NAA-SR-9777
SPECIAL
COPY

FINAL EVALUATION REPORT
HALLAM NUCLEAR POWER FACILITY
VOLUME V

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HNPF Field Publications

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.
P.O. BOX 309 CANOGA PARK, CALIFORNIA

CONTRACT: AT(11-1)-GEN-8
ISSUED: JUNE 12, 1964

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HALLAM NUCLEAR POWER FACILITY

Post Critical Test Procedure

AI-PC-510

Hydraulic Test of Main Heat Transfer System

Test Procedure Submitted for Approval

6/30/61
Date

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Test Procedure Reviewed

12/10/62
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Test Approved for Performance

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Test Completed

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HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
HYDRAULIC TEST OF MAIN HEAT TRANSFER SYSTEM

AI-PC-510

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A T O M I C S I N T E R N A T I O N A L
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Contract: AT(11-1)-GEN-8

February 7, 1964

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HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
HYDRAULIC TEST OF MAIN HEAT TRANSFER SYSTEM, AI-PC-510

I. PURPOSE OF TEST

The objective of this test was to determine the hydraulic characteristics of the Hallam Nuclear Power Facility (HNPF) main heat transfer system. Isothermal testing was performed to verify that the system could be expected to operate satisfactorily under normal operating conditions and to establish hydraulic parameters for comparison with data to be taken during initial power operation. Data were recorded during power operation to check for satisfactory system operation, and to establish base data for future comparison.

II. DISCUSSION OF TEST PERFORMANCE

This test was performed in two parts, i. e. , (1) under isothermal conditions and (2) during power operation. While testing at isothermal conditions, the main heat transfer system was operated under nearly all anticipated operating conditions. During each condition, all pertinent process instrumentation was recorded. During the power operation portion of the test, the system was operated only as necessary to meet plant requirements. Thus, the latter part of the test was of a monitoring nature.

System parameters recorded during isothermal testing were sodium pressures, pump speed, pump motor current, coupling current, sodium flow rates, moderator valve position, and sodium temperatures. Sodium flow rate in all six sodium loops was varied over the entire operating range. The primary loops were operated separately, in pairs, and simultaneously. The moderator throttle valve was tested over the entire valve operating range during three-loop operation with several different primary flow rates. No distinction was made between two- and three-loop operation for the secondary loops because these loops are hydraulically independent of each other. Scram actions were initiated by pushing the manual scram button. Sodium flow rate and throttle valve position for each loop were recorded on oscillographs. No special instrumentation other than the oscillographs was used for the tests.

The same data were recorded during power operation as during isothermal testing. These data were recorded periodically at steady-state system conditions at various power levels during power operation. Scram actions initiated during power operation are included in AI-PC-660, Scram and Loss-of-Load, and are not included in this test. The moderator throttle valve was adjusted to determine the proper mode of operation for the valve. The core temperatures which were recorded to analyze the moderator loop were reactor vessel wall, reactor temperature sensing element, instrumented moderator element, and fuel element temperatures.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

A. ISOTHERMAL TESTING

1. Two-Loop Operation

Initial operation of the main heat transfer system with two primary pumps was unsatisfactory. Partial closing of the throttle valves was required to maintain the sodium level above the pump low-level alarm points when operating at high flow rates. As reactor pressure drop was higher with three loops operating, it was not necessary to use the throttle valves. The pump level problem was determined to be due to improper sizing of the weep holes in the pump impellers. (These holes are provided to allow sodium to flow from the pump barrel to the suction side of the pump impeller.) With inadequate back pressure, leakage past the impeller and flow from the pump discharge through the lower pump bearing was insufficient to maintain pump level.

To reduce the flow from the pump barrel, four of the eight weep holes in each primary pump impeller were welded shut. The modified pumps operated satisfactorily together, in pairs, and individually. Although one-loop operation of the plant is not anticipated, this was considered the worst possible operating condition for causing low pump-level alarms; it was, therefore, a conservative test of the pumps. The check valve in each primary loop demonstrated proper operation when no reverse flow was measured in the non-operating loop during two-loop operation.

2. Moderator Coolant Loop

The ratio of the moderator coolant flow to the reactor flow was essentially the same with any two loops operating or with all three loops operating at any flow rate. Minimum moderator coolant flow rate is 5.3% of the total reactor flow rate with the throttle valve fully open. Moderator coolant loop flow is shown in Figures 510-1 and -2. The moderator flowmeter calibration used was 0.0452 mv/gpm. Primary and secondary sodium flowmeter calibration curves are shown in Figures 510-3 and -4, respectively. A slight variation in moderator flow can be seen as a function of primary flow and number of operating loops.

3. Heat Transfer Loop Characteristics

Hydraulic characteristics at isothermal conditions were very near predicted values as listed in AI-P-1167, Hot Sodium Circulation Test, Figure 1167-16, which is based on the original recommended flowmeter calibration. The head curves for each loop are shown in Figures 510-5 and -6. All component pressure drops are the difference between two gage readings and, thus, include piping. Due to a high oxide concentration in secondary loop No. 1, the pressure drop in this loop was considerably higher than in the other two loops. Intermediate heat exchanger (IHX) unit 1A had been removed from the system for repair of an internal tube rupture. Thus, after returning IHX-1A to service, secondary loop No. 1 was saturated with oxides. Reduction of the oxide concentration was followed by a reduction in pressure drop.

4. Scram Actions

Figures 510-7 and -8 illustrate the transient characteristics following a simulated scram from full flow and 25% of full flow, respectively. Operation of the throttle valves is shown in Figure 510-9. The valves automatically closed as programmed. After about 30 seconds, the valves were automatically switched to control of the convection control system, which signaled the valves to open. Mechanical failure of the throttle valve operators occurred during initial operation of the plant. Several modifications were made to prevent such failures and the system was retested. Some of the failures were due to improper setting of the torque limiter on the pneumatic valve drive motors. After adjustment of the torque limiter and modifications were made, the throttle valves closed properly during all 21 scrams which occurred during power operation for the post critical test program.

The primary loops exhibited sodium flow oscillations during flow decay after scram (see Figures 510-7 and -8). The oscillations occur at about the time the throttle valves stop closing. In addition, it was observed that the primary pump high-level alarms annunciated after each scram. The balancing legs change level when ever the pump suction pressure changes. (A balancing leg is a U-shaped vertical pipe which connects the pump barrel overflow with the pump suction line. The balancing leg prevents gas from being sucked into the pump suction line from the pump barrel, because the pump suction is not sufficient to pull the sodium level in the balancing leg down and around the base of the U-shaped pipe.) Therefore, it is probable that the oscillations are due to a combination of the stopping of the valves and the changing sodium levels in the pumps and balancing legs. Since no adverse effects have been observed, it can be concluded that the sodium levels do not reach levels which would cause a problem.

