	66
Steam Generator Level Loop 3 (%)	65	65
Steam Generator Level Loop 4 (%)	65	66
Steam Header Pressure (psig)	1045	1058
Steam Flow Loop 1 (pounds/hour)	0.7E6	1.1E6
Steam Flow Loop 2 (pounds/hour)	0.7E6	1.1E6
Steam Flow Loop 3 (pounds/hour)	0.75E6	1.05E6
Steam Flow Loop 4 (pounds/hour)	0.6E6	0.9E6
Feedwater Flow Loop 1 (pounds/hour)	0.75E6	1.1E6
Feedwater Flow Loop 2 (pounds/hour)	0.8E6	1.15E6
Feedwater Flow Loop 3 (pounds/hour)	0.8E6	1.15E6
Feedwater Flow Loop 4 (pounds/hour)	0.8E6	1.15E6
Feedwater Temperature Loop 1 (°F)	330	350
Feedwater Temperature Loop 2 (°F)	330	350
Feedwater Temperature Loop 3 (°F)	330	350
Feedwater Temperature Loop 4 (°F)	330	350
Feed Pump Discharge Pressure (psig)	1162	1175
Control Bank D Position (steps)	136.5	173.5
Control Bank C Position (steps)	227	227
Feedwater Pump 1-A Speed (rpm)	3900	4030

TABLE 3.4.2-4

10% LOAD DECREASE AT 50% POWER SUMMARY

	INITIAL CONDITION	FINAL CONDITION
Generator Load (MWe)	455	315
Nuclear Power (%)	47.5	33.5
Tavg Auctioneered (°F)	569	565
Tref (°F)	570	566
N-16 Power (%)	50	39
OPN16 Setpoint (%)	112	112
OTN16 Setpoint (%)	116	118
Pressurizer Pressure (psig)	2232	2240
Pressurizer Level (%)	39	34
Steam Generator Level Loop 1 (%)	66	66
Steam Generator Level Loop 2 (%)	66	67
Steam Generator Level Loop 3 (%)	65	65
Steam Generator Level Loop 4 (%)	65	65
Steam Header Pressure (psig)	1035	1060
Steam Flow Loop 1 (pounds/hour)	1.40E6	1.05E6
Steam Flow Loop 2 (pounds/hour)	1.50E6	1.15E6
Steam Flow Loop 3 (pounds/hour)	1.45E6	1.15E6
Steam Flow Loop 4 (pounds/hour)	1.35E6	1.05E6
Feedwater Flow Loop 1 (pounds/hour)	1.55E6	1.20E6
Feedwater Flow Loop 2 (pounds/hour)	1.55E6	1.25E6
Feedwater Flow Loop 3 (pounds/hour)	1.60E6	1.25E6
Feedwater Flow Loop 4 (pounds/hour)	1.65E6	1.30E6
Feedwater Temperature Loop 1 (°F)	372	345
Feedwater Temperature Loop 2 (°F)	374	348
Feedwater Temperature Loop 3 (°F)	372	345
Feedwater Temperature Loop 4 (°F)	372	346
Feed Pump Discharge Pressure (psig)	1160	1170
Control Bank D Position (steps)	182	144
Control Bank C Position (steps)	227	227
Feedwater Pump 1-A Speed (rpm)	4175	4170

TABLE 3.4.2-5

10% LOAD INCREASE AT 50% POWER SUMMARY

	INITIAL CONDITION	FINAL CONDITION
Generator Load (MWe)	315	452
Nuclear Power (%)	36.5	48
Tavg Auctioneered (°F)	568	571
Tref (°F)	567	571
N-16 Power (%)	38	49
OPN16 Setpoint (%)	112	112
OTN16 Setpoint (%)	114	114
Pressurizer Pressure (psig)	2230	2235
Pressurizer Level (%)	39	43
Steam Generator Level Loop 1 (%)	66	66
Steam Generator Level Loop 2 (%)	67	67
Steam Generator Level Loop 3 (%)	65	65
Steam Generator Level Loop 4 (%)	66	66
Steam Header Pressure (psig)	1060	1055
Steam Flow Loop 1 (pounds/hour)	1.07E6	1.40E6
Steam Flow Loop 2 (pounds/hour)	1.12E6	1.50E6
Steam Flow Loop 3 (pounds/hour)	1.10E6	1.45E6
Steam Flow Loop 4 (pounds/hour)	1.00E6	1.35E6
Feedwater Flow Loop 1 (pounds/hour)	1.15E6	1.50E6
Feedwater Flow Loop 2 (pounds/hour)	1.25E6	1.70E6
Feedwater Flow Loop 3 (pounds/hour)	1.20E6	1.65E6
Feedwater Flow Loop 4 (pounds/hour)	1.25E6	1.60E6
Feedwater Temperature Loop 1 (°F)	350	370
Feedwater Temperature Loop 2 (°F)	350	374
Feedwater Temperature Loop 3 (°F)	350	372
Feedwater Temperature Loop 4 (°F)	350	372
Feed Pump Discharge Pressure (psig)	1175	1180
Control Bank D Position (steps)	139	173.5
Control Bank C Position (steps)	227	227
Feedwater Pump 1-A Speed (rpm)	4200	4240

TABLE 3.4.2-6

10% LOAD DECREASE AT 100% POWER SUMMARY

	INITIAL CONDITION	FINAL CONDITION
Generator Load (MWe)	1135	1030
Nuclear Power (%)	100	88
Tavg Auctioneered (°F)	588	586
Tref (°F)	588	586
N-16 Power (%)	105	93
OPN16 Setpoint (%)	111	112
OTN16 Setpoint (%)	110	112
Pressurizer Pressure (psig)	2250	2240
Pressurizer Level (%)	63	58
Steam Generator Level Loop 1 (%)	66	66
Steam Generator Level Loop 2 (%)	66	66
Steam Generator Level Loop 3 (%)	62	63
Steam Generator Level Loop 4 (%)	65	66
Steam Header Pressure (psig)	986	982
Steam Flow Loop 1 (pounds/hour)	3.7E6	3.4E6
Steam Flow Loop 2 (pounds/hour)	3.8E6	3.4E6
Steam Flow Loop 3 (pounds/hour)	3.6E6	3.3E6
Steam Flow Loop 4 (pounds/hour)	3.75E6	3.35E6
Feedwater Flow Loop 1 (pounds/hour)	3.8E6	3.4E6
Feedwater Flow Loop 2 (pounds/hour)	3.8E6	3.4E6
Feedwater Flow Loop 3 (pounds/hour)	3.7E6	3.4E6
Feedwater Flow Loop 4 (pounds/hour)	3.75E6	3.35E6
Feedwater Temperature Loop 1 (°F)	440	430
Feedwater Temperature Loop 2 (°F)	440	430
Feedwater Temperature Loop 3 (°F)	440	430
Feedwater Temperature Loop 4 (°F)	440	430
Feed Pump Discharge Pressure (psig)	1144	1134
Control Bank D Position (steps)	188.5	153
Control Bank C Position (steps)	225	225
Feedwater Pump 1-A Speed (rpm)	4803	4658

TABLE 3.4.2-7

10% LOAD INCREASE AT 100% POWER SUMMARY

	INITIAL CONDITION	FINAL CONDITION
Generator Load (MWe)	1030	1135
Nuclear Power (%)	88	100
Tavg Auctioneered (°F)	586	588
Tref (°F)	586	588
N-16 Power (%)	93	105
OPN16 Setpoint (%)	112	112
OTN16 Setpoint (%)	112	110
Pressurizer Pressure (psig)	2240	2260
Pressurizer Level (%)	58	60
Steam Generator Level Loop 1 (%)	66	67
Steam Generator Level Loop 2 (%)	66	67
Steam Generator Level Loop 3 (%)	63	63
Steam Generator Level Loop 4 (%)	66	66
Steam Header Pressure (psig)	1020	984
Steam Flow Loop 1 (pounds/hour)	3.4E6	3.7E6
Steam Flow Loop 2 (pounds/hour)	3.4E6	3.8E6
Steam Flow Loop 3 (pounds/hour)	3.3E6	3.6E6
Steam Flow Loop 4 (pounds/hour)	3.35E6	3.75E6
Feedwater Flow Loop 1 (pounds/hour)	3.4E6	3.8E6
Feedwater Flow Loop 2 (pounds/hour)	3.4E6	3.8E6
Feedwater Flow Loop 3 (pounds/hour)	3.4E6	3.7E6
Feedwater Flow Loop 4 (pounds/hour)	3.35E6	3.75E6
Feedwater Temperature Loop 1 (°F)	430	440
Feedwater Temperature Loop 2 (°F)	430	440
Feedwater Temperature Loop 3 (°F)	430	440
Feedwater Temperature Loop 4 (°F)	430	435
Feed Pump Discharge Pressure (psig)	1163	1140
Control Bank D Position (steps)	153	187.5
Control Bank C Position (steps)	225	225
Feedwater Pump 1-A Speed (rpm)	4658	4840

3.4.3 - DYNAMIC RESPONSE TO A FULL LOAD REJECTION AND TURBINE TRIP - ISU-284A

OBJECTIVE

This test is performed to verify the ability of the primary and secondary plant and the plant automatic control systems to sustain a generator trip from full power and to bring the plant to stable conditions following the transient. The N-16 instrumentation response time is also determined. This test satisfies activities described by FSAR Table 14.2-3, Sheets 23, 24 and 28.

TEST METHODOLOGY

From a stable plant power of approximately 100%, a generator trip is initiated by opening both of the main generator output breakers. This directly causes a turbine trip and a reactor trip. The operators follow the permanent plant Emergency Operating Procedures to bring the plant to stable conditions. The data trending is terminated when Tavg is stabilized at approximately 557°F (no-load Tavg).

SUMMARY OF RESULTS

The generator output breakers were opened at 0930 hours and the RCS temperatures stabilized at 0932 hours. All of the following test acceptance criteria were met:

- o The pressurizer and steam generator safety valves did not lift.
- o Safety injection did not initiate.
- o All control and shutdown rods released and dropped to the fully inserted position.
- o The plant was stabilized in Mode 3.
- o The steam dump valves modulated closed in the proper sequence.
- o Feedwater isolation occurred immediately following the plant trip. This was prior to reaching the no-load average RCS temperature of 557°F, as expected.
- o Average RCS temperature stabilized at 558°F. This satisfied the >553^{0}i review criteria.

The response time of the N-16 instrumentation was 2.0 seconds. This satisfied the time response requirement of ≤2.17 seconds.

Nuclear flux dropped to less than 15% power in 1.6 seconds. This satisfied the ≤2 second response requirement.

3.4.3 - DYNAMIC RESPONSE TO A FULL LOAD REJECTION AND TURBINE TRIP - ISU-284A (Continued)

SUMMARY OF RESULTS (Continued)

Narrow range steam generator levels remained ≥21.6%. This satisfied the review criterion that levels may drop out of span (<0%) but should return to span (>0%).

Pressurizer level remained ≥24.92%. This satisfied the minimum level of ≥20% review criterion.

Pressurizer pressure remained ≥1977 psig. This satisfied the minimum pressure of ≥1950 psig review criterion.

Refer to Tahle 3.4.3-1 for detailed test results.

TABLE 3.4.3-1

TRIP FROM 100% POWER SUMMARY

	INITIAL CONDITION	FINAL CONDITION
Generator Load (MWe)	1160	0
Nuclear Power (%)	100	0
Tavg Auctioneered (°F)	587	557
Tref (°F)	589	557
N-16 Power (%)	101	2
OPN16 Setpoint (%)	112	112
OTN16 Setpoint (%)	108	107
Pressurizer Pressure (psig)	2250	2265
Pressurizer Level (%)	59	25
Steam Generator Level Loop 1 (%)	65	40
Steam Generator Level Loop 2 (%)	66	39
Steam Generator Level Loop 3 (%)	63	36
Steam Generator Level Loop 4 (%)	65	35
Steam Header Pressure (psig)	961	1070
Steam Flow Loop 1 (pounds/hour)	3.9E6	0.4E6
Steam Flow Loop 2 (pounds/hour)	3.9E6	0.2E6
Steam Flow Loop 3 (pounds/hour)	3.7E6	0.3E6
Steam Flow Loop 4 (pounds/hour)	3.8E6	0.0E6
Feedwater Flow Loop 1 (pounds/hour)	3.9E6	0.25E6
Feedwater Flow Loop 2 (pounds/hour)	3.9E6	0.25E6
Feedwater Flow Loop 3 (pounds/hour)	3.8E6	0.3E6
Feedwater Flow Loop 4 (pounds/hour)	3.8E6	0.05E6
Feedwater Temperature Loop 1 (°F)	445	440
Feedwater Temperature Loop 2 (°F)	445	440
Feedwater Temperature Loop 3 (°F)	445	440
Feedwater Temperature Loop 4 (°F)	445	435
Feed Pump Discharge Pressure (psig)	1120	510
Control Bank D Position (steps)	201.5	0
Control Bank C Position (steps)	225	0
Feedwater Pump 1-A Speed (rpm)	4870	2

3.4.4 - REMOTE SHUTDOWN CAPABILITY TEST - ISU-223A

OBJECTIVE

This test verifies that the unit can be taken from approximately 20% reactor power to Hot Standby conditions from outside the control room with a minimum shift crew. The potential to safely cool the unit to cold shutdown conditions from outside the control room is also demonstrated. This test satisfies activities described by FSAR Table 14.2-3, Sheets 25 and 26 and Regulatory Guide 1.68.2. The cool down to 350°F and remote switch over to Residual Heat Removal System cooling was performed as part of the preoperational test program.

TEST METHODOLOGY

From the condition of greater than 10% generator load and less than 25% reactor power, the reactor is manually tripped locally from the reactor trip breakers. Utilizing abnormal operating procedure ABN-905A, Loss of Control Room Habitability, the minimum shift crew establishes control of the reactor plant and stabilizes it in Mode 3 from the Remote Shutdown Panel (RSP). From the Mode 3 stabilized condition, a controlled cool down of at least 50°F is initiated from the RSP to demonstrate cool down capability. Upon completion, control of the plant is transferred back to the Main Control Room. A standby operations crew remains in the Main Control Room throughout the test to assume control, if needed.

SUMMARY OF RESULTS

The reactor was locally tripped at 0910 hrs from 20% reactor power and 150 MWe generator load. The minimum shift crew used ABN-905A to establish a stable, hot standby (Mode 3) condition by 1025 hrs. The Reactor Coolant System (RCS) was stable at approximately 558°F as indicated in the control room and 540°F as indicated at the RSP. The RSP temperature indications come from strap-on RTDs instead of RTDs in thermowells actually immersed the process fluid and are less accurate. They do, however, indicate trends in an acceptable manner. Following 40 minutes of data taking to ensure stable RCS conditions, a controlled cooldown was started in accordance with ABN-905A. This stable 40 minutes satisfied the requirement to maintain a stable, hot standby condition for at least 30 minutes.

The cool down was performed from 1107 hrs. to 1304 hrs. The cooldown was 50°F as indicated by the RSP strap-on RTDs, approximately 53°F based on the change in steam generator steam pressures indicated at the RSP, and 52°F to 60°F from various main control room temperature indications. This satisfied the greater than or equal to 50°F cooldown requirement. The cooldown rate did not exceed Technical Specification limits at any time.

3.4.4 - REMOTE SHUTDOWN CAPABILITY TEST - ISU-223A (Continued)

SUMMARY OF RESULTS (Continued)

All transfers of control to and from the RSP were properly done and the RSP equipment was verified to operate properly with the following minor exceptions:

- The failure of the #1 Motor Driven Auxiliary Feedwater Pump run light to illuminate had no adverse impact on test performance because alternate indications existed on the RSP to verify pump start (e.g. flows to steam generators and pump suction and discharge pressures).

- The failure of auxiliary feedwater flow indicator 1-FI-2466D to indicate flow for up to 48 minutes had no adverse impact on test performance because alternate indication of auxiliary feedwater flow to the #4 steam generator existed on the RSP for Train B, and based on verification of #4 steam generator level indications at the RSP.

- A problem with the #4 Main Steam Isolation Valve dual indication was associated with the valve's stem mounted limit switch, not the RSP indicator, and had no adverse impact on test results based on auxiliary operator verification of valve closure and limit switch operation. The valves were verified closed prior to the cooldown and were not required to be closed to establish or maintain stability following the reactor trip.

- The sticking meter movement on pressurizer level indicator 1-LI-460B had no adverse impact on test results because lightly tapping the meter bezel resulted in the proper indication and also based on the alternate 1-LI-459B indications. 1-LI-460B was not used to take cooldown data.

The above four items did not compromise the overall performance of the remote shutdown panel instrumentation and controls.

3.4.5 - LARGE LOAD REDUCTION TESTS - ISU-263A

OBJECTIVE

This test is performed to demonstrate the dynamic response of plant systems to automatically bring the plant to steady state conditions following a rapid 50% reduction in turbine load, and then to stabilize conditions at the reduced load. This test partially satisfies activities described by FSAR Table 14.2-3, Sheets 23 and 24.

TEST METHODOLOGY

With plant conditions stable at approximately 75% or 100% power, a 50% load decrease is manually initiated from the turbine-generator Electro-Hydraulic Controls (EHC) at a rate of approximately 200% power/minute and plant parameters are allowed to stabilize. The load decrease is performed by manually reducing the turbine-generator load limit setpoint to a value approximately 50% in power below the initial load reference operating power level. The load limit setpoint adjustment occurs at a rate of approximately 200% power/minute and is performed by main control board manual pushbutton operation of a motor driven potentiometer that is set to move at that rate. These pushbuttons are permanent plant control features and the related circuitry is closely associated with the built-in turbine-generator runback circuits. The 50% power load change is a nominal value and is actually specified to be 50% ±2% in magnitude. The 50% load change may result in reactor power changes of less than 50% power due to relatively low plant efficiency at lower power levels.

During the course of the test, strip chart recordings and Test Data Acquisition System (TDAS) recordings of key plant parameters are taken so that plant response can be analyzed. The principal parameters monitored included RCS Tavg, Tcold, Tref, pressurizer pressure and level, steam generator pressures and levels, steam and feedwater flows, control rod positions and speed, OTN16 and OPN16 setpoints, reactor power, feedwater pump speed and discharge pressure, N-16 power, safety and relief valve positions, and steam dump valve positions.

SUMMARY OF RESULTS

The first test was performed at the 75% power plateau from approximately 77% reactor power. The final test execution was from approximately 100% reactor power. Both test executions satisfied the following criteria:

- o The load decreases did not cause the reactor to trip nor the turbine to trip.

- o Safety injection did not initiate.

