THE DETRDIT EDISDN COMPANY

FERMI 2
NUCLEAR POWER PLANT

INTERIM STARTUP TEST REPORT

8603260122 860312
PDR ADOCK 05000341
P PDR

FERMI 2
NUCLEAR POWER PLANT

INTERIM STARTUP TEST REPORT

INDEX

Section		Page
1.0	Introduction	
	1.1 Purpose	1
	1.2 Test Report Format	1
	1.3 Plant Description	1
	1.4 Test Program Description	2
	1.5 References	3
2.0	General Test Program Information	
	2.1 Chronology of Major Events	6
	2.2 Matrix of Test Completion Dates	7
3.0	Test Results Summary	
	3.1 Chemical and Radiochemical	8
	3.2 Radiation Measurements	14
	3.3 Fuel Loading	54
	3.4 Full Core Shutdown Margin	66
	3.5 Control Rod Drive System	68
	3.6 SRM Performance and Control Rod Sequence Exchange	73
	3.7 IRM Performance	74
	3.8 LPRM Calibration	75
	3.9 APRM Calibration	76
	3.10 Process Computer	77
	3.11 RCIC System	78
	3.12 HPCI System	80
	3.13 Select Process Temperatures	82
	3.14 System Expansion	84
	3.15 Feedwater System	87
	3.16 MSIV Functional Test	88
	3.17 Relief Valves	89
	3.18 Piping System Vibration	91
	3.19 Reactor Water Cleanup System	92
	3.20 Residual Heat Removal System	94

FERMI 2
NUCLEAR POWER PLANT

INTERIM STARTUP TEST REPORT

1.0 Introduction

1.1 Purpose

The purpose of this Interim Startup Test Report is to provide a summary of the test results obtained in startup testing completed from initial fuel load through present heatup test condition testing. This report of plant startup and power ascension testing is submitted as required following receipt of the Operating License in compliance with Technical Specification 6.9.1.1. The testing activities performed through the period nine months after initial criticality are outlined herein in accordance with Technical Specification 6.9.1.3. This is an interim report that covers all testing applicable to the test conditions completed as described in FSAR Subsection 14.2.12.2. Supplementary reports will be issued to cover the remaining testing as required 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, or 1100 MWe, constructed in a Mark I containment with a toroidal suppression pool.

This plant is owned and operated by the Detroit Edison Company (90%) and the Wolverine Power Cooperative, Incorporated (10%).

1.4 Startup Test Program Description

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

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

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

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

1. To achieve an orderly and safe initial core loading

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

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

4. To conduct initial heatup and hot functional testing so that hot integrated operation of all systems is 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.

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

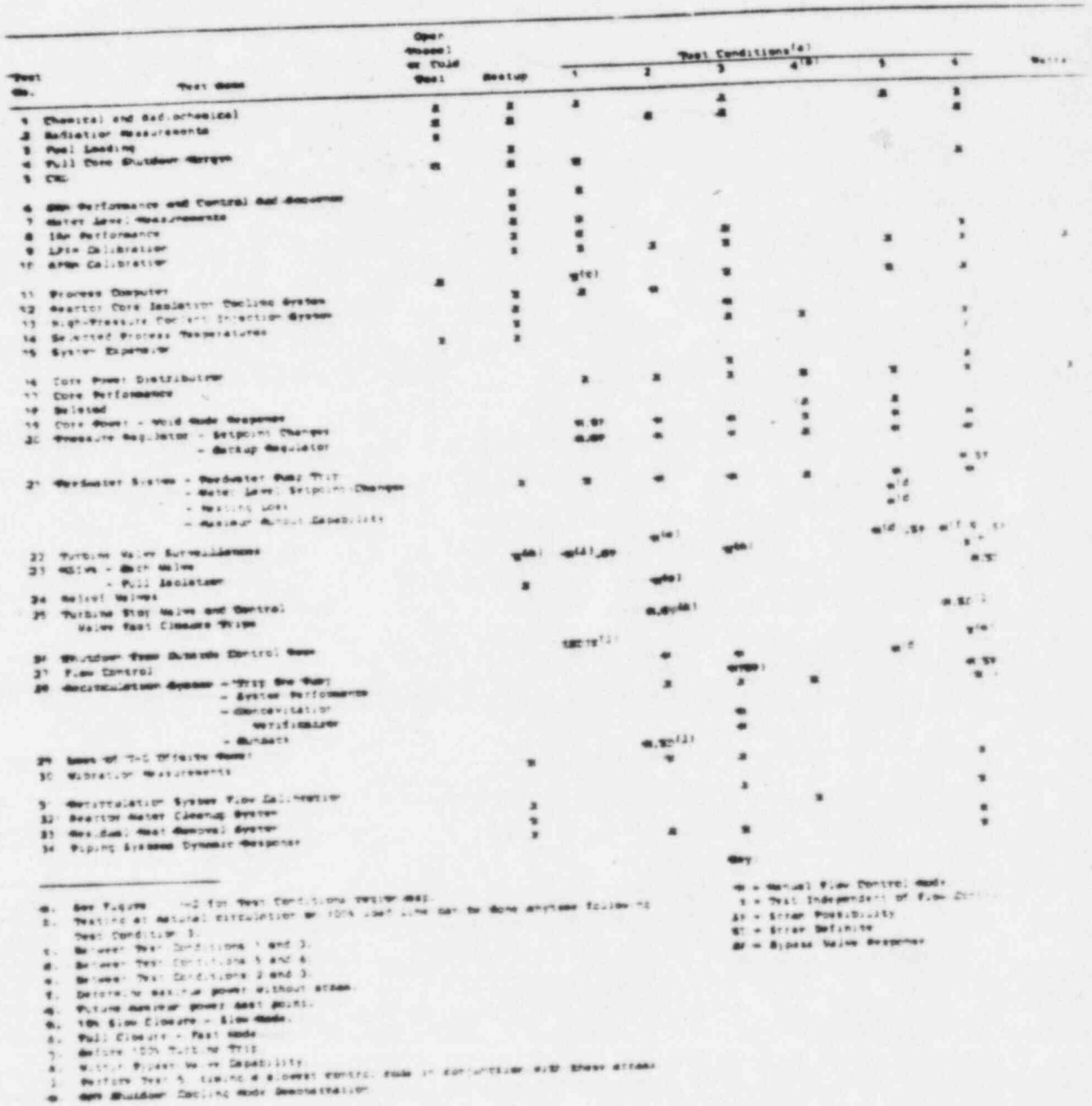
1.5 References

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.

FIGURE 1-1 STARTUP TEST PROGRAM



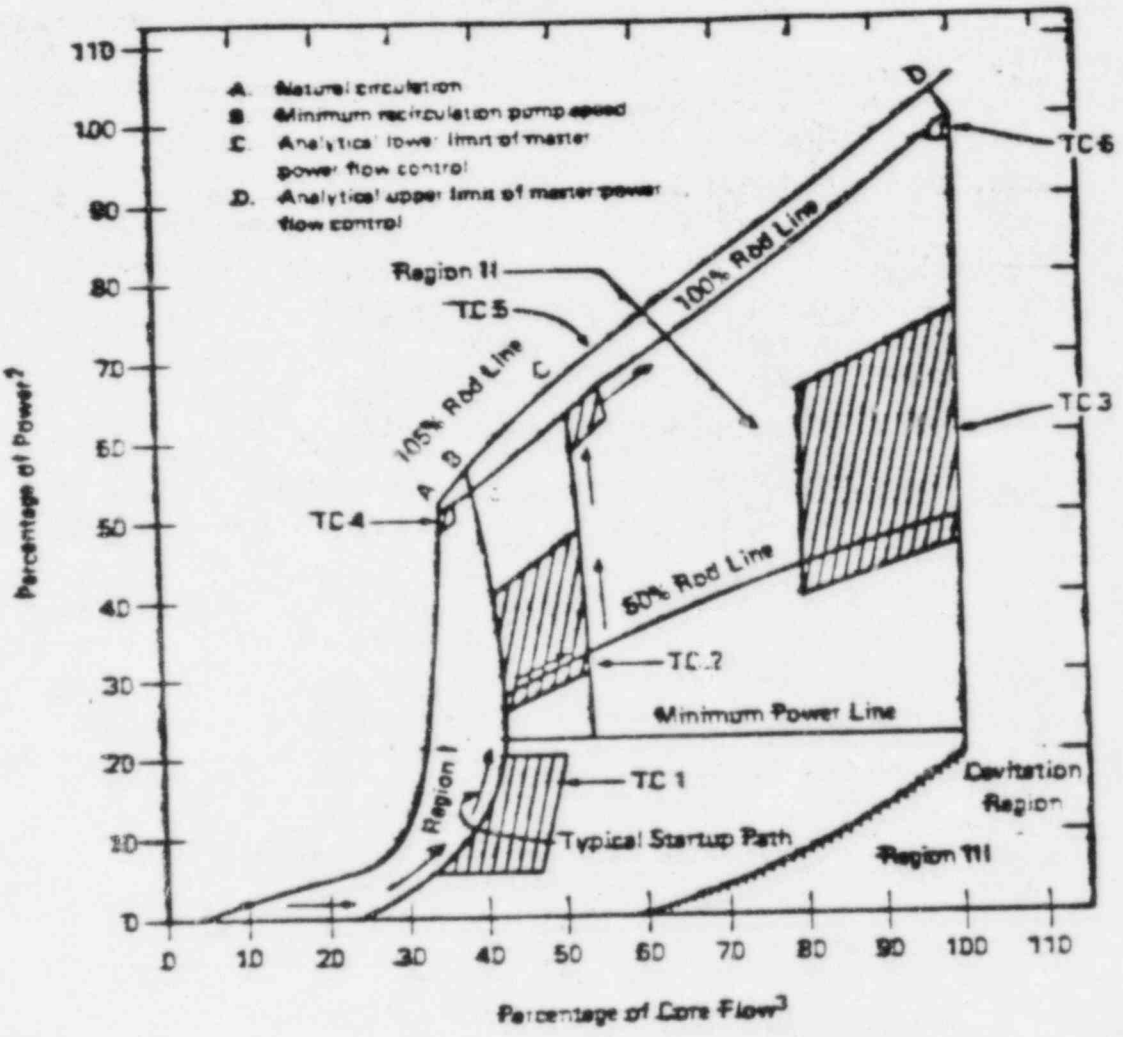
Test No.	Test Name	Open Vessel or Cold Test	Restup	Test Conditions[a] 1	2	3	4[b]	5	6	Water
1	Chemical and Radiochemical	X	X	X		X		X	X	
2	Radiation Measurements	X	X		X	X			X	
3	Fuel Loading	X								
4	Full Core Shutdown Margin		X						X	
5	CRD	X	X	X						
6	SRM Performance and Control Rod Sequence		X	X						
7	Water Level Measurements		X							
8	IRM Performance		X	X		X			X	
9	LPRM Calibration		X	X	X	X		X	X	X
10	APRM Calibration		X	X						
11	Process Computer	X		X[c]		X		X	X	
12	Reactor Core Isolation Cooling System		X	X	X					
13	High-Pressure Coolant Injection System		X			X			X	
14	Selected Process Temperatures		X			X	X		X	
15	System Expansion	X	X						X	
16	Core Power Distribution					X			X	
17	Core Performance			X	X	X	X	X	X	X
18	Deleted									
19	Core Power - Void Mode Response						X	X		
20	Pressure Regulator - Setpoint Changes			X,SP	X	X	X	X	X	
	- Backup Regulator			X,SP	X	X	X	X	X	
21	Feedwater System - Feedwater Pump Trip								X,SP	
	- Water Level Setpoint Changes		X	X	X	X	X	X	X	
	- Heating Loss							X[d]		
	- Maximum Runout Capability							X[d]		
22	Turbine Valve Surveillances				X[e]			X[d],SP	X[f],SP,SC	
23	MSIVs - Each Valve		X[h]	X[d],SP		X[h]			X[f]	
	- Full Isolation								X,SC	
24	Relief Valves		X		X[e]					
25	Turbine Stop Valve and Control Valve Fast Closure Trips				X,SP[h]				X,SC[i]	
26	Shutdown From Outside Control Room			SC[j]					X[e]	
27	Flow Control				X	X		X[d]		
28	Recirculation System - Trip One Pump					X[e]			X,SP	
	- System Performance				X	X	X		X	
	- Concavitation Verification					X				
	- Runback					X				
29	Loss of T-G Offsite Power				X,SC[j]					
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:

X = Manual Flow Control Mode
X = Test Independent of Flow Control
SP = Scram Possibility
SC = Scram Definite
SP = Bypass Valve Response

a. See Figure 1-2 for Test Conditions region map.
b. Testing at natural circulation on 100% 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. Between Test Conditions 2 and 3.
f. Determine maximum power without scram.
g. Future maximum power test point.
h. 10% Slow Closure - Slow Mode.
i. Full Closure - Fast Mode.
j. Before 100% Turbine Trip.
k. Within Bypass Valve Capability.
l. Perform Test 5, timing a slowest control rode in conjunction with these scrams.
m. [illegible] Shutdown Cooling Mode Demonstration.

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.

FIGURE 1-2

2.0 General Test Program Information

2.1 Chronology of Major Events

	Date
Received (5%) Facility Operating License No. NPF-33	3/20/85
Started Fuel Loading	3/20/85
Completed Fuel Loading	4/4/85
Completed Open Vessel Testing	6/1/85
Initial Criticality	6/21/85
Received (Full Power) Facility Operating License NPF-43	7/15/85
Completed Initial Turbine Roll	9/26/85
Fall Outage Begins	10/10/85

2.2 Matrix of Test Completion Dates

Test No.	Test Title	Pre-Fuel	Open Vessel	Heatup
1	Chemical and Radiochemical	1/14/85	---	7/16/85
2	Radiation measurements	1/24/85	4/19/85	7/14/85
3	Fuel Loading	---	4/5/85	---
4	Full Core Shutdown Margin	---	---	4/10/85
5	CRD	---	4/5/85	9/16/85
6	SRM Performance and Control Rod Sequence	---	---	8/25/85
7	Water Level Measurements	---	---	8/30/85
8	IRM Performance	---	---	8/19/85
9	LPRM Calibration	---	---	9/24/85
10	APRM Calibration	---	---	8/1/85
11	Process Computer	5/30/85	---	---
12	RCIC	---	---	7/14/85
13	HPCI	---	---	8/31/85
14	Selected Process Temperatures	---	---	9/25/85
15	System Expansion	---	6/12/85	9/1/85
* 16	Core Power Distribution	---	---	---
* 17	Core Performance	---	---	---
* 19	Core Power Void Mode Response	---	---	---
20	Pressure Regulator	---	---	---
21	Feedwater System	---	---	7/9/85
* 22	Turbine Valve Surveillance	---	---	---
23	MSIV	---	---	7/12/85
24	Relief Valves	---	---	7/3/85
* 25	Turbine Stop Valve and Control Valve Fast Closure	---	---	---
* 26	Shutdown from Outside Control Room	---	---	---
* 27	Flow Control	---	---	---
* 28	Recirculation System	---	---	---
* 29	Loss of Offsite Power	---	---	---
30	Vibration Measurements	---	---	7/3/85
* 31	Recirc. System Flow Calibration	---	---	---
32	Reactor Water Cleanup System	---	---	7/14/85
33	Residual Heat Removal System	---	---	8/31/85
34	Piping System Dynamic Response	---	---	9/24/85

--- = Test not performed in this test condition

* = Tests not addressed in this summary report

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.

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 3.1-1 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 3.1-2 for heatup chemistry data.

