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HALLAM NUCLEAR POWER FACILITY

MONTHLY OPERATING REPORT

NO. 26

SEPTEMBER, 1964

Contract No. AT(11-1)-513

SHELDON STATION
CONSUMERS PUBLIC POWER DISTRICT

CONTRIBUTORS

to

HNPF MONTHLY OPERATING REPORT NO. 26

G. L. Ballew	L. E. Keller
R. E. Buntain	T. M. Kyster
L. J. Cooper	M. C. Mauderly
R. H. Davidson	D. E. Monselle
A. J. Divis	R. L. Robb
H. V. Franklin	L. H. Sohl
R. E. Wilbur	

R. E. Durand - Technical Editor
I. O. Sunderman - General Editor

MONTHLY OPERATING REPORT

No. 26

September, 1964

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I. Plant Operating History

A. Summary of Operations, September, 1964

At the beginning of the period the reactor was carrying the entire Sheldon Station load. On September 2 a critical power breaker failure scrammed the reactor, which in turn resulted in a short turbine outage.

The reactor was restarted on September 2 and carried the station load until September 10. Due to the failure of moderator element M-51 on the evening of September 9 the conventional boiler was put in service in parallel with the reactor.

The conventional boiler tripped and the reactor scrammed on September 10 due to a temporary loss of instrument air in the conventional portion of the plant. The boiler was refired and the turbine was back on the line in eight minutes. The reactor was restarted and operated in parallel with the boiler until the evening of September 11 when the boiler was placed on hot standby.

The reactor carried the station load until September 15 when the boiler was returned to service in parallel with the reactor. This was done to use up coal in the bunkers in preparation for the plant October 4 outage. Burning of coal in the bunkers was completed on September 16; the boiler was placed on cold standby and the reactor carried the station load until September 22 when an electrical power failure off-site caused it to scram. The reactor was restarted and carried the station load until September 27. The reactor was shut down to run rod drop time tests in connection with the failure of moderator elements M-42 and M-67. Prior to this shut down, the conventional boiler was fired up and the station electrical output requirements were transferred to it.

Reactor thermal power (in percent of 256 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 items mentioned will be found in various appropriate sections of this report.

Operating Activity Summary

September 1 Five fuel element orifices were adjusted.

 Cold trapping of secondary loop No. 1 was started following a plugging temperature of 332°F.

Steam generation in loop No. 3 was increased to 251,000 lb/hr while monitoring evaporator level and steam purity.

September 2

Six fuel element orifice adjustments were attempted. C-85 was stuck and the others were adjusted.

Steam generation in loop No. 1 was increased to 251,000 lb/hr while monitoring evaporator water level and steam purity. Feedwater heater No. 4 was in service for this test.

The reactor scrammed due to failure of breaker 11X in the critical power system (see Outage Summary for details). Since the boiler was in cold standby, a hot startup of the reactor was performed.

Leak test of fuel storage pit No. 2 was discontinued and draining of the pit was started.

September 3

Steam generation in loop No. 3 was increased to 251,000 lb/hr with feedwater heater No. 4 in service while monitoring evaporator water level and steam purity. Steam generation was then slowly increased to 274,000 lb/hr.

Six fuel element orifice adjustments were attempted; five were successful and C-6 was stuck.

September 4

Draining of fuel storage pit No. 2 was completed.

Steam generation in loop No. 2 was increased to 251,000 lb/hr while monitoring evaporator water level and steam purity. Feedwater heater No. 4 was in service for this test.

Helium purging of the storage cells was completed following assembly of 20 uranium carbide elements.

A reactor physics test was performed to check the prompt power coefficient.

September 5

A program of placing all fuel element orifices in minimum torque was started. Twelve orifices were adjusted for temperature change. Seventeen others were adjusted to minimize torque only.

September 6

Six fuel element orifices were adjusted for temperature change. Eight others were adjusted to minimize torque only.

September 7

Thirty-three fuel element orifices were adjusted for temperature change. Twenty-nine others were adjusted to minimize torque only.

September 8

The detector of the pre-stack monitor was removed for maintenance and calibration.

The secondary cold trap was switched to loop No. 2. Final loop No. 1 plugging temperature was 250°F; initial loop No. 2 plugging temperature was 325°F.

Actuator housing seals on control rods CR-7, 13, and 17 were adjusted to reduce helium usage.

Nine fuel element orifices were adjusted for temperature change. Four others were adjusted to minimize torque only. Orifices in C-64, C-80, C-145, and C-171 were stuck and therefore not adjustable. This completed the program of positioning all orifices at minimum torque.

September 9 Actuator housing seals on control rods CR-5, 6, 10, 11, 14, 15, and 19 were adjusted to reduce helium usage to a minimum.

An attempt to adjust six fuel element orifices classified as stuck was unsuccessful. These orifices were C-6, C-63, C-64, C-80, C-87, and C-145.

A power depression in the elements surrounding moderator element M-51, a drop in reading on nuclear channel V, and movement of control rods on automatic control outward several inches all indicated a failure of moderator element M-51.

September 10 One fuel element orifice was adjusted.

The prestack monitor was returned to service.

The reactor scrammed due to loss of instrument air to the conventional portion of the plant (see Outage Summary for details).

September 11 Power range nuclear instrument channels V, VII, VIII, and IX were repositioned to make their output agree with reactor power level as calculated by heat balance. The unbalance was caused by failure of moderator element M-51 on September 9.

Control rod CR-14 in core position C-78 adjacent to failed moderator element M-51 was withdrawn to 110 inches from its banked position in Gang 1 of about 90 inches, and control rod CR-11 in Gang 2 was lowered to the banked rod position from full out. This swap was performed to partially offset the effect of the failed moderator element.

September 13 Cold trapping was shifted from secondary loop No. 2 to loop No. 3. Final loop No. 2 plugging temperature was 250°F. Initial loop No. 3 plugging temperature was 368°F.

September 14

Flow was established in the primary service system and a plugging run on primary fill tank No. 3 resulted in a plugging temperature of less than 250°F. Preheaters were then energized in preparation for a carbon trapping run. Flow was established through the carbon trap system from primary loop No. 1.

Steam generation in loop No. 1 was increased to 251,000 lb/hr with No. 4 feedwater heater in service while monitoring evaporator water level and steam purity.

Fill of fuel storage pit No. 2 with demineralized water was started.

A small accumulation of water on the pit No. 1 floor was drained to the maintenance cell holdup tanks.

September 15

A carbon trapping run was initiated at 1300 when the carbon trap temperature reached 1200°F.

September 16

Filling of fuel storage pit No. 2 with demineralized water was completed.

September 17

Cold trapping operations were shifted from secondary loop No. 3 to loop No. 2. Final plugging temperature of loop No. 3 was 250°F; initial loop No. 2 plugging temperature was 295°F.

An operational check of the laundry room washer, extractor, and dryer was performed.

Steam generator testing of loops No. 1 and No. 3 continued while evaporator water level and steam purity were monitored.

September 18

The carbon trapping run was stopped at the end of 80.7 hours due to an apparent electrical short in the heater.

Helium samples were taken from the superheater and evaporator of the No. 1 and No. 2 steam generator third fluid systems. A helium sample was also taken from the No. 2 secondary sodium expansion tank.

Calibration of evaporator water level controllers for loops No. 1 and No. 3 was completed.

Two fuel element orifices were adjusted.

Assembly of twenty UC fuel elements was completed. The elements were placed in storage cells and helium purging of their storage cells was begun. Six of these fuel elements were assembled with a new direct-drive orifice.

September 19 A permanent ladder has been installed on the wall of the maintenance cell. All cell plugs have been removed from the top of the maintenance cell for further work in it.

 A check of the flux controller was made at high and low loads with the feedback system inoperative.

September 21 Three fuel element orifice adjustments were attempted. Two were successful; C-80 was still stuck.

