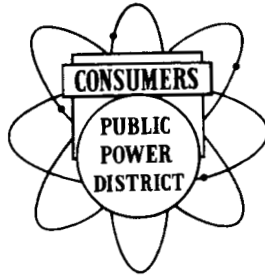


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# HALLAM NUCLEAR POWER FACILITY

## MONTHLY OPERATING REPORT

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Date: 11-9-06

NO. 22

MAY, 1964

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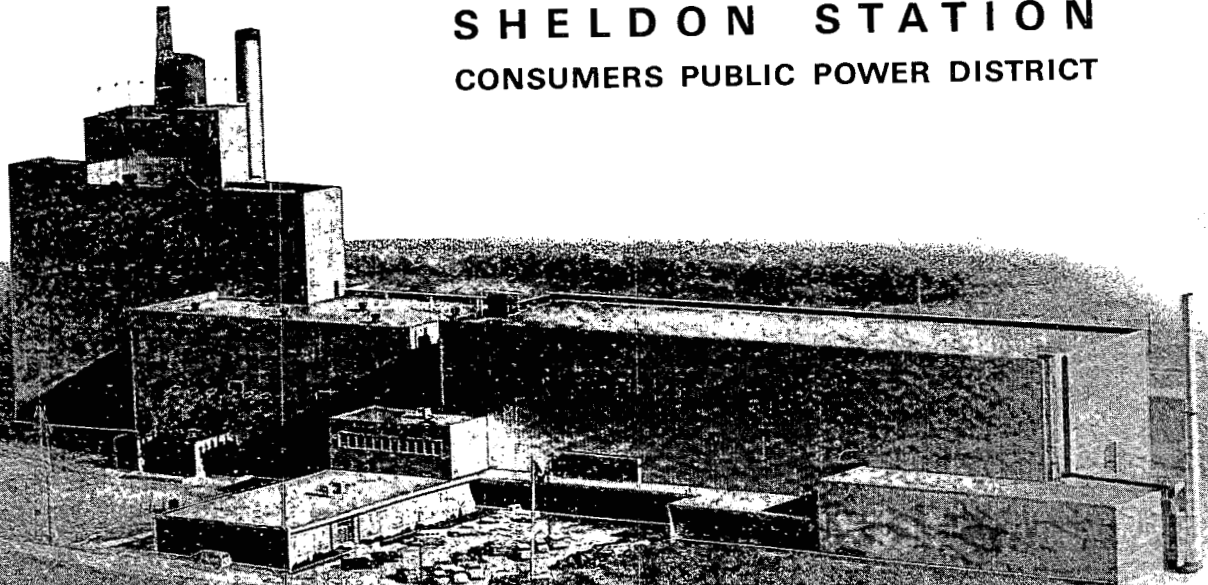
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# HNPF MONTHLY OPERATING REPORT

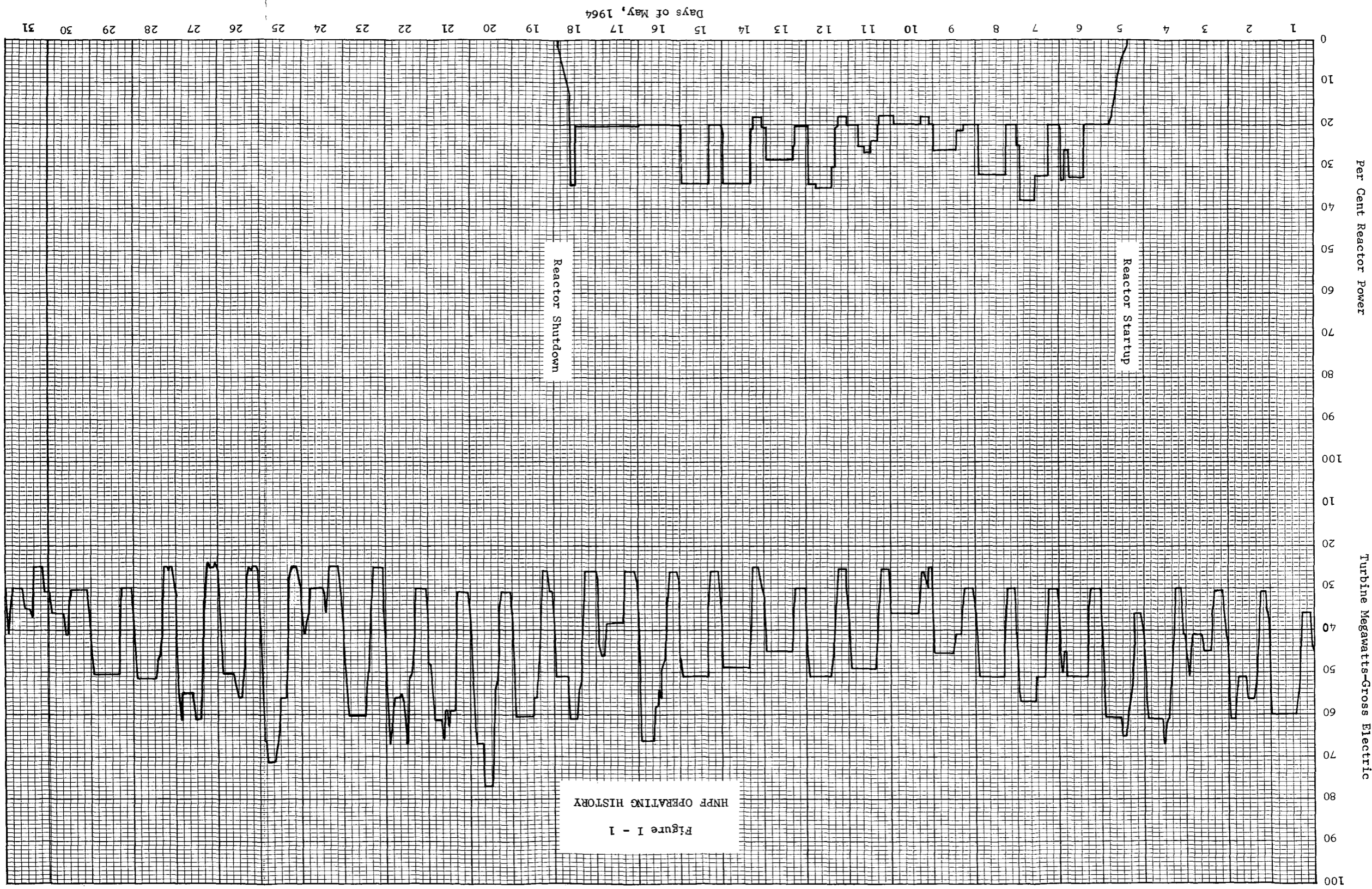
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May, 1964

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## I. Plant Operation

### A. Operating History, May, 1964

The reactor plant remained shut down until May 5 for completion of repair work on a bellows seal valve, then operated continuously in parallel with the conventional boiler until May 18. On May 18, the reactor plant was again shut down and remained shut down through the end of the month. The reason for the shut down was the rupture, on May 15, of an additional moderator can (M-58). During the shut down, control rod drop time tests were performed to verify that no serious misalignments in the core had been generated by the additional ruptured can. Following the successful completion of the control rod drop time tests shut down maintenance was performed on a number of items which were considered to be potential trouble makers.

No accidental scrams occurred during the month. Sheldon Station electrical output was uninterrupted.

Reactor thermal power (in percent of 250 Mwt) and gross station electrical output are shown on Figure I - 1.

Following is a condensed summary of significant operating activities during the month. Additional detail on the items mentioned will be found in various appropriate sections of this report.

May 1 - 5     HNPF remained shut down for repair of valve V-476.

Significant activities during this period.

May 5     Installation of new bellows and reinstallation of heaters and lagging on valve V-476 was completed. Preheat of the associated piping was started.

First part of special test on an orifice drive was performed. It was found that galling of the drive threads occurred (on the test element) at about 275 lb-in. torque.

May 6 - 18 HNPf operated at loads between 38 percent and 18 percent in accordance with CPPD load requirements.

Other significant activities during this period.

May 6 Second part of special test on an orifice drive was performed. A second drive assembly was lubricated and driven downward (in compression). Galling of the threads was observed at 150 - 200 lb-in. torque.

May 7 A leak alarm on valve V-476 was received at 1819. Preheat of the sodium service lines near the valve was discontinued.

May 8 The shield plug was removed and the primary service cell entered to inspect the leak detector on V-476. The leak detector was found to be shorted. It was replaced and the cell resealed. A nitrogen purge of the cell was initiated. Preheat of sodium service system lines was initiated.

May 11 Circulating flow from primary fill tank No. 3 through the primary service system was established at 1515. Forty-three minutes after the establishment of flow, another leak alarm was received on V-476. The system was immediately shut down and drained.

May 12 The primary sodium service cell plug was removed for a radiation survey and possible inspection of V-476. With the plug removed radiation level was 2.5 R/hr at floor level. Evidently, some radioactive sodium had leaked into the service system, perhaps when it was drained. The plug was therefore replaced. Action was taken to isolate this section of the piping by "freezing" lines leading back to the primary coolant loops. An apparent blockage of the reactor helium supply line was noticed. By observation of the reactor atmosphere pressure controller and a similar controller on the primary fill tanks, it was determined

that the blockage was either in the reactor helium supply vapor trap (VT-14), or between the vapor trap and the reactor. The "real" reactor atmosphere pressure remained known and in control, because the reactor atmosphere and the primary fill tank atmospheres communicate through a heated 3 inch vent line.

May 13 Another radiation survey, with the primary sodium service cell plug removed, gave 1.2 R/hr at floor level. The plug was replaced.

May 14 Blockage of reactor helium line was cleared by draining the line between VT-14 and the reactor to the primary sodium service drain tank. During this operation, a leak detector alarm was received on one of the valves involved (V-481). The alarm came in when the valve was opened, then cleared when the valve was closed. This was indicative of incorrect leak detector placement rather than an actual sodium leak.

May 15 The primary sodium service cell plug was again removed for a radiation survey. Radiation level at floor level was now 120 mr/hr. Radiation level in the cell near the working area was about 1.8 R/hr. No sodium could be seen.

A satisfactory test of the dry scrubber was performed.

At 2045, it was noted that control rods were moving out, and that fuel outlet temperatures around moderator can M-58 were dropping significantly. A "tilt" in core flux pattern as seen on the nuclear instruments was also observed. These symptoms were immediately attributed to rupture of the can.

Reactor power was lowered to 20 percent to accomodate a decrease in system demand and left at this power level for the weekend, while frequent fuel channel exit temperature maps were continued.

May 16 At 0655 a brief reactor setback was initiated by the plant protective system  $T_O$  computers. The cause of the setback was a spurious momentary decrease in the "measured" flux signal to the neutron flux controller. The flux controller, as a result, pulled rods until the  $T_O$  computers went into setback. The flux controller tripped to manual, and the reactor was leveled on manual control.

Difficulty with the reactor helium supply line continued. The line was periodically drained to restore it to normal.

May 18 The reactor was returned to the same power level ( $\sim 35$  percent) that existed prior to rupture of can M-58. A complete core temperature "map" was made for comparison purposes.

At 2250, the reactor was shut down, after being programmed to essentially zero power using a new shut down procedure. No scram was involved.

May 19 - 31 The HNPF remained shut down for control rod drop time tests and maintenance on a number of items in the plant which were potential causes for future shut down.

#### Significant activities during this period.

May 19 Leak detector for V-476 was replaced.

Drop time tests were run on all 16 movable control rods. All drop times were satisfactory.

May 20 All thimbles which had previously been removed from fuel storage pit No. 3 were reinstalled.

May 21 The reactor was brought to criticality at 0735 for a "zero power" reactivity check. It was found that no reactivity change had taken place since shut down on May 18. Reactor was shut down at 0838. The primary fill tank cell was entered to repair the pedestal

heaters on primary fill tank No. 3, and to replace a ruptured rupture disk between the primary fill tank cell and the primary pipeway.

Helium supply lines to the secondary sodium pump oil sump area were installed.

Fuel storage pit No. 3 was filled with demineralized water.

May 22 Fuel storage pit No. 3 was placed in service on the closed loop cooling water system.

The leak detector on V-476 was inspected and found to be in good condition.

The leak detector on V-481 was replaced. Primary service cell plug was replaced and a nitrogen purge initiated.

May 23 Sodium flow was established from the primary system through the primary plugging meter at 0535. A plugging run made at 0820 showed a primary plugging temperature of 260 F.

