Pennsylvania Power & Light Company

Susquehanna Steam Electric Station

Unit 1 Amended Startup Report

PENNSYLVANIA POWER AND LIGHT COMPANY

SUSQUEHANNA STEAM ELECTRIC STATION

UNIT NO. 1

AMENDED STARTUP REPORT

BY

ROBERT G. SHERANKO

APPROVED: H Keiser 5-9-85

H. W. KEISER (DATE)

SUPERINTENDENT OF PLANT

FOREWORD

This Amended Startup Report amends, consolidates and replaces the Susquehanna Steam Electric Station Unit No. 1 Startup Report and Supplementary Startup Report dated June 9, 1983, and September 6, 1983, respectively. Only those changes related to the Technical content of the Startup Report and Supplementary Startup Report are annotated by a vertical line in the margin. Minor changes, such as typographical corrections or editorial changes, are not identified.

ACKNOWLEDGEMENT

Many people from many work groups played a part in the Startup Test Program. My sincere thanks to all those people who contributed to the success of the Susquehanna 1 startup. My special thanks to Rick Wehry and Gary Merrill for providing the backbone of the Startup Test Group and to the Reactor Engineers, Shift Technical Advisors, Test Review Committee and Startup Test Group members who developed and ran the Startup Test Program:

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Wade Green
Ken Hart
Jim Klucar
Kathy Koval
Bob Lombard
Preston McDaniel
Dann McGann
Ken Mertes
Tom Oldenhage
Gordon Pedersen
Bernie Preisz
Don Roland
Marv Sager
Bob Schechterly
Cynthia Smith
Randy Sutt
Pat Taylor
Dennis Turner
Bill Weadon

Bob Sheranko

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SECTION 1

INTRODUCTION

1.1 REPORT ABSTRACT

This Amended Startup Report amends, consolidates and replaces the Startup Report and Supplemental Startup Report. This report consists of a summary of the Startup Test Program portion of the Initial Test Program performed at Unit 1 of the Susquehanna Steam Electric Station in compliance with Regulatory Guide 1.16 Revision 4, Section c.1.a, and Technical Specifications paragraphs 6.9.1.1 thru 6.9.1.3. This report covers the three major events of initial criticality, completion of the Startup Test Program, and commencement of commercial power operations, as well as the testing performed subsequent to the modification from partial arc to full arc turbine control valve steam admission.

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 Constructor 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.

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 were supplied in Bechtel Power Corporations Specifications 8856-M-392, Rev. 9, 8856-M-393, Rev. 7, and 8856-M-394, Rev. 6.

The remaining testing was specified in various sections of the Final Safety Analysis Report, Rev. 31.

1.5 MAJOR STARTUP TEST PROGRAM ADMINISTRATIVE CONTROLS

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

1. Initial Test Program Phase III - Initial Fuel Loading
2. Initial Test Program Phase IV - Initial Heatup and Low Power Testing
3. Test Condition 1
4. Test Condition 2
5. Test Condition 3
6. Test condition 5
7. 100% Rod Line Testing, which included:
   - Test Condition 4
   - Test Condition 6
   - Warranty Run

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

Test Plateaus 3 thru 7, inclusive, comprised Initial Test Program Phase V - Power Ascension Testing.

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 tests 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:

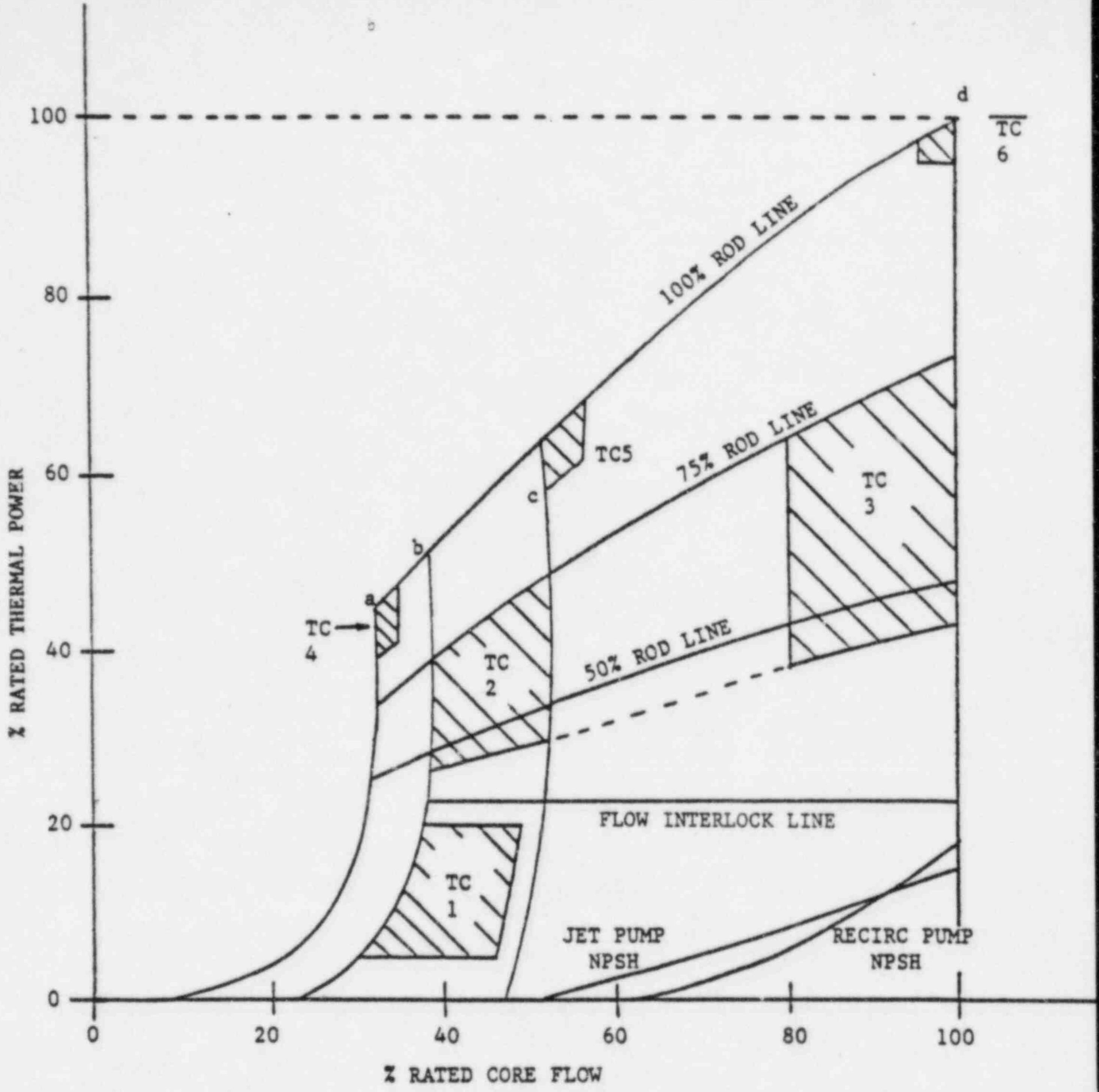
a. Elimination of any test identified as essential in Section 14 of the FSAR.

b. Modifications of test objectives, methods or Acceptance Criteria for any test identified as essential in Section 14 of the FSAR.

c. Performance of any test at a power level different from that described in the program.

d. Failure to complete any test included in the described program.

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 recirculation 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 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), Reactor Engineering Supervisor, Startup Test Group Supervisor, home office Nuclear Plant Engineering Mechanical Group Supervisor, and NSSS and AE representatives.

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

The Test Coordination Group was responsible for test implementation, pre-test preparations, on-shift test related activities coordination, test results data compilation and analysis, and evaluating and responding to all testing restraints.

The Test Director was responsible for briefing test personnel, ensuring that the test was performed in accordance with the procedure, verifying that all critical (Level I) Acceptance Criteria were satisfied and interfacing with plant operations personnel. Normally, the Shift Technical Advisor was utilized as the Test Director.

The Work Activities Review Committee (WARC) was established to review all Unit 1 facility construction, Preoperational Testing, System Testing, Hot Functional Testing and other System Demonstration activities performed concurrently with Unit I initial fuel loading or with Unit I Startup Test Program to assure that the activity would not affect the safe performance of Unit I fuel loading or the portion of the Unit I Startup Test Program currently being performed. The WARC was comprised of the Unit Coordinator (Chairman), Startup Test Group representative and an Operations representative.

General Electric personnel provided Technical Direction to operations and test personnel and operated the transient recording equipment.

Bechtel personnel provided Technical Direction for portions of the piping tests.

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

Nuclear Plant Engineering provided the formal interface between the Startup Test Group and the technical branches of Bechtel Power Corporation, provided technical resolutions to selected Test Exception Reports, and participated as a voting member of the TRC.

SECTION 2

SUMMARY

2.1 OVERALL EVALUATION

The Susquehanna Unit 1 Startup Test Program was a success. All tests were successfully completed, thus confirming the design bases. The Startup Test Program demonstrated, to the extent practical, that the plant does operate in accordance with design and does respond as designed to anticipated transients and postulated accidents. The Startup Test Program also demonstrated that conservative assumptions were made in Section 15 of the Final Safety Analysis Report in the analysis of those transients which were part of the test program.

The Startup Test Program took 254 days to complete starting with fuel loading on July 27, 1982, and ending with the completion of RHR Testing on April 7, 1983. During this time, the plant experienced only 22 scrams and a minimum amount of down time.

2.2 SUMMARY OF KEY EVENTS

July 17, 1982	Received Low Power Operating License
July 27, 1982	Commenced Fuel Loading
Aug. 8, 1982	Completed Fuel Loading
Sep. 10, 1982	Initial Criticality
Sep. 12, 1982	Completed Initial Fuel Loading Plateau Review
Sep. 20, 1982	Initially reached rated reactor pressure and temperature conditions
Oct. 4, 1982	Started Pre-Turbine Roll Outage
Oct. 31, 1982	Ended Pre-Turbine Roll Outage
Nov. 12, 1982	Completed Initial Heatup Plateau Review
Nov. 12, 1982	Received Full Power Operating License
Nov. 14, 1982	Initial Main Turbine Roll
Nov. 16, 1982	Initial Generator Synchronization
Nov. 21, 1982	Completed Plateau Review for Test Condition 1
Dec. 24, 1982	Completed Plateau Review for Test Condition 2
Jan. 16, 1983	Completed Plateau Review for Test Condition 3
Jan. 22, 1983	Completed Plateau Review for Test Condition 5
Feb. 4, 1983	Initial 100% Power Operations
Feb. 12, 1983	Started Turbine Strainer Outage
Feb. 24, 1983	Ended Turbine Strainer Outage
Mar. 8, 1983	Performed Natural Circulation Testing (Test Condition 4)
Mar. 29, 1983	Commenced Warranty Run
Apr. 4, 1983	Completed Warranty Run
Apr. 7, 1983	Completed "100% Power Testing"
Apr. 7, 1983	Started Pre-Commercial Operations Outage

May 23, 1983	Ended Pre-Commercial Operations Outage
June 1, 1983	Post Pre-Commercial Operations Outage Startup Retesting completed.
June 8, 1983	Susquehanna Unit 1 Declared Commercial

2.3 STARTUP TEST PROGRAM CHRONOLOGY

June 26, 1982	Started Pre-Fuel Load Baseline Radiation Survey
July 8, 1982	Started loading neutron sources into core
July 16, 1982	Transferred neutron sources from core to source storage rack. The reactor vessel and cavity were drained. This was done to permit re-radiographing several recirc riser welds which had Ultrasonic Test indications which were extremely difficult to evaluate.
July 17, 1982	Received Low Power Operating License from NRC
July 22, 1982	Refilled reactor vessel and cavity. Reloaded six neutron sources into core. Seventh neutron source holder was dropped when being moved from the storage rack.
July 27, 1982	Commenced Fuel loading at 12:20 p.m.
	Experienced first "RPS Trip" after second fuel bundle was loaded due to high SRM value.
July 30, 1982	Loaded seventh neutron source into core
Aug. 8, 1982	Last fuel bundle loaded at 12:59 p.m.
Aug. 20, 1982	Vessel drained to 30 inches on shutdown range
	Secondary containment established
	Progress from August 8th to 20th was hampered due to closeout of NCR's, problems encountered during LLRT's on core spray and RHR, ESW operability concerns, difficulties encountered in getting two ECCS loops operable, difficulties encountered during SGTS and CREOASS surveillances.
Aug. 23, 1982	Vessel assembly completed
Aug. 26, 1982	Commenced Operational Hydrostatic Test
Aug. 30, 1982	Completed Operational Hydrostatic Test
Sep. 8, 1982	Entered Operational Condition 2
	Progress from August 30th to September 8th was hampered due to replacement of O-rings on four CRD mechanisms, installation of TIP explosive valves, difficulties encountered with MSIV-LCS surveillance testing.

Sep. 10, 1982	Commenced reactor startup at 9:06 p.m.
	Initial criticality achieved at 11:17 p.m.
	SCRAM #1 occurred at 11:58 p.m. on IRM "H" Hi Hi Signal. Operator downranged IRM based on CRT reading which was giving a false low indication. (Shorting links were removed during this time).
Sep. 12, 1982	Completed Plateau Review for Initial Test Program Phase III-Initial Fuel Loading
Sep. 13, 1982	Heated reactor to 275° ± 30°F
	Performed inspection of piping in the drywell
Sep. 16, 1982	Increased reactor pressure to 150 psig
	Performed RCIC surveillances
	SCRAM #2. Manually scrammed reactor after two CRD accumulator low pressure alarms came up. CRD pumps had tripped on low suction pressure due to condensate reject valve cycling.
	Restarted reactor
Sep. 19, 1982	Increased reactor pressure to 600 psig
	Performed scram timing testing of selected CRD's
Sep. 20, 1982	Increased reactor pressure to 800 psig
	Performed scram timing testing of selected CRD's
	SCRAM #3 on low water level due to loss of reactor feed pump being used for level control. Pump tripped on loss of suction pressure caused by logic problem in Condensate Demineralizer System.
	Restarted reactor
	Initially reached rated reactor pressure and temperature conditions
Sep. 22, 1982	Emergency Plan activated due to fire in ESW pumphouse Motor Control Center at 9:37 a.m. Downgraded from ALERT to UNUSUAL EVENT at 10:35 a.m. Emergency Plan deactivated at 11:05 A.M.
	Manually shut down reactor to facilitate repair activities.
Sep. 24, 1982	Restarted reactor

Sep. 29, 1982	Manually shut down reactor in accordance with program to change rod sequence to enable testing of remaining half of rods
	Restarted reactor
Sep. 30, 1982	Developed condensor tube problems
Oct. 4, 1982	Manually shut down reactor and entered a pre-turbine roll outage to perform routine maintenance and design change items.
Oct. 31, 1982	Commenced reactor startup
Nov. 1, 1982	SCRAM #4 due to low water level. Technician grounded the EHC negative voltage bus resulting in a bypass valve transient with subsequent reactor vessel level transient.
Nov. 2, 1982	Restarted reactor
Nov. 2, 1982	Commenced main turbine shell warming
Nov. 10, 1982	SCRAM #5 due to low water level. Pressure transient occurred in the variable leg of the RPV level transmitters while being returned to service following surveillance testing.
Nov. 11, 1982	Restarted reactor
Nov. 12, 1982	Completed Plateau Review for Initial Test Program Phase IV- Initial Heatup
	Received Full Power Operating License
	Commenced Initial Test Program Phase V- Power Ascension Testing
	Increased power to 15% rated
Nov. 14, 1982	Completed initial main turbine roll to 100 rpm
Nov. 15, 1982	Completed initial main turbine roll to 1800 rpm
Nov. 16, 1982	Main generator initially synchronized to electrical grid and loaded to 20% rated
Nov. 20, 1982	SCRAM #6. Manually scrammed reactor from control room as part of ST 28.1, Shutdown From Outside The Main Control Room test. The plant was controlled and placed into cold shutdown from the Remote Shutdown Panel

Nov. 21, 1982	Restarted reactor
	Completed Plateau Review for Test Condition 1
Nov. 23, 1982	SCRAM #7 (unplanned SCRAM #6) occured when Technician was valving in pressure instrumentation
	Reactor restarted
Nov. 24, 1982	Performed ST 27.3, Generator Load Reject Within Bypass Capacity
	SCRAM #8, (unplanned SCRAM #7). Feedwater level control placed into Auto for first time. Resulting divergent level oscillations caused reactor feed pumps and main turbine to trip on high level.
Nov. 25, 1982	Initially reached 25% power
Nov. 26, 1982	Initially reached 30% power
Nov. 27, 1982	Initially reached 35% power
Nov. 28, 1982	SCRAM #9 (unplanned SCRAM #8) due to a control valve fast closure signal being generated during surveillances on the Combined Intermediate Stop Valves
	Reactor restarted
Dec. 1, 1982	Initially reached 40% power
Dec. 7, 1982	Commenced Process Computer testing
Dec. 8, 1982	SCRAM #10, (unplanned SCRAM #9) due to error made during surveillance testing
	Reactor restarted
Dec. 10, 1982	Initially reached 45% power
Dec. 21, 1982	Decided to postpone completion of Process Computer testing until Test Condition 3
Dec. 22, 1982	SCRAM #11. Performed ST 31.1, Loss of Turbine-Generator and Offsite Power test.
Dec. 23, 1982	Reactor restarted
Dec. 24, 1982	Completed Plateau Review for Test Condition 2
Dec. 25, 1982	Initially reached 55% power

Dec. 26, 1982	Initially reached 60% power
	Initial operation at 100% core flow
Dec. 27, 1982	Reduced power to 2% due to EHC fluid leak in the #4 turbine control valve actuator control pack
Dec. 28, 1982	Returned power to 60%
Dec. 29, 1982	Reduced power to 20% during ST 30.4, Recirculation System Limiter Verification
	Pulled rods to the 75% rod line
	Initially reached 65% power
Dec. 30, 1982	Initially reached 75% power
Jan. 6, 1983	SCRAM #12. Performed ST 27.1, Turbine Trip test from 75% power
	Main turbine shell shimmed to correct differential expansion problem
Jan. 8, 1983	Reactor restarted
Jan. 16, 1983	Completed Plateau Review for Test Condition 3
Jan. 17, 1983	Initial 100% Rod Line operation
Jan. 19, 1983	SCRAM #13 (unplanned SCRAM #10). Spurious actuation of Reactor Protection System.
Jan. 20, 1983	Restarted reactor
Jan. 22, 1983	Completed Plateau Review for Test Condition 5
Jan. 23, 1983	Initially reached 90% rated power
Jan. 25, 1983	SCRAM #14 (unplanned SCRAM #11) resulted from position switch problems encountered during surveillance testing of turbine stop valves
	Restarted reactor. Power level restricted until recalibration of recirc loop drive flow signals is complete
	Power at 75% rated. Experienced recirc pump runback due to circulating water pump trip
Feb. 2, 1983	Achieved Test Condition 6 plant conditions
Feb. 3, 1983	Power reduced to 82% rated due to high flow in the "D" condensate demineralizer

Feb. 4, 1983	Initially reached 100% power
Feb. 5, 1983	Replaced voltage regulator on "A" recirc pump motor-generator set. The pump had tripped three times during the startup
Feb. 9, 1983	Testing revealed that maximum feedpump runout capacity exceeded maximum allowable. The mechanical stops were lowered to prevent recurrence
Feb. 12, 1983	SCRAM #15. Performed ST 27.2, High Power Generator Load Rejection, from 100% power. Failure of electrical system to "fast transfer" resulted in the need to reperform this test. Refer to Section 4.27 for details
	Started Turbine Strainer Outage
Feb. 24, 1983	Ended Turbine Strainer Outage
	Restarted reactor
Feb. 26, 1983	Power restricted to 25% rated due to condenser tube leak repairs
Feb. 27, 1983	Resumed power ascension
Mar. 4, 1983	SCRAM #16. Performed ST 27.2, High Power Generator Load Rejection from 100% power.
Mar. 5, 1983	Restarted reactor
Mar. 8, 1983	Performed Natural Circulation testing (Test Condition 4)
Mar. 10, 1983	SCRAM #17 (unplanned SCRAM #12). All feedwater flow transmitters were valved out at one time causing both recirc pumps to runback resulting in a scram due to high water level.
Mar. 11, 1983	Restarted reactor. Power restricted to 50% rated due to indications of a loose part in the reactor vessel
Mar. 13, 1983	Plant placed into condition 2 and containment cooled to allow installation of additional loose parts monitoring instrumentation
Mar. 15, 1983	Resumed power ascension
Mar. 18, 1983	Achieved 100% power
Mar. 19, 1983	Reduced power to 60% rated as part of ST 34.1, Rod Sequence Exchange
Mar. 22, 1983	All preparations completed for Warranty Run including power stabilization at 100% rated

SCRAM #18 (unplanned SCRAM #13) due to high main steam line radiation levels caused by injection of air into reactor vessel via the condensate demineralizers.