SUMMARY OF RESULTS (Continued)

- o The steam generator safety and pressurizer safety valves did not lift during either of the load reductions.
- o No manual intervention was required to bring plant conditions to steady state.
- o Plant variables returned to steady state conditions without sustained or diverging oscillations.
- o Steam dump valves did not repeatedly cycle from open to closed position, although open position modulation did occur.

Numerical review criteria are summarized on Table 3.4.5-1. The 100% power test performance was without significant incident. However, two items were noted with respect to the 75% power test performance:

During the first test performance, from 77% reactor power, the turbine-generator load was reduced by approximately 570 MWe, 49% of full power. However, due to an urgent failure in the rod drive system approximately 2 minutes into the event, the control rods stopped their insertion. The effect of this failure was to limit the initial reactor power decrease to only approximately 28% of full power instead of a change of 49% to match the turbine-generator load change. This power mismatch was absorbed by the steam dump valves (turbine bypass valves) which stayed open for 48 minutes, instead of a more typical 5-8 minute duration. The open steam dumps created a "false" steam load not related to actual turbine-generator load such that the reactor, the steam generators and the feedwater system were not challenged to respond to the full 49% power change. The response of plant equipment, instrumentation and control circuitry to the imposed transient was excellent, even with the abrupt truncation of the control rod insertion. Plant process parameters behaved properly with no sustained or diverging oscillations. The principal reason for performing this test from approximately 75% power was to verify proper plant response to this type of transient prior to performing it from 100% power. If the plant control systems were not tuned properly for the test from 100% power, a much higher potential for a reactor trip would exist. This test from 77% power adequately demonstrated proper control system interactions and performance such that reperformance of this test from approximately 75% power was not justified.

SUMMARY OF RESULTS (Continued)

Also in the test from approximately 75% power, steam dump valve 1-TV-2370H failed to fully close. The problem was not related to the steam dump control circuitry. The circuitry demanded the valve to close, but the valve remained approximately 25% open. This valve was later repaired and was demonstrated to not "hang up" on closure.

Refer to Tables 3.4.5-1 through 3.4.5-3 for additional detailed data.

TABLE 3.4.5-1

Large Load Reduction Tests Summary

Power Plateau(%)	Peak Auctioneered Tavg(°F)	Expected Response(°F)	Auctioneered Tavg Undershoot(°F)	Expected Response(°F)
75	2	<7	0	<3
100	2	<7	0	<3

Power Plateau(%)	Auctioneered Tavg Peak to Valley(°F) During Steam Dump	Auctioneered Tavg Peak to Valley(°F) After Steam Dump	Expected Response(°F) During Steam Dump	Expected Response(°F) After Steam Dump	Pressurizer Pressure Swing(psig)	Expected Response(psig)
75	2	1	<3	<5	+15,-80	+100,-160
100	2	0	<3	<5	+25,-75	+100,-160

Power Plateau(%)	Steam Generator Level Swing(%)	Expected Response(%)	Duration of Max. Rod Speed (seconds)	Expected Time (seconds)	Steam Dump Duration (minutes)	Expected Time (minutes)
75	+8.65,-9.15	<±15	48	approx.<30	3	<8
100	+8.3,-10.6	<±15	45	approx.<30	5	<8

TABLE 3.4.5-2
LARGE LOAD REDUCTION FROM 75% POWER SUMMARY

	INITIAL CONDITION	FINAL CONDITION
Generator Load (MWe)	830	260
Nuclear Power (%)	77	33
Tavg Auctioneered (°F)	581	565
Tref (°F)	581	565
N-16 Power (%)	80	33
OPN16 Setpoint (%)	112	112
OTN16 Setpoint (%)	113	120
Pressurizer Pressure (psig)	2240	2250
Pressurizer Level (%)	51	40
Steam Generator Level Loop 1 (%)	65	66
Steam Generator Level Loop 2 (%)	66	66
Steam Generator Level Loop 3 (%)	63	63
Steam Generator Level Loop 4 (%)	65	66
Steam Header Pressure (psig)	1003	1030
Steam Flow Loop 1 (pounds/hour)	2.85E6	1.2E6
Steam Flow Loop 2 (pounds/hour)	2.8E6	1.2E6
Steam Flow Loop 3 (pounds/hour)	2.8E6	1.15E6
Steam Flow Loop 4 (pounds/hour)	2.8E6	1.2E6
Feedwater Flow Loop 1 (pounds/hour)	2.8E6	1.7E6
Feedwater Flow Loop 2 (pounds/hour)	2.85E6	1.7E6
Feedwater Flow Loop 3 (pounds/hour)	2.9E6	1.65E6
Feedwater Flow Loop 4 (pounds/hour)	2.8E6	1.6E6
Feedwater Temperature Loop 1 (°F)	416	350
Feedwater Temperature Loop 2 (°F)	418	350
Feedwater Temperature Loop 3 (°F)	418	350
Feedwater Temperature Loop 4 (°F)	416	350
Feed Pump Discharge Pressure (psig)	1188	1170
Control Bank D Position (steps)	208.5	110.5
Control Bank C Position (steps)	226	225
Feedwater Pump 1-A Speed (rpm)	4646	4065
Turbine Impulse Pressure (psig)	650	245

TABLE 3.4.5-3
LARGE LOAD REDUCTION FROM 100% POWER SUMMARY

	INITIAL CONDITION	FINAL CONDITION
Generator Load (MWe)	1130	595
Nuclear Power (%)	100	50
Tavg Auctioneered (°F)	588	570
Tref (°F)	589	572
N-16 Power (%)	101	62
OPN16 Setpoint (%)	110	111
OTN16 Setpoint (%)	110	120
Pressurizer Pressure (psig)	2250	2245
Pressurizer Level (%)	61	36
Steam Generator Level Loop 1 (%)	68	67
Steam Generator Level Loop 2 (%)	68	67
Steam Generator Level Loop 3 (%)	65	65
Steam Generator Level Loop 4 (%)	68	65
Steam Header Pressure (psig)	975	1018
Steam Flow Loop 1 (pounds/hour)	3.7E6	1.80E6
Steam Flow Loop 2 (pounds/hour)	3.8E6	1.80E6
Steam Flow Loop 3 (pounds/hour)	3.7E6	1.75E6
Steam Flow Loop 4 (pounds/hour)	3.75E6	1.85E6
Feedwater Flow Loop 1 (pounds/hour)	3.85E6	1.75E6
Feedwater Flow Loop 2 (pounds/hour)	3.8E6	1.75E6
Feedwater Flow Loop 3 (pounds/hour)	3.75E6	1.75E6
Feedwater Flow Loop 4 (pounds/hour)	3.75E6	1.70E6
Feedwater Temperature Loop 1 (°F)	439	352
Feedwater Temperature Loop 2 (°F)	440	353
Feedwater Temperature Loop 3 (°F)	441	353
Feedwater Temperature Loop 4 (°F)	438	352
Feed Pump Discharge Pressure (psig)	1133	1130
Control Bank D Position (steps)	195	90
Control Bank C Position (steps)	225	205
Feedwater Pump 1-A Speed (rpm)	4848	3928
Turbine Impulse Pressure (psig)	880	405

3.5 INSTRUMENTATION AND CALIBRATION TESTING

3.5.1 - CALIBRATION OF FEEDWATER AND STEAM FLOW INSTRUMENTATION AT POWER - ISU-202A

OBJECTIVE

The purpose of this test is to verify the calibration of Feedwater (FW) flow and Steam Flow (SF) instrumentation at each of the major reactor power levels. Calibration of the Feedwater flow instrumentation, because the characteristics of the Main Feedwater flow venturi are well known, is relatively straightforward. The differential pressure d/p transmitters are zero and span checked, and the downstream electronics conversion cards are shown to be in calibration. Steam flow is determined by measurement of steam generator pressure and the differential pressure developed by the flow of steam across the steam generator steam exit nozzle and associated piping to a downstream point on the main steamline. The major task is that of determining both the zero and the span of the steam flow differential pressure transmitters, because neither of these quantities are known precisely prior to actual power operations.

TEST METHODOLOGY

The test is comprised of three related activities. First, at hot, zero power, zero flow conditions, both FW and SF flow transmitters are verified to be at or adjusted to be at zero output. Provisions are made for high accuracy test d/p transmitters to be installed in parallel with the permanent plant FW d/p cells. As power is increased, the permanent plant instrumentation is verified to be within specified calibration tolerances with respect to the test d/p transmitters. Finally, the as-built spans of the SF measurement system are determined by requiring that the indicated SFs agree with the simultaneously measured FW flows.

With the plant in Mode 3 at hot, zero power conditions, an average reactor coolant temperature of approximately 557°F and steam generator pressures of approximately 1100 psig, the SF flow transmitters should have a zero output. Because the condensing pot for the low pressure side of the SF dp cell is at an elevation approximately three feet above the high pressure condensing pot, a zero offset has to be incorporated into the transmitter calibration to account for this static head difference. Due to differences in the thermal expansion of structures and piping to which these condensing pots are attached, and also due to high pressure static shift effects on the actual transmitters, the pre-test estimate of the zero offset usually has to be modified. The zero flow check of the FW flow transmitters is accomplished at low power operations (<10% reactor power) with the feedwater header pressurized by a Main Feedwater pump. Because the FW d/p cell instrument taps are

TEST METHODOLOGY (Continued)

installed in horizontal feedwater piping, the zero flow check is performed primarily to account for high pressure static shift of the transmitters.

At higher power levels up to and including 100%, data is taken which ultimately is used to determine the full span of the SF d/p transmitters as well as verifying the calibration of the FW flow instrumentation. With steam generator blowdown secured and the plant stable, a high accuracy measurement of FW flow is obtained. The most important parameters are the FW venturi differential pressures and these are obtained using precision, temporary test d/p cells which have both pre-test and post-test calibrations performed to minimize errors due to instrument drift or zero and span shifts. The output of the permanent plant d/p cells are compared to these values and adjustments are made to the permanent plant d/p cells, if necessary. Feedwater pressures and temperatures are also required for the determination of FW flow. As stated previously, only two parameters are used to measure SF, steam generator pressure and SF differential pressure. Unlike the FW permanent plant instrumentation, SF instrumentation has an allowance for density compensation based on steam generator pressure. Setting SF equal to FW flow, and extrapolating to full scale flows, allows the corresponding full span of the SF transmitter to be established. Recognizing that the most accurate extrapolation is that using data from higher power levels, the SF transmitters are not respanned until the 75% power data has been obtained. During steady operations at the 100% power plateau, a final set of data is taken and used to verify the calibration of both the SF and FW flow instrumentation. Adjustments and recalibration of the instrumentation is then undertaken, if found necessary.

SUMMARY OF RESULTS

Refer to Table 3.5.1-1 for detailed test results.

In Mode 3 at an RCS temperature of approximately 557°F and a corresponding steam generator pressure of approximately 1092 psig, the output of the SF transmitters at the NLP card was required to be 0.001, (+0.024,-0.000) volts at this zero flow condition. The as-found data ranged from -0.009 to +0.079 volts for these eight SF transmitters and all eight were recalibrated.

With the Main Feedwater header pressurized by a Main Feedwater pump, and the equalizing valve on the FW flow transmitter manifold

SUMMARY OF RESULTS (Continued)

open, the NLP card output voltage is also required to be 0.001 (+0.024,-0.000) volts. The collected data showed a range of -0.288 to +0.0034 volts, with only one of the eight transmitters satisfying the Acceptance Criterion. The remaining seven out-of-tolerance transmitters were recalibrated. The zero flow check of the SF and FW flow transmitters was therefore shown to be necessary and all sixteen transmitters were shown to be properly calibrated at zero flow conditions prior to ascending to significant power levels .

At-power testing was performed, with retesting as required, at the 30%, 50%, 75% and 100% plateaus. Primary and secondary plant systems were first shown to be stable by trending the following parameters: reactor power, average reactor coolant temperature, pressurizer pressure, steam generator levels, and SF and FW flows. After demonstrating adequate plant stability, the Test Data Acquisition System (TDAS) was used to collect the required data both from permanent plant and high accuracy test equipment.

The 30% and 50% data indicated that all eight of the SF transmitters would eventually have to be recalibrated. The SF/FW flow mismatches, however, were not so large that they adversely impacted plant operations. SF/FW flow mismatches serve as inputs to the steam generator level control system. The required level of agreement between SF and FW flows was ≤5% of full flow, with the differences ranging from 3.3% to 14.4% at 30% power and 6.7% to 16.6% at 50% power. Comparison of the permanent plant FW flow instrumentation with test instrumentation showed that only one transmitter, 1FT-540, required adjustment at the 30% power plateau. After recalibration of this d/p transmitter a retest of this instrument was performed with satisfactory results. At the completion of the 50% plateau testing, all FW flow loops were verified to be calibrated adequately and it was expected that the SF transmitters would have to be respanned after obtaining and evaluating the 75% power data.

Test data acquired at the 75% plateau, when combined with the lower power data, was used to derive the SF transmitter scaling shown on Table 3.5.1-1. The table also includes the scalings that were installed prior to power ascension testing. These were based on comparisons to other similar plants and best estimate engineering calculations. The SF transmitters were recalibrated to these new scalings and a retest at the 75% power level showed that the required 5% agreement level between indicated and calculated SF was achieved for all eight transmitters. This first retest at this power level showed that FW flow transmitters 1FT-530 and 1FT-541

3.5.1 - CALIBRATION OF FEEDWATER AND STEAM FLOW INSTRUMENTATION AT POWER - ISU-202A (Continued)

SUMMARY OF RESULTS (Continued)

required recalibration. After completion of these recalibrations, a second retest showed that while FW flow indications were satisfactory, the 0.5% agreement level between permanent plant and test equipment differential pressures was not satisfied for 1FT-520 and 1FT-530 (differences of 0.73% and 1.65%, respectively). The Test Review Group, after consideration of these results, decided that the magnitudes of out-of-calibration conditions were such that plant operations would not be adversely affected. Based on this and because at least one more complete set of data was to be collected at the 100% plateau, these two transmitters were not recalibrated at that time.

The initial set of data at 100% power indicated that three FW flow transmitters failed the Acceptance Criterion requiring no larger than a 0.5% difference in measured differential pressure between the permanent plant and test instrumentation. In addition to 1FT-520 and 1FT-530, previously identified at 75% power to be out-of-tolerance, 1FT-521 indicated a 0.7% difference. Additionally, 1FT-520 and 1FT-530 did not satisfy the required 1% agreement level between indicated and calculated flows (differences of 1.5% and 1.4%, respectively). After recalibration of these instruments the next retest yielded data which demonstrated that all eight FW flow transmitters and their associated circuitry were within the required levels of calibration.

At 100% power the indicated and calculated values for SF were required to be different by no more than 1%. The first set of data at this plateau showed that all SF transmitters except 1FT-543, with 3.5% difference, satisfied this requirement. A calibration check of this instrument was performed and it was found to be out of calibration. After recalibration using the new scaling shown on Table 3.5.1-1, the 1FT-543 retest data yielded acceptable results. However, in the retest 1FT-532 was found to have now failed the 1% agreement level, having a 1.1% difference. A detailed examination of all components in the SF instrument loops was undertaken at this time to determine why SF instruments or loops were, at different times, apparently drifting out of calibration. Although it had been noted that several of the steam generator pressure transmitters had been out of tolerance to varying degrees, the impact of this on indicated SF had not been fully evaluated. The calibration of these particular instruments is not directly evaluated by this procedure. As noted earlier, steam generator pressure is used to compensate for variations in steam density by the permanent plant instrumentation. Steam generator pressure values from test instrumentation were then used to avoid this potential error and provided the best evaluation of the

3.5.1 - CALIBRATION OF FEEDWATER AND STEAM FLOW INSTRUMENTATION AT POWER - ISU-202A (Continued)

SUMMARY OF RESULTS (Continued)

as-built calibration for the SF transmitters. This steam generator pressure compensation calculation was performed for the eight SF loops with the result that all scalings recommended at the 75% power plateau were found to be satisfactory with the exception of 1FT-543. As shown on Table 3.5.1-1 the recommended scaling is now -44 to 313 inches of water column (inWC) with a span of 357 inWC. It should be noted that although this is a significant change, the impact on indicated flow is only 2%. A Work Request was written to recalibrate 1FT-543 to correspond to this scaling. When the work has been completed, retesting of 1FT-543 is to be accomplished using permanent plant procedure PPT-P1-5001, "Calibration of Feedwater and Steam Flow Instrumentation at Power", which uses the same basic methodology as this test.

The last portion of testing performed by this procedure was to verify the SF transmitters zero flow outputs upon return to Mode 3 following the plant trip from 100% power test. Seven of the eight transmitters were found to be out-of-tolerance, ranging from -0.003 to +0.055 volts versus the required range of 0.001 (+0.024, -0.000) volt. The instruments were re-zeroed and retesting showed that all as-left values satisfied the Acceptance Criterion tolerance.

In summary, the testing was viewed as having been successful with the overall final results being:

- o The FW flow transmitters were left within their tightly specified calibration range
- o The downstream FW flow instrumentation was shown to be within calibration
- o The SF calibrations established in this test were acceptable, with only one transmitter requiring a new scaling.

Performance of the permanent plant procedure PPT-P1-5001 may be used in the future to finalize and/or enhance the scaling of the SF transmitters over time.

Table 3.5.1-1

Calibration of Steam Flow Transmitters

All values are given in units of inWC.

Transmitter	ORIGINAL SCALING Range	Span	75% POWER SCALING Range	Span
1FT-512	-29 to 447.6	476.6	-27 to 334	361
1FT-512	-45 to 431.6	476.6	-41 to 320	361
1FT-522	-30 to 427.8	457.8	-27 to 370	397
1FT-523	-28 to 429.8	457.8	-26 to 371	397
1FT-532	-30 to 489.2	519.2	-28 to 420	448
1FT-533	-31 to 488.2	519.2	-28 to 420	448
1FT-542	-30 to 432.6	462.6	-29 to 344	373
1FT-543	-46 to 416.6	462.6	-44 to 329	373 *

* Evaluation of the 100% data indicated that the best estimate range for 1FT-543 was -44 to 313 inWC with a span of 357 inWC.

3.5.2 - THERMAL POWER MEASUREMENT AND STATEPOINT DATA COLLECTION - ISU-224A

OBJECTIVE

This test is performed to determine reactor thermal power by a secondary plant calorimetric measurement and to collect control and protection instrumentation data at steady state power levels (statepoints).

TEST METHODOLOGY

From stable plant conditions, statepoint data is collected and calorimetric power measurements made at the approximate 0%, 30%, 50%, 75%, 90% and 100% power levels. Calorimetric data includes feedwater temperature, main feedwater flow venturi differential pressures, steam pressures, atmospheric pressure, and steam generator blowdown flows.

