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PENNSYLVANIA POWER AND LIGHT COMPANY

SUSQUEHANNA STEAM ELECTRIC STATION

UNIT NO. 2

STARTUP REPORT

BY.

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SUPERINTENDENT OF PLANT

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ACKNOWLEDGEMENT

In recognition to the many people who worked diligently to achieve successful completion of the Susquehanna Unit 2 Startup, I express my genuine appreciation. Although some groups were more directly involved than others, the entire Nuclear Department is to be congratulated for accomplishing the overall goal of testing Unit 2 in a safe and efficient manner, with the outcome being a plant that will operate safely and efficiently during it's commercial lifetime.

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SECTION 1

INTRODUCTION

1.1 REPORT ABSTRACT

This Startup Report, written to comply with Regulatory Guide 1.16 Revision 4, Section c.1.a, and Technical Specifications paragraph 6.9.1.1 thru 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 Susquehanna Steam Electric Station. It includes the events starting with initial fuel loading and ending with the completion of the Pre-Commercial Operations Outage. Since Susquehanna Unit 2 has not commenced commercial power operations, and since a few Startup Tests remain to be run, a supplementary report will be submitted later.

This 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 a brief abstract of each test procedure which also includes a test implementation matrix.

1.2 SUSQUEHANNA DESIGN PARAMETERS

The Susquehanna Steam Electric Station is a two unit nuclear power plant. The two units share a common control room, diesel generators, refueling floor, turbine operating deck, radwaste system, and other auxiliary systems. The 1075 acre plant site is located in Salem Township, Luzerne County, Pennsylvania, approximately 20 miles Southwest of Wilkes-Barre, 50 miles Northwest of Allentown and 70 miles Northeast of Harrisburg.

The Nuclear Steam Supply System for each unit consists of a General Electric Boiling Water Reactor, BWR/4 product line. The rated core thermal power for each unit is 3293 MWt. The corresponding net electrical output of each unit is 1050 MWe.

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 Contstructor was Bechtel Power Corporation.

The plant is owned and operated by The Pennsylvania Power and Light Company (90% ownership) and the Allegheny Electric Cooperative, Inc. (10%).

1.3 INITIAL TEST PROGRAM

The Initial Test Program encompasses the scope of events that commence with system/component turnover and terminate 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 power operation. Testing during the Initial Test Program is accomplished in five distinct and sequential phases:

a. Phase I - Component Inspection and Testing Phase

b. Phase II - Preoperational and Acceptance Testing Phase

c. Phase III - Initial Fuel Loading Phase

d. Phase IV - Initial Heatup and Low Power Testing Phase

e. Phase V - Power Ascension Test Phase

The Preoperational Test Program is defined as that part of the Initial Test Program that commences with system/component turnover and terminates with commencement of nuclear fuel loading. Component inspection and testing will insure that components and equipment are calibrated and checked, construction work on a particular system has been completed to the degree required, and the system is initially operated and prepared for subsequent testing. After component inspection and testing is complete on a system, formal tests denoted as preoperational or acceptance tests are conducted during the Preoperational and Acceptance Test phase. The Preoperational tests demonstrate, to the extent practicable, the capability of safety-related structures, systems, and components to meet their safety-related performance requirements. The completion of preoperational testing constitutes completion of Phase II of the Initial Test Program. Tests similar to preoperational tests denoted as acceptance tests may be conducted on additional non safety-related structures, systems, and components to demonstrate their capability to perform their non safety-related performance requirements.

The Startup Test Program is defined as that part of the Initial Test Program that commences with the start of nuclear fuel loading and terminates with the completion of power ascension testing. Formal tests, denoted as startup tests, are conducted during this program. These tests confirm the design bases and demonstrate, to the extent practicable, that the plant will operate in accordance with design and is capable of responding as designed to anticipated transients and postulated accidents. Startup testing is sequenced such that the safety of the plant is never totally dependent upon the performance of untested structures, systems, or components. The completion of startup testing constitutes completion of Phases III, IV, and V of the Initial Test Program.

1.4 STARTUP TEST PROGRAM SCOPE

The Susquehanna Startup Test Program was designed to comply with the requirements set forth in the following Regulatory Guides:

Reg. Guide 1.68 - Rev. 2

Reg. Guide 1.68.1 - Rev. 1

Reg. Guide 1.68.2 - Rev. 1

The acceptance criteria for the majority of the Startup Tests were based on General Electric supplied Startup Test Specifications, MPL Item Number A41-3610, Rev. 0, and modified by FDDR KR1-6013, Rev. 0, and FDDR-KR2-1041, Rev. 0.

The majority of additional testing concerned the thermal growth, steady state vibration and dynamic transient testing of ASME Section III Nuclear Class 1,2,3, and ANSI B31.1 piping. Specifications for this testing was supplied in Bechtel Power Corporations Specifications 8856-M-392, Rev. 14, 8856-M-393, Rev. 12, and 8856-M-394, Rev. 10.

The remaining testing was specified in various sections of the Final Safety Analysis Report, Rev. 35.

1.5 MAJOR STARTUP TEST PROGRAM ADMINISTRATIVE CONTROLS

Testing and power escalation was sequenced in six distinct Test Plateaus:

1. Test Plateau 0 - Open Vessel Testing
2. Test Plateau H - Heatup Testing
3. Test Plateau 1 - Test Condition 1 testing
4. Test Plateau 2 - Test Condition 2 testing
5. Test Plateau 3 - Test Condition 3 testing
6. Test Plateau 4 - 100% Rod Line Testing, which included:
   - Test Condition 5 testing
   - Test Condition 4 testing
   - Test Condition 6 testing

The definition of Test Condition is provided in Figure 1.5-1, sheets 1 and 2.

The final three phases of the Initial Test Program were comprised as follows:

Phase III - Initial Fuel Loading was comprised of Test Plateau 0

Phase IV - Initial Heatup and Low Power Testing was comprised of Test Plateau H

Phase V - Power Ascension Testing was comprised of Test Plateau 1 through 4

A Test Plateau Review is performed prior to escalating power above the maximum power associated with the current Test Plateau. The following items must be completed prior to the Test Plateau Review:

a. All Startup Tests scheduled for the current Test Plateau have been implemented, 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 test scheduled for the current Test Plateau have been resolved.

d. Quality Assurance has completed their review of the test and test results, or Test Exception Reports have been written to document and resolve exceptions.

A list of all tests approved to be run during a specific Test Plateau was contained in Startup Test 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 and assigned a required completion date relative to the different Test Plateaus.

Major modifications to the Startup Test Program as set forth in Section 14 of the FSAR could not be made without receiving prior NRC approval. Major modifications were defined as:

a. Elimination of any safety-related test.

b. Modifications of objectives, test methods or acceptance criteria for any safety-related test.

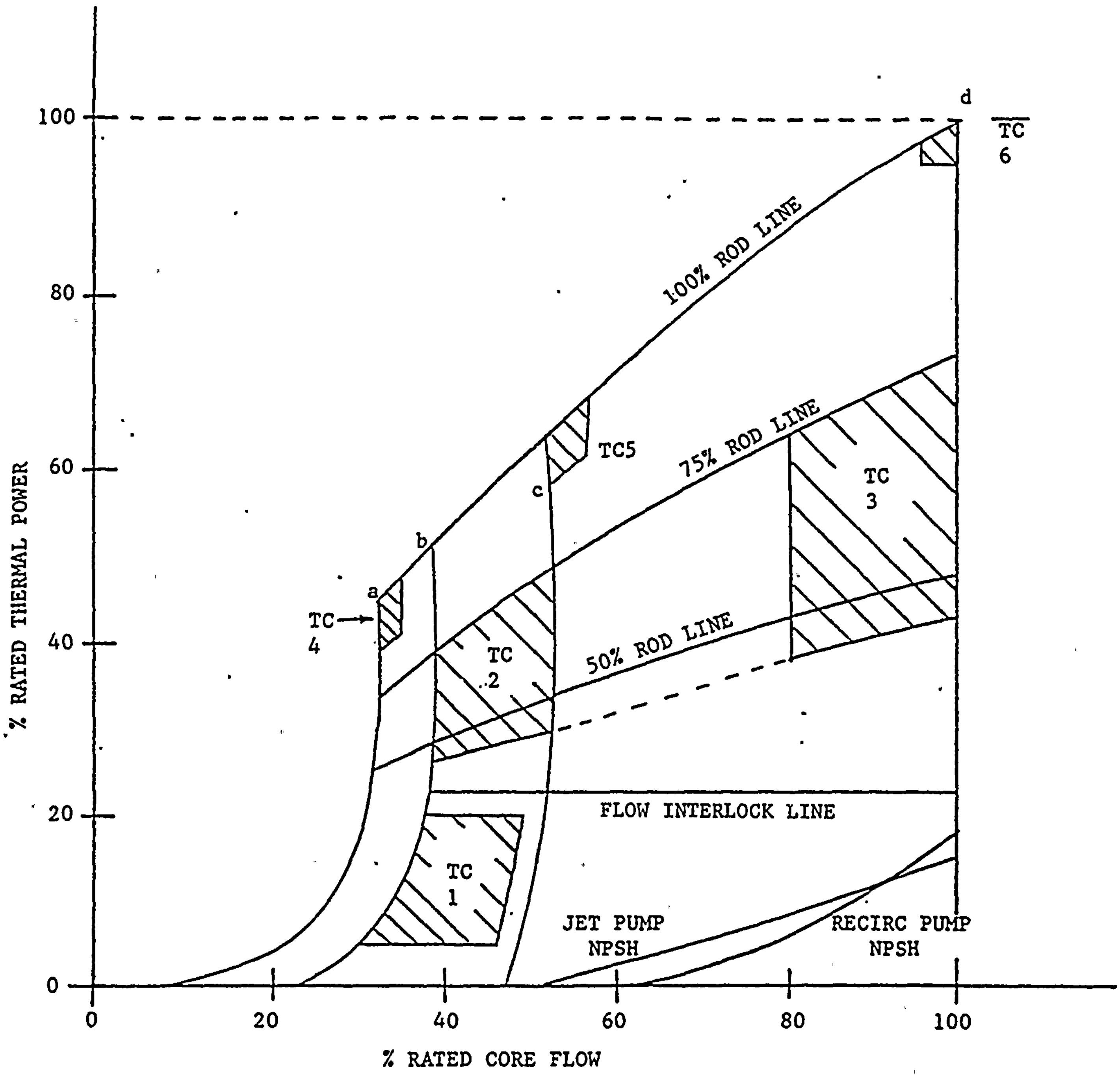
c. Performance of any safety-related test at a power level different from that stated in the FSAR by more than 5% of rated power.

d. Failure to satisfactorily complete the entire initial startup test program by the time core burnup equals 120 effective full power days.

e. Deviation from initial test program administrative procedures or quality assurance controls described in the FSAR.

f. Delays in the test program in excess of 30 days (14 days if power levels exceed 50 percent) concurrent with power operation.

The test program or individual test procedures could be made more restrictive or conservative without prior NRC approval.



NOTES

1. SEE FIGURE 1.5-1 SHEET 2 FOR DEFINITION OF TEST CONDITIONS
2. CONSTANT PUMP SPEED LINES:
   a. NATURAL CIRCULATION
   b. MINIMUM RECIRCULATION PUMP SPEED
   c. ANALYTICAL LOWER LIMIT OF MASTER FLOW CONTROL
   d. ANALYTICAL UPPER LIMIT OF MASTER FLOW CONTROL

FIGURE 1.5-1
SHEET 1 of 2
TEST CONDITIONS

Test Condition Number	Power-Flow Map Region and Notes
1	Core thermal power between approximately 5% and 20% rated. Recirculation pump speed within +10% of minimum pump speed. Before and after main generator synchronization.
2	Core thermal power between the 45% power rod line and 75% power rod line. Recirculation pump speed between minimum and lowest pump speed corresponding to Master Manual Mode. Lower power corner is within Turbine Bypass Valve capacity.
3	Core thermal power between 45% power rod line and 75% power rod line. Total core flow between 80% and 100% rated.
4	On the natural circulation core flow line within +0, -5% of the intersection with the 100% power rod line.
5	Core thermal power within +0, -5% of the 100% power rod line. Recirculation pump speed within +5% of the minimum recirculaiton pump speed corresponding to Master Manual Mode.
6	Core thermal power between 95% and 100% rated. Total Core flow +0, -5% rated core flow.

All testing is assigned to a specific Test Condition for convenience even though some testing, as described in the abstracts, is performed outside the bounds of the assigned Test Condition.

FIGURE 1.5 - 1
Sheet 2 of 2
Test Conditions

1.6 STARTUP TEST PROGRAM ORGANIZATION

The Susquehanna Startup Team was established as a Plant Staff Technical Group sub-section to prepare for and conduct the Startup Test Program on Susquehanna Unit 2. The Susquehanna Startup Team combined the on-shift testing and coordination roles performed by the Shift Startup Team and the off-shift technical, administrative and planning roles of the Startup Test Group into one group.

The Shift Startup Team was responsible for all on-shift facets of the Startup Test Program. This included test implementation, pre-test preparations, on-shift test related activities coordination, test results data compilation, analysis and independent review, and evaluating and responding to all testing restraints.

There were five Shift Startup Teams, each one assigned to a specific plant operations crew. Each team was composed of five members; two from the Plant Staff Technical Group, two from General Electric, and one from Nuclear Plant Engineering. All members of the Shift Startup Teams were qualified as Startup Testing Personnel per ANSI/ANS-3.1-1981. The Shift Startup Teams were augmented by qualified test directors from the Reactor Engineering, Chemistry and Health Physics groups as needed for specific specialized tests.

One member of each Shift Startup Team was designated as the Shift Test Engineer. This individual was responsible for all on-shift related activities and provided single-point contact with the operations Shift Supervisor. The Shift Test Engineer designated the Test Director and assistants for the preparation, conduct, data gathering, monitoring, analysis and review of each Startup Test.

The Test Director was responsible for briefing test personnel, ensuring that the test was performed in accordance with the approved procedure, and verifying that all Level 1 Acceptance Criteria were satisfied.

The Startup Test Group was responsible for the administrative control of the program, test procedure preparation and issuance, test results administrative review, test exception resolution coordination, test documentation, report preparation and interfacing with the NRC on Startup Testing related items.

The Test Review Committee (TRC) was established as a Plant Operations Review Committee (PORC) subcommittee to perform detailed reviews of test procedures, changes, exceptions, results and Test Plateau escalations, and recommend approval of these items as appropriate. The TRC's proceedings and recommendations were reviewed by the PORC along with their review of the above items. The Superintendent of Plant was responsible for approval of these items. The TRC was comprised of the Operations Supervisor (Chairman), Startup Test Group Supervisor, Reactor Engineering Supervisor, Nuclear Plant Engineering Group Supervisor and NSSS and AE representatives.

General Electric provided members for the Shift Startup Teams, provided Technical Direction to operations and other test personnel, and participated as a voting member of TRC on NSSS Startup Test activities.

Bechtel personnel provided Technical Direction for portions of the piping tests and participated as a voting member of TRC on non-NSSS Startup Test activities.

Nuclear Plant Engineering provided members for the Shift Startup Teams, provided the formal interface between the Susquehanna Startup Team and the technical branches of General Electric and the Bechtel Power Corporation, provided technical resolutions to all Test Exception Reports and participated as a voting member of TRC.

Nuclear Quality Assurance, in addition to their normal surveillance and audit activities, were responsible for reviewing all tests and test results.

SECTION 2

SUMMARY

2.1 OVERALL EVALUATION

The Susquehanna Unit 2 Startup Test Program has been successful to date. The Startup Test Program commenced with fuel loading on March 28, 1984. An extended pre-commercial outage was entered on October 27, 1984. As of that date, the Startup Test Program was 86% complete.

All testing identified in Chapter 14 of the FSAR has been completed with the exception of the tests listed below which will be run following completion of the outage:

- -ST 27.2, "High Power Generator Load Reject" and associated tests:
  - -ST 5.7, "Scram Timing of Selected Rods During Planned Scrams of Startup Test Program"
  - -ST 32.3, "Containment Temperature After Reactor Scram"
  - -ST 39.1, "Main Steam Piping Vibratory Response During Turbine Stop Valve Closure"
- -ST 8.3, "(RHR) Shutdown Cooling Mode" and associated tests:
  - -ST 17.7, "RHR System Piping (Expansion) - Shutdown Cooling"
  - -ST 33.3, "Steady State Vibration, Recirculation Piping"
- -ST 30.1, "Recirculation System One Pump Trip" and associated tests:
  - -ST 16.2, "Recirculation Pump Trip Recovery Data"
  - -ST 39.3, "Recirculation Piping Vibratory Response during Pump Trips and Restarts"
- -ST 23.5, "Feedwater Pump Trip"
- -ST 24.1, "Turbine Stop, Control, Combined Intermediate and Bypass Valve Testing"
- -ST 37.2, "Containment Inerting"

The following is a list of open Startup Test Program items as identified in open Test Exception Reports:

1. Adjust containment cooling system and retest to determine if better temperatures can be obtained in vessel skirt area.
2. Inspect offgas system, make adjustments and retest to determine if better guard bed flows and dewpoint temperatures can be obtained.
3. Retest two LPRMs which were bypassed during ST11.
4. Implement Environmental Upgrade Modification and retest HPCI after component changes are made.
5. Collect additional Radiation Base Point readings at 100% power and resolve outstanding test exceptions.

2.2 SUMMARY OF KEY EVENTS

Mar. 23, 1984	Received Low Power Operating License
Mar. 28, 1984	Commenced Fuel Loading
Apr. 13, 1984	Completed Fuel Loading
Apr. 19, 1984	Completed Plateau Review for Test Plateau 0, Open Vessel Testing
May 8, 1984	Initial Criticality
May 21, 1984	Initially reached rated reactor pressure and temperature conditions
May 28, 1984	Started RHR Outage
June 10, 1984	Ended RHR Outage
June 27, 1984	Received Full Power Operating License
June 28, 1984	Completed Plateau Review for Test Plateau H, Heatup Testing
July 1, 1984	Initial Main Turbine Roll
July 3, 1984	Initial Generator Synchronization
July 6, 1984	Completed Plateau Review for Test Plateau 1
July 26, 1984	Station Blackout Unusual Event
Aug. 7, 1984	Successful retest of Loss of Offsite Power Test
Aug. 8, 1984	Completed Plateau Review for Test Plateau 2
Sep. 9, 1984	Completed 50% Power Testing
Sep. 21, 1984	Completed Plateau Review for Test Plateau 3
Sep. 25, 1984	Completed Test Condition 5 testing
Sep. 28, 1984	Initial 100% Power Operations
Oct. 10, 1984	Performed Natural Circulation Testing (Test Condition 4)
Oct. 27, 1984	Started Pre-Commercial Operations Outage
Jan. 8, 1985	Ended Pre-Commercial Operations Outage

2.3 STARTUP TEST PROGRAM CHRONOLOGY

Mar. 20, 1984	Commenced first Startup Test, ST5.1, "CRD Insert-Withdrawal Checks"
Mar. 23, 1984	Received Low Power Operating License from NRC
Mar. 24, 1984	Loaded first neutron source into core.
	Detected piece of plastic pinched under the fuel support piece in the vicinity of the first neutron source holder. Suspended neutron source loading activities.
Mar. 25, 1984	Transferred neutron source from core to source storage rack.
Mar. 27, 1984	Retrieved plastic piece using mechanical device. Performed vessel cleanliness verification using underwater diver.
Mar. 28, 1984	Loaded neutron sources into core.
	Commenced Fuel Loading at 9:29 p.m.
Apr. 5, 1984	Experienced first "RPS Trip" due to faulty IRM "D" pre-amplifier.
Apr. 13, 1984	Last fuel bundle loaded at 12:31 a.m.
Apr. 15, 1984	Vessel cavity drained to about two feet below vessel flange
Apr. 17, 1984	Vessel assembly completed except for final head tensioning pass.
Apr. 18, 1984	Secondary containment restored
Apr. 19, 1984	Final head tensioning pass completed.
	Entered Operational Condition 4
	Completed Plateau Review for Test Plateau 0, Open Vessel Testing (Initial Test Program Phase III - Initial Fuel Loading)
Apr. 20, 1984	Commenced Operational Hydrostatic Test
Apr. 23, 1984	Completed Operational Hydrostatic Test
	Progress from April 23rd to May 2nd was hampered due to the rework of RHR valve 2F015A to reduce leakage to acceptable limits.

May 8, 1984	Entered Operational Condition 2
	Commenced reactor startup at 7:20 p.m.
	Initial Criticality achieved at 9:40 p.m.
May 10, 1984	Heated reactor to 275°F.
	Inspected drywell piping
May 11, 1984	Increased reactor pressure to 110 psig
	Progress between May 11th and May 17th hampered by HPCI flow problems caused by undersized flow orifice.
May 17, 1984	Increased reactor conditions to 450°F. at 411 psig
	Inspected drywell piping
	Progress between May 17th and May 20th hampered by problems with the CRD and "A" RFP control systems.
May 20, 1984	Increased reactor pressure to 600 psig
	Performed scram timing testing of selected CRDs
May 21, 1984	Increased reactor pressure to 800 psig
	Performed scram timing testing of selected CRDs
	Initially reached rated reactor pressure and temperature conditions
	Entered 7-day LCO due to dual indication on RHR valve 2F050B.
May 28, 1984	Commenced reactor shutdown to enter a RHR outage to correct RHR problems
	SCRAM #1. Experienced irregular main turbine bypass valve operation resulting in a power excursion and the #1 bypass valve being stuck open. Manually scrammed reactor. Upon disassembly of the valve, a chipping hammer was found between the valve's disc and seat.
June 10, 1984	Commenced reactor startup
June 11, 1984	Manually shutdown reactor to repair suppression pool vacuum breaker limit switches
June 12, 1984	Restart reactor
June 20, 1984	Reduced reactor pressure to 500 psig and closed MSIVs in order to replace RCIC lube oil drain line.

June 23, 1984	Returned reactor to rated pressure and temperature condition
June 27, 1984	Received Full Power Operating License
June 28, 1984	Completed Plateau Review for Test Plateau H, Heatup Testing (Initial Test Program Phase IV - Initial Heatup)
	Commenced Initial Test Program Phase V - Power Ascension Testing
	Entered Operational Condition 1
	Increased reactor power to 10% rated
June 29, 1984	Increased reactor power to 15% rated
	Started main turbine shell warming
July 1, 1984	Completed initial main turbine roll to 100 rpm
July 2, 1984	Completed initial main turbine roll to 1800 rpm
July 3, 1984	Main generator initially synchronized to electrical grid
July 4, 1984	Initially reached 20% rated power
July 5, 1984	SCRAM #2. Manually scrammed reactor from control room as part of ST28.1, Shutdown From Outside The Main Control Room. The plant was controlled and placed into cold shutdown from the Remote Shutdown Panel. Also ran ST28.2, Reactor Scram From Outside The Main Control Room, to demonstrate that a scram and MSIV isolation could be performed from outside the main control room.
July 6, 1984	Restarted reactor
	Completed Plateau Review for Test Plateau 1, Test Condition 1 testing
July 9, 1984	Performed ST27.3, Generator Load Reject Within Bypass Capacity
July 10, 1984	Initially reached 40% power
July 15, 1984	SCRAM #3 (unplanned SCRAM #2). Unit 2 was using the common hydrogen recombiner since the Unit 2 recombiner was out of service for maintenance. A scram on Unit 1 resulted in a loss of power to the common recombiner and subsequent loss of Unit 2 condenser vacuum and scram.
July 17, 1984	Restarted reactor

July 26, 1984 SCRAM #4. Performed ST31.1, Loss of Turbine-Generator and Offsite Power. Misalignment of DC knife switches on 4 Kv ESS buses resulted in a station blackout on Unit 2 and the declaration of an Unusual Event.

Aug. 1, 1984 Restarted reactor

Aug. 7, 1984 SCRAM #5. Successful retest of ST31.1

Aug. 8, 1984 Completed Plateau Review for Test Plateau 2, Test Condition 2 testing

Restarted reactor

Aug. 10, 1984 Initially reached 50% rated power

Reduced reactor power and closed MSIVs to install gaskets on main turbine steam piping

Aug. 18, 1984 Resumed testing at 50% rated power

Progress from July 26th through Aug. 18th was minimal due to station blackout resolution activities, RCIC EGR changeout and associated problems, water in the HPCI lube oil problem, RHR valve problems, and problems associated with the lack of gaskets on the main turbine steam lines.

Aug. 19, 1984 Initial operation at 100% core flow

Aug. 26, 1984 SCRAM #6. (unplanned SCRAM #3) caused by high water level in the "B" moisture seperator during surveillance testing on the main turbine CIV #4.

Restart reactor

Aug. 28, 1984 SCRAM #7 (unplanned SCRAM #4) Same as SCRAM #6

Sep. 2, 1984 Restarted reactor

Sep. 8, 1984 SCRAM #8 (unplanned SCRAM #5). Main turbine tripped during weekly check of turbine output mismatch logic caused by pressure transmitter being out of calibration.