Mar. 26, 1983	Restarted reactor. Delay in restart caused by: replacement of mode switch, change in location of loose parts monitoring instrumentation, full scale Emergency Plan Drill and CREOASS problems
Mar. 30, 1983	Commenced Warranty Run
Mar. 30, 1983	Halted Warranty Run. Reduced power to 70% rated to repair condenser tube leak
Apr. 1, 1983	Achieved 100% power. Resumed Warranty Run after power stabilization
Apr. 4, 1983	Completed Warranty Run
	SCRAM #19. Performed ST 25.3, (MSIV) Full Isolation, from 100% power
Apr. 7, 1983	Completed RHR testing, thus completing all "100% Power Tests"
	Started Pre-Commercial Operations Outage
May 23, 1983	Ended Pre-Commercial Operations Outage
	Restarted reactor
May 30, 1983	Completed Main Steam Isolation Valve F022C Retesting at 88.5% Power.
June 1, 1983	Achieved 100% power
	Pressure Regulator retesting completed
	Control and Stop Valve retesting completed
	RHR Steam Condensing Mode retesting completed
June 2, 1983	Testing of Common Offgas Recombiner completed
June 8, 1983	Susquehanna Unit 1 Declared Commercial

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. Because of 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 contains information necessary to successfully prepare for the implementation of Subtests and, as necessary, to provide adequate mechanisms for recording and analyzing data obtained during the implemention of Subtests. The Main Body consisted of at least the following sections:

1. Test Objectives
2. Test Description
3. Acceptance Criteria
4. References
5. Prerequisites
6. Precautions
7. Test Equipment
8. Procedure
9. 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. Initial Status
3. Test Instructions
4. Analysis
5. 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.

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 of the 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 OH123456W 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.

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.
1.2	6	No Reactor Water Cleanup This test evaluates the purification capacity of the Reactor Water Cleanup System. The system will be isolated for a period of time and then returned to operation. This test provides data on the cleanup removal constant, feedwater conductivity and impurity content and the behavior of soluble and insoluble species in the reactor water.
1.3	6	Steam Quality This test determines the quality of the nuclear steam by injection of sodium 24.
1.4	6W	Radiation Buildup on Piping This test provides baseline information on radiation buildup within certain piping and components.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
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.
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.

STARTUP TEST ABSTRACTS

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ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
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, after fuel loading but prior to initial criticality, upon initially reaching rated reactor pressure and temperature, upon initial generator synchronization at approximately 15-20% rated power, and at 50% and 100% rated power.
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 Cnambers 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.
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.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
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	0	In-Sequence Critical Refer to description of ST 4.0.
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 characteristics of the CRD system.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
5.1	0	Insert - Withdrawal Checks
	Refer to description for details	This test performs several functional tests for each control roc 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.
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	0	Zero Pressure Scram of Individual Rods
		Each CRD is withdrawn, scrammed and timed at zero reactor pressure.
5.4	H	Rated Pressure Scram Testing of individual rods
		Each fully withdrawn CRD is scrammed and timed at rated reactor pressure. This test is repeated for both Rod Withdrawal Sequences.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
5.5	OH Refer to description for details	Scram Testing of Selected Rods During this test, the four slowest CRD's from both rod withdrawal sequences 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 from both rod withdrawal sequences 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.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
5.8	3	Post Scram Differential Pressure Measurements This test functionally verifies the correct operation of the CRD Hydraulic System equalizing valves.
6.0		SRM PERFORMANCE AND CONTROL ROD SEQUENCE The objective of this test is to demonstrate that the operational neutron sources, SRM instrumentation, and rod withdrawal sequences provide adequate information to achieve criticality and increase power in a safe and efficient manner.
6.1	0	Signal to Noise Ratio and the Minimum Count Rate Determination This test verifies that Technical Specifications are met concerning signal to noise ratio and minimum count rate for each SRM.
6.2	H	SRM Response to Rod Withdrawal During rod withdrawal from all-rods-in to critical for both control rod sequences, the SRM channel responses are recorded when each control rod reaches the fully withdrawn position. After criticality, reactor coolant data is taken at points in the rod withdrawal sequence corresponding to completion of rod group withdrawal steps.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
6.3	6	Power Verses Rod Group Data This test establishes baseline data and demonstrates the continuous nature of the response of significant power related variables during power ascension. During a power increase to rated power using both rod withdrawal sequences, main generator output, steam flow, control valve position, and APRM values will be obtained at the end of each rod group movement. This test is conducted after the completion of Test Condition 3.
6.4	0	SRM Chamber Non-Saturation Demonstration This test demonstrates that the SRM Chambers do not saturate in their normal operating range.
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.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
7.2	3	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.
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-1R610. Data is collected at various drain flow rates and compared with total system flow as read by FI-1R609.
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.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
8.0		RESIDUAL HEAT REMOVAL SYSTEM The objectives of this test are to demonstrate the Residual Heat Removal (RHR) Systems ability to remove residual and decay heat from the nuclear system, to remove heat from the supression pool, and to condense steam while the reactor is isolated from the main condensor. ST's 8.1, 8.2, and 8.3, test the RHR System in the mode designated by the test's title.
8.1	2	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".
8.2	6	Steam Condensing Mode This test is performed after a major trip when the reactor is isolated from the main condenser. 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.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
8.3	6	Shutdown Cooling Mode 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	Steam Condensing Mode Stability Test This test demonstrates the stability of the controllers used in the Steam Condensing Mode.
8.5	W	Steam Condensing Mode This test is similar to ST 8.2 except that it is performed with the reactor at rated pressure and not isolated.
9.0		WATER LEVEL MEASUREMENTS The objective of this test is to verify consistent response of the narrow range and wide range level instrumentation, and to verify that the correct reference leg temperatures were used for instrument calibration.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
9.1	H123456 Refer to description for details	Water Level Data Comparison During steady state plant operation, reactor vessel level instrumentation readings are recorded and analyzed to verify that narrow and wide range level indications are tracking in a consistent manner. During the first run, actual reference leg temperature is taken and compared to the value assumed in the initial instrument calibration. During TC 3, this test is run at both 50% and 100% power.
10.0		IRM PERFORMANCE The objective of this test is to adjust the Intermediate Range Monitoring System to obtain the desired overlap with the SRM and APRM Systems. ST 10.1 verifies IRM overlap with the SRM System and ST 10.2 verifies overlap with the APRM System. If changes are made to the IRM System during 10.2, 10.1 will be repeated.
10.1	0 2	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.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
11.0		LPRM CALIBRATION The objective of this test is to verify proper installation of the LPRM detectors and to calibrate the Local Power Range Monitoring (LPRM) System.
11.1	1	Flux Response This test consists of observing and documenting the response of the individual LPRM channels to a change in local neutron flux caused by withdrawal or insertion of one of the directly adjacent control rods. This test may be performed in conjunction with ST 5.4, Rated Reactor Scram Testing of Individual Rods.
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.
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.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
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 56W	High Power APRM Calibration In this test, core thermal power is determined by a core heat balance. APRM's are then adjusted as necessary.
13.0		PROCESS COMPUTER The objective of this test is to verify that the process computer software is accurately and consistently performing its design calculations.
13.1	3	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-4, P-5, OD-1, OD-2, 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.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
13.3	3	Bundle Pow:r 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.
14.1	H	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	Reactor Vessel Injection In this test, RCIC is operated at rated reactor pressure and discharges into the reactor vessel.
14.3	1	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.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
14.4	H	Low Pressure Auto Quick Start to Vessel This test demonstrates the auto quick start capability of the RCIC System with reactor at 150 psig.
14.5	23	Surveillance Test Demonstration This test provides baseline data for RCIC surveillance tests and may be completed anytime after ST 14.3. This test is performed at both 150 psig and rated reactor pressures.
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.
15.1	H 3	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. ST 33.5 may be done concurrently with this test.
15.2	3	Reactor Vessel Injections, Rated Pressure In this test, HPCI is operated at rated reactor pressure discharging to the reactor vessel.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
15.3	3	Rated Pressure Auto Quick Starts to Vessel This test demonstrates the auto quick start capability of the HPCI System with reactor at rated pressure and the HPCI turbine and pump cold. This test must be satisfactorily completed twice in succession.
15.4	3 5	Surveillance Test Demonstration This test provides baseline information for HPCI surveillance testing and may be completed anytime after ST 15.3. This test is performed at both 150 psig and rated reactor pressures.
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 Hold Temperature Data Collection
	Refer to description for details	AT 275°F, a visual inspection of all piping inside containment and the reactor building portion of the steam tunnel is made. At 275°F and 450°F, remote instrumentation is used to verify all piping inside containment, except Feedwater, and main steam piping outside of containment.
17.3	H	Reactor at Normal Operating Temperature/Normal Operating Pressure Data Collection
		Same as ST17.2 at 450°F.
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 of Feedwater piping outside of containment 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 two to five heat-up/cooldown thermal cycles is as expected. Both visual and remotely monitored data is taken during this test.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
17.6	H	Thermal Expansion Examination of Selected RCIC, HPCI, and RWCU Systems Piping Outside Containment
	Refer to description for details	This test is a visual inspection of the piping systems listed in the title. This test is performed at 275°F, 450°F and rated reactor temperature.
17.7	6	RHR System Piping Outside Containment Data Collection
	Refer to description for details	This test is a visual inspection of RHR system piping while the system is in Shutdown Cooling or Steam Condensing modes.
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.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
19.1	12	BUCLE Calculations Refer to description in ST 19.0.
19.2	3456W	Process Computer Calculation Refer to description in ST 19.0.
20.0		STEAM PRODUCTION The objective of this test is to demonstrate that the Nuclear Steam Supply system is providing steam sufficient to satisfy all appropriate warranties as defined in the NSSS contract. ST 20.1 is performed near the beginning and end of ST 20.2.
20.1	W	2 Hour Demonstration This test consists of collecting and analyzing data collected every ten minutes while operating the reactor at the rated power level for a two hour period to demonstrate and document the satisfactory operation of the Nuclear Steam Supply system at warranted conditions.
20.2	W	100 Hour Demonstration This test consists of operating the reactor at or near the rated power level for a 100 hour period to demonstrate and document the satisfactory operation of the nuclear steam supply system for an extended period.

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.
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.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
22.1	23456	Pressure Regulator Test - Control Valves Controlling
	Refer to description for details	Refer to description in ST 22.0. During TC 3, this test was run at both 50% and 75% power.
22.2	23456	Pressure Regulator Test - Control Valves and Bypass Valves Controlling
		Refer to description in ST 22.0.
22.3	123456	Pressure Regulator Test - Bypass Valves Controlling
		Refer to description in ST 22.0.
23.0		FEEDWATER SYSTEM
		The 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.
23.1	1	Feedwater System and Startup Controller Level Step
		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	Feedwater System Manual Flow Step
		This test consists of initiating step changes in feedwater pump speed and demonstrating stable and proper response.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
23.3	23456	Feedwater System Level Setpoint Changes This test consists of initiating step changes in reactor water level and demonstrating stable and proper response.
23.4	6	Loss of Feedwater Heating 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	Feedwater Pump Trip 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	Maximum Feedwater Runout Capabilities This test demonstrates that the sum of the individual feed pump run out valves does not exceed the assumed value in the FSAR.
24.0		TURBINE VALVE SURVEILLANCE The objective of this test is to determine maximum power levels for recommended periodic surveillance testing of the Main Turbine Stop, Control, Bypass and Combined Intermediate Stop Valves without producing a reactor scram. Each subtest tests the valves indicated by its name and consists of stroking each valve until it's fully closed or open position, as appropriate, and monitoring plant response.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
24.1	3 56	Stop Valve Testing Refer to description in ST 24.0.
24.2	3 56	Control Valve Testing Refer to description in ST 24.0.
24.3	6	Bypass Valve Testing Refer to description in ST 24.0. This test is done only at maximum power level determined in ST 24.1.
24.4	6	Combined Intermediate Stop Valve Testing Refer to description in ST 24.0. This test is only done at maximum power level determined in ST 24.1.
25.0		MAIN STEAM ISOLATION VALVE The objectives of this test are to demonstrate the proper operation of the Main Steam Isolation Valves (MSIV), determine 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	H1	MSIV Functional Test During this test, each MSIV will be closed and timed.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
25.2	3 56	Full Closure of the Fastest MSIV This test uses the fastest closing MSIV as determined by ST 25.1 and determines the maximum power level at which a single MSIV can be closed without causing a reactor scram.
25.3	6	Full Isolation This test will demonstrate the reactor transient behavior that 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.

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ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
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.
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 1R101.
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 shutdown, maintained in a hot shutdown condition, and cooled down from outside the main control room using the emergency operating procedure.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
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.
28.2	3	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.
29.1	23 56	Response to Step Input This test consists of making step changes in recirculation pump speeds.
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.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
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. This test may be done in conjunction with ST 39.5.
30.4	3	Recirculation System Limitor Verification 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.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
31.0		LOSS OF TURBINE GENERATOR AND OFFSITE POWER 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	Loss of Turbine Generator and Off-Site Power Refer to description in ST 31.0.

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	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.
32.2	2 6	Containment Temperature at Steady State This test monitors temperature conditions during test conditions two and six.
32.3	6	Containment Temperature after Reactor Scram. This test monitors temperatures following the full isolation done in ST 25.3 with the drywell coolers being supplied from the Reactor Building Closed Cooling Water system.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
32.4	H	Main Steam Penetration Concrete Temperature At End of Heatup This test monitors surface temperature on the concrete surrounding the main steam line penetrations at the end of the initial reactor heatup.
33.0		STEADY STATE PIPING VIBRATION 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	Steady State Vibration for Main Steam Piping Inside Drywell This test is performed at approximately 25, 50, 75, and 100% rated steam flow.
33.2	23 6 Refer to description for details	Steady State Vibration, Main Steam Piping Outside Drywell and Feedwater Piping This test is performed in conjunction with ST33.1.
33.3	56	Steady State Vibration, Recirculation Piping This test is performed at approximately 50, 75 and 100% core flow on the 100% rod line, and during ST 8.3 when RHR is in the shutdown cooling mode.

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. This test may be done in conjunction with ST 15.2.
33.5	H	Steady State Vibration, HPCI, Suppression Pool to CST During this test, the suction for HPCI is transferred from the CST to the suppression pool. Piping steady state vibration data is collected. The HPCI turbine is then tripped for the purpose of obtaining dynamic transient data for analysis in ST 39.4. This test may be done in conjunction with ST 15.1.
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. This test may be done in conjunction with ST 14.2.
34.0		ROD SEQUENCE EXCHANGE The objective of this test is to perform a representative sequence exchange of control rod patterns at the power level presently planned for such exchanges during plant operation. It demonstrates that core limits and PCIOMR Threshold Limits will not be exceeded.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
34.1	6	Sequence Exchange Demonstration
		Refer to description in ST 34.0. The test is performed with core flow near minimum.
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	Recirculation System Flow Calibration
	Refer to Description for details	Refer to description in ST35.0. During TC 3, this test is run at both 50% and 75% power.
36.0		COOLING WATER SYSTEMS
		The objective of this test is to verify that the performance of the Reactor Building Closed Cooling Water, Turbine Building Closed Cooling Water and Service Water Systems are adequate with the reactor at rated temperature.
36.1	123 6	Cooling Water Systems Performance
		Refer to description in ST 36.0.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
37.0		GASEOUS RADWASTE SYSTEM 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	H123 56	Gaseous Radwaste Data Collection Refer to description in ST 37.0.
37.2	W	Containment Inerting Refer to description in ST 37.0. This test will be performed after the Warranty Run is completed.
38.0		BOP PIPING EXPANSION This test was incorporated into ST 17.
39.0		PIPING VIBRATORY RESPONSE DURING DYNAMIC TRANSIENTS 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.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
39.1	23 6	Main Steam Piping Vibratory Response During Turbine Stop Valve Closure This test monitors the main steam piping inside and outside the drywell during system transients caused by Turbine Stop Valve fast closures. This test is done in conjunction with ST 27.1, 27.2 and ST 31.1.
39.2	2	Main Steam and Safety Relief Valve Piping Vibratory Response During Safety Relief Valve Operation. This test is done during the safety relief valve testing performed in ST 26.2.
39.3	34 6	Recirculation Piping Vibratory Response During Pump Trips and Restarts. 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	HPCI Steam Supply Piping Vibratory Response During HPCI Turbine Trip This test analyzes the data collected during the HPCI turbine trip initiated in ST33.5.
39.5	3	Feedwater Piping Vibratory Response During Feedwater Pump Turbine Trips During this test, each Reactor Feedwater Pump will be tripped individually and the response of the feedwater piping analyzed.

ST NO.	TEST CONDITIONS	TITLE/DESCRIPTION
40.0		BOP Piping Steady State Vibration This test was incorporated into ST 33.

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 during each test condition (except TC4). These tests also verified the overall adequacy of sampling techniques, procedures and equipment.

Level 1

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 found to satisfy Acceptance Criteria.

ST1.5, Chemistry Data - Pre-Heatup, was performed at 0% power with 44.6% rated core flow. At this point a routine chemistry run was completed and all usual analyses performed. All Acceptance Criteria were met and all results were within the required specifications.

ST1.6 Chemistry Data, Heatup Tests, was also performed at this test condition with the reactor at 2% rated power. All routine analyses were run and all Acceptance Criteria were met. The initial running of feedwater created a problem due to high oxygen and total metals concentration from infrequent operation. This problem cleared as operations continued.