Pre-Fuel Load Chemistry Data

Date 1/15/85
CTP = 0
MWe = 0

System	Analysis	Results	Limit
REACTOR WATER	Conductivity (umho/cm) @ 25°C	0.43	≤ 10.0
	Chloride (ppb)	< 5	< 500
	Turbidity (NTU)	0.18	L.L.D.
	Boron (ppb)	< 10	< 50.0
	Silica (ppm)	0.023	< 5.0
	pH	6.2	5.3 ≤ pH ≤ 8.6
CONDENSATE	Conductivity (umho/cm) @ 25°C	0.5	≤ 0.5
	Chloride (ppb)	< 5	< 10.0
FEEDWATER	Insoluble Iron (ppb)	1.8	*
	Total Metals (ppb)	10	*
	Conductivity (umho/cm) @ 25°C	0.38	≤ 0.1
	Soluble Copper	1.4	*
	Insoluble Copper	0.1	*

*The limit of the soluble and insolubles (Total Metals) is that the total shall be ≤ 15 ppb of which can contain no more than 2 ppb copper.

L.L.D - Lowest Level of Detection

FIGURE 3.1-1

Heatup Test Condition Chemistry Data

Date 7/15/85
CIP = < 5%
MWe = 0

Reactor Water Temperature 540 °F

System	Analysis	Results	Limit
REACTOR WATER	Conductivity (umho/cm) @ 25°C	0.31	≤ 2.0
	Chloride (ppb)	8	< 200
	Turbidity (NTU)	3.6	**
	Iodine - 131 (uCi/ml)	<7.7 E-08	*
	- 133 (uCi/ml)	2.0 E-06	*
	2 hr Gross Activity - Filtrate (cpm/ml)	6.1 E-02	**
	- Crud (cpm/ml)	4.0 E-01	**
	7 dy Gross Activity - Filtrate (cpm/ml)	4.7	**
	- Crud (cpm/ml)	25	**
	Silica (ppm)	0.385	**
	pH	6.4	5.6 ≤ pH ≤ 8.6

*The dose equivalent Iodine-131 shall not exceed a concentration of 0.2 uCi/gm.
**This analysis is used to develop trend data.

FIGURE 3.1-2

System	Analysis	Results	Limit
CRD WATER	Conductivity (umho/cm) @ 25°C	0.08	≤ 0.1
	Oxygen (ppb)	500	≤ 50
RADIOACTIVE GASEOUS EFFLUENTS	Off-gas Vial Sample Analysis H21-P275A	Completed	*
	Off-gas Monitor Reading D11-K601A (mR/hr)	3.5	***
	Off-gas Monitor Reading D11-K601B (mR/hr)	3.5	***
	SGTS Exhaust, Div I D11-P275 (uCi/cc)	1.7 E-08	** Background Corrected
	SGTS Exhaust, Div II D11-P275 (uCi/cc)	1.2 E-08	** Background Corrected
	Turbine Bldg (uCi/cc) D11-P279	8.0 E-08	** Background Corrected
	Reactor Bldg (uCi/cc) D11-P280	4.05 E-07	** Background Corrected
	Radwaste Bldg (uCi/cc) D11-P281	9.1 E-07	** Background Corrected
	Service Bldg (uCi/cc) D11-P282	8.1 E-08	** Background Corrected
	Site Storage Bldg (uCi/cc) D11-P299	1.2 E-07	** Background Corrected

*Perform isotopic analysis of an Off-gas sample and attach data. This is for trend analysis.

**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.

***For trend analysis

FIGURE 3.1-2

System	Analysis	Results	Limit
DEMIN. EFFLUENT	Conductivity (umho/cm) @ 25°C	0.072	≤ 0.1
FEEDWATER	Insoluble Iron (ppb)	1	*
	Conductivity (umho/cm) @ 25°C	A 0.08 B 0.10	≤ 0.1 ≤ 0.1
	Insoluble Copper (ppb)	0.03	*
	Soluble Copper (ppb)	1	*
	Total Metals (ppb)	2.33	*
CONDENSATE	Conductivity (umho/cm) @ 25°C	0.14	≤ 0.5
	Chloride (ppb)	< 2	**
	Insoluble Iron (ppb)	3	**

*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 this Test Condition.

** For trend analysis.

FIGURE 3.1-2

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.

3.2.3 Results

Prior to fuel load, during open vessel and heatup test conditions, radiation measurements were taken in the form of process and area radiation monitor data and site surveys. This data was acceptable and personnel radiation protection was provided in full compliance with the criteria. See Figures 3.2-1 through 3.2-3 for applicable monitor and survey readings.

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 Dff 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

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).

FIGURE 3.2-1

Area Radiation Monitor Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod-Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

1	0.03 mr/hr	<0.1	0.05
2	0.02	<0.1	0.02
3	0.03	<0.1	0.02
4	0.02	<0.1	0.02
5	0.08	<0.1	0.08
6	0.03	<0.1	0.04
7	0.2	0.2	0.2
8	0.2	0.4	0.4
9	0.3	0.3	0.3
10	0.2	0.2	0.2
11	0.3	0.3	0.3
12	5	4	5
13	0.03	<0.1	0.04
14	3	3	4
15	0.02	<0.1	0.02
16	0.04	<0.1	0.05
17	0.02	<0.1	2
18	300	300	300
19	0.4	0.4	0.4
20	0.3	0.3	0.3
21	2	3	3
22	0.03	<0.1	0.03
23	0.02	<0.1	0.03
24	0.3	0.3	0.3
25	0.3	0.3	0.3
26	0.3	0.2	0.2
27	OOS	OOS	OOS
28	0.03	<0.1	0.03
29	0.06	<0.1	0.06

NOTE 1: These represent remote readings where possible
NOTE 2: OOS indicates that the monitor is "out of service".

FIGURE 3.2-1

Area Radiation Monitor Data
(Cont'd)

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod-Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

30	0.03 mr/hr	<0.1	0.04
31	0.2	0.2	0.2
32	5	4	5
33	0.02	<0.1	0.02
34	0.05	<0.1	0.07
35	0.03	<0.1	0.03
36	0.03	<0.1	0.03
37	0.02	<0.1	0.02
38	0.02	<0.1	0.02
39	0.05	<0.1	0.08
40	0.03	<0.1	0.03
41	0.03	<0.1	0.03
42	0.05	<0.1	0.04
43	0.04	<0.1	0.04
44	0.03	<0.1	0.02
45	OOS	0.2	OOS
46	0.2	0.4	0.4
47	3	3	0.3
48	3	4	3

NOTE 1: These represent remote readings where possible
NOTE 2: OOS indicates that the monitor is "out of service".

FIGURE 3.2-1

Process Radiation Monitor Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Off-Gas Radiation - A (D11-K601A)	3 mr/hr	2	3.5
Off Gas Radiation - B (D11-K601B)	3.5 mr/hr	3	3
Off Gas Radiation - Linear (D11-K602)	0 mr/hr	5E-8	0
Radwaste Effluent (D11-K604)	2 cps	2	2
GSW Effluent (D11-K605)	3.5 cps	4	4
RBCCW System (D11-K606)	3 cps	2	2
A Main Steam Line (D11-K603A)	1.2 mr/hr	1	OOS
B Main Steam Line (D11-K603B)	1.4 mr/hr	1	1
C Main Steam Line (D11-K603C)	1.6 mr/hr	2	1.2
D Main Steam Line (D11-K603D)	8.0 mr/hr	11	1
Div-I EECW HX Inlet (D11-K800A)	200 cpm	200	200
Div-I RHR S.W. (D11-K801A)	200 cpm	200	200
Div-I RB Vent Exh. (D11-K808)	50 cpm	50	40
Div-I Cont. Ctr. Makeup Air (D11-K809)	40 cpm	30	30
Two Min. Holdup Pipe Exh. (D11-K814)	250 cpm	200	400
Div-II EECW HX Inlet (D11-K800B)	200 cpm	200	200
Div-II RHR S.W. (D11-K801B)	150 cpm	200	200

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

Process Radiation Monitor Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Circ. Water Rev. Decant Line (D11-K802)	200 cpm	200	200
Div-II RB Vent Exh. (D11-K810)	60 cpm	60	40
Div-II Cont. Ctr. Makeup Air (D11-K813)	60 cpm	60	60
Two Min. Holdup Pipe Exh. (D11-K815)	300 cpm	200	400
1st. Flr. Inside Drywell (D21-K745)	0.3 mr/hr	0.2	DOS
1st Flr. On Site Stg. Bldg. Control Room (D21-K846)	0.2 mr/hr	0.4	0.5
1st Flr. On Site Stg. Bldg. Compactor Room (D21-K847)	3 mr/hr	4	4
1st Flr. On Site Stg. Bldg. Truck Unloading Station (D21-K848)	3 mr/hr	4	4
Fuel Pool Vent Exh. (DIV I) (D11-K609C/D11-R606)	4E-2 mr/hr	2.5E-2	4E-2
Fuel Pool Vent Exh. (DIV II) (D11-K609D/D11-R606)	6E-2 mr/hr	6E-2	4E-2
Pri. Cont. Rad.(gamma) (T50-K003/T50-R809)	40 cpm	45	90
Fuel Pool Vent Exh. (DIV I) (D11-K609A/D11-R605)	NA	NA	4E-2
Fuel Pool Vent Exh. (DIV II) (D11-K609B/D11-R605)	NA	NA	4E-2

NOTE 1: DOS indicates that the monitor is "Out of Service".
NOTE 2: NA = not available or applicable at this time.

FIGURE 3.2-2

Process Radiation Monitor Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

C.C. Emerg. Air (South), Div I (D11-K836A)	NA	NA	NA
C.C. Emerg. Air (North), Div I (D11-K837A)	NA	NA	NA
Cont. Area Hi-Range Monitor, Div I (D11-K816A)	NA	NA	NA
C.C. Emerg. Air (South), Div II (D11-K836B)	NA	NA	NA
C.C. Emerg. Air (North), Div II (D11-K837B)	NA	NA	NA
Cont. Area Hi-Range Monitor, Div II (D11-K816B)	NA	NA	NA
Reactor Bldg. Vent Plenum Exh. (D11-K610)	NA	NA	NA
Radwaste Bldg. Vent Exh. (D11-K610)	NA	NA	NA
SGTS Vent Exh. Div I (D11-K610)	NA	NA	NA
SGTS Vent Exh. Div II (D11-K610)	NA	NA	NA
OSB Machine Shop Vent Exh. (D11-K610)	NA	NA	NA
Turbine Bldg. Vent Exh. (D11-K610)	NA	NA	NA
On-Site Storage Bldg. Vent Exh. (D11-K610)	NA	NA	NA

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

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: General Site			
1	<0.2 mr/hr	<0.2	<0.2
2	<0.2 mr/hr	<0.2	<0.2
3	<0.2 mr/hr	<0.2	<0.2
4	<0.2 mr/hr	<0.2	.2
5	<0.2 mr/hr	<0.2	<0.2
6	<0.2 mr/hr	<0.2	<0.2
7	<0.2 mr/hr	<0.2	<0.2
8	<0.2 mr/hr	<0.2	<0.2
10	<0.2 mr/hr	<0.2	<0.2
11	<0.2 mr/hr	<0.2	<0.2
12	<0.2 mr/hr	<0.2	<0.2
13	<0.2 mr/hr	<0.2	<0.2
14	<0.2 mr/hr	<0.2	<0.2
15	<0.2 mr/hr	<0.2	<0.2
16	<0.2 mr/hr	<0.2	<0.2
17	<0.2 mr/hr	<0.2	<0.2
18	<0.2 mr/hr	<0.2	<0.2
19	<0.2 mr/hr	<0.2	<0.2

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

FIGURE 3.2-3

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: General Site

20	<0.2 mr/hr	<0.2	<0.2
21	<0.2 mr/hr	<0.2	<0.2
22	<0.2 mr/hr	<0.2	<0.2
23	<0.2 mr/hr	<0.2	<0.2
24	<0.2 mr/hr	<0.2	<0.2
25	<0.2 mr/hr	<0.2	<0.2
27	<0.2 mr/hr	<0.2	<0.2
28	<0.2 mr/hr	<0.2	<0.2
29	<0.2 mr/hr	<0.2	<0.2
30	<0.2 mr/hr	<0.2	<0.2

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

FIGURE 3.2-3

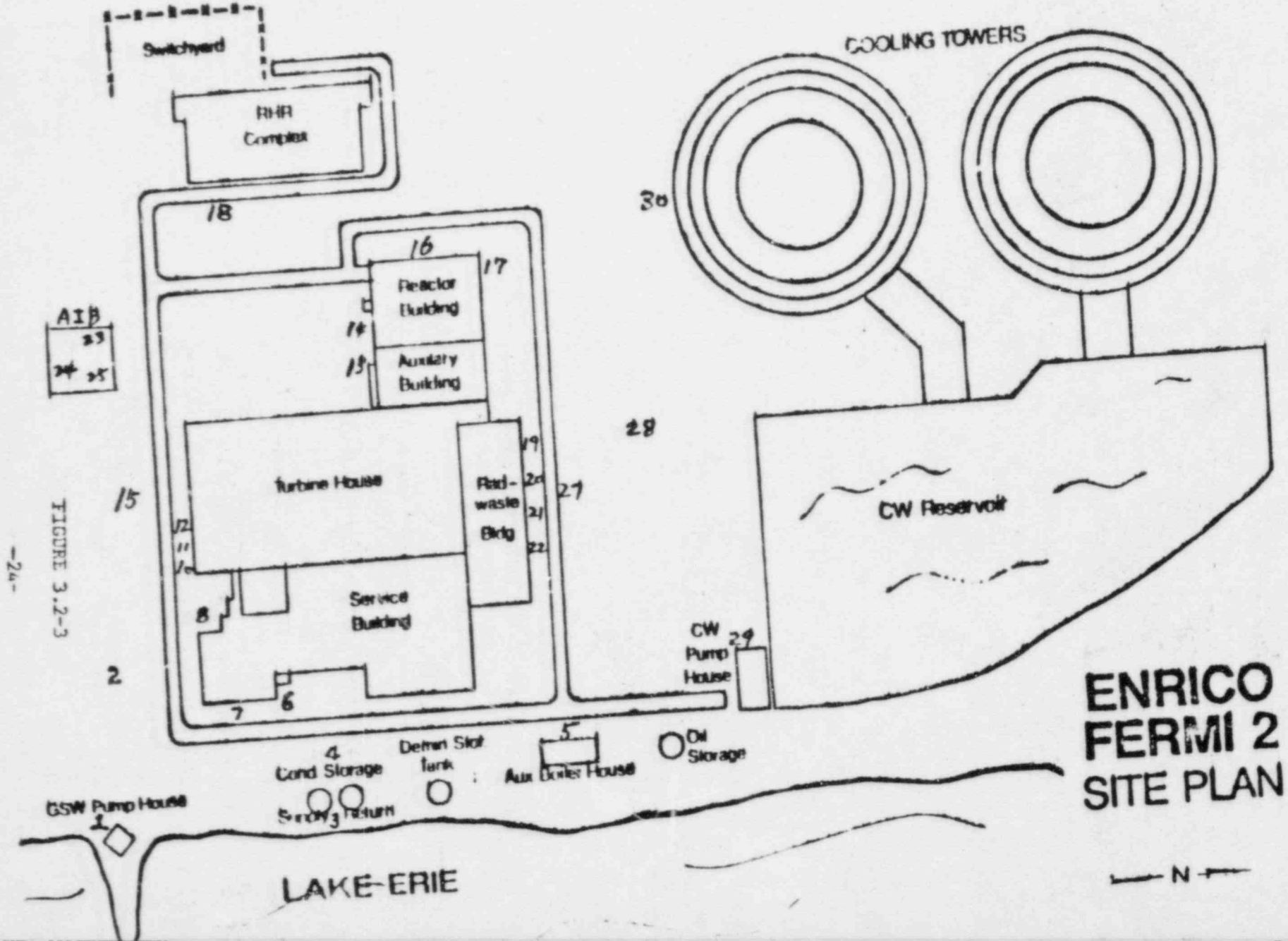


FIGURE 3.2-3

ENRICO FERMI 2
SITE PLAN

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: Reactor Building - Sub Basement