B. POWER OPERATION TESTING

1. Heat Transfer Loop Characteristics

The head curves for each loop are shown in Figures 510-10 and -11. The loop pressure drops based on volumetric flow rate were slightly less during power operation than during isothermal testing (compare Figures 510-10 and -11 with 510-5 and -6). This was expected, since a sodium density change with increasing temperature should cause a decrease in pressure drop. The head curves determined during power operation were very near predicted as listed in AI-P-1167, Hot Sodium Circulation Test, Figure 1167-16. Since the voltage signal from the flowmeters was used as a unit of flow in Figure 1167-16, it can be compared directly with Figures 510-10 and -11. New flowmeter calibration^{*} curves were made in AI-PC-530, Heat Transfer Tests. These calibration curves are shown in Figures 510-3 and -4 along with the originally recommended calibration curves. The later calibration curves are much different from the earlier curves and thus the system pressure drop was much different than predicted.

^{*}A. Divis and R. Large, "Heat Transfer Tests," AI-PC-530 (February 10, 1964), Figures 530-13 and -14.

2. Moderator Coolant Loop

The function of the moderator loop is to prevent temperature gradients across the various components within the reactor and thus minimize thermal stresses. Heat from the fuel elements is removed by sodium flow through the process tubes, whereas heat from the moderator elements, source, and control rods and dummy elements must be removed by the moderator coolant loop. Temperature of the various components within the reactor should be the same as that of the sodium within the fuel element channels at any elevation. Excessive temperatures would be controlled whenever the temperature gradients were controlled. The moderator elements are most susceptible to thermal stress.

Moderator coolant flow through the core is primarily convective flow. The space between the reactor core and the vessel wall offer a short circuit with practically no flow resistance for the moderator coolant. Convective flow through the core depends on a temperature difference between the core and the space between the core and the vessel wall. Sodium is free to flow across the core as well as to flow vertically.

The amount of heat removed by the moderator loop need not match the amount which is generated in the moderator, because heat is readily transferred between the moderator loop and the fuel element channels. During the test, the moderator coolant temperature adjacent to the fuel elements was almost independent of the moderator coolant flow rate as shown in Figure 510-12.

A gross temperature gradient was observed between the vessel wall and the center of the core. It has been predicted that local temperature gradient could exist due to mismatch of coolant flow rate to heat generation rate in individual flow channels around the source, around the control rods, and between the moderator elements. Only one moderator element had thermocouples installed; this element was in the approximate center of the core. The control rod thimbles do not have thermocouples attached. Other temperatures used in analyzing the moderator coolant system were taken from the reactor vessel wall, reactor temperature sensing elements, and the sodium within the fuel elements. The three reactor temperature sensing elements are outside the active core. Local temperature gradients could not be detected with the limited temperature instrumentation.

Figures 510-13 and -14 show that the vessel wall temperature decreased with an increase in moderator coolant flow rate due to either a change in moderator valve position and/or increase in sodium flow with reactor power. Figures 510-15 and -16 show a similar behavior for the temperature sensing elements which are near the outside of the core. The increase in moderator sodium flow tended to bypass the core by following the vessel wall until a convective head was produced to drive more sodium through the core. The convective head must have been caused by a decrease in the sodium temperature next to the vessel wall because moderator coolant temperatures within the core were controlled by heat transfer to the fuel channels.

Figures 510-17 and -18 show temperatures of various elements at two different power levels. Figure 510-17 exhibited a large temperature gradient across the moderator element at two elevations. Previous data did not exhibit this large temperature gradient. This large temperature gradient was not present on a geometrically similar part of the moderator element. Figure 510-19 shows that the temperature gradient was a function of power level.

Figures 510-20 and -21 show the temperatures of the sodium within the fuel channels. The other elements should match the temperature of the sodium within the fuel channels at any elevation. A cross reference of all the associated figures will show that it was not possible to eliminate temperature gradients across the moderator elements with any moderator valve setting when operating at higher power levels.

It can be concluded that the measured moderator flow rate should be less than that which would theoretically remove all of the heat generated in the moderator. The gross temperature gradient from the center of the core to the vessel wall may be reduced by reducing the moderator coolant flow rate, but this may cause local temperature gradients which were not measurable.

The optimum moderator flow rate is between that which would theoretically remove all of the heat generated in the moderator and that which would cause a large but permissible gross temperature gradient across the core. It should not be necessary to adjust the moderator flow control valve with changes in reactor power, because at low flow rates, where the ratio of the moderator coolant flow rate to total reactor flow changes most, the moderator temperature is least affected by changes in moderator coolant flow rate.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

The following are the test completion criteria associated with post critical test completion report AI-PC-510, Hydraulic Test of Main Heat Transfer System, as listed in NAA-SR-5700, Supplement 4, Page XI(c)-4, with remarks to explain compliance with the criteria.

a) Criterion: Sufficient operating data has been obtained at power levels, as required by the test, to allow determination of the following information regarding the hydraulic characteristics of the Sodium Heat Transfer System and its associated components at those power levels:

- 1) System sodium flow rates in pounds per hour.
- 2) Pressure rise in the system pumps.
- 3) System pump differential pressure in feet.
- 4) System hydraulic work factor.
- 5) System output work in horsepower.
- 6) System input power.
- 7) Approximate pump efficiency.
- 8) System pressure drop as a function of flow.
- 9) System efficiency as a function of flow.
- 10) Reactor pressure drop as a function of flow.
- 11) Pump transmitted horsepower as a function of coupling current.

Remarks:

- 1) Determination of system flow rate in pounds per hour was beyond the scope of this test. The relationship between sodium flow rate and sodium flowmeter reading was determined in AI-PC-530, Heat Transfer Test, on the basis of a heat balance. The sodium flowmeter was read as required by this test.
- 2) The pressure rise in the pumps is shown in Figures 510-7 and -8.

- 3) System pump differential pressure in feet is tabulated in Tables I through VI.
 - 4) System hydraulic work factor is tabulated in Tables I through VI.
 - 5) System output work in horsepower is tabulated in Tables I through VI.
 - 6) System input power is tabulated in Tables I through VI.
 - 7) Approximate pump efficiency is tabulated in Tables I through VI.
 - 8) System pressure drops are tabulated in Tables I through VI and shown in Figures 510-7 and -8.
 - 9) System efficiency is tabulated in Tables I through VI.
 - 10) Reactor pressure drop is shown in Figure 510-22.
 - 11) Pump transmitted horsepower is tabulated in Tables I through VI.
- b) Criterion: The system is demonstrated to perform in accordance with plant design parameters.

Remarks: The system operates satisfactorily as expected. Whether the moderator elements are cooled as expected was questionable, because of limited temperature instrumentation on the moderator elements.

