3907013190

T.S.6.9.1.1
T.S.6.9.1.2
T.S.6.9.1.3

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION
P. O. BOX A
SANATOGA, PENNSYLVANIA 19464
(215) 327-1200, EXT. 3000

GRAHAM M. LEITCH
VICE PRESIDENT
LIMERICK GENERATING STATION

April 2, 1990
Docket No. 50-353
License No. NPF-85

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: Limerick Generating Station, Unit 2
Startup Report - Cycle 1

Attached is the Limerick Generating Station Unit 2, Cycle 1, Startup Report. The report is being submitted in accordance with Technical Specifications Reporting Requirements 6.9.1.1, 6.9.1.2 and 6.9.1.3. This report contains pertinent information regarding the first cycle startup testing activities including; initial criticality, completion of the startup test program, and commencement of commercial operation.

If you have any questions, or require additional information, please do not hesitate to contact us.

Very truly yours,

M. J. McCormick for G. M. Leitch

KWM/kk

Attachment

cc: William T. Russell, Administrator, Region I, USNRC
T. J. Kenny, USNRC Senior Resident Inspector

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STARTUP REPORT FOLLOWING REFUEL OUTAGE
STATION LGS

CHAIRMAN - NRB	CB/53A-1	01 I	MGR. QA ~~PERF ASSESS~~	~~CB/53A-1~~ SB-4-3	01 I
EXEC VP NUC	CB/52C-7	01 I	PLANT MANAGER - LGS	LGS/A5-1	01 I
EXECUTIVE ASST - NRB	CB/53A-1	04 I	~~PLANT MANAGER - PB~~	~~PB/A4-1C~~	~~01 I~~
VP. LIMERICK	LGS-200	01 I	BR. HEAD-LGS LIC	CB/52A-5	01 I
VP. NUC SERVICES	CB51A-1	01 I	CORRESP RELEASE PT	LGS-340	01 I
VP. NUCLEAR ENGR	CB/~~53C-1~~ 62C-1	01 I	DIEFENDERFER, D.	OUTGOING	01 I
~~VP PBAPS~~	~~PB/SMO-1~~	~~01 I~~	LGS ENGR. REACTOR	LGS/A2-4	01 I
GEN. MGR NUC QA	CB/53A-1	01 I	LGS ENGR. REGUL SUP	LGS/SB-3-4	01 I
~~MGR ENGINEERING~~	~~CB/63B-5~~	~~01 I~~	PA DER BRP RES-LGS	LGS/~~SB-2~~ 335	01 I
~~MGR NUC TRAINING~~	~~CB/51A-9~~	~~01 I~~	~~SUPT. FUEL MGMT~~	~~MD/S14-4~~	~~01 I~~
~~MGR NUCL SUPPORT~~	~~CB/51A-1~~	~~01 I~~	SUPVR. ~~RSLR~~ Reporting	LGS/~~SB4-2~~ 338	01 I
~~MGR PB QUALITY DIV~~	~~PB/B-2-N~~	~~01 I~~	Constituent Coord.	CB-52A-5	01
			Secretary NCB	CB-52C-7	01

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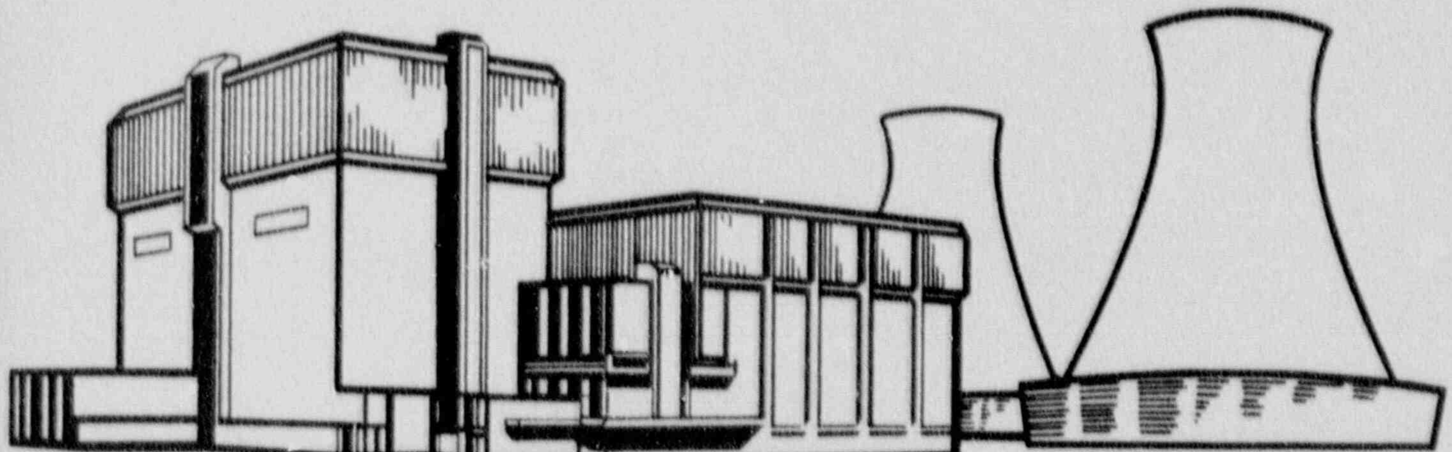


LIMERICK GENERATING STATION

3907013190

April 2, 1990

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION

UNIT NO. 2

STARTUP REPORT

Preparation Directed by:
M. J. McCormick, Manager
Limerick Generating Station

TABLE OF CONTENTS

SECTION 1

INTRODUCTION

1.1 REPORT ABSTRACT

This Startup Report, written to comply with Technical Specifications paragraphs 6.9.1.1 through 6.9.1.3, consists of a summary of the Startup Test Program portion of the Initial Test Program performed at Unit 2 of the Limerick Generating Station. This report includes events starting with initial fuel loading and ending with the completion of the Warranty Run.

The report addresses each of the Startup Tests identified in chapter 14 of the FSAR and includes a description of the measured values of the operating conditions or characteristics obtained during the test program with a comparison of these values to the Acceptance Criteria. Also included is a description of any corrective actions required to obtain satisfactory operation.

This report also provides a brief description of the plant, a description of the Startup Test Procedure format and the objectives of each test.

The report was prepared by various Startup Test Program personnel who prepared and implemented these tests and was reviewed by:

C. M. Sung - Test Results Supervisor
R. O. Hurd - Independent Reviewer
J. L. Klucar - Independent Reviewer
A. L. Jenkins - Startup Test Program Supervisor

1.2 LIMERICK PLANT DESCRIPTION

The Limerick Generating Station is a two unit nuclear power plant. The two units share a common control room, refueling floor, turbine operating deck, radwaste system, and other auxiliary systems.

The Limerick Generating Station is located on the east bank of the Schuylkill River in Limerick Township of Montgomery County, Pennsylvania, approximately 4 river miles downriver from Pottstown, 35 river miles upriver from Philadelphia, and 49 river miles above the confluence of the Schuylkill with the Delaware River. The site contains 595 acres - 423 acres in Montgomery County and 172 in Chester County.

Each of the LGS units employs a General Electric Company boiling water reactor (BWR) designed to operate at a rated core thermal power of 3293 MWt (100% steam flow) with a corresponding gross electrical output of 1092 MWe. Approximately 37 MWe are used for auxiliary power, resulting in a net electrical output of 1055 MWe. See Table 1.2-1 for Limerick Plant Parameters.

The containment for each unit is a pressure suppression type designated as Mark II. The drywell is a steel-lined concrete cone located above the steel-lined concrete cylindrical pressure suppression chamber. The drywell and suppression chamber are separated by a concrete diaphragm slab which also serves to strengthen the entire system.

The Architect Engineer and Constructor was Bechtel Power Corporation.

The plant is owned and operated by the Philadelphia Electric Company.

TABLE 1.2-1
Limerick 2 Plant Parameters

Parameter	Value
Rated Power (MWt)	3293
Rated Core Flow (Mlb/hr)	100
Reactor Dome Pressure (psia)	1020
Rated Feedwater Temperature (Deg. F)	420
Total Steam Flow (Mlb/hr)	14.159
Vessel Diameter (in)	251
Total Number of Jet Pumps	20
Core Operating Strategy	Control Cell Core
Number of Control Rods	185
Number of Fuel Bundles	764
Fuel Type	8 x 8 (Barrier)
Core Active Fuel Length (in)	150
Cladding Thickness (in)	0.032
Channel Thickness (in)	0.080
MCPR Operating Limit	1.22
Maximum LHGR (KW/ft)	13.4
Turbine Control Valve Mode	Partial Arc
Turbine Bypass Valve Capacity (% NBR)	25
Relief Valve Capacity (% NBR)	87.4
Number of Relief Valves	14
Recirculation Flow Control Mode	Variable Speed M/G Sets

1.3 INITIAL TEST PROGRAM

The Initial Test Program encompasses the scope of events that commences with system/component turnover and terminates with the completion of power ascension testing. The Initial Test Program is conducted in two separate and sequential subprograms: the Preoperational Test Program and the Startup Test Program. At the conclusion of these subprograms the plant is ready for normal commercial power operation. Testing during the Preoperational and Startup Test Programs is accomplished in four distinct and sequential phases.

<u>Major Test Phases</u> - Initial Test Program

a. Phase I - Preoperational Testing

b. Phase II - Initial Fuel Loading and Zero Power Testing

c. Phase III - Low Power Testing

d. Phase IV - Power Ascension Testing

Preoperational testing is completed during the Preoperational Test Program. Initial fuel loading and zero power testing, low power testing, and power ascension testing are completed during the Startup Test Program.

<u>Startup Test Program</u>

That part of the Initial Test Program which commences with the start of nuclear fuel loading and terminates with the completion of power ascension testing.

<u>Initial Fuel Loading and Zero Power Testing Phase</u>

That part of the Startup Test Program which includes chemical baseline data collection just prior to nuclear fuel loading, the movement of fuel assemblies from the fuel pool to the reactor core, radiological baseline data collection following completion of fuel loading, and reactor open vessel tests. Initial criticality is achieved in this test phase.

<u>Low Power Testing Phase</u>

That part of the Startup Test Program which includes the initial reactor heatup to rated reactor temperature and pressure and testing up to and including 5 percent rated reactor power.

Power Ascension Test Phase

That part of the Startup Test Program during which testing is performed at various power levels from 5 percent up to and including 100 percent rated reactor power. Testing during the Power Ascension Test Phase is accomplished in four distinct and sequential Test Plateaus.

Test Plateau A - Plant conditions cannot exceed those defined as Test Condition 1.

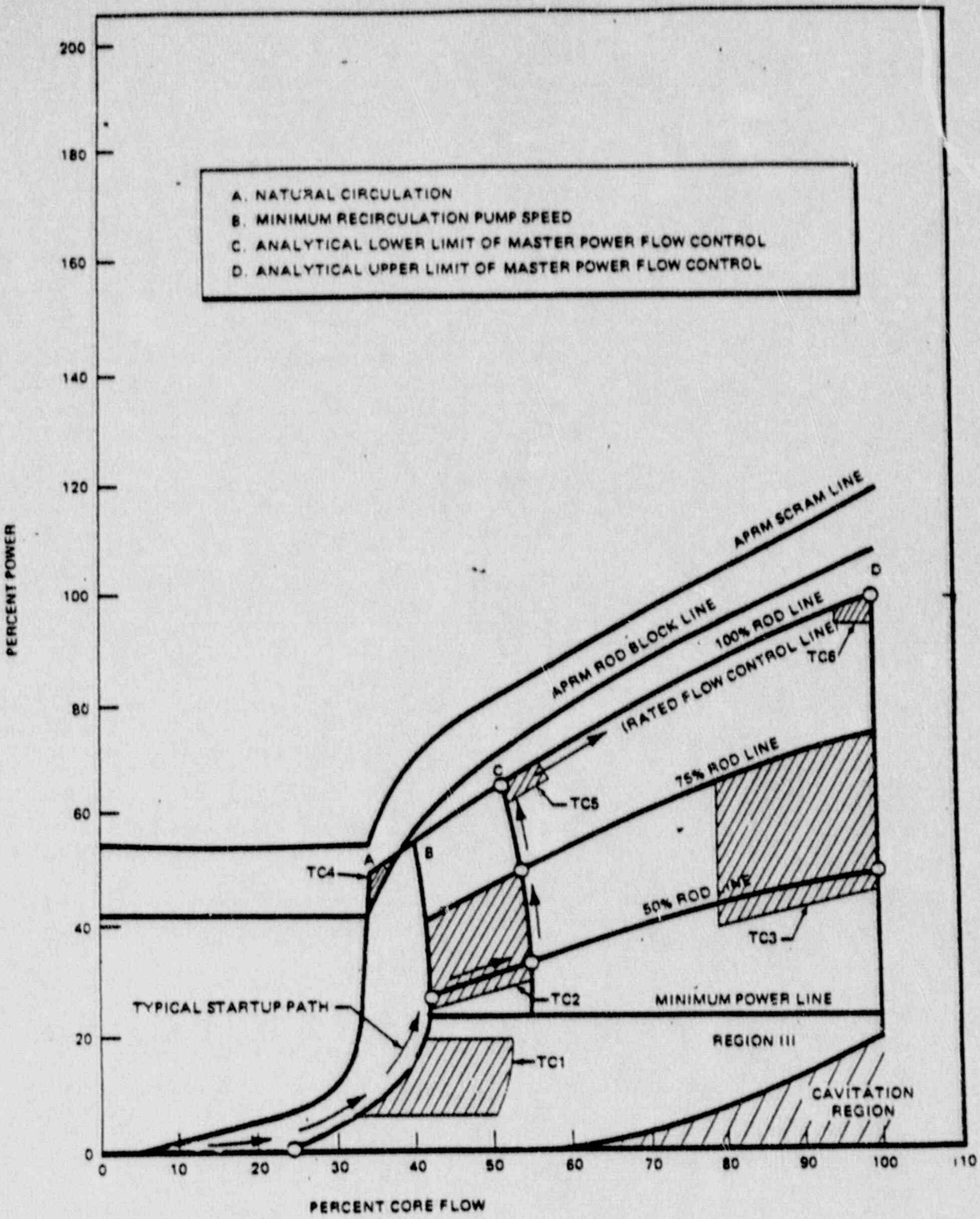
Test Plateau B - Plant conditions cannot exceed those defined as Test Condition 2.

Test Plateau C - Plant conditions cannot exceed those defined as Test Condition 3.

Test Plateau D - Testing at plant conditions up to and including 100% power (Test Conditions 4, 5, 6 and Warranty Run).

The definition of Test Condition is provided in Figure 1.3-1, Sheets 1 and 2.

The sequence of the Startup Test Program is provided in Figure 1.3-2.



Operational Power/Flow Map

Figure 1.3-1 Sheet 1

TEST CONDITION (TC) REGION DEFINITIONS

Test Condition No.	Power-Flow Map Region and Notes
1	Before or after main generator synchronization between 5% and 20% thermal power within +10, -0% of M-G Set minimum operating speed line in Local Manual mode.
2	After main generator synchronization between the 45% and 75% control rod lines between M-G Set minimum speeds for Local Manual and Master Manual modes.
3	From 45% to 75% control rod lines - core flow between 80% and 100% of its rated value.
4	* On the natural circulation core flow line - within +0, -5% of the intersection with the 100% power rod line.
5	Within +0, -5% of the 100% control rod line - within -0, +5% of the analytical lower limit of Master Flow Control.
6	Within +0, -5% of rated 100% power - within +0, -5% of rated 100% core flow rate.

* This region has been changed to +0, -20% before commencement of the Startup Test Program.

Figure 1.3-1 Sheet 2

TEST CONDITION (1)

TEST NO	PROCEDURE DESCRIPTION	OPEN VESSEL	HEAT UP	1	2	3	4	5	6	WARRANT
1	CHEMICAL AND RADIOCHEMICAL	X	X	X	X	X		X(6)	X	
2	RADIATION MEASUREMENTS	X	X		X	X			X	
3	FUEL LOADING	X								
4	FULL CORE SHUTDOWN MARGIN		X(4)							
5	CONTROL ROD DRIVE SYSTEM	X	X		X	X(11)			X	
6	SRM PERFORMANCE AND CONTROL ROD SEQUENCE		X(4)							
9	WATER LEVEL REFERENCE LEG TEMPERATURE		X	X	X	X	X	X	X	
10	IRM PERFORMANCE		X(4)	X						
11	LPRM CALIBRATION		X	X		X			X	
12	APRM CALIBRATION		X	X	X	X(10)		X	X	X
13	PROCESS COMPUTER PERFORMANCE VERIFICATION	X	X	X	X	X			X(5)	
14	RCIC SYSTEM PERFORMANCE VERIFICATION		X	X	X(5,6)					
14.1	RCIC SYSTEM STARTUP AFTER LOSS OF AC POWER TO THE SYSTEM			X(6)						
14.2	RCIC SYSTEM OPERATION WITH A SUSTAINED LOSS OF AC POWER TO THE SYSTEM			X(6)						
15	HPCI SYSTEM PERFORMANCE VERIFICATION		X			X				
16	SELECTED PROCESS TEMPERATURES VERIFICATION		X(10)			X	X(10)		X(10)	
17	SYSTEM EXPANSION	X	X		X(8)				X(6,9)	
18	TIP UNCERTAINTY					X			X	
19	CORE PERFORMANCE			X	X	X	X	X	X	X
20	STEAM PRODUCTION									X
21	DELETED									
22	PRESSURE REGULATOR RESPONSE			X	X	X	X(10)	X	X	
23	FEEDWATER CONTROL SYSTEM DEMONSTRATION			X	X	X	X	X	X(5)	
24	MAIN TURBINE VALVES SURVEILLANCE TEST					X			X(5,9)	
25	MAIN STEAM ISOLATION VALVES PERFORMANCE VERIFICATION		X	X		X(3)		X(3)	X, SD(2,5,9)	
26	MAIN STEAM RELIEF VALVES PERFORMANCE		X		X					
27	TURBINE TRIP AND GENERATOR LOAD REJECTION DEMONSTRATION				X(7)	X, SD(2,11)			X, SD(2)	
28	SHUTDOWN FROM OUTSIDE THE MAIN CONTROL ROOM DEMONSTRATION				X, SD(2)					
29	RECIRCULATION FLOW CONTROL DEMONSTRATION					X(5)			X(5)	
30	RECIRCULATION SYSTEM				X	X	X		X	
31	LOSS OF TURBINE-GENERATOR AND OFFSITE POWER				X, SD					
32	ESSENTIAL HVAC SYSTEM OPERATION AND CONTAINMENT HOT PENETRATION TEMPERATURE VERIFICATION		X			X			X	
33	PIPING STEADY STATE VIBRATION		X		X	X(5)		X	X(5,6,9)	
34	OFFGAS SYSTEM PERFORMANCE VERIFICATION		X	X		X			X(5)	
35	RECIRCULATION FLOW CALIBRATION					X(5)			X	
36	PIPING DYNAMIC TRANSIENT		X		X	X			X	
37	DELETED									
38	DELETED									
70	REACTOR WATER CLEANUP SYSTEM PERFORMANCE VERIFICATION		X							
71	RESIDUAL HEAT REMOVAL SYSTEM PERFORMANCE VERIFICATION				X(6)				X(8,9)	

LEGEND

X TEST INDEPENDENT OF FLOW CONTROLLER MODE

SD SCRAM DEFINITE

NOTES

(1) SEE FIGURE 14.2-9 TEST CONDITION REGION MAP
(2) PERFORM TEST 5, TIMING OF 4 SLOWEST CONTROL RODS IN CONJUNCTION WITH THESE SCRAMS
(3) FULL CLOSURE OF ONE VALVE ONLY
(4) MAY BE DONE DURING TEST CONDITION OPEN VESSEL
(5) SOME TESTS DONE DURING APPROACH TO TEST CONDITION
(6) MAY BE DONE DURING AN EARLIER TEST CONDITION IF CONDITIONS WARRANT
(7) DONE WITH STEAM BYPASS CAPACITY
(8) SOME TESTS DONE AFTER PLANNED TRIPS FROM POWER
(9) DETERMINE MAXIMUM POWER LEVEL TEST CAN BE PERFORMED WITHOUT CAUSING REACTOR SCRAM
(10) TEST NOT REQUIRED FOR UNIT 2
(11) TEST NOT REQUIRED FOR UNIT 2. NO TC-3 SCRAM DURING STP 27. NO STP 5 SCRAM TIMING TEST DUE TO NO SCRAM

Figure 1.3-2

Startup Test Program Sequence

1.4 MAJOR STARTUP TEST PROGRAM ADMINISTRATIVE CONTROLS

Startup testing and power escalation is sequenced in six distinct Test Plateaus.

1. Test Phase II - Initial Fuel Loading and Zero Power Testing (Test Condition Open Vessel)

2. Test Phase III - Low Power Testing (Test Condition Heatup)

3. Test Plateau A - Test Condition 1

4. Test Plateau B - Test Condition 2

5. Test Plateau C - Test Condition 3

6. Test Plateau D - 100% Rod Line Testing & Warranty Run

A Test Plateau Review is performed prior to commencing startup testing in the next higher plateau. The following items shall be completed prior to the Test Plateau Review:

a. All Startup Tests scheduled for the current Test Plateau have been implemented or deferred, the analyses have been completed, and the test results have been reviewed and approved.

b. All Startup Test Change Notices affecting tests scheduled for the current Test Plateau have been approved.

c. All Test Exception Reports affecting tests scheduled for the current Test Plateau have been resolved.

A list of all tests scheduled to be run during a specific Test Plateau is contained in Startup Test Procedure 99. This procedure was the primary means to document that all major administrative controls were satisfied.

Startup Test Change Notices (STCN) were written to document test procedure changes which were not made via a complete revision to the test procedure. STCN's were processed and approved independent of test results.

Test Exception Reports (TER) were written to document the description and resolution of all test exceptions as well as the subsequent actions required to close out the exception. The processing and approval of Test Exception Reports was independent of test results. All test exceptions which were resolved but not completely closed prior to the Plateau Review were evaluated for operational impact and carried over into subsequent test phases.

SECTION 2

SUMMARY

2.1 OVERALL EVALUATION

The Limerick Generating Station Unit 2 Startup Test Program has been successfully completed. The Startup Test Program commenced with fuel loading on June 23, 1989. Test Condition (TC) Heatup was completed on August 24, 1989. The full power license was granted on August 25, 1989 immediately followed by the commencement of TC 1 testing. Testing through TC 6 was successfully completed on December 6, 1989. Following a mini outage, Warranty Run was commenced on January 2, 1990 and was successfully completed on January 6, 1990.

All testing identified in Chapter 14 of the FSAR for Test Conditions Open Vessel, Heatup and TC 1 through 6 and Warranty Run have been performed. Individual test results are described in section 4. Open items resulting from test performance were documented by Test Exception Reports. There were five test exceptions with conditions not resolved at the close out of the Startup Test Program. These conditions were evaluated as not having a short term operational impact and were transferred as open items to other plant tracking mechanisms which will ensure that proper following and close out are accomplished.

TABLE 2-1

LIMERICK 2 MILESTONES

Jul - 1970	Started Construction, Temporary Permit
Jun - 1974	NRC Issued Construction Permit
Sep - 1977	RPV Set
Jan - 1988	Code Hydro
Feb - 1988	Started Preoperational Test Program (Energized High Voltage Switchgear)
Jul - 1989	Preoperational Test Program Completed
Jun 22, 1989	Received Fuel Load License
Jun 23, 1989	Started Fuel Load
Jul 4, 1989	Completed Fuel Load
Jul 10, 1989	Received Low Power License
Jul 11, 1989	Install RPV Head, Cold Shutdown (Operational Condition 4)
Jul 17, 1989	Completed Vessel Hydro
Aug 4, 1989	Open Vessel Testing Completed
Aug 8, 1989	Commenced Test Condition Heatup Testing
Aug 12, 1989	Initial Criticality
Aug 18, 1989	Established Initial Rated Pressure and Temperature
Aug 24, 1989	Completed Low Power Testing
Aug 25, 1989	Received Full Power License
Aug 27, 1989	Commenced Test Condition 1 Testing
Aug 30, 1989	Initial Main Turbhine Roll
Sep 1, 1989	Initial Generator Synchronization
Sep 3, 1989	Completed Test Condition 1 Testing Commenced Test Condition 2 Testing
Sep 28, 1989	Completed Test Condition 2 Testing
Oct 6, 1989	Commenced Test Condition 3 Testing

TABLE 2-1 (cont'd)

LIMERICK 2 MILESTONES

Nov 1, 1989	Completed Test Condition 3 Testing
	Commenced Test Condition 5 Testing
Nov 2, 1989	Completed Test Condition 5 Testing
	Commenced Test Condition 4 Testing
Nov 3, 1989	Completed Test Condition 4 Testing
	Commenced Test Condition 6 Testing
Dec 6, 1989	Completed Test Condition 6 Testing
Jan 2, 1990	Commenced Warranty Run Testing
Jan 6, 1990	Completed Warranty Run Testing
Jan 7, 1990	Completed Review of Startup Test Program
Jan 8, 1990	Declared Commercial Operation

TABLE 2-2

STARTUP TEST PROGRAM CHRONOLOGY

Jun 9, 1989	Commenced Test Condition Open Vessel.
Jun 15, 1989	Commenced first Startup Test, 2STP-1.1, "Pre-Fuel Load Data".
Jun 22, 1989	Received License to load fuel.
Jun 23, 1989	Commenced fuel loading at 2306 hours.
Jul 4, 1989	Last fuel bundle loaded at 1754 hours.
Jul 10, 1989	Received Low Power License.
Jul 11, 1989	RPV head installed. Entered Operational Condition 4 (Mode Switch in SHUTDOWN).
Jul 13, 1989	Commenced Operational Hydrostatic test.
Jul 15, 1989	Commenced 2STP-5.6, "Rated Reactor Pressure Scram Testing".
Jul 17, 1989	Completed Operational Hydrostatic test.
Jul 18, 1989	Completed 2STP-5.6, "Rated Reactor Pressure Scram Testing".
Aug 4, 1989	Completed Plateau Review of Test Condition Open Vessel (Phase II - Initial Fuel Loading and Zero Power Testing).
Aug 8, 1989	Commenced Test Condition Heatup.
Aug 11, 1989	Entered Operational Condition 2 (Mode Switch in STARTUP). Commenced reactor startup at 1650 hours.
Aug 12, 1989	Initial criticality achieved at 1226 hours.
Aug 13, 1989	Increased reactor pressure to 100 psig.
Aug 14, 1989	Heated reactor to 360 degrees F. Inspected drywell piping to evaluate freedom of expansion.
	Increased reactor pressure to 150 psig. Performed RCIC testing.
Aug 15, 1989	Increased reactor pressure to 200 psig. Performed HPCI testing.
	Completed ADS Valve testing.

TABLE 2-2 (cont'd)

STARTUP TEST PROGRAM CHRONOLOGY

Aug 16, 1989	Increased reactor temperature to 450 degrees F.
	Increased reactor pressure to 600 psig. Performed scram timing of selected CRD's.
Aug 17, 1989	Increased reactor pressure to 800 psig. Performed scram timing of selected CRD's.
Aug 18, 1989	Initially reached rated reactor pressure and temperature.
Aug 24, 1989	Completed Plateau Review of Test Condition Heatup (Phase III - Low Power Testing).
Aug 25, 1989	Received Full Power License.
Aug 27, 1989	Commenced Test Condition 1 Testing.
Aug 30, 1989	Initial Main Turbine Roll.
Aug 31, 1989	Completed Test Condition 1 Testing.
Sep 1, 1989	Initial Generator Synchronization.
	Completed Plateau Review of Test Plateau A (Test Condition 1).
Sep 3, 1989	Commenced Test Condition 2 Testing.
Sep 11, 1989	Main Generator runback followed by a Turbine trip within bypass valve capacity occurred due to a false sensing of low stator water cooling flow to a generator high voltage bushing.
Sep 12, 1989	2B Reactor Feedwater Pump Turbine trip. Reactor water level control was maintained by standby pump and level control system.
Sep 15, 1989	SRV capacity test at rated pressure, 2STP-26.2.
Sep 18, 1989	Turbine trip within bypass valve capacity at 21.2% power for 2STP-27.1.
	SCRAM #1. Completed 2STP-31.1, Loss of Offsite Power test, at 21.1% power. Reactor scrammed on low reactor water level.
Sep 20, 1989	Restarted reactor.

TABLE 2-2 (cont'd)

STARTUP TEST PROGRAM CHRONOLOGY

Sep 23, 1989	Turbine runback occurred due to a false sensing of low stator liquid cooling flow in the generator high voltage bushing. The transient initiated a recirculation pump runback. A second runback occurred during the restoration.
Sep 25, 1989	SCRAM #2. Manual scram from Remote Shutdown panel for 2STP-28.1.
	Commenced planned mini-outage.
Sep 26, 1989	Plant cooldown from Remote Shutdown panel for 2STP-28.2.
	Condenser leak identified by high hotwell conductivity, and subsequently the leaking tubes were plugged. A baffle was reoriented to prevent reccurrence.
Sep 28, 1989	Completed Plateau Review of Test Plateau B (Test Condition 2).
Oct 6, 1989	LGS Self-Assessment Review with NRC, and NRC concurred with Oversight Committee to proceed with testing above 50% power.
	Commenced Test Condition 3 Testing.
Oct 8, 1989	Restarted reactor.
	Completed planned mini-outage.
Oct 21, 1989	Recirc flow raised to 100%; 71% power.
Oct 24, 1989	Double Recirc Pump trip for 2STP-30.2
Oct 25, 1989	Completed 2STP-15.5, HPCI Cold Quick Start to Reactor Vessel.
Nov 1, 1989	Completed Plateau Review of Test Plateau C (Test Condition 3).
	Commenced Test Condition 5 Testing.
Nov 2, 1989	Completed Test Condition 5 Testing.
	Double Recirc pump trip to natural circulation. Commenced Test Condition 4 Testing.
Nov 3, 1989	Completed Test Condition 4 Testing.
	Commenced Test Condition 6 Testing.

TABLE 2-2 (cont'd)

STARTUP TEST PROGRAM CHRONOLOGY

Nov 10, 1989	SCRAM #3. Generator load rejection from 99.6% power occurred due to incorrect setpoint on a generator phase differential relay.
Nov 11, 1989	HPCI declared inoperable due to speed control problems observed during generator load rejection.
Nov 14, 1989	Restarted reactor.
	Increased reactor pressure to 200 psig. Performed HPCI testing. HPCI declared operable.
Nov 21, 1989	Reached 100% power (3293 MWt/1125 MWe).
Nov 22, 1989	Test results approved to take credit for the 2STP-27.4, Turbine Trip, from the 11/10/89 generator load rejection.
Nov 23, 1989	2B Reactor Feedwater Pump Turbine trip on high vibration caused by faulty vibration instrumentation.
Dec 1, 1989	SCRAM #4. Full MSIV Closure from 97.2% power for 2STP-25.3.
	Commenced planned mini-outage.
Dec 6, 1989	Completed Test Condition 6 Testing.
Dec 22, 1989	Completed planned mini-outage. Restarted reactor.
Jan 2, 1990	Commenced Warranty Run Testing.
Jan 6, 1990	Completed Warranty Run Testing.
Jan 7, 1990	Completed Plateau Review of Test Plateau D (Test Conditions 4, 5, 6 and Warranty Run).
Jan 8, 1990	Declared commercial operation.