1	<0.2 mr/hr	<0.4	0.4
2	<0.2 mr/hr	<0.4	0.2
3	<0.2 mr/hr	<0.4	0.3
4	<0.2 mr/hr	<0.4	0.2
5	<0.2 mr/hr	<0.4	0.4
6	<0.2 mr/hr	<0.4	0.3
7	<0.2 mr/hr	<0.4	0.6
8	<0.2 mr/hr	<0.4	0.4
9	<0.2 mr/hr	<0.4	0.4
10	<0.2 mr/hr	<0.4	0.4

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

FIGURE 3.2-3

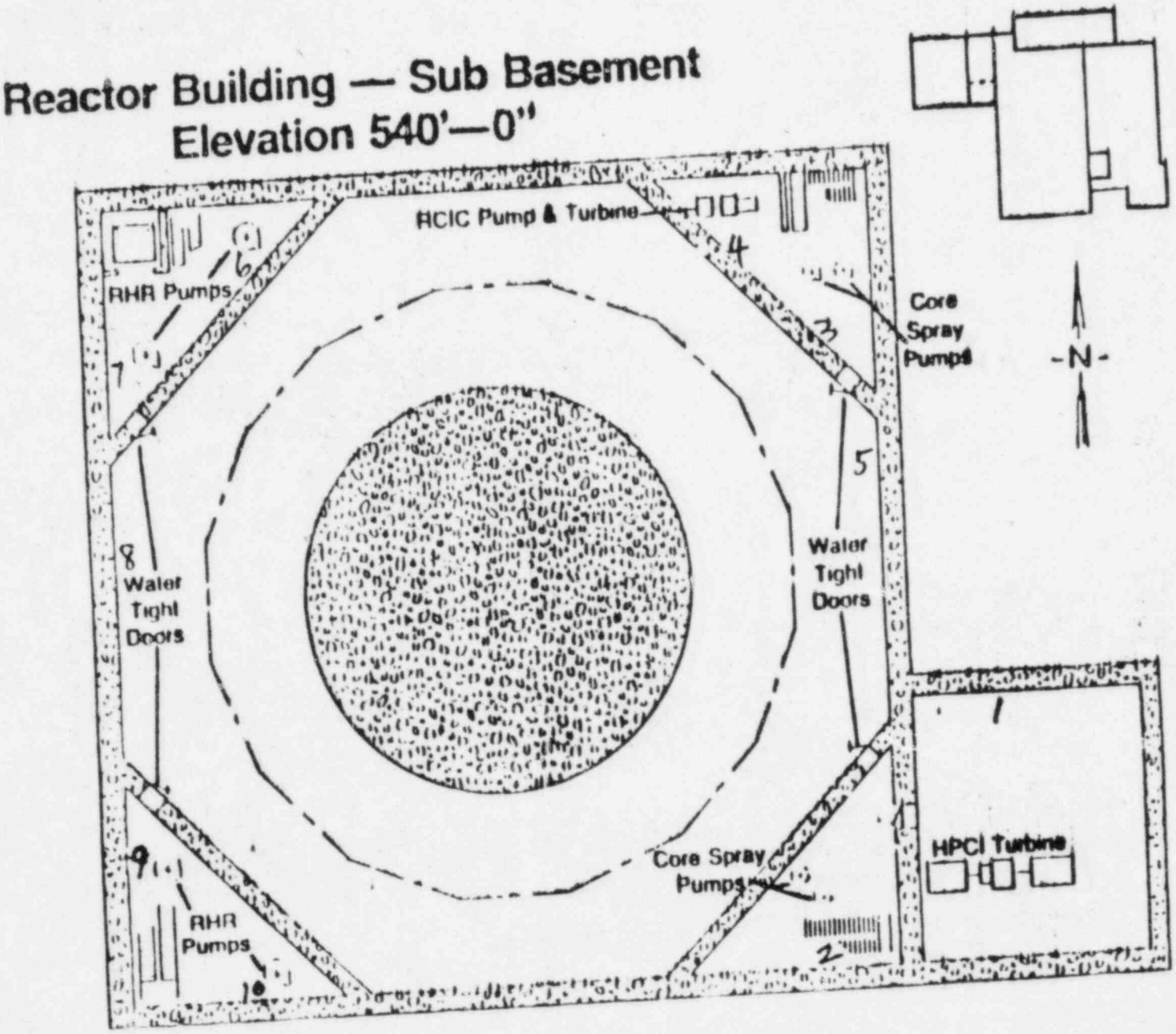

Reactor Building — Sub Basement
Elevation 540'—0"
RCIC Pump & Turbine
RHR Pumps
Core Spray Pumps
N
Water Tight Doors
Water Tight Doors
HPCI Turbine
Core Spray Pumps
RHR Pumps
1
2
3
4
5
6
7
8
9
10

FIGURE 3.2-3

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: Reactor Building - Basement

1	<0.2 mr/hr	<0.4	0.3
2	<0.2 mr/hr	<0.2	<0.2
4	<0.2 mr/hr	<0.2	<0.2
5	<0.2 mr/hr	<0.2	<0.2
6	<0.2 mr/hr	<0.4	0.2
7	<0.2 mr/hr	<0.4	0.2
8	<0.2 mr/hr	<0.4	0.2
9	<0.2 mr/hr	<0.4	0.2

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

FIGURE 3.2-3

Reactor Building — Basement
Elevation 562'—0"

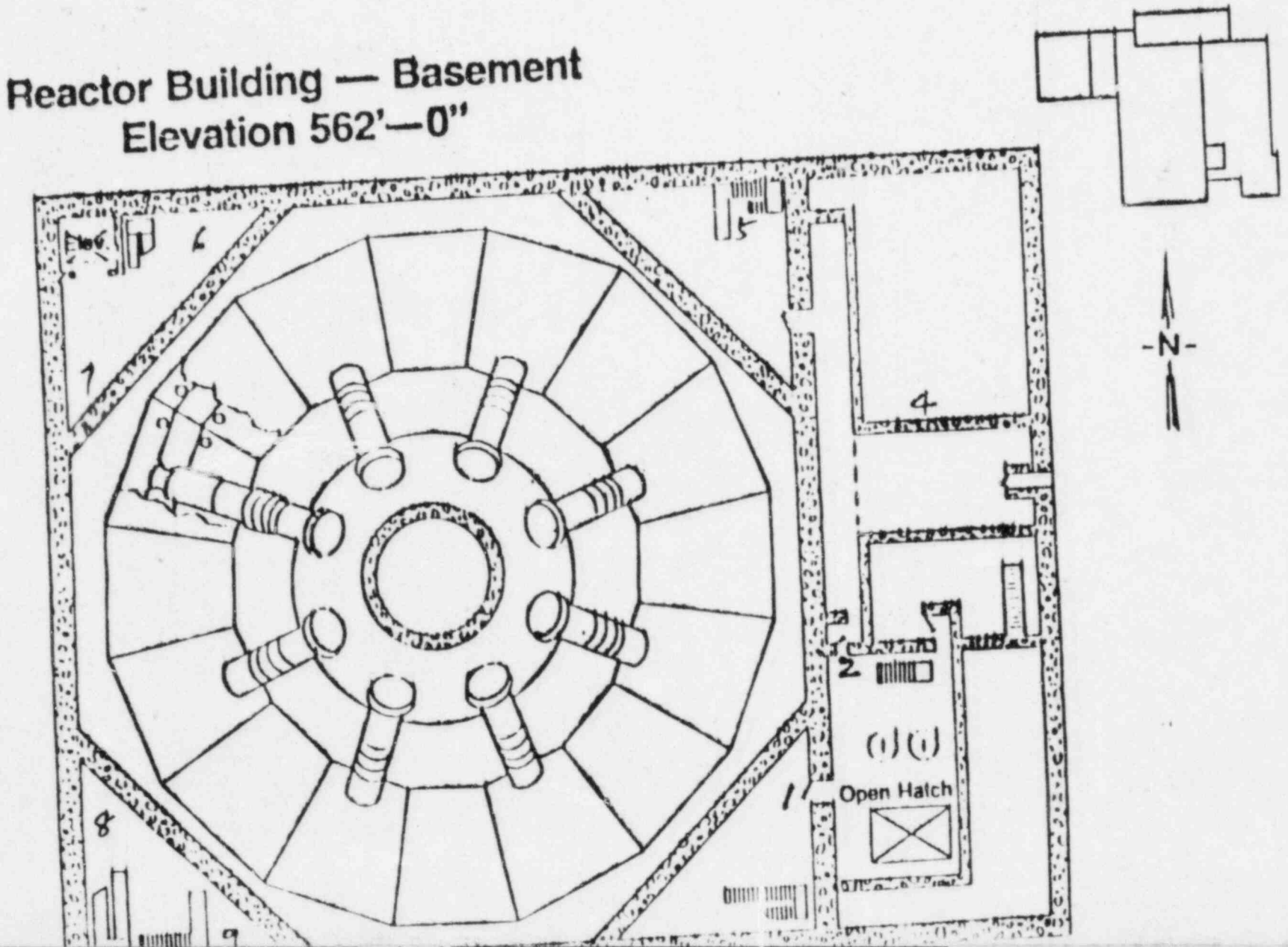


FIGURE 3.2-3

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: Reactor Building - First Floor

1	<0.2 mr/hr	<0.2	<0.2
2	<0.2 mr/hr	<0.2	<0.2
4	<0.2 mr/hr	<0.4	0.2
5	<0.2 mr/hr	<0.4	0.2
7	<0.2 mr/hr	<0.4	0.2
8	<0.2 mr/hr	<0.4	0.2
9	<0.2 mr/hr	<0.4	0.2
10	<0.2 mr/hr	<0.4	0.2
11	<0.2 mr/hr	<0.4	0.2
12	<0.2 mr/hr	<0.4	0.2
13	<0.2 mr/hr	<0.4	0.2
15	<0.2 mr/hr	<0.4	0.2
16	<0.2 mr/hr	<0.4	0.2
17	<0.2 mr/hr	<0.4	0.2
18	<0.2 mr/hr	<0.4	0.4
19	<0.2 mr/hr	<0.4	0.2
21	<0.2 mr/hr	<0.4	0.2
23	<0.2 mr/hr	<0.4	0.2

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

FIGURE 3.2-3

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/29/85	4/16/85	7/13/85

Location: Reactor Building - First Floor

25	<0.2 mr/hr	<0.2	<0.2
26	<0.2 mr/hr	<0.4	0.4
27	<0.2 mr/hr	<0.4	0.2
28	<0.2 mr/hr	<0.4	0.4
29	<0.2 mr/hr	<0.4	0.2
31	<0.2 mr/hr	<0.4	0.2
32	<0.2 mr/hr	<0.4	0.4
33	<0.2 mr/hr	<0.4	0.4
34	<0.2 mr/hr	<0.4	0.2
35	<0.2 mr/hr	<0.4	0.2
37	<0.2 mr/hr	<0.2	<0.2
38	<0.2 mr/hr	<0.2	<0.2
39	<0.2 mr/hr	<0.2	<0.2
40	<0.2 mr/hr	<0.2	<0.2

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

FIGURE 3.2-3

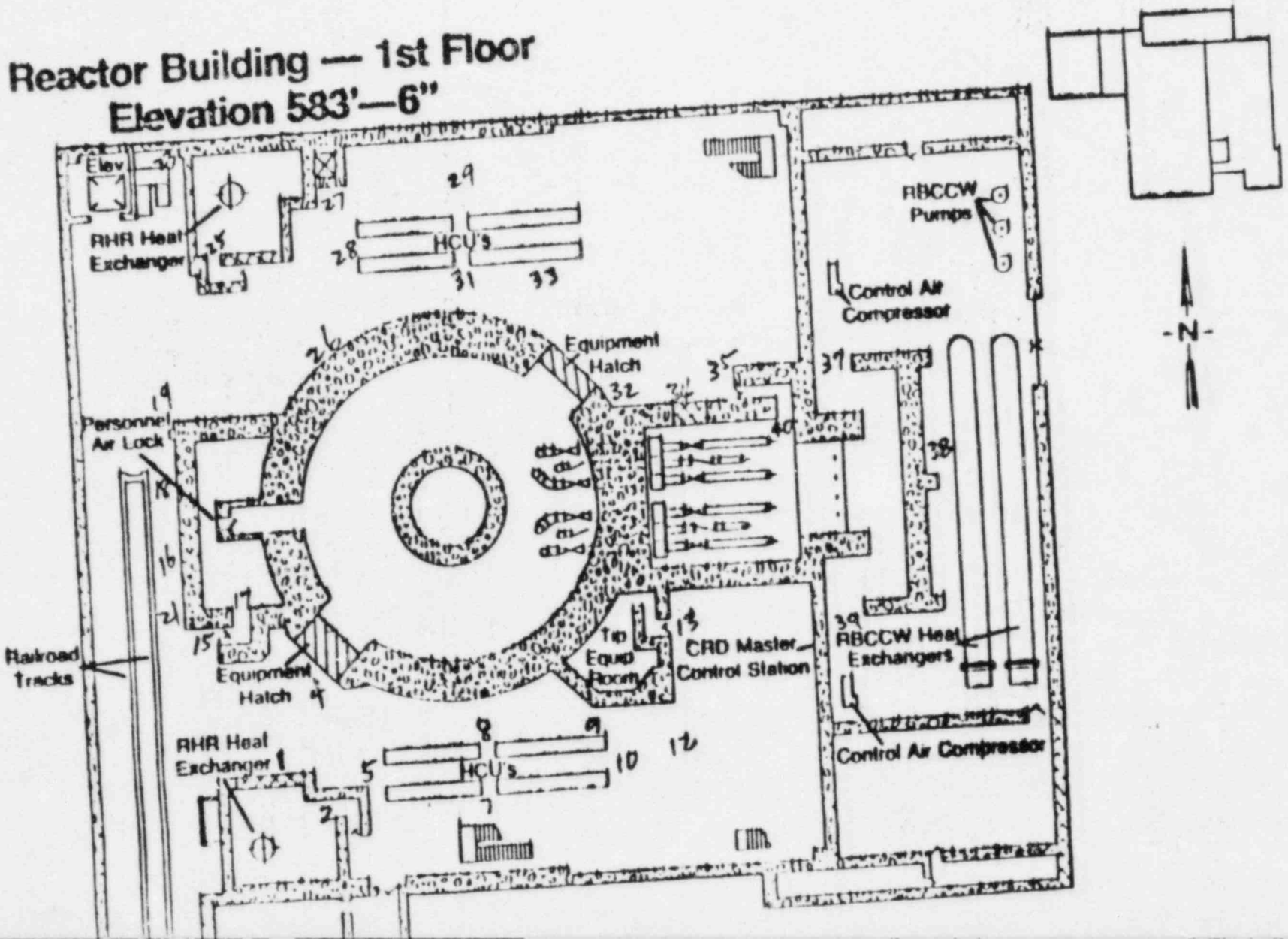

Reactor Building — 1st Floor
Elevation 583'—6"
Elev
RHR Heat Exchanger
HCU's
RBCCW Pumps
Control Air Compressor
Equipment Hatch
Personnel Air Lock
Railroad Tracks
Equipment Hatch
Top Equip Room
CRD Master Control Station
RBCCW Heat Exchangers
Control Air Compressor
RHR Heat Exchanger
HCU's
N