September 22 Three fuel element orifice adjustments were attempted. Two were successful; C-80 was still stuck.

 The reactor scrammed due to an electrical power failure off-site when a CPPD 115 Kv line to Lincoln was blown down by high winds (see Outage Summary for details). Since the boiler was in cold standby, a reactor hot startup was performed.

September 23 Cold trapping operations were switched from secondary loop No. 2 to loop No. 1. Final loop No. 2 plugging temperature was 250°F; initial loop No. 1 plugging temperature was 322°F.

 Draining of fuel storage pit No. 2 was started to permit additional repair work on the walls and floor.

September 24 Power ramp tests were performed between 100% and 80% power (at rates up to 5% change per minute) while the steam generators were monitored for moisture carryover.

 Three fuel element orifices were adjusted.

September 26 A power depression in the elements surrounding moderator elements M-42 and M-67, a slight flux tilt as shown on the nuclear instruments, and movement of control rods on automatic out several inches all indicated a failure of moderator elements M-42 and M-67.

September 27 Cold trapping operations were shifted to loop No. 3 secondary. Final loop No. 1 plugging temperature was 250°F; initial plugging temperature on loop No. 3 was 350°F.

 Filling of fuel storage pit No. 2 with demineralized water was started.

 The reactor was shut down at 0840 to perform rod drop time tests. While all rod times were within specified limits, control rod CR-1 rod-in light did not come on indicating that the rod may not have dropped to the full-in position. It was found that the rod drive when driving full down would make the rod-in light come on.

September 28

Rod drop time tests on CR-1 were repeated from full out and 12 inches out. In both instances the rod stopped approximately $1\frac{1}{2}$ inches from a full down position.

Filling of fuel storage pit No. 2 with demineralized water was completed.

At the end of the month the reactor remained shut down while the effect of the recent moderator element failure was being evaluated. Since the plant is scheduled for annual overhaul beginning October 4, the reactor will not be returned to power until replacement of ruptured moderator elements has been accomplished.

B. Outage Summary

Three unplanned scrams occurred during the month of September. The reactor scrambled on September 2 due to a critical power breaker failure, on September 10 due to a temporary loss of instrument air, and on September 22 due to an electrical power failure off-site. In addition the reactor was shut down on September 27 following the failure of moderator elements M-42 and M-67 and remained down for the rest of the month.

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

Table I - 1

OUTAGE SUMMARY SEPTEMBER, 1964

I. Duration of outage: 1539 to 1919, September 2, 1964.

Failure of the critical power breaker which supplies power to bus 11X of the reactor control board caused a reactor scram. The failure resulted in a loss of power to the controls of loop No. 1.

Lost Critical Time	1 hour 39 minutes
Lost Operating Time	3 hours 40 minutes

II. Duration of outage: 1137 to 1840, September 10, 1964.

Loss of instrument air on the conventional end of the plant reduced the air pressure signal to the boiler feed pumps which resulted in a reduction in feedwater flow. Reduction of feedwater level in evaporator No. 2 then scrambled the reactor.

Lost Critical Time	2 hours 32 minutes
Lost Operating Time	7 hours 3 minutes

III. Duration of outage: 1256 to 1615, September 22, 1964.

A power failure off-site of CPPD's 115 Kv line to Lincoln was caused when high winds blew down a portion of the line. Protective relaying caused the two oil circuit breakers to trip. When an attempt was made to reclose the breaker at the request of the system dispatcher, an apparent drop in house service voltage scrambled the reactor.

Lost Critical Time	1 hour 4 minutes
Lost Operating Time	3 hours 19 minutes

IV. Duration of outage: 0840, September 27, 1964, to end of period.

The reactor was shut down to run control rod drop time tests following failure of moderator elements M-42 and M-67. The reactor was still shut down at the end of the month.

Lost Critical Time	87 hours 20 minutes (thru September 30)
Lost Operating Time	92 hours 57 minutes (thru September 30)

C. Shutdown

The reactor was shut down on September 27 to do rod drop time tests in connection with the failure of moderator elements M-42 and M-67. Since a four-week annual plant shutdown was scheduled to begin October 4, the reactor was not returned to power.

The conventional portion of the plant will return to service approximately the end of October. The nuclear portion of the plant will be shut down for several months for replacement of ruptured moderator elements. A general description of the work required is given in Section VII, Moderator Replacement Program, of this report.

II. Plant Performance

A. Overall Thermal Performance

Gross reactor electrical generation for the month of August was 21,762 Mwh for an average gross load of 35.5 Mwe. Table II - 1 summarizes plant performance through September.

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, however, constituted a small portion of the reactor turbine service time logged during September.

Table II - 1

PLANT PERFORMANCE DATA

Total Station Electric Generation	<u>Current Month</u>	<u>This Year</u>	
Gross Generation (Mwh)	26,723	314,239	
Station Auxiliary Service (Mwh)	2,586	31,091	
Station Net Generation (Mwh)	24,137	283,148	
Average Gross Load (Mw)	37.5	47.8	
Plant Capacity Factor; 95 Mwe base (%) ¹	39.9	50.3	
Plant Availability Factor (%)	99.0	99.8	
Net Plant Heat Rate (Btw/Kwh)	12,440	---	
Turbine Cycle Heat Rate (Btu/Kwh)	10,294	---	
Operating Hours			<u>To Date</u>
Turbine Generator	712.9	6,564.8	
Reactor Critical	629.2	4,101.2	7,928.0
Reactor Turbine Service	613.0	3,942.8	6,270.6
Reactor			
Gross Thermal Generation (Mwd)	2,889	14,373	26,613
Average Thermal Power (Mwt)	110.4	84.0	80.5
Gross Reactor Electric Generation (Mwh)	21,762	106,519	192,458
Average Gross Load (Mwe)	35.5	27.1	30.7
Reactor Capacity Factor; 83 Mwe base (%) ³	42.8	32.6	37.0
Reactor to Turbine Availability Factor (%)	85.2	60.0	52.8

¹Plant Capacity Factor = Average Total Gross Load Mw x 100 ÷ 95 Mw

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

³Reactor Capacity Factor = Average Reactor Gross Load Mw x 100 ÷ 83 Mw

Average gross loads are based on actual operating hours

B. Equilibrium Operating Data

Equilibrium operating data are shown on Table II - 2. These data were taken at a net electrical power of 80.3 Mwe on September 24. At the time the data were taken the reactor was operating as the sole source of steam to the turbine and at full power level in one of a series of high power and ramp tests being conducted that date on the steam generator system. (Reference Section III of this Monthly Report)

Thermal heat balance calculations were made at each of the successive flux indicated power levels in the approach to full power operation. On the basis of these calculations, reactor power as indicated by Nuclear Instrument Channel IX was raised to a maximum value of 105% for the full power tests. A final heat balance calculation at this power level gave a value of reactor power of 248.5 Mwt, or 99.4% of a 250 Mwt base.

Plots of Channels IX and NLR 1/2 readings vs. calculated power are shown in Figure II - 1. An estimate of the error in power level indications by these two channels is given by the separation of the respective curves from that entitled "Theoretically Correct Indications". As shown, the extrapolated channel readings corresponding to 100% thermal power are roughly 105% and 107% for Channels IX and NLR 1/2 respectively. The fact that the value of calculated power was 99.4% (instead of 100%) at their indications can perhaps be explained by the possible range of error in heat balance data and calculations. Reactor thermal power is based on averaged steam conditions at the three generator outlets, feedwater conditions at the generator inlets, and a steam flow rate determined from the steam flow integrating meter.

Figure II - 1 also shows plots of gross electrical output as determined from the gross output Wh meter and as indicated from the gross output Mw meter. As shown, the maximum extrapolated values of gross generator output are 86.6 Mwe and 85.0 Mwe respectfully. Determinations of electrical output from the Wh meter readings were made only at the highest two power levels.