Statepoint data is taken from the main control board indicators, from the P2500 Process Computer, and from the Test Data Acquisition System (TDAS). Data recorded includes N-16 powers, RCS flows, RCS temperatures, pressurizer level and pressure, nuclear instrumentation outputs, steam pressures, main generator output, steam generator levels, feedwater flows, steam flows, and feedwater pressures and temperatures. The recorded values are compared against each other to ensure consistency.

Four data sets are taken within an approximate 20 minute time period for each of the parameters to assure good quality of data.

SUMMARY OF RESULTS

Reactor thermal power was determined based on calorimetric measurements at the 30%, 50%, 75%, 90% and 100% power testing regimes. Calorimetric measurements do not apply to the Mode 3 test at 0% power.

Statepoint data was taken at all power levels from all available channels. Certain P2500 process computer and TDAS channels were unavailable at some power levels due to software reconfiguration and hardware installation status. There were no absolute requirements that all channels be compared at all power levels. There were sufficient channels available at all power levels to adequately verify proper display of plant conditions.

Throughout the various performances of this test, a number of main control board indicators and associated P2500 process computer and TDAS channels were noted to differ by notable amounts. There were no specified agreement criteria in this test. All items were evaluated and satisfactorily dispositioned as being within calibration tolerances or corrected by recalibration.

3.5.3 - OPERATIONAL ALIGNMENT OF PROCESS TEMPERATURE AND N16 INSTRUMENTATION - ISU-226A

OBJECTIVE

This test is performed to align the N16 and Tavg process instrumentation, to verify the linearity of the N16 and Tavg instrumentation, to determine the optimum voltage setting for the N16 detector High Voltage power supplies, and to determine the N16 detector currents at various power levels. This test partially satisfies activities described by FSAR Table 14.2-3, Sheets 9, 10 and 22.

TEST METHODOLOGY

At Comanche Peak, the Reactor Coolant System (RCS) Resistance Temperature Detector (RTD) manifold hot leg temperature (Thot) and cold leg temperature (Tcold) measurement instrumentation has been replaced by a Nitrogen-16 (N16) power monitor and an in-line Tcold RTD. The N16 power monitor measures the thermal power of the reactor by detecting the amount of N16 present in the coolant. The concentration of N16 in the coolant is directly proportional to the fission rate in the core and is detected by measuring the high energy gamma flux from the N16 decay which penetrates the walls of the hot leg piping. The fast response in-line Tcold RTD is in a thin wall thermowell installed in the cold leg piping. The process control system uses these inputs to generate a Tavg signal which is used for input to Rod Control, pressurizer level control, and steam dump control. An N16 Power signal is also generated which inputs to the Reactor Protection System for Overtemperature and Overpower reactor trips.

This test is a collection of eleven different tests of the N16, Tcold and Tavg process instrument loops which are performed throughout the startup program from Mode 3 through 100% power. Refer to Table 3.5.3-1 for a matrix of which tests are performed at each plant condition.

The DETERMINATION/SETTING OF N16 DETECTOR HIGH VOLTAGE test is performed at approximately 50% reactor power. The N16 gamma detectors are tested one loop at a time. The current output from the N16 gamma detectors is measured by a picoammeter while the high voltage power supply output voltage is adjusted from 300 volts to 1200 volts. This data is plotted to determine the plateau region of the curve, the region of minimum output current change for a given voltage change. The power supply is then set to a value in this plateau region of the curve, nominally 800 volts.

The N16 CURRENT MEASUREMENT test is performed in Mode 3 and at the 30%, 50%, 75%, and 100% power plateaus. The input voltage and output voltage of each N16 power monitor module is recorded

TEST METHODOLOGY (Continued)

simultaneously with the reactor thermal power from a precision secondary calorimetric. At zero power (Mode 3) the power monitor module output is verified to be zero ±0.03 volts.

The RCS COLD LEG TEMPERATURE CHECKS are performed at every power plateau and in Mode 3. The active cold leg RTD temperature, as measured at the output of the NRA card in the process control rack, is compared to its associated spare RTD temperature, measured as resistance from the RTD. These temperatures are required to agree within 1.2°F for each RCS loop.

The VERIFICATION OF TAVG CIRCUITRY test is performed at every power plateau and in Mode 3. The process control system Tavg signal is compared to a calculated value generated by the following equation:

$$Tavg = Tcold + (K9)(N16\ Power)$$

where Tcold is the cold leg temperature from the active RTD circuitry, N16 is a power signal generated by the N16 process loop, and K9 is a constant equal to 1/2 the full power temperature difference of hot leg to cold leg. The calculated Tavg is verified to be within 0.5°F of the Tavg signal.

The NEUTRON STREAMING DETERMINATION test is performed at the 75% power plateau. The N16 gamma detectors in the RCS hot legs monitor gamma rays from the decay of N16 in the RCS water. Additionally, gamma rays streaming directly from the upper portion of the reactor core and secondary gamma rays generated by streaming neutrons add to the N16 power signal. This contribution to the N16 power signal comes primarily from the top region of the core. The signals from the top two detectors in each nuclear instrumentation power range channel are used to compensate the associated N16 power signal. During the Incore/Excore Detector Calibration test, an axial Xenon transient is initiated causing the neutron flux to shift axially in the core. The following data is taken during the transient: N16 power monitor output, the output from each of the top two power range detectors and precision calorimetric power measurements. Using the relationship:

$$Q = A_1(V^{\mu}N16) + A_2(V_A) + A_3\ (V_B)$$

where Q is calorimetric power, $V^{\mu}N16$ is the N16 Power monitor output, V_A is the top power range detector output and V_B is the next to top power range detector output, the constants A_1, A_2 and A_3 are determined by linear regression analysis. The neutron streaming compensation gains are then calculated and used to calibrate the process channels to negate the core streaming effects.

3.5.3 - OPERATIONAL ALIGNMENT OF PROCESS TEMPERATURE AND N16 INSTRUMENTATION - ISU-226A (Continued)

TEST METHODOLOGY (Continued)

The FULL-POWER DELTA-T (K-9) DETERMINATION test is performed at the 75% power plateau. The purpose of this test is to determine the 100% power hot leg to cold leg temperature difference by extrapolation of data collected at the previous power levels. Then a new K-9 constant (1/2 of the full power temperature difference) is calculated and used to recalibrate the N16 Tavg circuits. First, full power cold leg temperature is extrapolated using Tcold data from previous power levels. Then full power volumetric enthalpy (density compensated specific enthalpy) is extrapolated using the volumetric enthalpy from RCS flow measurements performed at previous power levels. The full power hot leg enthalpy and temperature are calculated based on these extrapolated values and the ASME Steam Tables. From this, the full power hot leg to cold leg temperature difference and K-9 constant are calculated.

The FULL POWER DELTA-T (K-9) VERIFICATION test is performed at the 100% power plateau. Cold leg and hot leg temperatures are determined from the RCS flow test procedure (hot leg temperature is determined by iteration of TTFM measurement of RCS flow, secondary precision calorimetric power, and Tcold). Then this temperature difference is compared to a temperature difference equivalent to twice the current value of K-9. Any loop which differs by more than 1% is recalibrated using the new K-9 value determined from the actual full power cold leg and hot leg temperatures.

The N16 POWER CHECK - K8 ADJUSTMENT test is performed at every power plateau and in Mode 3. The N16 power signal for each loop is compared to precision secondary calorimetric power and adjustments are made to make them match. Below 75% power the gain of the N16 power monitor module itself is adjusted. After the neutron streaming gains have been determined at 75% power (as described in a previous paragraph), adjustments are made to the K8 constant instead of the power monitor module. At zero power (Mode 3), the N16 power signal is verified to be zero ±0.05 volts.

The TEMPERATURE DECALIBRATION DATA test is performed at the 50% power plateau. This test is performed to determine the sensitivity of N16 power measurements and nuclear instrumentation power range measurements to changes in RCS temperature. The Automatic Reactor Control System test changes the RCS average temperature to 5°F above and 5°F below the normal average temperature, with reactor power held constant. At those plant conditions, this test collects the following data: N16 power, RCS temperatures, Nuclear Instrumentation outputs, and calorimetric power.

3.5.3 - OPERATIONAL ALIGNMENT OF PROCESS TEMPERATURE AND N16 INSTRUMENTATION - ISU-226A (Continued)

TEST METHODOLOGY (Continued)

The N16 POWER LINEARITY CHECKS are performed after the N16 power instrumentation has been adjusted at full power and during the power ascension following a plant trip. The power output from each loop is plotted against calorimetric power and evaluated for linearity by the NSSS vendor.

The N16 ELECTRICAL ZERO DETERMINATION test is performed at least four hours after a reactor shutdown from 100% power. Input and output voltages of each power monitor module are recorded. The output voltages are verified or adjusted to be zero ±0.033 volts. This ensures proper compensation for the background gamma flux.

This startup test also collects N16 data from the following transient tests: Full Load Rejection and Turbine Trip, Design Load Swings, Large Load Reduction, and Turbine/Generator Trip with Coincident Loss of Offsite Power. This data is evaluated by the NSSS vendor for proper response of N16 during transients.

SUMMARY OF RESULTS

The DETERMINATION/SETTING of N16 DETECTOR HIGH VOLTAGE test results indicated that the N16 detector current is independent of the power supply high voltage setting between 300 and 1200 volts. The high voltage power supplies were set to 800 volts.

The N16 CURRENT MEASUREMENT test results were:

Power Plateau	Mode 3	30%	50%	50% Retest #2
Calorimetric Power(%)	0	29.21	50.14	48.15
Loop 1 input (volts)	0.0002	-0.251	-0.4306	-0.415
Loop 1 output (volts)	0.0005	2.046	3.2645	3.153
Loop 2 input (volts)	0.0003	-0.244	-0.4182	-0.404
Loop 2 output (volts)	0.00004	2.082	3.3085	3.190
Loop 3 input (volts)	0.00025	-0.240	-0.4127	-0.397
Loop 3 output (volts)	0.0008	2.081	3.2929	3.169
Loop 4 input (volts)	0.00003	-0.242	-0.4203	-0.405
Loop 4 output (volts)	0.00035	2.067	3.3151	3.193

At zero power, the criterion is 0.0 ±0.03 volts

3.5.3 - OPERATIONAL ALIGNMENT OF PROCESS TEMPERATURE AND N16 INSTRUMENTATION - 1[illegible]U-226A (Continued)

SUMMARY OF RESULTS (Continued)

N16 CURRENT MEASUREMENT test results (continued)

Power Plateau	75%	100%
Calorimetric Power(%)	76.19	99.33
Loop 1 input (volts)	-0.668	-0.86
Loop 1 output (volts)	5.222	6.77
Loop 2 input (volts)	-0.645	-0.84
Loop 2 output (volts)	5.262	6.86
Loop 3 input (volts)	-0.633	-0.83
Loop 3 output (volts)	5.267	6.86
Loop 4 input (volts)	-0.638	-0.83
Loop 4 output (volts)	5.201	6.76

The loop inputs were all within 0.03 volts of zero in Mode 3. The retest at 50% power was performed due to a discrepancy in the calorimetric power measurements. All of the data collected was acceptable.

The RCS COLD LEG TEMPERATURE CHECKS test results were:

Power Plateau		Mode 3	30%	30% Retest#1	50%
Nuclear Power(%)		0	29.5	28.1	47.5
Loop 1	Active(°F)	557.57	555.56	556.24	558.34
Tcold	Spare(°F)	557.19	555.79	556.43	558.77
	ΔTcold(°F)	+0.38	-0.23	-0.19	-0.43
Loop 2	Active(°F)	557.13	555.34	555.92	558.48
Tcold	Spare(°F)	556.93	554.82	555.41	557.63
	ΔTcold(°F)	+0.20	+0.52	+0.51	+0.85
Loop 3	Active(°F)	557.36	555.53	556.10	558.51
Tcold	Spare(°F)	557.43	556.00	556.55	559.17
	ΔTcold(°F)	-0.07	-0.47	-0.45	-0.66
Loop 4	Active(°F)	557.53	555.36	556.01	558.23
Tcold	Spare (°F)	556.83	554.14	554.83	556.81
	ΔTcold (°F)	+0.70	+1.22	+1.18	+1.42

3.5.3 - OPERATIONAL ALIGNMENT OF PROCESS TEMPERATURE AND N16 INSTRUMENTATION - ISU-226A (Continued)

SUMMARY OF RESULTS (Continued)

Power Plateau		50% Retest #2	75%	100%	100% Retest #3
Nuclear power(%)		48%	76%	100%	97%
Loop 1	Active(°F)	559.62	560.16	561.42	559.10
Tcold	Spare(°F)	559.95	560.64	562.05	WR 559.76
	ΔTcold(°F)	-0.33	-0.48	-0.63	-0.66
Loop 2	Active(°F)	559.72	560.17	561.02	558.95
Tcold	Spare(°F)	558.87	559.10	559.79	WR 559.96
	ΔTcold(°F)	+0.85	+1.07	+1.23	-1.01
Loop 3	Active(°F)	559.85	560.09	561.08	558.68
Tcold	Spare(°F)	560.45	560.88	562.14	WR 559.56
	ΔTcold(°F)	-0.60	-0.79	-1.06	-0.88
Loop 4	Active(°F)	559.63	560.00	561.22	558.96
Tcold	Spare(°F)	558.24	558.44	559.43	WR 559.18
	ΔTcold(°F)	+1.39	+1.56	+1.79	-0.22

The values listed as WR for Retest #3 were from the Wide Range Tcold RTDs.

The acceptance criterion of this test was that the ΔTcold for each loop shall be 1.2°F or less. Retest #1 was performed at 30% due to the failure of loop 4 to meet this criterion and because the RCS temperature had changed 1.2°F during the test. Retest #2 was performed at 50% due to another failure of loop 4 and a discrepancy in the calorimetric measurement. The NSSS vendor recommended that power ascension continue to 100% power to gather more data since the 1.2°F criterion only applied at 100% power. At 100% power loops 2 and 4 failed the criterion. The NSSS vendor determined the measured difference was due to a physical temperature divergence in the cold leg piping (Cold Leg Streaming). Since the purpose of the 1.2°F criterion is to verify the NRA card is properly adjusted, the wide range RTD which has the same installation orientation as the active RTD was used for comparison in Retest #3 at 100% power. These results were satisfactory. The NSSS Vendor has reviewed the data and believes that the active RTDs provide the most accurate measurement of bulk coolant temperature in the cold leg piping. The Cold Leg Streaming issue is still under evaluation by the NSSS Vendor.

3.5.3 - OPERATIONAL ALIGNMENT OF PROCESS TEMPERATURE AND N16 INSTRUMENTATION - ISU-226A (Continued)

SUMMARY OF RESULTS (Continued)

The VERIFICATION OF TAVG CIRCUITRY test results were:

Tavg Error(°F)

Power Plateau	Loop 1	Loop 2	Loop 3	Loop 4	Criterion
Mode 3	-0.11	+0.07	-0.21	+0.07	<±0.5
Mode 3 Retest #1	N/A	N/A	-0.04	N/A	<±0.5
30%	-0.11	-0.07	-0.12	+0.08	<±0.5
50%	-0.02	-0.01	-0.15	0.00	<±0.5
75%	-0.141	-0.085	-0.161	-0.069	<±0.5
100%	-0.498	+0.06	-0.27	0.0	<±0.5
100% Retest #2	-0.26	-0.09	-0.46	0.0	<±0.5

Retest #1 in Mode 3 was required by recalibration of loop 3. Retest #1 at 100% was to be performed due to circuit calibrations for new K9 values identified in the FULL POWER DELTA T (K9) VERIFICATION test. Due to problems with test instrumentation, the data obtained was indeterminate and retest #2 at 100% power was performed. All loops calculated Tavg within the required 0.5°F accuracy.

The NEUTRON STREAMING DETERMINATION test results were:

Item	Loop 1	Loop 2	Loop 3	Loop 4
A_1	14.9855	14.7652	14.8892	15.0800
A_2	-0.8191	-0.4350	-0.8046	-0.8749
A_3	0	0	0	0
G_A	-0.0547	-0.0295	-0.0540	-0.0580
G_B	0	0	0	0

where $G_A = A_2/A_1$ and $G_B = A_3/A_1$, the gains for the top and next to top nuclear instrumentation detector signal compensation.

These values of gains were acceptable and used to calibrate the neutron streaming compensation circuits.

The FULL-POWER DELTA-T (K-9) DETERMINATION test results were:

Extrapolated Values	Loop 1	Loop 2	Loop 3	Loop 4
Tcold(°F)	559.71	559.78	559.66	559.55
Hot Leg Enthalpy (BTU/lbm)	634.76	635.10	634.12	635.95
Thot (°F)	614.59	614.81	614.17	615.37
DELTA-T(°F)	54.88	55.03	54.51	55.82
K-9 (°F)	27.44	27.52	27.26	27.91

3.5.3 - OPERATIONAL ALIGNMENT OF PROCESS TEMPERATURE AND N16 INSTRUMENTATION - ISU-226A (Continued)

SUMMARY OF RESULTS (Continued)

The K-9 constant used for preliminary calibration of N16 Tavg circuitry was 28.1°F. Since these extrapolated values were within 1.0°F of 28.1°F, no adjustments to K-9 were made at 75% power.

The FULL-POWER DELTA-T (K-9) VERIFICATION test results were as follows where the % Error = ((Full Power ΔT - (2xK9))/(2xK9))x100 and the New K9 value is equal to the Full Power ΔT/2:

Measured Values	Loop 1	Loop 2	Loop 3	Loop 4
Tcold (°F)	560.96	560.58	561.18	560.54
Thot (°F)	615.49	616.04	615.20	616.27
Full Power ΔT(°F)	54.89	55.66	54.34	55.97
2 x K9 (°F)	56.20	56.20	56.20	56.20
Error (%)	-2.33	-0.96	-3.31	-0.41
New K9 (°F)	27.45	27.83	27.17	N/A

NOTE: Tcold and Thot are measured values at 99%, ΔT is extrapolated to 100.0% power.

The error for loops 1 and 3 exceeded the allowed 1% and were recalibrated with new K-9 values. Loop 2 was also recalibrated with a new K-9 value since it was very close to the allowed 1%. The loop 4 value was left at 28.1°F. The VERIFICATION OF TAVG CIRCUITRY Retest #2 at 100% was performed following the adjustments to K-9 and was satisfactory.