FIGURE 3.2-3

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: Reactor Building - Second Floor

1	<0.2 mr/hr	<0.4	0.4
2	<0.2 mr/hr	<0.4	0.4
3	<0.2 mr/hr	<0.4	0.6
4	<0.2 mr/hr	<0.4	0.2
5	<0.2 mr/hr	<0.2	<0.2
6	<0.2 mr/hr	<0.4	0.6
7	<0.2 mr/hr	<0.4	0.6
8	<0.2 mr/hr	<0.4	0.6
10	<0.2 mr/hr	<0.4	0.8
11	<0.2 mr/hr	<0.4	0.6
12	<0.2 mr/hr	<0.4	0.4
13	<0.2 mr/hr	<0.4	0.5
14	<0.2 mr/hr	<0.4	1.4
15	<0.2 mr/hr	<0.4	0.8
16	<0.2 mr/hr	<0.4	0.2
18	<0.2 mr/hr	<0.2	<0.2
19	<0.2 mr/hr	<0.4	0.3
20	<0.2 mr/hr	<0.4	0.3

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

FIGURE 3.2-3

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: Reactor Building - Second Floor

21	<0.2 mr/hr	<0.4	0.4
22	<0.2 mr/hr	<0.4	0.2
23	<0.2 mr/hr	<0.4	0.4
24	<0.2 mr/hr	<0.4	0.3
25	<0.2 mr/hr	<0.4	0.2
26	<0.2 mr/hr	<0.4	0.2
28	<0.2 mr/hr	<0.4	<0.2
29	<0.2 mr/hr	<0.2	<0.2
30	<0.2 mr/hr	<0.2	2.5
31	<0.2 mr/hr	<0.2	<0.2
32	<0.2 mr/hr	<0.2	<0.2
33	<0.2 mr/hr	<0.2	<0.2

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

FIGURE 3.2-3

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

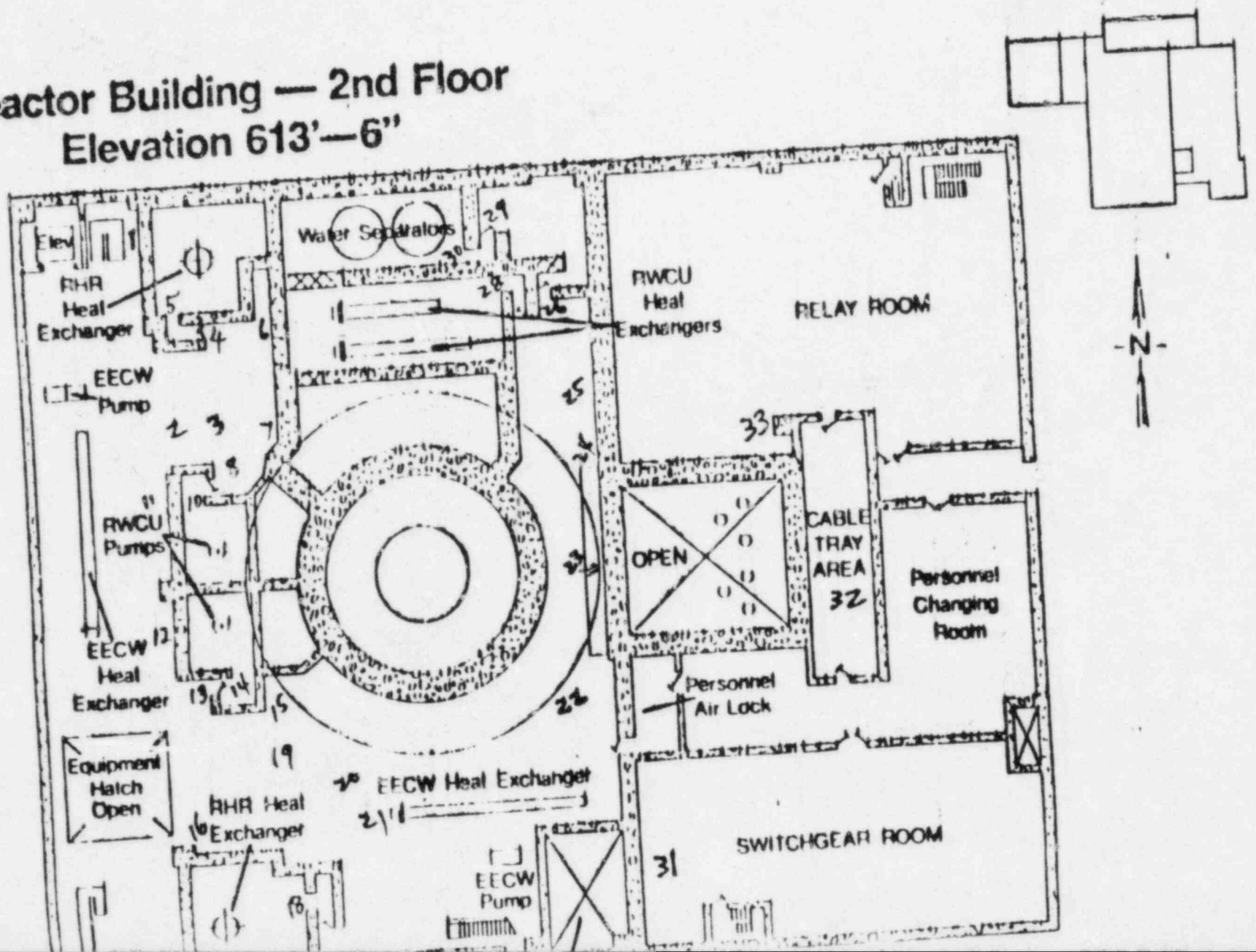


FIGURE 3.2-3

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: Reactor Building - Third Floor

1	<0.2 mr/hr	<0.4	0.3
2	<0.2 mr/hr	<0.4	0.3
3	<0.2 mr/hr	<0.4	0.3
4	<0.2 mr/hr	<0.4	0.2
5	<0.2 mr/hr	<0.4	0.3
6	<0.2 mr/hr	<0.2	<0.2
7	<0.2 mr/hr	<0.4	0.4
9	<0.2 mr/hr	<0.4	0.4
10	<0.2 mr/hr	<0.4	0.4
11	<0.2 mr/hr	<0.4	0.4
12	<0.2 mr/hr	<0.2	<0.2
13	<0.2 mr/hr	<0.2	<0.2
14	<0.2 mr/hr	<0.2	<0.2
15	<0.2 mr/hr	<0.2	<0.2

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

FIGURE 3.2-3

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

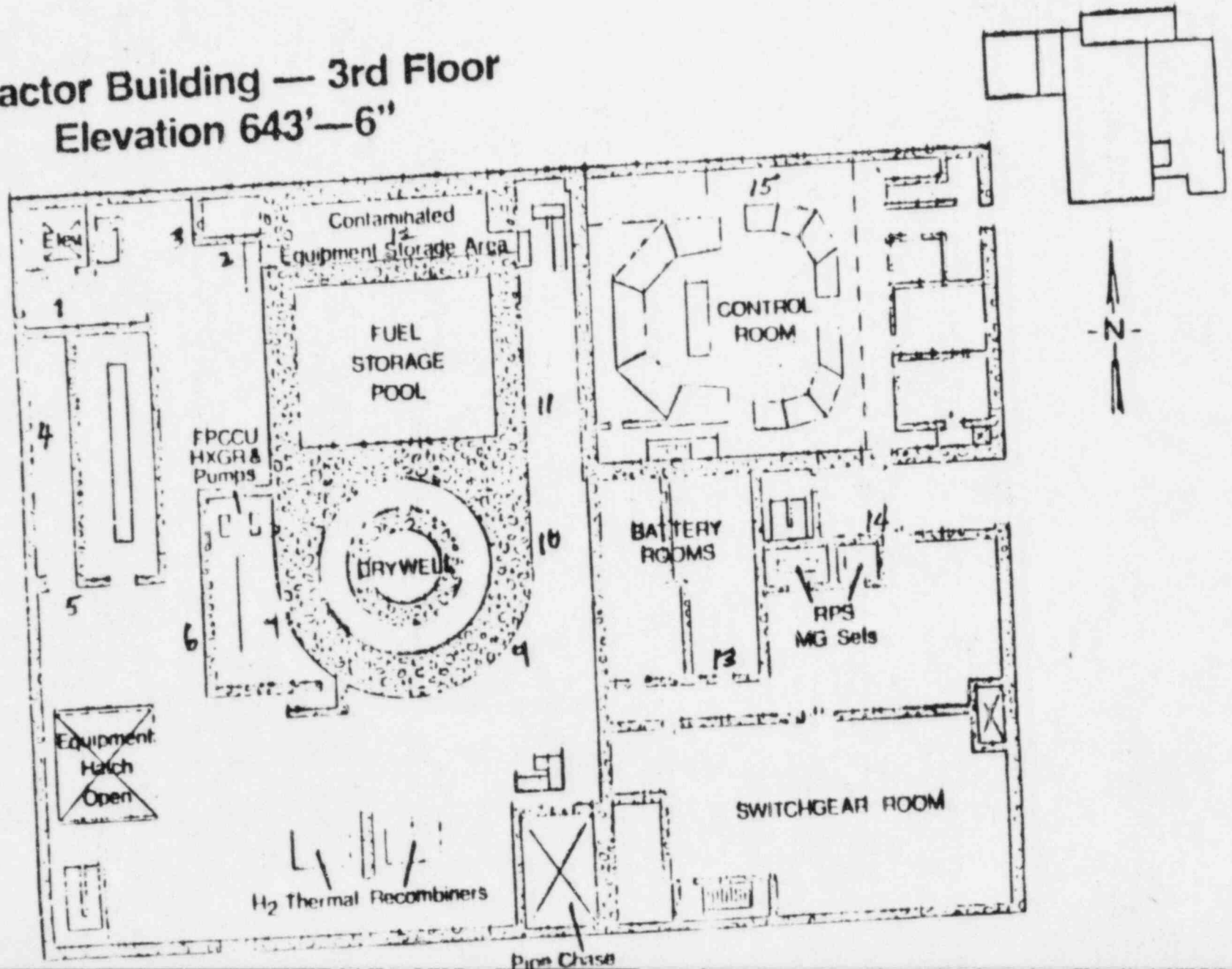


FIGURE 3.2-3

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: Reactor Building - Fourth Floor

2	<0.2 mr/hr	<0.2	<0.2
4	<0.2 mr/hr	<0.4	0.3
5	<0.2 mr/hr	<0.4	0.3
6	<0.2 mr/hr	<0.4	0.4
7	<0.2 mr/hr	<0.4	0.3
8	<0.2 mr/hr	<0.4	0.5
9	<0.2 mr/hr	<0.4	0.3
10	<0.2 mr/hr	<0.4	0.3
11	<0.2 mr/hr	<0.4	0.2
12	<0.2 mr/hr	<0.4	0.4
13	<0.2 mr/hr	<0.4	0.2

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

FIGURE 3.2-3

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

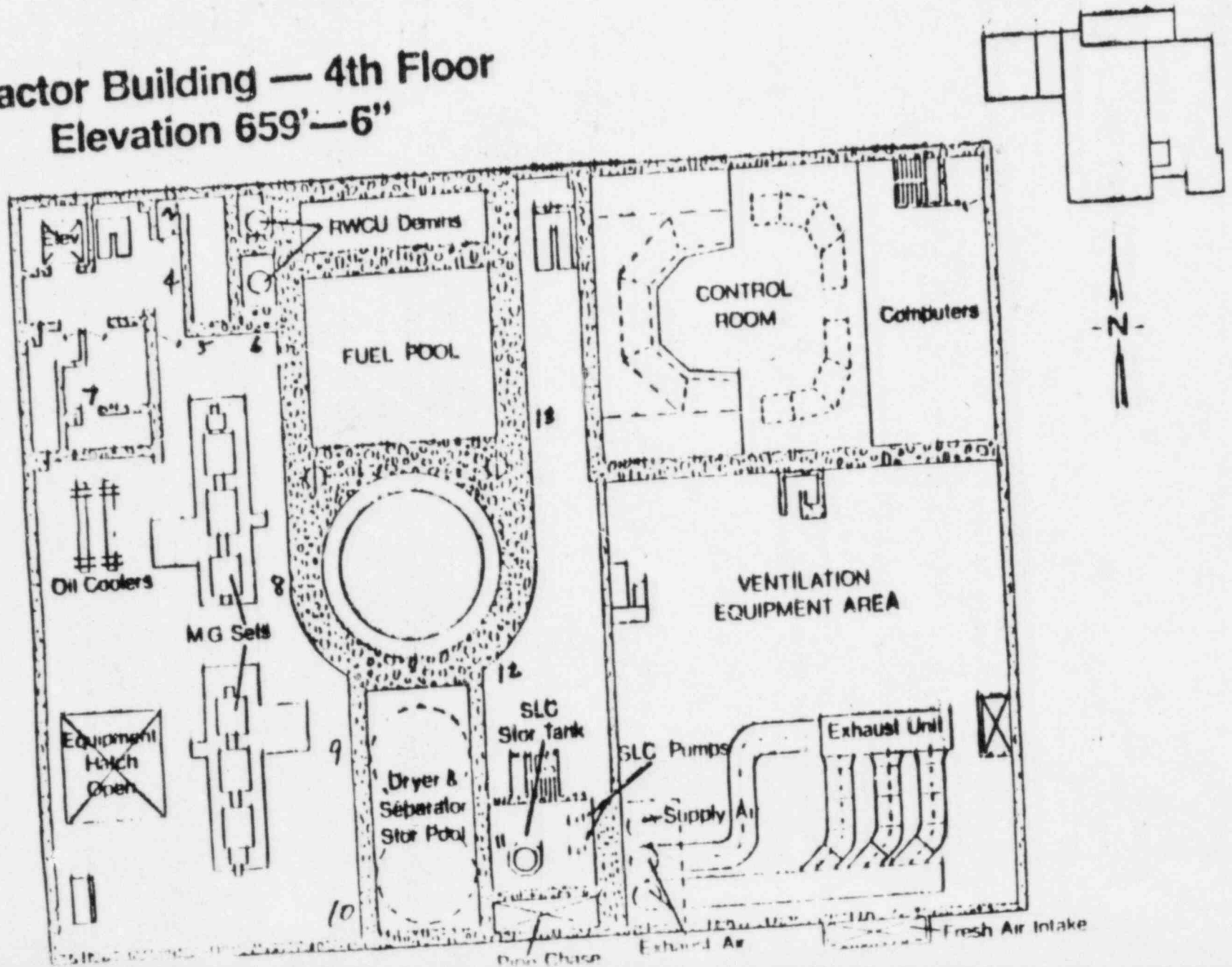


FIGURE 3.2-3

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod. Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: Reactor Building - Fifth Floor

1	<0.2 mr/hr	<0.4	0.2
2	<0.2 mr/hr	<0.4	0.2
3	<0.2 mr/hr	<0.4	0.2
4	<0.2 mr/hr	<0.4	0.3
5	<0.2 mr/hr	<0.4	0.3
6	<0.2 mr/hr	<0.4	0.3
7	<0.2 mr/hr	<0.4	0.2
8	<0.2 mr/hr	<0.4	0.3
9	<0.2 mr/hr	<0.2	<0.2
10	<0.2 mr/hr	<0.2	<0.2
11	<0.2 mr/hr	<0.2	<0.2
12	<0.2 mr/hr	<0.2	<0.2
13	<0.2 mr/hr	<0.2	<0.2
14	<0.2 mr/hr	<0.4	0.3