May 25 The reactor was brought to criticality at 1049 for criticality checks and operator training. The reactor was shut down at 1327. Cleaning of the raw water side of the steam dump was begun.

May 26 Cleaning of raw water side of the steam dump continued.

May 27 At 0900 a 100 hour carbon trap run was begun. Cleaning of the raw water side of the steam dump continued.

May 28 At 1625, lowering of reactor sodium temperature to 350 F was begun, in preparation for removal of splash baffles from the No. 3 evaporator.

Cleaning of the raw water side of the steam dump continued.

Tension and compression tests were run on a spare orifice drive assembly.

May 29 At 1145, draining of sodium from the No. 3 secondary loop was completed. At 1535, draining of water from the No. 3 evaporator was completed. The manhole was opened and forced air cooling initiated.

May 31 At 1300, a 100 hour carbon trap run was completed. Cool-down of the trap was begun.

## B. Outage Summary

No unplanned scrams occurred during the month of May. The outage started during April was terminated on May 5. Another shut down took place on May 18, but no scram was initiated.

Both outages are tabulated below. For the purposes of this summary, an outage is defined to be any time during which the reactor plant is not supplying steam to the turbine.

Table I - 1

### OUTAGE SUMMARY, MAY, 1964

Duration of Outage, 0000, May 1, 1964 to 1840, May 5, 1964.

Reactor remained shut down for repair of bellows seal valve V-476 in the primary sodium service cell.<sup>1</sup>

Lost Critical Time<sup>2</sup> 4 days 7 hours 37 minutes

Lost Operating Time 4 days 18 hours 40 minutes

Duration of Outage, 1600, May 18, 1964 to 2400, May 31, 1964.<sup>3</sup>

Reactor was shut down (without a scram) by driving control rods in to achieve subcriticality. Purpose of shut down was to perform control rod drop time tests and shut down plant maintenance.

Lost Critical Time<sup>4</sup> 12 days 21 hours 29 minutes

Lost Operating Time 13 days 8 hours

<sup>1</sup> See April HNPF Monthly Operating Report for previous data on this outage (page 8).

<sup>2</sup> Reactor returned to criticality at 0737, May 5, 1964.

<sup>3</sup> Reactor remained shut down at the end of the month.

<sup>4</sup> Reactor critical from 0735 to 0838, May 21, 1964 for zero power reactivity checks, and from 1049 to 1327, May 25, 1964, for operator training and criticality check.

## II. Plant Performance

### A. Overall Thermal Performance

Gross reactor electric generation for the month of May was 5,340 Mwh for an average gross load of 17.3 Mwe. Further details are listed on Table II - 1, Plant Performance Data.

The Sheldon Station turbine generator is designed to receive steam separately or in parallel from the Hallam Nuclear Power Facility and a conventional fossil fuel steam generator. Parallel operation constitutes a large portion of the accumulated reactor turbine service time. Parallel operation was the mode of operation for all the reactor turbine service time logged during May.

Table II - 1

## PLANT PERFORMANCE DATA

Total Station Electric Generation	<u>Current Month</u>	<u>This Year</u>	
Gross Generation (Mwh)	32,686	172,012	
Station Auxiliary Service (Mwh)	3,881	16,369	
Station Net Generation (Mwh)	28,805	155,643	
Average Gross Load (Mw)	43.9	47.2	
Plant Capacity Factor; 95 Mwe base (%) <sup>1</sup>	46.2	49.7	
Plant Availability Factor (%)	100.0	100.0	
Net Plant Heat Rate (Btw/Kwh)	13,300	----	
Turbine Cycle Heat Rate (Btu/Kwh)	9,830	----	
Operating Hours			<u>To Date</u>
Turbine Generator	744.0	3,647.6	---
Reactor Critical	330.9	2,792.9	6,619.7
Reactor Turbine Service	309.3	2,686.7	5,014.5
Reactor			
Gross Thermal Generation (Mwd)	816	8,848	21,088
Average Thermal Power (Mwt)	59.2	76.0	76.5
Gross Reactor Electric Generation (Mwh)	5,340	64,711	150,650
Average Gross Load (Mwe)	17.3	24.1	30.0
Reactor Capacity Factor; 83 Mwe base (%) <sup>3</sup>	20.8	29.0	36.1
Reactor to Turbine Availability Factor (%)	41.6	73.6	56.0

<sup>1</sup>Plant Capacity Factor = Average Total Gross Load Mw x 100 ÷ 95 Mw

<sup>2</sup>The "to date" reactor to turbine availability factor is based on hours accumulated since May 25, 1963 @ 0000, the approximate time of full power authorization.

<sup>3</sup>Reactor Capacity Factor = Average Reactor Gross Load Mw x 100 ÷ 83 Mw

Average gross loads are based on actual operating hours

B. Equilibrium Operating Data are shown on Table II - 2. The data were taken at a net electrical power of 22.6 Mwe on May 18. At the time the data were taken the reactor was operating in parallel with the conventional steam generator.

Due to parallel operation it is not possible to measure directly the electrical output of the HNPF. The values given for electrical generation and heat rate are therefore expected values rather than measured values.

The large variance in nuclear instrument readings is due to the apparent rupture of moderator can M-58. Adjustment of channel readouts will be made upon startup to compensate for this flux shift.

Table II - 2

## TYPICAL EQUILIBRIUM OPERATING DATA

Date May 18 Time 1200

Thermal Power (Mwt)	82.2
Gross Electrical Power (Mwe)	25.2
Reactor Auxiliary Power (Mwe)	2.6
Net Electrical Power (Mwe)	22.6
Net Plant Heat Rate (Btu/Kwh)	12,400

## Nuclear Instrument Channel Readings

Calculated (250 Mwt base) (%)	32.9
Channel No. 5 (%)	35.7
Channel No. 6 (%)	37.2
Channel No. 7 (%)	33.3
Channel No. 8 (%)	34.0
Channel No. 9 (%)	29.3
Neutron Flux Level Recorder (%)	34.3
Channel No. 10 (%)	32.7

## Steam &amp; Feed Water Conditions

Steam Gen No. 1 Outlet Temp (°F)	855
Steam Gen No. 2 Outlet Temp (°F)	855
Steam Gen No. 3 Outlet Temp (°F)	855
Steam Header Pressure (psig)	860
Turbine Throttle Temperature (°F)	850
Turbine Throttle Pressure (psig)	852
Turbine Throttle Flow (lb/hr)	236,500
Feed Water Supply Temperature (°F)	282

# MONTHLY OPERATING REPORT

Continued

## Steam Generator Conditions

	<u>Outlet (°F)</u>	<u>Pressure (psig)</u>	<u>Flow (lb/hr)</u>
Evaporator No. 1	530	870	74,800
Evaporator No. 2	530	866	87,400
Evaporator No. 3	530	869	74,300
Superheater No. 1	884		
Superheater No. 2	883		
Superheater No. 3	887		

## Sodium Conditions

	<u>Reactor Outlet (°F)</u>	<u>Reactor Inlet (°F)</u>	<u>Flow (lb/hr)</u>
Primary Loop No. 1	931	584	827,000
Primary Loop No. 2	934	578	937,000
Primary Loop No. 3	932	587	830,000
	<u>Stm Gen Inlet (°F)</u>	<u>Stm Gen Outlet (°F)</u>	<u>Flow (lb/hr)</u>
Secondary Loop No. 1	920	557	785,000
Secondary Loop No. 2	920	550	892,000
Secondary Loop No. 3	924	552	763,000

## Fuel Channel Outlet Temperature (°F)

Maximum	986	Channel No.	C-115
Minimum	882	Channel No.	C-145

## Instrumented Fuel Element Temperatures (°F)

Maximum Uranium Carbide	1072	T/C Location	C-41-8
Maximum Uranium Moly	959	T/C Location	C-39-1

## Instrumented Moderator Element

Maximum Flat to Scallop  $\Delta T$  (°F) 19

## Control Rod Configuration (156 inches is fully withdrawn)

1. 12	6. 77.8	11. 156.1	16. 78.9
2. 0	7. 156.1	12. 77.8	17. 78.9
3. 0	8. 156.1	13. 156.1	18. 78.9
4. 0	9. 77.8	14. 78.9	19. 78.5
5. 156.2	10. 156.1	15. 78.6	

## C. Core Performance

### 1. General

Evidence indicates that the rupture of moderator can M-58 occurred in mid-May. This can is located in the northeast quadrant near the core edge.

The core disturbance first became apparent about 1900 on Friday, May 15; the reactor was operating at about 33 percent power. At that time, fuel exit temperatures in the vicinity of M-58 had noticeably decreased -- this being the most immediate indication of the anomaly.

By 2300, May 15, fuel exit temperatures had dropped about 50 F. The compensating control rod withdrawal suggested about a 10 cent loss in reactivity. The resulting flux tilt was producing a signal to the protective signal which put the  $T_0$  computers very close to the alarm point.

At 2330 the system load demand required a reduction in power level to 20 percent. The fuel channel outlet temperatures on the opposite side of the core had increased as much as 12 F in compensation for the reduction around M-58. The average outlet temperature was reduced from 945 F to 930 F, to bring the maximum outlet temperatures down to normal levels. This temperature reduction tended to rebalance the  $T_0$  computer output signal. Fuel exit temperatures leveled out at about 0100, May 16.

Reactor power was maintained at 20 percent throughout the weekend and fuel exit temperature maps were made at four hour intervals. The fossil plant made the required load swings during this period.

On Monday morning, May 18, the power was returned to 33 percent, to duplicate as nearly as possible the conditions immediately prior

to the rupture. Data was then taken which was later analyzed, to more accurately evaluate the effects of the moderator can failure.

## 2. Reactivity History

A reactivity history was maintained as usual during the month. The graph for the pertinent portion of the month is shown on Figure II - 1. The reactivity loss associated with the rupture of M-58 is detailed in the insert of Figure II - 1. The primary reactivity loss is noted to have occurred between about 1700, May 15 and 0100, May 16 -- approximately an 8 hour period. The total loss attributed to the M-58 failure is determined from reactivity balance as 16 cents.

A summary of the reactivity effects of all moderator can ruptures is given in Table II - 3. Here it is seen that the true reactivity loss, when measured values are corrected for rod calibration error, is 19 cents.\*

The HNPF reactivity status at month's end is presented in Table II - 4.

## 3. Relative Power Distribution

The fuel channel exit temperature of each fuel element is normally recorded on a core temperature map once each day. This frequency may be increased during periods of particular importance -- as was the case during the rupture of M-58. From this fuel channel exit temperature data, a radial power map is obtained. Comparing the elemental radial power thus obtained from data taken before and after the rupture of M-58, a percentage change in relative power for each element was

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\* See the HNPF Monthly Operating Report for April, No. 21, page 20 for explanation of this correction.