Restarted reactor

Sep. 9, 1984 Reduced power by inserting rods while maintaining 100% core flow during performance of ST30.4, Recirculation System Limitor Verification

Completed 50% power testing

Sep. 10, 1984 Initially reached 75% rated power

Sep. 20, 1984 SCRAM #9. Performed ST27.1, Turbine Trip, from 75% power

Sep. 21, 1984	Completed Plateau Review for Test Plateau 3, Test Condition 3 testing
	Restarted reactor
Sep. 22, 1984	Retested "A" Recirculation Pump along 75% rod line
Sep. 25, 1984	Initial 100% rod line operation
	Completed Test Condition 5 testing
Sep. 26, 1984	Initially reached 80% rated power
Sep. 27, 1984	Initially reached 95% rated power
Sep. 28, 1984	Initially reached 100% rated power
Sep. 30, 1984	SCRAM #10 (unplanned SCRAM #6) caused by high water level in the"B" moisture seperator during runout testing of the "B" RFP per ST23.6
Oct. 1, 1984	Restarted reactor
	Progress after the September 30th scram was hampered due to the additional testing required to resolve problems on the moisture seperator.
Oct. 10, 1984	Performed Natural Circulation testing (Test Condition 4)
	Progress after October 10th was further hampered by HPCI oil pressure and vibration problems and "C" RFP bearing oil leaks.
Oct. 13, 1984	Manually shutdown reactor in order to change out CRD pilot scram valve disc holder sub-assemblies.
Oct. 17, 1984	Restarted reactor
Oct. 27, 1984	SCRAM #11. Performed ST25.3, MSIV Full Isolation, from 100% power.
	Started Pre-Commercial Operations Outage
Jan. 5, 1985	Restarted reactor
Jan. 8, 1985	Synchronized generator, thus ending Pre-Commercial Operations Outage

SECTION 3

STARTUP TEST PROCEDURES

3.1 STARTUP TEST PROCEDURE FORMAT AND CONTENT

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

The Main Body of a Startup Test establishes the overall Test Objectives and Acceptance Criteria for the associated Subtests. The Main Body consists of at least the following sections:

1. Test Objectives
2. Test 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, data acquisition, analysis of test results, and verification of acceptance criteria satisfaction. A Subtest consists of at least the following sections:

1. Discussion
2. Prerequisites
3. Initial Status
4. Test Instructions
5. Subsequent Actions
6. Group A Analysis
7. Group B Analysis
8. Appendices (optional)

A Startup Test 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 identical Subtest was performed more than once, provisions were made to identify plant conditions at which the Subtest was implemented. Startup test sections are written and laid out in such a manner that individual Subtests or the Main Body, including appendices to each, if any, can be removed and used independently of other sections.

Startup Tests are arbitrarily numbered 1 thru 99 - the number does not indicate sequence of implementation nor are all numbers necessarily used. Startup Test Sections (Main Bodies and Subtests) are numbered zero thru 99. The Main Body is always section zero. Subtests are arbitrarily numbered 1 thru 99 - the number does not indicate sequence of implementation nor are all numbers necessarily used, nor are all numbers used necessarilly successive.

Each Startup Test Section is considered as an individual procedure and thus is controlled independently of each other.

Acceptance criteria may be either quantitative or qualitative. Quantitative acceptance criteria specify test or equipment design values in accordance with design requirements (FSAR, equipment specifications, test specifications, etc.). These criteria state design 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 design functions (an event does or does not occur), such as automatic start, sequencing, or shutdown occuring under specified conditions.

Acceptance criteria are categorized into Level 1 and Level 2. 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 was not satisfied, the plant was placed in a suitable hold-condition until resolution was obtained. Tests compatible with this hold-condition were continued. Following resolution, applicable tests were repeated to verify that the requirements of the Level 1 criterion were now satisfied. A Level 2 criterion is associated with expectations relating to the performance of systems. If a Level 2 criterion was not satisfied, operating and testing plans were not necessarily altered. Investigations of the measurements and analytical techniques used for the predictions were started.

3.2 STARTUP TEST PROCEDURE ABSTRACTS

The abstracts on the following pages provide general information on the content of each Startup Subtest. The information given for the "zero" sections (i.e. 1.0) provide general objectives of the entire Startup Test. These abstracts in no way modify or replace the abstracts contained in Section 14 of the Final Safety Analysis Report. The letters and numbers OH123456 listed under the Test Conditions column indicate each Test Condition in which the Startup Subtest was run. For some subtests, additional implementation information is provided in the description.

STARTUP TEST ABSTRACTS

12/19/84

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
1.0		CHEMICAL AND RADIOCHEMICAL DATA The objective of this test is to demonstrate that the chemistry of all parts of the entire reactor system meets specifications and process requirements.
1.1	0	Chemistry Data - Pre Fuel Load This test consists of conducting specific chemical analyses on water samples drawn from Reactor Water Cleanup influent and Fuel Pool Cooling and Cleanup influent within 24 hours of starting fuel loading.
1.5	H	Chemistry Data-Pre Heatup This test consists of conducting specific chemical analysis on water samples drawn from Reactor Water Cleanup influent and the Control Rod Drive System within 24 hours of pulling the first rod.
1.6	H	Chemistry Data-Heatup Tests This test consists of conducting specific chemical and radiochemical analyses on water samples drawn from Reactor Water Cleanup influent and the Feedwater and CRD systems while the reactor is at rated pressure prior to exceeding 5% power.
1.7	123 56 Refer to description for details	Chemistry Data-Power Ascension Tests This test is similar to ST 1.6 except that it is done at various power levels during power ascension. During TC 3, the test is done at both 50% and 75% power.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
2.0		RADIATION MEASUREMENTS The objectives of this test are to determine the background radiation levels in the plant prior to operation for baseline data on activity buildup and to monitor radiation at selected power levels to ensure the protection of personnel during plant operation.
2.1	OH1 3 6 Refer to description for details	Startup Test Program Radiation Surveying A radiation survey is conducted prior to initial fuel loading, upon initially reaching rated reactor pressure and temperature, upon initial generator synchronization and during TC 3 and TC 6.
3.0		FUEL LOADING The objective of this test is to achieve the full and proper core complement of nuclear fuel assemblies through a safe and efficient fuel loading evolution.
3.1	0	Preparation and Installation of Neutron Sources and Fuel Loading Chambers This test prescribes the steps necessary to install all seven neutron sources and four fuel loading chambers into their initial position prior to beginning fuel loading.

STARTUP TEST ABSTRACTS

12/19/84

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
3.3	0	Fuel Loading During this test, the entire core compliment of fuel assemblies is moved from the fuel pool to the reactor core. Movement is governed by the Fuel And Core Component Transfer Authorization Sheet (FACCTAS). Partial core shutdown margin is also demonstrated during this test. ST 5.1, CRD-Insert Withdrawal Checks, is performed in conjunction with this test.
3.4	0	Core Verification After fuel loading, a verification of the location and orientation of each fuel bundle is made and reviewed to document correct loading.
4.0		FULL CORE SHUTDOWN MARGIN The objective of this test is to demonstrate that the reactor will be subcritical throughout the first fuel cycle with any single control rod fully withdrawn.
4.1	H	In-Sequence Critical Refer to description of ST 4.0.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
5.0		CONTROL ROD DRIVE SYSTEM The objective of this test is to demonstrate that the Control Rod Drive System operates properly and throughout the full range of primary coolant operating temperature and pressure and to determine the initial operating characterisitcs of the CRD system.
5.1	OH 2 Refer to description for details	Insert - Withdrawal Checks This test performs several functional tests for each control rod including: Insertion and withdrawal stroke time, drive water running and stall flow rate, rod position indication system operation, and control rod drive coupling. These checks are made prior to initial fuel loading and after fuel is loaded around each control rod. This test was also performed during normal reactor operating temperature and pressure conditions.
5.2	OH	Friction Measurements This test measures the differential pressure between drive water insert and withdrawal lines and continuous insertion of each CRD to determine if dynamic friction is within acceptable limits. This test is done at both zero and rated reactor pressure.
5.3	OH	Zero and Rated Pressure Scram of Individual Rods Each CRD is withdrawn, scrammed and timed at zero and rated reactor pressure.

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ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
5.5	OH	Scram Testing of Selected Rods
	Refer to description for details	During this test, the four slowest CRD's are scrammed and timed at various combinations of reactor and accumulator pressures. This ensures that any performance deterioration caused by heatup will be promptly discovered. The test is conducted at zero reactor pressure with accumulator pressure slightly above their low pressure alarm point, at 600 and 800 psig reactor pressure with accumulators normally charged, and at rated reactor pressure with zero accumulator pressure.
5.6	H	Insert-Withdrawal Checks of Selected Rods
		This test measures the time for insertion and withdrawal of the four slowest CRD's during normal reactor operating temperature and pressure conditions.
5.7	1 3 6	Scram Timing of Selected Rods During Planned Scrams of Startup Test Program
		The four slowest CRD's are timed during full scrams at various power levels during the Startup Test Program to determine the response characteristics of the CRD System during power operation and to demonstrate that no significant change has occurred between cold and power operating conditions.
5.8	2	Post Scram Differential Pressure Measurements
		This test functionally verifies the correct operation of the CRD Hydraulic System equalizing valves.

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ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
7.0		REACTOR WATER CLEANUP SYSTEM The objective of this test is to demonstrate specific aspects of the mechanical operability of the Reactor Water Cleanup System.
7.1	H	Blowdown Mode Performance Verification In this test, the Reactor Water Cleanup System will be operated in the Blowdown Mode with maximum cooling water flow through the non-regenerative heat exchangers. This test verifies the heat removing capabilities of both the regenerative and non-regenerative heat exchangers.
7.2	4	Hot Shutdown Mode Performance Verification In this test, the Reactor Water Cleanup System will be operated in the Hot Shutdown Mode with the Reactor Recirculation Pumps off. This test provides the Reactor Water Cleanup Pumps with the minimum net positive suction heads.
7.3	H	Normal Mode Performance Verification This test demonstrates that system design flow and temperatures can be met during the Normal Mode with cooling water temperatures within design limits.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
7.4	H	Calibration Verification of Reactor Bottom Head Flow Indicator In this test, all Reactor Water Cleanup System flow is directed through the bottom head drain to verify the operation of the bottom head flow indicator FI-2R610. Data is collected at various drain flow rates and compared with total system flow as read by FI-2R609.
7.5	3	Initial Drain Line Temperature Data Data is recorded on the bottom drain line temperature sensor and compared with recirculation loop coolant temperature to determine operability of the bottom drain line sensor.
8.0		RESIDUAL HEAT REMOVAL SYSTEM The objectives of this test are to demonstrate the Residual Heat Removal (RHR) System's ability to remove residual heat from the reactor pressure vessel and the suppression pool and operate in the suppression pool cooling mode, steam condensing mode and shutdown cooling mode.
8.1	H	Suppression Pool Cooling Mode During this test, each loop of the RHR system is placed in the Suppression Pool Cooling Mode to verify proper system operation and heat exchanger capacities. Since this test requires a relatively high temperature difference between RHR Service Water and the suppression pool, it may be done in conjunction with ST 26.2, "Relief Valve Rated Pressure Test".

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ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
8.2	6	<u>Steam Condensing Mode</u> This test is performed when the reactor is at power. During this test, the RHR loops are placed in the Steam Condensing Mode both singly and in combination to verify proper system operation and heat exchanger capacities.
8.3	H 6	<u>Shutdown Cooling Mode</u> This test is performed after a major trip when the reactor is at the required reduced pressure. During this test, the RHR loops are placed in the Shutdown Cooling Mode both singly and in combination to verify proper system operation and heat exchanger capacities.
8.4	6	<u>Steam Condensing Mode Stability Test</u> This test demonstrates the stability of the controllers used in the Steam Condensing Mode.
9.0		<u>WATER LEVEL MEASUREMENTS</u> The objective of this test is to determine actual reference leg temperature and to verify the correct reference leg temperature was used for calibration of the narrow range and wide range level instrumentation.
9.1	H	<u>Water Level Data Comparison</u> Refer to description in ST 9.0.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
10.0		IRM PERFORMANCE The objectives of this test are to demonstrate that the operational sources, SRM and IRM instrumentation and rod withdrawal sequences provide adequate information to achieve criticality and increase power in a safe and efficient manner for each of the specified rod withdrawal sequences and to adjust the Intermediate Range Monitor System as necessary to obtain the desired overlap with the SRM and APRM systems.
10.1	H	IRM - SRM Overlap Verification Refer to description in ST 10.0.
10.2	12	IRM - APRM Overlap Verification Refer to description in ST 10.0.
10.3	H.	SRM Signal to Noise Ratio and Minimum Count Rate Determination Refer to description in ST 10.0.
11.0		LPRM CALIBRATION The objective of this test is to calibrate the Local Power Range Monitoring (LPRM) System.
11.2	1	LPRM Calibration Without Process Computer In this test, the initial LPRM calibration is done using displayed data and the off-line computer program BUCLE for calculations.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
11.3	23 6	LPRM Calibration With Process Computer This test uses the process computer to supply the data needed and to perform an LPRM Calibration.
12.0		APRM CALIBRATION The objective of this test is to calibrate the Average Power Range Monitor (APRM) System.
12.1	H	Low Power APRM Calibration In this test, the reactor core.thermal power is determined based on the reactor recirculation pump suction water heatup at a constant rate and negligible steam flow from the vessel. APRM's are then adjusted as necessary.
12.2	123 56	High Power APRM Calibration In this test, core thermal power is determined by a core heat balance. APRM's are then adjusted as necessary. During TC 3, the test is done at both 50% and 75% power.
13.0		PROCESS COMPUTER The objective of this test is to verify the NSSS performance of the process computer under plant operating conditions.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
13.1	2	Dynamic Systems Test Case This test deals primarily with dynamic testing and verification of NSSS process computer programming, data storage and retrieval, array initialization, scan and alarms interfacing, and subroutine calling. After the successful completion of this test, the following programs will be considered operational: P-1, P-2, P-3, P-5, OD-1, OD-3, OD-7, OD-8, and OD-15.
13.2	3	Specified LPRM Substitute Value and BASE Distribution This test verifies that the new TIP Data and the BASE Values are properly calculated and stored after an OD-2 is performed.
13.3	3	Bundle Power Symmetry This test verifies the proper performance of the symmetry flags of the NSSS computer software.
14.0		REACTOR CORE ISOLATION COOLING SYSTEM The objectives of this test are to verify the proper operation of the Reactor Core Isolation Cooling (RCIC) System at the minimum and rated operating pressures and flow ranges, and to demonstrate system reliability in automatic starting from cold standby when the reactor is at power condition.

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ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
14.1	H 23	Condensate Storage Tank Injection During this test, RCIC is operated at 150 psig and rated reactor pressure while discharging to the condensate storage tank.
14.2	H 2	Reactor Vessel Injection In this test, RCIC is operated at rated reactor pressure and discharges into the reactor vessel.
14.3	23	Rated Pressure Auto Quick Start to Vessel This test demonstrates the auto quick start capability of the RCIC System with reactor at rated pressure and the RCIC turbine and pump cold. This test must be satisfactorily completed twice in succession.
14.4	H 3	Low Pressure Auto Quick Start to Vessel This test demonstrates the auto quick start capability of the RCIC System with reactor at 150 psig.
15.0		HIGH PRESSURE COOLANT INJECTION SYSTEM The objectives of this test are to verify the proper operation of the High Pressure Coolant Injection (HPCI) System at the minimum and rated operating pressures and flow ranges, and to demonstrate system reliability in automatic starting from cold standby when the reactor is at power condition.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
15.1	H 23	Condensate Storage Tank Injection During this test, HPCI is operated with the reactor vessel at 150 psig and at rated pressure with pump discharge to the condensate storage tank.
15.2	3 6	Reactor Vessel Injections, Rated Pressure In this test, HPCI is operated at rated reactor pressure discharging to the reactor vessel.
15.3	6	Rated Pressure Auto Quick Starts to Vessel This test demonstrates the auto quick start capability of the HPCI System which reactor at rated pressure and the HPCI turbine and pump cold. This test must be satisfactorily completed twice in succession.
16.0		SELECTED PROCESS TEMPERATURES The objective of this test is to determine the proper setting of the low flow control limiter for the recirculation pump and to obtain reactor pressure vessel bottom head region temperature data during recirculation pump trip and restart.
16.1	H	Minimum Recirculation Pump Speed Determination In this test, bottom head temperatures are monitored during a gradual decrease of the recirculation flow to determine if stratification occurs prior to reaching the lower speed limiter.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
16.2	3 6	Recirculation Pump Trip Recovery Data In this test, temperature data for the reactor pressure vessel bottom head region will be recorded at 10 minute intervals following planned reactor recirculation pump trips and restarts.
17.0		SYSTEM EXPANSION The objectives of this test are to verify that system piping during heatup and cooldown is free to move without unplanned obstruction or restraint, that system piping behaves in a manner consistent with assumptions of the stress analyses, and that there is agreement between calculated and measured values of displacement. All tests collect and analyze data on the following systems unless otherwise noted: Reactor Recirculation, Main Steam, Residual Heat Removal, Core Spray, Reactor Water Cleanup, High Pressure Coolant Injection, Reactor Core Isolation Cooling, and Feedwater. Remote instrumentation is used for portions of systems which are inaccessible during testing. Remainder of testing is done using local instrumentation or visual inspection.
17.1	H	Base Condition Data Collection This test is done prior to the initial heatup of system piping. Both visual and remotely monitored data is taken during this test.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
17.2	H	Intermediate and Rated Temperature Data Collection
	Refer to description for details	AT 275°F, 450°F and rated temperature, a visual inspection of all piping scoped for testing per the FSAR except Feedwater is made using this Subtest. Remote instrumentation is also used to verify piping inside containment and main steam piping outside of containment.
17.4	23	Feedwater at Normal Operating Temperature Data Collection
	Refer to description for details	This test collects and analyzes data for the Feedwater system only. Both visual inspection and remote instrumentation verification of Feedwater piping outside of containment is performed. This test is run at Feedwater temperatures of 260°F and 387°F.
17.5	6	Post Thermal Cycle Data Collection
		Piping systems are analyzed to verify that subsequent relaxing of piping systems after a heat-up/cooldown thermal cycle is as expected. Both visual and remotely monitored data is taken during this test.
17.7	2 6	RHR System Piping-Shutdown Cooling
		This test is a visual inspection of RHR system piping while the system is in the Shutdown Cooling mode.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
17.8	2 6	RHR System Piping-Steam Condensing This test is a visual inspection of the RHR system piping while the system is in the Steam Condensing mode.
18.0		TIP UNCERTAINTIES The objective of this test is to determine the mathematical uncertainty of TIP system readings.
18.1	3 6	TIP Uncertainty Determination Refer to description in ST 18.0.
19.0		CORE PERFORMANCE The objective of this test is to evaluate the core thermal power and flow and to demonstrate that the safety thermal limits are not exceeded. ST 19.1 performs this evaluation using the off line computer program BUCLE, and ST 19.2 uses the NSSS process computer.
19.1	12	BUCLE Calculations Refer to description in ST 19.0.
19.2	3456 Refer to description for details	Process Computer Calculation Refer to description in ST 19.0. During TC 3, this test is performed at both 50% and 75% power levels.

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ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
21.0		CORE POWER - VOID MODE RESPONSE The objective of this test is to verify the stability of the core power - mode dynamic response. Both tests monitor the reactor response to rapid void content changes. ST 21.1 produces these changes through control rod movement; ST 21.2 by initiating the simulated failure of the operating pressure regulator. Both tests are done during natural circulation testing at TC 4 and at minimum reactor recirculation pump speed.
21.1	4 6	Response of Power - Void Loop Through Control Rod Movement Refer to description in ST 21.0.
21.2	4 6	Response of Power - Void Loop Through Reactor Pressure Refer to description in ST 21.0. This test may be done in conjunction with ST 22.1.

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ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
22.0		PRESSURE REGULATOR The objectives of this test are to demonstrate stable controller settings, demonstrate the take over capability of the backup pressure regulator, and to demonstrate smooth pressure control transition between the turbine control valves and bypass valves when the reactor steam supply exceeds main turbine demand. ST 22.1, 22.2, and 22.3 differ only in the setting of the turbine Load Limit which effects which valves will control the pressure
22.1	23456	Pressure Regulator Test - Control Valves Controlling Refer to description in ST 22.0.
22.2	2 456	Pressure Regulator Test - Control Valves and Bypass Valves Controlling Refer to description in ST 22.0.
22.3	12 456	Pressure Regulator Test - Bypass Valves Controlling Refer to description in ST 22.0.
23.0		FEEDWATER SYSTEM The overall objectives of this test are to demonstrate that the feedwater system has been adjusted to provide acceptable water level and flow control and that licensing assumptions were conservative. Specific objectives are described in the Subtest abstracts.

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ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
23.1	1	<u>Feedwater System and Startup Controller Level Step</u> This test consists of introducing step changes in reactor water level while being controlled by the low load valve operating in auto.
23.2	23 6	<u>Feedwater System Manual Flow Step</u> This test consists of initiating step changes in feedwater pump speed and demonstrating stable and proper response.
23.3	23456	<u>Feedwater System Level Setpoint Changes</u> This test consists of initiating step changes in reactor water level and demonstrating stable and proper response.
23.4	6	<u>Loss of Feedwater Heating</u> This test consists of tripping the extraction steam to one of the feedwater heater trains at 85% reactor power and verifying that plant response is compatible with licensing assumptions.
23.5	6	<u>Feedwater Pump Trip</u> This test verifies that a low water level scram will not result due to the automatic run back feature in the recirculation system after a RFP trip.
23.6	6	<u>Maximum Feedwater Runout Capabilities</u> This test demonstrates that the sum of the individual feed pump run out valves does not exceed the assumed value in the FSAR.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
24.0		<u>TURBINE VALVE SURVEILLANCE</u> The objective of this test is to demonstrate acceptable margins to scram at maximum power levels recommended for periodic surveillance testing of the Main Turbine Stop, Control, Bypass and Combined Intermediate Stop Valves.
24.1	6	<u>Turbine Stop, Control, Combined Intermediate and Bypass Valve Testing</u> This subtest consists of stroking each valve until it's fully closed or open position, as appropriate, and monitoring plant response.
25.0		<u>MAIN STEAM ISOLATION VALVE</u> The objectives of this test are to demonstrate the proper operation of the Main Steam Isolation Valves (MSIV), demonstrate the maximum power level at which full closure of a single MSIV can be performed without causing a scram, and to demonstrate that licensing assumptions concerning the full isolation transient are conservative.
25.1	H 5 Refer to description for details	<u>MSIV Functional Test</u> During this test, each MSIV will be closed and timed and the resulting reactor response will be monitored. This test will be run at power levels slightly above TC 5 conditions and again at the predicted highest power level at which sufficient margins to scram exist.

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ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
25.3	6	Full Isolation This test will demonstrate the reactor transient behavior tnat results from the simultaneous full closure of all MSIV's at 100% power.
26.0		RELIEF VALVES The objectives of this test are to verify that the relief valves function properly, reseat properly after operation, contain no major blockages in discharge piping, and to demonstrate stable system response to relief valve operation. ST 26.1 is done at low reactor pressure and monitors bypass valve operation for determining proper response; ST 26.2 is done at rated reactor pressure and uses generator electrical output to determine proper response.
26.1	H	Relief Valve Low Pressure Test Refer to description in ST 26.0.
26.2	2	Relief Valve Rated Pressure Test Refer to description in ST 26.0.
27.0		TURBINE TRIP AND GENERATOR LOAD REJECTION The objective of this test is to demonstrate the reactor and its control systems response to trips of the main turbine and generator.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
27.1	3	Turbine Trip In this test, a turbine trip is initiated by depressing the main turbine trip pushbutton in the main control room.
27.2	6	High Power Generator Load Rejection In this test, a generator load rejection is initiated by opening the generator main breaker.
27.3	2	Generator Load Reject Within Bypass Capacity This test is similar to ST 27.2, except that the test is performed while steam production is within bypass valve capacity.
28.0		SHUTDOWN FROM OUTSIDE THE CONTROL ROOM The objective of this test is to demonstrate that the reactor can be scrammed, shutdown, maintained in a hot shutdown condition, and cooled down from outside the main control room using the emergency operating procedure.
28.1	1	Shutdown and Cooldown Demonstration Refer to description in ST 28.0. During this test the reactor is scrammed and isolated from inside the control room.

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ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
28.2	1	Reactor Shutdown From Outside the Control Room This test demonstrates that the reactor can be scrammed and isolated from outside the main control room.
29.0		RECIRCULATION FLOW CONTROL SYSTEM The objective of this test is to demonstrate the flow control capabilities of the plant over the entire recirculation pump speed range while operating in the Local Manual and Master Manual Modes.
29.1	23 56	Response to Step Input-Individual Manual This test consists of making step changes in recirculation pump speeds with the recirculation system in the Local Manual Mode.
29.3	3 56	Response to Step Input-Master Manual This test consists of making step changes in recirculation pump speeds with the recirculation system in the Master Manual Mode.
29.4	6	Recirc M/G Set High Speed Stop Settings Demonstrates that the electrical and mechanical high speed stops are properly set.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
30.0		RECIRCULATION SYSTEM The objective of this test is to demonstrate proper response of the recirculation system to various transients and to demonstrate that no recirculation system cavitation will occur in the operable region of the power - flow map.
30.1	3 6	Recirculation System One Pump Trip This test demonstrates that the feedwater control system can satisfactorily control water level on a single recirculation pump trip without a resulting turbine trip and associated scram. This test also demonstrates the validity of the restart procedure at the highest possible reactor power level.
30.2	3	Recirculation Pump Trip (RPT) of Two Pumps In this test, both reactor recirculation pumps will be tripped simultaneously using the RPT Breaker Trip Circuit. The data obtained from this test will be evaluated to verify pump coastdown performance prior to the scheduled turbine trips and generator load rejections at high power.
30.3	3	Recirculation Pump Runback In this test, proper conditions will be simulated to produce a recirculation pump runback to the #2 limiter setting. The results of the test will be analyzed to verify the adequacy of the recirculation runback to mitigate a scram.

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ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
30.4	3	<u>Recirculation System Limitor Verification</u> This test will demonstrate that the feedwater interlocks with the recirculation pump #1 limiter is set such that cavitation will not occur in the reactor recirculation system.
31.0		<u>LOSS OF TURBINE GENERATOR AND OFFSITE POWER</u> The objective of this test is to determine reactor transient performance during the loss of the main turbine generator coincident with the loss of all sources of off-site power. The objectives of this test are to demonstrate that the required safety systems will initiate and function properly without manual assistance, the electrical distribution and diesel generator systems will function properly, and the HPCI and/or RCIC systems will maintain water level if necessary, during a simultaneous loss of the main turbine - generator and offsite power. The loss of offsite power condition will be maintained for thirty minutes to demonstrate that necessary equipment, controls and indication are available following station blackout to remove decay heat from the core using only emergency power supplies and distribution system.
31.1	2	<u>Loss of Turbine Generator and Off-Site Power</u> Refer to description in ST 31.0.