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

FIGURE 3.2-3

Reactor Building 5th Floor
Elevation 677'—6" and 684'—8"

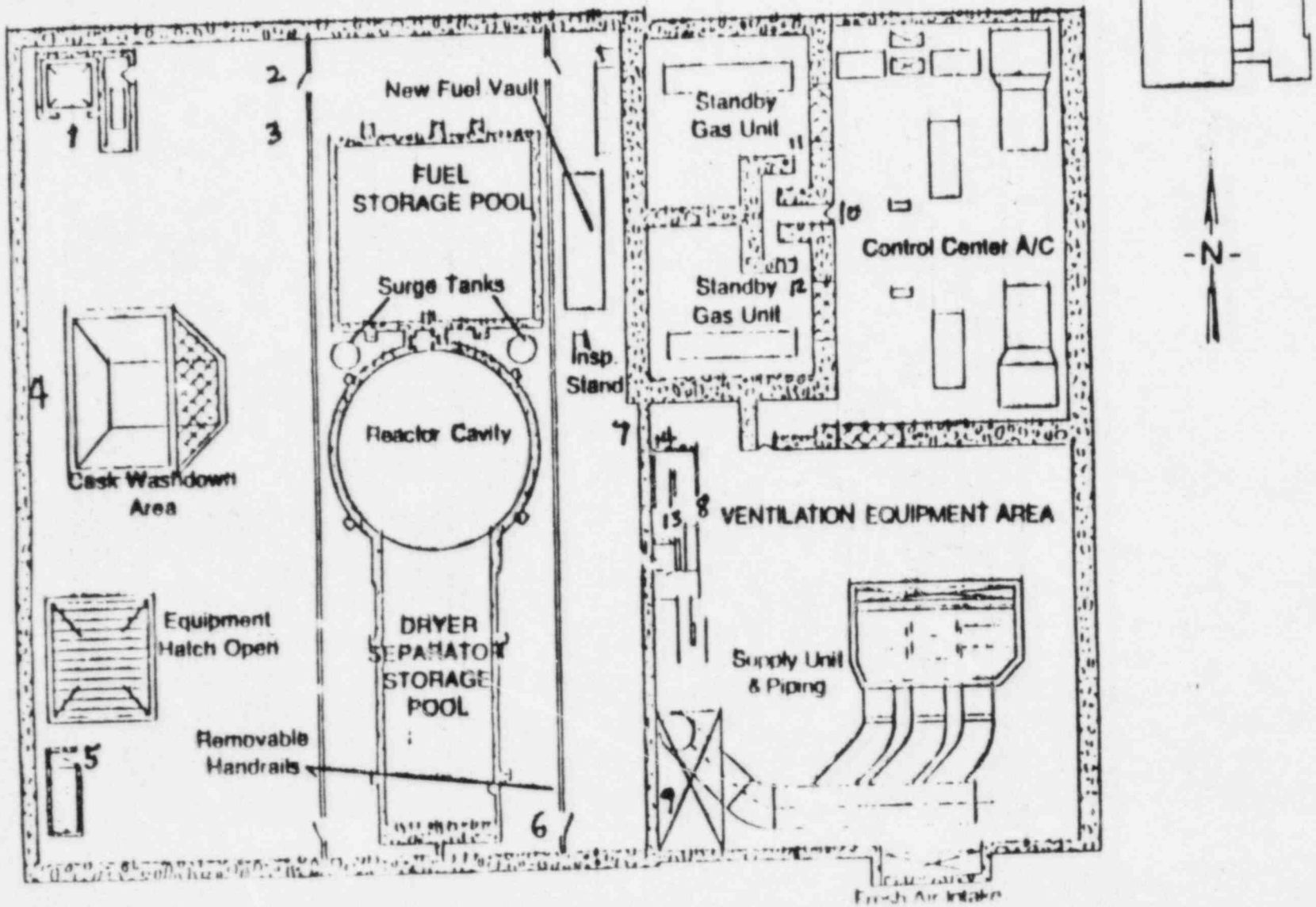


FIGURE 3.2-3

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: Turbine Building - Basement

1	<0.2 mr/hr	<0.2	<0.2
2	<0.2 mr/hr	<0.2	<0.2
3	<0.2 mr/hr	<0.2	<0.2
4	<0.2 mr/hr	<0.2	<0.2
5	<0.2 mr/hr	<0.2	<0.2
8	<0.2 mr/hr	<0.2	<0.2
9	<0.2 mr/hr	<0.2	<0.2
10	<0.2 mr/hr	<0.2	<0.2
11	<0.2 mr/hr	<0.2	<0.2
12	<0.2 mr/hr	<0.2	<0.2
13	<0.2 mr/hr	<0.2	<0.2
14	<0.2 mr/hr	<0.2	<0.2
15	<0.2 mr/hr	<0.2	<0.2
16	<0.2 mr/hr	<0.2	<0.2
17	<0.2 mr/hr	<0.2	<0.2

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

FIGURE 3.2-3

Turbine Building — Basement
Elevation 564'—0"

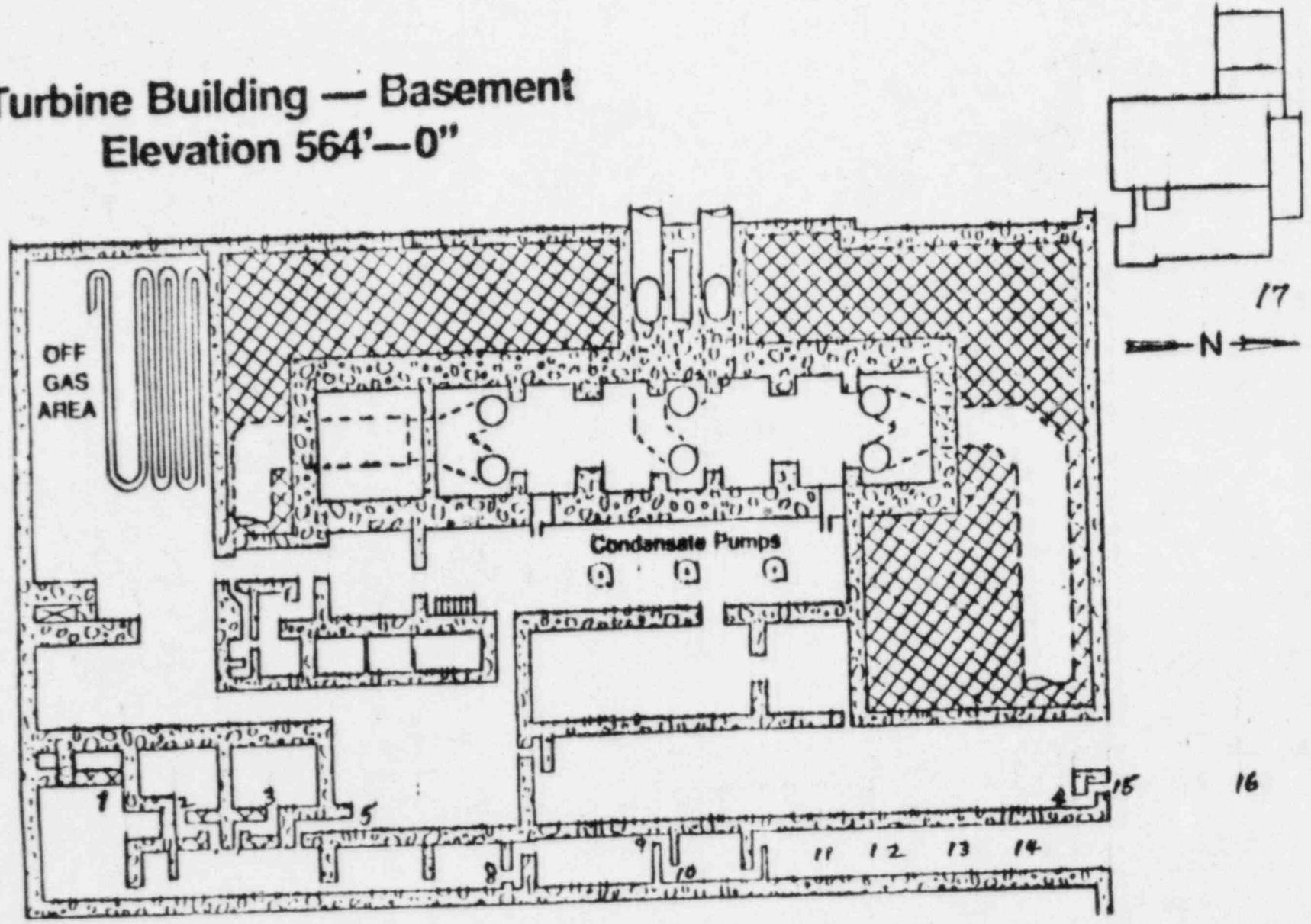


FIGURE 3.2-3

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: Turbine Building - First Floor

1	<0.2 mr/hr	<0.2	<0.2
2	<0.2 mr/hr	<0.2	<0.2
3	<0.2 mr/hr	<0.2	<0.2
4	<0.2 mr/hr	<0.2	<0.2
5	<0.2 mr/hr	<0.2	<0.2
6	<0.2 mr/hr	<0.2	<0.2
7	<0.2 mr/hr	<0.2	<0.2
8	<0.2 mr/hr	<0.2	<0.2
9	<0.2 mr/hr	<0.2	<0.2
10	<0.2 mr/hr	<0.2	<0.2
11	<0.2 mr/hr	<0.2	<0.2
12	<0.2 mr/hr	<0.2	<0.2
13	<0.2 mr/hr	<0.2	<0.2
14	<0.2 mr/hr	<0.2	<0.2
15	<0.2 mr/hr	<0.2	<0.2
16	<0.2 mr/hr	<0.2	<0.2
17	<0.2 mr/hr	<0.2	<0.2
18	<0.2 mr/hr	<0.2	<0.2

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

FIGURE 3.2-3

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: Turbine Building - First Floor

19	<0.2 mr/hr	<0.2	<0.2
20	<0.2 mr/hr	<0.2	<0.2
21	<0.2 mr/hr	<0.2	<0.2
22	<0.2 mr/hr	<0.2	<0.2
23	<0.2 mr/hr	<0.2	<0.2
24	<0.2 mr/hr	<0.2	<0.2
25	<0.2 mr/hr	<0.2	<0.2
26	<0.2 mr/hr	<0.2	<0.2
27	<0.2 mr/hr	<0.2	<0.2
28	<0.2 mr/hr	<0.2	<0.2
29	<0.2 mr/hr	<0.2	<0.2
30	<0.2 mr/hr	<0.2	<0.2
31	<0.2 mr/hr	<0.2	<0.2
32	<0.2 mr/hr	<0.2	<0.2
33	<0.2 mr/hr	<0.2	<0.2
34	<0.2 mr/hr	<0.2	<0.2
35	<0.2 mr/hr	<0.2	<0.2
36	<0.2 mr/hr	<0.2	<0.2

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

FIGURE 3.2-3

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: Turbine Building - First Floor

37	<0.2 mr/hr	<0.2	0.2
38	<0.2 mr/hr	<0.2	0.2
39	<0.2 mr/hr	<0.2	0.2
40	<0.2 mr/hr	<0.2	0.2
41	<0.2 mr/hr	<0.2	0.2
42	<0.2 mr/hr	<0.2	0.2
43	<0.2 mr/hr	<0.2	0.2
44	<0.2 mr/hr	<0.2	0.2

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

FIGURE 3.2-3

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

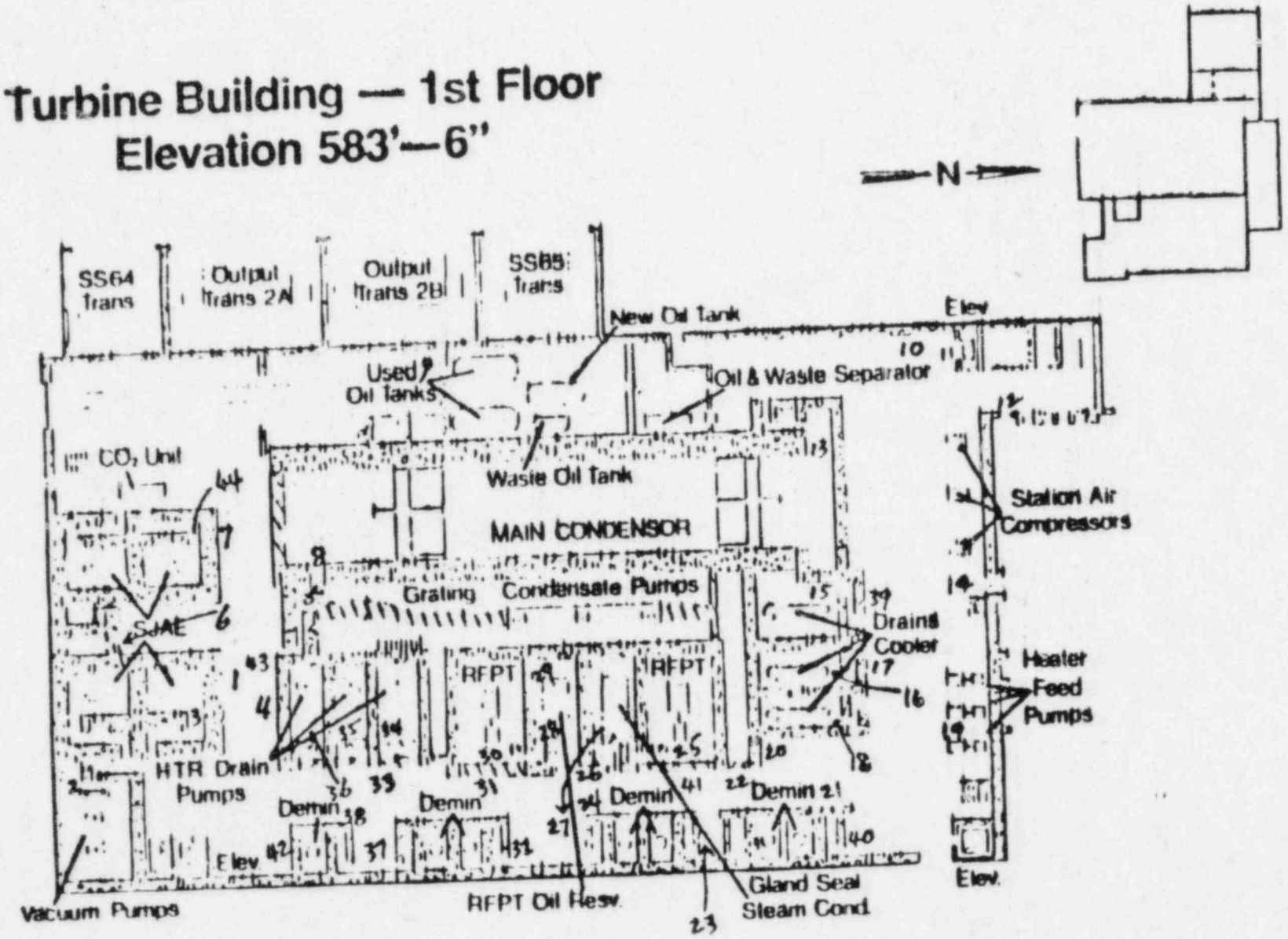


FIGURE 3.2-3

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: Turbine Building - Second Floor

