THE DETROIT EDISON COMPANY

FERMI 2
NUCLEAR POWER PLANT

INTERIM STARTUP TEST REPORT

SUPPLEMENT NO. 9

September 10, 1988

INTERIM STARTUP TEST REPORT
Supplement #9 Dated 09/10/88

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FOREWARD

This Supplementary Startup Test Report includes testing performed since Supplement #7 dated March 10, 1988 was transmitted to the NRC via NRC-88-0038 dated March 20, 1988. Letter NRC-88-0136 dated June 20, 1988 was transmitted to the NRC to serve as Supplement #8. There was no update at that time since the plant had been in a Local Leak Rate Testing Outage from February 1988 through May 1988, and as of that letter, plant power operation had not yet reached Test Condition Six power/flow requirements.

Since the plant's return to power operation, thirteen of the required twenty six Test Condition Six tests have been completed as well as all four of the required Test Condition Four tests. In addition, the balance of the feedwater system tuneup and dynamic response testing not possible during Test Condition Three has been successfully completed. This discrepant testing was identified as Inspection Report Open Item 341-88003-01 contained in Inspection Report No. 50-341/87046 (DRP).

In this supplement we are transmitting an updated copy of the entire test report. Revision bars have been added to show where changes have been made, except for changes which are only cosmetic in nature or which only involve renumbering sections or pages.

The results sections of this report will be filled in as the tests are completed in the future.

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INTERIM STARTUP TEST REPORT

1.0 Introduction

1.1 Purpose

The purpose of this Interim Startup Test Report and its associated supplements is to provide a summary of the test results obtained in startup testing completed from initial fuel load to the present. This report of plant startup and power ascension testing is submitted as required per Technical Specification 6.9.1.1. This interim report plus its supplements cover all testing applicable to the test conditions completed as described in UFSAR Subsection 14.1.4.8. Supplements will be issued as the remaining testing is completed, at the intervals specified per Technical Specification 6.9.1.3.

Included in this report are descriptions of the measured values of the operating conditions and characteristics obtained during the test program and any corrective actions that were required to obtain satisfactory operation.

1.2 Test Report Format

Sections 1.0 and 2.0 of this report provide general information about the Fermi 2 plant and the testing program. Section 3.0 provides a basic description of the testing we have performed along with a summary of the results and analysis obtained from each test. Each test summary is divided into three subsections covering the purpose, test criteria, and results of each test.

1.3 Plant Description

The Fermi 2 Nuclear Power Plant is located in Frenchtown Township, Monroe County, Michigan. The Nuclear Steam Supply System consists of a General Electric BWR 4 nuclear reactor rated at 3292 MWt, coupled to an English Electric Turbine/Generator rated at 1100 MWe, constructed in a Mark I containment with a toroidal suppression pool.

This plant is owned and operated by the Detroit Edison Company and the Wolverine Power Cooperative, Incorporated.

1.4 Startup Test Program Description

The Startup Test Phase began with preparation for fuel loading and will extend to the completion of the warranty demonstration. This phase is subdivided into four parts:

1. Fuel Loading and Open Vessel Tests
2. Initial heatup
3. Power tests
4. Warranty demonstration

The Startup Test Phase and all associated testing activities adhere closely to NRC Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."

The overall objectives of the Startup Test Phase are as follows:

1. To achieve an orderly and safe initial core loading

2. To perform all testing and measurements necessary to determine that the approach to initial criticality and the subsequent power ascension are accomplished safely and orderly

3. To conduct low-power physics tests sufficient to ensure that physics design parameters have been met

4. To conduct initial heatup and hot functional testing so that hot integrated operation of specified systems are shown to meet design specifications

5. To conduct an orderly and safe Power Ascension Program, with requisite physics and system testing, to ensure that when operating at power, the plant meets design intent

6. To conduct a successful warranty demonstration program

Tests conducted during the Startup Test Phase consist of Major Plant Transients and Stability Tests. The remainder of tests are directed toward demonstrating correct performance of the nuclear boiler and numerous auxiliary plant systems while at power. Certain tests may be identified with more than one part of the Startup Test Phase. Figure 1-1 shows a general view of the Startup Test Phase Program and should be considered in conjunction with

Figure 1-2 which shows, graphically, the various test areas as a function of core thermal power and flow. Note that Figure 1-1 has been modified to reflect certain tests which we presently intend to delete from the Startup Test Program, as discussed further in Reference 1.5.3.

For a more comprehensive description of the testing program refer to Reference 1.5.2.

1.5 References

The following is a list of documents that provide supplementary information of the Fermi 2 Startup Test Phase Program:

1. Fermi 2 Technical Specifications, Section 6.

2. Updated Final Safety Analysis Report, Fermi 2 Nuclear Power Plant, Section 14.

3. Letter VP-86-0141, "Startup Test Program Changes", dated October 17, 1986, from Frank E. Agosti to James G. Keppler.

4. Letter NRC-87-0179, "Initial Test Program Changes" dated September 30, 1987, from B. R. Sylvia to U. S. Nuclear Regulatory Commission, Washington, D.C.

5. Letter NRC-88-0181, "Change in Startup Test Program", dated July 14, 1988, from B. R. Sylvia to U. S. Nuclear Regulatory Commission, Washington, D.C.

FIGURE 1-1

STARTUP TEST PROGRAM

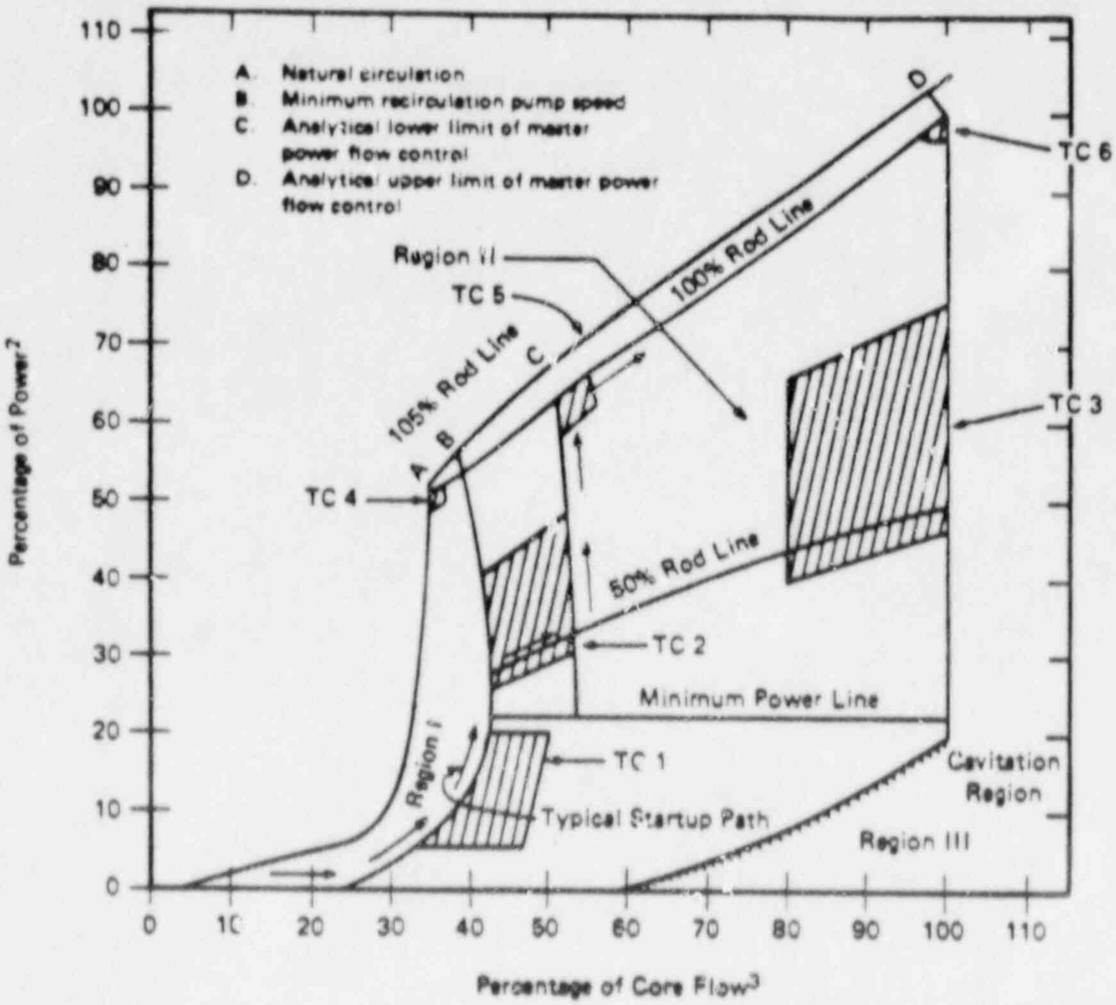
Test No.	Test Name	Open Vessel or Cold Test	Heatup	Test Conditions[a] 1	2	3	4[b]	5	6	Warranty
1	Chemical and Radiochemical	X	X	X		X		X	X	
2	Radiation Measurements	X	X		X	X			X	
3	Fuel Loading	X								
4	Full Core Shutdown Margin		X							
5	CRD	X	X	X					X	
6	SRM Performance and Control Rod Sequence		X	X						
7	Water Level Measurements		X							
8	IRM Performance		X	X						
9	LPRM Calibration		X	X		X			X	
10	APRM Calibration		X	X	X	X		X	X	X
11	Process Computer	X		X[c]		X		X	X	
12	Reactor Core Isolation Cooling System		X	X	M					
13	High Pressure Coolant Injection System		X			M				
14	Selected Process Temperatures		X				X		X	
15	System Expansion	X	X						X	
16	(Deleted)									
17	Core Performance			X	X	X	X	X	X	X
18	(Deleted)									
19	(Deleted)									
20	Pressure Regulator - Setpoint Changes			M,BP	M	M		M	M	
	- Backup Regulator			M,BP	M	M			M	
21	Feedwater System - Feedwater Pump Trip								M(SP)	
	- Water Level Setpoint Changes		X	X	M	M		M	M	
	- Heating Loss							M[d]		
	- Maximum Runout Capability							M[d]		
22	Turbine Valve Surveillances							M[d],SP	M[e,f],SP	
23	MSIVs - Each Valve		X[g]	M[h],SP		X[g]			X[g]	
	- Full Isolation								M,SD[k]	
24	Relief Valves		X		M[i]					
25	Turbine Stop Valve and Control Valve Fast Closure Trips				M,SP[j]				M,SD[k,l]	
26	Shutdown from Outside Control Room			(SD)X[l]					X[m]	
27	Flow Control				M	M		M[d]		
28	Recirculation System - Trip One Pump								M(SP)	
	- System Performance				X	X	X		X	
	- Noncavitation Verification					M				
29	Loss of T-G Offsite Power				M,SD[l]					
30	Vibration Measurements		X		X	X			X	
31	Recirculation System Flow Calibration					X			X	
32	Reactor Water Cleanup System		X				X			
33	Residual Heat Removal System		X						X	
34	Piping Systems Dynamic Response		X		X	X			X	

Key: M - manual flow control mode; X - test independent of flow control; SP - scram possibility; SD - scram definite; BP - bypass valve response.

[a]See Figure 14.1-2 for test conditions region map.
[b]Testing at natural circulation on 100 percent load line can be done anytime following Test Condition 3.
[c]Between Test Conditions 1 and 3.
[d]Between Test Conditions 5 and 6.
[e]Determine maximum power without scram.
[f]Future maximum power test point.
[g]10 percent slow closure-slow mode.
[h]Full closure-fast mode.
[i]Between Test Conditions 2 and 3.
[j]Within bypass valve capability.
[k]If an inadvertent full MSIV isolation or turbine/generator trip occurs at between 70 percent and 100 percent of core thermal power, credit may be taken for this test if supporting analysis shows that the results can be extrapolated to the higher power condition.
[l]Perform Test 5, timing four slowest control rods in conjunction with these scrams.
[m]RHR shutdown cooling mode demonstration.

FIGURE 1-2

APPROXIMATE POWER FLOW MAP
SHOWING STARTUP TEST CONDITIONS



Notes:

1. See Figure 1-1 for startup test titles.
2. Power in percentage of rated Thermal Power 3292 MWT.
3. Core flow in percentage of rated core recirculation flow. 100.0×10^6 lb/hr.
4. TC = test condition.

2.0 General Test Program Information

2.1 Chronology of Major Events

	Date
Received (5%) Facility Operating License No. NPF-33	03/20/85
Started Fuel Loading	03/20/85
Completed Fuel Loading	04/04/85
Completed Open Vessel Testing	06/01/85
Initial Criticality	06/21/85
Received (Full Power) Facility Operating License NPF-43	07/15/85
Completed Initial Turbine Roll	09/26/85
Bypass Line Replacement/ Environmental Qualification Equipment Upgrade Outage Begins	10/10/85
Neutron Source Changeout Complete	05/12/86
Outage Ends	07/24/86
Reactor Restarted	08/04/86
Completed Test Condition Heatup	09/03/86
Entered Test Condition One	09/10/86
Initial Synchronization to Grid	09/21/86
Condenser Repair Outage Begins	11/08/86
Reactor Restarted	12/18/86
Completed Test Condition One	01/07/87
Main Steam Line Instrument Tap Repair Outage Begins	01/09/87
Reactor Restarted	01/24/87
Entered Test Condition Two	02/24/87
Completed Test Condition Two with Loss of Offsite Power Test	03/16/87

Chronology of Major Events (Continued)

	Date
MSR Refit Outage Begins	03/16/87
Reactor Restarted	04/03/87
Main Steam Line Tap Repair Outage Begins	04/12/87
Reactor Restarted	05/10/87
South RFPT Damaged	05/13/87
Reactor Restarted	05/14/87
Commenced Test Condition Three Testing	06/10/87
Completed Core Flow Calibration at 50% Power	06/14/87
Outage to Repair Reactor Recirc MG Set "B"	06/25/87
Reactor Restarted	06/28/87
South Reactor Feedpump Returned to Service	07/02/87
Outage to Repair Feedwater Check Valve Begins	07/31/87
Reactor Restarted	10/09/87
Commenced Test Condition Three HPCI Test Sequence	10/14/87
Completed Test Condition Three HPCI Test Sequence	10/24/87
NRC Authorization to Exceed 50% Power Received	12/05/87
Resumed Test Condition Three Testing at 50-75% Power	12/09/87
Completed Core Flow Calibration at 71% Power	12/17/87
Completed Test Condition Three Testing	12/26/87

Chronology of Major Events (Continued)

	Date
Commenced Test Condition Five Testing	12/29/87
Completed Test Conditi n Five Testing	12/30/87
Outage to Investigate Reactor Feedpump Control problems	12/31/87
Reactor Restarted	01/08/88
NRC Authorization to Exceed 75% Power Received	01/15/88
Started 100 Hour Commercial Operation Demonstration Run at > 90% Net Generation	01/19/88
Completed Commercial Operation Demonstration Run	01/23/88
Completed Core Flow Calibration at 95% Power	01/28/88
Scheduled LLRT Outage Begins	02/27/88
Reactor Restarted	05/05/88
Reactor Manually Scrammed Due to Loss of 120 KV Bus 101	05/07/88
Reactor Restarted	05/08/88
Reactor Scram During HPCI Surveillance	05/08/88
Reactor Restarted	05/09/88
Reactor Scram Due to High RPV Pressure After Failure of Turbine Bypass Valves	05/10/88
Reactor Restarted	05/12/88
Commenced Test Condition Six Testing	07/11/88
Completed Core Flow Calibration at 97% Power	07/17/88

Chronology of Major Events (Continued)

	Date
Outage to Repair Unidentified Drywell Leakage 88-02	07/23/88
Reactor Restarted	08/06/88
Reactor Scram Due to False Turbine Vibration Signal	08/13/88
Reactor Restarted	08/14/88
Completed Feedwater Tuneup/ Dynamic Response Testing	08/16/88
Performed One Recirculation Pump Trip Test From Test Condition Six	08/20/88
Commenced and Completed Test Condition Four Testing	08/20/88
Plant Shutdown to Repair "B" Reactor Recirculation Pump Discharge Valve	08/21/88
Reactor Restarted	08/23/88
Plant Shutdown to Repair "B" Reactor Recirculation Pump Discharge Valve	08/29/88

2.2 Matrix of Test Completion Dates

Test No	Test Title	Pre-Fuel Load	Open Vessel	Heatup	TC One	TC Two	TC Three	TC Four	TC Five	TC Six
1	Chemical and Radiochemical	01/14/85	>>>	07/16/85	10/27/86	>>>	07/06/87	>>>	*12/21/87	$$$
2	Radiation measurements	01/26/85	04/19/85	07/14/85	10/17/86	03/08/87	12/17/87	>>>	>>>	07/13/88
3	Fuel Loading	>>>	04/04/85	>>>	>>>	>>>	>>>	>>>	>>>	>>>
4	Full Core Shutdown Margin	>>>	>>>	04/10/85	>>>	>>>	>>>	>>>	>>>	>>>
5	CRD	>>>	04/05/85	09/16/85	10/23/86	>>>	>>>	>>>	>>>	$$$
6	SRM Performance and Control Rod Sequence	>>>	>>>	08/25/85	10/05/86	>>>	>>>	>>>	>>>	>>>
7	Water Level Measurements	>>>	>>>	08/30/85	>>>	>>>	>>>	>>>	>>>	>>>
8	IRM Performance	>>>	>>>	06/04/86	01/07/87	>>>	>>>	>>>	>>>	>>>
9	LPRM Calibration	>>>	>>>	09/24/85	01/03/87	>>>	06/22/87	>>>	>>>	07/11/88
10	APRM Calibration	>>>	>>>	08/01/85	01/04/87	03/10/87	06/23/87	>>>	12/29/87	07/12/88
11	Process Computer	05/30/85	>>>	>>>	01/02/87	02/05/87	12/26/87	>>>	**11/12/87	07/12/88
12	RCIC	>>>	>>>	09/01/86	10/15/86	11/03/86	>>>	>>>	>>>	>>>
13	HPCI	>>>	>>>	09/01/86	12/29/86	>>>	10/24/87	>>>	>>>	05/06/88
14	Selected Process Temperatures	>>>	>>>	09/01/86	>>>	>>>	>>>	08/20/88	>>>	08/20/88
15	System Expansion	>>>	06/12/85	09/05/86	1[illegible]/04/86	>>>	>>>	>>>	>>>	07/11/88
16	Core Power Distribution (Deleted)	>>>	>>>	>>>	>>>	>>>	>>>	>>>	>>>	>>>
17	Core Performance	>>>	>>>	>>>	01/04/87	03/10/87	06/23/87	08/20/88	12/29/87	07/19/88
19	Core Power Void Mode Response (Deleted)	>>>	>>>	>>>	>>>	>>>	>>>	>>>	>>>	>>>
20	Pressure Regulator	>>>	>>>	>>>	10/21/86	03/10/87	12/18/87	>>>	12/29/87	$$$
21	Feedwater System	>>>	>>>	07/09/85	10/21/86	07/04/87	12/25/87	>>>	12/29/87	08/18/88
22	Turbine Valve Surveillance	>>>	>>>	>>>	>>>	>>>	>>>	>>>	12/30/87	08/19/88
23	MSIV	>>>	>>>	07/12/85	10/08/86	>>>	12/17/87	>>>	>>>	$$$
24	Relief Valves	>>>	>>>	07/03/85	>>>	03/11/87	>>>	>>>	>>>	>>>

$$$ = Testing to be performed in this test condition, but not yet completed.

>>> = No testing necessary for this test condition.

* = Test actually performed in TC-3 @ 66-71% power.

** = Test actually performed in TC-3 @ 49% power.

2.2 **Matrix of Test Completion Dates** (Continued)

Test No	Test Title	Pre-Fuel Load	Open Vessel	Heatup	TC One	TC Two	TC Three	TC Four	TC Five	TC Six
25	Turbine Stop Valve and Control Valve Fast Closure	>>>	>>>	>>>	>>>	03/16/87	>>>	>>>	>>>	***
26	Shutdown from Outside Control Room	>>>	>>>	>>>	10/23/86	>>>	>>>	>>>	>>>	$$$
27	Flow Control	>>>	>>>	>>>	>>>	03/08/87	12/26/87	>>>	>>>	$$$
28	Recirculation System	>>>	>>>	>>>	>>>	03/[illegible]/87	06/15/87	08/20/88	>>>	$$$
29	Loss of Offsite Power	>>>	>>>	>>>	>>>	03/16/87	>>>	>>>	>>>	>>>
30	Vibration Measurements	>>>	>>>	07/03/85	10/04/86	03/07/87	12/18/87	>>>	>>>	$$$
31	Recirc. System Flow Calibration	>>>	>>>	>>>	>>>	>>>	06/14/87*	>>>	>>>	01/28/88*
32	Reactor Water Cleanup System	>>>	>>>	07/14/85	>>>	>>>	>>>	08/20/86	>>>	>>>
33	Residual Heat Removal System	>>>	>>>	08/31/85	>>>	>>>	>>>	>>>	>>>	$$$
34	Piping System Dynamic Response	>>>	>>>	09/24/85	>>>	03/[illegible]	10/14/87	>>>	>>>	$$$

$$$ = Testing to be performed in this test condition, but not yet completed.

>>> = No testing necessary for this test condition.

* = Additional core flow calibrations performed on 12/17/87 and 07/17/88.

*** = In accordance with Reference Numbers [illegible] and 5 in Section 1.5 of this report, we have taken credit for an inadvertent turbine/generator trip from 74.8% CTP on 12/31/87.

3.0 Test Results Summary

3.1 Chemical and Radiochemical

3.1.1 Purpose

The principal purposes of this test are to collect information on the chemistry and radiochemistry of the Reactor Coolant and Support Systems, and to determine that the sampling equipment, procedures and analytic techniques are adequate to ensure specifications and process requirements are met.

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

3.1.2 Criteria

Level 1

Chemical factors defined in the Technical Specifications and Fuel Warrant must be maintained within the limits specified. Water quality must be known at all times and remain within the guidelines of the Water Quality Specifications.

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

Level 2

None

3.1.3 Results

Prior to loading fuel, appropriate chemistry data was taken. All data remained within criteria levels except for feedwater conductivity and feedwater copper concentration. These values could have been elevated due to no condenser vacuum, minimum Feedwater System flow, low sample flow rates and the normally expected higher corrosion product levels during initial plant systems operation.

During heatup test condition, these values were within acceptable limits. See Figure 3.1 for specific information on pre-fuel load chemistry data.

During the heatup test condition, all chemistry data taken fell within applicable limits except for Control Rod Drive (CRD) dissolved oxygen levels. These levels are expected to decrease during further test conditions with greater steam flow and the steam jet air ejectors in service which will more effectively purge gases from the condenser. Refer to Figure 3.1 for heatup chemistry data.

The Test Condition One data in general remained within acceptance criteria limits. Reactor water chemistry and radiochemistry measurements were made at a time when plant conditions were fairly stable. Reactor power was at 17%, the turbine was rolling but with no electrical output load. Analysis of the results showed the coolant to be well within the Technical Specification limits on all parameters. Radiochemistry analyses of the coolant showed activity levels and isotopes present to be normal for this power level and core exposure. The Dose Equivalent I-131 result was far below the Technical Specification limit of 0.2 uCi/gm. In Test Condition One, the steam jet air ejectors were in service resulting in low condensate, Condensate demineralizer effluent, and CRD dissolved oxygen levels. The high CRD dissolved oxygen level which was of concern during the heatup test condition is no longer considered to be a problem.

It should be noted that Reactor Conductivity varied considerably during the Test Condition One period. Conductivity has, on several occasions, even exceeded the Technical Specification values of 1.0 umho/cm for several hours. It was determined that the increase in conductivity was related to placing the Generator on line and increasing Generator load. One possible explanation was that operation of the Generator was causing the paint that was previously used to coat the internals of the Moisture Separator Reheater (MSR) and the Main Turbine to be carried into the condenser hotwell, thus causing the increase in Reactor conductivity. Another contributing factor was felt to be the Krylon coating that was previously used as a preservative coating for the turbine blades, which was being worn off the blades and into the condenser. Further investigation discounted the krylon coating (due to it's chemical makeup) as a cause of the conductivity increase. This situation seems to be improving as the plant continues to operate for longer periods at increasing power levels. Efforts were made during the condenser outage to remove paint from accessible areas in the MSRs and LP turbine exhausts. Mechanical cleaning by wire brushing and vacuuming was performed on the MSR's interior shell surface and hydro-lasing of the three LP turbine exhaust extensions to the condenser was performed.

Both Condensate Demineralizer Effluent and Feedwater dissolved oxygen levels at Test Condition One were less than 10 ppb, which are outside of the limits of $20 \leq O_2 \leq 200$ ppb. The 20 ppb minimum oxygen concentration has been recommended to establish and maintain a protective magnetite film on the inner surfaces of the carbon steel piping and equipment of these systems. The problem of low condensate/feedwater dissolved oxygen has occurred during the startup of other operating plants. The resolution at that time was to simply continue to monitor these parameters at higher power levels to see if the levels would increase with power. If dissolved oxygen levels do not increase to greater than 20 ppb by 100% power, it may become necessary to inject oxygen into the feedwater system. An assessment would first be made as to the corrosivity of the water to the carbon steel piping to determine if this is necessary.

All gaseous and liquid effluent samples obtained during performance of this procedure were within the license limitations. Various radioactive gaseous effluents were analyzed during Test Condition One. Grab samples were taken in an attempt to correlate analysis results with actual monitor readings. However, the activity levels being seen at the off-gas and ventilation sample points were still too low to provide meaningful data. Only one noble gas was detected, at a level which was just above the minimum detection limit. The off-gas monitor readings were also still quite low and variable. Low off-gas activity values are normal and expected at this power level and core exposure.

A measurement of radiolytic gas in steam was also made at Test Condition One. Analysis results were below the 0.06 cfm/MWt limit. Radiolytic gas is the hydrogen and oxygen formed in the reactor by radiation induced breakdown of water molecules. Values higher than 0.06 cfm/MWt could exceed the capacity of the off-gas system recombiners.

See Figure 3.1 for more detail regarding the chemistry data taken during Test Condition One.

The Test Condition Three data, in general, remained within acceptance criteria limits and satisfied Technical Specification requirements.

Reactor water chemistry and radiochemistry measurements were made at a time when plant conditions had been fairly stable for 48 hours. During this time period, the plant power level was held between 43 and 45 percent. Some of the chemistry results, while still acceptable, indicated problems with the primary system and especially with the reactor coolant chemistry. Approximately three hours prior to taking samples for this test, Condensate Filter Demineralizer (CFD) "B" was removed from service and CFD "F" was placed into service. Reactor water conductivity spiked, from 0.58 uS/cm up to 0.82 uS/cm. At the same time, sulfate levels increased in the coolant and the pH dropped. Since all of this occurred in the same time period, the conclusion was made that there was a resin intrusion and that the CFDs were the source of the resin. Numerous other chemistry excursions have occurred which support this conclusion.

Following those occurrences, progress was made in reducing and eliminating the source of the resin intrusions. The procedure for precoating the CFDs was changed to allow for a fiber underlay on the vessel septa. This inert underlay was used, on an interim basis, to reduce the amount of powdered resin which was escaping. Since that time, elements (septa) of a new design have been installed for each of the seven vessels. The new design septa utilizes a porous metal membrane which has a very small pore size, when compared to the old design wire screen mesh elements and the precoating procedure has been changed to eliminate the use of the inert fiber underlay. No further evidence of resin intrusion has been noted since the new septa have been installed and as a result, reactor coolant chemistry has shown significant improvement.

The higher than desired levels of sulfate in the reactor vessel were utilized to complete a reactor water cleanup (RWCU) test which could not be accomplished in TC1. This test was to determine the chloride removal rate of the demineralizers. A test procedure revision was made to allow other anions to be used as well as chloride, as they would have similar RWCU removal rates. The RWCU successfully demonstrated a removal capability of greater than 90% for sulfates.

Condensate and feedwater chemistry were also examined. All values obtained, with the exception of dissolved oxygen, were within the water quality specfications limits. Again, however, some of the results reflected the problems which were occurring in the primary system. Condensate conductivity was higher than would be normal, and this may have been attributable to carry-over of volatile resin breakdown products in the steam. Feedwater conductivity values were also somewhat higher than normal, and again this may have been partially the result of resin breakdown. Resin escaping from the condensate filter demineralizers would be exposed to high temperatures in the feedwater system, which can begin the process of degradation. The insoluble iron and total metals found in the condensate, condensate demineralizer effluent, feedwater and reactor water were within the specification limits and at levels expected for a plant startup.

The two exceptions noted during Test Condition Three testing are identical to two from Test Condition One. All are for low dissolved oxygen (< 10 ppb) in the condensate demineralizer effluent (CDE) and in the final feedwater (FFW). A minimum level of dissolved oxygen ($20 \geq O_2 \leq 200$ ppb) is desired in the feedwater system to promote and maintain a passive corrosion layer on the pipe walls. Low levels of dissolved oxygen can lead to excessive corrosion and higher corrosion products in the feedwater samples. Current corrosion product levels cannot yet be conclusively attributed to the low dissolved oxygen, but if the dissolved oxygen level does not increase with increases in power, it may be necessary to inject oxygen into the feedwater system. These parameters of dissolved oxygen and corrosion products will continue to be monitored closely in future test conditions.

All gaseous and liquid effluent samples obtained during the performance of this procedure were within the license limitations.

Various radioactive gaseous effluents were analyzed during TC3. Grab samples were taken in an attempt to correlate analysis results with actual monitor readings. However, the activity levels seen at the off-gas and ventilation points are still too low to provide meaningful data. The sum of six noble gasses is plotted against the off-gas monitor readings, but the plot has little meaning since present off-gas activity is too low to affect the monitor. However, the activity is sufficient to perform an analysis of the off-gas radionuclides and reactor water iodines. By normalizing the nuclide activities with respect to release rate, fission yield, and half-life, and then plotting the data, it was determined that the plant has a "recoil" pattern of release. Such a pattern indicates that there is no failed fuel.

A measurement of radiolytic gas in steam was made. Analysis results were below the 0.06 cfm/MWt limit. Radiolytic gas is the hydrogen and oxygen produced in the reactor by radiation induced breakdown of water molecules. It is a normal expected process, but values higher than the limit could cause the capacity of the off-gas system recombiners to be exceeded.

See Figure 3.1 for more detail regarding the chemistry data taken during Test Condition Three. Also note that identifying marks have been added to several data points in Test Conditions One and Three to note that sample dates are other than that of the main column heading. Reactor power conditions were, however, approximately the same as during the balance of sampling.

The Test Condition Five chemistry data was actually recorded during Test Condition Three as the plant achieved the required power level for the test (65-80% CTP). The requirements for sampling are based upon reactor power level only. In general, the data remained within acceptance criteria limits.

Reactor water chemistry and radiochemistry measurements were made immediately after raising power to greater than 65 percent. The test results showed that the reactor coolant chemistry was satisfactory at the time the samples were taken. Later data points taken after forward pumped drains (FPD) had been placed in service showed an increase in conductivity to about 0.38 uS/cm and a rise in sulfate levels to about 80 ppb. These relatively high values can be attributed to the FPD water being sent directly to the reactor. This was essentially the first time that the FPD had been utilized on a continuous basis.

The conductivity and sulfate values peaked at the levels previously mentioned and then began a slow decline. The source of the sulfate contamination is from the FPD system piping and equipment upstream of it. It is known that the internal surfaces of much of the piping and equipment in the plant had been painted with protective coatings. This coating and the impurities it contains is released with each increase in power, temperature and system flow. In the present plant condition the FPD water is no longer cycled back to the hotwell and filtered through the condensate filter demineralizers. Rather, it is pumped, untreated, into the reactor where the impurities are concentrated to the levels observed.

With the exception of this first period of FPD in service, the reactor coolant chemistry has been maintained at a very reasonable level as compared to previous test conditions. Prior to the startup in October 1987, the plant was in a several month outage. In that outage, all of the condensate filter demineralizer elements were replaced with a new design element (septa), as previously discussed. The new elements have prevented the resin intrusions which have occurred in the past, and which caused reactor coolant chemistry problems. Even with power operations up to 70 percent, conductivity was maintained below 0.15 uS/cm prior to the placing in service of FPD.

Condensate demineralizer influent, effluent, feedwater, and FPD chemistry were also examined as part of the test. All values obtained, with the exception of dissolved oxygen, were within the water quality specifications limits. Condensate demineralizer effluent water was excellent, with conductivity equal to that of theoretically pure water. FPD water and feedwater conductivities, although within specifications, were higher than that required to maintain the reactor coolant conductivity below the water chemistry guidelines value of 0.3 uS/cm.

The insoluble iron and total metals found in the condensate, condensate demineralizer effluent, feedwater, and reactor water were within the specification limits and at levels expected for a plant startup. FPD insoluble iron was elevated, as would be expected for the initial operation of a system.

All gaseous and liquid effluent samples obtained during performance of this procedure were within the license limitations.

Various radioactive gaseous effluents were analyzed during the test. Grab samples were taken in an attempt to correlate analysis results with actual monitor readings but the activity levels being seen at the ventilation points are still too low to provide meaningful data.

A sample of the offgas steam was taken, after the two minute delay pipe but prior to any further treatment. This sample was analyzed for its noble gas activity. The nuclide activities were normalized with respect to release rate, fission yield, and half-life, and then plotted. From this plot, it was determined that the plant has a recoil pattern of offgas release. Such a pattern indicates that there is no failed fuel.

Three test exceptions were taken during the testing. The test exceptions are similar to those from TC-1 and from TC-3. Two are for low dissolved oxygen in the condensate demineralizer effluent (CDE) and in the final feedwater (FFW). In addition, the third test exception documents the low dissolved oxygen observed in the forward pumped drain (FPD) sample. A minimum level of dissolved oxygen ($\geq$ 20 ppb) is desired in all of these systems to promote and maintain a passive corrosion layer on the pipe walls. Low levels of dissolved oxygen can lead to excessive corrosion and higher corrosion products levels in the process stream. Based on data obtained to date, current corrosion product levels cannot yet be conclusively attributed to the low dissolved oxygen.

Since the completion of the above noted testing, the plant has operated at power levels of up to 95%, most notably during the 100 hour Commerical Operation Run in late January of 1988. After the completion of that demonstration run, adjustments to the venting from the number five (5) north and south feedwater heaters were made by plant operations and chemistry personnel to attempt to increase the dissolved oxygen of the forward pumped drains water and, therefore, the final feedwater. This action was not successful in raising the dissolved oxygen in these two streams to acceptable levels, nor has there been any favorable indication that the dissolved oxygen in the CDE will increase with further increases in power and therefore, a design change has been prepared to add an oxygen injection system to the CDE if this proves necessary.

The parameters of dissolved oxygen and corrosion products levels will continue to be monitored closely in Test Condition Six.

See Figure 3.1 for more detail regarding the chemistry data taken between 66%-71% power.

During the Spring 1988 LLRT Outage, major repairs and modifications were made on the MSRs and during that period, while all removable internals were removed, all accessible interior surfaces with evidence of paint coating were thoroughly wire brushed to remove this material.

This has significantly decreased the rate of sulphate leachate entry into the feedwater stream when FPD are in service.

The remaining testing in this section not yet completed is the Test Condition Six steady state data collection and the Reactor Water No Cleanup Test.

Chemistry Data

System	Analysis	Pre Fuel Load	Heatup	Test Condition 1	Test Condition 3	Test Condition 5	Test Condition 6	Limit
	Date	1/15/85	7/15/85	10/17/86	6/11/87	12/19,20,21/87		
	CTP	0	<5%	17%	44%	66-71%		
	MWe	0	0	0	410	725-775		
	Mod-Temp	N/A	540°F	540°F	578°F	530°F		
Reactor Water	Conductivity							
	(umho/cm) @ 25°C	0.43	0.31	0.24	0.67	0.22		<1.0
	Chloride (ppb)	<5	8	<2	3	3		<200
	Turbidity (NTU)	0.18	3.6	0.15	0.15	0.06		See Note 2
	Iodine - 131 (uCi/ml)	N/A	-7.7 E-08	<MDA	<MDA	<MDA		See Note 1
	- 133 (uCi/ml)	N/A	2.0 E-06	P.5 E-6	3.1 E-5	6.2 E-5		See Note 1
	2 hr Gross Activity							
	- Filtrate (cpm/ml)	N/A	6.1 E02	7.86 E02	2.96 E3	2.60 E3		See Note 2
	- Crud (cpm/ml)	N/A	4.0 E01	1.30 E01	1.07 E2	2.65 E2		See Note 2
	7 day Gross Activity							
	- Filtrate (cpm/ml)	N/A	4.7	5.35	3.03 E1	5.78 E1		See Note 2
	- Crud (cpm/ml)	N/A	25	0.54	3.10 E1	2.03 E1		See Note 2
	Silica (ppm)	0.023	0.385	29	62	112		See Note 2
	pH	6.2	6.4	5.8	6.1	6.9		5.6 <ph <8.6
	Iron in Crud (ppb)	N/A	N/A	N/A	0.76	0.398		See Note 2
	Boron (ppb)	<10	N/A	N/P	N/A	N/A		<50.0

NOTE 1: The dose equivalent Iodine-131 shall not exceed a concentration of 0.2 uCi/gm.