TABLE 2-3 STARTUP TEST PERFORMANCE DATES (1 of 2)

STP No.		OPEN VESSEL	HEATUP	TC1	TC2	TC3	TC4	TC5	TC6	WARRANTY RUN
1	Chemical and Radiochemical	06/15/89 06/22/89	08/17/89 08/21/89	08/30/89 08/31/89	09/05/89 09/26/89	10/12/89 10/25/89	-	-	11/05/89 11/29/89	-
2	Radiation Measurements	07/07/89 07/13/89	08/17/89 08/20/89	-	09/05/89 09/09/89	10/18/89 10/24/89	-	-	11/06/89 11/27/89	-
3	Fuel Loading	06/23/89 07/08/89	-	-	-	-	-	-	-	-
4	Shutdown Margin Demonstrations	08/11/89 08/12/89	-	-	-	-	-	-	-	-
5	Control Rod Drive System	06/24/89 07/20/89	08/16/89 08/22/89	-	09/18/89 09/25/89	-	-	-	11/10/89 12/02/89	-
6	SRM Performance and Control Rod Sequence	-	08/11/89 08/12/89	-	-	-	-	-	-	-
9	Water Level Reference Leg Temperature	-	08/20/89 08/20/89	08/29/89 08/29/89	09/05/89 09/05/89	10/19/89 10/19/89	11/02/89 11/02/89	11/01/89 11/01/89	11/06/89 11/06/89	-
10	IRM Performance	-	08/11/89 08/12/89	08/27/89 08/27/89	09/21/89 09/21/89	10/08/89 10/08/89	-	-	-	-
11	LPRM Calibration	-	08/21/89 08/21/89	08/29/89 08/29/89	-	10/17/89 10/17/89	-	-	11/08/89 11/09/89	-
12	APRM Calibration	-	08/12/89 08/13/89	09/01/89 09/02/89	09/16/89 09/16/89	-	-	11/01/89 11/02/89	11/20/89 11/20/89	01/02/90 01/03/90
13	Process Computer	06/19/89 06/20/89	08/22/89 08/22/89	08/31/89 09/02/89	09/06/89 09/15/89	10/23/89 10/23/89	-	-	11/17/89 11/18/89	-
14	RCIC System	-	08/14/88 08/20/89	08/28/89 08/28/89	09/04/89 09/21/89	-	-	-	-	-
15	HPCI System	-	08/15/89 08/20/89	-	-	10/22/89 10/26/89	-	-	-	-
16	Selected Process Temperatures	-	-	-	-	10/21/89 10/21/89	-	-	-	-
17	System Expansion	07/09/89 07/25/89	08/13/89 08/18/89	-	09/09/89 09/17/89	10/02/89 10/02/89	-	-	11/06/89 12/06/89	-
18	TIP Uncertainty	-	-	-	-	10/17/89 10/18/89	-	-	11/08/89 11/10/89	-
19	Core Performance	-	-	09/02/89 09/02/89	09/15/89 09/15/89	10/17/89 10/17/89	11/02/89 11/02/89	11/01/89 11/01/89	11/08/89 11/09/89	01/02/90 01/03/90
20	Steam Production	-	-	-	-	-	-	-	-	01/02/90 01/06/90

TABLE 2-3 STARTUP TEST PERFORMANCE DATES (2 of 2)

STP No.		OPEN VESSEL	HEATUP	TC1	TC2	TC3	TC4	TC5	TC6	WARRANTY RUN
22	Pressure Regulator	-	-	08/29/89 08/29/89	09/13/89 09/13/89	10/20/89 10/20/89	-	11/02/89 11/02/89	11/08/89 11/21/89	-
23 A	Feedwater System Water Lvl Stpt Change	-	-	08/28/89 08/28/89	09/14/89 09/15/89	10/09/89 10/23/89	11/02/89 11/02/89	11/02/89 11/02/89	11/07/89 11/07/89	-
B	Loss of FW Heating	-	-	-	-	-	-	-	11/20/89 11/20/89	-
C	FW Pump Trip	-	-	-	-	-	-	-	11/27/89 11/27/89	-
D	Maximum Runout Capability	-	-	-	-	-	-	-	11/19/89 11/19/89	-
24	Turbine Valve Surveillance	-	-	-	-	10/13/89 10/17/89	-	-	11/04/89 11/[illegible]/89	-
25	Main Steam Isolation Valves	-	08/18/89 08/18/89	08/29/89 08/29/89	-	10/20/89 10/20/89	-	-	12/01/89 12/01/89	-
26	Relief Valves	-	08/15/89 08/15/89	-	09/15/89 09/15/89	-	-	-	-	-
27	Main Turbine Trip	-	-	-	09/18/89 09/25/89	-	-	-	11/10/89 11/10/89	-
28	Shutdown from Outside Control Room	-	-	-	09/25/89 09/26/89	-	-	-	-	-
29	Recirculation Flow Control System	-	-	-	-	10/23/89 10/24/89	-	-	11/05/89 11/06/89	-
30	Recirculation System	-	-	-	09/08/89 09/08/89	10/21/89 10/24/89	11/02/89 11/02/89	-	11/07/89 11/19/89	-
31	Loss of TG and Offsite Power	-	-	-	09/18/89 09/18/89	-	-	-	-	-
32	HVAC System	-	08/18/89 08/22/89	-	09/25/89 09/25/89	10/12/89 10/16/89	-	-	11/07/89 11/22/89	01/02/90 01/03/90
33	Piping Steady State Vibration	-	08/16/89 08/21/89	-	09/05/89 09/27/89	10/12/89 10/22/89	-	11/02/89 11/02/89	11/09/89 11/09/89	-
34	Offgas Performance Verification	-	08/16/89 08/21/89	08/30/89 08/31/89	-	10/12/89 10/24/89	-	-	11/21/89 11/28/89	-
35	Recirculation Flow Calibration	-	-	-	-	10/30/89 10/30/89	-	-	11/06/89 11/18/89	-
36	Piping Dynamic Transients	-	08/19/89 08/22/89	-	09/15/89 09/27/89	10/23/89 10/24/89	-	-	11/10/89 11/28/89	-
70	Reactor Water Cleanup System	-	08/21/89 08/23/89	-	-	10/09/89 10/09/89	-	-	-	-
71	Residual Heat Removal System	-	08/19/89 08/19/89	-	-	-	-	-	-	-

TABLE 2-4

SCRAM SUMMARY

No.	Date	T.C.	Cause
1	9/18/89	2	Planned #1 - Scram on low level during Loss of Turbine Generator and Offsite Power, 2STP-31.1
2	9/25/89	2	Planned #2 - Manual scram as part of the Remote Shutdown Test, 2STP-28.1.
3	11/10/89	6	Unplanned #1 - Generator load rejection due to incorrect setpoint on a generator phase differential relay. This unplanned trip was used to satisfy the requirements of the planned turbine trip, 2STP-27.4.
4	12/07/89	6	Planned #3 - Full MSIV isolation from 97.2% power for 2STP-25.3.

SECTION 3

STARTUP TEST PROCEDURES

3.1 STARTUP TEST PROCEDURE FORMAT AND CONTENT

Startup Test Procedures are generally written to demonstrate and verify the performance of a system or control system, to monitor the unit's response to a major transient, or to perform a specific activity. Because of the nature of Startup testing, and to facilitate procedure control, each Startup Test Procedure consists of a Main Body and one or more Subtests.

The Main Body of a Startup Test Procedure provides an overall test description, lists the test objectives, references and acceptance criteria and contains information necessary to successfully prepare for the implementation of Subtests. The Main Body consists of the following sections:

1. Objectives
2. Description
3. Acceptance Criteria
4. References
5. Procedure
6. Appendices (optional)

The Subtests contain the step-by-step instructions necessary for final preparations for the test, the actual performance of the test, and the analysis of data collected during the test. A Subtest consists of the following sections:

1. Discussion
2. Precautions
3. Test Equipment
4. Prerequisites
5. Initial Conditions
6. Test Instructions
7. Analysis
8. Appendices (optional)

A Startup Test Procedure contains as many Subtests as required to satisfy all the Acceptance Criteria listed in the Main Body and to effectively conduct testing at various plant conditions. If the same Subtest was performed more than once, provisions were made to identify plant conditions at which the Subtest was implemented.

3.2 ACCEPTANCE CRITERIA

Acceptance criteria may be either quantitative or qualitative. Quantitative acceptance criteria specify that test or equipment expected values are in accordance with test requirements (FSAR, equipment specification, test specifications, etc.). These criteria state expected values such as flows, temperatures, pressures, currents, voltages, etc., required under specific conditions. Such values are specified as maximums or minimums, or tolerances are provided. Qualitative acceptance criteria specify test or equipment functions (an event does or does not occur), such as automatic start, sequencing, or shutdown occurring under specified conditions.

Acceptance criteria are categorized as Level 1 or Level 2 which are defined below:

a. A Level 1 criterion normally relates to the value of a process variable assigned in the design of the plant, component, systems or associated equipment. If a Level 1 criterion were not satisfied, the plant would be placed in a suitable hold condition, until resolution was obtained. Tests compatible with the hold condition would be continued. Following resolution, applicable retesting would be reperformed to verify that the requirements of the Level 1 criterion were satisfied.

b. A Level 2 criterion is associated with expectations relating to the performance of systems. If a Level 2 criterion were not satisfied, operating and testing plans would not necessarily be altered. Investigations of the measurements and of the analytical techniques used for the predictions would be performed.

SECTION 4

RESULTS

`.1 2STP-1, CHEMICAL AND RADIOCHEMICAL

OBJECTIVE

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

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

ACCEPTANCE CRITERIA

Level 1

Chemical factors defined in the Technical Specifications must be maintained within the limits (chemical values and time intervals) specified.

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

Level 2

Chemical factors in the Fuel Warranty must be maintained within the specified limits.

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

RESULTS

2STP-1.1, Pre-Fuel Load Data

Chemical and radiochemical characteristics of reactor water, stored makeup water, storage and suppression pool water, standby liquid control, and closed cooling system water were measured. Results showed that all water chemistry values were within applicable limits, except for a pH value of 10.3 for the Reactor Enclosure Closed Cooling Water (the pH limit was 9.7). The pH value of 10.3 was within specifications per plant chemistry procedure (pH limit of 10.5). A follow-up Engineering evaluation concluded that a pH of up to 10.5 is acceptable for RECW operation.

The test results are summarized as follows:

PARAMETER	MEASURED VALUE	ACCEPTANCE CRITERIA
Reactor Water		
Conductivity, umho/cm at 25°C	0.24	≤3.0
Chloride, ppb	<20	≤500
pH at 25°C	6.59	5.3-8.6
Fuel and Equipment Storage Pools		
Conductivity, umho/cm at 25°C	0.87	≤3.0
Chloride, ppb	<20	≤500
pH at 25°C	5.91	5.3-7.5
Heavy Elements (Fe, Cu, Ni) ppb	29.3	≤100
Total Insolubles ppm	<0.1	≤1.0
Condensate Storage Tank		
Conductivity, umho/cm at 25°C	0.74	≤1.0

PARAMETER	MEASURED VALUE	ACCEPTANCE CRITERIA
Condensate Storage Tank (Continued)		
Chloride, ppb	<20	≤50
pH, at 25°C	6.04	6.0-8.0
Boron, ppb	13	≤100
Suppression Pool		
Chloride, ppb	<20	≤500
Standby Liquid Control System		
Sodium Pentaborate,lbs	5712.4	≥5500
Reactor Enclosure Cooling Water		
Corrosion Inhibitor, ppm	874	500-1000
pH at 25°C	10.3	9.0-9.7
Chloride, ppm	<0.02	≤10
Turbine Enclosure Cooling Water		
Corrosion Inhibitor, ppm	828	500-1000
pH at 25°C	9.7	9.0-9.7
Chloride, ppm	<0.02	≤10

2STP-1.2, Chemistry Data

Chemical and radiochemical characteristics of reactor water, control rod drive water, condensate demineralizer influent and effluent, feedwater, stored makeup water, floor drain water, and gaseous effluents were measured at various times during power ascension. In Test Condition 2, the Condensate Demineralizer Effluent dissolved oxygen was determined to be less than 10 ppb, with the lower limit for dissolved oxygen equal to 20 ppb per GE Water Quality Specifications. This Level 2 Acceptance Criterion failure was temporarily accepted in view of a scheduled oxygen injection modification which was eventually implemented on November 15, 1989, bringing the dissolved oxygen concentration within acceptance limits. In Test Condition 3, the pH of the Condensate Storage Tank was reported as 5.92 and 5.9 at two different times. These values were below the Level 2 Acceptance Criterion limit of 6.0. Since the CST is vented to the atmosphere, absorption of carbon dioxide may bring the pH value below 6. The GE Water Quality Specification specifies that the limit applies "after correction for dissolved CO_2". This correction, not normally applied in laboratory measurements in nuclear plants, would have raised the pH value above 6.0. A subsequent revision changed the lower acceptance limit to 5.7. Also during Test Condition 3, the total concentration of iron in feedwater exceeded the Level 2 Acceptance Criterion limit of 10 ppb. The reported total iron concentration was 37.72 ppb, most of which was insoluble iron. The plant apparently had not been at steady state condition long enough when the feedwater was sampled. The subsequent daily surveillance tests, however, showed that the concentration of iron in the feedwater was well below the 10 ppb limit, demonstrating that the deviating condition had been corrected. No Level 1 Acceptance Criteria failure was encountered during the performance of this subtest. Taking into account the test exceptions described in this paragraph and the corresponding resolutions, the results of all water chemistry tests were satisfactory. In addition, with the performance of this subtest, baseline data for North and South stack effluents and radiological dose rates were established.

The test results are summarized as follows:

	MEASURED VALUE						
Power Level	0%	25%	15-25%	45-55%	45-80%	90-100%	
PARAMETER	TC OV	TC HU	TC 2	TC 3	TC 3	TC 6	ACCEPTANCE CRITERIA
Reactor Water							
Conductivity,	0.66	0.13					≤ 2.0
umho/cm at 25 deg C			0.28	0.17	0.35	0.29	≤ 1.0
Chloride, ppb	< 20	< 2					≤ 100
			< 2	3.5	2.6	2.9	≤ 200
PH at 25 deg C	5.9	6.7	6.8	7.3	7.8	7.7	5.6-8.6
I-131 Dose Equivalent uCi/gm	N/A	1.21E-6	2.27E-5	2.70E-5	3.68E-5	4.12E-2	≤ 0.2
CRD Water							
Conductivity, umho/cm at 25 deg C	N/A	0.06	0.09	0.07	0.08	0.10	≤ 0.1
Dissolved O2, ppb	N/A	20	< 10	30	30	20	≤ 50
Cond Demin Effluent							
Conductivity, umho/cm at 25 deg C	N/A	0.06	0.09	0.07	0.06	0.06	≤ 0.1
Chloride, ppb	N/A	< 2	< 2	< 2	< 2	< 2	< 2
PH at 25 deg C	N/A	7.2	7.2	6.8	7.0	7.0	6.5-7.5
Total Metals ppb	N/A	1.09	3.87	5.76	2.24	0.95	≤ 9
Total Fe, ppb	N/A	0.22	1.97	4.95	0.83	0.34	≤ 5
Total Cu, ppb	N/A	0.55	0.28	0.13	0.18	0.17	≤ 2
Total Ni, ppb	N/A	0.04	0.17	0.09	0.06	0.04	≤ 2
Dissolved O2, PPb	N/A	20	10	60	40	30	20-200
Silica, ppb	N/A	< 10	< 10	< 10	< 10	< 10	< 10
Feedwater							
Conductivity, umho/cm at 25 deg C	N/A	0.093	0.095	0.066	0.064	0.066	≤ 0.1
PH at 25 deg C	N/A	6.7	7.0	6.7	7.0	7.1	6.5-7.5
Dissolved O2, ppb	N/A	10	10	40	30	40	10-200
Fe, Soluble, ppb	N/A	0.82	0.72	0.72	0.18	0.27	
Insoluble, ppb	N/A	0.31	0.39	37	0.65	0.75	≤ 10 Total
Cu, Soluble, ppb	N/A	0.47	0.27	0.49	0.19	0.21	
Insoluble, ppb	N/A	0.01	< 0.002	0.40	0.01	0.01	≤ 2 Total
Ni, Soluble, ppb	N/A	0.19	0.10	0.08	0.40	1.39	
Insoluble, ppb	N/A	0.01	0.01	0.03	< 0.005	0.01	
Cr, Soluble, ppb	N/A	0.16	0.05	< 0.004	0.01	0.02	
Insoluble, ppb	N/A	0.01	0.01	0.08	< 0.005	0.01	
Zn, Soluble, ppb	N/A	0.92	0.82	0.28	1.24	0.45	
Insoluble, ppb	N/A	0.02	0.01	0.13	< 0.01	<0.01	
Total Soluble, ppb	N/A	2.55	1.97	1.57	1.96	2.35	
Total Insoluble, ppb	N/A	0.36	0.42	37.5	0.68	0.78	
Total Metals	N/A	2.91	2.39	39.1	2.64	3.13	≤ 15

PARAMETER	TC OV	TC HU	TC 2	TC 3	TC 3	TC 6	ACCEPTANCE CRITERIA
CST							
Conductivity, umho/cm at 25 deg C	0.88	0.29	0.84	0.62	0.76	0.70	≤ 1.0
Chloride, ppb	< 20	< 2	8	< 2	4.5	2.3	≤ 50
PH at 25 deg C	6.08	6.25	6.03	5.92	5.9	6.0	6.0-8.0
Boron, ppb	< 10	< 20	< 10	< 10	< 10	< 10	≤ 100
Suppression Pool							
Chloride, ppb	30	< 2	< 2	< 20	< 20	42	≤ 500
Gaseous Effluents							
Offgas Activity, uCi/sec	N/A	N/A	78	343	221	299	≤ 330,000
Noble Gas Body Dose Rate, mRem/yr	< 0.1	0	0.08	2.19	0	2.23	≤ 500
Noble Gas Skin Dose Rate, mRem/yr	< 0.1	0	0.12	2.81	0	2.86	≤ 3,000
Floor Drain Sample Tank No. 2							
Liquid Effluent Activity	yes	yes	yes	yes	yes	yes	Below Tech Spec Limit

* N/A: Not Applicable

2STP-1.3, Gaseous Effluents Sampling and Analysis

In Test Condition 1, 3 and 6, offgas radiation monitor readings were compared with readings from grab samples taken at the same location to develop a correlation between the two. Additionally, the radiolytic gas production rate was determined. The latter results were found to be consistent at 52% and 95% reactor power.

Even though no acceptance criteria are associated with this subtest, some concern was expressed over the accuracy of the radiation monitor readings and corresponding activity release rates, and the resolution of which calls for a calibration check and possible troubleshooting and repair of the monitors.

2STP-1.4, Reactor Water No Cleanup Test

This subtest was performed in Test Condition 6 above 95% reactor power. The purpose of the test was to determine steam moisture carryover to the turbine. Carryover was determined by measuring the ratio of Na-24 concentration in the condensate to Na-24 concentration in reactor water. The measured carryover value, 0.0025%, is in excellent agreement with the corresponding value of 0.0024% measure for Unit 1.

No acceptance criteria were associated with this test.

2STP-1.6, Sample Station Operability

The purpose of this test was to demonstrate proper operation of those sample points that could not be included in the preoperational test. Part of 2STP-1.6 was performed in Test Condition 2, and part in Test Condition 6. Proper operation of all sample points was verified for the portion of the test performed in Test Condition 2. In Test Condition 6, however, two flow rates and two temperatures were outside the limits set by the procedure.

Although no acceptance criteria are associated with this subtest, test exception reports were written to document the observed deviations from the designed values.

4.2 2STP-2, RADIATION MEASUREMENTS

OBJECTIVES

The objectives of this test are to a) determine the background radiation levels in the plant environs prior to operation for base data to assess future activity buildup, b) monitor radiation at selected power levels to assure the protection of personnel during plant operation, and c) verify that general area dose rates and shield walls satisfy radiation zoning criteria.

ACCEPTANCE CRITERIA

Level 1

The radiation doses of plant origin and the occupancy times of personnel in radiation zones shall be controlled consistent with the guidelines of the standard for protection against radiation as outlined in 10CFR20, "Standard For Protection Against Radiation".

Level 2

None

RESULTS

2STP-2.1, Radiation Surveys

General area and shield wall surveys were performed throughout the plant subsequent to fuel load but prior to initial criticality, in Test Condition Heatup at rated temperature and pressure (critical at < 5% CTP, Core Thermal Power), in Test Condition 2 (approximately 45% CTP), in Test Condition 3 (approximately 60% CTP) and in Test Condition 6 (between 95 and 100% CTP).

All radiation dose rates in Test Condition Heatup and Test Condition 2 were within design zoning values, with a maximum of 40 mR/hr measured in the RWCU Backwash Receiving Tank Room in Test Condition 2. In Test Condition 3, there were two areas where the measured radiation levels failed to meet acceptance criteria. The TIP Room roof measured 300 mR/hr during TIP withdrawal from the core (zoned as < 100 mR/hr) and the east water tight door entrance to the condenser area (Turbine Enclosure 217') measured 4.0 mR/hr (zoned to be < 2.5 mR/hr). The TIP room measurement was not part of STP 2.1, "Radiation Survey Test," during TC-3. However, rezoning was still performed as a result of the measurements by Health Physics.

In Test Condition 6, there were several areas where the measured radiation levels failed to meet acceptance criterion. The CRD maintenance room general area measured 45 mR/hr (zoned as ≤ 15 mR/hr) and the RWCU sample sink area on Reactor Enclosure 253' measured 3.0 mR/hr (zoned as ≤ 2.5 mR/hr). No action was required for the CRD maintenance room as high dose rates were due to some Unit 1 CRDs temporarily stored in the room.

The shield wall of the 6B Feedwater Heater Room on Turbine Enclosure 269' and the west wall of the 2B Turbine Driven Reactor Feed Pump (TDRFP) Lube Oil Filter Area on Turbine Enclosure 200' (Rm 275) both measured 3.2 mR/hr at penetrations (zoned as ≤ 2.5 mR/hr). Engineering evaluations were performed. The penetration in the Lube Oil Filter area was determined to be acceptable with no modification or zoning change. Until the collar is added to the pipe, the penetration will be rezoned to ≤ 15 mR/hr.

All other areas and shield walls surveyed satisfied the zoning requirements at rated reactor power. Maximum dose rates of 1000 mR/hr were measured in Test Condition 6 in the Upper Valve Compartment (Reactor Enclosure 295') and the RWCU Isolation Valve Room (Reactor Enclosure 283'). Some rooms were not surveyed above 60% reactor power due to expected high dose rates in these areas.

The test results of room and area surveys are summarized as follows:

LOCATION: Turbine Enclosure

Room		Measured Dose Rate Gamma mR/hr					Acceptance
Number	Elev	PFL	TC HU	TC 2	TC 3	TC 6	Criteria
266	203	<0.2	<0.2	<0.2	1.0	4	≤100/≤15**
270	200	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
273	200	<0.2	<0.2	<0.2	0.2	1.0	≤100K/≤1K**
274	200	<0.2	<0.2	<0.2	<0.2	0.4	≤2.5
275	200	<0.2	<0.2	<0.2	0.5	0.4	≤2.5
276	200	<0.2	<0.2	<0.2	<0.2	1.5	≤2.5
277	200	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
335	217	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
346	217	<0.2	<0.2	<0.2	<0.2	0.4	≤2.5
346A	217	<0.2	<0.2	<0.2	<0.2	0.2	≤2.5
347	217	<0.2	<0.2	<0.2	0.4	0.4	<100/≤15**
350	217	<0.2	<0.2	<0.2	0.4	1.0	<100/≤15**
351	217	<0.2	<0.2	<0.2	<0.2	<0.2	<100/≤15**
353A	217	<0.2	<0.2	0.4	0.2	40	<100K/≤15**
353B	217	<0.2	<0.2	0.6	0.2	40	<100K/≤15**
353C	217	<0.2	<0.2	0.4	0.2	50	<100K/≤15**
354	217	<0.2	<0.2	<0.2	<0.2	0.2	≤2.5
354A	217	N/A	N/A	N/A	4.0	8.0	≤2.5
356	217	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
357	217	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
358	217	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
359	217	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
459	239	<0.2	<0.2	<0.2	<0.2	0.4	≤2.5

Room Number	Elev	Measured Dose Rate Gamma mR/hr PFL	TC HU	TC 2	TC 3	TC 6	Acceptance Criteria
460	239	<0.2	<0.2	<0.2	<0.2	0.2	≤2.5
461	239	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
463	239	<0.2	<0.2	<0.2	<0.2	0.4	≤2.5
465	239	<0.2	<0.2	<0.2	<0.2	<0.2	≤15
466	239	<0.2	<0.2	<0.2	<0.2	0.5	≤2.5
466A	239	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
543	269	<0.2	<0.2	<0.2	0.2	1.2	≤2.5
559	285	<0.2	<0.2	<0.2	<0.2	0.8	≤2.5
560	285	<0.2	<0.2	<0.2	<0.2	1.0	≤2.5
561	285	<0.2	<0.2	<0.2	<0.2	0.8	≤2.5
562	269	<0.2	<0.2	<0.2	<0.2	0.4	≤2.5
563	269	<0.2	<0.2	<0.2	<0.2	0.4	≤2.5
564	269	<0.2	<0.2	<0.2	<0.2	1.8	≤2.5
565	269	<0.2	<0.2	<0.2	0.2	2.0	≤2.5
565A	269	<0.2	<0.2	<0.2	0.2	1.5	≤2.5
566	269	<0.2	<0.2	<0.2	<0.2	0.5	≤2.5
528	302	<0.2	<0.2	<0.2	<0.2	0.6	≤2.5
629	302	<0.2	<0.2	<0.2	<0.2	<0.2	≤15

LOCATION: Reactor Enclosure

Room Number	Elev	Measured Dose Rate Gamma mR/hr PFL	TC HU	TC 2	TC 3	TC 6	Acceptance Criteria
173	177	<0.2	<0.2	<0.2	<0.2	0.2	<100K/<1K**
174	177	<0.2	<0.2	<0.2	0.4	1.0	<10K
177	177	<0.2	<0.2	<0.2	<0.2	<0.2	<2.5
178	177	<0.2	<0.2	<0.2	<0.2	<0.2	<2.5
179	177	<0.2	<0.2	<0.2	<0.2	<0.2	<100K/<100**
180	177	<0.2	<0.2	<0.2	<0.2	<0.2	<100K/<100**
181	177	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
182	177	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
183	177	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
184	190	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
184A	177	<0.2	<0.2	<0.2	<0.2	<0.2	≤15
185	177	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
186	177	<0.2	<0.2	<0.2	0.2	0.2	≤2.5
187	177	<0.2	<0.2	<0.2	<0.2	0.2	≤100K
188	177	<0.2	<0.2	<0.2	<0.2	0.2	≤2.5
189	177	<0.2	<0.2	<0.2	<0.2	0.2	≤2.5
279	201	<0.2	<0.2	<0.2	<0.2	0.2	≤2.5
280	201	<0.2	<0.2	<0.2	0.2	1.0	<10K/<1K**
281	201	<0.2	<0.2	<0.2	0.2	<0.2	<10K/1K**
282	201	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
283	201	<0.2	<0.2	<0.2	N/A	N/A	<10K/<100**
284	201	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
285	201	<0.2	<0.2	<0.2	*N/A	*N/A	<10K/<100**

Room Number	Elev	Measured Dose Rate Gamma mR/hr PFL	TC HU	TC 2	TC 3	TC 6	Acceptance Criteria
286	201	<0.2	<0.2	<0.2	*N/A	*N/A	<10K
287	201	<0.2	<0.2	<0.2	*N/A	*N/A	<100K
368	217	<0.2	<0.2	<0.2	*N/A	*N/A	<100K
369	217	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
370A	217	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
370B	217	<0.2	<0.2	<0.2	0.4	<0.2	≤2.5
370C	217	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
376	217	<0.2	<0.2	<0.2	*N/A	*N/A	<10K/<100**
475	253	<0.2	<0.2	<0.2	<0.2	0.2	≤2.5
475A	253	<0.2	<0.2	<0.2	<0.2	3.0	≤2.5
476	253	<0.2	<0.2	8.0	10	45	≤15
477	253	<0.2	<0.2	<0.2	<0.2	45	≤15
574	283	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
575	283	<0.2	<0.2	<0.2	<0.2	0.5	<1K/<15**
575A	283	<0.2	<0.2	<0.2	<0.2	<0.2	<100/<15**
580	283	<0.2	<0.2	<0.2	<0.2	0.2	≤2.5
581	283	<0.2	<0.2	14	8	20	<10K/<100**
582	283	<0.2	<0.2	10	<0.2	2.0	<10K/<100**
583	283	<0.2	<0.2	<0.2	8	240	<10K/<100**
584	283	<0.2	<0.2	28	40	280	<10K/<100**
585	283	<0.2	<0.2	<0.2	<0.2	<0.2	<1K
589	283	<0.2	<0.2	<0.2	<0.2	<0.2	<100/≤15**
593	295	<0.2	<0.2	<0.2	0.8	0.5	<100/≤2.5**
596	283	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5

Room Number	Elev	Measured Dose Rate Gamma mR/hr PFL	TC HU	TC 2	TC 3	TC 6	Acceptance Criteria
633	313	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
635	313	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
637	313	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
638	313	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
641	313	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
642	313	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
643	313	<0.2	<0.2	<0.2	<0.2	<2.0	≤2.5
647	331	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
648	331	<0.2	<0.2	<0.2	<0.2	<0.2	<100
649	331	<0.2	<0.2	<0.2	<0.2	<0.2	<100
650	331	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
651	331	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
652	331	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
653	331	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
645	352	<0.2	<0.2	<0.2	<0.2	<0.2	≤15
646	352	<0.2	<0.2	<0.2	<0.2	<0.2	≤15
654	352	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
707	352	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
708	352	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
709	352	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
710	352	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
710A	352	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5

LOCATION: Control Structure

Room Number	Elev	Measured Dose Rate Gamma mR/hr PFL	TC HU	TC 2	TC 3	TC 6	Acceptance Criteria
156	180	<0.2	<0.2	3.2	32	120	≤100/<1**
157	180	<0.2	<0.2	<0.2	<0.2	0.4	≤10
158	180	<0.2	<0.2	<0.2	<0.2	<0.2	≤10
159	180	<0.2	<0.2	<0.2	<0.2	<0.2	≤10
160	180	<0.2	<0.2	<0.2	<0.2	<0.2	≤15
166	180	<0.2	<0.2	<0.2	<0.2	0.2	≤2.5
259	200	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
263	200	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
263A	200	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
426	239	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
427	239	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
428	239	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
429	239	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
430	239	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
431	239	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
437	239	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
454	239	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
466A	239	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
450	254	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
453	254	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
529	269	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
530	269	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
531	269	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5

Room Number	Elev	Measured Dose Rate Gamma mR/hr PFL	TC HU	TC 2	TC 3	TC 6	Acceptance Criteria
532	269	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
533	269	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
542	289	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
619	304	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
624	332	<0.2	<0.2	<0.2	<0.2	<0.2	≤15
625	350	<0.2	<0.2	<0.2	0.4	0.2	≤2.5

LOCATION: Radwaste and Offgas Enclosure

Room Number	Elev	Measured Dose Rate Gamma mR/hr PFL	TC HU	TC 2	TC 3	TC 6	Acceptance Criteria
213	RW 191	<0.2	<0.2	<0.2	<0.2	0.2	≤2.5
214	OG 195	<0.2	<0.2	<0.2	<0.2	0.5	≤2.5
215	OG 195	<0.2	<0.2	<0.2	<0.2	<0.2	≤2.5
225	OG 195	<0.2	<0.2	<0.2	<0.2	0.4	≤2.5

* Not performed due to high radiation. Dose rate may be determined by extrapolation.