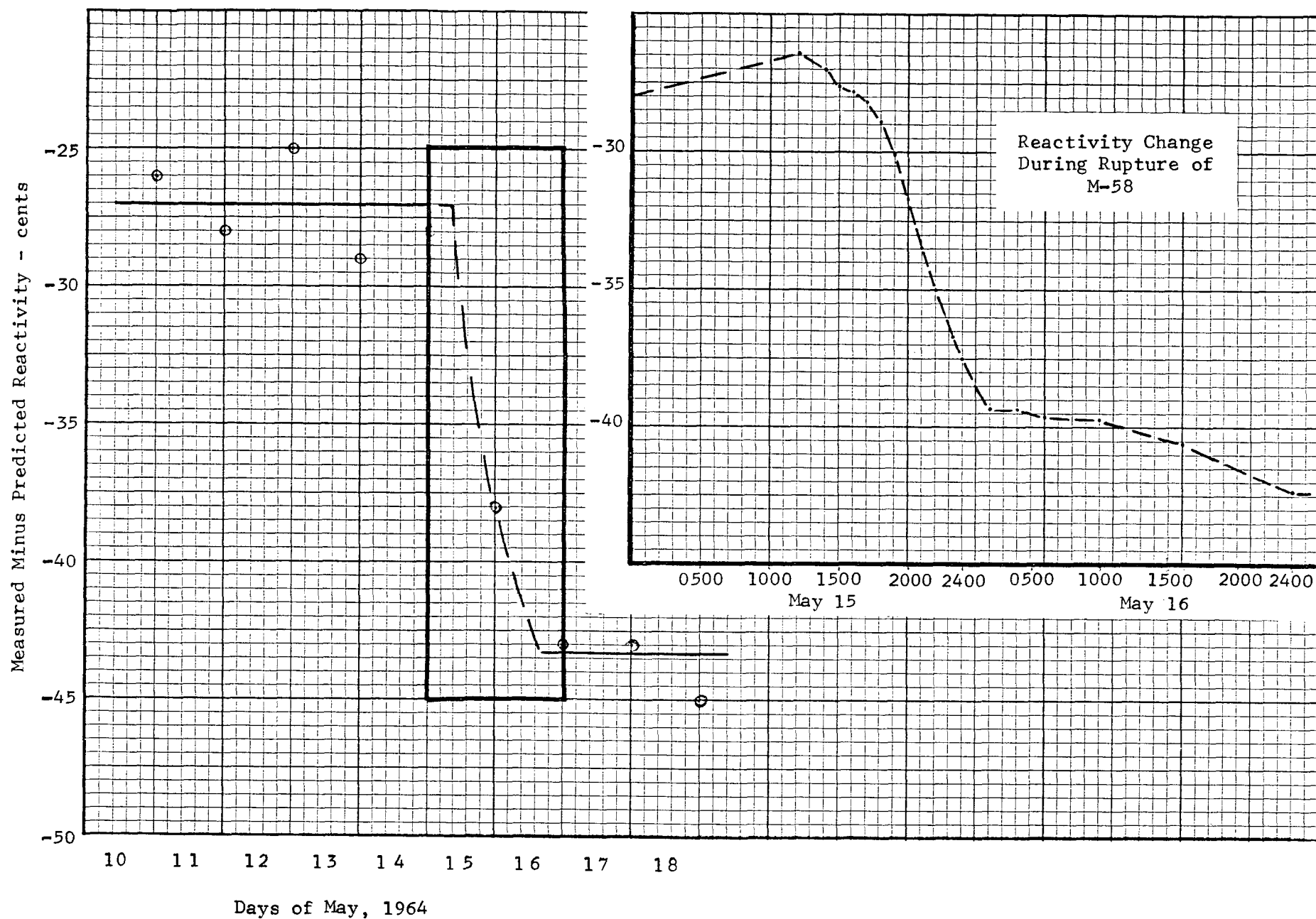


Figure II - 1  
 REACTIVITY CHANGE DURING RUPTURE  
 OF MODERATOR CAN M-58

Moderator Can	Date Ruptured	Date Add'l Disturbance	Actually Observed	Observed - Corrected for Rod Cal Error	Predicted Worth*
M-94	Feb 2	---	3 ¢	4 ¢	4 ¢
M-38 & M-62	Feb 20 & Feb 20-22	March 6-12	9	10 1/2	20 (M-38)
		March 29-April 5	9	10 1/2	11 (M-62)
			7	8	
M-38 & M-62	Total		<u>25</u>	<u>29</u>	<u>31</u>
M-3	March 4	---	18	21	21
M-58	May 15	---	16	19	18
TOTALS			62 ¢	73 ¢	74 ¢

\* Predicted worth is flux weighted to corrected-observed worth of M-3 as base.

Table II - 3

SUMMARY OF FAILED MODERATOR CAN WORTHS

Table II - 4

## HNPF Reactivity Status

Date	Period Prior to Power Operation January 1963		Change Produced by:	October 3, 1963		Change Produced by:	May 31, 1964	
Item	Cents	% $\Delta$ K		Cents	% $\Delta$ K		Cents	% $\Delta$ K
Total rod worth	1840	12.8	20.5 full power days operation; 19 Zr thimbles replaced by ss; addition of 10 UC elements— total elements 150.	1633	11.3	63.9 additional full power days operation, making a total of 84.4 full power days.	1633	11.3
Measured Excess Reactivity	983	6.8		845	5.8		708*	4.9*
Shutdown Margin	857	6.0		788	5.5		925	6.4
Na Worth + 1%	735	5.1		735	5.1		735	5.1
"Extra" Shutdown Margin	122	0.9		53	0.4		190	1.3

\* Reactivity losses associated with moderator can ruptures not included.

evaluated. The pertinent portion of this data is shown in Figure II - 2. A maximum change of 15 - 20 percent is noted for the elements adjacent to the failed can.

4. Fuel Element Orifices

No adjustments were made to fuel element orifices during May. The status of orifices for the month's end is thus the same as that reported in the April report.

5. Fuel Status

The status of fuel at the end of the month is given by Table II - 5.

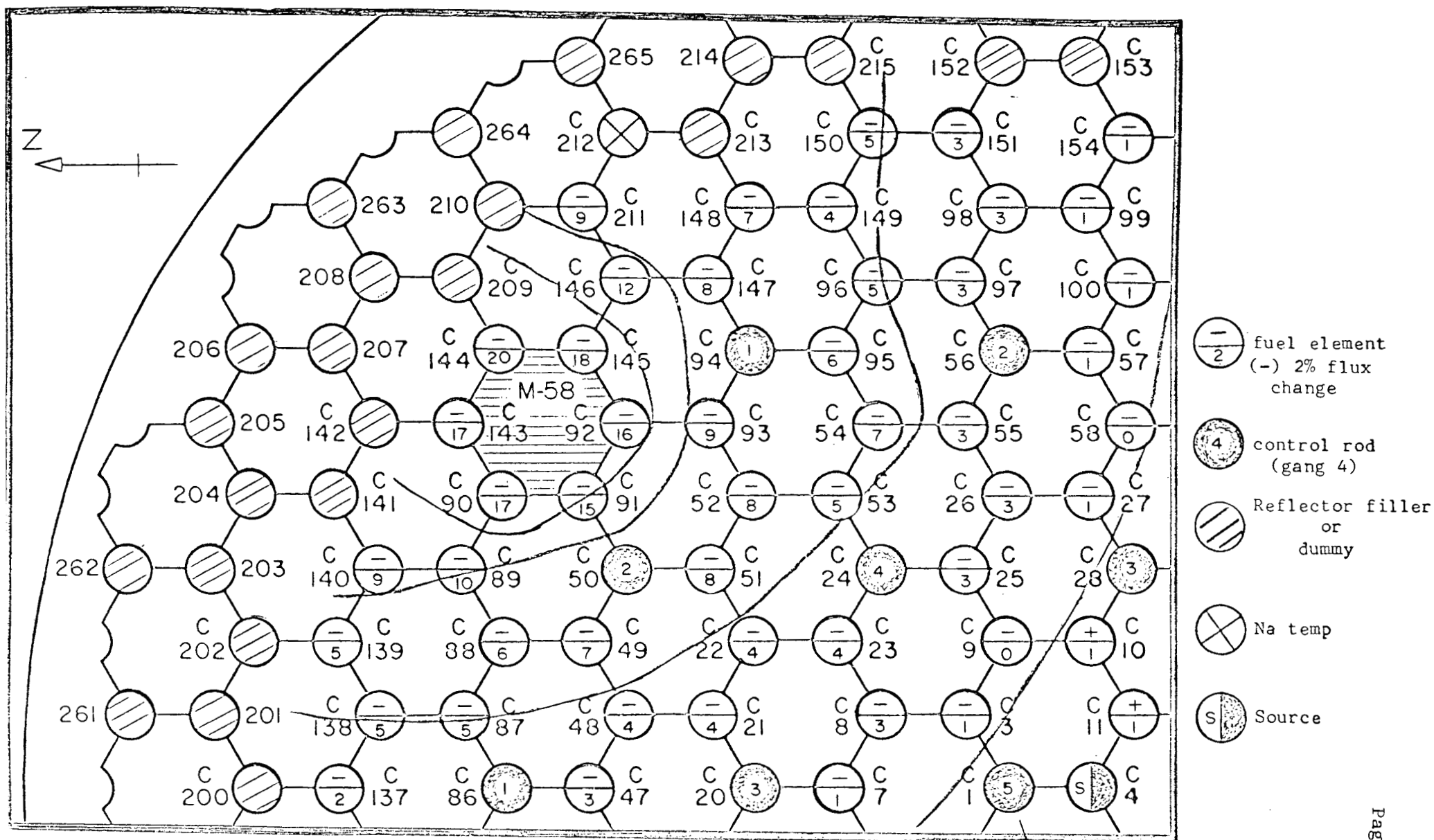


Table II - 5

## FUEL STATUS

May 31, 1964

## Fuel Inventory

## Location

	<u>I</u>	<u>II</u>	<u>III</u>	<u>IV</u>	<u>Total</u>
Reactor	139	1	8	2	150
Storage	3	10			13

Assembly Type

I	U-10 Moly (3.6 w/o)
II	U-10 Moly (3.45 w/o)
III	UC (3.7 w/o)
IV	UC (4.9 w/o)

## Fuel Isotopic Changes

Consumed (Kg)	<u>Current Month</u>	<u>Cumulative Since January 1963</u>
U-235	1.0	26.8
Pu	0.03	0.72
Accumulated (Kg)		
Pu	0.35	9.71

Cumulative Since January 1963	Element No.	Core Position	Average Exposure (Mwd)	Peak Exposure <sup>1</sup> (Mwd)	Average Burnup (Mwd/T)	Peak Burnup <sup>1</sup> (Mwd/T)
Core Average	---	---	---	---	723	---
U-Moly Element w/highest exposure	35 <sup>2</sup> 66	C-51 C-43	208	389	1055	1980
U-Moly Pre-measured Element w/highest Exposure <sup>3</sup>	31 152 158	C-26 C-34 C-18	184	344	935	1740
U-C Element w/highest Exposure <sup>4</sup>	5	C-69	128	239	815	1520

<sup>1</sup> An axial peak/average flux of 1.7 (integrated average) and an elemental disadvantage factor of 1.1 are assumed.

<sup>2</sup> Two elements have approximately equal exposure.

<sup>3</sup> Element in C-39 had premeasured rods and has greater exposure, but this element is thermocoupled.

<sup>4</sup> All U-C elements have premeasured rods.

## D. Sodium Purity

### 1. Cold Trap Performance

#### a. Primary System

There was no cold trapping of the primary system during May. Total oxide collected by primary cold trap number one therefore remains at 65.7 pounds.

A manual plugging run was performed on May 23 with a resulting plugging temperature of 260 F. The previous plugging temperature which was 250 F was obtained on March 31. Normally, plugging temperatures are obtained more frequently but maintenance work in the primary sodium service cell would not permit circulation of sodium through the primary sodium service system. The small increase in plugging temperature indicates that the oxide buildup in the primary system is very slow.

On May 25 a plugging temperature of 311 F was obtained for primary fill tank number three.