NOTE 2: This data is used to support trend analysis.

N/A: Sample not taken and/or equipment not in service.

FIGURE 3.1
(Page 1 of 5)

Chemistry Data

System	Analysis	Pre Fuel Load	Heatup	Test Condition 1	Test Condition 3	Test Condition 5	Test Condition 3	Limit
	Date	1/15/85	7/15/85	10/17/86	6/11/87	12/19,20,21/87		
	CTP	0	<5%	17%	44%	66-71%		
	MWe	0	0	0	410	725-775		
	Mod-Temp	N/A	540°F	540°F	528°F	530°F		
CRD Water	Conductivity (umho/cm) @ 25°C	N/A	0.08	* 0.09	0.059	0.072		<0.1
	Oxygen (ppb)	N/A	500	* 50	20	5		<50
Radioactive Gaseous Effluents	Off-gas Vial Sample Analysis H21-P275A	N/A	Completed	Completed	Completed	Completed		See Note 3
	Off-gas Monitor Reading D11-K601A (mr/hr)	N/A	3.5	20	4.0	4.0		See Note 2
	Off-gas Monitor Reading D11-K601B (mr/hr)	N/A	3.5	2	3.0	3.4		See Note 2
	SGTS Exhaust, Div I D11-P275 (uCi/cc)	N/A	1.7 E-08	4.6 E-08	N/A	2.3 E-07		See Note 4
	SGTS Exhaust, Div II D11-P275 (uCi/cc)	N/A	1.2 E-08	5.0 E-08	N/A	1.9 E-07		See Note 4
	Turbine Bldg D11-P279 (uCi/cc)	N/A	8.0 E-08	3.4 E-08	<MDA	3.3 E-07		See Note 4

NOTE 2: This data is used to support trend analysis.

NOTE 3: Perform Isotopic analysis of an Off-gas sample and attach data. This is for trend analysis.

NOTE 4: Readout of Channel 5 on SPING panel, Noble Gas, Xe-133 equivalent (uCi/cc). This data is for trend analysis. If the SGTS is not in service enter NA.

*: Sample Date 10/21/86.

N/A: Sample not taken and/or equipment not in service.

FIGURE 3.1
(Page 2 of 5)

Chemistry Data

System	Analysis	Pre Fuel Load	Heatup	Test Condition 1	Test Condition 3	Test Condition 5	Test Condition 6	Limit
	Date	1/15/85	7/15/85	10/17/86	6/11/87	12/19,20,21/87		
	CTP	0	<5%	17%	44%	66-71%		
	MWe	0	0	0	410	725-775		
	Mod-Temp	N/A	540°F	540°F	528°F	530°F		
Radioactive Gaseous Effluents (continued)	Reactor Bldg (uCi/cc) D11-P280	N/A	4.05 E-07	6.25 E-07	<MDA	1.2 E-06		See Note 4
	Radwaste Bldg (uCi/cc) D11-P281	N/A	9.1 E-07	8.2 E-07	<MDA	<MDA		See Note 4
	Service Bldg (uCi/cc) D11-P282	N/A	8.1 E-08	6.4 E-08	<MDA	2.1 E-07		See Note 4
	Site Storage Bldg D11-P299 (uCi/cc)	N/A	1.2 E-07	5.2 E-08	<MDA	5.5 E-07		See Note 4
Demin. Effluent	Conductivity (umho/cm) @ 25°C	N/A	0.072	0.059	0.058	0.055		<0.1
	Oxygen (ppb)	N/A	N/A	<10	<10	<10		$20<O_2<200$
	Insoluble Iron (ppb)	N/A	N/A	* 0.098	** 0.59	0.033		See Note 2
	Total Iron (ppb)	N/A	N/A	* 0.73	** 2.78	0.153		See Note 2

NOTE 2: This data is used to support trend analysis.

NOTE 4: Readout of Channel 5 on SPING panel for Noble Gas, Xe-133 equivalent (uCi/cc). This data is for trend analysis. If the SGTS is not in service enter NA.

*: Sample Date 10/20/86-->10/21/86.

**: Sample Date 06/22/87-->06/23/87.

N/A: Sample not taken and/or equipment not in service.

FIGURE 3.1

(Page 3 of 5)

Chemistry Data

System	Analysis	Pre Fuel Load	Heatup	Test Condition 1	Test Condition 3	Test Condition 5	Test Condition 6	Limit
	[illegible]l Condition	Pre Fuel Load	Heatup	Test Condition 1	Test Condition 3	Test Condition 5	Test Condition 6	
	Date	1/15/85	7/15/85	10/17/86	6/11/87	12/19,20,21/87		
	CTP	0	<5%	17%	44%	66-71%		
	MWe	0	0	0	410	725-775		
	Mod-Temp	N/A	540°F	540°F	528°F	530°F		
System	Analysis							Limit
Feedwater	Soluble Iron (ppb)	N/A	N/A	* 1.22	** 0.58	0.43		See Note 2
	Insoluble Iron (ppb)	1.8	1	* 6.16	** 0.35	0.83		See Note 5
	Conductivity(umho/cm) A	0.38	0.08	0.065	0.070	0.067		<0.1
	@ 25°C B	N/A	0.10	0.065	0.089	0.062		<0.1
	Oxygen (ppb)	N/A	N/A	<10	<10	<10		$20<O_2<200$
	Insoluble Copper (ppb)	0.1	0.03	* 0.26	** 0.02	0.028		See Note 5
	Soluble Copper (ppb)	4.4	1	* 0.45	** 0.07	0.088		See Note 5
	Total Metals (ppb)	10	2.33	*12.98	** 1.51	2.13		See Note 5
Condensate	Conductivity (umho/cm) @ 25°C	0.5	0.14	0.16	0.091	0.068		<0.5
	Chloride (ppb)	<5	<2	<2	<2	<2		See Note 2
	Insoluble Iron (ppb)	N/A	3	3.05	24	14.4		See Note 2

NOTE 2: This analysis is used to develop trend data.

NOTE 5: The limit of the solubles and insolubles (Total Metals) is that the total shall be ≤15 ppb of which can contain no more than 2 ppb copper. These limits may be exceeded in Test Condition 1.

*: Sample date 10/20/86-->10/22/86.

**: Sample date 06/21/87-->06/23/87.

N/A: Sample not taken and/or equipment not in service.

FIGURE 3.1
(Page 4 of 5)

Chemistry Data

System	Analysis	Pre Fuel Load	Heatup	Test Condition 1	Test Condition 3	Test Condition 5	Test Condition 6	Limit
	Date	1/15/85	7/15/85	10/17/86	6/11/87	12/19,20,21/87		
	CTP	0	<5%	17%	44%	66-71%		
	MWe	0	0	0	410	725-775		
	Mod-Temp	N/A	540°F	540°F	528°F	530°F		
Heater Drains	Insoluble Iron (ppb)	N/A	N/A	N/A	N/A	9.83		See Note 2
	Soluble Iron (ppb)	N/A	N/A	N/A	N/A	0.37		See Note 2
	Conductivity (umho/cm) @ 25°C B	N/A	N/A	N/A	N/A	0.080		<0.1
	Oxygen (ppb)	N/A	N/A	N/A	N/A	10		$20<O_2<200$
Liquid Radwaste Sample Tank	Principal Gamma Emitter (prior to discharge, uci/ml)	N/A	N/A	N/A	N/A	N/A		See Note 2

NOTE 2: This analysis is used to develop trend data.

N/A: Sample not taken and/or equipment not in service.

FIGURE 3.1
(Page 5 of 5)

3.2 Radiation Measurements

3.2.1 Purpose

The purpose of this test is to determine the background radiation levels in the plant environs for baseline data and activity build-up during power ascension testing to ensure the protection of plant personnel during plant operation.

3.2.2 Criteria

<u>Level 1</u>

The radiation doses of plant origin and the occupancy times of personnel in radiation zones shall be controlled consistent with the guidelines of the standards for protection against radiation outlined in 10CFR20, "Standards for Protection Against Radiation", and NRC General Design Criteria.

<u>Level 2</u>

None

3.2.3 Results

Radiation measurements were taken in the form of process and area radiation monitor data and site surveys. To date, all data taken has been acceptable and personnel radiation protection has been provided in full compliance with the criteria.

See Figures 3.2-1 through 3.2-3 for applicable monitor and survey readings. These Figures reflect the results of this test for all the test conditions for which this data was required.

FIGURE 3.2-1
(Page 1 of 5)

Area Radiation Monitor Sensor Locations

Channel No.	Location (Col.) Floor-Bldg.
1	(F-10) 2nd Flr. Reac. Bldg. (RB) Pers. Air Lock
2	(B-9) 1st Flr. RB Equip. Air Lock
3	(J-13) 2nd Flr. Aux. Bldg. (AB) Access Control
4	(G-10) 2nd Flr. AB Change Area Control
5	(B-13) 3rd Flr. RB CRD Storage and Maintenance Area
6	(G-13) 3rd Flr. AB Main Control Room (CR)
7	(F-9) Sub Base. RB S.E. Corner
8	(B-10) Sub Base. RB S.W. Corner
9	(B-15) Sub Base. RB N.W. Corner
10	(G-17) Sub Base. RB N.E. Corner
11	(G-11) Sub Base. RB HPCI Rm.
12	(F-11) 1st Flr. RB Neut. Mon. Eq. Rm.
13	(F-10) 1st Flr. RB Neut. Mon. Control Panel.
14	(A-11) Sub Base. RB Supp. Pool
15	(F-15) 5th Flr. RB Fuel Stor. Pool
16	(F-15) 4th Flr. RB New Fuel Vault
17	(F-12) 5th Flr. RB Refuel Area Near Reactor
18	(F-13) 5th Flr. RB Refuel Area Near Reactor (High Range)
19	(L-12) 3rd Flr. Turbine Bldg. (TB) Turbine Inlet End
20	(R-10) Base. TB Sump
21	(N-7) 2nd Flr. TB Main Cond. Area
22	(J-4) 1st Flr. TB Decon. Area
23	(M-17) 1st Flr. Rad. Waste Bldg. (RWB) Control Rm.
24	(N-17) Base. RWB Equip. Drain S. Pump
25	(P-16) Base. RWB Floor Drain S. Pump
26	(R-17) 1st Flr. RWB Drum Conveyor Aisle Operating Area
27	Spare
28	(G-11) 4th Flr. AB Vent. Equip. Rm.
29	(B-15) 4th Flr. RB Change Rm.
30	(H-12) RB Basement Air Lock
31	(B-12) 1st Flr. RB Drywell Air Lock Labyrinth
32	(G-13) 1st Flr. AB Near Blowout Pnl.
33	(C-9) 1st Flr. RB South Air Lock
34	(N-2) 2nd Flr. TB Near Off Gas Equip.
35	(R-2) 1st Flr. TB Near S.J.A.E. Area
36	(K-1) 1st Flr. TB S.W. Corner
37	(M-2) 3rd Flr. TB South End
38	(R-14) Base. RWB Scrap Cement Recovery
39	(L-13) 1st Flr. RWB H.P. Lab
40	(P-16) 1st Flr. RWB Receiving Area
41	(S-17) 1st Flr. RWB Bailing Room
42	(N-16) 1st Flr. RWB Filter Demin. Area
43	(S-17) Mezz. RWB Washdown Area
44	(S-12) 1st Flr. Service Bldg. (SB) Mach. Shop.

FIGURE 3.2-1
(Page 2 of 5)

Area Radiation Monitor Sensor Locations

Channel No.	Location (Col.) Floor-Bldg.
*45	1st Flr. Inside Drywell
*46	1st Flr. On Site Stg. Bldg. Control Room
*47	1st Flr. On Site Stg. Bldg. Compactor Room
*48	1st Flr. On Site Stg. Bldg. Truck Unloading Station

*The remote indicator is located on Process Radiation Monitor Panel H11-P884 (Relay Room).

Area Radiation Monitor Data

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88
Channel Number:				Radiation Levels:			
1	0.03	<0.1	0.05	0.04	0.05	0.06	0.07
2	0.02	<0.1	0.02	0.02	0.08	0.06	0.10
3	0.03	<0.1	0.02	0.02	0.03	0.06	0.03
4	0.02	<0.1	0.02	0.02	0.02	0.03	0.04
5	0.08	<0.1	0.08	0.07	0.08	0.08	0.10
6	0.03	<0.1	0.04	0.02	0.03	0.02	0.03
7	0.2	0.2	0.2	0.2	0.02	0.1	1.0
8	0.2	0.4	0.4	0.3	0.04	0.3	1.0
9	0.3	0.3	0.3	0.3	0.04	0.2	1.0
10	0.2	0.2	0.2	0.2	0.02	0.1	0.3
11	0.3	0.3	0.3	0.3	0.03	0.1	0.5
12	5	4	5	1000	20.0	900	2000
13	0.03	<0.1	0.04	0.04	0.05	0.02	0.3
14	3	3	4	6	6.0	10.0	20.0
15	0.02	<0.1	0.02	0.06	0.06	0.05	0.06
16	0.04	<0.1	0.05	0.04	0.03	0.03	0.03
17	0.02	<0.1	2	0.06	0.06	0.04	0.06

NOTE 1: These represent remote readings where possible.

NOTE 2: OOS indicates that the monitor is "out of service".

NOTE 3: All radiation readings above in units of millirems/hour.

FIGURE 3.2-1

(Page 3 of 5)

Area Radiation Monitor Data

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
Mwe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	537°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88

Channel Number:	Radiation Levels:						
18	300	300	300	300	200	300	300
19	0.4	0.4	0.4	0.3	0.05	1.0	2.0
20	0.3	0.3	0.3	0.3	0.3	2.0	3.0
21	2	2	3	3	40.0	200	OOS
22	0.03	<0.1	0.03	0.04	0.03	0.02	0.03
23	0.02	<0.1	0.03	0.03	0.03	0.02	0.03
24	0.3	0.3	0.3	0.3	0.2	0.2	0.5
25	0.3	0.3	0.3	0.2	0.3	0.1	0.4
26	0.3	0.2	0.2	0.2	0.2	0.1	0.2
27	OOS	OOS	OOS	OOS	OOS	OOS	OOS
28	0.03	<0.[illegible]	0.03	0.03	0.02	0.02	0.03
29	0.06	<0.1	0.06	0.05	0.05	0.04	0.07
30	0.03	<0.1	0.04	0.3	6.0	15.0	0.4
31	0.2	0.2	0.2	0.4	1.0	4.0	10.0
32	5	4	5	5	5.0	3.0	5.0
33	0.02	<0.1	0.02	0.02	0.02	0.02	0.04
34	0.05	<0.1	0.07	0.06	0.07	0.2	0.8

NOTE 1: These represent remote readings where possible.

NOTE 2: OOS indicates that the monitor is "out of service".

NOTE 3: All radiation readings above in units of millirems/hour.

FIGURE 3.2-1
(Page 4 of 5)

Area Radiation Monitor Data

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/[illegible]6/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88
Channel Number:				Radiation Levels:			
35	0.03	<0.1	0.03	0.03	0.03	0.03	0.05
36	0.03	<0.1	0.03	0.03	0.03	0.02	0.04
37	0.02	<0.1	0.02	0.02	0.03	0.04	0.10
38	0.02	<0.1	0.02	0.02	0.02	0.02	0.02
39	0.05	<0.1	0.08	0.05	0.05	0.04	0.05
40	0.03	<0.1	0.03	0.03	0.03	0.02	0.03
41	0.03	<0.1	0.03	0.04	0.03	0.02	0.03
42	0.05	<0.1	0.04	0.03	0.05	0.03	0.04
43	0.04	<0.1	0.04	0.04	0.04	0.02	0.04
44	0.03	<0.1	0.02	0.03	0.03	0.02	0.02
45	OOS	0.2	OOS	OOS	OOS	OOS	OOS
46	0.2	0.4	0.4	0.5	0.4	0.4	0.5
47	3	3	0.3	2	3.0	3.0	4.0
48	3	4	3	2	3.0	5.0	4.0

NOTE 1: These represent remote readings where possible.

NOTE 2: OOS indicates that the monitor is "out of service".

NOTE 3: All radiation readings above in units of millirems/hour.

FIGURE 3.2-1
(Page 5 of 5)

Process Radiation Monitor Data

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88
Off-Gas Radiation - A (D11-K601A) [mr/hr]	3	2	3.5	11	2.0	4.0	4.0
Off-Gas Radiation - B (D11-K601B) [mr/hr]	3.5	3	3	2	3.0	4.0	6.0
Off-Gas Radiation - Linear (D11-K602) [mr/hr]	0	5E-8	0	1E-8	3.16E-6	75 E-9	5E-8
Radwaste Effluent (D11-K604) [cps]	2	2	2	4	7.0	50	200
GSW Effluent (D11-K605) [cps]	3.5	4	4	4	4.0	5.0	5.0
RBCCW System (D11-K606) [cps]	3	2	2	2	2.0	25	3.0
A Main Steam Line (D11-K603A) [mr/hr]	1.2	1	OOS	8	4.41E2	6.56 E2	9.55 E-2
B Main Steam Line (D11-K603B) [mr/hr]	1.4	1	1	30	3.83E2	6.71 E2	9.90 E-2
C Main Steam Line (D11-K603C) [mr/hr]	1.6	2	1.2	9	4.01E2	6.78 E2	9.80 E-2
D Main Steam Line (D11-K603D) [mr/hr]	8.0	11	1	16	3.89E2	7.61 E2	8.60 E-2
Div-I EECW Hx Inlet (D11-K800A) [cpm]	200	200	200	300	550	5000*	3000
Div-I RHR S.W. (D11-K801A) [cpm]	200	200	200	200	450	200	1500
Div-I RB Vent Exh. (D11-K808) [cpm]	50	50	40	50	45	50	50

NOTE 1: OOS indicates that the monitor is "Out of Service".

NOTE 2: NA = not available or applicable at this time.

* = Additional shielding has since been placed by this detector and readings are now consistent with Div. II.

FIGURE 3.2-2
(Page 1 of 5)

Process Radiation Monitor Data

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88
Div-I Cont. Ctr. Makeup Air (D11-K809) [cpm]	40	30	30	30	40	50	40
Two Min. Holdup Pipe Exh. (D11-K814) [cpm]	250	200	400	1000	10000	30000	45000
Div-II EECW Hx Inlet (D11-K800B) [cpm]	200	200	200	300	300	500	1000
Div-II RHR S.W. (D11-K801B) [cpm]	150	200	200	200	200	250	600
Circ. Water Rev. Decant Line (D11-K802) [cpm]	200	200	200	300	300	250	200
Div-II RB Vent Exh. (D11-K810) [cpm]	60	60	40	50	50	50	50
Div-II Cont. Ctr. Makeup Air (D11-K813) [cpm]	60	60	60	50	60	60	60
Two Min. Holdup Pipe Exh. (D11-K815) [cpm]	300	200	400	3000	2500	30000	100000
1st Flr. Inside Drywell (D21-K745) [mr/hr]	0.3	0.2	OOS	0.1	0.1	OOS	OOS
1st Flr. on Site Stg. Bldg. Control Room (D21-K846) [mr/hr]	0.2	0.4	0.5	0.5	0.5	0.3	0.5

NOTE 1: OOS indicates that the monitor is "Out of Service".

NOTE 2: NA = not available or applicable at this time.

FIGURE 3.2-2
(Page 2 of 5)

Process Radiation Monitor Data

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88

1st Flr. On Site Stg. Bldg. Compactor Room (D21-K847) [mr/hr]	3	4	4	4	4	3	4
1st Flr. On Site Stg. Bldg. Truck Unloading Station (D21-K848) [mr/hr]	3	4	4	4	4	3	4
Fuel Pool Vent Exh. (DIV I) (D11-K609C/D11-R606) [mr/hr]	4E-2	2.5E-2	4E-2	3.5E-2	4E-[illegible]	7 E-2	5 E-2
Fuel Pool Vent Exh. DIV II) (D11-K609D/D11-R606) [mr/hr]	6E-2	6E-2	4E-2	2.5E-2	3E-2	7 E-2	5 E-2
Pri. Cont. Rad (gamma) (T50-K003/T50-R809) [cpm]	40	45	90	50	70	175	350
Fuel Pool Vent Exh. (DIV I) (D11-K609A/D11-R605) [mr/hr]	NA	NA	4E-2	2E-2	3E-2	6 E-2	6 E-2
Fuel Pool Vent Exh. (DIV II) (D11-K609B/D11-R605) [mr/hr]	NA	NA	4E-2	3.5E-2	5E-2	6 E-2	7 E-2
C.C. Emerg. Air (South) Div I (D11-K836A) [cpm]	NA	NA	NA	30	30	50	40

NOTE 1: OOS indicates that the monitor is "Out of Service".

NOTE 2: NA = not available or applicable at this time.

FIGURE 3.2-2

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Process Radiation Monitor Data

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	726-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88
C.C. Emerg. Air (North), Div I (D11-K837A) [cpm]	NA	NA	NA	20	OOS	20	30
Cont. Area Hi-Range Monitor, Div I (D11-K816A) [mr/hr]	NA	NA	NA	2000	2500	5000	7000
C.C. Emerg. Air (South), Div II (D11-K836B) [cpm]	NA	NA	NA	30	30	50	30
C.C. Emerg. Air (North), Div II (D11-K837B) [cpm]	NA	NA	NA	60	25	50	50
Cont. Area Hi-Range Montior, Div II (D11-K816B) [mr/hr]	NA	NA	NA	2000	3000	6000	8000
Reactor Bldg. Vent Plenum Exh. (D11-K610) [uCi/cc]	NA	NA	NA	6.65 E-7	1.07E-6	3.1 E-4	7.51 E-7
Radwaste Bldg. Vent Exh. (D11-K610) [uCi/cc]	NA	NA	NA	8.83 E-7	8.35E-7	3.1 E-4	-8.63 E-7
SGTS Vent Exh. Div I (D11-K610) [uCi/cc]	NA	NA	NA	1.32 E-7	3.92E-8	2.3 E-3	3.06 E-8
SGTS Vent Exh. Div II (D11-K610) [uCi/cc]	NA	NA	NA	6.75 E-7	5.19E-8	1.24 E-7	2.53 E-8
OSB Machine Shop Vent Exh. (D11-K610) [uCi/cc]	NA	NA	NA	4 E-8	2.07E-7	1.13 E-3	2.34 E-7

NOTE 1: OOS indicates that the monitor is "Out of Service".

NOTE 2: NA = not available or applicable at this time.

FIGURE 3.2-2
(Page 4 of 5)

Process Radiation Monitor Data

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88
Turbine Bldg. Vent Exh. (D11-K610) [uCi/cc]	NA	NA	NA	7.67 E-8	1.85 E-7	3.92 E-6	1.70 E-7
On-Site Storage Bldg. Vent Exh. (D11-K610) [uCi/cc]	NA	NA	NA	2.64 E-7	1.[illegible]5 E-7	2.35 E-4	4.17 E-7

NOTE 1: OOS indicates that the monitor is "Out of Service".

NOTE 2: NA = not available or applicable at this time.

FIGURE 3.2-2
(Page 5 of 5)

Survey Data

Location: General Site

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100^0F	<100^0F	540^0F	10^0F	510^0F	527^0F	530^0F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88

1	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
3	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
4	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
5	<0.2	<0.2	[illegible]	<0.2	<0.2	<0.2	<0.2
6	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
7	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
8	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
10	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
11	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
12	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
13	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
14	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
15	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
16	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
17	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
18	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2

NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

FIGURE 3.2-3
(Page 1 of 32)

Survey Data

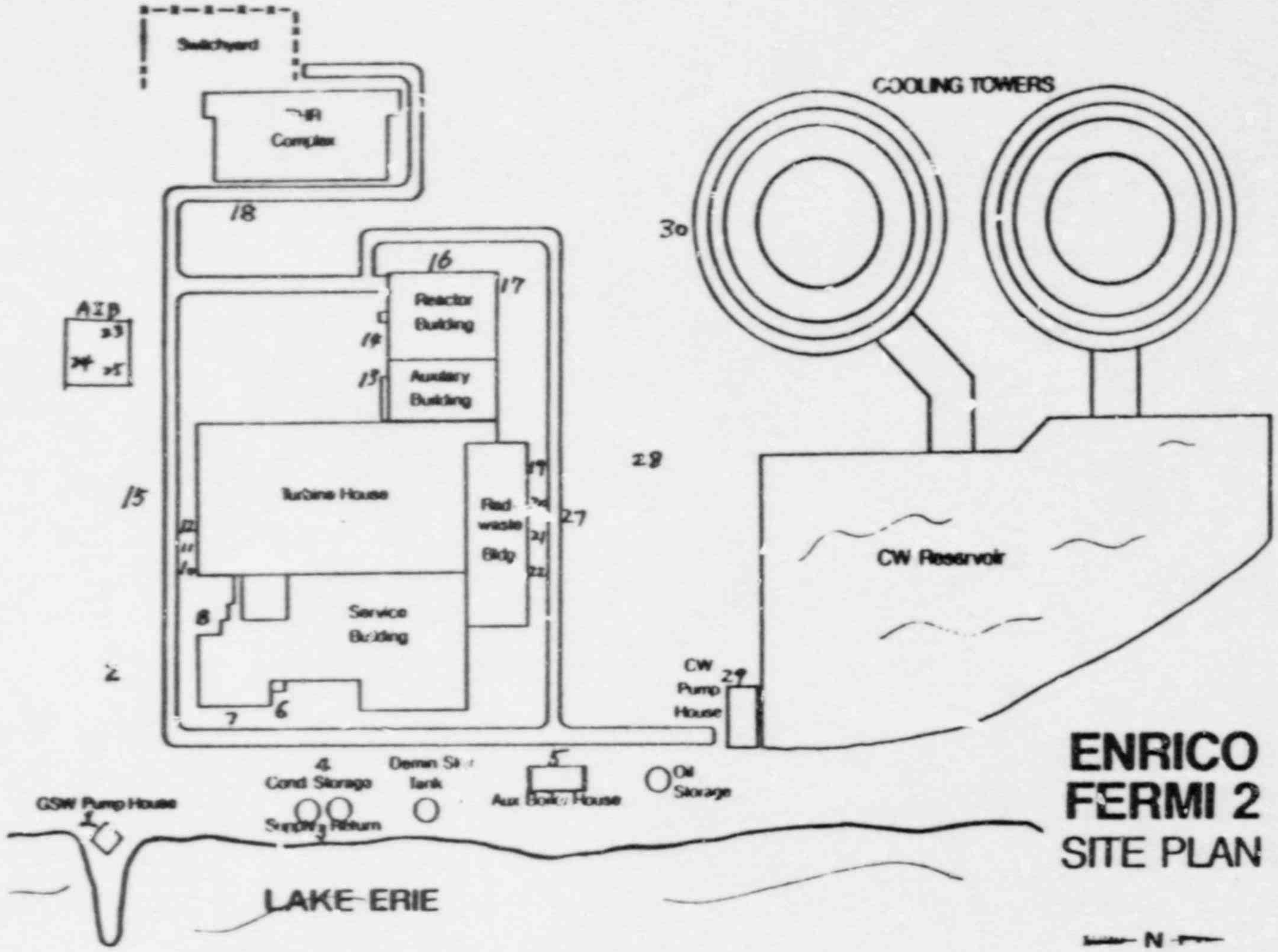
Location: General Site

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88
19	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
20	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
21	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
22	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
23	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
24	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
25	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
27	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
26	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
29	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
30	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2

NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

FIGURE 3.2-3
(Page 2 of 32)



(FIGURE 3.2-3)

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Survey Data

Location: Reactor Building - Sub Basement

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88
1	<0.2	<0.4	0.4	<0.2	<0.2	<0.2	0.5
2	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	1.0
3	<0.2	<0.4	0.3	<0.2	<0.2	<0.2	<0.2
4	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	<0.2
5	<0.2	<0.4	0.4	<0.2	2.5	1.2	3.6
6	<0.2	<0.4	0.3	<0.2	<0.2	<0.2	2.0
7	<0.2	<0.4	0.6	<0.2	<0.2	<0.2	0.8
8	<0.2	<0.4	0.4	<0.2	<0.2	<0.2	3.0
9	<0.2	<0.4	0.4	<0.2	0.4	0.2	1.4
10	<0.2	<0.4	0.4	<0.2	0.2	0.2	1.0

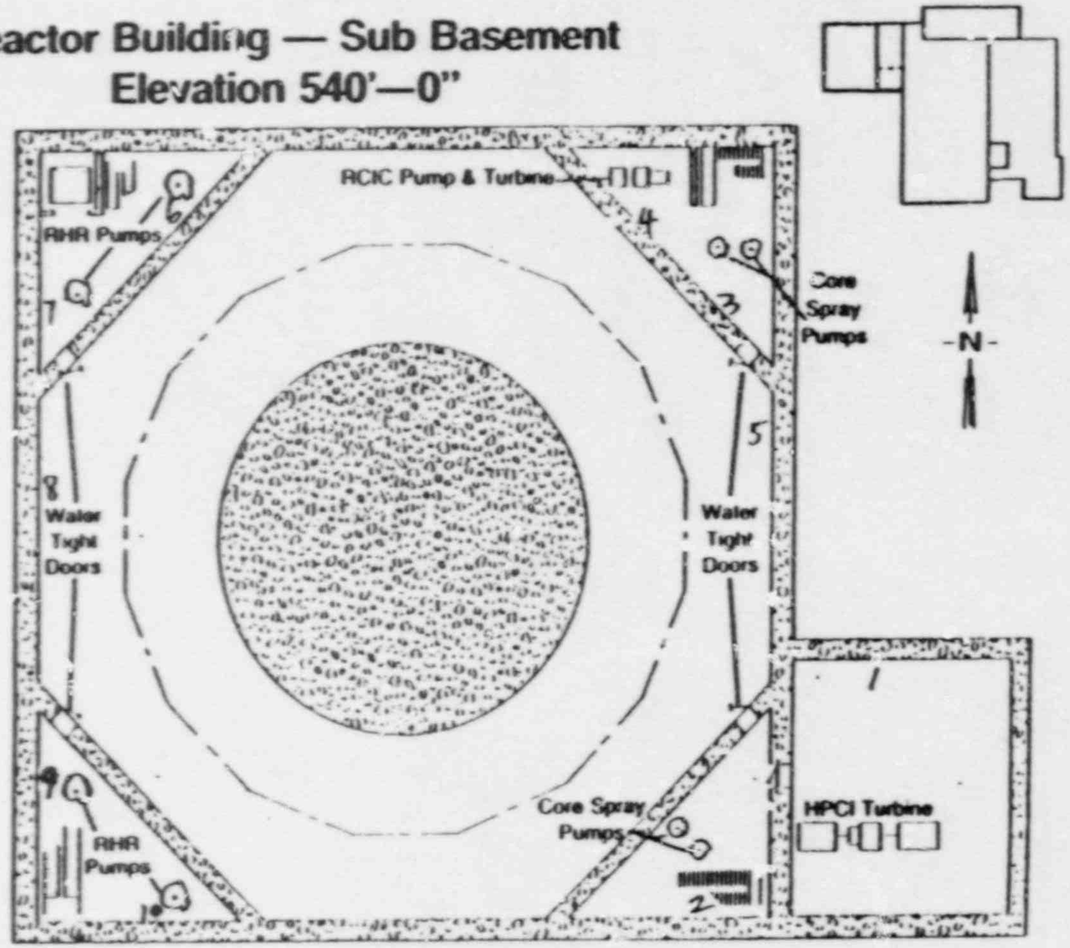
NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

FIGURE 3.2-3

(Page 4 of 32)

Reactor Building — Sub Basement
Elevation 540'—0"



(FIGURE 3.2-3)
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Survey Data

Location: Reactcr Building - Basement

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88

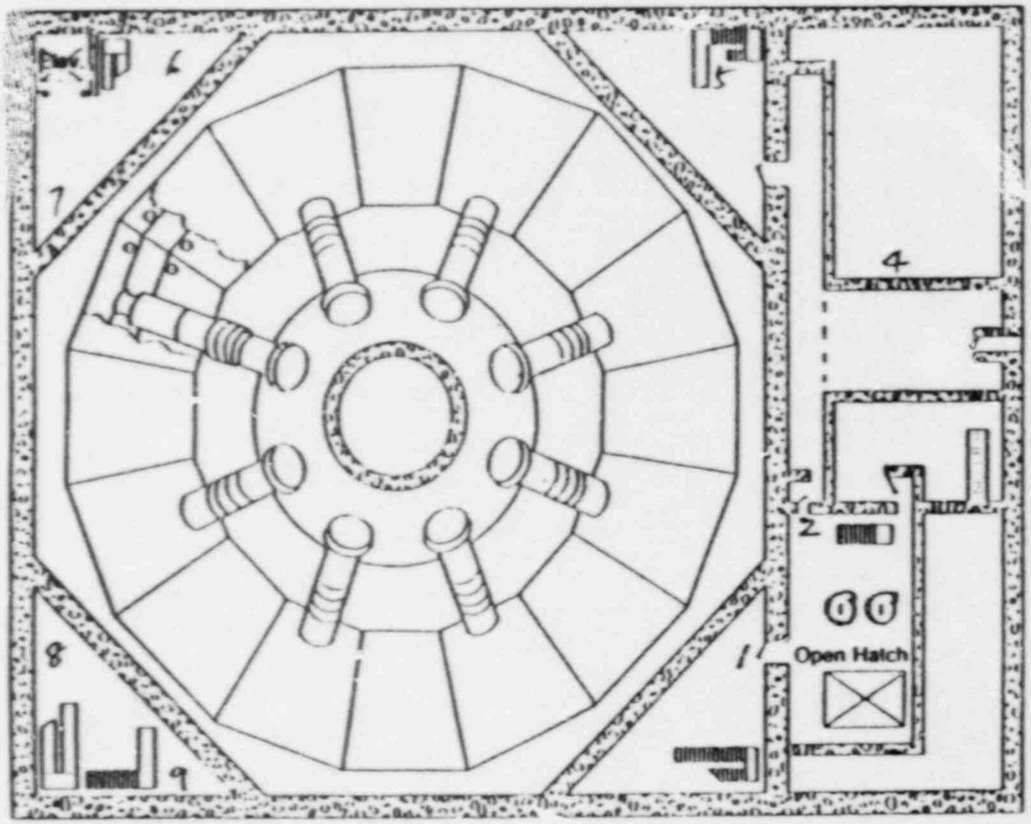
1	<0.2	<0.4	0.3	<0.2	<0.2	<0.2	<0.2
2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
4	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
5	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
6	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	<0.2
7	<0.2	<0.4	0.2	<0.2	0.2	<0.2	<0.2
8	<0.2	<0.4	0.2	<0.2	1.0	0.3	<0.2
9	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	<0.2

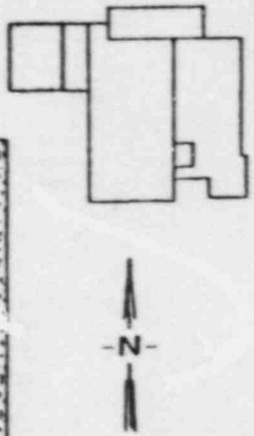
NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

FIGURE 3.2-3
(Page 6 of 32)

Reactor Building — Basement
Elevation 562'—0"





(FIGURE 3.2-3)
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Survey Data

Location: Reactor Building - First Floor

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88

1	<0.2	<0.2	<0.2	<0.2	0.4	0.2	0.9
2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.3
4	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	0.4
5	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	0.4
7	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	0.4
8	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	0.3
9	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	0.2
10	<0.2	<0.1	0.2	<0.2	<0.2	<0.2	<0.2
11	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	<0.2
12	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	0.7
13	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	<0.2
15	<0.2	<0.4	0.2	<0.2	0.4	<0.2	0.2
16	<0.2	<0.4	0.2	<0.2	0.4	0.2	1.2
17	<0.2	<0.4	0.2	<0.2	<0.2	0.4	1.0
18	<0.2	<0.4	0.4	<0.2	0.6	0.4	2.0
19	<0.2	<0.4	0.2	<0.2	1.5	0.6	3.8
21	<0.2	<0.4	0.2	<0.2	0.4	0.2	1.2
23	<0.2	<0.4	0.2	<0.2	<0.2	0.2	0.4

NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

FIGURE 3.2-3
(Page 8 of 32)

Survey Data

Location: Reactor Building - First Floor

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	3	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88
25	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.3
26	<0.2	<0.4	0.4	<0.2	1.0	0.5	4.0
27	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	0.2
28	<0.2	<0.4	0.4	<0.2	<0.2	<0.2	0.8
29	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	0.4
31	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	0.3
32	<0.2	<0.4	0.4	<0.2	<0.2	<0.2	<0.2
33	<0.2	<0.4	0.4	<0.2	<0.2	<0.2	0.3
34	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	0.2
35	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	0.2
37	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
38	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
39	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
40	<0.2	<0.2	<0.2	4.4	30	60	N/A

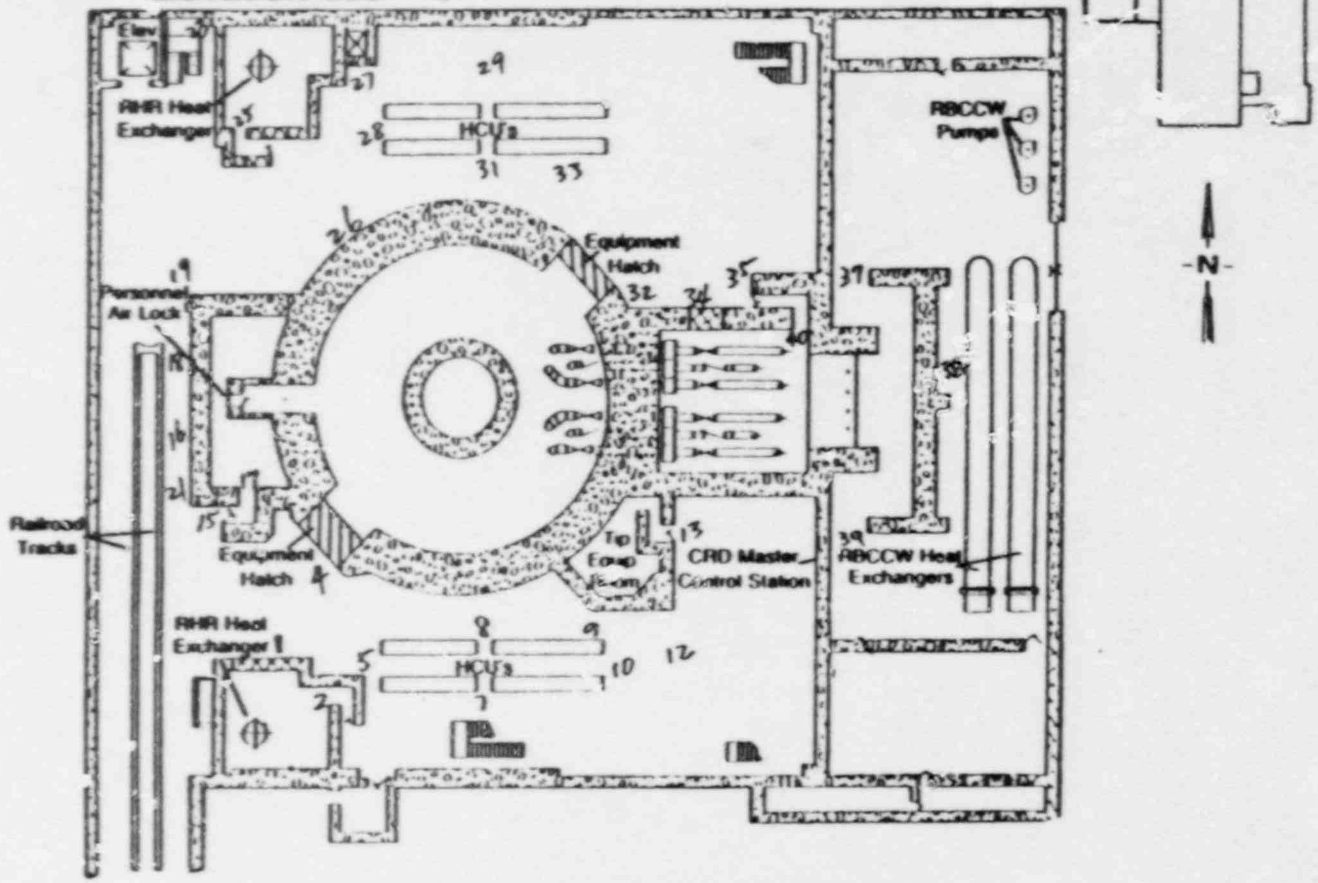
NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

N/A = Reading not taken, High Radiation Area, 0.2 millirems/hour at locked door.

