THE DETROIT EDISON COMPANY

FERMI 2
NUCLEAR POWER PLANT

INTERIM STARTUP TEST REPORT

SUPPLEMENT NO. 4

June 10, 1987

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FOREWARD

This Supplementary Startup Test Report includes the testing performed since the previous interim summary report dated March 10, 1987. This report was transmitted to the NRC via VP-NO-87-0055 dated March 20, 1987. Since that report was issued Fermi 2 has completed all of the tests required for Test Condition Two except the balance of the feedwater system testing. Plant operation is presently limited to 50% power and selected tests in the lower region of the Test Condition Three envelope are planned.

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.

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

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

FIGURE 1-1

STARTUP TEST PROGRAM

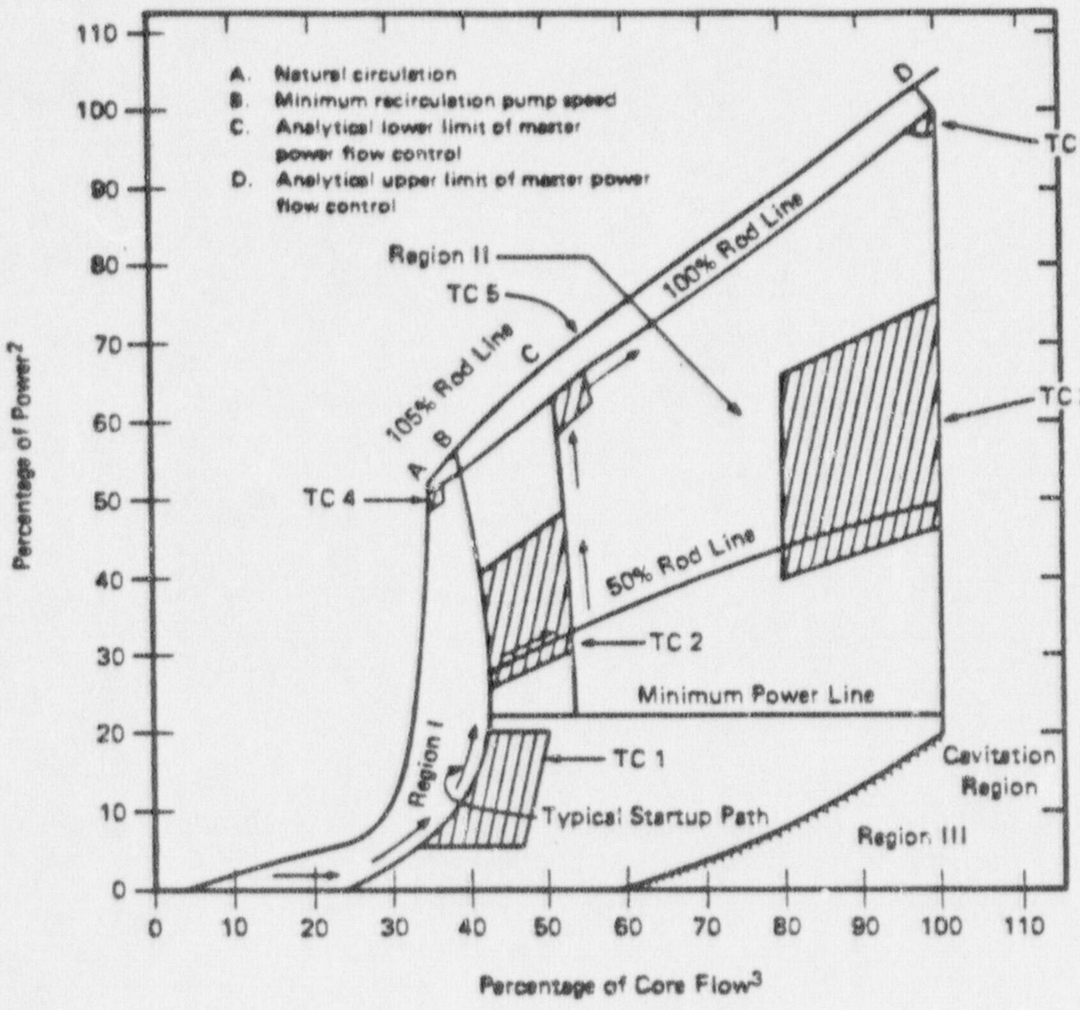
Test No.	Test Name	Open Vessel or Cold Test	Heatup	Test Conditions[a] 1	2	3	4^b	5	6	Warranty
1	Chemical and Radiochemical	X	X	X		X		X	X	
2	Radiation Measurements	X	X		X	X			X	
3	Fuel Loading	X								
4	Full Core Shutdown Margin		X							
5	CRD	X	X	X					X	
6	SRM Performance and Control Rod Sequence		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	09/16/86
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
Reacter Restarted	05/14/87

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	>>>		>>>		
2	Radiation measurements	01/24/85	04/19/85	07/14/85	10/17/86	03/08/87		>>>	>>>	
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	>>>	>>>	08/04/86	01/07/87	>>>	>>>	>>>	>>>	>>>
9	LPRM Calibration	>>>	>>>	09/24/85	01/03/87	>>>		>>>	>>>	
10	APRM Calibration	>>>	>>>	08/01/85	01/04/87	03/10/87		>>>		
11	Process Computer	05/30/85	>>>	>>>	01/02/87	02/05/87	>>>			
12	RCIC	>>>	>>>	09/01/86	10/15/86	11/03/86	>>>	>>>	>>>	>>>
13	HPCI	>>>	>>>	09/01/86	12/29/86	>>>		>>>	>>>	>>>
14	Selected ProcessTemperatures	>>>	>>>	09/01/86	>>>	>>>			>>>	
15	System Expansion	>>>	06/12/85	09/05/86	11/04/86	>>>	>>>	>>>	>>>	
16	Core Power Distribution (Deleted)	>>>	>>>	>>>	>>>	>>>	>>>	>>>	>>>	>>>
17	Core Performance	>>>	>>>	>>>	01/04/87	03/10/87				
19	Core Power Void Mode Response (Deleted)	>>>	>>>	>>>	>>>	>>>	>>>	>>>	>>>	>>>
20	Pressure Regulator	>>>	>>>	>>>	10/21/86	03/10/87		>>>		
21	Feedwater System	>>>	>>>	07/09/85	10/21/86	$$$		>>>		
22	Turbine Valve Surveillance	>>>	>>>	>>>	>>>	>>>	>>>	>>>		>>>
23	MSIV	>>>	>>>	07/12/85	10/08/86	>>>		>>>	>>>	
24	Relief Valves	>>>	>>>	07/03/85	>>>	03/11/87	>>>	>>>	>>>	>>>
25	Turbine Stop Valve and Control Valve Fast Closure	>>>	>>>	>>>	>>>	03/16/87	>>>	>>>	>>>	
26	Shutdown from Outside Control Room	>>>	>>>	>>>	10/23/86	>>>	>>>	>>>	>>>	

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

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

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
27	Flow Control	>>>	>>>	>>>	>>>	03/08/87		>>>	>>>	
28	Recirculation System	>>>	>>>	>>>	>>>	03/08/87			>>>	
29	Loss of Offsite Power	>>>	>>>	>>>	>>>	03/16/87	>>>	>>>	>>>	>>>
30	Vibration Measurements	>>>	>>>	07/03/85	10/04/86	03/07/87		>>>	>>>	
31	Recirc. System Flow Calibration	>>>	>>>	>>>	>>>	>>>		>>>	>>>	
32	Reactor Water Cleanup System	>>>	>>>	07/14/85	>>>	>>>	>>>		>>>	>>>
33	Residual Heat Removal System	>>>	>>>	08/31/85	>>>	>>>	>>>	>>>	>>>	
34	Piping System Dynamic Response	>>>	>>>	09/24/85	>>>	03/16/87		>>>	>>>	

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

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

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 Warranty 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 low condenser vacuum, minimum filter demineralizer flow and low sample flow rates.

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 directly related to placing the Generator on line and increasing Generator load. One possible explanataion is that operation of the Generator is 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 is felt to be the Krylon coating that was previously used as a preservative coating for the turbine blades, which is now being worn off the blades and into the condenser. Further investigation is continuing to determine the exact 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 problem of low condensate/feedwater dissolved oxygen has occurred during the startup of other operating plants. The resolution at this time is to simply continue to monitor these parameters at higher power levels to see if the levels will 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.

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 are 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 to date.

Chemistry Data

System	Test Condition	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				
	CTP	0	<5%	17%				
	MWe	0	0	0				
	Mod-Temp	N/A	540°F	540°F				
System	**Analysis**							**Limit**
Reactor Water	Conductivity							
	(umho/cm) @ 25°C	0.43	0.31	0.24				<2.0
	Chloride (ppb)	<5	8	<2				<200
	Turbidity (NTU)	0.18	3.6	0.15				See Note 2
	Iodine - 131 (uCi/ml)	N/A	<7.7 E-08	<MDA				See Note 1
	- 133 (uCi/ml)	N/A	2.0 E-06	5.5 E-6				See Note 1
	2 hr Gross Activity							
	- Filtrate (cpm/ml)	N/A	6.1 E02	7.86 E02				See Note 2
	- Crud (cpm/ml)	N/A	4.0 E01	1.30 E01				See Note 2
	7 day Gross Activity							
	- Filtrate (cpm/ml)	N/A	4.7	5.35				See Note 2
	- Crud (cpm/ml)	N/A	25	0.54				See Note 2
	Silica (ppm)	0.023	0.385	29				See Note 2
	pH	6.2	6.4	5.8				5.6 <ph <8.6
	Iron in Crud (ppb)	N/A	N/A	N/A				See Note 2
	Boron (ppb)	<10	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.

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 6	Limit
	Test 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				
	CTP	0	<5%	17%				
	MWe	0	0	0				
	Mod-Temp	N/A	540^{o}F	540^{o}F				
CRD Water	Conductivity (umho/cm) @ 25^{o}C	N/A	0.08	0.09				<0.1
	Oxygen (ppb)	N/A	500	50				<50
Radioactive Gaseous Effluents	Off-gas Vial Sample Analysis H21-P275A	N/A	Completed	Completed				See Note 3
	Off-gas Monitor Reading D11-K601A (mr/hr)	N/A	3.5	20				See Note 2
	Off-gas Monitor Reading D11-K601B (mr/hr)	N/A	3.5	2				See Note 2
	SGTS Exhaust, Div I D11-P275 (uCi/cc)	N/A	1.7 E-08	4.6 E-08				See Note 4
	SGTS Exhaust, Div II D11-P275 (uCi/cc)	N/A	1.2 E-08	5.0 E-08				See Note 4
	Turbine Bldg D11-P279 (uCi/cc)	N/A	8.0 E-08	3.4 E-08				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.

FIGURE 3.1
(Page 2 of 5)

Chemistry Data

System	Test Condition	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				
	CTP	0	<5%	17%				
	MWe	0	0	0				
	Mod-Temp	N/A	540°F	540°F				
System	Analysis							Limit
Radioactive	Reactor Bldg (uCi/cc)							
Gaseous	D11-P260	N/A	4.05 E-07	6.25 E-07				See Note 4
Effluents	Radwaste Bldg (uCi/cc)							
(continued)	D11-P281	N/A	9.1 E-07	8.2 E-07				See Note 4
	Service Bldg (uCi/cc)							
	D11-P282	N/A	8.1 E-08	6.4 E-08				See Note 4
	Site Storage Bldg							
	D11-P299 (uCi/cc)	N/A	1.2 E-07	5.2 E-08				See Note 4
Demin.	Conductivity							
Effluent	(umho/cm) @ 25°C	N/A	0.072	0.059				<0.1
	Oxygen (ppb)	N/A	N/A	<10				$20<O_2<200$
	Insoluble Iron (ppb)	N/A	N/A	0.098				See Note 2
	Total Iron (ppb)	N/A	N/A	N/A				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.

FIGURE 3.1
(Page 3 of 5)

Chemistry Data

System	Test Condition	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				
	CTP	0	<5%	17%				
	MWe	0	0	0				
	Mod-Temp	N/A	540°F	540°F				
System	Analysis							Limit
Feedwater	Soluble Iron (ppb)	N/A	N/A	N/A				See Note 2
	Insoluble Iron (ppb)	1.8	1	6.16				See Note 5
	Conductivity(umho/cm) A	0.38	0.08	0.065				<0.1
	@ 25°C B	N/A	0.10	N/A				<0.1
	Oxygen (ppb)	N/A	N/A	<10				$20<O_2<200$
	Insoluble Copper (ppb)	0.1	0.03	0.26				See Note 5
	Soluble Copper (ppb)	4.4	1	0.45				See Note 5
	Total Metals (ppb)	10	2.33	12.98				See Note 5
Condensate	Conductivity (umho/cm) @ 25°C	0.5	0.14	0.16				<0.5
	Chloride (ppb)	<5	<2	<2				See Note 2
	Insoluble Iron (ppb)	N/A	3	3.05				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.

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				
	CTP	0	<5%	17%				
	MWe	0	0	0				
	Mod-Temp	N/A	540^{0}F	540^{0}F				
Heater Drains	Insoluble Iron (ppb)	N/A	N/A	N/A				See Note 2
	Soluble Iron (ppb)	N/A	N/A	N/A				See Note 2
	Conductivity (umho/cm) @ 25^{0}C B	N/A	N/A	N/A				<0.1
	Oxygen (ppb)	N/A	N/A	N/A				$20<O_2<200$
Liquid Radwaste Sample Tank	Principal Gamma Emitter (prior to discharge, uci/ml)	N/A	N/A	N/A				See Note 2

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

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

Level 1

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

Level 2

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 has been completed.