Table II - 2

TYPICAL EQUILIBRIUM OPERATING DATA

Date September 24, 1964 Time 1245

Thermal Power (Mwt)	243.5
Gross Electrical Power (Mwe)	85.8
Reactor Auxiliary Power (Mwe)	5.5
Net Electrical Power (Mwe)	80.3
Net Plant Heat Rate (Btu/Kwh)	10,300

Nuclear Instrument Channel Readings

Calculated (250 Mwt base) (%)	97.3
Channel No. 5 (%)	99.5
Channel No. 6 (%)	101.3
Channel No. 7 (%)	99.2
Channel No. 8 (%)	102
Channel No. 9 (%)	103
Neutron Flux Level Recorder (%)	104
Channel No. 10 (%)	103.5

Steam & Feed Water Conditions

Steam Gen No. 1 Outlet Temp (°F)	876
Steam Gen No. 2 Outlet Temp (°F)	886
Steam Gen No. 3 Outlet Temp (°F)	877
Steam Header Pressure (psig)	884
Turbine Throttle Temperature (°F)	875
Turbine Throttle Pressure (psig)	821.8
Turbine Throttle Flow (lb/hr)	760,000
Feed Water Supply Temperature (°F)	378

Steam Generator Conditions

	<u>Outlet (°F)</u>	<u>Pressure (psig)</u>	<u>Flow (lb/hr)</u>
Evaporator No. 1	533.3	912	253,400
Evaporator No. 2	535.7	927	261,300
Evaporator No. 3	533.6	915	245,300
Superheater No. 1	879		
Superheater No. 2	885		
Superheater No. 3	888		

Sodium Conditions

	<u>Reactor Outlet (°F)</u>	<u>Reactor Inlet (°F)</u>	<u>Flow (lb/hr)</u>
Primary Loop No. 1	940.3	607.0	2,720,000
Primary Loop No. 2	948.0	604.9	2,740,000
Primary Loop No. 3	946.7	609.9	2,635,000
	<u>Stm Gen Inlet (°F)</u>	<u>Stm Gen Outlet (°F)</u>	<u>Flow (lb/hr)</u>
Secondary Loop No. 1	907.0	564.0	2,500,000
Secondary Loop No. 2	912.9	559.7	2,540,000
Secondary Loop No. 3	915.2	557.5	2,375,000

Figure II - 1

CHANNEL READINGS AND GROSS GENERATOR OUTPUT
Vs.
CALCULATED REACTOR THERMAL POWER

C. Core Performance

1. General

Reactor operation during August and September demonstrated that return to full power generation could be accomplished. The major objectives in this demonstration as outlined in the August monthly report were met:

- a. A new core anomaly did occur on September 9-10 (rupture of moderator can M-51); this anomaly was detected and could be tolerated, with power operation continuing until the September 27 shutdown.
- b. Nuclear instruments were adjusted during September to remain in agreement with heat balance measurements.
- c. Fuel element orifices were adjusted to maintain all fuel and fuel exit temperatures within acceptable operating limits (Note: Though orifice adjustment alone could not accomplish this, orifice adjustment combined with adjustment of control rod pattern did provide normal fuel and fuel exit temperatures at all power levels).
- d. Even with the adjustments to rod pattern which were required, the position of banked control rods was acceptable for operation at all Xenon concentrations.

On September 24, the reactor power was increased to 100 percent for the first time since July, 1963. The duration of 100 percent power operation was short, not because of an incapability of the reactor, but because the increased electrical generation was not required.

With the reactor operating at 65 Mwt, two additional moderator can failures occurred on September 26, M-42 and M-67. These failures brought the total number of moderator can ruptures to eight. Shortly after these final failures, the reactor was shutdown to make preparations for the removal and replacement of the eight failed cans. Shutdown of the reactor was 7 days ahead of schedule and was in fact produced by the additional can ruptures. While continuation of operation could have been made, additional adjustments to control rod pattern and nuclear instruments would have been necessary.

2. Moderator Element Failures

The reactivity loss associated with the failure of moderator can M-51 on September 9-10 was found to be 23 cents. Figure II - 2 shows this loss occurring within a two hour period, typical of the reactivity change associated with most previous ruptures. Figure II - 3 shows the change in relative power with the M-51 rupture. The failure of cans M-42 and M-67 on September 26 had associated a 25 cent total reactivity loss. With the exception of being a double failure, the rupture of these cans was again typical of previous occurrences.

Results of reactivity history determinations throughout the month of September may be seen in Figure II - 4. Significant events of the month are noted on this figure. Table II - 3 summarizes the eight failed moderator cans and the reactivity changes noted. A total of 139 cents change was measured for the eight cans.

3. Nuclear Instruments

Rupture of M-51 on September 9-10 necessitated the adjustment of nuclear instruments. Figure II - 5 shows the percentage deviation of the nuclear instruments from heat balance measurements made frequently during the month. The instruments were adjusted on September 11 with both power level and Xenon concentration lower than normal. The gradual shift over the next six days is attributed to the effect of fission product poison redistribution, in turn requiring control rod withdrawal; both effects produce a detectable shift in flux as noted in the nuclear instruments. This flux shift was also seen in fuel and fuel channel exit temperatures. Such a flux shift is a predictable effect and produces no operating difficulties.

4. Fuel Element Orifices

The effort to restore all fuel exit temperatures to within ± 15 degrees of the average was continued in early September. Just prior to the failure of M-51 on September 9-10 this small deviation in fuel channel exit temperatures was achieved in all but 12 elements. The 12 elements are those with fully closed, stuck, or broken orifices. This pattern of fuel exit temperatures was disturbed by the rupture

Figure II - 2
 REACTIVITY HISTORY DURING RUPTURE
 OF MODERATOR CAN NO. M-51

Table II - 3

SUMMARY OF FAILED MODERATOR CAN WORTHS

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	Mar 6-12 Mar 29 - Apr 5	9 9 7 <u>25</u>	10½ 10½ 8 <u>29</u>	20 (M-38) 11 (M-62) <u>31</u>
M-38 & M-62	Totals				
M-3	Mar 4	---	18	21	21
M-58	May 15	---	16	19	18
M-51	Sept 9	---	23	23	20
M-42 & M-67	Sept 26	Sept 27-30	25 18 <u>43</u>	25 18 <u>43</u>	23 (M-42) 20 (M-67) <u>43</u>
M-42 & M-67	Totals				
TOTALS			128¢	139¢	137¢

* Predicted worth is determined by flux weighting to corrected-observed worth of M-3 as base.

Figure II - 5

NUCLEAR INSTRUMENT READINGS
Deviation from Calculated Power vs. Time

of M-51. From this point, the fuel exit temperatures were adjusted as usual to maintain:

- 1) Values less than 1000°F, and
- 2) Delta T across moderator cans of less than 50°F.

No effort was made to re-establish the ± 15 degree pattern of fuel exit temperatures.

Status of fuel element orifice positions on September 30 is given in Table II - 4.

5. Fuel Status

Status of HNPF fuel on September 30 is shown in Table II - 5. In this table isotopic fuel changes as well as fuel exposure conditions of importance are noted.

6. Reactivity Status

The status of core reactivity at month's end is given in Table II - 6. The reactivity losses associated with rupture of moderator cans (139 cents) are not included in this table. Current reactivity conditions are compared with two other periods in HNPF operation.