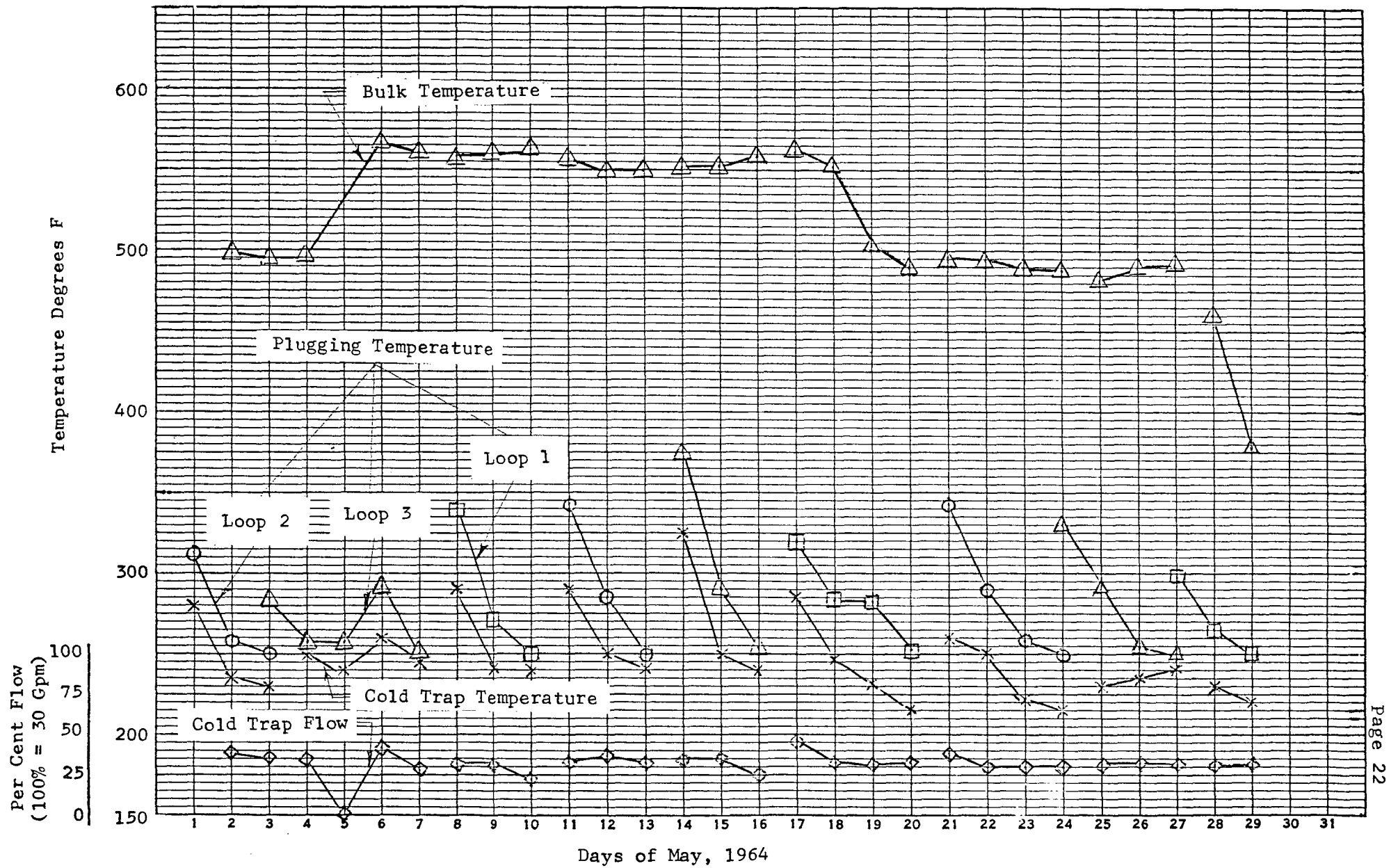
#### b. Secondary System

All the secondary loops were cold trapped during April as shown on Figure II - 3. The amount of oxide removed was calculated based on plugging temperature and the average rate of oxide going into solution of 0.3 pounds per day.

Table II - 6

<u>Loop</u>	<u>Total Oxide Removed (lbs)</u>
1	10.1
2	12.4
3	13.3
Total oxides removed by trap through April	<u>53.3</u>
Total	89.1 lbs

Figure II - 3  
SECONDARY COLD TRAP PERFORMANCE



## D. Carbon Trap Performance

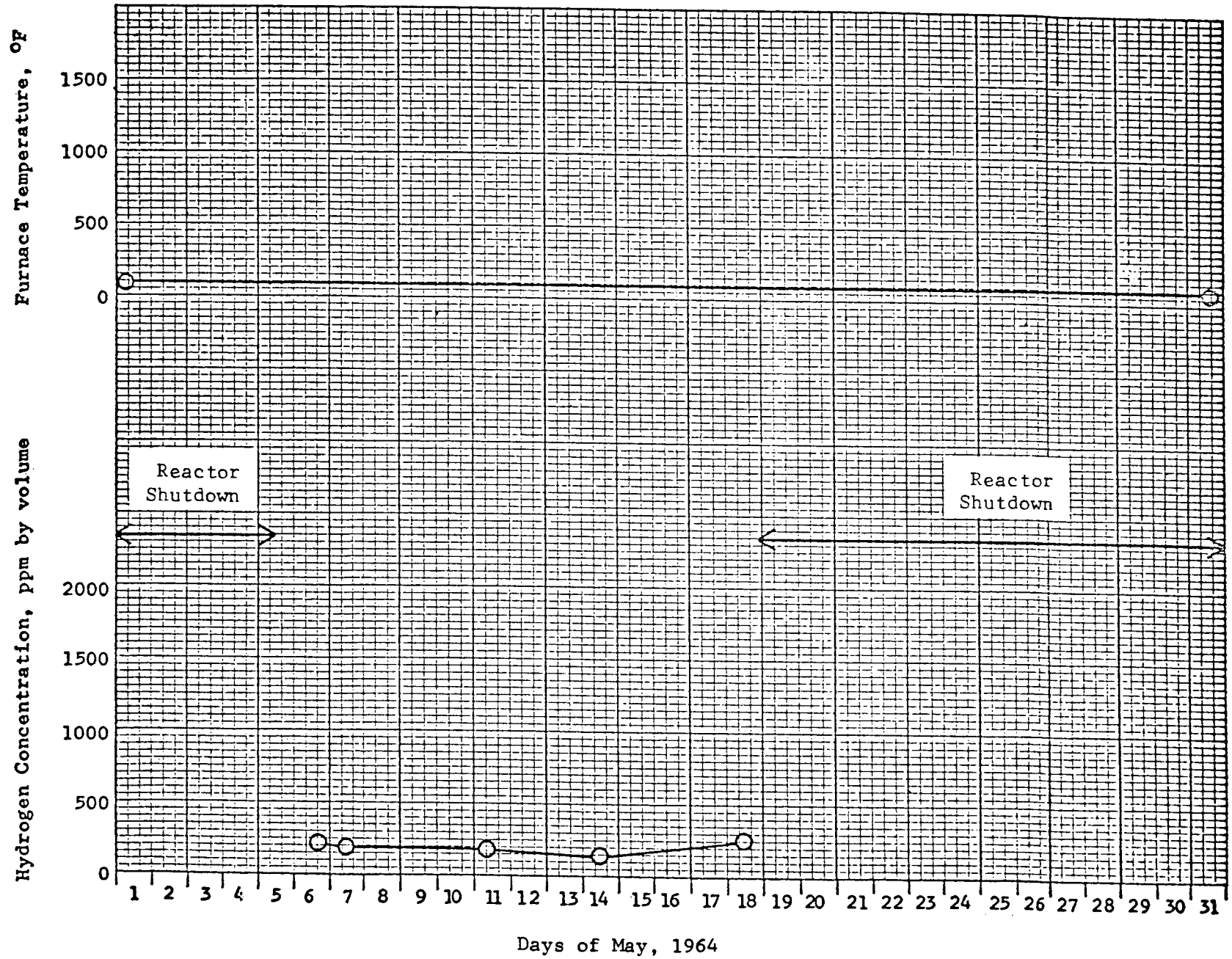
Preheating of the carbon trap commenced on May 23. The carbon trap run was initiated on May 27, at 0900. On May 31, at 1300 the 100 hour run was completed and the carbon trap was allowed to cool down.

## E. Control Rod Thimble Helium Atmosphere

Helium was circulated through the control rod thimbles throughout the month. The heaters in the hydrogen removal furnace were off at all times. During the period of reactor operation from May 5 to May 18, the hydrogen concentrations varied from 130 to 250 ppm by volume as shown in Figure II - 4.

Figure II - 4

CONTROL ROD THIMBLE HELIUM ATMOSPHERE



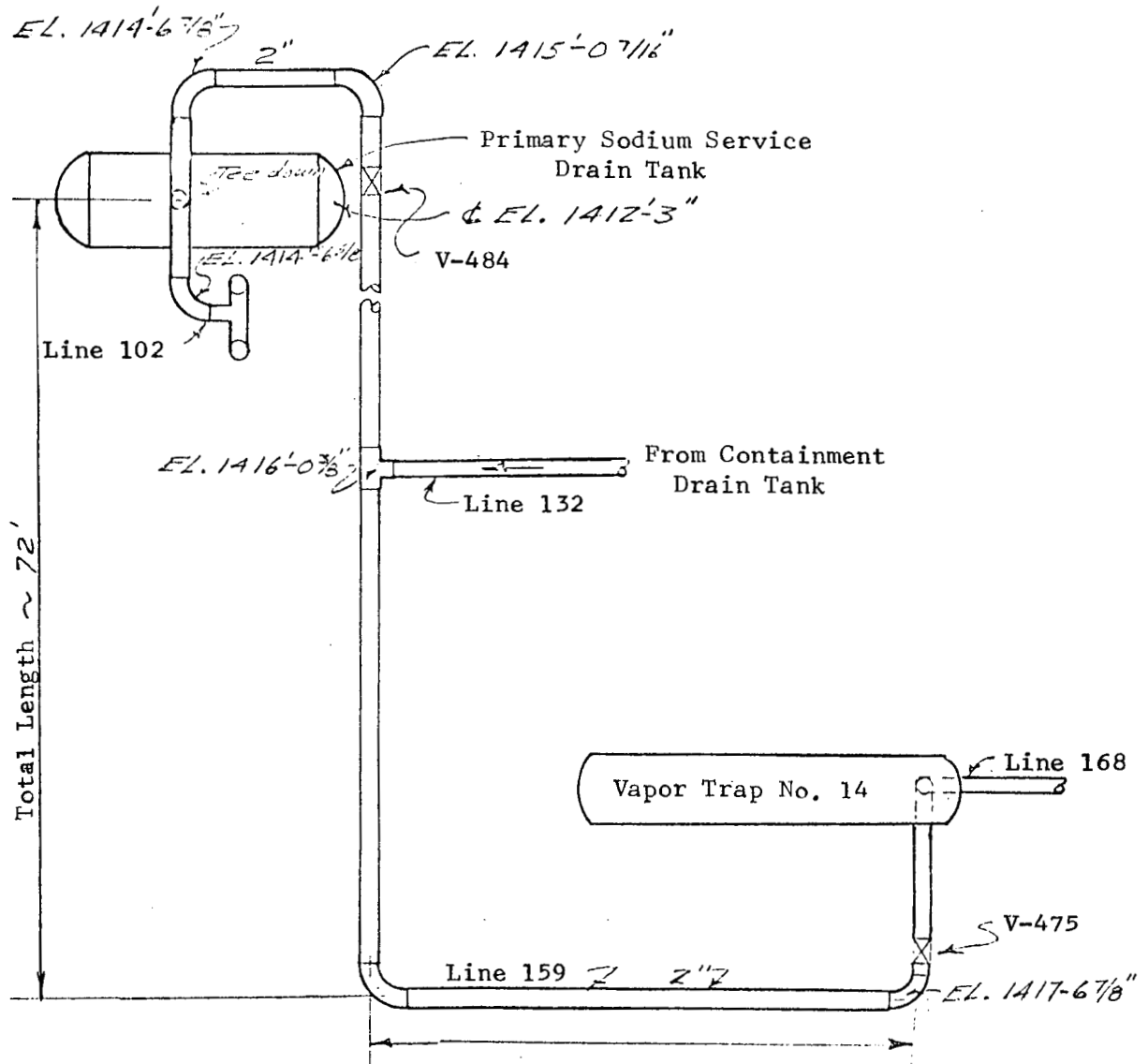
#### F. Reactor Atmosphere Pressure

A study was conducted of the problems encountered from May 11 - 14 concerning reactor atmosphere pressure control. In connection with this study sketches were drawn of the interconnecting helium supply line between vapor trap VT-14 and the reactor, and the drain line from VT-14 to the primary sodium service drain tank. These sketches, Figures II - 5 and II - 6, serve to illustrate the piping layout and also to show elevations of the various sections of piping.

The first sign of trouble was a difference in pressure indications of reactor atmosphere pressure indicator controller PIC-803 and primary fill tank atmosphere PIC-805, even though the vents of these two systems were interconnected. It soon became apparent that there was an obstruction in the line connecting the reactor and VT-14. Since the discharge from radiation detector RD-1 ties into the helium supply line this served to pump up the pressure in this section of piping. PIC-803 was indicating the pressure in the obstructed helium supply line rather than the actual reactor pressure. Actual reactor atmosphere pressure was indicated by PIC-805 and remained within its normal range during this period.

At this time valves V-475, V-484 and V-481 were opened. This served to drain VT-14 to the primary service drain tank and also interconnected reactor helium supply and reactor vent. The step level indication on the drain tank showed an increase in level at this time and PIC-803 and PIC-805 were in agreement. When any of the valves were closed pressure as indicated by PIC-803 again began to increase. This indicated that the obstruction was after the point where the VT-14 drain line ties into line 168.