STARTUP TEST ABSTRACTS

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ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
32.0		CONTAINMENT ATMOSPHERE AND MAIN STEAM TUNNEL COOLING The objective of this test is to verify the ability of the drywell coolers/recirculation fans and the Reactor Building portion of the Main Steam Tunnel Coolers to maintain design conditions in the drywell and reactor building portion of the main steam tunnel pipeway during operating conditions and post-scram conditions. This test also verifies that containment Main Steamline penetrations do not overheat adjacent concrete.
32.1	H 6	Containment Temperature at end of Heatup This test consists of monitoring temperatures near the end of the initial approach to rated reactor temperature and pressure. This test was repeated after rebalancing of the containment cooling system.
32.2	23 56	Containment Temperature at Steady State This test monitors steady-state temperature conditions.
32.3	23 6	Containment Temperature after Reactor Scram. This test monitors temperatures following planned reactor scrams during the Startup Test Program.

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ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
32.4	6	<u>Main Steam Penetration Concrete Temperature</u> This test monitors surface temperature on the concrete surrounding the main steam line penetrations after reactor heatup.
33.0		<u>STEADY STATE PIPING VIBRATION</u> The objective of this test is to verify that steady state vibration levels on Main Steam, Reactor Recirculation, Feedwater, HPCI, and RCIC piping, are within acceptable limits. The title of the tests indicate which piping is being monitored.
33.1	23 6 Refer to description for details	<u>Steady State Vibration for Main Steam Piping Inside Drywell</u> This test is performed at approximately 25, 50, 75, and 100% rated steam flow.
33.2	23 6 Refer to description for details	<u>Steady State Vibration, Main Steam Piping Outside Drywell and Feedwater Piping</u> This test is performed in conjunction with ST33.1.
33.3	H 3 56	<u>Steady State Vibration, Recirculation Piping</u> This test is performed at approximate 50, 75 and 100% core flow on the 100% rod line, and during ST 8.3, ST 16.1 and ST 30.4.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
33.4	3	Steady State Vibration, HPCI, CST To Vessel This test is performed with HPCI taking suction from the CST and discharging to the reactor vessel at rated pressure and rated flow.
33.6	H	Steady State Vibration, RCIC, Reactor Steam Supply This test is run with RCIC taking suction from the CST and injecting to the vessel at rated reactor pressure and rated flow.
35.0		RECIRCULATION SYSTEM FLOW CALIBRATION The objective of this test is to perform a complete calibration of recirculation flow instrumentation.
35.1	3 6 Refer to Description for details	Recirculation System Flow Calibration Refer to description in ST35.0. During TC 3, this test is run at both 50% and 75% power.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
37.0		<u>GASEOUS RADWASTE SYSTEM</u> The objective of this test is to demonstrate that the gaseous radwaste system operates within technical specifications and design limits during a full range of plant power operation, and to demonstrate the proper operation of the containment inerting system.
37.1	H1 3 56	<u>Gaseous Radwaste Data Collection</u> This test demonstrates that the gaseous radwaste system operates within technical specifications.
37.2	6	<u>Containment Inerting</u> This test demonstrates the proper operation of the containment inerting system.
37.3	H 6	<u>Gaseous Radwaste System Performance</u> This test demonstrates that the gaseous radwaste system operates within design limits.
39.0		<u>PIPING VIBRATORY RESPONSE DURING DYNAMIC TRANSIENTS</u> The objective of this test is to demonstrate that the vibrational response of selected piping is within acceptable limits when the piping is subjected to selected controlled system transients.

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ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
39.1	23 6	<u>Main Steam Piping Vibratory Response During Turbine Stop Valve Closure</u> This test verifies proper response of the main steam piping inside and outside the drywell during system transients caused by Turbine Stop Valve fast closures in conjunction with other STs.
39.2	2	<u>Main Steam and Safety Relief Valve Piping Vibratory Response During Safety Relief Valve Operation.</u> This test is done in conjunction with the safety relief valve testing performed in ST 26.2.
39.3	34 6	<u>Recirculation Piping Vibratory Response During Pump Trips and Restarts.</u> This test is done in conjunction with the recirculation pump trips and restarts of ST 30.1 and ST 30.2 and other times during planned pump trips and restarts.
39.4	H	<u>HPCI Steam Supply Piping Vibratory Response During HPCI Turbine Trip</u> This test analyzes the data collected during the HPCI turbine trip initiated in ST15.1.
39.5	3	<u>Feedwater Piping Vibratory Response During Feedwater Pump Turbine Trips</u> During this test, each Reactor Feedwater Pump will be tripped individually and the response of the feedwater piping analyzed.

SECTION 4

STARTUP TEST RESULTS

4.1 (ST1) CHEMICAL AND RADIOCHEMICAL DATA

The principal objective of this test was to verify that chemical parameters of the reactor coolant and selected support systems met the acceptance criteria. These tests also demonstrated the overall adequacy of sampling techniques, procedures and equipment.

<u>Level 1</u>

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

2. The concentration of activity of liquid effluents must conform to the Technical Specifications.

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

ST1.1, Chemistry Data - Pre-Fuel Load, was performed at open vessel conditions, 0% power. The initial readings of chlorides, conductivity and pH were sampled and found to satisfy Acceptance Criteria.

ST1.5, Chemistry Data - Pre-Heatup, was performed at 0% power with 44.34% rated core flow. At this point a routine chemistry run was completed with all usual analysis. All Acceptance Criteria were met with no out of spec chemistry.

ST1.6 Chemistry Data, Heatup Tests, was also performed with the reactor at 3% rated power. All routine analysis were run and all Acceptance Criteria were met.

ST1.7, Chemistry Data - Power Ascension Tests, was run at 17% (TC1), 40% (TC2), 47.5% (TC3), 73.6% (TC3), 74.5% (TC5) and 99.8% rated power (TC6) to successfully demonstrate that samples from Reactor Water Cleanup influent, Feedwater and Control Rod Drive water were within Acceptance Criteria.

4.2 (ST2) RADIATION MEASUREMENTS

Radiation Measurements was concerned with the activity produced in the confines of the plant. The first set of data that was taken was to determine the background activation produced from cosmic interaction and non-organic matter in the strata. The second set of readings were taken at various stages of reactor power. This provided for a basis of seeing that the plant could operate without endangering personnel by exceeding limits as set forth in 10CFR20.

The Acceptance Criteria expressed in these tests are as follows:

Level 1

1. The radiation doses of plant origin and the occupancy times of personnel in radiation zones shall be controlled consistant with the guidelines in 10CFR20, Standards for Protection Against Radiation.

Level 2

1. The radiation doses of plant origin shall meet the following limits depending upon which Radiation Zone the radiation base survey point is located:

RADIATION ZONE	LIMIT
I	Less than or equal to 0.5 mRem/hr
II	Less than or equal to 2.5 mRem/hr
III	Less than or equal to 15 mRem/hr
IV	Less than or equal to 100 mRem/hr

A radiation survey using ST2.1 "Startup Test Program Radiation Surveying" was conducted prior to initial fuel loading, upon initially reaching rated pressure and temperature, and at 17% (TC1), 73.6% (TC3)and 99.4% (TC6)rated power.

During the test performance prior to fuel load and upon initially reaching rated pressure and temperature, the area around the CRD pumps was higher than the Zone II limits due to the CRD suction filter. A Zone II - Zone III boundary was established around the pumps. Readings higher than the Acceptance Criterion was also found at Radiation Base Points 2R044 and 2R069. Resolution of these exceptions is ongoing at the time of this report.

4.3 (ST3) FUEL LOADING

The initial core of Susquehanna Unit 2 was successfully loaded with 764 fuel assemblies in 16 days (March 28, 1984 to April 13, 1984). Adequate shutdown margin was demonstrated after 144 bundles were loaded. Control rod functional tests and friction tests were performed in parallel with loading the fuel. The full core verification was performed to show that all fuel assemblies were properly loaded, oriented, and seated in the core. Both the Level 1 and Level 2 Acceptance Criteria were satisfied.

The Level 1 Acceptance Criterion stated that the partially loaded core must be subcritical by at least 0.38% $\Delta K/K$ with the analyticallly highest worth control rod fully withdrawn. After 144 fuel assemblies were loaded, the analytically highest worth rods 26-27 and 34-27 were withdrawn one notch at a time while observing the nuclear instrumentation. The nuclear instrumentation did not indicate a continuous positive period, thus demonstrating subcriticality.

The Level 2 Acceptance Criterion stated that the fully loaded core must be installed and configured as specified. This verification was independently verified by Reactor Engineering, Quality Assurance and the NSSS supplier.

Prior to the start of fuel loading, four fuel loading chambers (Type FLC NA08) were assembled, placed in the core, and connected to the permanent SRM preamplifiers. The scram setpoint was set at $1X10^5$ CPS and rod block was set at $5 X 10^4$ CPS. The reactor protection system was placed in the non-coincidence scram mode (shorting links removed). High voltage and discriminator curves were obtained for each FLC. FLC moves during fuel loading are depicted in figure 4.3.1.

The first SbBe neutron source was installed into the Unit 2 RPV on March 24, 1984. During the installation of the first Source Holder, a piece of plastic was discovered pinched under the fuel support piece in the vicinity of the Source Holder. The Source Holder was removed from the RPV to allow recovery of the piece of plastic.

A diver removed the piece of plastic from the RPV on March 27, 1984 and performed a visual inspection of the core after several unsuccessful attempts to remove the plastic using the pincher tool from the refuel bridge. The Source Holder installation was completed on March 28, 1984 without any other complications.

The entire core complement of fuel assemblies was prepared, inventoried and stored in the fuel pool prior to the start of fuel loading. Before fuel was loaded, each control rod was tested for position indication, coupling and scram time verifying proper operation of the control rod and ensuring that the blade guides did not interfere with control rod travel. Fuel loading commenced using the PP&L Fuel And Core Component Transfer Authorization Sheet (FACCTAS) as the guiding document. Starting near the center of the core, four fuel assemblies were loaded around the central neutron source. The loading continued in the control cell units that sequentially completed each face of the ever increasing square core.

A plot of inverse count rate (1/m) was taken during fuel load to verify subcriticality through the entire fuel load. The plot was taken after loading each fuel assembly until 16 assemblies were loaded. Subsequent to that, 1/m plots were taken every 4 assemblies until 256 fuel assemblies were loaded. After 256 assemblies were loaded 1/m plots were taken every 16 assemblies. Plotting frequencies were increased if the current 1/m plot indicated that criticality would occur prior to the next planned 1/m plot.

On several occasions during the early stages of fuel loading, criticality was predicted by the 1/m plot before the next scheduled plotting point. The reason for this was the geometrical effects encountered when less than four control cells are loaded and the strong effects as fuel is loaded adjacent to the neutron sources. The interpretation of the geometry affected 1/m plots allow disregarding one or more 1/m intercepts because the obvious geometeric effect invalidates the theoretical basis for the 1/m plots.

Several problems were encountered with fuel loading equipment. A brief summary is given:

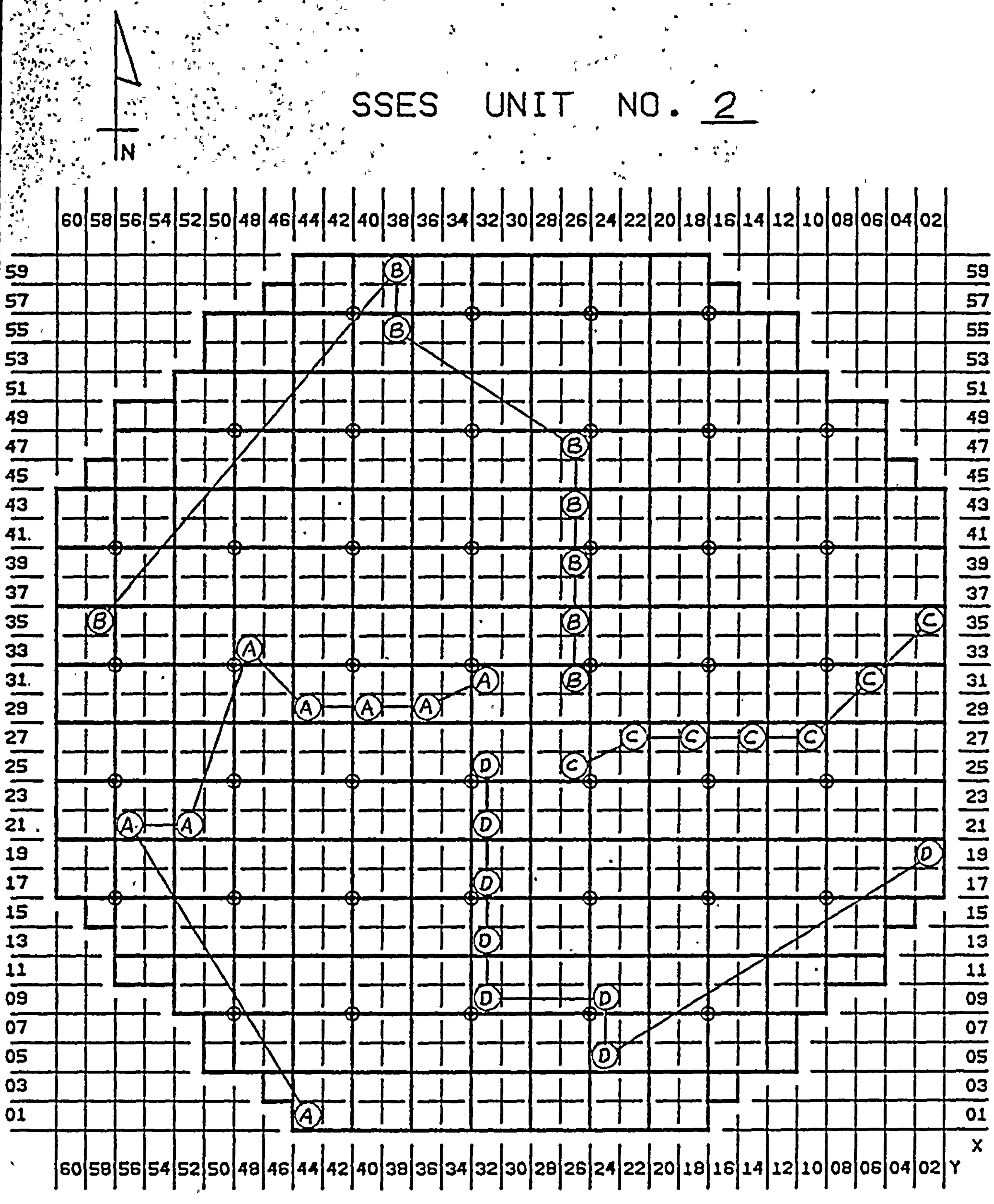
DATE	PROBLEM	SOLUTION
March 29, 1984	Refueling hoist drifting down on its own	Replaced switch which controls up-down movements
March 31, 1984	Grapple would not unlatch	Reset relief valve on compressor
March 31, 1984	Bent boom due to operator error	Replaced boom with Unit 1's
April 3, 1984	High spikes on SRM count and period while attempting to remove blade guide	Replace ampherrol connector for grapple latching mechanism which was causing a short

The fully loaded core was verified to be installed and configured properly on April 14, 1984. The fuel load arrangements is shown on Figure 4.3-2.

FIGURE 4.3-1

FLC MOVES DURING FUEL LOAD

SSES UNIT NO. 2



A-184077-c

SSES CORE / A POOL LOCATION

G½FØ1

LOCATION CODE

APPROVED BY CH. REVISION NO. DATE ER. DESCRIPTION

FRACTIONAL DECIMAL

FIGURE 4.3-2
SHEET 1 OF 10
FUEL LOAD ARRANGEMENT

```
PP&L  SSES  UNIT 2                             DATE:  1/ 4/85      PAGE:    1
CORE MAP

JY/IX        2         4         6         8        10        12        14        16        18        20       IX/JY
         1         3         5         7         9        11        13        15        17        19

                                                                                NI
 21                                                                          TAEYE6-5                          21
       ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 20    :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :      20
       :   18:   :  715:   :  595:   :  596:   :  489:   :  288:   :  411:   :  222:   :  378:   :  182:
       ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
 19          B2103               B2203               B2076               B2091               B2202             19
       ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 18    :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :      18
       :   19:   :  714:   :  597:   :  594:   :  490:   :  488:   :  223:   :  410:   :  183:   :  379:
       ........  ........  ........  ........LS........  ........  ........  ........LS........  ........
 17                                       TEART224                                TEART2-9                     17
                 ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 16              :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :      16
                 :   92:   :  581:   :  582:   :  475:   :  476:   :  458:   :  283:   :  448:   :  276:
                 ........  ........CR........  ........CR........  ........CR........  ........CR........
 15                              B2204               B2077               B2200               B2151             15
                           ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 14                        :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :      14
                           :    8:   :  580:   :  477:   :  474:   :  459:   :  457:   :  277:   :  449:
                           ........  ........  ........  ........  ........  ........NI........  ........NI
 13                                                                          TAEYC7-5            SOURCE-1      13
                           ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 12                        :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :      12
                           :   88:   :   84:   :  565:   :  567:   :  550:   :  552:   :  537:   :  331:
                           ........  ........  ........CR........  ........CR........  ........CR........
 11                                                  B2127               B2238               B2196             11
                                               ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 10                                            :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :      10
                                               :    3:   :  566:   :  553:   :  551:   :  539:   :  538:
                                               ........  ........  ........  ........LS........  ........
  9                                                                          TEART210                           9
                                                         ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
  8                                                      :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :       8
                                                         :   80:   :  701:   :  703:   :  686:   :  688:
                                                         ........  ........CR........  ........CR........
  7                                                                      B2222               B2141              7
                                                         ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
  6                                                      :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :       6
                                                         :   76:   :   11:   :  702:   :  689:   :  687:
                                                         ........  ........  ........  ........  ........
  5                                                                                                             5
                                                                             ...FA...  ...FA...  ...FA...
  4                                                                          :LJW   :  :LJW   :  :LJW   :       4
                                                                             :   72:   :  760:   :  761:
                                                                             ........  ........CR........
  3                                                                                          B2231              3
                                                                                       ...FA...  ...FA...
  2                                                                                    :LJW   :  :LJW   :       2
                                                                                       :   65:   :   64:
                                                                                       ........  ........

JY/IX        2         4         6         8        10        12        14        16        18        20.      IX/JY
         1         3         5         7         9        11        13        15        17        19
```

FIGURE 4.3-2
SHEET 2 OF 10
FUEL LOAD ARRANGEMENT

```
PP&L  SSES  UNIT 2                      DATE:  1/ 4/85      PAGE:    2
CORE MAP

JY/IX           22        24        26        28        30        32        34        36        38        40    IX/JY
             21        23        25        27        29        31        33        35        37        39

                                                                                                         NI
21                                                                                                    THHHL519   21
          ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
20        :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :    20
          :  370:   :  174:   :  366:   :  162:   :  152:   :  154:   :  142:   :  354:   :  180:   :  376:
          ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
19               B2098               B2137               B2248               B1739               B2130           19
          ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
18        :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :    18
          :  175:   :  371:   :  163:   :  367:   :  155:   :  153:   :  355:   :  143:   :  377:   :  181:
          ........  ........LS........  ........  ........  ........LS........  ........  ........  ........LS
17                      TEART231                                TEART238                              TEART221   17
          ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
16        :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :    16
          :  440:   :  268:   :  432:   :  260:   :  246:   :  248:   :  236:   :  424:   :  228:   :  416:
          ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
15               B2084               B2172               B2168               B2219               B2159           15
          ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
14        :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :    14
          :  269:   :  441:   :  261:   :  433:   :  249:   :  247:   :  425:   :  237:   :  417:   :  229:
          ........  ........  ........  ........  ........  ........  ........  ........NI........  ........
13                                                                              SOURCE-2                         13
          ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
12        :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :    12
          :  525:   :  327:   :  516:   :  320:   :  306:   :  308:   :  296:   :  508:   :  291:   :  498:
          ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
11               B2232               B2164               B2209               B2143               B2230           11
          ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
10        :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :    10
          :  527:   :  526:   :  321:   :  517:   :  309:   :  307:   :  509:   :  297:   :  499:   :  497:
          ........  ........LS........  ........  ........  ........LS........  ........  ........  ........LS
9                       TEART226                                TEART21                               TEART230   9
          ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
8         :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :    8
          :  670:   :  672:   :  654:   :  656:   :  638:   :  640:   :  622:   :  624:   :  606:   :  608:
          ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
7                B2097               B2224               B2109               B2002               B2082           7
          ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
6         :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :    6
          :  673:   :  671:   :  657:   :  655:   :  641:   :  639:   :  625:   :  623:   :  609:   :  607:
          ........  ........  ........  ........  ........  ........  ........  ........  ........  ........
5                                                                                                                5
          ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
4         :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :    4
          :  752:   :  753:   :  744:   :  745:   :  736:   :  737:   :  728:   :  729:   :  720:   :  721:
          ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
3                B2166               B2210               B1952               B2120               B1965           3
          ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
2         :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :    2
          :   57:   :   56:   :   49:   :   48:   :   41:   :   40:   :   33:   :   32:   :   25:   :   24:
          ........  ........  ........  ........  ........  ........  ........  ........  ........  ........

JY/IX           22        24        26        28        30        32        34        36        38        40    IX/JY
             21        23        25        27        29        31        33        35        37        39
```

FIGURE 4.3-2
SHEET 3 OF 10
FUEL LOAD ARRANGEMENT

```
PP&L  SSES  UNIT 2                                   DATE:  1/ 4/85      PAGE:    3
CORE MAP

JY/IX           42        44        46        48        50        52        54        56        58        60    IX/JY
           41        43        45        47        49        51        53        55        57        59

                                              NI
  21                                       SOURCE-3                                                             21
        ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
  20    :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :      20
        :   214:  :   403:  :   274:  :   447:  :   330:  :   536:  :   682:  :   685:  :   758:  :    63:
        ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
  19           B1962               B2239               B2214               B1948               B1868            19
        ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
  18    :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :      18
        :   402:  :   215:  :   446:  :   275:  :   535:  :   534:  :   684:  :   683:  :   759:  :    62:
        ........  ........  ........  ........LS........  ........  ........  ........LS........  ........
  17                                       TEART219                                 TEART22                     17
        ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
  16    :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :                16
        :   220:  :   408:  :   282:  :   456:  :   546:  :   549:  :   698:  :   700:  :    71:
        ........CR........  ........CR........  ........CR........  ........CR........  ........
  15           B2119               B2142               B2115               CA030                                15
        ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
  14    :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :                          14
        :   409:  :   221:  :   455:  :   454:  :   548:  :   547:  :   699:  :    10:
        ........  ........  ........  ........NI........  ........  ........  ........
  13                                       TAEYC8-5                                                             13
        ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
  12    :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :                          12
        :   287:  :   486:  :   470:  :   472:  :   562:  :   564:  :    79:  :    75:
        ........CR........  ........CR........  ........CR........  ........  ........
  11           B2075               B2099               B2234                                                    11
        ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
  10    :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :                                              10
        :   487:  :   485:  :   473:  :   471:  :   563:  :     2:
        ........  ........  ........  ........LS........  ........
   9                                       TEART239                                                              9
        ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
   8    :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :                                                        8
        :   590:  :   592:  :   577:  :   578:  :    83:
        ........CR........  ........CR........  ........
   7           B2100               B2106                                                                         7
        ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
   6    :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :                                                        6
        :   593:  :   591:  :   579:  :     7:  :    87:
        ........  ........  ........  ........  ........
   5                                                                                                             5
        ...FA...  ...FA...  ...FA...
   4    :LJW   :  :LJW   :  :LJW   :                                                                            4
        :   712:  :   713:  :    91:
        ........CR........  ........
   3           B2110                                                                                             3
        ...FA...  ...FA...
   2    :LJW   :  :LJW   :                                                                                      2
        :    17:  :    16:
        ........  ........

JY/IX           42        44        46        48        50        52        54        56        58        60    IX/JY
           41        43        45        47        49        51        53        55        57        59
```

FIGURE 4.3-2
SHEET 4 OF 10
FUEL LOAD ARRANGEMENT

```
PP&L  SS[illegible] UNIT 2                          DATE:  1/ 4/85      PAGE:    4
CORE MAP

JY/IX       2         4         6         8        10        12        14        16        18        20        IX/JY
       1         3         5         7         9        11        13        15        17        19

                                           LS                                      LS
41                                      TEART217                                TEART242                       41
     ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
40   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     40
     :    58:  :   755:  :   675:  :   676:  :   528:  :   529:  :   271:  :   442:  :   185:  :   380:
     ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
39         B2258               B2220               B2129               B2024               B2217               39
     ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
38   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     38
     :    59:  :   754:  :   677:  :   674:  :   530:  :   328:  :   443:  :   270:  :   381:  :   184:
     ........  ........  ........  ........  ........  ........NI........  ........  ........  ........
37                                                          SOURCE-5                                           37
     ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
36   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     36
     :    50:  :   747:  :   659:  :   660:  :   323:  :   518:  :   263:  :   434:  :   177:  :   372:
     ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
35         B2063               B2251               B1967               B2207               B2192               35
     ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
34   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     34
     :    51:  :   746:  :   661:  :   658:  :   519:  :   322:  :   435:  :   262:  :   373:  :   176:
     ........  ........  ........  ........LS........  ........  ........  ........LS........  ........
33                                      TEART2-6                                TEART237                       33
     ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
32   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     32
     :    42:  :   739:  :   643:  :   644:  :   311:  :   312:  :   251:  :   252:  :   165:  :   166:
     ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
31         B2155               B2180               B2145               B2081               B2187               31
     ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
30   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     30
     :    43:  :   738:  :   645:  :   642:  :   313:  :   310:  :   253:  :   250:  :   167:  :   164:
     ........  ........  ........  ........  ........  ........  ........  ........  ........  ........
29                                                                                                             29
     ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
28   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     28
     :    34:  :   731:  :   627:  :   628:  :   511:  :   298:  :   427:  :   238:  :   361:  :   156:
     ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
27         B2083               B2171               B1930               B2140               B2206               27
     ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
26   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     26
     :    35:  :   730:  :   629:  :   626:  :   299:  :   510:  :   239:  :   426:  :   157:  :   360:
     ........  ........  ........  ........LS........  ........  ........  ........LS........  ........
25                                      TEART232                                TEART222                       25
     ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
24   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     24
     :    26:  :   723:  :   611:  :   612:  :   501:  :   292:  :   419:  :   230:  :   357:  :   144:
     ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
23         B2149               B2165               B2074               B1794               B2112               23
     ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
22   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     22
     :    27:  :   722:  :   613:  :   610:  :   502:  :   500:  :   231:  :   418:  :   145:  :   356:
     ........  ........  ........  ........  ........  ........  ........  ........  ........  ........

JY/IX       2         4         6         8        10        12        14        16        18        20        IX/JY
       1         3         5         7         9        11        13        15        17        19
```