V. EVALUATION

The purpose of post critical test AI-PC-510 was to determine the hydraulic characteristics of the main heat transfer system. Testing was conducted under both isothermal and power generation conditions. Except for determination of sodium flow rates, the test results comply with the test completion criteria listed in NAA-SR-5700, Final Summary Safeguards Report for the Hallam Nuclear Power Facility (Supplement 4). Calibration of the sodium flowmeters was performed under AI-PC-530, Heat Transfer Tests.

A. INITIAL TESTING

Post critical test AI-PC-510 is a continuation of the main heat transfer system hydraulic testing performed under preoperational tests AI-P-1103^{*} and AI-P-1167.[†] Several major pump and system deficiencies were uncovered and corrected during preoperational testing activities.

When the primary pumps were operated at high flow rates at less than design head conditions, helium cover gas above the free sodium surface in the pump barrel was entrained in the flowing sodium because of low pump barrel sodium levels. The low levels resulted from a condition of net flow reversal to the pump barrel. Flow from the barrel through weep holes to the pump suction side was greater than flow from the pump discharge side through the hydraulic bearing and barrel peripheral seal to the pump barrel. The condition was successfully corrected by plugging four of the eight weep holes in each of the primary pump impellers.

Progressive gas entrainment in the secondary sodium loops expansion tanks was eliminated by modifying the piping to bypass approximately 96% of the sodium flow around the expansion tanks. Continuous degasification is provided by the 4% flow through the expansion tank.

Testing after the modifications were performed verified that the deficiencies were corrected and the system was ready for low-power operation.

B. ISOTHERMAL TESTING

Isothermal testing, after correction of the deficiencies, provided the following required data:

- 1) Isothermal hydraulic characteristics of the main heat transfer system
- 2) Reactor pressure drop
- 3) Relationship between moderator loop and primary loop flow
- 4) Flow decay characteristics after scram.

^{*}P. F. Shaw, "Main Heat Transfer System," AI-P-1103, Atomics International (1963)

[†]P. F. Shaw, "Hot Sodium Circulation Test," AI-P-1167, Atomics International (1962)

The use of a flowmeter calibration that was found to be inaccurate during subsequent work caused the system head curves to be initially interpreted as agreeing with those predicted. The actual relationship between the large permanent magnet flowmeter millivolt signals and sodium flow rate was not determined until power testing during the heat transfer tests performed per AI-PC-530. Actual flows were about 23% greater than those expected for the primary sodium system and 17.7% greater for the secondary system.

The moderator loop flow to reactor flow ratio, with all loops operating, was found to be substantially constant over most of the flow range but the curves reveal a tendency for the ratio to increase at the lower flows. Moderator flows reported are indicated loop flow and do not include any estimate of leakage from the reactor inlet plenum across the grid plate to the moderator plenum.

The freeze seal throttle valves in both the primary and secondary sodium loops have caused operational difficulties and an excessive amount of maintenance. Minor modifications to increase valve operational reliability have not been completely successful. The valve topworks have been redesigned and new parts are on order to improve the reliability of these important components.

C. POWER OPERATION

1. Test Completion Criteria

Experimental and calculated data that show compliance with the test completion criteria are given in the form of tables except for the system and reactor pressure drops which are plotted versus flow in millivolts.

a. Pressure Measurements

Below about 40% of system full flow, pressure measurement data tend to scatter because of the low differential pressures and inherent inaccuracies in the process pressure instrumentation. The pressure instrumentation is also subject to zero point drift.

b. Hydraulic Work Factors

The system hydraulic work factors found in the tables are often termed the liquid horsepower or the hydraulic horsepower when expressed in horsepower units.

c. System and Pump Efficiencies

The system (motor-coupling-pump) and pump efficiencies are only approximate because of their dependence on pressure measurements and calculated motor input powers. Motor input power was calculated from phase current, an approximate motor voltage of 4160 v, and power factors obtained from the motor manufacturer's literature.

2. System Head Curves

The system head curves of Figures 510-10 and -11 are plotted as dashed lines in Figures 510-23 and -24. These figures are predicted pump and system characteristic curves provided with the test procedure. Also plotted are several points from Table I through Table VI, and a design data point from system description S. D. A81-070-1.

It is clear that the system head curves are different than those predicted. System heads at a given flow rate are roughly $1/2$ to $2/3$ of those predicted. Measured pump speeds are reasonably close to what might be predicted for the actual head curve intercepts.

The reasons for the discrepancy between predicted and actual system head curves are not known precisely. A 10% safety factor was allowed on the pipe friction pressure drops and at the time (December, 1959) the predicted curves were prepared there existed some uncertainty with respect to reactor core and steam generator pressure drops. Component pressure drops determined during the heat transfer test, AI-PC-530, are in good agreement with those predicted when corrections are applied for line pressure drop to the point of pressure measurement. In any event, the pumps have more than sufficient capacity to meet the lower than design head requirements of the main heat transfer system.

3. Moderator Cooling

Core component temperatures given in Figures 510-12 through -21 reveal that the instrumented moderator element and fuel element temperatures at any elevation are essentially independent of moderator loop valve position over the position range studied. However, reactor vessel wall and sodium temperature element temperatures are a function of moderator valve setting. Control rod and source temperatures at the time these data were obtained are not reported.

Undesirable temperature differences are shown to exist at full power between the various core components (Figure 510-17) although satisfactory differences exist at the lower power level of 14% (Figure 510-18). Although data were not obtained above the 80% valve closed position, examination of Figures 510-13 and -15 indicates that the full power temperature differences may be reduced by operating at a moderator valve position of about 85% closed. However, moderator temperatures probably will not be improved by a change from 75 to 85% closed because of their independence of moderator valve position (see Figure 510-12).

TABLE I
PRIMARY LOOP NO. 1 OPERATING PARAMETERS

Parameter	Units	Run Number					
		1	2	3	4	5	6
Sodium flow	mv	13.7	16.1	22.0	26.3	27.8	28.4
Sodium flow	lb/hr	1.42 (10) ⁶	1.67 (10) ⁶	2.30 (10) ⁶	2.82 (10) ⁶	3.04 (10) ⁶	3.13 (10) ⁶
Pump differential pressure	psi	7.0	8.8	17.4	24.9	27.8	29.5
Reactor differential pressure	psi	6.2	8.4	15.0	20.9	24.0	25.5
IHX differential pressure	psi	0.8	0.4	2.4	4.0	3.8	4.0
Pump head	ft	19.3	24.3	48.0	68.7	76.7	81.5
Hydraulic work factor	ft-lb/min	4.56 (10) ⁵	6.76 (10) ⁵	18.4 (10) ⁵	32.3 (10) ⁵	38.8 (10) ⁵	42.5 (10) ⁵
Pump output power	hp	13.8	20.5	56.4	97.9	118	129
Motor output power	hp	88	95	144	193	208	220
Pump speed	rpm	316	360	498	590	620	642
Pump input power	hp	31	38	78	127	144	158
Pump efficiency	%	45	54	72	77	82	82
Motor input power	hp	101	111	160	222	228	242
System efficiency	%	14	18	35	44	52	53
Coupling current	amp	1.02	1.24	1.78	2.10	2.24	2.40
Motor current	amp	22.3	22.9	26.3	30.5	32.0	33.4