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%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87		
Channel Number:				Radiation Levels:			
1	0.03	<0.1	0.05	0.04	0.05		
2	0.02	<0.1	0.02	0.02	0.08		
3	0.03	<0.1	0.02	0.02	0.03		
4	0.02	<0.1	0.02	0.02	0.02		
5	0.08	<0.1	0.08	0.07	0.08		
6	0.03	<0.1	0.04	0.02	0.03		
7	0.2	0.2	0.2	0.2	0.02		
8	0.2	0.4	0.4	0.3	0.04		
9	0.3	0.3	0.3	0.3	0.04		
10	0.2	0.2	0.2	0.2	0.02		
11	0.3	0.3	0.3	0.3	0.03		
12	5	4	5	1000	20.0		
13	0.03	<0.1	0.04	0.04	0.05		
14	3	3	4	6	6.0		
15	0.02	<0.1	0.02	0.06	0.06		
16	0.04	<0.1	0.05	0.04	0.03		
17	0.02	<0.1	2	0.06	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%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87		
Channel Number:			Radiation Levels:				
18	300	300	300	300	200		
19	0.4	0.4	0.4	0.3	0.05		
20	0.3	0.3	0.3	0.3	0.3		
21	2	2	3	3	40.0		
22	0.03	<0.1	0.03	0.04	0.03		
23	0.02	<0.1	0.03	0.03	0.03		
24	0.3	0.3	0.3	0.3	0.3		
25	0.3	0.3	0.3	0.2	0.3		
26	0.3	0.2	0.2	0.2	0.2		
27	OOS	OOS	OOS	OOS	0.05		
28	0.03	<0.1	0.03	0.03	0.02		
29	0.06	<0.1	0.06	0.05	0.05		
30	0.03	<0.1	0.04	0.3	6.0		
31	0.2	0.2	0.2	0.4	1.0		
32	5	4	5	5	5.0		
33	0.02	<0.1	0.02	0.02	0.02		
34	0.05	<0.1	0.07	0.06	0.07		

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%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87		
Channel Number:				Radiation Levels:			
35	0.03	<0.1	0.03	0.03	0.03		
36	0.03	<0.1	0.03	0.03	0.03		
37	0.02	<0.1	0.02	0.02	0.03		
38	0.02	<0.1	0.02	0.02	0.02		
39	0.05	<0.1	0.08	0.05	0.05		
40	0.03	<0.1	0.03	0.03	0.03		
41	0.03	<0.1	0.03	0.04	0.03		
42	0.05	<0.1	0.04	0.03	0.05		
43	0.04	<0.1	0.04	0.04	0.04		
44	0.03	<0.1	0.02	0.03	0.03		
45	OOS	0.2	OOS	OOS	OOS		
46	0.2	0.4	0.4	0.5	0.4		
47	3	3	0.3	2	3.0		
48	3	4	3	2	3.0		

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%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87		

Off-Gas Radiation - A (D11-K601A) [mr/hr]	3	2	3.5	11	2.0		
Off-Gas Radiation - B (D11-K601B) [mr/hr]	3.5	3	3	2	3.0		
Off-Gas Radiation - Linear (D11-K602) [mr/hr]	0	5E-8	0	1E-8	3.16E-6		
Radwaste Effluent (D11-K604) [cps]	2	2	2	4	7.0		
GSW Effluent (D11-K605) [cps]	3.5	4	4	4	4.0		
RBCCW System (D11-K606) [cps]	3	2	2	2	2.0		
A Main Steam Line (D11-K603A) [mr/hr]	1.2	1	OOS	8	4.41E2		
B Main Steam Line (D11-K603B) [mr/hr]	1.4	1	1	30	3.83E2		
C Main Steam Line (D11-K603C) [mr/hr]	1.6	2	1.2	9	4.01E2		
D Main Steam Line (D11-K603D) [mr/hr]	8.0	11	1	16	3.89E2		
Div-I EECW Hx Inlet (D11-K800A) [cpm]	200	200	200	300	550		
Div-I RHR S.W. (D11-K801A) [cpm]	200	200	200	200	450		
Div-I RB Vent Exh. (D11-K808) [cpm]	50	50	40	50	45		

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 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%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87		

Div-I Cont. Ctr. Makeup Air (D11-K809) [cpm]	40	30	30	30	40		
Two Min. Holdup Pipe Exh. (D11-K814) [cpm]	250	200	400	1000	10E4		
Div-II EECW Hx Inlet (D11-K800B) [cpm]	200	200	200	300	300		
Div-II RHR S.W. (D11-K801B) [cpm]	150	200	200	200	200		
Circ. Water Rev. Decant Line (D11-K802) [cpm]	200	200	200	300	300		
Div-II RB Vent Exh. (D11-K810) [cpm]	60	60	40	50	50		
Div-II Cont. Ctr. Makeup Air (D11-K813) [cpm]	60	60	60	50	60		
Two Min. Holdup Pipe Exh. (D11-K815) [cpm]	300	200	400	3000	2500		
1st Flr. Inside Drywell (D21-K745) [mr/hr]	0.3	0.2	OOS	0.1	0.1		
1st Flr. on Site Sfg. Bldg. Control Room (D21-K846) [mr/hr]	0.2	0.4	0.5	0.5	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%		
MWe	0	0	0	0	425		
Mod-Temp	<100^{o}F	<100^{o}F	540^{o}F	540^{o}F	510^{o}F		
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87		

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

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 3 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%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87		

C.C. Emerg. Air (North), Div I (D11-K837A) [cpm]	NA	NA	NA	20	OOS		
Cont. Area Hi-Range Monitor, Div I (D11-K816A) [mr/hr]	NA	NA	NA	2000	2500		
C.C. Emerg. Air (South), Div II (D11-K836B) [cpm]	NA	NA	NA	30	30		
C.C. Emerg. Air (North), Div II (D11-K837B) [cpm]	NA	NA	NA	60	25		
Cont. Area Hi-Range Montior, Div II (D11-K816B) [mr/hr]	NA	NA	NA	2	3.0		
Reactor Bldg. Vent Plenum Exh. (D11-K610) [uCi/cc]	NA	NA	NA	6.65 E-7	1.07E-6		
Radwaste Bldg. Vent Exh. (D11-K610) [uCi/cc]	NA	NA	NA	8.83 E-7	8.35E-7		
SGTS Vent Exh. Div I (D11-K610) [uCi/cc]	NA	NA	NA	1.32 E-7	3.92E-8		
SGTS Vent Exh. Div II (D11-K610) [uCi/cc]	NA	NA	NA	6.75 E-7	5.19E-8		
OSB Machine Shop Vent Exh. (D11-K610) [uCi/cc]	NA	NA	NA	4 E-8	2.07E-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%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87		

Turbine Bldg. Vent Exh. (D11-K610) [uCi/cc]	NA	NA	NA	7.67 E-8	1.85E-7		
On-Site Storage Bldg. Vent Exh, (D11-K610) [uCi/cc]	NA	NA	NA	2.64 E-7	1.25E-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%		
MWe	0	0	0	0	425		
Mod-Temp	<100^{o}F	<100^{o}F	540^{o}F	540^{o}F	510^{o}F		
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87		
1	<0.2	<0.2	<0.2	<0.2	<0.2		
2	<0.2	<0.2	<0.2	<0.2	<0.2		
3	<0.2	<0.2	<0.2	<0.2	<0.2		
4	<0.2	<0.2	<0.2	<0.2	<0.2		
5	<0.2	<0.2	<0.2	<0.2	<0.2		
6	<0.2	<0.2	<0.2	<0.2	<0.2		
7	<0.2	<0.2	<0.2	<0.2	<0.2		
8	<0.2	<0.2	<0.2	<0.2	<0.2		
10	<0.2	<0.2	<0.2	<0.2	<0.2		
11	<0.2	<0.2	<0.2	<0.2	<0.2		
12	<0.2	<0.2	<0.2	<0.2	<0.2		
13	<0.2	<0.2	<0.2	<0.2	<0.2		
14	<0.2	<0.2	<0.2	<0.2	<0.2		
15	<0.2	<0.2	<0.2	<0.2	<0.2		
16	<0.2	<0.2	<0.2	<0.2	<0.2		
17	<0.2	<0.2	<0.2	<0.2	<0.2		
18	<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

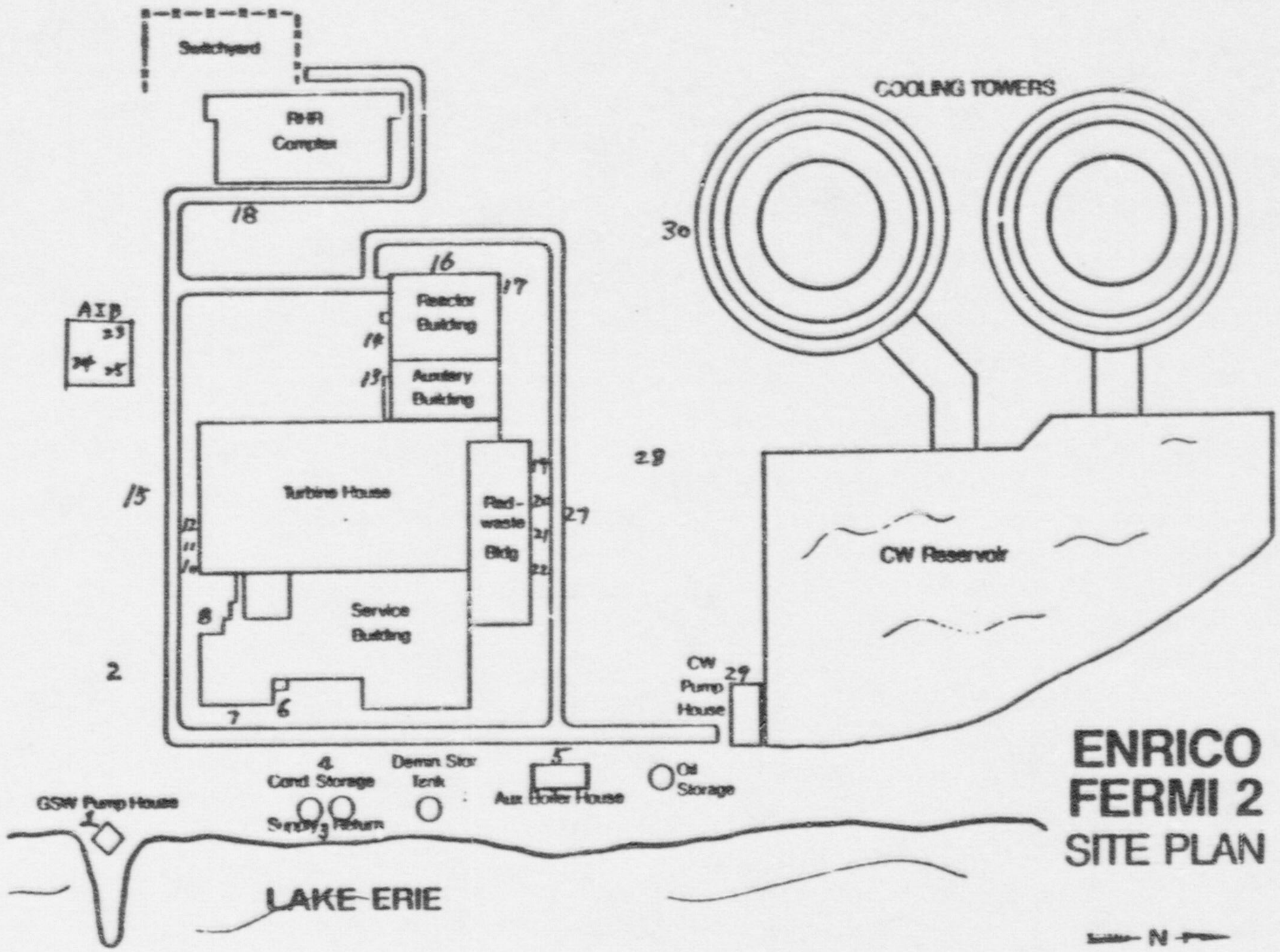
Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Tes. Conditic 3	Test Condition 6
CTP	0	0	<5%	17%-20%	48%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/16,17/86	3/8/87		

19	<0.2	<0.2	<0.2	<0.2	<0.2		
20	<0.2	<0.2	<0.2	<0.2	<0.2		
21	<0.2	<0.2	<0.2	<0.2	<0.2		
22	<0.2	<0.2	<0.2	<0.2	<0.2		
23	<0.2	<0.2	<0.2	<0.2	<0.2		
24	<0.2	<0.2	<0.2	<0.2	<0.2		
25	<0.2	<0.2	<0.2	<0.2	<0.2		
27	<0.2	<0.2	<0.2	<0.2	<0.2		
28	<0.2	<0.2	<0.2	<0.2	<0.2		
29	<0.2	<0.2	<0.2	<0.2	<0.2		
30	<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)
(Page 3 of 32)

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%	19%	48%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/17/86	3/8/87		

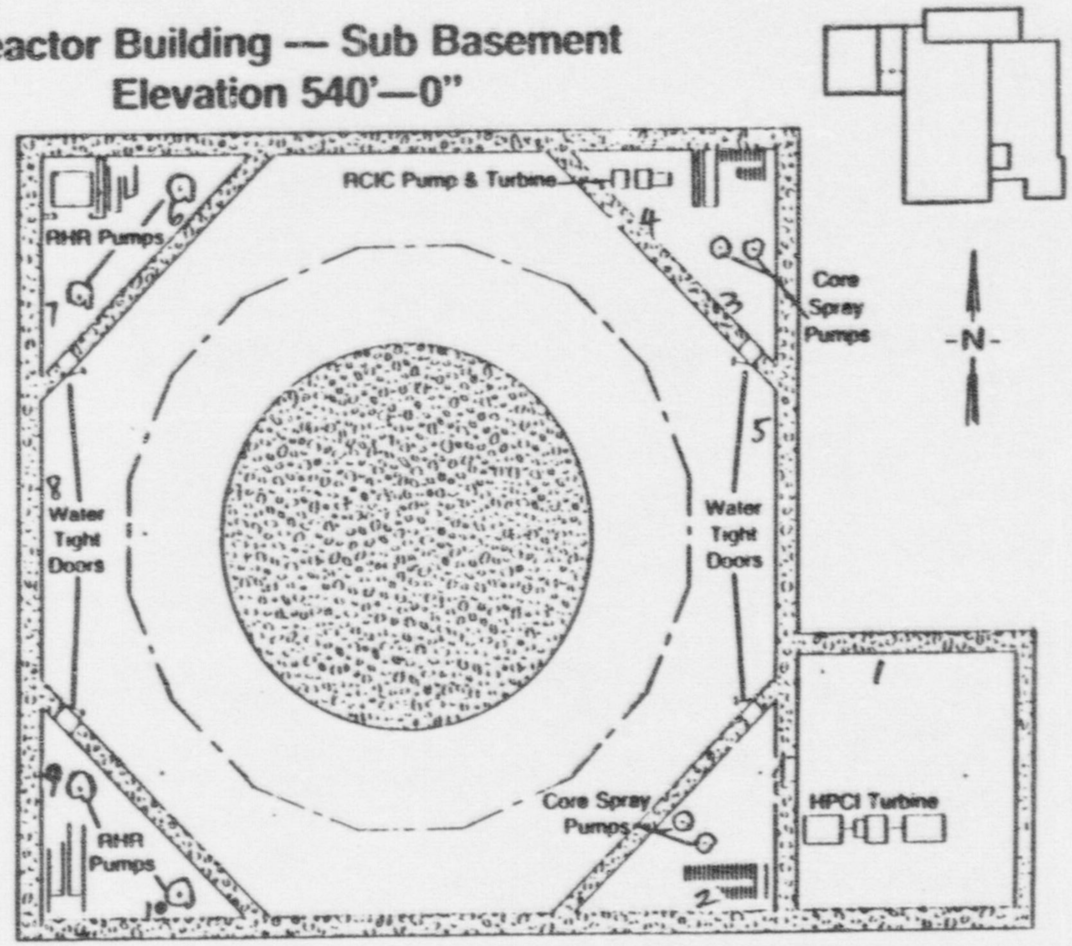
1	<0.2	<0.4	0.4	<0.2	<0.2		
2	<0.2	<0.4	0.2	<0.2	<0.2		
3	<0.2	<0.4	0.3	<0.2	<0.2		
4	<0.2	<0.4	0.2	<0.2	<0.2		
5	<0.2	<0.4	0.4	<0.2	2.5		
6	<0.2	<0.4	0.3	<0.2	<0.2		
7	<0.2	<0.4	0.6	<0.2	<0.2		
8	<0.2	<0.4	0.4	<0.2	<0.2		
9	<0.2	<0.4	0.4	<0.2	0.4		
10	<0.2	<0.4	0.4	<0.2	0.2		