FIGURE 4.3-2
SHEET 5 OF 10
FUEL LOAD ARRANGEMENT

```
PP&L  SSES  UNIT 2                                   DATE:  1/ 4/85      PAGE:     5
CORE MAP

JY/IX            22        24        26        28        30        32        34        36        38        40        IX/JY
             21        23        25        27        29        31        33        35        37        39

                               LS                                      LS                                      LS
 41                         TEART225                                TEART28                                 TEART240      41
        ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 40     :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :          40
        :   147:  :   359:  :   159:  :   363:  :   169:  :   171:  :   375:  :   179:  :   382:  :   187:
        .......CR.........  .......CR.........  .......CR.........  .......CR.........  .......CR.........
 39             B2105               B2117               CA045               B2144               B2010             39
        ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 38     :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :          38
        :   358:  :   146:  :   362:  :   158:  :   170:  :   168:  :   178:  :   374:  :   186:  :   383:
        ........  .......NI.......  ........  ........  .......NI.......  ........  ........  ........
 37                         TAEYC8-2                                TAGGK2-2                                          37
        ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 36     :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :          36
        :,  137:  :   348:  :   119:  :   339:  :   129:  :   131:  :   350:  :   139:  :   352:  :   141:
        .......CR.........  .......CR.........  .......CR.........  .......CR.........  .......CR.........
 35             B2062               B2246               B2126               B2228               B2147             35
        ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 34     :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :          34
        :   349:  :   136:  :   338:  :   118:  :   130:  :   128:  :   138:  :   351:  :   140:  :   353:
        ........  .......LS.......  ........  ........  .......LS.......  ........  ........  .......LS
 33                         TEART211                                TABVB9-1                                TEART216      33
        ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 32     :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :          32
        :   125:  :   126:  :   105:  :   107:  :    99:  :    98:  :   108:  :   111:  :   148:  :   151:
        .......CR.........  .......CR.........  .......CR.........  .......CR.........  .......CR.........
 31             B2195               B2086               B2108               B1897               B2153             31
        ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 30     :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :          30
        :   127:  :   124:  :   106:  :    97:  :    94:  :   100:  :   110:  :   109:  :   150:  :   149:
        ........  .......NI.......  .......NI.......  .......NI.......  ........  ........  ........
 29                         TAEYC7-7          SOURCE-4              TAEYC7-4                                          29
        ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 28     :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :          28
        :   337:  :   116:  :   335:  :    96:  :    95:  :   102:  :   120:  :   341:  :   160:  :   365:
        .......CR.........  .......CR.........  .......CR.........  .......CR.........  .......CR.........
 27             B2191               B2225               B2092               B1921               B2205             27
        ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 26     :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :          26
        :   117:  :   336:  :   104:  :   334:  :   103:  :   101:  :   340:  :   121:  :   364:  :   161:
        ........  .......LS.......  ........  ........  .......LS.......  ........  ........  .......LS
 25                         TEART233                                TEART243                                TEART218      25
        ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 24     :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :          24
        :   347:  :   134:  :   342:  :   122:  :   112:  :   114:  :   132:  :   345:  :   172:  :   369:
        .......CR.........  .......CR.........  .......CR.........  .......CR.........  .......CR.........
 23             B2215               B2211               B2102               B2005               B1817             23
        ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 22     :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :          22
        :   135:  :   346:  :   123:  :   343:  :   115:  :   113:  :   344:  :   133:  :   368:  :   173:
        ........  ........  ........  ........  ........  ........  ........  ........  ........  ........

JY/IX            22        24        26        28        30        32        34        36        38        40        IX/JY
             21        23        25        27        29        31        33        35        37        39
```

FIGURE 4.3-2
SHEET 6 OF 10
FUEL LOAD ARRANGEMENT

```
PP&L S[illegible]S UNIT 2                                   DATE:  1/ 4/85       PAGE:    6
CORE MAP

JY/IX       42        44        46        48        50        52        54        56        58        60     IX/JY
        41        43        45        47        49        51        53        55        57        59

                                          LS                                      LS
 41                                     TEART23                                 TEART234                     41
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 40   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     40
      :   386:  :   191:  :   414:  :   227:  :   494:  :   496:  :   602:  :   605:  :   718:  :    23:
      ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
 39          B2096               B2087               B2152               B2193               B1964           39
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 38   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     38
      :   190:  :   387:  :   226:  :   415:  :   290:  :   495:  :   604:  :   603:  :   719:  :    22:
      ........  ........NI........  ........  ........  ........  ........  ........  ........  ........
 37                 SOURCE-6                                                                                 37
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 36   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     36
      :   390:  :   195:  :   422:  :   235:  :   506:  :   295:  :   618:  :   621:  :   726:  :    31:
      ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
 35          B2059               B2057               B2160               B2170               B2139           35
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 34   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     34
      :   194:  :   391:  :   234:  :   423:  :   294:  :   507:  :   620:  :   619:  :   727:  :    30:
      ........  ........  ........  ........LS........  ........  ........  ........LS........  ........
 33                                     TEART214                                TEART212                     33
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 32   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     32
      :   198:  :   201:  :   242:  :   245:  :   302:  :   305:  :   634:  :   637:  :   734:  :    39:
      ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
 31          B2118               B2197               B2188               B2116               B2094           31
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 30   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     30
      :   200:  :   199:  :   244:  :   243:  :   304:  :   303:  :   636:  :   635:  :   735:  :    38:
      ........  ........  ........  ........  ........  ........  ........  ........  ........  ........
 29                                                                                                          29
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 28   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     28
      :   206:  :   395:  :   258:  :   431:  :   318:  :   515:  :   650:  :   653:  :   742:  :    47:
      ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
 27          B2229               B2212               B2104               B1827               B1838           27
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 26   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     26
      :   394:  :   207:  :   430:  :   259:  :   514:  :   319:  :   652:  :   651:  :   743:  :    46:
      ........  ........  ........  ........LS........  ........  ........  ........LS........  ........
 25                                     TEART27                                 TEART229                     25
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 24   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     24
      :   210:  :   399:  :   266:  :   439:  :   326:  :   524:  :   666:  :   669:  :   750:  :    55:
      ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
 23          B2163               B2125               B1966               B2079               B1839           23
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 22   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :     22
      :   398:  :   211:  :   438:  :   267:  :   523:  :   522:  :   668:  :   667:  :   751:  :    54:
      ........  ........  ........  ........  ........  ........  ........  ........  ........  ........

JY/IX       42        44        46        48        50        52        54        56        58        60     IX/JY
        41        43        45        47        49        51        53        55        57        59
```

FIGURE 4.3-2
SHEET 7 OF 10
FUEL LOAD ARRANGEMENT

```
PP&L  SS[illegible] UNIT 2                          DATE:  1/ 4/85     PAGE:    7
CORE MAP

JY/IX    2    4    6    8   10   12   14   16   18   20     IX/JY
       1    3    5    7    9   11   13   15   17   19

  61                                                                                             61
                                                                              ...FA... ...FA...
  60                                                                          :LJW   : :LJW   :  60
                                                                              :    20: :    21:
  59                                                                          ........CR........ 59
                                                                                     B2128
                                                                     ...FA... ...FA... ...FA...
  58                                                                 :LJW   : :LJW   : :LJW   :  58
                                                                     :    93: :   717: :   716:
                                                                     ........LS........ ........
  57                                                                        TEART213             57
                                                   ...FA... ...FA... ...FA... ...FA... ...FA...
  56                                               :LJW   : :LJW   : :LJW   : :LJW   : :LJW   :  56
                                                   :    89: :     9: :   585: :   599: :   601:
                                                   ........ ........CR........ ........CR........
  55                                                               B2121             B2134       55
                                                   ...FA... ...FA... ...FA... ...FA... ...FA...
  54                                               :LJW   : :LJW   : :LJW   : :LJW   : :LJW   :  54
                                                   :    85: :   584: :   583: :   600: :   598:
                                                   ........ ........ ........NI........ ........
  53                                                                        TAGGK2-3             53
                                          ...FA... ...FA... ...FA... ...FA... ...FA... ...FA...
  52                                      :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   :  52
                                          :     4: :   569: :   479: :   481: :   491: :   493:
                                          ........CR........ ........CR........ ........CR........
  51                                             B2189             B2093             B2208       51
                        ...FA... ...FA... ...FA... ...FA... ...FA... ...FA... ...FA... ...FA...
  50                    :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   :  50
                        :    77: :    81: :   570: :   568: :   480: :   478: :   492: :   289:
                        ........ ........LS........ ........ ........ ........LS........ ........
  49                                  TEART223                              TEART236             49
                        ...FA... ...FA... ...FA... ...FA... ...FA... ...FA... ...FA... ...FA...
  48                    :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   :  48
                        :    12: :   705: :   555: :   556: :   460: :   461: :   225: :   413:
                        ........CR........ ........CR........ ........CR........ ........CR........
  47                           B2253             B1902             B2113             B2167       47
               ...FA... ...FA... ...FA... ...FA... ...FA... ...FA... ...FA... ...FA... ...FA...
  46           :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   :  46
               :    73: :   706: :   704: :   557: :   554: :   462: :   284: :   412: :   224:
               ........ ........ ........ ........ ........ ........ ........NI........ ........
  45                                                                        TAGGK6-1             45
      ...FA... ...FA... ...FA... ...FA... ...FA... ...FA... ...FA... ...FA... ...FA... ...FA...
  44  :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   :  44
      :    66: :   763: :   691: :   692: :   540: :   541: :   279: :   450: :   189: :   385:
      ........CR........ ........CR........ ........CR........ ........CR........ ........CR........
  43         B2095             B2146             B2185             B1872             B2198       43
      ...FA... ...FA... ...FA... ...FA... ...FA... ...FA... ...FA... ...FA... ...FA... ...FA...
  42  :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   : :LJW   :  42
      :    67: :   762: :   693: :   690: :   542: :   332: :   451: :   278: :   384: :   188:
      ........ ........ ........ ........ ........ ........ ........ ........ ........ ........

JY/IX    2    4    6    8   10   12   14   16   18   20     IX/JY
       1    3    5    7    9   11   13   15   17   19
```

FIGURE 4.3-2
SHEET 8 OF 10
FUEL LOAD ARRANGEMENT

```
PP&L  S     UNIT 2                          DATE:  1/ 4/85      PAGE:    8
CORE MAP

JY/IX       22         24         26         28         30         32         34         36         38         40        IX/JY
        21         23         25         27         29         31         33         35         37         39

 61                                                                                                                        61
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 60   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :               60
      :    28:  :    29:  :    36:  :    37:  :    44:  :    45:  :    52:  :    53:  :    60:  :    61:
      ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
 59           B2233               B2058               B2157               B2237               B2249                    59
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 58   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :               58
      :   725:  :   724:  :   733:  :   732:  :   741:  :   740:  :   749:  :   748:  :   757:  :   756:
      ........  ........LS........  ........  ........  ........LS........  ........  ........  ........LS
 57                  TABVB9-4                                  TABVB9-8                                   TEART25          57
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 56   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :               56
      :   615:  :   617:  :   631:  :   633:  :   647:  :   649:  :   663:  :   665:  :   679:  :   681:
      ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
 55           B2227               B2156               B2078               B2154               B2135                    55
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 54   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :               54
      :   616:  :   614:  :   632:  :   630:  :   648:  :   646:  :   664:  :   662:  :   680:  :   678:
      ........  ........  ........  ........  ........  ........  ........  ........  ........  ........
 53                                                                                                                        53
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 52   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :               52
      :   503:  :   505:  :   301:  :   513:  :   315:  :   317:  :   521:  :   325:  :   532:  :   533:
      ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
 51           B2177               B1941               B2122               B2107               B2111                    51
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 50   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :               50
      :   504:  :   293:  :   512:  :   300:  :   316:  :   314:  :   324:  :   520:  :   329:  :   531:
      ........  ........LS........  ........NI........  ........LS........  ........  ........  ........LS
 49                  TEART2-4            SOURCE-7              TEART215                                   TEART241         49
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 48   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :               48
      :   233:  :   421:  :   241:  :   429:  :   255:  :   257:  :   437:  :   265:  :   445:  :   273:
      ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
 47           B2136               B2150               B2080               B2194               B2169                    47
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 46   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :               46
      :   420:  :   232:  :   428:  :   240:  :   256:  :   254:  :   264:  :   436:  :   272:  :   444:
      ........  ........  ........  ........  ........  ........  ........  ........  ........  ........NI
 45                                                                                                       TAEJT7-8         45
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 44   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :               44
      :   193:  :   389:  :   197:  :   393:  :   203:  :   205:  :   397:  :   209:  :   401:  :   213:
      ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
 43           B2213               B2124               B2179               B2252               B2199                    43
      ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 42   :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :               42
      :   388:  :   192:  :   392:  :   196:  :   204:  :   202:  :   208:  :   396:  :   212:  :   400:
      ........  ........  ........  ........  ........  ........  ........  ........  ........  ........

JY/IX       22         24         26         28         30         32         34         36         38         40        IX/JY
        21         23         25         27         29         31         33         35         37         39
```

FIGURE 4.3-2
SHEET 9 OF 10
FUEL LOAD ARRANGEMENT

```
PP&L  S[illegible]  UNIT 2                          DATE:  1/ 4/85      PAGE:     9
CORE MAP

JY/IX         42        44        46        48        50        52        54        56        58        60        IX/JY
          41        43        45        47        49        51        53        55        57        59

  61                                                                                                              61
       ...FA...  ...FA...
 60    :LJW   :  :LJW   :                                                                                        60
       :   68:   :   69:
       ........CR........
  59         B1944                                                                                                59
       ...FA...  ...FA...  ...FA...
 58    :LJW   :  :LJW   :  :LJW   :                                                                              58
       :  765:   :  764:   :   70:
       ........  ........  ........
  57                                                                                                              57
       ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 56    :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :                                                          56
       :  695:   :  697:   :  708:   :   13:   :   74:
       ........CR........  ........CR........  ........
  55         B2250               B2223                                                                            55
       ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 54    :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :                                                          54
       :  696:   :  694:   :  709:   :  707:   :   78:
       ........  ........  ........  ........NI........
  53                                       TAGGK2-1                                                               53
       ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 52    :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :                                                52
       :  544:   :  545:   :  559:   :  561:   :  572:   :    5:
       ........CR........  ........CR........  ........CR........
  51         B2173               B2162               B2052                                                        51
       ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 50    :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :                            50
       :  333:   :  543:   :  560:   :  558:   :  573:   :  571:   :   82:   :   86:
       ........  ........  ........  ........LS........  ........  ........  ........
  49                                       TEART220                                                               49
       ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 48    :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :                            48
       :  453:   :  281:   :  464:   :  465:   :  466:   :  469:   :  574:   :    6:
       ........CR........  ........CR........  ........CR........  ........CR........
  47         B2088               B2158               B2101               B2221                                    47
       ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 46    :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :                  46
       :  280:   :  452:   :  285:   :  463:   :  468:   :  467:   :  576:   :  575:   :   90:
       ........  ........  ........  ........  ........  ........  ........  ........  ........
  45                                                                                                              45
       ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 44    :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :        44
       :  404:   :  217:   :  406:   :  219:   :  482:   :  484:   :  586:   :  589:   :  710:   :   15:
       ........CR........  ........CR........  ........CR........  ........CR........  ........CR........
  43         B2085               B2183               B2190               B2114               B2218                43
       ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...  ...FA...
 42    :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :  :LJW   :        42
       :  216:   :  405:   :  218:   :  407:   :  286:   :  483:   :  588:   :  587:   :  711:   :   14:
       ........  ........  ........  ........  ........  ........  ........  ........  ........  ........

JY/IX         42        44        46        48        50        52        54        56        58        60        IX/JY
          41        43        45        47        49        51        53        55        57        59
```

FIGURE 4.3-2

SHEET 10 of 10

FUEL LOAD ARRANGEMENT

SSES UNIT 2 CYCLE 1
ENRICHMENT BY SERIAL NUMBER

0.71 bundles (92)	LJW002 thru LJW093
1.76 bundles (240)	LJW094 thru LJW333
2.19 bundles (432)	LJW334 thru LJW765

4.4 (ST4) FULL CORE SHUTDOWN MARGIN

The objective of this test is to demonstrate that the reactor will be subcritical throughout the first fuel cycle with any single control rod fully withdrawn. The results demonstrated that the measured shutdown margin was 2.647% delta k/k and criticality occurred within 1% delta k/k of the predicted critical rod pattern. All level 1 and level 2 acceptance criteria were satisfied.

This test was performed by withdrawing control rods in the B-2 sequence until criticality and then establishing a steady positive period. The reactor went critical on rod 18-43 notch position 6 for a total of 2290 notches.

The period as calculated from the equation $T = \dfrac{t}{\ln \frac{(P)}{(PO)}}$

was 233 sec. Average coolant temperature during this period was 111.4°F.

The equation used to calculate shutdown margin is:

$$\text{rho (SDM)} = \frac{\text{Keff (RODS)}}{\text{Keff (RODS)}} - \frac{\text{Keff (SRO)}}{\text{Keff (SRO)}} + \text{rho (temp)} - \text{rho (period)}$$

Keff (SRO) is the value of Keff predicted with the strongest rod out (.9705), and Keff (RODS) is the value of Keff predicted with the stable period rod pattern (.99880). Based on a period of 233 sec. and 111.4°F moderator temperature, rho (temp) and rho (period) is -23.87×10^{-4} DELTA k/k and 3.41×10^{-4} DELTA k/k respectively.

The Level 1 Acceptance Criterion for this test was: The shutdown margin of the fully loaded, cold (68°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 determined strongest rod (or its reactivity equivalent) withdrawn. If the shutdown margin is measured at some time during the cycle other than the most reactive time, compliance with the above criterion is shown by demonstrating that the shutdown margin is 0.38% delta k/k plus an exposure dependent correction factor which corrects the shutdown margin at that time to the minimum shutdown margin". Analysis by General Electric has determined that the minimum core shutdown margin for the initial fuel load occurrs at the beginning of life; therefore, the exposure dependent correction factor was zero. The calculated minimum shutdown margin based on test results was 2.647% delta k/k thus satisfying the Level 1 Acceptance Criterion.

The Level 2 Acceptance Criterion for this test was: "Criticality should occur within 1.0% delta k/k of the predicted critical rod configuration". Criticality was achieved on 2290 notches which is between 1488 and 2568 notches which represents predicted critical rod configuration ± 1.0% delta k/k. Thus, the Level 2 Acceptance Criterion was satisfied.

4.5 (ST5) CONTROL ROD DRIVE SYSTEM

The control rod drive system was tested before fuel load, during fuel load, during heatup and at rated pressure to show that there was no significant binding of the control rods or drive mechanisms either initially or during plant heatup. After freedom of movement was verified at zero and rated reactor pressure; each individual control rod was scrammed to obtain the scram times. The slowest rods were then identified and tested further during the program by scramming and stroke timing to verify system reliability. Adequate performance of the CRD equalizing valves was also verified. The following Acceptance Criteria were verified during this test:

LEVEL 1

1. Each CRD must have a normal withdraw speed indicated by a full 12-foot stroke in greater than or equal to 40 seconds.

2. The mean (average) 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-enerigzed.)

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

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

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

4. The maximum scram time of each CRD from the fully withdrawn position to notch position 05, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

LEVEL 2

5. Each CRD must have normal insert and withdrawal speed indicated by a full 12-foot stroke in 40 to 60 seconds.

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

7. The differential pressure as measured between the cooling water header and the exhaust water header will be limited to 90 psid, with the cooling water header pressure referenced as the high side, measured two minutes following a scram reset from rated conditions.

ST5.1 Insert and Withdraw Checks

The control rod insert and withdraw times were checked for each control rod prior to fuel load with control rod blade guides installed and during fuel load after fuel was loaded around each control rod. Acceptable stroke times demonstrated control rod freedom of movement and proper operation of the directional control valves. Although not part of the Acceptance Criteria, other parameters were checked during this test such as rod coupling, Rod Position Indication System operation and drive water flow with the rod moving and stalled. Acceptance Criteria 1 and 5 were successfully verified during this test.

ST5.2 Friction Measurements

The friction test detects defects in directional control valves and excessive CRD friction by the measurement, analysis and comparison of CRD piston-over (PO) and piston-under (PU) differential pressure (dp). The dp measurements are obtained by connecting special test equipment to each HCU. The friction test was conducted by measuring the differential pressure between drive water insert and withdraw lines during the continuous insertion of each CRD. For any CRD whose differential pressure variation exceeded 15 psid during a continuous insertion between notch positions 48 and 02, a settling test was performed. ST5.2 was performed at zero and rated reactor pressure for all control rods.

At zero reactor pressure, all control rods passed the friction test except rod 50-47. A settling test was performed on this rod with satisfactory results.

At rated reactor pressure, 23 control rods failed the friction test. A settling test was performed on these rods with satisfactory results on all rods but 50-47. GE engineering reviewed the data and considered the results acceptable because the differential pressures exceeded the criteria by only a small amount and there was no indication of a system malfunction. Acceptance Criterion 6 was verified in this test.

ST5.3 Zero and Rated Reactor Pressure Scram Testing of Individual Rods

Existing test switches at the HCU were used to scram each individual control rod. Measurement of the scram time of each rod was obtained through the use of a chart recorder at the scram timing panel.

At zero reactor pressure the slowest control rod time was 1.58 seconds from position 48 to position 05. At rated reactor pressure, the slowest control rod time was 3.29 seconds from position 48 to position 05. The scram times on each instance were small enough that Acceptance Criteria 2, 3 and 4 were easily met. The slowest four control rods were selected for further testing in ST 5.5, 5.6 and 5.7.

ST5.5 Scram Testing of Selected Rods

The test method was the same as for ST5.3. This test was performed at the following test conditions: at zero reactor pressure with accumulator pressure just above the low pressure alarm point; at 600 ± 50 psig reactor pressure with normal accumulator pressure; at 800 ± 50 psig reactor pressure with normal accumulator pressure; and at rated reactor pressure with the accumulator at 0 psig. Each control rod was scrammed three times at every test condition. The greatest elapsed scram time to position 05 observed during these individual control rod scrams was 2.85 seconds. Therefore, the scram times easily met the 7 second maximum. Acceptance Criterion 4 was verified during this test.

ST5.6 Insert - Withdraw checks of Selected Rods

The test method for this test was the same as for ST5.1. The control rods tested were those selected in ST5.3. When this test was performed at rated reactor pressure, control rod 50-51 required an adjustment at the HCU in order to obtain an acceptable stroke time. Acceptance Criteria 1 and 5 were verified during this test.

ST5.7 Scram Timing of Selected Rods During Planned Scrams of Startup Test Program

This test measured the scram time of the slowest control rods selected in ST5.3. This data was collected from various power levels during the Startup Test Program in conjunction with planned full core scrams. The planned scrams were ST28.1, Shutdown and Cooldown Demonstration; ST27.1, Turbine Trip; and ST25.3, Full Isolation. The data taken during ST28.1 was lost due to operator error while recording with the Transient Monitoring System. The scram times were obtained simultaneously for all rods which were fully withdrawn. Of the selected test rods, the greatest elapsed scram time to position 05 was 2.4 seconds. Therefore the scram times easily met the 7 second maximum. Acceptance Criterion 4 was verified during this test.

ST5.8 Post - Scram Differential Pressure Measurements

This test consisted of measuring the differential pressure between the cooling water header and the exhaust water header during the period following a scram and scram reset. The test was performed at rated pressure. The test verified the correct functional operation of the CRD hydraulic system equalizing valves. Acceptance Criterion 7 was successfully verified during this test with a differential pressure at 45 psid versus the maximum Acceptance Criterion of 90 psid.

ELAPSED SCRAM TIME TO POSITION 05 IN SECONDS

[2] ROD	[3] 0 PSIG			[1] ROD	[4] 600 PSIG			[5] 800 PSIG			[6] RATED			27.1 (TC-3)		25.3 (TC-6)	
	1	2	3		1	2	3	1	2	3	1	2	3	[7] ROD	TIME	[7] ROD	TIME
26-35	1.59	1.60	1.59	26-35	2.4	2.4	2.4	2.64	2.68	2.54	2.63	2.49	2.47	14-11	2.56	14-11	2.46
30-31	1.59	1.59	1.58	30-31	2.4	2.3	2.3	2.49	2.51	2.53	2.49	2.51	2.49	30-19	2.53	30-19	[8]
30-43	1.61	1.61	1.62	30-23	2.6	2.5	2.5	2.77	2.60	2.60	2.85	2.47	2.50	34-23	2.56	34-23	2.18
34-55	1.62	1.62	1.63	50-51	2.3	2.4	2.3	2.25	2.27	2.33	2.40	2.49	2.56	34-47	2.48	34-47	2.47
														50-47	2.28	50-47	2.17

NOTES:

1. Four slowest rods which were completely withdrawn at time of test selected from results of ST 5.3 at zero reactor pressure.
2. Four slowest rods selected by ST 5.3 at zero reactor pressure.
3. Zero reactor pressure with accumulator pressure just above the low pressure alarm point.
4. 586 psig reactor pressure with normal accumulator pressure.
5. 850 psig reactor pressure with normal accumulator pressure.
6. Rated reactor pressure with the accumulator at 0 psig.
7. Four slowest rods which were completely withdrawn at time of test selected from results of ST 5.3 at rated reactor pressure.
8. Rod did not exhibit all notches, therefore time is unavailable. Rod will be retimed in a subsequent scram after the precommercial outage.