ST1.7, Chemistry Data - Power Ascension Tests, was run at 19% (TC1), 27% (TC2), 54% (TC3), 69% (TC5) and 97% rated power (TC6) to successfully demonstrate that samples from Reactor Water Cleanup influent, Feedwater and Control Rod Drive water were within the Acceptance Criteria.

Other tests were performed to obtain baseline data. There were no Acceptance Criteria associated with these tests.

ST1.2, No Reactor Water Cleanup, was run at 100% power to evaluate the purification capacity of the Reactor Water Cleanup system.

ST1.3, Steam Quality, was run at 100% power to determine steam quality leaving the reactor vessel.

ST1.4, Radiation Buildup On Piping was performed after two shutdowns from 100% power to provide baseline data on radiation buildup on selected piping.

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 ensuring 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 radiat'on zones shall be controlled consistant with the guidelines in 10CFR 0, 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, after fuel loading but prior to initial criticality, upon initially reaching rated pressure and temperature, and at 17%, 47% and 100% rated power. All points surveyed were within the limits specified for their location and all Acceptance Criteria were satisified.

4.3 (ST3) FUEL LOADING

The initial core of Susquehanna Unit 1 was successfully loaded with 764 fuel assemblies in 12 days (July 27, 1982 to August 8, 1982). 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% ΔK/K with the analytically determined highest worth control rod fully withdrawn. After 144 fuel assemblies were loaded, the analytically determined 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. (General Electric Co.)

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 4.5×10^4 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 original FLC cable installation was inadequate and noisy FLC readings resulted. Permanent cables were installed sorted in cable trays from the refuel floor to the SRM preamplifier to be utilized to hook up the FLCS to the SRM Electronics. The noise problem was traced to the connection box on the refueling floor. It was found that when the connection box was covered as designed, the cables were bent in such a way as to exceed the minimum bend radius and thus allowed noise to be induced on the cables. New connectors were attached and the cables laid out in such a way as to eliminate the bends. Noise curves were run which confirmed that the noise problem had cleared up. Additionally, each SRM fission chamber detector was "bugged" with a neutron source holder whose strength was about 1760 curies to check for detector response. SRMS 'A' and 'B' exhibited a classic count/discriminator curve but were very low in amplitude (1.5 CPS) where SRM detectors B & D failed to respond at all. The SRM detectors were rechecked against neutron flux after sufficient fuel bundles were loaded into the core to provide coupling between the installed neutron sources and the SRM detectors. Noise and discrimination curves were generated. SRM 'B' failed to respond and was replaced. The SRM high voltage settings had to be reset because the initial settings were >550V and it was decided that the HV setting should be lowered to some value less than 550V.

The SSES Unit 1 SbBe neutron sources arrived at SSES on 7/4/82, were placed in Source Holders, and initially installed into the Unit 1 RPV by 7/11/82.

Unfortunately, all Source Holders had to be removed from the RPV on 7/16/82 in order to drain the RPV to perform Reactor Recirculation System piping radiography. The Source Holders were in the process of reinstallation into the SSES Unit 1 RPV on 7/22/82 when the first Source Holder dropped off the Instrument Handling Tool (most likely because of not being properly grappled) and fell to the bottom of the Spent Fuel Pool. A decision was made to continue to load the remaining 6 Source Holders but during the installation of the next Source Holder into the RPV the underwater camera struck the Instrument Handling Tool and broke a one inch circular piece of plastic off the camera's attached, underwater light shield (Hydro Projects Model No. AQ-1000) which disappeared into the core area beneath the Top Guide. At this point, all Source Holder installation ceased. On 7/23/82, Source Holder installation continued and 6 Source Holders were successfully installed and the dropped Source Holder S/N 6624568 was recovered, inspected, and found damaged (bowed about 2 inches). The two source pins were removed and installed in a Unit 2 Source Holder. This Unit two Source Holder was then inserted in position 12-37 of Unit 1 core. Regarding the piece of plastic, Nuclear Plant Engineering has depositioned this to accept as is since the plastic will melt during heatup. The average initial pin strength (6-26-82) was 1018 curies/pin. The average source strength at the start of fuel loading was 703 curies/pin.

The entire core complement of fuel assemblies was prepared, inventoried, and stored in the fuel pool prior to the start of fuel loading. Fuel was loaded into the core from the center out in a roughly spiral pattern of increasing size.

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 s urces. The interpretation of the geometry affected 1/M plots allow disregard ng one or more 1/M intercepts because the obvious geometric 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
7/29/82	Failure of refueling bridge main grapple motor. Grapple motor sparked followed by release of smoke.	Brake on Grapple hoist was replaced.
7/29/82	TV camera not functioning.	Repaired-loose pin jack.
7/31/82	Refueling bridge defective power cable. Could not move bridge over the core.	Cable repaired.
8/2/82	Bridge power cable problem.	Cable repaired.

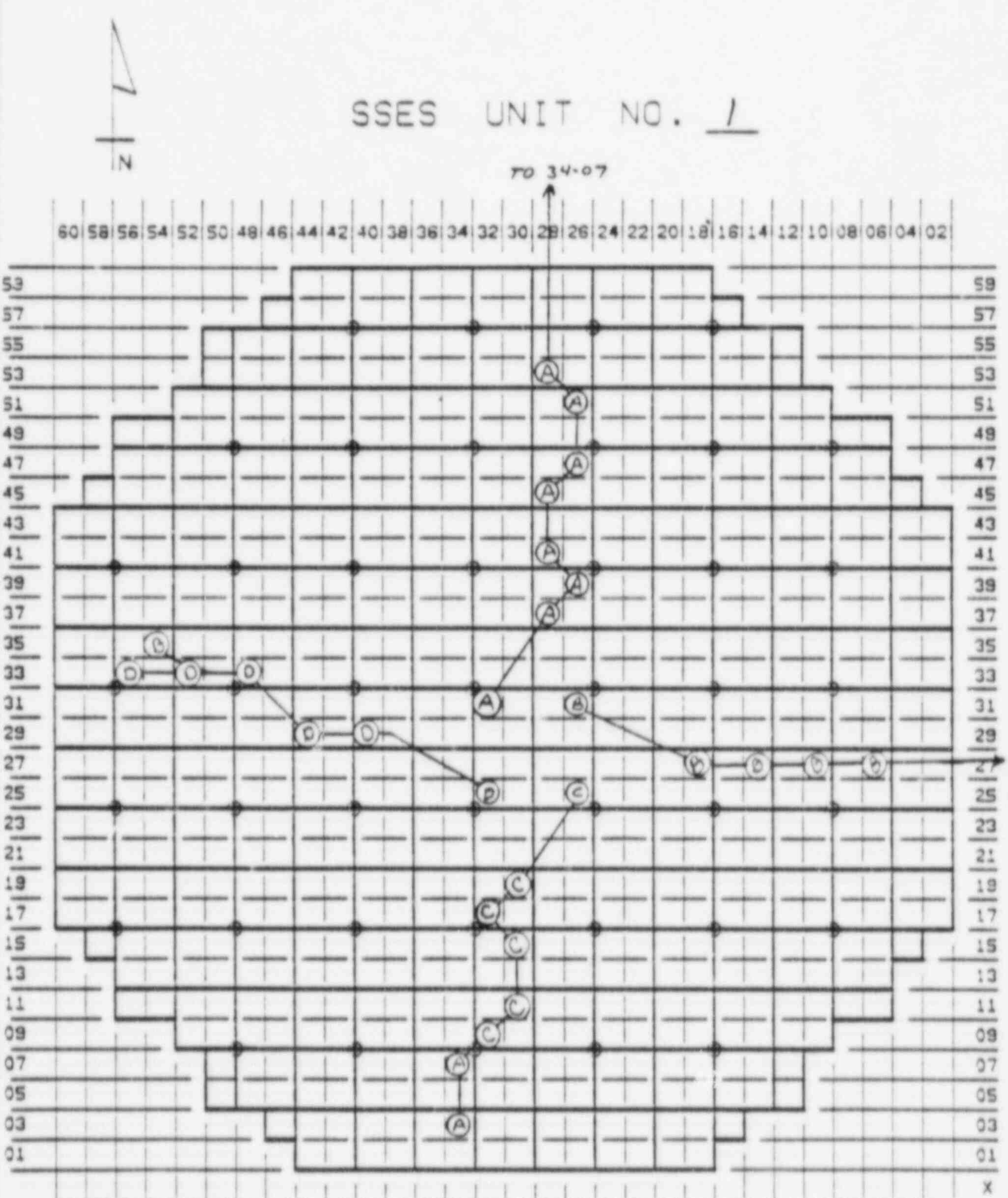
The fully loaded core was verified to be installed and configured properly on August 11, 1982. The fuel load arrangement is shown on Figure 4.3-2.

PP&L FORM 801 (8 1/2" x 11")

FIGURE 4.3-1

FLC MOVES DURING FUEL LOAD

SSES UNIT NO. 1



SSES CORE / ▲ POOL LOCATION

G 1/2 F Ø 1

4-7

A-184077-0

FRACTIONAL 0 1 2 2 1 0 DECIMAL

REGION 1
CORE MAP

SSES UNIT 1
CYCLE 1

Y \ X	1	3	5	7	9	11	13	15	17	19
20	LJM 782	LJN 422	LJN 424	LJN 427	LJN 431	LJM 808	LJN 441	LJM 805	LJN 453	LJM 800
18	LJM 781	LJN 423	LJN 425	LJN 428	LJN 432	LJN 436	LJM 807	LJN 447	LJM 803	LJN 460
16		LJM 773	LJN 426	LJN 429	LJN 433	LJN 437	LJN 442	LJM 806	LJN 454	LJM 801
14			LJM 772	LJN 430	LJN 434	LJN 438	LJN 443	LJN 448	LJM 804	LJN 461
12			LJM 771	LJM 762	LJN 435	LJN 439	LJN 444	LJN 449	LJN 455	LJM 802
10					LJM 761	LJN 440	LJN 445	LJN 450	LJN 456	LJN 462
8						LJM 728	LJN 446	LJN 451	LJN 457	LJN 463
6						LJM 725	LJM 721	LJN 452	LJN 458	LJN 464
4								LJM 720	LJN 459	LJN 465
2									LJM 719	LJM 718

FIGURE 4.3-2
SHEET 1 of 10
FUEL LOAD ARRANGEMENT

REGION 2
CORE MAP

SSES UNIT 1
CYCLE 1

Y \ X	21	23	25	27	29	31	33	35	37	39
20	LJS 981	LJS 747	LJS 988	LJS 753	LJS 756	LJS 762	LJS 768	LJS 997	LJS 774	LJT 4
18	LJS 745	LJS 985	LJS 750	LJS 991	LJS 757	LJS 763	LJS 994	LJS 771	LJT 1	LJS 777
16	LJS 962	LJS 748	LJS 989	LJS 754	LJS 758	LJS 764	LJS 769	LJS 998	LJS 775	LJT 5
14	LJS 746	LJS 986	LJS 751	LJS 992	LJS 759	LJS 765	LJS 995	LJS 772	LJT 2	LJS 778
12	LJS 983	LJS 749	LJS 990	LJS 755	LJS 760	LJS 766	LJS 770	LJS 999	LJS 776	LJT 6
10	LJS 984	LJS 987	LJS 752	LJS 993	LJS 761	LJS 767	LJS 996	LJS 773	LJT 3	LJT 7
8	LJS 971	LJS 972	LJS 973	LJS 974	LJS 975	LJS 976	LJS 977	LJS 978	LJS 979	LJS 980
6	LJS 961	LJS 962	LJS 963	LJS 964	LJS 965	LJS 966	LJS 967	LJS 968	LJS 969	LJS 970
4	LJS 951	LJS 952	LJS 953	LJS 954	LJS 955	LJS 956	LJS 957	LJS 958	LJS 959	LJS 960
2	LJS 638	LJS 639	LJS 640	LJS 641	LJS 642	LJS 643	LJS 644	LJS 645	LJS 646	LJS 647

X

FIGURE 4.3-2
SHEET 2 of 10
FUEL LOAD ARRANGEMENT

SSES UNIT 1
CYCLE 1

REGION 3
CORE MAP

Y \ X	41	43	45	47	49	51	53	55	57	59
20	LJM 918	LJS 881	LJM 921	LJS 894	LJM 924	LJS 905	LJS 910	LJS 914	LJS 917	LJS 671
18	LJS 875	LJM 919	LJS 888	LJM 923	LJS 900	LJS 906	LJS 911	LJS 915	LJS 918	LJS 672
16	LJS 779	LJS 882	LJM 922	LJS 895	LJS 901	LJS 907	LJS 912	LJS 916	LJS 673	
14	LJS 876	LJM 920	LJS 889	LJS 896	LJS 902	LJS 908	LJS 913	LJS 674		
12	LJS 780	LJS 883	LJS 890	LJS 897	LJS 903	LJS 909	LJS 676	LJS 675		
10	LJS 877	LJS 884	LJS 891	LJS 898	LJS 904	LJS 677				
8	LJS 878	LJS 885	LJS 892	LJS 899	LJS 678					
6	LJS 879	LJS 886	LJS 893	LJS 680	LJS 679					
4	LJS 880	LJS 887	LJS 681							
2	LJS 683	LJS 682								

X

Y

FIGURE 4.3-2
SHEET 3 of 10
FUEL LOAD ARRANGEMENT

SSES UNIT 1
CYCLE 1

REGION 4
CORE MAP

Y \ X	1	3	5	7	9	11	13	15	17	19
40	LJM 797	LJN 84	LJN 94	LJN 104	LJN 396	LJN 402	LJN 28	LJN 411	LJN 16	LJN 418
38	LJM 796	LJN 85	LJN 95	LJN 387	LJN 397	LJN 34	LJN 406	LJN 22	LJN 414	LJN 10
36	LJM 795	LJN 86	LJN 96	LJN 388	LJM 896	LJN 403	LJN 29	LJN 410	LJN 17	LJN 419
34	LJM 794	LJN 87	LJN 97	LJN 389	LJN 398	LJN 35	LJN 407	LJN 23	LJN 415	LJN 11
32	LJM 793	LJN 88	LJN 98	LJN 390	LJM 897	LJN 36	LJN 30	LJN 24	LJN 18	LJN 12
30	LJM 792	LJN 89	LJN 99	LJN 391	LJM 898	LJN 37	LJN 31	LJN 25	LJN 19	LJN 13
28	LJM 791	LJN 90	LJN 100	LJN 392	LJN 399	LJN 38	LJN 408	LJN 26	LJN 416	LJN 14
26	LJM 786	LJN 91	LJN 101	LJN 393	LJM 899	LJN 404	LJN 32	LJN 412	LJN 20	LJN 420
24	LJM 785	LJN 92	LJN 102	LJN 394	LJN 400	LJN 39	LJN 409	LJN 27	LJN 417	LJN 15
22	LJM 784	LJN 93	LJN 103	LJN 395	LJN 401	LJN 405	LJN 33	LJN 413	LJN 21	LJN 421

X

Y

FIGURE 4.3-2
SHEET 4 of 10
FUEL LOAD ARRANGEMENT

REGION 5
CORE MAP

SSES UNIT 1
CYCLE 1

Y \ X	21	23	25	27	29	31	33	35	37	39
40	LJM 994	LJS 923	LJS 690	LJS 931	LJM 976	LJS 711	LJS 935	LJS 727	LJS 943	LJS 739
38	LJS 919	LJS 684	LJS 927	LJS 696	LJS 702	LJS 712	LJS 721	LJS 939	LJS 733	LJS 947
36	LJM 995	LJS 924	LJS 691	LJS 932	LJS 703	LJS 713	LJS 936	LJS 728	LJS 944	LJS 740
34	LJS 920	LJS 685	LJS 928	LJS 697	LJS 704	LJS 714	LJS 722	LJS 940	LJS 734	LJS 948
32	LJM 996	LJS 686	LJS 692	LJS 698	LJS 705	LJS 715	LJS 723	LJS 729	LJS 735	LJS 741
30	LJM 997	LJS 687	LJS 693	LJS 699	LJS 706	LJS 716	LJS 724	LJS 730	LJS 736	LJS 742
28	LJS 921	LJS 688	LJS 929	LJS 700	LJS 707	LJS 717	LJS 725	LJS 941	LJS 737	LJS 949
26	LJM 998	LJS 925	LJS 694	LJS 933	LJS 708	LJS 718	LJS 937	LJS 731	LJS 945	LJS 743
24	LJS 922	LJS 689	LJS 930	LJS 701	LJS 709	LJS 719	LJS 726	LJS 942	LJS 738	LJS 950
22	LJM 999	LJS 926	LJS 695	LJS 934	LJS 710	LJS 720	LJS 938	LJS 732	LJS 946	LJS 744

X

FIGURE 4.3-2
SHEET 5 of 10
FUEL LOAD ARRANGEMENT

REGION 6
CORE MAP

SSES UNIT 1
CYCLE 1

Y \ X	41	43	45	47	49	51	53	55	57	59
40	LJS 819	LJM 931	LJS 827	LJM 944	LJS 835	LJS 839	LJS 845	LJS 855	LJS 865	LJS 657
38	LJM 925	LJS 823	LJM 937	LJS 831	LJM 949	LJS 840	LJS 846	LJS 856	LJS 866	LJS 656
36	LJS 820	LJM 932	LJS 828	LJM 943	LJS 836	LJM 955	LJS 847	LJS 857	LJS 867	LJS 655
34	LJM 926	LJS 824	LJM 938	LJS 832	LJM 950	LJS 841	LJS 848	LJS 858	LJS 868	LJS 654
32	LJM 927	LJM 933	LJM 939	LJM 945	LJM 951	LJM 956	LJS 849	LJS 859	LJS 869	LJS 653
30	LJM 928	LJM 934	LJM 940	LJM 946	LJM 952	LJM 957	LJS 850	LJS 860	LJS 870	LJS 652
28	LJM 929	LJS 825	LJM 941	LJS 833	LJM 953	LJS 842	LJS 851	LJS 861	LJS 871	LJS 651
26	LJS 821	LJM 935	LJS 829	LJM 947	LJS 837	LJM 958	LJS 852	LJS 862	LJS 872	LJS 650
24	LJM 930	LJS 826	LJM 942	LJS 834	LJM 954	LJS 843	LJS 853	LJS 863	LJS 873	LJS 649
22	LJS 822	LJM 936	LJS 830	LJM 948	LJS 838	LJS 844	LJS 854	LJS 864	LJS 874	LJS 648

FIGURE 4.3-2
SHEET 6 of 10
FUEL LOAD ARRANGEMENT

REGION 7
CORE MAP

SSES UNIT 1
CYCLE 1

Y \ X	1	3	5	7	9	11	13	15	17	19
60									LJS 626	LJS 627
58								LJS 625	LJN 46	LJN 40
56						LJS 623	LJS 624	LJN 53	LJN 47	LJN 41
54						LJS 622	LJN 59	LJN 54	LJN 48	LJN 42
52					LJS 621	LJN 65	LJN 60	LJN 55	LJN 49	LJN 43
50			LJS 619	LJS 620	LJN 70	LJN 66	LJN 61	LJN 56	LJN 50	LJN 1
48			LJS 618	LJN 75	LJN 71	LJN 67	LJN 62	LJN 57	LJN 4	LJN 44
46		LJS 617	LJN 79	LJN 76	LJN 72	LJN 68	LJN 63	LJN 6	LJN 51	LJN 2
44	LJS 616	LJN 82	LJN 80	LJN 77	LJN 73	LJN 69	LJN 8	LJN 58	LJN 5	LJN 45
42	LJM 798	LJN 83	LJN 81	LJN 78	LJN 74	LJN 9	LJN 64	LJN 7	LJN 52	LJN 3