TABLE II
PRIMARY LOOP NO. 2 OPERATING PARAMETERS

Parameter	Units	Run Number					
		1	2	3	4	5	6
Sodium flow	mv	13.7	16.1	21.9	26.1	27.6	28.8
Sodium flow	lb/hr	1.40 (10) ⁶	1.68 (10) ⁶	2.24 (10) ⁶	2.68 (10) ⁶	2.89 (10) ⁶	3.01 (10) ⁶
Pump differential pressure	psi	7.1	9.0	15.1	23.4	27.0	29.4
Reactor differential pressure	psi	5.4	7.5	12.4	19.3	21.9	24.3
IHX differential pressure	psi	1.7	1.5	2.7	4.1	5.1	5.1
Pump head	ft	17.6	24.8	41.6	64.5	74.5	81.4
Hydraulic work factor	ft-lb/min	4.97 (10) ⁵	6.95 (10) ⁵	15.5 (10) ⁵	28.8 (10) ⁵	35.4 (10) ⁵	40.8 (10) ⁵
Pump output power	hp	15.1	23.0	51.5	95.0	117	138
Motor output power	hp	81	95	140	193	208	226
Pump speed	rpm	322	372	508	600	632	658
Pump input power	hp	29.1	39	79	129	147	167
Pump efficiency	%	52	59	65	74	80	83
Motor input power	hp	96	110	157	212	229	249
System efficiency	%	16	21	33	45	51	55
Coupling current	amp	1.22	1.50	2.00	2.40	2.54	2.70
Motor current	amp	22.0	22.7	26.2	30.5	32.0	33.9

TABLE III
PRIMARY LOOP NO. 3 OPERATING PARAMETERS

Parameter	Units	Run Number					
		1	2	3	4	5	6
Sodium flow	mv	12.8	15.1	20.6	24.6	26.0	27.4
Sodium flow	lb/hr	1.37 (10) ⁶	1.62 (10) ⁶	2.20 (10) ⁶	2.62 (10) ⁶	2.78 (10) ⁶	2.98 (10) ⁶
Pump differential pressure	psi	6.6	8.9	17.0	23.7	26.0	28.8
Reactor differential pressure	psi	5.7	8.0	14.5	20.4	22.5	25.1
IHX differential pressure	psi	0.9	0.9	2.5	3.3	3.5	3.7
Pump head	ft	18.2	24.6	46.9	65.4	71.7	79.5
Hydraulic work factor	ft-lb/min	4.15 (10) ⁵	6.65 (10) ⁵	17.2 (10) ⁵	28.5 (10) ⁵	33.2 (10) ⁵	39.5 (10) ⁵
Pump output power	hp	13.7	22.0	56.7	94.0	110	130
Motor output power	hp	95	101	147	196	210	228
Pump speed	rpm	304	350	476	558	582	610
Pump input power	hp	32.2	39	78	122	137	156
Pump efficiency	%	43	56	73	77	80	83
Motor input power	hp	111	117	165	215	230	250
System efficiency	%	12	19	34	44	48	52
Coupling current	amp	0.9	1.10	1.60	2.00	2.10	2.30
Motor current	amp	22.9	23.3	26.6	30.9	32.2	34.0

TABLE IV
SECONDARY LOOP NO. 1 OPERATING PARAMETERS

Parameter	Units	Run Number					
		1	2	3	4	5	6
Sodium flow	mv	12.9	15.2	20.7	24.6	26.0	27.0
Sodium flow	lb/hr	-	-	-	-	-	-
Pump differential pressure	psi	7.1	8.6	16.5	22.7	25.9	28.0
IHX differential pressure	psi	3.0	4.5	7.2	10.2	11.6	12.5
Evaporator differential pressure	psi	-	-	2.3	4.5	6.3	5.5
Superheater differential pressure	psi	4.6	4.4	7.0	6.0	8.0	10.0
Pump head	ft	19.5	23.7	45.5	62.5	71.4	77.1
Hydraulic work factor	ft-lb/min	4.45 (10) ⁵	6.24 (10) ⁵	15.4 (10) ⁵	24.5 (10) ⁵	30.0 (10) ⁵	34.4 (10) ⁵
Pump output power	hp	13.5	18.9	46.6	74.3	90.9	104
Motor output power	hp	84	95	133	173	186	203
Pump speed	rpm	284	316	424	492	520	542
Pump input power	hp	24	33	63	95	108	123
Pump efficiency	%	56	57	74	78	84	85
Motor input power	hp	99	109	149	192	206	222
System efficiency	%	14	17	31	39	44	47
Coupling current	amp	1.20	1.44	1.90	2.30	2.40	2.60
Motor current	amp	22.2	22.8	25.5	28.8	30.0	31.5

TABLE V
SECONDARY LOOP NO. 2 OPERATING PARAMETERS

Parameter	Units	Run Number					
		1	2	3	4	5	6
Sodium flow	mv	13.1	15.4	21.0	25.0	26.4	27.5
Sodium flow	lb/hr	1.30 (10) ⁶	1.60 (10) ⁶	2.02 (10) ⁶	2.35 (10) ⁶	2.48 (10) ⁶	2.60 (10) ⁶
Pump differential pressure	psi	7.4	23.5*	16.4	22.4	24.7	26.6
IHX differential pressure	psi	2.4	18.7	7.8	10.4	11.5	11.3
Evaporator differential pressure	psi	2.2	3.0	3.6	6.0	6.4	7.4
Superheater differential pressure	psi	2.8	1.8	5.0	6.0	6.8	6.8
Pump head	ft	20.4	64.7	45.1	61.6	68.0	73.2
Hydraulic work factor	ft-lb/min	4.42 (10) ⁵	17.3 (10) ⁵	15.2 (10) ⁵	24.2 (10) ⁵	28.1 (10) ⁵	31.7 (10) ⁵
Pump output power	hp	13.4	52.5	46.1	73.4	85.1	96.0
Motor output power	hp	88	140	137	179	187	205
Pump speed	rpm	260	462	416	492	508	514
Pump input power	hp	25	72	64	99	107	118
Pump efficiency	%	53	73	72	74	80	81
Motor input power	hp	102	157	179	199	206	224
System efficiency	%	13	33	26	37	41	43
Coupling current	amp	1.18	1.90	1.80	2.20	2.30	2.42
Motor current	amp	22.4	26.2	25.7	29.4	30.1	31.7

*Sodium throttle valve partially closed.