TABLE 4.5-1

CRD SCRAM TIMES

4.6 (ST7) REACTOR WATER CLEANUP SYSTEM

The Reactor Water Cleanup (RWCU) System was operated in the Blowdown, Hot Standby and Normal Modes. Satisfactory performance was demonstrated by comparing actual plant data during this operation with values from the G.E. process diagram. The different flow paths tested the capacity of the pumps, regenerative and non-regenerative heat exchangers and the bottom head drain line. The following Acceptance Criteria were verified during this test:

<u>Level 1</u>

None

<u>Level 2</u>

1. The temperature at the tube side outlet of the NRHXs shall not exceed 120°F in the Normal mode.

2. The RWCU pump available NPSH will be a minimum of 13 feet during the Hot Standby Mode as defined in the process diagram.

3. The cooling water flow to the non-regenitive heat exchangers shall be limited to 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 cooling water outlet temperature shall not exceed 180°F.

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

5. Bottom head flow indicator FI-2R610 shall indicate within 25 gpm of RWCU flow indicator FI-2R609 when total system flow is thru the bottom head drain.

6. The temperature at the tube side outlet of the NRHX's shall not exceed 130°F in the blowdown mode.

<u>7.1 Blowdown Mode Performance Verification</u>

The Reactor Building Closed Cooling Water (RBCCW) was aligned to provide the allowed flow to the non-regenerative heat exchanger. Then, the RWCU system was operated in the Blowdown Mode with partial system flow returning to the vessel to test the regenerative heat exchanger capacity. Next, the system was operated in the Blowdown Mode with no flow returning to the vessel to test the non-regenerative heat exchanger capacity. Acceptance Criteria 3 and 6 were verified during this test.

<u>7.2 Hot Shutdown Mode Performance Verification</u>

With the recirculation pumps off, the RWCU system was operated in the Hot Shutdown Mode. Data was collected and calculations showed that adequate NPSH existed for both pumps. Acceptance Criterion 2 was verified during this test.

7.3 Normal Mode Performance Verification

The RWCU system was operated in the normal flow path. The collected data demonstrated that system design flow could be met with cooling water temperatures within their design limits. Acceptance Criteria 1 and 3 were verified during this test.

7.4 Calibration Verification of Reactor Bottom Head Flow Indicator

The RWCU system was aligned so that all system flow went through the bottom head drain flow line. A comparison was made at four different flows to verify that the bottom head drain flow indicator read within 25 gpm of the system flow indicator. Acceptance Criterion 5 was verified during this test.

7.5 Initial Drain Line Temperature Data

With 100% core flow and the RWCU system operating in its normal mode, the Bottom Head Drain Valve, HV-2F101, was opened to increase the flow from the bottom head region. The bottom head drain line temperature sensor was found to read within 8°F of the recirculation loop suction temperature. Acceptance Criterion 4 was verified during this test.

The reactor water cleanup system met the operating requirements specified in the Acceptance Criteria thereby demonstrating acceptable capacity of the pumps and heat exchangers and acceptable operation of various temperature and flow indicators.

4.7 (ST8) RESIDUAL HEAT REMOVAL SYSTEM

The objectives of this test were to demonstrate the ability of the Residual Heat Removal (RHR) System to remove heat from the reactor system so that refueling and nuclear system servicing can be performed, and to condense steam while the reactor is isolated from the main condenser.

The following Acceptance Criteria were verified during this test.

Level 1

1. The transient response of any system-related variable to any test input must not be divergent.

Level 2

2. The RHR System shall be capable of operating in the SUPPRESSION POOL COOLING MODE at 35.2 MBTU/HR. Each RHR loop shall be tested independently in this mode.
3. The RHR System shall be capable of operating in the STEAM CONDENSING MODE at 107 MBTU/HR per heat exchanger for dual HX operation and 155 MBTU/HR for single HX operation. Both simultaneous operation of RHR loops and single loop operation shall be tested in this mode.
4. The RHR System shall be capable of operating in the SHUTDOWN COOLING MODE. Both dual loop operation and single loop operation shall be tested in this mode.
5. The decay ratio for system related variables containing oscillatory modes of response must be less than or equal to 0.25.

ST8.1 Suppression Pool Cooling Mode

The RHR heat exchanger capacity in the Suppression Pool Cooling Mode was demonstrated to be 41×10^6 Btu/hr for the A loop and 47×10^6 Btu/hr for the B loop when the suppression pool temperature was between 80 and 82°F. These capacities exceeded the required minimum. Acceptance Criterion 2 was verified during this test.

ST8.2 Steam Condensing Mode

The objective of this test was to demonstrate that RHR could be operated in the Steam Condensing Mode to demonstrate sufficient RHR heat exchanger capacity.

This test was performed with the reactor at rated pressure. The RHR Heat Exchangers were accepting steam from the reactor via the HPCI steam line. Condensate was returned to the reactor via the RCIC system. The RHR heat exchangers were tested singly and simultaneously to verify the heat exchanger cooling capacities.

The RHR system was demonstrated to operate in the STEAM CONDENSING MODE with a heat removal capacity of $\geq$107 MBTU/HR per heat exchanger operation and $\geq$155 MBTU/HR in single heat exchanger operation. Acceptance Criterion 3 was verified during this test.

ST8.3 Shutdown Cooling Mode

This test demonstrates the operation of the RHR system in the SHUTDOWN COOLING MODE. The test has been performed satisfactorily for the A loop and is scheduled to be performed on the B loop and in dual loop operation.

Acceptance Criteria 4 will be verified in these tests.

ST8.4 Steam Condensing Mode Stability Test

This test was performed at 96% reactor thermal power individually for each RHR heat exchanger. Maximum allowable step changes were made to heat exchangers level and pressure and the response was recorded by the transient recording system (GETARS). The transient plots were then analyzed to verify that all control system related variables behaved in a manner consistent with design parameters. Acceptance Criteria 1 and 5 were verified during this test.

4.8 (ST 9) WATER LEVEL REFERENCE LEG TEMPERATURE

The results of the testing verified the accuracy of the reference leg temperature value used during the reactor water level instrument calibrations. Acutal references leg temperatures were measured and compared to assumed values used in the calibrations. This difference was correlated to a percent of scale endpoint span error.

The Acceptance Criteria were as follows:

<u>Level 1</u>

None

<u>Level 2</u>

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

<u>ST9.1 Water Level Instrument Calibration Verification</u>

In this subtest, temperatures were recorded inside and outside the drywell in the vicinity of the level instrument reference legs for comparison to the assumed temperatures used in instrument calibration. The test was performed during heatup at rated temperature and pressure. For each level instrument, the temperature difference was less than that required for a 1% scale endpoint error thus verifying the Acceptance Criterion.

4.9 (ST10) SRM AND IRM PERFORMANCE AND CONTROL ROD SEQUENCE

The objectives of this test were to demonstrate that the operational sources, SRM and IRM instrumentation and rod withdrawal sequences provide adequate information to achieve criticality and increase power in a safe and efficient manner for each of the specified rod withdrawal sequences and to adjust the Intermediate Range Monitor System as necessary to obtain the desired overlap with the SRM and APRM systems.

The Acceptance Criteria were as follows:

Level 1

1. The overlap between the SRM and IRM shall be at least 1/2 decade.

2. The overlap between the IRM and APRM shall be at least 1/2 decade.

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

4. There must be a minimum count rate of 3 counts/second on the required operable SRMs.

5. The IRMs must be on scale before the SRMs exceed the rod block setpoint.

ST 10.1 IRM-SRM Overlap Verification

The results of this subtest verified the overlap between the IRMs and SRMs. During the first performance of this subtest, a relatively high noise level was noted in the IRMs. After maintenance, the subtest was reperformed and proper overlap was clearly verified. Acceptance Criteria 1 and 5 were verified in this test.

ST 10.2, IRM-APRM Overlap Verification

The results of this subtest verified correct overlap between the IRM and APRM system. The subtest was performed during initial power increase and following initial calibration of the APRMs. IRM gain adjustments were unnecessary in both cases and Acceptance Criterion 2 was verified in this test.

ST 10.3 Signal to Noise Ratio and Minimum Count Rate Determination

The results of this subtest verified adequate SRM signal-to-noise ratio and minimum count rate. The lowest signal-to-noise ratio was 28.8:1 and the lowest count rate on a fully inserted SRM channel was 19 counts/second. Acceptance Criteria 3 and 4 were verified in this subtest.

4.10 (ST 11) LPRM CALIBRATION

The purpose of this test (ST 11) was to calibrate the Local Power Range Monitoring (LPRM) System such that the meters read proportional to the thermal neutron flux at the location of the detectors.

The Acceptance Criteria are as follows:

<u>Level 1</u>

None

<u>Level 2</u>

Each LPRM will be within 10% of its calculated value

At Test Condition 1, a complete LPRM calibration was performed without the aid of the process computer. A full set of TIP traces were made, and these were digitized and manually input into BUCLE (Backup Core Limits Evaluation) to calculate initial LPRM "Gain Adjustment Factors (GAF). The amplifier input calibration currents for each LPRM detector divided by its GAF would result in the input amplifier current which would yield a final GAF equal to 1.00. Based on the GAF from BUCLE, the input amplifier currents for the appropriate detectors were adjusted and another full set of TIP traces was taken to verify the calibration. Of the 172 LPRMs, 87 showed GAFs of 1 ± .1.

At Test Condition 2, an LPRM calibration was performed with the process computer. Program OD-1 was used to determine and store LPRM computer calibration constants that are proportional to the TIP readings at the time of the OD-1. Program P-1 was then used to calculate LPRM GAFs. The input amplifier currents were then adjusted and OD-1/P-1 was repeated. All but 4 bypassed LPRMs satisfied level 2 criteria. The 4 bypassed LPRMs were 32-17C, 08-33A,
48-41C and 32-49C.

At Test Condition 3 another LPRM calibration was done with same procedure as Test Condition 2. 168 LPRMs satisfied level 2 acceptance criteria. The 4 exceptions were the same bypassed LPRMs as during Test Condition 2.

At Test Condition 6 a calibration was again performed. Only LPRMs 32-17C and 08-33A were bypassed and all other 170 LPRMs met the level 2 criteria.

This test demonstrated the ability to calibrate the LPRM system with and without the aid of the process computer.

4.11 (ST 12) APRM CALIBRATION

The objectives of this test was to calibrate the Average Power Range Monitor (APRM) System. The Level 1 Acceptance Criterion was that "APRM Channels must be calibrated to read greater than or equal to actual core thermal power". This Acceptance Criterion was satisfied at all Test Conditions tested.

The APRM channels are calibrated by calculating the core thermal power (based on heat balance data) and adjusting the individual APRM channels amplifiers to indicate this value in units of percent of rated thermal power (3293 Mwt). However, a gain adjustment factor (scaling factor) will be used for the APRM's when the maximum fraction of limiting power density (MFLPD) is greater than the fraction of rated power (FRP). The purpose of the APRM gain adjustment factor is to effectively lower the scram and rod block setpoints as required by Technical Specification 3.2.2. When the MFLPD is greater than the FRP, the APRM channels are adjusted to indicate 100 times the MFLPD value.

Prior to initial operation of the reactor, the gain of the APRM channels is set at the maximum value to ensure response during the initial plant heatup. To allow this gain value to be reduced and enable the plant to be brought to rated temperature and pressure it is necessary to perform an initial calibration. Note that in the STARTUP mode the APRM scram trip is at 15% on the APRM scale. This initial APRM calibration is based on heat balance data using the reactor coolant system temperature heatup rate. Due to the uncertainty of the data values in this calculation, this method of APRM calibration is used only during the initial heatup. When the power level approaches 20%, the uncertainty in the heat balance is greatly reduced and the normal steady state heat balance and data acquisition methods are used. After the initial startup it is not necessary to use the heatup rate heat balance method since the APRM channels were calibrated prior to shutdown and the amplifier gains are not changed during the shutdown period. Therefore it is not necessary to calibrate the APRM channels during the subsequent plant startup until 25% power is reached.

This test consists of two subtests. Subtest 12.1, Low Power APRM Calibration, was performed only during the initial heatup to enable the initial increase in power to 25% of rated. Subtest 12.2, High Power APRM Calibration, was performed at Test Conditions 1, 2, 3, 5 and 6 using the plant procedure SR-278-002 to perform the APRM Calibration. The succeeding discussion is divided into 2 parts to distinguish between initial heatup calibration and subsequent calibrations. The test methodologies used are different and the discussion is divided to emphasize this.

<u>ST 12.1 Low Power APRM Calibration</u>

The purpose of this test is to do an initial calibration of the APRM's while the moderator temperature is increasing. This was done by performing a heat balance on the reactor vessel to determine the core thermal power. This resulted in a calculated core thermal power of 19.87 MWt which is .603 percent of the rated core thermal power. After this calculation, the APRM readings were divided by the initial percent of rated power to come up with an APRM adjustment factor for each APRM. The APRM's were then adjusted by using this adjustment factor to

read higher than the actual percent of rated power. The APRM readings were all done on the "expand X10 scale" because of the low power level involved in performing this test. The APRM's were adjusted higher than the calibrated APRM value to ensure that the acceptance criterion would be met. Test results are shown below:

APRM	Initial Value (Expanded X10 Scale)	Adjustment Factor	Calibrated APRM Value	Final Value Expanded X10 Scale
A	2.55	.150	.382	.85
B	2.15	.176	.378	.7
C	2.65	.143	.379	.9
D	2.05	.176	.361	.7
E	2.15	.167	.359	.7
F	2.30	.162	.373	.75

By adjusting the APRM's to the final values as shown above, the level 1 criterion was met since this final value is higher than the power level indicated by the calibration APRM value.

ST 12.2 High Power APRM Calibration

This subtest requires the performance of Plant Reactor Engineering procedure SR-278-002 for calibration of the APRM channels based on core thermal power determined by core heat balance during the Startup Test Program. APRM calibration surveillance procedure SR-278-002 is normally performed weekly and may be performed on a more frequent basis and after each major change in power level.

The calibration of an APRM channel consists of adjusting the APRM amplifier gain to cause the indicated APRM value to be the desired value. Although the APRM channels are normally calibrated to indicate percent of rated core thermal power, a gain adjustment factor will be used when the maximum fraction of limiting power density (MFLPD) of any reactor fuel type is greater than the fraction of rated power (FRP). The purpose of the scale factor is to effectively lower the scram and rod block setpoints as required by Technical Specification 3.2.2. When a gain adjustment factor is applied the APRM channels are adjusted to indicate 100 times the MFLPD value.

The reactor core thermal power and the MFLPD was determined by on-line process computer programs OD-3 and P-1, or the appropriate backup methods before the on-line process computer program verification (ST 13) was completed. This backup method consisted of performance of RE-OTP-002, Core Thermal Power Evaluation (Backup Method). The backup method for MFLPD consisted of performance of RE-OTP-004 using the BUCLE program on an off-line computer.

Test conditions during which ST 12.2 was run and results of each test is tabulated in Tables 4.11-1 and 4.11-2. The Acceptance Criterion was met at all Test Conditions tested.

TABLE 4.11-1

TEST CONDITIONS FOR ST 12.2

TEST CONDITION	1	2	3	5	6
CORE FLOW (MLB/HR)	90	45	84.2	60	99.56
Rx POWER (%)	17.9	39.9	47	70.6	98.2
Rx DOME PRESSURE (psig)	945	940	976	969	987
GENERATOR POWER (MWe)	160	410	795	760	1079
DATE PERFORMED	7/04/84	12/02/84	9/08/84	9/25/84	9/29/84

TABLE 4.11-2

ST 12.2 RESULTS (In % Of Full Power, 3293 MWt)

	TC 1		TC 2		TC 3		TC 5		TC 6	
APRM CHANNEL	DESIRED VALUE	FINAL SET VALUE	DESIRED VALUE	FINAL SET VALUE	DESIRED VALUE	FINAL SET VALUE	DESIRED VALUE	FINAL SET VALUE	DESIRED VALUE	FINAL SET VALUE
A	17.9	36	39.9	39.9	47	48	70.6	70.9	98.2	98.3
C	17.9	38.5	39.9	39.9	47	47	70.6	71.6	98.2	99.1
E	17.9	39.5	39.9	39.9	47	47	70.6	71.3	98.2	98.7
B	17.9	37	39.9	39.9	47	47	70.6	71.4	98.2	98.7
D	17.9	38	39.9	39.9	47	47	70.6	71.2	98.2	98.3
F	17.9	34	39.9	39.9	47	48	70.6	71.1	98.2	98.6

4.12 (ST 13) PROCESS COMPUTER

The purpose of ST-13 is to verify the NSSS performance of the process computer under plant operating conditions. In particular, this test dealt with the dynamic system test case (DSTC), the OD-2 checkout and verification of the correct operation of the control rod symmetry flag used in P-1. The thermal limits from P-1 and LPRM GAFs are compared to the results from BUCLE.

The following Acceptance Criteria were verified during the performance of these tests:

<u>Level 1</u>

None

<u>Level 2</u>

1. The MCPR calculated by BUCLE and process computer either:

   a. Are in the same fuel assembly and do not differ in value by more than 2% or

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

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

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

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

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

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

   b. For the case in which the MAPLHGR calculated by the process computer is in a different assembly than that calculated by BUCLE, for both assemblies, the MAPLHGR and APLHGR calculated by the two methods shall agree within 2% for the same assembly.

4. The LPRM calibration factors calculated by BUCLE and the process computer agree to within 2%.

ST 13.1 Dynamic System Test Case

In this subtest, proper operation of the following NSS programs was verified by demonstration of operation and comparison to manual calculations of computer values:

OD-1, Whole Core LPRM Calibration and BASE Distribution
OD-3, Core Thermal Power and APRM Calibration
OD-7, Present Control Rod Positions
OD-8, Present LPRM Readings
OD-15, Computer Shutdown and Outage Recovery Monitor
P-1, Periodic Core Evaluation
P-2, Daily Core Performance Summary
P-3, Monthly Core Performance Summary
P-5, Drifting LPRM Diagnostic

The thermal limit and LPRM GAF comparisons were performed twice in this subtest and are tabulated as follows:

DSTC First Comparison

	Location	*Location Code	P/C Value	BUCLE Value	Difference	Acceptance Criteria
MCPR	25-26	1, 2	2.757	2.771	0.51%	2%
MLHGR	25-44-11	1	5.26	5.28	0.38%	2%
MLHGR	25-44-12	2	5.226	5.28	1.03%	2%
MAPLHGR	19-36-12	1, 2	4.55	4.57	0.44%	2%

Maximum LPRM GAF Difference = 1.16% 2%

*Location Code: 1 - location calculated by Process Computer (P/C)
2 - location calculated by BUCLE

DSTC Second Comparison

	Location	*Location Code	P/C Value	BUCLE Value	Difference	Acceptance Criteria
MCPR	25-26	1	2.723	2.755	1.17%	2%
MCPR	43-34	2	2.753	2.746	0.25%	2%
MLHGR	25-44-11	1, 2	5.30	5.31	0.19%	2%
MAPLHGR	46-26-12	1	4.59	4.58	0.22%	2%
MAPLHGR	17-28-12	2	4.58	4.59	0.22%	2%

Maximum LPRM GAF Difference = 1.16% 2%

The process computer accurately performed its design calculations and Acceptance Criteria 1, 2, 3 and 4 were verified in this subtest.

ST 13.2 Specified LPRM Substitute Value and Base Distribution

The purpose of this subtest was to verify that new TIP data and new BASE values are properly calculated and stored after an OD-2 is performed. No Level 1 or Level 2 Acceptance Criteria are verified in this subtest. The test was performed with satisfactory results.

ST 13.3 Bundle Power Symmetry

This subtest consists of comparing the process computer calculated values to the BUCLE calculated values performed with the symmetry flag set to represent both symmetric and asymmetric control rod pattern. Results are tabulated below:

Asymetric Comparison

	Location	*Location Code	P/C Value	BUCLE Value	Difference	Acceptance Criteria
MCPR	51-40	1,2	1.996	1.994	0.1%	2%
MLHGR	9-20-4	1,2	9.44	9.45	0.1%	2%
MAPLHGR	9-20-4	1,2	8.20	8.21	0.1%	2%
Maximum LPRM GAF Difference = 1.02%						2%

The process computer accurately performed its design calculations and Acceptance Criteria 1, 2, 3 and 4 were verified in this subtest.

Overall, the Unit 2 computer testing went much smoother than Unit 1 computer testing. This is due to the incorporation of Unit 1 software revisions into the Unit 2 software.

4.13 (ST 14) REACTOR CORE ISOLATION COOLING SYSTEM

The Reactor Core Isolation Cooling (RCIC) system demonstrated proper operation at the minimum and rated operating pressures and flow ranges. Reliability in the automatic quick starting mode from cold conditions was also demonstrated with the reactor at rated conditions and at 150 psig.

The following Acceptance Criteria were verified during this test:

Level 1

1. The average pump discharge flow must be equal to or greater than 600 gpm after 30 seconds have elapsed from automatic initiation at any reactor pressure between 150 (+15, -0) psig and rated.

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

Level 2

1. In order to demonstrate a margin to overspeed and isolation trips, the speed peak resulting from the initial start and subsequent speed peaks shall be less than or equal to 4809 rpm.

2. 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.

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

4. The delta P switch for the RCIC steam supply line high flow isolation trip shall be calibrated to a differential pressure corresponding to less than or equal to 300% of the maximum required steady state flow, with the reactor assumed to be near the pressure for main steam relief valve actuations.

The RCIC system demonstrated its reliability by always achieving rated flow within the allowed 30 seconds, and by never tripping during auto start tests. The turbine did trip once during a manual start, which was attributed to air in the servo control valve following maintenance to the control valve. The other minor problems that did occur were all Level 2 Acceptance Criteria failures and are described below.

A steam leak was observed at the RCIC turbine high pressure end during initial testing. This leak was small enough so as not to affect turbine operation and could not be found during subsequent testing.

During heatup while performing ST 14.2 a speed peak of 5167 RPM was observed which failed the maximum speed peak of 4809 critera. Investigations led to a damaged speed sensing pickup, which was replaced, and the decision was made to continue testing after inspections of the EGR and governor linkage. In TC-2 the speed peak repeated during two runs of ST 14.3. A determination was made to shorten the governor valve stroke to 5/8" and to replace the EGR mechanism. A complete set of retests was performed. All further testing resulted in acceptable speed peaks.

For all but the final two rated pressure RCIC tests, the delta pressure steam line switches, PDSH-2N017 and PDSH-2N018, were set higher than the calculated setpoints. The plant setpoints were lower than the setpoint listed in Tech Spec Table 3.3.2-2 as required. The Tech Spec setpoint Table states that the value listed is an initial setpoint, and the final setpoint is to be determined during the startup test program. The final setpoints are still under evaluation, and any changes shall be submitted to the commission.

Dates, Test Conditions and results of RCIC testing is shown on Table 4.13-1. All Acceptance Criteria were satisfied.

				LEVEL 1		LEVEL 2			
DATE	TEST CONDITION	TEST	PRESSURE	TIME TO RATED FLOW < 30 SEC	TRIP ?	SPEED PEAK < 4809	OSCILLATIONS	NO SEAL LEAKAGE	CALC. ΔP SWITCH SETTING > SETPOINT
6-14-84	HEATUP	14.1	150 #	YES	NO	2650	NONE	NONE	----------
6-14-84	HEATUP	14.4	150 #	YES	NO	2550	NONE	NONE	----------
6-24-84	HEATUP	14.1	RATED	YES	NO	4733	NONE	NONE	SEE NOTE 1
6-24-84	HEATUP	14.2	RATED	YES	NO	FAILURE 5167	NONE	VERY SMALL	SEE NOTE 1
7-11-84	TC-2	14.3	RATED	YES	NO	FAILURE 4913	NONE	NONE	SEE NOTE 1
7-24-84	TC-2	14.3	RATED	YES	NO	FAILURE 4900	NONE	NONE	SEE NOTE 1
8-03-84	TC-2	14.1	RATED	YES	FAILURE	3840	NONE	NONE	SEE NOTE 1
8-04-84	TC-2	14.1	RATED	YES	NO	4520	NONE	NONE	SEE NOTE 1
8-05-84	TC-2	14.2	RATED	YES	NO	4480	NONE	NONE	SEE NOTE 1
8-18-84	TC-3	14.3	RATED	YES	NO	3936	NONE	NONE	YES
8-21-84	TC-3	14.3	RATED	YES	NO	3959	NONE	NONE	YES
9-03-84	TC-3	14.1	150 #	YES	NO	2564	NONE	NONE	----------
9-03-84	TC-3	14.4	150#	YES	NO	2500	NONE	NONE	----------

NOTES:

1. CALCULATED SETTINGS WERE BELOW SETPOINT. PLANT SETPOINT IS BELOW TECH SPEC SETPOINT AS REQUIRED. HOWEVER, THE TECH SPEC HAS A FOOTNOTE STATING THAT THE FINAL TECH SPEC SETPOINT IS TO BE DETERMINED BY THE STARTUP TEST PROGRAM.

----- INDICATES CRITERION NOT APPLICABLE TO THIS TEST.

DESCRIPTIONS:

14.1 CST TO CST FLOW STEPS AND AUTO QUICK START TO CST.

14.2 VESSEL INJECTION. FLOW STEPS AND AUTO QUICK START TO THE VESSEL

14.3 VESSEL INJECTION AT RATED. AUTO QUICK START TO THE VESSEL WITH RCIC TURBINE COLD.

14.4 VESSEL INJECTION AT 150 #. AUTO QUICK START TO THE VESSEL.

TABLE 4.13-1

RCIC TEST CONDITIONS AND RESULTS

4.14 (ST 15) HIGH PRESSURE COOLANT INJECTION SYSTEM

Proper operation of the High Pressure Coolant Injection (HPCI) system was demonstrated at the minimum and rated operating pressures and flow ranges. Reliability in the automatic quick starting mode from cold conditions was also demonstrated with the reactor at rated conditions.