Core Position	Orifice Plug Position	Orifice Calibration Setting	Core Position	Core Position	Orifice Plug Position	Orifice Calibration Setting
C-2	4-3/4	1-1/16	46	170	3-3/8	1-3/32
C-3	4-15/16	1-5/32	47	171*	3-1/2	1-1/16
C-5	4-21/32	1-0	48	174	3-7/32	0-21/32
C-6*	4-19/32	0-3/4	49 6	o 175	2-25/32	0-23/32
C-7	4-5/8	1-9/64	51 2	176	3-13/16	1-1/8
C-8	4-23/32	1-3/16	52 2	o 177	2-31/32	1-1/8
C-9	4-1/2	1-1/16	+ 53 2	178	3-3/4	1-1/8
C-10	4-13/32	1-1/32	54 2	o 179	3-1/32	1-3/16
C-11	4-13/16	1-1/16	55 }	180	3-3/8	1-1/8
C-12	4-3/4	1-1/16	+ 57	o 211	1-1/32	1-1/32
C-13	4-23/64	1-1/8	58 6	+ - U-Carbide Fuel o - Displaced U-Mo Fuel * Stuck ** 76 & 81 Interchanged # Full Open September 30, 1964		
C-14	5-0	1-1/16	59			
C-15	4-27/32	1-1/32	60 }			
C-17 #	4-5/8	0-5/8	+ 61 6			
C-18	4-31/32	1-3/32	63 6			
C-19	4-13/16	1-1/8	64 *			
C-21	4-3/4	1-1/8	+ 65			
C-22	4-3/4	1-1/8	66 #			
C-23	4-29/32	1-3/16	67 #			
C-25	4-7/8	1-3/32	+ 69 #			
C-26	4-1/2	1-1/8	70 6			
C-27	4-31/64	1-3/16	71			
C-29	4-17/32	1-1/32	72			
C-30	4-27/32	1-1/8	+ 73 }			
C-31	4-61/64	1-3/16	74			
C-33	5-3/64	1-1/16	75			
C-34 #	4-15/16	1-1/16	76 * 2			
C-35	4-3/4	1-1/16	77			
C-37	4-27/32	1-7/32	79			
C-39 #	5-1/8	1-1/8	80 * 6			
C-40 #	5-1/16	1-1/16	81 * 6			
+ C-41 #	5-1/8	1-1/8	82 2			
C-42 #	5-7/32	1-1/8	83 6			
C-43 #	4-23/32	0-23/32	84			
+ C-45	5-1/8	1-5/32	85 2			

Table II - 5

FUEL STATUS

Fuel Inventory

Location

	<u>I</u>	<u>II</u>	<u>III</u>	<u>IV</u>	<u>V</u>	<u>Total</u>
Reactor	139	1	8	2		150
Storage	3	10			20	33

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)
 V UC-Core II

Fuel Isotopic Changes

Consumed (Kg)	<u>Current Month</u>	<u>Cumulative Since January 1963</u>
U-235	3.7	33.9
Pu	0.10	0.90
Accumulated (Kg)		
Pu	1.36	12.29

Cumulative Since January 1963	Element No.	Core Position	Average Exposure (Mwd)	Peak Exposure ¹ (Mwd)	Average Burnup (Mwd/T)	Peak Burnup ¹ (Mwd/T)
Core Average	---	---	---	---	912	---
U-Moly Element w/highest exposure	66	C-43	261	488	1323	2475
U-Moly Pre-measured Element w/highest Exposure ³	152 ² 158	C-34 C-18	232	434	1178	2200
U-C Element w/highest Exposure ⁴	5	C-69	184	344	1171	2190

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

²Two elements have approximately equal exposure.

³Element in C-39 has premeasured rods and has greater exposure, but this element is thermocoupled.

⁴All U-C elements have premeasured rods.

Table II - 6

HNPF Reactivity Status

Date	Period Prior to Power Operation January 1963		Change Produced by:	October 3, 1963		Change Produced by:	September 30, 1964	
Item	Cents	% Δ K		Cents	% Δ K		Cents	% Δ 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	85.9 additional full power days operation; making a total of 106.4 full power days.	1610	11.2
Measured Excess Reactivity	983	6.8		845	5.8		675*	4.7*
Shutdown Margin	857	6.0		788	5.5		935	6.5
Na Worth + 1%	735	5.1		735	5.1		735	5.1
"Extra" Shutdown Margin	122	0.9		53	0.4		200	1.4

* Reactivity losses associated with ruptured moderator cans not included.

D. Sodium Purity

1. Cold Trap Performance

a. Primary System

No cold trapping of the primary systems was done during September nor were any plugging runs initiated. The last plugging run was on August 30 at which time the plugging temperature was 265°F.

On September 14 flow was established to the plugging meter from primary fill tank No. 3. The plugging temperature was measured to be less than 250°F.

The amount of oxide removed by primary cold trap No. 1 remains at 71.5 pounds.

b. Secondary System

All the secondary loops were cold trapped during September as shown on Figure II - 6 . The amount of oxide removed was calculated based on plugging temperatures and an average rate of oxide going into solution of 0.3 pounds per day.

Table II - 7

<u>Loop</u>	<u>Total Oxide Removed (lbs)</u>
1	11.8
2	8.1
3	11.4
Total oxide removed by	
trap through August	119.3
Total	150.6

2. Carbon Trap Performance

A carbon trap run was commenced on September 15 and continued until approximately 2100 on September 18 at which time the heater circuit breaker opened. The circuit breaker would not hold when the heater was re-energized. The run was terminated with a time of operation of approximately 80 hours.

Figure II - 6
SECONDARY COLD TRAP PLUGGING TEMPERATURE

E. Control Rod Thimble Helium Atmosphere

Helium was circulated through all control rod thimbles throughout the month of September. The hydrogen concentration in this system was measured on September 15 and on September 22; the results indicated 310 ppm and 250 ppm respectively. The maximum allowable hydrogen concentration is 700 ppm.

F. Flux Controller Position Feedback

The flux controller was checked on September 18 and 19 to study the effect of the flux controller position feedback. The plant control system operation without flux controller position feedback was observed at 35 and 47 percent power.

Thirty-five percent more rod movement was noted when the flux controller was operated without position feedback. The reason for the increase in rod movement is that, without position feedback, the rods will move until there is a decrease in flux deviation. The resultant change in flux is too large for steady state operation.

With feedback, the rods will stop moving with a change in position feedback in addition to flux. This prevents the large change in flux which requires corrective action by the flux controller.

Although there was additional rod movement without position feedback, there was no apparent difference in the amount of deviation in steam pressure, fuel outlet temperature, or flux.

G. Miscellaneous Systems Data

Other system data are shown in the table below.

Table II - 8

MISCELLANEOUS SYSTEMS DATA

Make-up Usage:

Sodium

Primary: None

Secondary: None

Nitrogen Delivered: 247,300 cu. ft.

Helium Usage: 36,850 cu. ft.

Steam Generator Evaporator Water Analysis:

Typical Chemical Equilibrium Operating Conditions

	Steam Generator #1	Steam Generator #2	Steam Generator #3
Date	Sep 8, 1964	Sep 8, 1964	Sep 8, 1964
Time	1005	1005	1005
pH	10.2	10.4	10.3
Conductivity (mmho)	43.0	54.0	44.0
OH Alkalinity (ppm CaCO_3)	7.5	8.5	6.5
Silica (ppm SiO_2)	1.5	1.6	1.2
Hydrazine (ppm N_2H_4)	0.0120	0.0090	0.0056
Phosphate (ppm PO_4)	5.1	5.9	4.9

III. Testing

A. Steam Generator Testing

1. Introduction

The results of the steam generator tests accomplished in 1963 were reported in the HNPF Monthly Operating Report No. 17, December, 1963. The following objectives were not completed at that time:

- a. Determine the correlation between indicated evaporator water level and steam flow rate for all three steam generators at steam flow rates from 0 to 251,000 lb/hr.
- b. Define the combinations of evaporator water level and steam flow rate which cause excessive moisture carryover in all three steam generators.
- c. Study the performance of the steam generators during positive and negative power ramps near 100 percent reactor power.
- d. Evaluate the performance of the steam generators with and without No. 4 feedwater heater in service.

Prior to the recent steam generator tests the "T" bar splash baffles were removed from evaporator No. 3 and an automatic water level program was added to the water level control system for each steam generator.