FIGURE 3.2-3
(Page 9 of 32)

Reactor Building — 1st Floor
Elevation 583'—6"



(FIGURE 3.2-3)
(Page 10 of 32)

Survey Data

Location: Reactor Building - Second Floor

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88

1	<0.2	<0.4	0.4	<0.2	<0.2	<0.2	<0.2
2	<0.2	<0.4	0.4	<0.2	<0.2	<0.2	0.8
3	<0.2	<0.4	0.6	<0.2	3.8	2.0	8.0
4	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	<0.2
5	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.8
6	<0.2	<0.4	0.6	<0.2	<0.2	<0.2	0.2
7	<0.2	<0.4	0.6	<0.2	<0.2	<0.2	<0.2
8	<0.2	<0.4	0.6	<0.2	1.5	0.8	3.8
10	<0.2	<0.4	0.8	1.0	25	38	*0.4
11	<0.2	<0.4	0.6	<0.2	0.5	0.2	1.4
12	<0.2	<0.4	0.4	<0.2	<0.2	0.2	1.7
13	<0.2	<0.4	0.5	<0.2	0.5	0.4	0.4
14	<0.2	<0.4	1.4	3.6	100	100	N/A
15	<0.2	<0.4	0.8	<0.2	<0.2	0.2	<0.2
16	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	<0.2
18	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.2
19	<0.2	<0.4	0.3	<0.2	<0.2	<0.2	0.2
20	<0.2	<0.4	0.3	<0.2	<0.2	<0.2	<0.2

NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

* = RWCU Shut down.

N/A = Reading not taken. High Contamination Area.

FIGURE 3.2-3

(Page 11 of 32)

Survey Data

Location: Reactor Building - Second Floor

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88

21	<0.2	<0.4	0.4	<0.2	<0.2	<0.2	<0.2
22	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	<0.2
23	<0.2	<0.4	0.4	<0.2	<0.2	<0.2	<0.2
24	<0.2	<0.4	0.3	<0.2	<0.2	<0.2	<0.2
25	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	<0.2
26	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	<0.2
28	<0.2	<0.4	<0.2	0.4	5.0	2.0	N/A
29	<0.2	<0.2	<0.2	<0.2	<0.2	0.8	N/A
30	<0.2	<0.2	2.5	6	34	100	N/A
31	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
32	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
33	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2

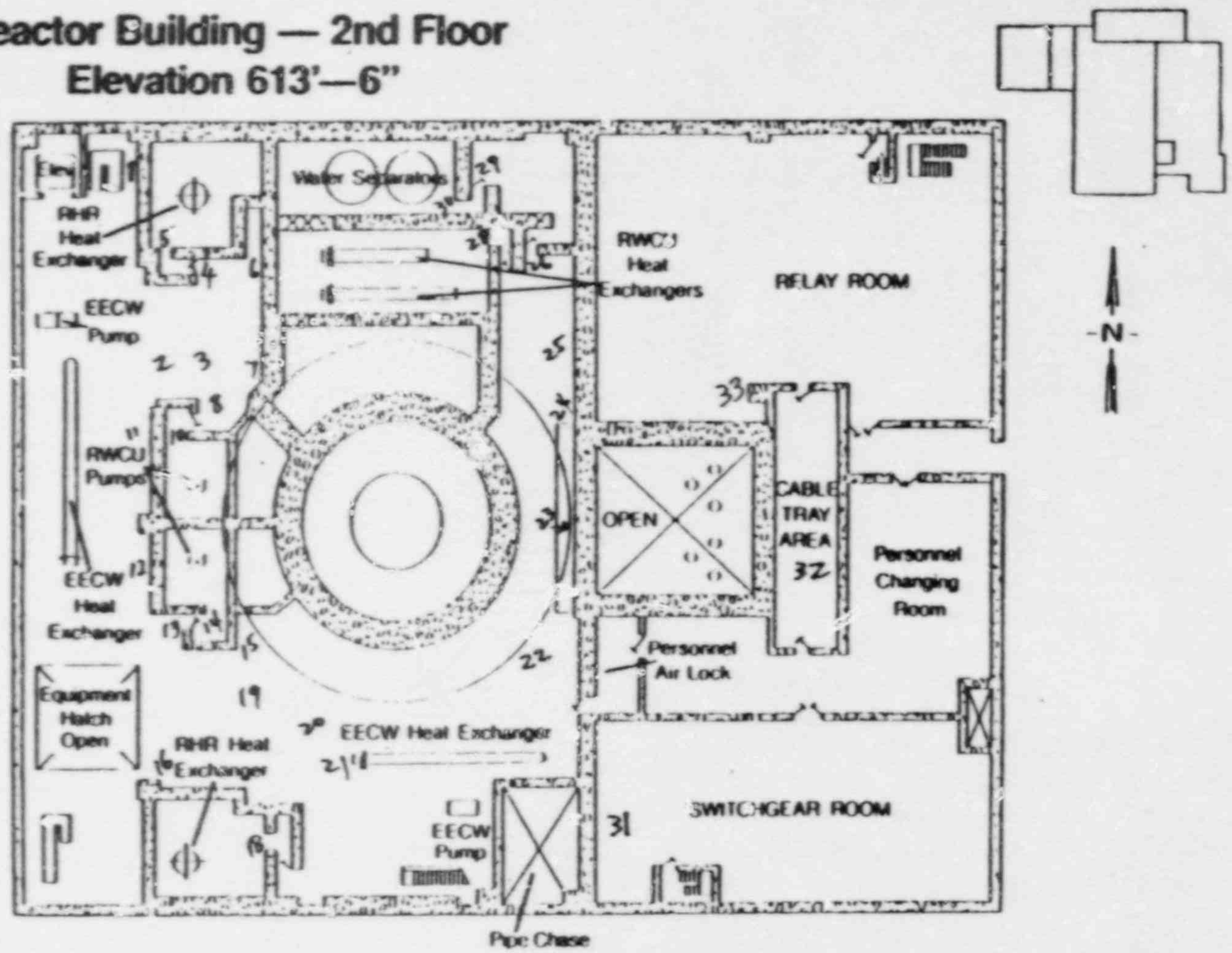
NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

N/A = Reading not taken. High Contamination Area.

FIGURE 3.2-3
(Page 12 of 32)

Reactor Building — 2nd Floor
Elevation 613'—6"



(FIGURE 3.2-3)
(Page 13 of 32)

Survey Data

Location: Reactor Building - Third Floor

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-temp	<100^{o}F	<100^{o}F	540^{o}F	540^{o}F	510^{o}F	527^{o}F	530^{o}F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88

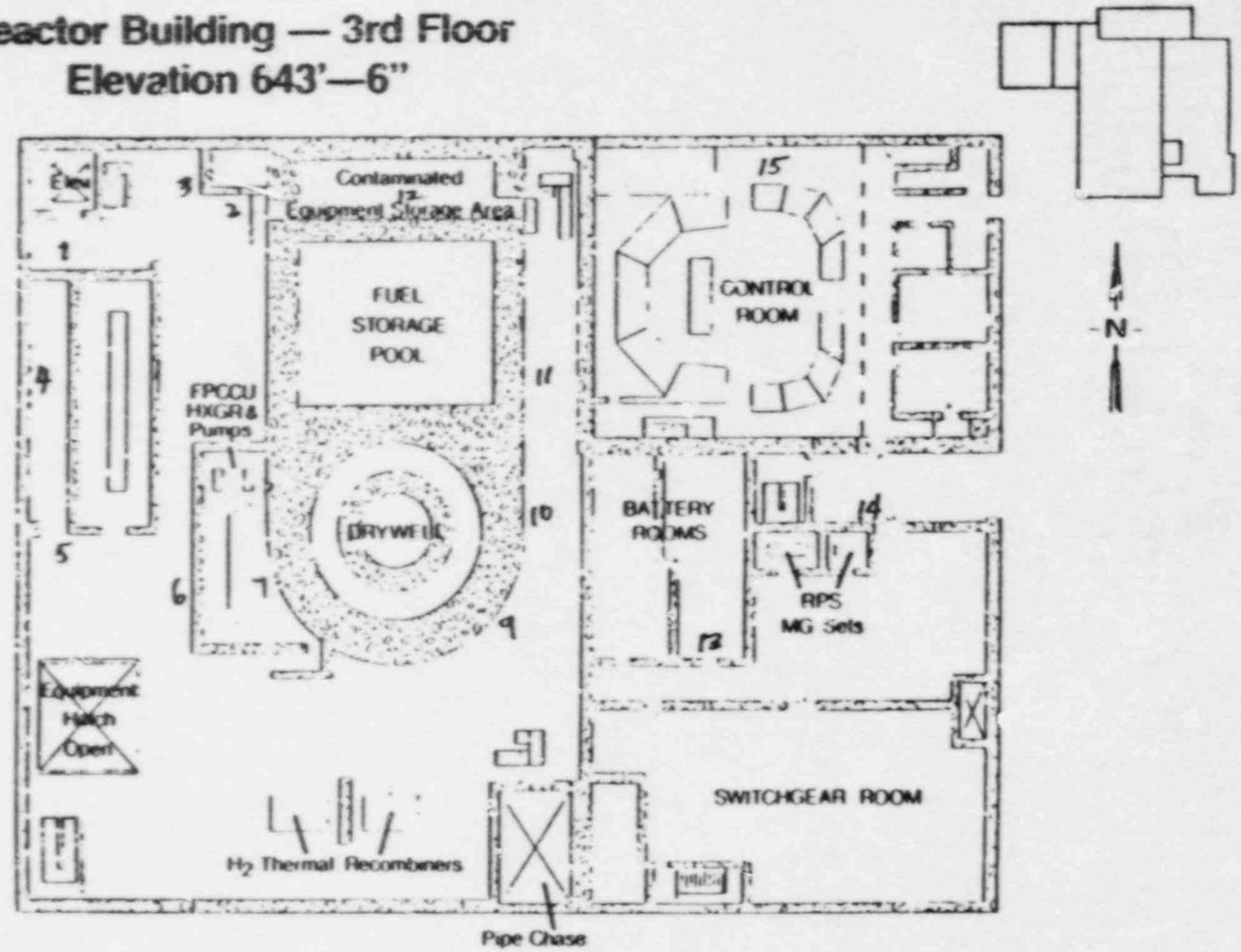
1	<0.2	<0.4	0.3	<0.2	<0.2	<0.2	<0.2
2	<0.2	<0.4	0.3	<0.2	<0.2	<0.2	<0.2
3	<0.2	<0.4	0.3	<0.2	<0.2	<0.2	<0.2
4	<0.2	<0.4	0.3	<0.2	<0.2	<0.2	<0.2
5	<0.2	<0.4	0.3	<0.2	<0.2	<0.2	<0.2
6	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
7	<0.2	<0.4	0.4	0.4	1.0	1.6	3.8
9	<0.2	<0.4	0.4	<0.2	<0.2	<0.2	1.3
10	<0.2	<0.4	0.4	<0.2	<0.2	<0.2	0.4
11	<0.2	<0.4	0.4	<0.2	<0.2	<0.2	<0.2
12	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
13	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
14	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
15	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2

NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

FIGURE 3.2-3
(Page 14 of 32)

Reactor Building — 3rd Floor
Elevation 643'—6"



(FIGURE 3.2-3)
(Page 15 of 32)

Survey Data

Location: Reactor Building - Fourth Floor

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88

2	<0.2	<0.2	<0.2	0.3	4.2	8.0	* <0.2
4	<0.2	<0.4	<0.3	0.2	<0.2	<0.2	<0.2
5	<0.2	<0.4	<0.3	0.2	<0.2	<0.2	0.8
6	<0.2	<0.4	<0.4	0.2	<0.2	<0.2	<0.2
7	<0.2	<0.4	<0.3	0.2	<0.2	<0.2	<0.2
8	<0.2	<0.4	<0.5	0.2	<0.2	<0.2	<0.2
9	<0.2	<0.4	<0.3	0.2	<0.2	<0.2	<0.2
10	<0.2	<0.4	<0.3	0.2	<0.2	<0.2	<0.2
11	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	<0.2
12	<0.2	<0.4	0.4	<0.2	<0.2	<0.2	<0.2
13	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	<0.2

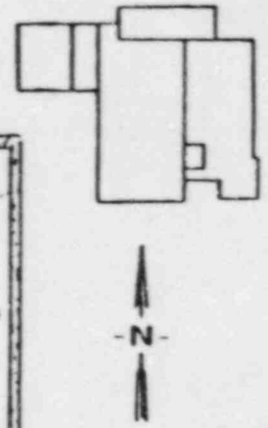
NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

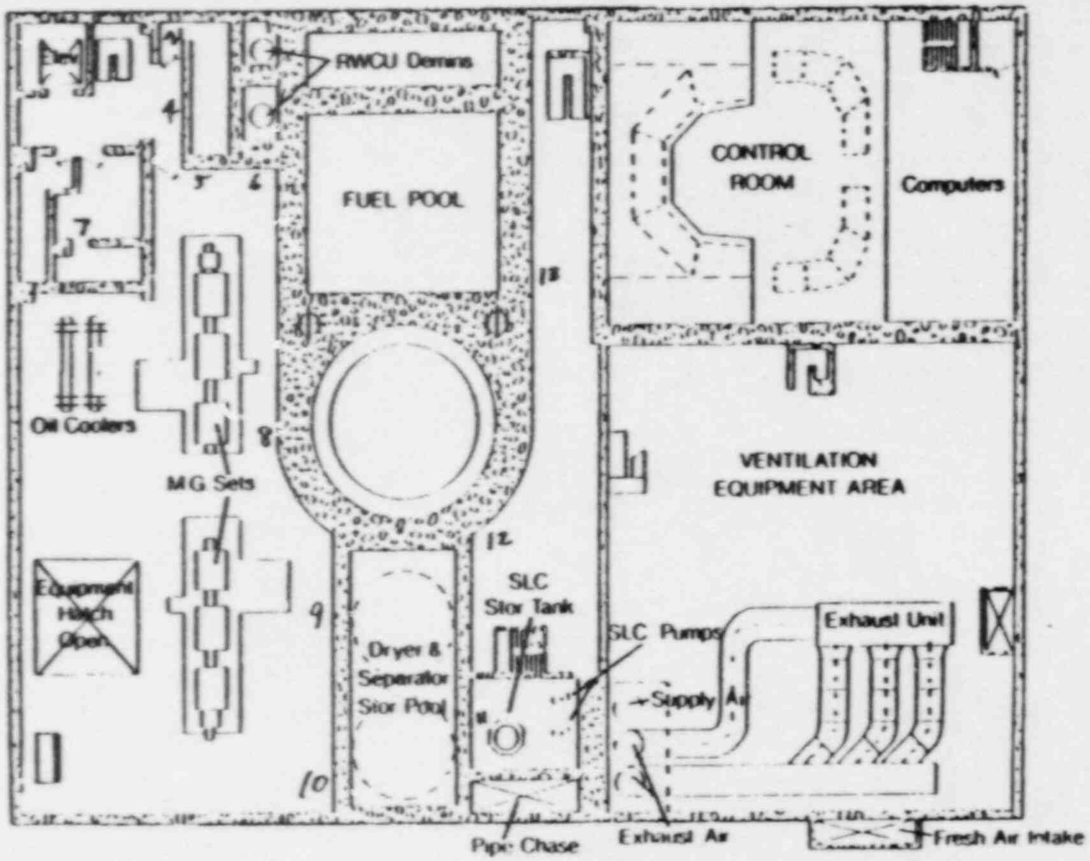
NOTE 2: All radiation readings above in units of millirems/hour.

* Reading taken at Outer Door.

FIGURE 3.2-3
(Page 16 of 32)

Reactor Building — 4th Floor
Elevation 659'—6"





(FIGURE 3.2-3)
(Page 17 of 32)

Survey Data

Location: Reactor Building - Fifth Floor

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88

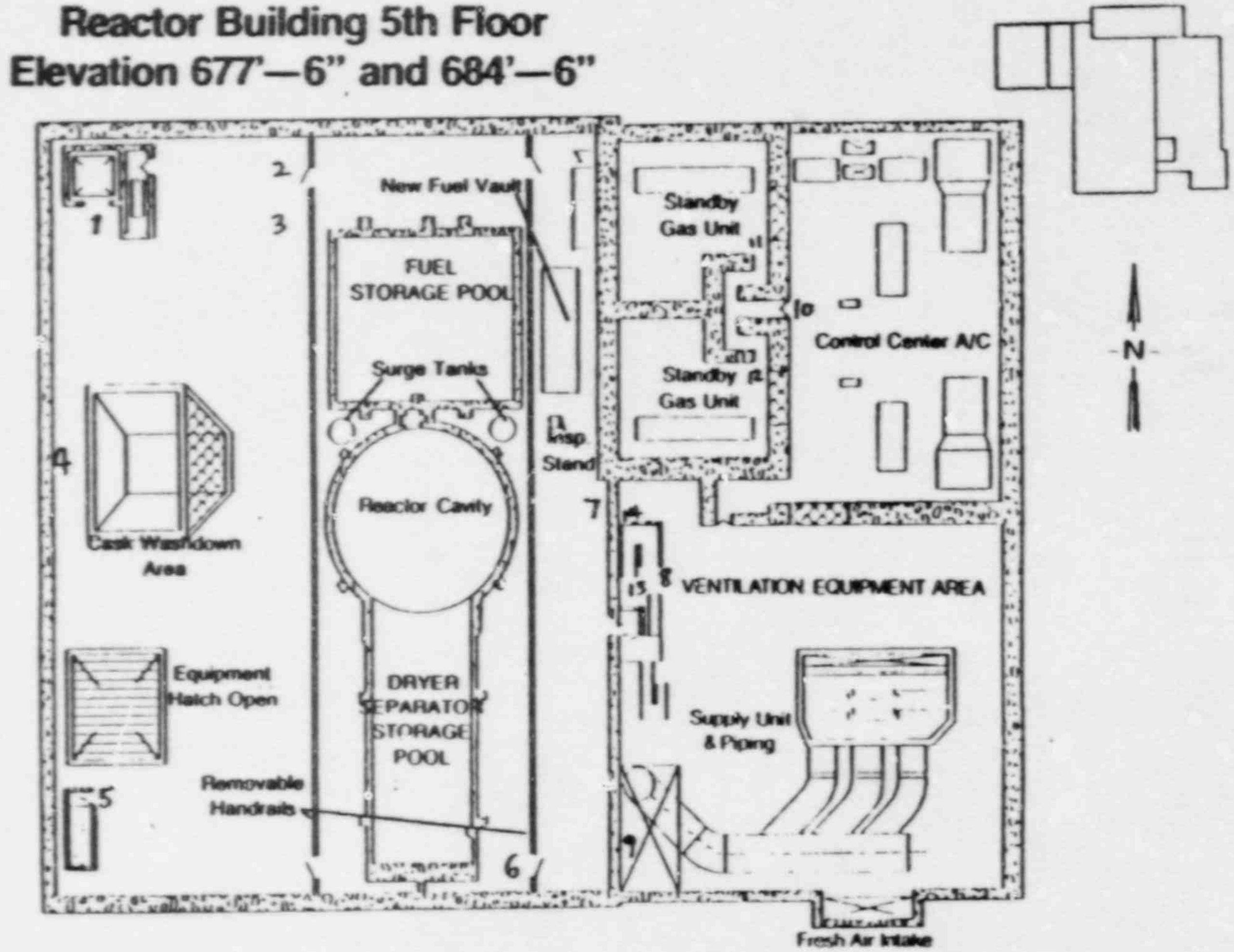
1	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	<0.2
2	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	<0.2
3	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	<0.2
4	<0.2	<0.4	0.3	<0.2	<0.2	<0.2	<0.2
5	<0.2	<0.4	0.3	<0.2	<0.2	<0.2	<0.2
6	<0.2	<0.3	<0.3	<0.2	<0.2	<0.2	<0.2
7	<0.2	<0.4	0.2	<0.2	<0.2	<0.2	<0.2
8	<0.2	<0.4	0.3	<0.2	<0.2	<0.2	<0.2
9	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
10	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
11	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
12	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
13	<0.2	<0.2	<0.2	<0.2	<0.2	3.0	1.0
14	<0.2	<0.4	0.3	<0.2	<0.2	<0.2	<0.2

NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

FIGURE 3.2-3
(Page 18 of 32)

Reactor Building 5th Floor
Elevation 677'—6" and 684'—6"



(FIGURE 3.2-3)

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Survey Data

Location: Turbine Building - Basement

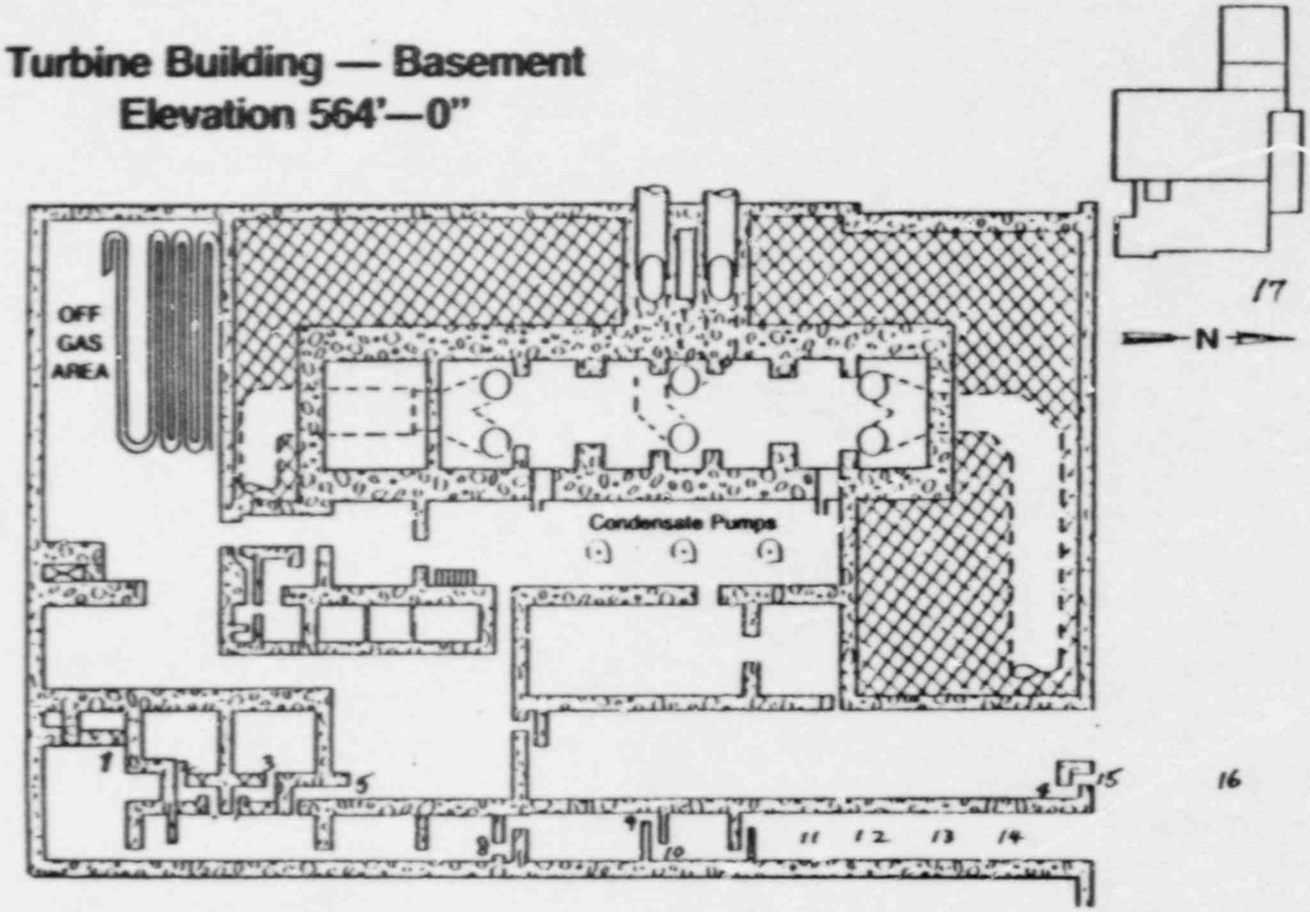
Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88
1	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
3	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
4	<0.2	<0.2	<0.2	<0.2	<0.2	0.7	<0.2
5	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
8	<0.2	<0.2	<0.2	<0.2	0.3	<0.2	<0.2
9	<0.2	<0.2	<0.2	<0.2	<0.2	0.5	1.5
10	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
11	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
12	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.2
13	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
14	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
15	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
16	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
17	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2

NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

FIGURE 3.2-3
(Page 20 of 32)

Turbine Building — Basement
Elevation 564'—0"



(FIGURE 3.2-3)
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Survey Data

Location: Turbine Building - First Floor

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	3/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88
1	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
3	<0.2	<0.2	<0.2	<0.2	2.0	2.0	* 10
4	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
5	<0.2	<0.2	<0.2	<0.2	0.3	0.6	2.0
6	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
7	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
8	<0.2	<0.2	<0.2	<0.2	0.9	2.0	2.8
9	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
10	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
11	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
12	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
13	<0.2	<0.2	<0.2	<0.2	1.2	3.5	8.0
14	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
15	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
16	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
17	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
18	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2

NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

* Survey performed in SJAE #1 Room vs SJAE #3 Room since SJAE #3 OOS.

FIGURE 3.2-3
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Survey Data

Location: Turbine Building - First Floor

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88
19	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.2
20	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
21	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.2
22	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.2
23	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
24	<0.2	<0.2	<0.2	<0.2	<0.2	0.3	0.2
25	<0.2	<0.2	<0.2	<0.2	0.3	1.0	1.8
26	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.5
27	<0.2	<0.2	<0.2	<0.2	2.0	5.0	5.0
28	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.4
29	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.2
30	<0.2	<0.2	<0.2	<0.2	0.2	1.0	1.5
31	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.2
32	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.2
33	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.2
34	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.2
35	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	1.4
36	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	1.2

NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

FIGURE 3.2-3
(Page 23 of 32)

Survey Data

Location: Turbine Building - First Floor

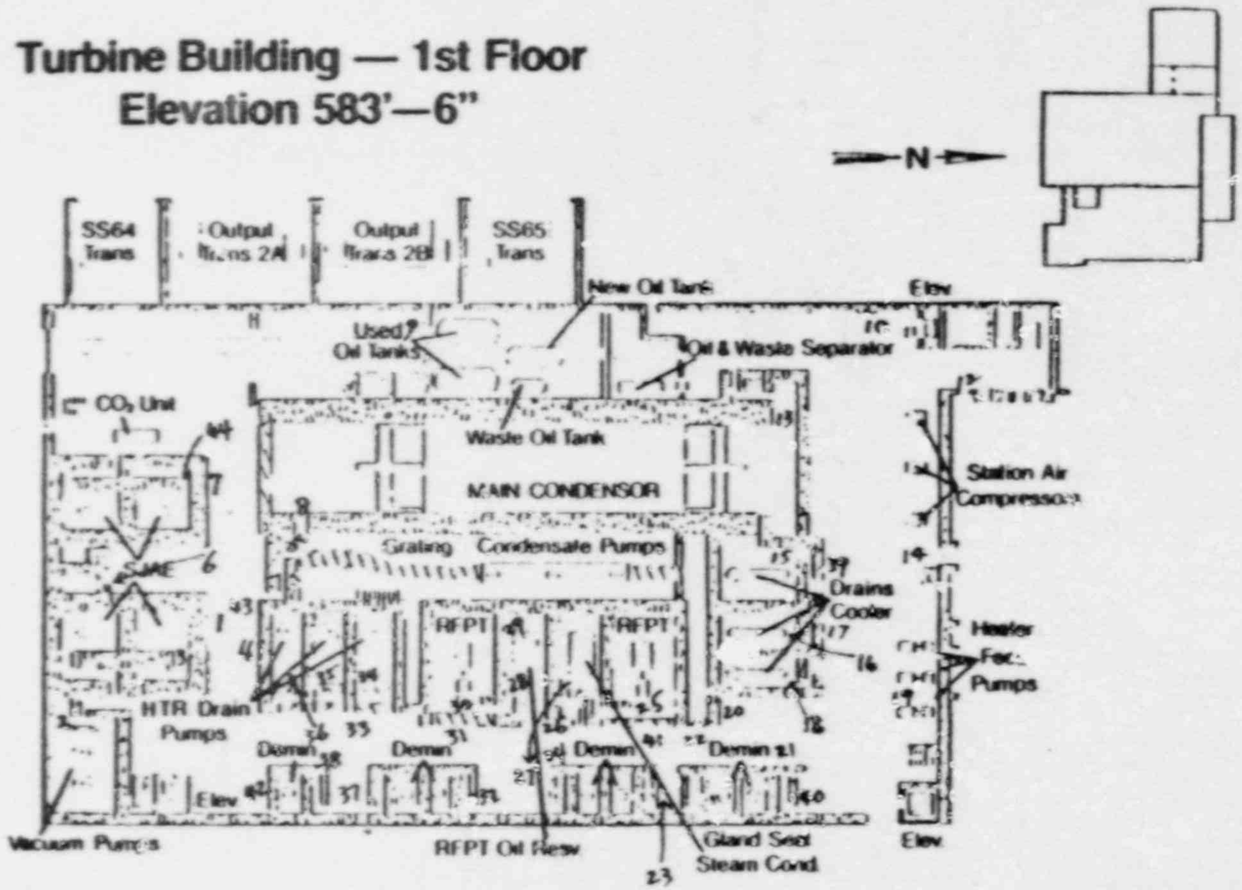
Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88
37	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
38	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
39	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
40	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
41	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.4
42	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
43	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
44	<0.2	<0.2	<0.2	<0.2	<0.2	4.0	10.0

NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

FIGURE 3.2-3
(Page 24 of 32)

Turbine Building — 1st Floor
Elevation 583'—6"



(FIGURE 3.2-3)
(Page 25 of 32)

Survey Data

Location: Turbine Building - Second Floor

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/ 6/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88

1	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
3	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
4	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
5	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
6	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
7	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
8	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
9	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
10	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
11	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
12	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
13	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
14	<0.2	<0.2	<0.2	<0.2	<0.2	26	48
15	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
16	<0.2	<0.2	<0.2	<0.2	10	30	60
17	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
18	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2

NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

FIGURE 3.2-3
(Page 26 of 32)

Survey Data

Location: Turbine Building - Second Floor

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88
19	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
20	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
21	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
22	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
23	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
24	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
25	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
26	<0.2	<0.2	<0.2	1.2	6.0	<0.2	30
27	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
28	<0.2	<0.2	<0.2	<0.2	<0.2	6.0	15
29	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
30	<0.2	<0.2	<0.2	<0.2	<0.2	20	30
31	<0.2	<0.2	<0.2	<0.2	<0.2	20	<0.2
32	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2

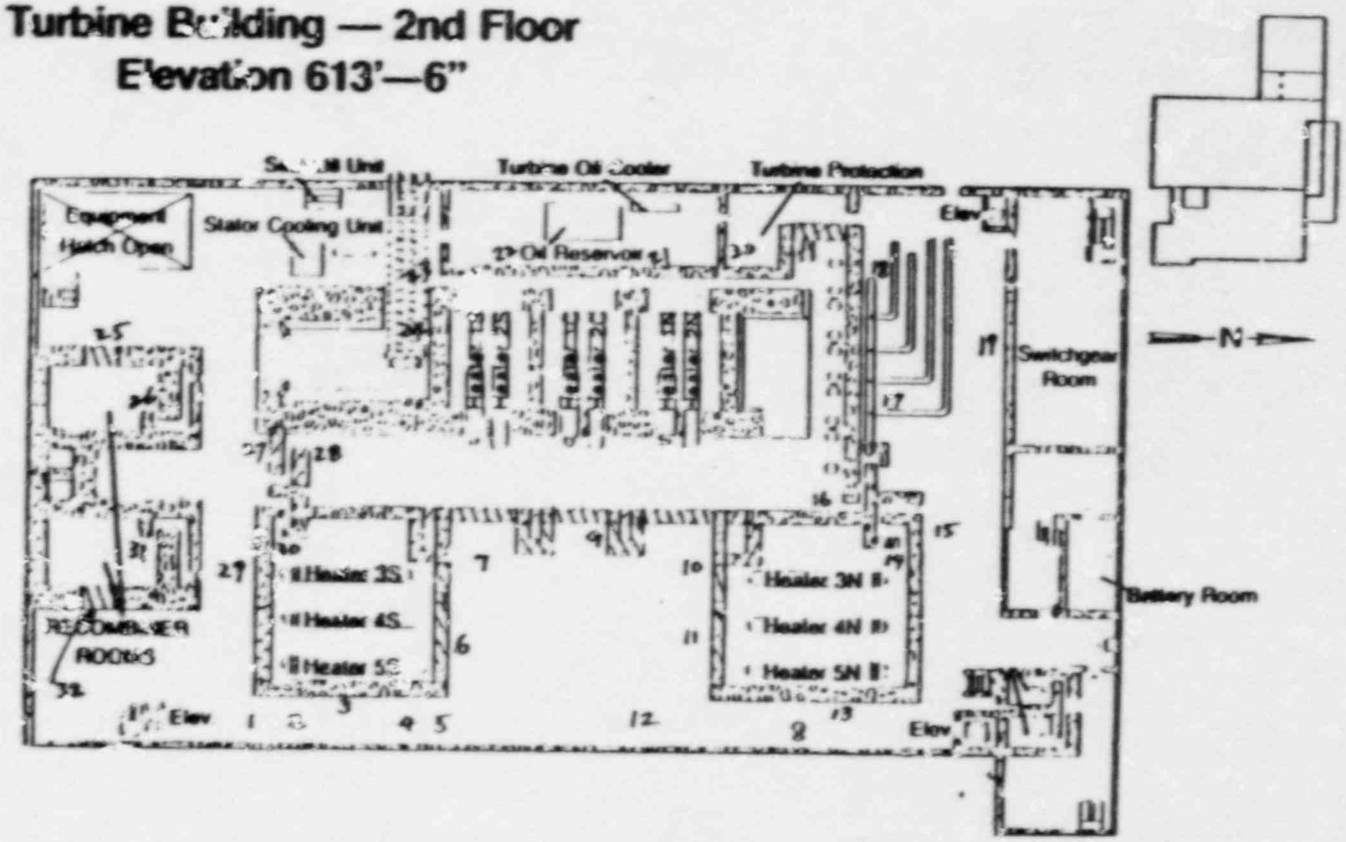
NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