TABLE VI
SECONDARY LOOP NO. 3 OPERATING PARAMETERS

Parameter	Units	Run Number					
		1	2	3	4	5	6
Sodium flow	mv	12.9	15.2	20.9	24.7	26.2	27.4
Sodium flow	lb/hr	1.24 (10) ⁶	1.45 (10) ⁶	1.93 (10) ⁶	2.26 (10) ⁶	2.41 (10) ⁶	2.53 (10) ⁶
Pump differential pressure	psi	5.3	7.7	15.0	20.4	22.5	25.1
IHX differential pressure	psi	1.8	3.6	7.5	9.9	11.0	12.0
Evaporator differential pressure	psi	1.6	2.4	3.6	5.2	6.0	6.2
Superheater differential pressure	psi	1.9	1.7	3.9	5.3	5.5	6.9
Pump head	ft	14.6	21.2	41.3	56.1	62.0	69.1
Hydraulic work factor	ft-lb/min	3.02 (10) ⁵	5.12 (10) ⁵	13.3 (10) ⁵	21.2 (10) ⁵	24.9 (10) ⁵	29.2 (10) ⁵
Pump output power	hp	9.2	15.5	40.3	64.3	75.5	88.5
Motor output power	hp	70	81	119	161	168	185
Pump speed	rpm	292	328	440	504	530	544
Pump input power	hp	23	30	67	91	100	112
Pump efficiency	%	40	52	60	71	76	79
Motor input power	hp	83	95	135	180	188	202
System efficiency	%	11	16	30	36	40	44
Coupling current	amp	1.00	1.22	1.80	2.08	2.20	2.40
Motor current	amp	21.4	21.9	24.5	27.8	28.7	29.9

APPENDIX

CALCULATION METHODS

1. Sodium Pressures

The sodium pressures were corrected for zero point drift. All of the pressure gages were read with zero sodium flow. The pressures were assumed to be zero with no sodium flow.

2. Sodium Flow

The sodium flow in lb/hr was obtained by converting the mv reading with the use of Figure 510-3, HNPF Primary Sodium Flowmeter Calibration, and Figure 510-4, HNPF Secondary Sodium Flowmeter Calibration.

3. Hydraulic Work Factor

The hydraulic work factor is the product of the pump head in feet and the sodium flow rate in lb/min. It is the power output of the pump.

4. Motor Output Power

The motor output power was found directly from the motor manufacturer's curve as a function of motor current.

5. Pump Input Power

The pump input power was calculated as the motor output power multiplied by the pump speed and divided by the motor speed. This calculation neglects friction losses.

6. Motor Input Power

The motor input power was calculated from the motor current and power factor. The power factor was obtained from the motor manufacturer's curve as a function of motor current.

Figure 510-1. Moderator Flow vs Moderator Throttle Valve Position

RATIO OF MODERATOR COOLANT FLOW TO REACTOR FLOW

Figure 510-2. Moderator Flow vs Reactor Flow

Figure 510-3. HNPf Primary Sodium Flowmeter Calibration

Figure 510-4. HNPf Secondary Sodium Flowmeter Calibration

Figure 510-5. Primary Pump Differential Pressure vs Sodium Flow

Figure 510-6. Secondary Pump Differential Pressure vs Sodium Flow

SODIUM FLOW RATE

Figure 510-7. Primary Sodium Flow Decay from Full Flow After a Simulated Scram

SODIUM FLOW RATE

Figure 510-8. Primary Sodium Flow Decay from 25% of Full Flow After a Simulated Scram

Figure 510-10. Primary Loop Pressure Drop

Figure 510-11. Secondary Loop Pressure Drop

Figure 510-12. Moderator Instrumented Element Temperatures vs Moderator Valve Position

Figure 510-13. Vessel Wall Temperature vs Moderator Valve Position

Figure 510-14. Vessel Wall Temperature vs Reactor Flow Rate (Power Level)

Figure 510-15. Temperature Element Temperatures vs Moderator Valve Position

Figure 510-16. Temperature Element Temperature vs Primary Flow Rate (Power Level)

Figure 510-17. Reactor Temperatures vs Core Elevation, Full Power Operation

Figure 510-18. Reactor Temperatures vs Core Elevation, 14% Power Operation

Figure 510-19. Moderator Temperatures vs Primary Flow Rate (Power Level)

Figure 510-20. Fuel Element Sodium Temperature vs Moderator Valve Position

Figure 510-21. Fuel Element Sodium Temperature vs Primary Flow Rate (Power Level)

Figure 510-22. Reactor Pressure Drop

Figure 510-23. Pump Performance and Predicted Primary Sodium System Characteristics

Figure 510-24. Pump Performance and Predicted Secondary Sodium System Characteristics

HALLAM NUCLEAR POWER FACILITY

Post Critical Test Procedure

AI-PC-530

Intermediate Heat Exchanger Performance

Test Procedure Submitted for Approval

6/30/61 James E. Rhens
Date Atomic International

Test Procedure Reviewed

12/18/62 D. E. Longman
Date Atomic Energy
Commission

Test Approved for Performance

12/31/62 James E. Rhens
Date Atomic International

Test Completed

7/29/63 James E. Rhens
Date Atomic International

Test Approved for Performance

12/31/62 R. C. Burtain
Date Atomic Energy
Commission

Test Completed

7/20/63 R. C. Burtain
Date Atomic Energy
Commission

HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
HEAT TRANSFER TESTS

AI-PC-530

Test and Analysis

by

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(Consumers Public Power District)

Evaluation

by

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Contract: AT(11-1)-GEN-8

February 10, 1964

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HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
HEAT TRANSFER TESTS, AI-PC-530

I. PURPOSE OF TEST

The purpose of this test was to obtain and correlate performance data on the HNPF heat transfer system, in order to verify that the steam generators and intermediate heat exchangers (IHX) were functioning properly. The test was also to verify the indicated reactor power level during the rise-in-power tests. Results obtained served a four-fold purpose in that:

- a) Satisfactory performance at each power level provided the basis for the next rise-in-power step.
- b) Heat transfer data provided a basis for comparison of indicated reactor power and computed reactor power.
- c) The computed primary and secondary sodium flows were used to calibrate the main heat transfer sodium flow meters.
- d) Resultant performance data were plotted for use as a permanent record of "new clean" performance for future comparative purposes in analyzing plant performance.

II. DISCUSSION OF TEST PERFORMANCE

A. SUMMARY

A total of 10 heat transfer tests were run in the rise-to-power testing program. The first four tests were performed at very low power levels, the highest being 38 Mwt or 15% of full load. All steam produced during this low level testing was discharged to the steam dump system. These initial tests served as a basis for obtaining Atomic Energy Commission (AEC) authorization to proceed to power levels above 15%. Since these tests were at such low power levels, the results are not plotted on the performance graphs. Values obtained for the heat transfer coefficient of each evaporator, superheater, and IHX at 38 Mwt are included in the tables. Results of the other low power tests were uncertain because of instrumentation inaccuracy at low readings.