** Acceptance Criteria for Plant Operation/Acceptance Criteria for Plant Shutdown.

4.3 2STP-3, FUEL LOADING

OBJECTIVE

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

ACCEPTANCE CRITERIA

Level 1

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

Level 2

None

RESULTS

2STP-3.1, Fuel Loading

The initial core of Limerick Unit 2 (764 fuel bundles) was successfully loaded in 12 days. Control rod drive system insert/withdraw checks per 2STP-5.1 were performed in parallel during the loading sequence. The Level 1 Acceptance Criteria for demonstrating adequate partial core shutdown margin with the highest worth rod fully withdrawn was satisfied. After the entire core was loaded, a full core verification surveillance test was performed to show that all fuel assemblies were properly loaded and oriented in the core.

The partial core shutdown margin test demonstrates at least 0.38% delta K/K with the analytically determined highest worth control rod fully withdrawn. With 144 fuel assemblies loaded, the highest worth rod (30-43) and diagonally adjacent rod (26-47) were fully withdrawn while observing the nuclear instrumentation. Subcriticality was demonstrated; there was no indication of a continuous positive period.

There were several significant differences in the fuel loading testing between Unit 1 and Unit 2. Installed SRM's were used to monitor the neutron count rate vice Fuel Loading Chambers (FLC's) to save time associated with installing, calibrating, and moving the FLC's. Neutron sources are installed in their alternate locations in order to be closer to the SRM's. In addition, the fuel loading sequence began between an SRM and a neutron source and continued in a spiral pattern around the initial SRM, vice starting from the center of the core and working outward.

The extensive prerequisites to fuel loading included verifying the operability of all nuclear instrumentation, setting SRM rod block setpoints at 1 x 10^4 CPS and scram setpoints at 2 x 10^4 CPS, placing RPS in the non-coincidence scram mode (shorting links removed), verifying completion of control rod preoperational testing, and ensuring reactor coolant chemistry met the Fuel Warranty Specifications.

Fuel loading commenced in accordance with the LGS Core Component Transfer Authorization Sheet (CCTAS). Another method that differed for Unit 2 is that SRM monitoring requirements were exempted for the first 16 bundles based on a detailed analysis which demonstrated that subcriticality is maintained for up to 16 bundles even with all rods withdrawn.

A plot of inverse count rate (1/M) was maintained through the rest of the fuel load to verify subcriticality and to predict the maximum number of fuel assemblies which may be added to the core during a loading step while maintaining subcriticality. Any sizeable changes that occurred in this plot were due to the geometrical effects of fuel being loaded adjacent to the neutron sources.

4.4 2STP-4, SHUTDOWN MARGIN DEMONSTRATION

OBJECTIVE

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

ACCEPTANCE CRITERIA

Level 1

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

Level 2

Criticality should occur within ± 1.0% delta k/k of the predicted critical.

RESULTS

2STP-4.1, In Sequence Critical

This subtest was performed by withdrawing rods in sequence to criticality, then establishing a stable positive period. This period and the moderator temperature at criticality are noted. Shutdown margin (SDM) is determined as the reactivity of the withdrawn control rods, minus the reactivities of the single highest worth rod, and the temperature and period reactivities. An exposure correction factor can be applied if the minimum core shutdown margin occurs later in core life, but for the first fuel cycle of Unit 2 this was at the beginning of the cycle, therefore the exposure correction factor was zero.

The shutdown margin determined for the initial fuel loading was 2.09% delta k/k, which satisfied the Level 1 Acceptance Criteria value of at least 0.38% delta k/k. The value of 2.09 percent included the corrections for temperature of 109.9°F vice 68°F for the cold core, and the correction for the reactivity associated with the 275 second reactor period. Initial criticality occurred at 2246 notches withdrawn. This satisfied the Level 2 Acceptance Criteria of being within ± 1.0% of the predicted critical, which was at 2236 notches with an acceptable range of 1637 to 2397 notches withdrawn.

Subtests performed in conjunction with this were 2STP-6.2, Approach to Criticality - SRM Response to Control Rod Withdrawal and 2STP-10.1, SRM/IRM Overlap.

4.5 2STP-5, CONTROL ROD DRIVE SYSTEM

OBJECTIVES

The objectives of this test are to demonstrate that the Control Rod Drive (CRD) System operates properly over the full range of primary coolant operating temperatures and pressures, and to determine the initial operating characteristics of the CRD system.

ACCEPTANCE CRITERIA

Level 1

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

The mean scram time of all operable CRD's must not exceed the following times (Scram time is measured from the time the pilot scram valve solenoids are de-energized):

Position Inserted to From Fully Withdrawn	Scram Time (Seconds)
45	0.43
39	0.86
25	1.93
05	3.49

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

Position Inserted to From Fully Withdrawn	Scram Time (Seconds)
45	0.45
39	0.92
25	2.05
05	3.70

Level 2

Each CRD must have normal insert and withdrawn speeds of 3.0 $\pm$ 0.6 inches per second, indicated by a full 12 foot stroke in 40 to 60 seconds.

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

RESULTS

2STP-5.1, Insert - Withdraw Checks

Following the completion of the loading of each control cell, functional checks were performed on the associated CRD. These checks consisted of measuring CRD insertion and withdrawal times, measuring insertion and withdrawal drive flows (running and stall), checking for proper coupling, and verifying proper Rod Position Information System (RPIS) operation. All rods met the Level 1 and Level 2 stroke time criteria with insert times ranging between 41.7 and 55.2 seconds and withdraw times ranging between 42.8 and 57.4 seconds. One rod failed to indicate Position 38 on the Four Rod Display. The position indication probe was subsequently replaced, and the rod was retested satisfactorily.

2STP-5.2, Zero Reactor Pressure Friction Testing

Following the completion of fuel loading and CRD functional checks, each CRD was friction tested. All CRDs satisfied the Level 2 Acceptance Criteria without a dp variation greater than 15 psid during a continuous insertion. The measured dp variation ranged between 5.3 and 14.3 psid.

2STP-5.3, Zero Reactor Pressure Scram Testing

Following completion of friction testing, each CRD was scram tested. All applicable Level 1 Acceptance Criteria were satisfied since the mean scram times to position 45, 39, 25 and 05 for all control rods tested were less than 0.43, 0.86, 1.93 and 3.49 seconds, respectively, and the mean scram times of the three fastest rods in every 2 x 2 array to position 45, 39, 25 and 05 were less than 0.45, 0.92, 2.05 and 3.70 seconds, respectively. The mean scram time of all CRDs tested and associated criteria are listed below:

Position Inserted to From Fully Withdrawn	Measured Mean Scram Time (Seconds)	Level 1 Criteria (Seconds)
45	0.23	0.43
39	0.40	0.86
25	0.83	1.93
05	1.50	3.49

2STP-5.5, Rated Reactor Pressure Friction Testing

At rated temperature and pressure, four selected rods were individually friction tested. Each rod satisfied the applicable Level 2 Acceptance Criteria with a dp variation less than 15 psid for a continuous insertion.

The test results are as follows:

Selected Rod	dp Variation (psid)	Level 2 Criteria (psid)
22-11	6.46	$\leq$ 15
26-23	6.10	$\leq$ 15
34-19	9.21	$\leq$ 15
58-43	7.19	$\leq$ 15

2STP-5.6, Rated Reactor Pressure Scram Testing

At rated pressure, during the vessel hydrostatic test (TC Open Vessel), all CRDs were individually scram tested. All CRDs satisfied the applicable Level 1 Acceptance Criteria. The mean scram times of all CRDs are as follows:

Position Inserted to From Fully Withdrawn	Measured Mean Time (Seconds)	Level 1 Criteria (Seconds)
45	0.29	$\leq$ 0.43
39	0.57	$\leq$ 0.86
25	1.24	$\leq$ 1.93
05	2.24	$\leq$ 3.49

The average insertion time of the three fastest rods in each 2 x 2 array also satisfied the applicable Level 1 criteria.

2STP-5.7, Rated Reactor Pressure Insert/Withdraw Checks and Scram Testing of Selected Rods

From the results of 2STP-5.5 and 5.6, four rods were selected at rated reactor pressure for insert and withdraw checks and scram time measurement.

2STP-5.4, Scram Testing of Selected Rods

From the results of previous CRD testing, the four rods with the slowest scram times to position 05 or with unusual operating characteristics-were selected for further testing.

This subtest was performed at the following test conditions: at zero reactor pressure with accumulator pressure just above the low pressure alarm point, at 600 psig reactor pressure with normal accumulator pressure, and at 800 psig reactor pressure with normal accumulator pressure. Each control rod was scrammed three times at every test condition. There were no acceptance criteria verified in this subtest, but each selected rod scram time was verified against Technical Specifications (TS) when performing this subtest during Test Condition Heatup.

The scram times of selected control rods are as follows:

Selected Rod	Measured Time to Position 05 (sec) 0 psig (TC OV)	600 psig (TC HU)	800 psig (TC HU)	TS Limit (sec)
22-11	1.78	2.33	2.47	≤ 7
	1.55	2.56	2.71	≤ 7
	1.56	2.32	2.58	≤ 7
26-11	1.65			≤ 7
	1.69			≤ 7
	1.66			≤ 7
18-39	1.71			≤ 7
	1.69			≤ 7
	1.71			≤ 7
26-47	1.65			≤ 7
	1.69			≤ 7
	1.69			≤ 7
26-23		2.65	2.64	≤ 7
		2.52	2.79	≤ 7
		2.53	2.75	≤ 7
30-15		2.36	2.51	≤ 7
		2.34	2.32	≤ 7
		2.38	2.39	≤ 7
34-07		2.32	2.32	≤ 7
		2.30	2.39	≤ 7
		2.28	2.33	≤ 7

One rod (34-19) did not satisfy the applicable Level 1 and Level 2 Acceptance Criteria on insert and withdrawal speeds. The withdrawal needle valve was adjusted and the rod was retested satisfactorily. The insert and withdrawal speeds are summarized below:

Selected Rod	Insert (sec)	Withdraw (sec)	Level 1 Criteria (sec)	Level 2 Criteria (sec)
34-19	41.7	47.5	> 40	40-60
58-43	43.7	50.4	≥ 40	40-60
22-11	43.7	50.0	≥ 40	40-60
26-23	40.8	50.8	≥ 40	40-60

There was no scram timing acceptance criteria verified in this subtest.

2STP-5.8, Scram Timing of Selected Rods During Planned Scrams of The Startup Test Program

The scram time of four selected rods was measured in conjunction with 2STP-31.1, Loss of Turbine Generator and Offsite Power, 2STP-28.1, Reactor Shutdown from Outside the Control Room, 2STP-25.3, MSIV Full Isolation, and 2STP-27.4, Turbine Trip at TC 6. The four rods tested in 2STP-5.7 were tested during 2STP-25.3 and 2STP-27.4. During 2STP-31.1 and 2STP-28.1, rod 34-19 was not fully withdrawn, therefore, rod 02-43 was substituted.

There were no acceptance criteria verified in these subtests, but each selected rod scram time was verified against Technical Specifications for satisfactory performance.

Selected Rod	Measured Time to Position 05 (sec) 2STP-31.1	2STP-28.1	2STP-25.3	2STP-27.4	TS Limit (sec)
22-11	2.50	2.37	2.29	2.42	≤ 7
26-23	2.62	2.44	2.62	2.51	≤ 7
58-43	2.32	2.24	2.15	2.22	≤ 7
02-43	3.34	2.29			≤ 7
34-19			2.16	2.37	≤ 7

4.6 2STP-6, SRM PERFORMANCE AND CONTROL ROD SEQUENCE

OBJECTIVES

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

ACCEPTANCE CRITERIA

Level 1

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

There must be a minimum of .7 counts per second provided that the signal to noise ratio is at least 2:1.

Level 2

None

RESULTS

2STP-6.1, SRM Signal to Noise Ratio and Minimum Count Rate

Prior to initial critical testing, the shorting links were removed placing the Reactor Protection System (RPS) in the noncoincident scram mode. In addition, the SRM High Flux Alarm and SRM High-High Flux Trip were conservatively adjusted one decade less than their normal values (set to 1×10^4 cps and 2×10^4 cps, respectively).

Prior to rod withdrawal, each SRM was withdrawn to demonstrate SRM signal to noise ratio and minimum count. For each SRM, the observed minimum count rate and signal to noise ratio is identified in the following table:

SRM	Minimum Count Rate	Signal to Noise Ratio
A	200	1999
B	100	999
C	150	1499
D	190	1899

These results satisfy the Level 1 acceptance criteria.

2STP-6.2, Approach to Criticality - SRM Response to Control Rod Withdrawal

Control rods were withdrawn in accordance with the approved Rod Worth Minimizer (RWM) rod sequence for startup. During control rod withdrawals, to avoid rod blocks or scrams, SRM detectors were partially withdrawn, as required, to maintain the observed count rate greater than 100 CPS and less than 1 x 10^4 CPS. In addition, during the control rod withdrawals from all rods-in to criticality, SRM channel readings were recorded for each control rod withdrawal. Upon achieving criticality, the SRM count rate was increased until SRM/IRM overlap was demonstrated.

There were no acceptance criteria verified in this subtest; however, the data recorded had provided baseline data for later comparison of SRM channel response during rod withdrawal.

2STP-6.3, SRM Non Saturation Demonstration

Reactor power was maintained in the intermediate range and the shorting links were installed returning the RPS to the coincident scram mode. SRM nonsaturation was then demonstrated by bypassing each SRM and inserting it into the core until the observed count rate exceeded 3 x 10^5 CPS. SRM rod block and scram setpoints were then restored to their normal values.

There were no acceptance criteria verified in this subtest.

4.7 2STP-9, WATER LEVEL REFERENCE LEG TEMPERATURE

OBJECTIVES

The objectives of this test are to measure the level instrumentation reference leg temperature, recalibrate the water level instruments if the measured temperature is significantly different from the value assumed during the initial end points calibration, and to obtain baseline data on the Narrow Range and Wide Range water level instrumentation.

ACCEPTANCE CRITERIA

Level 1

None

Level 2

The difference between the actual reference leg temperature(s) and the value(s) assumed during initial calibration shall be less than that amount which will result in a scale end point error of 1% of the instrument span for each range.

RESULTS

2STP-9.1, Reference Leg Temperature Comparison

With the reactor at rated temperature and pressure in Test Condition Heatup (TC HU), the following parameters were recorded from various plant instruments and temporary test equipment and subsequently analyzed: reactor water level, reactor building temperature, and drywell temperature readings.

The difference between the measured reference leg temperatures and the temperatures assumed during the initial instrument calibration were less than the amounts that produced a scale end point error of 1% of the measured instrument span for each range, thereby satisfying the acceptance criterion.

This subtest was performed in TC 1, 2, 3, 4, 5 and 6 to determine whether changes in plant conditions had affected reactor water level end point calculations. In each test condition, 1 through 6, the temperatures of the reference leg and the reactor building were within the ranges calculated not to produce an end point error of 1%. Therefore, the applicable acceptance criteria were satisfied.

The test results are as follows:

	Temperature (deg. F)							
	TC HU	TC 1	TC 2	TC 3	TC 4	TC 5	TC 6	Level 2 Criteria
Drywell Temp.	115	120	116	115	118	120	120	90-155
	130	130	129	130	120	120	122	90-155
Rx Bldg Temp.	79	84	78	70	78	78	78	60-100
	80	83	80	72	77	78	77	60-100
	82	85	81	74	78	80	80	60-100
	84	88	84	77	82	83	81	60-100
Ref Leg Temp A	134	133	133	136	131	132	134	90-155
Ref Leg Temp B	132	132	131	133	128	129	131	90-155
Ref Leg Temp C	136	135	137	136	132	132	134	90-155
Ref Leg Temp D	137	124	125	135	129	131	132	90-155

4.8 2STP-10, IRM PERFORMANCE

OBJECTIVES

The objectives of this test are to demonstrate Intermediate Range Monitoring (IRM) System response to neutron flux and to adjust the IRM system to obtain an optimum overlap with the SRM and APRM systems.

ACCEPTANCE CRITERIA

Level 1

Each IRM channel must be on scale before the SRM's exceed their rod block setpoint.

Each APRM must be on scale before the IRM's exceed their rod block setpoint.

Level 2

Each IRM channel must be adjusted so that one-half decade overlap with the SRM's is assured

Each IRM channel must be adjusted one decade overlap with the APRM's is assured.

RESULTS

2STP-10.1, SRM/IRM Overlap

SRM/IRM Overlap was demonstrated during the sequence of testing that began with initial criticality and ended with SRM non-saturation testing. Rods were withdrawn and the SRM's were partially withdrawn when the count rates approached the lowered SRM rod block setpoint (1×10^4 CPS). Following the final detector withdrawal, a normalized count rate was calculated and utilized to determine the count rate of the fully inserted SRM. Rods were then withdrawn until all IRM downscale lights cleared and the increase in count rate was terminated. Data was then taken which adequately demonstrates the SRM/IRM overlap. Once overlap was satisfactorily demonstrated, RPS was taken out of the noncoincident scram mode by the installation of the shorting links.

The following indications were recorded after all IRM readings were above the downscale value of 5/125 (1.6 on 0-40 scale) and SRM count rates were stabilized:

SRM	Normalized Reading	IRM	Range	Reading
A	3.5×10^4 CPS	A	1	4/40
B	2.6×10^4 CPS	B	1	3/40
C	1.8×10^4 CPS	C	1	3/40
D	1.7×10^4 CPS	D	1	3/40
		E	1	4/40
		F	1	4/40
		G	1	3/40
		H	1	3/40

Similar results were obtained after final gain adjustments were made during Test Condition 2.

The applicable Level 1 criterion was satisfied when each IRM channel was on scale before the SRM's exceeded the normal rod block setpoint of 1×10^5 CPS (normalized reading).

The applicable Level 2 criterion was satisfied when each IRM downscale light cleared and all SRM's indicated less than 5×10^4 CPS (half decade from rod block setpoint).

2STP-10.2, IRM Range 6-7 Continuity

During the initial reactor heatup, with IRM's A through H on Range 6, reactor power was increased and stabilized to acquire readings between 50 to 80/125. Then each IRM was switched to Range 7 and the readings observed. If the readings on channels 6 & 7 did not agree within $\pm$ 5%, the IRM in question was bypassed and the high frequency preamplifier was adjusted as necessary.

All IRM's were left with a Range 7 reading within ±5% full scale of the corresponding Range 6 reading. Each high frequency amplifier for IRM Ranges 7 through 10 had to be adjusted to satisfy the ±5% test objective. Following the adjustment of all IRM channels, the as-left readings were recorded as indicated below:

IRM	Range 6 Reading (0-125 Scale)	Range 7 Reading (0-40 Scale)
A	92.0	9.5
B	80.0	8.0
C	80.0	8.0
D	80.0	8.1
E	57.0	6.0
F	80.0	8.5
G	80.0	8.2
H	80.0	8.0

2STP-10.3, IRM/APRM Overlap

IRM/APRM overlap was demonstrated during the initial power increase in Test Condition 1, and following the initial calibration of the Average Power Range Monitors (APRM's) by a reactor heat balance.

All IRM's except E were left with adequate IRM/APRM overlap. Each IRM high frequency amplifier gain had to be adjusted to satisfy the test objective.

The as-left Test Condition 1 readings were recorded as indicated below:

IRM	Range 8 Reading (0-125 Scale)	APRM Reading
A	100	8.1
B	50	7.7
C	105	8.5
D	90	7.4
*E	124	9.0
F	105	8.0
G	76	
H	88	

With the exception of IRM E, all applicable acceptance criteria were satisfied. IRM E was retested in a subsequent test condition.

Prior to Test Condition 2 testing, clarification of the Acceptance Criteria was incorporated into the test. This change included the specific values from Tech Specs and allowed recording of data without going off range by using Range 10 instead of Range 8.

Readings recorded as a result of testing these same objectives during Test Condition 2 demonstrated that only one-half decade between IRMS and ARPMS was assured. Level 2 acceptance criteria of one decade overlap between IRMS and APRMS was not met. This exception was accepted since Technical Specification requires only a one-half decade overlap of the IRM and APRM.

The test results for Test Condition 2 are summarized as follows:

IRM	Range 10 Reading (0-125 Scale)	APRM Reading
A	31	3.5
B	12	3
C	13	3.5
D	17	4
E	24	4
F	17	3
G	20	
H	21	

The applicable Level 1 criterion was satisfied.

4.9 2STP-11, LPRM CALIBRATION

OBJECTIVES

The objectives of this test are to calibrate the Local Power Range Monitoring (LPRM) System and to verify LPRM flux response.

ACCEPTANCE CRITERIA

Level 1

None

Level 2

Each LPRM reading will be within 10% of its calculated value.

RESULTS

2STP-11.1, Verification of Proper Connection of LPRM Detectors and Readout Equipment

The purpose of this subtest was to observe and document Local Power Range Monitor (LPRM) response to neutron flux changes caused by movement of directly adjacent control rods. As each control rod was individually stroked, the response of each LPRM detector in the nearest LPRM string was observed from the panel and on the Plant Monitoring System (PMS) computer.

All 172 LPRM detectors responded properly to local changes in neutron flux, thus assuring proper connection of LPRM detectors and readout equipment.

There were no acceptance criteria associated with this subtest.

2STP-11.2, LPRM Calibration Without Process Computer

This subtest was deleted for Unit 2. The content of this subtest was incorporated into 2STP-13.3, Program Testing at Test Condition 1.

2STP-11.3, LPRM Calibration with Process Computer

The purpose of this subtest was to provide documentation and verification of proper LPRM calibration using the Process Computer in accordance with Plant Surveillance Test Procedure, Traversing In-Core Probe (TIP) Calibration of LPRM's. LPRM values and a complete set of TIP traces are stored using the process computer program, OD-1. The individual amplifier input calibration currents required to produce a full scale meter reading on each LPRM meter are then determined. The process computer program, P-1, calculates the correct LPRM readings for that core condition. New input calibration currents are determined by dividing the original amplifier input currents by the Gain Adjustment Factors (GAF's). These new currents are then applied to give the standard meter reading. The OD-1 is reperformed, and new GAF's are determined. These GAF's must be verified to satisfy the acceptance criteria.

This subtest was performed during Test Condition 3 and 6 at 51% and 97% of rated core thermal power, respectively. The Gain Adjustment Factors (GAF's) were all adjusted to between 0.9 and 1.1 for all LPRM's. The acceptance criterion was satisfied for all LPRM's during each performance.

2STP-11.4, LPRM Operational Verification During Rod Withdrawal

The purpose of this subtest was to document the response to local changes in neutron flux of any LPRM's which failed to properly respond during the performance of 2STP-11.1, Verification of Proper Connection of LPRM Detectors and Readout Equipment, in Test Condition Heatup. Although all 172 LPRM detectors responded satisfactorily in Test Condition Heatup, the Plant Monitoring System (PMS) plots used to document the response for several LPRM's were not obtained. These LPRM's were retested with this subtest. Control rods adjacent to these LPRM's were moved, and all detectors responded properly to the changes in neutron flux.

There were no acceptance criteria associated with this subtest.

4.10 2STP-12, APRM CALIBRATION

OBJECTIVES

The objective of this test is to calibrate the Average Power Range Monitor (APRM) System.

ACCEPTANCE CRITERIA

Level 1

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

Technical specification allowable limits for APRM Scram and Rod Block setpoints shall not be exceeded.

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

Level 2

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

RESULTS

2STP-12.1, Constant Heatup Rate APRM Calibration

The purpose of this subtest was to perform an initial calibration of the APRMs and to verify APRM rod block and scram setpoints. The Gain Adjustment Factors used for the calibration were calculated using a core thermal power determined from a constant reactor coolant heatup rate heat balance. All acceptance criteria were satisfied.

The first part of this test involved taking plant data every 10 minutes during a reactor heatup. The heatup was established and maintained for 1 hour by withdrawing control rods. The heatup rate was held constant during this 1 hour period with the average being 40 degrees F/hr.

For each data set in this 1 hour period, a core thermal power (CTP) was calculated. By dividing the CTP in percent of rated by the individual APRM reading, an APRM Gain Adjustment Factor (AGAF) was calculated for each APRM, for each data set. An average AGAF was then calculated for each APRM channel.

Steady state plant conditions were established for the APRM calibration. Since both the AGAF (1) and the AS FOUND APRM READING (2) were low, a direct adjustment of the APRMs to the desired reading would be impractical; therefore, each APRM was first calibrated by taking the AS FOUND APRM READING and multiplying this by an artificial AGAF of 0.5 to yield the desired AS LEFT APRM READING. The APRM amplifier gain was adjusted to make the APRM read as close to but not less than the desired AS LEFT APRM READING. This APRM reading is the AS LEFT APRM READING (3).

The AGAF USED (4) was determined by dividing the AS LEFT APRM READING (3) by the AS FOUND APRM READING (2). The ADDITIONAL AGAF (5) was calculated by dividing the AGAF by the AGAF USED, and it would be used for next APRM calibration. When reactor power was increased to a higher level, the APRM readings were then recorded as the AS FOUND APRM READING (6). The DESIRED APRM READING (7) for each channel was calculated by multiplying the ADDITIONAL AGAF by the AS FOUND APRM READING. The APRM amplifier gain was adjusted to make the APRM read as close as possible to but not less than the DESIRED APRM READING.

The following are the results of the APRM Calibration:

APRM	(1) AGAF	(2) AS FOUND APRM READING	(3) AS LEFT APRM READING	(4) AGAF USED	(5) ADDITIONAL AGAF
A	0.159	1.15	0.6	0.522	0.305
B	0.218	1.4	0.7	0.50	0.436
C	0.202	1.1	0.6	0.545	0.371
D	0.179	1.5	0.75	0.50	0.358
E	0.201	1.95	1.0	0.513	0.392
F	0.194	1.3	0.65	0.50	0.388

APRM	(6) AS FOUND APRM READING	(7) DESIRED APRM READING	(8) AS LEFT APRM READING
A	3.9	1.19	2.5
B	3.6	1.57	2.4
C	3.5	1.30	2.5
D	3.5	1.25	2.4
E	4.0	1.57	2.6
F	3.5	1.36	2.4

The rod block and scram setpoints for each APRM Channel were checked to verify that they would cause a rod block and scram at 12% and 15% indicated CTP, respectively. All APRMs satisfied this criteria.

The scram and rod block setpoints on each APRM channel were recorded as follows:

APRM	ROD BLOCK SETPOINT % CTP		SCRAM SETPOINT % CTP	
	Measured	Required	Measured	Required
A	11.98	$\leq$ 12	14.91	$\leq$ 15
B	11.80	$\leq$ 12	14.75	$\leq$ 15
C	11.90	$\leq$ 12	14.92	$\leq$ 15
D	11.90	$\leq$ 12	14.90	$\leq$ 15
E	11.89	$\leq$ 12	14.89	$\leq$ 15
F	11.95	$\leq$ 12	14.76	$\leq$ 15

2STP-12.2, Low Power APRM Calibration

This subtest was performed during Test Condition 2 at approximately 20% power. The purpose of this subtest was to calibrate the APRM channels against core thermal power (CTP) which was determined from a manual steady state heat balance calculation and to verify APRM rod block and scram setpoints. All acceptance criteria were satisfied.

Data was taken while the plant was at steady state conditions and then core thermal power was calculated using this data. The APRM Gain Adjustment Factor (AGAF) for each channel was calculated by dividing the CTP in % of rated by the initial APRM reading. The APRMs were calibrated by recording the present APRM reading as the AS FOUND APRM READING. The DESIRED APRM READING was calculated by multiplying the AS FOUND APRM READING by the channel's AGAF. The APRM amplifier gain was then adjusted to make the ARPM read as close to but not less than the DESIRED APRM READING.

The following are the results of the APRM calibration:

APRM	INITIAL APRM READING	AGAF	AS FOUND APRM READING	DESIRED APRM READING	AS LEFT APRM READING
A	24.5	0.597	24.5	14.62	15.0
B	27.5	0.532	27.5	14.62	15.0
C	28.0	0.522	28.0	14.62	15.0
D	25.5	0.573	25.5	14.62	15.0
E	27.0	0.541	27.0	14.62	15.0
F	25.0	0.585	25.0	14.62	15.0

The scram and flow biased rod block setpoints for each APRM channel were verified to be within technical specification limits. All APRMs satisfied technical specification limits.

2STP-12.3, High Power APRM Calibration

The purpose of this subtest was to calculate the APRM channels using a core thermal power (CTP) which was determined from a steady state heat balance. The steady state heat balance (which used the on-line process computer programs OD-3 and P-1) also determined the Core Maximum Fraction of Limiting Power Density (CMFLPD).

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The APRMs were calibrated by adjusting the amplifier gain to make the APRMs read the desired value. This value was CTP in percent of rated and is defined as the product of the current APRM reading and the AGAF. However, when CMFLPD was greater than the Fraction of Rated Thermal Power (FRTP), the APRM was adjusted to read CMFLPD in percent to meet the requirements of Technical Specification 3/4.2.2.

This subtest was performed during Test Conditions 2, 3, 5, 6 and Warranty Run. All APRM's were calibrated to read equal to or greater than actual core thermal power.

The following are the results of APRM calibration performed during Test Condition Warranty Run:

APRM	AS FOUND APRM READING	DESIRED APRM READING	AS LEFT APRM READING
A	99.8	99.6	99.8
B	99.8	99.6	99.8
C	99.7	99.6	99.7
D	99.7	99.6	99.7
E	99.8	99.6	99.8
F	99.7	99.6	99.7

In addition, the flow biased scram and rod block setpoints were verified to be less than the allowable values given in Technical Specifications. All applicable acceptance criteria were satisfied.