FIGURE 3.2-3

(Page 27 of 32)

Turbine Building — 2nd Floor
Elevation 613'—6"



(FIGURE 3.2-3)
(Page 28 of 32)

Survey Data

Location: Turbine Building - Third Floor

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	[illegible]	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88

1	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
3	<0.2	<0.2	<0.2	<0.2	8.0	70	90
4	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
5	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
6	<0.2	<0.2	<0.2	<0.2	<0.2	1.0	<0.2
7	<0.2	<0.2	<0.2	<0.2	0.4	<0.2	1.4
8	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
9	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
10	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
11	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
12	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
13	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
14	<0.2	<0.2	<0.2	<0.2	0.5	<0.2	<0.2
15	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.2
16	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
17	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.2
18	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2

NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

FIGURE 3.2-3
(Page 29 of 32)

Survey Data

Location: Turbine Building - Third Floor

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88

19	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
20	<0.2	<0.2	<0.2	<0.2	2.0	80	100
21	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
22	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
23	<0.2	<0.2	<0.2	<0.2	<0.2	6.0	10
24	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
25	<0.2	<0.2	<0.2	<0.2	12	100	100
26	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
27	<0.2	<0.2	<0.2	<0.2	<0.2	5.0	8.0
28	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
29	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
30	<0.2	<0.2	<0.2	<0.2	0.2	<0.2	<0.2
31	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.5
32	<0.2	<0.2	<0.2	<0.2	0.2	<0.2	1.5
33	<0.2	<0.2	<0.2	<0.2	0.2	<0.2	0.4
34	<0.2	<0.2	<0.2	<0.2	3.0	90	120
35	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.3
36	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	0.2

NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

FIGURE 3.2-3
(Page 30 of 32)

Survey Data

Location: Turbine Building - Third Floor

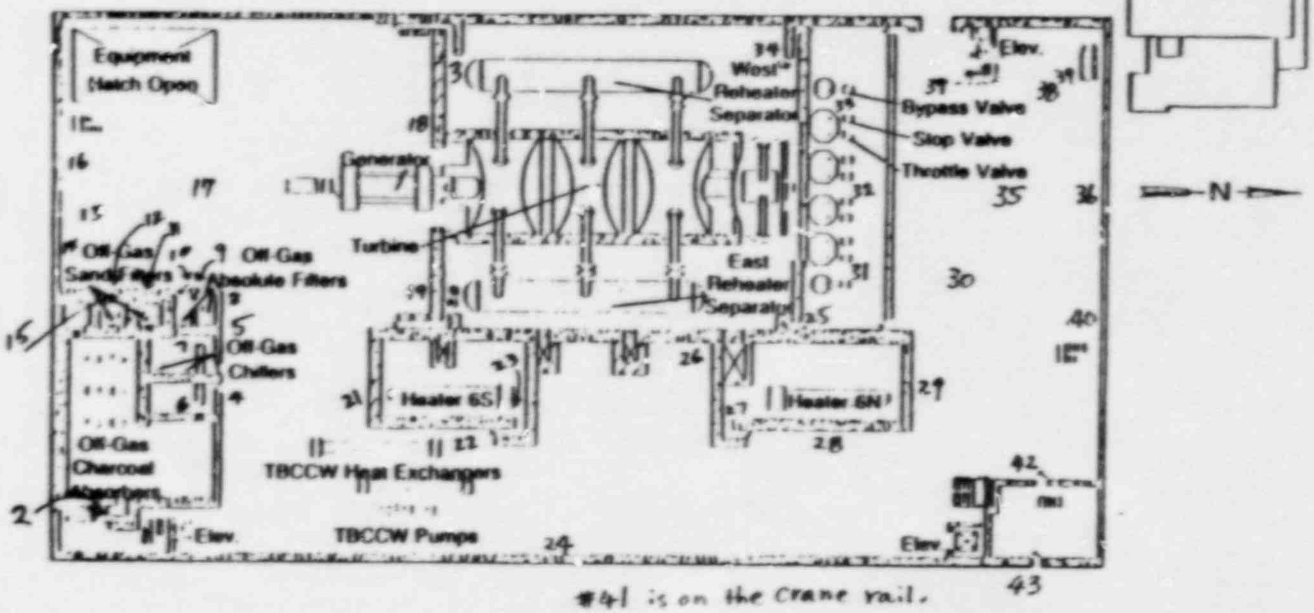
Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%	70.2-71%	96.9%
MWe	0	0	0	0	425	729-737	1076
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F	527°F	530°F
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87	12/15,16,17/87	7/12,13/88
37	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
38	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
39	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
40	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	1.2
41	<0.2	<0.2	<0.2	<0.2	1.0	5.0	1.2
42	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
43	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2

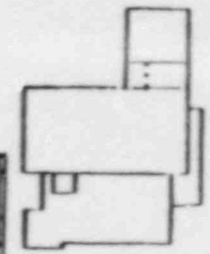
NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

NOTE 2: All radiation readings above in units of millirems/hour.

FIGURE 3.2-3
(Page 31 of 32)

Turbine Building — 3rd Floor
Elevation 643'—6"





#41 is on the Crane rail.

(FIGURE 3.2-3)
(Page 32 of 32)

3.3 Fuel Loading

3.3.1 Purpose

The purpose of this test was to load fuel safely and efficiently to the full core size (764 assemblies).

3.3.2 Criteria

Level 1

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

There must be a neutron signal count-to-noise count ratio of at least 2:1 on the required operable SRMs or fuel loading chambers (FLC). The minimum count rate, as defined by the Technical Specifications, must be met on the required operable SRMs or fuel loading chambers.

Level 2

None

3.3.3 Results

Prior to fuel loading, all fuel assemblies were inspected and then stored in the fuel pool in such a way that no rotation of fuel assemblies would be required during their transfer to the reactor vessel and also that no assembly would pass over any other assembly in the fuel pool during fuel loading. The only exception to this was bundle LJK 954 which was oriented SW instead of SE in the fuel pool, but was verified to be properly oriented in the core. Before the start of fuel load, all control rods were fully inserted, all blade guides were positioned as shown on Figure 3.3-1. Seven Sb-Be neutron sources were installed at locations shown on Figure 3.3-1.

All applicable initial conditions were verified prior to the start of fuel loading. Four times during the fuel loading process, fuel loading was suspended for greater than eight hours, and all applicable initial conditions were reverified before fuel loading was resumed.

The Bottom head drain temperature indication was used to obtain the Reactor Coolant Temperature at least once every eight hours ($\pm$ 15 minutes) during the fuel loading process.

Detailed fuel loading sheets, approved by the Reactor Engineer, provided the instructions on each individual fuel assembly to be moved from a specific location in the fuel pool to a pre-assigned location in the core. It also provided the instructions on what control rods were to be exercised for functional and sub-criticality checks for pre-defined core configurations. FLC moves to be made during the fuel loading were also included. Most of the changes required to the fuel loading sheets during fuel loading were to move the FLCs earlier due to high count rates experienced when fuel assemblies and/or the neutron sources were too close to the FLCs. The only other change involved using Control Rod 10-27 (instead of 06-27) for a sub-criticality check due to an accumulator problem with Rod 06-27.

Four FLCs (one per quadrant) were used to monitor the count rate from the start of fuel loading up to the point when 532 bundles were loaded in the core. In order to keep the FLC count rate within a desirable range and to accommodate an increasing core size, it was necessary to move the FLCs outward by approximately one cell routinely as fuel loading progressed. The location of FLCs was selected to ensure that each quadrant of the core was adequately monitored. (See Figure 3.3-4)

The upscale alarm setpoint was set at 1×10^5 cps and the upscale trip setpoint was set at 2×10^5 cps for each FLC. The downscale rod block setpoint was 3 cps. The FLCs were checked for flux response either by control rod pulls during scheduled sub-criticality checks or by lifting the FLCs partially out of the core. These flux response checks were made at least once every eight hours during fuel loading and prior to the resumption of fuel loading when fuel loading was delayed for eight hours or more. In addition, the Signal-to-Noise ratio was calculated for each FLC prior to start of fuel load, during any required reverification of plant initial conditions and every time the FLCs were moved to a new location. (See Figure 3.3-2)

Four SRMs (one per quadrant) were used to monitor the neutron count rate starting from the point when 532 bundles were loaded in the core to the completion of fuel load (764 bundles). With the SRM detectors connected to the SRM instrument channels, the rod block and the upscale trip setpoints were set down to 1×10^4 and 2×10^4 respectively, since no previous saturation test was performed on the SRM detectors. The down scale rod block setpoint was 3 cps. The SRM flux response check was performed at least once every eight hours during the fuel loading process by partially withdrawing each SRM.

Fuel loading commenced on March 20, 1985 with the loading of four fuel assemblies around the central neutron source. The loading continued in control cell units that sequentially completed each face of an increasing square core, loading in a clockwise direction until a 12 x 12 square was completed with symmetry about the center source. The thirteen control cells (52 bundles) needed to form a 14 x 14 square array of bundles around the center Control Rod (30-31) were loaded next. The remaining control cells were loaded, one on each face at a time, in a clockwise manner, such that the core was rotationally symmetric after every four control cells had been loaded. (See Figure 3.3-3)

Control rod functional and sub-criticality checks were performed either after every cell (first 4 cells in the core), or after every two or four cells as dictated by the detailed fuel loading sheets. The purpose of the sub-criticality checks was to ensure that it was safe to load the next control cell(s).

For each bundle a visual verification was performed to ensure that the bundle was properly grappled before the bundle was lifted from the fuel pool racks, that there was adequate clearance on all sides while the bundle was being moved to the reactor cavity and that it was loaded in the core in the proper location with the proper orientation. Also, physical verification was made of the fact that the bundle was ungrappled before the hoist was raised. Similar verifications were made for the blade guides lifted out of the core and the FLC moves made during the fuel loading process.

A day-by-day account of the fuel load progress is given in Figure 3.3-5. Most of the problems that caused delays were related to the refueling bridge (limit switch, power loss, grapple indication, air hose break, etc.). Fuel loading was halted on Sundays in order to perform required weekly surveillances on FLC/SRMs, IRMs, APRMs and the refueling bridge.

During the fuel loading process, FLC/SRM count rates were monitored periodically and 1/M calculations were performed and plotted for each FLC/SRM and for the average of the four FLC/SRMs (See Figure 3.3-6). The average 1/M plot was used to project the estimated number of bundles for criticality. If criticality was projected during the next loading increment then the increment size was reduced between 1/M calculations. Strong geometric effects were seen, particularly during the first few bundles loaded in the core and also when the bundles were loaded near and FLC. These geometric effects resulted in erronious (but highly conservative) projections which often resulted in very small increment sizes (1 - 2 bundles) between 1/M calculations. After eighty bundles were loaded in the core, the maximum increment size between 1/M calculations was reduced to one cell (4 bundles except for the peripheral locations where a maximum of five bundles were loaded between 1/M calculations).

Bundle LJK 677 was identified to have a rusted channel fastener that had to be replaced. Some debris was identified in the core on bundles LKJ 398, LJK 506 and LJK 957. After fuel loading was completed, these bundles were pulled out of the core to correct the respective problems and reinserted back into the core.

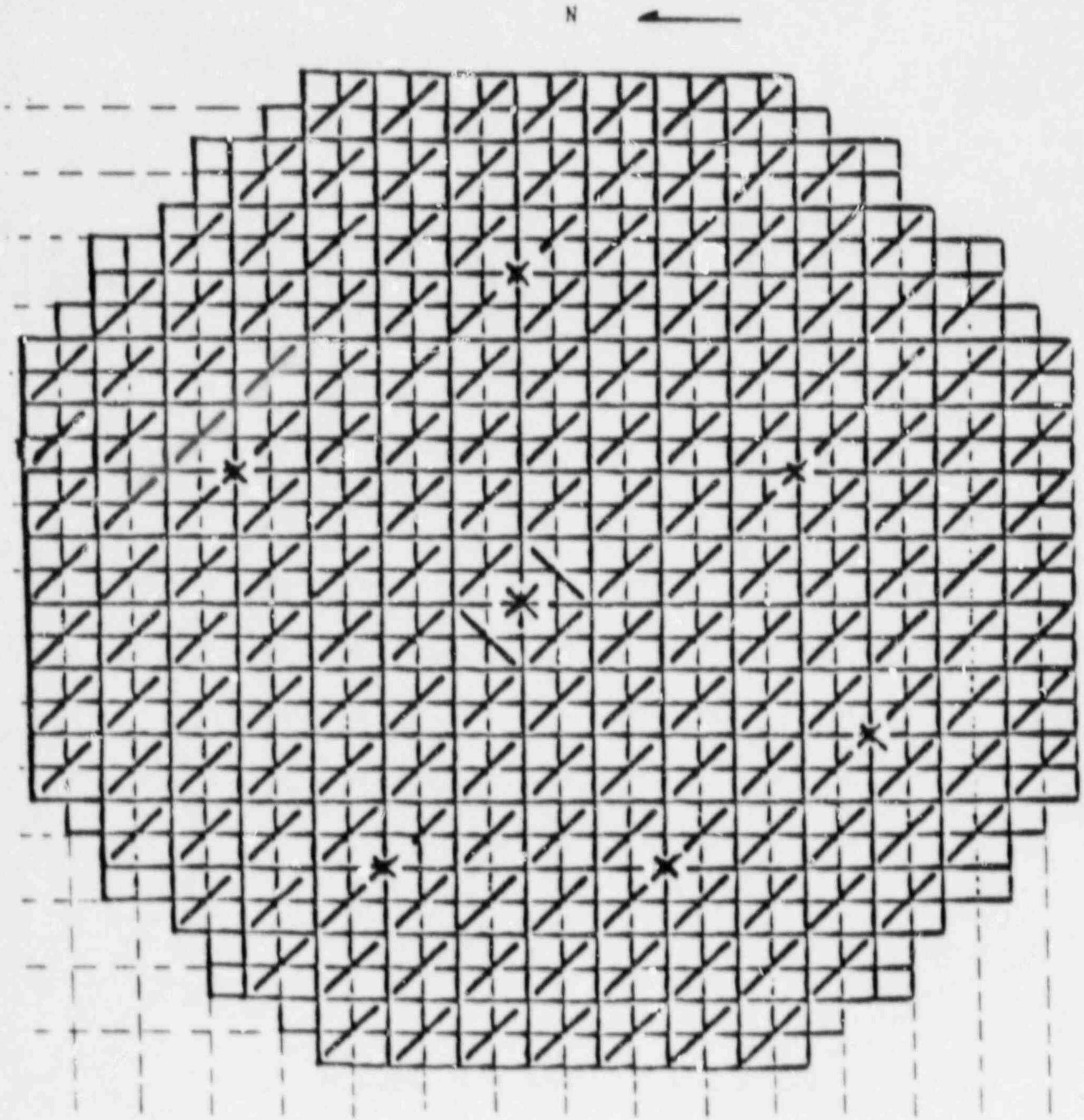
After the 12 x 12 square array of bundles was completed, a partial core shutdown margin (SDM) demonstration was performed by withdrawing the analytically determined strongest Rod (26 - 27) and a diagonally adjacent Rod (22- 23) out of the core. Sub-criticality with these two rods withdrawn demonstrated that there was at least a 0.38% delta K/K shutdown margin for the existing core configuration. Because the calculated Keff for the 12 x 12 array with the two rods withdrawn was 0.9758, and the calculated Keff for the full core with only the strongest rod withdrawn is 0.97, sub-criticality for the partial core demonstrated that the shutdown margin would be met throughout the remaining fuel loading process.

The fuel loading was completed after fifteen days on April 4, 1985. All criteria were satisfied.

FIGURE 3.3-1

NEUTRON SOURCE LOCATION AND BLADE GUIDE ORIENTATION

PRIOR TO FUEL LOADING



* SOURCE (7)   /BLADE GUIDE (185)

FIGURE 3.3-2

Signal to Noise Measurement

DATE (TIME)	DETECTOR	A CPS	A S/N	B CPS	B S/N	C CPS	C S/N	D CPS	D S/N	# OF BUNDLES LOADED
03-20-85 (2019)	FLC	10	24	10	99	10	32.3	10	24	Prior to fuel load
03-21-85 (0005)	FLC*	50	49	60	59	50	49	80	79	4
03-22-85 (0340)	FLC*	50	249	50	99	60	149	70	174	48
03-22-85 (2005)	FLC*	6.6	16	3.8	9.8	6.5	64	6.0	5	96
03-22-85 (2227)	FLC*	-	-	7.0	3.4	-	-	-	-	96
03-23-85 (2110)	FLC*	5	4	12	11	-	-	-	-	144
03-25-85 (1420)	FLC	10	19.0	11	14.7	12	19.0	12	14.0	156
03-26-85 (0020)	FLC*	10	49.0	20	89.9	-	-	-	-	196
03-26-85 (1915)	FLC*	38	189	32	159	40	159	4.8	15	260
03-28-85 (1116)	FLC*	30	99	4	39	35	116	2.5	7.3	388
03-29-85 (0907)	FLC	300	999	100	999	150	374	90	299	440
04-01-85 (1528)	SRM	16	159	12	119	40	399	15	149	532

*S/N Ratios obtained during FLC moves
-FLC not moved

FIGURE 3.3-3

CORE LOADING SEQUENCE

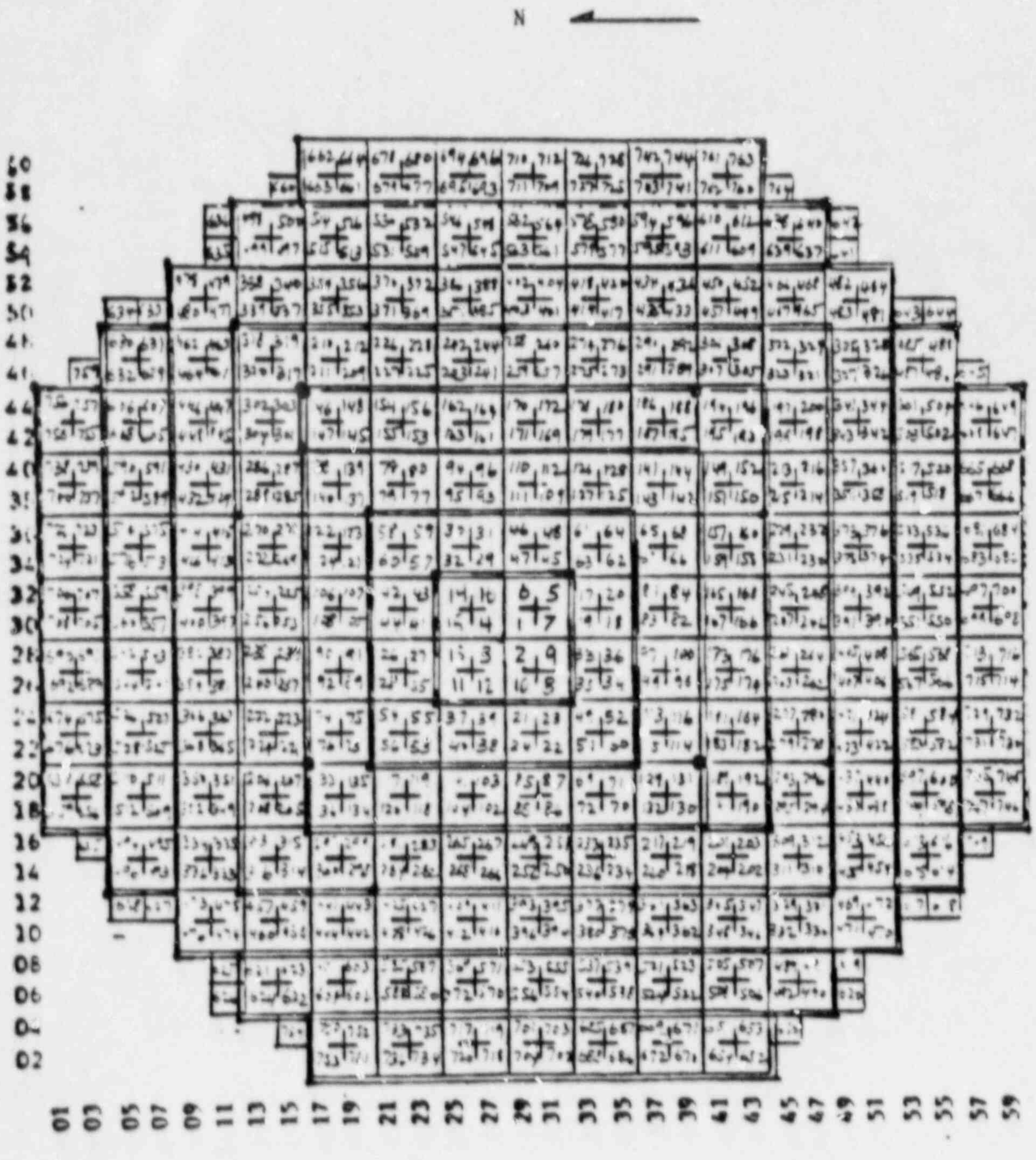


FIGURE 3.3-4

FLC MOVES DURING CORE LOADING

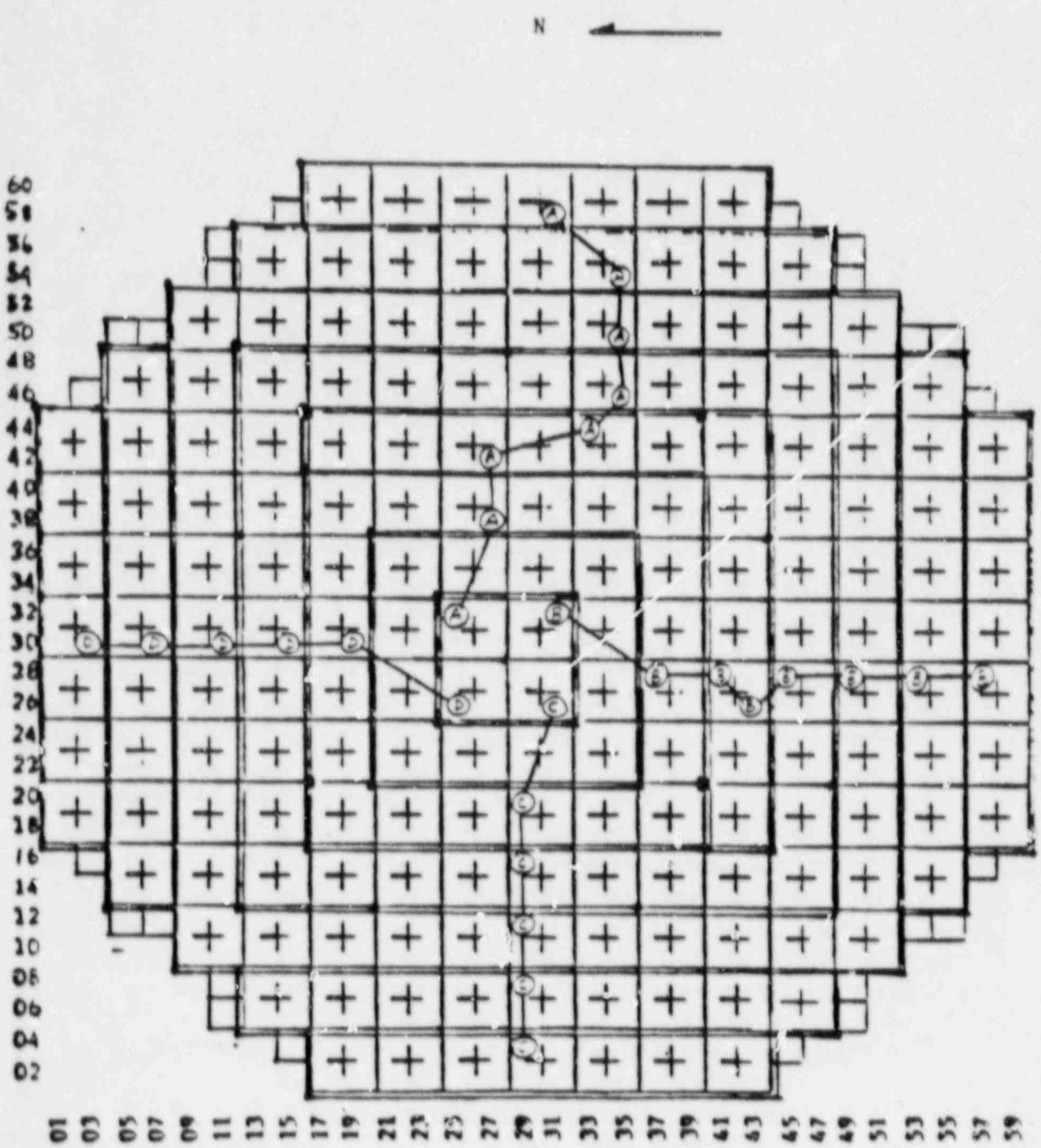


FIGURE 3.3-5

Daily Fuel Loading Progress

DATE	BUNDLES LOADED DAY	BUNDLES LOADED TO DATE	COMMENTS
03-20-85	4	4	Fuel load started at 2130.
03-21-85	32	36	Rod Block limit switch malfunction.
03-22-85	62	98	
03-23-85	58	156	
03-24-85	0	156	Weekly surveillance on SRMs, IRMs, APRMs and Refueling Bridge.
03-25-85	38	196	Fuel load resumed at 1500.
03-26-85	82	278	
03-27-85	84	362	
03-28-85	76	438	
03-29-85	66	504	Transformer #64 lost due to initiation of its deluge (fire protection) system.
03-30-85	28	532	0400 refuel bridge power cable problem. Cable cut and re-termed to restore the system.
03-31-85	0	532	Weekly surveillance. FLC to SRM switchover.
04-01-85	14	546	Fuel load resumed at 2000.
04-02-85	74	620	
04-03-85	48	668	Air hose damaged when stuck center section of the mast was released and dropped.
04-04-85	96	764	Fuel load completed at 2350.

FIGURE 3.3-6

NUMBER OF BUNDLES LOADED

1/M Plot

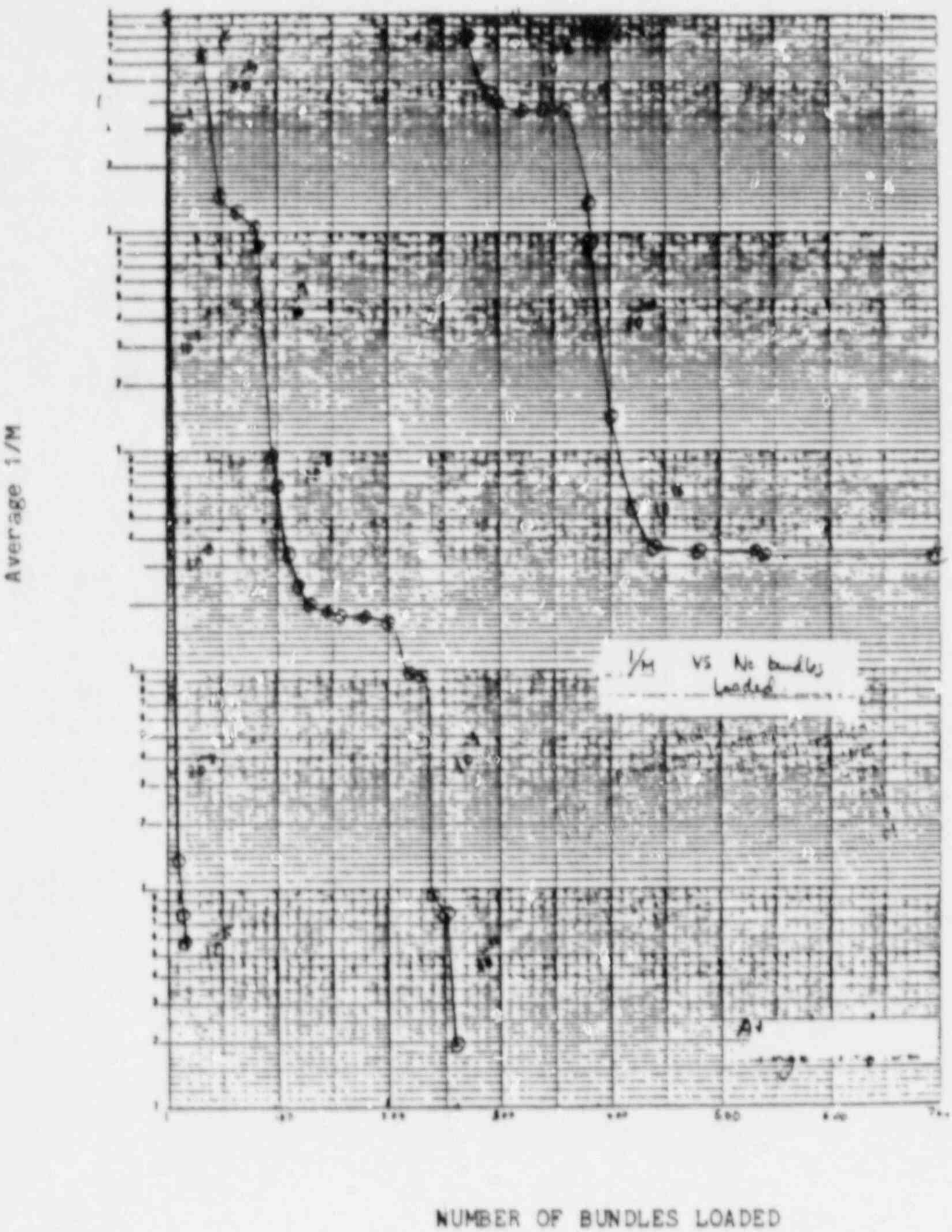


FIGURE 3.3-7
(Page 1 of 2)

FUEL LOCATION VERIFICATION

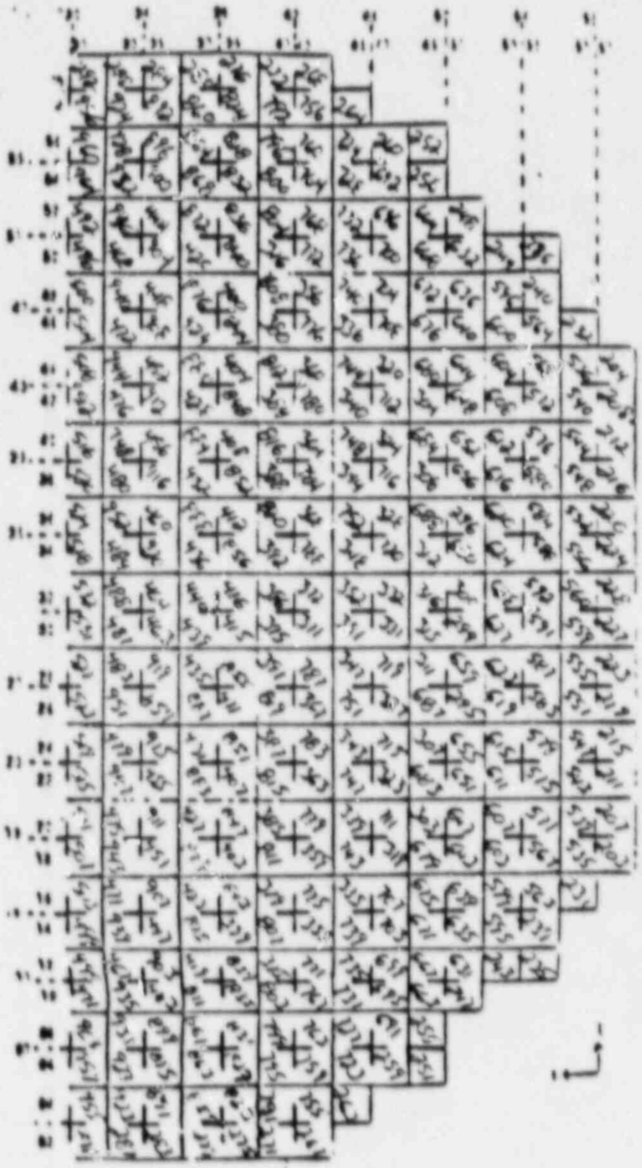
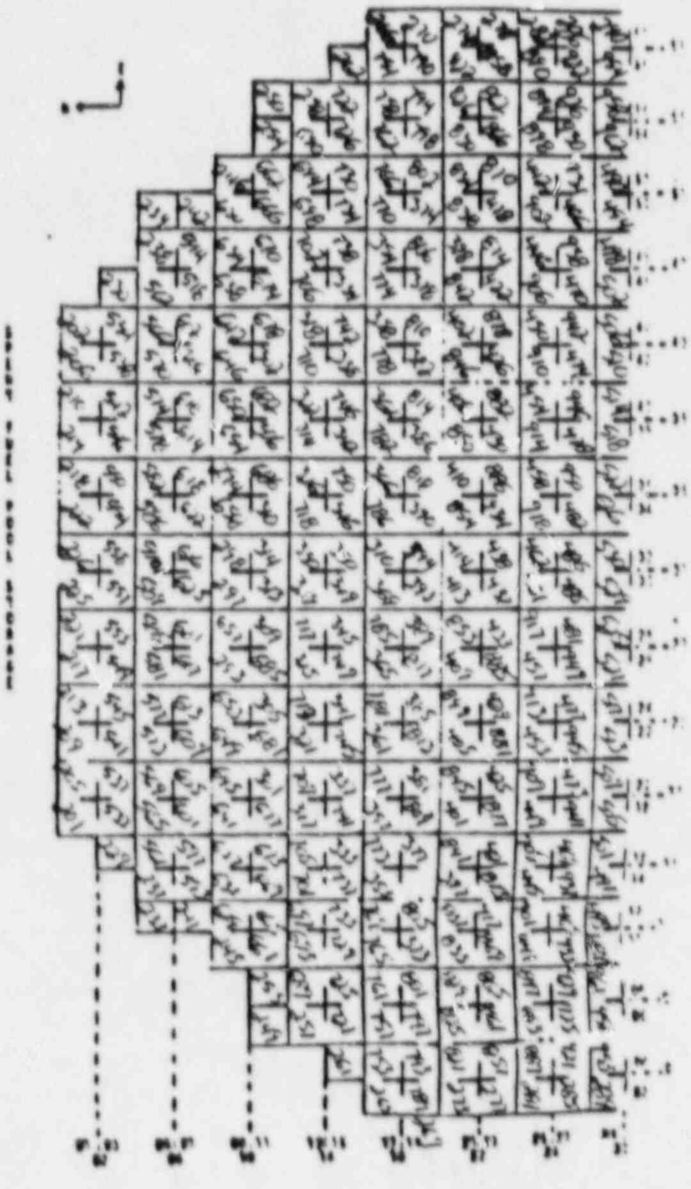


FIGURE 3.3-7
(Page 2 of 2)

FUEL LOCATION VERIFICATION



3.4 Full Core Shutdown Margin

3.4.1 Purpose

The purpose of this test is to assure that the reactor will be subcritical throughout the first cycle with any single control rod fully withdrawn and all other rods fully inserted with the core in its maximum reactivity state.

3.4.2 Criteria

Level 1

The shutdown margin of the fully loaded core with the analytically determined strongest rod withdrawn must be at least 0.38 percent delta k/k plus R (an additional margin for exposure) where R = 0.5 percent delta k/k.

Level 2

Criticality should occur within $\pm$ 1.0 percent delta k/k of the predicted critical.

3.4.3 Results

The fully loaded core was made critical by withdrawing control rods following the B sequence, using the Reduced Notch Worth Procedure. This sequence contained the analytically strongest Rod 06-39, which was fully withdrawn before reaching criticality. Prior to performing the shutdown margin demonstration, as required by Technical Specifications, the shorting links were removed to put the Reactor Protection System in the non-coincidence scram mode.

The point of criticality was demonstrated by withdrawing control rods following the order given in the rod pull sheets until an (approximate) 300 second period was observed with Group 3 Rod 18-51 withdrawn to notch Position 08. Moderator temperature was recorded at 96°F. Later, with moderator temperature still at 96°F, the reactor was then made supercritical by withdrawing Control Rod 10-43 to Position 08. SRM A,B,C and D measurements were taken every 30 seconds for three and one half minutes. Period analysis was performed by fitting the data linearly on a semi-log plot and measuring time to increase one decade from which period was calculated. The average period was determined to be 76.5 seconds.