2. Presentation of Test Results

During August and September, 1964, the objectives listed above were accomplished. In addition, some tests were made at steam flow rates greater than 251,000 lb/hr per steam generator.

a. Indicated Evaporator Water Level vs. Steam Flow Rate

An automatic water level program was added to each of the evaporator water level control systems. This program maintained the evaporator water level approximately at "normal water level", or two inches above the sodium tubes, at all steam flow rates. Manual adjustments to the program were made, when necessary, to achieve normal water level and also to produce higher water levels during the tests.

Experience to date has shown that it is necessary to take a sample from one of the internal sample lines, or probes, in order to be certain of the evaporator water level.

b. Moisture Carryover Determinations

The recent test data has been added to that which was presented in the December, 1963, Monthly Operating Report and is presented in Figures III - 1, III - 2, and III - 3.

Steam generator No. 1 (see Figure III - 1) can produce steam with an acceptable moisture content even at 260,000 lb/hr steam flow. The proper evaporator water level, however, is very important and only two or three inches variation in water level is available.

Steam generator No. 2 (see Figure III - 2) produced steam with an acceptable moisture content at steam flows as high as 272,000 lb/hr. The evaporator water level is not nearly as critical as in steam generator No. 1.

Steam generator No. 3 (see Figure III - 3) produced steam with an acceptable moisture content at steam flows as high as 268,000 lb/hr. The evaporator water level appeared to have approximately the same effect on moisture carryover in steam generator No. 3 as in steam generator No. 2. (NOTE: The data shown for the two highest steam flows in Figure III - 3 were taken after the "T" bar steam baffles were removed.)

Since it is not possible to obtain a representative evaporator water sample, the curves in Figures III - 1, -2, and -3 may be displaced vertically from their true positions. The shapes of these curves, however, are indicative of the conditions which exist inside the evaporators since the relative moisture carryover during each test was measured very accurately. At low steam flow rates the amount of moisture carryover is nearly constant over a wide range of water levels. Under these conditions the interface between the vapor phase and the steam-water transition zone is quite sharp, and when the water level is finally raised high enough for this interface to reach the steam outlet baffle boxes, the steam leaving the evaporator is flooded with water. As the steam flow rate is increased, the interface between the vapor phase and the steam-water transition zone becomes less distinct and the transition zone extends to a higher elevation; under these conditions the moisture carryover increases more gradually as

HNPF STEAM GENERATOR NO. 1 PERFORMANCE

Figure III - 1

HNPF STEAM GENERATOR NO. 2 PERFORMANCE

Figure III - 2

HNPF STEAM GENERATOR NO. 3 PERFORMANCE

Figure III - 3

the evaporator water level is raised. If the steam flow rate is increased high enough, this interface will be raised to the elevation of the steam outlet baffle box and primary separation of steam and water by gravity will no longer occur. This condition has nearly been reached in steam generator No. 1 at 260,000 lb/hr steam flow.

c. Power Ramps Near 100 Percent Reactor Power

The reactor power was raised to approximately 100 percent power with No. 4 feedwater heater in service and allowed to remain at 100 percent for approximately one-half hour. All three steam generators performed satisfactorily with the water level slightly below normal. The reactor power was then lowered to approximately 80 percent in one step at a rate of five percent per minute and allowed to remain at 80 percent for approximately one-half hour. Reactor power was then raised to 100 percent in one step at a rate of five percent per minute, and allowed to remain at 100 percent for approximately one-half hour; this sequence of power ramps was repeated three times and each steam generator was monitored for carryover during one of the ramps. During the power ramps the evaporator water levels were maintained slightly below normal water level and the conductivity of the sample from the 2-S probe of each steam generator was recorded continuously. Under these conditions it was possible to detect and manually correct for deviations in evaporator water level. During these ramp tests the amount of carryover was well below 0.25 percent and no indication of instability was observed.

d. Effect of No. 4 Feedwater Heater

Moisture carryover tests were conducted both with and without No. 4 feedwater heater. The results of these tests indicate that feedwater temperature is not a significant factor.

3. Summary

The following conclusions can be made concerning the steam generator performance:

- 1) The steam generator performance is satisfactory at all power levels up to and including 100 percent. Moisture carryover can be maintained below 0.25 percent under all normal operating conditions.

- 2) Evaporator water level is very important; the bottom of the steamwater transition zone must be controlled at approximately two inches above the sodium tubes at the cold end of the evaporator (near the helium-water tubesheet).
- 3) The amount of moisture carryover is not dependent upon feedwater temperature to any significant extent.
- 4) No instability was experienced with respect to moisture carryover during power ramps between 80 percent and 100 percent reactor power with No. 4 feedwater heater in service.

B. Steam Generator Superheaters

1. Purpose

The purpose of this test was to determine the pressure drop across the superheaters of each steam generator during operation at 100% power.

2. Methods

Test gages indicating evaporation pressure and superheater outlet pressure were calibrated and installed on each generator unit. Base data were obtained at power levels of approximately 28% of full power. Load was then increased to 100% and several readings were taken to give a value of increased pressure drop for each superheater. This value of increased pressure drop was then added to a value of 2 psi which was assumed to be the correct value of pressure drop across each superheater at 28% power. The value of 2 psi is based on the design pressure drop of 26 psi at 100% power and the relationship that pressure drop is proportional to the square of the flow.

This relative method of obtaining pressure drop was used to eliminate inaccuracy of the pressure gages. Even when calibrated the gages usually are in error by several pounds after installation. This is the result of high temperatures in the steam generator room and the fact that at pressures approaching 1000 psi the gages cannot be expected to give readings closer than 2 or 3 psi. Appropriate water leg corrections were incorporated in the computation. The differential pressures reported are those between evaporator steam pressure and superheater outlet. This includes pressure drop through the eliminator which should be minor.

3. Results and Conclusions

The pressure drop values are as follows:

Steam Generator No. 1	22.7 psi
Steam Generator No. 2	28.7 psi
Steam Generator No. 3	23.7 psi

These values compare quite well to the design value of 26 psi. The higher drop through superheater No. 2 was due partly to a slightly higher steam flow through this steam generator.

C. Control Rod Drop Time Test

Control rod drop time tests were run on September 27 following reactor shutdown.

1. Purpose of Test

The test was conducted to demonstrate whether the indicated rupture of additional moderator cans has caused any binding problems in the control rods.

2. Presentation and Discussion of Data

Table III - 1 shows all five sets of rod drop time data taken in 1964. It can be seen from the table that all the recently obtained (9/27/64) drop times meet Technical Specification values, except that control rod number one had an exceptionally short overall drop time (ΔT_3). Some of the other drop times are somewhat shorter than those previously measured, but this is not a trend in a dangerous direction and still within technical specification limits.

The short drop time for control rod No. 1 was found to be due to the fact that this rod did not insert the last 1 to $1\frac{1}{2}$ inches. This type of problem has been encountered before (ref. HNPF July Monthly Report, Section III, page 13), but is not deemed to be dangerous since the last few inches of rod movement have very little reactivity worth. It appears that in these last few inches of travel, when the snubbers are already slowing the fall of the rods, slight changes in the frictional loads in the system can cause the rod to not fully insert. Further investigations will be conducted as to the cause of this additional friction prior to reloading this control rod into the reactor following moderator replacement.