The following Acceptance Criteria were verified during this test:

<u>Level 1</u>

1. The average pump discharge flow must be equal to or greater than 5000 gpm after 25 seconds have elapsed from automatic initiation at any reactor pressure between 150 psig and rated.

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

<u>Level 2</u>

1. In order to demonstrate a margin to overspeed and isolation trips, the following criteria shall be met: (a) the speed peak resulting from the initial start shall be less than or equal to 4543 RPM and (b) subsequent speed peaks shall be less than or equal to 5% above the rated speed, or 4336 rpm.

2. 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.

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

4. The delta P switch for the HPCI steam supply line high flow isolation trip shall be calibrated to actuate at no greater than 300% of the maximum required steady state flow, with the reactor assumed to be near the pressure for main relief valve actuation.

5. The HPCI Pump Net Positive Suction Head (NPSH) shall be at least 21 feet at a flow rate of at least 5000 gpm with the cooling water valve open while taking suction from the Condensate Storage Tank (CST). The NPSH calculations must be corrected for 100°F suction temperature and the CST water level at the level where the HPCI suction automatically swaps to the suppression pool.

The HPCI system demonstrated its reliability by never tripping or isolating during testing and by achieving rated flow within the allowed 25 seconds in 5 out of 6 tests. In the sixth test, ST 15.3 on 9-25-84, the system required 26.3 seconds to achieve rated flow. The Tech Spec limit of 30 seconds not violated. Investigation into the problem resulted in an Environmental Upgrade Modification and a replacement of the mechanical overspeed trip mechanism. The Environmental Upgrade Modification involved replacing the EGR, the servo on the control valve, the temperature control valve on the lube oil cooler and the turbine trip solenoid valve and was done during the Pre-Commercial Operations Outage. All HPCI testing will be repeated after the Pre-Commercial Outage.

The two other problems that did occur were both Level 2 Acceptance Criteria failures. The initial run of ST 15.1 yielded a subsequent speed peak of 4440 rpm, which was above the limit of 4336 rpm. The HPCI flow controller was tuned up, and since ST 15.1 at 150# had already been run, ST 15.1 at both 150# and rated were repeated. The other problem which also surfaced during the initial test concerned a low NPSH value caused by the startup strainer never being removed from the suction line. Upon removal, the NPSH value was acceptable.

Dates, Test Conditions and results of HPCI testing is shown on Table 4.14-1. All Acceptable Criteria were satisfied except for the time to rated flow failure mentioned previously. All HPCI testing will be repeated after the Pre-Commercial Outage.

				LEVEL 1		LEVEL 2				
DATE	TEST CONDITION	TEST	PRESSURE	1 TIME TO RATED FLOW <25 SEC	2 TRIP ?	1 SPEED PEAK INITIAL / SUBSEQUENT 4543 4356	2 OSCILLATIONS	3 NO SEAL LEAKAGE	4 ΔP SWITCH SETTING	5 NPSH >21 FEET
6-27-84	HEATUP	15.1	RATED	YES	NO	FAILURE <4543 / 4440	NONE	NONE	PASS	FAILURE 19.0 FEET
7-07-84	TC-2	15.1	150 #	YES	NO	2950 / 2950	NONE	NONE	----	45 FEET
8-06-84	TC-2	15.1	RATED	YES	NO	2950 / 4210	NONE	NONE	PASS	44 FEET
9-03-84	TC-3	15.1 [1]	150 #	YES	NO	2900 / 2900	NONE	NONE	----	45 FEET
9-20-84	TC-3	15.2	RATED	YES	NO	1900 / 4050	NONE	NONE	PASS	--------
9-27-84	TC-6	15.3	RATED	FAILURE 26.3 SEC	NO	3933 / 4041	NONE	NONE	PASS	--------

NOTES:

1. ST 15.1 at 150 # repeated due to HPCI flow controller tune-up as per TER 076.

---- Indicates criterion not applicable to this test.

DESCRIPTIONS:

15.1 CST TO CST. FLOW STEPS AND AUTO QUICK START TO CST.

15.2 VESSEL INJECTION. FLOW STEPS AND AUTO QUICK START TO VESSEL.

15.3 VESSEL INJECTION. AUTO QUICK START TO THE VESSEL WITH THE HPCI TURBINE COLD.

TABLE 4.14-1
HPCI TEST CONDITIONS AND RESULTS

4.15 (ST 16) SELECTED PROCESS TEMPERATURES

The objectives of this test was to identify any reactor operating modes that cause temperature stratification and to determine the proper setting of the low flow control limiter for the recirculation pumps to avoid coolant temperature stratification in the reactor pressure vessel bottom head region. The Acceptance Criteria which were proven by this test are as follows:

<u>Level 1</u>

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

2. The recirculation pump in an idle loop must not be started unless the loop suction temperature is within 50°F of the active loop.

3. The recirculation pump in an idle loop must not be started unless the operating loop flow rate is less than or equal to 50% of rated loop flow.

4. When both loops have been idle, an idle recirculation loop shall not be started unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant within the reactor pressure vessel is less than or equal to 50°F.

At Initial Heatup, ST 16.1, Minimum Recirculation Pump Speed Determination was performed to establish the minimum allowable recirculation pump speed and the setting of electrical low speed limiters on the scoop tube positioners. The data for this test was gathered by decreasing pump speed to the minimum where there is no sudden increase in different temperature or unstable pump speed control and before and after recirculation pump trips. There were no Acceptance Criterion associated with this test.

During Test Conditions 3, 4 and 6, ST 16.2, Recirculation Pump Trip Recovery Data, was performed during all planned and unplanned recirculation pump trips. This test verified in all Test Conditions that adequate mixing is occurring in the vessel bottom head during recirculation pump trips such that there was not the potential for thermal shock to the vessel when the pump was restarted. There were no outstanding problems during the running of this test and all Acceptance Criteria were verified.

4.16 (ST 17) SYSTEM EXPANSION

The results of the testing showed that the main steam inside containment piping, reactor recirculation system piping and balance-of-plant piping scoped for system expansion testing in the Startup Test Program per FSAR Table 3.9-33 was free to move without unplanned obstruction or restraint during heatup and cooldown, that the system piping behaved in a manner consistent with assumptions of the stress analysis, and that there was agreement between calculated and measured values of displacement.

System expansion monitoring of piping systems and pipe restraining devices took place during the initial plant heatup, initial heatup and cooldown of designated systems, and subsequent to plant cooldown during the Unit 2 Pre-Commercial Outage. Data was recorded on GETARS (transient recording system) from remotely mounted displacement instrumentation located on piping for system expansion testing. Recorded data was compared with design calculated values to determine acceptable piping movement. For balance-of-plant systems scoped for system expansion testing in the Startup Test Program, per FSAR Table 3.9-33, that were accessible during plant operation and hence need not be remotely instrumented, examination and manual measurements were performed by the qualified test engineers to determine acceptable piping movement.

The Acceptance Criteria were as follows:

<u>Level 1</u>

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

The measured displacements at the established remote instrumented locations on Main Steam (inside drywell) and recirculation piping shall not exceed the allowable values calculated for the specific points.

Balance-of-plant piping systems scoped for testing per FSAR Table 3.9-33 will not be restrained against thermal expansion during the test, except by design intent.

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

Snubbers shall not become extended or compressed to the limits of their total travel.

<u>Level 2</u>

The measured displacements at the established remote instrumented locations on Main Steam (inside drywell) and recirculation piping shall not exceed the expected values calculated for the specific points.

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

For balance-of-plant piping systems scoped for testing per FSAR Table 3.9-33, the measured deflections, when plotted against the calculated deflections for the specific points, shall fall within their calculated acceptable range.

The change of location of the balance-of-plant piping systems scoped for testing per FSAR Table 3.9-33, after the testing had been completed and the piping has returned to its start-of-test temperature, will not be more than ±25% of the total measured deflection during the testing.

System expansion testing was performed for the piping systems or portions of piping systems listed below during the Startup Test Program:

1. Main Steam piping inside and outside primary containment.

2. Reactor Recirculation system piping.

3. Reactor Water Cleanup system piping inside and outside primary containment.

4. High Pressure Coolant Injection system steam supply piping inside and outside primary containment.

5. Reactor Core Isolation Cooling system steam supply piping inside and outside primary containment.

6. Core Spray system pump discharge piping inside primary containment.

7. Residual Heat Removal system supply, return and head spray piping inside containment.

8. Feedwater system piping inside and outside primary containment.

9. High Pressure Coolant Injection system pump discharge piping to feedwater line outside containment.

10. Reactor Core Isolation Cooling system pump discharge piping outside containment.

11. Residual Heat Removal system outside primary containment.

All piping remote displacement instrumentation was initially zeroed prior to commencement of initial reactor heatup. Piping not remotely instrumented was reference marked at each observation point in its cold condition.

System expansion testing for (1) through (7), listed above, was performed during initial reactor heatup at reactor coolant temperatures of 275°F, 450°F and rated reactor temperature and pressure.

System expansion testing for (8) through (10), listed above, was performed when feedwater system temperature was 260°F and 387°F (rated). These system temperatures occurred during Test Conditions 2 and 3, respectively.

System expansion testing for (11), listed above, was performed when the Residual Heat Removal system was operated in its Steam Condensing and Shutdown Cooling modes of operation, with the exception of the RHR B Loop of Shutdown Cooling which will be performed and discussed in a supplement to this report.

All piping tested during the Startup Test Program, as stated above, was finally re-examined following reactor shutdown during the Pre-Commercial Outage to determine that subsequent relaxing of piping systems after the heatup/cooldown thermal cycle was as expected.

Problems encountered during the Start-up Test Program System Expansion testing were for the most part minor in nature and include the following:

1. R.W.C.U. Piping Rework - This rework was performed to eliminate a possible interference that existed between R.W.C.U. piping/supports and the Drywell Sumps. Work included modification of the sumps and the concrete blocks inside the sumps as well as rearrangement of R.W.C.U. supports in this area. Completion of the rework eliminated the interference.

2. Inspection/Exercise Snubber - Hanger DCA-210-H11 on RHR showed no movement during ST 17 testing. Action was to inspect and exercise snubber. During this work snubber found to be "bound", could not be exercised, and was replaced. Subsequent analysis was performed to evaluate to effect a rigid support would have on the piping, and no detrimental piping fatigue occurred during the subsequent heat-ups.

3. Reset of One Hanger - One hanger had to have its cold setting adjusted.

All supports reworked or adjusted account for less than 1% of those examined.

The performance, thus far, of ST 17 proved that piping design met all test objectives set forth in the FSAR.

4.17 (ST18) TIP UNCERTAINTY

The purpose of this test was to determine the total uncertainty of the TIP system readings. The test was conducted at Test Condition 3 (9/12/84)and at Test Condition 6 (10/7/84). The average total uncertainty for all test sets was 2.62%. Level 2 criteria ion that total TIP uncertainty obtained by averaging the uncertainties for all sets shall be less than 6%. The level 2 criterion was thus met. Plant conditions are given on table 4.17.1 and detailed results are given on table 4.17.2.

This test consisted of operation of the TIP system in conjunction with the process computer programs OD-1 and OD-2 to obtain and edit the TIP data necessary to determine TIP value uncertainties. All TIP data was taken with the reactor at steady state conditions and an octant symmetric rod pattern which only occurs in rod withdrawal Sequence A.

The random noise uncertainty was determined from successive TIP runs made at the common location (32-33) with each of the TIP machines making six runs at index position 10. The TIP data was obtained by simultaneous operation of the process computer OD-2 program which provides 24 nodal TIP values for each TIP traverse. The TIP values are in units of full power adjusted BASE values. The standard deviation of the random noise is derived by taking the square root of the average of the variances at nodal levels 5 through 22, where the nodal variance is obtained from the fractional deviations of the successive TIP values about their nodal mean value. Data analysis is performed using an off-line computer program. This program requires the manual input of 17 nodal values from each of the TIP runs edited by OD-2.

The total TIP uncertainty is determined by performing a complete set of TIP traverses as required by process computer program OD-1. The total TIP uncertainty is obtained by dividing the standard deviation of the symmetric TIP pair nodal ratios by the square root of 2. The nodal TIP ratio is defined as the nodal BASE value of the TIP in the lower right half of the core divided by its symmetric counterpart in the upper left half. Data analysis is performed using the off-line computer program. The program requires the input of the Process Computer Security Log (SECLOG) generated following the completion of OD-1.

TIP reprodücibly consists of a random noise component and a geometric component. The geometric component of TIP reproducibility is obtained by statistically subtracting the random noise component from the total TIP reproducibility. The geometric component is due to variation in the water gap geometry and TIP tube orientation from one TIP location to another. Measurement of these components is obtained by taking repetitive TIP readings at a single TIP location, and by analyzing pairs of TIP readings taken at TIP locations which are symmetrical about the core diagonal of fuel loading symmetry.

TABLE 4.17-1

PLANT CONDITIONS

Test Condition:	TC 6	Test Condition:	TC 3
Date Performed:	10/7/84	Date Performed:	9/12/84
Core Power (MWt):	3165.7 (96.1%)	Core Power (MWt):	2332 (70.8)
Generator Output (MWe):	1041.7 (94.9)	Generator Output (MWe):	753 (69.4)
Core Flow (Mlb/hr):	98.9 (98.9%)	Core Flow (Mlb/hr):	98 (98%)
Dome Pressure (PSIG):	982	Dome Pressure (PSIG):	958

TABLE 4.17-2

RESULTS

%POWER	RANDOM NOISE	GEOMTRICAL UNCERTAINTY	TOTAL UNCERTAINTY	TEST CONDITION
70.8%	2.514%	.808%	2.641%	3
96.1%	1.608%	2.413%	2.679%	6

The average total uncertainty is: 2.62%

Level 2 Criteria: $\leq$ 6%

4.18 (ST19) CORE PERFORMANCE

The core performance test is used to document the determination of the principal thermal and hydraulic parameters associated with core behavior. At each test condition the core thermal power and performance parameters were evaluated using the appropriate Reactor Engineering procedure. These values were compared to the test acceptance criteria which are based on Technical Specification limits for core performance parameters and the core thermal power limit based on the design flow control line. All test acceptance criteria were met for all Subtests (conducted at Test Conditions 1,2,3,4,5 and 6).

This Startup Test consists of two Subtests:

Subtest 19.1, BUCLE Calculation, documents the performance of RE-OTP-002 and RE-OTP-004 to determine core thermal power and core performance parameters respectively. RE-OTP-002, Core Thermal Power Evaluation (Backup Method) uses a manual calculation to compute core thermal power based on heat balance data from plant instrumentation. RE-OTP-004, Core Thermal Hydraulic Performance Evaluation (Backup Method) uses the off-line computer BUCLE (Backup Core Limits Evaluation) program to determine the core performance parameters. This off-line program requires core power, flow, inlet subcooling and reactor pressure determined in RE-OTP-002 and power distribution data from a complete set of Traversing Incore Probe (TIP) scans, LPRM readings and control rod position data. The actual calculation is identical to that performed by the process computer program, P1. This Subtest was performed at Test Conditions 1 and 2.

Subtest 19.2, Process Computer Calculation, documents the performance of RE-OTP-001, Core Thermal Power Evaluation (Computer Method) and RE-OTP-003, Core Thermal Hydraulic Performance Evaluation (Computer Evaluation) to determine thermal power and core performance parameters. These Reactor Engineering procedures use the process computer programs OD-1, OD-3 and P1, to store the core power distribution data from the TIP traverses, heat balance data from plant instruments, and perform the necessary calculation. This Subtest was performed at Test Conditions 3, 4, 5 and 6.

Acceptance Criteria for ST19 follows:

Level 1

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

The steady-state Minimum Critical Power Ratio (MCPR) shall not be less than the required Technical Specification value times the value of K(f).

The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) shall not exceed the limits given in Table 4.18-1.

Steady-state reactor power shall be limited to the rated MWt (3293 MWt) and values on or below the design flow control line.

ST19.1, Bucle Calculations

This Subtest documents the determination of the following parameters prior to the completion of Process Computer verification.

- Core Thermal Power (CTP)
- Maximum Linear Heat Generation Rate (MLHGR)
- Minimum Critical Power Ratio (MCPR)
- Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

Using RE-OTP-002, the core thermal power is determined by recording heat balance data for flows into and out of the reactor pressure vessel, available from plant instrumentation, and performing the calculations detailed on form RE-OTP-002-1. Core flow is available from jet pump instrumentation and is recorded on form RE-OTP-002-1.

Using RE-OTP-004, the core performance parameters (MLHGR, MCPR and MAPLHGR) are determined from the off-line computer program, BUCLE. This program requires input of the Traversing Incore Probe (TIP) data for each LPRM location, control rod position, core thermal power, core flow, inlet subcooling and reactor pressure. The TIP traces are obtained by using RE-OTP-011 to run a traverse of every TIP machine location including the reference channel in the common location for each TIP machine. The TIP trace data is entered into the BUCLE program as 24 nodal values for each TIP trace. The Control Rod Pattern is obtained by editing OD-7. The core thermal power, core flow, inlet subcooling and reactor pressure are entered or calculated on form RE-OTP-002-1.

The BUCLE program calculates the MLHGR, MCPR and MAPLHGR and then compares these values to the limits in Section 3.2 of the Technical Specification and determines a ratio of the calculated value divided by the limit, with the exception of MCPR which uses the limit value divided by calculated value. These ratios are MFLCPR for MCPR, MAPRAT for MAPLHGR, and MFLPD for MLHGR. Test conditions are provided in table 4.18-2 and test results are provided in table 4.18.3.

ST19.2, Process Computer Calculation

This Subtest documents the determination of the following parameters using the process computer to monitor plant data and perform the calculation.

- Core Thermal Power (CTP)
- Maximum Linear Heat Generation Rate (MLHGR)
- Minimum Critical Power Ratio (MCPR)
- Maximum Average Planar Linear Heat Generator Rate (MAPLHGR)

Using RE-OTP-001, the process computer program OD-3 is performed to determine and edit core thermal power based on plant heat balance data. This program also monitors and edits the core flow at the time the heat balance data is recorded.

Using RE-OTP-003, the process computer program P1 is performed to determine the core performance parameters (MLHGR, MCPR and MAPLHGR). The P1 program is edited to obtain the Periodic Core Performance Log. The P1 Program uses a stored data array to describe the core power distribution.

This data array, defined as BASE (L,K), is obtained by program OD-1 which uses the TIP values recorded during a scan of all TIP locations including the common location (Reference channel) for all TIP machines. The BASE values are then modified by changes occurring in the LPRM values following the operation of program OD-1. If significant changes occur in the BASE values at core location at or near the maximum LHGR value due to LPRM changes the edit of the P1 program contains BASE CRIT CODES. Program OD-2 can be used to update the BASE value for that TIP/LPRM location and "clear" the BASE CRIT CODES.

The P1 Program will compare the MLHGR, MCPR and MAPLHGR to the limits in section 3.2 of Technical Specification and determine a ratio of the calculated value divided by the limit with the exception of MCPR which uses the limit value divided by the calculated value. These ratios are MFLCPR for MCPR, MAPRAT for MAPLHGR and MFLPD for MFHGR.

Test conditions are provided in table 4.18-4 and test results are provided in table 4.18-5.

MAPLHGR Limit Versus Average Planar Exposure

Table 4.18-1

Average Planar Exposure MWD/T	8CR711 MAPLHGR KW/FT	8CR183 MAPLHGR KW/FT	8CR233 MAPLHGR KW/FT
200	11.5	12.0	11.9
1000	11.4	12.2	12.0
5000	11.4	12.6	12.1
10000	11.5	12.8	12.1
15000	11.5	12.9	12.2
20000	11.0	12.6	12.1
25000	10.4	11.7	11.6
30000	9.7	10.8	11.2

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPES:
8CR711 - LOW ENRICHMENT (0.711%)
8CR183 - MEDIUM ENRICHMENT (1.76%)
8CR233 - HIGH ENRICHMENT (2.19%)

Table 4.18-2

ST 19.1 Plant Conditions

Test Condition	1	2
Core Power (MWT)	586 (17.8%)	1314 (40%)
Generator Output (MWe)	141	360
Core Flow (Mlb/hr)	42.1	45.6
Dome Pressure (psig)	921	931
Date Performed	7/4/84	7/10/84

Table 4.18-3

ST 19.1 Results
(Most Limiting Thermal Limits
From BUCLE)

MFLPD	TC
.255	1
.385	2

MFLCPR	TC
.331	1
.556	2

MAPRAT	TC
.247	1
.373	2

Table 4.18-4

ST 19.2 Plant Conditions

Test Conditions	3	4	5	6
Core Power (MWT)	2346(71.2%)	1507(45.8%)	2331(90.8%)	3248(98.6%)
Generator Output (MWe)	750	449	942	1086
Core Flow (Mlb/hr)	98.3	34.6	62.2	99.6
Dome Pressure (psig)	958	934	958	988
Date Performed	9/12/84	10/10/84	9/25/84	9/27/84

Table 4.18-5

ST 19.2 Results
(Most Limiting Thermal Limits)

MFLPD	TC
.646	3
.399	4
.647	5
.875	6

MFLCPR	TC
.609	3
.678	4
.774	5
.814	6

MAPRAT	TC
.631	3
.387	4
.619	5
.878	6

4.19 (ST21) CORE POWER-VOID MODE RESPONSE

This test demonstrated the stability of the reactor core power-void response in two ways: (1) Initiation of a rapid change in core pressure (by completing positive and negative 10 psi steps) and (2) By inserting a control rod a few notches. These Subtests were performed at Test Condition 4 and on the 100% rod line at minimum flow. Criteria for this test was that the transient response of any system related variable to any test input must not diverge. Test results confirm that system related variables did not diverge. Hence ST21 criteria was satisfied.

A control rod was selected near the most limiting CPR bundle as detemined by the process computer program, P1. The control rod selected was rod 46-43. LPRM 48-41A near the rod tip was chosen as the selected LPRM. Rod 46-43 was notched in 2 notches and LPRM 48-41A indicated a 5.2% local flux depression. Plant stability was adequately demonstrated. The test was conducted at min flow and the 100% rod line. ST21.1 was repeated in Test Condition 4 using rod 14-15 and LPRM 16-17B. The results were similar. Local flux depression was about 8.8% of the steady state value. Again, plant stability was adequately demonstrated.

The stability of the reactor core power void dynamic response to pressure transients was demonstrated on the 100% rod line at minimum flow. The chosen LPRM string near the most limiting CPR bundle was 48-41. The pressure transient consisted of ±10 psig step changes. The test was repeated at Test Condition 4 with LPRM string 16-17 chosen again as the monitored string. ST21.2 was performed in conjunction with ST22.1 during TC-4. ST22.1 consists of a 10 psi negative and positive step change in Pressure Regulator setpoint followed by simulated failure of the operating Pressure Regulator. Failure of the operating pressure regulator caused a pressure increase yielding a slight decrease in void content and subsequent increasing neutron flux and damped response of the power-void loop. The transients were initiated on the Reactor Pressure Test Card in the Lower Relay Room. Both "A" and "B" regulators were exercised.

In summary, the stability of the reactor core power-void response was adequately demonstrated at Test Condition 4 and at the 100% rod line/minimum flow operating point. Neutron flux transients were very well damped.

4.20 (ST22) PRESSURE REGULATOR

The Pressure Regulator startup tests were performed to demonstrate stable controller settings and that the settings would provide a smooth response. The "takeover" capability was demonstrated as well as the smooth pressure control transition between the turbine control valves and bypass valves when the reactor steam supply exceeded main turbine demand.

The stable response of pressure control system variables was demonstrated in this test by introducing approximately ±10 psi step changes in the pressure setpoint of the controlling pressure regulator. At each test condition, Load Limit, Load Set and Maximum Combined Flow were adjusted to demonstrate pressure control by combined Turbine Control and Bypass Valve response and by Bypass Valve response alone. A pressure regulator failure was also simulated through the use of the Test Fail Switch in the control circuitry. The test results analysis showed the margins to scram vs. reactor pressure and neutron flux.

The Acceptance Criteria were as follows:

<u>Level 1</u>

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

<u>Level 2</u>

1. 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 when operated above the lower limit of the automatic load following range.

2. When in the recirculation manual mode, the pressure response time from initiation of pressure setpoint step change to the turbine inlet pressure peak shall be less than or equal to 10 seconds.

3. Pressure control system dead band, decay, etc., shall be small enough that steady state limit cycles (if any) shall produce steam flow variations no larger than ±0.5 percent of rated steam flow.

4. The normal difference between regulator set points must be small enough that the neutron flux remains below its scram value by a margin of 7.5 percent.

5. The normal difference between regulator set points must be small enough that peak vessel pressure remains below the scram setting by a margin of 10 psi.

ST22.1, Pressure Regulator Test - Control Valves Controlling was run at TC-2, TC-3, TC-4, TC-5 and TC-6. All Acceptance Criteria were verified with acceptable margins to scram.

ST22.2, Pressure Regulator Test - Control Valve and Bypass Valves Controlling was performed at TC-2, TC-4, TC-5 and TC-6. All Acceptance Criteria were proven with acceptable margin to scram.

ST22.3, Pressure Regulator Test - Bypass Valves Controlling, was run at TC-1, TC-2, TC-4, TC-5 and TC-6. All Acceptance Criteria were proven with acceptable margin to scram.