The shutdown margin of the fully loaded core at 68°F with the analytically strongest rod withdrawn was determined to be 2.72% delta k/k. Level 1 criteria were satisfied since the measured shutdown margin was larger than R + 0.38% = 0.88% delta k/k where R is defined here as the analytical difference in shutdown margin (cold) at the most limiting point in the cycle and Beginning of Life - of the core.

The difference in keff between the theoretical critical configuration and the actual measured critical configuration was found to be 0.28% delta k/k. This satisfies Level 2 criteria since criticality occured within 1% delta k/k of the theoretical critical eigenvalue.

3.5 Control Rod Drive System

3.5.1 Purpose

Each control rod drive (CRD) was tested to measure insert/withdraw and scram times and friction dP levels in the CRD hydraulic system. This was done to demonstrate that the CRD system operates properly over the full range of primary coolant temperatures and pressures.

3.5.2 Criteria

Level 1

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

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

Position Inserted From Fully Withdrawn	Scram Time (Seconds)
46	0.358
36	1.096
26	1.860
6	3.419

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

Position Inserted From Fully Withdrawn	Scram Time (Seconds)
46	0.379
36	1.161
26	1.971
6	3.642

Level 2

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

If the differential pressure variation exceeds 15 psid for a continuous drive-in, a settling test must be performed. In this 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.

3.5.3 Results

Insert/withdraw timing, friction testing, and scram timing were performed on the CRDs at the conditions specified in Figure 3.5-1.

All of the individual control rods were scram time tested, friction tested and insert/withdraw timed during the Open Vessel test condition. Adjustments to some CRDs had to be done in some cases to bring insert/withdraw times into acceptance limits. During the friction testing, no pressure differential measurements exceeded the criteria of 15 psid and no settling tests had to be performed. The four slowest rods in each sequence were also scrammed at reduced accumulator pressure. All test criteria were satisfied.

During Heatup, the four slowest rods in each sequence were scram timed at 600 psig and at 800 psig. Upon reaching rated temperature and pressure conditions, all CRDs were scram timed. The eight slowest rods determined during Open Vessel and Heatup testing were then insert/withdraw timed, friction tested, and scrammed at reduced accumulator pressure. Figure 3.5-2 shows the average scram time of the eight slowest rods, four in each sequence, at various reactor pressures compared to the maximum permissible.

The specific results from our rated pressure testing are as follows:

	Mean Scram Times			
Rod Position	46	36	26	06
Mean Scram Time for all 80 Seq. B rods (sec)	0.302	0.852	1.398	2.501
Mean Scram Time for all 97 Seq. A rods (sec)	0.288	0.802	1.340	2.436
Mean Scram Time for ALL rods, Seq. A and Seq. B (sec) (core average)	0.295	0.826	1.368	2.467
Mean Scram Time of the 3 fastest CRDs in a two-by-two array for ALL rods, Seq. A and Seq. B (core average)	0.325	0.900	1.481	2.655

In conjunction with the planned scram for the Shutdown from Outside the Control Room test performed in Test Condition One, the scram times for the four (4) slowest Sequence "A" control rods were determined. All the scram times were within the acceptance criteria.

FIGURE 3.5-1

CONTROL-ROD-DRIVE SYSTEM TESTS

Test Description	Accumulator Pressure	Preop Tests	Reactor Pressure with Core Loaded psig 0	600	800	rated
Position Indication		All	All			
Normal Stroke Times Insert/Withdraw		All	All			4(a)
Coupling		All	All			
Friction			All			4(a)
Scram	Normal	All	All	4(a)	4(a)	All
Scram	Minimum		4(a)			4(a)
Scram	Zero					4(a)
Scram (scram discharge volume high level)(c)	Normal					
Scram	Normal					4(b)

a. Refers to four CRDs selected for continuous monitoring based on slow normal accumulator pressure scram times, or unusual operating characteristics, at zero reactor presssure. The four selected CRDs must be compatible with rod worth minimizer, RSCS systems, and CRD sequence requirements.

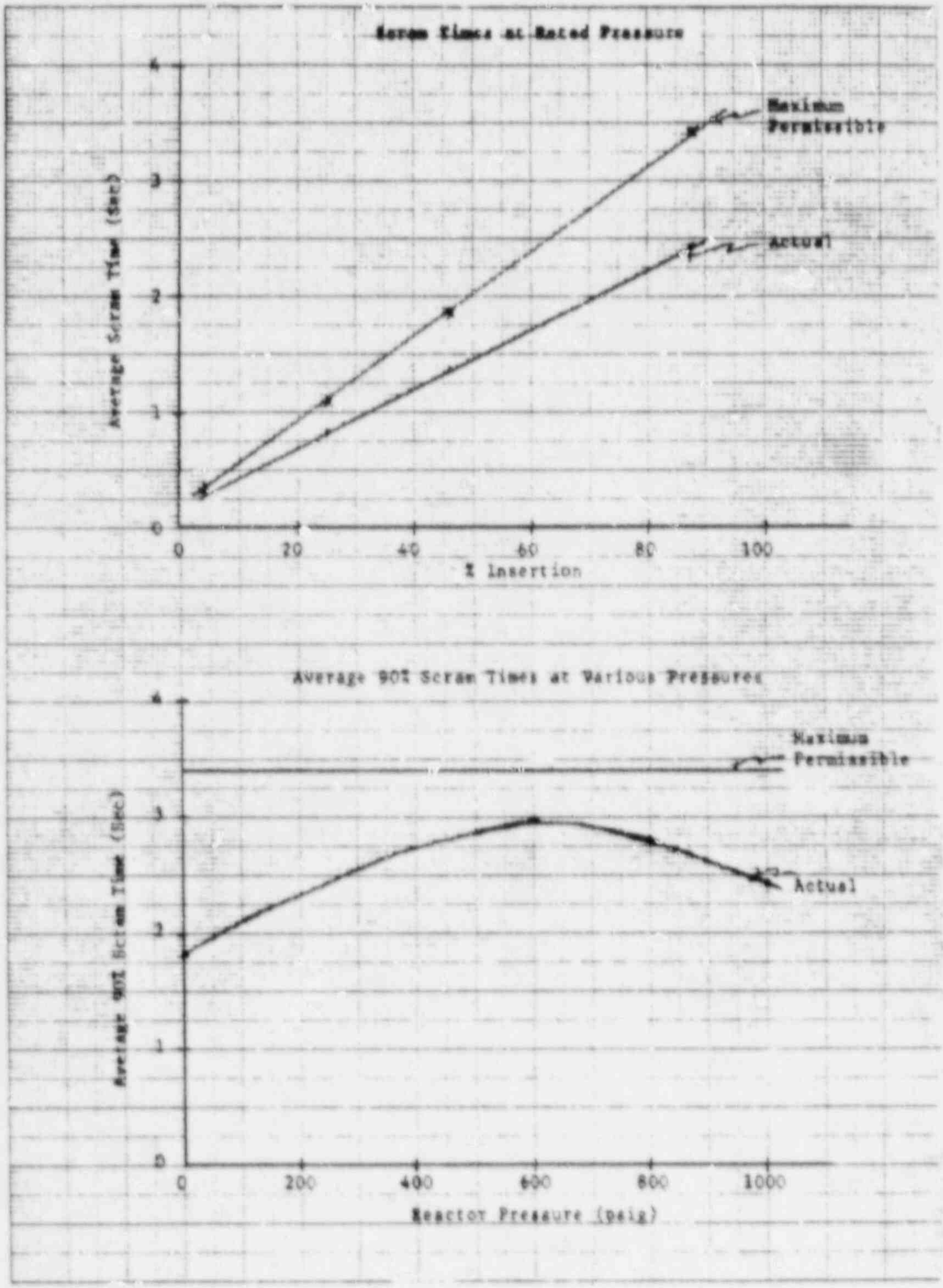
b. Scram times of the four slowest CRDs will be determined at Test Conditions 1 and 6 during planned reactor scrams.

c. The scram discharge volume fill time will be determined at Test Conditions 1 and 6 during planned reactor scrams.

Note: Single CRD scrams should be performed with the charging valve closed (do not ride the charging pump head).

FIGURE 3.5-2

Scram Testing Results



3.6 Source Range Monitor Performance and Control Rod Sequence Exchange

3.6.1 Purpose

The purpose of this test was to demonstrate that the operational sources, source range monitor (SRM) instrumentation, and rod withdrawal sequences provide adequate information to achieve criticality and increase power in a safe and efficient manner. The effect of typical rod movements on reactor power was also determined.

3.6.2 Criteria

Level 1

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

There must be a minimum count rate as defined by Technical Specification on the required operable SRMs.

Level 2

None

3.6.3 Results

Prior to the initial criticality in sequence B, the count-to-noise ratio for SRM (A, B, C and D) were 43, 149, 199 and 49 respectively. These ratios were well above the Level 1 criteria of 2:1. The minimum counts on the SRMs (A, B, C and D) were 20, 15, 40 and 15 cps respectively. These were well above the minimum Level 1 criteria required of 0.7 cps.

SRM readings were also taken periodically during initial criticality in both sequences and IRM readings were obtained during the initial heatup in sequence B. All test criteria were satisfied.

Performance data was gathered during power ascension to 20% in Control Rod Sequence A and Sequence B. At the end of each rod worth minimizer group, APRM, feed flow, and steam flow values were recorded.

3.7 Water Level Measurement

3.7.1 Purpose

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

3.7.2 Criteria

<u>Level 1</u>

None

<u>Level 2</u>

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

3.7.3 Results

Testing of the level instrumentation accuracy showed that scale end point errors when actual drywell temperatures and assumed calibration temperatures were compared were 0.708%, 0.554%, 1.0507% and 0.320% for wide range (Div. I), wide range (Div. II), narrow range (Div. I) and narrow range (Div. II), respectively. The slight Level 2 criteria violation for Div. I narrow range level instrumentation was found acceptable following an evaluation performed by General Electric.

It was previously intended to repeat this test to obtain another set of data with all the drywell coolers in operation. However, based on an evaluation performed by General Electric, the test results are acceptable and no further testing is required.

3.8 IRM Performance

3.8.1 Purpose

The purpose of this test is to adjust the intermediate range monitor system to obtain an optimum overlap with the SRM and APRM systems.

3.8.2 Criteria

Level 1

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

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

Level 2

None

3.8.3 Results

During the initial criticality, all IRMs except IRM D showed response prior to the SRM's reaching 5 x 10^4 cps. IRM D was repaired and tested satisfactorily at a later date. Range 6/7 overlap calibration was also completed for each IRM, except IRM G which was reading erratically. This IRM was replaced and retested successfully.

IRMs G and H underwent repairs during the outage that required retesting of the range 6/7 overlap. After some adjustments, overlap was again sucessfully demonstrated for both.

All APRMs were shown to be onscale prior to any IRM exceeding its rod block setpoint during a plant shutdown in Test Condition One.

It was noted that IRM channels C, E, F and H were not reading one-half decade below their range 9 rod block setpoints. Although Technical Specification verification of overlap was satisfactorily performed in conjunction with Plant Surveillance procedures, the test was reperformed after APRMs were adjusted at a higher power level. Results of this reperformance follow below.

IRM G Range 6/7 Overlap Calibration was reperformed successfully during the plant restart in October 1987. This calibration was necessary due to the replacement of the preamplifier for this IRM.

During the startup of the reactor after the Spring 1988 LLRT Outage, IRM/APRM overlap was checked and adjustments were made to the IRM gains to provide additional margin to the IRM high rod block setpoint. The Level I criteria that each APRM must be on scale before the IRMs exceed their rod block setpoint (< 108/125 on range 10) was satisfied as is shown by the following final settings:

	APRM Reading All GAFs = 1.0	IRM Reading After Adjustment	Range
A	8.3	20/125	10
B	7.9	13/125	10
C	8.0	60/125	10
D	7.9	40/125	10
E	8.1	44/125	10
F	8.6	26/125	10
G		20/125	10
H		29/125	10

Since the IRMs had been adjusted, the SRM/IRM overlap was reverified during a reactor startup in August 1988, and it was shown that each IRM was on scale before the SRMs exceeded their rod block setpoint.

3.9 LPRM Calibration

3.9.1 Purpose

The purpose of this test is to verify LPRM response to flux changes and proper LPRM connection to neutron monitoring electronics and to calibrate the LPRM's to their calculated valves.

3.9.2 Criteria

Level 1

None

Level 2

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

3.9.3 Results

The initial LPRM verification test was performed while the Reactor was at rated pressure in the heatup test condition, in conjunction with scram time testing. Specific control rods were selected to be used for flux response checks based on their proximity to the LPRM strings. The withdrawal of these rods from Position 00 (FULL IN) to Position 48 (FULL OUT) was observed in terms of the LPRM flux response as the rod was withdrawn past each of the four LPRMs for the associated LPRM string. All 172 LPRMs (43 LPRM strings with 4 LPRMs per string) were observed, using Brush Recorders and STARTREC System for flux response. Initially, no flux response was observed on 25 of the 172 LPRMs For the LPRMs that showed flux response. the proper order of the LPRM response (D, C, B, A) was observed.

During supplemental testing, it was found that some LPRM detectors were connected in reverse order and these were corrected. One detector was found damaged and had to be repaired. During Test Condition One all remaining LPRMs were observed to show proper flux response following repair efforts.

An initial LPRM calibration utilizing the Traversing In-Core Probe (TIP) System and the Backup Core Limits Evaluation (BUCLE) program was conducted in Test Condition One. Utilizing TIP traces, local LPRM readings, and heat balance information, a gain adjustment factor (GAF) was determined for each LPRM.

These GAFs were then used to adjust the gains of the LPRMs and a followup test was performed to verify criteria. Due to non-steady state conditions, a total of four full sets of TIP traces were made. Upon completion of the test, a total of 23 LPRMs did not meet the above criteria. The majority of the failures were reasonably close to the criteria, or were in the low power region of the core where criteria can be ignored.

During Test Condition Three relevant portions of REP 54.000.05, LPRM Calibration - Computer Determination, were performed. This entailed performing an OD-1 with a complete set of TIP traces, running a P1 to update the LPRM GAFs, obtaining an OD-10 Option 7 GAF edit, and obtaining the initial LPRM flux amplifier input currents.

All 172 GAFs were reviewed, and it was determined that eight (8) GAF adjustments on the following LPRMs were necessary.

16-33A	48-17A
48-33A	24-25A
16-57A	08-17D
16-09A	32-49D

These eight GAF values were outside of the 0.95 to 1.05 range, and were used to calculate new LPRM flux amplifier input currents.

Following these eight (8) LPRM GAF adjustments, an OD-1 with TIP traces was performed, a P1 was run and an OD-10. Option 7 GAF edit was obtained.

Upon review of the GAF edit only one LPRM GAF was outside of the 1.00 $\pm$ 0.10 required range. LPRM 32-49D was reading 0.0, and was diagnosed as a drifter on the latest P1 edit. IGAF was manually set, a P1 was run, and the LPRM 32-49D had a GAF of 1.0.

Upon completion of REP 54.000.05, all 172 LPRM readings were verified to be within 10 percent of their calculated readings, thus satisfying the Level 2 criteria.

During Test Condition 6, relevant portions of REP 54.000.05, LPRM Calibration - Computer Determination, were performed. This entailed performing an OD-1 with a complete set of TIP traces, running a P1 to update the LPRM GAFs, obtaining an OD-10 Option 7 GAF edit, and obtaining the initial LPRM flux amplifier input currents.

All 172 GAFs were reviewed, and it was determined that 52 GAF adjustments were necessary. LPRM 40-17A is failed low and is bypassed.

These 52 values were outside of the 0.95 to 1.05 range, and were used to calculate new LPRM flux amplifier input currents.

Following these 52 LPRM GAF adjustments, an OD-1 with TIP traces was performed, a P1 was run and an OD-10, Option 7 GAF edit was obtained.

Upon review of the GAF edit, all but three LPRM GAFs were between .91 - 1.11, the required range, thus satisfying the Level 2 criteria. LPRM 40-17A is failed low and bypassed, therefore has a GAF of 0.0. LPRM 08-17A has a GAF of .91 and 24-33D has a GAF of 1.11. Since these GAFs were equal to the criteria value and not outside of the tolerance, no further adjustments were made based on that and the location of these LPRMs not being near critical fuel segments.

This concludes the planned LPRM Calibrations during the Startup Test Program.

3.10 Average Power Range Monitor Calibration

3.10.1 Purpose

The purpose of this test is to calibrate the APRM system.

3.10.2 Criteria

Level 1

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

The APRM channels must be calibrated to read equal to, or greater than the actual core thermal power. Recalibration of the APRM system is not necessary from a safety standpoint if at least two APRM channels per RPS trip circuit have readings greater than or equal to core power. Technical Specification and fuel warranty limits on APRM scram and rod block shall not be exceeded.

Level 2

If the above criteria are satisfied, then the APRM channels will be considered to be reading accurately if they agree with the heat balance to within (+7, -0) percent of rated power.

3.10.3 Results

During heatup, each APRM channel was calibrated to read greater than or equal to a manual calculation of Core Thermal Power based upon a constant heatup rate analysis. The APRM scram trip setpoints were also adjusted to produce a scram at less than 15% of rated power. The Level 1 criteria was satisfied.

An initial APRM calibration was performed during Test Condition One at a Reactor Power of 13.3%. All APRMs were adjusted to read within (+3, -0)% of calculated core thermal power, as determined by a manual heat balance calculation. A second APRM calibration was performed later in Test Condition One when core thermal power (CTP) was determined to be 15.56% as determined from a manual heat balance calculation. APRM gain adjustments were then evaluated and the APRMs adjusted to read 16.0% which is +0.44% above CTP and satisfies the above Level 2 criteria.

During Test Condition Two, following a full core LPRM calibration, each APRM channel was calibrated to a reactor power of 48.4%. This reactor core thermal power was calculated by heat balance, and the six APRMs were calibrated to read within (+7, -0)% of the 48.4% power, thus satisfying Level 2 criteria. This also ensured that the Level 1 criteria requiring that the APRM channels be calibrated to read equal to, or greater than the actual core thermal power was met. Finally, the Scale Factor was determined to be equal to 1.0 since no APRM gain adjustments were imposed. This satisfied the Level 1 criteria requiring that Technical Specifications and fuel warranty limits on APRM scram and rod block shall not be exceeded.

During Test Condition Three, the Process Computer was used to determine a core thermal power of 48.3%. No APRM gain adjustments were imposed which allowed the Scale Factor to be set equal to 1.0. Therefore, the six APRM desired readings were determined to be 48.3%.

The six APRM readings taken locally at Relay Room Panel H11-P608 revealed that the absolute differences between the desired and current APRM readings were within (+2%, -0%) except for APRM B which initially read 48.2%. Therefore, APRM B was adjusted by changing the setting of the R16 gain potentiometer to read greater than 48.3% CTP.

The final APRM readings at that power were as follows:

APRM A 50.0	APRM D 49.2
APRM B 48.8	APRM E 48.6
APRM C 49.0	APRM F 49.2

The scale Factor was determined to be equal to 1.0 and all the APRMs are reading greater than core thermal power. This satisfied the Level 1 criteria.

As seen by the data above, the Level 2 criteria is also satisfied.

During Test Condition Five, the Process Computer was used to determine the core thermal power of 71.7%. No APRM gain adjustments were imposed which allowed the Scale Factor to be set equal to 1.0 and, therefore, the desired APRM readings were determined to be 71.7%. The actual APRM readings taken locally at the Relay Room panel H11-P608 were between 70.0%

to 71% of rated power. All six (6) APRMs were adjusted to read greater than 71.7% CTP by changing the setting of the R16 gain potentiometer.

The final APRM readings at that power were as follows:

APRM A 72.0%	APRM D 72.0%
APRM B 72.0%	APRM E 72.0%
APRM C 72.0%	APRM F 72.0%

Since the Scale Factor was determined to be equal to 1.0 and all APRMs are reading greater than core thermal power, the Level 1 criteria is satisfied. The Level 2 criteria is also satisfied by the above readings.

During Test Condition 6, the Process Computer was used to determine the core thermal power of 96.8%. No APRM gain adjustments were imposed which allowed the Scale Factor to be set equal to 1.0, and therefore the desired APRM readings were determined to be 96.8%. The actual APRM readings taken locally at the Relay Room Panel H11-P608 were acceptable except APRM "E" which was adjusted from 96.0% to 97.0% by changing the setting of the R16 gain potentiometer.

The final APRM readings at that power were as follows:

APRM A 97.0%	APRM D 97.5%
APRM B 97.0%	APRM E 97.0%
APRM C 97.0%	APRM F 97.5%

Since the Scale Factor was determined to be 1.0 and all APRMs were reading greater than the calculated core thermal power of 96.8%, both the Level I and Level II criteria are satisfied.

3.11 Process Computer

3.11.1 Purpose

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

3.11.2 Criteria

<u>Level 1</u>

None

<u>Level 2</u>

Programs OD-1, P1, and OD-6 are considered operational when the MCPR, the maximum LHGR, the maximum APLHGR, and the LPRM gain adjustment factors calculated by BUCLE and the process computer agree with the tolerances specified in the FSAR.

Remaining programs will be considered operational on the successful completion of the static and dynamic testing.

3.11.3 Results

The TIP System consists of five identical probes used to measure and record the axial neutron flux profile at 43 radial core locations. The recorded information is used by the Process Computer to calibrate the fixed in-core Local Power Range Monitors. Each probe is driven into and withdrawn from the core by its associated drive mechanism.

In order to operate automatically, the TIP drive control units must be programmed with the probe position at top and bottom of the core. These top and bottom limits are programmed and verified in the TIP cold alignment. This portion of the test was performed successfully by hand-cranking the TIPS to the top of the core and setting the core limits based on the resulting position readings.

In order to follow and read data from the TIP machines, the Process Computer must receive position information and flux signals from the TIP System. This interface is tested in the Static System Test Case by running the TIP machines in various configurations and verifying the proper responses on the Process Computer.

The Static System Test Case had two objectives: verification of the program logic and checkout of the TIP interface. The first objective was successfully achieved, but the TIP interface checkout was unsuccessful due to a problem with the TIP System that resulted in the loss of TIP position indication. This original position indication problem was repaired.

As part of the Test Condition One testing, the TIP top and bottom core limits were reverified under hot conditions, and the TIP interface with the X-Y plotter was also verified to function properly. Following repairs to TIP "C" ball valve, a process computer interface problem, and TIP "B" Logic, a successful OD-1 was obtained from the process computer. It was noted that a three (3) second delay was occurring between X-Y plotter traces and the machine normalized, full power adjusted TIP array. This problem was corrected prior to the OD-1 portion of the Dynamic System Test Case.

The Dynamic System Test Case was performed during steady state conditions with reactor power at approximatley 20%. The testing included:

1. Verification of the Computer Outage Recovery Monitor (CORM) to initialize necessary variables and exposure arrays as part of initial plant computer startup and to allow for controlled set of data in further system testing.

2. Verification that all required plant sensors for NSS programs are being properly scanned.

3. Verification of the heat balance subroutine used by OD-3 and P1 by comparing it with a manually calculated heat balance.

4. Performing an LPRM calibration to verify the operation of OD-1 prior to the verification of thermal limit calculations.

5. Verification of thermal limits calculations and core power distribution.

6. Verification of the exposure updating programs P4 (10 Minute Core Energy Increment), P1 (Periodic Core Evaluation), P2 (Daily Core Performance Summary) and P3 (Monthly Core Performance Summary).

7. Verifying key variable memory locations and performing manual calculations to verify the remaining NSS software at steady state operation and symmetric rod pattern.

Thermal limit and LPRM calibration factor calculations were verified in conjunction with the DSTC. The verification was performed by taking the same data that is input to the P1 program, for its calculation, and inputting it into an approved offline computer program (Backup Core Limits Evaluation (BUCLE), which also performs the P1 calculations. The resulting thermal limits and LPRM calibration factors were verified against the criteria. In all instances the results were in the same fuel assembly and the results are as follows:

Parameter	Location	P1 Results	Bucle Results	% Error
Max LHGR	33-52-13	3.78	3.78	0%
Max MAPLHGR	27-10-13	3.30	3.30	0%
Min CPR	27-10	3.877	3.876	.02%

$$\% \text{ Error} = \frac{\text{P1 Result - Bucle Result}}{\text{P1 Result}} * 100\%$$

The Local Power Range Monitor (LPRM) gain adjustment factors calculated by BUCLE and the process computer were verified to agree within 2%.

Programs OD-1, P1, OD-6 and the remaining NSS programs were considered operational upon the satisfactory performance of this procedure.

During Test Condition Three, a Process Computer - BUCLE Comparison was performed at steady-state conditions at 48.4% reactor power and 93% core flow. With P1 blocked, the following list of process computer edits were obtained and compared to the respective BUCLE edits:

RCAL
GAF
W
PBUN
EBUN
NSS Core Performance Log
Thermal Data in Fuel Assembly IX, JY
The 12 Bundles Closest to CPR Limits
The 12 Highest Ratios of a Bundle MAPLHGR to its LIMLHGR
Target Exposure and Power Data

Each process computer value was verified to agree with each BUCLE value to within ± 2% (FSAR tolerances).

An MCPR of 2.819 was calculated by P1, and an MCPR of 2.821 was calculated by BUCLEs PINEWRP, each for bundle 17-18. These values are within 0.07% of each other, therefore satisfying the Level 2 criteria.

An MLHGR of 5.76 was calculated by P1, and an MLHGR of 5.75 was calculated by BUCLEs PINEWRP, each for bundle 17-26-11. These values are within 0.17% of each other, therefore satisfying the Level 2 criteria.

An MAPLHGR of 5.05 was calculated for bundle 17-26-11 by both P1 and BUCLEs PINEWRP. Therefore, the Level 2 criteria was satisfied.

The process computer OD-10, Option 7 GAF edit was compared to the BUCLEs EDITMAP GAF array. The values were verified to agree within ± 2%, therefore satisfying the Level 2 criteria.

During Test Condition Three, the Process Computer - Power Change Verification was performed to demonstrate the performance of the OD-4 and OD-5 programs during power changes.

The test was performed in three sections where two of those sections dealt with the performance of OD-4 and OD-5 programs after a large power change (> 20% of rated power) from either recirculation flow alone or control rods alone, as compared to P1 program. One section dealt with comparison of symmetric and non-symmetric P1s.

Symmetric P1 and Non-Symmetric P1 Comparison

Once steady-state conditions were established, P1 program was run with the symmetry flag set to reflect mirror symmetric conditions. After P1 run was completed, the Bundle Power array (PBUN) was obtained from OD-10 Option 22. Next, the symmetry flag was set to 3 (asymmetric) and P1 run again and MFLCPR and MFLPD were compared against the last P1 (symmetric) run. Also the Bundle Powers were compared against the last edit and all these parameters were observed to be within 5% rms. Once all the required edits were obtained, the core symmetry flag was restored to mirror symmetry.

Large Power Change (using flow only)

This section was performed by establishing steady-state conditions, running OD-4s for at least 10 different rods, running OD-5, running P1 and then blocking P1. With P1 blocked, power was raised by flow alone, at least 20% of rated power and once steady-state conditions were established at the new power level, OD-4s were run for the same rods as before, OD-5 was run and then P1 was run. OD-4 and OD-5 edits were then compared against P1 before and after the power change and the results were as follows:

	OD-5 vs P1 (% diff)		OD-4 vs P1 (% diff)	
Power (%)	MFLCPR	MFLPD	MFLCPR	MFLPD
53.49	0.348	0.05	0.632	0.202
73.57	0.265	0.141	1.550	0.572

The overall power increase was 20.08% of rated power and the "SML NSS" video alarm flag was observed during power ramp. OD-18 Verification Data was completed and the core flow (WT) was verified to be between WLO and WHI. OD-19 Verification Data was completed and the APRM A value was verified to be within the proper power band. The high and low powers pertaining to the power band from Data Class 15 were compared to the OD-3 edit and were found to be within 1% of each other.

Large Power Change (using control rods only)

This portion of the test was performed after steady-state conditions were established at 26.08% power and all required edits (OD-4s, OD-5 and P1) were obtained before blocking P1. Power was increased by rods alone to 48% power to provide an overall power change of 21.92% of rated. Once steady-state conditions were reached, the required edits (OD-4s, OD-5 and P1) were obtained for comparison. The results were as follows:

	OD-5 vs P1 (% diff)		OD-4 vs P1 (% diff)	
Power (%)	MFLCPR	MFLPD	MFLCPR	MFLPD
26.08	0.1	0.167	0.13	0.13
48.00	1.63	4.57	1.82	4.83

At the conclusion of this segment of the test, P1 was restored to normal operation.

Based on the above successful testing, OD-4 and OD-5 programs are considered operational, which satisfies the Level 2 criteria.

During Test Condition Three, with the reactor operating at a steady-state power of approximately 49%, the Process Computer PCIOMR (Preconditioning Interim Operating Management Recommendation) Verification was performed to verify the correct operation of the OD-11 program.

There are fifteen options associated with the OD-11 program: Options 1 through 11, Options 66, 77, 88 and 99. OD-11 calculates and edits data pertinent to the monitoring and applications of the PCIOMR. This program has six options (1 through 5 and 11) which edit information concerning the present power distribution and the stored preconditioned envelope, two options (6 and 7) concerning predicted power increases due to control rod withdrawals, four options (8, 77, 88 and 99) which permit monitoring of the preconditioning ramp rate on a model basis, and two options (9 and 10) which allow operators to establish and maintain the preconditioned envelope. In addition, Option 66 is available for automatic editing or suppression of the Option 3 and 6 edits. The verification testing was performed as follows:

Data Interrogation

The first step was to save PCIKON, KWTH, IEXPC and IPC arrays on magnetic tape. During the course of the test, several of the arrays were changed to force messages on the alarm typer or other edits which would facilitate checkout of the OD-11 software. At the end of the test, this magnetic tape was used to restore these arrays back to their original values.

Since this test had originally been scheduled for Test Condition Five and was now being conducted at a relatively low power level in Test Condition Three, the threshold nodal power was reduced to 5 kW/ft (from 14 kW/ft) so that there would be some nodes in the core that would exceed this artificially low threshold value.

Nodal power edits on OD-11 Option 4 were checked against hand calculation for selected controlled and uncontrolled nodes and were found to be in general agreement. Nodal exposure and control rod positions were satisfactorily compared against OD-10 edits. Nodal powers, pre-conditioned power values, the envelope power values and exposure edits on OD-11 Options 1, 2, 3, 4 and 5 were compared for consistency.

Envelope Updating

OD-11 Option 1 was run to obtain the edit of Nodes versus DELTA-E intervals. OD-11 Option 9 was run for a selected non-zero DELTA-E interval and verified to properly update the nodes in that interval. Next, OD-11 Option 2 was run to get an edit of P-PC versus number of nodes in each interval. Since the nodal power for all nodes was below the pre-set minimum pre-conditioned value, process computer parameter PF was increased for a selected node to provide a situation for which the nodal power exceeded its pre-conditioned value and would be a candidate for envelope updating. OD-11 Option 10 was run and this selected node was verified to be properly updated before PF was restored to its original value.

Predictive Overpower Model

OD-11 Option 6 was run to get control rod notch positions and rod withdrawal permissives then compared against OD-7 and OD-11 Option 5 and Option 7 for consistency. PCIKON (1) was lowered from 7.945 to 5.5 to create a situation where rod withdrawal would not be permitted on OD-11 Option 6. OD-11 Option 7 was run to verify that the predictive model was working properly before PCIKON (1) was restored to its orignal value.

Automatic Alarm and Initiation

This segment of the test involved checking out OD-11 Option 66 which turns on (or off) the OD-11 Option 3 and Option 6 edits. These options (3 and 6) only run if the fraction of feedwater flow is greater than PCIKON (3). This portion was checked by actually reducing PCIKON (3) to a value below the fraction of feedwater flow and either obtaining the Option 3 and 6 edits or verifying the edits were suppressed.

The Feedwater Alarm setpoint calculated by the OD-11 program was verified against the hand calculations.

PCIKON (5) was changed from 0.3 to -10 to create a situation which would provide an overpower alarm. Once this was verified, PCIKON (5) was changed to 10 which produced a message that the overpower alarm had cleared before the original value of PCIKON (5) was restored. Control rods were maneuvered to check out alarms associated with overpower situations due to rod pulls. PCIKON (4), and PCIKON (7) and WTFACT array was changed to create situations which would be identified by the OD-11 program as a potential overpower condition resulting in an alarm and/or P1 initiation. The effect of KNOT variable was checked by setting it to -2 and observing that P1 aborted after initiation due to "asymmetry" as expected. It was also verified that for a significant change in Core Thermal power, OD-11 Option 3 and 6 edits will be printed after P1 even if these edits are turned off by Option 66.

Static Test of OD-11 Ramp Monitor

The first portion of the OD-11 Ramp Monitor check was to determine the proper value of threshold power, PCIKON (1), PCIKON (3) and IPC array which would enable the selection of five nodes representing margin to envelope (P - max (PC, KWTH)) in the following five segments:

Less than -.055 kW/ft
Greater than -.055 kW/ft but less than 0.0
Between 0.0 and 0.2 kW/ft
Slightly above 0.2 kW/ft
Largest P - max (PC, KWTH) value

Once these five nodes were selected, OD-11 Option 8 was initiated by running OD-11 Option 77 and OD-11 Option 88. The subsequent OD-11 Option 8 and Option 3 edits were compared for consistency. The parameters that were checked for consistency/accuracy were the number of overpower nodes, margin to envelope, peak nodal power, ramp rate, nodal exposure and that the PC value falls into the proper segment for ramping. Upon running another P1 it was verified that all the previously flagged overpower nodes were properly initialized. Of the five selected nodes, four of the nodes were observed to behave predictably but the P-PC value for the highest power node was somewhat lower than expected. The calculated P-PC was 1.2 which was outside the 1.32 $\pm$ .055 kW/ft range. Upon further investigation it was observed that the original setup to perform this portion of the test was based on the peak nodal power of 6.66 kW/ft. However, after approximately five hours when this portion of the test was being performed, the maximum nodal power was 6.56 kW/ft which is 0.1 kW/ft lower than the original nodal power and acceptably explains this discrepancy.

The Process Computer overpower alarm setpoint based on feedwater flow was checked against hand calculations. The alarm setpoint was reduced to check the program that provides the "potential overpower alarm" prior to restoring the setpoint back to its original value.

The pre-conditioned power values and the ramp rates were successfully checked for accuracy against hand calculations. The OD-11 Option 8 ramp monitoring program was observed to properly account for step change in nodal power due to OD-2 runs and the nodal PC values from the OD-11 Option 4 edits were successfully checked against hand calculations. Finally, the OD-11 Option 8 auto termination feature based on low ramp rate was verified to be properly functioning before the PCIKON, KWTH, IEXPC and IPC arrays were restored to their original values.

Based on the above successful testing, the PCIOMR program, OD-11, is considered operational which satisfies the Level 2 criteria.

During Test Condition 6, a Process Computer - BUCLE Comparison was performed at steady-state conditions at 96.8% reactor power and 97.7% core flow. With P1 blocked, the following list of process computer edits were obtained and compared to the respective BUCLE edits:

- RCAL
- GAF
- W
- PBUN
- EBUN
- NSS Core Performance Log
- Thermal Data in Fuel Assembly IX, JY
- The 12 Bundles Closest to CPR Limits
- The 12 Highest Ratios of a Bundle MAPLHGR to its LIMLHGR
- Target Exposure and Power Data

Each process computer value was verified to agree with each BUCLE value to within ± 2%.

An MCPR of 1.483 was calculated by P1, and an MCPR of 1.479 was calculated by BUCLEs P1NEWRP, each for bundle 15-48. These values are within 0.27% of each other, therefore satisfying the Level 2 criteria.

An MLHGR of 11.63 was calculated by P1, and an MLHGR of 11.67 was calculated by BUCLEs P1NEWRP, each for bundle 43-46-4. These values are within 0.34% of each other, therefore satisfying the Level 2 criteria.

An MAPLHGR of 10.13 was calculated by P1, and an MAPLHGR of 10.12 was calculated by BUCLEs P1NEWRP, each for bundle 17-50-8. These values are within 0.09% of each other, therefore satisfying the Level 2 criteria.

The process computer OD-10, Option 7 GAF edit was compared to the BUCLEs EDITMAP GAF array. The values were verified to agree within $\pm$ 2%, therefore satisfying the Level 2 criteria.

The following discrepancies were noted:

When comparing the value of Fuel Segment Quality (QUAL), Segment Void Fraction (VF) and Segment Power (POW) for the incore limiting bundle used for the comparison (43-46), the BUCLE value and the OD-6, Option 2 value differed by greater than 2% for several nodes.

Parameter	Node	OD-6, Opt. 2	BUCLE	Error
POW	24	0.0735	0.072	2.04%
QUAL	3	-0.0076	-0.0073	3.95%
QUAL	4	0.0073	0.0077	5.20%
VF	2	0.035	0.036	2.78%

Since the computed values were very small, the small relative differences resulted in large percentage differences. Since the actual absolute differences are small, these discrepancies are not considered significant.