The reduced drop times for some of the control rods could be due to many effects such as:

- 1) Different temperatures in the reactor during the various tests.
- 2) Use of different measuring devices to obtain data. Though the same type equipment was used, the same recorder unit was not necessarily used. This can cause slight deviations in chart drive time scale speed giving slightly different results when measured from the chart.
- 3) Continued use of the control rods will cause some wear between moving parts. This wear could possibly cause looser fits between moving parts without much change in roughness and

Table III - 1
CONTROL ROD DROP TIME TEST
(All Times in Milliseconds)

Control Rod No.	Core Position	3/8/64			4/8/64			4/27/64			5/19/64			9/27/64		
		ΔT_1	ΔT_2	ΔT_3	ΔT_1	ΔT_2	ΔT_3	ΔT_1	ΔT_2	ΔT_3	ΔT_1	ΔT_2	ΔT_3	ΔT_1	ΔT_2	ΔT_3
1	C-1	52	780	974	53	779	975	54	779	977	55	781	975	47	778	**
5	C-38	54	789	965	56	791	967	56	788	962	55	788	964	55	776	952
6	C-16	55	781	956	55	784	959	53	780	956	51	779	954	57	777	951
7	C-44	57	774	965	57	779	965	51	770	957	58	773	959	51	767	947
8	C-50	55	774	953	53	778	953	51	768	947	53	766	943	47	775	955
9	C-24	54	784	983	57	791	981	56	783	972	55	788	973	55	790	979
10	C-56	59	785	961	58	781	959	53	779	958	53	781	960	58	*	970
11	C-62	57	773	962	53	772	962	54	771	955	51	770	956	46	772	961
12	C-32	54	770	954	53	771	959	52	767	951	52	767	950	53	766	952
13	C-68	55	790	960	54	790	964	56	788	961	59	788	962	53	783	961
14	C-78	59	796	977	59	802	983	57	792	973	58	790	970	49	796	976
15	C-86	58	787	970	59	796	980	57	779	960	58	781	963	53	767	947
16	C-94	57	776	965	57	779	966	57	776	963	55	774	962	56	776	965
17	C-102	57	781	957	55	787	968	57	777	955	54	779	957	49	786	964
18	C-110	58	774	965	53	773	962	53	767	953	54	769	957	58	771	962
19	C-118	55	772	959	54	780	968	54	766	951	54	769	955	49	763	947

ΔT_1 = Time from magnet de-energized to when the "rod latched" switch breaks contact

ΔT_2 = Time from magnet de-energized to 9 feet of free fall.

ΔT_3 = Time from magnet de-energized to full snubbed position

* Coil broken -- measurement not made

** Short drop time

Technical Specification limits:

ΔT_1 : 45-60 milliseconds

ΔT_2 : ≤ 803

ΔT_3 : 900-1015

friction and thus cause faster drop times. However, wear could also cause greater roughness and friction without loosening the fit between parts. This would cause slower drop times in which case it would not explain the current results. The drop times as measured on 9/27/64 are considered reasonable compared to past data within the accuracy obtainable by present testing methods.

3. Conclusion

The data obtained in September demonstrate that the present number of ruptured moderator cans in the core has not adversely affected the scram capability of the control rods.

IV. Radiation Control

A. Radiation and Contamination Surveys and Monitoring

The plant site environmental air filter was changed and counted on a weekly schedule during September. 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 changed from 3×10^{-12} to 5×10^{-12} $\mu\text{c/cc}$ beta-gamma, and from 5×10^{-13} to 9×10^{-13} $\mu\text{c/cc}$ alpha. These initial activities 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 changed and counted weekly. No activity in excess of normal atmospheric airborne levels was detected.

Periodic radiation surveys of reactor vent system components (CV-1004 and FI-849) and reactor cover gas monitor (RD-1) in-line filter housing were performed. No evidence of sodium carryover was detected.

Radiation surveys of the control rod actuator housings were performed immediately following reactor scrams on September 2, 10, and 22. No significant elevation of radiation levels at the housings was detected. Periodic radiation surveys of control rod actuator housings which were performed throughout September showed the maximum radiation level to be 1.7 mr/hr (contact) which was due principally to Argon-41 in the control rod actuator housing atmosphere.

On September 4 and 24 radiation surveys of fuel storage pit No. 2 were performed in preparation for entry for continued waterproofing work. Maximum radiation level detected in working areas, was 1.5 mr/hr.

Frequent radiation surveys were performed in connection with removal of the fuel storage pit No. 1 access plug and inspection for water leakage into the pit. Maximum radiation level at deck level over the open access hole was 18 mr/hr.

Periodic radiation surveys and smear surveys were performed of fuel storage and fuel element assembly areas in the assembly building. Maximum radiation level was 2.5 mr/hr and no surface contamination in excess of 30 dpm per 100 cm^2 was detected.

Smear surveys of individual fuel rods were performed during the continuing assembly of fuel elements. No contamination in excess of 30 dpm/100 cm² was detected.

Continuous Health and Safety coverage was provided for the transfer of 40 new fuel elements from the assembly building to storage in the reactor building.

Periodic smear surveys and radiation surveys were performed throughout the plant area during this report period. No surface contamination in excess of 30 dpm per 100 cm² or abnormal radiation levels were detected outside of barriered working areas.

Additional radiation surveys and monitoring operations were performed in connection with items listed in D "Other Health and Safety Activities" of this section.

B. Sampling and Analysis of Process Systems

During September one gas decay tank was sampled prior to release. Sample activity was 2.4×10^{-6} $\mu\text{c/cc}$. The tank volume was released via the main building stack at a rate of less than 50 cfm.

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 7.2×10^{-3} $\mu\text{c/cc}$. Results of gamma spectrum analysis show the changes of constituent isotope ratios with changing reactor power and operating time to be consistent with previous observations. No cladding abnormalities other than the previously detected pin hole leaks are indicated by these data.

On September 3, a water sample was obtained from fuel storage cell No. 2 in preparation for drainage of the pit. Maximum sample activity was less than 1.7×10^{-7} $\mu\text{c/ml}$.

The closed loop cooling water system was sampled on September 10 and counted prior to the performance of chemical analyses. Maximum sample activity was less than 1×10^{-9} $\mu\text{c/ml}$.

On September 24, samples of the control rod actuator housing helium atmosphere were obtained for counting and gamma spectrum analysis. Sample activity was 1.3×10^{-4} $\mu\text{c/cc}$ due principally to Argon-41.

On September 28, air sampling was performed with portable instruments due to the shutdown of the R/A waste facility stack monitor for repairs. Initial sample activity was less than 4×10^{-10} $\mu\text{c/cc}$ and decayed in a manner normal to atmospheric air samples in this area.

Microcuries Per Cubic Centimeter $\mu\text{c/cc}$

C. Health Physics Instruments

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

All portable survey instruments were checked for functional abnormalities and recalibrated using incapsulated sources.

Two additional encapsulated cobalt 60 calibration sources, consisting of one 700 mc, and one 250 mc, were received.

D. Other Health and Safety Activities

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

1. Modification of storage plugs for use in new fuel storage.
2. Decontamination of tools and equipment.
3. Replacement of corrosion test strips in the closed loop cooling system.
4. Transfer of new fuel to storage in the reactor building.
5. Maintenance of prestack monitor (RD-1002).
6. Continuing fuel orifice adjustment work.
7. Inspection and repair of control rod actuator drive seals and helium leak testing.
8. Fuel storage pit No. 2 waterproofing rework.
9. Fuel handling machine grapple maintenance.
10. Removal of master slave manipulators from maintenance cell for through tube seal modification.
11. Removal of maintenance cell sealing plug prior to entry for equipment modification.
12. Fuel handling machine maintenance and checkout in preparation for reactor unloading.
13. Decontamination of control rod SN-20 boron shroud and packaging for storage.
14. Installation of new relief valve and radioactive liquid waste disposal pump discharge line.
15. Lee manipulator clutch repair in maintenance cell.
16. Maintenance cell in-cell crane modification.
17. Control rod CR-1 drop testing.
18. Storage cell plug quad ring cleaning and replacement.
19. Maintenance of control rod CR-7 drive brake.
20. Maintenance cell in-cell equipment installation.

Solid radioactive waste was packaged periodically for storage and ultimate disposal.