Test Condition 1 results:

TEST	% POWER	MAX RESPONSE TIME	MAX VARIATION	MARGIN - TO - SCRAM NEUTRONFLUX	MARGIN - TO - SCRAM HIGH PRESS
22.3	13%	5.0 sec.	0%	85%	115 psi psi

Test Condition 2 results:

TEST	% POWER	MAX RESPONSE TIME	MAX VARIATION	MARGIN - TO - SCRAM NEUTRONFLUX	MARGIN - TO - SCRAM HIGH PRESS
22.1	40%	3.7 sec.	0%	72%	108 psi
22.2	40%	4.4 sec.	0%	73%	108 psi
22.3	40%	4.3 sec.	0%	73%	108 psi

Test Condition 3 results:

TEST	% POWER	MAX RESPONSE TIME	MAX VARIATION	MARGIN - TO - SCRAM NEUTRONFLUX	MARGIN - TO - SCRAM HIGH PRESS
22.1	47.2%	4.5 sec.	0%	65%	103 psi psi

Test Condition 4 results:

TEST	% POWER	MAX RESPONSE TIME	MAX VARIATION	MARGIN - TO - SCRAM NEUTRONFLUX	MARGIN - TO - SCRAM HIGH PRESS
22.1	44.4%	3.8 sec.	0%	70.2%	105 psi
22.2	44.4%	5.0 sec.	0%	70%	105 psi
22.3	44.4%	4.2 sec.	0%	68.6%	105 psi

Test Condition 5 results:

TEST	% POWER	MAX RESPONSE TIME	MAX VARIATION	MARGIN - TO - SCRAM NEUTRONFLUX	MARGIN - TO - SCRAM HIGH PRESS
22.1	70%	3.75 sec.	0%	42%	81 psi
22.2	70%	3.7 sec.	0%	*	* psi
22.3	70%	3.7 sec.	0%	*	* psi

* Pressure regulated failure testing not performed at TC-5 per approved program; therefore, this data not available.

Test Condition 6 results:

TEST	POWER	MAX RESPONSE TIME	MAX VARIATION	MARGIN - TO - SCRAM NEUTRONFLUX	MARGIN - TO - SCRAM HIGH PRESS
22.1	99.3%	5.5 sec.	0%	11.4%	52 psi
22.2	99%	3.8 sec.	0%	10.8%	52 psi
22.3	100%	4.0 sec.	0%	13.3%	52 psi

The overall operation of the Pressure Regulator Control System was excellent. All Acceptance Criteria were satisfied and there were no oscillatory responses to pressure changes at any operating power level. There were no steady state limit cycles.

4.21 (ST23) FEEDWATER SYSTEM

The objectives of this test are (a) to demonstrate acceptable response to the feedwater control system for reactor water level control, (b) to demonstrate stable reactor response to subcooling changes, i.e., loss of feedwater heating, (c) to demonstrate the capability of the automatic core flow runback feature to prevent low water level scram following the trip of one feedwater pump, and (d) to demonstrate that the maximum feedpump runout capability is compatable with licensing assumptions.

These objectives were successfully demonstrated by the performance of the following Subtests:

ST 23.1 at Test Condition (TC) 1 - With the water level being automatically controlled using the low load valve and the recirculation system in manual, ±5 inch step changes in the water level setpoint were made to demonstrate proper response and operability of the feedwater system at low reactor power.

ST 23.2 at TC 2, 3 and 6 - With one feedwater pump in manual and the others in auto, a ±5% change in the manually controlled feed pump was made. The response of the feedwater system to these steps was analyzed and compared to the applicable acceptance criteria. The recirculation system was in manual for these tests.

ST 23.3 at TC 2, 3, 4, 5 and 6 - With the recirculation system in manual, ±5 inch changes in the water level setpoint were made to demonstrate proper response and stability of the feedwater system.

ST 23.4 at approximately 80% power - A simulated turbine trip signal to the extraction steam valves was initiated which would result in the most severe restriction of extraction steam to one feedwater heater string. Recordings of the transient were analyzed and compared to the predicted response and acceptance criteria.

ST 23.5 at TC 6 - One feedwater pump is tripped to demonstrate the capability to avoid a scram and prevent a low reactor water level trip due to the loss of one feedwater pump. (Not yet performed)

ST 23.6 - A maximum feedwater runout capability test was done to demonstrate that the actual capability is compatible with licensing assumptions.

The following Acceptance Criteria were verified during the performance of these tests:

Level 1

1. The transient response of any level control system-related variable to any test input must not diverge. (1) (2) (3)

2. For the feedwater heater loss test, the maximum feedwater temperature decrease due to a single failure case must be less than or equal to 100°F. The resultant MCPR must be greater than the fuel thermal safety limit.(4)

3. For the Feedwater heater loss test, the increase in heat flux cannot exceed the predicted Level 2 value by more than 2%. The predicted value will be based on the actual test values of feedwater temperature change and power level. (4)

4. The feedwater flow runout capability must not exceed 18.15 MLB/hr at 1060 psig Reactor pressure. (6)

<u>Level 2</u>

5. 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. (2) (3)

6. The open loop dynamic flow response of each feedwater actuator (turbine or valve) to small (10%) step disturbance shall be:

Maximum time to 10% of a step disturbance	1.1 sec.
Maximum time from 10% to 90% of a step disturbance	1.9 sec.
Peak overshoot (% of step disturbance)	15% (2)

7. The average rate of response of the feedwater actuator to large (greater than or equal to 20% of pump flow) step disturbances shall be between 10 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. (2)

8. For the feedwater heater loss test, the increase in heat flux cannot exceed the predicted value referenced to the actual feedwater temperature changes and the initial power level. (4)

9. A scram must be avoided from low water level with at least a 3 inch margin following a trip of one of the operating feed water pumps. (5)

(1) Applicable to ST 23.1

(2) Applicable to ST 23.2

(3) Applicable to ST 23.3

(4) Applicable to ST 23.4

(5) Applicable to ST 23.5

(6) Applicable to ST 23.6

<u>ST 23.1 Startup Controller Level Step</u> - At TC-1, with a reactor power level of 12%, ±5 inch level setpoint changes were made with the Low Load Valve Controller controlling level in automatic. Transient signals were recorded, analyzed for divergence and found to be acceptable.

ST 23.2 Feedwater System Manual Flow Step - At TC-2, with reactor power at 39%, TC-3, with reactor power at 71% and TC-6, with reactor power at 100%, manual step changes of 25% feedwater flow were made to each feedwater pump controller with the remaining feedwater pumps in automatic.

Transient parameters were measured to determine rise time, peak overshoot and stability. For testing at TC-2 and TC-3 many of the Level 2 Acceptance Criteria were not met, although the overall system response was deemed acceptable at these power levels. At TC-6, all 3 pumps exceeded the overshoot criteria and RFP "A" also exceeded the rate of change criteria for large step changes. Based on the overall system response the present feedwater pump performance is acceptable.

ST 23.3 Feedwater System Level Setpoint Changes - At TC-2 with reactor power at 35%, TC-3 with power at 71%, TC-4 with power at 43%, TC-5 with power at 74% and TC-6 with power at 100%, 5 inch increases and decreases in level in both single and three element control were made. Transient signals were monitored and analyzed for divergence and oscillatory behavior. All Acceptance Criteria satisfied. ST 23.3 at TC-6 to be retested and results addressed in a supplement to this report.

ST 23.4 Loss of Feedwater Heating - At TC-6 with reactor power level at 82%, a turbine trip signal to the feedwater heater extraction steam valves was simulated resulting in the isolation of extraction steam to the last three heaters of one feedwater train. This resulted in a feedwater temperature decrease of approximately 34°F and a heat flux increase of approximately 6%. Test results confirm that conservative assumptions were made in the analysis of this incident in Section 15 of the FSAR.

ST 23.5 Feedwater Pump Trip - Not performed at this time. Results will be discussed in a supplement to this report.

ST 23.6 Maximum Feedwater Runout Capability - At TC-6, each feedwater pump was placed in manual one at a time and speed increased to its high speed stop and feedwater flows were recorded. The calculated value of runout flow was 15.92 Mlb/hr versus the maximum allowable value of 18.15 Mlb/hr.

Overall, the Feedwater System met the objectives of the test and satisfied the Acceptance Criteria.

4.22 (ST24) TURBINE VALVE SURVEILLANCE

Turbine Valve Surveillances will be performed after the pre-commercial outage to determine acceptable maximum power levels for periodic surveillance testing of the Main Turbine Stop, Control, Bypass and Combined Intermediate Stop Valves without causing a reactor scram. These surveillances will verify the margin to scram for reactor pressure, heat flux, and neutron flux and the margin to main steam line Isolation due to peak steam flow. The tests consist of the opening and closing of the valves individually and recording the parameter changes which were affected by this operation. The following Acceptance Criteria will be proven during these tests:

Level 1

NONE

Level 2

1. Peak neutron flux must remain at least 7.5% below the neutron flux scram trip value (118%).

2. Peak vessel pressure must remain at least 10 psi below the high pressure scram setting (1037 psig).

3. Heat Flux must remain at least 5% less than its flow biased scram value (113.5%).

4. Peak steam flow in each line must remain at least 10% below the high flow isolation trip setting (132%).

4.23 (ST25) MAIN STEAM ISOLATION VALVES

The objectives of this test were (a) to functionally check the main steam isolation valves (MSIVs) for proper operation at selected power levels, (b) to determine reactor behavior during and following simultaneous full closure of all MSIVs, (c) to determine isolation valve closure time and (d) to demonstrate the maximum power at which a single valve closure can be made without a scram.

These objectives were satisfied by the performance of Subtest 25.1-MSIV Functional Test during Heatup Testing, Test Condition TC-5 and TC-6; and Subtest 25.3 - Full Isolation at TC-6.

The acceptability of the fast criteria (3 seconds) is determined by extrapolating the full stroke time from measured stroke times between nominal 10% closed and 90% closed. The acceptability of the slow criteria (5 seconds) is determined by utilizing the full stroke from solenoid deenergization to 90% closed and extrapolating the final 10% of stroke.

The following acceptance criteria were verified during these tests:

<u>Level 1</u>

1. The positive change in vessel dome pressure occurring within 30 seconds after closure of all MSIVs must not exceed predicted values by more than 25 psi. (2)

2. The positive change in heat flux following closure of all MSIVs shall not exceed predicted values by more than 2% of rated value. (2)

3. Following the closure of all MSIV's, the reactor must scram. (2)

4. Feedwater control settings must prevent flooding the main steam lines during the full isolation test. (2)

5. The closure time for any MSIV shall not be less than 3.0 seconds nor greater than 5.0 seconds. (1)(2)

<u>Level 2</u>

6. The positive change in vessel dome pressure occurring within the first 30 seconds after the closure of all MSIVs must not exceed the predicted values. Predicted values will be referenced to actual test conditions of initial power level, scram timing and dome pressure and will use beginning of life nuclear data. (2)

7. The positive change in heat flux occurring within the first 30 seconds after the closure of all MSIVs 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. (2)

8. If water level reaches Level 2 setpoint during the MSIV full closure test, RCIC shall automatically initiate and reach rated flow. (2)

9. During the MSIV full closure test, the relief valves must reclose properly (without any detectable leakage) following the pressure transient. (2)

10. During full closure of individual MSIVs, peak vessel dome pressure must remain at least 10 psi below the scram value. (1)

11. During full closure of individual MSIVs, peak neutron flux must remain at least 7.5% below its scram value. (1)

12. During full closure of individual MSIVs, steam flow in individual lines must remain at least 10% below the high flow isolation trip setting. (1)

13. During full closure of individual MSIVs, the peak simulated heat flux must remain at least 5% less than its flow biased scram setpoint. (1)

(1) Applicable to ST 25.1

(2) Applicable to ST 25.3

Subtest 25.1

During Initial Heatup at rated pressure, TC-5 at approximately 64% power and TC-6 at approximately 89% power, each MSIV was individually closed to demonstrate proper operation and to measure its closure time, and to determine the maximum power level at which a valve may be tested. Proper operation was demonstrated and closure times were within limits. Margins to scram or isolation were calculated and used to extrapolate the highest power level at which a valve could be tested. This power level was determined to be 88%. Neutron flux, reactor pressure, heat flux and steam flow margins to scram or isolation were calculated and results are listed in Table 4.23-1.

Subtest 25.3

A full MSIV isolation was initiated from 100% power and the parameters of heat flux and reactor pressure were recorded and compared to predicted values. These results are shown in Table 4.23-1. The actual pressure rise experienced during this test was such that no safety/relief valves lifted. RCIC auto started and restored reactor water level to normal. The maximum water level experienced was 76.2".

All Acceptance Criteria were met during the test. Test results confirm that conservative assumptions were made in the analysis of this incident in Section 15 of the FSAR.

Overall, the MSIVs met the objectives of the test and satisfied the Acceptance Criteria.

SUBTEST/TC/POWER LEVEL	APRM MARGIN TO SCRAM %	PRESSURE MARGIN TO SCRAM-psi	HEAT FLUX MARGIN TO SCRAM %	STEAM FLOW MARGIN TO ISOLATION-%	AVERAGE CLOSURE TIME (SEC)	FASTEST CLOSURE TIME (SEC)
25.1/INITIAL HEATUP/1%	N/A	N/A	N/A	N/A	4.0	3.5
25.1/TC-5/64%	46	78	20	49	3.9	3.5
25.1/TC-6/89%	13	36	12.6	9.6	3.7	3.3

	PREDICTED HEAT FLUX INCREASE	ACTUAL HEAT FLUX INCREASE	PREDICTED PRESSURE INCREASE	ACTUAL PRESSURE INCREASE	AVERAGE CLOSURE TIME (SEC)	FASTEST CLOSURE TIME (SEC)	MAXIMUM WATER LEVEL
25.3/TC-6/100%	$<$1%	0%	109.3 psi	50 psi	4.0	3.4	76.2"

TABLE 4.23-1

ST 25 TEST DATA

4.24 (ST26) RELIEF VALVES

The results of the testing showed that all relief valves functioned properly and reseated properly after operation. The testing also demonstrated plant pressure control system stability during relief valve operation and showed that no blockages existed in relief valve discharge piping.

The Acceptance Criteria were as follows:

Level 1

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

Level 2

1. 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.

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

3. During the low pressure functional tests, the change in bypass valve position for each SRV opening shall be greater than or equal to a value corresponding to the average change minus 10% of one bypass valve.

4. During the rated pressure tests, the change in MWe for each SRV opening shall be greater than or equal to a value corresponding to the average change minus 0.5% of MWe.

The testing was accomplished in two distinct Subtests:

ST26-1, Relief Valve Low Pressure Test, was implemented with reactor dome pressure at 138 psig during the Heatup Test Plateau. Each relief valve was manually cycled to verify proper operation with each valve held open for approximately 10 seconds to allow pressure control system related variables to stabilize. All applicable acceptance criteria were met with the following exceptions: The change in bypass valve positions during the opening of relief valves "F" and "K" were not greater than the average change in bypass valve position minus 10% calculated from the opening of all relief valves, one at a time. The average change less 10% of bypass valve position during relief valve operation was 68%. The change in bypass valve position during relief valve "F" operation was 67% and 66% during relief valve "K" operation, thus not meeting the acceptance criteria. This exception was resolved when General Electric and Nuclear Plant Engineering concluded that the performance of relief valve "F" and "K" was adequate for the existing low pressure plant conditions and that the operation of the valves would be re-examined during the performance of ST26.2, Relief Valve Rated Pressure Test at Test Condition 2. Also, the discharge temperature of relief valve "L" stabilized at a value 101°F which was greater than 10°F difference from its initial temperature of 114°F, thus not meeting the Acceptance Criteria. Since the final temperature was 3°F cooler than the specified criteria, it was concluded by Engineering that the intent of the Acceptance Criteria was verified and further testing was unnecessary.

ST26.2, Relief Valve Rated Pressure Test, was implemented at 41 percent rated reactor thermal power with reactor dome pressure at 930 psig during Test Condition 2. Each relief valve was manually cycled to verify proper operation at rated pressure. The decrease in main generator electric output during each relief valve actuation was compared to the generator electric output average change, calculated after all relief valves had been actuated, to verify that no major blockages in valves or tailpipes existed. Pressure control system related variables were again observed for stability during relief valve actuation and the relief valve tailpipe temperatures were monitored after actuation to verify that each relief valve had properly reseated. All acceptance criteria were met during the test.

The testing overall showed that the objectives as set forth in the Final Safety Analysis Report were satisfied.

4.25 (ST27) TURBINE TRIPS AND GENERATOR LOAD REJECTION

The objective of ST27 is to demonstrate the response of the reactor and its control systems to protective trips in the turbine and generator. This was accomplished by performing a manually initiated turbine trip at Test Condition 3 (Subtest ST 27.1). During this transient, reactor water level, pressure, and simulated heat flux were recorded and compared to predicted results and Acceptance Criteria. At 20% power, a generator load rejection within bypass capacity (Subtest 27.3) was manually initiated by opening the generator output breaker to demonstrate the ability to ride through a load rejection within bypass capacity without a scram. During both transients, main turbine stop, control and bypass valve positions and reactor water level were recorded and compared to Acceptance Criteria. After the pre-commercial outage ST27.2 will be performed where a turbine trip will be initiated by opening the generator output breaker at Test Condition 6. The Acceptance Criteria verified in these tests are as follows:

<u>Level 1</u>

1. For turbine and generator trips there should be a delay of no more than 0.1 seconds following the beginning of control or stop valve closure before the beginning of bypass valve opening. (1)(2)(3)

2. For turbine and generator trips the bypass valves should be opened to a point corresponding to greater than or equal to 80 percent of full open within 0.3 seconds from the beginning of control or stop valve closure motion. (1)(2)

3. Feedwater system settings must prevent flooding of the steam line following these transients. (1)(2)(3)

4. The positive change in vessel dome pressure occuring within 30 sec. after either generator or turbine trip must not exceed the Level 2 criteria by more than 25 psi.(1)(2)

5. After either a generator or turbine trip the positive change in heat flux shall not exceed the Level 2 criteria by more than 2% of rated value.(1)(2)

6. The two pump drive flow coastdown transient during the first three seconds must be greater than or equal to 3 second but less than or equal to 4.5 second time constants for the pump and motor. If coupled to the M-G sets the coastdown must have a time constant longer than 5 seconds. (1)(2)

<u>Level 2</u>

1. There shall be no MSIV closure in the first 3 minutes of the transient. (1)(2)

2. There shall be no operator action taken to prevent a MSIV trip within the first three (3) minutes after the transient.(1)(2)

3. The positive change in vessel dome pressure occurring within the first 30 seconds after the closure of all stop/control valves must not exceed the predicted value. Predicted values will be referenced to actual test conditions of initial power level and dome pressure, scram timing, and the time for the start of stop/control valve motion to start of control rod motion, and will use beginning of life nuclear data.(1)(2)

4. The positive change in heat flux occurring within the first 30 sec. after the closure of all stop/control valves must not exceed 0%. (1)(2)

5. For the generator trip within the bypass valve capacity, (initial thermal power less than or equal to 25% of rated) the reactor shall not scram. (3)

6. The total delay from the initiation of a Turbine Stop Valve Closure or Turbine Control Valve Fast Closure to complete suppression of the elecric arc between the fully open contacts of the Recirculation Pump Trip Breaker shall be less than 175 milliseconds. (2)

7. Feedwater level control shall maintain water level above the L2 level trip setpoint for HPCI, RCIC and ATWS RPT. (1)(2)

8. Feedwater level control shall avoid the loss of feedwater flow due to a high level (L8) trip. (1)(2)

(1) Applicable to ST 27.1

(2) Applicable to ST 27.2

(3) Applicable to ST 27.3

<u>Subtest 27.1 - Turbine Trip</u>

In this subtest a turbine trip was initiated from 74% power by actuating the manual Turbine Trip pushbutton which trips closed the four Main Turbine Stop Valves. The EHC system immediately opened the bypass valves to limit the reactor vessel pressure rise. The Feedwater Control System reacted to maintain water level. See Table 4.25-1 for a summary of the results of this test.

<u>Subtest 27.2 - High Power Generator Load Rejection</u>

This subtest will be performed in TC-6 after the pre-commercial outage.

Subtest 27.3 - Generator Load Reject Within Bypass Capacity

With the reactor operating at 20% of rated power level, so that the reactor scram signals on Turbine Control Valve Fast Closure and Turbine Stop Valve Trip were bypassed, the Main Generator Breaker was opened. This resulted in a Turbine Trip and Control Valve Fast Closure without causing a reactor scram. The bypass valves opened to control reactor pressure and the feedwater system maintained water level constant although a slight oscillitory response in water level was noted. The overall response was uneventful as anticipated. The delay time from the start of control or stop valve closure to the start of bypass valve opening was 0.05 seconds which was less than the maximum allowed of 0.1 seconds.

With the exception of Level 2 criteria number 6, which has not yet been tested, all Acceptance Criteria have been verified between Test Conditions 2 and 3. ST 27.2 will be performed in TC-6 after the pre-commercial outage to complete the acceptance criteria verifications.

SUBTEST	POWER LEVEL	PREDICTED PRESSURE RISE	ACTUAL PRESSURE RISE	MAXIMUM WATER LEVEL	ACTUAL HEAT FLUX RISE	CV/SV CLOSE TO BPV OPEN DELAY	TIME TO 80% OF BPV OPEN
27.1	74%	117.6 psi	69 psi	34"	0	.055 sec	.22 sec

TABLE 4.25-1

ST 27 TEST RESULTS

4.26 (ST28) SHUTDOWN FROM OUTSIDE THE CONTROL ROOM

The results of the testing showed that the reactor could be scrammed and the main steam isolation valves closed from outside the control room and that the reactor could be successfully cooled down using control devices located outside of the control room, utilizing the minimum shift complement of control room operating personnel per the Technical Specifications.

The Acceptance Criteria were as follows:

Level 1

NONE

Level 2

1. The reactor must be capable of being scrammed and isolated from outside the control room.
2. The reactor can be maintained in hot shutdown conditions from outside the control room.
3. 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 outside the control room. This test is deemed successful when reactor pressure is less than 98 psig (permissive setpoint) and the RHR Shutdown Cooling Mode has been put into operation.
4. The reactor can be safely cooled down from outside the control room.

The demonstration of shutting down from outside the control room was accomplished in two distinct Subtests.

The first Subtest, ST28.1, was performed July 5, 1984, in Test Condition 1, at 19% rated reactor thermal power. Using the minimum shift complement of control room operating personnel per the Technical Specifications, the reactor was scrammed and the MSIVs closed from the control room. This crew then evacuated the control room and assumed their various station assignments. Using the remote shutdown panel control devices, reactor pressure, temperature and level were first stabilized and then a slow cooldown was begun. The final part of this Subtest involved the operation of the Shutdown Cooling Mode of the Residual Heat Removal System from the remote shutdown panel which successfully demonstrated that the reactor could be safely cooled down from outside the control room.

The second Subtest, ST28.2, was performed solely to demonstrate that operations personnel could initiate a scram and Main Steam Isolation Valve closure from outside the control room. This demonstration took place at 0% reactor thermal power during Test Condition 1, July 5, 1984. Breakers on the Reactor Protection System power distribution panels, located outside of the control room, were opened which caused the reactor to scram and the Main Steam Isolation Valves to close, thus successfully demonstrating that a reactor scram and isolation could be initiated from outside the control room.

The testing completed per both Subtests successfully demonstrated that all test objectives as set forth in the Final Safety Analysis Report could be met.

4.27 (ST29) RECIRCULATION FLOW CONTROL SYSTEM

The results of the testing demonstrated the flow control capability of the plant over the entire reactor recirculation pump speed range in individual pump local manual mode of control and the combined pump master manual mode of control. The testing also determined that the electrical compensator and controller settings were set for desired system performance and stability.

The Acceptance Criteria were as follows:

Level 1

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

Level 2

2. A scram shall not occur due to recirculation flow control manuevers.

3. The APRM neutron flux trip avoidance margin shall be greater than or equal to 7.5% when the power maneuver effects are extrapolated to those that would occur on the 100% rod line.

4. The decay ratio of any oscillatory controlled variable must be less than or equal to 0.25.

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

6. The heat flux trip avoidance margin shall be greater than or equal to 5% when the power manuever effects are extrapolated to those that would occur along the 100% rod line.

ST29.1 and 29.3 - Response to Step Inputs

The testing was performed during plant power ascension from low to high power levels, beginning in Test Condition 2 (43% rated reactor thermal power) and ending in Test Condition 6 at 94% rated reactor thermal power and 90% rated core flow. During all implementations (Test Conditions 2, 3, 5 and 6), all acceptance criteria were met and the recirculation flow control system was shown to be stable and responsive. Table 4.27-1 tabulates the extropolated margin-to-scram values determined during each implementation of this test.

Two different methods were used for checking the system response to step changes. ST29.1 introduced step changes with flow control in Local Manual and ST29.3 introduced the step changes with flow control in combined Master Manual. This was necessary due to the lower range limit of Master Manual control. Table 4.27-1 tabulates the flows at which each test was performed.

The testing demonstrated that the objectives as set forth in the Final Safety Analysis Report were satisfied. Acceptance Criteria 1, 2, 3, 4, 5 and 6 were verified in these subtests.

29.4 - Verification of Recirculation MG Set High Speed Stops

This test demonstrated that the Recirc MG set electrical and mechanical high speed stops are set at less than or equal to 102.5% and 105% respectively of rated core flow. The actual results were 101.6% and 102.2% for the "A" MG set, and 101.8% and 103.6% for the "B" MG set. No Acceptance Criteria were verified in this subtest.

LOCAL MANUAL MODE						
TEST CONDITION	REACTOR THERMAL POWER (% RATED)	CORE FLOW (% RATED)	APRM NEUTRON FLUX MARGIN-TO-SCRAM AVOIDANCE (%)		HEAT FLUX MARGIN-TO-SCRAM AVOIDANCE (%)	
			ACCEPTANCE CRITERIA MINIMUM	TEST RESULTS	ACCEPTANCE CRITERIA MINIMUM	TEST RESULTS
2	43	50	7.5	52.3	5	23.2
3	60	85	7.5	18	5	10.7
5	70	63	7.5	37.9	5	16.1
6	94	90	7.5	21.3	5	14.7
MASTER MANUAL MODE						
3	51	95	7.5	19.5	5	19.9
5	61	90	7.5	10.6	5	5.9
6	94	90	7.5	21	5	7

TABLE 4.27-1

ST 29 TEST RESULTS

4.28 (ST30) RECIRCULATION SYSTEM

The objectives of this test are to:

a. Obtain recirculation system performance data during pump trip, flow coastdown, and pump restart.

b. Verify that the feedwater control system can satisfactorily control water level without a resulting turbine trip and associated scram.

c. Record and verify acceptable performance of the recirculation two pump circuit trip system.

d. Verify the adequacy of the recirculation runback to mitigate a scram.

e. Verify that no recirculation system cavitation will occur in the operable region of the power-flow map.