The next step was to close V-484 and pressurize VT-14 to try to blow out the obstruction in line 168 by momentarily opening the bypass around CV-803. This was done and when the system was valved back to normal PIC-803 and PIC-805 were in agreement. It appeared at this time that the problem had been taken care of; however, the same situation arose



Plan View

Figure II - 5

VAPOR TRAP VT-14 DRAIN LINE

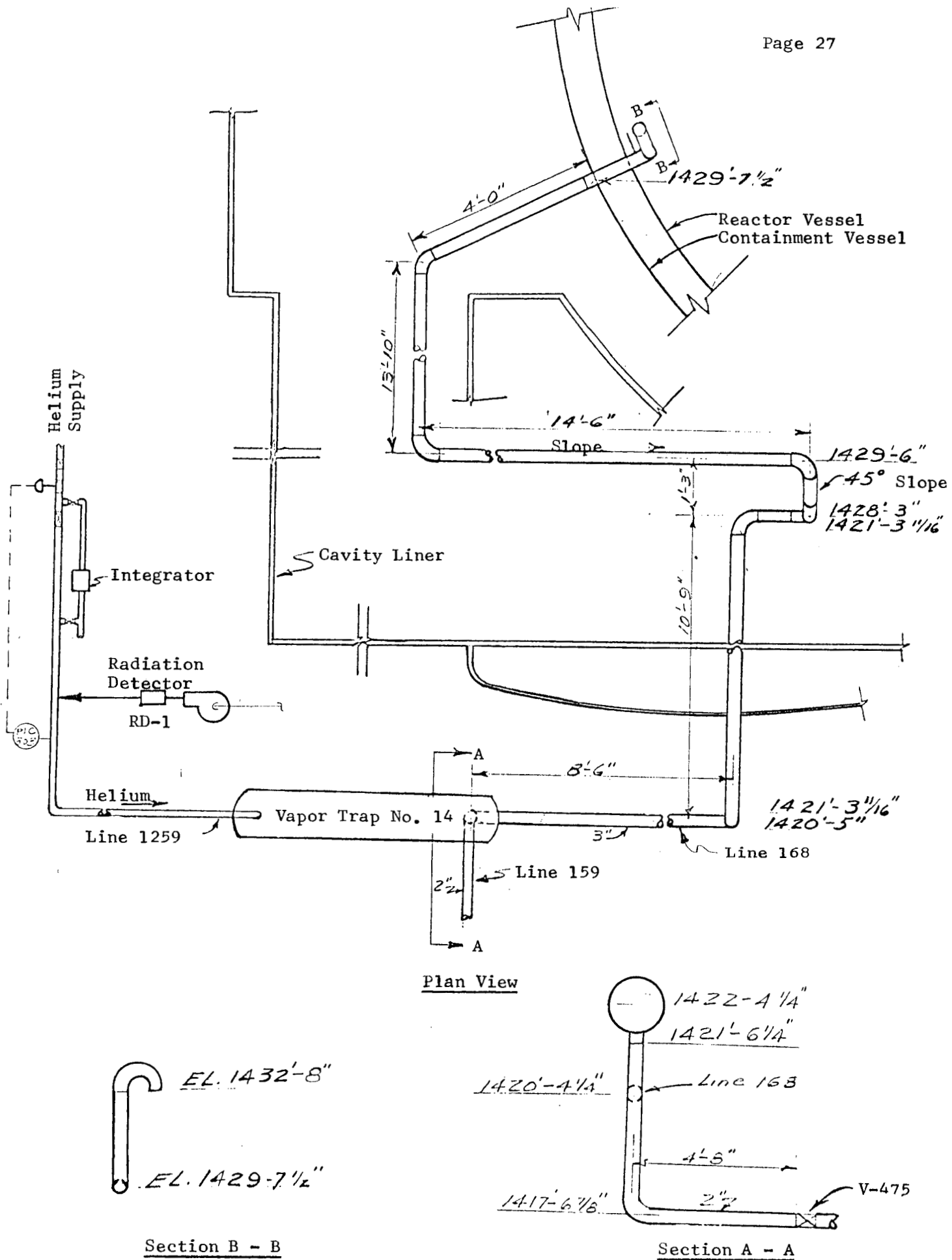


Figure II - 6

REACTOR HELIUM SUPPLY

again and line 168 had to again be blown clear. This was done five times over a period of several days. Since then the system has operated normally.

The question arises as to how sodium found its way into line 168. There are two possible routes. One is that sodium vapors from the reactor collected in line 168 and sodium from these vapors eventually blocked the line. This is very unlikely since due to radiation detector RD-1, there is a constant flow through this line in the opposite direction.

The more plausible answer is that sodium was forced back up the vapor trap drain line. This is supported by the fact that on May 11 just prior to the first indication of trouble the primary sodium service system was drained to the primary service drain tank. During the course of the SOP the drain tank was pressurized to 10 pounds to force the sodium into the primary fill tank. Since the pressure required to overcome the elevation difference between the drain tank and line 168 is about 3 pounds, the pressure used was more than sufficient to force sodium into line 168.

In order for sodium to be forced up line 159 from the drain tank, the tank has to be filled completely. Operations personnel indicated that they were quite careful to prevent the tank from filling completely, and that the high level alarm did not come in at any time. The alarm and the level indicators were checked by adjusting setpoints and appeared to be functioning properly. Even with this check it is still possible that the indicator is in error and the tank was actually filled. Another possibility is that there was sodium in the drain line above valve V-484 from some previous operation. Any helium that leaked past V-484 would force the sodium up this line. The sodium would have had to get past V-475 which was closed but it could very well be that this valve was actually not fully closed. It took strong arm methods to open this valve and since it is a bellows seal valve it appears that there was binding in the actuator mechanism. This binding could have prevented the valve from being fully closed.

When VT-14 was valved to drain, all the sodium apparently did not drain out. There may have been a low spot in the piping or a cold spot which prevented complete drainage. The most likely point would be at the tee just below the vapor trap. A cold spot at this point would block line 168 as sodium slowly collected at this point. By pressurizing the vapor trap some of the sodium may have been forced down the drain line thereby clearing line 168.

When the opportunity arises V-484 will be checked and the binding alleviated.

## G. Other Systems

The table below shows other plant data.

Table II - 7

## MISCELLANEOUS SYSTEMS DATA

## Make-up Usage:

Sodium

Primary: None

Secondary: None

Nitrogen Delivered: 609,900 scf

Helium Usage: 36,300 scf

## Steam Generator Evaporator Water Analysis:

## Typical Chemical Equilibrium Operating Conditions

	Steam Generator #1	Steam Generator #2	Steam Generator #3
Date	5/15/64	5/15/64	5/15/64
Time	1025	1025	1025
pH	10.4	10.2	10.4
Conductivity (mmho)	47.0	32.0	49.0
OH Alkalinity (ppm $\text{CaCO}_3$ )	6.0	5.0	8.0
Silica (ppm $\text{SiO}_2$ )	1.6	0.9	1.2
Hydrazine (ppm $\text{N}_2\text{H}_4$ )	0.0096	0.0073	0.0045
Phosphate (ppm $\text{PO}_4$ )	6.9	2.4	5.9

### III. Testing

#### A. Mechanical Test on Fuel Element Orifice Drive Assembly

##### 1. Purpose

The purpose of this test is to evaluate the performance of the orifice assembly while the orifice drive is operated both in the upward and downward directions and with the lower orifice drive shaft under restraint. The results of this test will be used to set allowable torque limits for orifice adjustment.

##### 2. Introduction

Some difficulty has been experienced with the orifice drive mechanism in the reactor. On several occasions, while orifice adjustments were being made, it was apparent that the orifice bulb was not moving and that the orifice assembly was sticking. Conditions which substantiate these conclusions are:

- 1) The fuel outlet temperature did not change when the orifice position indicator was moved.
- 2) The orifice drive assembly did not move freely.

In order not to damage the orifice assembly, the following torque limits were imposed:

- 1) Maximum upward torque (tension) of 75 lb-in.
- 2) Maximum downward torque (compression) of 300 lb-in.

This test was designed to establish the maximum allowable torque which could be applied, in both directions, without damaging the orifice assembly.

##### 3. Method

A spare, unused reactor plug (RP-200) was mounted vertically in a support frame over pit No. 19. The orifice drive shaft was restrained by placing a steel pin horizontally through two holes in the lower end of the hanger tube. Prior to the tension test

the orifice was lowered until the bottom spacer on the orifice drive shaft was below the restraining pin location and the pin was then inserted. As the orifice was driven upward, the pin prevented the spacer from moving. For the compression test the spacer was initially positioned above instead of below the pin.

#### 4. Results

##### a. Tension Test (May 5, 1964)

A torque wrench was used to drive the orifice upward, incrementally, in the following sequence:

- Step 1 100 lb-in. of torque applied
- 2 torque released
- 3 150 lb-in. of torque applied
- 4 torque released
- 5 200 lb-in. of torque applied
- 6 torque released
- 7 250 lb-in. of torque applied
- 8 torque released
- 9 300 lb-in. of torque applied
- 10 350 lb-in. of torque applied
- 11 torque released

The deformation of the orifice drive shaft and of the lower spacer was observed throughout these steps and did not appear to be excessive. The following observations were made during this portion of the test:

Steps 1 and 2: The 100 lb-in. torque could be applied and released by hand.

Step 4: The 150 lb-in. torque was released by using the wrench for the first two 1/4 turns, then released by hand.

- Step 6: The 200 lb-in. torque was released by using the wrench for the first eight 1/4 turns, then released by hand.
- Step 8: The 250 lb-in. torque was released by using the wrench for the first thirteen 1/4 turns, then released by hand.
- Step 11: The 350 lb-in. torque was released by using 275 lb-in. on the wrench. After the restraining pin was removed, 275 lb-in. torque was required to move the drive assembly in either direction.

The upper drive assembly was then removed.

Visual inspection of the threads on the brass shaft assembly showed evidence of galling.

b. Compression Test (May 6, 1964)

The upper drive assembly was removed from spare reactor plug RP-104 (which has a damaged orifice drive shaft) and the stainless steel nut and brass shaft assembly were installed into RP-200. Molybdenum disulfide lubricant was applied to the orifice drive threads. A torque wrench was used to drive the orifice downward, incrementally, in the following sequence:

- Step 1 100 lb-in. of torque applied
- 2 torque released
- 3 150 lb-in. of torque applied
- 4 torque released
- 5 200 lb-in. of torque applied
- 6 torque released
- 7 250 lb-in. of torque applied
- 8 torque released

The deformation of the orifice drive shaft and of the lower spacer was observed and did not appear to be excessive. The drive shaft did not appear to be bowed. The torque required to release the orifice drive was approximately the same as that observed in the tension test for the same torque.

During this test the strain on the lower extension of the flexible cable was measured by placing a pencil mark on the rod at the bottom of the shield plug with the orifice lowered but with no torque applied. This mark was displaced downward from the shield plug as follows:

Step 1 1.0 mm with 100 lb-in. torque applied

Step 3 1.5 mm with 150 lb-in. torque applied

Step 5 1.3 mm with 200 lb-in. torque applied

Step 7 1.3 mm with 250 lb-in. torque applied.

These strain measurements show that the maximum force on the orifice drive shaft occurred at Step 3 when 150 lb-in. of torque was applied. After that time much of the torque was transmitted only to the drive threads and less stress was applied to the lower drive shaft than in Step 3.

## 5. Summary

These tests show that at least a portion of the orifice sticking problem is attributable to galling of the threads on the brass shaft assembly.

Further testing is planned.

B. Sodium Oxide Scrubber Test

1. Purpose of Test

To verify the operability of the sodium oxide removal system.

2. Results

The sodium oxide scrubber was tested in accordance with SOP-5806 on May 15. No malfunctions occurred and the test sequence proceeded automatically upon actuation from the control room.

3. Conclusion

The sodium oxide scrubber system is operable upon call.

### C. Control Rod Drop Test

A control rod drop time test was conducted on May 19. This makes five such tests that have been run since the control rod thimbles were changed.

#### 1. Purpose of Test

The test was conducted to again determine if moderator can ruptures are having any effect on control rod dropping capability. Indications received from operation of the reactor on May 15 suggested that another moderator can may have ruptured and it was desired to see if this had any effect on control rod drop times.

#### 2. Presentation and Discussion of Data

Table III-1 shows the data obtained on May 19, 1964. As can be seen, all the times are within Technical Specification values. Also shown on Table III-1 are the arithmetic mean value (rounded to nearest millisecond) and range (largest minus smallest value) for the three time quantities for each rod. The mean and range values are based on all five sets of test data that have been obtained since control rod thimbles were changed including the data obtained on May 19, 1964. It can be seen from the table that, generally speaking, the data of May 19, 1964 agree with mean values to within  $\sim 1/2$  or less the range for that time value for that rod. The greatest spread in data remains 20 milliseconds as was true in April.

#### 3. Conclusion

The data obtained in May again demonstrates that the rupture of moderator cans has had no measurable effect on control rod drop time. The data shows that rod drop times have been unaffected (within statistically reasonable limits) even by an increasing number of ruptured moderator cans present in the reactor core.