On September 30, a shipping cask (B of E No. 362 in crash shield No. 3) was received for use in the shipment of a startup source for re-irradiation.

E. Radioactive Waste

The August inventory shipment and disposal of radioactive waste are tabulated in Table IV - 1.

Table IV - 1

RADIOACTIVE WASTE: INVENTORY, SHIPMENT AND DISPOSAL

Radioactive Waste

<u>Inventory</u>	<u>Quantity</u>	<u>Activity (Curies)</u>
Solid (cu ft)	7.3×10^2	2×10^{-1}
Liquid (gal)	6.9×10^3	7.4×10^{-4}
	1.3×10^3	*
Gas (scf)	3.8×10^3	*

Shipments

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

Other DisposalDisposal Method

Solid (cu ft)	none	---	
Liquid (gal)	none	---	
Gas (scf)	9.4×10^3	6.4×10^{-4}	Released**

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

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

V. Maintenance

A. Instrument Maintenance

1. Plant Control System

An auto-manual switch was replaced on hand control station HCS-6 because of erratic operation.

Variable gain (false bias) of the neutron flux controller was adjusted to increase low power gain.

Temperature recorders TR-3A and TR-3B associated with the fuel outlet temperature scanner were checked for calibration.

Maintenance work was performed on the special fuel element recorder TR-12. A defective chart drive capacitor and a broken drive string were replaced.

In order to provide a linear steam flow anticipation signal, the input to "B" of computer FCCp-500 was changed from components 1 of 29 to 6 of 29. The proportional band was changed from 50 to 100.

Adjustments on pressure recorder controller PRC-501 included changing the proportional band from 300 to 200, changing the reset from 7 to 10, and changing the rate from 0.1 to 0.0. This was done to improve the operation of the plant control system.

2. Calibration of Level Recorder Controller

A complete recalibration of level recorder controller LRC-151 and LRC-351 evaporator level measurement systems was completed September 18. This check was made because the recorded level did not agree with the actual level, as determined from the 2-S conductivity probes.

The largest error was corrected by blowing down the sensing lines to the level transmitters. It is assumed air was trapped in the top connection of the temperature equalizing columns. Small errors were also found in the reversing dividers, computing relays, and pressure transmitters.

When the "T" bar splash baffles were removed from No. 3 steam generator it was assumed this would help correct the false water level problem. Actual level as determined by the 2-S conductivity probe indicated that removing these baffles did not improve the water level indication.

The demand level computer was calibrated to program IRC-351 as shown in Figure V - 1. The calibration adjustments will control the water level at two inches above the sodium tubes at the cold end of the evaporator.

3. Plant Protective System

A patch panel cord was replaced on signal plug to the "B" dT/dt computer.

Power-to-flow ratio computer T₀ S/N-1 was apparently noisy and indicated alarm conditions. Reconditioned amplifiers were installed in positions A2 and A4. The computer was realigned and used as a spare.

A noisy fluctuating output condition of power-to-flow ratio computer T₀ S/N-4 was initiating alarm conditions. Reconditioned amplifiers were installed in seven of eight positions, A1, A2, A4, A5, A6, A7, and A8. The computer was recalibrated.

11 operational amplifiers for the protective system computers were reconditioned.

4. Sodium Systems

Tubes were replaced in the amplifier of the miscellaneous sodium temperature indicator TI-400.

Stationary contacts and wipers were replaced on the system "D" preheat No. 3 stepping switch.

Erratic operation of the carbon trap temperature indicator TI-419 was corrected by cleaning the selector switch.

Secondary sodium expansion tank level indicators were checked. Null and gain was adjusted.

The power oscillators to No. 1 and No. 3 secondary sodium fill tank level alarm units were replaced with a new model.

Tubes and batteries were replaced in the vapor trap temperature indicator TI-800 because of lack of response.

5. Radiation Detection and Monitoring System

An incorrect count rate indication of radiation detector RD-1 was corrected by replacing a rectifier tube.

Two tubes were replaced in the high voltage power supply of the constant air monitor unit.

Figure V - 1

HNPf EVAPORATOR WATER LEVEL PROGRAM

The detector of the reactor building pre-stack monitor RD-1002 was removed for maintenance and calibration. The electrometer tube was replaced and the detector chamber was recharged with nitrogen. The stack monitor remained in service during this period.

Radiation detectors RD-900 and RD-1000 were recharged with nitrogen.

6. Health Physics Instruments

Batteries were replaced in the survey meter 2612 S/N-2535.

Batteries, probe, and GM tube were replaced on the survey meter 2612 S/N-2536.

Batteries were replaced in the Juno meter S/N-2369 because of incorrect indication.

One tube and a GM tube were replaced on the count rate meter (Labitron) because of no radiation detection.

A GM tube was replaced on the count rate meter because it did not detect radiation.

7. Miscellaneous Instruments

A test and reconditioning apparatus was fabricated for maintenance of the electro-mechanical choppers used in the Sanborn recorder pre-amplifiers.

Diaphragms in the solenoid valves of the steam/water mixer to the laundry washing machine were replaced.

B. Miscellaneous Maintenance

1. Control Rod Actuators

The actuator housing seals of the following control rods were adjusted to reduce the helium leakage:

CR-1	CR-11
CR-5	CR-13
CR-6	CR-14
CR-7	CR-15
CR-10	CR-17
	CR-19

The brake was replaced on control rod drive motor CR-17.

Control rod CR-7 brake lining was replaced.

2. Other Maintenance

Grapple quad rings on the fuel handling machine were replaced in order to improve the cask atmosphere seals.

A fifteen micron filter was replaced on the closed loop cooling water system.

The clutch on the Lee manipulator located in the maintenance cell was removed for replacement.

Quad rings were cleaned or renewed as required on six storage plugs of pit No. 2.

Operation of the laundry equipment disclosed a defective level control valve. A diaphragm was replaced in the valve.

VI. Modifications

A. Fuel Storage Pit No. 2

Since the leak rate from pit No. 2 was determined to be excessive during the month of August, 1964, pit No. 2 was drained on September 2 and 3. On September 9 a coating of Ironite plaster was applied to the floor and the lower 15 feet of all four walls. On September 10 a coating of Ironite plaster was applied to the floor and the lower one foot of all four walls.

Pit No. 2 was then filled with demineralized water and a leak test was initiated. On September 23 the leak test was terminated; the measured leak rate was approximately 0.5 gallons per hour. Pit No. 2 was drained and two coatings of Ironite plaster were applied to the entire inside surface of the pit. Demineralized water was used to fill pit No. 2 on September 27 and 28; another leak test was started. This leak test was in progress at the end of the month.

B. Evaporator Feed and R/A Waste Transfer Pump Relief Valve

A 20 psig relief valve was installed in the discharge line of the evaporator feed and R/A waste transfer pump. The relief valve relieves back to the pump suction and is in parallel with the existing 100 psig relief valve RV 1102 (see Figure VI - 1).

This installation is designed to prevent back-up of liquid waste into the dilution water line, should the check valve fail to operate properly.

RADIO-ACTIVE LIQUID WASTE DILUTION SYSTEM

Figure VI - 1

C. Maintenance Cell

1. Ladder

A 40 foot ladder that incorporates a safety slide device was installed near the tunnel on the west side of the maintenance cell. The ladder extends from the tool lay down shelf to the maintenance cell floor. The new installation permits easier personnel access to the lower level of the cell.

A 4 foot access ladder bridges the gap between the tool lay down shelf and the 40 foot ladder.

The installation is complete except for painting.

2. Master Slave Manipulator

Air flow restrictors were installed on the master slave manipulator. Cable sizes to the manipulator on the in-cell slave side were increased from $\frac{1}{32}$ inch to $\frac{1}{16}$ inch diameter. New stronger tong jaws were installed. These modifications provided for a transfer of the weak link of the system from the slave side to the master side of the maintenance cell wall.