These objectives were satisfied by the successful performance of Subtest 30.1 - Recirculation System One Pump Trip at TC-3; Subtest 30.2 - Recirculation Pump Trip (RPT) of Two Pumps at TC-3; Subtest 30.3 Recirculation Pump Runback at TC-3; and Subtest 30.4 Recirculation System Limiter Verification at TC-3. Subtest 30.1 - Recirculation System One Pump Trip will be reperformed at TC-6 after the pre-commercial outage.

The Acceptance Criteria verified during this test are as follows:

<u>Level 1</u>

1. The response of any level related variables during a single pump trip must not diverge.(1)

2. The two pump drive flow coastdown transient, during the first 3 seconds of an RFT trip, must fall within the specified bounds.(2)

<u>Level 2</u>

3. The reactor shall not scram during the one pump trip.(1)

4. The APRM margin to avoid a scram shall be at least 7.5% during the one pump trip recovery.(1)

5. The reactor water level margin to avoid a high level trip shall be at least 3.0 inches during the one pump trip.(1)

6. During a single recirculation pump trip and recovery, the simulated heat flux must remain at least 5% less than its flow biased scram value (clamped at 113.5%). (1)

7. Runback logic shall have settings adequate to prevent recirculation pump operation in areas of potential cavitation.(4)

8. The recirculation pumps shall runback upon a trip of the runback circuit.(3)

(1) Applicable to ST 30.1 only

(2) Applicable to ST 30.2 only

(3) Applicable to ST 30.3 only

(4) Applicable to ST 30.4 only

Subtest 30.1 Recirculation System One Pump Trip

A 72% power, 96% core flow, a recirculation MG Set drive motor breaker was tripped from the control room. During this trip, reactor parameters were recorded during the ensuing transient and were analyzed to verify non-divergence of oscillatory responses, adequate margins to RPS setpoints and capability of the feedwater system to prevent a high water level trip. The capability to restart the recirculation pump at a high power level was also demonstrated. The margins to scram measured during the pump trip and pump restart are presented in Table 4.28-1. Acceptance Criteria 1, 3, 4, 5 and 6 were verified in this subtest. This subtest will be reperformed at TC-6 after the pre-commercial outage.

Subtest 30.2 Recirculation Pump Trip (RPT) of Two Pumps

At 72% power and 99% core flow, the RPT breakers were simultaneously tripped using a temporary test switch. Parameters were monitored during the transient and analyzed to demonstrate acceptable pump coastdown performance. The pump coastdown time met the coastdown criteria. Acceptance Criterion 2 was verified in this subtest.

Subtest 30.3 Recirculation Pump Runback

At 71% power, 98% core flow, a feedwater pump was tripped and reactor water level allowed to drop below level 4, causing a runback of both recirculation pumps to the No. 2 Limiter setting of 45% of rated speed. The runback occurred, producing a smooth transient for all parameters measured. Acceptance Criterion 8 was verified in this subtest.

Subtest 30.4 Recirculation System Limiter Verification

This test demonstrates that the Feedwater Flow interlocks with the Recirculation Pump No. 1 Limiter are set such that cavitation will not occur in the Recirculation Pumps or Jet Pumps. The absence of pump cavitation is verified by observation of normally installed instrumentation to monitor the differential pressure across each recirculation pump, loop flow elbow tap and double tap jet pumps.

With reactor power at 51% and core flow at 95% of rated, the No. 1 Limiter was bypassed so the actual runback would not take place and control rods were inserted until the No. 1 Limiter actuated. This occurred at 20% of Total Feedwater Flow for each limiter. Cavitation was not observed. Acceptance Criterion 7 was verified in this subtest.

Overall, all objectives of the test were met and all Acceptance Criteria were satisfied.

TABLE 4.28-1

ST 30.1 TEST RESULTS

	PUMP TRIP	PUMP RESTART	
SUBTEST/TC	MARGIN TO HIGH WATER LEVEL TRIP	APRM MARGIN TO SCRAM	MARGIN TO FLOW BIAS SCRAM
30.1/3	13 in.	66.5%	17.5%

4.29 (ST31) LOSS OF OFFSITE POWER

In both performances of this subtest the plant was on line at approximately 30% power, reactor recirc pumps were set near minimum speed and the Feedwater Control System was in AUTO. Electrical power to Unit 2 from Unit 1 was prevented by racking out breaker 0A10502 (Unit 1 - Unit 2 Tie Breaker) and the incoming supply breakers 2A20101, 2A20201, 2A20301 and 2A20401. Load centers feeding Reactor and Turbine Building heating loads, S&A Building, Radwaste Building, SCC Building and Control Structure loads, and the River Intake Structure 4.16 KV buses were transferred to receive power from Unit 1 buses only. The turbine-generator was manually tripped and isolated from the Control Room, simultaneously with a manual trip of breaker 0A10401 (SV XFMR 20 Bus 20 Breaker) to cause a total loss of offsite power to Unit 2. Transient monitoring equipment was set up to record the dynamic response of selected plant variables.

The following Acceptance Criteria were verified during the performance of these tests:

<u>Level 1</u>

1. All safety systems such as the Reactor Protection System, the Diesel Generators, RCIC and HPCI must function properly without manual assistance.

2. HPCI and/or RCIC system action, if necessary, shall keep reactor water level above the initiation level of Core Spray, LPCI and Automatic Depressurization Systems (RPV Low Level 1).

<u>Level 2</u>

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

4. Permanent instrumentation for reactor power, reactor pressure, water level, control rod position, suppression pool temperature, high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) shall be demonstrated operable following re-energization of the 4KV buses by the diesel generators.

During the first performance of ST31.1 on 7/26/84, an incorrect line-up of the DC control power to the ESS buses prevented the diesel generators from automatically starting and energizing the 4KV ESS buses 2A201, 2A202, 2A203 and 2A204 as required. The test was reperformed on 8/7/84 and plant response during the transient was as expected. There were no test exceptions or any unordinary operator actions required during the 30 minute period following initiation. Acceptance Criteria 1, 2, 3 and 4 were verified during this performance of the test.

4.30 (ST32) CONTAINMENT ATMOSPHERE AND MAIN STEAM TUNNEL COOLING

The objective of this test is to verify the ability of the Drywell Coolers/recirculation fans and the Reactor Building portion of the Main Steam Tunnel Coolers to maintain design conditions in the drywell and reactor building portion of the main steam tunnel pipeway during operating conditions and post scram conditions. This test also verifies that containment main steam line penetrations do not overheat adjacent concrete.

The Acceptance Criteria were as follows:

Level 1

1.. The area under the reactor vessel in the Control Rod Drive area is maintained at or below 185°F.(1)(2)(3)

Level 2

1. The general drywell area is maintained at an average temperature less than or equal to 135°F, with maximum local temperature not to exceed 150°F.(1)(2)(3)

2. The area beneath the reactor vessel in the CRD area is maintained at an average temperature less than or equal to 135°F. (1)(2)(3)

3. The inside base of the shield wall in the RPV skirt area is maintained at temperatures greater than 100°F. (1)(2)(3)

4. The area around the recirculation pump motors is maintained at an average temperature less than or equal to 128°F, with maximum local temperature not to exceed 135°F. (1)(2)(3)

5. The reactor building portion of the main steam pipeway is maintained at or below 125°F. (1)(2)(3)

6. The concrete temperature surrounding primary containment main steamline penetrations is maintained less than 200°F. (4)

7. The reactor pressure vessel support skirt flange shall be maintained at or below 150°F. (1)(2)(3)

8. The area surrounding the drywell head shall have an average temperature equal to or greater than 135°F. (2)

9. The area surrounding the drywell head shall have a maximum local temperature not to exceed 150°F. (1)(2)(3)

10. The area beneath the reactor vessel in the CRD area shall have a minimum local temperature above 100°F. (1)(2)

    (1) Applicable to ST32.1

    (2) Applicable to ST32.2

(3) Applicable to ST32.3

(4) Applicable to ST32.4

The testing provided a means to prove design temperature standards inside primary containment and the reactor building portion of the main steam tunnel. The process computer (utilizing permanent plant temperature sensors), temporary temperature elements used during the Integrated Leak Rate Testing, and other special instrumentation provided temperature data from the different areas. The data was collected during initial reactor heatup, while in steady state operating conditions at Test Conditions 2, 3, 5 and 6 and following the reactor scram in Test Conditions 2, 3 and 6.

The individual Subtest results were as follows:

<u>ST-32.1 Containment Temperature At End of Heatup</u>

Temperature data was collected during the initial reactor heatup from a reactor pressure of approximately 800 psig to approximately 920 psig, continuing to record data until containment temperatures stabilized. All acceptance criteria were met with the following exceptions:

(1) The average temperature in the reactor recirculation pump "A&B" motor areas exceeded their acceptance criterion value of 128°F. Maximum average temperature during the testing was 132°F in the "A" motor area, and 130°F in the "B" motor area. The exception was resolved by the cognizant engineering groups by deferring any action until the Test Condition 2 steady state operation implementation could be implemented to obtain additional data. The 131°F average temperature was therefore conditionally acceptable. Test results from the Test Condition 2 testing subsequently showed that the average temperature per the acceptance criterion was met.

(2) The undervessel temperature in the CRD area did not meet its minimum acceptance criterion value of 100°F. The minimum temperature in this area during the testing was 71°F. This exception was resolved by General Electric letter GB-83-034 to PP&L which stated that 70°F or greater is acceptable for the air temperature adjacent to the outside surface of the RPV support skirt for all schedule startups and operation with the air velocity no greater than 6 feet per second. This test will be reperformed after flow balancing during the pre-commercial outage.

<u>ST-32.2 @ TC-2, Containment Temperature At Steady State</u>

Temperatures were monitored inside primary containment and in the reactor building portion of the main steam tunnel with the plant in steady state operating conditions at 39% reactor thermal power. All acceptance criteria were met with the following exceptions:

(1) The temperature in the drywell head area did not meet its Acceptance Criterion minimum temperature of 135°F for the first set of data. The average temperature was initially 132°F but then rose to greater than 135°F and remained there for the duration of the test.

(2) The temperature inside the base of the shield wall in the RPV skirt area did not meet its acceptance criterion minimum temperature of 100°F. The minimum measured temperature during the test in this area was 74.5°F.

(3) The temperature below the reactor vessel in the CRD area did not meet its acceptance criterion minimum temperature of 100°F. The minimum temperature during the test in this area was 71°F.

(4) The test was terminated before the support skirt flange temperature stabilized. Although the temperature remained below the maximum allowed by the acceptance criterion, verification of an equilibrium temperature was not obtained.

These exceptions were resolved by the cognizant engineering groups based on continued operation in accordance with the technical specifications, instrument installation and data collection and reperformance of the test.

<u>ST32.2 @ TC-3 and TC-5 Containment Temperature At Steady State</u>

Temperatures were monitored inside primary containment and in the reactor building portion of the main steam tunnel with the plant in steady state operating conditions at 38% reactor thermal power and again at 72%. In both cases, all Acceptance Criteria were met with the following exception:

(1) The temperature inside the base of the shield wall in the RPV skirt area did not meet its acceptance criterion minimum temperature of 100°F. The minimum measured temperature during the test was 78°F in the first case and 82°F in the second. Based on the previously mentioned letter GB-83-034, and on a scheduled repeat of the test, the test results were accepted and analysis of test data by the congnizant engineering groups continued.

<u>ST-32.2 @ TC-6 Containment Temperature At Steady State</u>

Temperatures were monitored inside primary containment and in the reactor building portion of the main steam tunnel with the plant in steady state operating conditions at 99% reactor thermal power and again at 100%.

All acceptance criteria were met with the following exceptions:

(1) In one case the temperature of the RPV support skirt flange did not meet its maximum acceptance criterion of 150°F. The maximum measured temperature was 153°F.

(2) In one case the temperature inside the base of the shield wall in the RPV skirt area did not meet its minimum acceptance criterion of 100°F. The minimum measured temperature was 86.5°F.

(3) In one case the average temperature in the "A" recirc pump motor area did not meet its acceptance criterion of less than or equal to 128°F. The maximum measured temperature was 128.7°F.

These exceptions were resolved by the cognizant engineering groups based on a flow balance of the containment cooling system and scheduled reperformance of the test after the pre-commercial outage.

At the time of this report, a flow balance of the containment cooling system has been performed and ST32.2 is scheduled to be performed after the completion of the outage.

ST-32.3 Containment Temperature After Reactor Scram

This Subtest monitored temperatures inside containment and in the reactor building portion of the main steam tunnel preceeding and following a reactor scram. The test was performed three times with initial core thermal power levels of 31%, 74% and 100% (TC-2, 3 and 6). All Acceptance Criteria were satisfied with the following exceptions:

(1) In two cases (TC-2 and 3) the temperature inside the base of the shield wall in the RPV skirt area did not meet its minimum acceptance criterion of 100°F. The measured minimum temperatures were 81°F and 82°F respectively.

(2) In one case (TC-6), data was missed and the temperature inside the base of the shield wall in the RPV skirt area could not be verified.

These exceptions were resolved by the cognizant engineering groups based on the previously mentioned letter GB-83-034, scheduled system flow balance and reperformance of the test after the pre-commercial outage.

ST32.4 Main Steam Penetration Concrete Temperature

The surface temperature on the concrete surrounding the main steamline penetrations obtained after the reactor had been at greater than 95% power for greater than 24 hours. The acceptance criterion concrete temperature of less than 200°F was met with the temperature recorded being 122°F.

4.31 (ST33) PIPING STEADY STATE VIBRATION

The results of the testing showed that steady state vibratory response for Main Steam inside containment and Reactor Recirculation piping and all Balance-of-Plant piping scoped for steady state vibration testing in the Startup Test Program, per FSAR Table 3.9-33, was within the acceptable design limits.

Data was recorded on GETARS (transient recording system) from remotely mounted vibration sensors. Recorded data was processed, as applicable, and compared with design calculated values. The Acceptance Criteria were as follows:

<u>Level 1</u>

1. The measured amplitude (peak to peak) of each remotely monitored point on the main steam inside containment and reactor recirculation lines shall not exceed the allowable value for that point.

<u>Level 2</u>

1. The measured amplitude (peak to peak) of each remotely monitored point on the main steam inside containment and reactor recirculation lines shall not excced the expected value for that point.

2. The maximum amplitude of the piping response for each remotely monitored point on Balance-of-Plant systems, identified in FSAR Table 3.9-33, shall not induce a stress in the pipe more than 50% of the endurance limit of the material.

For Balance-of-Plant systems scoped for testing in the Startup Test Program, per FSAR Table 3.9-33, that were accessible during plant operation and hence need not be remotely instrumented, examination was performed by the qualified test engineers to determine steady state vibratory response acceptability. The Acceptance Criterion in this case was as follows:

<u>Level 2</u>

3. The vibratory response of Balance-of-Plant non-remotely monitored systems or portions of systems identified in FSAR Table 3.9-33 shall be judged to be within acceptable limits by a qualified test engineer.

Steady state vibration testing was performed for the piping systems listed below:

(1) Main Steam system piping inside and outside primary containment.

(2) Feedwater system piping inside and outside primary containment.

(3) Reactor Recirculation system piping.

(4) High Pressure Coolant Injection system piping (steam supply, turbine exhaust and pump discharge)

(5) Reactor Core Isolation Cooling system piping (steam supply and pump discharge piping)

Testing for (1) and (2) listed above, was performed at approximately 25, 50, 75 and 100% rated steam flow with the plant operating at steady state conditions in Test Conditions 2, 3 and 6.

Testing for (3) listed above, was performed at approximately 50, 75 and 100% core flow on the 100% rod line, with each Division of the Residual Heat Removal system operating in the shutdown cooling mode in conjunction with ST8.3, and during the performance of ST16.1 and ST30.4.

Testing for (4) listed above, was performed with the High Pressure Coolant Injection system in steady state operation, discharging to the reactor vessel at its rated flow rate of 5000 (+100,-0) gpm. This occurred during Startup Test Condition 3.

Testing for (5) listed above, was performed with the Reactor Core Isolation Cooling system in steady state operation, discharging to the reactor vessel at its rated flow rate of 600 (+10,-0) gpm. This testing occurred during the Heatup Test Plateau with the reactor at rated temperature and pressure.

No piping steady state vibratory response problems were encountered during any of the testing. The only testing related problem was the apparent failure of one of the remotely mounted sensors on Main Steam inside containment. The responsible design organization determined the piping vibratory response to be acceptable based on data collected from other sensors that were mounted adjacent, or in proximity, to the failed sensor.

The performance of ST33 proved that the piping design met all test objectives as set forth in the FSAR.

4.32 (ST35) RECIRCULATION SYSTEM FLOW CALIBRATION

The objective of this test is to perform a complete calibration of the installed recirculation system flow instrumentation. This test was performed twice during TC-3 at 48% power and at 71% power and once during TC-6 at 99% power.

The following Acceptance Criteria were verified during this test:

Level 2

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

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

At 51% power and 90.9% indicated core flow, single tap jet pump, double tap jet pump and recirculation loop elbow tap data was taken and a manual calculation was performed to determine total core flow. Calculated core flow was found to be 97.6%. GEs offline computer code JRPMP calculated core flow to be 98.3% and PP&Ls Core Flow Calibration computer code calculated core flow to be 97.0%. Investigation showed that the differences between the computer calculation and the manual calculations were the assumed M-Ratio in the first iteration of the calculation and the constants used in JRPMP for the calibrated jet pump spans. The jet pump loop flow summers were adjusted to give the correct loop flows with a total core flow of 97.6%.

At 71% power and 98.7% indicated core flow, the manually calculated value of core flow was 98.4% so no further adjustments were made.

At 99% power and 99.4% indicated core flow the manually calculated value of core flow was 97.9%. The JRPMP calculated value was 97.3% and the Core Flow Calibration computer calculated value was 97.1%. The small differences between the computer calculated and manually calculated values were due to the M-Ratio used in the initial iteration. The difference in the indicated value versus the calculated value was due to the variation in M-Ratio as a function of power level. Adjustments were made to the loop flow summers to give a correct indication of total core flow and individual loop flows. 100% rated drive flow was found to be 29.4 Mlb/hr. The APRM flow units were adjusted such that 100% rated drive flow is equal to 30.4 Mlb/hr (38,810 gpm per loop) to allow a factor of 1.0 Mlb/hr for conservatism in the flow biased scram and rod block values. The average M-Ratio at 100% was calculated to be 2.4. The maximum jet pump riser plugging value was 0.039 which was less than the maximum allowable value of 0.100. The maximum jet pump nozzle plugging value was .068 which was less than the maximum allowable value of 0.120. The maximum loop flow variation was 0.010 which was less than the maximum allowable value of 0.030.

The value of total core flow as calculated from the calibrated jet pumps indication is 8% less than actual core flow. Although there are no requirements to adjust this, adjustments to the span of the calibrated jet pump flow transmitters will be made so that the calibrated jet pump flow indication can be used to determine total core flow.

4.33 (ST37) GASEOUS RADWASTE SYSTEM

The objective of this test was to demonstrate that the Gaseous Radwaste System operates within the Technical Specification and design limits during a full range of plant power operation and to demonstrate the proper operation of the containment nitrogen inerting system during plant operation. The objectives of this test were satisfied by performing Subtest ST-37.1 - Gaseous Radwaste Data Collection and Subtest ST-37.3 - Gaseous Radwaste System Performance. Subtest ST 37.2 - Containment Inerting will be performed following the pre-commercial outage.

The following Acceptance Criteria were verified during this test:

<u>Level 1</u>

1. The release of radioactive gaseous and particulate effluents must not exceed the limits specified in section 3.11.2.1, 3.11.2.2 and 3.11.2.3 of the SSES Technical Specifications.(1)

<u>Level 2</u>

1. The system flow, pressure, temperature and dew point shall not exceed design specifications.(3)

2. The catalytic recombiner, the hydrogen analyzer, activated carbon beds and the filters shall be performing their required functions.(3)

3. There shall be no less than 8,000 lbs/hr of dilution steam flow when the steam jet air ejectors are pumping.(3)

4. The containment nitrogen inerting system shall be capable of inerting the primary containment free volume within 24 hours from the start of the test and the resulting oxygen concentration shall be less than 4%.(2)

   (1) Applicable to ST-37.1

   (2) Applicable to ST-37.2

   (3) Applicable to ST-37.3

<u>ST37.1 Gaseous Radwaste Data Collection</u>

This test consisted of taking gaseous grab samples to monitor the release of radioactive gaseous and particulate effluents at various power levels during the Startup Test Program.

ST37.1 was performed during Initial Heatup at 2.5% power, Test Condition (TC)1 at 13% power, TC-3 at 57% power, TC-5 at 74% power and TC-6 at 100% power. All level 1 Acceptance Criteria were met.

ST37.2, Containment Inerting

This test has not yet been completed and will be performed during the startup after the pre-commercial outage.

ST37.3, Gaseous Radwast System Performance

ST37.3 was performed during heatup. Offgas system operational parameters were recorded and compared to design values to verify proper system operation. All parameters were within design limits except for guard bed flows and guard bed dewpoints which were higher than their design limits. Both of these problems were encountered during the Unit 1 startup and are still being evaluated. A plant modification is planned for the Unit 1 refueling outage which is intended to reduce the moisture being carried over from the chiller to the mist eliminator and to the charcoal beds. If this modification solves the high dewpoint problems, it will be incorporated into Unit 2. The high guard bed flow has been attributed to condenser in-leakage. A helium leakage test program is underway to identify and correct sources of in-leakage on both units.

4.34 (ST39) PIPING VIBRATORY RESPONSE DURING DYNAMIC TRANSIENTS

The results of the testing showed that the dynamic vibratory response during selected controlled system transients for the Main Steam inside containment and Reactor Recirculation piping and all Balance-of-Plant piping scoped for dynamic transient testing in the Startup Test Program, per FSAR Table 3.9-33, was within the acceptable design limits.

Data was recorded on GETARS (transient recording system) from remotely mounted sensors prior to, during and following each transient. Recorded data was processed, as applicable, and compared with design calculated values. The Acceptance Criteria were as follows:

<u>Level 1</u>

1. The measured vibration amplitude (peak to peak) for each remotely monitored point of main steam inside containment and reactor recirculation piping shall not exceed the allowable value for each specific point.

<u>Level 2</u>

1. The measured vibration amplitude (peak to peak) for each remotely monitored point of main steam inside containment and reactor recirculation piping shall not exceed the expected value for each specific point.

For Balance-of-Plant systems scoped for testing in the Startup Test Program, per FSAR Table 3.9-33, examination was performed by the qualified test engineers to determine dynamic vibratory response acceptability. The Acceptance Criteria in this case were as follows:

<u>Level 2</u>

1. The vibratory response of systems identified in FSAR Table 3.9-33 shall be judged acceptable by a qualified test engineer.

2. No signs of excessive piping response (such as damaged insulation, markings on piping, structural or hanger steel, or walls, damaged pipe supports, etc.) shall be present during a post transient walkdown of systems listed in FSAR Table 3.9-33.

<u>ST39.1</u> verified the proper response of main steam piping inside and outside the drywell during the following planned transients:

(1) ST27.3, Generator Load Reject Within Bypass Capacity @ 20% reactor thermal power (Test Condition 2).

(2) ST31.1, Loss of Turbine-Generator and Offsite Power @ 30.5% reactor thermal power (Test Condition 2).

(3) ST27.1, Turbine Trip @ 74% reactor thermal power (Test Condition 3).

(4) ST25.3, MSIV Full Isolation @ 100% Reactor Thermal Power (Test Condition 6).

This test will be repeated during ST27.2, High Power Generation Load Rejection @ 100% Reactor Power. This will be discussed in a supplement to this report.

ST39.2 inspected the main steam relief valve discharge piping after the performance of ST26.2, Relief Valve Rated Pressure Testing @ 40% reactor thermal power (Test Condition 2).

ST39.3 verified the proper response of reactor recirculation piping during the following planned transient tests:

(1) ST30.1, Recirculation System One Pump Trip and subsequent restart @ 72% reactor thermal power (Test Condition 3).

(2) ST30.2, Recirculation Pump Trip (RPT) of Two Pumps and subsequent restarts @ 71% reactor thermal power (Test Condition 3).

(3) Manual Trip of Recirculation Pumps to enter Test Condition 4 (Natural Circulation Testing) @ 50% reactor thermal power and subsequent pump restarts.

This test will be repeated during ST30.1, Recirculation System One Pump Trip and subsequent restart during Test Condition 6. This testing will be discussed in a supplement to this report.

ST39.4 verified the proper response of High Pressure Coolant Injection steam supply piping during a planned HPCI turbine trip from its rated flow of 5000 gpm during ST15.1, HPCI CST Injection. This occurred at approximately 3.5% reactor thermal power with the reactor pressure vessel at rated pressure during the Heatup Test Phase.

ST39.5 verified the proper response of feedwater system discharge piping. This testing was accomplished by manually tripping each reactor feedwater pump, operating at its normal pump flow rate, one at a time. This testing occurred at approximately 70% reactor thermal power (Test Condition 3).

No piping dynamic transient vibratory response problems were encountered during any of the testing. The only testing related problem was the apparent failure of one of the remotely mounted sensors on Main Steam inside containment. The responsible design organization determined the piping vibratory response to be acceptable based on data collected from other sensors that were mounted adjacent, or in proximity to, the failed sensor.

The performance of ST39 proved that the piping design met all test objectives as set forth in the FSAR.