The N16 POWER CHECK - K8 ADJUSTMENT test results were:

For zero power, the N16 output voltage must be -0.05 volts to +0.05 volts

Test	Loop 1	Loop 2	Loop 3	Loop 4
Mode 3 (volts)	0.01752	0.00951	0.07667	0.01082
Mode 3 Retest #1(volts)	N/A	N/A	0.00723	N/A

For at-power measurements, if N16 power differs from calorimetric by 1% or more, the N16 gain is to be adjusted. (Listed Values are calorimetric power (%) - N16 power(%))

Power Plateau	Loop 1	Loop 2	Loop 3	Loop 4
30%	-0.48	-0.63	-0.60	-0.64
50%	2.21	1.95	2.13	2.07
50% Retest #2 (before adjustment)	1.78	1.59	1.84	1.59
50% Retest #2 (after adjustment)	0.02	0.01	0.16	0.16
75%	-0.28	-0.37	-0.25	+0.44
100%	-0.46	-0.16	-0.91	-0.01

3.5.3 - OPERATIONAL ALIGNMENT OF PROCESS TEMPERATURE AND N16 INSTRUMENTATION - ISU-226A (Continued)

SUMMARY OF RESULTS (Continued)

Retest #1 in Mode 3 was performed due to recalibration of loop 3 to correct an out of specification voltage. Following recalibration all voltages were within 0.05 volts of zero. At the 50% power plateau, a discrepancy in the calorimetric data resulted in Retest #2. The results of Retest #2 were all greater than the required maximum 1% difference between the calorimetric and N16 power. All four of the N16 power monitor modules gains were readjusted and following readjustment, all results were within 1%. The results of testing at 75% and 100% power were within 1% and no further adjustments were made to the power monitor modules and none were made to K8 constants.

The TEMPERATURE DECALIBRATION DATA test results were:

Item	At Normal Tavg	Tavg +5°F	Tavg -5°F	DELTA +5 to -5°F
Calorimetric Power (%)	48.19	48.21	48.62	-0.41
N16 Power (%)	48.02	48.51	48.05	0.46
Nuclear Power(%)	47.36	49.16	46.24	2.92
Tcold (°F)	558.81	564.03	553.54	10.49
Thot (°F)	588.32	593.51	583.35	10.16
Tavg (°F)	572.47	578.08	567.28	10.80

The test results indicate that N16 power measurements are less sensitive to temperature changes than Nuclear Instrumentation power range measurements. The small change of 0.46% power/10°F shows that the N16 instrumentation is sufficiently insensitive to temperature changes to not require compensation circuitry.

The N16 POWER LINEARITY CHECKS were performed following the large load reduction test. Reactor power was reduced to 30% and then returned to 100% with hold points at 50% and 75% power to collect N16 power data. The NSSS vendor evaluated the data and determined it to be acceptable.

The N16 ELECTRICAL ZERO DETERMINATION test results were:

Loop	Input(volts)	Output(volts)	As Left Output(volts)
1	0.000	0.004	0.004
2	0.000	0.003	0.003
3	0.000	0.004	0.004
4	0.000	0.003	0.003

3.5.3 - OPERATIONAL ALIGNMENT OF PROCESS TEMPERATURE AND N16 INSTRUMENTATION - ISU-226A (Continued)

SUMMARY OF RESULTS (Continued)

The N16 Electrical Zero Determination was performed approximately four and one-half hours after the Full Load Rejection and Turbine Trip from 100% power. The output voltages were within the required ± 0.033 volts of zero tolerance and no adjustments were made.

The N16 Transient Response Data was collected and provided to the NSSS vendor. The evaluation determined that the N16 system responded properly to the transients.

TABLE 3.5.3-1

PROCESS TEMPERATURE/N16 TESTS VS. PLANT CONDITIONS MATRIX

	MODE 3	30%	50%	75%	100%
DETERMINATION/SETTING OF N16 DETECTOR HIGH VOLTAGE			X		
N16 CURRENT MEASUREMENTS	X	X	X Retest2	X	X
RCS COLD LEG TEMPERATURE CHECKS	X	X Retest1	X Retest2	X	X Retest3
VERIFICATION OF TAVG CIRCUITRY	X Retest1	X	X	X	X Retest2
NEUTRON STREAMING DETERMINATION				X	
FULL POWER DELTA-T(K-9) DETERMINATION				X	
FULL POWER DELTA-T(K-9) VERIFICATION					X
N16 POWER CHECK - K8 ADJUSTMENT	X Retest1	X	X Retest2	X	X
TEMPERATURE DECALIBRATION DATA			X		
N16 POWER LINEARITY CHECKS	Performed during Power Ascension following 100% power testing				
N16 ELECTRICAL ZERO DETERMINATION	Performed at zero power following 100% power testing				
N16 TRANSIENT RESPONSE DATA	N16 Data collected during: Full Load Rejection and Turbine Trip; Design Load Swings; Large Load Reduction; Turbine/Generator Trip With Coincident Loss of Offsite Power				

3.5.4 - OPERATIONAL ALIGNMENT OF NUCLEAR INSTRUMENTATION - ISU-204A

OBJECTIVE

This test is performed to verify that the excore Nuclear Instrumentation System (NIS) functions per design. This test satisfies activities described by FSAR Table 14.2-3, Sheets 9 and 10.

TEST METHODOLOGY

Selected parameters are evaluated, monitored, and determined during various testing phases.

Prior to and at the time of Initial Criticality, Source Range (SR) to Intermediate Range (IR) channel overlap data is taken to verify how much overlap exists between them. Data is recorded simultaneously from both SR and both IR channels as the reactor neutron flux increases during the approach to criticality. This data permits the calculation of whether or not the IR channels begin to indicate at a sufficiently low flux level such that adequate margin exists to be able to deenergize the SR channels prior to reaching the SR reactor trip setpoint. Data is also recorded simultaneously from both IR and all four Power Range (PR) channels. This data, when combined with similar data at full power, permits the calculation of IR and PR channel overlap.

During power escalation, at approximately 30%, 50%, 75% and 98% power, the % power outputs from all four PR channels are aligned to be within ±1% of reactor thermal power (calorimetric power). The 98% power execution provides additional assurance that the PR channels are properly calibrated so as not to exceed 100% power when power is increased from 98% to 100%. Additional data is recorded for use in the IR and PR overlap calculations. The measured PR channel detector currents are extrapolated to 120% power, the full instrument span, for use as needed for Quadrant Power Tilt Ratio (QPTR) calculations. The measured PR channel detector currents are also plotted as a function of calorimetric power for use in verification of detector linearity.

At approximately full power, final data of the type taken during power ascension is recorded. The PR channels are aligned to be within ±1% of calorimetric power. The full power currents are combined with those during power ascension to verify PR detector linearity. The full power IR and PR current data is evaluated to demonstrate adequate IR and PR overlap, such that adequate margin exists to be able to block the IR reactor trip. The IR data is extrapolated to 100% power and this calculated 100% power value is used to compute the IR high level rod stop and high level trip setpoints and reset values.

3.5.4 - OPERATIONAL ALIGNMENT OF NUCLEAR INSTRUMENTATION - ISU-204A (Continued)

TEST METHODOLOGY (Continued)

After Shutdown from Power Operations of at least 800 MWD/MTU, the operating high voltages and discriminator bias voltages for the SR channels and the compensating voltages for the IR channels are determined and set.

Prior to core loading the SR channel high voltages and discriminator bias voltages were set using neutron sources to produce detector currents. This test has these voltages readjusted to properly correspond to actual reactor neutron and gamma spectra producing the detector currents.

The IR detector consists of two concentric detector volumes, one sensitive to neutrons and gamma rays and one sensitive only to gammas. The current outputs from the two detector volumes are placed in opposition to one another, i.e., bucked against each other, such that the current signal components that are proportional to gammas cancel and the net current corresponds only to neutrons. To compensate for size, geometry and efficiency differences between the two detector volumes a bias current is also applied between the two volumes. The IR compensating voltages, which provide these proper bias currents, are initially set to -40 volts to ensure complete elimination of the gamma signal, even at the price of losing a portion of the neutron signal. This ensures that the channel output is forced low enough following a reactor trip to permit the SR channels to automatically reenergize. If improperly set, the large gamma signal present following a trip could cause the IR channel output to remain abnormally high for an extended period of time and prevent automatic reenergization of the SR channels. The compensating voltages are set using actual reactor neutron and gamma spectra as detector inputs to ensure proper screening of the gamma signal.

SUMMARY OF RESULTS

A minimum overlap of 1.5 decades was observed on all SR/IR and IR/PR channel combinations. Specifically, the overlaps for the four SR/IR channel combinations were observed to be more than 1.57 decades and the overlaps for all eight IR/PR channel combinations were observed to be more than 1.97 decades. Refer to Table 3.5.4-1 for detailed results. The high voltages for the SR channels were set to 1880 VDC. The discriminator bias voltages for the SR channels were set to -0.500 VDC (N31) and -0.599 VDC (N32). The compensating voltages for the IR channels were set at -23.551 VDC (N35) and -10.94 VDC (N36) with a core burnup of approximately 1400 MWD/MTU.

SUMMARY OF RESULTS (Continued)

The PR channel outputs were either found to be or were aligned to be within ±1% of calorimetric power at all power plateaus and at full power. 120% power detector currents were calculated but were not actually used for QPTR calculation. The PR channels were verified to demonstrate acceptable linearity of outputs. Refer to Figure 3.5.4-1 for a plot of channel N41 summed top and bottom detector currents as a function of calorimetric power. Similar plots were made for the other three channels. Refer to Table 3.5.4-1 for detailed results.

The 100% power IR channel outputs were calculated and the IR rod stop and trip setpoints and reset values were calculated. Refer to Table 3.5.4-1 for detailed results. Refer to Figure 3.5.4-2 for a plot of average IR detector current as a function of calorimetric power.

Only one significant problem occurred during test performance. In the 50% power plateau test execution, it was discovered that the calorimetric power value was in error due to instrument tubing leaks associated with the precision feedwater flow instrumentation. This precision instrumentation, which was installed specifically for calorimetric measurements, was separate from the permanently installed flow instruments. The leaks were repaired and the 50% power plateau test portion was repeated. Only the final 50% power plateau results are included in Table 3.5.4-1. The 30% power results were also slightly affected by this problem, but the uses of this data did not merit repetition of this data at 30% power.

Table 3.5.4-1
Nuclear Instrumentation Results Summary

	Testing Plateau				
	30%	50%	75%	98%	100%
Calorimetric Power (%)	29.31	48.04	77.59	97.7	99.6
IR N35 output (amps)	1.5E-4	2.2E-4	3.2E-4	4.5E-4	4.2E-4
IR N36 output (amps)	1.5E-4	2.3E-4	3.3E-4	4.8E-4	4.8E-4
PR N41 summed current (μ amps)	155	248	376	480	486
PR N42 summed current (μ amps)	198	320	484	606	618
PR N43 summed current (μ amps)	172	278	421	531	540
PR N44 summed current (μ amps)	172.1	280	424	534	544
*PR N41 Power (%)	29.8	46.5	78.0	97.4	100.0
*PR N42 Power (%)	29.8	46.0	78.0	96.5	100.1
*PR N43 Power (%)	30.0	46.5	78.0	98	100.0
*PR N44 Power (%)	30.0	46.5	78.0	97.6	100.0

* Values recorded are the as found values. The as left values were either the same as the as found values or adjusted to be within ±1% of calorimetric power.

Table 3.5.4-1
Nuclear Instrumentation Results Summary (Continued)

Source Range vs. Intermediate Range Overlaps

Channels	Overlap (decades)
N31 vs. N35	1.57
N31 vs. N36	1.57
N32 vs. N35	1.6
N32 vs. N36	1.6

Intermediate Range vs. Power Range Overlaps

Channels	Overlap (decades)
N35 vs. N41	1.97
N35 vs. N42	2.54
N35 vs. N43	2.08
N35 vs. N44	2.23
N36 vs. N41	1.98
N36 vs. N42	2.51
N36 vs. N43	2.09
N36 vs. N44	2.27

Intermediate Range Currents

	Channel	
	N35	N36
Full Power Current (amps)	4.22E-4	4.82E-4
High Level Trip Setpoint (amps)	1.06E-4	1.21E-4
High Level Trip Technical Specification Limit (amps)	1.33E-4	1.52E-4
High Level Rod Stop (amps)	0.84E-4	0.96E-4

NOTE: The IR High Level Trip Setpoint is the current equivalent to ≤25% of full power, the Rod Stop is at ≤20% of full power and the Technical Specification Limit is at ≤31.5% of full power. The reset values are nominally calculated to be 1/2 of the actuation values.

Figure 3.5.4-1
Power Range Current vs. Calorimetric Power

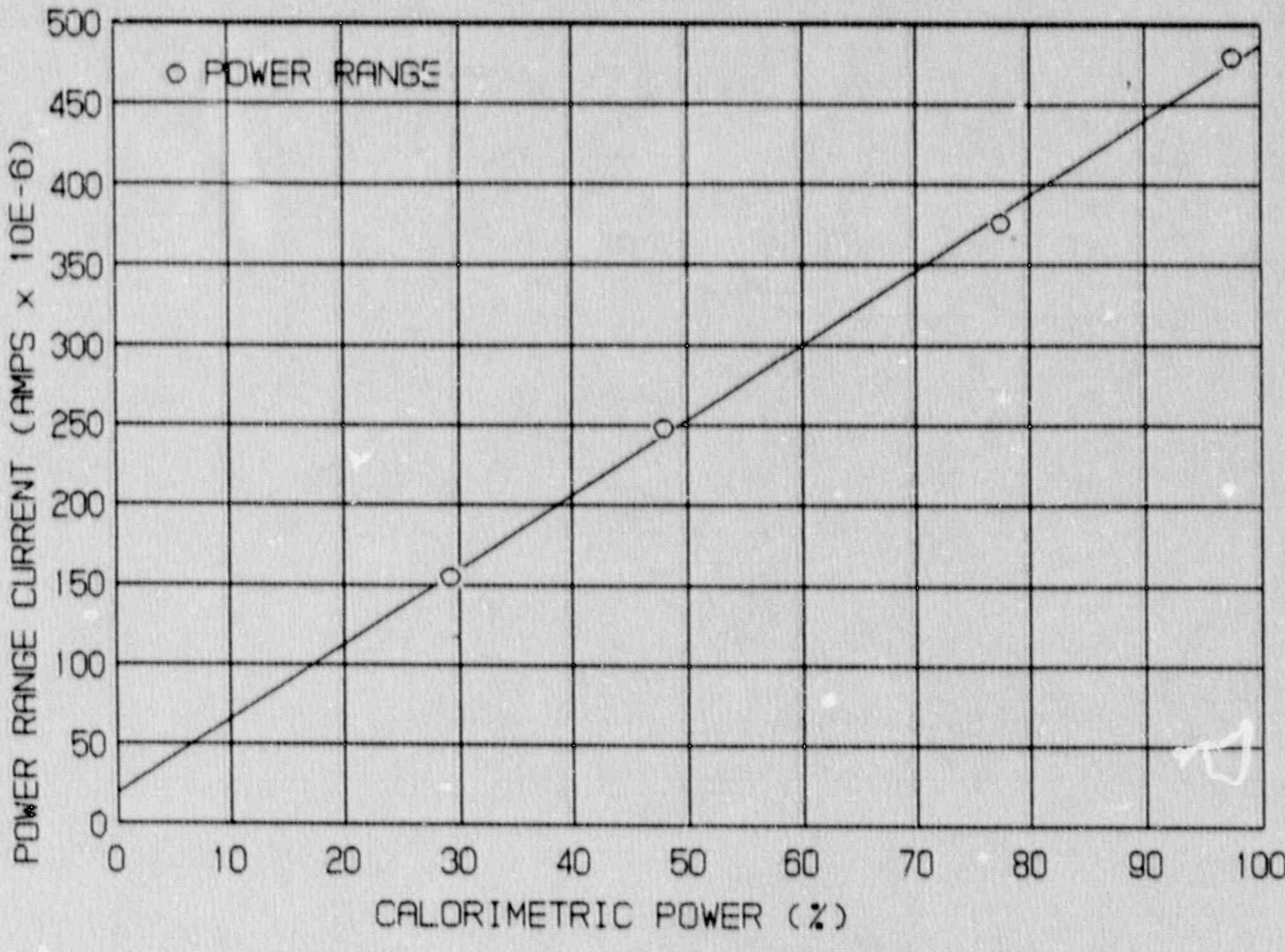
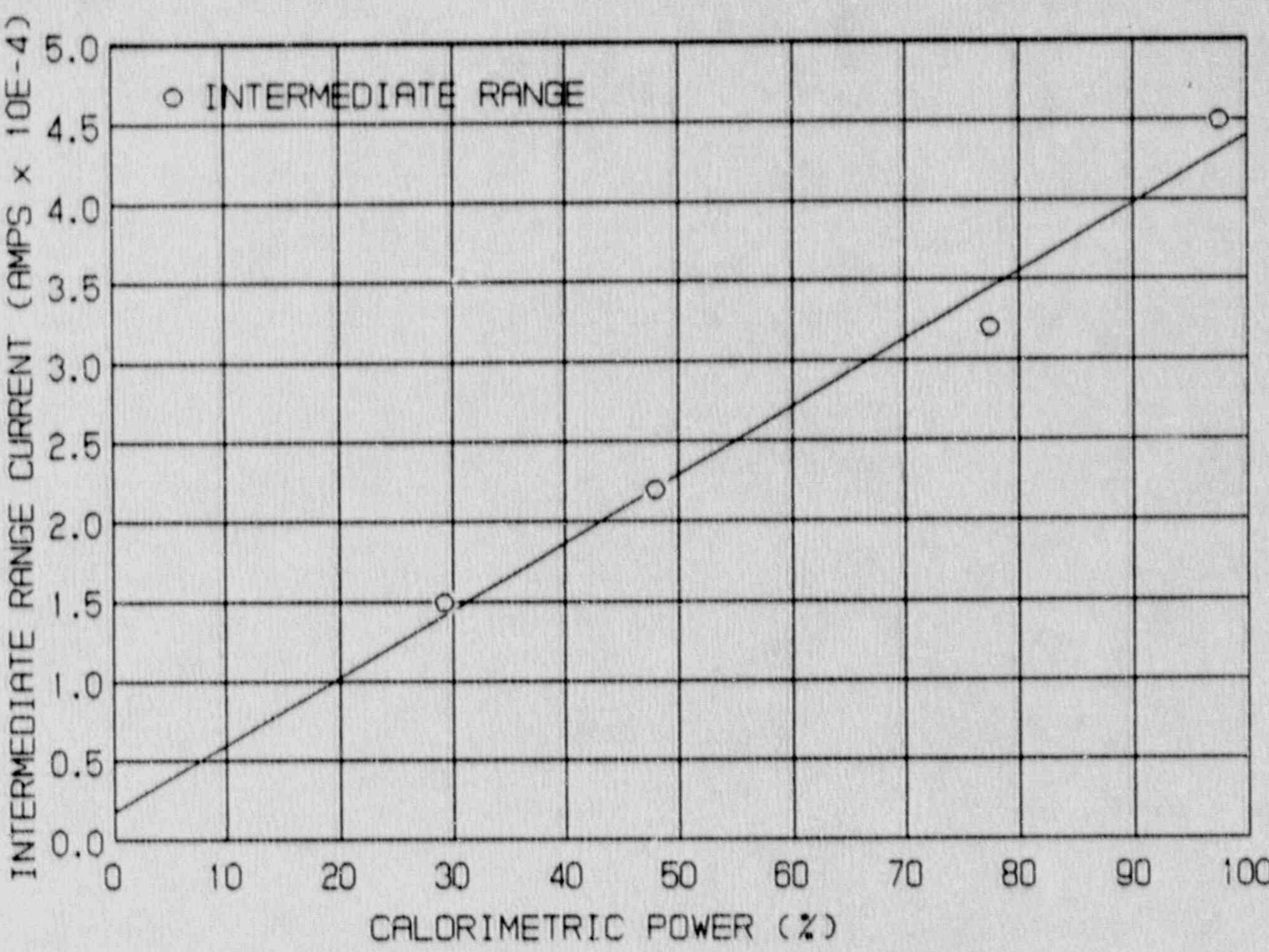


Figure 3.5.4-2
Intermediate Range Current vs. Calorimetric Power



3.5.5 - INCORE/EXCORE DETECTOR CALIBRATION - NUC-203

OBJECTIVE

This permanent plant procedure is performed to assure that a linear relationship exists between the excore neutron currents and the incore Axial Flux Difference (AFD). Once established, this excore current/AFD relationship is used to perform various calibrations of the excore channels, the OTN16 AFD inputs, Axial Flux Difference indications and plant process computer inputs. This procedure partially satisfies activities described by FSAR Table 14.2-3, Sheet 22 and Technical Specification 3/4.3.1.1.