Table III - 1

CONTROL ROD DROP TIME TEST  
(All Times in Milliseconds)

Control Rod No.	Core Position	Test Data 5/19/64			Five Test Statistics					
		$\Delta T_1$	$\Delta T_2$	$\Delta T_3$	$\Delta T_1$		$\Delta T_2$		$\Delta T_3$	
					Mean	Range	Mean	Range	Mean	Range
1	C-1	55	781	975	53	3	779	3	976	4
5	C-38	55	788	964	55	4	789	3	966	8
6	C-16	51	779	954	53	4	781	5	958	12
7	C-44	58	773	959	56	7	774	9	962	9
8	C-50	53	766	943	53	4	771	12	951	17
9	C-24	55	788	973	55	4	787	8	977	11
10	C-56	53	781	960	55	6	782	6	961	10
11	C-62	51	770	956	53	6	772	5	960	8
12	C-32	52	767	950	53	2	770	7	955	12
13	C-68	59	788	962	56	5	789	2	962	4
14	C-78	58	790	970	58	4	794	12	975	13
15	C-86	58	781	963	57	6	785	17	968	20
16	C-94	55	774	962	57	4	776	5	964	4
17	C-102	54	779	957	55	3	781	10	960	13
18	C-110	54	769	957	54	6	771	7	961	14
19	C-118	54	769	955	54	1	772	14	958	17
Technical Specifications		45- 60	$\leq 803$	900- 1015						

$\Delta T_1$  = Time from magnet de-energized to when the "rod latched" switch breaks contact

$\Delta T_2$  = Time from magnet de-energized to 9 feet of free fall

$\Delta T_3$  = Time from magnet de-energized to full snubbed position

D. Area Fire Alarm Test

1. Purpose

The purpose of the test was to operate the four area fire alarm horns and to adjust the four horns to desired audible output.

2. Method

Site personnel were notified by loudspeaker that the test of the area fire alarm system would be made.

During the sounding of the fire alarm, maintenance personnel adjusted each of the four horns to desired audible output.

Refer to the HNPf Monthly Operating Report, No. 21, April, 1964, page 49.

3. Conclusions

- a. The area fire alarm system is operable on call.
- b. The four horns located in pairs on top of the reactor building are each sounding with satisfactory volume.

#### IV. Radiation Control

##### A. Radiation and Contamination Surveys and Monitoring

The plant site environmental air sampler filter was changed and counted on a weekly schedule during May. Initial counting of these filters was performed and followed by recounting at 4 hour and 24 hour decay times. Results of the initial counting of these continuous one week collections ranged from  $5 \times 10^{-12}$  to  $9 \times 10^{-12}$   $\mu\text{c/cc}$  beta-gamma and from  $6.9 \times 10^{-13}$  to  $9 \times 10^{-13}$   $\mu\text{c/cc}$  alpha. These initial activity levels and their associated decay characteristics are within normal limits for environmental airborne particulate samples in this area.

The main building and radioactive waste facility stack monitor filters were also changed and counted weekly. No activity in excess of normal atmospheric airborne particulate levels was detected.

Radiation surveys of reactor vent system components (CV-1004 and FI-849) were performed weekly and RD-1 in-line filter housing was surveyed at the time of each reactor cover gas sampling. No evidence of sodium carry-over was detected.

On May 1 and 4 re-surveys of the primary sodium service vault were performed and Health and Safety coverage was continued for valve V-476 bellows replacement work. Working area radiation level was 6 mr/hr. Smear surveys were performed within the cell periodically as work progressed. Maximum contamination on exterior surfaces of piping and deck areas was 52 d/m/100  $\text{cm}^2$  which was on the bonnet of valve V-476. Smear surveys of the area following completion of the work showed no surface contamination in excess of 30 d/m/100  $\text{cm}^2$ . On completion of this work the plug was re-installed and the system returned to service.

On May 12 Health and Safety coverage was provided for momentary removal of the primary sodium service vault access plug to permit remote

visual inspection of valve V-476 following an alarm of the sodium leak detector in that valve. No entry into the cell was made and nitrogen atmosphere in the cell was maintained during this inspection. No external evidence of sodium leak was observed. Radiation level at elevation 1440 ft 6 in. floor level over the open access hole was 2.5 r/hr. Maximum personnel exposure for this operation was 5 mrem. Radiation monitoring was repeated in the same manner on May 13, 15 and 19 in connection with re-inspections and to assess the effectiveness of isolation of the system from the main heat transfer loops.

On May 20 the primary sodium service cell was entered and a radiation survey performed prior to inspection and repair of the valve V-476 sodium leak detector. Maximum radiation level was 155 mr/hr (contact) at line 149. The radiation level in the working area at valve V-476 was 15 mr/hr.

On May 4 and 6 measurements of radiation levels in the primary pipe vault were performed using the west instrument plug. The purpose of these measurements was to confirm that such measurements could be performed safely at maximum activation conditions if required, and to obtain additional reference data for future use. A vertical traverse through the depth of the pipe vault showed maximum gamma radiation levels at elevation 1423 ft to be 375 mr/hr on May 4 approximately 10 days after reactor shutdown of April 24, and  $10^3$  r/hr on May 6 at approximately 50% equilibrium activation for 20% reactor power level.

On May 18 the third shipment of Core II fuel rods was received at the plant site. Smear surveys of the truck and exterior surfaces of the shipping boxes disclosed no contamination in excess of 30 dpm/100 cm<sup>2</sup> (total). No alpha contamination was detected. Radiation surveys of the shipment and individual boxes were performed. No radiation in excess of twice normal background level was detected.

A radiation survey of the accessible major components of No. 2 secondary sodium loop and areas of steam generator room No. 2 was performed on May 18 in response to a high radiation alarm indicated by the No. 2 secondary sodium radiation monitor (RM-200). No elevated radiation levels were detected. Investigation of the detector disclosed an electronic component failure, repair of which cleared the alarm.

A radiation survey of the primary fill tank vault was performed on May 21 in preparation for inspection and repair of tank No. 3 pedestal heaters and replacement of a rupture disc between the tank vault and the primary pipe vault. Maximum radiation level was 4 mr/hr. Air breathing apparatus was required for entry due to the nitrogen atmosphere hazard existing prior to and during the rupture disc replacement.

Periodic smear surveys and area radiation surveys were performed throughout the plant areas. No contamination in excess of 30 dpm/100 cm<sup>2</sup> was detected outside of control barriers. No abnormal radiation levels were detected in areas of routine access by plant personnel.

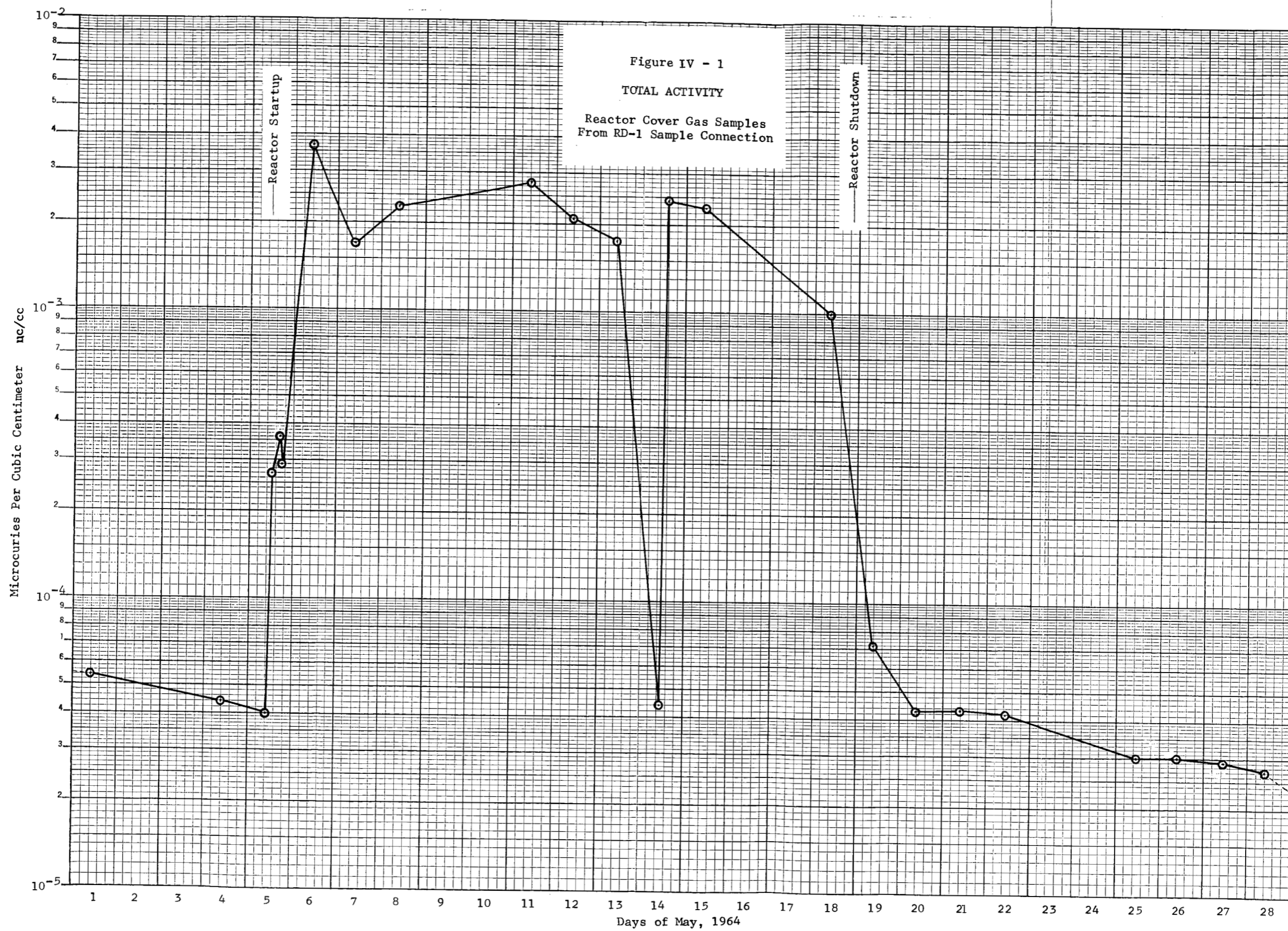
Additional radiation surveys, smear surveys and monitoring operations were performed in connection with items listed in Table IV - 1 and item D-"Other Health and Safety Activities" of this section.

#### B. Sampling and analysis of process systems

During May one gas decay tank was sampled prior to release. Activity of this sample was  $1.25 \times 10^{-6}$   $\mu\text{c/cc}$ . The tank volume was released via the main building stack at a rate of less than 50 scfm.

Twenty-four samples of reactor cover gas were obtained from the RD-1 sample connection for counting (Figure IV - 1) and gamma-spectrum analysis (Figure IV - 2). Maximum sample activity was  $3.7 \times 10^{-3}$   $\mu\text{c/cc}$ .

The Xenon-133 peaking following reactor startup and the decline of constituent activities after scram agree with previous observations. No appreciable change in fuel cladding integrity is indicated.