4.11 2STP-13, PROCESS COMPUTER

OBJECTIVES

The objective of this test is to verify the performance of the Plant Monitoring System (PMS) Process Computer software under plant operating conditions.

ACCEPTANCE CRITERIA

Level 1

None

Level 2

The Minimum Critical Power Ratio (MCPR) calculated by BUCLE (the off-site Mark III computer system program) and by PMS either:

- are in the same fuel assembly and do not differ in value by more than 2% or

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

The maximum Linear Heat Generation Rate (LHGR) calculated by BUCLE and by PMS either:

- are in the same fuel assembly and do not differ in value by more than 2%, or

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

The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) calculated by BUCLE and by PMS either:

- are in the same fuel assembly and do not differ in value by more than 2%, or

- for the case in which the MAPLHGR calculated by PMS is in a different assembly than that calculated by BUCLE, for each assembly, the MAPLHGR and APLHGR calculated by the two methods shall agree within 2%.

The Local Power Range Monitor (LPRM) gain adjustment factors calculated by BUCLE and PMS agree to within 2%.

Each LPRM reading will be within 10% of its calculated value.

RESULTS

2STP-13.1, Static System Test Case

The Static System Test Case associated with PMS/TIP machine interface was satisfactorily performed during Test Condition Open Vessel. Proper OD-1 operation, including interface with the TIP machines, agreement between computer and TIP machine index settings, and generation of CRT and typer messages, was demonstrated.

There were no acceptance criteria associated with this test.

This subtest consisted of using the Nuclear System Test Simulator to input various control rod patterns and using PMS software to simulate plant conditions prior to actual plant operation during Test Condition Open Vessel. OD-1 was then run with various simulated plant conditions to verify that the appropriate failure checks were made and the correct CRT and typer messages were generated. The TIP machines were then operated to verify proper computer/TIP machine interface. The TIP indexers were switched to each position to verify that the computer correctly monitored the index settings. Various TIP operation failure checks, such as waiting too long to start a traverse, stopping the traverse mid-core, moving a control rod, failing the simulated TIP signal, and varying the APRM signal during traverses, were also tested. Finally, a complete set of TIP traverses was performed. Other programs utilized during the test included OD-14, Control Rod Positions and OD-15, Computer Outage Recovery Monitor.

2STP-13.2, TIP Alignment at Rated Temperature

The TIP Alignment test at Test Condition Heatup was performed with the reactor operating at rated temperature and pressure.

The purpose of this test was to determine if the core top and core bottom (NCCB) limits or the x-y plotter span required adjustments. Each of the TIP guide tubes were probed, and the full-in index position at hot conditions was verified to be greater than or equal to the value at cold conditions. Three TIP channels were adjusted per Calibration Procedure for the Traversing In-Core Probe System. No X-Y plotter adjustments were required.

There were no acceptance criteria associated with this subtest.

2STP-13.3, Program Testing at Test Condition 1

Program Testing was performed during Test Condition 1 at 14% of rated core thermal power. During this subtest the TIP core limits were checked against the limits set in 2STP-13.2, TIP Alignment at Rated Reactor Pressure, performed during Test Condition Heatup. The average difference between the axial TIP traces, and the design values, were found to be within criteria; therefore, no changes to the TIP core limits were necessary.

A complete OD-1, Whole Core LPRM Calibration and BASE distribution, was performed confirming correct TIP-Computer interface. OD-7, Control Rod Positions, and OD-8, LPRM readings were verified.

The Level 2 Acceptance Criteria for each LPRM reading to be within 10% of its calculated value was met for all but 3 LPRM's. The results were accepted, and a subsequent LPRM calibration during TC 3 was performed with satisfactory results.

2STP-13.4, Dynamic System Test Case

This subtest was performed during Test Condition 2 at 23% MWt rated.

Proper operation of the NSS PMS programs were verified by comparing their outputs against manual calculations, off-line computer calculations (BUCLE), and outputs from other programs.

The following are the results of core thermal limits and LPRM GAF calculations:

MCPR COMPARISON

Limiting Bundle	MCPR (P1)	MCPR (BUCLE)	Difference	Level 2 Criteria
33-32	3.837	3.812	0.66%	≤ 2%

MLHGR COMPARISON

Limiting Bundle	MLGHR (P1)	MLGHR (BUCLE)	Difference	Level 2 Criteria
33-32	3.98	3.98	0.00	≤ 2%

MAPLHGR COMPARISON

Limiting Bundle	MAPRHGL (P1)	MAPLHGR (BUCLE)	Difference	Level 2 Criteria
33-32	3.29	3.29	0.00	≤ 2%

LPRM GAF COMPARISON

Limiting LPRM	GAF (OD-10)	GAF (BUCLE)	Difference	Level 2 Criteria
32-41-B	0.935	0.930	0.54%	≤ 2%

The following checks were performed during the performance of this subtest:

1. Correct initialization of PMS was verified including verification that all exposure data was zero.

2. Proper scanning by plant sensors.

3. PMS was proven to be able to initialize data using OD-15.

4. The ability of PMS to correctly perform a whole-core LPRM calibration was verified by checking the results against manual calculations.

5. PMS power distribution and core thermal limits calculations were verified to be correct.

6. The proper operation of the LPRM digital filtering initialization function and the LPRM drift diagnostic test were verified.

The following PMS programs were declared operational upon successful completion of this subtest:

P1	Periodic Core Performance Calculations
P2	Daily Core Performance Summary
P3	Monthly Core Performance Summary
P4	10 Minute Core Energy Increment
P5	Drifting LPRM Diagnostic and Digital Filtering
P6	15 Second Database Update
OD-3	Core Thermal Power and APRM Calibration
OD-6	Thermal Limit Data
OD-10	Edit Specified Data Arrays
OD-13	LPRM Sensitivity
OD-14	Substitute and Unknown Control Rod Positions
OD-15	Computer Outage Recovery
OD-16	Target Exposure and Power Data
OD-18	LPRM Alarm Trip Settings
OD-20	Refueling Update Monitor

2STP-13.5, Program Testing at Test Condition Two

This subtest was performed during Test Condition 2 at 38.5% core thermal power.

This subtest performed an operability check on OD-2, Specified LPRM Substitute Value and BASE Distribution, and OD-9, Axial Interpolation Data, by comparing the outputs of these programs with each other and with the data stored in various NSS arrays in PMS. OD-2 and OD-9 were declared operational upon successful completion of this test.

There were no acceptance criteria for this test.

2STP-13.6, Program Testing at Test Condition Three

This subtest was performed at 70.2% core thermal power during Test Condition 3. The purpose of this subtest is to verify the operation and calculations of the P-1 program and OD-10, Option 22 edits for asymmetric rod pattern conditions.

The test compared values of the symmetric and asymmetric modes for the P-1 program and the OD-10, Option 22 edit. All asymmetric values were within 15% of the symmetric values verifying the operability of these programs in the asymmetric mode.

There were no acceptance criteria for this test.

2STP-13.9, Program Testing During Power Changes

This subtest was performed during the ascent to Test Condition 6. The purpose of this subtest was to verify that PMS is capable of following power and core flow changes and can accurately calculate LPRM trip levels, and thermal limits during substantial changes in CTP. These checks were made during a power increase from 73% to 95%. PMS operated satisfactorily in all areas for both changes.

There were no acceptance criteria for this test.

4.12 2STP-14, RCIC SYSTEM

OBJECTIVES

The objectives of this test are to verify the proper operation of the Reactor Core Isolation Cooling (RCIC) System over its expected operating pressure and flow ranges, and to demonstrate reliability in automatic starting from cold standby when the reactor is at power conditions.

ACCEPTANCE CRITERIA

Level 1

The average pump discharge flow must be equal to or greater than 100% rated value after 30 seconds have elapsed from automatic initiation at any reactor pressure between 150 psig and rated.

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

Level 2

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

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

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

The delta P switches for the RCIC steam supply line high flow isolation trip shall be calibrated to actuate at the value specified in the plant technical specifications (about 300%).

The RCIC system must have the capability to deliver specified flow against normal rated reactor pressure without the normal AC site power supply.

RESULTS

2STP-14.1, RCIC Functional Demonstration CST to CST at 150 psig

The RCIC run at a reactor pressure of 150 psig from Condensate Storage Tank (CST) to CST, was attempted 3 times. The first attempt was aborted due to a steam leak at a piping flange joint. The second attempt was aborted due to a governor control valve malfunction. The third attempt was completed successfully. The test consisted of a manual start, flow steps in manual and automatic, and quick start.

No acceptance criteria are directly applicable to this subtest.

2STP-14.2, Functional Demonstration and Controller Optimization at Rated Pressure CST to CST

This subtest was a RCIC run at 920 psig reactor pressure from CST to CST. This subtest consisted of a manual quick start, inner and outer loop control system tuning, flow steps in manual and automatic, and a quick start. All acceptance criteria were satisfied.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
Time to Rated Flow	14.5 seconds	$\leq$ 30 seconds
Turbine Trip or Isolation After Initiation	None	None
Maximum Turbine Speed	4317 rpm	<4725 rpm
Steam Leakage	None	None

2STP-14.3, Stability Check CST to CST at 150 psig

This subtest was not performed since system response was deemed acceptable based on previous runs.

2STP-14.4, Controller Optimization During RPV Injection at Rated Pressure

This subtest was a RCIC vessel injection at 920 psig reactor pressure. The test consisted of a manual start, flow steps in manual and automatic, and a quick start. This test also monitored turbine steam exhaust pressure to verify it is below the high exhaust pressure turbine trip setpoint. All acceptance criteria were met with the exception of the Level 2 criteria for RCIC system related variables to have a decay ratio of not greater than 0.25. The variables that failed this criteria are as follows: RCIC Flow Controller Output, Turbine Speed, and RCIC System Flow. This exception was acceptable based on EGM output and control valve response meeting the acceptance criteria, and aforementioned criteria failures were attributed to feedwater system intervention.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
Time to Rated Flow	17.8 seconds	$\leq$ 30 seconds
Turbine Trip or Isolation After Initiation	None	None
Maximum Turbine Speed	4250 rpm	<4725 rpm
Steam Leakage	None	None
Decay Ratio	0.13 - 0.53	$\leq$ 0.25

2STP-14.5, Stability Check CST to RPV at 150 psig

This subtest was a RCIC vessel injection at 150 psig and was performed during Test Condition 2. The test was unable to be completed due to the turbine being manually tripped because of reactor water level problems caused by the feedwater system not being able to compensate for the additional RCIC System Flow at the power level at which the test was performed. Enough data was collected to satisfy all Level 1 acceptance criteria. Level 2 acceptance criteria for steam leakage to atmosphere and decay ratios of RCIC system related variables were unable to be verified (not recorded); however, reperformance of the test was deemed unnecessary based on previous RCIC runs and data collected.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
Time to Rated Flow	10.9 seconds	$\leq$ 30 seconds
Turbine Trip or Isolation After Initiation	None	None
Maximum Turbine Speed	2450 rpm	<4725 rpm
Steam Leakage	None	None
Decay Ratio	Not Recorded	$\leq$ 0.25

2STP-14.6, RCIC Cold Quick Start at Rated Pressure - CST to RPV and 2STP-14.8, RCIC Endurance Run

2STP-14.6 consisted of two cold quick starts at rated pressure with injection to the reactor vessel with no RCIC operation for 72 hours beforehand. The first cold quick start was completed successfully in Test Condition 1. All Level 1 acceptance criteria were satisfied. RCIC system flow exceeded the Level 2 acceptance criteria of a decay ratio not greater than 0.25. This was acceptable based on EGM output and control valve position meeting the acceptance criteria. All other Level 2 acceptance criteria were met. The second cold quick start was performed in Test Condition 2 and 2STP-14.8, RCIC Endurance Run, was performed concurrently. All Level 1 acceptance criteria for 2STP-14.6 were met. As in the first cold quick start, RCIC system flow exceeded the Level 2 acceptance criteria of a decay ratio not greater than 0.25 and was deemed acceptable for the same reasons. All other Level 2 acceptance criteria were met. No acceptance criteria were associated with 2STP-14.8 which consisted of a quick start followed by 2 hours of continuous operation CST to CST, and finally two consecutive quick starts to the reactor vessel.

The test results are summarized as follows:

Parameter	MEASURED VALUE TC 1	TC 2	Acceptance Criteria
Time to Rated Flow	17.8 sec	17.8 sec	≤ 30 seconds
Turbine Trip or Isolation After Initiation	None	None	None
Maximum Turbine Speed	4330 rpm	4250 rpm	<4725 rpm
Steam Leakage	None	None	None
Decay Ratio	0 - 0.80	0 - 1.0	≤ 0.25
Steam Line Delta P		375" H_2O	≤ 213" H_2O*

*This value is a setpoint recommendation and would require a Technical Specfication change if implemented.

2STP-14.7, Surveillance Test CST to CST and
2STP-14.9, Loss of AC Power

These two subtests were performed concurrently during Test Condition 2. 2STP-14.7 consisted of a quick start injecting to the CST, this lead into 2STP-14.9 which consisted of a quick start followed by 2 hours of continuous operation CST to CST, and finally two consecutive quick starts to the reactor vessel. All phases of the test were performed without AC power being supplied to RCIC components, including the room cooler. All acceptance criteria for both tests were satisfied and oil temperature, room temperature and battery voltage remained within the prescribed limits.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
Time to Rated Flow	16.6 seconds	≤ 30 seconds
Turbine Trip or Isolation After Initiation	None	None
Maximum Turbine Speed	4400 rpm	<4725 rpm
Decay Ratio	0	≤ 0.25
Satisfactory System Operation without AC Power	Yes	Yes

4.13 2STP-15, HPCI SYSTEM

OBJECTIVES

The objectives of this test are to verify the proper operation of the High Pressure Coolant Injection (HPCI) System over its expected operating pressure and flow ranges, and to demonstrate reliability in automatic starting from cold standby when the reactor is at rated pressure conditions.

ACCEPTANCE CRITERIA

Level 1

The average pump discharge flow must be equal to or greater than 100% rated value after 30 seconds have elapsed from automatic initiation at any reactor pressure between 200 psig and rated.

The HPCI turbine shall not trip or isolate during auto or manual start tests.

Level 2

In order to provide an overspeed isolation trip margin, the transient first peak shall not come closer than 15% (of rated speed) to the overspeed trip, and subsequent speed peaks shall not be greater than 5% above the rated turbine speed.

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

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

The delta P switches for the HPCI steam supply line high flow isolation trip shall be calibrated to actuate at the value specified in plant technical specifications (about 300%).

RESULTS

2STP-15.1, Functional Demonstration CST to CST at 200 psig

This subtest was a HPCI run at a reactor pressure of 200 psig from the Condensate Storage Tank (CST) to CST during Test Condition Heatup. This test consisted of a manual quick start, flow steps in manual and automatic, and a quick start. Test was completed satisfactorily.

No acceptance criteria were associated with this test.

2STP-15.2, Functional Demonstration and Controller Optimization at Rated Pressure CST to CST

This subtest was a HPCI run at 920 psig reactor pressure from CST to CST during TC Heatup. This test consisted of a manual start, inner and outer loop tuning, flow steps in both manual and automatic, and a quick start. This test also monitored turbine steam exhaust pressure to verify it remains below the high turbine exhaust pressure setpoint. This test was completed successfully and all Level 1 acceptance criteria were met. All Level 2 acceptance criteria were met with the exception of the subsequent speed peak not being greater than 5% above the rated turbine speed. Subsequent speed peak obtained was 4566 rpm while the limit was 4399 rpm. This was determined to be caused by the position to which HPCI Test Loop Shutoff Valve was positioned. Results were deemed to be acceptable.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
Time to Rated Flow	17.9 seconds	$\leq$ 30 seconds
Turbine Trip or Isolation After Initiation	None	None
First Turbine Peak Speed	2684 rpm	<4609 rpm
Subsequent Turbine Peak Speed	4566 rpm	< 4399 rpm
Steam Leakage	None	None

2STP-15.3, Stability Check CST to CST at 200 psig

This subtest was optional and was not performed since the performance of HPCI system was adequately demonstrated and documented during the performance of the other HPCI tests.

2STP-15.4, Controller Optimization During RPV Injection at Rated Pressure

This subtest was a HPCI run at rated reactor pressure injecting to the reactor vessel from the CST during TC 3. This subtest consisted of a manual start, flow steps in both manual and automatic, and a quick start. Erratic flow response occurred during step changes at reduced flow conditions due to the relative size of the step change induced (approximately 38%). The turbine was manually tripped when the suction valves shifted from the CST to the suppression pool. The test recommenced with the performance of the quick start and it was determined that enough data was collected to satisfy the intent of the test. All Level 1 acceptance criteria were met. All Level 2 acceptance criteria were met with the exception of the decay ratios of greater than 0.25 for HPCI system flow and HPCI discharge pressure. This was determined not to be a function of inner or outer loop control system and the results were acceptable.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
Time to Rated Flow	19.9 seconds	$\leq$ 30 seconds
Turbine Trip or Isolation After Initiation	None	None
First Turbine Peak Speed	4175 rpm	<4609 rpm
Subsequent Turbine Peak Speed	4150 rpm	<4399 rpm
Steam Leakage	None	None
Decay Ratio	0 - 1.33	$\leq$ 0.25
Steam Line Delta P	2012" H_2O	>343" H_2O

2STP-15.5, HPCI Cold Quick Start at Rated Pressure CST to RPV

This subtest was a HPCI run at rated reactor pressure injecting to the reactor vessel from the CST performed during TC 3. This subtest consisted of a cold quick start (no HPCI operation for at least 72 hours) to the reactor vessel. The test was completed successfully, and all acceptance criteria were met.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
Time to Rated Flow	22.5 seconds	$\leq$ 30 seconds
Turbine Trip or Isolation After Initiation	None	None
First Turbine Peak Speed	4110 rpm	<4609 rpm
Subsequent Turbine Peak Speed	4200 rpm	<4399 rpm

2STP-15.6, HPCI Surveillance Test CST to CST

This subtest was an optional test that was not performed since the performance of the HPCI System was adequately demonstrated and documented during the performance of the other HPCI tests.

2STP-15.7, HPCI Endurance Run

This subtest was a HPCI run at rated reactor pressure with the HPCI system taking a suction from the CST and discharging to the CST during TC Heatup. This test was to consist of continuous operation following a quick start for 2 hours or until oil temperature stabilized; however, the test was terminated after 1 1/2 hours due to rising suppression pool temperatures. It was determined that since the oil temperatures were nearly stable, the intent of the test was satisfied.

No acceptance criteria were associated with this test.

4.14 2STP-16, SELECTED PROCESS TEMPERATURES

OBJECTIVES

The objective of this Subtest is to assure that the measured bottom hr d drain temperature corresponds to bottom head coolant temperature during normal operations.

ACCEPTANCE CRITERIA

Level 1

None

Level 2

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

RESULTS

2STP-16.1, Minimum Recirculation Pump Speed Determination

This Subtest was deleted for Unit 2 since it is not required per the FSAR.

2STP-16.2, Bottom Head Drain Temperature

This subtest was performed during Test Condition 3 at approximately 65% power. The accuracy of the bottom head drain temperature was verified by comparing its measurement with the recirculation loop coolant temperature at rated flow when adequate mixing in the vessel lower head can be assumed.

The average difference in the temperatures was 24.6 degrees F. Thus the applicable acceptance criterion was satisfied.

2STP-16.3, Recirculation Pump Trip Recovery Data

This Subtest was not performed for Unit 2 since it is not required per the FSAR.

4.15 2STP-17, SYSTEM EXPANSION

OBJECTIVES

This Subtest verifies that safety related piping systems and other piping systems as identified in the FSAR expand in an acceptable manner during plant heatup and power escalation. Specific objectives are to verify that:

Piping thermal expansion is as predicted by design calculations.

Snubbers and spring hangers remain within operating travel ranges at various piping temperatures.

Piping is free to expand without interferences.

ACCEPTANCE CRITERIA

Level 1

There shall be no obstructions which will interfere with the thermal expansion of the Main Steam (inside drywell) and Reactor Recirculation piping systems.

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

Level 2

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

Snubbers do not become extended or compressed to within 1/4 inch of their total allowable travel (on scale reading) and they retain swing clearance.

Measured displacements compared with the calculated displacements are within the specified range.

Residual displacements, when measured following system return to ambient temperature, do not exceed the greater of $\pm$ 1/8 in. or $\pm$ 25% of the maximum displacements measured during system initial heatup.

All hangers and snubbers shall be within their normal operating range.

Spring hangers do not become extended or compressed beyond their working range (on scale reading).

RESULTS

2STP-17.1, Measured Pipe Displacements (Selected BOP Systems)

System expansion testing was performed on selected segments of the following BOP piping systems:

a. Main Steam (loops B and C, outside drywell)

b. Residual Heat Removal (shutdown cooling mode supply/return and LPCI)

c. Core Spray (Loop A, inside drywell)

d. High Pressure Coolant Injection (turbine steam supply)

e. Reactor Core Isolation Cooling (turbine steam supply)

f. Reactor Water Cleanup (return from the regenerative heat exchangers to the RPV; RWCU suction from reactor recirculation line to drywell wall).

The thermal movements of system piping were measured during Test Condition Open Vessel (baseline), Test Condition Heatup, and following reactor initial cooldown from normal operating temperature.

Initial piping positions were determined, relative to structural reference points, prior to reactor heatup in order to establish baseline data.

Piping movements were measured using both remotely monitored instrumentation and direct manual/visual methods. Spring hangers and snubbers on specified piping segments were inspected to verify that these devices did not become extended or compressed beyond their working range.

System expansion testing for Main Steam, High Pressure Coolant Injection and Reactor Core Isolation Cooling was performed at reactor moderator temperatures of 360 $\pm$ 25 degrees F, 450 $\pm$ 25 degrees F, and at rated reactor temperature (535 degrees F) and pressure.

System expansion testing for Residual Heat Removal, Core Spray, and Reactor Water Cleanup was performed at rated reactor temperature and pressure.

Residual displacements for all tested systems were determined subsequent to the cooldown from the initial reactor heatup.

Several problems were encountered and some data points did not meet acceptance criteria during the performance of this subtest. This subtest only had level 2 acceptance criteria. They were minor in nature and were reviewed by Project Engineering and were deemed acceptable and therefore, required no further action.

The test results were summarized as follows:

Parameter	Measured Displacement (inch)				Acceptance Criteria
	360°F	450°F	535°F	Cooldown	
Main Steam					
DT.YA.01	-0.39				-0.47 ± 0.40
		-0.57			-0.64 ± 0.40
			-0.78		-0.84 ± 0.42
				-0.03	± 0.2
DT.ZA.02	-0.42				-0.65 ± 0.40
		-0.62			-0.85 ± 0.42
			-0.80		-1.09 ± 0.55
				-0.28	± 0.2
DT.ZA.03	1.92				2.18 ± 1
		2.28			2.98 ± 1
			2.52		3.93 ± 1
				-0.05	± 0.63
DT.XA.04	4.21				4.26 ± 1
		3.12			5.82 ± 1
			3.72		7.67 ± 1
				-2.53	± 1.06
DT.XA.05	1.03				0.99 ± 0.49
		1.52			1.35 ± 0.67
			1.79		1.74 ± 0.87
				0.27	± 0.45
DT.XA.06	0.84				0.85 ± 0.42
		1.22			1.16 ± 0.58
			1.45		1.53 ± 0.76
				0.17	± 0.36
DT.XA.07	4.33				3.93 ± 1
		0.54			5.36 ± 1
			3.53		7.07 ± 1
				-3.54	± 1.08

Parameter	Measured Displacement (inch)				Acceptance Criteria
	360°F	450°F	535°F	Cooldown	
Main Steam (cont.)					
DT.ZA.08	2.48				1.50 ± 0.75
		3.65			2.05 ± 1
			N/A		2.70 ± 1
				-0.002	+ 0.91
DT.ZA.09	-0.92				-1.18 ± 0.59
		-1.27			-1.56 ± 0.78
			-1.54		-2.01 ± 1
				-0.29	± 0.39
DT.YA.10	-0.43				-0.35 ± 0.40
		-0.62			-0.48 ± 0.40
			-0.83		-0.64 ± 0.40
				-0.007	± 0.21
DT.ZA.11	0.88				0.61 ± 0.40
		1.10			0.83 ± 0.41
			1.34		1.10 ± 0.55
				-0.11	± 0.34
DT.XA.12	0.76				0.72 ± 0.40
		1.17			0.98 ± 0.49
			1.41		1.29 ± 0.64
				0.25	± 0.35
DT.XA.13	0.45				0.57 ± 0.40
		0.78			0.78 ± 0.40
			0.92		1.02 ± 0.51
				0.26	± 0.23
DT.ZA.14	-0.71				-0.80 ± 0.40
		-0.97			-1.10 ± 0.55
			-1.19		-1.44 ± 0.72
				-0.15	± 0.3

Parameter	Measured Displacement (inch)				Acceptance Criteria
	$360^{o}F$	$450^{o}F$	$535^{o}F$	Cooldown	
Main Steam (cont.)					
DT.XA.15	0.24				0.27 ± 0.40
		0.51			0.36 ± 0.40
			0.90		0.47 ± 0.40
				0.14	± 0.13
HPCI					
DT.ZD.01	1.05				0.89 ± 0.44
		1.41			1.23 ± 0.61
			1.69		1.64 ± 0.82
				-0.27	± 0.42
DT.XD.02	-0.89				-0.91 ± 0.45
		-1.18			-1.18 ± 0.59
			-1.35		-1.52 ± 0.76
				-0.12	± 0.34
DT.YD.03	-0.63				-0.66 ± 0.40
		-0.83			-0.88 ± 0.44
			-0.99		-1.17 ± 0.58
				0.06	± 0.25
RCIC					
DT.YF.01	-0.23				-0.62 ± 0.40
		-0.26			-0.90 ± 0.45
			-0.51		-1.23 ± 0.61
				0.15	± 0.13
DT.XF.02	0.16				0.83 ± 0.41
		0.21			1.10 ± 0.55
			0.35		1.42 ± 0.71
				-0.02	± 0.09

Parameter	Measured Displacement (inch)				Acceptance Criteria
	360°F	450°F	535°F	Cooldown	
RCIC (cont.)	-				
DT.ZF.03	0.76				0.98 ± 0.49
		0.99			1.35 ± 0.67
			1.30		1.78 ± 0.89
				0.16	± 0.33
Core Spray					
DT.XG.01			-0.27		-0.48 ± 0.40
				0.032	± 0.07
DT.ZG.02			-0.53		-0.64 ± 0.40
				0.04	± 0.13
DT.YG.03			1.04		1.21 ± 0.60
				-0.74	± 0.26
RHR LPCI					
DT.XH.01			0.77		0.76 ± 0.40
				0.02	± 0.19
DT.YH.02			0.75		0.79 ± 0.40
				0.02	± 0.19
DT.ZH.03			0.41		0.47 ± 0.40
				0.01	± 0.10
RHR SDC Supply					
DT.SI.01			0.47		0.68 ± 0.40
				-0.11	± 0.12
DT.YI.02			-0.74		-0.79 ± 0.40
				-0.08	± 0.19
DT.XI.03			-1.11		-1.16 ± 0.58
				0.09	± 0.28

Parameter	Measured Displacement (inch)				Acceptance Criteria
	360°F	450°F	535°F	Cooldown	
RHR SDC Return					
DT.ZJ.01			0.90	0.03	0.99 ± 0.49 ± 0.23
DT.XJ.02			-0.85	0.10	-0.96 ± 0.48 ± 0.21
DT.YJ.03			-0.28	0.06	-0.65 ± 0.40 ± 0.07
RWCU					
DT.ZL.01			0.60	0.23	1.23 ± 0.61 ± 0.15
DT.XM.01			-0.17	-0.02	-0.25 ± 0.40 ± 0.04
DT.YM.02			0.08	-0.02	0.07 ± 0.40 ± 0.02
DT.ZM.03			-0.42	0.02	-0.46 ± 0.40 ± 0.11
DT.ZN.01			-0.63	1.13	1.45 ± 0.72 ± 0.16
DT.YN.02			-0.23	-0.07	-0.30 ± 0.40 ± 0.06
DT.XN.03			-0.79	-0.19	-1.44 ± 0.72 ± 0.2

Parameter	Measured Value	Acceptance Criteria
Snubbers	No	Not extended or compressed to within 1/4" of total travel
Spring Hangers	Yes	Remained within working range

2STP-17.2, Measured Pipe Displacements (Feedwater and RWCU Systems)

This subtest monitored the thermal expansion of feedwater system piping, downstream of the high pressure heaters, as system temperature increased during power ascension. Sections of RWCU system piping, connecting to feedwater and affected by feedwater system temperature, were also monitored.

Piping movements were measured using both remotely monitored instrumentation and direct manual/visual methods. Spring hangers on specified piping segments were inspected to verify that they did not become extended or compressed beyond their working range.

Thermal expansion data was taken at Test Condition Open Vessel (Baseline Measurements), Test Condition 2 (275 ± 28 degrees F feedwater temperature) and Test Condition 6 (420 ± 40 degrees F feedwater temperature). Residual displacements were determined following feedwater system cooldown during an outage in TC 6.

Several problems were encountered and some data points did not meet acceptance criteria during the performance of this subtest. This subtest only had Level 2 acceptance criteria. They were minor in nature and were reviewed by Project Engineering and were deemed acceptable and therefore, required no further action.

The test results are summarized as follows:

Parameter	Measured Displacement (inch) 275°F	Measured Displacement (inch) 420°F	Measured Displacement (inch) Cooldown	Acceptance Criteria
Feedwater				
DT.XB.01	0.1	0.25	0.029	0.23 ± 0.40 0.42 ∓ 0.40 ± 0.063
DT.YB.02	0.37	0.27	-0.018	0.44 ± 0.40 0.31 ∓ 0.40 ± 0.093
DT.SB.03	-0.82	1.05	0.269	0.71 ± 0.40 1.03 ∓ 0.51 ± 0.263
DT.YB.04	1.06	0.88	0.124	0.91 ± 0.45 0.74 ∓ 0.40 ± 0.265
DT.SB.05	-1.72	2.11	0.867	0.88 ± 0.44 1.42 ∓ 0.71 ± 0.528
DT.YB.06	1.24	1.26	-0.049	1.17 ± 0.58 1.19 ∓ 0.59 ± 0.315
DT.ZC.01	0.48	-0.82	-0.136	-0.49 ± 0.40 -0.80 ∓ 0.40 ± 0.205
RWCU				
DT.YL.02	0.14	0.12	-0.038	0.36 ± 0.40 0.29 ∓ 0.40 ± 0.035

Parameter	Measured Value	Acceptance Criteria
Spring Hangers	Yes	Remained within working range

2STP-17.3, Measured Pipe Displacements (Main Steam Inside Drywell and Reactor Recirculation)

This subtest provides the means for collecting thermal expansion data on the Main Steam lines (inside the drywell) and Reactor Recirculation (RR) piping under specific conditions, verifying that the observed thermal expansion is consistent with that predicted by analysis. Data collection was accomplished using the Plant Monitoring System (PMS) and the specific system remote monitoring instrumentation (Lanyard Potentiometers and Resistance Temperature Devices, RTD's) installed on each Main Steam line and Recirculation loop.