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 4 of 32)

Reactor Building — Sub Basement
Elevation 540'—0"



(FIGURE 3.2-3)
(Page 5 of 32)

Survey Data

Location: Reactor Building - Basement

Test Condition	Pre Fuel Load	Open Vessel	Heatup	Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	19%	48%		
MWe	0	0	0	0	425		
Mod-Temp	<100^{o}F	<100^{o}F	540^{o}F	540^{o}F	510^{o}F		
Date	1/19/85	[illegible]/16/85	7/13/85	10/17/86	3/8/87		

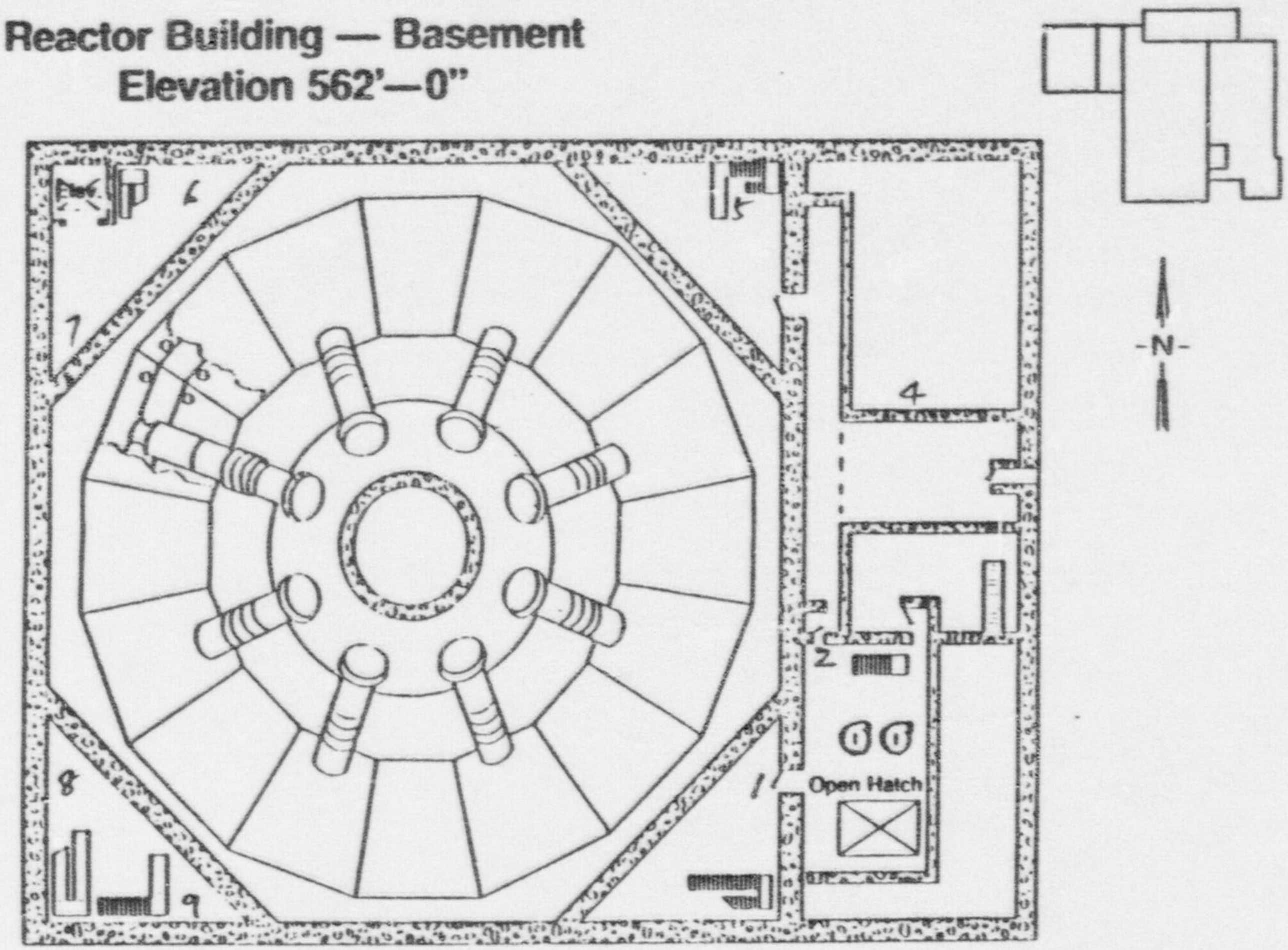
1	<0.2	<0.4	0.3	<0.2	<0.2		
2	<0.2	<0.2	<0.2	<0.2	<0.2		
4	<0.2	<0.2	<0.2	<0.2	<0.2		
5	<0.2	<0.2	<0.2	<0.2	<0.2		
6	<0.2	<0.4	0.2	<0.2	<0.2		
7	<0.2	<0.4	0.2	<0.2	0.2		
8	<0.2	<0.4	0.2	<0.2	1.0		
9	<0.2	<0.4	0.2	<0.2	<0.2		

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)
(Page 7 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	0	0	<5%	19%	48%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/17/86	3/8/87		

1	<0.2	<0.2	<0.2	<0.2	0.4		
2	<0.2	<0.2	<0.2	<0.2	<0.2		
4	<0.2	<0.4	0.2	<0.2	<0.2		
5	<0.2	<0.4	0.2	<0.2	<0.2		
7	<0.2	<0.4	0.2	<0.2	<0.2		
8	<0.2	<0.4	0.2	<0.2	<0.2		
9	<0.2	<0.4	0.2	<0.2	<0.2		
10	<0.2	<0.4	0.2	<0.2	<0.2		
11	<0.2	<0.4	0.2	<0.2	<0.2		
12	<0.2	<0.4	0.2	<0.2	<0.2		
13	<0.2	<0.4	0.2	<0.2	<0.2		
15	<0.2	<0.4	0.2	<0.2	0.4		
16	<0.2	<0.4	0.2	<0.2	0.4		
17	<0.2	<0.4	0.2	<0.2	<0.2		
18	<0.2	<0.4	0.4	<0.2	0.6		
19	<0.2	<0.4	0.2	<0.2	1.5		
21	<0.2	<0.4	0.2	<0.2	0.4		
23	<0.2	<0.4	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-

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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%	19%	48%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/17/86	3/8/87		
25	<0.2	<0.2	<0.2	<0.2	<0.2		
26	<0.2	<0.4	0.4	<0.2	1.0		
27	<0.2	<0.4	0.2	<0.2	<0.2		
28	<0.2	<0.4	0.4	<0.2	<0.2		
29	<0.2	<0.4	0.2	<0.2	<0.2		
31	<0.2	<0.4	0.2	<0.2	<0.2		
32	<0.2	<0.4	0.4	<0.2	<0.2		
33	<0.2	<0.4	0.4	<0.2	<0.2		
34	<0.2	<0.4	0.2	<0.2	<0.2		
35	<0.2	<0.4	0.2	<0.2	<0.2		
37	<0.2	<0.2	<0.2	<0.2	<0.2		
38	<0.2	<0.2	<0.2	<0.2	<0.2		
39	<0.2	<0.2	<0.2	<0.2	<0.2		
40	<0.2	<0.2	<0.2	4.4	30		

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

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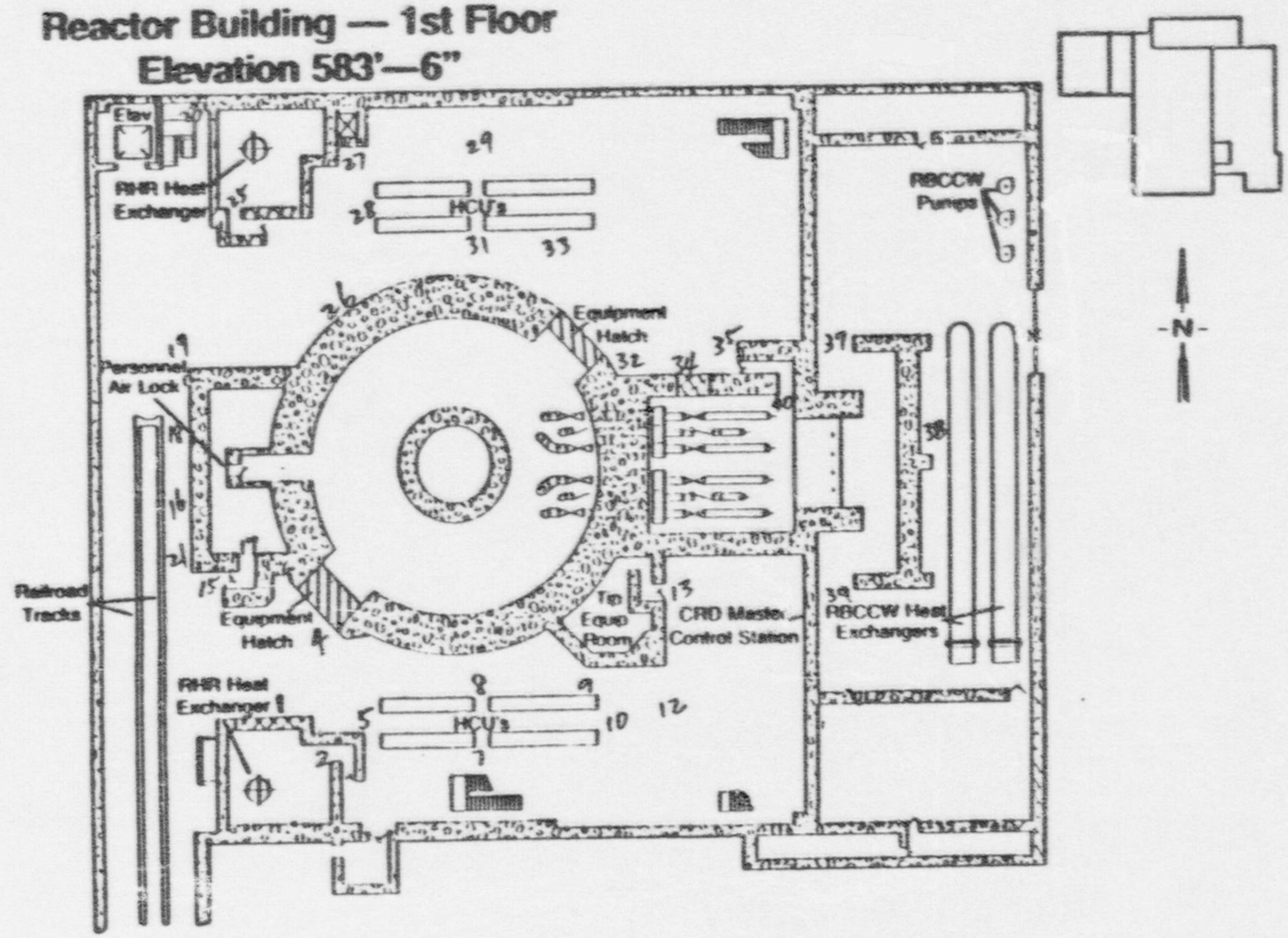
Reactor Building — 1st Floor
Elevation 583'—6"



(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%	19%	48%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/17/86	3/8/87		

1	<0.2	<0.4	0.4	<0.2	<0.2		
2	<0.2	<0.4	0.4	<0.2	<0.2		
3	<0.2	<0.4	0.6	<0.2	3.8		
4	<0.2	<0.4	0.2	<0.2	<0.2		
5	<0.2	<0.2	<0.2	<0.2	<0.2		
6	<0.2	<0.4	0.6	<0.2	<0.2		
7	<0.2	<0.4	0.6	<0.2	<0.2		
8	<0.2	<0.4	0.6	<0.2	1.5		
10	<0.2	<0.4	0.8	1.0	25		
11	<0.2	<0.4	0.6	<0.2	0.5		
12	<0.2	<0.4	0.4	<0.2	<0.2		
13	<0.2	<0.4	0.5	<0.2	0.5		
14	<0.2	<0.4	1.4	3.6	100		
15	<0.2	<0.4	0.8	<0.2	<0.2		
16	<0.2	<0.4	0.2	<0.2	<0.2		
18	<0.2	<0.2	<0.2	<0.2	<0.2		
19	<0.2	<0.4	0.3	<0.2	<0.2		
20	<0.2	<0.4	0.3	<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 11 of 32)