TEST METHODOLOGY

For the 50% power execution of this procedure, a base case full core flux map is taken at stable core conditions. A small reactor coolant dilution is made and Control Bank D is inserted 15 to 25 steps to compensate for this reactivity change, with reactor power held constant. The effect is to push neutron flux toward the bottom of the core which makes AFD more negative. With AFD more negative than the base case, a quarter core flux map is taken. The reactor coolant is then borated to restore Control Bank D to its original position. The small magnitude (approximately 5% AFD) and short duration (approximately 1 hour) of this AFD change does not result in any significant residual Xenon transient effects on the core. The following data is taken during both flux maps: calorimetric power, excore nuclear detector currents and main control board and P2500 process computer AFD and Axial Offset indications. The flux map axial power distribution (top half of core vs. lower half of core) results are combined with the other data to compute the proper calibration constants.

The incore flux map results are assumed to represent the correct axial power distribution and are used as the basis for all AFD indications. However, the axial power distribution, AFD, indications are based on outputs from the excore power range detectors. Due to changes in core radial power distributions the flux that the excore detectors see may not accurately represent the actual core averaged conditions. To compensate for this, the calibrations of the excore detector based AFD indications are based on incore flux map results. This procedure plots actual excore detector outputs as a function of incore flux map Δq. Δq is the relative top vs. bottom incore power distribution while AFD is the excore detector measured top vs. bottom core power distribution. When the channels are calibrated, Δq = AFD. The resulting plots allow calculation of the excore currents that would be expected to be present if the core were to be at selected incore Δq values. These selected incore Δq values would be the calibration points. These values are supplied to Instrumentation and Controls for use

TEST METHODOLOGY (Continued)

with their normal plant calibration procedures. They inject the specified current signals into the power range circuitry inputs and adjust the outputs and indications to correspond to the selected incore Δq.

The direct relationship between P2500 process computer Axial Offset and incore flux map Axial Offset is calculated as the slope of the curve for the plot of the P2500 values as a function of the incore flux map values. These slopes are input to the P2500 process computer as conversion constants to convert the excore computer inputs to correspond to incore Δq values used for reactor monitoring.

The results of the excore current vs. incore AFD plots are also used to calculate the full span (120% power) currents that would exist at the 0% AFD condition for use in Quadrant Power Tilt Ratio (QPTR) calculations.

For the 75% power execution of this procedure, an axial Xenon oscillation is created by a significant insertion of Control Bank D (up to 40 steps) in response to a reactor coolant dilution, holding this inserted position for approximately two hours and then borating the reactor coolant to restore Control Bank D to its starting position. While Control Bank D was deeply inserted, Xenon is preferentially depleted and Iodine preferentially produced in the lower half of the core. When Control Bank D is withdrawn the neutron flux shifts toward the top of the core over time as the Iodine decays to Xenon in the lower half of the core. While the flux moves upward, with the corresponding positive change in AFD, numerous quarter core flux maps are taken. Full core flux maps are taken prior to the Control Bank D insertion and with Control Bank D at its maximum inserted position. The same plant data is taken during the flux maps as was done at 50% power. The same calculations are also performed as at 50% power. The 75% power results are generally expected to be more accurate than those at 50% power due to a reduction of the adverse temperature redistribution effects with increasing core delta temperature. Following completion of data acquisition, the axial Xenon transient is suppressed using permanent plant procedure NUC-118, "Xenon Oscillation Dampening".

For the 100% power execution of this procedure, a full core flux map is performed at stable core conditions and the same plant data is taken as done at lower power levels. A comparison is made

TEST METHODOLOGY (Continued)

between the indicated AFD values and the incore flux map AFD. If the comparison is satisfactory, no adjustments are made to the AFD circuitry. If the comparison is not satisfactory, the AFD circuitry would be recalibrated based on a combination of 75% power and the 100% power data or by generation of a much smaller axial Xenon transient at 100% power with data acquisition performed as at 75% power.

SUMMARY OF RESULTS

The excore detector data and incore flux map results were successfully used to calculate the calibration parameters for the OTN16 inputs, P2500 process computer inputs and Axial Flux Difference indications at both the 50% and 75% reactor power levels. The 100% power results were satisfactory with no recalibrations required.

During the 50% power test, performed at approximately 47% power, two retests were performed. The Δq values for the full core flux map and the quarter core flux map were not sufficiently far enough apart to yield reliable results. A third flux map was taken as Retest #1 and used in conjunction with the first map, the full core map. The full core flux map had an indicated extrapolated incore Δq of -3.09%, the first quarter core map had -5.6% and the Retest #1 quarter core map -25.4%. Retest #2 did not involve the acquisition of new data but was performed to repeat P2500 process computer input calculations due to a software methodology change. The calculations were revised to correspond to the new methodology before Retest #2 was performed. This 50% power test performance ensured that the AFD circuitry was properly calibrated prior to exceeding 50% power, where Technical Specification 3/4.2.1 first applies.

During the 75% power test, performed at approximately 77.5% power, ten flux maps were obtained ranging from an incore Δq of -22.6% to +9.5%. No retests were performed. This test performance was also used to satisfy Technical Specification Surveillance Requirement 4.3.1.7.

During the 100% power test, the differences between the excore AFD values and the incore flux map Δq were all less than the allowed maximum of 3% and, when statistically combined using the square root of the sum of the squares method, the difference was less than the allowed maximum of 8%. Therefore, no instrumentation adjustments were necessary. No retests were performed. This test performance was also used to satisfy Technical Specification Surveillance Requirement 4.3.1.1.2a.

3.5.5 - INCORE/EXCORE DETECTOR CALIBRATION - NUC-203 (Continued)

SUMMARY OF RESULTS (Continued)

Refer to Table 3.5.5-1 for detailed test results. Refer to Figures 3.5.5-1 through 3.5.5-3 for example plots of AFD and Control Bank D position during the 75% power axial Xenon transient and for example results for Power Range Channel N41.

Table 3.5.5-1
Incore/Excore Detector Calibration Summary

50% Power (Incore Δq value in %)

NIS Channel	Upper Detector Current(μamps)	Lower Detector Current(μamps)	Slope of Incore vs. Excore Axial Offset
N-41	(2.9038 x Incore Δq) +310.42	(-2.1716 x Incore Δq) +326.56	1.0080
N-42	(4.0111 x Incore Δq) +433.01	(-2.3289 x Incore Δq) +388.18	1.0317
N-43	(3.2495 x Incore Δq) +352.14	(-2.3233 x Incore Δq) +354.36	1.0166
N-44	(3.3056 x Incore Δq) +354.58	(-2.3481 x Incore Δq) +358.07	1.0095

75% Power (Incore Δq value in %)

NIS Channel	Upper Detector Current(μamps)	Lower Detector Current(μamps)	Slope of Incore vs. Excore Axial Offset
N-41	(2.3463 x Incore Δq) +283.63	(-1.9417 x Incore Δq) +304.78	1.1248
N-42	(3.2688 x Incore Δq) +397.03	(-2.0720 x Incore Δq) +362.81	1.1682
N-43	(2.6572 x Incore Δq) +322.10	(-2.0290 x Incore Δq) +331.44	1.1414
N-44	(2.7179 x Incore Δq) +325.71	(-2.0759 x Incore Δq) +335.08	1.1270

100% Power

NIS Channel	Excore AFD (%)	Incore Δq(%)	Difference(%)
N41	-7.56	-8.535	0.975
N42	-7.14	-8.535	1.395
N43	-7.31	-8.535	1.225
N44	-7.35	-8.535	1.185
		Square root of the sum of the Squares	2.409%

Figure 3.5.5-1
Incore/Excore Calibration - Plot of AFD vs. Time

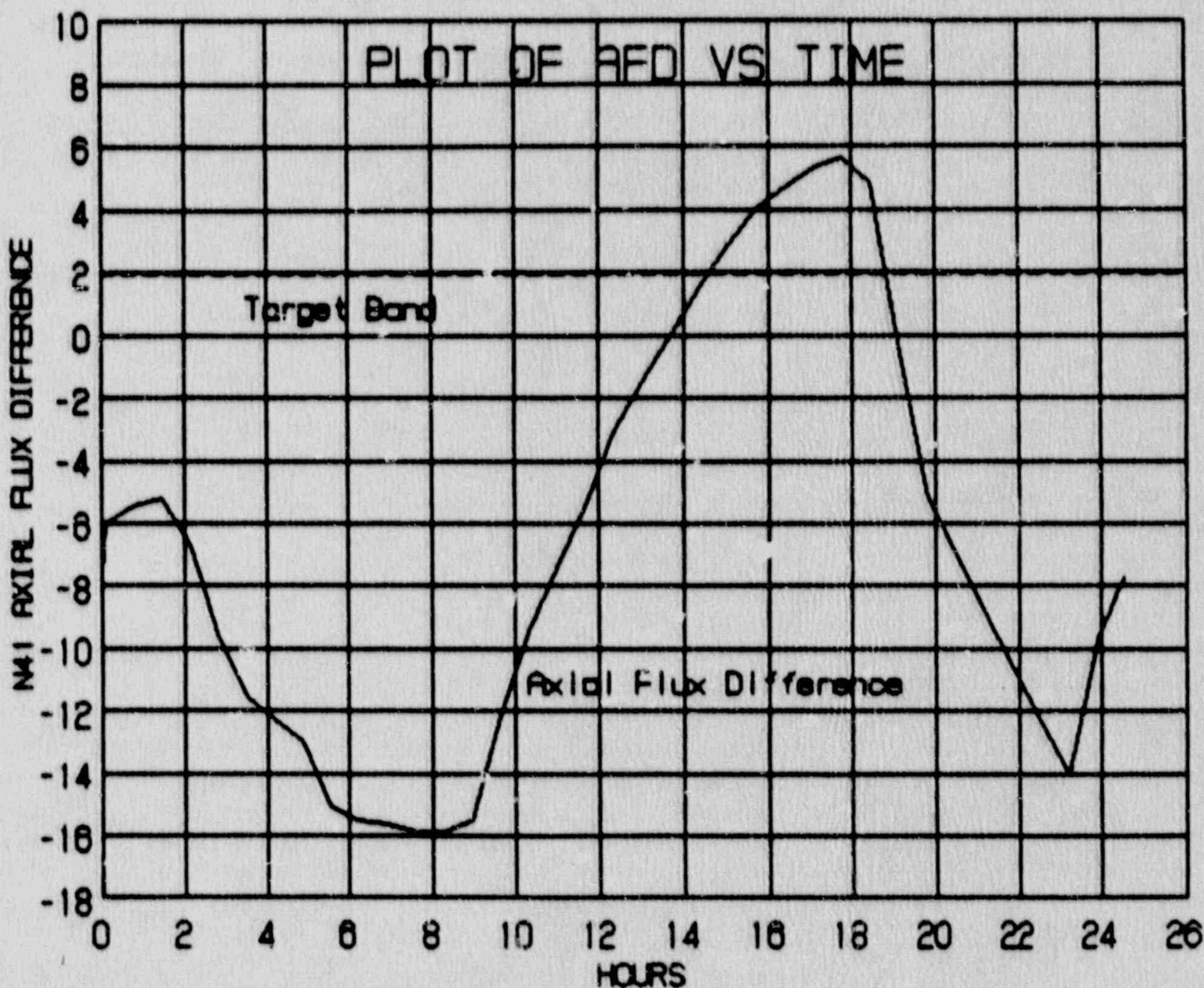


Figure 3.5.5-2
Incore/Excore Calibration - Control Bank D Position vs. Time

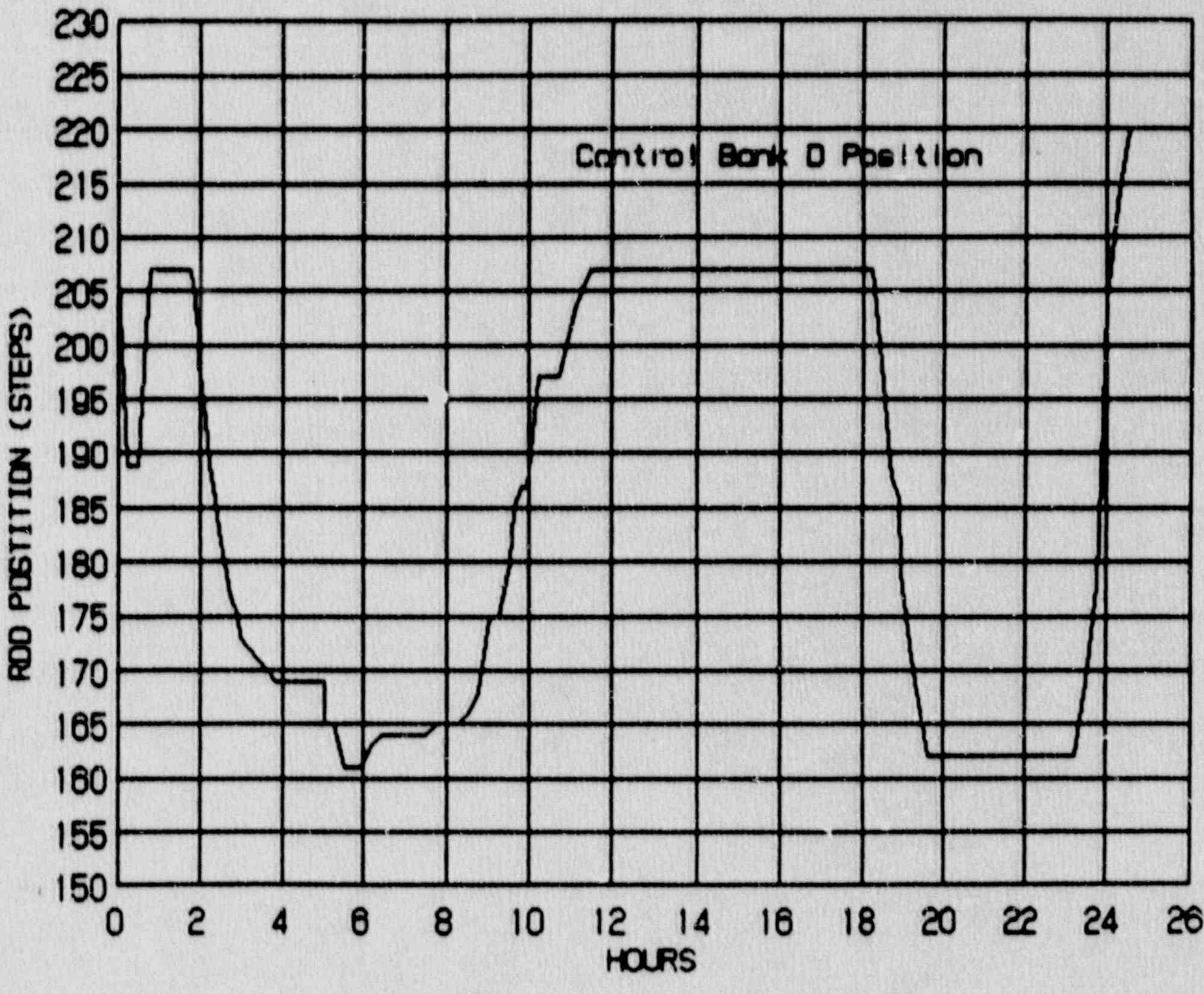
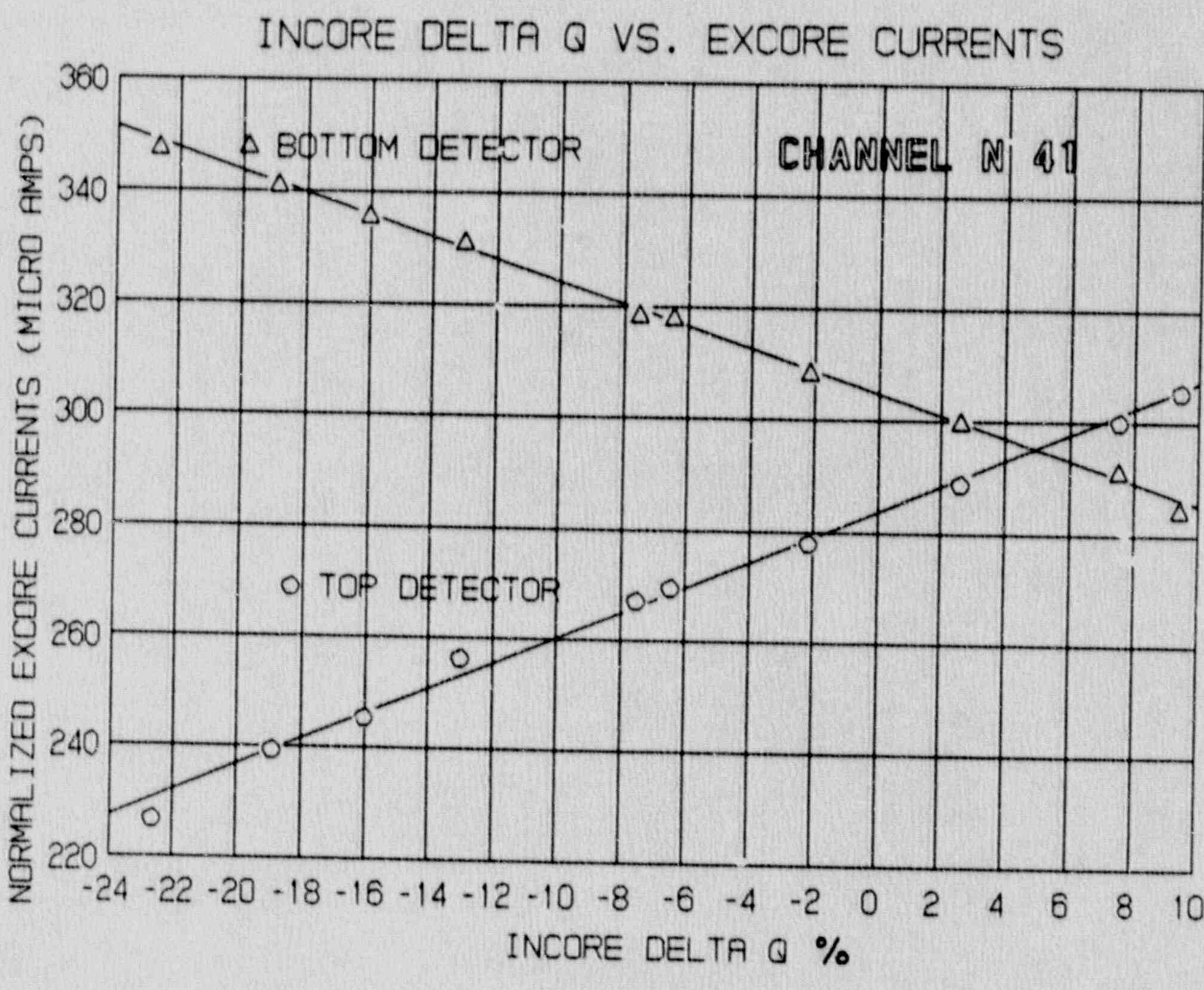


Figure 3.5.5-3
Example Axial Flux Difference Calibration - Channel N41



3.5.6 - LOOSE PARTS MONITORING BASELINE DATA - ISU-211A

OBJECTIVE

This test is performed to gather noise frequency response data in Modes 5 and 3 and at approximately 0%, 50%, 75% and 100% reactor power. This data is used as a reference baseline for analyzing suspected loose parts in the NSSS and to verify proper alarm levels and noise filter settings.