FIGURE 4.3-2
SHEET 7 of 10
FUEL LOAD ARRANGEMENT

REGION 8
CORE MAP

SSES UNIT 1
CYCLE 1

Y \ X	21	23	25	27	29	31	33	35	37	39	
60	LJS 628	LJS 629	LJS 630	LJS 631	LJS 632	LJS 633	LJS 634	LJS 635	LJS 636	LJS 637	60
58	LJU 200	LJU 201	LJU 202	LJU 203	LJU 204	LJU 205	LJU 206	LJU 207	LJU 208	LJU 209	58
56	LJU 190	LJU 191	LJU 192	LJU 193	LJU 194	LJU 195	LJU 196	LJU 197	LJU 198	LJU 199	56
54	LJU 180	LJU 181	LJU 182	LJU 183	LJU 184	LJU 185	LJU 186	LJU 187	LJU 188	LJU 189	54
52	LJT 11	LJT 12	LJM 964	LJT 18	LJM 970	LJM 977	LJT 23	LJM 989	LJU 175	LJU 176	52
50	LJT 10	LJM 961	LJT 17	LJM 967	LJM 971	LJM 978	LJM 983	LJT 24	LJM 988	LJU 177	50
48	LJM 959	LJT 13	LJM 965	LJT 19	LJM 972	LJM 979	LJT 22	LJM 986	LJU 174	LJM 992	48
46	LJT 9	LJM 962	LJT 16	LJM 968	LJM 973	LJM 980	LJM 984	LJU 171	LJM 990	LJU 178	46
44	LJM 960	LJT 14	LJM 966	LJT 20	LJM 974	LJM 981	LJT 21	LJM 987	LJU 173	LJM 993	44
42	LJT 8	LJM 963	LJT 15	LJM 969	LJM 975	LJM 982	LJM 985	LJU 172	LJM 991	LJU 179	42
	21	23	25	27	29	31	33	35	37	39	

Y

X

FIGURE 4.3-2
SHEET 8 of 10
FUEL LOAD ARRANGEMENT

REGION 9
CORE MAP

SSES UNIT 1
CYCLE 1

Y \ X	41	43	45	47	49	51	53	55	57	59
60	LJS 658	LJS 659								
58	LJS 781	LJS 785	LJS 660							
56	LJS 782	LJS 786	LJS 790	LJS 661	LJS 662					
54	LJS 783	LJS 787	LJS 791	LJS 795	LJS 663					
52	LJS 784	LJS 788	LJS 792	LJS 796	LJS 800	LJS 664				
50	LJM 909	LJS 789	LJS 793	LJS 797	LJS 801	LJS 805	LJS 665	LJS 666		
48	LJN 468	LJM 912	LJS 794	LJS 798	LJS 802	LJS 806	LJS 810	LJS 667		
46	LJM 910	LJN 469	LJM 914	LJS 799	LJS 803	LJS 807	LJS 811	LJS 814	LJS 668	
44	LJN 467	LJM 913	LJN 470	LJM 916	LJS 804	LJS 808	LJS 812	LJS 815	LJS 817	LJS 669
42	LJM 911	LJN 466	LJM 915	LJN 471	LJM 917	LJS 809	LJS 813	LJS 816	LJS 818	LJS 670

FIGURE 4.3-2
SHEET 9 of 10
FUEL LOAD ARRANGEMENT

FIGURE 4.3-2
SHEET 10 of 10
FUEL LOAD ARRANGEMENT

ENRICHMENT BY SERIAL NUMBER

0.71 bundles (92)

(24) LJM716, 719-721, 725, 728, 761, 762, 771-773, 781, 782, 784-786, 791-798
(68) LJS616-683
(92)

1.76 bundles (240)

(39) LJN001 - LJN039
(104) LJM896 - LJM999
(97) LJS684 - LJS780
(240)

2.19 bundles (432)

(65) LJN040 - LJN104
(85) LJN387 - LJN471
(219) LJS781 - LJS999
(24) LJT001 - LJT024
(39) LJU171 - LJU209
(432)

4.4 (ST.4) 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 2.716% 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 26-51 notch position 6 for a total of 2306 notches.

The period as calculated from the equation $T = \dfrac{t}{\ln \dfrac{(P)}{(PO)}}$ was 140.8 sec. Average coolant temperature during this period was 105.9°F.

The equation used to calculate shutdown margin is:

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

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 (1.00028). Based on a period of 140.8 sec. and 105.9°F moderator temperature, rho (temp) and rho (period) is 20.85×10^{-4} DELTA k/k and 5.236×10^{-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 2.716% 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 2306 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 (ST.5) 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 10-11. A settling test was performed on this rod with satisfactory results.

At rated reactor pressure, 37 control rods failed the friction test. A settling test was performed on these rods with satisfactory results on all rods but two, 30-03 and 58-27. 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 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. The slowest control rod time was 1.86 seconds from position 48 to position 05. The scram times were small enough that Acceptance Criteria 2,3 and 4 were easily met. The slowest four control rods in each rod withdrawal sequence were selected for reliability testing in ST5.5.

ST5.4 Rated Reactor Pressure Scram Testing of Individual Rods

The test method was the same as for ST5.3 except the test was performed at rated pressure. The slowest control rod time was 3.26 seconds from position 48 to position 05. The scram times were small enough that Acceptance Criteria 2,3 and 4 were easily met. The slowest four control rods in each withdrawal sequence were selected for further testing in ST5.5, 5.6 and 5.7.

ST5.5 Scram Testing of Selected Rods

The test method was the same as for ST5.3 and 5.4. 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. The rods selected in ST5.3 were tested in the first three test conditions and the rods selected in ST5.4 were tested at the last test condition. Each control rod was scrammed three times at every test condition. Test results are compiled in Table 4.5-1. The greatest elapsed scram time to position 05 observed during these 96 individual control rod scrams was 3.42 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 ST5.4. When this test was performed at rated reactor pressure, seven control rods had a satisfactory stroke time. The eighth control rod, 10-43, required an adjustment at the HCU in order to obtain an acceptable str ke time. Acceptance Criteria 1 and 4 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 ST5.4. 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 scram times were obtained simultaneously for the four rods being tested. The greatest elapsed scram time to position 05 was 2.7 seconds. Therefore, the scram times easily met the 7 second maximum. Acceptance Criterion 4 was verified during this test. This data has been added to Table 4.5-1 for ease of comparison.

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 10 psid versus the Acceptance Criterion of 90 psid.

ELAPSED SCRAM TIME TO POSITION 05 IN SECONDS

	[4] 0 PSIG			[5] 600 PSIG			[6] 800 PSIG		
[1] "A" SEQUENCE	1	2	3	1	2	3	1	2	3
18-39	1.54	1.58	1.58	2.56	2.46	2.38	2.62	2.39	2.43
30-51	1.66	1.66	1.66	2.41	2.46	2.38	2.38	2.28	2.38
38-43	1.75	1.75	1.78	2.99	2.70	2.64	3.00	2.58	2.68
54-27	1.59	1.55	1.57	2.40	2.80	2.46	2.21	2.58	2.42

"B" SEQUENCE	1	2	3	1	2	3	1	2	3
02-43	1.55	1.75	1.60	2.67	2.91	2.69	2.88	3.02	2.96
18-27/34-51[3]	1.82	1.79	1.58	2.76	2.55	2.76	2.61	2.58	2.66
34-11	1.71	1.70	1.70	2.78	2.78	2.79	2.74	2.74	2.70
50-19/46-15[3]	1.66	1.66	1.68	2.52	2.63	2.47	2.55	2.37	2.58

	[7] RATED			[8] SPECIAL TEST		
[2] "A" SEQUENCE	1	2	3	28.1	27.1	25.3
18-15	2.27	2.42	2.40	[9]	2.3	[9]
18-39	2.51	2.38	2.38	[9]	2.4	[9]
26-23	2.40	2.31	2.32	[9]	2.4	[9]
34-47	2.65	2.68	2.93	[9]	2.7	[9]

"B" SEQUENCE	1	2	3	28.1	27.1	25.3
10-43	3.42	3.02	2.61	2.42	[9]	2.38
18-35	3.36	2.71	2.61	2.61	[9]	2.41
42-43	2.88	2.78	2.54	2.38	[9]	2.42
50-35	2.65	2.60	2.58	2.60	[9]	2.18

NOTES:

1. Four slowest rods selected by ST 15.3.
2. Four slowest rods selected by ST 15.4.
3. Rods 18-27 and 50-19 were not fully withdrawn at 600# and 800# test conditions. The next two slowest rods, 34-51 and 46-15 were substituted for use in testing at 600# and 800#.
4. Zero reactor pressure with accumulator pressure just above the low pressure alarm point.
5. 600 ± 50 psig reactor pressure with normal accumulator pressure.
6. 800 ± 50 psig reactor pressure with normal accumulator pressure.
7. Rated reactor pressure with the accumulator at 0 psig.
8. During performance of ST 28.1, 27.1 or 25.3.
9. Rods not fully withdrawn during test.

TABLE 4.5-1

CRD SCRAM TIMES

4.6 (ST6) SRM PERFORMANCE AND CONTROL ROD SEQUENCE

The results of the testing successfully demonstrated that the operational neutron sources, SRM instrumentation, and rod withdrawal sequences provided adequate information to achieve criticality and increase power safely and efficiently.

The Acceptance Criteria were as follows:

Level 1

1. There must be a neutron signal count-to noise count ratio of at least 2:1 on the required operable SRM's.
2. There must be a minimum count rate of three counts/second on the required operable SRM's.
3. The IRM's must be on scale before the SRM's exceed the rod block set point.

Level 2

None

The testing consisted of four subtests that recorded neutron monitoring system and plant performance data from the "all rods in" condition to rated reactor power. Subtests 6.1 and 6.2 covered the approach to criticality and the subsequent heatup of the reactor coolant system to rated temperature. Subtest 6.3 data was taken at the end of each Rod Worth Minimizer rod group movement. Subtest 6.4 demonstrated SRM chamber non-saturation.

The results of Subtest 6.1 testing, Determination of Source Range Monitor Signal to Noise Ratio and Minimum Count Rate, performed during Open Vessel test plateau, showed that the signal count-to-noise count ratios for SRM channels A,B,C and D were 359:1, 409:1, 539:1 and 329:1, respectively, thus meeting the required ratio of 2:1. The testing also showed that the count rates, with each SRM fully inserted, for SRM channels A,B,C and D were 36, 41, 54 and 33 counts per second, respectively, thus meeting the required fully inserted minimum count rate of 3 counts per second.

The results of Subtest 6.2 testing, SRM Response to Rod Withdrawal, performed during Initial Heatup Test Plateau showed that all IRM channels indicated on scale readings before the SRM channels exceeded the normal rod block setpoint of 1×10^5 counts per second with the SRM detectors partially retracted. IRM channel indications ranged from 20/40 on Range 5 to 32/125 on Range 8.

The results of Subtest 6.3 testing, Power Versus Rod Group, performed during Test Conditions 5 and 6, collected data for baseline establishment and demonstrated the continuous nature of the response of significant power - related parameters during power ascension. No Acceptance Criteria were verified on this Subtest.

The results of Subtest 6.4 testing, Source Range Monitor Chamber Non-Saturation Demonstration, performed during the Open Vessel Test Plateau showed that the SRM chambers did not saturate below 7.5×10^5 counts per second, indicative that SRM chamber saturation would not occur in their normal operating range. No Acceptance Criteria were verified in this Subtest.

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

4.7 (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-1R610 shall indicate within 25 gpm of RWCU flow indicator FI-1R609 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 a 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 Line Bypass Valve, HV-1F001, was opened to increase the flow from the bottom head region. The bottom head drain line temperature sensor was found to read within 5°F of the average recirculation loop suction temperatures. 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.

Some difficulty was encountered in interpreting the system's flow indicators and comparing this data to the Acceptance Criteria. The RWCU system has a high differential flow isolation circuit which monitors system inlet, system outlet and system blowdown flow indicators. The normal operating temperatures at these flow indicators is 530°F, 435°F and 120°F, respectively. Due to the density effects on water, the flow rate in gpm increases or decreases as the water expands or contracts while proceeding through the system's heat exchangers. I&C has calibrated the system flow indicators to read accurately at cold conditions because of the high differential flow circuit. The flows indicated on the GE process diagram assume that the system is at operating temperature and that the flow indicators are calibrated for operating temperatures. Flow data for the startup test was obtained by installing a local d/p cell at the system flow element. This differential pressure reading was used to obtain system flow indication at operating conditions. Having resolved this apparent system flow indication discrepancy, the RWCU testing was completed without further difficulty.

4.8 (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 heat exchanger capacity specified in process diagrams. 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 the heat exchanger capacity specified in process diagrams. 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 at the heat exchanger capacity specified in process diagrams. Both simultaneous operation of RHR loops and single 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.

6. The time to place the RHR Heat Exchangers in the steam condensing mode with RCIC using the heat exchanger condensate flow for suction shall average one half hour or less.

ST8.1 Suppression Pool Cooling Mode

The RHR heat exchanger capacity in the Suppression Pool Cooling Mode was demonstrated to be 88×10^6 Btu/hr for the A loop and 81×10^6 Btu/hr for the B loop when the suppression pool temperature was greater than 95°F. These capacities exceeded the required minimum by a factor of two. Acceptance Criterion 2 was verified during this test.

ST8.2 Steam Condensing Mode

The test was performed with the reactor isolated from the main condensor after performance of ST25.3, (MSIV) Full Isolation, from 100% power. Both simultaneous and single loop operation of RHR was tested. There were no divergent or unacceptable oscillatory responses during the test, thus Acceptance Criteria 1 and 3 were satisfied.

The capacity of the RHR heat exchanger was not verified in this Subtest. The failure to demonstrate 100% heat exchanger capacity was due to a low decay heat load resulting in a reactor pressure of less than 600 psig. Startup Test ST8.5 was written to demonstrate proper heat exchanger capacity and was successfully run after the Pre-Commercial Operations Outage.

RHR Division II was placed in service in 30 minutes. RHR Division I required 43 minutes. A review of the functional design bases for the RHR Steam Condensing Mode concluded that these times were acceptable thus satisfying Acceptance Criterion 6.

ST 8.3 Shutdown Cooling Mode

The "A" RHR heat exchanger capacity in the Shutdown Cooling Mode was demonstrated to be 373 x 10^6 Btu/hr. This capacity exceeded the required minimum by 65%.

Prior to implementing ST 8.3, it was known that the 100°F/hr cooldown rate could be exceeded during this test. Because of this, the procedure cautioned the operator to monitor the cooldown rate continuously and to stop the reactor cooldown after a temperature change of 60° was observed. These actions would prevent exceeding the reactor cooldown limit of 100°F change in one hour. A 60° temperature change was observed in the "A" recirculation loop within eight minutes of commencing the test and the "A" recirculation loop temperature cooled down about 90°F. While the operator was securing the reactor cooldown, vessel level dropped to Level 3 (13") due to the cooldown, causing a reactor scram and a shutdown cooling system isolation. Because of the severity of the transient on the plant, it was decided not to repeat the test on the "B" RHR heat exchanger. An engineering analysis concluded that the "B" heat exchanger capacity was similar to the "A" due to heat exchanger design similarities and performance similarities demonstrated in ST 8.1.

Heat Exchanger capacity was not demonstrated during simultaneous operation of both loops because of the excessive cooldown rates observed during single loop operation. The two RHR loops operate independently of each other with the exception that they share a common suction line in the shutdown cooling mode. Satisfactory heat exchanger capacity for simultaneous operation was demonstrated by showing satisfactory heat exchanger capacity for single loop operation and by showing that the flow rate for two loop operation was obtained through the suction line. Acceptance Criterion 4 was verified during this test.

ST 8.4 Steam Condensing Mode Stability Test

This test was performed at 96% rated reactor thermal power individually for each RHR heat exchanger. Maximum allowable step changes were made to heat exchanger 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.

ST 8.5 Steam Condensing Mode

With the reactor at rated pressure, the RHR Heat Exchangers accepted steam from the reactor via the High Pressure Coolant Injection steam supply line and discharged condensate initially to the suppression pool. Upon acceptable water quality from the heat exchanger discharge, the Reactor Core Isolation Cooling System was lined up to take suction from the heat exchanger, discharging condensate into the reactor. Data was collected in both single and dual heat exchanger operation and analysis performed to show that the heat exchanger capacities as specified in the process diagram could be met, thus satisfying Acceptance Criterion 3.

4.9 (ST 9) WATER LEVEL MEASUREMENTS

The results of the testing showed that throughout the Startup Test Program, the reactor narrow range level indicators agreed within ±1.5 inches of their average reading and the wide range level indicators agreed within ±6 inches of their average reading. The testing performed during initial reactor heatup also showed that the difference between the actual reference leg temperatures and the values assumed during level instrument calibration was less than the amount which would result in a scale end point error of 1% of the instrument span for each range.

Reactor water level data was recorded from upset range, narrow range and wide range indicating instrumentation at reactor rated temperature and pressure during Initial Heatup Test Condition and while at steady state operating conditions in each of Test Conditions 1 through 6. At Test Condition 3, the test was run at both 44% and 73% power. Data for each indicating instrument was compared, in each case, to the specific range's calculated average value to verify uniform calibration and operation of that channelized instrument. Results are tabulated in Table 4.9-1.

Also, during the Initial Heatup Test Condition, with the reactor vessel at rated temperature and pressure, instrument reference leg area temperatures inside and outside the drywell were recorded. Calculations were then performed using the collected temperature data and the results were compared with initial calibrations which had been performed using predicted reference leg area temperatures to verify that any difference between the predicted and actual reference leg area temperatures would not result in a level indicator scale end point error greater than or equal to 1% of the instrument span for each range.

The Acceptance Criteria were as follows:

Level 1

None

Level 2

1. The narrow range level indicators should agree with ±1.5 inches of their average reading.

2. The wide and upset range level indicators should agree within ±6 inches of their average reading.

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

One problem encountered during the testing was the Upset Range level indicator reading outside of the allowable ±6 inches from the Wide and Upset Range calculated average level reading (+8 inches at Test Condition 3 and -9 inches at Test Condition 4). Both excessive differences from the calculated average were documented as Startup Test Exception Reports. The exceptions were resolved when

it was determined that the Upset Range level indication should not have been included as part of the Wide Range Level instrumentation since its calibration is based on a different reference leg length than that of the Wide Range reference leg. With the Upset Range reading removed from the Wide Range level average calculation, there was good agreement among all Wide Range indicators and between the Wide Range and Narrow Range calculated averages. Readings for the Upset Range level indication were also determined to be acceptable based on the existing plant operating conditions and the instrumentation tolerances for that instrument.