Survey Data

Location: Reactor Building - Second Floor

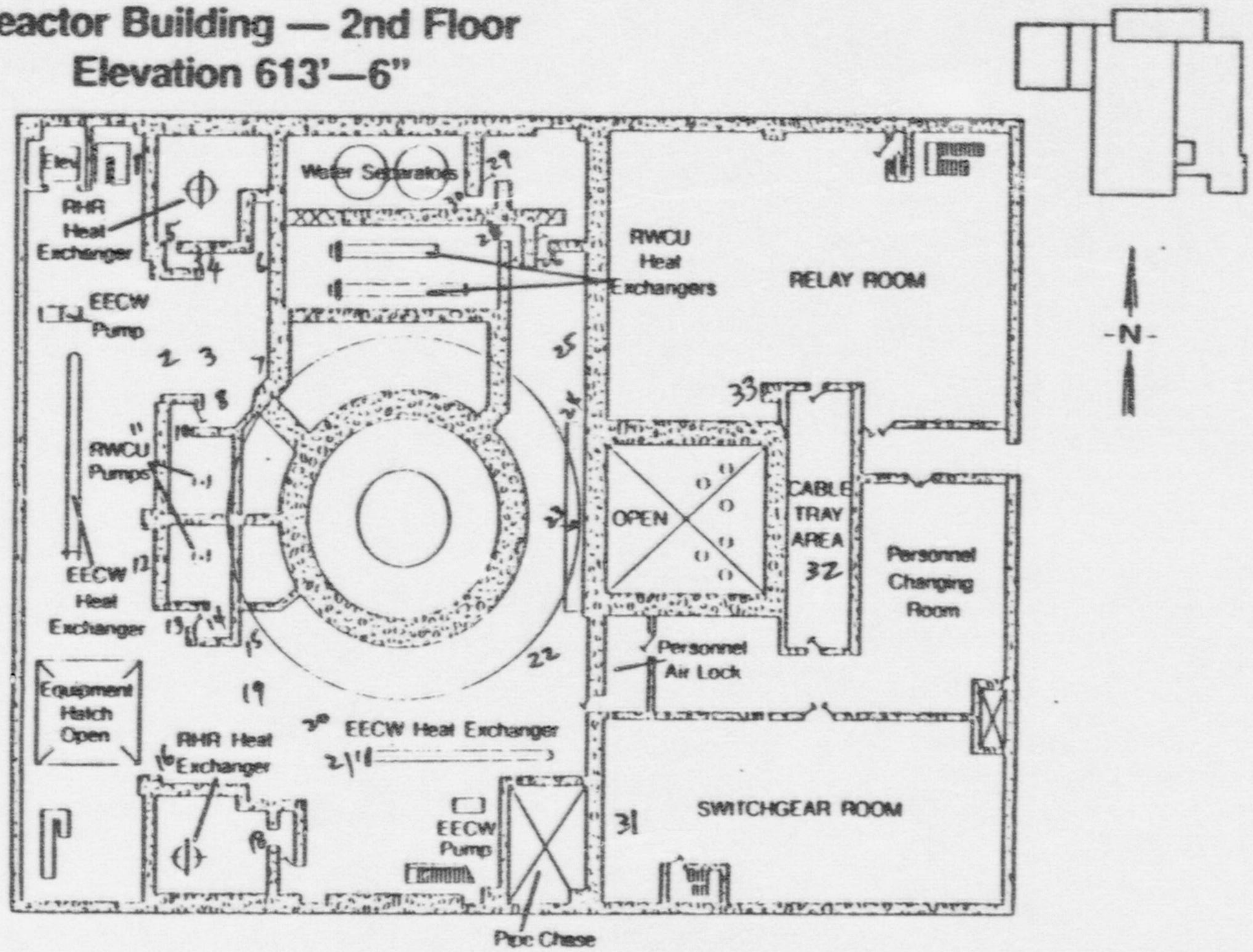
Test Condition	Pre Fuel Load	Open Vessel	Heatup	Test Condition 1	Test Condition 2	Test Condition 3	Test Condition 6
CTP	0	0	<5%	19%	48%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/17/86	3/8/87		
21	<0.2	<0.4	0.4	<0.2	<0.2		
22	<0.2	<0.4	0.2	<0.2	<0.2		
23	<0.2	<0.4	0.4	<0.2	<0.2		
24	<0.2	<0.4	0.3	<0.2	<0.2		
25	<0.2	<0.4	0.2	<0.2	<0.2		
26	<0.2	<0.4	0.2	<0.2	<0.2		
28	<0.2	<0.4	<0.2	0.4	5.0		
29	<0.2	<0.2	<0.2	<0.2	<0.2		
30	<0.2	<0.2	2.5	6	34		
31	<0.2	<0.2	<0.2	<0.2	<0.2		
32	<0.2	<0.2	<0.2	<0.2	<0.2		
33	<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 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%	19%	48%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/17/86	3/8/87		

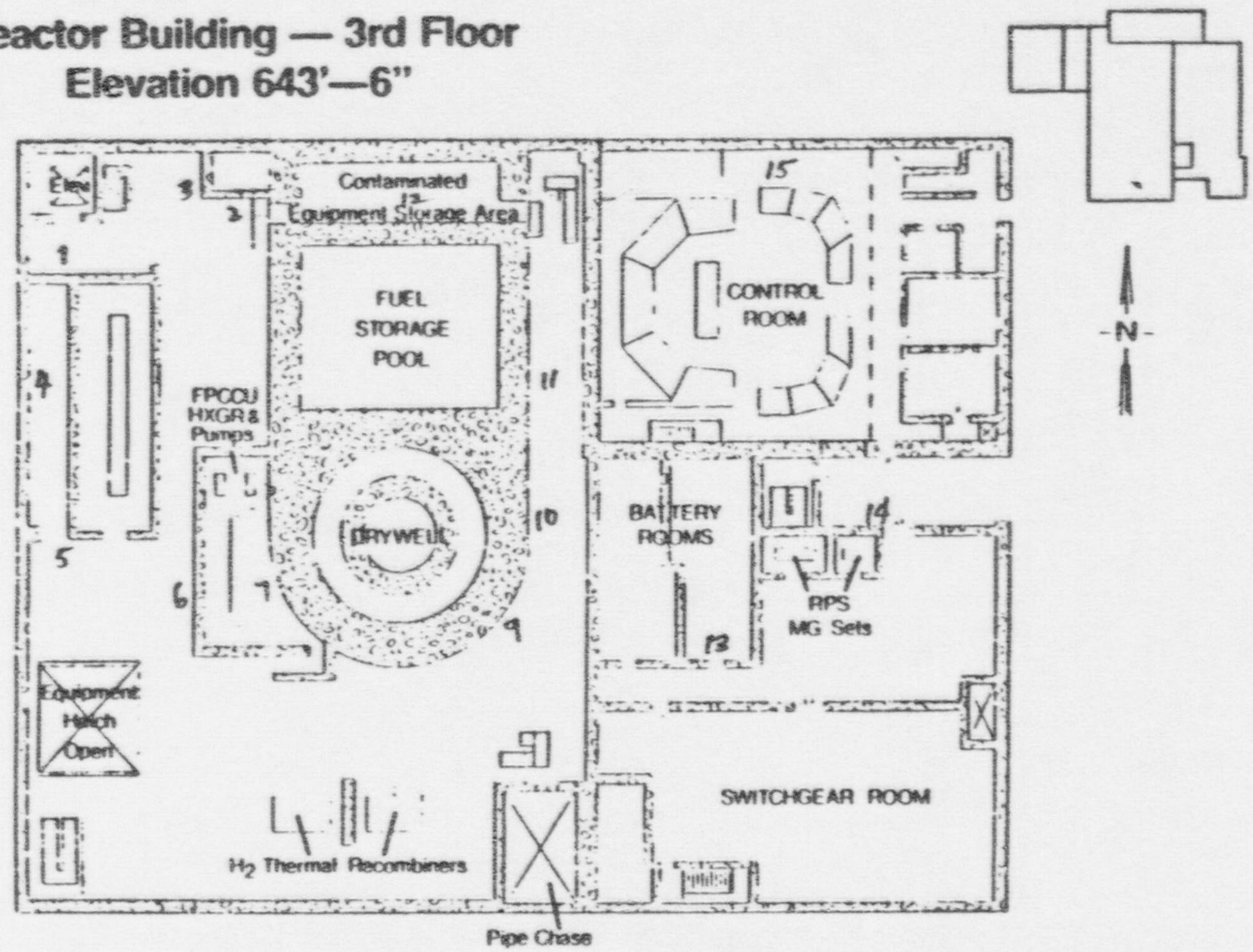
1	<0.2	<0.4	0.3	<0.2	<0.2		
2	<0.2	<0.4	0.3	<0.2	<0.2		
3	<0.2	<0.4	0.3	<0.2	<0.2		
4	<0.2	[illegible]	0.2	<0.2	<0.2		
5	<0.2	<0.4	0.3	<0.2	<0.2		
6	<0.2	<0.2	<0.2	<0.2	<0.2		
7	<0.2	<0.4	0.4	0.4	1.0		
9	<0.2	<0.4	0.4	<0.2	<0.2		
10	<0.2	<0.4	0.4	<0.2	<0.2		
11	<0.2	<0.4	0.4	<0.2	<0.2		
12	<0.2	<0.2	<0.2	<0.2	<0.2		
13	<0.2	<0.2	<0.2	<0.2	<0.2		
14	<0.2	<0.2	<0.2	<0.2	<0.2		
15	<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%	19%	48%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/16/86	3/8/87		

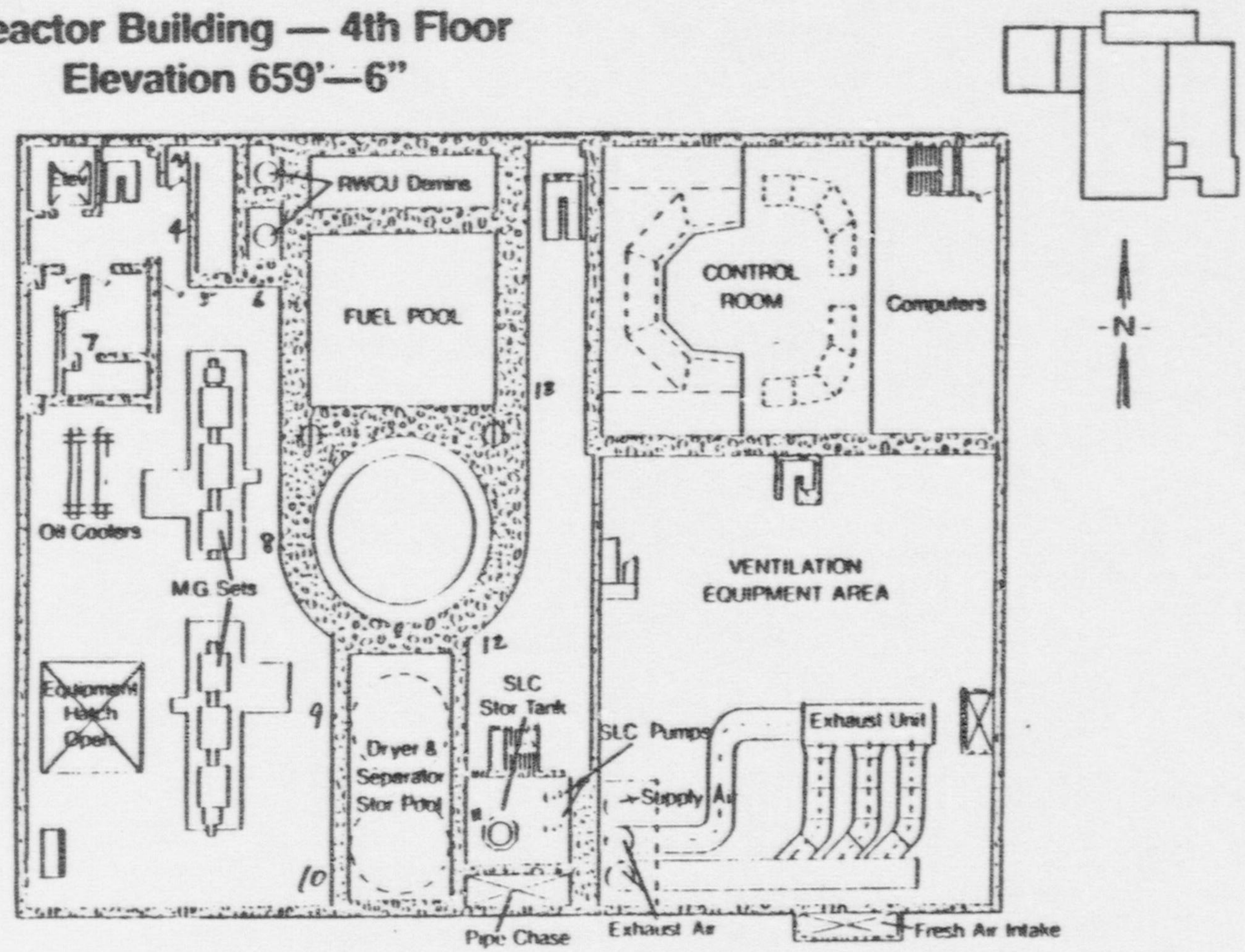
2	<0.2	<0.2	<0.2	0.3	4.2		
4	<0.2	<0.4	<0.3	0.2	<0.2		
5	<0.2	<0.4	<0.3	0.2	<0.2		
6	<0.2	<0.4	<0.4	0.2	<0.2		
7	<0.2	<0.4	<0.3	0.2	<0.2		
8	<0.2	<0.4	<0.5	0.2	<0.2		
9	<0.2	<0.4	<0.3	0.2	<0.2		
10	<0.2	<0.4	<0.3	0.2	<0.2		
11	<0.2	<0.4	0.2	<0.2	<0.2		
12	<0.2	<0.4	0.4	<0.2	<0.2		
13	<0.2	<0.4	0.2	<0.2	<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 16 of 32)