Thermal expansion data was collected during Test Condition Open Vessel at ambient temperature conditions before the head was installed, during Test Condition Heatup at reactor water temperatures of $360 \pm 25^{\circ}F$, $450 \pm 25^{\circ}F$ (intermediate temperature), and $535 \pm 25^{\circ}F$ (rated temperature), and again at Test Condition 6 at rated feedwater temperature ($420 \pm 20^{\circ}F$).

Remotely monitored displacement and temperature instrumentation is installed on the MS Lines and on the RR Loops. The MS displacement instruments are located at the lower and upper elbow tap of each MS Line. The MS RTD's are located downstream of the MSIV on MS Line A and C. The RR displacement instruments are in four locations on each loop: 1) on the suction line near the nozzle, 2) beneath the RR Pump, 3) on the RR Pump Discharge line first upward elbow, and 4) the discharge header at the RHR tap. The RR RTD's are located on the RR Loop A and B suction lines and on RR Loop A only discharge line.

For these NSSS triaxial transducers, Level 1 limits are calculated for the existing pipe temperature and Level 2 limits apply only at rated conditions. All Level 1 limits were met for all the temperature plateaus. For TC-Heatup at rated reactor vessel conditions, 15 remotely monitored points fell outside of their Level 2 limits (10 on Main Steam and 6 on Reactor Recirculation). These test exceptions were documented and judged to be acceptable based on the measured values would not create thermal expansion stresses beyond the ASME III Code stress limits.

At Test Condition 6 during rated feedwater temperatures (420 degrees F), the thermal expansion data was obtained from remotely monitored instrumentation and the results yielded no Level 1 criteria violations. Twenty one (13 on Main Steam and 8 on Reactor Recirculation) of the remotely monitored points fell outside of their Level 2 limits. The resolution to the exceptions was that the test results were acceptable and satisfied the startup test specification requirements since the measured values would not create thermal expansion stresses beyond the ASME III Code stress limits.

The test results are summarized as follows:

Parameter	Measured Displacement (inch) 360°F	450°F	535°F	420°F	Acceptance Criteria Level 1 MIN MAX	Level 2 MIN MAX
RR Piping						
RA-SX	0.487				-0.895 2.120	N/A
		0.671			-0.700 2.315	N/A
			0.832	0.813	-0.539 2.476	0.875 1.062
RA-SY	0.059				-0.274 0.550	N/A
		0.068			-0.230 0.594	N/A
			0.069	0.042	-0.194 0.630	0.124 0.312
RA-SZ	-0.005				-0.135 0.199	N/A
		0.002			-0.125 0.209	N/A
			0.004	0.000	-0.117 0.217	-0.044 0.144
RA-PX	0.049				-0.[illegible]84 0.826	N/A
		0.046			-0.313 0.897	N/A
			0.041	0.103	-0.255 0.955	-0.256 0.444

Parameter	Measured Displacement (inch) 360°F	450°F	535°F	420°F	Acceptance Criteria Level 1 MIN MAX	Level 2 MIN MAX
RR Piping (cont'd)						
RA-PY	-1.232				-3.500	
					1.720	N/A
		-1.601			-3.500	
					1.324	N/A
			-1.917	-1.904	-3.500	-2.059
					0.997	-1.872
RA-PZ	0.011				-0.272	
					0.545	N/A
		0.011			-0.229	
					0.558	N/A
			0.006	-0.031	-0.193	0.122
					0.624	0.309
RA-DX	-0.208				-0.660	
					0.337	N/A
		-0.282			-0.710	
					0.286	N/A
			-0.343	-0.328	-0.776	-0.371
					0.221	-0.184
RA-DY	-0.789				-2.686	
					1.206	N/A
		-1.040			-2.916	
					0.976	N/A
			-1.250	-1.215	-3.215	-1.363
					0.677	-1.175
RA-DZ	-0.287				-0.925	
					0.451	N/A
		-0.365			-0.999	
					0.377	N/A
			-0.444	-0.475	-1.095	-0.501
					0.281	-0.313
RA-RX	-0.074				-0.267	
					0.168	N/A
		-0.107			-0.282	
					0.153	N/A
			-0.129	-0.189	-0.302	-0.178
					0.133	0.009

Parameter	Measured Displacement (inch) 360°F	450°F	535°F	420°F	Acceptance Criteria Level 1 MIN MAX	Level 2 MIN MAX
<u>RR Piping</u> (cont.)						
RA-RY	-0.221				-0.761	
					0.380	N/A
		-0.287			-0.820	
					0.371	N/A
			-0.347	-0.350	-0.897	-0.420
					0.244	-0.233
RA-RZ	-0.425				-1.544	
					0.716	N/A
		-0.568			-1.672	
					0.587	N/A
			-0.702	-0.660	-1.840	-0.804
					0.420	-0.616
RB-SX	-0.500				-2.305	
					1.106	N/A
		-0.709			-2.504	
					0.907	N/A
			-0.875	-0.865	-2.671	-1.059
					0.740	-0.872
RB-SY	0.036				-0.324	
					0.597	N/A
		0.044			-0.279	
					0.642	N/A
			0.047	0.028	-0.241	-0.126
					0.680	0.313
RB-SZ	-0.026				-0.198	
					0.141	N/A
		-0.049			-0.207	
					0.132	N/A
			-0.065	-0.058	-0.215	-0.139
					0.124	0.048
RB-PX	-0.095				-0.813	
					0.423	N/A
		-0.119			-0.678	
					0.358	N/A
			-0.136	-0.136	-0.932	-0.408
					0.304	-0.220

Parameter	Measured Displacement (inch) 360°F	450°F	535°F	420°F	Acceptance Criteria Level 1 MIN MAX	Level 2 MIN MAX
RR Piping (cont'd)						
RB-PY	-1.249				-3.500	
					2.055	N/A
		-1.631			-3.500	
					1.669	N/A
			-1.950	-1.944	-3.500	-1.965
					1.346	-1.778
RB-PZ	0.056				-0.113	
					0.137	N/A
		0.064			-0.109	
					0.141	N/A
			0.060	0.023	-0.106	-0.075
					0.144	0.113
RB-DX	0.247				-0.441	
					0.814	N/A
		0.321			-0.383	
					0.872	N/A
			0.380	0.369	-0.308	0.226
					0.948	0.414
RB-DY	-0.795				-2.921	
					1.457	N/A
		-1.067			-3.149	
					1.229	N/A
			-1.271	-1.248	-3.443	-1.350
					0.934	-1.162
RB-DZ	0.280				-0.601	
					1.145	N/A
		0.376			-0.516	
					1.230	N/A
			0.430	0.436	-0.406	0.373
					1.340	0.561
RB-RX	0.061				-0.262	
					0.384	N/A
		0.080			-0.243	
					0.402	N/A
			0.102	0.123	-0.163	0.010
					0.371	0.198

Parameter	Measured Displacement (inch) 360°F	450°F	535°F	420°F	Acceptance Criteria Level 1 MIN MAX	Level 2 MIN MAX
RR Piping (cont.)						
RB-RY	-0.202				-1.140	
					0.775	N/A
		-0.261			-1.196	
					0.719	N/A
			-0.314	-0.372	-0.928	-0.406
					0.303	0.219
RB-RZ	0.459				-0.896	
					1.758	N/A
		0.618			-0.762	
					1.892	N/A
			0.752	0.734	-0.588	0.645
					2.066	0.833
Main Steam Line						
SA-LX	-0.503				-1.883	
					0.737	N/A
		-0.676			-2.061	
					0.558	N/A
			-0.796	-0.763	-2.230	-1.014
					0.389	-0.827
SA-LY	0.019				-0.107	
					0.098	N/A
		0.029			-0.108	
					0.097	N/A
			0.035	0.073	-0.109	-0.100
					0.096	0.087
SA-LZ	-0.044				-0.129	
					0.190	N/A
		-0.055			-0.119	
					0.200	N/A
			-0.063	-0.062	-0.110	-0.044
					0.209	0.143

	Measured Displacement (inch)				Acceptance Criteria	
Parameter	360°F	450°F	535°F	420°F	Level 1	Level 2
					MIN	MIN
Main Steam Line (cont.)					MAX	MAX
SA-UX	-0.752				-2.802	
					1.067	N/A
		-1.011			-3.072	
					0.797	N/A
			-1.204	-1.171	-3.328	-1.487
					0.541	-1.300
SA-UY	0.020				-0.131	
					0.198	N/A
		0.046			-0.121	
					0.208	N/A
			0.065	0.050	-0.111	-0.040
					0.218	0.147
SA-UZ	0.304				-0.629	
					1.583	N/A
		0.419			-0.481	
					1.731	N/A
			0.502	0.552	-0.340	0.672
					1.872	0.860
SB-LX	-0.496				-2.585	
					1.359	N/A
		-0.670			-2.777	
					1.167	N/A
			-0.790	-0.752	-2.955	-1.077
					0.988	-0.889
SB-LY	-0.048				-0.140	
					0.117	N/A
		-0.040			-0.144	
					0.113	N/A
			-0.036	0.026	-0.147	-0.112
					0.110	0.075
SB-LZ	0.036				-0.162	
					0.228	N/A
		0.049			-0.152	
					0.238	N/A
			0.054	0.071	-0.142	-0.040
					0.248	0.147

Parameter	Measured Displacement (inch) 360°F	450°F	535°F	420°F	Acceptance Criteria Level 1 MIN MAX	Level 2 MIN MAX
Main Steam Line (cont.)						
SB-UX	-0.955				-3.500	
					2.480	N/A
		-1.267			-3.500	
					2.118	N/A
			-1.574	-1.532	-3.500	-1.949
					1.782	-1.761
SB-UY	-0.035				-0.187	
					0.279	N/A
		0.011			-0.173	
					0.293	N/A
			0.015	-0.107	-0.160	-0.021
					0.306	0.167
SB-UZ	0.557				-1.363	
					2.594	N/A
		0.724			-1.171	
					2.786	N/A
			0.894	0.920	-0.991	0.893
					2.965	1.080
SC-LX	-0.520				-1.461	
					-0.149	N/A
		-0.696			-1.665	
					-0.056	N/A
			-0.831	-0.791	-1.856	-1.145
					-0.247	-0.958
SC-LY	-0.058				-0.121	
					0.095	N/A
		-0.026			-0.125	
					0.091	N/A
			-0.024	0.005	-0.129	-0.115
					0.087	0.073
SC-LZ	-0.146				-0.324	
					0.104	N/A
		-0.189			-0.359	
					0.069	N/A
			-0.215	-0.215	-0.391	-0.271
					0.036	-0.083

Parameter	Measured Displacement (inch) 360°F	450°F	535°F	420°F	Acceptance Criteria Level 1 MIN MAX	Level 2 MIN MAX
Main Steam Line-(cont.)						
SC-UX	-1.023				-2.542	
					0.192	N/A
		-1.365			-2.910	
					-0.176	N/A
			-1.635	-1.620	-3.2[illegible]1	-1.978
					-0.517	-1.790
SC-UY	0.055				-0.103	
					0.305	N/A
		0.098			-0.071	
					0.337	N/A
			0.122	0.046	-0.041	0.068
					0.366	0.256
SC-UZ	-0.589				-1.542	
					0.152	N/A
		-0.805			-1.759	
					-0.065	N/A
			-0.921	-0.874	-1.961	-1.208
					-0.267	-1.020
SD-LX	-0.505				-2.720	
					1.575	N/A
		-0.692			-2.898	
					1.397	N/A
			-0.827	-0.793	-3.067	-1.013
					1.227	-0.826
SD-LY	0.001				-0.122	
					0.143	N/A
		0.004			-0.118	
					0.147	N/A
			-0.001	0.014	-0.115	-0.076
					0.150	0.111
SD-LZ	0.005				-0.240	
					0.176	N/A
		0.028			-0.250	
					0.166	N/A
			0.023	0.027	-0.259	-0.145
					0.157	0.043

Parameter	Measured Displacement (inch) 360°F	450°F	535°F	420°F	Acceptance Criteria Level 1	Level 2
					MIN	MIN
Main Steam Line (cont.)					MAX	MAX
SD-UX	-0.755				-3.500	
					2.299	N/A
		-1.044			-3.500	
					2.034	N/A
			-1.264	-1.225	-3.500	-1.464
					1.782	-1.276
SD-UY	0.033				-0.253	
					0.376	N/A
		0.050			-0.234	
					0.395	N/A
			0.057	0.055	-0.215	0.005
					0.413	0.192
SD-UZ	-0.505				-2.364	
					1.374	N/A
		-0.647			-2.518	
					1.220	N/A
			-0.770	-0.775	-2.664	-0.889
					1.074	-0.701

2STP-17.4, Visual Pipe Inspections (Main Steam Inside Drywell and Reactor Recirculation)

This subtest monitored the main steam inside drywell and recirculation piping systems by visual inspections of the piping, hangers and snubbers during Test Condition Open Vessel (baseline data), Test Condition Heatup (at 360 $\pm$ 25 deg F and 535 $\pm$ 25 deg F), and following two complete heatup cycles.

Visual inspections of the Recirculation and Main Steam piping and supports at TC Open Vessel showed no evidence of obstructions to normal system expansion. No cables were found stretched, no position indicators were out of their travel range, and no hangers were bottomed out. All snubbers and hangers were within their normal operating range. No hangers were found fully extended or compressed and no cables were found stretched. No restrictions to thermal expansion were noted.

The test results are summarized as follows:

Parameter	TC OV	360°F	535°F	After Two Cycles	Acceptance Criteria
	Measured Value				
Main Steam Line	No	No	No	No	No obstruction which interfere with the thermal expansion
RR Piping	No	No	No	No	
Pipe Supports	No	No	No	No	Not bottomed out nor fully extended
Pipe Support	Yes	Yes	Yes	Yes	Within operating range

4.16 2STP-18, TIP UNCERTAINTY

OBJECTIVES

The objective of this test is to determine the reproducibility of the Traversing Incore Probe (TIP) system readings.

ACCEPTANCE CRITERIA

Level 1

None

Level 2

The total TIP uncertainty (including random noise and geometrical uncertainties) obtained by averaging the uncertainties for all data shall be less than 6.0%.

RESULTS

2STP-18.1, TIP Uncertainty Determination

In this Subtest the total TIP uncertainty was calculated from TIP data taken during TC 3 and TC 6 when the TIP system was operated in conjunction with the Process Computer programs OD-1, OD-2, and OD-10. Calculation of the random noise and geometric TIP uncertainties was not required since the total TIP uncertainty in both test conditions was less than 6%.

The applicable Level 2 criterion was satisfied in both test conditions. The TIP uncertainties were 1.4% in TC 3 and 1.3% in TC 6, and the average of the total TIP uncertainties was 1.35%.

4.17 2STP-19, CORE PERFORMANCE

OBJECTIVES

The objectives of this test are to:

a) Evaluate the core thermal power and core flow rate; and

b) Evaluate whether the following core performance parameters are within limits:

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

ACCEPTANCE CRITERIA

Level 1

The Maximum Linear Heat Generation Rate (MLHGR) of any rod during steady-state conditions shall not exceed the limit specified by the Plant Technical Specifications (13.4 kw/ft).

The steady-state Minimum Critical Power Ratio (MCPR) shall exceed the minimum limit specified by the Plant Technical Specifications.

The Maximum Average Linear Heat Generation Rate (MAPLHGR) shall not exceed the limits specified by the Plant Technical Specifications.

Steady-state reactor power shall be limited to the rated core thermal power (3293 MWt).

Core flow shall not exceed 105% of its rated value (100 Mlb/hr).

Level 2

None

RESULTS

2STP-19.1, Core Performance - BUCLE Calculation

In Test Condition 2 the off-line computer program, Backup Core Limits Evaluation (BUCLE), was used to calculate the core thermal limit parameters MLHGR, MCPR, and MAPLHGR. All acceptance criteria were satisfied.

The reactor core thermal power and core flow rate during the test were 482 MWt and 42 Mlb/hr, respectively. These were less than the Level 1 criterion limits of 3293 MWt and 105 Mlb/hr.

The values of MFLPD (MLHGR/MLHGR Limit), MFLCPR (MCPR Limit/MCPR), and MAPRAT (MAPLHGR/MAPLHGR Limit) were calculated to be 0.166, 0.235, and 0.157, respectively, using the off-line computer program BUCLE. Since all of these thermal limit parameter ratios were less than 1.0, the Level 1 acceptance criteria were satisfied.

2STP-19.2, Process Computer Calculation

This subtest was performed at 39%, 50%, 41%, 59%, 97%, and 100% core thermal power (CTP) during Test Conditions 2, 3, 4, 5, 6, and Warranty Run, respectively. The purpose of this subtest is to verify the process computer calculation of thermal limits using core performance parameters and heat balance data. All acceptance criteria were satisfied as shown below:

	Test Condition						
	2	3	4	5	6	WR	Acceptance Criteria
CTP(%)	38.8	50.5	41.0	58.8	97.0	99.7	< 100
Core Flow(%)	45.27	56.3	38.5	48.3	95.3	99.9	≤ 105
MFLPD	0.420	0.472	0.385	0.560	0.666	0.924	≤ 1.00
MFCP	0.526	0.608	0.595	0.778	0.875	0.887	≤ 1.00
MAPR	0.386	0.435	0.353	0.554	0.845	0.906	≤ 1.00

4.18 2STP-20, STEAM PRODUCTION

OBJECTIVES -

The objectives of this test are to demonstrate that the Nuclear Steam Supply System (NSSS) can provide steam sufficient to satisfy all appropriate warranties as defined in the NSSS contract.

ACCEPTANCE CRITERIA

Level 1

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

The NSSS shall be capable of supplying 14,159,000 pounds per hour of steam of not less than 99.7% quality at a pressure of 985 psia at the discharge of the second main steam isolation valve, as based upon a final reactor feedwater temperature of 420 degrees F and a control rod drive feed flow of 32,000 pounds per hour at 80 degrees F. The reactor feedwater flow must equal the steam flow less the control rod drive feed flow.

Level 2

None

RESULTS

2STP-20.1, Two Hour Demonstration

This subtest was performed in conjunction with STP-20.2, 100 Hour Demonstration, at the beginning (0 hr.), middle (50 hr.) and end (98 hr.) of that demonstration. In each case, data was taken at ten minute intervals for two hours then averaged. The averaged data was used in heat balance calculations to determine core thermal power. Steam moisture content was determined by the carryover from the reactor and steam line pressure drop. NSSS steam production performance was evaluated by adjusting the warranted steam flow for actual plant operating conditions and comparing it to actual steam flow.

Hand calculations of core thermal power showed that the process computer heat balance calculation was approximately 35 MWt higher. The discrepancy in the hand calculations was traced to improper converting of temporary feedwater flow instrument signals from VDC to inch water dp, and an error was also discovered in one of the terms used to calculate feedwater flow loop A. After these corrections have been made, the difference in core thermal power was reduced to approximately 15 MWt, within the tolerance allowed by feedwater flow measuring instruments.

Results of the two hour runs are summarized below. All criteria were satisfied.

	Steam Quality	Calculated Steam Flow (Mlb/hr)	Calculated Core Thermal Power (MWt)
RUN-1	99.81%	14.043 (99.18%)	3269.51 (99.29%)
RUN-2	99.80%	14.034 (99.12%)	3260.92 (99.21%)
RUN-3	99.80%	14.054 (99.26%)	3271.41 (99.34%)

2STP-20.2, 100 Hour Demonstration

This subtest consists of operating the reactor at or near rated core thermal power for a 100 hour period. The OD-3 program was performed every hour to verify thermal limits and rated core thermal power were not exceeded during the demonstration.

The most limiting thermal limit values recorded by the process computer during the 100 hour demonstration were:

Parameter	Acceptance Criteria	Process Computer Value
(CMAPR) MAPRAT	$\leq$ 1.0	.905
(CMFLCP) MFLCPR	$\leq$ 1.0	.893
(CMFLPD) MFLPD	$\leq$ 1.0	.939
(PCTPWR X0.01) FRTP	$\leq$ 1.0	.998

Six of the 100 OD-3 edits showed the core thermal power exceeding the rated power level of 3293 MWt due to the use of instantaneous data. The maximum deviation was 6.2 MWt. This exception was accepted since the plant was operated within the guidelines of GP-5, "Power operation", which requires operator to achieve a shift average of 3293 MWt with small excursions permitted.

4.20 2STP-22, PRESSURE REGULATOR

OBJECTIVES

The objectives of this test are as follows:

To demonstrate optimized controller settings for the pressure control loop by analysis of the transients induced in the reactor pressure control system by means of set point changes to the pressure regulators.

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

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

ACCEPTANCE CRITERIA

Level 1

The transient response of any pressure control system related variable to any test input must not diverge.

Level 2

Pressure control system related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25. (This criterion does not apply to tests involving simulated failure of one regulator with the backup regulator taking over.)

The pressure response time from initiation of pressure setpoint change to the turbine inlet pressure peak shall be $\leq$10 seconds.

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

The peak neutron flux and/or peak vessel pressure shall remain below the scram settings by 7.5 percent and 10 psi respectively for all pressure regulator transients performed at Test Condition 6.

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

Steam Flow Region	Variation
0 to 85%	≤4:1
85% to 97%	≤2:1
97% to 99%	≤5:1

RESULTS

2STP-22.1, Pressure Regulator Response - Control Valve Operation

This subtest consists of performing approximately 10 psi negative and 10 psi positive step changes in Pressure Regulator setpoint and demonstrating the backup capability of the Standby Pressure Regulator upon simulated failure of the Controlling Pressure Regulator. During the transients caused by pressure regulator setpoint step changes and failure demonstrations, the Turbine Load Limit and Load Set were set high to allow only control valves to control reactor pressure. The response of reactor and regulator variables following the step changes was recorded on the Plant Monitoring System (PMS) equipment. The step changes were performed with Pressure Regulator A in control and again with Pressure Regulator B in control. The Pressure Regulator gain, lead and lag settings were set during the tuneup of the pressure regulator system prior to performing the subtest. Following the step change test of each Pressure Regulator, the simulated failure was initiated from the Reactor Pressure Test cards to demonstrate the capability of the Standby Pressure Regulator in a backup capacity to take over without approaching scram setpoints.

This subtest was performed at Test Condition 2, 3, 5 and 6 with the exception that the "failure to backup regulator" testing was not performed at Test Condition 5.

All Level 1 acceptance criteria were satisfied. All Level 2 acceptance criteria except the variation in incremental regulation were satisfied.

The test results are summarized as follows:

Parameter	Measured Value TC 2	TC 3	TC 5	TC 6	Acceptance Criteria
Transient Response	No	No	No	No	Not Divergent
Decay Ratio	0-0.25	0-0.25	0-0.25	0	≤ 0.25
Response Time to Pressure Peak	3.6 sec	3.1 sec	3.6 sec	3.4 sec	≤ 10 sec
Limit Cycle Oscillations	0%	0%	0%	0%	≤ 0.5%
Margin to Flux Scram* for Step Change	N/A	N/A	16.8%	10%	> 7.5%
Margin to Flux Scram* for Failure Test	N/A	N/A	N/A	10.9%	> 7.5%
Margin to Pressure* Scram for Step Change	N/A	N/A	83 psi	47.8 psi	> 10 psi
Margin to Pressure* Scram for Failure Test	N/A	N/A	N/A	41 psi	> 10 psi

* Applicable only for TC 6

The variation in incremental regulation was calculated as follows:

Steam Flow Region	Incremental Regulation Maximum	Minimum	Variation	Limit
0-85%	118.52	67.98	1.743	≤4:1
85%-97%	116.57	56.16	2.076	≤2:1
85%-99%	This range not reached during testing			≤5:1

This exception was evaluated and found to be acceptable. The 2.076 value was a small deviation from the limit and was calculated by a conservative method. All dynamic system testing showed excellent system response to perturbations during the performance of 2STP-22. The 97% - 99% range was not reached during testing due to overall plant performance. 100% rated power was reached with lower steam flows.

2STP-22.2 Pressure Regulator Response - Control Valve and Bypass Valve Operation

This subtest consists of performing approximately 10 psi negative and 10 psi positive step changes in Pressure Regulator setpoint. During the transients caused by pressure regulator setpoint changes, the Turbine EHC Load Set was at a point where bypass valve action was incipient, and the Load Limit was set above the Load Set. Both control and bypass valves were operated to control reactor pressure. The response of reactor and regulator variables following the step changes was recorded on the Plant Monitoring System (PMS) equipment. The step changes were performed with Pressure Regulator A in control and with Pressure Regulator B in control. The Pressure Regulator gain, lead and lag settings were set during the tuneup of the pressure regulator system prior to performing the subtest.

This subtest was performed at Test Condition 3.

All Level 1 and Level 2 acceptance criteria were satisfied.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
Transient Response	No	Not Divergent
Decay Ratio	0	$\leq$ 0.25
Response Time to Pressure Peak	3.5 sec	$\leq$ 10 sec
Limit Cycle Oscillations	0	$\leq$ 0.5%

2STP-22.3 Pressure Regulator Response - Bypass Valve Operation

This subtest consists of performing approximately 10 psi negative and 10 psi positive step changes in Pressure Regulator setpoint and demonstrating the backup capability of the Standby Pressure Regulator upon the simulated failure of the Controlling Pressure Regulator. During the transients caused by pressure regulator setpoint step changes and failure demonstrations, the Turbine EHC Load Set was at a point where the bypass valves were initially passing about 10% of rated steam flow and the Load Limit was set above the Load Set. The response of reactor and regulator variables following the step changes was recorded on the Plant Monitoring System (PMS) equipment. The step changes were performed with Pressure Regulator A in control and again with Pressure Regulator B in control. The Pressure Regulator gain, lead and lag settings were set during the tune-up of the pressure regulator system prior to performing the subtest. Following the step change test of each Pressure Regulator, the simulated failure was initiated to demonstrate the capability of the Standby Pressure Regulator in a backup capacity to take over without approaching scram setpoints.

This subtest was performed at Test Condition 1, 2, 5 and 6 except that the Pressure Regulator failure portions of this subtest was not performed at Test Condition 5.

All Level 1 and Level 2 acceptance criteria except the decay ratio less than or equal to 0.25 were satisfied.

At Test Condition 1, one pressure transmitter output had a decay ratio of 0.29. This was accepted due to the fact that overall system response was excellent and three other pressure transmitter signals met the criteria.

The test results are summarized as follows:

Parameter	Measured Value TC 1	TC 2	TC 5	TC 6	Acceptance Criteria
Transient Response	No	No	No	No	Not Divergent
Decay Ratio	0-0.29	0-0.23	0-0.25	0-0.18	$\leq$ 0.25
Response Time to Pressure Peak	4.5 sec	4.2 sec	3.6 sec	4.2 sec	$\leq$ 10 sec
Limit Cycle Oscillations	0%	0%	0%	0%	$\leq$ 0.5%
Margin to Flux Scram* for Step Change	N/A	N/A	13.7%	10.1%	> 7.5%
Margin to Flux Scram* for Failure Test	N/A	N/A	N/A	9.6%	> 7.5%
Margin to Pressure* Scram for Step Change	N/A	N/A	82.5 psi	42.8 psi	> 10 psi
Margin to Pressure* Scram for Failure Test	N/A	N/A	N/A	38.4 psi	> 10 psi

* Applicable only for TC 6

4.21 2STP-23, FEEDWATER SYSTEM

OBJECTIVES

The objectives of this test are:

To demonstrate that the feedwater system has been adjusted to provide acceptable reactor water level control.

To demonstrate an adequate response to a feedwater temperature reduction.

To demonstrate the capability of the automatic core flow runback feature to prevent low water level scram following the trip of one feedwater pump at high power operation.

To demonstrate that the maximum feedwater runout capability is compatible with the licensing assumptions.

ACCEPTANCE CRITERIA

Level 1

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

For the feedwater heater loss test, the maximum feedwater temperature decrease due to a single failure case must be ≤100 degrees F. The resultant MCPR must be greater than the fuel thermal safety limit.

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

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

127% NBR at 1020 psig

Level 2

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

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

a.	Maximum time to 10% of a step disturbance	≤1.2 sec
b.	Maximum time for 10% to 90% of a step disturbance	≤2.5 sec
c.	Peak overshoot (% of step disturbance)	≤15%
d.	Settling time, 100% ±5%	≤14 sec

The average rate of response of the feedwater actuator to large (>20% of pump flow) step disturbances shall be between 5 percent and 25 percent rated feedwater flow/second. This average response rate will be assessed by determining the time required to pass linearly through the 10 percent and 90 percent response points.

At steady-state generation for the 3/1 element systems, the input scaling to the mismatch gain should be adjusted such that the level error due to biased mismatch gain output should be within ±1 inch when operating within Test Condition 6.

The increase in simulated heat flux cannot exceed the predicted value referenced to the actual feedwater temperature change and initial power level.

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

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

a. With rated complement of pumps - 115.5% NBR at 1075 psia

b. One feedwater pump tripped conditions - 80% NBR at 1025 psia.

RESULTS

2STP-23.1, FW System Startup Controller Level Step

This subtest observed Feedwater Control System response to step changes while operating in the automatic mode on the Startup Controller. This subtest was performed in Test Condition 1 and in Test Condition 3 at Test Condition 1 power levels. The first performance satisfied all Level 1 and Level 2 acceptance criteria. The second performance, to demonstrate the stability of a new level controller, satisfied all Level 1 and Level 2 acceptance criteria.

The test results are summarized as follows:

Parameter	Measured Value TC 1	TC 3	Acceptance Criteria
Transient Response	No	No	Not Divergent
Decay Ratio	0	0	$\leq$ 0.25

2STP-23.2, Feedwater System Manual Flow Step

This subtest was performed in TC 2, TC 3, and TC 6 for all three Turbine Driven Reactor Feed Pumps (TDRFP) speed controllers. In this subtest, positive and negative speed (flow) steps were initiated using a step generator. The TDRFP response was slow in TC 2 for delay time (max. Time from 0% to 10%) and rise time (max. Time from 10% to 90%). After review, the system was deemed adequately tuned to support TC 3 testing. In TC 3, the delay time was still slow. The Feedwater System response was viewed satisfactory and would support TC 6 testing. TC 6 delay time for all three TDRFP's and the rise time for the "C" TDRFP did not meet the criteria, but overall system response was evaluated and determined to be acceptable based on other startup test results.