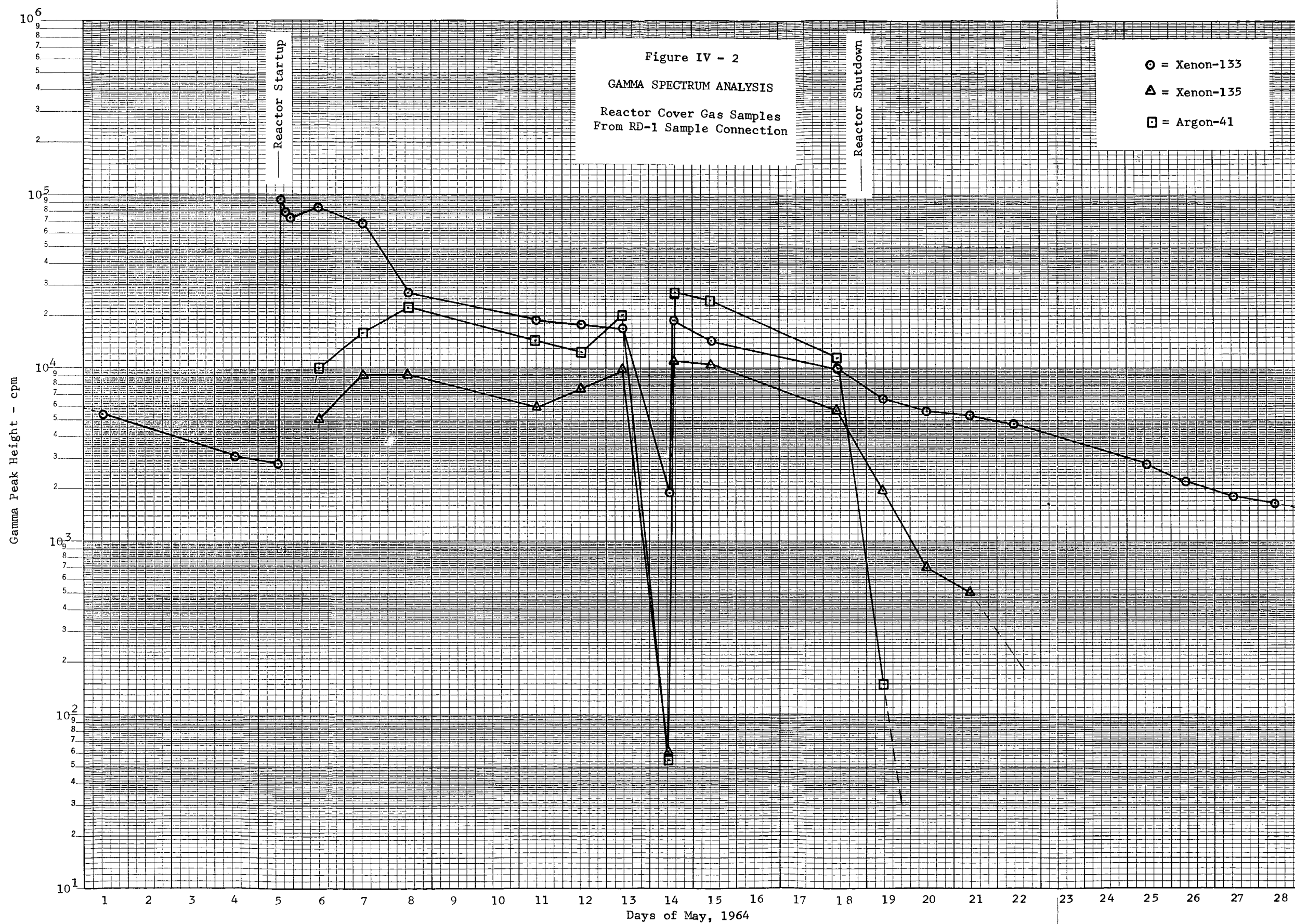


Table IV - 1

RADIATION SURVEYS  
OFF-NORMAL CONDITIONS OR ACTIVITIES

Location	Radiation Level (mrem/hr)	Contamination (dpm/100 cm <sup>2</sup> )
Primary sodium service vault valve V-476 (bellows) repair	6 (working area)	52
Primary sodium service vault - initial valve inspection by remote visual observation	1.9 x 10 <sup>5</sup> (max) 1.2 x 10 <sup>3</sup> (at aux bay floor level)	-- --
Primary sodium service vault. Repair of valve V-476 leak detector	155 (max) 15 (working area)	-- --
Steam generator room No. 2 -- investigation of RM-200 alarm	Background	--

During the period of May 11 through 14 intermittent draining of sodium from vapor trap No. 14 produced major disturbances of reactor cover gas circulation through RD-1. These flow disturbances undoubtedly affected the validity of sample data during that period.

On May 8 gas samples were collected from the area of the upper "O" ring seal of stationary control rod No. 4 thimble, where leak checking had detected the presence of a helium leak. Sample activity was  $3 \times 10^{-5}$   $\mu\text{c/cc}$  due principally to Argon-41. The absence of Xenon confirmed that the leak was from the control rod thimble atmosphere and not reactor cover gas. Repair of the seal terminated the leak.

Seven water samples were obtained from the closed loop cooling water system and counted prior to the performance of chemical analyses. Maximum sample activity was  $6.5 \times 10^{-9}$   $\mu\text{c/ml}$ .

The normal level R/A liquid waste storage tank was sampled in preparation for release to the leaching ponds. Sample activity was less than  $3 \times 10^{-6}$   $\mu\text{c/ml}$  (total) and no alpha activity was detected.

#### C. Health Physics Instruments

During May, backgrounds of all counting room instruments were counted and recorded daily. Periodic functional and calibration checks were performed.

All survey meters were checked for functional abnormalities and calibration using encapsulated calibration sources.

#### D. Other Health and Safety Activities.

Health and safety coverage was provided for the following work items:

1. Loading face shield nitrogen compressor shaft seal replacement.
2. Removal of reactor plugs RP-104 and RP-200 from storage and inspection of orifice plug drive mechanism.

3. Primary sodium service system valve V-476 bellows and sodium leak detector repair.
  4. Nitrogen cooler No. 8 blower bearing repair.
  5. Control rod actuator drive shaft seals helium leak check and repair of No. 14, 15, 17 and 19.
  6. Control rod drop time testing.
  7. Reinstallation of helium sample connections on stationary control rods.
  8. Fuel handling machine load-cell testing and re-setting.
  9. Fuel storage pit thimble replacement.
  10. Removal of blind flange from closed loop cooling water system -- fuel storage pit No. 3 circulation line No. 1510.
  11. Removal and inspection of orifice plug drive mechanisms from stored reactor plugs RP-130 and 183.
  12. Removal of plastic covering from pit No. 19 and packaging of waste.
  13. Removal and repair of stack monitor pump.
  14. Repair of stationary control rod No. 4 helium atmosphere seal.
  15. Decontamination of tools and equipment.
- Solid radioactive waste was packaged periodically for storage.

#### E. Radioactive Waste

The May inventory, shipment and disposal of radioactive wastes are tabulated in Table IV - 2.

Table IV - 2

## RADIOACTIVE WASTE: INVENTORY, SHIPMENT AND DISPOSAL

## Radioactive Waste

<u>Inventory</u>	<u>Quantity</u>	<u>Activity (Curies)</u>
Solid (cu ft)	$2.3 \times 10^2$	$4.23 \times 10^{-2}$
Liquid (gal)	$4.5 \times 10^3$	$6.86 \times 10^{-4}$
	$1.3 \times 10^3$	*
Gas (scf)	$1.2 \times 10^2$	*

Shipments

Solid (cu ft)	None	---
Liquid (gal)	None	---
Gas (scf)	None	---

Other DisposalDisposal Method

Solid (cu ft)	None	---	---
Liquid (gal)	$4.02 \times 10^3$	$4.04 \times 10^{-5}$	Released**
Gas (scf)	$9.3 \times 10^3$	$3.29 \times 10^{-4}$	Released***

\* Tank volumes in process of accumulation. These had not been sampled; therefore activity levels had not yet been determined.

\*\* Liquid waste released to leaching field.

\*\*\* Gaseous waste released to atmosphere via main building stack.

## V. Maintenance and Modification

### A. Primary Vault Maintenance

The ruptured disk from the primary fill tank cell to the primary pipe way was replaced. Previous cell pressure tests had indicated the atmospheres of the two areas were common, as noted in the HNPF Monthly Operating Report, No. 18, January, 1964, page 47. The rupture disk had failed. The other rupture disc (pipe tunnel to fill tank area) was inspected and no visual cracks were observed. There was no nitrogen leakage noted from a higher pressured pipeway through the rupture disc into the primary fill tank area.

Power to the pedestal heaters on No. 3 primary fill tank was interrupted when the circuit breaker tripped and could not be reclosed indicating a fault. The fault was found in the junction box on the fill tank. After repair the heaters were placed in service.

### B. Control Rod Drive Actuators

To decrease helium loss seals were adjusted on control rod actuators of CR-13, 14, 15, 17 and 19.

The drive in limit switch on actuator No. 4 of CR-16 was repositioned, as it had slipped out of adjustment, rendering it inoperative.

### C. Sodium Valves and Leak Detectors

Valve V-476 bellows seal had ruptured and was replaced. The sodium block valve is located on No. 2 EM pump discharge. A new leak detector was installed in the bellows. When the valve was brought to operating temperature the leak detector went into alarm condition. Further inspection showed the bellows had not leaked, but the leak detector had faulted due to moisture in the magnesium oxide insulation. The leak detector was replaced with one which had been dried in a drying oven at 175 F for 96 hours.

A leak detector alarm occurred on primary sodium service drain tank valve V-481, when the valve was opened to full open position. By closing the valve one turn the alarm cleared. Inspection showed the leak detector apparently was placed too near the valve stem. A new leak detector was installed, as the original was damaged in removal.

A leak detector alarm was received on No. 2 primary sodium block valve. A check of circuitry indicated an erroneous alarm due to one electrode being grounded to the sheath. Transposition of the two leads cleared the alarm. Further investigation will be made when the primary pipe way is accessible.

#### D. Waste Facility Stack Monitor

The radioactive waste facility radiation monitor pump failed due to frozen bearings. The bearings were replaced with bearings containing high temperature grease.

#### E. Miscellaneous Maintenance

Removal and cleaning scale from the raw water side of the steam dump was started.

Rollo-matic filters on all heating and ventilating units were replaced this month.

The shaft seals were replaced on No. 2 loading face shield compressor as leakage increased beyond capacity of one recompressor.

The diaphragm was replaced on the helium block valve V-5158 to No. 3 evaporator third fluid system. The diaphragm had ruptured.

## F. Instrument Maintenance

## 1. Plant Control System

## a. Reactor and nuclear instrumentation

Channels V, VI, VII, VIII and IX flux amplifiers were recalibrated.

A wiring error in the rectifier circuit of the log N and period amplifier was causing spurious voltage regulation conditions. This resulted in sporadic short period alarm and power level shifts. The rectifier circuit was rewired.

Log count rate meter S/N 463 was giving an unstable count rate indication and a noisy period. A loose connection was found on the chassis.

Log count rate meter S/N 407 would not calibrate within specifications. Two tubes were replaced and the unit was recalibrated and installed in Channel II.

Erratic operation of the nuclear power computer was caused by poor contacts in auto-manual switch of HCS-6. The switch was replaced.

Temperature recorder TRC-1 was calibrated.

Calibration of the measured pen was checked on temperature recorders TRC-1, TRC-2 and TRC-4.

## b. Heat transfer system

Temperature indicator TI-22 was reading low. The recorder was recalibrated.

All primary and secondary sodium flow recorder controllers were calibrated.

All primary reactor inlet sodium temperature recorder controllers were calibrated.

### c. Feedwater and Steam System

The square root converters of all feedwater and steam flow recorders were calibrated.

Differential pressure cell transmitters of loop 1 steam and feedwater flow were calibrated.

The low level scram alarm from level indicator LXL-250 differential pressure cell transmitter was coming in at 9.8 inches instead of 8 inches. The transmitter was recalibrated.

All steam and feedwater flow differential cell transmitters were rezeroed.

### 2. Sodium Instrumentation

Pressure indicators PI-110 and PI-111 were rezeroed as they were giving high indications.

Secondary sodium expansion tank level indicators LI-100, LI-200 and LI-300 were calibrated. Tubes were replaced in LI-100 and LI-300.

Tubes were replaced in No. 2 secondary fill tank sodium level indicator, to restore the proper gain in the amplifier system.

### 3. Radiation Detection and Monitoring System

Check valves were changed on the pump unit of radiation detector RD-1 because of low flow.

Count rate meter RIC-1102 in the radioactive waste facility had stopped counting. The selenium rectifier was replaced with a silicon diode. A filter capacitor and two tubes were also replaced.

The high voltage transformer of the portal personnel monitor near the Health Physics office was replaced because it was drawing excessive current and blowing fuses.

Relay contacts on the portal personnel monitor to the high bay were cleaned. No. 6 window had not alarmed during the monitor checkout.

The constant air monitor, CAM, in the decontamination room was not giving a counting indication. Tubes were replaced in the high voltage power supply and in the count rate meter, and connections were repaired in the preamplifier.

#### 4. Health Physics and Survey Instruments

Batteries were replaced in the following instruments:

Eberline E-500A GM counter  
Radector portable ion chamber at the guard house  
Juno S/N 2331  
Portable GM counter at the guard house

A GM tube was shorted and replaced in the portable GM counter Nuclear Chicago 2612 S/N 2536.

#### 5. Preheat System

Replacement of RT relay in system A recorder was made because of relay failure.

High temperature mercury control switches were replaced in systems D and H recorders.

A gear was replaced in system J recorder because the chart drive was slipping.

Contacts were replaced in relay E-29.

Slide wire contacts were replaced on system D recorder.