Upon receiving authorization to exceed 15% power, power levels were increased in steps until full power was reached. During this time, six more sets of data were taken and analyzed. Steam produced while testing above 15% of full power (steam dump system capacity) was fed directly to the turbine-generator. Five of the six tests were run with the nuclear reactor supplying all the steam to the turbine-generator. The remaining test was performed with the nuclear reactor supplying steam to the turbine-generator in parallel with the conventional boiler. During five of the tests it was therefore possible to measure, by use of station wattmeters, the amount of electrical energy being produced by the nuclear reactor and the power required by auxiliary equipment. With this information, net and gross plant heat rates were calculated (Table IV). Data obtained from these tests were plotted and are to be used for future comparative purposes. After steady-state conditions had been attained, data for all tests were taken at one-half hour intervals for 3 hours. The resulting seven sets of readings were then averaged and used to obtain one set of data for that particular load.

B. INSTRUMENTATION

Since most of the sodium temperatures were read out on multipoint indicators which could not be read accurately enough to be of any value, it was necessary to read the thermocouple emf on a potentiometer. This was accomplished in such a manner as to allow one person, using a laboratory potentiometer, to obtain a complete set of sodium system temperatures. By using a thermostatically controlled oven, the potentiometer cold junction was maintained at a constant 150°F. The potentiometer read out in millivolts which were converted to degrees Fahrenheit. Prior to the commencement of testing, a set of temperature data was obtained at isothermal conditions. These data were then used to correct thermocouple readings obtained during the testing period. Another set of isothermal data was obtained at the end of the testing period and compared to that previously acquired.

Prior to testing, all sodium system pressure indicators were zeroed. This consisted of adjusting all gages in a particular loop to read the same at no-flow conditions but with the system full of sodium. In this way, pressure differentials due to differences in elevation were eliminated. After testing was completed, all gages were again read at the no-flow condition and values obtained were compared to data previously taken. No attempt was made to determine the pressure drops

attributable to pipe friction; pipe friction is therefore incorporated in the pressure drops across the elements in the system. (Figure 530-15 illustrates the sodium pressure indicators used and their relative location in the system.)

C. POWER LEVEL DETERMINATION

Power level was obtained from feedwater and/or steam flow and the known enthalpy rise across the steam generator from evaporator inlet to superheater-attemperator outlet. Steam flow was determined by measuring feedwater flow which was equivalent to steam flow. The two were equal as there was no blow-down and sampling flow was negligible. The heat transferred from secondary sodium to the steam and feedwater in each steam generator was calculated from these data. System losses were added to the heat transferred to obtain total reactor power.

D. ENERGY LOSSES

Losses to the cooling water system were determined by measuring water flow and temperature differential. These losses were determined for the three intermediate heat exchanger cell nitrogen coolers, primary pipe tunnel nitrogen cooler, loading face shield nitrogen cooling system, and the reactor cavity cooling system. By measuring the air flow and temperature rise through the ventilation system for the steam generator rooms, the heat given off by the steam generators was accounted for.

III. ANALYSIS AND PRESENTATION OF RESULTS

A. EVAPORATORS

The heat transfer coefficients calculated for the evaporators are listed in Table I, along with the expected (clean) heat transfer coefficients. These values vs total reactor power are plotted in Figure 530-1, and in Figure 530-2, vs heat transferred for each loop. Figure 530-1 also shows the expected coefficients for fouled conditions. The large discrepancy between calculated and expected values results from the method used in calculating log-mean temperature difference. At low loads, the values obtained for log-mean temperature difference are much greater than actual. This difference becomes smaller at higher loads as evidenced by the fact that expected and calculated values approach each other at high loads.

TABLE I
EVAPORATOR HEAT TRANSFER COEFFICIENTS

Evaporator No.	Power		Heat Transferred (Btu/hr)	Heat Transfer Coefficient (Btu/hr-°F-ft ²)	
	Mwe	Mwt		Expected (clean)	Calculated
1	-	38	32.98 x 10 ⁶	525	67.8
2	-	-	30.15 x 10 ⁶	525	64.7
3	-	-	34.53 x 10 ⁶	525	71.2
1	15	52	47.01 x 10 ⁶	580	117.2
2	-	-	51.33 x 10 ⁶	580	126.7
3	-	-	47.01 x 10 ⁶	580	117.2
1	25	81	74.57 x 10 ⁶	642	194.8
2	-	-	77.16 x 10 ⁶	642	206.5
3	-	-	72.15 x 10 ⁶	642	184.0
1	40	125	115.66 x 10 ⁶	700	350.0
2	-	-	115.76 x 10 ⁶	700	346.0
3	-	-	109.94 x 10 ⁶	700	322.8
1	50	148	136.44 x 10 ⁶	727	357.0
2	-	-	137.40 x 10 ⁶	727	391.0
3	-	-	134.23 x 10 ⁶	727	379.0
1	65	197	183.31 x 10 ⁶	760	513.0
2	-	-	182.69 x 10 ⁶	760	551.0
3	-	-	176.41 x 10 ⁶	760	535.0
1	80	244	220.69 x 10 ⁶	790	652.0
2	-	-	223.18 x 10 ⁶	790	697.0
3	-	-	214.70 x 10 ⁶	790	678.0

This deviation in log-mean temperature difference is due to two factors, one being sodium-to-sodium heat transfer and the other being the fact that at lower loads all the heat transfer is accomplished in a short portion of the tubes, rather than along the entire length. Because of the bayonet tube design, with the resulting sodium-to-sodium heat transfer, the sodium temperatures measured external to the evaporator are not the same as the temperatures leaving and entering the area where sodium-to-water heat transfer takes place. This regenerative heat transfer results in a calculated log-mean temperature difference which is greater than actual. Since the heat transfer coefficient is inversely proportional to the log-mean temperature difference, the values obtained for the coefficients are low. During operation at lower loads much of the heat transfer is accomplished along a portion of the total tube length. As the log-mean temperature is an integrated value along the whole length of the tube, the portion of the tube where heat transfer takes place is penalized by the remainder of the tube where very little transfer takes place. In other words, the tube area which is used in calculating the coefficients at less than full load should be some portion of the total area rather than the total area. As there is no positive means available for determining just what the active tube length is, or the amount of sodium-to-sodium heat transfer, the method used here is the best available at this time. Since the calculated values tend to approach predicted values at full load, the evaporators appear to be operating satisfactorily. No attempt was made, therefore, to correct the calculated values in order that these values would better agree with those predicted.