The test results are summarized as follows:

Parameter	Measured Value TC 2	TC 3	TC 6	Acceptance Criteria
Transient Response	No	No	No	Not Divergen
Decay Ratio	0	0-0.16	0	$\leq$ 0.25
Max. Time from 0% to 10%	1.4-2.4 sec	1.6-1.7 sec	1.4-1.7 sec	$\leq$ 1.2 sec
Max. Time from 10% to 90%	3.4-4.2 sec	1.9-2.3 sec	1.6-2.7 sec	$\leq$ 2.5 sec
Peak Overshoot	0%	2.0-7.0%	0%	$\leq$ 15%
Settling Time	7.5-10.6 sec	5.9-8.9%	3.5-9.1%	$\leq$ 14 sec
Rate of Response to Large Step	*12.5-13.9%/sec	N/A	N/A	5-25%/sec

* Use 2P-45.1, Feedwater System preoperational test results

2STP-23.3 Feedwater System Level Setpoint Changes

This subtest was performed in TC 2, TC 3, TC 4, TC 5, and TC 6. In this subtest, the Feedwater Level Control (Master Feedwater Controller) was-used to perform negative and positive level demand step changes for both single and three element control. All Level 1 and Level 2 criteria was satisfied in all Test Conditions. Level error, seen when switching between single and three element in TC 6, was minimal and within tolerance.

The test results are summarized as follows:

	Measured Value					
Parameter	TC 2	TC 3	TC 4	TC 5	TC 6	Acceptance Criteria
Transient Response	No	No	No	No	No	Not Divergent
Decay Ratio	0-0.25	0-0.22	0	0-0.18	0	$\leq$ 0.25
Level Change * following 3/1 Element Switch-over	1.3"	0.1"	0.3"	1.7"	0.5"	< 1"

* Applicable only for TC 6

2STP-23.4, Loss of Feedwater Heating

This subtest was performed in TC 6. In this subtest, adequate response to a feedwater temperature reduction was demonstrated. The failure was simulated by isolating extraction steam to the sixth stage feedwater heaters. All of the acceptance criteria were satisfied in this subtest.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
Feedwater Temperature Decrease	52.9°F	$\leq$ 100°F
MCPR	1.54	> 1.06
Heat Flux Increase - Level 1	107.85%	< 111.15%
Heat Flux Increase - Level 2	107.85%	< 109.15%

2STP-23.5, Feedwater Pump Trip

This subtest demonstrated the capability of the automatic recirculation pump runback feature to prevent a low water level scram following a trip of one reactor feedpump. This subtest also demonstrated the RFPT speed controllers' ability to prevent high and low water level trips as discussed under 2STP-23.2, Feedwater System Manual Flow Steps. The subtest was performed in TC 6 by manually tripping the "C" Turbine Driven Reactor Feed Pump (TDRFP). The acceptance criterion required that the reactor avoid a low water level scram by three inches from an initial water level halfway between the high and low level alarm setpoints. This criterion was satisfied.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
Margin to Low Water Level Scram	13.4"	> 3"

2STP-23.6, Feedwater - RFPT Data

This subtest was deleted for Unit 2. The content of this subtest was incorporated into 2STP-23.7, Maximum Feedwater Runout Capacity.

2STP-23.7, Maximum Feedwater Runout Capability

This subtest consists of determining if the maximum feedwater runout capability is compatible with the licensing assumptions by verifying that maximum feedwater flows do not exceed the flows specified in the FSAR.

During this subtest, each feedwater pump was separately operated in the manual speed control mode in parallel with the other two pumps operating in the automatic speed control mode. While in Manual, the speed of the feedwater pump being tested was gradually increased to maximum controller output and associated flows and pressures recorded. The two pumps in AUTO decreased flows accordingly with no change in total feedwater flow or reactor water level. This process was repeated for each feedwater pump to determine the individual pump runout speed. Finally, the individual flows were corrected to the FSAR pressures and compared with the FSAR maximum flows.

The subtest was performed in TC 6. The Level 2 criteria were satisfied. The Level 1 criteria was exceeded when the final results indicated a maximum FW runout of 128%. A feedwater runout sensitivity study was conducted which concluded that an increase to 135.4% NBR (Nuclear Boiler Rated) is acceptable for all licensed operating domains. FSAR Sections 15.0, 15.1, Table 15.0-1, Table 15.0-5, and Table 15.1-3 were reviewed in making the determination. A change has been made to revise the feedwater design spec data sheet, and an LDCN is in progress to update Chapter 15 to reflect the 135.4% flow.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
Maximum Feedwater Flow with 3 Pumps Running	128% NBR at 1020 psig	< 127% NBR at 1020 psig
Maximum Feedwater Flow with 3 Pumps Running	123% NBR at 1075 psia	> 115.5% NBR at 1075 ps
Maximum Feedwater Flow with 2 Pumps Running	108% NBR at 1025 psia	> 80% NBR at 1025 psia

4.22 2STP-24, TURBINE VALVE SURVEILLANCE

OBJECTIVES

The objectives of this test are to demonstrate acceptable procedures and maximum power levels for periodic surveillance testing of the main turbine control, stop and bypass valves without producing a reactor scram.

ACCEPTANCE CRITERIA

Level 1

None

Level 2

Peak neutron flux must be at least 7.5% below the scram trip setting.

Peak vessel pressure must remain at least 10 psi below the high pressure scram setting.

Peak steam flow in each line must remain 10% below the high flow isolation trip setting.

RESULTS

2STP-24.1, Stop Valve Testing

The Stop Valve Testing was performed in Test Condition 3 and 6. In this subtest, each Main Turbine Stop Valve (MSV) was stroked from full open to full closed and then returned to full open position to verify a 7.5% peak neutron trip margin, a peak vessel pressure margin of 10 psi below the trip setpoint, and a peak steam flow of 10% below the high flow isolation setting. This was accomplished using the test pushbuttons on the EHC Turbine Control Panel. All acceptance criteria were satisfied at 95.4% core thermal power.

This subtest was performed initially in Test Condition 3 at 56% reactor power and then repeated twice during the ascension to Test Condition 6 at 71% and 77% reactor power. Extrapolation of the results of stop valve testing in Test Condition 3 and 6 shows that periodic surveillance testing can be performed at 100% core thermal power without violating Level 2 acceptance criteria. The last subtest was performed in Test Condition 6 at 96% reactor power.

The test results are summarized as follows:

Parameter	Measured Value TC 3	TC 6 (71%)	TC 6 (77%)	TC 6 (96%)	Acceptance Criteria
Margin to Flux Scram	31.5%	19.5%	1[illegible].8%	16.4%	> 7.5%
Margin to Pressure Scram	87.3 psi	74.2 psi	60.3 psi	34.6 psi	> 10 psi
Margin to Steam Flow Isolation	77.5%	51.3%	40.0%	28.4%	> 10%

2STP-24.2, Control Valve Testing

The Control Valve Testing was performed in Test Condition 3 and 6. This subtest called for individual cycling of each Main Turbine Control Valve (CV) from its initial position to fully closed and then returning to its initial position. Reactor pressure is maintained by the automatic opening of the other CVs or Bypass Valves as demanded by the pressure regulator.

This subtest was performed initially in Test Condition 3 at 56% reactor power and then repeated twice during the ascension to Test Condition 6 at 85% and 90% reactor power. Extrapolation of the results of control valve testing in Test Condition 3 and 6 shows that periodic surveillance testing can be performed at 100% core thermal power without violating Level 2 acceptance criteria.

The Test Condition 6 subtest was initially attempted at 76% reactor thermal power. A much larger than expected flux spike (approximately 13%) was observed during stroking of the first control valve to be tested (CV #1) and the subtest was aborted. This response was subsequently determined to be in accordance with EHC system design.

At 76% power, when the #1 control valve is closed for testing, the #2 and #3 valves go from 35% to 95% open and the #4 control valve goes from full closed to 25% open. When the #1 control valve is reopened, the #2, #3, and #4 control valves start to close. Reactor pressure is controlled until the #4 control valve is closed and then drops about 20 psi because the #2 and #3 control valves have to travel from 95% to 50% open before they have an appreciable effect on steam flow.

When turbine control valve testing is done at a higher power, the #4 control valve will maintain reactor pressure for a longer portion of the transient and the #2 and #3 control valves will resume pressure control sooner than they do at 76% power.

The test results are summarized as follows:

Parameter	Measured Value TC 3	TC 6 (85%)	TC (90%)	Acceptance Criteria
Margin to Flux Scram	33.0%	11.6%	15.1%	> 7.5%
Margin to Pressure Scram	83.0 psi	63.1 psi	45.7 psi	> 10 psi
Margin to Steam Flow Isolation	82.3%	49.6%	44.8%	> 10%

2STP-24.3, Bypass Valve Testing

The Bypass Valve Testing was performed in Test Condition 6. In this test, each bypass valve was stroked from full closed to full open and then returned to full closed position. This was accomplished using the selector switch and test pushbutton on the EHC Turbine Control Panel. All acceptance criteria were satisfied at 94.6% core thermal power.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
Margin to Flux Scram	19.2%	> 7.5%
Margin to Pressure Scram	45.1 psi	> 10 psi
Margin to Steam Flow Isolation	42.2%	> 10%

4.23 2STP-25, MAIN STEAM ISOLATION VALVES

OBJECTIVES -

The objectives of this test are to functionally check the Main Steam Isolation Valves (MSIV's) for proper operation at selected power levels, to determine the MSIV closure times, and to determine the maximum power level at which full closure of a single MSIV can be performed without causing a reactor scram.

The full isolation is performed to determine the reactor transient behavior that results from the simultaneous full closure of all MSIV's at a high power level.

ACCEPTANCE CRITERIA

Level 1

MSIV stroke time shall be no faster than 3.0 seconds. MSIV closure time shall be no slower than 5.0 seconds.

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

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

Reactor must scram to limit the severity of the neutron flux and simulated heat flux transients.

Level 2

The reactor shall not scram. The peak neutron flux must be at least 7.5 percent below the trip setting. The peak vessel pressure must remain at least 10 psi below the high pressure scram setting.

The reactor shall not isolate. The peak steam flow on each line must remain 10 percent below the high steam flow isolation trip setting.

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

The positive change in vessel dome pressure and simulated heat flux occurring within the first 30 seconds after the closure of all MSIV valves must not exceed the predicted values. Predicted values will be referenced to actual test conditions of initial power level and dome pressure and will use beginning of life nuclear data.

If water level reaches the reactor vessel low water level (Level 2) setpoint, RCIC and HPCI shall automatically initiate and reach rated system flow.

Recirculation pump trip shall be initiated if water Level 2 is reached.

RESULTS

2STP-25.1, MSIV Functional Test

This subtest was performed during Test Condition Heatup and Test Condition 1. Although the fastest MSIV tested in Test Condition 1 was within the acceptance criterion, the MSIV was adjusted to slow the closure time during an outage following Test Condition 2. This was to ensure the closure time remained within requirements at higher power levels when the closure time would be faster.

The test results are summarized as follows:

	Measured Value		
Parameter	TC HU	TC 1	Acceptance Criteria
MSIV Stroke Time	NA	3.2-4.0 sec	3-5 sec
Margin to Flux Scram	10.56%	23.7%	> 7.5%
Margin to Pressure Scram	110.3 psi	103 psi	> 10 psi
Margin to Steam Flow Isolation	125.8%	115.1%	> 10%

All applicable acceptance criteria were satisfied.

2STP-25.2, Full Closure of Fastest MSIV

This subtest was performed during Test Condition 3. A subsequent subtest that was to have been conducted in Test Condition 5 was not performed. Two tests were initially required to provide enough data points to estimate the maximum power at which the MSIV could be closed without scramming the reactor. Limerick does not conduct testing which fully closes an MSIV while at power, therefore this extrapolation is not required. No estimation of maximum power was made.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
MSIV Stroke Time	3.6-4.1 sec	3-5 sec
Margin to Flux Scram	35.9 %	> 7.5%
Margin to Pressure Scram	63 psi	> 10 psi
Margin to Steam Flow Isolation	48.4%	>10%

2STP-25.3, Full MSIV Closure

The purpose of this subtest is to demonstrate the reactor's transient behavior to a full closure of all MSIVs near 100% power.

The MSIV closure times were all greater than or equal to 3.0 seconds and less than or equal to 5.0 seconds. The positive change in vessel dome pressure occurring within 30 seconds after closure of all MSIVs was less than the Level 2 criteria by >75 psi. The positive change in simulated heat flux did not exceed the Level 2 criteria. The main steam lines did not become flooded and the reactor scrammed, as required.

Water level monitored by PMS reached a level of -44.5 inches and was less than the Level 2 setpoint for 6 seconds. Both Reactor Recirc Pump ATWS channels tripped, HPCI received an initiation signal, and RCIC received an initiation signal. HPCI and RCIC were manually secured by the operator prior to reaching rated flows due to level swell caused by recirculation pump trip. This exception was accepted with no further action required since the performance of HPCI and RCIC systems had been proven in previous tests.

During the initial pressure transient, no SRV's opened; however, later pressurization resulted in eight SRV's opening, and one additional SRV (A) had a tailpipe temperature increase of only 43 degrees F. For three SRV's, the tailpipe temperature did not return to the initial temperature for approximately 11 hours, the remaining SRV temperatures returned to initial temperature in approximately 3 hours. The plant was depressurized to approximately 277 psi prior to all temperatures returning to the initial temperature. This exception was accepted with no further action required since the acoustic monitors and other plant parameters indicated that all SRV's were closed.

Several SRV's with lift setpoints of 1150 psi opened although the maximum pressure observed was 1132 psi. This exception was accepted and no further actions were required.

Since the SRV's did not open within 30 seconds, the actual reactor pressure rise, rather than predicted reactor pressure rise was used to determine maximum vessel dome pressure changes as the SRV's had no effect on the actual reactor pressure rise within 30 seconds.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
MSIV Stroke Time	3.4-4.2 sec	3-5 sec
Vessel Dome Pressure Increase	62.2 psi	173.5 psi
Steam Line	No (32")	Not flooded (< 118")
Reactor Scram	Yes	Initiated
SRV Tailpipe Temperature	0-10 deg F	Returned to within 10 deg F of initial temperature
Vessel Dome Pressure Increase	62.2 psi	≤148.5 psi
HPCI	Yes	Initiated if RPV Level 2 is reached
RCIC	Yes	Initiated if RPV Level 2 is reached
Recirc Pumps	Yes	Tripped if RPV Level 2 is reached

4.24 2STP-26, RELIEF VALVES

OBJECTIVES

The objectives of this test are a) to verify that the Relief Valves function properly (can be manually opened and closed), b) to verify that the Relief Valves reset properly after actuation, c) to verify that there are no major blockages in the Relief Valve discharge piping, and d) to demonstrate system stability to Relief Valve operation.

ACCEPTANCE CRITERIA

Level 1

There should be a positive indication of steam discharge during the manual actuation of each Relief Valve.

The flow through each Relief Valve shall compare favorably with value assumed in the FSAR accident analysis at normal operating Reactor pressure.

Level 2

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

The temperature measured by the thermocouples on the discharge side of the valves shall return to within 10 Deg. F of the temperature recorded before the valve was opened.

During the rated pressure functional test, the steam flow through each Relief Valve, as measured by Generator Gross MWe, shall not be lower than the average valve response by more than 0.5% of rated MWe.

RESULTS

2STP-26.1, ADS Valve Low Pressure Test

During Test Condition Heatup with reactor pressure at 286 psig, each ADS Valve was manually cycled to verify proper operation. Each valve was maintained open for approximately 10 seconds to allow system variables to stabilize.

Positive indication of Relief Valve discharge was verified by review of transient plots of Bypass Valve position.

All applicable acceptance criteria were satisfied with the following exception: Relief Valve E failed to meet the Level 2 criterion for discharge side temperatures to return to within 10 Deg. F of the initial temperature. Valve position, as indicated by the Acoustic Monitoring System and other plant parameters, indicated that Relief Valve E was shut.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
Steam Through Each ADS Valve	Yes	Yes
SRV Tailpipe Temperature	2-99 deg F	Returned to within 10 deg F of initial temperature

2STP-26.2, Relief Valve Rated Pressure Test

This subtest was performed during Test Condition 2. Each relief valve was manually cycled and maintained open for approximately 10 seconds. This was later determined to be inadequate to allow parameters to stabilize; however, the data collected was adequate to satisfy the intent of this test. This was based upon all pressure control related variables demonstrating satisfactory damping following SRV closure and that by using peak values for gross MWe and steam flow rather than steady state values, the results were more conservative. Positive indication of Relief Valve discharge was verified by the change in gross generator output (MWE).

All relief valves actuated and flow through each valve compared favorably with the value assumed in the FSAR accident analysis at normal operating pressure satisfying the Level 1 criteria.

All Level 2 criteria were satisfied with the following exceptions: 1) Relief Valves A and H did not meet the criterion for discharge side temperatures returning to within 10 Deg. F of initial temperature. 2) The thermocouple for Relief Valve L was inoperable so temperature data collected was inaccurate. Acoustic Monitoring System for Relief Valves A, H, L, and other plant parameters indicated that the relief valves were shut. The test results were evaluated and determined to be acceptable.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
Steam Flow Through Each SRV	6.85-6.99% NBR	$\leq$ 7% NBR
Average Steam Flow Through Each SRV	Yes	Within average by < 0.5%
Decay Ratio	0-0.25	$\leq$ 0.25
SRV Tailpipe Temperature	0-19 deg F	Returned to within 10 deg F of initial temperature

4.25 2STP-27, MAIN TURBINE TRIP

OBJECTIVES

The objectives of this test are to demonstrate the response of the reactor and its control systems to protective trips of the Main Turbine and to evaluate the response of the bypass and safety/relief valves.

ACCEPTANCE CRITERIA

Level 1

For turbine and generator trips at power levels greater than 50% nuclear boiler rated, there should be a delay of less than 0.1 seconds following the beginning of control or stop valve closure before the beginning of bypass valve opening. The bypass valves should be opened to a point corresponding to greater than or equal to 80% of their capacity within 0.3 seconds from the beginning of control or stop valve closure motion.

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

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

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

The recirculation pump and motor time constants for the two-pump drive flow coastdown transient should be ≤4.5 seconds from 1/4 to 2 seconds after the pumps are tripped.

The total time delay from the start of the turbine stop valve or control valve motion to the complete suppression of the electrical arc between the fully open contacts of the RPT circuit breakers shall be less than or equal to 175 milliseconds.

Level 2

There shall be no MSIV closure during the first three minutes of the transient and operator action shall not be required during that period to avoid the MSIV closure.

The positive change in vessel dome pressure occurring within the first 30 seconds after the initiation of either generator or turbine trip must not exceed predicted values.

The positive change in simulated heat flux occurring within the first 30 seconds after the initiation of either Generator or turbine trip must not exceed predicted values.

Feedwater level control shall avoid loss of feedwater flow due to a high (L8) water level trip during the event.

Low (L2) water level recirculation pump trip, HPCI and RCIC shall not be initiated.

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

For the turbine trip within the bypass valves capacity, the reactor shall not scram.

The measured bypass valve capability shall be equal to or greater than that used in the FSAR analysis (25% of nuclear boiler rated steam flow).

RESULTS

2STP-27.1, Turbine Trip Within Bypass Valve Capacity

This subtest was performed at 21% core thermal power during Test Condition 2. The main turbine was tripped manually by depressing the Turbine Trip pushbutton which shut the main turbine stop and control valves. The bypass valves opened to maintain pressure control and the reactor did not scram, thus satisfying the single Level 2 acceptance criterion.

2STP-27.2, Bypass Valve Capacity Check

This subtest was performed between 30% and 42% core thermal power during Test Condition 2. The main turbine bypass valves were gradually opened by increasing reactor power while maintaining generator output constant. The increase in reactor power was determined for each 25% increase in individual bypass valve opening. This was repeated until four bypass valves were fully opened. The change in reactor power was then plotted versus bypass valve position. The capacity of the bypass valves was determined from this graph. Total bypass valve capacity was calculated to be 31.1% of rated core thermal power, thus satisfying the acceptance criterion that measured bypass valve capability be greater than or equal to that used in the FSAR analysis (25% of nuclear boiler rated steam flow).

2STP-27.3, Turbine Trip at Test Condition 3

This subtest was deleted for Unit 2 per simplified startup test program.

2STP-27.4, Turbine Trip at Test Condition 6

An inadvertent turbine load reject which occurred at 99.6% rated thermal power during Test Condition 6 was used to verify the requirements of this test. The initial plant conditions that the turbine trip test requires were met. The nature of the transient was similar to the test scenario except that the operator took early action to trip feedwater pumps.

Level 1 acceptance criteria were satisfied with the exception of verifying that feedwater system settings would prevent flooding of the steam lines. This acceptance criteria could not be verified because the operator tripped the feedpumps before reactor level had peaked during the transient. An engineering evaluation determined that the feedwater system settings would have prevented flooding of the main steamlines based on data prior to the operator intervention and based on the past performance of the feedwater system during the power ascension program.

Level 2 acceptance criteria were satisfied with the exception of verifying that feedwater level control would avoid loss of feedwater flow due to a high water level trip. This acceptance criteria could not be verified because the operator tripped the feedpumps before reactor level had peaked during the transient. Insufficient data was available to demonstrate that the high water level trip would have been avoided. An engineering evaluation determined that if the high water level trip had occurred, it would not involve a safety issue. The feedwater system is not a safety related system and the high level trip is not a safety related function.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
Bypass Valve Opening Delay	-0.02 sec	< 0.1 sec following the Beginning of SV or CV Closure
Bypass Valve Reached 80% Opening	0.19 sec	< 0.3 sec following the Beginning of SV or CV Closure
Main Steam Line	No	Not Flooded
Vessel Dome Pressure Increase - Level 1	113 psi	< 171 psi
Vessel Dome Pressure Increase - Level 2	113 psi	< 146 psi
Flux Increase - Level 1	0%	≤ 2.5%
Flux Increase - Level 2	0%	≤ 0.5%
Recirc Pump Coastdown Time	Yes	Below the 4.5 sec Pump Inertia Time Constant Curve
RPT Breaker Response Time	89 ms	≤ 175 ms
MSIV	No	Not Isolated
Maximum Reactor Water Level	32.5"	< 54" (L8)
Minimum Reactor Water Level	-1"	> -38" (L2)
SRV Tailpipe Temperature	* N/A	Returned to Within 10°F of Initial Temperature

* SRV's did not lift during the test.

4.26 2STP-28, SHUTDOWN FROM OUTSIDE THE CONTROL ROOM

OBJECTIVES

The objectives of this test are to demonstrate that the Reactor a) can be safely shutdown from outside the control room, b) can be maintained in a Hot Standby condition from outside the control room and c) can be safely cooled from hot to cold shutdown from outside the control room. In addition, it will provide an opportunity to demonstrate that the procedures for remote shutdown are clear and comprehensive and that operational personnel are familiar with their applications.

ACCEPTANCE CRITERIA

Level 1

None

Level 2

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

The reactor can be safely shutdown to a hot standby condition from outside the control room using the minimum shift crew complement.

The reactor coolant temperature and pressure can be lowered sufficiently (at a rate that does not exceed the Technical Specification limit) from outside the control room to permit operation of the Shutdown Cooling Mode of the Residual Heat Removal System.

The Shutdown Cooling Mode of the Residual Heat Removal System can be initiated from outside the control room with a heat transfer path established to the Ultimate Heat Sink.

The Shutdown Cooling Mode of the Residual Heat Removal System can be used to reduce Reactor coolant temperature at a rate which does not exceed the Technical Specification limit.

RESULTS

2STP-28.1, Reactor Shutdown to Hot Standby Demonstration

This subtest was implemented in Test Condition 2 at 21% rated thermal power. A reactor scram, full MSIV isolation and turbine trip was initiated from the auxiliary equipment room in accordance with the plant special event procedure with the remote shutdown panel manned.

From an initial pressure of 920 psig, reactor pressure peaked at 940 psig approximately 10 seconds following the scram. Positive pressure control from the remote shutdown panel was demonstrated 20 minutes following the scram by manually opening a safety relief valve and lowering reactor pressure from 820 psig to 770 psig.

Reactor vessel water level reached a minimum of minus one inch approximately 10 seconds following the scram. RCIC was started from the remote shutdown panel three minutes after the scram and was used to maintain water level between 32 and 40 inches during the cooldown.

The reactor was cooled down 21 degrees F in 44 minutes using RCIC and safety relief valves controlled from the remote shutdown panel.

The applicable Level 2 acceptance criteria were satisfied during the performance of this subtest. All system operations from the remote shutdown panel were satisfactory.

The test results are summarized as follows:

Parameter	Measure Value	Acceptance Criteria
Reactor Cooldown	Yes	Can be initiated and controlled from outside the control room
Reactor Shutdown to a Hot Standby Condition	Yes	Can be initiated from outside the control roo
Shutdown Cooling Mode of The Residual Heat Removal System	Yes	Can be initiated from outside the control roo

2STP-28.2, Reactor Cooldown Demonstration

This subtest was implemented in Test Condition 2, separately from 2STP-28.1, Reactor Shutdown to Hot Standby Demonstration. Initial RPV parameters were 42 psig pressure and >60 inch indicated (at Remote Shutdown panel) water level.

The shutdown cooling mode of the residual heat removal system was placed in operation from outside the control room and a controlled cooldown was initiated without any aid from inside the control room.

Vessel temperature was lowered from 295 degrees F to 243 degrees F in one hour and fifteen minutes for a cooldown rate of 42 degrees F per hour.

During this subtest, the remaining Level 2 acceptance criteria were satisfied.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
Shutdown Cooling Mode of the Residual Heat Removal System	Yes	Can be initiated from outside the control room
Cooldown Rate	$42^{o}F$	$\leq 100^{o}F/hr$

4.27 2STP-29, RECIRCULATION FLOW CONTROL SYSTEM

OBJECTIVES

The objectives of this test are to demonstrate the flow control capability of the plant over the entire pump speed range, in both local manual and master manual operation modes and to determine that the controllers are set for the desired system performance and stability.

ACCEPTANCE CRITERIA

Level 1

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

Level 2

A scram shall not occur due to recirculation flow control maneuvers. The APRM neutron flux trip avoidance margin shall be ≥7.5% when the power maneuver effects are extrapolated to those that would occur along the 100% rated rod line.

The decay ratio of any oscillatory controlled variable must be ≤0.25.

Steady-state limit cycles (if any) shall not produce turbine steam flow variations greater than ±0.5% of rated steam flow.

RESULTS

2STP-29.1, Local Manual Recirculation Flow Control

The local manual recirculation flow control subtests were performed during the ascension to Test Conditions 3 and 6. In these subtests, the recirculation flow control system's responses to step changes in generator speed demand, together with related reactor parameters response, were recorded to verify stability. Nominal ±5% generator speed demand steps were injected into the recirculation flow control loops where the delta speed versus delta demand curves show the greatest gain. A voltage step generator was used to introduce the transients.

For the "A" and "B" Loop, there were no divergent oscillations. No scram occurred, and the APRM neutron flux trip avoidance margin was acceptable. The decay ratios of the oscillatory variables were acceptable. Dynamic and steady state oscillations were tested acceptable.

The test results are summarized as follows:

Parameter	Measured Value TC 3	Measured Value TC 6	Acceptance Criteria
Transient Response	No	No	Not Divergent
Margin to Flux Scram	14.2%	15.6%	> 7.5%
Decay Ratio	0	0	≤ 0.25

2STP-29.2, Master Manual Recirculation Flow Control

The Master Manual Recirculation Flow Control subtests were performed in Test Conditions 3 and 6. The subtests were performed by introducing steps of approximately +5% speed demand using the slow (first) detent of the Master Controller.

This testing was successfully performed with all acceptance criteria satisfied.

The test results are summarized as follows:

	Measured Value		
Parameter	TC 3	TC 6	Acceptance Criteria
Transient Response	No	No	Not Divergent
Margin to Flux Scram	8.2%	14.4%	> 7.5%
Decay Ratio	0	0	≤ 0.25
Limit Cycle Oscillations	0	0	0.5%

4.28 2STP-30, RECIRCULATION SYSTEM

OBJECTIVES

The objectives of this test are to:

Obtain recirculation system performance data during steady-state conditions, pump trip, flow coastdown, and pump restart.

Verify that the feedwater control system can satisfactorily control water level on a single recirculation pump trip without a resulting turbine trip and associated scram.

Record and verify acceptable performance of the circuit for a two-recirculation pump trip.

ACCEPTANCE CRITERIA

Level 1

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

The recirculation pump and motor time constant for the two pump drive flow coastdown transient should be $\leq$4.5 seconds from 1/4 to 2 seconds after the pumps are tripped and $\geq$3.0 seconds from 1/4 to 3 seconds after the pumps are tripped.

Level 2

The reactor water level margin to avoid a high level trip shall be $\geq$3.0 inches during the one pump trip.

The APRM margin to avoid a scram shall be $\geq$7.5% during the pump trip recovery.

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

The measured core delta P shall not be >0.6 PSI above prediction.

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

The measured recirculation pump efficiency shall not be >8% points below the vendor tested efficiency.

RESULTS

2STP-30.1, Recirculation System, One Pump Trip

One purpose of this subtest is to demonstrate the ability to avoid a high reactor water level main turbine and reactor feedwater pump turbine trip when a single recirculation pump is tripped. Secondly, this subtest demonstrates the capability to restart the tripped recirculation pump without causing a reactor scram.

This subtest was performed in Test Condition 3 at 71% reactor power and in Test Condition 6 at 96% reactor power. In Test Condition 3, the margin to high level turbine trip was 13.2 inches, and the margin to APRM scram was 32%. During the Test Condition 6 performance, the margin to high level turbine trip was 10.3 inches, and the margin to APRM scram was 29%. The reactor did not scram at either test condition during the pump trip recovery. All acceptance criteria were satisfied.

2STP-30.2, Recirculation Pump Trip (RPT) of Two Pumps

The purpose of this subtest is to monitor the coastdown rate of the recirculation pumps following the simultaneous tripping of both recirculation pumps. The flow coastdown rate is monitored to verify that the flow reduces quickly enough to limit the reactor power spike and not so quickly that the core flow reduction precedes the drop in heat flux which could cause a limiting Critical Power Ratio (CPR) transient.

During Test Condition 3 at approximately 73% reactor power, both recirculation pumps were simultaneously tripped using the RPT circuitry. Subsequent data reduction and analysis showed that the flow coastdown minimum criteria, based on maintaining nucleate boiling at the high power node during a LOCA, was satisfied. The maximum coastdown criteria based on maintaining adequate margin to thermal limits for transient events initiated by or resulting in a turbine trip or generator load rejection at the end of core life, however, was not satisfied. Some of the data fell cutside of the upper curve for a brief period of time. Analysis of the Level 1 criteria violation has determined that the magnitude of the variation is not significant and is well within the uncertainty assumed in the analysis. The actual test results are bounded by Limerick's FSAR RPT/EOC analyses and are therefore acceptable.

Additionally, the recirculation pump flow coastdown was reanalyzed from data obtained during the Test Condition 6 turbine trip. This data satisfied all flow coastdown criteria.