1	<0.2 mr/hr	<0.2	<0.2
2	<0.2 mr/hr	<0.2	<0.2
3	<0.2 mr/hr	<0.2	<0.2
4	<0.2 mr/hr	<0.2	<0.2
5	<0.2 mr/hr	<0.2	<0.2
6	<0.2 mr/hr	<0.2	<0.2
7	<0.2 mr/hr	<0.2	<0.2
8	<0.2 mr/hr	<0.2	<0.2
9	<0.2 mr/hr	<0.2	<0.2
10	<0.2 mr/hr	<0.2	<0.2
11	<0.2 mr/hr	<0.2	<0.2
12	<0.2 mr/hr	<0.2	<0.2
13	<0.2 mr/hr	<0.2	<0.2
14	<0.2 mr/hr	<0.2	<0.2
15	0.2 mr/hr	<0.2	<0.2
16	<0.2 mr/hr	<0.2	<0.2
17	<0.2 mr/hr	<0.2	<0.2
18	<0.2 mr/hr	<0.2	<0.2

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

FIGURE 3.2-3

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: Turbine Building - Second Floor

19	<0.2 mr/hr	<0.2	<0.2
20	<0.2 mr/hr	<0.2	<0.2
21	<0.2 mr/hr	<0.2	<0.2
22	<0.2 mr/hr	<0.2	<0.2
23	<0.2 mr/hr	<0.2	<0.2
24	<0.2 mr/hr	<0.2	<0.2
25	<0.2 mr/hr	<0.2	<0.2
26	<0.2 mr/hr	<0.2	<0.2
27	<0.2 mr/hr	<0.2	<0.2
28	<0.2 mr/hr	<0.2	<0.2
29	<0.2 mr/hr	<0.2	<0.2
30	<0.2 mr/hr	<0.2	<0.2
31	<0.2 mr/hr	<0.2	<0.2
32	<0.2 mr/hr	<0.2	<0.2

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

FIGURE 3.2-3

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

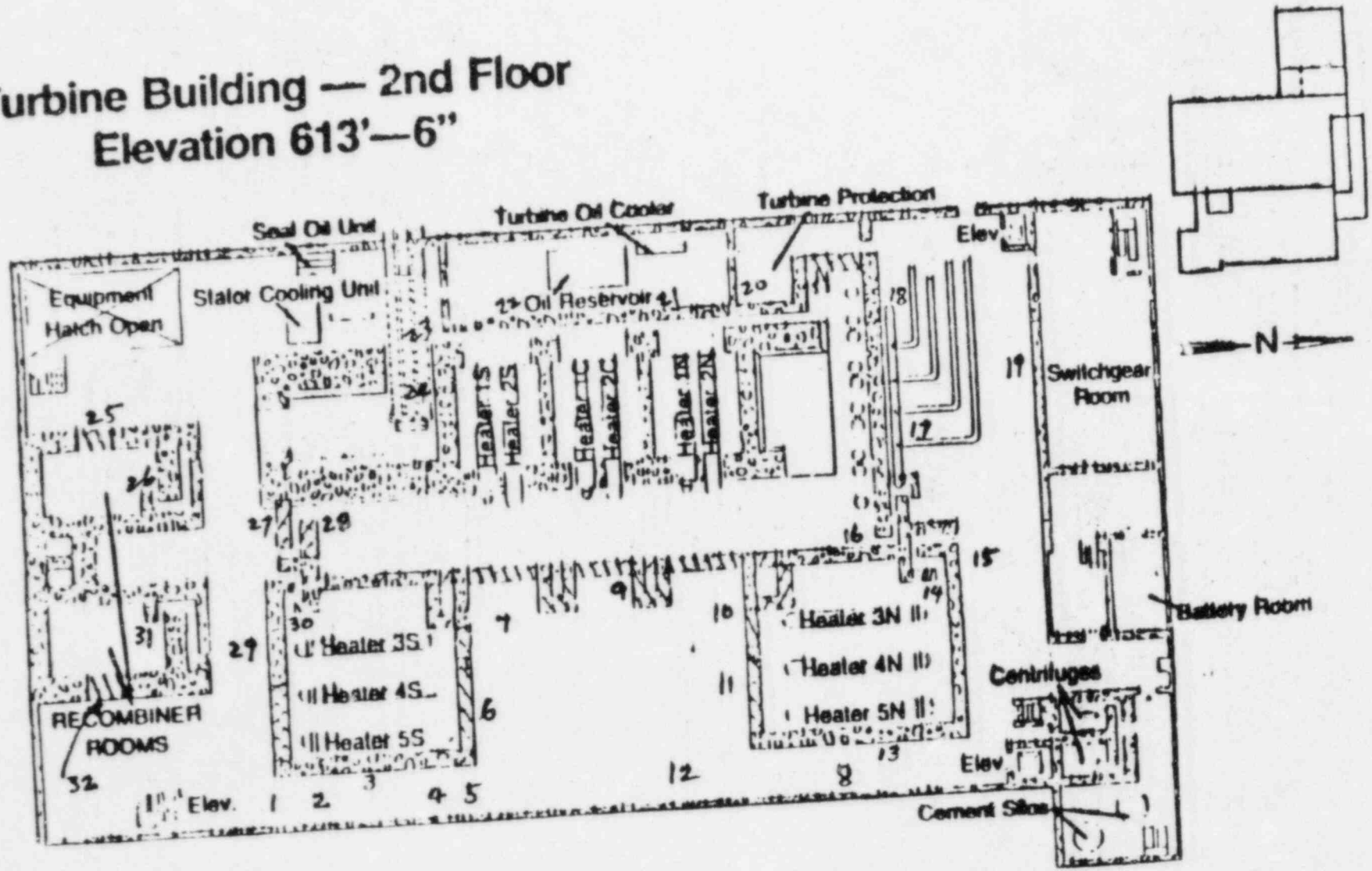


FIGURE 3.2-3

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: Turbine Building - Third Floor

1	<0.2 mr/hr	<0.2	<0.2
2	<0.2 mr/hr	<0.2	<0.2
3	<0.2 mr/hr	<0.2	<0.2
4	<0.2 mr/hr	<0.2	<0.2
5	<0.2 mr/hr	<0.2	<0.2
6	<0.2 mr/hr	<0.2	<0.2
7	<0.2 mr/hr	<0.2	<0.2
8	<0.2 mr/hr	<0.2	<0.2
9	<0.2 mr/hr	<0.2	<0.2
10	<0.2 mr/hr	<0.2	<0.2
11	<0.2 mr/hr	<0.2	<0.2
12	<0.2 mr/hr	<0.2	<0.2
13	<0.2 mr/hr	<0.2	<0.2
14	<0.2 mr/hr	<0.2	<0.2
15	<0.2 mr/hr	<0.2	<0.2
16	<0.2 mr/hr	<0.2	<0.2
17	<0.2 mr/hr	<0.2	<0.2
18	<0.2 mr/hr	<0.2	<0.2

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

FIGURE 3.2-3

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: Turbine Building - Third Floor

19	<0.2 mr/hr	<0.2	<0.2
20	<0.2 mr/hr	<0.2	<0.2
21	<0.2 mr/hr	<0.2	<0.2
22	<0.2 mr/hr	<0.2	<0.2
23	<0.2 mr/hr	<0.2	<0.2
24	<0.2 mr/hr	<0.2	<0.2
25	<0.2 mr/hr	<0.2	<0.2
26	<0.2 mr/hr	<0.2	<0.2
27	<0.2 mr/hr	<0.2	<0.2
28	<0.2 mr/hr	<0.2	<0.2
29	<0.2 mr/hr	<0.2	<0.2
30	<0.2 mr/hr	<0.2	<0.2
31	<0.2 mr/hr	<0.2	<0.2
32	<0.2 mr/hr	<0.2	<0.2
33	<0.2 mr/hr	<0.2	<0.2
34	<0.2 mr/hr	<0.2	<0.2
35	<0.2 mr/hr	<0.2	<0.2
36	<0.2 mr/hr	<0.2	<0.2

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

FIGURE 3.2-3

Survey Data

Test Condition	Pre-fuel load	Open Vessel	Heatup
CTP	0	0	<5%
MWe	0	0	0
Mod.Temp	<100°F	<100°F	540°F
Date	1/19/85	4/16/85	7/13/85

Location: Turbine Building - Third Floor

37	<0.2 mr/hr	<0.2	<0.2
38	<0.2 mr/hr	<0.2	<0.2
39	<0.2 mr/hr	<0.2	<0.2
40	<0.2 mr/hr	<0.2	<0.2
41	<0.2 mr/hr	<0.2	<0.2
42	<0.2 mr/hr	<0.2	<0.2
43	<0.2 mr/hr	<0.2	<0.2

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

FIGURE 3.2-3

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

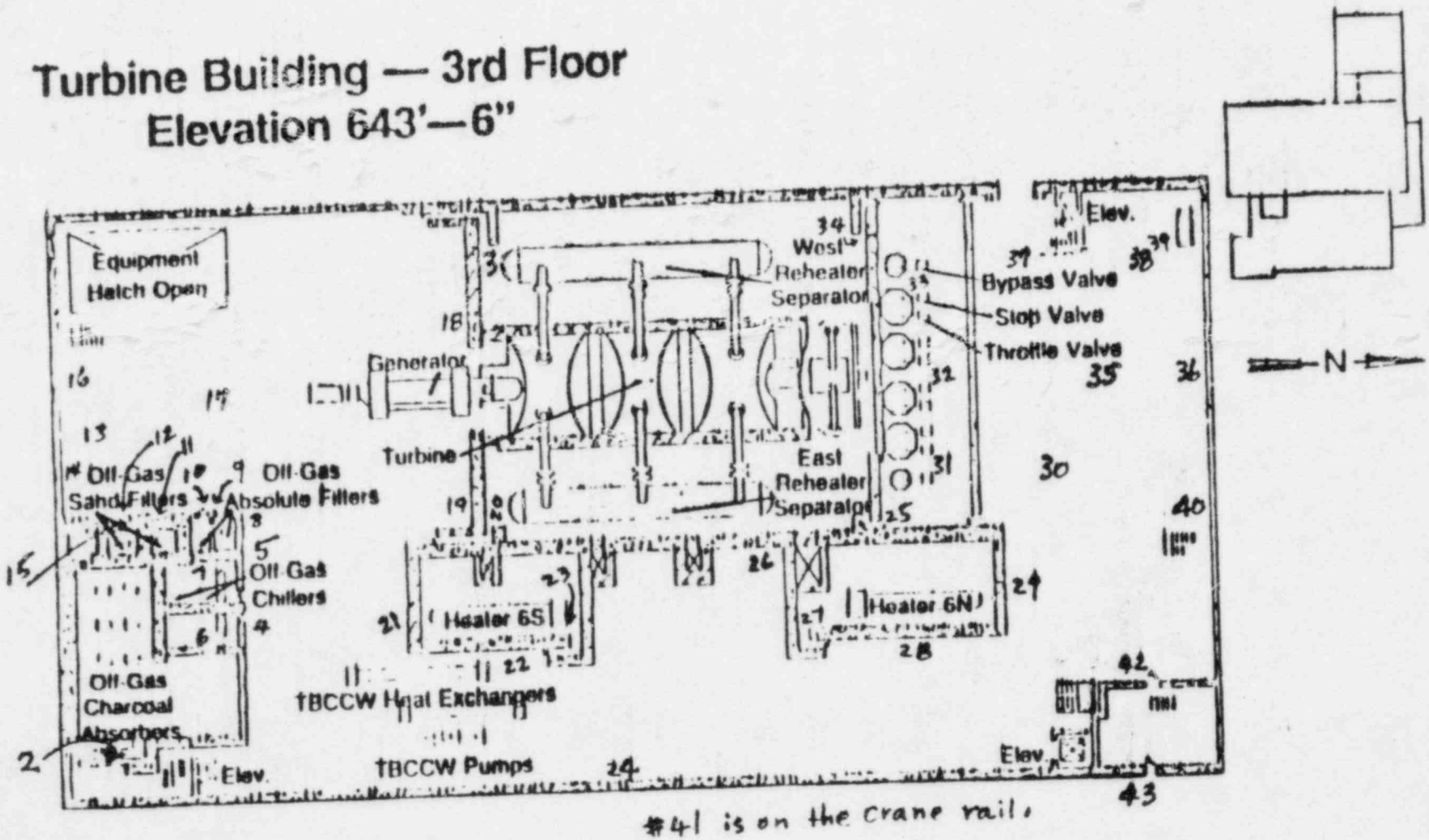


FIGURE 3.2-3

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.

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 an FLC. These geometric effects resulted in erronious (but highly conservative) projections which often resulted in very small increment sizes (1 - 2 bundles) between 1/M calculations. After eighty bundles were loaded in the core, the maximum increment size between 1/M calculations was reduced to one cell (4 bundles except for the peripheral locations where a maximum of five bundles were loaded between 1/M calculations).

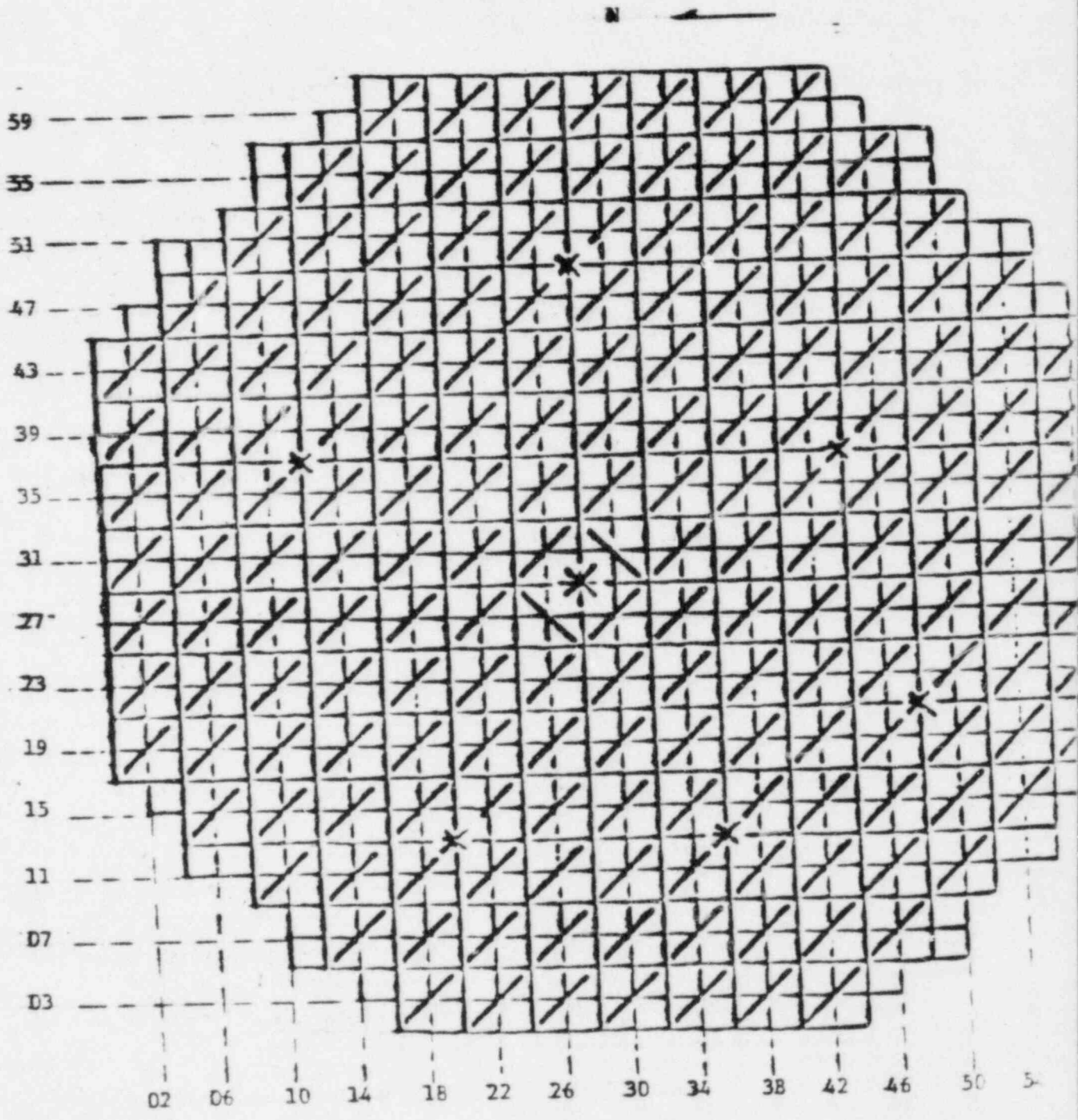
Bundle LJK 677 was identified to have a rusted channel fastener that had to be replaced. Some debris was identified in the core on bundles LKJ 398, LJK 506 and LJK 957. After fuel loading was completed, these bundles were pulled out of the core to correct the respective problems and reinserted back into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