Referring to the sodium temperature summary sheets (Figures 530-18 thru 530-23), note that there is a large discrepancy between evaporator outlet temperature and IHX inlet temperature. This temperature difference is much too great to be due to heat losses. It was concluded that the evaporator outlet temperature is affected by stratification; therefore, the IHX inlet temperature was used for determining evaporator heat transfer coefficients. Using IHX inlet temperature and the superheater inlet temperature, sodium flow was calculated from the known heat input to the feedwater and/or steam. Using the calculated sodium flow, a sodium evaporator inlet temperature was calculated. This calculated temperature was used rather than the measured temperature as it appeared that the measured temperature was in error.

Figure 530-3 shows the predicted and actual sodium pressure drops through the evaporators. Points are plotted only for the three highest loads because the sodium pressure instruments are not capable of measuring the low differential pressures which exist at lower loads. The actual pressure drops agree quite closely with those expected.

B. SUPERHEATERS

Table II lists the expected and calculated heat transfer coefficients for the superheaters. A plot of these values vs reactor power is illustrated in Figure 530-4, and in Figure 530-5, vs heat transferred for each loop. Here again, the coefficient is affected by sodium-to-sodium heat transfer, although not to as great an extent as in the evaporators. The regenerative heat transfer results in low coefficients at the lower loads. However, as load increased and regeneration had less effect, the coefficients actually exceeded the predicted values. It appears, therefore, that the superheaters are performing satisfactorily. Since the first three tests were performed with four feedwater heaters and the last three tests with three feedwater heaters, and as the number of feedwater heaters in service does affect the coefficients slightly, "expected" curves are shown for both three and four feedwater heater operation.

Referring to the sodium temperature summary sheets, it is noted that the temperature of sodium to the superheater is measured at two points. In calculating heat transfer coefficients, the temperature nearest the superheater was used. The superheater outlet temperature was a calculated temperature as explained under Section A, Evaporators.

Figure 530-6 shows the predicted and actual sodium pressure drops through the superheaters. The slightly higher than expected pressure drops are due to pipe friction which is included in the values plotted.

C. ATTEMPERATOR VALVE EFFECT

Figure 530-17 illustrates the data obtained by varying attemperator valve position from full closed to full open while maintaining constant reactor power output. Attemperator valve position was varied by use of HCS-151, -251, and -351 which are located on the main control board. The attemperating valve on loop No. 3 failed to close completely. There is, therefore, a sizeable temperature differential between superheater outlet and attemperator outlet at indicated full closed position on HCS-351 (Loop No. 3).

TABLE II
SUPERHEATER HEAT TRANSFER COEFFICIENTS

Superheater No.	Power		Heat Transferred (Btu/hr)	Heat Transfer Coefficient (Btu/hr-ft ² -°F)	
	Mwe	Mwt		Expected (clean)	Calculated
1	-	38	7.02×10^6	65	27.5
2	-	-	6.54×10^6	65	24.7
3	-	-	7.13×10^6	65	28.4
1	15	52	10.67×10^6	82	49.1
2	-	-	11.39×10^6	82	49.3
3	-	-	11.13×10^6	82	48.8
1	25	81	17.72×10^6	107	81.6
2	-	-	18.39×10^6	107	82.0
3	-	-	17.13×10^6	107	82.0
1	40	125	28.30×10^6	137	142.0
2	-	-	28.35×10^6	137	131.8
3	-	-	26.92×10^6	137	133.5
1	50	148	30.42×10^6	152	141.2
2	-	-	32.89×10^6	152	142.9
3	-	-	30.36×10^6	152	147.8
1	65	197	41.59×10^6	180	183.5
2	-	-	44.35×10^6	180	185.0
3	-	-	40.69×10^6	180	185.0
1	80	244	56.06×10^6	207	233.0
2	-	-	56.96×10^6	207	224.0
3	-	-	54.44×10^6	207	226.0

D. INTERMEDIATE HEAT EXCHANGERS (IHX)

Table III lists the expected and calculated heat transfer coefficients for the combined (both units in each loop) intermediate heat exchangers. These values are plotted vs reactor power in Figure 530-7, and in Figure 530-8, vs heat transferred for each loop. Loop No. 3 IHX coefficient is below the predicted values, while loop No. 2 is above predicted and loop No. 1 is about equal to predicted. Since the units are essentially identical and should have equal heat transfer coefficients, the variation is due either to fouling or inaccuracies in temperature measurement. The latter is the most likely, since even a small error in temperature may result in a much larger error in heat transfer coefficient. In general, the combined heat exchangers appear to be performing satisfactorily as the heat transfer coefficients are all within 10% of that predicted. Heat transfer coefficients were not calculated for the individual heat exchangers because there was no means of determining the division of flow between the two units.

In the sodium temperature summary sheets, the secondary sodium temperatures used in calculating IHX heat transfer coefficients are the same temperatures used for superheater inlet and evaporator outlet as explained under Sections A, Evaporators, and B, Superheaters. This does not hold true for the periods that the cold trap was in operation. With the cold trap in operation, the secondary IHX outlet temperature used was the one shown before the cold trap return. Primary sodium temperatures used were the reactor outlet and reactor inlet temperatures which are those shown closest to the reactor. The temperatures around the individual units of each pair of units compare quite closely, indicating that approximately the same amount of heat transfer was being accomplished in each unit. Only in loop No. 1 is there much discrepancy in that the secondary outlet temperatures differ by as much as 12°F. This is due to the fact that unit "A" had 16 of its tubes sealed off during modification of the IHX's. The plugged tubes reduce the amount of primary sodium flow through unit "A" thereby reducing the secondary outlet temperature. The primary sodium outlet temperature from unit "A" was not available due to malfunction of the thermocouple.

Figures 530-9 and 530-10 show the actual and predicted pressure drops across the secondary and primary sides of the IHX's. Here again the losses due to pipe friction add to the actual IHX pressure drop. Pipe friction in the secondary loop is especially high due to the fact that the pressure drop across the

TABLE III
IHX HEAT TRANSFER COEFFICIENTS

Loop No.	Power		Heat Transferred (Btu/hr)	Heat Transfer Coefficient (Btu/hr-ft ² -°F)	
	Mwe	Mwt		Expected (clean)	Calculated
1	-	38	40.3 x 10 ⁶	600 *	634 *
2	-	-	37.2 x 10 ⁶	400	397
3	-	-	42.2 x 10 ⁶	420	427
1	15	52	58.5 x 10 ⁶	485	555
2	-	-	60.7 x 10 ⁶	500	530
3	-	-	58.1 x 10 ⁶	485	485
1	25	81	92.2 x 10 ⁶	615	585
2	-	-	95.4 x 10 ⁶	630	675
3	-	-	88.5 x 10 ⁶	605	593
1	40	125	143.9 x 10 ⁶	765	800
2	-	-	144.0 x 10 ⁶	765	835
3	-	-	136.9 x 10 ⁶	750	748
1	50	148	166.9 x 10 ⁶	870	815
2	-	-	170.2 x 10 ⁶	880	928
3	-	-	164.6 x 10 ⁶	870	804
1	65	197	224.8 x 10 ⁶	945	975
2	-	-	227.0 x 10 ⁶	950	1001
3	-	-	217.2 x 10 ⁶	930	912
1	80	244	278.9 x 10 ⁶	1025	1073
2	-	-	280.0 x 10 ⁶	1025	1120
3	-	-	269.2 x 10 ⁶	1010	1008

*Intermediate heat exchanger unit 1A was not in service during this test. Therefore, unit 1B was operating at approximately 30% of its design full power level.

throttle valve is included. Although the actual pressure drops exceed the predicted values, there does not appear to be any reason for concern.