2STP-30.3, Recirculation System Performance

During this test, data is collected at various steady state plant conditions to determine the Recirculation System performance. Data was recorded at steady state conditions during Test Condition 2, 3, 4 and 6 at reactor powers of 24%, 65%, 42% and 96%, respectively. Calculations were performed to determine pump efficiencies, core flow shortfall, core delta P, and drive flow shortfall. All acceptance criteria were satisfied.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
Core Flow Shortfall	4.4%	≤ 5%
Core Delta P	18.1 psid	≤ 22.5 psid
Drive Flow Shortfall	2.8%	≤ 5%
Recirc Pump Efficiency	90%	≥ 72%

4.29 2STP-31, LOSS OF TURBINE-GENERATOR AND OFFSITE POWER

OBJECTIVES

This test determines electrical equipment and reactor system transient performance during a trip of the main turbine-generator coincident with the loss of all sources of offsite power.

ACCEPTANCE CRITERIA

Level 1

All safety systems, such as the reactor protection system, the diesel generators and, if necessary, HPCI and/or RCIC system actuation occurs during this transient, reactor water level shall be maintained above the initiation level of the Low Pressure Core Spray, LPCI, Automatic Depressurization System, and MSIV Closure. Manual intervention may be performed to prevent HPCI/RCIC actuation and to terminate the transient provided the following are satisfied: 1) adequate pressure control is demonstrated by relief valve action (if necessary), and 2) the rate of vessel water level decrease is established well enough to estimate when automatic initiation of HPCI and RCIC systems would occur. Diesel generators shall start automatically.

Level 2

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

Reactor pressure shall not exceed 1250 psig.

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

Normal cooling systems shall be capable of maintaining adequate drywell cooling and adequate suppression pool water temperature.

RESULTS

2STP-31.1, Loss of Turbine-Generator and Offsite Power

This subtest was performed in Test Condition 2 at 21.1% of rated thermal power. Prior to the trip, reactor pressure was 924 psig, drywell temperature was 119 degrees F and drywell pressure was reduced to -0.2 psig. Plant services busses 222, 214D, 224D, 234D and 244D were aligned to Unit 1 power supplies and the 101 safeguard power supply to Unit 2 safeguard busses was racked out. The main turbine and the 500 KV substation breaker 205 were simultaneously tripped to simulate a loss of turbine-generator with a coincident loss of all offsite power.

The diesel generators auto started and restored power to the four 4 KV safeguard busses within 9.4 seconds. The reactor protection system properly inserted a scram on reactor vessel level 3 in 51 seconds and the level remained above the HPCI, RCIC, LPCI, Core Spray, and MSIV level setpoints.

All control room instrumentation necessary for safe reactor shutdown remained energized. Reactor pressure did not increase above its initial value; no safety relief valve actuation was required. Drywell average temperature increased to 129.1 degrees in 22 minutes. Peak drywell pressure reached 0.29 psig. Suppression pool water temperature remained at 79.1 degrees F throughout the test and subsequent recovery. All acceptance criteria were satisfied.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
All Safety Systems	Yes	Functioned as required
Minimum Water Level	-30.7"	> -129"
Maximum Reactor Pressure	925 psig	< 1250 psig
Drywell Pressure	0.29 psig	< 1.68 psig
Bulk Drywell Temperature	129.1°F	< 135°F
Suppression Pool Temperature	79.1°F	< 120°F
SRV Tailpipe Temperature	N/A	Returned to within 10°F of initial temperature if lifted

4.30 2STP-32, ESSENTIAL HVAC SYSTEM OPERATION AND CONTAINMENT HOT PENETRATION TEMPERATURE VERIFICATION

OBJECTIVES

The objectives of this test are to demonstrate that under actual/normal operating conditions during Test Condition Heatup, Test Condition 3 and Test Condition 6, the various HVAC systems are capable of maintaining specified ambient temperatures and, where required, relative humidity within the following areas:

a) Primary Containment (drywell and suppression chamber) (for temperature only)

b) Reactor Enclosure and Main Steam Tunnel (for temperature only)

c) Control Room

d) Control Enclosure

e) Radwaste Enclosure (for temperature only)

This test also verifies that the concrete temperature surrounding containment hot penetrations remains within specified limits.

In addition, this test shall verify that the reactor pressure vessel support skirt surrounding air temperature and air flow impingement velocities, remain within specified limits.

ACCEPTANCE CRITERIA

Level 1

The drywell area volumetric average air temperature does not exceed 135 degrees F.

Level 2

The drywell area and suppression chamber are maintained between 65 degrees F and 150 degrees F in safety related equipment areas and not to exceed 160 degrees F in non-safety related equipment areas.

The reactor pressure vessel (RPV) support skirt surrounding air temperature is maintained above a minimum of 55 degrees F.

The concrete temperatures surrounding primary containment main steam line and feedwater line penetrations are maintained at less than or equal to 200 degrees F.

The following areas of the control enclosure are maintained between 65 degrees F and 104 degrees F: rooms 164, 258, 263, 336, 428, 429, 430, 431, 432, 433, 434, 435, 449, 450, 452, 453, 454, 540, 619, 625, 624A and 624B.

The control room is maintained at a temperature between 65 degrees F and 78 degrees F and relative humidity between 30% R.H. and 90% R.H.

The following areas of the reactor enclosure are maintained between 65 degrees F and 104 degrees F: rooms 182, 189, 279, 284, 287, 370, 475, 479, 574, 580B, 580C, 580D, 580G, 581, 582, 583, 585, 594, 637, 638, 641, 651 and 653.

The following areas of the reactor enclosure are maintained between 65 degrees F and 110 degrees F: rooms 576, 577, 578 and 579.

The following areas of the reactor enclosure are maintained between 65 degrees F and 115 degrees F: rooms 173, 174, 179, 180, 181, 184, 185, 188, 280, 281, 283, 285, 575, 584, 589, 593 and 597.

The following areas of the reactor enclosure are maintained between 65 degrees F and 120 degrees F: rooms 286, 374, 375, 376, 480 and 587.

The following areas of the radwaste enclosure are maintained between 65 degrees F and 76 degrees F: rooms 410, 411, 412, 415, 416, 417 and 418.

The following battery control rooms are to be maintained between 65 degrees F and 104 degrees F: rooms 323, 324, 360, 361, 425, 426, 427 and 436.

The auxiliary equipment room 542 is to be maintained between 60 degrees F and 82 degrees F, with the relative humidity maintained between 30% R.H. and 90% R.H.

The reactor pressure vessel support skirt flow impingement velocity is less than 15 feet per second.

RESULTS

2STP-32.1, Primary Containment Temperature

This subtest monitored temperatures at various locations in the drywell and the suppression chamber. The subtest was repeated four times each in TC Heatup (rated reactor temperature and pressure), TC 3 and TC 6 to obtain the following combinations of primary containment chilled water loops/unit cooler fans:

a) Chilled water loop "A" / Fans "1"

b) Chilled water loop "A" / Fans "2"

c) Chilled water loop "B" / Fans "1"

d) Chilled water loop "B" / Fans "2"

For Test Condition Heatup with the "A" chilled water loop and the "1" fans in service, and with "B" chilled water loop and the "1" fans in service, fan 2A2V212 was substituted for fan 2A1V212 due to equipment availability. For Test Condition 6 with the "B" chilled water loop and the "1" fans in service, fan 2A2V212 was substituted for fan 2A1V212 due to equipment availability, and the "A" chilled water loop subtests were performed during the Warranty Run due to chilled water loop valving problem.

All acceptance criteria were satisfied except the air temperature at RPV skirt area could not be verified during the Warranty Run since the temporary temperature sensors were removed during the pre warranty run outage. This exception was evaluated and accepted since all pertinent data has been verified acceptable in previous test conditions and by 2STP-32.7, Reactor Pressure Vessel Support Skirt Surrounding Air Temperature and Impingement Velocity.

The test results are summarized as follows:

Test Condition Heatup

	Measured Value ($^{\circ}F$)				
	Loop A		Loop B		
Parameter	Fans 1	Fans 2	Fans 1	Fans 2	Acceptance Criteria
Maximum Drywell Temperature	*154	*157	*155	*155	<150
Minimum Drywell Temperature	107	110	112	110	> 65
Volumetric Avg. Drywell Temperature	120	123	124	124	<130
Maximum Suppression Pool Temperature	86	90	90	90	<150
Minimum Suppression Pool Temperature	85	90	90	90	> 65
RPV Skirt Temperature	76	77	82	82	> 55

* These values are not Drywell Volumetric average temperature (which acceptance Criteria of <150 is measured against) but are the highest localized Drywell Temperature readings.

Test Condition Three

Parameter	Measured Value (°F)				Acceptance Criteria
	Loop A		Loop B		
	Fans 1	Fans 2	Fans 1	Fans 2	
Maximum Drywell Temperature	130	131	130	131	<150
Minimum Drywell Temperature	103	105	104	105	> 65
Volumetric Avg. Drywell Temperature	115	117	117	117	<130
Maximum Suppression Pool Temperature	100	101	101	103	<150
Minimum Suppression Pool Temperature	99	99	99	98	> 65
RPV Skirt Temperature	71	72	77	77	> 55

Test Condition Six or Warranty Run

Parameter	Measured Value (°F)				Acceptance Criteria
	Loop A		Loop B		
	Fans 1	Fans 2	Fans 1	Fans 2	
Maximum Drywell Temperature	138	140	131	133	<150
Minimum Drywell Temperature	110	111	104	106	> 65
Volumetric Avg. Drywell Temperature	125	124	120	122	<130
Maximum Suppression Pool Temperature	92	92	100	100	<150
Minimum Suppression Pool Temperature	87	87	90	95	> 65
RPV Skirt Temperature	N/A	N/A	79	79	> 55

2STP-32.2, Hot Penetration Concrete Temperature

This subtest consists of monitoring and recording the concrete surface temperature surrounding the main steam line penetrations and feedwater penetrations outside primary containment.

For Test Conditions Heatup through Test Condition 6, concrete temperatures remained well under the 200 degree limit with the maximum recorded temperature of 158 degrees F on feedwater line "B" (0 degree quadrant) and minimum recorded temperature of 83 degrees F on main steam line "A" (180 degree quadrant).

2STP-32.3, Control Enclosure Temperature and Relative Humidity

In this subtest, temperature is monitored at the various locations in the control enclosure. The subtest was repeated twice in each designated test condition, once with the loop "A" Control Enclosure chilled water in operation, and then with the loop "B" Control Enclosure chilled water in operation.

For Test Condition Heatup, data was not obtained for loop "B" due to equipment availability and the associated subtest was deleted from the test condition. When the loop "A" test was performed, two data points had temperature values below the acceptance criteria minimum of 65°F. One of these was an instrumentation problem and backup data was taken using a hand-held test instrument. The second data point, located in the Remote Shutdown room, had a temperature value of 63°F on the first of two data sets. This temperature was evaluated and accepted by engineering.

For test conditions 3 and 6, all test data for both loops "A" and "B" was within the required ranges. The Remote Shutdown room was above the minimum temperature and well below the maximum. The instrument problem noted above for TC Heatup, was corrected prior to the TC 3 test and the associated room temperature was acceptable.

The test results are summarized as follows:

	Measured Temperature (°F)					
	TC HU	TC 3		TC 6		
Room No.	Loop A Min-Max	Loop A Min-Max	Loop B Min-Max	Loop A Min-Max	Loop B Min-Max	Acceptance Criteria Min-Max
164	74-84	68-79	77-81	76-82	76-81	65-104
258	75-76	69-71	80-80	78-79	79-79	65-104
263	82-83	78-80	83-84	84-84	82-82	65-104
336	78-79	75-76	80-81	81-82	81-81	65-104
323	83-83	79-79	82-83	81-81	81-82	65-104
324	80-81	77-78	81-82	80-80	80-81	65-104
360	77-78	77-77	79-80	78-79	79-79	65-104
361	76-77	75-76	78-80	78-78	79-79	65-104
425	80-80	79-79	81-82	79-79	80-80	65-104
426	75-74	75-76	78-80	76-77	77-77	65-104
427	77-78	77-77	79-80	79-79	79-79	65-104
436	79-79	78-78	81-81	77-78	79-79	65-104
428	85-86	83-84	87-87	86-86	86-87	65-104
429	83-82	82-81	84-84	84-84	84-85	65-104
430	83-84	82-82	85-85	84-84	84-85	65-104
431	80-80	78-79	81-82	80-80	80-81	65-104
432	84-85	82-83	86-86	84-85	85-85	65-104
433	84-84	82-82	85-85	84-84	84-84	65-104
434	81-82	80-80	84-84	81-82	83-83	65-104
435	81-82	81-81	84-84	82-82	82-82	65-104
449	82-83	79-80	83-83	81-82	82-82	65-104
450	81-82	77-78	81-82	81-82	82-82	65-104
454	78-79	76-77	81-81	80-80	79-80	65-104
452	81-83	86-87	86-87	83-83	86-86	65-104
453	82-82	82-83	85-85	83-84	85-86	65-104
540	63-65	69-69	70-71	68-68	66-67	65-104
542	61-71	64-74	66-75	61-72	62-73	60- 82
619	75-76	72-73	75-77	72-74	73-73	65-104
625	84-84	76-79	83-85	80-82	79-81	65-104
624A	81-82	74-74	78-79	77-77	75-75	65-104
624B	82-82	74-74	78-78	76-77	76-77	65-104

	Measure Relative Humidity (%)					
	TC HU	TC 3		TC 6		
Room No.	Loop A Min-Max	Loop A Min-Max	Loop B Min-Max	Loop A Min-Max	Loop B Min-Max	Acceptance Criteria Min-Max
Room 542	54-69	43-56	56-72	39-57	52-64	30-90

2STP-32.4, Control Room Temperature and Relative Humidity

This subtest consists of monitoring temperatures in the control room, offices and related inhabited areas. The subtest was repeated twice in each designated test condition, once with the loop "A" chilled water in operation, and then with the loop "B" chilled water in operation.

For Test Condition Heatup, data was not obtained for loop "B" due to equipment availability and the associated subtest was deleted from the test condition. When the test was performed for loop "A", all data was within the required ranges.

For Test Condition 3, all loop "A" test data was within the required ranges. When the test was performed for loop "B", all temperatures were within the required ranges but control room return air relative humidity indicated low. The return air instrumentation was checked for calibration and found to be reading low. The instrumentation was then recalibrated.

For Test Condition 6, all test data was within the required ranges for both loops "A" and "B".

The test results are summarized as follows:

	Measured Temperature (°F)					
	TC-HU	TC 3		TC 6		
Room No.	Loop A Min-Max	Loop A Min-Max	Loop B Min-Max	Loop A Min-Max	Loop B Min-Max	Acceptance Criteria Min-Max
Office	72-73	74-74	73-75	72-73	72-73	65-78
Shft Super Office	72-73	71-71	72-74	71-71	71-72	65-78
Shop	70-72	70-71	72-74	70-71	70-70	65-78
Instrument Lab.	70-71	69-70	72-74	70-70	70-70	65-78
Control Rm	72-72	71-71	73-74	71-72	71-71	65-78
Control Rm	73-74	73-73	74-76	72-73	72-73	65-78
Control Rm	73-73	73-74	75-76	72-73	72-73	65-78
Control Rm	70-71	70-71	71-73	70-71	69-71	65-78
Control Rm	71-72	72-73	73-74	73-74	71-72	65-78
Control Rm	71-71	69-70	71-73	70-71	69-70	65-78
Control Rm	71-73	72-73	73-75	73-74	71-72	65-78
Control Rm	69-70	69-70	71-72	70-71	68-70	65-78
Control Rm	69-71	71-72	72-74	73-73	71-71	65-78
Control Rm	70-70	69-70	71-73	71-72	70-71	65-78
00C681	71-72	72-73	72-73	72-72	71-72	65-78
00C681	71-71	70-72	71-72	71-72	70-70	65-78

	Measure Relative Humidity (%)					
	TC HU	TC 3		TC 6		
Room No.	Loop A Min-Max	Loop A Min-Max	Loop B Min-Max	Loop A Min-Max	Loop B Min-Max	Acceptance Criteria Min-Max
Control Rm	59-61	36-45	56-67	39-43	51-53	30-90
Control Rm	60-60	36-45	55-66	39-42	50-52	30-90
Control Rm	60-60	36-45	58-67	39-43	52-52	30-90
Control Rm	60-64	38-46	60-70	41-44	53-55	30-90
Control Rm	58-63	35-42	55-64	37-41	49-50	30-90
Control Rm	61-64	38-46	60-69	40-45	53-55	30-90
Control Rm	58-63	35-42	55-64	37-41	50-50	30-90
Control Rm	62-63	39-45	59-68	40-44	52-54	30-90
Control Rm	60-63	35-43	56-64	37-42	50-51	30-90
Control Rm	63-63	36-44	58-66	37-42	51-52	30-90
00C101C	61-62	24-37	59-60	41-43	54-56	30-90
00C101C	58-59	24-33	57-57	41-48	57-57	30-90

2STP-32.5, Reactor Enclosure and Main Steam Tunnel Temperature

This subtest consists of monitoring and recording the ambient air temperatures within various rooms of the reactor enclosure.

For Test Condition Heatup, all recorded room/area temperatures were within acceptance criteria limits. Fifteen rooms/areas however, had differential (supply vs. return) temperatures in excess of the specified limits. Twelve rooms/areas were evaluated by Engineering as acceptable with no further action required, and in the other areas the data was acceptable to continue power ascension testing, with further evaluation to occur when this subtest was performed again during Test Condition 3.

For Test Condition 3, all recorded room/area temperatures were within acceptance criteria limits. Eight rooms/areas had differential temperatures in excess of the specified limits. Engineering accepted test results with no further action required.

For Test Condition 6, all recorded room/area temperatures were within acceptance criteria limits. Twenty-six rooms/areas had differential temperatures in excess of the specified limits. Engineering reviewed the test results and concluded that the higher than expected differential temperatures would have no impact or equipment operability and no significant effect on equipment qualifications. This is based on a study of outdoor air temperatures for the Philadelphia area which showed that the occurrence of outside air temperatures high enough to cause a room temperature to exceed its upper limit is extremely small (1% or less). Room/area temperatures would therefore be within the temperature ranges specified for the equipment qualifications 99% of the time. A follow-up study has been initiated to determine what, if any, increased preventative maintenance activities are required to preserve the qualified life of safety-related equipment within the affected areas.

The test results are summarized as follows:

Room No.	Measured Temperature (°F) TC HU MIN-MAX	TC 3 MIN-MAX	TC 6 MIN-MAX	Acceptance Criteria MIN-MAX
182	84-84	77-77	81-81	65-104
189	84-84	75-75	77-78	65-104
279	85-85	76-78	80-82	65-104
284	80-84	73-79	73-77	65-104
287	92-93	80-80	80-80	65-104
370	83-84	75-77	78-81	65-104
475	82-85	74-81	78-85	65-104
479	84-85	75-76	83-83	65-104
574	81-82	72-74	76-76	65-104
580	83-83	74-76	80-81	65-104
581	87-87	77-78	83-83	65-104
582	85-86	70-72	77-78	65-104
583	80-81	76-78	84-84	65-104
584	92-92	81-83	92-92	65-104
585	80-81	73-76	73-76	65-104
594	82-83	74-78	77-81	65-104
637	80-81	72-74	77-77	65-104
638	80-81	73-75	75-76	65-104
641	81-82	72-74	77-80	65-104
653	83-83	76-76	80-80	65-104
651	83-83	74-74	75-75	65-104
576	84-84	75-75	85-86	65-104
577	89-90	76-77	87-88	65-104
578	82-83	72-73	82-82	65-104
579	85-86	74-75	79-79	65-104
173	78-82	72-77	72-77	65-115
174	81-82	70-73	73-74	65-115
179	87-88	88-90	97-98	65-115
180	87-92	86-87	91-91	65-115
181	83-84	78-79	80-80	65-115
184	82-84	77-79	77-79	65-115
185	82-83	75-77	77-78	65-115
188	82-84	75-77	77-79	65-115
280	80-81	78-78	78-78	65-115
281	82-83	75-76	76-76	65-115
283	106-106	102-102	107-108	65-115
285	99-100	97-97	107-108	65-115
575	88-88	80-80	91-91	65-115
584	92-92	81-83	92-92	65-115
597	94-95	83-85	94-94	65-115
593	88-88	81-82	91-96	65-115
589	85-85	74-74	82-83	65-115

Measured Temperature (^{o}F)

Room No.	TC HU MIN-MAX	TC 3 MIN-MAX	TC 6 MIN-MAX	Acceptance Criteria MIN-MAX
286	84-85	75-76	78-78	65-120
374	96-96	89-89	97-97	65-120
375	100-100	92-93	100-100	65-120
376	95-95	87-87	94-94	65-120
480	86-109	79-102	86-115	65-120
587	103-109	100-104	110-115	65-120

Measured Differential Temperature (^{o}F)

Instrument No.	TC HU	TC 3	TC 6	Acceptance Criteria
TE-RR-182	1	1	2	≤4
TE-RR-189	4	0.5	3.5	≤4
TE-RR-279A	3	1.5	3	≤4
TE-RR-279B	2	-1	2	≤4
TE-RR-284A	0.5	-4	-1	≤4
TE-RR-284B	4	2	5	≤4
TE-RR-284C	1.5	-0.5	1	≤4
TE-RR-287	10.5	3	2	≤4
TE-RR-370A	4	0	5	≤4
TE-RR-370B	4	-1	3.5	≤4
TE-RR-370D	4	-1	2.5	≤4
TE-RR-370E	3	-1	5	≤4
TE-RR-475A	4	2.5	6.5	≤4
TE-RR-475B	1	1	2	≤4
TE-RR-475C	2.5	4	5.5	≤4
TE-RR-475D	1	0.5	5.5	≤4
TE-RR-475E	1	0	2.5	≤4
TE-RR-475F	1	0	1	≤4
TE-RR-475G	1.5	2.5	2	≤4
TE-RR-475H	0.5	-0.5	1	≤4
TE-RR-475J	1	-1.5	0.5	≤4
TE-RR-475K	5	0	4.5	≤4
TE-RR-479	5.5	-1.5	5.5	≤4
TE-RR-574	0.5	-1.5	-0.5	≤4
TE-RR-580B	1.5	-2	2.5	≤4
TE-RR-580C	1	0.5	2	≤4
TE-RR-580D	1.5	-1	0.5	≤4
TE-RR-580E	0.5	0	2	≤4
TE-RR-580F	1.5	0	0.5	≤4
TE-RR-581	6	3.5	4.5	≤4
TE-RR-582	4.5	-3	-1	≤4
TE-RR-583	0.5	3	5.5	≤4
TE-RR-585	2	-3	4.5	≤4

Measured Differential Temperature (^{o}F)

Instrument No.	TC HU	TC 3	TC 6	Acceptance Criteria
TE-RR-594A	1.5	2.5	2.5	≤4
TE-RR-594B	2	-0.5	2	≤4
TE-RR-594C	1.5	1	2.5	≤4
TE-RR-637A	0.5	-1	0.5	≤4
TE-RR-637B	0.5	-1	0.5	≤4
TE-RR-638	1.5	1	1	≤4
TE-RR-641A	3	-1	2	≤4
TE-RR-641B	3.5	-0.5	6	≤4
TE-RR-653	5	2.5	4.5	≤4
TE-RR-651	5	0.5	-0.5	≤4
TE-RR-576	3	0.5	9	≤10
TE-RR-577	8.5	2	11	≤10
TE-RR-578	1.5	-2	5.5	≤10
TE-RR-579	2.5	0	2.5	≤10
TE-RR-280	1.5	1	0	≤15
TE-RR-281	0.5	-1.5	-2	≤15
TE-RR-284	24	25	29.5	≤15
TE-RR-285	17.5	20	29.5	≤15
TE-RR-575	7	5.5	14.5	≤15
TE-RR-584	11	8	13.5	≤15
TE-RR-597	13.5	10	15.5	≤15
TE-RR-593	6	6.5	16.5	≤15
TE-RR-589	5	0	9.5	≤15
TE-RR-286	4.5	-1	6	≤20
TE-RR-374	16	12.5	21	≤20
TE-RR-375	20.5	15.5	28.5	≤20
TE-RR-376	14.5	11	19	≤20
TE-RR-480A	26	27	36.5	≤20
TE-RR-480B	9	9	13.5	≤20
TE-RR-480C	5	5.5	8.5	≤20
TE-RR-480D	22	25	35.5	≤20
TE-RR-587A	21	23.5	30	≤20
TE-RR-587B	24	25.5	32	≤20
TE-RR-587C	20	22.5	30	≤20
TE-RR-587D	24	23	29	≤20

2STP-32.6, Radwaste Enclosure Temperature

This subtest was deleted for Unit 2 since radwaste enclosure is common to both Unit 1 and Unit 2.

2STP-32.7, Reactor Pressure Vessel Support Skirt Surrounding Air Temperature and Impingement Velocity

This subtest was performed at the end of Test Condition 2, concurrent with reactor trip and cooldown per 2STP-28.1, Reactor Shutdown to Hot Standby Demonstration, and 2STP-28.2, Reactor Cooldown Demonstration. No problems were encountered and all test data was within the required ranges.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
Skirt air temperature	81-84°F	> 55°F
Air flow velocity	13-14 Ft/sec.	< 15 Ft/sec.

4.31 2STP-33, PIPING STEADY STATE VIBRATION

OBJECTIVE

The objective of this test is to verify that the steady state vibration of Main Steam, Reactor Recirculation and selected BOP piping systems is within acceptable limits.

ACCEPTANCE CRITERIA

Level 1

Operating Vibration: The measured amplitude (peak to peak) of each remotely monitored point shall not exceed the allowable value for that point.

Level 2

Operating Vibration: The measured amplitude (peak to peak) of each remotely monitored point shall not exceed the expected value for that point.

The steady state vibrations of visually examined balance of plant piping are acceptable if the vibration levels are judged by a qualified test engineer to be negligible. Vibration levels judged to be potentially significant are evaluated as determined necessary by BPC Project Engineering.

The vibration measured by a remote accelerometer is acceptable if the acceleration frequency spectrum falls in the negligible region of the acceptance chart for that accelerometer. If the acceleration frequency spectrum crosses the negligible region boundary, the test results shall be evaluated by BPC Project Engineering.

RESULTS

2STP-33.1, Main Steam Piping (Inside Drywell) Steady State Vibration

This subtest provided the means for collecting vibration data on Main Steam piping at steady state conditions with various nominal main steam flows. Data was recorded by the Plant Monitoring System (PMS) from the remote monitoring instrumentation (24 lanyard potentiometers and 2 resistance temperature devices). Data was collected at Test Condition 2 (25% rated main steam flow), Test Condition 3 (50% and 75% rated main steam flow) and Test Condition 6 (100% rated main steam flow). One data point exceeded Level 2 limits in TC 6, but it was accepted since the value did not exceed the Level 1 criteria which is based on the piping material endurance stress for infinite number of fatigue life cycles.

The test results are summarized as follows:

Location	Measured Vibration (mil) TC 2	TC 3 (50%)	TC 3 (75%)	TC 6	Acceptance Criteria Level 1	Level 2
SA-LX-AC	1.2	2	8	16.9	< 20	< 10
SA-LY-AC	1.2	2	2	3.5	< 100	< 68
SA-LZ-AC	1.7	2	2	1.7	< 32	< 16
SA-UX-AC	1.8	2	2	2.9	< 656	<328
SA-UY-AC	1.7	2	2	2.3	< 654	<328
SA-UZ-AC	1.2	1	2	1.8	< 782	<392
SB-LX-AC	6.4	6	6	7.6	< 20	< 10
SB-LY-AC	6.4	6	6	5.8	< 100	< 66
SB-LZ-AC	4.7	5	5	5.2	< 20	< 10
SB-UX-AC	1.2	2	1	1.2	< 378	<190
SB-UY-AC	1.7	2	1	1.2	< 470	<236
SB-UZ-AC	5.8	5	12	20.4	< 234	<118
SC-LX-AC	6.4	6	5	6.4	< 158	< 80
SC-LY-AC	5.3	5	5	5.3	<1068	<534
SC-LZ-AC	6.5	6	7	6.5	< 202	<100
SC-UX-AC	7.0	7	6	7	< 62	< 30
SC-UY-AC	5.9	7	6	6.4	< 64	< 32
SC-UZ-AC	5.3	5	6	7.6	< 126	< 62
SC-LX-AC	6.0	6	7	7.2	< 20	< 10
SD-LY-AC	6.9	8	8	13.3	< 100	< 60
SD-LZ-AC	6.4	7	6	6.4	< 20	< 10
SD-UX-AC	6.4	6	8	9.3	< 72	< 36
SD-UY-AC	7.0	6	6	5.8	< 88	< 44
SD-UZ-AC	5.8	6	5	7.0	< 78	< 38

2STP-33.2, Recirculation Piping Steady State Vibration

This subtest provided the means for collecting vibration data on Recirculation piping at steady state conditions with various nominal recirculation pump flows. Data was recorded by the Plant Monitoring System (PMS) from the remote monitoring instrumentation (24 lanyard potentiometers and 3 resistance temperature devices). Data was collected at Test Condition Heatup (minimum recirc flow), Test Condition 3 (50% and 75% rated recirc flow), Test Condition 5 (minimum recirc flow) and Test Condition 6 (100% recirc flow). All vibration criteria were satisfied.

The test results are summarized as follows:

Location	Measured Vibration (mil) TC 2	TC 3 (50%)	TC 3 (75%)	TC 5	TC 6	Acceptance Criteria Level 1	Level 2
RA-SX-AC	1.2	2	1.7	1.7	1.7	<64	<32
RA-SY-AC	1.7	2	1.7	1.7	1.7	<84	<42
RA-SZ-AC	1.2	1	1.2	1.2	1.2	<78	<38
RA-PX-AC	1.7	2	1.7	1.7	1.7	<68	<34
RA-PY-AC	1.8	2	1.8	1.8	1.8	<80	<40
RA-PZ-AC	1.7	1	1.2	1.2	1.7	<68	<34
RA-DX-AC	1.7	2	1.7	1.7	1.7	<414	<212
RA-DY-AC	1.8	2	1.7	1.7	1.8	<386	<192
RA-DZ-AC	1.7	2	1.7	1.7	2.3	<484	<242
RA-RX-AC	1.7	2	1.2	1.7	1.7	<160	<80
RA-RY-AC	1.7	2	1.7	1.7	1.7	<138	<70
RA-RZ-AC	1.7	2	1.7	1.2	1.7	<208	<104
RB-SX-AC	1.2	2	1.7	1.8	1.8	<70	<34
RB-SY-AC	1.8	2	1.8	1.8	1.8	<92	<46
RB-SZ-AC	1.8	3	1.8	7.0	8.8	<54	<28
RB-PX-AC	1.8	1	1.8	1.2	1.8	<146	<74
RB-PY-AC	1.7	2	1.7	1.7	1.7	<226	<114
RB-PZ-AC	1.8	2	1.8	1.8	1.8	<140	<70
RB-DX-AC	6.4	7	7.6	7.0	7.6	<498	<250
RB-DY-AC	5.8	6	5.8	5.2	6.3	<398	<200
RB-DZ-AC	1.2	2	4.1	4.7	7.0	<422	<210
RB-RX-AC	0.6	4	0.6	3.5	0.6	<248	<124
RB-RY-AC	1.8	2	1.8	1.8	1.8	<150	<76
RB-RZ-AC	1.2	2	1.8	1.8	1.8	<234	<118

2STP-33.3, Main Steam (Outside Drywell), Main Steam Bypass, and Feedwater Piping Steady State Vibration

In this subtest, steady state vibration of specified segments of Main Steam (outside drywell), Main Steam Bypass and Feedwater piping was measured or visually examined.