Roller mounts complete with seals were installed on both the inside and the outside joints of the manipulator port in the maintenance cell wall. A nitrogen pressure is applied to the manipulator port to assure that in-leakage into the cell is nitrogen gas.

D. Component Transfers

During September a number of components were transferred within the storage pits. Movements were made to prepare storage pits Nos. 1, 2, and 3 for the unloading of fuel from the reactor. Figure VI - 2 shows the plan view of the storage pit.

Component transfers

1. 10 storage plugs were transferred from pit No. 1 to pit No. 3.
2. 12 concrete filler plugs were transferred from pit No. 3 to pit No. 2. This will allow full utilization of pit No. 3 for storage of irradiated elements.
3. 78 storage plugs were removed from storage pits, bagged, and placed on the high bay floor. These 78 storage plugs were modified by drilling two holes at the bottom end of the plug. At a later date a hanger rod will be attached by means of a pin. This modification will allow the attachment of either new fuel or spent fuel to storage plugs for storage. See Figure VI - 3.
4. 10 irradiated dummy elements, previously removed from the core during loading of uranium carbide (UC) fuel, were removed from pit No. 3 and placed in pit No. 1

E. UC Fuel Element

The first transfer of the new uranium carbide fuel elements was received in the reactor building from the fuel assembly area. Twenty elements were placed in fuel storage pit No. 2. Each element has 8 rods which are enriched to ~4.9 percent U^{235} . Figure VI - 4 shows the configuration of the UC elements.

Figure VI - 2

FUEL AND COMPONENT STORAGE AT HNPF
Reactor Building High Bay Area

Figure VI - 3

FOUR MODIFICATIONS TO 78 FUEL ELEMENT STORAGE CELL SHIELD PLUGS

Figure VI - 4
UC FUEL ELEMENT

II. Moderator Replacement Program

The progressive failure of moderator elements (cans) in the reactor core led to a decision in March, 1964 to shut the reactor down at the earliest feasible date, remove the damaged cans, replace them with good ones, and examine the damaged cans to determine cause of failure and methods of correcting the problem.

The possibility that moderator elements might rupture was, of course, considered in the design stages of the plant. However, only minimal equipment was provided for element replacement. The primary areas in which the equipment was lacking were in providing reasonably "tight" containment of reactor atmosphere (and hence contamination), and in keeping radiation levels to a practical minimum. Therefore work was started in March, 1964 to provide additional equipment for moderator replacement, and to provide procedures for operation of the additional equipment. This work proceeded at a rate compatible with a scheduled plant shutdown on October 4, 1964. Atomics International, under contract with CPPD and the AEC, was responsible for equipment design, fabrication, and procedures.

During September, 1964, some of the new and modified equipment began to arrive at the reactor site. Most of the draft procedures for use of the equipment had also arrived. The remainder of the equipment and procedures are scheduled to be delivered to the site during October, 1964. Subsequent monthly reports will describe progress made in checkout of the equipment, and in removal and replacement of moderator cans.

At the end of September, 1964, eight moderator cans were known to have failed, based on nuclear instrument perturbations, local power depressions, and reactivity losses as described in previous monthly reports. A ninth can (M-79) is suspected, but cannot definitely be listed as a failed can. Can failures, the dates (all in 1964) on which they occurred, and the reactivity decrease associated with each can failure, are tabulated below. Refer to Figure VII - 1 for a map showing location of the failed elements.

Table VII - 1

HNPF MODERATOR ELEMENT FAILURES

<u>Date</u>	<u>Can No.</u>	<u>Reactivity Decrease (cents)</u>
Feb 2	M-94	4
Feb 20	M-38	29
Feb 20-22	M-62	
Mar 4	M-3	21
May 15	M-58	19
Sept 9	M-51	23
Sept 9	M-79*	--
Sept 26	M-42	25
Sept 26	M-67	<u>18</u>
Total		139

* This can failure is regarded as a "possible" failure. Any reactivity decrease associated with its failure is small and has been included in the 23 cents associated with failure of M-51.

Figure VII - 1

HNPf KNOWN FAILED MODERATOR CANS

Special crews made up of reactor operations and maintenance personnel, supplemented by mechanics and health physicists provided under contract by Atomics International, will perform the actual work of removing and replacing the moderator elements. There will be four crews to permit 24-hour a day, 7-day a week operations, each headed by a crew leader.

In outline, the sequence of activities intended is as follows:

1. All fuel, dummy elements, and other components will be removed from the core. In general, these components will be removed in reverse order from which they were loaded. The dummy elements, except for those adjacent to a suspected ruptured can will be removed first. After several such elements have been removed, lights and a viewing device will be installed to permit observation of the sodium surface. Next, fuel elements, except those adjacent to suspected ruptured cans, will be removed. Then dummy elements adjacent to ruptured moderator cans will be removed, followed by removal of fuel adjacent to ruptured cans. Then, control rods and finally the neutron source will be removed.
2. The reactor will be drained of sodium. During the draining, the pressure within the experimental moderator can (M-3) will be monitored in an attempt to determine the approximate elevation of the leak. As the sodium level reaches the elevation of the tops of the moderator cans, the optical viewing devices will be used to determine if any of the ruptured moderator cans have expanded longitudinally a detectable amount.
3. After draining, a borescope will be inserted into various process channels to further define the degree of moderator can swelling.
4. A special induction coil probe will be inserted, within a thimble, in approximately 50 process channels. A vertical traverse of each of these channels will be made with the induction probe, which is excited by an A-C voltage. The current through the induction probe is measured, being dependent upon the electrical conductivity of the material surrounding the probe. A different reading is obtained when the probe traverses a channel adjacent to a ruptured can than when it traverses a channel adjacent to a "good" can. It is expected that the probing will verify the suspected ruptured cans, and locate others which may be ruptured.

5. The instrument extension on the top of moderator can M-3 will be cut off by means of a special tungsten arc cutting tool. The cut-off portion will be removed using the fuel handling machine or a gas lock. It will then be possible to rotate the top shield.
6. The fuel handling machine will be dismantled and converted to a moderator handling machine.
7. The reactor top shield will be rotated as required, and one of three 58" diameter moderator removal plugs removed, after installation of a 59" diameter gate valve.
8. With the 58" access opening over the moderator can to be replaced, the moderator handling machine will be positioned over the selected damaged can and sealed to the 59" gate valve.
9. The selected damaged can will then be removed, transported to the maintenance cell, placed in a canister, then withdrawn from the maintenance cell and placed in a moderator storage cell.
10. At least one damaged can will be retained in the maintenance cell for thorough physical examination concurrently with moderator can removal.
11. All damaged cans will be removed one by one following a similar sequence. It may be necessary, because of graphite swelling, to remove as many as three "good" cans adjacent to each damaged can to permit removal of the damaged can. One "good" can, not adjacent to any damaged cans, will also be removed as a "control" sample. Assuming that graphite swelling is an important factor, to remove only the suspected 9 damaged cans and one "good" one, the top shield will have to be rotated at least 8 times. Each of the three 58" access plugs will have to be removed and inserted at least once. At least 25 moderator cans will have to be removed and replaced with something, either another moderator can or a special dummy spacer can. At least 9 reflector-filler elements will have to be removed and replaced.
12. As each damaged can is removed, the adjacent "good" cans will be optically inspected for possible damage to the skin. This will require insertion of an optical device through the top plug on the moderator handling machine. Some inspection of the moderator can support pedestals may also be performed using the optical device.

13. After all known damaged moderator cans have been replaced, the reactor will be filled with sodium, the temperature raised to about 500°F, then drained.
14. The induction probe will then be used to determine whether any more ruptured moderator cans are present in the core. If not, preparations will be made to reload fuel.

Errata for HNPF Monthly Operating Report No. 25, August, 1964

Page 37, line 1 should read:

"is the result of compounds of sodium and hydrogen rather
than sodium oxide"