When comparing the values from the LPRM RCAL Array, the BUCLE value and the OD-10 value for LPRMs 16-57-D and 56-33-D differed by 2.57% and 2.19%, respectively, thereby exceeding the requirement for agreement to within 2%. The raw LPRM readings manually inputted into the BUCLE Program were rounded values (18% vice 18.45% for 16-57-D and

23% vice 23.5% for 56-33-D). As a result, the BUCLE Program yielded RCAL values which exceeded the 2% requirement. Had the inputs been made using at least the first decimal, the BUCLE RCAL Array values would have been very close to the OD-10 values (to much less than a 2% difference) and, therefore, this deviation is acceptable.

This concludes the series of tests performed on the Process Computer during the Startup Test Program.

3.12 RCIC System

3.12.1 Purpose

The purpose of this test is to verify the proper operation of the RCIC system over its expected operating pressure range.

3.12.2 Criteria

Level 1

The average pump discharge flow must be equal to or greater than the 100-percent-rated value after 50 seconds have elapsed from initiation on all auto starts at any reactor pressure between 150 psig and rated. With pump discharge at any pressure between 250 psig and 100 psi above rated pressure, the required flow is 600 gpm. (The 100 psi is a conservatively high value for line losses. The measured value may be used if available).

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

Level 2

To provide a margin on the overspeed trip and isolation, the first and subsequent speed peaks on the transient start shall not exceed the rated speed of the RCIC turbine by more than 5 percent.

For small speed or flow changes in either manual or automatic mode, the decay ratio of each recorded RCIC system variable must be less than 0.25.

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

The delta P switch for the RCIC steam supply line high-flow isolation trip shall be adjusted to actuate at 300 percent of the maximum required steady state flow, with the Reactor assumed to be near the pressure for main relief valve actuation.

3.12.3 **Results**

During the Heatup Test Condition, the RCIC pump suction and discharge was lined-up in a closed loop with the condensate storage tank. The system was subjected to negative and positive 10% step changes in flow at system flows of 600 gpm and 270 gpm using both a step generator and the RCIC flow controller. Minimum flow data was also taken at a speed of 2000 rpm and a RCIC quickstart was performed.

The RCIC system was able to supply 600 gpm at a discharge pressure of 1140 psig in 35 seconds when automatically started using 940 psig steam from the vessel. The K72 time delay relay was set down from 10 sec to 5 sec to prevent the RCIC turbine from coasting down excessively before the opening of the Steam Admission Valve, thus reducing the experienced transient. The RCIC turbine did not isolate or trip during the auto and maunal starts. In addition, there were no RCIC turbine speed peaks or oscillations in RCIC system variables in the transient testing.

The RCIC system was also subjected to an extended run at rated flow conditions. RCIC performed satisfactorily with all system temperatures stabilized below alarm levels and a negative pressure maintained on the gland seal condenser system.

All Level 1 and Level 2 criteria were satisfied except the RCIC steam supply high flow isolation trip setting. During the Outage for the replacement of the Main Steam Bypass Lines, engineering modifications to the instrument lines were completed that were expected to solve the problems found with the instrument sensing lines.

Upon recommencing Heatup in August of 1986, the RCIC EGM module was found malfunctioning and was replaced. Because of this and the instrument line modifications discussed above, the RCIC system was subjected to further testing including 10% positive and negative step changes in both speed and flow, and a quickstart.

With the reactor pressure at 955 psig, the RCIC system was able to supply 600 gpm at a discharge pressure of 1143 psig in 33 seconds. All Level 1 and Level 2 criteria were satisfied except the turbine gland seal system verification and the RCIC steam supply high flow isolation trip setting.

Due to a failure of the RCIC Barometric Condenser Vacuum Pump, data did not show the existance of a vacuum on the vacuum tank as required by the test criteria. Subsequent work on the Barometric Condenser Pump corrected the problems and it was retested successfully.

Data was also taken during this test to determine the actual 300% value for the RCIC steam supply line high flow isolation trip setpoint. However, the trip setpoints were not adjusted to these settings, but are being left at the current trip setpoints given in the Technical Specifications. The current settings as specified by the Technical Specification are set conservatively compared to the value calculated by the performance of this testing, yet provide ample margin to prevent spurious RCIC isolations on system automatic initiations.

During Test Condition One, RCIC system testing consisted of a hot manual vessel injection, two (2) cold quick start vessel injections, a 150 psig CST to CST run, a 150 psig vessel injection, and a CST to CST run at rated pressure for baseline data. The only problem of any significance during any of these runs was a turbine speed peak 29 rpm above the Level 2 limit of 4725 rpm, which occurred during the initial hot manual vessel injection. Minor adjustments were made to the RCIC control circuitry and the problem did not reoccur in subsequent tests.

For the hot manual vessel injection, with the reactor supplying steam at a pressure of 915 psig, the RCIC pump delivered a flowrate of $\geq$ 600 gpm at a discharge pressure of 965 psig in 28.4 seconds. As discussed above, the turbine reached a maximum speed peak of 4764 rpm, which exceeded the Level 2 criteria. Based on data taken in conjunction with this test, it was determined that the actual line loss value for the RCIC system was 50 psid.

For the first cold vessel injection, with the reactor supplying steam at a pressure of 918 psig, the RCIC pump delivered a flowrate of $\geq$ 600 gpm at a discharge pressure of 970 psig in 28.5 seconds. The maximum speed peak was 4686 rpm for the RCIC turbine.

For the second cold vessel injection, with the reactor supplying steam at a pressure of 910 psig, the RCIC pump delivered a flowrate of ≥ 600 gpm at a discharge pressure of 970 psig in 29.2 seconds, with a maximum speed peak of 4488 rpm.

During the 150 psig CST to CST run, with the reactor supplying steam at a pressure at 165 psig, the RCIC pump delivered a flowrate of ≥ 600 gpm at a discharge pressure of 271 psig in 22.0 seconds, with a maximum speed peak of 2818.

During the rated reactor pressure CST to CST run, with the reactor supplying steam at a pressure of 920 psig, the RCIC pump delivered a flowrate of ≥ 600 gpm at a discharge pressure of 1095 psig in 29 seconds, with no discernable speed peak as the turbine ramped up smoothly to a final speed of 4500 rpm.

The 150 psig vessel injection was conducted with the reactor supplying steam at 160 psig. The system reached ≥ 600 gpm in an elapsed time of 21.5 seconds at a discharge pressure of 215 psig, with a maximum speed peak of 2641 rpm.

RCIC testing was successfully completed with a 150 psig cold CST to CST baseline data test. With the reactor supplying steam at a pressure of 165 psig, the RCIC pump delivered a flowrate of ≥ 600 gpm at a discharge pressure of 360 psig in 19.5 seconds, with an initial speed peak of 1418 rpm followed by a smooth ramp to a final maximum speed of 2766 rpm.

Subsequent to the completion of testing, data gathered to determine the RCIC steam supply line high flow isolation trip setpoint was evaluated further by Nuclear Engineering. The results of this evaluation validate the flow equation used to determine the initial trip setpoint as presently listed in Tech. Spec. Table 3.3.2-2 and, therefore, no adjustment is necessary. This evaluation is detailed in Design Calculation #4595 Revision B.

3.13 HPCI System

NOTE: As discussed in memorandum NRC-87-0179, "Initial Test Program Changes", dated September 30, 1987, from B. R. Sylvia to U.S. Nuclear Regulatory Commission, Washington, D.C., the Level 1 criteria for system response time to rated flow has been modified to agree with Plant Technical Specifications. The Level 2 criteria for margin to overspeed trip has been modified to reflect the control system hydraulic modifications which improved the stop and control valve response to a quick start.

3.13.1 Purpose

The purpose of this test is to verify proper operation of the High Pressure Coolant Injection (HPCI) system over its expected operating pressure range.

3.13.2 Criteria

Level 1

The average pump discharge flow must be equal to or greater than the 100-percent-rated value with a system response time of less than or equal to 30 seconds as defined in Technical Specifications at any reactor pressure between 150 psig and rated. With pump discharge at any pressure between 250 psig and 100 psi above rated pressure, the flow should be at least 5000 gpm. (The 100 psi is a conservatively high value for line losses. The measured value may be used if available).

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

Level 2

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

The delta P switch for the HPCI steam supply line high flow isolation trip shall be adjusted to actuate at 300 percent of the maximum required steady-state flow with the reactor assumed to be near main relief valve actuation pressure.

For small speed or flow changes in either manual or automatic mode, the decay ratio of each recorded HPCI system variable must be less than 0.25.

The margin to avoid the overspeed trip shall be at least 10% of the nominal overspeed trip setpoint of 5000 rpm, during all auto starts of the HPCI system.

3.13.3 Results

Following setup of the control system, initial coupled turbine performance runs were performed on the HPCI system during initial heatup. Dynamic stability checks were conducted with the HPCI pump suction and discharge lined-up in a closed loop to the CST. Flow step changes of ± 500 gpm were introduced by the flow controller in automatic, with HPCI system flows at 5000 gpm and 2700 gpm.

During automatic initiation testing of HPCI, a discharge flow of 5000 gpm was reached in 23.4 seconds. Twenty-five seconds after the automatic initiation HPCI flow had reached 5310 gpm at a discharge pressure of 1140 psig, 190 psig greater than reactor pressure. HPCI did not trip or isolate during any of the manual or automatic starts. Adequate margin was demonstrated on turbine speed peaks and oscillations of system variables. An extended run was performed in which system temperatures stabilized at acceptable levels and the gland seal system performed satisfactorily.

All Level 1 and Level 2 criteria were satisfied except for the steam supply isolation trip setpoint. During the extended Outage which started in the Fall of 1985, engineering modifications were completed that were expected to correct the problems experienced with the instrument sensing lines. Because of this modification, the EGR bypass line installation, and other modifications that were made to the HPCI System during the Outage, the Startup Tests were repeated for this system when the plant restarted in August of 1986.

Dynamic Stability checks were again completed using 500 gpm step changes introduced in both manual and automatic flow control modes with the HPCI System operating in a closed loop to the CST. Level 2 criteria was exceeded when HPCI System flow had a measured decay ratio of 0.28 resulting from a mid-flow speed decrease step change in the manual mode. This is currently considered to be acceptable but will be examined closely in HPCI testing at higher test conditions.

During a HPCI automatic initiation in the CST closed loop lineup, a HPCI System flow of 5000 gpm was achieved in 21.2 seconds. Twenty-five seconds after the automatic initiation occurred, HPCI flow was 5003 gpm at 1185 psig pump discharge pressure, 225 psig greater than the 960 psig reactor pressure.

Data was also taken during this test to determine the actual 300% value for the HPCI steam supply line high flow isolation trip setpoint. However, the trip setpoints were not adjusted to these settings, but are being left at the current trip setpoints given in Technical Specifications. The current isolation settings as specified in Technical Specifications are considered acceptable as they are conservative yet provide ample margin to prevent spurious HPCI isolations on system automatic initiations.

All other Level 1 and 2 criteria were met.

During retesting of HPCI in September of 1986, a sluggish response was noted in the HPCI control valve. In an attempt to make the HPCI System more responsive, it was decided to replace the EGR in the hydraulic portion of the HPCI control system. As a result, the 1000 psig hot CST injection was repeated to verify proper control system operation. HPCI was successfully quick started and HPCI discharge flow reached the 100-percent-rated value (5000 gpm) in 21.0 seconds. Following the automatic initiation, HPCI flow leveled out at 5100 gpm with a discharge pressure of 1190 psig. The initial speed peak was 2134 rpm and the maximum peak was 4114 rpm. All other Level 1 and Level 2 criteria were met.

In June of 1987, following the February 1987 turbine rotor replacement (reference LER 87-006-00) and prior to the scheduled Test Condition Three HPCI test sequence, tuning of the HPCI governor control system was performed. During this tuning, a RCIC turbine trip occurred on low suction pressure when the HPCI turbine was Quick Started. To prevent recurrence, HPCI and RCIC suctions were aligned to different sources.

During the initial vessel injection attempt, the HPCI turbine underwent a total of five overspeed trip/reset actions, violating Level 1 criteria, prior to being secured. Two diagnostic CST to CST runs determined the overspeed conditions were minimum flow related, and consequently, the second vessel injection attempt was to provide an immediate flowpath to the vessel by manually opening the injection valve immediately following the Quick Start.

The second vessel injection attempt was aborted when a logic problem caused the injection valve to cycle closed, creating a water hammer damaging the suction relief valve, suction pressure instrumentation and the flow transmitter. In addition, the RGSC was found to be defective.

Following repairs to the suction relief valve and replacement/recalibration of the RGSC, suction and flow instrumentation, retuning was performed.

Once the governor control system had been retuned, a third vessel injection attempt and dynamic stability checks were performed, this time successfully. Time to rated flow was 25.2 seconds, exceeding the Level 1 criteria of 25 seconds. The initial speed peak was 1096 rpm and the maximum speed peak was at 3991 rpm. All speed and flow step changes exhibited acceptable decay ratios. At no time did the gland seal condenser system allow steam leakage to atmosphere.

Following the required 72 hour cooldown period, a cold vessel injection attempt resulted in two overspeed trip/reset actions, a Level 1 criteria violation.

Per GE recommendation, the control valve hydraulic assist valve was fully closed and retuning was performed. After the retuning effort, another HPCI vessel injection and dynamic stability checks were performed, resulting in a time to rated flow of 22.3 seconds with initial and maximum speek peaks of 1222 and 4303 rpm, respectively. This exceeded the Level 2 criteria for a maximum speed peak of 4200 rpm. Several speed and flow steps at mid flow conditions failed to achieve Level 2 quarter damping criteria. At no time did the gland seal condenser system allow steam leakage to atmosphere.

After the required 72 hour cooldown period, HPCI was Cold Quick Started to the vessel. Time to rated flow was 27.5 seconds, exceeding the Level 1 criteria of 25 seconds. The initial and maximum speed peaks were 1095 and 4461 rpm, respectively. This exceeeded the Level 2 criteria of a maximum speed peak of 4200 rpm. At no time did the gland seal condenser system allow steam leakage to atmosphere.

The second Cold Quick Start to the vessel occurred 286 hours after the previous Cold Quick Start, far in excess of the required 72 hour cooldown period. Time to rated flow was 30.85 seconds, exceeding the Technical Specification allowable value of 30 seconds and the Level 1 criteria of 25 seconds. The initial and maximum speed peaks were 2918 and 4328 rpm, respectively, exceeding Level 2 criteria for a maximum speed peak of 4200 rpm. At no time did the gland seal condenser system allow steam leakage to atmosphere.

During a diagnostic test to investigate HPCI performance after a 24 hour cooldown period, the HPCI turbine tripped on overspeed. In order to further investigate HPCI performance, five diagnostic HPCI CST to CST test runs were performed.

As a result of this and other investigations, the HPCI turbine control oil system was disassembled, cleaned, and inspected and the HPCI EGR was replaced. During the HPCI outage, the HPCI discharge check valve was changed from a lift check to a swing check in an attempt to improve closing times to mitigate suction piping overpressure transients observed during HPCI turbine trips.

Following HPCI operability checks, tuning was again performed resulting in acceptable turbine performance. HPCI [illegible] art performance was further improved [illegible] changing out the HPCI stop valve limit switches, r[illegible] delay to the RGSC ramp start.

In October of 1987, the Test Condition Three HPCI Vessel Injection test sequence was reperformed in its entirety, beginning with the Hot Vessel Injection.

Following a manual start to the vessel, dynamic stability checks were performed. Two of the average flow steps, ± 500 gpm at 2200-2700 gpm, did not meet the Level 2 criteria for quarter damping. This condition was accepted because of the high degree of stability at higher flow rates.

Following the manual start a Hot Quick Start to the vessel was performed, with rated flow occurring after 20.5 seconds. The maximum transient speed peak was 4117 rpm. All other Level 1 and Level 2 criteria were met.

Following a 91 hour cooldown, the first HPCI Cold Quick Start was performed, with rated flow occurring after 21.5 seconds. The maximum transient speed peak was 4130 rpm. All other Level 1 and Level 2 criteria were met.

The final HPCI Cold Quick Start was performed following a 74 hour cooldown period. The maximum transient speed peak was 4123 rpm and rated flow was obtained 21.4 seconds after initiation. Approximately one minute into the test, the HPCI turbine tripped on High RPV Water Level (Level 8). Because of the short duration of the test, Gland Seal System data could not be taken. This Level 2 criteria violation was accepted based on acceptable Gland Seal System performance on all prior tests.

The HPCI turbine trip on Level 8 was avoidable with a more rapid feedwater turbine speed adjustment and was not the result of any HPCI System component malfunction and, therefore, was not considered to be a violation of the Level 1 criteria. All other Level 1 and Level 2 criteria were satisfied.

Following the completion of HPCI Vessel Injection testing, the final Cold CST Quick Start test was performed to collect baseline data for the Operations Surveillance Testing Program. After a 72 hour cooldown period, HPCI was Quick Started to the CST, with rated flow occurring after 19.9 seconds. The maximum transient speed peak was 414[illegible] and steady state flow stabilized at 5400 gpm [illegible] discharge pressure at 1[illegible]60 psig.

Data gathered during the above testing to determine the HPCI steam line high flow isolation trip setpoint has been further evaluated by Nuclear Engineering. The results of this evaluation [illegible]date the flow equation used to determine the [illegible]ial trip setpoint as presently listed in Tech. [illegible]c. Table 3.3.2 [illegible] and therefore no adjustment is necessary. This evaluation is detailed in Design Calculation #4372 Revision C.

During the startup following the Spring 1988 LLRT Outage, the final Cold and Hot CST Injections and stability checks were performed at a reactor pressure of 150 psig.

During the Cold Quick Start, HPCI discharge flow reached greater than 5000 gpm within 19.2 seconds. Pump discharge pressure was 275 psig which was greater than 100 psig above reactor pressure. HPCI speed reached a maximum value of 2816 rpm which is well below the 10% margin to the overspeed trip setpoint of 5000 rpm (4420 rpm). All applicable test criteria were met.

Following the above testing, HPCI was started manually and flow steps in both manual (speed control) and auto (flow control) modes were performed with pump flow between 4500 and 5100 gpm. Stability was successfully demonstrated by this testing and all HPCI system variable responses were shown to have decay ratios less than 0.25. A Hot Quick Start was then performed and HPCI discharge flow reached greater than 5000 gpm within 16.1 seconds with a discharge pressure of 290 psig which was greater than 100 psi above reactor pressure. The maximum speed reached by HPCI was 2816 rpm. All applicable test criteria were met.

This concludes all required HPCI testing during the Startup Test Program.

3.14 Selected Process Temperatures

3.14.1 Purpose

The purposes of this procedure are to establish the proper setting of the low speed limiter for the recirculation pumps to avoid coolant temperature stratification in the reactor pressure vessel bottom head region, to provide assurance that the measured bottom head drain temperature corresponds to bottom head coolant temperature during normal operations, and to identify any reactor operating modes that cause temperature stratification.

3.14.2 Criteria

Level 1

The reactor recirculation pumps shall not be restarted nor flow increased unless the coolant temperatures between the steam dome and bottom head drain are within 145°F. The recirculation pump in an idle loop must not be started, active loop flow must not be raised, and power must not be increased unless the idle loop suction temperature is within 50°F of the active loop suction temperature. If two pumps are idle, the loop suction temperature must be within 50°F of the steam dome temperature before pump startup.

Level 2

During operation of two recirculation pumps 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.

3.14.3 Results

For the initial testing conducted in 1985, the coolant temperatures measured at 30% Recirculation pump speed satisfied the Level 1 criteria. The instability of the recirc. speed controller that occurred during this test precluded an effective investigation of the stratification phenomenon at low flows. The test also allowed setting of the low speed limiter based on flow controller variations of $\pm$ 2% of rated speed. Flow controller variations of $\pm$ 5% were experienced prior to stratification so the test was terminated.

The minimum recirculation pump speed data collection was resumed in August, 1986 following completion of the preceding Outage. In subsequent heatup testing, the Recirc MG Sets were hand cranked down to speeds of about 20%. The Level 1 criteria was satisfied at all times during this test. The low speed limiter setting was chosen to be 28% speed based on the previously observed controller instability below that level.

During Test Condition Six, with the reactor operating at 95.9% CTP and 97% CF, the bottom head temperature as measured from the bottom drain line thermocouple was within 17°F of the recirculation loop temperature thereby satisfying the Level 2 criteria of ≤ 30°F delta temperature. Following entry into and completion of Test Condition 4 testing, the differential temperature between the steam dome and the bottom head temperature was verified to be within 44°F prior to the restart of the first recirculation pump, thereby satisfying the Level 1 criteria that this difference be ≤ 145°F. Additionally, the loop suction temperature was verified to be within 1°F of the steam dome temperature prior to this first pump restart which satisfies the Level 1 criteria that this difference be ≤ 50°F.

Although the second pump was not restarted due to the failure of its discharge valve to close and permit restart, the suction temperature of this idle loop was verified to be within 3°F of the active loop suction temperature and therefore would have satisfied the Level 1 criteria that this difference be ≤ 50°F had the pump been restarted.

This concludes the required selected process temperatures section of the Startup Test Program.

3.15 System Expansion

3.15.1 Purpose

The purpose of this test is to verify that selected plant piping systems are free and unrestrained with regard to thermal expansion, and to verify that the thermal movement of the piping and associated support system components is consistent with the analytical prediction of the piping system stress analysis.

3.15.2 Criteria

Level 1

The measured displacements at the instrumented locations shall be within the greater of the specified allowable tolerance of the calculated values, or ± 0.25 inches for the specific points.

There shall be no obstruction which will interfere with the expected thermal expansion of the piping system.

Electrical cables shall be able to accommodate expected thermal expansion of the piping system.

Instrumentation and branch piping can accommodate expected thermal expansion of the piping system.

The constant hanger shall not be bottomed or topped out.

The spring hanger shall not be bottomed or topped out.

The snubber shall not be bottomed or topped out.

Level 2

The measured displacements at the instrumented locations should be within the greater of the specified expected tolerance of the calculated values, or ± 0.25 inches for the specific points.

The installed cold position of the constant hanger must be within ± 5% of the design cold load.

The installed cold position of the spring hanger must be within ± 5% of the design cold load.

The snubber may deviate from its design cold position setting $\pm$ 1/2", providing the position is not less than 1/2" from bottoming out.

3.15.3 Results

Piping Inspection Results

Selected piping systems were walked down at various plant conditions to identify possible restraints to projected thermal expansion. These walkdowns occured at ambient temperature, 250°F and rated temperature. Hanger and snubber settings were recorded and thermal expansion (PVDET) sensors were verified to be intact.

No restraints to projected thermal expansion were identified. One-hundred and forty-three (143) supports were identified as being out of tolerance or topped or bottomed out. Following re-verification and engineering evaluation, sixteen (16) supports were adjusted or modified and the remainder accepted as is.

The East and West Main Steam Bypass Lines were replaced during the Outage which started in the fall of 1985, because of cracks which were discovered in these lines. During subsequent testing following reactor restart in August, 1986 these lines were visually inspected to verify that they were unrestrained with regards to projected thermal expansion. These walkdowns occured at ambient temperature; and at recirc loop temperatures of 350° and rated.

No restraints to bypass line thermal expansion were identified. Five supports were found out of tolerance, and upon engineering evaluation were accepted as-is.

Third thermal cycle visual inspections and hanger readings were made on all system piping including the replaced Main Steam Bypass Lines. There were no restraints to thermal expansion identified. Two-hundred-ninety-five (295) supports were identified as not being within their proper working range. Following engineering evaluation and reverification, eight (8) supports were reset and the remaining supports accepted as-is".

System Expansion Results

Selected points on the piping systems were wired with remote sensors to monitor the thermally induced piping movements during system operation. The monitored points were expected to undergo large movements or experience large thermal stresses.

After establishing initial readings for the sensors at ambient conditions, the sensors were monitored during the initial heatup of the plant. Data was recorded at 50°F intervals until the reactor reached operating temperature. The evaluations found several criteria exceedances, but upon engineering evaluation of the exceedances, all were found acceptable.

In addition, initial ambient sensor readings taken before Heatup were compared to ambient sensor readings after a Heatup and cooldown cycle was completed. No appreciable difference in the before and after readings were noted, indicating piping movement was not restrained.

Thermal Expansion data was again taken at 50°F intervals at moderator temperatures beginning at 100°F during the subsequent heatup cycle following initial heatup. The data was evaluated at each temperature plateau before proceeding to the next level. Upon reaching rated temperature, four Level 2 criteria violations existed, but these were very minor and accepted as-is.

The East and West Main Steam Bypass Lines that were replaced in the fall of 1985 were also monitored for expected thermal expansion during the subsequent heatup after the Outage. The heatup and cooldown sensor readings satisfied all Level 1 and Level 2 criteria except at the 350°F recirc loop temperature plateau. At that point there was one Level 2 failure which resulted from inadequate heating of the bypass piping due to the bypass valves being closed at the time the test was performed. At higher temperatures data was taken with the bypass valves open, and all criteria were satisfied.

During Test Condition Six, with the reactor operating at 96.6% CTP and 97.5% CF, additional sensor readings were taken. Movement was determined from baseline readings taken at cold conditions during the previous plant shutdown.

There were two apparent Level 1 criteria exceedances and twenty three Level 2 criteria exceedances associated with this data collection. Displacement sensor D-203 located on the RCIC Steam Supply Line had a reading of 580 mils vs an allowable of 241 mils; however, sensors D-201, D202 and D-204, also located on the RCIC Steam Line, were very close to the analytical prediction. Additionally, displacements of sensors on the B main steam line are close to their analytically predicted displacement which indicates that the overall header is moving in the predicted direction and therefore sensor D-203 may not be working properly; however, if this is not the case, preliminary calculations show that the maximum stress in the RCIC system caused by this exceedance is 7% of the allowable stress and is therefore acceptable.

The load measured by force sensor K004A was 28,321 lbs vs an allowable Level 1 value of 12,515 lbs. Previous to this test during initial heatup of the plant in 1985, a load of 35,687 lbs. was measured by this sensor and detailed calculations at that time deemed that load to be acceptable. Since the present load is less than that previously evaluated and found acceptable, this load is also acceptable.

Detailed evaluations by Sargent & Lundy are ongoing for the above Level 1 violations and the twenty three Level 2 violations but not yet complete as of this report date.

3.16 Core Power Distribution

NOTE: As discussed in memorandum VP-86-0141, "Startup Test Program Changes", dated October 17, 1986, from Frank E. Agosti to James G. Keppler, it is our intention to delete this test.

3.17 Core Performance

3.17.1 Purpose

a. To evaluate the core thermal power.

b. To evaluate the following core performance parameters:

1. Maximum linear heat generation rate (MLHGR)

2. Minimum critical power ratio (MCPR)

3. Maximum average planar linear heat generation rate (MAPLHGR).

3.17.2 Criteria

Level 1

The maximum linear heat generation rate (MLHGR) during steady-state conditions shall not exceed the allowable heat flux as specified in the Technical Specifications.

The steady-state minimum critical power ratio (MCPR) shall be maintained greater than, or equal to, the value specified in the Technical Specifications.

The maximum average planar linear heat generation rate (MAPLHGR) shall not exceed the limits given in the plant Technical Specifications.

Steady-state reactor power shall be limited to full rated maximum values on or below the design flow control line.

Core flow should not exceed its rated value.

Level 2

None

3.17.3 Results

BUCLE computer analysis of whole core TIP traces obtained at 15.6% reactor power showed that all criteria were met, during Test Condition One.

The Core Performnance parameters during Test Condition Two were determined using the Process Computer programs P1 (Periodic Core Evaluation) and

OD-3 (Core Thermal Power/APRM Calibration). All Level 1 criteria were satisfied upon the determination and verification of the following parameters:

Core Thermal Power (CMWT)
Percent of Rated Core Thermal Power (PCT PWR)
Core Flow (WT)
Maximum Linear Heat Generation Rate (MLHGR)
Minimum Critical Power Ratio (MCPR)
Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

During Test Condition Three, the Process Computer programs (P1 and OD-3 Option 2) were again run to determine the above parameters:

The Process Computer edits were utilized to determine that all requirements associated with the test were satisfied as follows:

The Core Maximum Fraction of Limiting Power Density was 0.43 which satisfied the acceptance criteria that this value be less than or equal to 1.0.

The Core Maximum Fraction of the Limiting Critical Power Ratio was 0.44 which satisfies the acceptance criteria that this value be less than or equal to 1.0.

The Core Maximum Average Planar Linear Heat Generation Rate Ratio was 0.42 which satisfies the acceptance criteria that this value be less than or equal to 1.0.

The rated maximum value for reactor power at 95.3% of rated core flow was determined to be is 96.5% of rated Core Thermal Power based on the design flow control line. The actual calculated CTP was 48.6% which was below the design flow control line.

Measured core flow was 95.3% of rated core flow which satisfies the criteria that core flow does not exceed its rated value.

During Test Condition Five, the Process Computer programs (P1 and OD-3 Option 2) were run during the performance of the Reactor Engineering procedure 54.000.07 (Core Performance Parameter Check). The Process Computer edits were utilized to determine that all requirements associated with the test were satisfied as follows.

The Process Computer value of Core Maximum Fraction of Limiting Power Density was 0.663, which satisfies the acceptance criteria that requires this value to be less than or equal to 1.0.

The Process Computer value of Core Maximum Fraction of the Limiting Critical Power Ratio was 0.704, which satisfies the acceptance criteria that requires this value to be less than or equal to 1.0.

The Process Computer value of Core Maximum Average Planar Linear Heat Generation Rate Ratio was 0.665, which satisfies the acceptance criteria that requires this value to be less than or equal to 1.0.

The rated maximum value for reactor power at 61.4% of rated core flow was determined to be 74% of rated Core Thermal Power based on the design flow control line. The actual CTP was 71.8% which was below the design flow control line.

Measured core flow was 61.4% of rated core flow which satisfies the criteria that core flow does not exceed its rated value.

During Test Condition Six, Process Computer Programs P1 and OD-3 Option 2 were run during the performance of the Reactor Engineering procedure 5?.000.07 (Core Performance Parameter Check). The Process Computer edits were utilized to determine that all requirements associated with the test were satisfied as follows:

The Process Computer value of Core Maximum Fraction of Limiting Power Density was 0.877. This satisfies the acceptance criteria which requires this value to be less than or equal to 1.0.

The Process Computer value of Core Maximum Fraction of the Limiting Critical Power Ratio was 0.849. This satisfies the acceptance criteria which requires this value to be less than or equal to 1.0.

The Process Computer value of Core Maximum Average Planar Linear Heat Generation Rate Ratio was 0.861. This satisfies the acceptance criteria which requires this value to be less than or equal to 1.0.

The rated maximum value for reactor power at 99.8% of rated core flow was determined to be 100% of rated Core Thermal Power based on the design flow control line. The actual CTP was 98.4% which was below the design flow control line.

Measured core flow was 99.8% of rated core flow which satisfies the criteria that core flow does not exceed its rated value.

During Test Condition Four, Process Computer Programs P1 and OD-3 Option 2 were run during the performance of the Reactor Engineering procedure 34.000.07 (Core Performance Parameter Check). The Process Computer edits were utilized to determine that all requirements associated with the test were satisfied as follows:

The Process Computer value of Core Maximum Fraction of Limiting Power Density was 0.39. This satisfies the acceptance criteria which requires this value to be less than or equal to 1.0.

The Process Computer value of Core Maximum Fraction of the Limiting Critical Power Ratio was 0.582. This satisfies the acceptance criteria which requires this value to be less than or equal to 1.0.

The Process Computer value of Core Maximum Average Planar Linear Heat Generation Rate Ratio was 0.375. This satisfies the acceptance criteria which requires this value to be less than or equal to 1.0.

The rated maximum value for reactor power at 35% of rated core flow was determined to be 50% of rated Core Thermal Power based on the design flow control line. The actual CTP was 39.6% which was below the design flow control line.

Measured core flow was 35% of rated core flow which satisfies the criteria that core flow does not exceed its rated value.

3.18 Steam Production

This test was previously deleted from the FSAR (Section 14.1.4.8.18).

3.19 Core Power-Void Mode Response

NOTE: As discussed in memorandum VP-86-0141, "Startup Test Program Changes", dated October 17, 1986, from Frank E. Agosti to James G. Keppler, it is our intention to delete this test.

3.20 Pressure Regulator

3.20.1 Purpose

The purpose of this test is to:

a. Determine the optimum settings for the pressure control loop by analysis of the transients induced in the reactor pressure control system by means of the pressure regulators.

b. To demonstrate the takeover capability of the backup pressure regulator on failure of the controlling pressure regulator and to set spacing between the setpoints at an appropriate value.

c. To demonstrate smooth pressure control transition between the control valves and bypass valves when the reactor generates more steam than is used by the turbine.

3.20.2 Criteria

Level 1

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

Level 2

In all tests the decay ratio must be less than or equal to 0.25 for each process variable that exhibits oscillatory response to pressure regulator changes when the plant is operating above the lower limit setting of the master flow controller.

Pressure control deadband, delay, etc., shall be small enough for steady-state limit cycles, if any, to produce turbine steam flow variations no larger than 0.5 percent of rated flow.

During the simulated failure of the controlling pressure regulator along the 100 percent rod line, the backup regulator shall control the transient so that the peak neutron flux or peak vessel pressure remains below the scram settings by 7.5 percent and 10 lb/in.^{2}, respectively.

After a pressure setpoint adjustment, the time between the setpoint change and the occurrence of the pressure peak shall be 10 seconds or less. (This applies to pressure setpoint changes made with the recirculation system in the master or local manual control mode.)

3.20.3 Results

Proper pressure regulator operation was demonstrated in Test Condition One by analysis of system response to step increases and decreases in pressure demand with the bypass valves open and generator not on the line. Additional steady-state measurements were taken with the generator loaded and bypass valves closed. All Level 1 and Level 2 criteria were met.

The pressure setpoint changes on each regulator, while significant in magnitude (11-13 psig), were stable and well damped. As such no system tuning was performed in this test condition.

The Regulator failure tests yielded significantly different responses (14 psig change for failure of #1; 6 psig change for failure of #2). This discrepancy in response is likely attributable to differences in the time delay circuitry for each channel in the High Value Gate and difference of 1.7 psig in the sensed pressure being fed to each regulator channel. The time delay component in the regulator high value gates has since been removed.

The testing performed for the Pressure Regulator during Test Condition Two consisted of introducing 10 psig step change and simulated regulator failures in the Pressure Control System.

The Level 1 criteria for this test during Test Condition Two was satisfied when no process variables were found to be divergent and all decay ratios were less than 1.0 during the 10 psig step changes and simulated regulator failures.

Steady-state steam flow variations were monitored by measuring generator electrical output limit cycling due to pressure controller operation. The Level 2 criteria requiring that these variations are no larger than 1.0 percent peak-to-peak of rated flow was satisfied by analysis of the generator output which showed a maximum variation of 0.9 percent peak-to-peak of rated flow.

The other Level 2 criteria associated with this test required that, after a pressure setpoint adjustment, the time between the change and the occurrence of the pressure peak shall be 10 seconds or less. Analysis of this test's 10 psig steps showed peak pressures between 3.6 and 5.2 seconds, satisfying the criteria.

Finally, the elimination of the time delay to backup regulator takeover resulted in significant improvement over Test Condition One results in response to both normal transfers and regulator failures. At no time did the bypass valves enter their "FAST" mode and all transients were controlled and strongly damped.

Pressure regulator testing during Test Condition Three was performed at 71.4 CTP to verify the optimum settings for the pressure control loop by analysis of system response to step decreases and increases in pressure demand. In addition, the takeover capability of the backup pressure regulator upon failure of the controlling pressure regulator was demonstrated. Proper pressure regulator operation was demonstrated with both the Turbine Control Valves alone and with "incipient" conditions, defined as that condition where the load demand has just barely closed the bypass valves. Additional steady-state measurements were taken with the generator loaded and the bypass valves closed.

For the 10.0 psig down and up steps and regulator failures performed in this test, no process variables were found to be divergent and all decay ratios were less than 1.0, thereby satisfying the Level 1 criteria.

An analysis of the steady state generator output data recorded during this test shows a maximum peak to peak value of 9.3 MWe which is less than 11.54 MWe (1% peak-to-peak) and, therefore satisfies the Level 1 criteria.

An analysis of the 10.0 psig down and up steps show peak pressures occur between 4.3 and 6.0 seconds after step initiation, well within the ten (10) second Level 2 criteria for this test.