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



(FIGURE 3.2-3)
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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%	19%	48%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/17/86	3/8/87		
1	<0.2	<0.4	0.2	<0.2	<0.2		
2	<0.2	<0.4	0.2	<0.2	<0.2		
3	<0.2	<0.4	0.2	<0.2	<0.2		
4	<0.2	<0.4	0.3	<0.2	<0.2		
5	<0.2	<0.4	0.3	<0.2	<0.2		
6	<0.2	<0.3	<0.3	<0.2	<0.2		
7	<0.2	<0.4	0.2	<0.2	<0.2		
8	<0.2	<0.4	0.3	<0.2	<0.2		
9	<0.2	<0.2	<0.2	<0.2	<0.2		
10	<0.2	<0.2	<0.2	<0.2	<0.2		
11	<0.2	<0.2	<0.2	<0.2	<0.2		
12	<0.2	<0.2	<0.2	<0.2	<0.2		
13	<0.2	<0.2	<0.2	<0.2	<0.2		
14	<0.2	<0.4	0.3	<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
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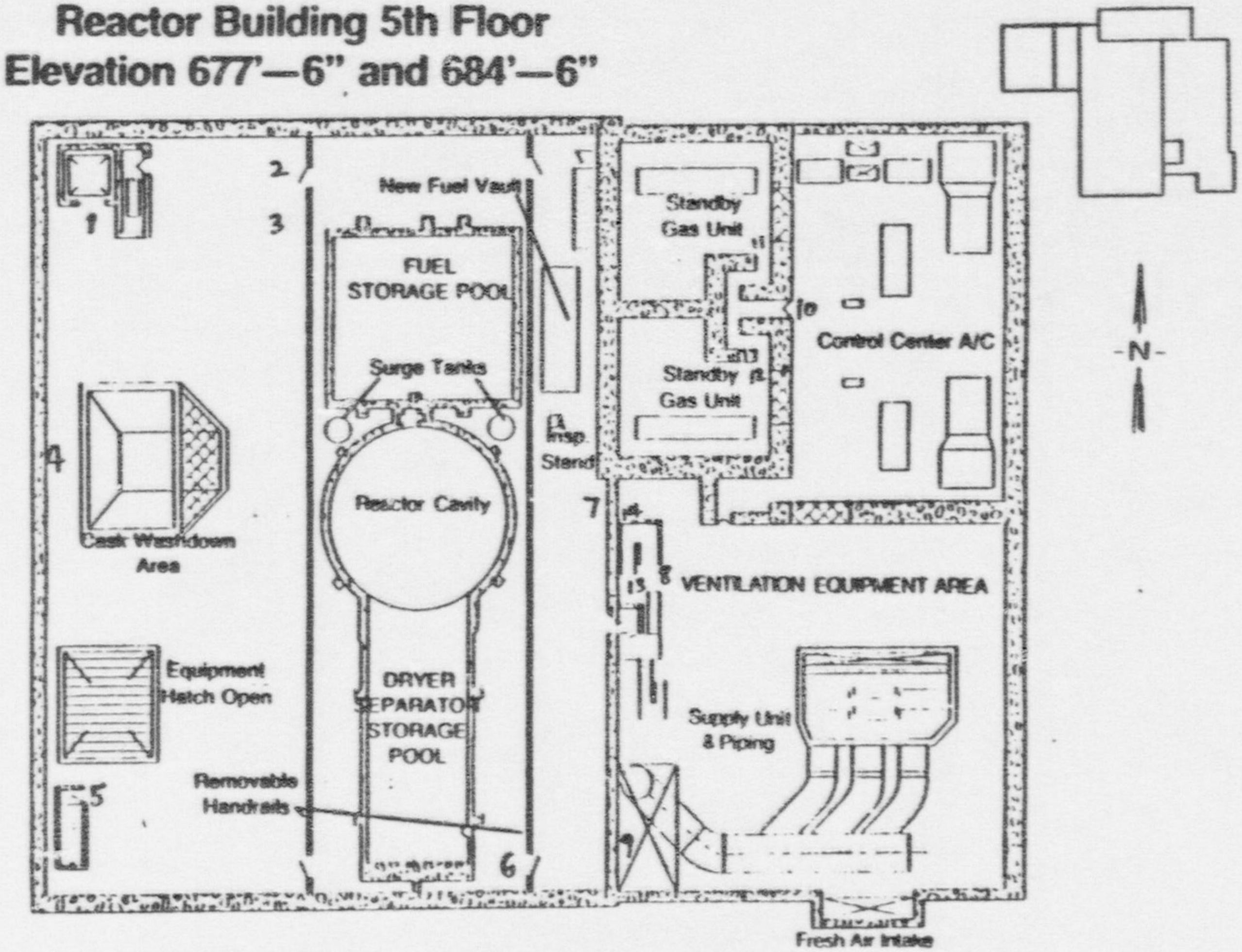
Reactor Building 5th Floor
Elevation 677'—6" and 684'—6"



(FIGURE 3.2-3)
(Page 19 of 32)

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%	19%	48%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/16/86	3/8/87		

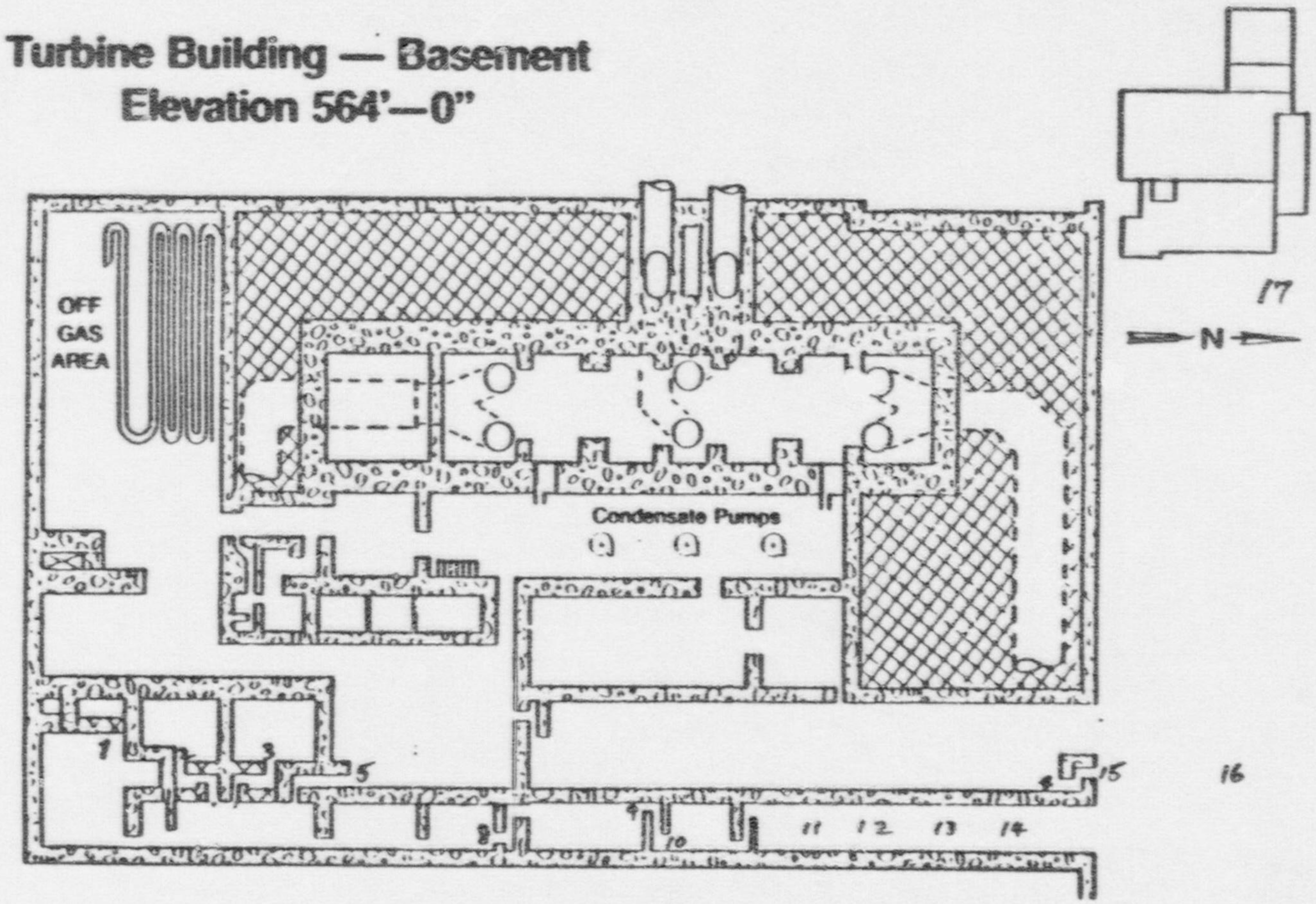
1	<0.2	<0.2	<0.2	<0.2	<0.2		
2	<0.2	<0.2	<0.2	<0.2	<0.2		
3	<0.2	<0.2	<0.2	<0.2	<0.2		
4	<0.2	<0.2	<0.2	<0.2	<0.2		
5	<0.2	<0.2	<0.2	<0.2	<0.2		
8	<0.2	<0.2	<0.2	<0.2	0.3		
9	<0.2	<0.2	<0.2	<0.2	<0.2		
10	<0.2	<0.2	<0.2	<0.2	<0.2		
11	<0.2	<0.2	<0.2	<0.2	<0.2		
12	<0.2	<0.2	<0.2	<0.2	<0.2		
13	<0.2	<0.2	<0.2	<0.2	<0.2		
14	<0.2	<0.2	<0.2	<0.2	<0.2		
15	<0.2	<0.2	<0.2	<0.2	<0.2		
16	<0.2	<0.2	<0.2	<0.2	<0.2		
17	<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)
(Page 21 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%	48%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/16/86	3/8/87		

1	<0.2	<0.2	<0.2	<0.2	<0.2		
2	<0.2	<0.2	<0.2	<0.2	<0.2		
3	<0.2	<0.2	<0.2	<0.2	2.0		
4	<0.2	<0.2	<0.2	<0.2	<0.2		
5	<0.2	<0.2	<0.2	<0.2	0.3		
6	<0.2	<0.2	<0.2	<0.2	<0.2		
7	<0.2	<0.2	<0.2	<0.2	<0.2		
8	<0.2	<0.2	<0.2	<0.2	0.9		
9	<0.2	<0.2	<0.2	<0.2	<0.2		
10	<0.2	<0.2	<0.2	<0.2	<0.2		
11	<0.2	<0.2	<0.2	<0.2	<0.2		
12	<0.2	<0.2	<0.2	<0.2	<0.2		
13	<0.2	<0.2	<0.2	<0.2	1.2		
14	<0.2	<0.2	<0.2	<0.2	<0.2		
15	<0.2	<0.2	<0.2	<0.2	<0.2		
16	<0.2	<0.2	<0.2	<0.2	<0.2		
17	<0.2	<0.2	<0.2	<0.2	<0.2		
18	<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 22 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%	48%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/16/86	3/8/87		

19	<0.2	<0.2	<0.2	<0.2	<0.2		
20	<0.2	<0.2	<0.2	<0.2	<0.2		
21	<0.2	<0.2	<0.2	<0.2	<0.2		
22	<0.2	<0.2	<0.2	<0.2	<0.2		
23	<0.2	<0.2	<0.2	<0.2	<0.2		
24	<0.2	<0.2	<0.2	<0.2	<0.2		
25	<0.2	<0.2	<0.2	<0.2	0.3		
26	<0.2	<0.2	<0.2	<0.2	<0.2		
27	<0.2	<0.2	<0.2	<0.2	2.0		
28	<0.2	<0.2	<0.2	<0.2	<0.2		
29	<0.2	<0.2	<0.2	<0.2	<0.2		
30	<0.2	<0.2	<0.2	<0.2	0.2		
31	<0.2	<0.2	<0.2	<0.2	<0.2		
32	<0.2	<0.2	<0.2	<0.2	<0.2		
33	<0.2	<0.2	<0.2	<0.2	<0.2		
34	<0.2	<0.2	<0.2	<0.2	<0.2		
35	<0.2	<0.2	<0.2	<0.2	<0.2		
36	<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 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%	48%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/16/86	3/8/87		
37	<0.2	<0.2	<0.2	<0.2	<0.2		
38	<0.2	<0.2	<0.2	<0.2	<0.2		
39	<0.2	<0.2	<0.2	<0.2	<0.2		
40	<0.2	<0.2	<0.2	<0.2	<0.2		
41	<0.2	<0.2	<0.2	<0.2	<0.2		
42	<0.2	<0.2	<0.2	<0.2	<0.2		
43	<0.2	<0.2	<0.2	<0.2	<0.2		
44	<0.2	<0.2	<0.2	<0.2	<0.2		

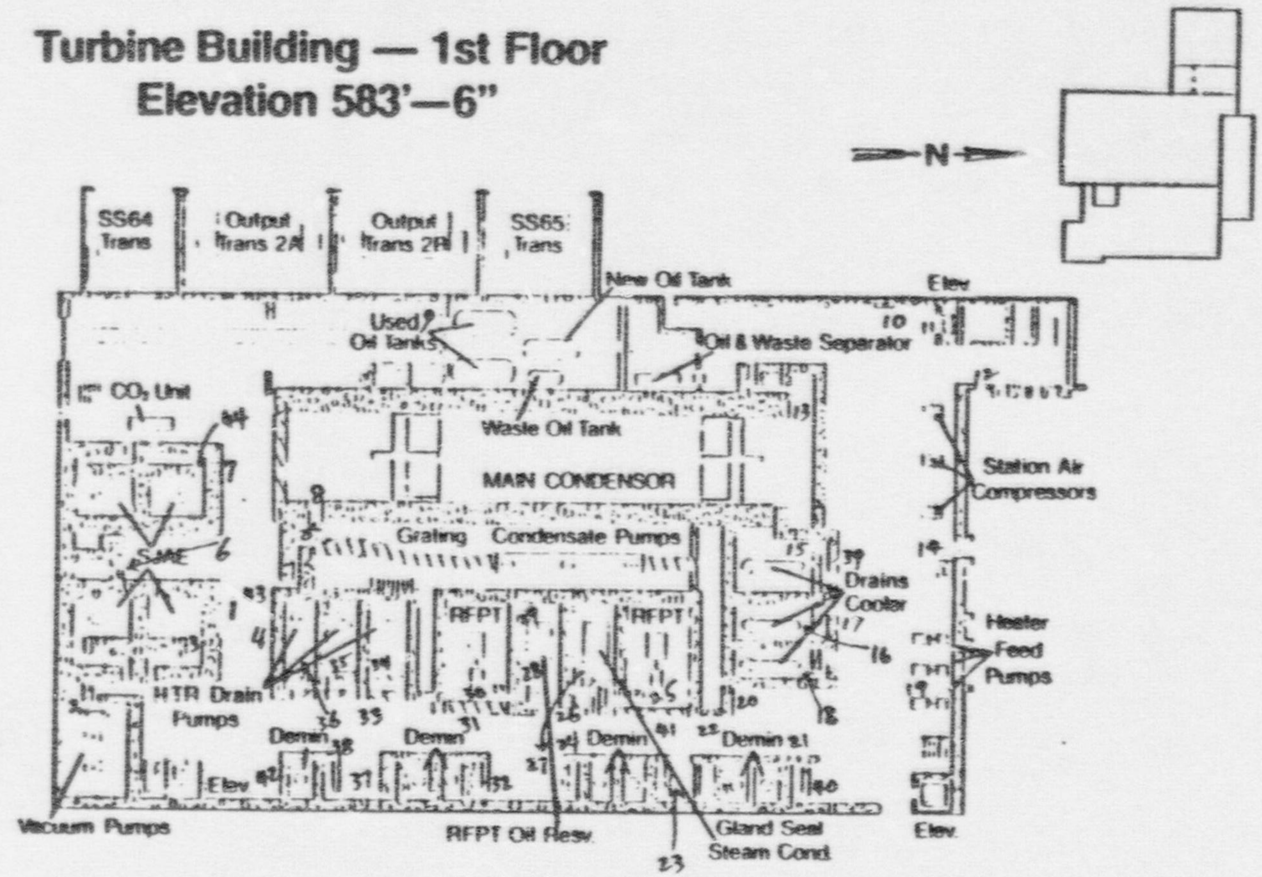
NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