Test Condition	Reactor Power %	All Acceptance Criteria Met ?	Narrow Range Indicator Average Level (in)	Narrow Range Greatest Difference From Average (in)	Wide & Upset Range Indicator Average Level (in)	Wide & Upset Range Greatest Difference from Average (in)	Wide Range Only Indicator Average Level (in) (1)	Wide Range Only Greatest Difference from Average (in) (1)
Heatup	1.5	YES	35.5	±1.0	36.1	+3.9	36.2	+3.8
1	21	YES	33.7	-1.4	35.2	-5.2	35.7	-3.7
2	40	YES	34.6	-1.1	36.7	-4.7	37.1	-3.1
3	44.2	NO (1)	33.9	+1.1	24	+8 (1)	23.3	-3.3
3	73	YES	35.2	-1.2	29.9	+3.1	29.9	+3.1
4	46	NO (1)	34.3	-1.3	39	-9 (1)	39.8	+5.2
5	69	YES	35.4	+1.1	37.5	±5.5	38.0	+5.0
6	99	YES	33.6	+1.4	29.5	±3.5	29.5	±3.5

Notes:

(1) The differences of +8 and -9 inches were recorded from the Upset Range Level Indicator and documented as Startup Test Exceptions. Per resolutions to the Test Exception Reports, the Upset Range Indicator should not have been compared with the Wide Range Level Indication average since its operating range is different. The differences of +8 and -9 inches were determined to be acceptable based on the plant operating conditions and instrument tolerances. The last two columns on this chart show the average and difference from average for Wide Range instrumentation, non-inclusive of Upset Range.

Table 4.9-1
Reactor Water Level Measurements

4.10 (ST 10) IRM PERFORMANCE

The results of the testing showed that adequate overlap existed between the Intermediate Range Monitor (IRM) and the Source Range Monitor (SRM) and between the IRM and Average Power Range Monitor (APRM) systems.

The Acceptance Criteria were as follows:

Level 1

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

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

Level 2

None

The testing was completed in two distinct Subtests.

ST 10.1, IRM-SRM Overlap Verification, demonstrated by direct observation and documentation the initial rise in neutron flux from the source range to the intermediate range during the Open Vessel test plateau following the initial critical operation of the reactor and again following critical operation of the reactor during subsequent startup in Test Condition 2. During both implementations, the SRM's with the SRM detectors partially withdrawn, indicated less than 1×10^5 counts per second when all IRM downscale alarms had cleared, the maximum count rate being 8×10^4 counts per second.

ST 10.2, IRM-APRM Overlap Verification, demonstrated by direct observation and documentation the initial rise in neutron flux from the intermediate range to the power range in Test Condition 1 and following the initial calibration of the APRM's by a reactor heat balance (ST 12.2) in Test Condition 2. The results of the testing at Test Condition 1 showed that all IRM's indicated less than 50/125 on Range 10 when all APRM downscale alarms were cleared, the highest reading being 44/125, proving that an overlap of at least 1/2 decade existed between the IRM's and APRM's. The results of the testing at Test Condition 2 showed that all IRM's with the exception of IRM channel "D" detector indicated less than 50/125 on Range 10 when all APRM downscale alarms were cleared. The highest reading, exclusive of IRM channel "D", was 49/125 proving that an overlap of at least 1/2 decade existed between the IRM's and APRM's.

IRM channel "D" did not respond properly to neutron flux increase during the Test Condition 2 testing. This channel was declared inoperable and was bypassed until the detector was replaced during the turbine strainer outage following Test Condition 3. On February 25, 1983, the IRM/APRM overlap on channel "D" detector was demonstrated to be at least 1/2 decade, the reading being 49/125 on Range 10.

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

4.11 (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.

Prior to attaining Test Condition 1, the LPRMs were verified operable by observing that each chamber responded to flux changes caused by the movement of an adjacent control rod. This was attempted during initial heatup (< 5% power); however, the flux level at the bottom of the core was not sufficient to produce a response on the "A" level detectors. All LPRM chambers were observed operable at a higher power level with exception of detector 40-17-B which did not respond.

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, 171 were operable, and 141 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 5 LPRMs satisfied level 2 criteria. The input amplifier current was again adjusted on these 5 LPRMs.

At Test Condition 3 another LPRM calibration was done with same procedure as Test Condition 2. LPRM detector 16-49-B in addition to 40-17-B was failed at the time of the calibration. All remaining 170 LPRMs satisfied level 2 Acceptance Criteria.

At Test Condition 6 a calibration was again performed. Only LPRM 40-17-B was failed and all 171 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. All Acceptance Criteria were satisfied.

4.12 (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-78-002 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.

ST 12.1 Low Power APRM Calibration

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 18.88 MWt which is .573 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	1.6	.147	.235	.50
B	1.65	.164	.232	.50
C	1.25	.141	.220	.40
D	1.25	.182	.205	.35
E	1.25	.176	.228	.45
F	1.4	.182	.255	.45

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-78-002 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-78-002 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-TP-002, Core Thermal Power Evaluation (Backup Method). The backup method for MFLPD consisted of performance of RE-TP-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 4.12-1 and 4.12-2. The Acceptance Criterion was met at all Test Conditions tested.

TABLE 4.12-1

TEST CONDITIONS FOR ST 12.2

	TEST CONDITIONS					
TEST CONDITION	1	2	3	5	6	Warranty
Core Flow (%)	90	45	84.2	60	97.5	100
Rx Power (%)	18.2	39.0	71.4	69	97	99.9
Rx Dome Pressure (psig)	945	940	976	969	998.7	1005
Generator Power (MWe)	160	410	795	760	1072.5	1091.2
Heat Balance Method	RE-TP-002	RE-TP-002	OD3	P-1	OD3	OD3
Date Performed	11/18/82	12/2/82	12/30/82	1/18/83	2/7/83	4/4/83

TABLE 4.12-2

ST 12.2 RESULTS (in % of Full Power, 3293 MWt)

APRM CHANNEL	TC 1 DESIRED VALUE	TC 1 FINAL SET VALUE	TC 2 DESIRED VALUE	TC 2 FINAL SET VALUE	TC 3 DESIRED VALUE	TC 3 FINAL SET VALUE	TC 5 DESIRED VALUE	TC 5 FINAL SET VALUE	TC 6 DESIRED VALUE	TC 6 FINAL SET VALUE	WARRANTY DESIRED VALUE	WARRANTY FINAL SET VALUE
A	18.2	18.5	39.0	40	71.4	71.9	69	69	97.4	97.5	99.7	100
C	18.2	19.5	39.0	40	71.4	71.5	69	69	97.4	97.5	99.7	100
E	18.2	19.5	39.0	39	71.4	71.5	69	69	97.4	98	99.7	100
B	18.2	18.5	39.0	39	71.4	71.7	69	69	97.4	97.5	99.7	100
D	18.2	18.5	39.0	39	71.4	72.0	69	69	97.4	98	99.7	100
F	18.2	19.5	39.0	39.5	71.4	71.8	69	69	97.4	98	99.7	100

4.13 (ST 13) PROCESS COMPUTER

The purpose of ST-13 is to verify the proper operation of the NSS 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 GAF's are compared to the results from BUCLE.

The thermal limits and LPRM GAF comparison between the process computer (P/C) and BUCLE were performed twice during the Dynamic System Test Case (DSTC) and once when verifying proper P1 symmetry flag operation. The results are as follows:

DSTC First Comparison

	P/C Location	P/C Value	BUCLE Location	BUCLE Value	DIFFERENCE	Acceptance Criteria
MCPR	23-36	2.618	23-36	2.615	0.1%	2%
MLHGR	31-36-5	5.92kw/ft	31-36-5	5.92kw/ft	0.0%	2%
MAPLHGR	31-26-5	5.02kw/ft	31-26-5	5.62kw/ft	0.0%	2%

Maximum LPRM GAF Difference = 1.24% 2%

DSTC Second Comparison

	P/C Location	P/C Value	BUCLE Location	BUCLE Value	DIFFERENCE	Acceptance Criteria
MCPR	23-36	2.604	23-36	2.598	0.23%	2%
MLHGR	31-36-5	5.92kw/ft	31-36-5	5.93kw/ft	0.17%	2%
MAPLHGR	29-26-5	5.02kw/ft	29-26-5	5.03kw/ft	0.20%	2%

Maximum LPRM GAF Difference = 1.23% 2%

Asymetric Comparison

	P/C Location	P/C Value	BUCLE Location	BUCLE Value	DIFFERENCE	Acceptance Criteria
MCPR	39-52	1.872	39-52	1.871	0.05%	2%
MLHGR	35-18-5	9.88	35-18-5	9.82kw/ft	0.61%	2%
MAPLHGR	41-44-4	8.34	43-26-4	8.26kw/ft	0.96%	2%

Maximum LPRM GAF Difference = 1.05% 2%

All the level 2 criteria were satisfied. The process computer accurately performed its design calculation.

Subtest 13.1, Dynamic System Test Case (DSTC), deals primarily with the dynamic testing and verification of NSSS process computer programming, data storage and retrieval, array initialization, scan and alarms interfacing, and subroutine calling. Included in the DSTC are the checkouts of P-1, P-2, P-3, P-4, P-5, OD-1, OD-3, OD-7, OD-8 and OD-15.

Subtest 13.2, Specified LPRM Substitute Value and BASE Distribution (OD-2), verifies that the new TIP data and new BASE values are properly calculated and stored after an OD-2 is performed.

Subtest 13.3, Bundle Power Symmetry, involves comparing the NSSS process computer calculated values to the BUCLE calculated values. This test is performed with the symmetry flag set to represent both symmetric and asymmetric control rod pattern.

The initial checkout with ST 13.1 began on 12/6/82 with the plant in Test Condition 2. The initial checkouts of OD-3, OD-7 and OD-8 performed at the beginning of the DSTC were successful. Core thermal power (CTP) from OD-3 was within 1.7% of the manually calculated value. When OD-15, option 2 was run, the core energy (ECOR) and generator energy (EGEN) were not calculated properly. The software was fixed and OD-20 had to be rerun. A subsequent check of OD-3 showed that CTP from OD-3 and the manually calculated value were within 0.02% of each other. When the P-4 verification was attempted, it was found that ECOR and EGEN were not being calculated correctly. This was caused by a combination of two problems. First was that a computer conversion factor, CPM, was incorrectly set, and secondly, the generator megawatt metering was miscalibrated. When both of these were fixed, P-4 ran correctly. The feedwater flows were then manually calculated and verified against the computer values as being performed correctly. When OD-18 was demanded, the message "CORE FLOW UNKNOWN" was received. This was traced back to OD-18 looking at the wrong computer address for core flow, and the problem was easily fixed. However, after this, when OD-18 was demanded, it still would not bracket the core flow correctly. This problem was fixed and verified with subsequent testing outside of the DSTC. At this point, the OD-3 value of CTP and the manually calculated value were shown to be 1.8% apart.

At this point in the test, OD-1 was run. A problem was encountered in being able to only run one TIP in the core at a time. This did not effect the actual OD-1 program, and the eventual fix was to remove a capacitor on the interface cards for the TIPs. During OD-1 operations, many computer crashes were encountered due to what was ultimately determined to be a missing instruction in the computer operating system. The missing instruction was inserted and now there is no problem with running OD-1 to completion. Nor is there a problem with demanding other NSS programs while OD-1 (or P-1) is running. All of the OD-1 calculations were verified as being done correctly; however, after OD-1 was run, there was a "DIB BLOCK" that prevented OD-[illegible] from being run. A change was made in the OD-1 logic to reset the DIB block upon completion of OD-1. At this point, P-1 was demanded for the first time and its execution time was 10 minutes to perform 3 iterations.

On the P-1 edit, all LPRM's were listed as being failed. After running OD-20, it is necessary to run OD-14, option 9, before demanding P-1. Unfortunuately, shortly after P-1 was run, a reactor scram occurred and the DSTC was restarted, however, certain steps that were already verified were not reperformed.

When the DSTC was restarted, a higher power level was achieved at Test Condition 3 and another OD-1 was run before demanding P-1. To verify the proper operation of P-1, edits from OD-6, OD-9, OD-10, OD-11 and OD-16 were necessary. It was found that some of the data on the OD-9 edits did not match what was on the OD-1 edits despite the fact that the locations listed on the edits were the same. This was traced back to OD-9 not looking at the correct computer address to print out the data, and the problem was fixed. After P-1 was run, a Security Log is required which is to be later loaded into BUCLE. A problem was encountered while this was attempted. Susquehanna is the first site to use cassette magnetic tapes for obtaining Security Logs. Due to the interface between the Alarm typer and the cassette unit, everything that gets printed out on the Alarm typer also gets put onto the tape if the tape unit is operating. Hence, it is important to get a Security Log without first blocking P-4. This problem was ultimately resolved with software modifications to use the OD-typer as the cassette interface with input and output prompts directed to the alarm typer while the SECLOG is being written. Once a Security Log was obtained, BUCLE was run to do the P-1 BUCLE comparisons. Differences were noted in bundle flows and bundle powers. Also, large differences on the order of 10% were seen in the thermal limit comparisons. It was noted that there was a direct correlation between high bundle flows and high bundle powers. This problem turned out to be the most elusive encountered during the DSTC due to the compounded errors encountered in the code. First, the computer was using control cell core logic instead of what was required. When this was fixed (only a data book change), the MFLPD values matched, but everything else was still wrong. The remainder of the problem was because a symbol being used in P1-2 was intended to be a local parameter, but elsewhere in P-1 it was listed under a common statement. The result was that the data array was being written over. The correction was easy once the problem was identified and P-1 was verified as being correct. The DSTC was restarted from a point just after the OD-1 was performed. During the subsequent P-1 verification, the P-1 vs. BUCLE data compared very well. It was noted that the bundle fuel segment void fractions (OD-10, option 68) for some bundle decreased near the core top. This was seen on both the OD-10 edits and the BUCLE edits and was accepted as being correct.

After P-1 was verified, the attempt to verify P-2 and P-3 was made. This was rather difficult to do because of the number of problems and restarts encountered during the DSTC. The results showed P-2 to be operating correctly except if a computer restart is performed during the time span covered by P-2. If a computer restart is performed, the daily maximums editted by P-2 only include data acquired after the restart. This problem was traced back to OD-15 option 2 "Computer Outage Recovery Monitor". This program was designed with 3 options, but was set up to use only one option, the option that happened to zero P-2 data when the program was executed. OD-15 was modified and the applicable steps of ST 13.1, P-2 checkout were successfully repeated.

The last part of the DSTC was to verify that a Security Log could be input into the NSS computer and was completed successfully.

In subsequent testing, ST 13.2 (OD-2) and ST 13.3 (P-1 symmetry flag) were both successfully verified.

In general, all Acceptance Criteria were satisfied. There were a number of problems encountered during the performance of these tests but these were resolved satisfactorily and the process computer accurately performed its design calculations.

4.14 (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 in 30 seconds or less from automatic initiations 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 4580 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.

5. The RCIC Steam Isolation Valves HV-1F007, HV-1F008, and Steam Admission Valve HV-1F045 are capable of opening against a differential pressure in the direction of flow of at least 970 psig.

6. The RCIC Minimum Flow Valve HV-1F019 is capable of opening and closing against a discharge pressure of at least 1100 psig.

The RCIC system demonstrated its reliability by never tripping or isolating during testing and by always achieving rated flow within the allowed 30 seconds. The few minor problems that did occur were all Level 2 Acceptance Criteria failures and are described below.

The maximum speed peak limit in the FSAR was 4580 rpm. GE revised their Acceptance Criteria to allow speed peaks up to 4809 after the FSAR was written. Although FSAR limit was exceeded during several tests, the GE limit was never exceeded. A change to the FSAR covering this has been submitted.

Small steps in flow demand were inserted at the flow controller in manual and automatic, injecting to the vessel and CST and at high and low flow. The RCIC system was found to be very stable throughout the entire range of testing with the following exception - some small flow oscillations (approximately 25 gpm peak to peak) were observed with the RCIC system injecting to the vessel at approximately 300 gpm flow. Although not 1/4 wave damped, the flow was stable and was judged to be acceptable by GE.

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.

Dates, Test Conditions and results of RCIC testing is shown on Table 4.14-1. All Acceptance Criteria were satisfied.

				LEVEL 1		LEVEL 2					
DATE	TEST CONDITION	TEST	PRESSURE	[1] TIME TO RATED FLOW < 30 sec.	[2] TRIP ?	[1] SPEED PEAK < 4580	[2] OSCILLATIONS	[3] NO SEAL LEAKAGE	[4] ΔP SWITCH SETTING	[5] STROKE STEAM VALVES	[6] STROKE MIN FLOW VALVE
11-7-82	Heatup	14.1	Rated	see note 1.	NO	FAIL	------	-----	------	------	------------
11-10-82	Heatup	14.1	Rated	18 sec.	NO	[2] 4710	NONE	SMALL	PASS	----	PASS
11-10-82	Heatup	14.2	Rated	18.5 sec.	NO	4534	ACCEPTABLE	SMALL	PASS	PASS	----
11-11-82	Heatup	14.4	150#	8.1 sec.	NO	2416	NONE	NONE	----	----	----
11-12-82	Heatup	14.1	150#	8 sec.	NO	2500	NONE	NONE	----	----	----
11-15-82	TC-1	14.3	Rated	19 sec.	NO	4533	NONE	NONE	PASS	----	----
11-19-82	TC-1	14.3	Rated	19.7 sec.	NO	[2] 4651	NONE	NONE	PASS	----	----
12-2-82	TC-2	14.5	150#	----	NO	----	NONE	NONE	----	----	----
1-11-83	TC-3	14.5	Rated	----	NO	----	NONE	NONE	----	----	----

NOTES:

1. Test used for the two hour demonstration only
2. The GE Acceptance Criteria allowed a speed peak of 4809 RPM

---- 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.
14.5 Demonstration Of 18 Month RCIC System Logic Functional Check, SO-50-003.

TABLE 4.14-1

RCIC TEST CONDITIONS AND RESULTS

4.15 (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:

Level 1

1. The average pump discharge flow must be equal to or greater than 5000 gpm in 25 seconds or less 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.

Level 2

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 the rated speed of 4130 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 Steam Line Isolation Valves HV-1F002, HV-1F003 and Steam Admission Valve HV-1F001 are capable of opening against a differential pressure in the direction of flow of at least 970 psig.

6. The HPCI Minimum Flow Valve HV-1F012 is capable of opening and closing against a pump discharge pressure of at least 1100 psig.

7. 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 140°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 9 out of 10 tests. In the tenth test, ST 15.1 on 1-1-83, the system started in 25.1 seconds with flow exceeding 4900 gpm during the interval between 17 and 25 seconds. Evaluation by General Electric has determined that the results were acceptable.

Some problems were experienced in tuning the HPCI flow controller. Difficulty was experienced in trying to find the optimum controller settings so that the system would start in 25 seconds but not trip, yet would still be stable for step changes in flow demand. As a result, ST 15.1 and 15.2 had to be repeated.

Small steps in flow demand were inserted at the flow controller in manual and automatic, injecting to the vessel and CST, at high and low flow. The HPCI system was found to be very stable throughout the entire range of testing with the following exception - during flow step changes below 3000 gpm, the HPCI system was observed to overshoot the demanded flow by 300 gpm for a 500 gpm step. This characteristic was analyzed by GE and determined to be acceptable since the flow did converge on the demanded flow and the flow was stable.