The following Table summarizes the pressure data resulting from the step changes and regulator failures:

Test Description	Initial Press	Max/Min Press	Final Press
TCV			
Reg #1 Failure	963.7	973.5	967.2
Reg #2 Failure	963.7	973.7	967.5
Reg #1 10 psig downstep	963.6	951.0	952.9
Reg #1 10 psig upstep	952.9	965.6	963.7
Reg #2 10 psig down step	964.3	951.7	953.1
Reg #2 10 psig upstep	953.3	966.0	963.9
Incipient			
Reg #1 10 psig downstep	963.9	953.3	953.9
Reg #1 10 psig upstep	953.7	964.8	963.5
Reg #2 10 psig downstep	964.3	953.3	954.2
Reg #2 10 psig upstep	954.2	965.4	964.3

Pressure regulator testing during Test Condition Five was performed to verify the optimum settings for the pressure control loop by analysis of system response to step decreases and increases in pressure demand. Proper pressure regulator operation was demonstrated with both the Turbine Control Valves and the Bypass Valves controlling pressure. Additional steady-state measurements were taken with the generator loaded and bypass valves closed.

For the 10 psig down and up steps performed in this test, all process variables were strongly damped and no decay ratios were found to exceed 0.25, satisfying both the Level 1 and Level 2 criteria.

An analysis of the 10 psig down and up steps show peak pressures occur between 3.2 and 5.1 seconds after step initiation, well within the ten (10) second Level 2 criteria for this test.

Rated turbine steam flow is equivalent to 1154 MWe, consequently, 1% peak to peak variations must be less than 11.54 MWe. An analysis of the steady state generator output shows a maximum peak to peak value of 9.928 MWe, thus satisfying the Level 2 criteria.

The following Table summarizes the pressure data resulting from the pressure setpoint step changes.

Test Description	Initial Press	Max/Min Press	Final Press
TCV Steps			
Reg #1 10 psig downstep	964.8	952.3	953.9
Reg #1 10 psig upstep	953.7	966.8	964.7
Reg #2 10 psig downstep	964.5	952.1	953.4
Reg #2 10 psig upstep	953.7	966.8	964.5
BPV Steps			
Reg #1 10 psig downstep	965.0	953.3	954.5
Reg #1 10 psig upstep	955.8	968.3	965.2
Reg #2 10 psig downstep	965.2	953.1	954.4
Reg #2 10 psig upstep	954.5	967.7	965.6

Pressure Regulator testing has not yet been completed in Test Condition Six.

3.21 Feedwater System

3.21.1 Purpose

a. To adjust the feedwater control system for acceptable reactor water level control.

b. To demonstrate stable reactor response to subcooling changes.

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.

d. To demonstrate adequate response to feedwater heating loss.

e. To determine the maximum feedwater runout capability.

3.21.2 Criteria

Level 1

The response of any level-related variable to any test input change, or disturbance, must not diverge during the setpoint changes.

For the feedwater temperature 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.

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

The feedwater flow runout capability must not exceed the assumed value in the FSAR.

Level 2

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

A scram must not occur from low water level following a trip of one of the operating feedwater pumps. There should be a greater than 3-in. water-level margin to scram for the feedwater pump trip.

For the feedwater temperature loss test, the increase in simulated heat flux cannot exceed the predicted value referenced to the actual feedwater temperature change and power level, which will be taken from the Transient Safety Analysis Design Report.

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

The dynamic flow response of each feedwater actuator (turbine or valve) to small (<10 percent) step disturbances shall be the following:

a. Maximum time to 10 percent of a step disturbance ≤1.1 sec.

b. Maximum time from 10 to 90 percent of a step disturbance ≤1.9 sec.

c. Peak overshoot (percentage of step disturbance) ≤15 percent.

3.21.3 Results

During the initial heatup, the feedwater system performed satisfactorily in both the manual and automatic modes. All level-related variables did not diverge during testing and all system related variables did not exceed a 0.25 decay ratio for their oscillatory responses in the level setpoint changes. All applicable test criteria were satisfied.

During Test Condition One, as previously done during the heatup testing, the Startup Level Controller setpoint was adjusted to simulate step changes of three inches for Reactor water level. During the setpoint increase water level increased in a smooth manner with little overshoot and stabilized within 75 seconds. During the setpoint decrease water level decreased and overshot the three inch down step by 2 to 3 additional inches. This overshoot dampened rapidly and water level stabilized within 110 seconds.

The Test Condition One test was completed satisfactorily. The criteria that the decay ratio of level control system-related variables being less than .25 was met for all portions of this test.

During Test Condition Two, feedwater system testing was limited to single element master level controller step changes due to equipment problems with the Dynamic Compensator Lead/Lag Network Computation Module. The dynamic flow response of the Reactor feed pump turbines was not able to be checked because the flow to the Reactor was insufficient to allow automatic level control with two pumps operating with both minimum flow bypass valves shut. Both minimum flow bypass valves are required to be closed to adequately measure the flow response of the feedwater actuators to step inputs.

Feedwater system response to five inch Reactor level changes using setpoint tape manipulations in single element automatic control were smooth and controlled. All applicable acceptance criteria were met for the conditions tested.

In Test Condition Three, at a reactor power of 48%, testing was conducted in both One Element and Three Element modes, with each feedpump feeding the vessel and the other in standby. This satisfied the above noted Test Condition Two testing that could not be completed earlier due to the inoperative Dynamic Computation module.

Both SRFPT Control Systems (System #1 and System #2) were tuned and ± 10% speed demand steps with the pump in the recirculation mode were performed.

After the completion of SRFPT Speed Control System testing, the NRFP was then placed in standby after the SRFP was placed into service feeding the vessel. Level setpoint tape changes of up to ± 5 inches were performed in both One Element and Three Element modes. Once proper Level Control System response was verified, the ± 5 inch level setpoint adjustment ramps were performed in both One and Three Element modes.

Following completion of SRFP testing, both of the NRFPT Speed Control Systems were tuned and tested, again with the pump in the recirculation mode. Once the NRFP was placed in service feeding the vessel, level setpoint change testing was performed in the same manner as the SRFP.

The STARTREC traces for both One and Three Element Control mode were analyzed for quarter damped response. The following signals were deemed to be Level Control System-related:

Feedwater Control Function Generator Output - NRFP
Feedwater Control Function Generator Output - SRFP
Master Feedwater Controller Output
North Reactor Feed Pump Flow
South Reactor Feed Pump Flow
North RFPT Speed
South RFPT Speed

All of the above signals showed quarter damped (0.25) response to $\pm$ 5 inch level setpoint changes which satisfies the Level 1 criteria of non-divergence and the Level 2 criteria of decay ratio.

In Test Condition Three at a reactor power of 71.4%, additional water level setpoint changes ($\pm$ 5 inches) in both Single and Three Element modes were performed. The applicable Level 1 criteria for no divergence and Level 2 criteria for quarter damping were met for the testing performed.

Planned testing to verify the dynamic flow response and rate of response and the feedwater turbine actuators (Level 2 criteria) could not be performed due to Feed Pump Turbine speed control and hydraulic control oil system problem.

In Test Condition Five at a reactor power of 71.2%, water level setpoint changes ($\pm$ 5 inches) were again performed in both Single and Three Element modes. The Level 1 criteria for no divergence and the Level 2 criteria for quarter damping of level control system related signals were met.

This testing provided confidence that the Feed Pumps would adequately respond to expected demands until hydraulic control oil system repairs and turbine speed control system modifications could be made during the Spring 1988 LLRT Outage.

Prior to the shutdown for the Spring 1988 LLRT Outage, diagnostic tests were performed to determine the physical response of the RFPTs to open loop step changes. This response information resulted in a redesign of the Woodward Governor control amplifier cards, one of which was installed in the NRFP Turbine System #2 on an experimental basis. During the LLRT outage, the Governor pilot actuators on both RFPTs were replaced with new units and the control oil systems were modified to install hydraulic accumulators. The entire feedwater/governor control system was also recalibrated on both RFPTs.

During the power ascension in the startup following the LLRT outage, limited inner speed loop (± 60 rpm) step response testing of the original and experimental speed control amplifiers verified that the newly modified amplifier cards would be necessary since speed control stability could not be achieved with the original amplifier cards. An experimental amplifier card was then also installed in the SRFP Turbine System #2 and stability of the feedwater control system was demonstrated by the performance of ± 60 rpm inner speed loop steps on both RFPTs and ± 5 inch level controller setpoint changes in 3 element mode only.

Following the completion of Test Condition Six steady state testing during the startup after Outage 88-02, the entire Test Condition Three feedwater tuneup optimization and test sequence were performed at approximately 75% CTP with the newly designed speed control amplifier cards installed in both speed control channels of both the North and South RFPTs.

The results of the speed control system testing is tabulated below.

Dynamic Flow Response to small (< 10%) step disturbances (100 rpm = ~ 7% flow)

RFPT	System Number	Step Size (rpm)	Delay Time (Sec) ≤ 1.1	Rise Time (Sec) ≤ 1.9	Peak Overshoot (%) ≤ 15%	Settle Time (Sec) ≤ 14
North	#1	100 up	0.36	0.80	23	4.14
North	#1	100 dn	0.52	0.70	23	5.96
North	#2	100 up	0.44	0.90	18	4.42
North	#2	100 dn	0.48	0.84	22	5.06
South	#1	100 up	0.46	0.80	23	5.32
South	#1	100 dn	0.40	0.74	20	4.66
South	#2	100 up	0.50	0.80	23	3.30
South	#2	100 dn	0.52	0.76	21	5.76

As can be seen from the above, all applicable Level 2 criteria with the exception of Peak Overshoot were met; however, this has been deemed acceptable by GE based upon the acceptable rate of response to the steps and the high degree of stability following the steps.

Average Rate of Response to large (> 20% flow) disturbance 250 rpm = ~ 20-21% flow)

RFPT	System Number	Step Size (rpm)	Rate of Response % Flow/Sec >10 <25% Flow/Sec
North	#1	250 up	20.0
North	#1	250 dn	15.6
North	#2	250 up	18.0
North	#2	250 dn	17.2
South	#1	250 up	18.8
South	#1	250 dn	12.1
South	#2	250 up	18.0
South	#2	250 dn	11.7

As can be seen from the above, the Level 2 criteria has been met.

Following the completion of the above RFPT tuning, $\pm$ 5 inch Reactor Water level setpoint changes were performed with the feedwater controller in both 3 element and single element modes. The Level 1 criteria for non divergence and the Level 2 criteria for quarter dampening of all level control related variables was met.

With the reactor operating in Test Condition Six at 96.8% CTP and 97% CF, $\pm$ 5 inch Reactor Water level setpoint changes were again performed with the feedwater controller in both 3 element and single element modes. The applicable Level 1 and Level 2 criteria were met.

The remaining feedwater system testing required for the Startup Test Program which includes the Maximum Feedwater Runout Capability, One Pump Trip and Loss of Feedwater Heating tests has not yet been performed.

3.22 Turbine Valve Surveillance

3.22.1 Purpose

To demonstrate acceptable procedures and maximum power levels for surveillance testing of the main turbine control and stop valves without producing a reactor scram.

3.22.2 Criteria

Level 1

None

Level 2

Peak neutron flux must be at least 7.5 percent below the scram trip setting. Peak vessel pressure must remain at least 10 lb/in.^{2} below the high-pressure scram setting. Peak heat flux must remain at least 5.0 percent below its scram trip point.

Peak steam flow in the high-flow lines must remain 10 percent below the high-flow isolation trip settings.

3.22.3 Results

Turbine Control and Stop Valve Surveillance testing has been performed up to a power level of 91.4% CTP. All criteria to that point have been satisfied and from a reactor physics standpoint this test could be performed at higher power levels; however, due to a balance of plant considerations with fluctuating heater levels and turbine control valves nearing 100% open, the highest power level recommended to perform this test was determined to be ≤ 90% CTP.

The results of the testing performed are tabulated below for the most limiting control valve.

	Rx Power (%)	Margin to Neutron Flux Scram (%)	Margin to High Pressure Scram (psi)	Margin to Heat Flux Scram (%)	Margin to High Steam Flow (Mlb/hr)
Acceptance Criteria		≥ 7.5	≥ 10	≥ 5.0	≥ 0.354
STUT.050.024	70.8	40.2	95.8	8.68	2.34
* STUT.050.024 Supplement 1	*76.8	33.8	93.9	13.29	2.03
Information Only Data Set	80.0	33.0	91.0	8.00	1.98
STUT.050.024 Supplement 2	84.5	29.2	84.8	7.34	1.40
STUT.050.024 Supplement 3	87.8	21.6	80.0	6.98	1.17
STUT.050.024 Supplement 4	90.0	22.3	80.0	9.52	1.11
STUT.050.024 Supplement 5	91.4	21.8	72.7	9.14	1.08

* This data set taken was not on the 100% rod line, resulting in the higher margin to heat flux scram (flow biased).

Additionally, the East and West Bypass Valves were tested at 70.8% CTP; however, the criteria, although met at this power level, does not apply to this testing since the valves are manually stroked slowly open then slowly closed in turn and at no time do the valves enter a fast open/trip closed mode. Therefore, there is no transient associated with pressure regulator response, and the only effect seen is due to the reduction in feedwater temperature due to the diversion of ~ 15% steam flow to the condenser. A confirmatory Bypass Valve test at approximately 90% CTP is planned to evaluate the effects of Bypass Valve testing on balance of plant equipment at that power level.

3.23 Main Steam Isolation Valves

3.23.1 Purpose

a. To check functionally the main steam line isolation valves (MSIVs) for proper operation at selected power levels.

b. To determine reactor transient behavior during and after simultaneous full closure of all MSIVs.

c. To determine isolation valve closure time.

3.23.2 Criteria

Level 1

The MSIV stroke time (t_s) shall be no faster than 3.0 seconds (average of the fastest valve in each steamline) and for any individual valve 2.5 seconds $\leq t_s \leq 5$ seconds. Total effective closure time for any individual MSIV shall be t_{sol} plus the maximum instrumentation delay time and shall be ≤ 5.5 seconds.

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

Flooding of the main steam lines shall not occur following the full MSIV closure test.

The reactor must scram during the full simultaneous MSIV closure test to limit the severity of the neutron flux and simulated fuel surface heat flux transient.

Level 2

During full closure of individual valves, peak vessel pressure must be at least 10 psi below scram, peak neutron flux must be at least 7.5 percent below scram, and steam flow in individual lines must be at least 10 percent below isolation trip setting. The peak heat flux must be at least 5 percent less than its trip point. The reactor shall not scram or isolate as a result of individual valve testing.

The relief valves must reclose properly (without leakage) following the pressure transient resulting from the simultaneous MSIV full closure.

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

After the full MSIV closure, the initial action of the RCIC and HPCI shall be automatic if L2 is reached, with RCIC capable of establishing an average pump discharge flow equal to or greater than 600 gpm within the first 50 seconds after automatic initiation and HPCI capable of establishing an average pump discharge flow equal to or greater than 5000 gpm within the first 25 seconds after automatic initiation.

If the low-low set pressure relief logic functions after the simultaneous full MSIV closure test, the open/close actions of the SRVs shall occur within ±20 psi of the low-low set design setpoints. The total number of opening cycles, for the safety/relief valves opening on low-low setpoint, after initial blowdown is not to exceed four times during the initial 5 minutes following isolation. If any safety relief valves open as a result of this test, only one valve may reopen after the first blowdown.

Recirculation pump trip shall be initiated if L2 is reached after the MSIV full closure test.

3.23.3 Results

During the Heatup Test Condition, with the RPV at rated temperature and pressure conditions, each of the inboard and outboard isolation valves were successfully closed slowly to the approximately 90% open position and then fully reopened, without any noticeable change in reactor pressure, APRM readings or reactor water level.

In Test Condition One, with the Reactor at 7% power, a fast full closure of each individual MSIV was performed. All applicable Level 1 and Level 2 criteria were met. The closure times are shown in the table below, using a calculated maximum instrument delay time of 0.299 seconds.

Test Condition One*

MSIV	t_s	t_{sol}	Total
F022A	4.298	4.611	4.910
F022B	3.505	3.703	4.002
F022C	4.798	4.904	5.203
F022D	3.205	3.301	3.600
F028A	4.294	4.387	4.686
F028B	3.809	3.839	4.138
F028C	3.617	3.899	4.198
F028D	4.057	4.226	4.525

* All recorded times are measured in seconds.

During Test Condition Three, with the reactor at 69.2% CTP, each of the inboard and outboard isolation valves were successfully closed slowly to the approximately 90% open position and then fully reopened without any noticeable change in reactor pressure, APRM readings or reactor water level.

During Test Condition Six with the reactor at 96.8% CTP, each of the inboard and outboard isolation valves were successfully closed slowly to the approximately 90% open position and then fully reopened without any noticeable change in reactor pressure, APRM readings or reactor water level.

The remaining Level 1 and Level 2 criteria are associated with the MSIV simultaneous full closure and will be verified when that test is performed.

3.24 Relief Valves

3.24.1 Purpose

The purposes of this test are to verify that the Safety Relief Valves (SRV) function properly (can be opened and closed manually), reset properly after operation, and that there are no major blockages in the relief valve discharge piping.

3.24.2 Criteria

Level 1

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

Level 2

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

The temperature measured by thermocouples on the discharge side of the valves shall return to within $10^{\circ}F$ of the temperature recorded before the valve was opened. If pressure sensors are available, they shall return to their initial state upon valve closure.

During the 250 psig functional test, the steam flow through each relief valve as measured by the initial and final bypass valve (BPV) position shall not differ by more than 10 percent from the average relief valve steam flow as measured by bypass valve position.

During the rated pressure test, the steam flow through each relief valve as measured by change in MW(e) is not to differ by more than 0.5 percent of rated MW(e) from the average of all the valve responses.

3.24.3 Results

During the heatup testing, all 15 SRVs were manually actuated. There was positive indication of steam discharge upon actuation of each SRV. As each SRV was operated there was a sudden temperature rise on the SRV discharge tailpipe, the appropriate pressure switch responded, and BPV position decreased to control reactor pressure. The Level 1 criteria was satisfied.

All pertinent variables related to pressure control did not exhibit any oscillatory responses with decay ratios greater than 0.25.

The SRV discharge line temperatures for five SRVs did not return to within 10°F of the temperature recorded prior to actuation as quickly as the other discharge lines; however, they did cool down sufficiently to indicate that the SRVs were not leaking. Shortly after the performance of this test a reactor scram occurred and on the subsequent startup, the SRV tailpipe temperatures remained low, further verifying that the SRVs did properly reclose.

Three SRVs had steam flow values, as measured by BPV position change, that differed from the average relief valve steam flow by greater than 10%. The bypass valve position was inadequate to get a proper value of steam flow from BPV position change. Upon the actuation of each SRV the BPV closed completely. Had there been more bypass steam flow, the BPV would not have closed completely and there would be a more accurate value of SRV steam flow. This steam flow variance was reevaluated during the Test Condition Two SRV testing.

All fifteen SRVs were manually actuated with the plant at rated pressure during Test Condition Two. Plant parameters related to pressure control were monitored on the GETARS computer, as well as other plant parameter responses, including generator load decreases.

The Level 1 criteria was met based on three positive indications of steam discharge during the actuation of each valve. They were the sudden temperature rise in the discharge tailpipe, the positive indication of a MWe decrease during the valve actuations, and the response from the tailpipe pressure sensor of each valve being tested.

The Level 2 criteria requiring that Pressure Control System variables did not exhibit any oscillatory responses with decay ratios greater than 0.25, was verified by the analysis of the GETARS data of the following variables:

Pressure Regulator Output
Control Valve Demand
Control Valve #1 Position
Narrow Range Pressure
Generator Output (Gross MWe)

GETARS data was also used to verify that the change in the plant's MWe following each SRV lift did not differ by more than 0.5% of the rated MWe from the average of all valves responses. All SRVs exhibited a less than 5.5 MWe variation from the 68.5 MWe average variation, thus satisfying the Level 2 criteria.

SRVs B21-F013J and B21-F013M did not return to within 10°F of their initial tailpipe temperature values during the test. However, the temperatures did return to within 10°F of their initial values when checked at a later time, thus satisfying a Level 2 criteria.

Finally, part of the Licensing Commitment 2.c.5 of the full power operating license was satisfied by this Test Condition Two relief valve test. It was demonstrated that all adjacent temperature readings were within 45°F of each other following a 10 second SRV lift with a suppression pool mixing system in operation.

This concludes the relief valve testing to be performed during the Startup Test Phase Program.

3.25 Turbine Stop Valve and Control Valve Fast Closure Trips

3.25.1 Purpose

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

3.25.2 Criteria

Level 1

For turbine/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. The bypass valves should be opened to a point corresponding to greater than or equal to 80 percent of their capacity within 0.3 seconds from the beginning of control or stop valve closure motion.

Flooding of the main steam lines shall not occur following the turbine/generator trips.

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

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

Level 2

There shall be no MSIV closure in the first 3 minutes of the transient, and operator action shall not be required in that period to avoid the MSIV trip.

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

For the turbine/generator trip within the bypass valves capacity, the reactor shall not scram for initial thermal power values less than or equal to 25 percent of rated.

If the low-low set pressure relief logic functions, the open/close actions of the SRVs shall occur within ± 20 psi of their design setpoints. If any safety relief valves open, only one valve may reopen after the first blowdown.

3.25.3 Results

During the Test Condition Two testing with a reactor power of 21.8%, a turbine/generator trip was initiated with a generator output of 151 MWe, by opening both generator output breakers CM and CF.

A reactor scram did not occur following the tubine/generator trip with the reactor at 21.8% power. This is required at a reactor power < 25%, therefore, satisfying the Level 2 criteria.

The East and West bypass valves began opening within 0.04 seconds and 0.06 seconds, respectively, following the beginning of the control and stop valve closure. This satisfied the ≤ 0.1 second opening time required for the Level 1 criteria.

The Level 1 criteria (applicable to Test Condition Six) requiring that the bypass valves open to a point corresponding to ≥ 80% of their capacity within 0.3 seconds from the beginning of the control and stop valves closure motions was not satisfied during the Test Condition Two testing. The valves only opened to 56.3% of their combined capacity at 0.3 seconds with the West Bypass Valve open 99.8%, and the East Bypass Valve open 12.7%. Repairs and off-line response time testing of the East Bypass Valve Unitized Actuator were performed successfully during the MSR outage. The effects of steam flow on bypass valve response time was further evaluated following an inadvertent turbine trip from 50% reactor power on 7/20/87. The East and West Bypass Valves began opening within 0.025 seconds and 0.065 seconds, respectively, and had reached ≥ 80% of their capacity within 0.2 seconds which would have satisfied the above Level 1 criteria if it had been applicable.

Following the completion of Test Condition Five, with the reactor operating at 74.6% CTP with 73.6% core flow, an inadvertent turbine/generator trip/reactor scram was experienced. The date of this occurrence was 12/31/87. In accordance with Reference Number 3 in Section 1.5 of this report, an analysis of the data recorded during this event has been performed to determine if this inadvertent trip can be substituted for the full power turbine/generator trip scheduled during Test Condition Six. This event occurred as a result of a Water Level 8 trip signal rather than from normal operating condition. Therefore, a code simulation of a trip from full power and normal operating conditions was performed. It was concluded that all test criteria would have been satisfied if the test was performed as scheduled. Consequently, we have taken credit for this situation in accordance with 10CFR50.59 and this change to the Startup Test Program has been submitted by letter NRC-88-0181, dated 7-14-88, as required by Fermi 2 License Condition 2.C(14).

3.26 Shutdown from Outside the Control Room

3.26.1 Purpose

To demonstrate that the reactor can be brought from a normal, initial, steady-state power level to the hot shutdown condition and to verify that the plant has the potential for being safely cooled from hot shutdown to cold shutdown conditions from outside the control room.

3.26.2 Criteria

Level 1

None

Level 2

During the cold shutdown demonstration, the reactor must be brought to the point where cooldown is initiated and under control.

During the simulated control room evacuation and hot shutdown demonstration, the reactor vessel pressure and water level are controlled using equipment and controls outside the control room.

3.26.3 Results

During the simulated control room evacuation and hot shutdown test performed during Test Condition One, the designated Shutdown Crew, consisting of the minimum shift complement, performed all activities associated with the reactor shutdown and control of the reactor vessel water level and pressure from outside the Control Room.

The reactor vessel pressure and water level were controlled for a period of over thirty minutes following successful reactor shutdown and isolation from outside the Control Room by the minimum shift complement, which successfully meets all test criteria and performance objectives of the applicable governing documents.

The test sequence of events was as follows:

Time	Event
1223	Test Start Time (Hi Comm Announcement)
1224	"Shutdown Crew" Evacuation of Control Room
1224	APRMs A&B to Standby (to initiate Reactor Scram)
1224	Relay TTR-2 manually tripped (to initiate Main Turbine Trip)
1225	Main Steam Line Radiation Monitors to Standby (to initiate MSIV Isolation)
1226	Restoration of APRMs A&B and the Main Steam Line Radiation Monitors to the Operate positions
1226	Exit Relay Room
1228	Transfer Switches operated at Remote Shutdown Panel (RSP) (RSP Control)
1230	RHRSW started at Remote Shutdown Panel (RHR Service Water Pumps A&C)
1233	RHR Pump A started at Remote Shutdown Panel
1233	Div II Transfer Switch operated (Div II D.C. ESF Power)
1234	RCIC initiated from Remote Shutdown Panel
1235	RCIC at rated flow (600 gpm)
1237	"A" SRV cycled from Remote Shutdown Panel (Open for approximately seven seconds)
1238	"B" SRV cycled from Remote Shutdown Panel (Open for approximately nine seconds)
1239	Start of Stable Control Period in Hot Shutdown
1313	Completion of Stable Control Period in Hot Shutdown

Time	Event
1313	Transfer Switches operated (RSP Transfer to Control Room Control)
1313	Test Termination

The remaining testing within this section, involving a demonstration of the plant's capability to reach cold shutdown conditions from outside the control room, is scheduled to be performed following the MSIV simultaneous full closure test in Test Condition Six.

3.27 Flow Control

3.27.1 Purpose

a. To determine the correct gain settings for the individual recirculation controllers.

b. To demonstrate plant response to changes in recirculation flow in both local manual and master manual mode.

c. To set the limits of range of operation for the recirculation pumps.

3.27.2 Criteria

Level 1

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

Level 2

The decay ratio of the speed loop response shall be ≤ 0.25 at any speed.

Flow control system limit cycles (if any) must produce a turbine steam flow variation no larger than ± 0.5 percent of the rated steam flow value.

The APRM neutron flux trip avoidance margin shall be ≥ 7.5 percent, and the heat flux trip avoidance margin shall be ≥ 5.0 percent as a result of the recirculation flow control maneuvers.

3.27.3 Results

In Test Condition Two, $\pm$ 4% step change testing was performed on both recirculation system speed control loops in the local manual mode at 38.8% Reactor power and 47.5% core flow.

A review of the data recorded indicates no variables related to the recirculation system were divergent.

A qualitative review of the speed response of the A Reactor Recirculation MG Set verified that the decay ratio was $<$ 0.25 for the $\pm$ 4% speed steps performed.

The B Reactor Recirculation (RR) MG Set exhibited a limit cycle of approximately 2 1/2% speed peak-to-peak when operating at 38% speed. Due to this limit cycle, the "B" speed loop response Decay Ratio could not be verified and will be retested when controller optimization is performed in Test Condition Three.

Flow control system limit cycles were verified and the peak-to-peak change in gross generator output during steady-state conditions was less than ± 0.5% of rated generator output or 11.5 MWe peak-to-peak. This criteria was satisfied with the largest observed generator output limit cycle of 10.55 MWe peak-to-peak (± .46% of rated output).

The peak APRM neutron flux was 57.71%. This APRM reading includes an APRM gain adjustment factor of 1.25 which was required due to a high core peaking factor. The calculated APRM neutron flux trip avoidance margin was 60.29%, satisfying the ≥ 7.5% criteria.

The minimum heat flux trip avoidance margin was 22.39% for the increasing speed steps, satisfying the criteria of ≥ 5.0%.

In Test Condition Three, testing was performed to demonstrate that the plant response to changes in recirculation flow was stable following flow control system tuning. Initial settings were also input to the dual limiter portion of the Master Flow Controller control circuitry in the Master Manual Mode of recirculation flow control. Jet pump baseline data for compliance with Technical Specifications Surveillance 4.4.1.2, Jet Pumps Operability, along the 75% rod line was obtained.

The test was conducted in three segments to support the above objectives.

The first segment consisted of individual Recirculation MG Set ± 9-11% speed step tests in local manual mode at the applicable region of highest gain for each Recirculation MG Set. The speeds associated with these regions of high gain were previously identified as 65% for RR MG Set "A" and 69% for RR MG Set "B". The second segment consisted of ± 4-6% speed steps in the Master Manual Mode of operation at RR MG Set speed/flow points of 96% flow, 70% RR MG Set speed and 50% RR MG Set speed.

The third segment consisted of adjustment of the dual limiter portion of the Master Flow Controller and obtaining baseline jet pump d/p data at 2% speed steps between 61% to 95% core flow along the 75% rod line. The sequence of testing was to reduce the Master Flow Controller M/A station output slowly until no further core flow decrease occurred. This point was then labelled as the minimum core flow point and loop speeds, core flow and "lo-pot" setting of the dual limiter were recorded. The point was verified by Master Flow Controller M/A station output increases until the core flow just started to respond. The Master Flow Controller was then adjusted in 2% increasing speed steps with flow allowed to reach steady state for each step. Jet pump baseline data was recorded when steady state conditions were obtained at each new speed plateau. This increase was halted at the electrical speed stop on RR MG Set "A". The "hi-pot" setting of the dual limiter for the Master Flow Controller was then adjusted to slightly lower the RR MG Set speeds to be less than the electrical high speed stop setting. This point was then labelled as the maximum core flow point and loop speeds, core flow and "hi-pot" setting of the dual limiter were recorded.

The dual limiter setpoints at the 75% rod line are tabulated below:

Minimum Flow Data:		Maximum Flow Data	
Core Flow:	61%	Core Flow:	95%
MG Set "A" Speed:	46%	MG Set "A" Speed:	78.5%
MG Set "B" Speed:	45.5%	MG Set "B" Speed:	78.5%
"Lo-Pot" Setting:	31	"Hi-Pot" Setting:	68

These settings will be evaluated again during the Step Change Testing/Ramp Test scheduled during Test Condition Six.

A review of the STARTREC traces for the above testing indicated that no variables related to the Recirculation System were divergent, thereby satisfying the Level 1 criteria for this test.

A qualitative review of the applicable STARTREC traces showed that the speed loop response of "A" RR MG Set loop controller, "B" RR MG Set loop controller and the Master Flow Controller to the required steps all met the Level 2 $\leq$ 0.25 damping criteria.

All of the post step steady state turbine steam flow variations were less than 11.5 MW(e) peak to peak, satisfying the applicable Level 2 criteria. The results are tabulated below:

Step	MW(e) Variation	Meets Criteria of <11.5 MW(e) Peak to Peak
Loop A 9-11% decrease (65% speed)	5.31	Yes
Loop A 9-11% increase (65% speed)	7.45	Yes
Loop B 9-11% decrease (69% speed)	6.83	Yes
Loop B 9-11% increase (69% speed)	8.07	Yes
Both Loops 4-6% decrease (96% flow)	6.83	Yes
Both Loops 4-6% increase (98% flow)	7.45	Yes
Both Loops 4-6% decrease (70% speed)	6.83	Yes
Both Loops 4-6% increase (70% speed)	7.45	Yes

Step	MW(e) Variation	Meets Criteria of <11.5 Mw(e) Peak to Peak
Both Loops 4-6% decrease (50% speed)	8.69	Yes
Both Loops 4-6% increase (50% speed)	6.83	Yes

All scram avoidance margins were met as shown by the values tabulated below:

Step	APRM High Flux Margin Margin ≥7.5%	Heat Flux Trip Avoidance Margin ≥5.0%	Meets Criteria
Loop A 9-11% increase (65% speed)	38.1%	36.1%	Yes
Loop B 9-11% increase (69% speed)	37.0%	34.9%	Yes
Both Loops 4-6% increase (96% flow)	37.1%	35.4%	Yes
Both Loops 4-6% increase (70% speed)	41.6%	28.2%	Yes
Both Loops 4-6% increase (50% speed)	52.6%	25.7%	Yes

With respect to the previously noted limit cycle (2 1/2% speed peak-to-peak) on the B Reactor Recirculacion MG Set speed loop at 38% speed, this problem has since disappeared after Scoop Tube cam shaping and has been attributed to an anomaly of the original cam.

Both A and B MG Sets do, however, exhibit speed oscillations of approximately 3% speed peak-to-peak in the 24-28% and 52-56% speed ranges. These oscillations are due to an inherent anomaly in the fluid coupler hydraulic system and this phenomenon is not specific to Fermi 2 MG Sets. The oscillations in the 24-28% speed range has been overcome by prohibiting operations below 28% speed.

Investigation as to the effect of the oscillation in the 52-56% speed region was conducted and the effect of these oscillations have been minimized by unbalancing the Reactor Recirculation MG Set speeds within the limitation of Tech. Spec. 3.4.1.3 while ramping flow upward through this region.

In Test Condition Six, step change testing in both local manual and master manual was performed on the 100% Load Line to demonstrate that the plant response to changes in recirculation flow was stable. Individual Recirculation MG Set ± 4-6% speed steps in local manual mode were performed at the previously identified highest gain region speeds of 65% for Reactor Recirculation MG Set "A" and 69% for Reactor Recirculation MG Set "B". Master manual mode speed steps of ± 2-3% were performed at Reactor Recirculation MG Set speed/flow points of 100% flow, 75% Reactor Recirculation MG Set speed and 65% Reactor Recirculation MG Set speed.

A qualitative review of the applicable STARTREC traces showed that the response of "A" RR MG Set Speed Loop Controller, "B" RR MG Set Speed Loop Controller and the Master Flow Controller to the required steps all met the Level 1 criteria for non divergence and the Level 2 ≤ 0.25 damping criteria.

All of the post step steady state turbine steam flow variations were less than 11.5 MW(e) peak to peak, satisfying the applicable Level 2 criteria. The results are tabulated below:

Step	MW(e) Variation	Meets Criteria of <11.5 MW(e) Peak to Peak
Loop A 4-6% decrease (65% speed)	7.4	Yes
Loop A 4-6% increase (65% speed)	6.8	Yes
Loop B 4-6% decrease (69% speed)	9.3	Yes

Step	MW(e) Variation	Meets Criteria of <11.5 MW(e) Peak to Peak
Loop B 4-6% increase (69% speed)	6.8	Yes
Both Loops 2-3% decrease (100% flow)	6.2	Yes
Both Loops 2-3% increase (100% flow)	6.2	Yes
Both Loops 2-3% decrease (75% speed)	5.6	Yes
Both Loops 2-3% increase (75% speed)	7.4	Yes
Both Loops 2-3% decrease (65% speed)	5.6	Yes
Both Loops 2-3% increase (65% speed)	8.1	Yes

All scram avoidance margins were met as shown by the values tabulated below:

Step	APRM High Flux Margin Margin ≥7.5%	Heat Flux Trip Avoidance Margin ≥5.0%	Meets Criteria
Loop A 4-6% increase (65% speed)	26.9%	20.2%	Yes
Loop B 4-6% increase (69% speed)	22.7%	21.6%	Yes
Both Loops 2-3% increase (100% flow)	12.1%	10.4%	Yes

Step	APRM High Flux Margin Margin ≥7.5%	Heat Flux Trip Avoidance Margin ≥5.0%	Meets Criteria
Both Loops 2-3% increase (75% speed)	19.2%	20.2%	Yes
Both Loops 2-3% increase (65% speed)	28.0%	19.7%	Yes

The remaining testing in this section to obtain Jet Pump baseline data along the 100% Rod Line and the verification/adjustment of the dual limiter setpoints of the master flow controller has not yet been completed.

3.28 Recirculation System

3.28.1 Purpose

a. To verify that the feedwater control system can satisfactorily control the water level without a resulting turbine trip/scram and obtain actual pump speed/flow.

b. To verify recirculation pump startup under pressurized reactor conditions.

c. To obtain recirculation system performance data.

d. To verify that no recirculation system cavitation occurs in the operable region of the power-flow map.

3.28.2 Criteria

Level 1

The response of any level-related variables during pump trips must not diverge.

Level 2

The simulated heat flux margin to avoid a scram shall be greater than or equal to 5.0 percent during the one pump trip recovery.

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

During the noncavitation verification, runback logic shall have settings adequate to prevent operation in areas of potential cavitation.

During the one pump trip, the reactor water level margin to avoid a high-level trip (L8) shall be greater than or equal to 3.0 inches.

3.28.3 Results

During Test Condition Two, recirculation system baseline performance data was recorded at 38.8% reactor power and 47.5% core flow and at 48% reactor power and 55.7% core flow.

Baseline Recirculation System Performance data at Test Condition Three power - flow conditions was collected at 47% power and 100% core flow.

Also during Test Condition Three, a test was run to verify that the recirculation pump runback limits are sufficient as to prevent operation where recirculation pump or jet pump cavitation is predicted to occur.