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

FIGURE 3.2-3

(Page 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%	48%		
MWe	0	0	0	0	425		
Mod-Temp	<100^{o}F	<100^{o}F	540^{o}F	540^{o}F	510^{o}F		
Date	1/19/85	4/16/85	7/13/85	10/16/86	3/8/87		

1	<0.2	<0.2	<0.2	<0.2	<0.2		
2	<0.2	<0.2	<0.2	<0.2	<0.2		
3	<0.2	<0.2	<0.2	<0.2	<0.2		
4	<0.2	<0.2	<0.2	<0.2	<0.2		
5	<0.2	<0.2	<0.2	<0.2	<0.2		
6	<0.2	<0.2	<0.2	<0.2	<0.2		
7	<0.2	<0.2	<0.2	<0.2	<0.2		
8	<0.2	<0.2	<0.2	<0.2	<0.2		
9	<0.2	<0.2	<0.2	<0.2	<0.2		
10	<0.2	<0.2	<0.2	<0.2	<0.2		
11	<0.2	<0.2	<0.2	<0.2	<0.2		
12	<0.2	<0.2	<0.2	<0.2	<0.2		
13	<0.2	<0.2	<0.2	<0.2	<0.2		
14	<0.2	<0.2	<0.2	<0.2	<0.2		
15	<0.2	<0.2	<0.2	<0.2	<0.2		
16	<0.2	<0.2	<0.2	<0.2	10.00		
17	<0.2	<0.2	<0.2	<0.2	<0.2		
18	<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%	48%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/16/86	3/8/87		

19	<0.2	<0.2	<0.2	<0.2	<0.2		
20	<0.2	<0.2	<0.2	<0.2	<0.2		
21	<0.2	<0.2	<0.2	<0.2	<0.2		
22	<0.2	<0.2	<0.2	<0.2	<0.2		
23	<0.2	<0.2	<0.2	<0.2	<0.2		
24	<0.2	<0.2	<0.2	<0.2	<0.2		
25	<0.2	<0.2	<0.2	<0.2	<0.2		
26	<0.2	<0.2	<0.2	1.2	6.0		
27	<0.2	<0.2	<0.2	<0.2	<0.2		
28	<0.2	<0.2	<0.2	<0.2	<0.2		
29	<0.2	<0.2	<0.2	<0.2	<0.2		
30	<0.2	<0.2	<0.2	<0.2	<0.2		
31	<0.2	<0.2	<0.2	<0.2	<0.2		
32	<0.2	<0.2	<0.2	<0.2	<0.2		

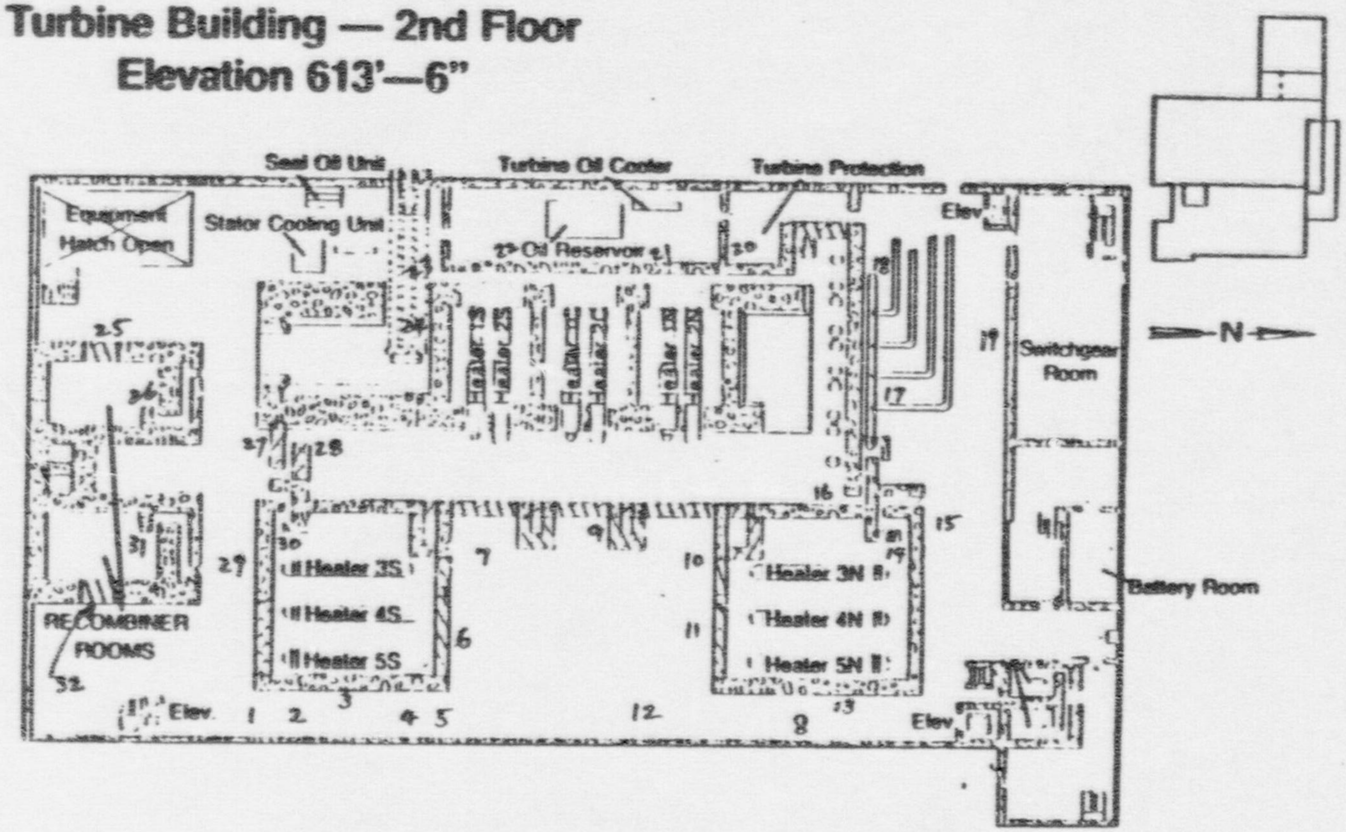
NOTE 1: Readings are total readings where possible. Total reading = gamma + neutron

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%	48%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/65	10/16/86	3/8/87		

1	<0.2	<0.2	<0.2	<0.2	<0.2		
2	<0.2	<0.2	<0.2	<0.2	<0.2		
3	<0.2	<0.2	<0.2	<0.2	8.0		
4	<0.2	<0.2	<0.2	<0.2	<0.2		
5	<0.2	<0.2	<0.2	<0.2	<0.2		
6	<0.2	<0.2	<0.2	<0.2	<0.2		
7	<0.2	<0.2	<0.2	<0.2	0.4		
8	<0.2	<0.2	<0.2	<0.2	<0.2		
9	<0.2	<0.2	<0.2	<0.2	<0.2		
10	<0.2	<0.2	<0.2	<0.2	<0.2		
11	<0.2	<0.2	<0.2	<0.2	<0.2		
12	<0.2	<0.2	<0.2	<0.2	<0.2		
13	<0.2	<0.2	<0.2	<0.2	<0.2		
14	<0.2	<0.2	<0.2	<0.2	0.5		
15	<0.2	<0.2	<0.2	<0.2	<0.2		
16	<0.2	<0.2	<0.2	<0.2	<0.2		
17	<0.2	<0.2	<0.2	<0.2	<0.2		
18	<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%	48%		
[illegible]	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/16/86	3/8/87		
19	<0.2	<0.2	<0.2	<0.2	<0.2		
20	<0.2	<0.2	<0.2	<0.2	2.0		
21	<0.2	<0.2	<0.2	<0.2	<0.2		
22	<0.2	<0.2	<0.2	<0.2	<0.2		
23	<0.2	<0.2	<0.2	<0.2	<0.2		
24	<0.2	<0.2	<0.2	<0.2	<0.2		
25	<0.2	<0.2	<0.2	<0.2	12		
26	<0.2	<0.2	<0.2	<0.2	<0.2		
27	<0.2	<0.2	<0.2	<0.2	<0.2		
28	<0.2	<0.2	<0.2	<0.2	<0.2		
29	<0.2	<0.2	<0.2	<0.2	<0.2		
30	<0.2	<0.2	<0.2	<0.2	0.2		
31	<0.2	<0.2	<0.2	<0.2	<0.2		
32	<0.2	<0.2	<0.2	<0.2	0.2		
33	<0.2	<0.2	<0.2	<0.2	0.2		
34	<0.2	<0.2	<0.2	<0.2	3.0		
35	<0.2	<0.2	<0.2	<0.2	<0.2		
36	<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%	48%		
MWe	0	0	0	0	425		
Mod-Temp	<100°F	<100°F	540°F	540°F	510°F		
Date	1/19/85	4/16/85	7/13/85	10/16/86	3/8/87		

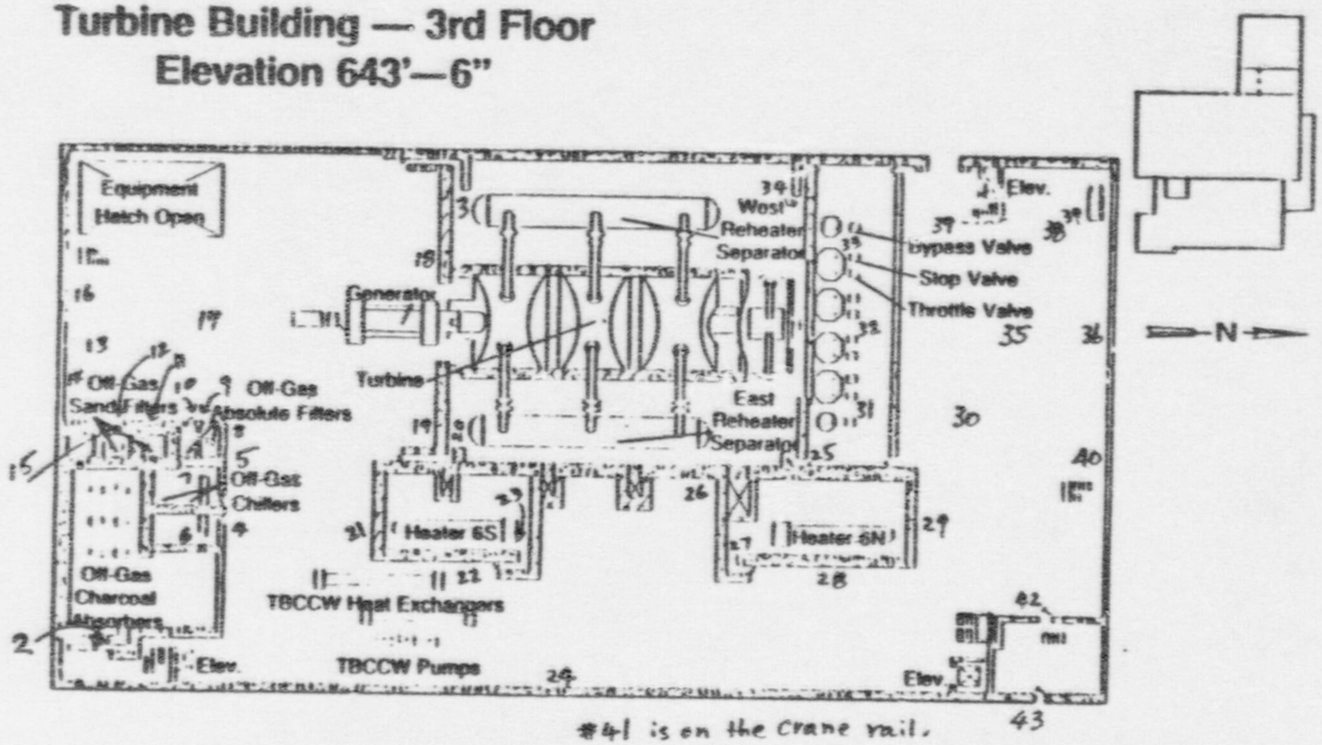
37	<0.2	<0.2	<0.2	<0.2	<0.2		
38	<0.2	<0.2	<0.2	<0.2	<0.2		
39	<0.2	<0.2	<0.2	<0.2	<0.2		
40	<0.2	<0.2	<0.2	<0.2	<0.2		
41	<0.2	<0.2	<0.2	<0.2	1.0		
42	<0.2	<0.2	<0.2	<0.2	<0.2		
43	<0.2	<0.2	<0.2	<0.2	<0.2		

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 (± 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