Dates, Test Conditions and results of HPCI testing is shown on Table 4.15-1. All Acceptance Criteria were satisfied.

				LEVEL 1		LEVEL 2						
DATE	TEST CONDITION	TEST	PRESSURE	[1]TIME TO RATED FLOW < 25 SEC.	[2]TRIP ?	[1]SPEED PEAK INITIAL/SUBSEQUENT 4[illegible]43 4130	[2]OSCILLATIONS	[3]NO SEAL LEAKAGE	[4]Δ P SWITCH SETTING	[5]STROKE STEAM VALVES	[6]STROKE MIN FLOW VALVE	[7]NPSH >21 ft.
11-8-82	Heatup	[1]15.1	Rated	24 sec.	NO	------------	--------	----	----	----	----	52 ft.
11-10-82	Heatup	15.1	Rated	21 sec.	NO	3594 / 3927	NONE	NONE	PASS	----	----	----
11-12-82	Heatup	15.1	150#	16 sec.	NO	1875 / 2875	NONE	NONE	----	----	----	----
12-31-82	TC-3	[2]15.2	Rated	22.7 sec.	NO	3300 / 3900	ACCEPTABLE	NONE	PASS	----	----	----
1-1-83	TC-3	15.1	Rated	[3]25.1 sec.	NO	3200 / 3900	NONE	NONE	PASS	----	PASS	----
1-4-83	TC-3	15.3	Rated	24.5 sec.	NO	3895 / 3958	NONE	NONE	PASS	----	----	----
1-8-83	TC-3	15.4	150#	14 sec.	NO	2750 / 2750	NONE	NONE	----	----	----	----
1-8-83	TC-3	15.1	150#	15 sec.	NO	2800 / 2750	NONE	NONE	----	----	----	----
1-12-83	TC-3	15.3	Rated	20.5 sec.	NO	3700 / 4000	NONE	NONE	PASS	----	----	----
1-16-83	TC-5	15.4	Rated	24.5	NO	3400 / 3912	NONE	NONE	----	----	----	----
2-3-83	TC-6	15.2	Rated	----	----	---- ----	----	----	----	PASS	----	----

NOTES:

1. Adjustments were made to the HPCI controller following this test to improve HPCI performance
2. 15.2 was performed on 12/29/82 in TC-3. Oscillatory behavior was observed. The controller settings were changed requiring all previously completed HPCI tests to be repeated.
3. Evaluation by General Electric determined that results are acceptable.

---- 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.
15.4 18 MONTH HPCI SYSTEM AND LOGIC FUNCTIONAL TEST SO-52-003.

TABLE 4.15-1

HPCI TEST CONDITIONS AND RESULTS

4.16 (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:

Level 1

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.

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 the plant would not scram during recirculation pump trips and restarts, and would maintain a temperature gradient as 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.

4.17 (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 at the end of Test Condition 6. 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:

Level 1

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.

Level 2

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 during and at the end of Test Condition 6.

All piping tested during the Startup Test Program, as stated above, was finally re-examined following reactor shutdown at the end of Test Condition 6 to determine that subsequent relaxing of piping systems after the heatup/cooldown thermal cycle was as expected.

Problems encountered during the Startup Test Program System Expansion testing were very minimal. Only 15 spring hangers (less than 1% of those examined) required re-adjustment to design load when they were observed to be out of their normal operating range.

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

4.18 (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 (1/1/83, 1/2/83) and at Test Condition 6 (2/7/83). The average total uncertainty for all test sets was 1.53%. Level 2 criterion is 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 4.18.1 and detailed results are given on table 4.18.2.

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 reproducibly 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.

One set of TIP data was taken at the 66% power level and two other sets at 97% power. All calculations were performed using the Mark III program "TIPREPRO". Two sets of data were taken in Test Condition 6 and averaged with the data set taken at Test Condition 3. Only one set of data was used during TC-3 because the second security log did not contain a complete data set. This was acceptable as the NSSS supplier only required one data set per test condition.

TABLE 4.18-1

PLANT CONDITIONS

Test Condition:	TC 6	Test Condition:	TC 3
Date Performed:	2/7/83	Date Performed:	1/1/83 - 1/2/83
Core Power (MWt):	3201.4 (97.4%)	Core Power (MWt):	2167 (65.8%)
Generator Output (MWe):	1074.75 (99%)	Generator Output (MWe):	739.5(68.1%)
Core Flow (Mlb/hr):	97.5 (97.5%)	Core Flow (Mlb/hr):	89.4 (89.4%)
Dome Pressure (PSIG):	999.5	Dome Pressure (PSIG):	970

TABLE 4.18-2

RESULTS

%POWER	RANDOM NOISE	GEOMTRICAL UNCERTAINTY	TOTAL UNCERTAINTY	TEST CONDITION
65.8%	1.46%	.78%	1.657%	3
97.4%	1.08%	1.08%	1.53%	6
97.4%	1.08%	.90%	1.41%	6

The average total uncertainty is: 1.53%

Level 2 Criteria: ≤ 6%

4.19 (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-TP-002 and RE-TP-004 to determine core thermal power and core performance parameters respectively. RE-TP-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-TP-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-TP-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-TP-001 , Core Thermal Power Evaluation (Computer Method) and RE-TP-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 4.19-1.

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

Test exceptions were written at several test conditions stating that the value of K(f) used was overly conservative because sufficient data was not available to determine the appropriate manual flow control curve to use. In all cases the value of K(f) used was the most conservative value. Because the most conservative value was used, the MCPR thermal limit calculations were conservative during the test program. The flow adjustment factor K(f) is used as follows:

$$FLCPR = \frac{CPR\ Limit * K(f)}{CPR}$$

The flow adjustment factor adjusts FLCPR at core flows less than the rated core flow. The factor K(f) insures that the operating MCPR limit (automatic control) or safety limit MCPR (manual control) is not exceeded should flow increase to the maximum flow rate. The automatic flow control line is the most conservative and was used during the Startup Test Program.

It is noted that margin to thermal limits at 100% power (equilibrium xeron conditions) is about 12%. This margin provides for operating flexibility as well as helping to maintain fuel integrity during plant transients.

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-TP-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-TP-002-1. Core flow is available from jet pump instrumentation and is recorded on form RE-TP-002-1.

Using RE-TP-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-TP-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-TP-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 4.19-2 and test results are provided in table 4.19-3.

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-TP-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-TP-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 4.19-4 and test results are provided in table 4.19-5.

MAPLHGR Limit Versus Average Planar Exposure

Table 4.19.1

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 4.19-2

ST 19.1 Plant Conditions

Test Condition	1	2
Core Power (MWT)	642 (19.5%)	1131 (42.5%)
Generator Output (MWe)	160	340
Core Flow (Mlb/hr)	32 (32%)	42.5 (42.5)
Dome Pressure (psig)	950	960
Date Performed	11/17/82	11/28/82

Table 4.19-3

ST 19.1 Results
(Most Limiting Thermal Limits
From BUCLE)

x-y-z	MFLPD	TC
55-28-15*	.273	1
27-28-6	.349	2

x-y	MFLCPR	TC
23-36	.394	1
25-36	.559	2

x-y-z	MAPRAT	TC
5-34-15	.273	1
27-28-6	.333	2

* The reader is advised that coordinates represent four symmetric locations in the core.

Table 4.19-4

ST 19.2 Plant Conditions

Test Conditions	3	4	5	6	Warranty
Core Power (MWT)	2156(65.5%)	1518(46%)	2211(69%)	3195(95%)	3290(99.9%)
Generator Output (MWe)	736	462	760	1072.5	1074
Core Flow (Mlb/hr)	92.4(92.4%)	36(36%)	60(60%)	97.5(97.5%)	100(100%)
Dome Pressure (psig)	970	948	969	999	1005
Date Performed	12/31/82	3/8/83	1/18/83	2/7/83	4/4/83

Table 4.19-5

ST 19.2 Results
(Most Limiting Thermal Limits)

x-y-z	MFLPD	TC
39-08-12	.649	3
23-26-5	.442	4
15-48-12	.622	5
7-32-12	.829	6
25-8-12	.876	W

x-y	MFLCPR	TC
39-08	.607	3
23-10	.684	4
39-10	.784	5
17-48	.833	6
23-10	.873	W

x-y-z	MAPRAT	TC
39-08-12	.654	3
23-36-4	.425	4
95-14-12	.624	5
7-32-12	.837	6
25-8-12	.882	W

4.20 (ST20) WARRANTY RUN

The Warranty Run successfully demonstrated that the Nuclear Steam Supply System provided sufficient steam to satisfy the NSSS contract. This test was implemented with the reactor operating at or near rated power (3293 MWt) using temporary and permanent instrumentation and the plant process computer to monitor heat balance data.

The Acceptance Criteria were as follows:

<u>Level 1</u>

1. The average reactor core thermal power (CTP) shall not exceed 3293 MWt.

2. The Maximum Average Planar Ratio (MAPRAT) shall be less than or equal to 1.0.

3. The Maximum Fraction of Limiting Critical Power Ratio (MFLCPR) shall be less than or equal to 1.0.

4. The Maximum Fraction of Limiting Power Density (MFLPD) shall be less than or equal to 1.0.

<u>Level 2</u>

1. The NSSS shall be capable of supplying 13,483,000 pounds per hour of steam of not less than 99.7% quality at a pressure of 985 psia at the outlet of the second main steam line isolation valve, as based upon a final Feedwater temperature of 383°F, measured as near the reactor pressure vessel as practicable, and a control rod drive feed flow of 32,000 pounds per hour at 80°F.

ST 20.1, Two Hour Demonstration, was performed three times during the 100 hour run. All Acceptance Criteria were satisfied.

ST 20.2, Warranty Run 100 Hour Demonstration was run concurrently with the three two hour demonstrations. All Acceptance Criteria were satisfied with final readings for MAPLHGR=.891, MCPR=.938 and LHGR=.881. No problems were encountered.

4.21 (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 by causing failure of the operating pressure regulator) and (2) By inserting a control rod two 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 26-07. LPRM 24-09B near the rod tip was chosen as the selected LPRM. Rod 26-07 was notched in 2 notches and LPRM 24-09B indicated an 11% local flux depression. Plant stability was adequately demonstrated. The test was conducted at min flow and the 100% rod line. This test was repeated in Test Condition 4 using rod 22-35 and LPRM 24-33A. The results were similar. Local flux depression was about 12% 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 24-09. The pressure transient included step changes and simulated failure of the operating pressure regulator. The test was repeated at Test Condition 4 with LPRM string 24-09 chosen again as the monitored string. 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. 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. 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.

4.22 (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:

Level 1

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

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 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.

ST 22.3, Pressure Regulator Test-Bypass Valves Controlling, was run at Test Condition 1, 15% power. At Test Condition 1 there was an oscillatory response seen due to bypass valves going completely closed. This was not considered to be a problem because at higher power levels the Bypass Valves would not go completely closed due to pressure regulator failure. It was verified at this test condition that the pressure response time was less than 10 seconds from the

initiation of a pressure setpoint step change. There were no steady state limit cycles apparent. The Flow Biased Scram (Heat Flux) margin to scram was 44% and the High Pressure margin to scram was 109 psi. There were no exceptions which violated Acceptance Criteria at this Test Condition.

ST 22.1, Pressure Regulator Test - Control Valves Controlling, was run at 40% power. All Acceptance Criteria were proven with acceptable margins to scram. ST 22.2, Pressure Regulator Test - Control Valve and Bypass Valves Controlling, was performed at Test Condition 2 (40% rated). All Acceptance Criteria were proven with a 3.2 second maximum response time, no variations in steady state limit cycles and margins to scram of 25.5% for Flow Biased Scram and 104 psi for High Pressure scram. ST 22.3, Pressure Regulator Test - Bypass Valves Controlling, also was run at 40% rated power. The Acceptance Criteria were proven with margin to scram of 24% for the flow Biased setpoint, and 104 psi for the high pressure scram.

ST 22.1, 22.2 and 22.3 were run at Test Condition 3 with the following results:

				MARGIN - TO - SCRAM	
TEST	% POWER	MAX RESPONSE TIME	MAX VARIATION	NEUTRONFLUX	HIGH PRESS
22.1	75%	4.5 sec.	0%	38.7%	76 psi
22.2	63%	6.9 sec.	0%	20.0%	90 psi
22.3	60%	4.5 sec.	0%	28.0%	90.5 psi

All Acceptance Criteria were proven; there were no steady state limit cycles.

At Test Condition 4, ST 22.1, 22.2 and 22.3 were performed. The following table shows the results of these tests:

				MARGIN - TO - SCRAM	
TEST	% POWER	MAX RESPONSE TIME	MAX VARIATION	NEUTRONFLUX	HIGH PRESS
22.1	43%	3.88 sec.	0%	54.4%	74.6 psi
22.2	43%	4.1 sec.	0%	64.25%	74.5 psi
22.3	43%	3.88 sec.	0%	55.7%	75 psi

All Acceptance Criteria were satisfied.

At Test Condition 6, ST 22.2, 22.2 and 22.3 were performed with the following results:

TEST	POWER	MAX RESPONSE TIME	MAX VARIATION	MARGIN - TO - SCRAM	
				NEUTRONFLUX	HIGH PRESS
22.1	97.5%	4.06 sec.	0%	13.9%	43.8 psi
22.2	97.5%	4.12 sec.	0%	13.5%	42.5 psi
22.3	97.5%	3.75 sec.	0%	13.2%	41.5 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.

4.22-A (ST22) PRESSURE REGULATOR (Testing Following Conversion From Partial to Full Arc Steam Admission)

The results of the testing showed that the Pressure Regulator controller settings provided smooth and stable response to pressure changes and simulated Pressure Regulator failures. This retesting was performed for the purpose of demonstrating response and stability following the conversion of control valve steam admission from partial arc to full arc.

The Acceptance Criteria were as follows:

Level 1

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

Level 2

1. Pressure control system related variables may contain oscillatory modes of response. In these cases, the decay ration 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.

ST 22.1, Pressure Regulator Test - Control Valves Controlling, and ST 22.2, Pressure Regulator Test - Control Valves and Bypass Valves Controlling, were conducted at 20%, 75% and 100% rated reactor thermal power following the startup subsequent to the Pre-Commercial Operation Outage. The response of the regulator to step changes and simulated regulator failures was stable and no steady state limit cycles were observed. The response times and margins-to-scram were follows:

TEST	% POWER	MAX RESPONSE TIME	MARGIN - TO NEUTRON FLUX	-SCRAM HIGH PRESSURE
22.1	20	4.0 Sec	95.2%	102 psi
22.2	20	5.0 Sec	96.2%	102.1 psi
22.1	75	4.8 Sec	35.8%	66 psi
22.2	75	6.5 Sec	37.2%	66.7 psi
22.1	100	5.4 Sec	11.8%	40.2 psi
22.2	100	4.0 Sec	11.2%	40.5 psi
Acceptance Criterion		$\leq$10.0 Sec	$\geq$7.5%	$\geq$10.0 psi

All Acceptance Criteria and test objectives as set forth in the Final Safety Analysis Report were satisfied.

4.23 (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 was 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.

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. 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 the assumed value in the FSAR. (6)

Level 2

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. 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

ST 23.1 Startup Controller Level Step - 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 45%, TC-3, with reactor power at 75% and TC-6, with reactor power at 97%, 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. At TC-3, step changes of only 15% maximum could be made without deadheading the pump. It was found that during the performance of these tests, it was necessary to keep the controllers for the minimum flow valves for the feedwater pumps in the manual mode to prevent these valves from opening up during the transient, thus causing a more severe transient to occur. On the B&C Feedwater Pumps, the peak overshoot exceeded the Acceptance Criteria and some small limit cycling was observed. It was determined, based on the overall performance of the feedwater system, that this overshoot and limit cycling was acceptable.

ST 23.3 Feedwater System Level Setpoint Changes - At TC-2 with reactor power at 45%, TC-3 with power at 75%, TC-4 with power at 43%, TC-5 with power at 69% and TC-6 with power at 97%, 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. At TC-6 some small irregular oscillatory behavior was noted, however, the overall response was considered acceptable.

ST 23.4 Loss of Feedwater Heating - At TC-6 with reactor power level at 85%, 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 44°F and a heat flux increase of approximately 5%. 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 - At TC-6 with reactor power level of 97%, the "B" feedwater pump was tripped. The recirculation pump speeds ran back to the #2 Limiter (approximately 46% speed), preventing a reactor scram from low water level.

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. Initially, the runout flow as calculated exceeded the Level 1 criteria. The Reactor Feed Pump High Speed Stop settings were readjusted and that portion of the test was re-run. The final calculated value of runout flow was 18.86 Mlb/hr versus the maximum allowable value of 19.05 Mlb/hr.

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

4.24 (ST24) TURBINE VALVE SURVEILLANCE

Turbine Valve Surveillances were performed during the Startup Test Program 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 were performed at various power levels to 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 consisted of the opening and closing of the valves individually and recording the parameter changes which were affected by this operation. The following Acceptance Criteria were 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 (134%).

ST 24 was composed of four Subtests:

ST24.1 - Stop Valve Testing

ST24.2 - Control Valve Testing

ST24.3 - Bypass Valve Testing

ST24.4 - Combined Intermediate Stop Valve Testing

ST24.1 and 24.2 were performed during Test Condition 3 at 56% power, Test Condition 5 at 69% power, and Test Condition 6 at 100% power. Once it was determined that Stop Valve and Control Valve testing could be performed at 100% power, ST24.3 and 24.4 were also run. The results of turbine valve testing is tabulated in Table 4.24-1.

No problems were encountered during turbine valve testing. All Acceptance Criteria were satisfied with a large margin.

TEST PARAMETERS			MARGIN TO SETPOINTS			
ST	TEST CONDITION	REACTOR POWER(%)	NEUTRON FLUX (%)	VESSEL PRESSURE	FLOW BIASED HEAT FLUX(%)	STEA FLOW
24.1	3	56	59.5	87.4	49.7	78.
	5	69	46.3	81.0	17.4	62.
	6	100	15.5	37.8	14.6	28.
24.2	3	56	57.5	77.4	48.1	75.
	5	69	45.1	66.2	15.8	58.
	6	100	13.1	13.2	28.5	25.
24.3	6	100	15.6	38.2	14.3	29.
24.4	6	100	15.8	38.6	14.6	30.
ACCEPTANCE CRITERIA			7.5	10.0	5.0	10.

TABLE 4.24-1

TURBINE VALVE TEST RESULTS

4.24A (ST24) TURBINE VALVE SURVEILLANCE (Testing Following Conversion From Partial To Full Arc Steam Admission)

The results of the testing showed that acceptable margins-to-scram existed, following the conversion of control valve steam admission from partial arc to full arc, to enable periodic surveillance testing of the control and stop valves up to the 100% rated reactor thermal power level.

The Acceptance Criteria were as follows:

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 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 (134%).