TEST METHODOLOGY

At each specified plant condition, a background noise recording of each of the 20 loose parts accelerometer channels is made using the permanently installed recording equipment. The recorded data is then played back through an oscilloscope to verify adequate signal quality.

SUMMARY OF RESULTS

A comprehensive summary of the results of Loose Parts Monitoring System testing including preoperational impact testing results, filter settings and background noise spectra is provided in Attachment A pursuant to Regulatory Guide 1.133 requirements for loose parts monitoring system testing.

Baseline data was satisfactorily collected in this test to provide a baseline for each of the 20 accelerometer channels. Testing was performed without incident except for several minor problems as follows:

During the initial test performance in Mode 5, the output signals were distorted and erratic. The recorder heads were cleaned and demagnetized and the data was retaken. Also, the alarms associated with module LPM-8 would not clear. The sensor cable and line driver associated with LPM-8 were replaced and the channel was successfully retested.

At 30% power, one of the two installed recorders was noted to be making excessive noise. The recorder was repaired and testing resumed. It was noted that the output from accelerometer #4 was sporadically erratic. The output was not erratic during the recording of the output signal from this accelerometer. The cable connectors for this accelerometer were cleaned and tightened and the problem did not recur.

3.5.7 - STARTUP ADJUSTMENTS OF REACTOR CONTROL SYSTEMS - ISU-020A

OBJECTIVE

This test is performed to determine the average RCS temperature (Tavg) value which results in establishment of the design steam pressure at full load within the temperature limits for the maximum allowable Tavg. This is accomplished by making adjustments to the reference Tavg (Tref) program and rescaling the turbine impulse pressure instrumentation, as necessary. Pressurizer level is also verified to correspond to the proper programmed value as a function of power.

TEST METHODOLOGY

At approximately 30%, 50%, 75%, 90% and 100% power, plant data is taken for use in evaluation and extrapolation of the Tref program and turbine impulse pressure (Pimp) program value. This data consists of calorimetric power, Tavg values, pressurizer level and level setpoints, Tref, steam pressures, turbine impulse pressures, main generator electrical output and feedwater flows. Data is also taken to verify proper response of the pressurizer level control program as a function of power.

A change in Tavg results in a change to average steam generator saturation temperature which directly affects steam generator saturation pressure. The rod control system automatically functions to maintain Tavg at, or very close to, Tref. The value of Tref increases as a programmed function of power. In order to optimize steam pressure, the test evaluates Tavg, Tref and steam generator pressure and calculates what change in Tref would be necessary to alter Tavg by the proper amount to result in the optimal steam generator pressure. The optimal steam generator pressure is assumed to be the full power design pressure of 1000 psia. There is an upper limit on Tref of 589.2°F, the highest Tref value assumed in the accident analyses. Full power Tref is initially set to 589.2°F.

The power input to the Tref program comes from turbine impulse pressure. Pimp increases linearly with turbine-generator output and the predicted values may require rescaling to correspond to actual impulse pressures. Pimp data is taken and extrapolated to full power for comparison with the vendor supplied Pimp predictions. Any significant deviations in the Pimp program would have to be corrected by recalibration of the Pimp channel or taken into account in the calculation of the new Tref program. These Tref and Pimp extrapolations are made at 75%, 90% and 100% power only.

3.5.7 - STARTUP ADJUSTMENTS OF REACTOR CONTROL SYSTEMS - ISU-020A (Continued)

TEST METHODOLOGY (Continued)

Actual pressurizer levels are compared to calculated program levels based on the actual power levels. The level control setpoint values are thus also verified to be proper. This comparison is made at 75%, 90% and 100% power only.

Refer to Figure 3.5.7-1 for example plots of Tref, Pimp and Pressurizer Level as functions of power and Figure 3.5.7-2 for an example plot of steam pressure vs. power.

SUMMARY OF RESULTS

At the 75% power plateau, Tref was extrapolated to a full power value of 587.12°F which was verified to be below the design maximum of 589.2°F. Steam generator pressure was extrapolated to 1017.5 psia which did not match the design range of 1000 ±10 psia. Turbine impulse chamber pressure was extrapolated to 865 psia at 100% power. This compared favorably with the 880 psia vendor supplied prediction.

At 90% power, Tref extrapolated to 587.24°F. This was again verified to be below the design maximum of 589.2°F. Steam generator pressure was extrapolated to 1016.3 psia which was also not within the design range of 1000 ±10 psia. Turbine impulse chamber pressure was extrapolated to 893 psig at 100% power. This also compared favorably with the 880 psia vendor supplied prediction.

At the 100% power plateau, Tref extrapolated to 587.3°F. This was again verified to be below the design maximum of 589.2°F. Steam generator pressure was extrapolated to 1015.0 psia which was not within the range of 1000 ±10 psia. Therefore, an adjustment of approximately -1.9°F to the Tref program value was required to reduce the steam generator pressures by approximately 15 psi. The full power extrapolated turbine impulse pressure of 907 psia was higher than the 880 psia vendor supplied prediction. This 3% difference did not adversely affect the Tref program use of the Pimp signal because the output clips at 100% Pimp power. Therefore, even though Pimp power would indicate as 103% when actual power was at 100%, the Tref output would be the 100% value. This 3% error was considered insignificant and the full power impulse pressure adjustments were not performed.

3.5.7 - STARTUP ADJUSTMENTS OF REACTOR CONTROL SYSTEMS - ISU-020A (Continued)

SUMMARY OF RESULTS (Continued)

Pressurizer level was found to deviate from the calculated program value by -0.2% at 75% power, by -0.61% at 90% power and by -1.07% at 100% power. This satisfied the $\leq \pm 3\%$ allowed deviation criterion.

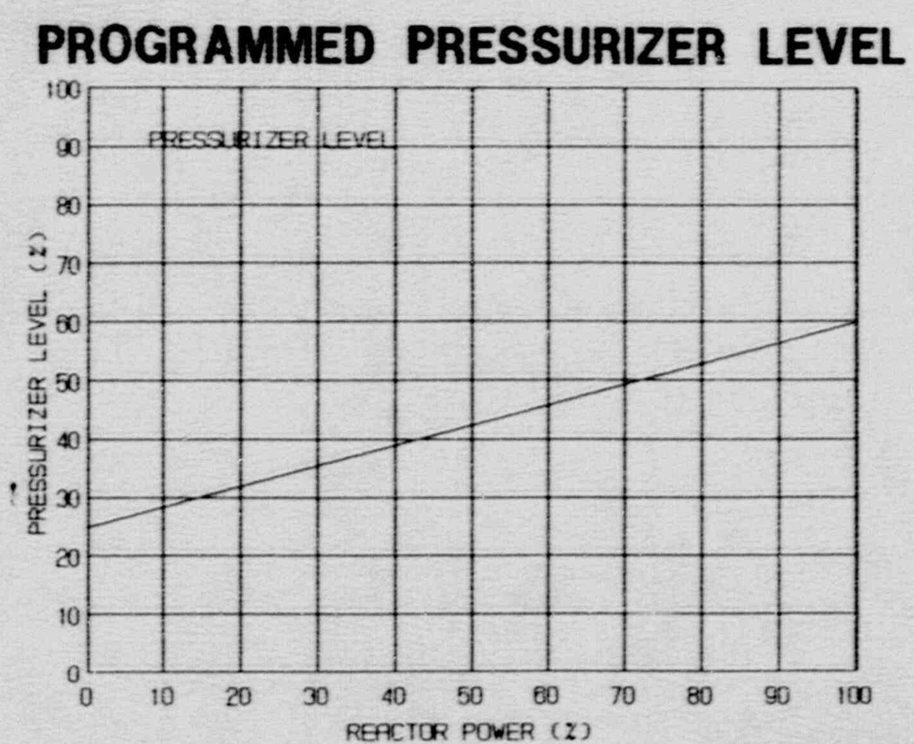
Refer to Table 3.5.7-1 for detailed results.

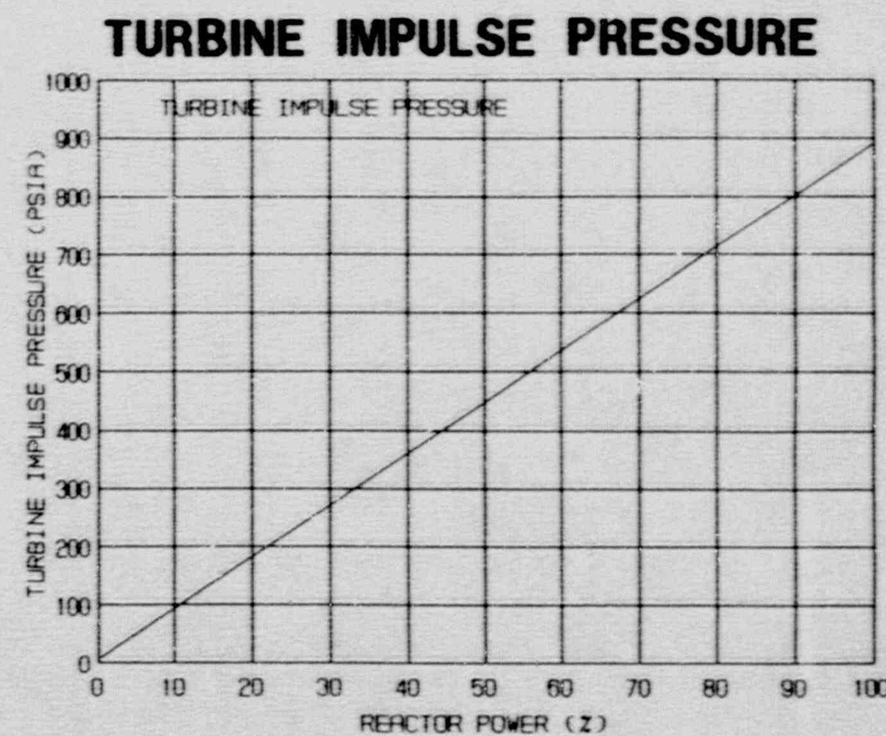
TABLE 3.5.7-1

Startup Adjustments Summary

	30%	50%	75%	90%	100%
Calorimetric Power (%)	29.30	48.14	76.57	89.36	99.77
Tref (°F)	564.1	572.9	581.2	586.2	589.5
Tavg (°F)	564.3	572.2	581.3	584.5	588.6
Pressurizer Level (%)	32.90	41.96	51.64	55.50	60.46
Calculated Pressurizer Level (%)	N/A	N/A	51.44	54.89	59.39
Actual Pressurizer Level Setpoint (%)	33.04	42.01	50.89	56.05	60.40
Average impulse pressure (psia)	198.1	438.9	662.3	798.0	905.0
Average steam generator pressure (psia)	1068.0	1070.0	1044.4	1020.7	1015.2
100% Extrapolated steam generator pressure (psia)	N/A	N/A	1017.5	1016.3	1015.2
Gross Elcctric Output(MWe)	230	520	921	1128	1155

Figure 3.5.7-1


PROGRAMMED PRESSURIZER LEVEL
PRESSURIZER LEVEL
PRESSURIZER LEVEL (%)
REACTOR POWER (%)
0
10
20
30
40
50
60
70
80
90
100


TURBINE IMPULSE PRESSURE
TURBINE IMPULSE PRESSURE
TURBINE IMPULSE PRESSURE (PSIA)
REACTOR POWER (%)
0
100
200
300
400
500
600
700
800
900
1000

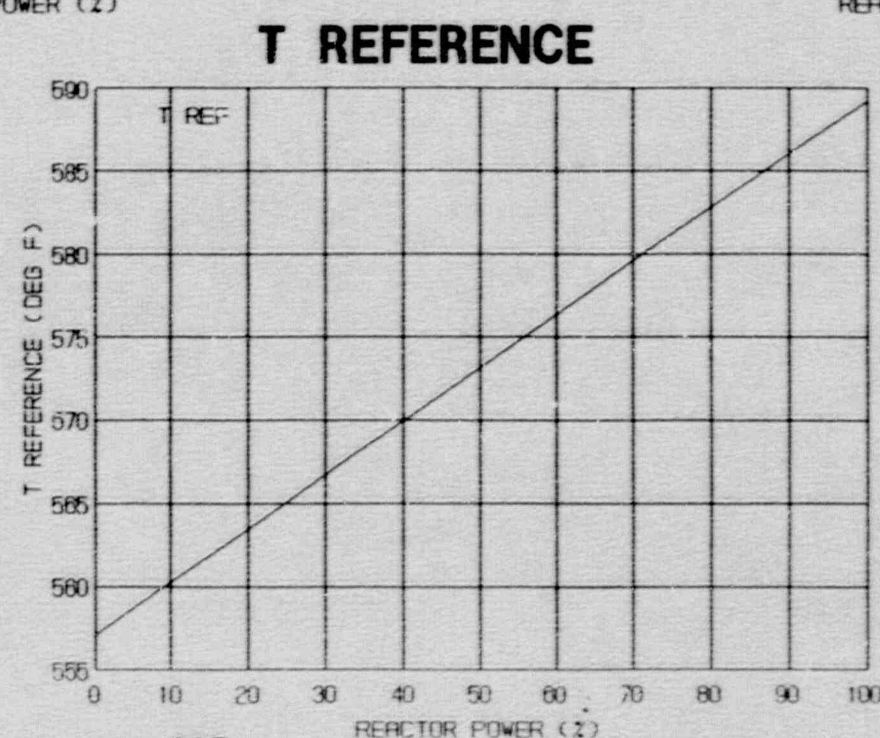
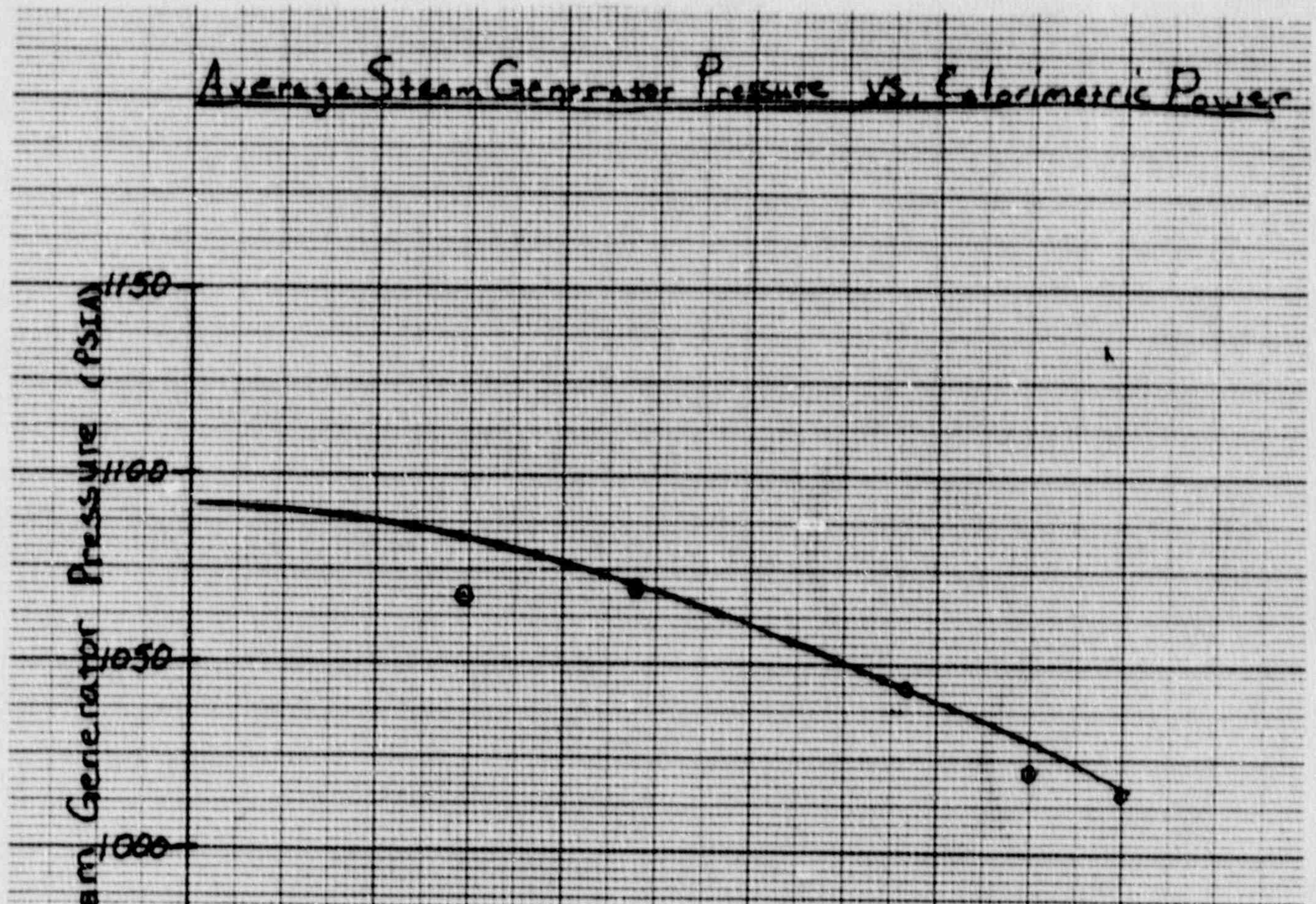

T REFERENCE
T REF
T REFERENCE (DEG F)
REACTOR POWER (%)
555
560
565
570
575
580
585
590

Figure 3.5.7-2
Steam Pressure vs. Power



3.5.8 - FULL POWER PERFORMANCE TEST - ISU-281A

OBJECTIVE

This test is performed to demonstrate the reliability of the Nuclear Steam Supply System (NSSS) to maintain its warranted output of 3425 MWth (+0, -5%) for 100 hours without a load reduction or plant trip resulting from an NSSS malfunction, to demonstrate the ability of the plant to generate 1150 MWe (+0, -5%) for 100 consecutive hours and to demonstrate the ability of the NSSS to develop 3425 MWth (+0, -1%) at a steam generator pressure of $\geq$990 psia.