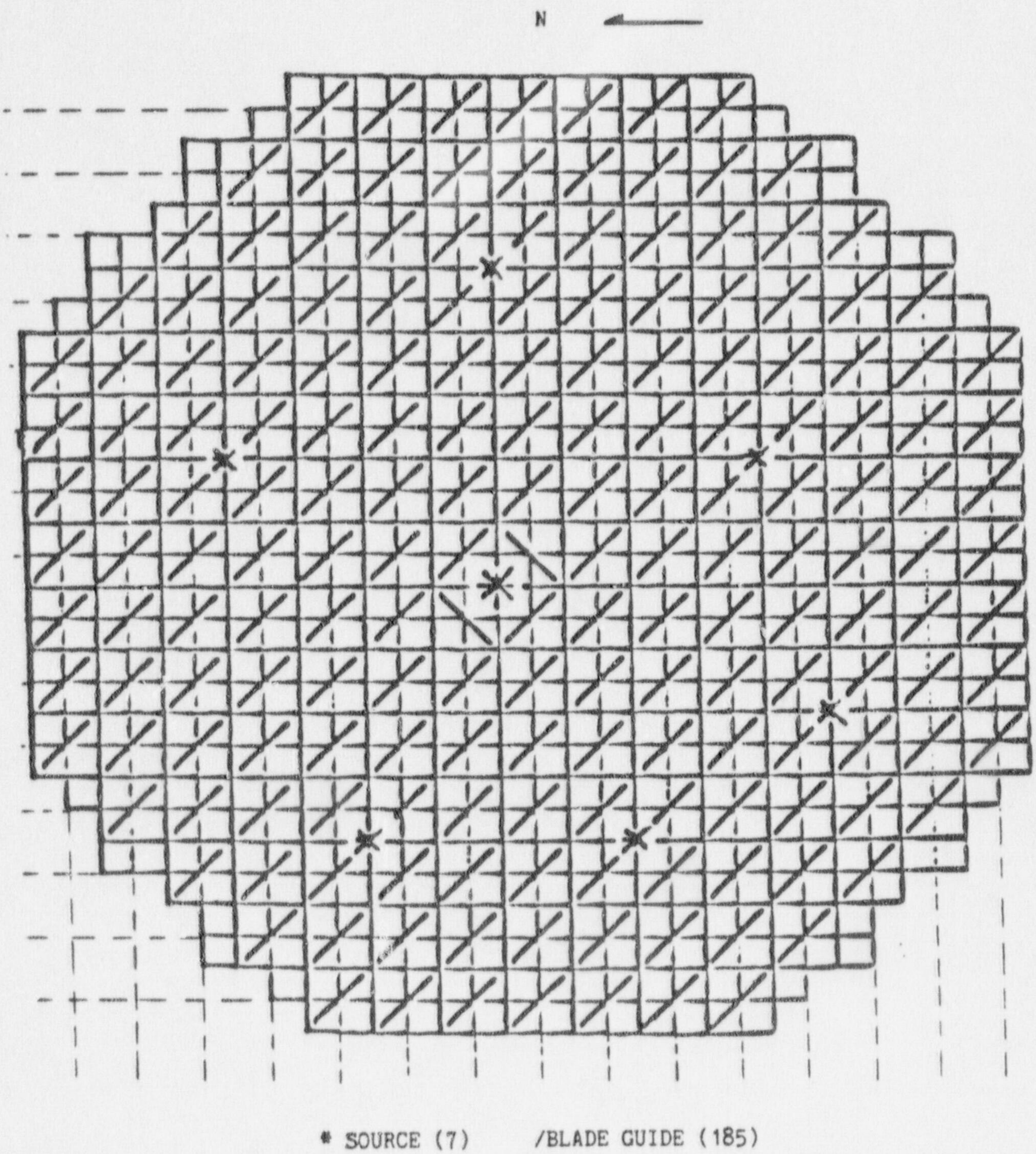
with only the strongest rod withdrawn is 0.9',
sub-criticality for the partial core demonstrated that the shutdown margin would be met throughout the remaining fuel loading process.

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

FIGURE 3.3-1

MEUTRON 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*	6.8	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

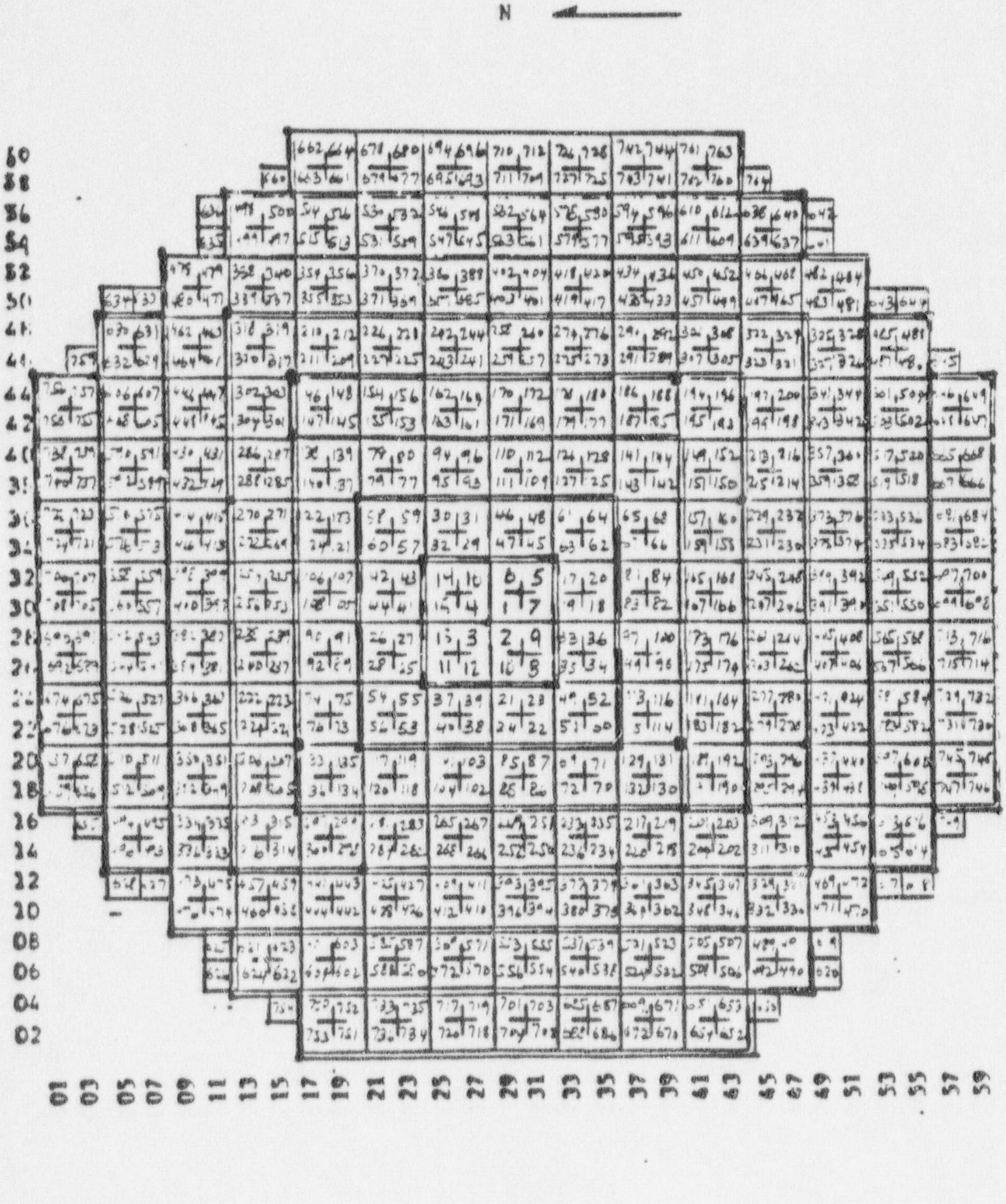


FIGURE 3.3-4

FLC MOVES DURING CORE LOADING

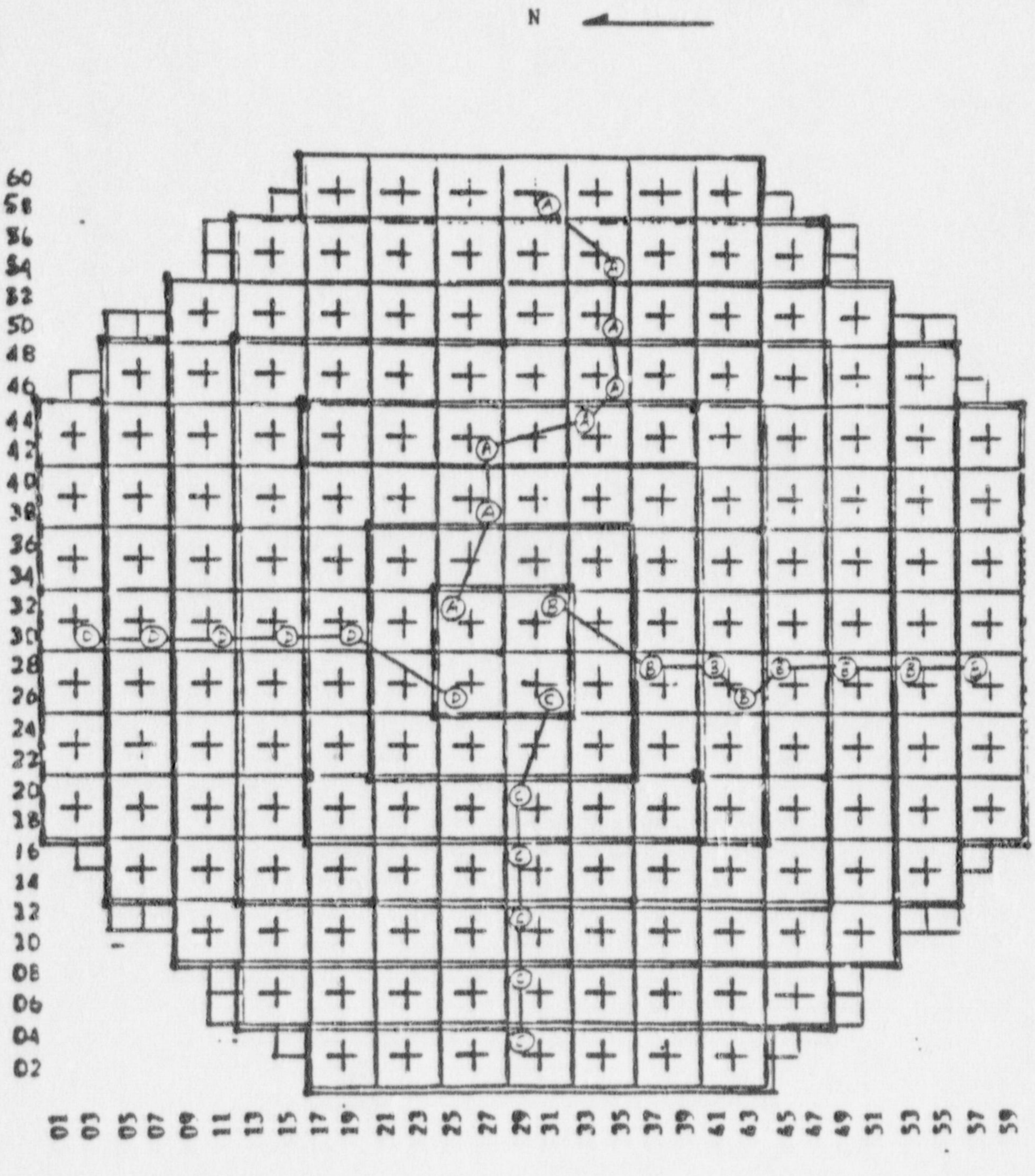


FIGURE 3.3-5

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

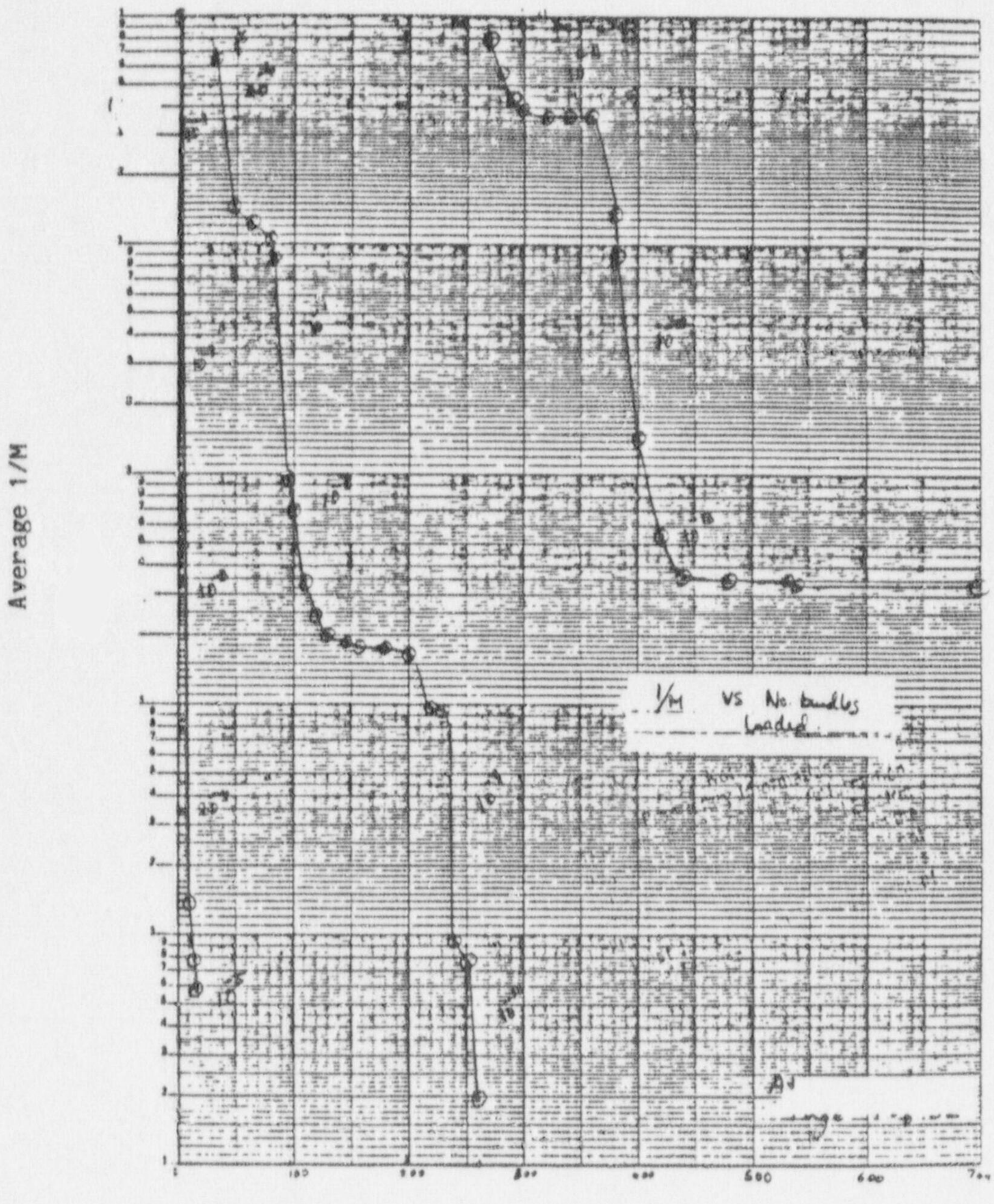


FIGURE 3.3-7
(Page 1 of 2)