The testing consisted of the opening and closing of the valves individually, recording the parameter changes and calculating the margin to scram for reactor pressure, heat flux, neutron flux and peak steam line flow isolation. The margins-to-scram were as follows:

		MARGINS - TO - SCRAM			
TEST	% POWER	PRESSURE	NEUTRON FLUX	HEAT FLUX	HIGH STEAM LINE FLOW
24.1	100	44 psi	14.5%	12.6%	27.2%
24.2	100	37 psi	13.4%	11.4%	26%
Acceptance Criteria		≥10 psi	≥7.5%	≥5.0%	≥10.0%

All Acceptance Criteria and test objectives as set forth in the Final Safety Analysis Report were satisfied.

4.25 (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 determine 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 and Test Condition (TC)1; Subtest 25.2-Full Closure of Fastest MSIV during TC-3, TC-5, and TC-6 at the highest power level during which a single valve could be tested without causing a scram; and Subtest 25.3-Full Isolation at TC-6.

The acceptability of the fast criteria (3 seconds) is determined by utilizing the full stroke time without delay extrapolated 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 time with delay extrapolated for 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. (1)

2. The positive change in heat flux following closure of all MSIVs shall not exceed predicted values by more than 2% of rated value. (1)

3. Following the closure of all MSIV's, the reactor must scram. (1)

4. The average of the closure times for the fastest MSIV in each steam line, exclusive of delay, shall not be less than 3.0 seconds. (2)

5. Closure time for any MSIV, including delay, shall not be greater than 5.0 seconds.

6. Closure time for the fastest MSIV shall be greater than or equal to 2.5 seconds.

7. Feedwater control settings must prevent flooding the main steam lines during the full isolation test. (1)

8. The time delay between the close initiation signal and the extrapolated initial valve movement from 100% open for any MSIV shall be less than or equal to 0.5 seconds.

9. The closure time for any MSIV shall not be less than 30 seconds.(3)

Level 2

1. 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. (1)

2. 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. (1)

3. If water level reaches Level 2 setpoint during the MSIV full closure test, RCIC shall automatically initiate and reach rated flow. (1)

4. During the MSIV full closure test, the relief valves must reclose properly (without any detectable leakage) following the pressure transient. (1)

5. During full closure of individual MSIVs, peak vessel dome pressure must remain at least 10 psi below the flow biased scram setting value.

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

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

8. During full closure of individual MSIVs, the peak simulated heat flux must remain at least 5% less than its scram value.

   (1) Applicable to ST 25.3 only

   (2) Not applicable to ST 25.2

   (3) Applicable to ST25.2 run after Pre-Commercial Operations Outage only.

Subtest 25.1

During Initial Heatup at rated pressure and during TC-1 at approximately 19% power, each MSIV was individually closed to demonstrate proper operation and to measure its closure time. Proper operation was demonstrated and closure times were within limits; however, calculation of the delay time from initiation signal to start of valve movement resulted in two of eight cases in a negative number. Negative delay times are not possible. The negative number results from in-accuracies in the location of the limit switches and non-linear motion of the valve. Neutron flux, reactor pressure, heat flux and steam flow margins to scram or isolation were calculated and results are listed in Table 4.25.1.

Subtest 25.2

Prior to the Pre-Commercial Operations Outage, full closure testing of the fastest MSIV (1F022C) was performed at 56%, 69% and 86% rated reactor thermal power. During the outage, an oil leak was detected and repaired on the 1F022C valve actuator. During the subsequent startup, this testing was repeated at 5%, 85% and 88.5% rated reactor thermal power. This testing also demonstrated that closure time for the fastest MSIV was consistent with the Plant Technical Specifications and the Final Safety Analysis Report. During testing at the lower power levels, margins to scram or isolation from neutron flux, reactor pressure, heat flux and steam flow were calculated and used to extrapolate the highest power level at which a valve should be tested. The highest power level at which a single MSIV could be tested and still yield acceptable margins to scram and isolation was extrapolated and demonstrated to be 88.5%. Results of all testing are listed in Table 4.25-1.

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 4.25.1. The actual pressure rise experienced during this test was such that no safety/relief valves lifted. RCIC and HPCI auto started and restored reactor water level to normal. The maximum water level experienced was +65".

A Level I criteria failure occurred when MSIV 1F022C closed in 2.1 seconds. Subsequent investigation found an oil leak in a fitting on the oil reservoir in the valve actuator. The loss of oil explains the gradual decrease in closure time seen on this valve (see Table 4.25.1). The oil leak was repaired and the valve closure time adjusted to within acceptable limits.

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.

Table 4.25-1

ST 25 TEST DATA

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/2%	12.1	127	N/A	125	3.5	3.2	
25.1.TC-1/39%	98	120	42	107	3.8	3.7	
25.2/TC-3/5%	58	83	49	62	N/A	3.2	
25.2/TC-5/69%	41	62	14	40	N/A	2.9	
25.2/TC-6/86%	19	33	9	12	N/A	2.8	
						Min.	Max.
25.2/*/ 5%	112	116	54.8	122	N/A	3.46	3.84
25.2/*/85%	19	34	11.3	23	N/A	3.32	3.70
25.2/*/88.5%	18.5	37	5.7	19	N/A	3.39	3.76
Acceptance Criteria	≥7.5%	≥10.0 psi	≥5.0%	≥10%	---	≥3.0	≤5.0

* - Startup follow Unit 1 Pre-Commercial Outage

	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%	1.1%	0	116	50	3.2	2.1	64.6

4.26 (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:

<u>Level 1</u>

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

<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.

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 at approximately 3 percent rated reactor thermal power with reactor dome pressure at 170 psig during the Initial 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 Acceptance Criteria were met with one exception. The change in bypass valve position during the opening of relief valve "S" was 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 78.1%. The change in bypass valve position during relief valve "S" operation was 76.7%, thus not meeting the Acceptance Criteria. This exception was resolved when General Electric San Jose Engineering concluded that the performance of relief valve "S" was adequate for the existing low pressure plant conditions and that the valve operation would be re-examined during the performance of ST26.2, Relief Valve Rated Pressure Test at Test Condition 2.

ST26.2, Relief Valve Rated Pressure Test, was implemented at 45 percent rated reactor thermal power with reactor dome pressure at 944 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.

4.27 (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) and by opening the generator output breaker at Test Condition 6 (Subtest 27.2). During these transients, reactor water level, pressure, and simulated heat flux were recorded and compared to predicted results and Acceptance Criteria. At 25% 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 all three transients, main turbine stop, control and bypass valve positions and reactor water level were recorded and compared to Acceptance Criteria. The Acceptance Criteria verified in these tests are as follows:

Level 1

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.

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.

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

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)

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)

Level 2

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

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

3. The positive change in vessel dome pressure occurring within the first 30 seconds after the closure of all MSIVs 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)

4. The positive change in heat flux occurring within the first 30 sec. after the closure of all MSIVs 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 from the start of stop/control valve motion to start of control rod motion, and will use beginning of life nuclear data.(1)

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.

(1) Not applicable to ST 27-3 - Generator Trip Within Bypass Capacity.

Subtest 27.1 - Turbine Trip

In this subtest a turbine trip was initiated from 75% 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 4.27.1 for a summary of the results of this test.

A Level 1 Criteria failure resulted during this test in that the recirculation pump coastdown time was faster than that predicted. A revised calculation was performed using the correct Susquehanna sensor time constant. This calculation showed that the recirculation pumps coastdown was acceptable.

During ST27.1 dated 1/6/83, total feedwater flow increased to approximately 16 millon lbs/hr to recover water level which had dropped to 7". As a result of this high flow, Reactor Feed Pump suction pressure was reduced to approximately 280 psig. RFP's B&C tripped after the designated five second time delay while RFP A recovered water level to normal after peaking at 47". Re-occurrence has been prevented by lowering the high speed stop settings on the RFP's to prevent the high flow condition.

Subtest 27.2 - High Power Generator Load Rejection

In this subtest, a generator load rejection was initiated by opening the Main Generator Breaker 230 KV OCB 1R101. This action initiates a fast closure of the Main Turbine Control Valves to limit the turbine overspeed.

During the Startup Test Program, this subtest was performed twice due to the fact that during the first performance of the subtest, the transfer of the plant electrical loads did not occur, thus invalidating the test. However, since the resulting transient was more severe than the planned transient, the results are reported here.

Subtest 27.2-1 First High Power Generator Load Rejection

In this subtest, a generator load rejection was initiated from 98% power. A sync-check relay sensed an out of phase condition between the power to the auxiliary bus and the startup transformer and prevented fast transfer from occurring. Slow transfer of the bus did take place, however, auto restart of the major loads did not occur because the "Trip/Enable" feature of these breakers was not selected to allow auto restart. This resulted in loss of all

Condensate Pumps, Circulating Water Pumps and Service Water Pumps. The loss of condensate resulted in the trip of all Feedwater Pumps on low suction pressure. Water level decreased rapidly due to void collapse and SRV/BPV openings. RCIC and HPCI auto started at -21" and -35" respectively and restored reactor water level. The MSIV's closed 27 seconds after the trip due to low water level at -28". Recovery from the trip was hampered by the following conditions:

- Loss of Service Water to the Reactor Building Chillers resulted in a drainage of the Service Water side of the heat exchangers. After Service Water was restored, an air binding condition occurred at the Service Water flow switches preventing restart of the chillers until the system was vented.

- Condensate demineralizer valves, which closed under no flow conditions following loss of condensate, had to be hand-cranked off their seats during restart.

- Reactor building Zone III isolation could not be restored due to blown fuses in the damper circuit.

- In addition it was discovered that the plant process computer generates significant amounts of data updates at 2400 hrs. which tend to interfere with sequence of events logs.

The plant response to this high power generator load rejection accompanied by loss of Condensate/Feedwater, Circulating Water and Service Water followed by closure of all MSIVs, was as would have been expected had the transient been anticipated. Pressure was controlled by opening SRV's. The lack of decay heat required the SRVs be manually opened only three times to control pressure. HPCI was used to maintain level until the Condensate System was restored at approximately 500 psi RPV pressure.

Subtest 27.2-2 Second High Power Generator Load Rejection

Prior to this trip, a plant modification was made to remove the sync-check relay from the fast transfer logic of the auxiliary bus and to select the Trip/Enable switches in the slow transfer logic such that the major loads would restart after the bus is re-energized from the startup transformer. The load rejection was performed at 100% power. Fast transfer of the auxilairy bus from the unit auxiliary transformer to the startup transformer occurred. The plant responded as expected and a summary of the results is presented in Table 27.1.

A Level 1 failure of the Acceptance Criteria on recirculation pump flow coastdown occurred. Since the coastdown rate was too slow, the Technical Specification operational MCPR limit for End of Cycle RPT inoperable was input into the process computer to permit continued operation. Investigation into flow sensor response times is ongoing at the time of this report.

Subtest 27.3 - Generator Load Reject Within Bypass Capacity

With the reactor operating at 25% 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.

A failure of the Level 1 criteria which states that the bypass valves should be opened to a point corresponding to greater than or equal to 80% of full open within 0.3 seconds from the beginning of control or stop valve closure motion was encountered during this test. This failure resulted because power level at which the test was performed only required the bypass valves to open 73% to maintain pressure after the turbine trip. This response occurred in 0.2 seconds which was determined to be acceptable.

Overall results confirm that conservative assumptions were made in the analysis of these events in Section 15 of the FSAR. The objectives of the test were met, and all Acceptance Criteria were satisfied.

SUBTEST	POWER LEVEL	PREDICTED PRESSURE RISE	ACTUAL PRESSURE RISE	MAXIMUM WATER LEVEL	PREDICTED HEAT FLUX RISE	ACTUAL HEAT FLUX RISE	CV/SV CLOSE to BPV OPEN DELAY	TIME TO 80% OF BPV OPEN
27.1	75%	123 psi	63 psi	47"	.09%	0	.02 sec.	.18 sec.
27.2-1	98%	95.7	80	35"	.04%	0	.004 sec.	.135 sec.
27.2-2	100%	105.1	81	36"	.04%	0	.02 sec.	.16 sec.

TABLE 4.27-1

ST 27 TEST RESULTS

4.28 (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 single unit 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 November 20, 1982, in Test Condition 1, at 19% rated reactor thermal power. Using the minimum single unit complement of control room operating personnel per the Technical Specifications, the reactor was scrammed 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 3, January 7, 1983. 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.

4.29 (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

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

2. 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.

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

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

5. 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.

The testing was performed during plant power ascension from low to high power levels, beginning in Test Condition 2 (39.5% rated reactor thermal power) and ending in Test Condition 6 at 97.7% rated reactor thermal power and 97.4% 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 very stable and responsive. Table 4.29-1 tabulates the margin-to-scram values determined during each implementation of this test.

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

TEST CONDITION	REACTOR THERMAL POWER (% rated)	CORE FLOW (% rated)	MARGIN - TO - SCRAM AVOIDANCE (%)			
			APRM NEUTRON FLUX		HEAT FLUX	
			ACCEPTANCE CRITERIA MINIMUM	TEST RESULTS	ACCEPTANCE CRITERIA MINIMUM	TEST RESULT
2	39.5	46.3	7.5	80	5	26.6
3	70	100	7.5	46.6	5	42.1
5	74	60.8	7.5	45.1	5	16.1
6	97.7	97.4	7.5	18.3	5	9.8

TABLE 4.29-1

ST 29 TEST RESULTS

4.30 (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 and TC-6; 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.

The Acceptance Criteria verified during this test are as follows:

Level 1

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)

Level 2

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. Peak simulated heat flux must remain at least 5% below its scram tip point.(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 70% power, 100% core flow, a recirculation MG Set drive motor breaker was tripped from the control room. At 98% power, 98% core flow a failure in the MG Set voltage regulator board initiated an MG Set breaker trip. During each of these trips, 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 4.30-1.

Subtest 30.2 Recirculation Pump Trip (RPT) of Two Pumps

At 75% power and 100% core flow, the RPT breakers were simultaneously tripped using a temporary test switch. Parameters were monitored during the transient to be analyzed to demonstrate acceptable pump coastdown performance. The pump coastdown time did not initially meet the coastdown criteria. However, after a more detailed analysis was performed, it was determined that the coastdown time was acceptable.

Subtest 30.3 Recirculation Pump Runback

At 75% power, 100% core flow, a circulating water pump trip was simulated in the recirculation flow control system to cause 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. A summary of the initial and final reactor conditions is presented in Table 4.30-2.

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 57% and core flow at 100% 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.

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

TABLE 4.30-1

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	13.2 in.	71.7%	23.0%
30.1 / 6	7.0 in.	74.5%	12.1%

TABLE 4.30-2

ST 30.3 TEST RESULTS

PARAMETER	INITIAL CONDITIONS	AFTER RUNBACK	UNITS
REACTOR POWER	75	55	%
REACTOR PRESSURE	960	945	psig
FEEDWATER FLOW	9.8	7.0	mlb/hr
STEAM FLOW	10.2	7.5	mlb/hr
CORE FLOW	99	63	mlb/hr
FEEDWATER TEMP	364	340	°F
CORE Δ P	12.5	4.2	psid

4.31 (ST31) LOSS OF OFFSITE POWER

The results of the testing showed that during a simultaneous loss of the main turbine generator and offsite power, the electrical distribution and diesel generator systems functioned properly, the required safety systems, with the exception of the Standby Gas Treatment (SBGTS) and Emergency Service Water "B" Loop Systems, initiated and functioned properly without manual assistance, the reactor vessel water level was automatically maintained above the initiation level of Core Spray, Low Pressure Coolant Injection and Automatic Depressurization Systems, the permanent instrumentation for reactor power, pressure, water level, control rod position, suppression pool temperature, High Pressure Coolant Injection and Reactor Core Isolation Cooling were demonstrated to be operable following re-energization of the 4KV busses by the diesel generators, and the temperature on the discharge side of the Safety Relief Valve that lifted returned to within 10°F of the temperature recorded before the valve opened.

Subsequent to the performance of the test, investigation of the electrical control wiring for the "B" Loop Emergency Service Water pumps uncovered several problems:

(1) Loose wiring on time-delay starting relays on both "B" loop pumps.

(2) Instantaneous contact on time-delay starting relay out of adjustment on one "B" loop pump (OP504D).

(3) Open states link (discontinuity) in control scheme for one "B" loop pump (OP504D).

Following corrective maintenance on both pump control circuits, an operability test was performed and it demonstrated that both pumps automatically started following re-energization of the 4KV busses by the diesel generators.

Also subsequent to the performance of the test, a special test, TP-70-001, was performed to determine the cause of SBGTS failure to start on a loss of power situation. The test revealed the SBGTS was failing to meet the minimum required time to achieve the ΔT across its heater. Further investigation showed the temperature controllers for the heaters failed low on loss of power and had a very slow response time when power was restored. PMR 83-316 changed the heater control logic so the controller would fail high on a loss of power. The SBGTS was retested under normal and loss of power condition with satisfactory results.

The Acceptance Criteria were as follows:

Level 1

1. All safety systems such as the Reactor Protection System, the Diesel Generators, Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (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, Low Pressure Coolant Injection and Automatic Depressurization Systems.

Level 2

1. The temperature measured by the thermocouples on the discharge side of the Safety Relief Valves shall return to within 10°F of the temperature recorded before the valve was opened.

2. Permanent instrumentation for reactor power, reactor pressure, water level, control rod position, suppression pool temperature, HPCI and RCIC shall be demonstrated operable following re-energization of the 4KV busses by the diesel generators.

The turbine-generator was manually tripped and isolated simultaneously with a manual trip of the 13KV breaker which feeds Startup Bus 10 causing a total loss of offsite power to Unit 1. Prior to the performance of the testing, any circuit breakers that could supply power from Unit 2 power sources to Unit 1 were racked out such that all electrical distribution busses on Unit 1 could only be energized from Unit 1. During and following the tripping of the turbine-generator and loss of offsite power to Unit 1, the transient recording system (GETARS) recorded the transient and dynamic response of selected plant variables. Plant emergency operating procedures were implemented and any unit safety systems requiring manual assistance were recorded in the Startup Test 31.1 Official Test Copy. All four diesel generators successfully started and energized their respective 4 KV electrical distribution busses. Safety Relief Valve "B" opened and closed to control reactor pressure. The lowest reactor water level reached during the testing was -27.5 inches which was considerably above the -129 inches initiation level of Core Spray, Low Pressure Coolant Injection and Automatic Depressurization Systems.

Overall results confirm that conservative assumptions were made in the analysis of this event in Section 14 of the FSAR. The objectives of the test were met, and all Acceptance Criteria were satisfied.

4.32 (ST32) CONTAINMENT ATMOSPHERE AND MAIN STEAM TUNNEL COOLING

The results of the testing showed that the following temperatures could be maintained during all operating conditions and post scram conditions:

(1) General drywell area average temperature less than or equal to 135°F.

(2) General drywell area maximum local temperature less than or equal to 150°F.

(3) The area beneath the reactor vessel in the CRD area average temperature less than or equal to 135°F.

(4) Area beneath the reactor vessel in the CRD area maximum local temperature less than or equal to 165°F and minimum local temperature greater than 100°F.

(5) Reactor building portion of the main steam tunnel temperature less than or equal to 120°F.

(6) Reactor pressure vessel support skirt flange temperature less than or equal to 150°F.

(7) Concrete surrounding primary containment main steamline penetrations temperature less than 200°F.