TEST METHODOLOGY

The test is initiated with the plant operating within 5% of its rated NSSS output as determined by power range nuclear instrumentation which is calibrated to correspond to calorimetric power and at a steam pressure of $\geq$990 psia. Plant conditions are then stabilized at their design values for 35 to 65 hours with stability verified by calorimetric data acquisition. Power is verified to be or is increased to be within 1% of the 3425 MWth design NSSS output and a four hour performance measurement is initiated to verify actual NSSS power by collecting calorimetric data every 5 minutes and computing power hourly. The remaining hours of the 100 hour minimum duration are then completed.

SUMMARY OF RESULTS

The 100 hour run was started at 2100 hours on 7-17-90 and completed at 0100 hours on 7-22-90. However, plant conditions were maintained and the test was continued until 0900 nours on 7-23-90. This resulted in a total documented run of 132 hours. The first 48 hours of the run occurred prior to the formal start of the test. Credit was taken for these 48 hours based on Reactor Operator logs and data acquisition system records. The minimum and maximum hourly NSSS power measurements were 99.11% and 100.58% of the rated power of 3425 MWth over the entire 132 hour duration. There was no load reduction or plant trip during the 132 hour duration.

A net electrical output of between 1093.4 and 1108.6 MWe (95.08% to 96.40% of 1150 MWe) was demonstrated over the entire 132 hour duration. This satisfied the 1150 MWe (+0, -5%) criterion. The average output over the 100 hour duration was 1102.6 MWe, 95.88% of 1150 MWe.

The four hourly calorimetrics demonstrated NSSS output to be between 3415.9 and 3420.3 MWth, corresponding to a range of 99.73% to 99.86% power, at a steam generator outlet pressure of $\geq$990.42 psia. This satisfied the 3425 MWth (+0, -1%) at $\geq$990 psia criterion.

3.5.9 - P2500 PROCESS COMPUTER SOFTWARE VERIFICATION - ISU-019A

OBJECTIVE

The P2500 Process Computer Software Verification test is performed to verify that the process computer receives correct inputs from selected process variables in the field and to validate selected performance calculations performed by the process computer. This verification and validation is performed by comparing the output from the process computer to permanent plant instrumentation and the Test Data Acquisition System (TDAS) computer output.

TEST METHODOLOGY

At plant power levels of 0%, 50%, 75% and 100%, selected analog and digital plant parameters monitored by the P2500 Westinghouse process computer are compared to Main Control Board instruments and/or outputs from the TDAS computer to ensure they agree within specified tolerances. The parameters selected were those judged to be most important to the operators in monitoring plant conditions and evaluating equipment performance. The tolerances are based on the accuracy of the instrumentation and associated instrument loops.

The following five programs are the performance calculations verified by this test:

The Total Thermal Power algorithm used by the process computer calculates the thermal output of each of the four steam generators as well as a value for total secondary calorimetric power. The results of these calculations are compared to the precision calorimetric performed by the TDAS computer.

The Heater Differential algorithm performed by the process computer calculates the temperature difference across each of the twelve feedwater heaters based on inputs from thermocouples in the feedwater system. The TDAS also takes temperature readings from the same locations and calculates the temperature differences. These temperature difference values are then compared.

The Percent Turbine Power algorithm performed by the process computer calculates turbine electrical load based on a linear function conversion of the turbine impulse pressure. The output is provided to a digital meter on the Main Control Board. Data is recorded in this test to correlate turbine impulse pressure to reactor thermal power and to correlate calorimetric power to generator megawatts. The data is provided to Engineering for determination of the proper calibration constants to be used in the P2500 process computer for this linear function conversion.

The Calibration Check of Power Range Nuclear Channels algorithm performed by the process computer compares a calculated average

3.5.9 - P2500 PROCESS COMPUTER SOFTWARE VERIFICATION - ISU-019A (Continued)

TEST METHODOLOGY (Continued)

Nuclear Instrumentation System (NIS) power range power output with the reactor thermal power calculated in the Total Thermal Power algorithm. This comparison is provided to appraise the operator of a possible drift in the power range channel outputs. The calorimetric power calculated by the TDAS is used in this test to evaluate the comparison made by the P2500 process computer.

The Quadrant Power Tilt Ratio algorithm performed by the process computer calculates upper, lower, and average radial flux tilts on a quadrant basis using inputs from the NIS power range detectors. The results of this calculation are compared to the manual calculations performed in accordance with the permanent plant surveillance procedure.

SUMMARY OF RESULTS

During the 0% power level test, all of the digital inputs to the P2500 were verified correct. Approximately one third of the analog signals could not be verified at this power level for various reasons including instruments out of service, instruments off-scale due to plant conditions, TDAS non-availability, instruments out of calibration, and incorrect scaling or processing by the P2500 process computer. Instruments were calibrated as necessary and retesting was performed at higher power levels.

During the 50% power level test, nine of the computer addresses failed the instrument correlation. Three required modification to the computer database to correct the engineering ranges, three failed due to the inaccuracies of the Main Control Board indicators at the low end of their scales, two could not be tested due to unavailability of the TDAS channels, and one had a wiring error in the field. The remainder of the inputs passed the instrument correlation.

During the 75% power level test, ten of the computer addresses failed the instrument correlation. Seven needed recalibration of the input devices, two could not be tested due to unavailability of the TDAS channels, and one needed a modification to the computer database to correct the engineering range. The remainder of the inputs passed the instrument correlation.

During the 100% power level test, four of the computer addresses failed the instrument correlation. Three needed recalibration of the input devices and one needed a modification to the computer database to correct the engineering range. The remainder of the inputs passed the instrument correlation.

3.5.9 - P2500 PROCESS COMPUTER SOFTWARE VERIFICATION - ISU-019A (Continued)

SUMMARY OF RESULTS (Continued)

The comparisons of the process computer's performance calculations at the various power levels are listed in Table 3.5.9-1. The values of % Mean Deviation listed are calculated from the following equation:

$$\%\ \text{Mean Deviation} = \left|\frac{\text{Comparison Value} - \text{P2500 Value}}{\text{Comparison Value} + \text{P2500 Value}}\right| \times 100$$

The recorded results values have been rounded off. The % Mean Deviation values were calculated using the not rounded raw values.

During the 50% power test, the Total Thermal Power algorithm was satisfactory. The Heater Difference algorithm was not successful because the feedwater heater thermocouples were incorrectly wired to the process computer. The thermocouple wiring was corrected and the process computer was determined to have been calculating correctly but had been receiving incorrect inputs. The Power Range Nuclear Channels Calibration algorithm was successfully tested at 50% power. The data was collected for the Percent Turbine Power Meter for transmittal to Engineering.

During the 75% power test, the Total Thermal Power algorithm was satisfactory. The Heater Difference algorithm was not successful because the feedwater heater system was not in the normal operating configuration and one channel of TDAS was out of service. This testing was deferred to a higher power level. The Power Range Nuclear Channels Calibration algorithm was successfully tested at 75% power. The Quadrant Power Tilt Ratio algorithm was tested for the first time at 75% power and was satisfactory. Additional data was collected for the Percent Turbine Power Meter for transmittal to Engineering.

During the 100% power test, the Total Thermal Power algorithm passed the review criterion. However, a difference of 98.7 MWth existed between the P2500 process computer and the TDAS precision calorimetric. The major contributor to this difference was an error in the feedwater temperature input to the P2500 program that resulted in readings that were approximately 15°F high at 100% power. This was caused by errors in the linear approximation of the thermocouple curve used in the analog circuits. The Heater Difference algorithm was successfully verified at 100% power. The temperature differences across feedwater heaters 2A and 6B were not within the review criteria but were accurate to within 2%, and also 2°F, which was judged acceptable. The Power Range Nuclear Channels Calibration and Quadrant Power Tilt Ratio algorithms were successfully verified. The data for the Percent Turbine Power Meter was transmitted to Engineering for calculation of the calibration constants.

TABLE 3.5.9-1

PROCESS COMPUTER ALGORITHM COMPARISONS

TOTAL THERMAL POWER

The Review Criterion for % Mean Deviation is ≤2%

Power Level	P2500 Process Computer(MWth)	TDAS(MWth)	% Mean Deviation
50%	1621.8	1627.0	0.16
75%	2628.8	2662.1	0.63
100%	3318.7	3417.4	1.45

HEATER DIFFERENCE

The Review Criterion for % Mean Deviation is ≤1%

100% Power Temperature Differences

Feedwater Heater	P2500 Process Computer(°F)	TDAS(°F)	% Mean Deviation
1A	41.7	41.3	0.45
1B	41.5	42.0	0.59
2A	42.8	44.1	1.6
2B	41.7	42.2	0.67
3A	81.1	80.3	0.44
3B	82.4	81.9	0.29
4A	54.0	53.9	0.07
4B	54.6	54.3	0.28
5A	69.0	69.0	0.05
5B	69.4	69.6	0.11
6A	38.0	38.1	0.14
6B	23.7	24.3	1.1

NOTES: 1) Thermocouples were incorrectly wired for 50% power test and the data was inconclusive.

2) The Feedwater Heater System was not in its normal configuration at 75% power and the data was not representative of actual plant conditions.

3) The % mean deviations for Feedwater Heaters 2A and 6B exceeded the 1% Review Criterion for 100% power. It was determined that the process computer calculations for such low temperature differences was acceptable.

4) The temperature difference values recorded were rounded off. The % Mean Deviation values were calculated based on not rounded values.

TABLE 3.5.9-1

PROCESS COMPUTER ALGORITHM COMPARISONS
(Continued)

CALIBRATION CHECK OF POWER RANGE NUCLEAR CHANNELS

The Review Criterion is % Mean Deviation <2% OR Actual % Power Difference < ±2.5% (for 50%, 75% results) and % Power Difference <1.5% (for 100% results)

Power Level	P2500 (NIS)	TDAS Calorimetric	% Mean Deviation
50%	46.8%	47.7%	1%
75%	77.7%	78.05%	0.22%
100%	99.7%	99.93%	0.11%

QUADRANT POWER TILT RATIO

The Review Criterion for % Mean Deviation is ≤2%

75% Power

Upper Radial Tilt	P2500 Process Computer	Hand Calculation	% Mean Deviation
N41	0.995	0.998	0.15
N42	1.002	0.996	0.30
N43	1.001	0.996	0.25
N44	1.002	1.006	0.20

Lower Radial Tilt	P2500 Process Computer	Hand Calculation	% Mean Deviation
N41	0.999	0.997	0.10
N42	0.996	0.994	0.10
N43	1.003	1.008	0.25
N44	1.002	0.997	0.25

100% Power

Upper Radial Tilt	P2500 Process Computer	Hand Calculation	% Mean Deviation
N41	1.005	1.0017	0.16
N42	0.995	0.9952	0.01
N43	0.999	0.9990	0.0
N44	1.001	1.0041	0.15

Lower Radial Tilt	P2500 Process Computer	Hand Calculation	% Mean Deviation
N41	1.007	1.0034	0.18
N42	0.991	0.9896	0.07
N43	1.001	1.0081	0.35
N44	1.002	0.9989	0.15

3.5.10 - AUTOMATIC REACTOR CONTROL SYSTEM TEST - ISU-203A

OBJECTIVE

This procedure is performed to demonstrate the capability of the automatic reactor control system to maintain Reactor Coolant System average temperature (Tavg) within an acceptable tolerance about the reference Tavg (Tref) under steady state and transient conditions. Tref is the programmed Tavg setpoint as a function of power. This procedure satisfies activities described by FSAR Table 14.2-3, Sheets 4 and 33.

TEST METHODOLOGY

With reactor power stabilized at approximately 50% and Tavg matched to Tref, rod control is placed in automatic to monitor Tavg for oscillations. After approximately ten minutes, Tavg is manually increased to approximately 5°F higher than Tref by manual withdrawal of Control Bank D. Rod control is then placed in automatic and Tavg is allowed to return to and stabilize within approximately ±1.5°F of Tref by automatically controlled Control Bank D motion. After Tavg has stabilized, rod control is again placed in manual to decrease Tavg to approximately 5°F lower than Tref by manual insertion of Control Bank D. Rod control is then again placed back in automatic and Tavg again allowed to return to and stabilize within approximately ±1.5°F of Tref. Various plant parameters and instrumentation signals within the automatic reactor control loops are monitored on strip chart recorders during these temperature transients. Values recorded are Tavg, Tref, nuclear flux, turbine power, turbine impulse pressure, steam header pressure, pressurizer pressure and rod control mismatch and error signals.

SUMMARY OF RESULTS

During steady state operation, it was found that Tavg was maintained within ±1.5°F of Tref with no problems. When Tavg was increased by 5°F, it took approximately 79 seconds to return Tavg to within ±1.5°F of Tref. When Tavg was decreased by 5°F, it took approximately 68 seconds to return Tavg to within ±1.5°F of Tref.

3.6 DEFERRED PREOPERATIONAL TESTING

3.6.1 - PROCESS SAMPLING SYSTEM - ISU-028A

OBJECTIVE

The Process Sampling System test is performed to demonstrate the capability of the sampling system to provide liquid and gas samples through the correct flow paths from the primary and secondary systems, to demonstrate the adequacy of plant sampling procedures and to verify sample line holdup times. The test verifies acceptable flow rates at design temperatures and pressures and verifies the operability of automatic on-line analyzers and sample coolers. This test satisfies activities described by FSAR Table 14.2-2, Sheets 6 and 6a, and the deferred preoperational testing in System Test Matrix 1-2200.

TEST METHODOLOGY

The operability of the sampling system is demonstrated by obtaining samples from the primary and secondary systems and measuring the sample flow rates, pressures, and temperatures. The on-line analyzers are compared to grab sample analyses. Plant chemistry procedures are used to obtain samples and are thus verified adequate. The reactor coolant hot leg sample lines are purged at the maximum flow rate and verified to be delayed at least 60 seconds inside the missile barrier.

SUMMARY OF RESULTS

Refer to Table 3.6.1-1 for detailed test results.

Safety Injection Accumulators #2 and #3 sample flow rates were initially too low. The sample lines were flushed and retested with satisfactory results.

Reactor Coolant Hot Leg Loops #1 and #4 sample flow rates were initially too high, resulting in holdup times inside the missile barrier of less than 60 seconds. A new valve control rod, cut to a length of 3 1/2 inches, was installed in drag valve 1PS-0252 to provide sufficient flow resistance. The final flow rates were 0.8 gpm in the purge mode and 0.73 and 0.72 gpm in the grab sample mode, respectively. These correlate to hold up times of 64.4 seconds in the purge mode and 70.6 and 71.6 seconds in the grab sample mode, respectively.

The sample line from the pressurizer steam space was initially found to be blocked. A faulty quick-disconnect fitting was repaired and the flow rate was determined to be satisfactory.

3.6.1 - PROCESS SAMPLING SYSTEM - ISU-028A (Continued)

SUMMARY OF RESULTS (Continued)

Upon completion of this test, the flowrates, pressures and temperatures were satisfactory. Initial test results indicated that the original acceptance criteria were too restrictive. These criteria were changed and FSAR Amendment 79 incorporated the modified criteria.

The plant chemistry sampling procedures were used to sample each of the sample points and were demonstrated to be satisfactory.

The Steam Generator Blowdown analyzers were verified operable by comparison with analyzed grab samples for specific conductivity, cation conductivity and sodium.