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

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

NEUTRON SOURCE LOCATION AND BLADE GUIDE ORIENTATION

PRIOR TO FUEL LOADING



* SOURCE (7) /BLADE GUIDE (185)

Figure 3.3-1

Signal to Noise Measurement

DATE (TIME)	DETECTOR	A		B		C		D		# OF BUNDLES LOADED
		CPS	S/N	CPS	S/N	CPS	S/N	CPS	S/N	
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	150	12	119	40	399	15	149	532

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

Figure 3.3-2

CORE LOADING SEQUENCE

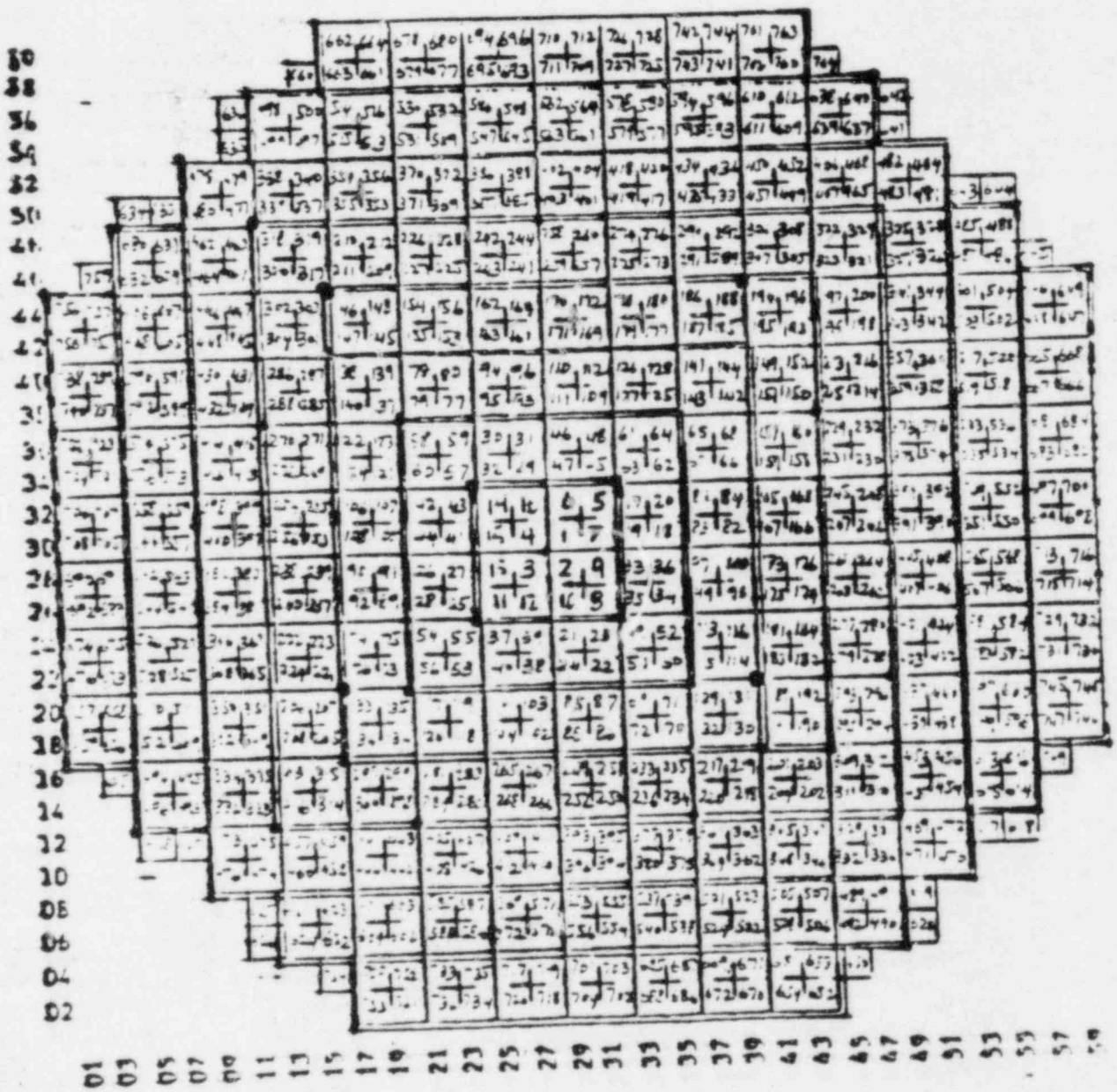


Figure 3.3-3

FLC MOVES DURING CORE LOADING

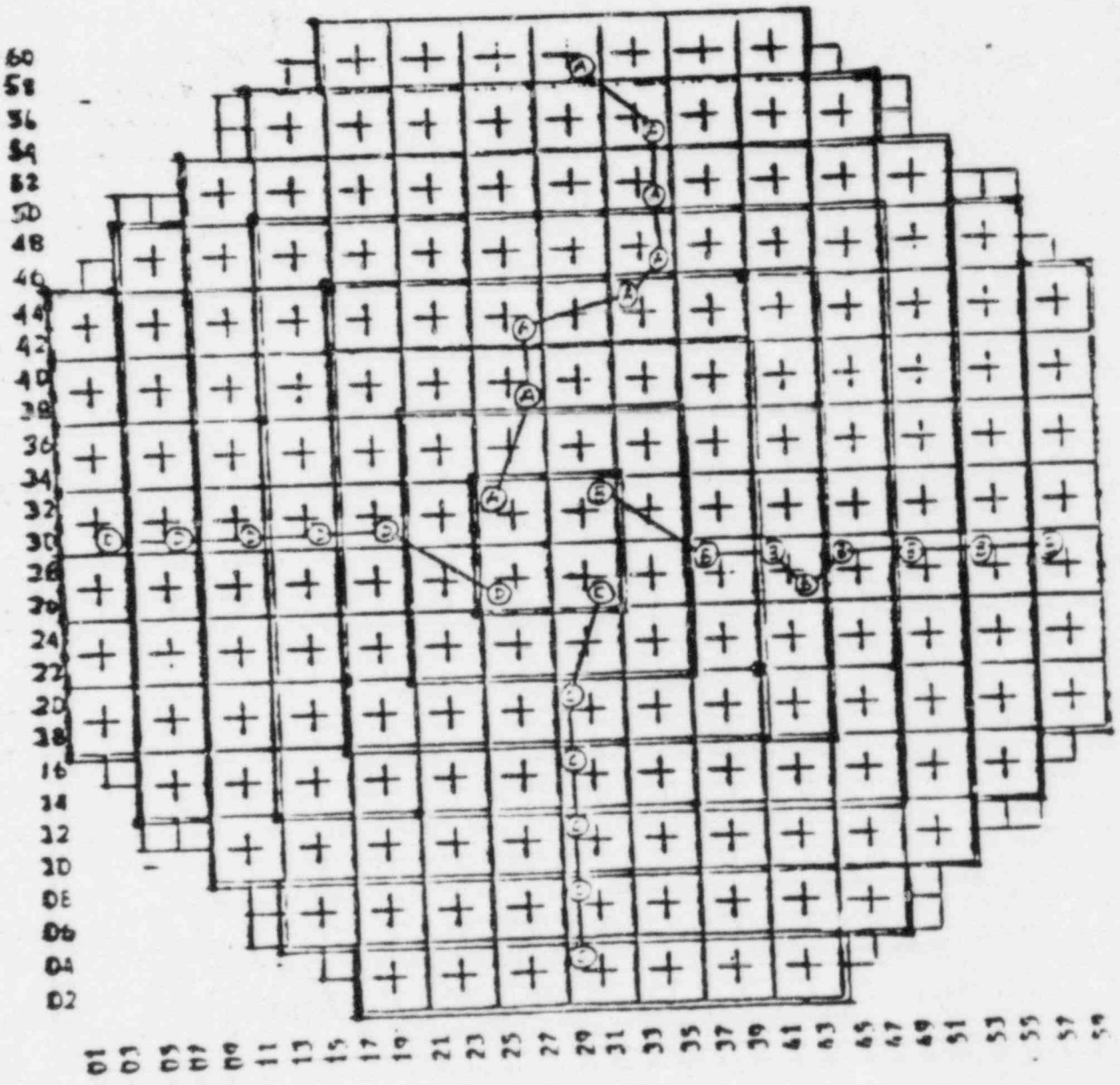


Figure 3.3-4

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 & 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 3.3-5