For sections of piping not normally accessible, data was obtained from remotely mounted vibration sensors and recorded on the Plant Monitoring System (PMS). Recorded data was then processed as required and compared with design limits. Normally accessible piping was visually inspected and evaluated by qualified test engineers using visual and tactile judgement and hand held vibration monitors.

The subtest was conducted during Test Condition 2 (25% rated power), Test Condition 3 (50% and 75% rated power), and Test Condition 6 (100% rated power). All three feedwater loops were tested at each power level.

No piping steady state vibratory response problems were encountered during the tests. All vibration criteria were satisfied.

2STP-33.4, HPCI Steam Piping Steady State Vibration

In this subtest, steady state vibrations of the HPCI turbine steam supply and exhaust piping are measured with the turbine operating on nuclear steam at rated temperature and pressure with the HPCI pump discharging to the CST at rated head and flow.

The subtest was performed during Test Condition Heatup, concurrent with 2STP-15.2, HPCI Functional Demonstration. Data was recorded on the Plant Monitoring System (PMS) from remotely mounted vibration sensors. Recorded data was processed as required, and compared with design limits.

No piping steady state vibratory response problems were encountered during the test. All vibration criteria were satisfied.

2STP-33.5, RCIC Steam Piping Steady State Vibration

In this subtest, steady state vibrations of the RCIC turbine steam supply and exhaust piping are measured with the turbine operating on nuclear steam at rated temperature and pressure with the RCIC pump discharging to the CST at rated head and flow.

The subtest was performed during Test Condition Heatup, concurrent with 2STP-14.2, RCIC Functional Demonstration. Data was recorded on the Plant Monitoring System (PMS) from remotely mounted vibration sensors. Recorded data was processed as required and compared with design limits.

No piping steady state vibratory response problems were encountered during the test. All vibration criteria were satisfied.

2STP-33.6, Reactor Water Cleanup Piping Steady State Vibration

In this subtest, steady state vibrations of specified segments of Reactor Water Cleanup piping are measured with the reactor at rated temperature and pressure and with the reactor water cleanup system operating in the normal mode.

The subtest was conducted during Test Condition Heatup, concurrent with 2STP-70.3, Reactor Water Cleanup Normal Mode Performance Verification. Data was recorded on the Plant Monitoring System (PMS), from remotely mounted vibration sensors. Recorded data was processed as applicable, and compared with design limits.

No piping steady state vibratory response problems were encountered during the test. All vibration criteria were satisfied.

2STP-33.7, RHR Shutdown Cooling Mode Piping Steady State Vibration

In this subtest, steady state vibrations of specified segments of the RHR Loop A piping are measured or visually evaluated with the RHR system operating in the shutdown cooling at rated flow.

For sections of piping not normally accessible, data was obtained from remotely mounted vibration sensors and recorded on the Plant Monitoring System (PMS). Recorded data was then processed as required and compared with design limits. Normally accessible piping was visually inspected and evaluated by qualified test engineers using visual and tactile judgement and hand held vibration monitors.

The test was conducted following reactor shutdown at the conclusion of Test Condition 2, concurrent with 2STP-28.2, Shutdown Cooling Demonstration.

No piping steady state vibratory response problems were encountered during the test. All vibration criteria were satisfied.

2STP-33.8, EHC System Piping Steady State Vibration

This subtest is to measure steady state vibrations of specified segments of the EHC system piping. The test was conducted during Test Condition 2 (25% rated power), Test Condition 3 (50% and 75% rated power), transition to Test Condition 6 from Test Condition 4 (87.5% rated power) and Test Condition 6 (100% rated power).

Data was obtained from remotely mounted vibration sensors and recorded on the Plant Monitoring System (PMS). Recorded data was processed as required and evaluated.

No piping steady state vibratory response problems were encountered during the tests. There are no acceptance criteria associated with this subtest.

4.32 2STP-34, OFFGAS PERFORMANCE VERIFICATION

OBJECTIVES

The objectives of this test are to verify that the Offgas Recombination and Ambient Charcoal System operates within the technical specification limits and expected operating conditions.

ACCEPTANCE CRITERIA

Level 1

The allowable dose and dose rates from releases of radioactive gaseous and particulate effluents to areas at and beyond the SITE BOUNDARY shall not be exceeded.

Allowable limits on the radioactivity release rates of the six noble gases measured at the aftercondenser discharge shall not be exceeded.

The hydrogen content o. the offgas effluent downsteam of the recombiner shall be equal to, or less than, 4% by volume.

The total flow rate of dilution steam plus offgas when the steam jet air ejectors are in operation shall exceed 9555 lbs/hr.

Level 2

System flows, pressures, temperatures and dewpoint shall be within expected performance values.

The preheater, catalytic recombiner, aftercondenser, hydrogen analyzers, cooler condenser, activated carbon beds and the HEPA filter shall be performing their required functions adequately. The automatic drain systems function adequately.

TEST RESULTS

2STP-34.1, Offgas Performance Verification

This subtest verified proper operation of the Offgas System. During the test, operating parameters, including flows, pressures, temperatures, dewpoint, conductivity and annunciator status were observed and recorded. Gaseous grab samples and iodine/particulate samples were obtained and radiochemical analysis performed. Levels were observed at specified locations to verify proper drain system performance.

The subtest was performed during Test Condition Heatup, Test Condition 1 (15% - 20% power), Test Condition 3 (45% - 50% power and 70% - 80% power) and Test Condition 6 (95% - 100% power).

Dose and dose rates from releases of radioactive gaseous and particulate effluents at the site boundary have all been within Technical Specification Limits. Isotopic analysis in most cases indicated Lower Limit of Detection (LLD).

Radioactive release rates of the six noble gases measured at the recombiner aftercooler discharge were all well below the Technical Specification limit of 330,000 uCi/sec. The hydrogen content of the offgas effluent downstream of the recombiner was below the allowable level 1 acceptance maximum of 4% by volume. For all test conditions, the actual hydrogen concentration was below 1%. The total flow rate of dilution steam plus offgas exceeded the required minimum value of 9,555 lb/hr for all test conditions.

In both Test Condition 3 and Test Condition 6, the cooler condenser discharge temperature and dewpoint exceeded level 2 acceptance criteria limits. This condition was evaluated and accepted for operation until at least the first refueling outage. An ongoing task force consisting of NED and plant staff is following this problem.

The test results are summarized as follows:

Parameter	Measured Value TC HU	TC 1	TC 3	TC 3	TC 6	Acceptance Criteria
Radioactive Release Rate at SITE BOUNDARY	Yes	Yes	Yes	Yes	Yes	Less than Tech Spec. Limits
Radioactivity Release Rate at After condenser Discharge	0.2	43	343	191	170	$\leq$330000 uci/sec.
H_2 Content After Recombiner	0.8	0.5	1.4	1.2	1.2	$\leq$4%
Total Steam and Gas Flow	13400	14000	14000	14000	14800	$\geq$9555 lb/hr
Preheater Inlet Temperature	N/A	N/A	280	280	280	250 $\pm$30°F
Preheater Outlet Temperature	N/A	N/A	340	340	340	340 $\pm$ 50°F
Recombiner Inlet	N/A	N/A	1.8	1.9	1.7	$\leq$4.8 psig
Recombiner Outlet	N/A	N/A	423	476	593	$\leq$900°F
After Condenser Gas Pressure	N/A	N/A	1.5	2.3	2.3	$\leq$4.8 psig
H_2 Content at After condenser	N/A	N/A	1.1	0.9	1.0	<1%

Parameter	Measured Value					Acceptance Criteria
	TC HU	TC 1	TC 3	TC 3	TC 6	
Cooler Condenser Discharge Temperature	N/A	N/A	52.5	44	45.5	40 $\pm 5^{\circ}F$
Cooler condenser Dewpoint	N/A	N/A	51	49	51	$\leq 45^{\circ}F$
Charcoal Adsorber Bed Temperature	N/A	N/A	64	64	64.6	65 $\pm 5^{\circ}F$
Offgas Flow	N/A	N/A	30	60	50	$\leq$75 SCFM

N/A: Not Applicable

4.33 2STP-35, RECIRCULATION SYSTEM FLOW CALIBRATION

OBJECTIVES

The objectives of this test are to perform a complete calibration of the recirculation system flow instrumentation, including specific signals to the plant process computer and to adjust the recirculation flow control system to limit maximum core flow to 107% of rated core flow.

ACCEPTANCE CRITERIA

Level 1

None

Level 2

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

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

The flow control system shall be adjusted to limit maximum core flow to 107% of rated.

The calculated jet pump M-Ratio shall not be < 0.2 points below prediction.

The nozzle and riser plugging criteria shall not be exceeded.

RESULTS

2STP-35.1, Recirculation System Flow Calibration

In Test Condition 3 at 72.8% power and 93.5% indicated core flow, single tap jet pump, double tap jet pump and recirculation loop data was recorded and a calculation was performed to determine total core flow. Calculated core flow was 93.23%. The jet pump loop flow summers were adjusted to provide correct loop and total core flows and the APRM/RBM flow bias instrumentation was adjusted to function properly at rated core flow conditions. All acceptance criteria were met with the exception of jet pumps 17/18 and 19/20 exceeding the nozzle and riser plugging criteria.

In Test Condition 6 at 96.1% power and 98.6% indicated core flow, the core flow was again calculated. Calculated core flow was 97.3%. The jet pump loop summers and APRM/RBM flow bias instrumentation were readjusted to function properly at rated power, rated flow conditions. All acceptance criteria were met with the exception of jet pump 17/18 exceeding the nozzle and riser plugging criteria.

Engineering has continued to review jet pump flow data and investigate the cause of flow differences between jet pump 17/18 as well as general flow patterns between recirculation loops.

The test results are summarized as follows:

Parameter	Measured Value TC 3	Measured Value TC 6	Acceptance Criteria
Jet Pump Flow Instrumention	Yes	Yes	Adjusted to Provide Correct Flow Indication
APRM/RBM Flow Bias Instrumentation	Yes	Yes	Adjusted to Function Properly at Rated Conditions
M-Ratio	2-16	2.00	$\geq$1.80
Nozzle Plugging	0.019-0.152	0.011-0.130	<0.12
Riser Plugging	0.001-0.039	0.001-0.042	<0.10

2STP-35.2, Recirculation System Flow Limiter Adjustment

In Test Condition 6, the recirculation pump MG set scoop tube mechanical and electrical high speed stops were adjusted to limit core flow to less than 109 and 107 percent respectively. The actual settings for both the "A" and "B" recirculation pumps were 107% for the mechanical stops and 106% for the electrical stops. The applicable acceptance criterion for this test was satisfied.

4.34 2STP-36, PIPING DYNAMIC TRANSIENTS

OBJECTIVES

The objectives of this test are to verify that the following pipe systems are adequately designed and restrained to withstand the following respective transient loading conditions:

Main Steam - Main Turbine Stop Valve/Control Valve closures at 20-25% and 95-100% of rated thermal power.

Main Steam and Relief Valve Discharge - Main Steam Relief Valve actuations.

Recirculation - Recirculation Pump trips and restarts.

High Pressure Coolant Injection steam supply - High Pressure Coolant Injection turbine trip.

Feedwater - Reactor feed pump trip/coastdown.

ACCEPTANCE CRITERIA

Level 1

Operating Transients: The measured amplitude (peak to peak) of each remotely monitored point shall not exceed the allowable value for that point.

Level 2

Operating Transients: The measured amplitude (peak to peak) of each remotely monitored point shall not exceed the expected value for that point.

The maximum measured loads, accelerations, displacements, and/or velocities are less than or equal to the acceptance limits specified.

In the judgment of the qualified test engineers, no signs of excessive piping response (such as damaged insulation, markings on piping, structural steel, hanger steel, or walls, damaged pipe supports, etc.) are found during a post-transient walkdown and visual inspection of the piping tested and associated branch lines.

RESULTS

2STP-36.1, Main Steam Piping Vibration during Main Turbine Stop Valve and Control Valve Closure

These subtests were performed during TC 2 in conjunction with STP-27.1, Turbine Trip Within Bypass Valve Capacity and during TC 6 following an unplanned generator load rejection and subsequent generator trip.

Data was recorded on the Plant Monitoring System (PMS) from remotely mounted transducers, including lanyard pots, load sensing clevis pins, pressure transducers and accelerometers. Recorded data from the lanyard pots and load sensing clevis pins was processed as required and evaluated against acceptance criteria limits. Data obtained from accelerometers and pressure transducers was also processed for evaluation by Engineering on an informational basis. Closure time of Turbine Stop Valve #2 was also recorded for information. Following completion of the test, a walkdown of the system was conducted and a visual inspection made to ensure that no signs of excessive piping transient response were present.

The results of the testing at Test Condition 2 demonstrated that dynamic loading and vibratory response of the main steam piping during main turbine stop and control valve closure was within acceptable design limits. No problems were discovered on the post-test walkdowns following the trip.

During TC 6, an unplanned generator load Rejection and subsequent main turbine trip occurred on 11/10/89. All vibration readings for the event were within limits for the NSSS scope MS piping. However, for the 7 load sensing clevis pins installed on the BOP steam lines and the lanyard pot installed on MSV #2, the Process Computer sampling rate was too slow to obtain the frequency resolution required (100 HZ). In addition, even at the slow sampling rates, two of the 7 Clevis Pins exceded allowable loads.

The two Clevis Pins exceeding allowable limits were acceptable as-is based on load levels experienced and post transient walkdown. To gather the required data at a future (Post Power Ascension) planned, or unplanned, Turbine Trip, the 7 Clevis Pins and Lanyard Pot will be left installed in the plant with appropriate data acquisition support. Data acquisition and analysis will be in accordance with Special Procedure SP-099.

The test results are summarized as follows:

Location	Measured Vibration (mil) TC 2	TC 6	Acceptance Criteria Level 1	Level 2
SA-LX-AC	1.2	32.1	< 64	< 60
SA-LY-AC	1.2	4.1	≤ 136	≤ 74
SA-LZ-AC	1.2	8.1	≤ 70	≤ 66
SA-UX-AC	12.3	33.3	≤ 656	≤ 328
SA-UY-AC	8.1	36.6	≤ 654	≤ 328
SA-UZ-AC	11.7	34.4	≤ 782	≤ 392
SB-LX-AC	6.4	18.2	≤ 68	≤ 60
SB-LY-AC	7.0	13.9	≤ 134	≤ 66
SB-LZ-AC	4.6	13.4	≤ 68	≤ 60
SB-UX-AC	5.8	24.0	≤ 378	≤ 190
SB-UY-AC	2.9	32.5	≤ 470	≤ 236
SB-UZ-AC	15.7	38.5	≤ 234	≤ 118
SC-LX-AC	5.8	6.9	≤ 158	≤ 80
SC-LY-AC	4.7	11.7	≤1068	≤ 538
SC-LZ-AC	5.9	28.2	≤ 202	≤ 100
SC-UX-AC	20.5	43.9	≤ 106	≤ 80
SC-UY-AC	7.0	10.5	≤ 596	≤ 74
SC-UZ-AC	30.4	53.3	≤ 126	≤ 82
SD-LX-AC	6.6	6.6	≤ 60	≤ 60
SD-LY-AC	9.8	25.5	≤ 122	≤ 72
SD-LZ-AC	23.4	38.4	≤ 60	≤ 60
SD-UX-AC	15.7	36.5	≤ 74	≤ 72
SD-UY-AC	8.7	21.5	≤ 88	≤ 84
SD-UZ-AC	31.9	48.7	≤ 82	≤ 80

2STP-36.2, Main Steam and Relief Valve Discharge Piping Vibration during SRV Operation

This subtest was performed in conjunction with STP-26.2, Relief Valve Rated Pressure Test in TC 2.

Data was recorded on the Plant Monitoring System (PMS) from remotely mounted sensors. Recorded data was processed as required, and evaluated against the applicable acceptance criteria limits.

As each Relief Valve was cycled at rated reactor pressure, transient vibration was recorded for main steam piping inside the drywell. Remotely mounted sensors, 24 lanyard potentiometers in total, were installed and monitored on the main steam lines. All transient vibration results obtained satisfied the applicable level 1 and 2 acceptance criteria.

In addition, during the cycling of Relief Valve "J", the dynamic vibratory response of the discharge piping was monitored. All transient vibration results were within acceptable design limits.

A walkdown of the affected piping following the transient confirmed that no signs of excessive movement were present.

The test results are summarized as follows:

Location	Measured Vibration (Mil)			Acceptance Criteria	
	SRV A	SRV E	SRV J	Level 1	Level 2
MSL A					
SA-UX-AC	20	19	41	≤656	≤324
SA-UY-AC	13	23	17	≤654	≤324
SA-UZ-AC	20	24	44	≤782	≤392
SA-LX-AC	9	9	8	≤64	≤60
SA-LY-AC	4	5	8	≤136	≤70
SA-LZ-AC	3	6	9	≤72	≤68

Location	Measured Vibration (Mil)				Acceptance Criteria	
	SRV B	SRV F	SRV K	SRV N	Level 1	Level 2
MSL B						
SB-UX-AC	9	22	6	16	≤378	≤190
SB-UY-AC	23	13	11	13	≤470	≤236
SB-UZ-AC	15	31	14	23	≤234	≤118
SB-LX-AC	7	6	7	6	≤66	≤60
SB-LY-AC	7	10	8	8	≤134	≤66
SB-LZ-AC	5	6	5	9	≤66	≤60

Location	Measured Vibration (Mil)			Acceptance Criteria	
	SRV C	SRV G	SRV L	Level 1	Level 2
MSL C					
SC-UX-AC	18	30	19	≤74	≤66
SC-UY-AC	9	11	8	≤102	≤86
SC-IZ-AC	28	56	25	≤126	≤102
SC-LX-AC	6	9	6	≤158	≤80
SC-LY-AC	8	12	5	≤1086	≤534
SC-LZ-AC	7	9	6	≤202	≤100

	Measured Vibration (Mil)				Acceptance Criteria	
Location	SRV D	SRV H	SRV M	SRV S	Level 1	Level 2
MSL D						
SD-UX-AC	24	31	21	21	≤116	≤116
SD-UY-AC	12	13	16	21	≤124	≤124
SD-UZ-AC	43	35	42	35	≤126	≤126
SD-LX-AC	6	6	7	7	≤60	≤60
SD-LY-AC	16	11	21	21	≤122	≤60
SD-LZ-AC	26	35	29	33	≤60	≤60

Location	Measured Maximum Transient Load (lb)	Acceptance Criteria
SRV J		
357	- 4412	± 7776
365	10282	±18758
380	4951	± 8174
381	4779	± 6375
390	1626	± 7829

2STP-36.3, Recirculation Piping Vibration during Selected Transients

This subtest provided the means for collecting vibration data for the recirculation piping for the following transients:

Event	Test Condition
Recirc Pump B Trip	3
Recirc Pump B Restart	3
Recirc Pump A Trip	6
Recirc Pump A Restart	6
Two Pump Trip	3
RHR A SDC Initiation & Shutdown	6
RHR B SDC Initiation & Shutdown	6

Data collection was accomplished using the Plant Monitoring System (PMS) and remote monitoring instrumentation (24 lanyard potentiometers and 3 resistance temperature devices). For all tests, vibration acceptance criteria were satisfied.

The test results are summarized as follows:

Recirculation Pump Trip and Restart

	Measured Vibration (Mil)					Acceptance Criteria	
Location	Trip A	Restart A	Trip B	Restart B	Trip Two Pumps	Level 1	Level 2
RA-SX-AC	1.7	1.7	1.7	1.7	2	≤68	≤68
RA-SY-AC	1.7	1.7	1.7	1.2	2	≤84	≤78
RA-SZ-AC	1.2	1.7	1.2	1.2	2	≤78	≤74
RA-PX-AC	1.7	4.0	1.7	1.7	3	≤68	≤64
RA-PY-AC	1.8	1.8	1.8	1.8	2	≤80	≤80
RA-PZ-AC	2.9	7.6	1.7	1.2	5	≤68	≤64
RA-DX-AC	3.5	5.2	1.2	1.2	10	≤424	≤212
RA-DY-AC	1.7	2.9	1.7	1.7	2	≤386	≤192
RA-DZ-AC	6.4	33.0	1.7	1.7	16	≤484	≤242
RA-RX-AC	3.5	5.8	1.7	1.7	6	≤160	≤80
RA-RY-AC	1.7	2.3	1.7	1.2	2	≤138	≤70
RA-RZ-AC	1.2	5.2	1.7	1.2	1	≤208	≤104
RB-SX-AC	1.7	1.2	1.2	1.7	2	≤70	≤70
RB-SY-AC	1.8	1.8	1.8	1.8	2	≤92	≤82
RB-SZ-AC	4.7	2.3	1.2	1.8	8	≤60	≤60
RB-PX-AC	2.3	1.2	1.2	1.2	8	≤146	≤74
RB-PY-AC	1.2	1.7	1.2	1.2	2	≤226	≤114
RB-PZ-AC	1.2	1.2	1.8	1.2	6	≤140	≤70
RB-DX-AC	7.6	5.8	5.2	5.8	11	≤498	≤250
RB-DY-AC	5.2	5.8	5.8	5.8	6	≤398	≤200
RB-DZ-AC	6.4	2.9	1.7	1.7	17	≤422	≤218
RB-RX-AC	1.2	0.6	0.6	0.6	3	≤248	≤124
RB-RY-AC	1.8	1.8	1.8	1.8	2	≤150	≤76
RB-RZ-AC	1.2	1.2	1.8	1.8	2	≤234	≤118

RHR Shutdown Cooling Initiation and Shutdown

Location	Measured Vibration (Mil) RHR A Start	Shutdown	RHR B Start	Shutdown	Acceptance Criteria Level 1	Level 2
RA-SX-AC	1.7	2	1	1.7	≤68	≤68
RA-SY-AC	1.7	2	2	1.7	≤84	≤78
RA-SZ-AC	1.2	0.6	1	1.2	≤78	≤74
RA-PX-AC	4.0	1	1	1.2	≤68	≤64
RA-PY-AC	2.9	1	1	1.2	≤80	≤80
RA-PZ-AC	4.1	2	1	1.2	≤68	≤64
RA-DX-AC	2.3	1	2	1.2	≤424	≤212
RA-DY-AC	4.7	2	1	1.7	≤386	≤192
RA-DZ-AC	3.5	1	2	1.2	≤484	≤242
RA-RX-AC	11.0	2	2	1.7	≤160	≤80
RA-RY-AC	3.5	3	1	1.7	≤138	≤70
RA-RZ-AC	4.0	3	2	1.7	≤208	≤104
RB-SX-AC	1.7	1	1	2.3	≤70	≤70
RB-SY-AC	2.3	2	2	6.4	≤92	≤82
RB-SZ-AC	13.4	2	4	2.3	≤60	≤60
RB-PX-AC	4.7	1	1	11.0	≤146	≤74
RB-PY-AC	1.7	1	1	1.7	≤226	≤114
RB-PZ-AC	2.9	1	1	9.4	≤140	≤70
RB-DX-AC	8.7	5	6	9.3	≤498	≤250
RB-DY-AC	6.9	5	5	5.2	≤398	≤200
RB-DZ-AC	6.4	2	5	13.0	≤422	≤210
RB-RX-AC	0.6	0.6	1	1.8	≤248	≤124
RB-RY-AC	1.2	0.6	1	0.6	≤150	≤76
RB-RZ-AC	1.8	2	2	1.8	≤234	≤118

2STP-36.4, HPCI Steam Supply Piping Vibration During HPCI Turbine Stop Valve Closure

This test was performed in conjunction with STP-15.2, Functional Demonstration and Controller Optimization at Rated Pressure, CST to CST, during Test Condition Heatup. Closure time of the HPCI turbine stop valve was also recorded for information.

The results of this test showed that the dynamic vibratory response of the HPCI steam supply piping during a stop valve closure was within acceptable design limits.

Data was recorded on the Plant Monitoring System (PMS) from remotely mounted vibration sensors prior to, and following, a trip of the HPCI turbine. Recorded data was processed as required, and evaluated against acceptance criteria limits.

A review of the compensated data indicated that four accelerometers yielded piping response measurements in excess of acceptance criteria limits. Acceptance criteria limits were conservatively established for the overall system based on an envelope of system weakness factors for all piping components within the scope of the test. For the four points in excess of this conservative value, new limits were calculated based on the specific piping configurations at those points. Test results were accepted, based on the revised limits.

A walkdown of the affected piping following the transient confirmed that no signs of excessive movement were present.

The test results are summarized as follows:

Data Point	Measured Value	Acceptance Criteria
217	-215 lb	±3996 lb
318	-267 lb	±4585 lb
200E	- 1.26 in/sec	± 1.01 in/sec
205E	1.01 in/sec	± 1.01 in/sec
150E	- 0.58 in/sec	± 1.07 in/sec
138	0.54 in/sec	± 1.07 in/sec
120A	1.00 in/sec	± 1.07 in/sec
105E	1.09 in/sec	± 1.07 in/sec
69	95 lb	±1000 lb
56	- 54 lb	±1000 lb
40	* 10.8 in/sec	± 1.07 in/sec
75E	- 1.59 in/sec	± 1.07 in/sec
80E	1.01 in/sec	± 1.07 in/sec
84A	- 1.00 in/sec	± 1.07 in/sec

* Value was corrected to 1.5 in/sec after correction of baseline drifting problem.

2STP-36.5, Feedwater Piping Vibration during Reactor Feedpump Trip/Coastdown

This subtest was performed in conjunction with STP-23.5, Reactor Feedpump Trip at Test Condition 6. "C" Feedpump was tripped with rated flow in each loop.

Data was recorded on the Plant Monitoring System (PMS) from remotely mounted sensors. Recorded data was processed as required, and compared with the acceptance criteria values.

One pressure sensor had an invalid signal. Since this signal was for information only and did not affect analysis of the test results, there was no impact on the test. All other data was within acceptable design limits.

A walkdown of the affected piping following the transient was performed. Based on the walkdown, as completed, and the analysis of the test data the dynamic loading was evaluated as acceptable.

The test results are summarized as follows:

Data Point	Measured Valve	Acceptance Criteria
64B	7.6 in/sec^2	<14.5 in/sec^2
96	- 7.1 in/sec^2	<14.5 in/sec^2
110B	5.5 in/sec^2	<14.5 in/sec^2
162B	- 5.4 in/sec^2	<14.5 in/sec^2
162B	-12.5 in/sec^2	<14.5 in/sec^2
65	0.9 in/sec	≤1.63 in/sec
145	1.11 in/sec	≤1.63 in/sec

4.35 2STP-70, REACTOR WATER CLEANUP SYSTEM

OBJECTIVES

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

ACCEPTANCE CRITERIA

Level 1

None

Level 2

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

The cooling water supplied to the non-regenerative heat exchangers shall be less than 6% above the flow corresponding to the heat exchanger capacity (as determined from the process diagram) and the existing temperature differential across the heat exchangers. The outlet temperature shall not exceed 180 Deg. F.

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

RESULTS

2STP-70.1, Blowdown Mode Performance Verification

The RWCU System was tested during Test Condition Heatup at rated temperature and pressure in the Blowdown Mode with two RWCU pumps running, and one RWCU NRHX group in service. The RWCU System was aligned to divert all flow to the main condenser and the system flow was then increased until more than 111.4 gpm (indicated flow rate) was obtained. The steady state RWCU NRHX outlet temperature was less than 130 Deg. F and the steady state NRHX RECW outlet temperature was less than 180 Deg. F when the system flow reached 148 gpm. The other NRHX was placed into service and testing repeated with similar results. All applicable acceptance criteria were satisfied.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
NRHX Outlet Temperature	10[illegible]	<130°F
Cooling Water Flow to NRHX	97%	<106% of Design Value
NRHX Cooling Water Outlet Temperature	157°F	<180°F

2STP-70.2, Hot Shutdown Mode Performance Verification

This test was deleted for Unit 2. The data required has been obtained by 2P-62.1, RWCU preoperational test.

2STP-70.3, Normal Mode Performance Verification

The RWCU System was tested in the Normal Mode with two RWCU pumps running, two Filter/Demineralizers (F/D's) in service, and RWCU NRHX B group in service. While maintaining balanced F/D flow, F/D flow was adjusted until more than 267 gpm (indicated flow rate) system flow was obtained. The steady state RWCU NRHX outlet temperature was less than 150 Deg. F when the RWCU System indicated flow rate reached 267 gpm.

The RWCU NRHX A group was placed in service and testing repeated. The indicated flow for NRHX A only reached a flow of 265 gpm versus the required flow of 267 gpm. This minor difference was accepted since process temperatures indicated that additional flow through the NRHX A would not violate any process temperature limits.

Vibration measurements were then taken on the A and B RWCU pumps - pump bearing housing vibration in the horizontal, vertical and axial directions and shaft vibration on the coupling end. All applicable acceptance criteria were satisfied for the RWCU NRHX A and B groups and the A and B RWCU pumps.

Vibration data and analysis for the C RWCU pump were performed at a later date when C RWCU pump was placed in service. All applicable acceptance criteria were satisfied.

The test results are summarized as follows:

Parameter	Measured Value	Acceptance Criteria
NRHX Outlet Temperature	$105^{\circ}F$	$<130^{\circ}F$
Cooling Water Flow to NRHX	89%	<106% Design Value
Pump Vibration	0.1-1.0 mil	<2 mils peak-to-peak

4.36 2STP-71, RESIDUAL HEAT REMOVAL SYSTEM

OBJECTIVES

The objectives of this test are to demonstrate the ability of the Residual Heat Removal (RHR) System to remove residual and decay heat from the nuclear system so that refueling and nuclear servicing can be performed. Additionally, this test will demonstrate the ability of the RHR System to remove heat from the suppression pool.

ACCEPTANCE CRITERIA

Level 1

The RHR System shall be capable of operating in the Suppression Pool Cooling Mode at the heat exchanger capacity specified.

Level 2

None

RESULTS

2STP-71.1, Suppression Pool Cooling Mode

The Residual Heat Removal (RHR) System was demonstrated for heat exchanger performance capacity in the suppression pool cooling mode at Test Condition Heatup. Inlet and outlet temperatures were recorded from the RHR system and RHR Service Water System streams every five minutes during a twenty minute duration test. Heat exchanger capacities for RHR loops A and B successfully met the Level 1 acceptance criteria.

As shown in the table below, the average heat removal rate for both heat exchangers were higher than the process diagram values. As a result, the actual performance of the heat exchangers is greater than the design performance.

	Measured Value		
Parameter	HXA	HXB	Acceptance Criteria
RHR System Heat Removal Rate (MBtu/hr)	70.5	50.6	≥ 26.0

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