TABLE 3.6.1-1

PROCESS SAMPLING SYSTEM SUMMARY

GRAB SAMPLE MODE

SAMPLE POINT	FLOWRATE (GPM) ACTUAL	FLOWRATE (GPM) REQ'D.	PRESSURE (PSIG) ACTUAL	PRESSURE (PSIG) REQ'D.	CONDITIONED TEMPERATURE (°F) ACTUAL	CONDITIONED TEMPERATURE (°F) REQ'D.
#1 SG Blowdown	0.63	0.4-1.0	51.5	5-80	95	≤115
#2 SG Blowdown	0.69	0.4-1.0	52	5-80	105	≤115
#3 SG Blowdown	0.69	0.4-1.0	51	5-80	114	≤115
#4 SG Blowdown	0.79	0.4-1.0	49	5-80	90	≤115
Downstream SG Blowdown Cation Demin	0.32	0.05-1.0	9	5-10	80	N/A
Downstream SG Blowdown Mixed Bed Demin	0.29	0.05-1.0	9	5-10	80	N/A
CVCS Letdown Downstream of Demineralizer	0.91	≤1.0	N/A	N/A	N/A	N/A
CVCS Letdown Upstream of Demineralizer	0.77	≤1.0	N/A	N/A	N/A	N/A
RHR Train A	0.87	0.15-1.0	40	5-80	95	≤115
RHR Train B	0.50	0.15-1.0	30	5-80	70	≤115
SI Accum #1	0.81	0.15-1.0	49	5-80	83	≤115
SI Accum #2	0.78	0.15-1.0	60	5-80	84	≤115
SI Accum #3	0.75	0.15-1.0	54	5-80	85	≤115
SI Accum #4	0.85	0.15-1.0	58	5-80	84	≤115
Pzr Steam Space	0.45	0.15-1.0	62	5-80	95	≤115
RCS Hot Leg #1	0.73	0.15-1.0	74	5-80	99	≤115
RCS Hot Leg #4	0.72	0.15-1.0	75	5-80	103	≤115
Pzr Liquid	0.48	0.15-1.0	58	5-80	95	≤115
SFP Demin #1 Inlet	0.76	0.75-1.0	N/A	N/A	N/A	N/A
SFP Demin #1 Outlet	0.77	0.75-1.0	N/A	N/A	N/A	N/A
SFP Demin #2 Inlet	0.81	0.75-1.0	N/A	N/A	N/A	N/A
SFP Demin #2 Outlet	0.77	0.75-1.0	N/A	N/A	N/A	N/A

TABLE 3.6.1-1 (Continued)

PROCESS SAMPLING SYSTEM SUMMARY
(Continued)

CONTINUOUS PURGE MODE

SAMPLE POINT	FLOWRATE (GPM) ACTUAL	FLOWRATE (GPM) REQ'D.	PRESSURE (PSIG) ACTUAL	PRESSURE (PSIG) REQ'D.	CONDITIONED TEMPERATURE (°F) ACTUAL	CONDITIONED TEMPERATURE (°F) REQ'D.
#1 SG Blowdown	0.48	0.4-1.0	51	5-80	102	≤115
#2 SG Blowdown	0.64	0.4-1.0	55	5-80	101	≤115
#3 SG Blowdown	0.59	0.4-1.0	55	5-80	100	≤115
#4 SG Blowdown	0.64	0.4-1.0	54.5	5-80	90	≤115
Downstream SG Blowdown Cation Demin	0.05	≤1.0	9.5	5-10	N/A	N/A
Downstream SG Blowdown Mixed Bed Demin	0.05	≤1.0	9.5	5-10	N/A	N/A
CVCS Letdown Downstream of Demineralizer	0.55	≤1.0	N/A	N/A	N/A	N/A
CVCS Letdown Upstream of Demineralizer	0.4	≤1.0	N/A	N/A	N/A	N/A
RHR Train A	0.95	0.75-1.0	59	5-80	92	≤115
RHR Train B	0.9	0.75-1.0	54	5-80	80	≤115
SI Accum #1	0.8	0.75-1.0	58	5-80	83	≤115
SI Accum #2	0.8	0.75-1.0	45	5-80	84	≤115
SI Accum #3	0.75	0.75-1.0	45	5-80	83	≤115
SI Accum #4	0.9	0.75-1.0	45	5-80	84	≤115
Pzr Steam Space	1.0	0.75-1.0	64	5-80	97	≤115
RCS Hot Leg #1	0.8	0.75-1.0	70	5-80	95.3	≤115
RCS Hot Leg #4	0.8	0.75-1.0	66	5-80	103	≤115
Pzr Liquid	0.9	0.75-1.0	50	5-80	110	≤115

3.6.2 - IN-PLACE ATMOSPHERIC CLEANUP FILTER TEST - PRIMARY PLANT - ESF - EGT-751X

OBJECTIVE

The In-place Atmospheric Cleanup Filter Test is performed to demonstrate proper operation and integrity of the Primary Plant Ventilation ESF filtration units, including the High Efficiency Charcoal Absorbers (HECA), High Efficiency Particulate Air (HEPA) filters and unit heaters. This test satisfies activities described by FSAR Table 14.2-2, Sheet 29 and the deferred preoperational testing in System Test Matrix 1-2400.

TEST METHODOLOGY

With one ESF train in service, air flow through each filtration unit is determined by traverse air velocity measurements or from in-line flow elements and compared to Technical Specification limits. The total pressure drop across the filter housing is measured with a manometer and is also compared to Technical Specification limits.

The power input to each unit heater is determined by measurements of current and voltage. The rate of heat added to the air by each heater is determined by measurements of upstream and downstream wet and dry bulb temperatures and air flow measurements. The ratio of heat output to power input then determines the heater's efficiency.

The penetration and bypass leakage is determined for each HEPA filtration unit by injection of dioctyl phthalate (DOP) aerosol and measuring the concentration upstream and downstream of each filter. There are two HEPA filters per unit, one referred to as the upstream filter and one referred to as the downstream filter.

Each HECA filter unit is leak tested by injection of R-11 refrigerant as a tracer gas and measuring the upstream and downstream concentrations.

SUMMARY OF RESULTS

Refer to Table 3.6.2-1 for detailed test results.

This test was performed in conjunction with a separate test procedure which verified filter air flow distribution and air-aerosol mixing uniformity.

The initial testing of heater CPX-VAFUPK-01 resulted in a power calculation of 94.9 KW which failed the acceptance criterion of 100 ±5 KW. The heater was retested over a longer duration with satisfactory results. All heaters dissipated 100 ±5 KW.

3.6.2 - IN-PLACE ATMOSPHERIC CLEANUP FILTER TEST - PRIMARY PLANT - ESF - EGT-751X (Continued)

SUMMARY OF RESULTS (Continued)

All primary plant ESF filtration units were tested and satisfied the air flow requirement of 15,000 cfm ±10% with a pressure drop across the combined HEPA and HECA filters of less than 8.5 inches water gauge.

The refrigerant gas (R-11) penetration and bypass leakage of each HECA bank was less than the required maximum of 1.0% at rated flow. The DOP penetration and bypass leakage of each HEPA bank was less than the required maximum of 1.0% at rated flow.

Balancing of the HVAC air flow distributions was performed prior to this test to ensure adequate flow rates to the areas served by these HVAC systems.

TABLE 3.6.2-1

IN-PLACE ATMOSPHERIC CLEANUP FILTER TEST SUMMARY

Item	Required Performance	CPX-VAFUPK-01	CPX-VAFUPK-02	CPX-VAFUPK-15	CPX-VAFUPK-16
Air Flow(cfm)	13500-16500 at <8.5"WC	15915 at 6.2"WC	16216 at 6.3"WC	15251 at 6.1"WC	14928 at 6.3"WC
Heater Power Dissipation(KW)	95-105	96.04	97.6	96.5	99
Upstream HEPA Filter Penetration & Bypass Leakage (%)	<1.0	<0.025	<0.05	<0.05	<0.05
Downstream HEPA Filter Penetration & Bypass Leakage (%)	<1.0	<0.1	<0.05	<0.10	<0.05
HECA Leakage(%)	<1.0	0.042	0.02	0.01	<0.01

3.6.3 - CONTAINMENT & PENETRATION ROOMS TEMPERATURE SURVEY - ISU-282A

OBJECTIVE

The Containment and Penetration Rooms Temperature Survey is performed to verify that the Reactor Coolant pipe penetrations, air supply to Reactor Vessel Supports, Neutron Detector Well discharge air, containment air, Steam Generator compartment air, Pressurizer room air, CRDM shroud air, CRDM platform area air, and Feedwater and Main Steam penetration rooms are maintained at or below their design temperatures when the RCS is at normal operating temperature and also when the RCS is at nominal full power conditions. This test satisfies activities described in FSAR Section 9.4.A, and the deferred preoperational testing in System Test Matrix 1-3600.

TEST METHODOLOGY

The concrete temperature around each Reactor Coolant System (RCS) pipe penetration is measured with a thermocouple when the RCS is at normal operating temperature in Mode 3. Temperatures are recorded from permanent plant instrumentation for Neutron Detector Well exhaust air, CRDM shroud exhaust air and containment air. Local readings using thermocouples or resistance temperature detectors are recorded for containment areas, Pressurizer room, Feedwater and Main Steam penetration areas, both inside and outside containment, and the Reactor Vessel Support supply air.

The same measurements are repeated with the reactor in operation at approximately 100% power.

SUMMARY OF RESULTS

After the required plant conditions were verified to have existed for a minimum of 24 hours, three sets of measurements were taken, each set at least two hours apart. The highest reading of each parameter was then compared to the acceptance criterion. All temperatures were within the acceptance criteria of the test at both 100% reactor power and when in Mode 3.

Criterion	TEST RESULTS Mode 3	100% Power
Concrete temperatures in each RCS Pipe Penetration are less than or equal to 200°F	140.9°F	166.0°F
Containment average air temperature is less than or equal to 120°F	92°F	105°F

3.6.3 - CONTAINMENT & PENETRATION ROOMS TEMPERATURE SURVEY - ISU-282A (Continued)

SUMMARY OF RESULTS (Continued)

Criterion	TEST RESULTS Mode 3	100% Power
Steam Generator compartment air temperatures are less than or equal to 120°F	98.2°F	06.4°F
Pressurizer room temperature is less than or equal to 120°F	99.0°F	108.0°F
In containment, Main Steam and Feedwater penetration area temperatures are less than or equal to 120°F	85.2°F	105.6°F
Outside containment, Main Steam and Feedwater penetration room temperatures are less than or equal to 104°F	97.2°F	100.4°F
Neutron Detector Well and reactor vessel support area exhaust air temperature is less than or equal to 150°F	145°F	143°F
CRDM Shroud Exhaust air temperature is less than or equal to 163°F	133°F	131°F
CRDM Platform area temperature is less than or equal to 140°F	104.7°F	108.3°F
Reactor Vessel Support supply air temperatures are less than or equal to 90°F. (Mode 3 only)	83.4°F	N/A

3.6.4 - TURBINE DRIVEN AUXILIARY FEEDWATER PUMP ACTUATION AND RESPONSE TIME TESTS - EGT-768A and EGT-769A

OBJECTIVE

The Turbine-Driven Auxiliary Feedwater Pump (TDAFP) test is performed to demonstrate the capability to deliver flow to the steam generators within the acceptable time after an initiating signal. The main steam header isolation valves are stroked and verified to open within the required time. These valves supply steam to the TDAFP turbine to drive the pump. This test satisfies activities described by FSAR Table 14.2-2, Sheet 51 and the deferred preoperational testing in System Test Matrix 1-3700.

TEST METHODOLOGY

The TDAFP is lined up to recirculate back to the Condensate Storage Tank with its discharge isolated from the steam generators. The pump is started by simulation of an Auxiliary Feedwater Actuation signal from the Train A Solid State Protection System and the response time is measured from the time of relay actuation to when the pump flow exceeds the minimum design flow of 860 gpm. The pump is shut down and placed in standby. The pump is then restarted by simulation of an Auxiliary Feedwater Actuation signal from the Train B Solid State Protection System and the response time is again measured from the time of relay actuation to when the pump flow exceeds 860 gpm. This pump response time is required to be less than or equal to 58.0 seconds.

The stroke open time of the Main Steam Header Isolation Valves are recorded and verified to be between 9.0 and 11.0 seconds.

SUMMARY OF RESULTS

The TDAFP was started from the Train A Auxiliary Feedwater Actuation signal (Relay K641) and the time to reach a pump flow of greater than 860 gpm was 26.2 seconds. This satisfied the acceptance criterion of 58.0 seconds or less. The stroke open time for Main Steam Header Isolation Valve 1-HV-2452-1 was 4.0 seconds which did not satisfy the review criterion of 9.0 to 11.0 seconds. The actuator on the valve was readjusted and the valve was retested resulting in a stroke open time of 9.02 seconds.

The TDAFP was restarted from the Train B Auxiliary Feedwater Actuation signal (Relay K641) and the time to reach a pump flow of greater than 860 gpm was 29.2 seconds. This satisfied the acceptance criterion of 58.0 seconds or less. The stroke time for the Main Steam Header Isolation Valve 1-HV-2452-2 was 5.9 seconds which did not satisfy the review criterion of 9.0 to 11.0 seconds. The actuator on the valve was readjusted and the valve was retested resulting in a stroke open time of 10.4 seconds.

3.6.4 - TURBINE DRIVEN AUXILIARY FEEDWATER PUMP ACTUATION AND RESPONSE TIME TESTS - EGT-768A and EGT-769A (Continued)

SUMMARY OF RESULTS (Continued)

The TDAFP was also retested using the Train B Auxiliary Feedwater Actuation signal to determine the impact on the pump response time of changing the steam header isolation valve stroke open time. The pump response time was 22.84 seconds after the adjustment was made to valve 1-HV-2452-2. Thus, the steam header isolation valve stroke time adjustments did not adversely affect overall TDAFP response time.

3.6.5 - MSIV ISOLATION RESPONSE TIME TESTS - EGT-764A and EGT-765A

OBJECTIVE

The MSIV Isolation Response Time Tests are performed to demonstrate that the Main Steam Isolation Valves (MSIVs) close within the maximum allowed time upon initiation of a close signal from the Solid State Protection System. This test satisfies activities described by FSAR Table 14.2-2, Sheets 50 and 50a and the deferred preoperational testing in System Test Matrix 1-6400.

TEST METHODOLOGY

This test is performed in Mode 3 or in Mode 4 above 300°F. Strip chart recorders are connected to the MSIV position indication circuits and to a test switch which has an input to the Train A Solid State Protection System that can actuate slave relay K627. The MSIVs are then closed by operation of the test switch and the response times from the test switch actuation to the MSIV fully closed indications are determined from the recorder traces.

The test is then repeated with actuation of the Train B Solid State Protection System K627 slave relay.

SUMMARY OF RESULTS

The Main Steam Isolation Valves were response time tested from the Train A Solid State Protection System. The recorded closure times were as follows:

Valve Number	Closure Time
1-HV-2333A	4.04 seconds
1-HV-2334A	4.48 seconds
1-HV-2335A	4.44 seconds
1-HV-2336A	3.92 seconds

The valves were then response time tested from Train B Solid State Protection System and the recorded closure times were as follows:

Valve Number	Closure Time
1-HV-2333A	3.78 seconds
1-HV-2334A	3.62 seconds
1-HV-2335A	3.82 seconds
1-HV-2336A	4.68 seconds

The valves all satisfied the maximum allowed closure time criterion of 5.0 seconds.

3.6.6 - REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVE LEAKAGE TESTING - EGT-712A

OBJECTIVE

The Reactor Coolant System Pressure Isolation Valve Leakage Testing is performed to demonstrate that the leakage past these valves is within the limits required by CPSES Technical Specification 3.4.5.2.f. The leakage tests of the Train A RHR Hot Leg Injection Valve 1-8841A, and the four RHR Cold Leg Injection Valves 1-8818A, 1-8818B, 1-8818C, and 1-8818D satisfy the deferred preoperational testing described in System Test Matrix 1-5700.

TEST METHODOLOGY

With the plant in Mode 5 and the Train A RHR and SI Hot Leg Injection flowpaths not in use, a valve lineup is established to route all leakage from Valve 1-8841A through the SI test header. Either actual RCS pressure or a temporary hydrostatic pressure pump is connected to apply pressure against valve 1-8841A from the RCS side. The leakage is measured by flow through the SI test header flowmeter and then mathematically converted to the leakage that would exist at the normal RCS pressure of 2235 psig.

With the plant in Mode 5 and the applicable loop of RHR and SI Cold Leg Injection flowpaths not in use, a valve lineup is established to route all leakage through the RHR Cold Leg Injection valve under test to the SI test header. Either actual RCS pressure or a temporary hydrostatic pressure pump is connected to apply pressure against the valve from the RCS side. The leakage is measured by flow through the SI test header flowmeter and is then also converted to leakage that would exist at the normal RCS pressure of 2235 psig.

SUMMARY OF RESULTS

Each of the RCS Pressure Isolation Valves (check valves) were verified operable by forward flow prior to the leak test.

The Train A RHR Hot Leg Injection alve 1-8841 A was tested in Mode 5 with a hydrostatic pressure pump supplying a test pressure of 230 psig. The recorded SI test header flow (leakage flow) was 0.0 gpm. The leakage, converted to the leak rate that would exist at 2235 psig, was also 0.0 gpm. This satisfied the leakage flow acceptance criterion for this valve of 3.0 gpm or less when at 2235 psig.

3.6.6 - REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVE LEAKAGE TESTING - EGT-712A (Continued)

SUMMARY OF RESULTS (Continued)

The RHR Cold Leg Injection valves, 1-8818A, 1-8818B, 1-8818C and 1-8818D, were individually tested in Mode 5. The test results were as follows:

Valve	Test Pressure	Measured Flow	Measured Flow Converted to 2235psig
1-8818A	225 psig	0.0 gpm	0.0 gpm
1-8818B	340 psig	0.1 gpm	0.26 gpm
1-8818C	290 psig	0.0 gpm	0.0 gpm
1-8818D	290 psig	0.0 gpm	0.0 gpm

These valves also satisfied the leakage rate acceptance criterion for these valves of 3.0 gpm or less when at 2235 psig. Valve 1-8818B was tested using actual RCS pressure. The other three valves were tested using a temporary hydrostatic pressure pump.

3.6.7 - CONDENSATE REJECT VALVE TEST - EGT-TP-90A-002

OBJECTIVE

The Condensate Reject Valve test is performed to demonstrate that condensate reject and makeup isolation valves 1-HV-2484 and 1-HV-2485 are capable of stroking to the fully open and closed positions under dynamic operational conditions. This test satisfies activities described by the deferred preoperational testing in System Test Matrix 1-9505.

TEST METHODOLOGY

With the plant in Modes 4, 5, or 6 and condenser vacuum established, makeup flow from the Condensate Storage Tank (CST) to the condenser hotwell is established. Each of the isolation valves is then closed and reopened. Then, condensate reject flow from the condenser hotwell to the CST is established and each of the isolation valves is again individually closed and reopened.

SUMMARY OF RESULTS

With the plant in Mode 5, condenser vacuum established, and with condensate makeup flow to the hotwell, valve 1-HV-2484 failed to fully close from the handswitch operation. The valve's limit switches and torque switches were readjusted. The condensate reject isolation valves were then retested and both properly opened and closed under makeup flow and also under reject flow conditions.

4.0 - REFERENCES

1) Comanche Peak Steam Electric Station Final Safety Analysis Report

2) Regulatory Guide 1.68, Revision 2

3) Regulatory Guide 1.68.2, Revision 1

4) Regulatory Guide 1.133, Revision 1

5) Comanche Peak Technical Specifications

6) Comanche Peak Operating License NPF-28

7) Comanche Peak Operating License NPF-87

8) WCAP-9806, Rev. 2, The Nuclear Design and Core Physics Characteristics of the Comanche Peak Unit 1 Nuclear Power Plant, Cycle 1

9) Westinghouse NSSS Startup Manual

10) Letter, N.R. Metcalf, Westinghouse Electric Corporation, to B. W. Coss, TU Electric, "Comanche Peak Unit 1 Cycle 1 Boron Worth", 90TB -G-007, April 12, 1990.