1/M Plot

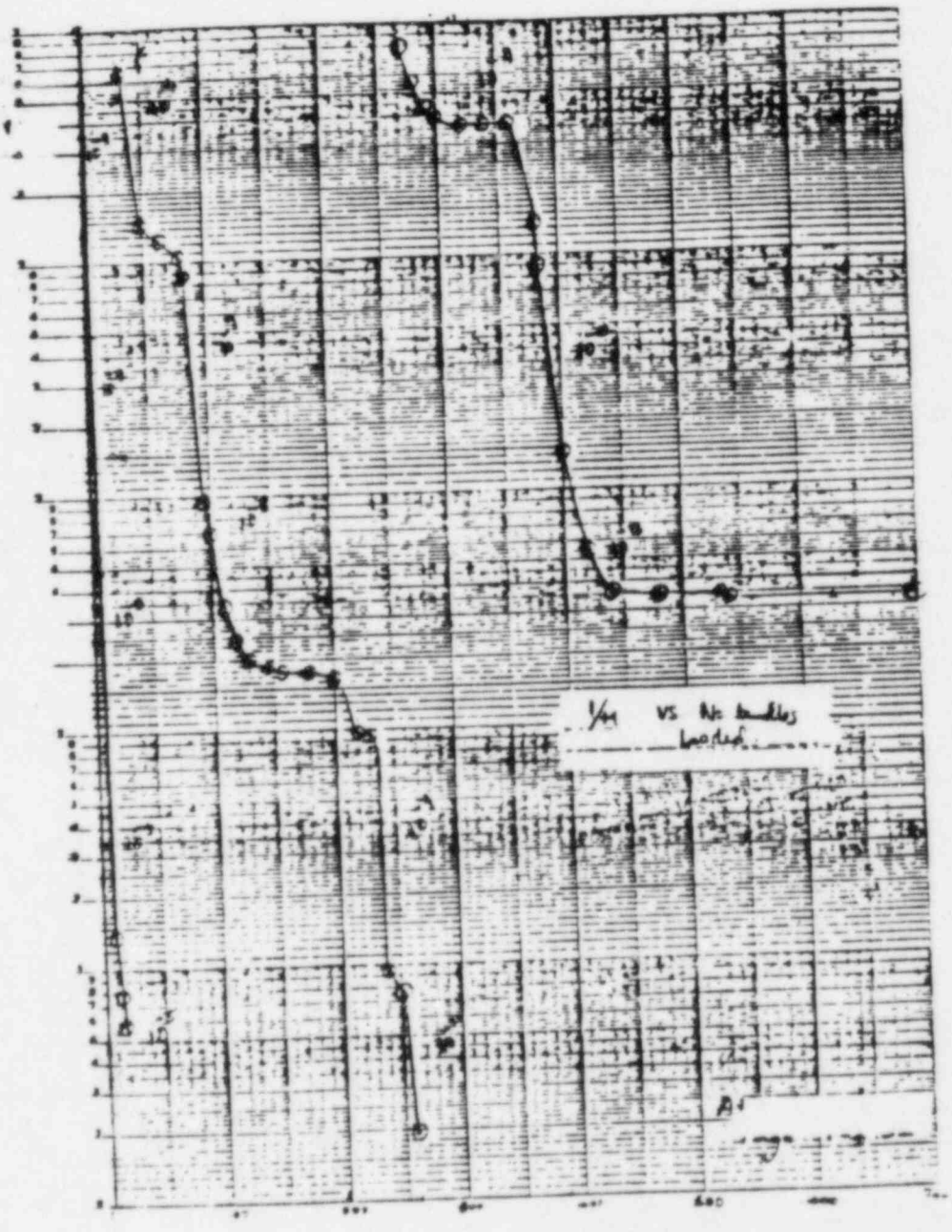


Figure 3.3-6

FUEL LOCATION VERIFICATION

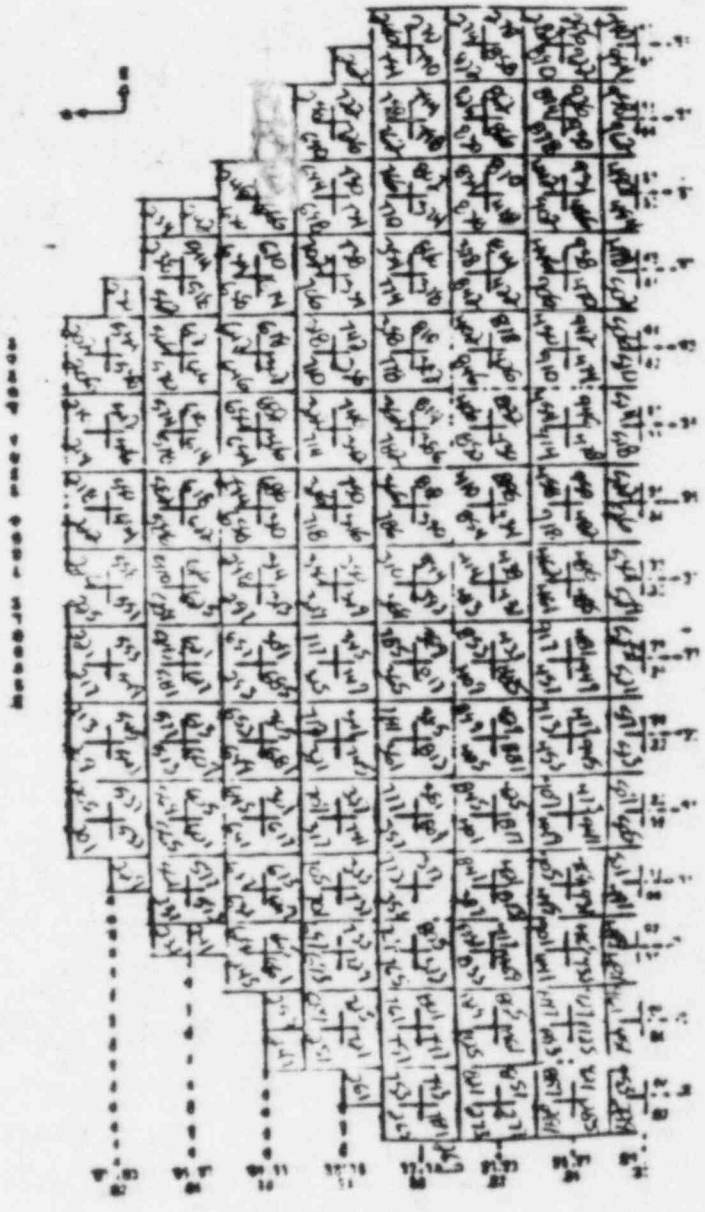


Figure 3.3-7

FUEL LOCATION VERIFICATION

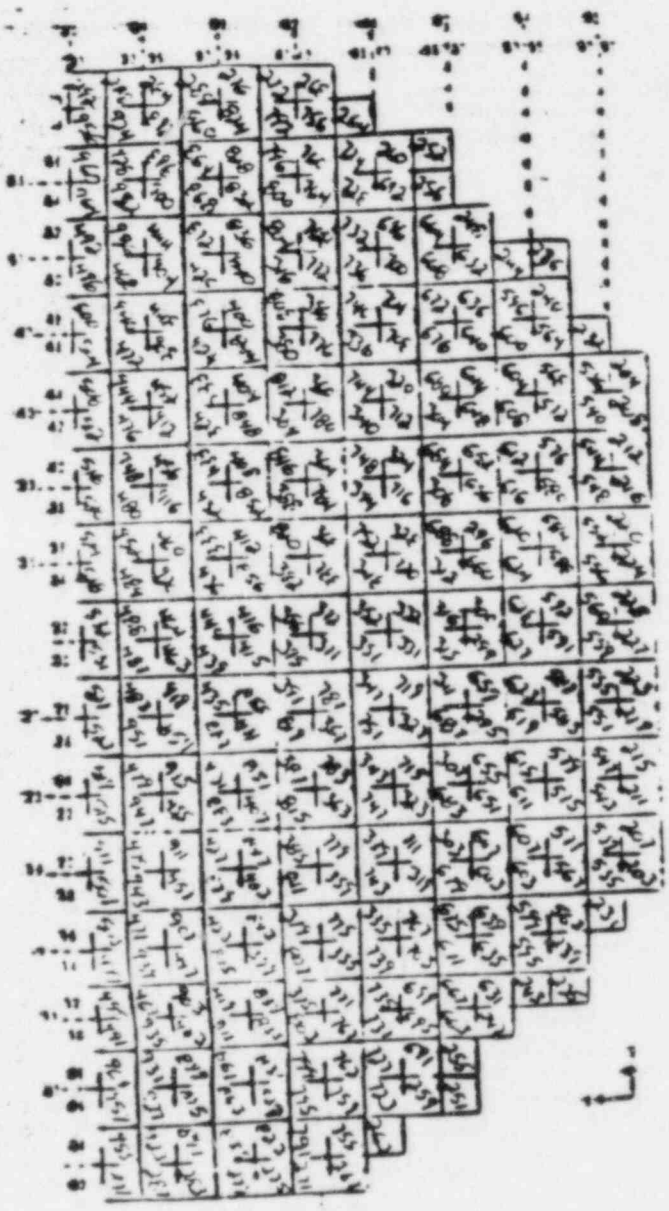


Figure 3.3-7

3.4 Full Core Shutdown Margin

3.4.1 Purpose

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

3.4.2 Criteria

Level 1

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

Level 2

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

3.4.3 Results

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

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

The shutdown margin of the fully loaded core at 68°F with the analytically strongest rod withdrawn was determined to be 2.72% delta k/k. Level 1 criteria were satisfied since the measured shutdown margin was larger than R + 0.38% - 0.88% Δ 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% Δ k/k. This satisfies Level 2 criteria since criticality occurred within 1% Δ 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 withdrawal speed less than or equal to 3.6 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.

3.5.2 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

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 pressure. 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 3.5-1

Scram Testing Results

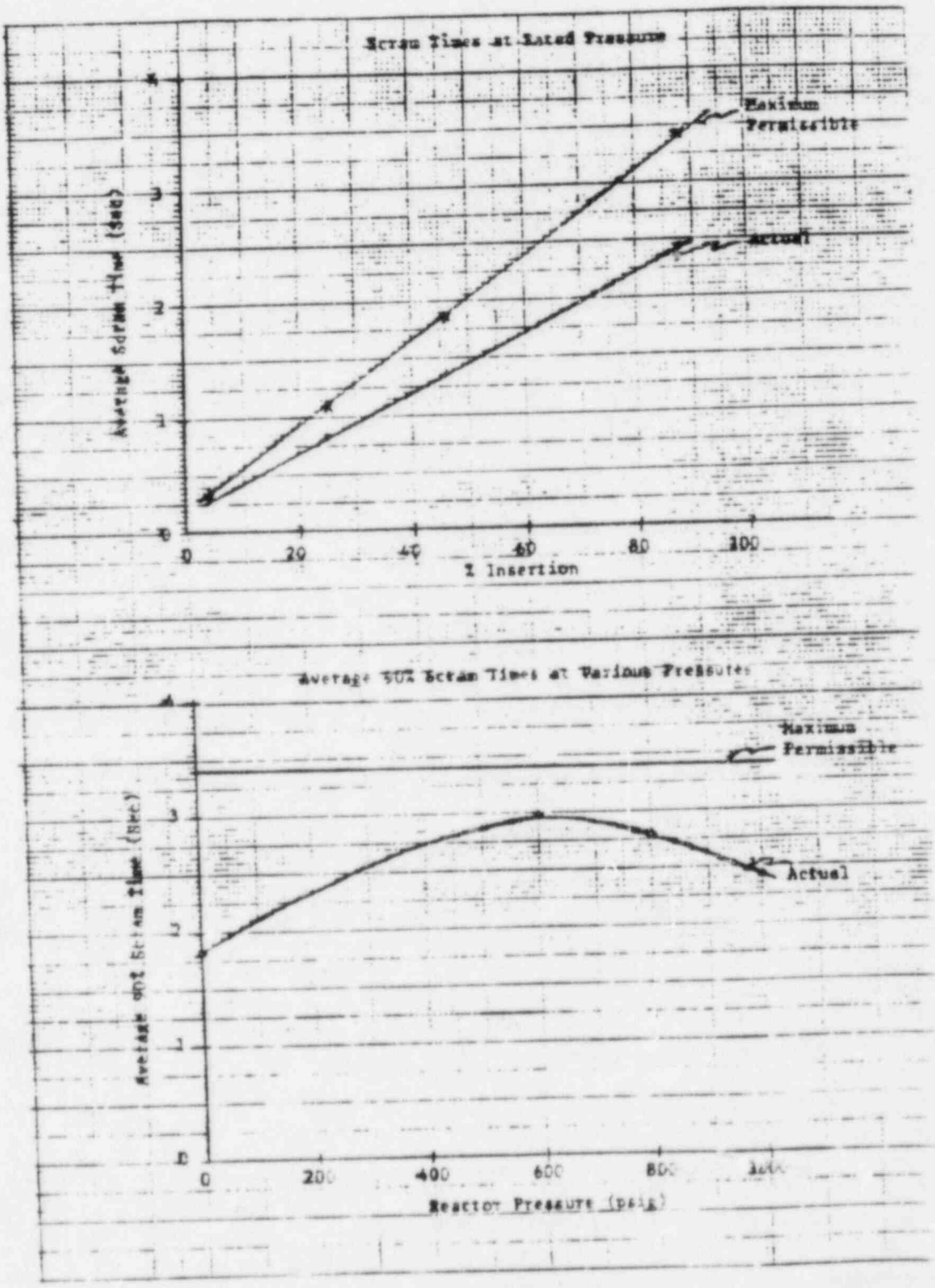


FIGURE 3.5-2

3.6 Source Range Monitor Performance and Control Rod Sequence Exchange

3.6.1 Purpose

The purpose of this procedure was to demonstrate that Source Range Monitor (SRM) count-to-noise ratio, minimum count rate, and saturation requirements are satisfied and to monitor Intermediate Range Monitor (IRM) readings during heatup. This procedure was performed successfully.

3.6.2 Criteria

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.

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.

3.7 IRM Performance

3.7.1 Purpose

The purpose of this test is to verify that the Intermediate Range Monitor (IRM) System is functional by increasing the neutron flux in the reactor following initial criticality, until all eight channels come on scale with significant overlap to the SRM's and to subsequently ensure adequate overlap with the APRM system.

3.7.2 Criteria

Level 1

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

NOTE: Tech Spec table 4.3.1.1-1 requires 1/2 decade overlap between the SRMs and IRMs.

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

NOTE: Tech Spec Table 4.3.3.1-1 requires 1/2 decade of overlap during a controlled shutdown.

3.7.3 Results

During the initial criticality, all IRM's except IRM D showed response prior to the SRM's reaching 5×10^4 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. The second criteria could not be verified but will be done upon reaching higher power levels in Test Condition 1.

3.8 LPRM Calibration

3.8.1 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 values.

3.8.2 Criteria

Level 2

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

3.8.3 Results

This 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. At the present time four LPRMs located on the periphery of the core have failed to respond. These peripheral detectors are expected to respond during further testing at higher power levels in Test Condition 1. Also at that time, the LPRMs will be calibrated to the Level 2 criteria.

3.9 Average Power Range Monitor Calibration

3.9.1 Purpose

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

3.9.2 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.

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.

3.9.3 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.

The Level 2 criteria could not be proven as reactor power and heatup rate decreased after completion of the heat balance. Since the adjusted power readings were far below 7%, this criteria is not a major concern at low power levels, and will be applied to APRM testing for Test Condition 1 and beyond.

3.10 Process Computer

3.10.1 Purpose

The purpose of this test was to perform the Transversing In-Core Probe (TIP) cold alignment and Static System Test Case (SSTC) after installation of the TIP System.

3.10.2 Criteria

Level 2

The cold TIP alignment setting and the Static System Test Case (SSTC) testing is completed by the satisfactory performance of this test procedure.

3.10.3 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 SSTC 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 design error in the TIP System that resulted in the loss of TIP position indication. The TIP interface problem will be corrected and the criteria will be verified in the Test Condition 1 testing.

3.11 RCIC System

3.11.1 Purpose

The purpose of this test was to set up the RCIC control system and to verify the proper operation of the Reactor Core Isolation Cooling (RCIC) System while the reactor is at rated pressure conditions.

3.11.2 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.

3.11.3 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 manual 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. This will be verified later after engineering modifications to the instrument lines are completed.

3.12 HPCI System

3.12.1 Purpose

The purpose of this test was to verify proper operation of the High Pressure Coolant Injection (HPCI) system with the reactor operating at rated pressure.

3.12.2 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.

3.12.3 Results

Following setup of the control system, initial coupled turbine performance runs were performed on the HPCI system. 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. This will be verified later after engineering modifications to the instrument lines are completed.

3.13 Select Process Temperatures

3.13.1 Purpose

The purpose of this test is to ensure the measured bottom head drain temperature corresponds to the bottom head coolant temperature and to verify that temperature stratification in the bottom head region is avoided by proper setting of the Recirculation low speed limiter. In addition, the calibration of the reactor water level indicators was checked as well as the temperature of the reference legs.

3.13.2 Criteria

Level 1

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

Level 2

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

3.13.3 Results

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 ± 2% of rated speed. Flow controller variations of ± 5% were experienced prior to stratification so the test was terminated. Further investigation of the low flo region will be conducted during the completion of remaining heatup testing.

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. However, subsequent to the performance of this test, a drywell cooler that was out of service at the time has been repaired and additional insulation on some drywell support beams has been added which should improve on the previous results. As such, this data will be re-taken during the remaining heatup testing and the appropriate analysis redone.

3.14 System Expansion

3.14.1 Purpose

The purpose of this test is to verify that selected plant piping systems are free and unrestrained with regard to thermal expansion.

3.14.2 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.

3.14.3 Results

Piping Inspection

Selected piping systems were walked down at various plant conditions to identify possible restraints to projected thermal expansion. These walkdowns occurred at ambient temperature, 250° 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.

System Expansion

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 the 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.

3.15 Feedwater System

3.15.1 Purpose

The purpose of this test was to verify the operability of the Feedwater Control System in the manual and aut modes of operation using the Startup Level Controller.

3.15.2 Criteria

Level 1

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

Level 2

The 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.

3.15.3 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 criteria were satisfied.

3.16 MSIV Functional Test

3.16.1 Purpose

The purpose of this test was to functionally check the main steam line isolation valves (MSIVs) for proper operation by performing a slow closure from the 100% open position to approximately 90% open position followed by a return to 100% open position on each valve.

3.16.2 Criteria

None

3.16.3 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.

3.17 Relief Valves

3.17.1 Purpose

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

3.17.2 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°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.

3.17.3 Results

During the heatup testing, all 15 SRV's 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 as indicated on the GETARS off line plots. 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 SRV's did not return to within 10°F of the temperature recorded prior to actuation as soon as the others, however, they did cool down sufficiently to indicate that they were not leaking by. 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 SRV's did properly reclose.

Three SRV's had steam flow values, as measured by BPV position change, that differed from the average relief valve steam flow by greater than 10%. The bypass valve position was inadequate to get a proper value of steam flow from BPV position change. Upon the actuation of each SRV the BPV closed completely. Had there been more bypass steam flow, the BPV would not have closed completely and there would be a more accurate value of SRV steam flow. This steam flow variance will be reevaluated during the TC2 SRV testing.

3.18 Piping System Vibration

3.18.1 Purpose

The purpose of this test is to determine the flow induced transient vibration responses of the Main Steam lines inside the drywell and Safety Relief Valve (SRV) lines during planned SRV operation for dynamic transient vibration hardwired testing.

3.18.2 Criteria

Level 1

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

Level 2

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

3.18.3 Results

All vibration data taken in the conduction of the Heatup SRV testing was well below all specified Level 1 and Level 2 criteria.

3.19 Reactor Water Cleanup System

3.19.1 Purpose

The purpose of this test was to verify the proper operation of the RWCU system in the Blowdown and Normal modes. Included in this was a calibration verification of the RWCU bottom head flow transmitter.

3.19.2 Criteria

Level 2

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

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

The bottom head flow indicator will be recalibrated against the RWCU flow indicator if the deviation is greater than 25 gpm.

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

3.19.3 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.

The RWCU system was operated in both the normal and blowdown modes with the reactor at rated temperature and pressure. Process variables were recorded in order to demonstrate the proper performance of the RWCU system in each of these modes. The non-regenerative heat exchange tube side outlet temperatures for the normal and blowdown mode were 112°F and 122°F respectively. These values were within the Level 2 criteria limits of 120°F and 130°F for each mode. Using temperature measurements from the RBCCW side of the non-regenerative heat exchangers (NRHX) the cooling water flow was calculated to be less than 6% above the NRHX capacity. The non-regenerative heat exchanger cooling water outlet temperatures were well within our Level 2 criteria of 180°F. All Level 2 criteria were satisfied.

3.20 Residual Heat Removal System

3.20.1 Purpose

The purpose of this test is to demonstrate the ability of the Residual Heat Removal (RHR) System to remove heat deposited in the suppression pool.

3.20.2 Criteria

Level 2

The RHR System is capable of operating in the suppression pool cooling mode at the flow rates and temperature differentials indicated on the process diagrams.

3.20.3 Results

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 had 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.