The test was conducted by establishing total core flow at 90% ($\pm$ 3%) of rated at a reactor power of 44.2%. Both Recirculation MG Set Scoop Tubes were locked and while reducing reactor power by the insertion of control rods, jet pump dp, recirculation pump vibration, drive flow, pump delta pressure, and pump suction temperatures were continuously monitored for indications of pump cavitation. Throughout the power reduction to the actuation of Limiter #1 at 23.6% of rated feedwater flow and 27.4% of rated reactor power, no indications of pump cavitation were observed. Reactor power was further reduced to 21.7% rated feedwater flow and 25.3% of reactor power at which point the power reduction was stopped due to indication of an increasing width of the recording of reactor core delta P which could be an early indication of cavitation. Therefore, it may be concluded that the runback logic settings are conservatively adjusted such that operation in areas of potential cavitation is prevented and that the Level 2 criteria has been satisfactorily met.

During Test Condition Six, recirculation system baseline performance data was recorded at 98.6% CTP and 100% CF.

Also during Test Condition Six, the one recirculation pump trip test from 95.9% CTP and 97% CF was conducted to verify that the feedwater control system can satisfactorily control the reactor water level without a resulting turbine trip/scram and to obtain data for actual recirculation MG Set speed vs flow in the single recirculation loop configuration.

The test was conducted from the above initial conditions by placing the Reactor Recirculation MG Set "A" motor CMC switch to the Off position.

Following the transient and after stable conditions were reached, the "B" Reactor Recirculation MG Set speed was reduced to 75% and speed/flow data in the single loop configuration was gathered in 2% (+ 1%) decreasing increments until 30% speed was reached. Test Condition Four was then entered by placing the "B" Reactor Recirculation MG Set motor CMC switch to the Off position. Upon completion of the required Test Condition Four tests (as described elsewhere in this report), "A" Reactor Recirculation MG Set was restarted to exit from Test Condition Four. Upon attempting the restart of "B" Reactor Recirculation MG Set, its discharge valve B31-F031B failed to close, and therefore restart logic could not be satisfied. The plant was shut down to effect repairs to this valve and after restart and power ascension to the same approximate initial conditions of 59.9% CTP and 46.9% CF, "B" Reactor Recirculation MG Set was tripped and a restart attempted with the same resulting failure of B31-F031B to close. The plant is currently shut down to further investigate the problem.

A qualitative review of the response of level related variables recorded during the one pump trip from 95.9% CTP showed that the Level 1 criteria for this test was met since none were divergent. The margin to the reactor water high level (L8) trip was 5.1 inches thereby satisfying the Level 2 criteria of $\geq$ 3.0 inches.

The Level 2 criteria of APRM and Simulated Heat Flux Margins to Scram could not be verified for the one pump trip recovery due to the inability to restart the "B" Reactor Recirculation MG Set.

The restart of the "A" Reactor Recirculation MG Set from Test Condition Four was, however, evaluated to this criteria with the APRM Margin to Scram being 79.6% and the Simulated Heat Flux Margin to Scram being 22.4% thereby satisfying the Level 2 criteria.

The balance of this test to verify the restart of recirculation pumps under pressurized reactor conditions with the other pump running will be performed when the plant returns to operation.

3.29 Loss of Turbine-Generator and Offsite Power

3.29.1 Purpose

a. To determine the reactor transient performance during the loss of the main generator and all offsite power.

b. To demonstrate acceptable performance of the station electrical supply system.

3.29.2 Criteria

Level 1

The reactor protection system, the diesel-generator, RCIC and HPCI must function properly without manual assistance. HPCI and/or RCIC system action, if necessary, shall keep the reactor water level above the initiation level of low-pressure core spray, LPCI, and automatic depressurization systems.

Level 2

If the low-low set pressure relief logic functions, the open/close actions of the SRVs shall occur within $\pm$20 psi of their design setpoints. If any safety relief valves open, only one may reopen after the first blowdown.

3.29.3 Results

The test was initiated during Test Condition Two by isolating the plant from off-site power by simultaneously opening both the 345 KV and 13.2 KV feeds to the in-plant busses.

It was demonstrated that the following actions occurred once the test was initiated without any operator assistance:

1. The Reactor Protection System automatically scrammed the reactor.

2. The Turbine/Generator Protection System automatically initiated a trip and fast closure of the Main Turbine steam admission valves.

3. The Emergency Diesel Generators automatically started and properly loaded the ESF busses, and

4. Control of reactor water level and pressure during transient conditions were maintained.

It was also demonstrated that the required equipment and support systems operated satisfactorily without dependence on off-site power sources for the extended test duration of 30 minutes. No automatic initiation signal/setpoint was received for either HPCI or RCIC. The lowest reactor water level reached during the test was 138.8 inches. The Level 1 setpoint of 31.8 inches, at which Core Spray, LPCI and ADS are initiated, was therefore avoided by a significant margin. Based on the above, the Level 1 criteria for this test was successfully met.

Following the first blowdown, only SRV B21-F013A reopened. This satisfies the Level 2 criteria requirement that specifies only one SRV may open at that time.

The low-low set pressure relief function for two low-low set valves, SRV "A" and SRV "G" was actuated during the test. On increasing reactor pressure, six SRVs lifted at a pressure of 1100.1 psi. These actuations were in accordance with the Level 2 criteria required for this test.

This concludes all Loss of Turbine/Generator and Off-Site Power testing during the Startup Test Phase program.

3.30 Steady-State Vibration

3.30.1 Purpose

To determine the vibration characteristics of the primary pressure boundary piping (NSSS) and ESF (ECCS) piping systems for vibrations induced by recirculation flows, hot two-phase forces, and hot hydrodynamic transients; and to demonstrate that flow-induced vibrations, similar in nature to those expected during normal and abnormal operation, will not cause damage and excessive pipe movement and vibration.

3.30.2 Criteria

Level 1

The measured vibration levels of the piping shall not exceed the acceptable specified values.

Level 2

The measured vibration levels of the piping must not exceed the expected specified values.

3.30.3 Results

During Test Condition One, the RCIC Steam Supply Line inside the drywell and tne RCIC Pump Discharge Line near its connection to the Feedwater Line were monitored for vibration using installed sensors during a vessel injection at rated conditions. Evaluation of the data showed that all vibration levels were within acceptable values.

During Test Condition Two, steady state vibration was measured for selected piping systems at 25% (± 5%) of rated steam flow and at 50% (± 5%) of rated core flow. Data was initially gathered for seven piping systems consisting of Feedwater, Main Steam, Reactor Recirculation, RHR, SRVs D&J, HPCI and RCIC. More data was collected at a later date for eight locations on the Main Steam piping and one location on the RCIC piping at 25% and 29% rated steam flow.

This extra testing was necessary because the Level 1 criterion for six of these locations were exceeded in the initial set of data. Also, more data was needed to determine the impact of the removal of snubbers from piping between the Turbine Control Valves and the High Pressure Turbine.

A total of eight Level 1 criterions for instruments D-015, D-016, D-017, A-014, A-015, and A-016, were exceeded in this second set of data. However, based on hand held vibration measurements and/or detailed pipe stress analysis by Sargent and Lundy, all criteria violations were found acceptable.

Revised criteria levels for selected sensor locations were incorporated into future test plans.

During Test Condition Three, vibration data was collected to determine the flow induced vibration responses of the Main Steam Lines, Reactor Recirculation Loops, Feedwater, HPCI, RCIC, RHR and Safety Relief Valve piping during steady-state vibration hardwired testing. Steady-state vibration data was obtained and analyzed for 80 (± 5)% and 100 (± 5)% of rated core flow, and 50% (± 5%) and 75% (± 5%) of rated steam flow. Post transient steady-state data was also obtained following two HPCI RPV injection for HPCI piping sensors.

There was a total of two (2) exceedences to the Level 1 criteria as follows during the 80% core flow data collection:

Sensor	Level 1 mils p-p	Measurement mils p-p
A-014	10	11.3
A-015	14	49.6

For sensors A-014 and A-015, it was determined that their readings were unreliable, and that vibration for this area of piping is acceptable based on the readings of sensors D-009, D-010 and D-011.

There was one exceedence to the Level 2 criteria during the 100% Core Flow data collection.

Sensor	Level 2 inch p-p	Measurement inch p-p
SA-RZ	0.024	0.027

The Level 2 criteria exceedence for sensor SA-RZ was evaluated and considered to be acceptable. Review of the same sensor data at 25% steam flow and 50% core flow showed satisfactory peak-to-peak amplitude.

There was one exceedance to the Level 1 criteria during the 75% Steam Flow data collection.

Sensor	Level 1 inch p-p	Measurement inch p-p
A-006	50	77

For sensor A-006, it was determined after an evaluation by Sargent and Lundy given the as-built location of the sensor, reviewing the design calculations and similar test data, that the vibration criteria of A-006 would be acceptable to a new increased value of 84 mils peak-to-peak.

Although sensor A-601 (RHR Head Spray Line) appeared to have exceeded its Level 1 criteria based on vibration data taken at 70.1% steam flow, it had previously been determined from data at approximately 60% steam flow that this sensor was not giving a true reading. This diagnostic data was collected to determine sensor operability prior to raising power and it was noted that this particular sensor's reading was not consistent with the other eight (8) neighboring sensors on the same line. An evaluation was performed at that time by Sargent and Lundy and it was determined that this sensor reading was not correct and would not be a restraint to raising reactor power.

There was also one exceedance to the Level 2 criteria during the 75% Steam Flow data collection.

Sensor	Level 2 inch p-p	Measurement inch p-p
SA-RZ	0.024	0.026

The Level 2 criteria exceedance for sensor SA-RZ was previously evaluated during the 100% Core Flow data collection and found to be acceptable.

There were no piping vibration criteria exceedances during the 50% (± 5%) Steam Flow data collection.

Post transient steady-state data following the HPCI RPV injection was analyzed and found acceptable; however, this data collection was repeated due to the subsequent replacement of E41-F005, HPCI Discharge Check Valve. Results from this additional data collection during a HPCI RPV Injection in October of 1987 were also acceptable.

During Test Condition Six, vibration data was collected to determine the flow induced vibration responses of the Main Steam Lines, Reactor Recirculation Loops, Feedwater, RCIC, HPCI, RHR and Safety/Relief Valve piping during steady state operation of the plant with Main Steam flow between 95 and 100% of rated steam flow.

Vibration data was collected during the following steady-state conditions:

Date Performed	Power	Core Flow	Main Steam Flow
July 9, 1988	97.1	97	95.3
July 11, 1988	96.74	97.5	95.3

In the first data set, there were two (2) apparent Level 1 criteria exceedances associated with accelerometers A-052 and A-053. These accelerometers are located on Safety Relief Valve B21-F013E piping. However, based on Sargent and Lundy recalculations of allowables transmitted to the site on May 20, 1988, the Level 1 criteria associated with these sensors were being revised and the sensor readings were within the new Level 1 criteria.

Sensor	Level 1 Mils p-p	Measurement Mils p-p	New Level 1 Mils p-p
A-052	9	16	17
A-053	8	12	16

There were no Level 2 criteria exceedances.

In the second set of data performed to the revised procedure, no Level 1 and only one (1) Level 2 criteria violation were found.

Sensor	Measurement Mils p-p	Level 1 Allowable Mils p-p	Level 2 Expected Mils p-p
SB-RZ2	76	110	56

This exceedance has been reviewed by General Electric and has been found acceptable.

Also during the Test Condition Six steady state testing, the vibration traces corresponding to Lanyard Potentiometers RA-SX1 (Recirculation Loop A), and RB-HX4 (Recirculation Loop B) and Accelerometer A-108 were indicative of bad sensors. Based on other adequate data from nearby sensors, the omission of these three from the data set was found acceptable by both Sargent and Lundy and General Electric.

3.31 Recirculation System Flow Calibration

3.31.1 Purpose

To perform a complete calibration of the installed recirculation system flow instrumentation.

3.31.2 Criteria

Level 1

None

Level 2

Jet pump flow instrumentation is adjusted so that the jet pump total flow recorder provides a correct core flow indication at rated conditions.

The APRM/RBM flow-bias instrumentation is adjusted to function properly at rated conditions.

The flow control system shall be adjusted to limit maximum core flow to 102.5 percent of rated flow by limiting MG set scoop tube position.

3.31.3 Results

During Test Condition Three at a reactor power of 45%, a total core flow calibration was performed using Reactor Engineering procedure 56.000.02. This was the first core flow calibration performed and therefore, approximately 5% margin was established between rated and indicated core flow.

During the initial run, the jet pump milli-volt readings were found to be varying making it difficult to obtain accurate readings. Several readings were taken at each square rooter. The highest and lowest readings were averaged together and the average value was recorded. The Reactor Engineering procedure required that the jet pump square rooter output be within .25 ma of the expected output based upon measured input. This requirement was not initially met. The Reactor Engineering in-house code calculated a total core flow of 97.7%. This value compared well against the General Electric code, JRPUMP, (which calculated core flow to be 97.6%). This is a very good agreement since the Reactor Engineering code used jet pump instrument span from I&C calibration sheets, while JRPUMP used the design instrument span of 10-50 ma.

An RC network was developed to filter the jet pump milli-volt readings and the Reactor Engineering procedure was run a second time. Milli-volt readings were taken simultaneously from the input and output jacks of the square root extractors. This method enabled us to meet the requirement that the output of the square root be within .25 ma of the expected output. The filter helped, but did not prevent, the oscillations in the milli-volt readings.

The Reactor Engineering procedure was run a third time using a different filter with a 4-5 second time constant. The milli-volt readings were still unstable but average values were recorded. Core flow was calculated to be 100.0% by the Reactor Engineering code, while JRPUMP calculated core flow to be 99.8%. The flow calibration was completed by adjusting B21-602 A, B and B31-607 A, B, C, D summers, which satisfies the Level 2 criteria for the adjustment of instrumentation providing core flow indication and APRM/RBM flow-bias. The recirculation system was placed in MASTER MANUAL. Speed and flow data was collected while flow was decreased from 100% to 80%. Flow vs speed data was plotted for this range. This data was extrapolated out to 102.5% core flow to obtain the corresponding speed. Flow was increased to 95%. The recirculation system was placed in the LOCAL MANUAL mode. MG Set "A" speed was increased to 850 rpm (equivalent to 102.5% flow). The mechanical stop was set at this speed. The electrical stop was set 6/64" before the mechanical stop based upon the scoop tube positioner. This position is at 840 rpm (equivalent to 101% flow). The mechanical and electrical stops were set in a similar manner on MG Set "B". MG Set "A" speed was reduced and MG Set "B" was increased. The mechanical stop was set at 868 rpm (equivalent to 102.5% flow). The electrical stop was set at 855 rpm (equivalent to 100.7% flow). This is 14/64" before the mechanical stop based upon the scoop tube positioner. This satisfies the Level 2 criteria of limiting the maximum core flow to 102.5% of rated by limiting the MG Set Scoop Tube positions.

Later during Test Condition Three at a reactor power of 71% and 94.5% core flow, a total core flow calibration was performed using Reactor Engineering Procedure 56.000.02. This power and core flow were sufficiently high to obtain data at the upper end of the Test Condition Three window.

The latest revision of the Reactor Engineering Procedure incorporated the lessons learned from the previous core flow calibration performed at 45% CTP. The RC filter network was modified for ease of connection and proper resistor configuration and the procedure was modified to ensure that this filter was properly applied. This resolved previous concerns with APRM flow unit GAFs and jet pump summer adjustments and facilitated the adjustment portion of this test. The basic sequence was that data was gathered following a calibration check of jet pump loop instrumentation. The data was used in the in-house Reactor Engineering code to calculate total core flow, jet pump loop flows and APRM Flow Unit Gain Adjustment Factors. Total core flow correction was not required. Jet pump loop flows required adjustments as one loop was indicating greater than calculated flow and the other was indicating less than calculated flow. The APRM flow units also required adjustment because of the increased drive flow required for the same core flow (decreased M-ratio). These adjustments were calculated as Gain Adjustment Factors (GAFs) and were successfully applied to the flow units. The existing settings of the Reactor Recirculation MG Sets scoop tube high speed electrical stops required that a small adjustment be made to the high speed stops so that the required speed to obtain 100% indicated flow could be reached. This limited the available data for extrapolation of the final 102.5% flow equivalent speed and coupled with the requirements for conservatism with respect to Technical Specifications setpoints combined to provide sufficient error in the settings of the electrical high speed stops to preclude obtaining 100% flow at predicted high speed stop settings. The procedure allows for readjustment of the high speed stops if required to obtain 100% core flow, but does not require the adjustment. Due to the nonlinear behavior of two phase flow losses from the 50% rod line to the 75% rod line, it was determined that further adjustments would not be done at this time, but rather would be adjusted at the 100% rod line at approximately 90% power.

The Level 2 criteria associated with this test were satisfied as follows:

a. The B21-K602A/B jet pump flow loop summers were adjusted in accordance with the flow calculated in Reactor Engineering Procedure 56.000.02.

   The computer code output indicated the following gain adjustment factors for B21-K602A/B:

   B21-K602A (GAF) = 0.9520
   B21-K602B (GAF) = 1.0380

   Composite gains were calculated per Reactor Engineering Procedure 56.000.02 and entered into the Reactor Engineering data book.

   The composite gains for the jet pump loops were:

   B21-K602A (CGAF) = 1.000
   B21-K602B (CGAF) = 1.040

b. The B31-K607 A,B,C,D summers (APRM/RBM flow-bias) were adjusted using the GAFs calculated in Reactor Engineering Procedure 56.000.02.

   The computer code output indicated the following Gain Adjustment Factors for flow units B31-K607A through B31-K607D:

   Flow Unit A - B31-K607A (GAF) = 1.021
   Flow Unit B - B31-K607B (GAF) = 1.039
   Flow Unit C - B31-K607C (GAF) = 1.034
   Flow Unit D - B31-K607D (GAF) = 1.019

c. The mechanical high speed stops were set at 102.5% core flow which equated to 81.4% speed (912 rpm) for RR MG Set A and 82.3% (922 rpm) for RR MG Set B. The electrical high speed stops were set at 100% core flow for the "A" MG Set which equated to 80.0% speed (896 rpm), and 100% for the "B" MG Set which equated to 80.5% speed (902 rpm).

After completion of the 100 hour commercial run at > 90% CTP but prior to entry into Test Condition Six, another calibration of the recirculation system flow instrumentation was run at 95.5% CTP to determine the scoop tube electrical and mechanical stop positions.

Reactor Engineering Procedure 56.000.02 was used to determine the actual core flow at the time of this test. The Process Computer program OD-3 Option 2 was run to obtain the initial conditions for this test, and based on that OD-3 Option 2 edit, the core flow was 96.21 Mlbm/hr. The core flow as indicated on the control room recorder B21-R613 was 94 Mlbm/hr. However, the actual core flow was determined to be 91.82 Mlbm/hr and the jet pump summers and flow units were adjusted to bring the indicated flow within acceptable agreement (within ± 2% of rated flow) of the calculated flow. After the adjustments were made, the indicated core flow on the control room recorder B21-R613 was 91.0 Mlbm/hr and the core flow obtained via the Process Computer was 92.77 Mlbm/hr. Reactor Engineering Procedure 56.000.02 was also used to determine the Gain Adjustment Factors to be applied to the recirculation system flow units which were used to adjust the Technical Specification required flow biased rod block and scram setpoints. Prior to these adjustments, the flow biased APRM rod blocks were set approximately at the rated load line causing rod blocks when approaching the rated load line. The average GAF applied to the flow units was 1.09 which raised the APRM flow biased rod block line about 6% above the rated load line. This provided the margin to be able to operate at or near the rated load line with the rod block annunciator cleared. The calculation portion of REP 56.000.02 was run twice: first by using an estimated number of 33.0 Mlbm/hr for the rated recirculation drive flow and the second time by using the actual value of the rated drive flow as determined from the new M-ratio. The new M-ratio was determined to be 2.198 for loop A and 2.168 for loop B. The design M-ratio is 1.84 and the difference in the design versus actual M-ratios suggest that at the present time, the actual rated recirculation drive flow is 31.4 Mlbm/hr and not 35.2 Mlbm/hr. An evaluation run of REP 56.000.02 will be performed later at rated plant conditions to obtain the updated M-ratio, core dP and indicated versus actual core flow values.

Once the core flow and recirculation flow unit adjustments were made, core flow was reduced from 91 Mlbm/hr indicated (92.93 Mlbm/hr from the Process Computer) to 81 Mlbm/hr indicated (83.28 Mlbm/hr from the Process Computer) in approximately 2% core flow steps. Core flow versus recirculation MG Set speed data was collected at each step.

The core flow versus percent of rated MG Set speed was graphed and used to extrapolate the Recirculation MG Set speed corresponding to 102.5% of rated core flow.

The extrapolated speed was determined using the core flow signal from the Process Computer instead of Control Room recorder B31-R613 to set the mechanical stops as this resulted in a more conservative (lower) Recirculation MG Set speed.

Once the MG Set speeds corresponding to 102.5% core flow was determined, the mechanical and electrical stops for both A and B MG Sets were set. This was accomplished by first increasing the recirculation speed to 79% in individual manual mode. At this point, A MG Set speed was increased to 86.2% speed (equivalent to 102.0% core flow) using the M/A station B31-R621A. The previous electrical and mechanical stop settings were adjusted to allow this speed increase. The "A" MG Set scoop tube was then locked and the speed manually cranked up to 970 rpm (86.6% speed) corresponding to 102.5% core flow and the mechanical stops set. The MG Set A speed was manually cranked down to 953 rpm (85.1%) to set the electrical stops. This was the speed associated with 100.9% of rated core flow. Once this was accomplished, the A MG Set scoop tube was unlocked and the speed reduced to 79% speed to match the B MG Set speed. The same procedure was repeated for B MG Set, and the mechanical stop was set at 960 rpm (85.7%) corresponding to 102.2% core flow and the electrical stop was set at 948 rpm (84.6%) which is equivalent to 101.0% of rated core flow.

The Master flow control limiter Hi-pots were set to provide a maximum flow (in Master Manual mode) of 98% as indicated on the B31-R613 recorder or 100% core flow per Process Computer Program OD-3 Option 2.

The Level 2 criteria associated with this test were met as follows:

a. The computer code output indicated the following gain adjustment factors for B21-K602A/B jet pump flow loop summers at 91.82% core flow:

   B21-K602A (GAF) = 0.9672
   B21-K602B (GAF) = 0.9762

Composite GAFs for the jet pump loops were calculated per of Reactor Engineering Procedure 56.000.02 and were as follows:

B21-K602A (CGAF) = 1.038
B21-K602B (CGAF) = 1.015

An evaluation run of 56.000.02 is planned at rated conditions to verify that this Level 2 criteria is indeed satisfied.

b. The B31-K607 A,B,C,D summers (APRM/RBM flow-bias) were adjusted using the GAFs calculated in Reactor Engineering Procedure 56.000.02.

The previous GAFs were all set at 1.00 so the Composite Gain Adjustment Factors (CGAF) were equal to the calculated (GAFs).

The computer code output indicated the following Gain Adjustment Factors for flow units B31-K607A through B31-K607D:

Flow Unit A - B31-K607A (GAF) = 1.083
Flow Unit B - B31-K607B (GAF) = 1.092
Flow Unit C - B31-K607C (GAF) = 1.098
Flow Unit D - B31-K607D (GAF) = 1.077

c. As previously noted, the mechanical stops for MG Sets scoop tubes were set at 86.6% speed (970 rpm) for "A" and 85.7% (960 rpm) for "B". This translates to an equivalent flow of 102.5% for "A" and 102.2% for "B".

During Test Condition Six, another complete calibration of the recirculation system flow instrumentation was performed at 96.97% CTP and 96.54% CF to resolve issues raised in the previous core flow calibration.

REP 56.000.02, Core Flow Calibration, was performed in the performance mode for this test. As a prerequisite, the zero and full span settings of single tapped and double tapped jet pump transmitters and square root converters were checked and calibrated as necessary. In addition, the Recirculation Flow Units were calibrated as required per REP 56.000.02 and the zero and full span settings of the Recirculation Flow Nozzle transmitters were obtained and documented.

A complete set of single tapped and double tapped jet pump delta Pressure measurements were obtained, and core flow evaluations were performed per this procedure. These evaluations revealed that the Loop Flow Variations for Loop A was 3.18% which exceeded the 3% criteria as stated. To resolve this issue, another set of measurements was obtained and evaluated per this procedure, and this time the Loop Flow Variation criteria for Loops A and B were 2.06% and 0.08%, respectively, which met the 3% criteria. This time, however, the Nozzle Plugging criteria, as stated in REP 56.000.02, associated with jet pumps 9/10 and jet pumps 11/12 was exceeded. To resolve this discrepancy, an attempt was made to obtain a set of readings associated with jet pumps 9 and 10 and for jet pumps 11 and 12 such that these readings were taken at the same time every 10 seconds for 2 minutes and averaged before they were compared. Based on this, the nozzle plugging criteria was satisfied. This is due to the fluctuating nature of these readings, and the outcome is quite sensitive to how and when these readings are taken. The actual core flow was determined to be 96.54 Mlbm/hr by performing the required calculations in REP 56.000.02. The indicated core flow on the control room recorder B21-R613 was reading 96 Mlbm/hr and the process computer OD3 Option 2 edit was showing 97.73 Mlbm/hr. The loop drive flows at the time of this test were calculated to be 14.87 Mlbm/hr for Loop A and 15.06 Mlbm/hr for Loop B for a total drive flow of 29.93 Mlbm/hr. New M ratios were determined to be 2.159 for Loop A and 2.288 for Loop B. Total Rated drive flow was determined to be 31 Mlbm/hr and the jet pump summers and flow units were adjusted based on the Gain Adjustment Factors derived in REP 56.000.02.

When core flow was increased to 100% of rated core flow, the recirculation MG set speeds were measured using a Strobe-o-tac to determine how this data point would compare with the MG Set speed versus core flow projection made during the previous core flow calibration. The results were as follows:

	MG Set Speed (rpm)	
	A	B
100% Core Flow	928	943
Existing Electrical Stop	953	948
Existing Mechanical Stop	970	960

Using the projections obtained from previous test and superimposing the 100% core flow data point from this test on those points, the projected core flows associated with the electrical and mechanical stops are as follows:

Projected Core Flows (%)

	MG Set A	MG Set B	Average
At Electrical Stop	102.0	101.0	101.5
At Mechanical Stop	103.7	102.2	102.95

Based on this, both MG sets at their respective electrical stops would result in a projected core flow of 101.5% of rated. The MG sets at the mechanical stops would result in a projected core flow of 102.95% of rated. These are within the requirements of Technical Specification Surveillance 4.4.1.1.2 which states that the electrical and mechanical stops be set at less than or equal to 102.5% and 105.0% of rated core flow, respectively. Based on this, it was decided that the electrical and mechanical stops were not required to be adjusted.

Also at 100% of Rated flow, the core dP was found to be 20.26 psid which is 1.61 psi below prediction which supports supplemental data analysis requirements.

The Level 2 criteria associated with this test were met as follows:

The computer code output indicated the following gain adjustment factors for B21-K602A/B at 96.54% core flow.

B21-K602A (GAF) = 1.025
B21-K602B (GAF) = 1.017

Composite GAFs for the jet pump loops were calculated per Reactor Engineering Procedure 56.000.02 and were as follows:

B21-K602A (CGAF) = 1.064
B21-K602B (CGAF) = 1.032

Jet Pump Summers Gain Adjustments were made per REP 56.000.02 to satisfy this criteria.

The B31-K607 A,B,C,D summers were adjusted using the GAFs calculated in Reactor Engineering Procedure 56.000.02.

The computer code output indicated the following Gain Adjustment Factors for flow units B31-K607A through B31-K607D:

Flow Unit A B31-K607A GAF = 1.020
Flow Unit B B31-K607B GAF = 1.020
Flow Unit C B31-K607C GAF = 1.007
Flow Unit D B31-K607D GAF = 1.014

Composite GAFs for Flow Units B31-K607A through B31-K-607D were calculated per REP 56.000.02 and were as follows:

Flow Unit A B31-K607A GAF = 1.105
Flow Unit B B31-K607B GAF = 1.114
Flow Unit C B31-K607C GAF = 1.106
Flow Unit D B31-K607D GAF = 1.092

Adjustments were made per REP 56.000.02, and this Level 2 criteria has been satisfied.

The mechanical stops for MG sets were set at 86.6% speed for "A" and 85.7% for "B". This translates to an equivalent flow of 103.7% for "A" and 102.2% for "B" or an average of 102.95 for both which exceeds the criteria for this test but is within the requirements of Technical Specifications, and therefore no adjustments were made as previously discussed.

This concludes the recirculation system flow calibrations during the Startup Test Program.

3.32 Reactor Water Cleanup System

3.32.1 Purpose

The purpose of this test is to demonstrate specific aspects of the mechanical operability of the reactor water cleanup system.

3.32.2 Criteria

Level 1

None

Level 2

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

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

The bottom head flow indicator will be recalibrated against the RWCU flow indicator if the deviation is greater than 25 gpm.

The pump available NPSH is 13 feet or greater during the hot shutdown with loss of RPV recirculation pumps mode defined in the process diagrams.

3.32.3 Results

During the Heatup test condition, the RWCU system was placed in a configuration so that flow was taken from the bottom drain and directly fed back to the vessel, bypassing the demineralizers. In this configuration G33-610, bottom drain flow, should read the same as G33-609, system inlet flow. Our data showed a maximum deviation of 62 gpm. Bottom drain flow was recalibrated such that the Level 2 criteria could be satisfied.

Also during Heatup, the RWCU system was operated in both the normal and blowdown modes with the reactor at rated temperature and pressure. Process variables were recorded in order to demonstrate the proper performance of the RWCU system in each of these modes. The non-regenerative heat exchange tube side outlet temperatures for the normal and blowdown mode were 112°F and 122°F respectively. These values were within the Level 2 criteria limits of 120°F and 130°F for each mode. Using temperature measurements from the RBCCW side of the non-regenerative heat exchangers (NRHX) the cooling water flow was calculated to be less than 6% above the NRHX capacity. The non-regenerative heat exchanger cooling water outlet temperatures were well within our Level 2 criteria of 180°F. All applicable Level 2 criteria were satisfied.

During Test Condition Four, the Hot Standby Mode of the Reactor Water Cleanup System was used to prevent temperature stratification in the reactor vessel while the recirculation pumps were not running. Data was taken in this mode to determine the RWCU pump NPSH during natural circulation conditions where NPSH is most limiting. The actual NPSH was calculated to be 702 feet and is consistent with the NPSH measured at other BWR plants. This satisfies the Level 2 criteria that NPSH be $\geq$ 13 feet in the most limiting mode of operation.

This concludes the required Startup Test Program testing on the Reactor Water Cleanup System.

3.33 Residual Heat Removal System

3.33.1 Purpose

The purpose of this test is to demonstrate the ability of the Residual Heat Removal (RHR) System to remove residual and decay heat from the nuclear system so that refueling and nuclear servicing can be performed.

3.33.2 Criteria

Level 1

None

Level 2

The RHR System is capable of operating in the suppression pool cooling and shutdown cooling modes at the flow rates and temperature differentials indicated on the process diagrams.

3.33.3 Results

During the Heatup test phase, each division of the RHR system was placed in the Suppression Pool Cooling Mode and process data was taken for a 30 minute time period. The extrapolated heat capacity for both heat exchangers indicated an excess capacity of 67.5%. This was expected since in early heat exchanger life the heat transfer coefficient is larger and capacity was determined to accommodate some deterioration.

The remaining testing in this section to demonstrate the operation of RHR in the shutdown cooling mode has not yet been performed. It is anticipated that this test will be coordinated with the MSIV full isolation and Shutdown from Outside the Control Room/Cold Shutdown Demonstration tests.

3.34 Piping System Dynamic Response Testing

3.34.1 Purpose

Verify that piping system structural behavior under probable transient loadings is acceptable and within the limit predicted by analytical investigations.

3.34.2 Criteria

Level 1

The measured vibration levels of the piping shall not exceed the acceptable specified values.

Level 2

The measured vibration levels of the piping must not exceed the expected specified values.

3.34.3 Results

Piping dynamic transient vibrations were monitored during Heatup, in conjunction with Relief Valve testing, for two SRV lines and selected Main Steam Lines. All vibration data recorded was within the acceptable and expected limits as defined by the Level 1 and Level 2 criteria.

Piping dynamic transient vibrations were monitored during Test Condition Two in conjunction with relief valve actuations during relief valve testing, and during the planned Turbine/Generator Load Reject (Within Bypass) test. Data for the two SRV lines and the Main Steam Lines showed all vibration data was within Level 1 and Level 2 criteria except D-001, which was inoperable, and D-003, D-005 and D-008 which did not meet Level 2 criteria. All violations were reviewed and evaluated by Sargent and Lundy and were found to be acceptable. It is worth noting that the original criteria for these instruments were given as "information only" and were mistakenly incorporated into the procedure as Level 2 criteria.

During Test Condition Three, data was collected to determine the flow induced vibrational response of the High Pressure Coolant Injection (HPCI) system piping during a planned HFCI System cold vessel injection to the reactor.

During the first successful HPCI cold vessel injection to the reactor the load on force pin F-155, located at the HPCI discharge, exceeded its Level 1 criteria. After a detailed walkdown of the HPCI supports and upon completion of further analysis of the HPCI System pipe supports by Sargent and Lundy, three additional strain gauge networks on three other HPCI supports were installed to monitor strains during the next cold injection.

During that cold injection, all Level 1 and 2 criteria were satisfied. The additional strain gauges were monitored and these values were provided to Nuclear Engineering for evaluation and were found acceptable.

Subsequent to this test, a HPCI vessel injection was performed on 7-5-87 which resulted in a HPCI overspeed trip. During that event, a water hammer and suction line overpressurization transient occurred (reference LER-87-030-00) which, after engineering analysis, has resulted in the replacement of E41-F005, HPCI Discharge Check Valve and several HPCI System hanger modifications.

Due to these changes in HPCI piping configuration, this testing was reperformed to evaluate HPCI piping response during the next planned HPCI Quick Start testing sequence.

That additional test was performed in conjunction with a Hot Quick Start Vessel Injection on October 14, 1987 and resulted in all criteria being met. The results from the three tests are summarized below:

Sensor	6/22/87	7/4/87	10/14/87	Level 1 Allowable
F-155	13966 lb_f	4929 lb_f	2164 lb_f	10000 lb_f

As described in Reference 1.5.3 of this report, the initial test program was changed to allow taking credit for certain inadvertent scrams if a supporting analysis of the collected data demonstrated that the test results are valid when extrapolated to higher power levels.

On December 31, 1987 an inadvertent turbine/generator trip did occur while at 74.8 percent reactor power. The UFSAR required analysis of the data collected from the turbine/generator trip was subsequently performed by General Electric. The analysis demonstrated that although the trip occurred at a lower power level, the purpose of the test and all test criteria were satisfactorily met for the NSSS (GE) portion of the vibration data. This is documented in Reference 1.5.5 of this report.

It should be noted also that the Sargent and Lundy required data was also recorded for this test. However, a failed sensor at the time of the inadvertent turbine trip induced noise in the sensor readings for the Sargent and Lundy data. This sensor, F-006, has been confirmed as the source of the unusual sensor readings and has subsequently been removed from service.

Consequently, the Sargent and Lundy data recorded is suspect as are any associated criteria violations. The Sargent and Lundy vibration data will be recorded during a future turbine trip either inadvertent or scheduled. The Main Steam instrumentation in the drywell was unaffected by the failed F-006 and the data collected was found satisfactory by the General Electric NSSS piping analysis.

During Test Condition Six prior to the entry into Test Condition Four, piping vibration data was recorded for the recirculation loops when Recirc. Pump A was tripped with the reactor operating at 97% power. There were two Level 2 criteria violations that were evaluated by General Electric and found acceptable.

The violations were as follows:

Sensor	Level 1 Inches p-p	Level 2 Inches p-p	Measured Vibration Inches p-p
RA-SX	1.626	0.038	0.202
RA-HY	0.318	0.008	0.029

After Recirc. Pump B was reduced to minimum speed, vibration data was obtained when it was tripped to bring the plant to natural circulation conditions. All criteria was satisfied.

Subsequently, Recirc. Pump A was restarted on the recovery from Test Condition Four. Vibration data taken during the restart satisfied all criteria. The Recirc. Pump B restart could not be performed at this time due to a torque switch problem with the Recirc. Pump B Discharge Valve, B31-F031B and will be completed when the plant returns to power operation and this portion of testing is repeated.

B. Ralph Sylvia
Senior Vice President

Detroit
Edison

6400 North Dixie Highway
Newport, Michigan 48166
(313) 586-4150

Plant Technical
Specifications

September 20, 1988
NRC-88-0212

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

References: (1) Fermi 2
NRC Docket No. 50-341
Facility Operating License No. NPF-43

(2) Detroit Edison Letter to NRC "Startup
Report Supplement 8" NRC-87-0136, dated
June 20, 1988.

Subject: Startup Report - Supplement 9

This is Supplement 9 of the Startup Report for Fermi 2. As required by Fermi 2 Technical Specification 6.9.1.3, a supplement is being submitted every 3 months until completion of the Startup Test Program. A supplemental report will be submitted by December 20, 1988.

If you have any questions regarding this report, please contact Patricia Anthony, Compliance Engineer at (313) 586-1617.

Sincerely,

B Ralph Sylvia

cc: A. B. Davis
R. C. Knop
T. R. Quay
W. G. Rogers

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