#### 6. Other Instrument Maintenance

The transmitter of flow indicator 1001 in the radioactive vent system was checked for calibration.

Pressure indicator controller PIC-803T, reactor atmosphere transmitter in the helium system was checked for calibration.

Oxygen analyzer AnO<sub>2</sub>900 was calibrated on May 5 and May 20.

Grapple torque units on the fuel handling machine were calibrated and alarms were set.

A resistor was replaced on No. 2 grapple alarm controller.

## G. Modifications

### 1. Maintenance Cell Modification

Contract work on the maintenance cell modification for the fuel examination program was nearing completion. The modification, which provides additional in cell non-destructive test equipment is designed to be remotely movable and replaceable.

Fuel distortion, length and diameter, will be measured, and results will provide an indication of remaining fuel life.

Figure V - 1 shows the maintenance cell with existing equipment detailed on the right side of the figure. Equipment added in this modification are outlined to the left of the figure.

A fuel rod stage assembly was installed in the cell at the first (upper) operating level. The stage will be utilized to raise, lower and otherwise locate the fuel rod for advantageous position during the examination, measurement and inspection program. At the end of this report period, the stage assembly was scheduled for wiring modification.

The modifications at the second level which were all contractor completed at the end of this report period, are listed;

- 1) One additional master slave manipulator
- 2) Periscope for view and photography, with camera
- 3) Cutting machine to cut the hanger casting of the fuel element assembly
- 4) Fuel rod diameter measuring device with recorder
- 5) Fuel rod position transmitter device with digital recorder
- 6) Gamma scanner with through tube and two pin hole cameras

Selected uranium molybdenem premeasured fuel elements of Core I are scheduled to be examined at some future time.

# MODIFICATION ADDITIONS

## FIRST LEVEL

STAGE ASSEMBLY

## SECOND LEVEL

MASTER SLAVE MANIPULATOR

PERISCOPE AND CAMERA

CUTTING TOOL

ROD DIAMETER GAUGE, RECORDER

ROD POSITION TRANSMITTER, RECORDER

GAMMA SCANNER AND CAMERAS

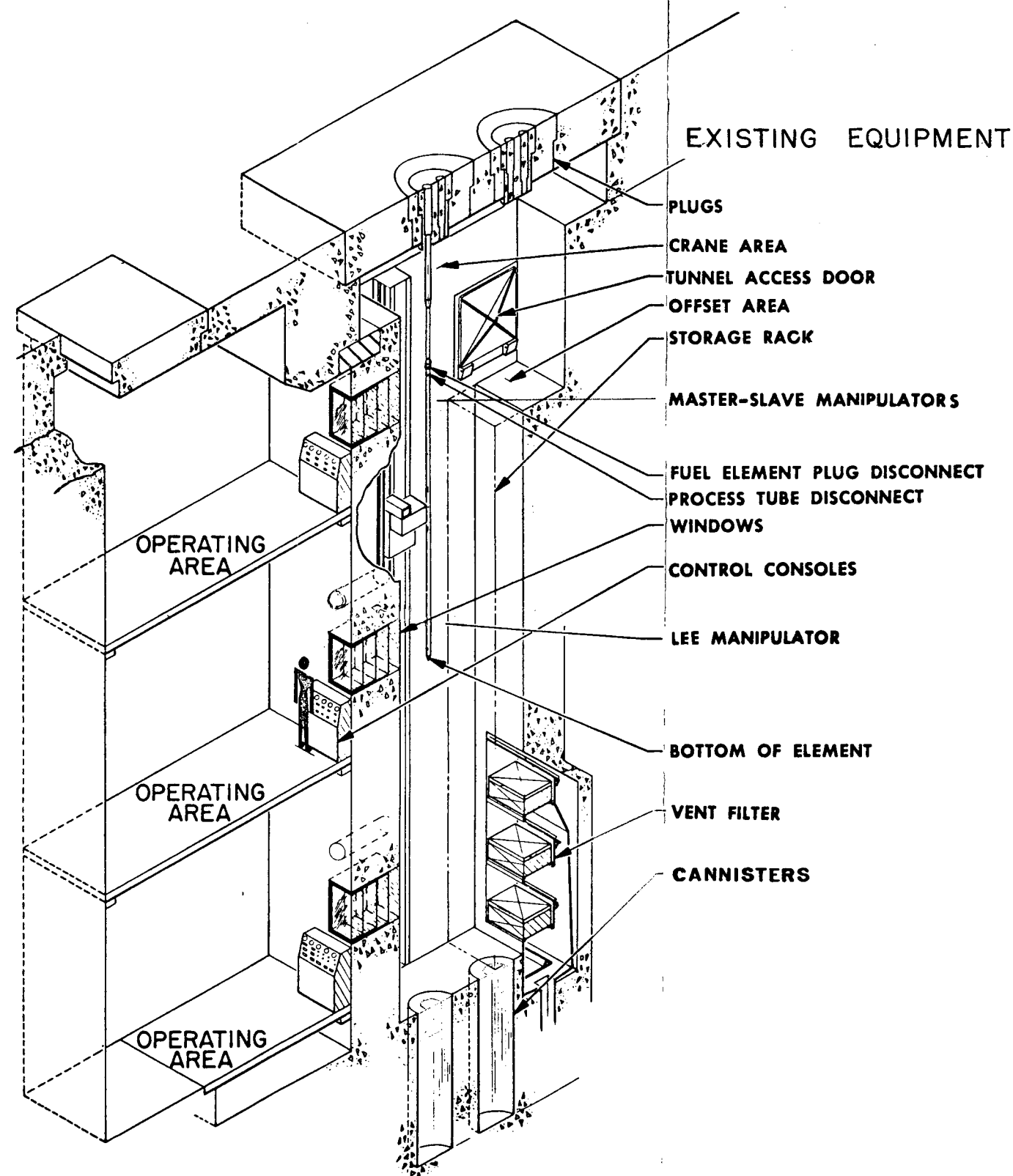


Figure V - 1

Maintenance Cell and Operating Areas

2. Modifications to the Reactor Plant Heating and Ventilating System

During the week of May 4 a representative from Atomics International was on site to check the operability of the modified heating and ventilation system. The modifications, as reported in the February Monthly Operating Report, No. 19, page 81, consist of a booster fan in the exhaust ductwork in the auxiliary bay, a new cooling duct in the secondary sodium service area, and ductwork to each of the main sodium pumps.

### 3. Secondary Sodium Pump Helium Supply Modification

Upon arrival on site of the required valves the modification of the helium supply piping to the secondary sodium pumps was completed. The modified system provides a separate helium supply line to the lower oil sump area of each secondary pump. Figure V - 2 depicts schematically the typical arrangement of the piping system at each pump.

The necessity for this modification is explained in the October, 1963, Monthly Operating Report. Essentially, the separate helium supply to the lower oil sump area assures that the pressure in this area will always be the same as that in the barrel section of the pump.

Under these conditions sodium cannot be forced up the shaft annulus and into the lower oil sump area as occurred in October of 1963, and recorded on page 44 of report No. 15.

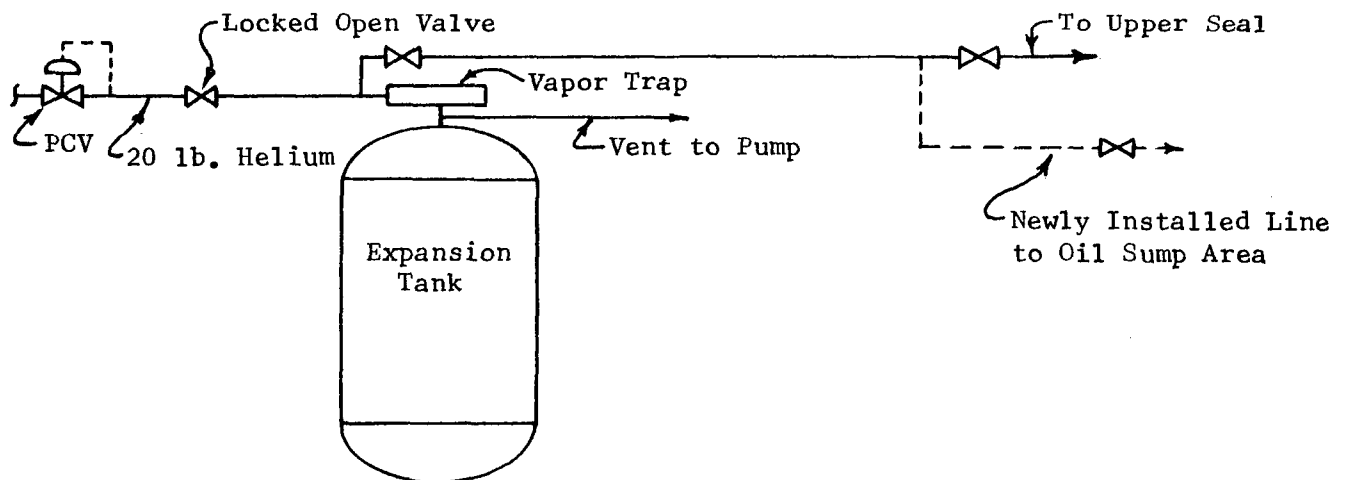


Figure V - 2  
SECONDARY SODIUM PUMP HELIUM  
SUPPLY MODIFICATION

## VI. Training Program for Senior Operator Licenses

In preparation for the Atomic Energy Commission's examination for Senior (Reactor) Operator License in May thirteen applicants attended a one hour a day, six week refresher course during January and February of 1964. Several other Consumer employees audited this course, which covered mathematics, nuclear physics, health physics and safety. This course was taught by Sheldon Station Staff personnel.

A formal study course was initiated on February 17. The two hour per day, 5 day a week class periods were scheduled for eleven consecutive weeks and were conducted by an Atomics International Training Instructor. Some of the classes were taught by Consumers personnel.

Considerable home work and outside study together with periodic review tests were included in the training program. In addition to the Senior Operator Applicants several additional Consumer employees and other site personnel also audited the course. An outline of the course is listed on Table VI - 1.

On May 11, twelve Consumers personnel, listed below, successfully completed the examination and were issued Senior Operator Licenses, effective June 1, 1964:

- Chemical Engineer
- Health Physicist
- Maintenance Supervisor
- Office Engineer
- Performance Engineer
- 4 Shift Supervisors
- 3 Unit Operators

One of the performance engineers, not previously licensed, passed the written Senior Operator License examination and will be required to take an oral examination plus a written operator examination at a future date.

Four of Sheldon Station staff personnel (the Plant Superintendent, the Assistant Plant Superintendent for Nuclear Engineering, one Shift Supervisor, and the Reactor Engineer) had received their Senior Operator License at an earlier date.

Table VI - 1

## STUDY COURSE OUTLINE

Senior Operator License Training Program  
Sheldon Station -- Hallam Nuclear Power Facility

Mathematics and General Physics Review  
Atomic and Nuclear Physics  
Interaction of Radiation with Matter  
Radiation Units, Shielding and Dose Calculation  
Radiation Detection Devices and Nuclear Instrument Theory  
Radiation Protection Regulations  
Biological Effects of Radiation  
Health Physics Procedures and Radioactive Waste Handling and Disposal  
Introduction Reactor Technology  
The Four Factor Formula  
Introduction to Reactor Kinetics Topics  
Criticality Computation, Critical Mass Determination  
In Hour and Rod Calibration Curves  
Xenon and Samarium Poisoning  
Subcritical Multiplication Effects  
Flux Traces and their Interpretation  
Fuel Loading, Handling Procedures and Precautions  
Moderator Can Handling  
Administrative Procedures, Limits, Technical Specifications  
Specific Operating Characteristics of Reactor and Auxiliary Systems  
and Hazards Analysis  
Review of Radiation Detection and Monitoring System Operation  
Review of Radioactive Gas Handling System  
Review of Radioactive Liquid Waste System  
General Review of HNPF Systems  
Fuel Cycle, Uranium Carbide Fuel  
Reactor Procedures  
Facility Authorization

Errata for HNPF Monthly Operating Report No. 20, March, 1964

Page 10, line 5, Gross Electrical Power (Mwe) should read:

28.9

Page 10, line 7, Net Electrical Power (Mwe) should read:

26.2

Page 10, line 8, Net Plant Heat Rate (Btu/Kwh) should read:

12,000

Errata for HNPF Monthly Operating Report No. 21, April, 1964

Page 14, line 15, Reactor Turbine Service To Date (hrs) should read

4,705.2

Page 14, line 19, Gross Reactor Electrical Generation (Mwh)

This Year should read:

59,371