FUEL LOCATION VERIFICATION

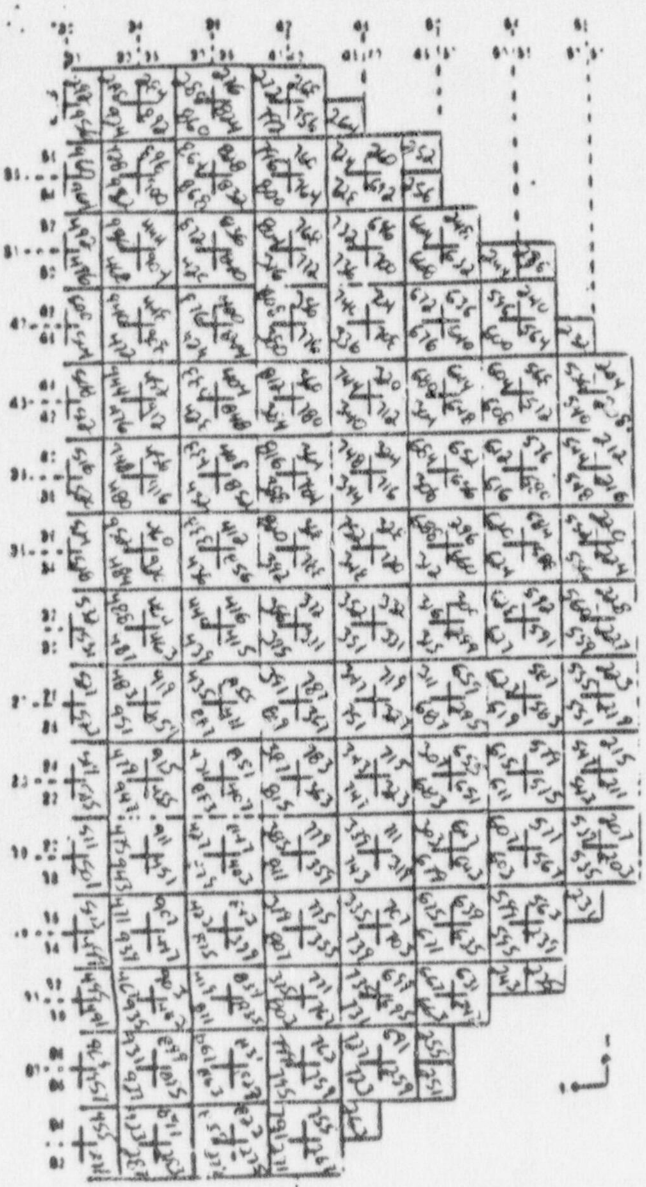
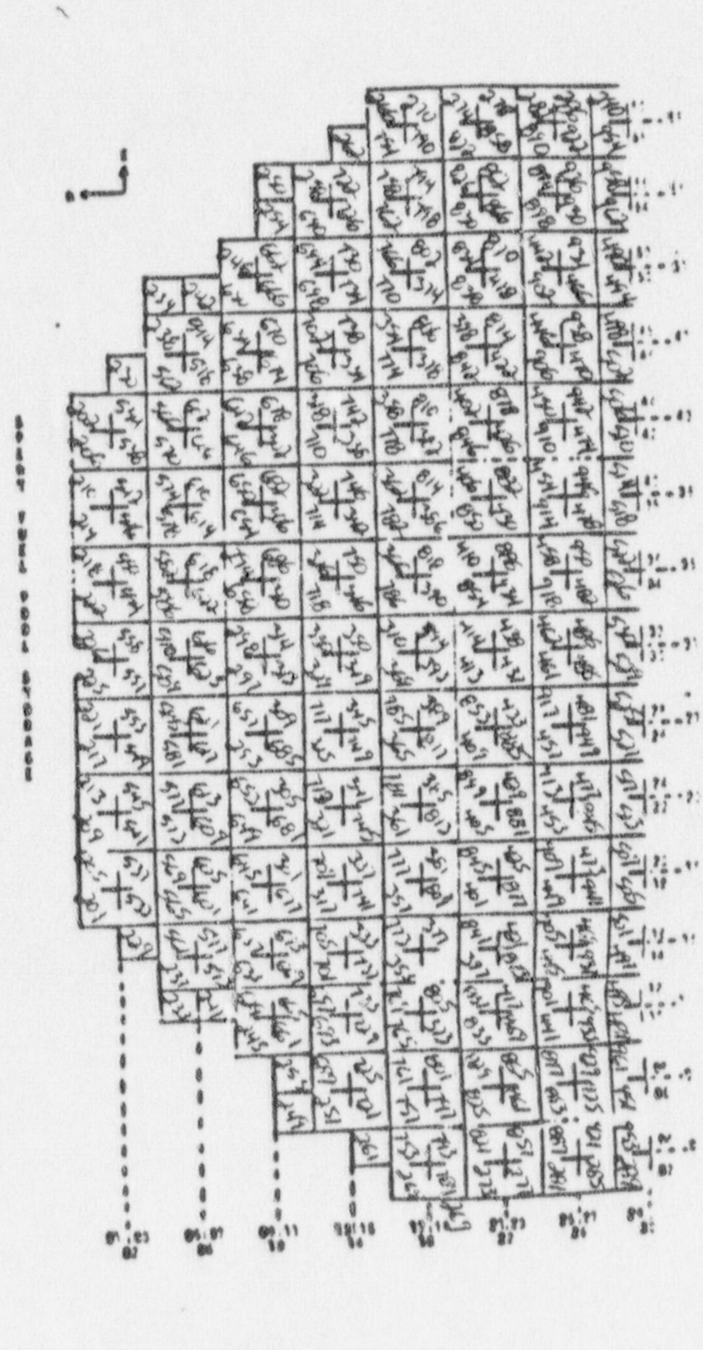


FIGURE 3.3-7
(Page 2 of 2)

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 ± 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 (± 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 88 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

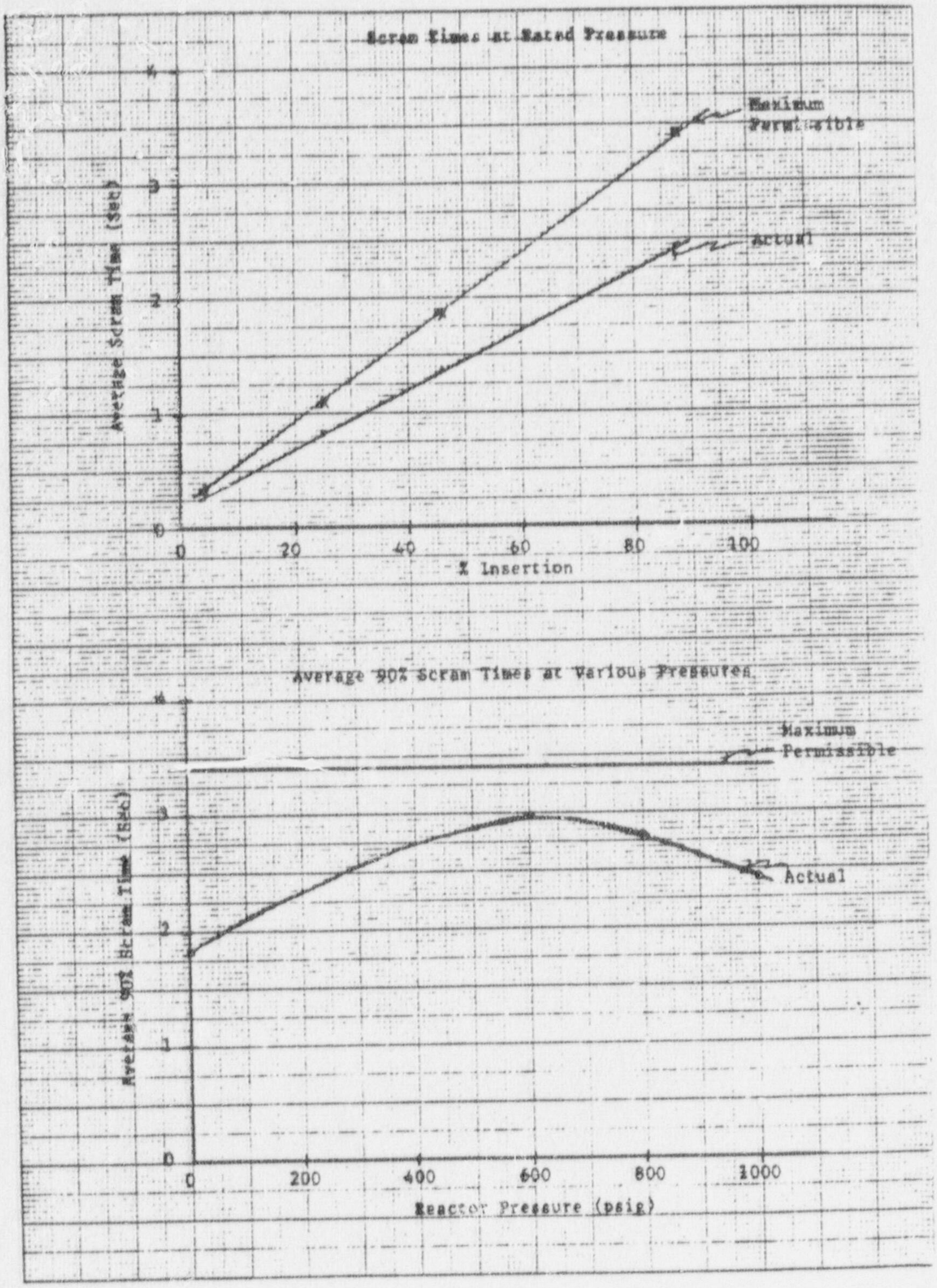
Test Description	Accumulator Pressure	Preop Tests	Reactor Pressure with Core Loaded psig			
			0	600	800	rated
Position Indication		All	All			
Normal Stroke Times Insert/Withdraw		All	All			4(a)
Coupling		All	All			
Friction			All			4(a)
Scram	Normal	All	All	4(a)	4(a)	All
Scram	Minimum		4(a)			4(a)
Scram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

Level 1

None

Level 2

The difference between the actual reference leg temperature(s) and the value(s) assumed during initial calibration shall be less than that amount 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 will be reperformed after APRMs are adjusted at a higher power level.

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.

This test will be reperformed at a higher power level during Test Condition Three.

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.

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%

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

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.

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 $28.\bar{4}$ 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 $28.\bar{5}$ 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.

3.13 HPCI System

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 after 25 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 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.

To provide a margin on the overspeed trip and isolation, the transient start first speed peak shall not come closer to the overspeed trip than 15 percent of rated speed, and subsequent speed peaks shall not be greater than 5 percent above the rated turbine speed.

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 with the CST in which 500 gpm flow step changes were manually and automatically introduced by the flow controller with HPCI system flows at 5000 gpm and 2700 gpm.

During the 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 manual or automatic starts. There was also adequate margin on turbine speed peaks and oscillations of system variables. An extended run was also performed in which system temperatures stabilized at acceptable levels and the gland seal system performed satisfactorily.

All Level 1 and Level 2 criteria are satisfied except 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 when a mid-flow speed decrease step change was inserted 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 the 9/86 retesting of HPCI, sluggish response was noted in the HPCI control valve. As a result, it was decided to replace the EGR component 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. On the quick start 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.

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.

The remaining testing in this section will be completed at higher test conditions, including those tests intended to verify the Level 2 criteria at rated core flow.

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 ± 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.

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
Percent of Rated Core Thermal Power
Core Flow
Maximum Linear Heat Generation Rate
Minimum Critical Power Ratio
Maximum Average Planar Linear Heat Generation Rate

These parameters are monitored in this manner throughout the testing program.

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.

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. The balance of feedwater system testing not completed during Test Condition Two has been incorporated into the Test Condition Three feedwater system test progression.

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

The Turbine Valve Surveillance test has not been completed to date.

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.

The remaining Level 1 and Level 2 criteria are associated with the MSIV simultaneous full closure and will not be verified until that test is performed during a higher test condition.

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 $10^{O}F$ 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 val 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, and the effects of steam flow on bypass valve response time will be further evaluated during the generator load rejection test in Test Condition Six.

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 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 $\leq$0.25 at any speed.

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

The APRM neutron flux trip avoidance margin shall be $\geq$7.5 percent, and the heat flux trip avoidance margin shall be $\geq$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 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%.

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.

Further testing will be performed in Test Conditions Three, Four and Six which relate to the above criteria.

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 ±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 the RCIC Pump Discharg 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 have been incorporated into future test plans.

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

The testing associated with this section is performed in Test Conditions Three and Six.

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^{o}F in the blowdown mode and shall not exceed 120^{o}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^{o}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^{o}F and 122^{o}F respectively. These values were within the Level 2 criteria limits of 120^{o}F and 130^{o}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^{o}F. All applicable Level 2 criteria were satisfied.

The remaining testing for the Reactor Water Cleanup System (Hot Standby Operation) will be completed in Test Condition Four.

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.

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.

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Plant Technical
Specifications

Nuclear Operations

June 19, 1987
NRC-87-0084

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Reference: (1) Fermi 2
NRC Docket No. 50-341
Facility Operating License No. NPF-43

(2) Detroit Edison Letter to NRC
"Startup Report - Supplement 3"
VP-NO-87-0055, dated March 20, 1987

Subject: Startup Report - Supplement 4

This is Supplement 4 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 September 20, 1987.

If you have any questions regarding this report, please contact Mr. Girija Shukla, Compliance Engineer at (313) 586-5313.

Sincerely,

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