Some problems were encountered maintaining design operating temperatures in the reactor recirculation pump motor area, in the area surrounding the drywell head and in the area inside the biological shield surrounding the reactor pressure vessel. These problems were resolved and are elaborated upon further on in this report.

The Acceptance Criteria were as follows:

<u>Level 1</u>

1. The area under the reactor vessel is maintained below 185°F.

<u>Level 2</u>

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.

2. The area beneath the reactor vessel is maintained at an average temperature less than or equal to 135°F, maximum local temperature not to exceed 165°F, with minimum local temperature above 100°F.

3. The inside base of the shield wall on the RPV skirt area is maintained greater than 100°F.

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.

5. The reactor building portion of the main steam pipeway is maintained at or below 120°F.

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

7. The reactor pressure vessel support skirt flange shall be maintained at or below 150°F.

8. The area surrounding the drywell head shall have a minimum temperature no less than 135°F.

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 and 6 and following the Main Steam Isolation Valve full isolation test from 100% power at the end of Test Condition 6.

The individual Subtest results were as follows:

ST-32.1 Containment Temperature At End of Heatup

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 for two hours after reaching 920 psig. All Acceptance Criteria were met with the following exceptions:

(1) The average temperature in the reactor recirculation pump "B" motor area exceeded its Acceptance Criterion value of 128°F. Average temperature during the testing was 131°F. 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 temperature inside the shield wall in the vessel support skirt area did not meet its minimum Acceptance Criterion value of 100°F. The temperature in this area during the testing was 96°F. This exception was resolved by General Electric letter EAG-4215 to PP&L which stated that when the reactor is at operating temperature, the minimum air temperature requirements beneath the reactor vessel in the vessel support skirt and control rod drive areas shall be greater than 90°F. The testing showed that this minimum temperature was met.

ST 32.4 Main Steam Penetration Concrete Temperature

The surface temperature on the concrete surrounding the main steamline penetrations was monitored by a contact thermistor device during the initial reactor heatup from a reactor pressure of approximately 800 psig to approximately 920 psig. The Acceptance Criterion concrete temperature of less than 200°F was met with the maximum recorded being 101.2°F.

ST-32.2 @ TC-2, Containment Temperature At Steady State

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 45% reactor thermal power. All Acceptance Criteria were met with the following exceptions:

(1) The average and maximum local temperatures in the reactor recirculation pump "A" motor area exceeded their Acceptance Criteria values of 128°F and 135°F respectively. Average temperature during the testing was 129.5°F. Maximum local temperature during the test was 140°F. These exceptions were resolved by the cognizant engineering groups based on the fact that both criteria failures were related to the same permanent temperature instrument which was subsequently determined to have been misinstalled. An inspection of the drywell showed that this temperature element had been installed adjacent to a structural beam in a "dead space" where it could not provide accurate representation of the area temperature near the recirculation pump motor. A modification request to relocate this element was initiated by the PP&L home office engineering group. Based on the indicated temperatures of the other temperature elements located in the recirculation pump "A" motor area (average temperature of 124.2°F and max local of 127.3°F) the temperature in that vicinity was shown to be acceptable.

(2) The temperature in the drywell head area did not meet its Acceptance Criterion minimum temperature of 135°F. Average temperature indication on one temperature element was 127.5°F during the testing. This exception was resolved by the cognizant engineering groups based on the average temperature from all four temperature elements in the drywell head area being 136.4°F. In addition, General Electric and Bechtel engineering groups concluded that there is no minimum temperature requirement in the drywell head area and the only requirement is that the temperature in this area remain at or below 150°F, which was maintained throughout all the testing.

ST-32.2 @ TC-6 Containment Temperature At Steady State

Temperatures were again monitored inside primary containment and in the reactor building portion of the main steam tunnel with the plant in steady state operating conditions at 97% reactor thermal power.

All Acceptance Criteria were met with the following exception in addition to those which were resolved identical to the exceptions encountered during the steady state testing performed at TC-2:

(1) The temperature in the reactor building portion of the main steam tunnel did not meet its Acceptance Criterion maximum of 120°F. The average temperature on one temperature indicator during the test was 128.7°F. Since the other three elements in this same area indicated an average temperature of 96°F, the exception was resolved by issuing a recalibration request for the out of spec instrument since it was highly suspected that the instrument was providing faulty indication. The average of all instrument indications for the area was 104°F which was well below the Acceptance Criterion limit.

Following calibration of the out of spec instrument and temperature monitoring during subsequent startup and power ascension to rated reactor power on May 28, 1983, this instrument indicated 112°F which is below the acceptable operating maximum temperature of 120°F.

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 the main steam isolation valve full closure and scram from 99.8% rated reactor thermal power. All Acceptance Criteria were satisfied with the exception of the same criteria that were not met during steady state operating conditions at Test Condition 6. The resolutions were identical to those during the steady state testing.

4.33 (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:

Level 1

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.

Level 2

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 exceed the acceptable value for that point.

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:

Level 2

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) High Pressure Coolant Injection system (suction piping from suppression pool)

(6) 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 during the approach to and while in Test Condition 6 at approximately 50, 75 and 100% core flow and with each Division of the Residual Heat Removal system operating in the shutdown cooling mode.

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 High Pressure Coolant Injection system taking suction from the suppression pool and discharging to the Condensate Storage Tank at its rated flow rate of 5000 (+100,-0) gpm while in steady state operation. This testing occurred during the Initial Heatup Test Phase with the reactor at rated temperature and pressure.

Testing for (6) 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 Initial Heatup Test Phase 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 failure of a few of the remotely mounted sensors on Main Steam inside containment and Reactor Recirculation piping, which in each case, were repaired or replaced at the next opportunity. In each case where a failed sensor existed, 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.

4.34 (ST34) ROD SEQUENCE EXCHANGE

A rod sequence exchange at 58% power from sequence B2 to sequence A2 was performed on 3/19/83. Level 1 criteria that the "completion of the exchange of one control rod pattern for the complementary pattern with continual satisfaction of all licensed core limits" was satisfied. In addition level 2 criteria that "all nodal powers shall remain below the PCIOMR threshold limit (14.0 Kw/ft)" was also satisfied. The peak LHGR attained was 9.57 Kw/ft.

This Subtest required a performance of a cont d sequence exchange using the plant procedure RE-TP-009, Rod Sequence Exchange. To demonstrate the use of this procedure at the condtions anticipated during normal plant operations, the test began on the design flow control line. The core power was decreased to 70% of rated power by reducing core flow to the minimum recirculation pump speed of the Master Manual control mode. The core power was further reduced by control rod insertion to flatten the flux. At this point, power was above the low power setpoint (20% power) of the Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS), where it was maintained throughout the rod sequence exchange. The actual rod pattern exchange was performed on a row by row basis until the complete rod pattern had been exchanged. Throughout this procedure the core performance parameters of Maximum Linear Heat Generations Rate (MLHGR), Minimum Critical Power Ratio (MCPR) and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) were monitored using the LPRM's, TIP's or Process Computer program P1 to ensure the Technical Specifications limits of T.S. 3.2.4, 3.2.3, and 3.2.1 respectively were not exceeded. The PCIOMR threshold limit for each fuel node is dependent on the Nodal Average Exposure (NAE) with the initial threshold value for exposures less than 3300 MWD/T of 14.0 KW/FT. Since this value is greater than the Technical Specifications limit of 13.4 KW/FT, the Level 2 Criterion was less restrictive. However, data recorded during this test will be useful in planning for normal operational Rod Sequence Exchange. The specific core flow, control rod movements, control rod exchange movements and core performance parameters monitoring methods are contained in RE-TP-009, based on the actual operating rod pattern and the desired rod pattern in the new sequence.

The basic methodology of performing the sequence exchange follows. After core flow was reduced, rods were inserted to achieve a relatively flat flux distribution. Next the rods are exchanged one row at a time starting from one side of the core and continuing row by row in row sequence until all rods are in the new pattern. The basic rules are:

1. All control rods are inserted at least two notches deeper than either of the adjacent, in-row control rods on each side of the inserted rods. If the "two-notch" position places an inserted control rod less than two notches from an adjacent, adjacent-row control rod, then the inserted control rod is inserted further until it reaches a position at which it satisfies the "two-notch" minimum criteria for all four adjacent control rods.

2. Before a control rod is withdrawn, both adjacent in-row control rods must have been inserted per item 1 above.

3. Following withdrawal of the in-row control rods, the inserted control rods' positions are readjusted so as to match the post-sequence exchange control rod pattern rod density and power distribution with that of the initial pattern.

A step by step procedure for the sequence exchange was prepared prior to execution. During the sequence exchange core thermal power was monitored. In addition, TIP traces were taken prior to and after the row exchange to verify that flux distribution had not changed greatly. Also LPRM readings were monitored to ensure the LHGR limit of 13.4 KW/ft was not exceeded.

The rod sequence exchange was executed smoothly with no major problems. All Acceptance Criteria were satisfied. The worst case values experienced during the exchange were:

MFLPD	.533
MFLCPM	.672
MAPRAT	.534
LHGR	9.57 KW/ft

4.35 (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 61% power and at 75% power and twice during TC-6 at 98% power and 100% 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 61% power and 98.5% indicated core flow, single tap jet pump, double tap jet pump and recirculation elbow nozzle data was taken and input into the General Electric - JRPMP computer code. This calculation showed that the actual total core flow was 111%. Due to this high flow rate, the total core flow instrumentation was not adjusted at this time. The high flow rate was due to a larger measured M-ratio than designed.

At 75% power and 89% indicated core flow, the JRPMP calculation showed the actual core flow to be 102%. The jet pump loop flows were recalibrated such that indicated core flow was in agreement with actual core flow. It was also discovered that the original calibration span of the recirculation flow transmitters was not correct. An extrapolation of the recirculation drive flow which would be expected at 100% core flow, 100% power was made and input into the APRM/RBM flow units. Plots of recirculation drive flow versus core flow along various rod lines was made as power level was increased to monitor actual data versus the predicted values. These plots showed agreement within 2%.

At 100% power, 102% indicated flow with both recirc pumps in individual manual mode at 928 rpm, data was taken using a strip chart recorder to record the jet pump summer inlet and outlet signals in order to obtain a more accurate averaging of the oscillatory jet pump signals. Using this method the JRPMP calculations resulted in an actual core flow of 99.6%. An extrapolation indicated that at 100% flow, the total recirculation drive flow would be 78,800 gpm. Adjustments were made to the total core flow instrumentation and to the APRM/RBM flow units. In addition, the electrical stops on the Bailey positioners were set at 102%.

At 98.5% rated power and 99.7 Mlb/hr indicated core flow, ST 35.1 was performed again to verify the changes made following the calibration at 100% power. At this time, the recirculation system was in Master Manual mode. In this mode, the "B" recirc pump runs at 1% higher speed than the "A" pump. Again, data was taken using a strip chart recorder. The results of the JRPMP calculation showed total core flow to be 98.5% which is within 1% agreement with indicated flow. This calibration also showed recirculation flow extrapolated to 100% power, 100% flow to be 80,200 gpm which was 2.7% lower than previously calculated. The recirculation flow transmitter spans will be adjusted during the next scheduled surveillance.

4.36 (ST36) COOLING WATER SYSTEMS

The overall evaluation of the testing showed that the performance of the Reactor Building Closed Cooling Water (RBCCW), Turbine Building Closed Cooling Water (TBCCW) and Service Water (SW) systems provided adequate cooling capability with the reactor at rated temperature.

The Acceptance Criteria were as follows:

Level 1

NONE

Level 2

1. The Service Water Pump discharge header temperature is less than 95°F.

2. The RBCCW Heat Exchanger RBCCW outlet temperature is maintained at 100±5°F.

3. The TBCCW Heat Exchanger TBCCW outlet temperature is maintained at 100±5°F.

4. The TBCCW Heat Exchanger shall be capable of transferring 2,110,000 BTU/Hr. of heat with a transfer rate of 220 BTU/Hr/Ft^2/°F.

5. The RBCCW Heat Exchanger shall be capable of transferring 18,020,000 BTU/Hr. of heat with a transfer rate of 213.8 BTU/Hr./Ft^2/°F.

ST36.1, Cooling Water Systems Performance, was implemented during Test Conditions 1, 2, 3 and 6. Acceptance Criteria for Service Water discharge temperature, RBCCW heat exchanger RBCCW outlet temperature and TBCCW heat exchanger TBCCW outlet temperature were satisfied during all test implementations. Problems were encountered during all test implementations in meeting the RBCCW and TBCCW heat transfer values and heat transfer rates as dictated by the Acceptance Criteria. These exceptions were resolved following the last test implementation at Test Condition 6 by the cognizant engineering groups when it was determined that the data collected during the best obtainable present temperature and flow conditions indicated that the equipment was performing properly. Extrapolation of the collected data further indicated that adequate heat exchanger capacity existed for both the RBCCW and TBCCW systems to perform properly at all anticipated operating conditions, thus meeting the test objective as set forth in the Final Safety Analysis Report.

4.37 (ST37) GASEOUS RADWASTE SYSTEM

The objectives of this test were to demonstrate that the Gaseous Radwaste System, operating with either the Unit 1 or Common Recombiner, 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. These objectives were satisfied by performing Subtest ST-37.1 - Gaseous Radwaste Data Collection Utilizing Unit 1 Recombiner, ST 37.1 - Gaseous Radwaste Data Collection Utilizing the Common Recombiner, and Subtest ST-37.2 - Containment Inerting.

The following Acceptance Criteria were verified during this test:

Level 1

1. The release of radioactive gaseous and particulate effluents must not exceed the limits specified in the site Technical Specifications.(1)

2. There shall be no less than 8,000 lbs/hr of dilution steam flow when the steam jet air ejectors are pumping.(1)

Level 2

1. The system flow, pressure, temperature and dew point shall comply with design specifications.(1)

2. The catalytic recombiner, the hydrogen analyzer, the activated carbon beds and the filters shall be performing their required functions.(1)

3. 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 or equal to 4%.(2)

   (1) Applicable to ST-37.1

   (2) Applicable to ST-37.2

ST37.1 Gaseous Radwaste Data Collection (Unit 1 Recombiner)

ST37.1 was performed during Initial Heatup at 2% power, Test Condition (TC)1 at 20% power, TC-3 at 60% power, TC-5 at 69% power and TC-6 at 97% power. All level 1 Acceptance Criteria were met. During the performance of this test, the following problems were encountered:

The hydrogen analyzers were frequently found to be inoperable. This problem is apparently due to excessive moisture entering the sampling system upon system shutdown. A plant modification will be made to automatically isolate the analyzers following a scram or offgas isolation.

High differential pressure on the inlet HEPA filters was traced to a water saturation of the filter. An engineering evaluation determined that the HEPA filter was not needed so it was removed from the system.

The range of the offgas flow meter was found to be smaller than the value of offgas flow actually encountered. A plant modification will be made to install meters with a larger range so that offgas flow may be read directly.

Various instrumentation problems were encountered which required repair and/or recalibration of the instrument.

High dewpoint readings in the Offgas System were originally thought to be inaccurate and/or uncalibrated moisture probes. It has since been determined that the instrumentation is correct and that the mist eliminators are not performing their intended function. Moisture is being carried over from the chiller to the mist eliminator and to the charcoal beds. A plant modification to replace the mist eliminators will be made as soon as practicable. Continued operation until the modification can be implemented is being justified by the low effluent radiation levels encountered thus far with the system in operation.

The ability to maintain acceptable offgas system guard bed inlet dewpoint temperatures will be demonstrated after design modifications are made and prior to Cycle 2 Startup.

ST 37.1 Gaseous Radwaste Data Collection (Common Recombiner)

This test for the Common Recombiner in service was performed at 99.7% rated reactor thermal power. Offgas system operational parameters were monitored and recorded to verify proper system operation. Gaseous grab samples were collected for radioactivity analysis. Both Level 1 Acceptance Criteria were satisfied along with Level 2 Acceptance Criterion No. 2. Problems were encountered similar to those during the performance of ST 37.1 on the Unit 1 Recombiner. They were:

Various instrumentation problems were encountered which will require modification, repair and/or recalibration of the instruments.

The mist eliminators are not performing their intended function. Moisture is being carried over from the chiller to the mist eliminator and to the charcoal beds. A plant modification to replace the mist eliminators will be made as soon as practicable. Continued operation until the modification can be implemented is being justified by the low effluent radiation levels encountered thus far with the system in operation.

The ability to maintain acceptable offgas system guard bed inlet dewpoint temperatures will be demonstrated after design modifications are made and prior to Cycle 2 Startup.

ST 37.2 Containment Inerting

ST 37.2 was performed at the end of a maintenance outage following the last of the 100% power trip tests. The containment was inerted in 13.5 hours to an oxygen concentration of less than 3.8% in the drywell and less than 2.8% in the suppression chamber, thus satisfying the Acceptance Criterion.

4.38 (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:

Level 1

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.

Level 2

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.

2. The maximum measured loads, displacements, accelerations and/or pressures for Balance-of-Plant systems, identified in FSAR Table 3.9-33, shall be less than or equal to the design maximum expected values.

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 dynamic vibratory response acceptability. The Acceptance Criteria in this case were as follows:

Level 2

1. The vibratory response of Balance-of-Plant non-safety related piping, per FSAR Table 3.9-33, shall be judged acceptable by the 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 Balance-of-Plant non-related piping, per FSAR Table 3.9-33.

Piping dynamic transient testing was performed for the piping systems or portions of piping systems listed below:

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

(2) Main Steam Safety Relief Valve discharge piping and associated Main Steam piping inside primary containment.

(3) Reactor Recirculation system piping.

(4) High Pressure Coolant Injection system steam supply piping.

(5) Feedwater system discharge piping inside and outside primary containment.

Testing for (1) listed above was performed in conjunction with the following planned transient tests:

(I) Generator Load Reject Within Bypass Capacity @ 25% reactor thermal power (Test Condition 2).

(II) Loss of Turbine-Generator and Offsite Power @ 38% reactor thermal power (Test Condition 2).

(III) Turbine Trip @ 75% reactor thermal power (Test Condition 3).

(IV) High Power Generator Load Rejection @ 100% reactor thermal power (Test Condition 6).

(V) MSIV Full Isolation @ 100% reactor thermal power (Test Condition 6).

Testing for (2) listed above was performed in conjunction with Relief Valve Rated Pressure Testing @ 45% reactor thermal power (Test Condition 2).

Testing for (3) listed above was performed in conjunction with the following planned transient tests:

(I) Recirculation System One Pump Trip and subsequent restart @ 70% reactor thermal power (Test Condition 3).

(II) Recirculation Pump Trip (RPT) of Two Pumps and subsequent restarts @ 76% reactor thermal power (Test Condition 3).

(III) Recirculation System One Pump Trip and subsequent restart @ 98% reactor thermal power (Test Condition 6).

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

Testing for (4) listed above was performed with a High Pressure Coolant Injection system turbine trip from its rated flow rate of 5000 gpm. This occurred at approximately 4% reactor thermal power with the reactor pressure vessel at rated pressure during the Initial Heatup Test Phase.

Testing for (5) listed above, 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 73% 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 failure of a few of the remotely mounted sensors on Main Steam inside containment which were repaired or replaced at the next opportunity. In each case where a failed sensor existed, 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 ST 39 proved that the piping design met all test objectives as set forth in the FSAR.