E. SODIUM LOOPS PRESSURE DROP

Primary and secondary sodium loop head curves are presented in Figures 530-11 and -12, along with the predicted head curves. As can be seen, the actual head for a given flow is approximately equal to that predicted. The sodium flow is in terms of flowmeter signal (millivolts) to simplify future comparison. Figures 530-13 and -14 show calibration curves for the primary and secondary sodium flowmeters. These curves were developed from heat balance data obtained during this test and represent the best available calibration curves for the flowmeters.

F. REACTOR THERMAL OUTPUT

Reactor thermal output was calculated as previously described under Section II, Discussion of Test Performance. Table IV summarizes the thermal and electrical output data obtained during the tests. Losses from the heat transfer system to the cooling water system were essentially constant, as was expected, and were in the order of 0.7 Mwt. The loss from the steam generators was determined only at full load and was assumed constant throughout. This loss was 0.78 Mwt.

Reactor thermal output as calculated was compared to indicated power output from nuclear instruments. These data appear on Figure 530-16. Indicated and calculated thermal power compared quite closely.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with post critical test AI-PC-530, Heat Transfer, as listed in NAA-SR-5700, Supplement 4, pages XI(c)-5, XI(c)-9, and XI(c)-11 with remarks regarding the test to explain compliance with the criteria.

TABLE IV
REACTOR AND TURBINE-GENERATOR SUMMARY

Load Mwt	53.30 †				82.33				125.96			
Loop No.	Loop No. 1	Loop No. 2	Loop No. 3	Total	Loop No. 1	Loop No. 2	Loop No. 3	Total	Loop No. 1	Loop No. 2	Loop No. 3	Total
Steam Flow (lb/hr)	48,790	53,210	48,790	150,790	79,750	82,550	77,140	239,440	127,900	127,900	121,700	377,500
Secondary Na Flow (lb/hr)	534,000	555,980	526,100	1,616,080	809,380	816,700	776,100	2,402,180	1,384,200	1,303,000	1,267,800	3,955,000
Primary Na Flow (lb/hr)	541,400	561,660	556,000	1,659,060	870,000	895,000	849,460	2,614,460	1,439,600	1,461,800	1,402,000	4,303,400
Reactor Output (Mwt)	-	-	-	53.30	-	-	-	82.33	-	-	-	125.96
Losses (Mwt)	-	-	-	1.40	-	-	-	1.43	-	-	-	1.46
Generator Gross (Mwe)	-	-	-	15.04	-	-	-	25.21	-	-	-	40.96
Generator Net (Mwe)	-	-	-	12.47	-	-	-	22.63	-	-	-	37.69
Auxiliary Power (Mwe)	-	-	-	2.57	-	-	-	2.58	-	-	-	3.27
Net Heat Rate (Btu/kw)	-	-	-	14,587	-	-	-	12,416	-	-	-	11,406
Condenser Pressure In.Hg	-	-	-	1.46	-	-	-	1.30	-	-	-	1.47

Load Mwt	148.43 §				197.48				244.15			
Loop No.	Loop No. 1	Loop No. 2	Loop No. 3	Total	Loop No. 1	Loop No. 2	Loop No. 3	Total	Loop No. 1	Loop No. 2	Loop No. 3	Total
Steam Flow (lb/hr)	145,100	146,200	142,800	434,100	195,500	194,800	188,100	578,400	238,890	241,670	232,470	713,030
Secondary Na Flow (lb/hr)	1,470,000	1,467,000	1,452,700	4,389,700	2,025,000	2,000,000	1,939,000	5,964,000	2,586,000	2,523,800	2,442,800	7,552,600
Primary Na Flow (lb/hr)	1,652,000	1,637,000	1,588,700	4,877,700	2,330,000	2,246,000	2,181,000	6,757,000	2,866,000	2,831,000	2,755,000	8,452,000
Reactor Output (Mwt)	-	-	-	148.43	-	-	-	197.48	-	-	-	244.15
Losses (Mwt)	-	-	-	1.45	-	-	-	1.49	-	-	-	1.50
Auxiliary Power (Mwe)	-	-	-	-	-	-	-	66.04	-	-	-	81.32
Generator Gross (Mwe)	-	-	-	*	-	-	-	61.58	-	-	-	76.16
Generator Net (Mwe)	-	-	-	-	-	-	-	4.46	-	-	-	5.16
Net Heat Rate (Btu/kw)	-	-	-	-	-	-	-	10,944	-	-	-	10,941
Condenser Pressure In.Hg	-	-	-	-	-	-	-	2.06	-	-	-	2.75

*Parallel operation - Gross output not meaningful, as generator was being fed steam simultaneously from the Nuclear and Conventional Plants.

†53.30, 82.33, and 125.96 Mwt loads used four feedwater heaters;

§148.43, 197.48, and 244.15 Mwt loads used three feedwater heaters.

a) Criterion: (IHX)

Sufficient operating data has been obtained at power levels as required by the test to allow determination of the following information regarding the heat exchangers at those power levels:

- 1) Primary sodium enthalpy loss through the heat exchangers.
- 2) Secondary sodium enthalpy gain through the heat exchangers.
- 3) Calculated reactor thermal output.
- 4) Accuracy of the calibration of the nuclear power computer.
- 5) The mean ΔT associated with the heat exchangers.
- 6) System heat transfer coefficient.
- 7) Pressure drop versus flow for each primary and secondary sodium loop.

Intermediate heat exchangers perform in accordance with plant design parameters.

Remarks: The heat transfer coefficients for the intermediate heat exchangers were calculated. This necessitated the determination of primary and secondary enthalpy change through the heat exchangers and the log-mean temperature difference. Primary enthalpy change across the heat exchangers was used in calculating reactor thermal power. Reactor thermal power was then compared to the nuclear power computer indicated power. Pressure drops were determined and compared to predicted values. The heat exchangers operated satisfactorily throughout the testing period.

b) Criterion: (Evaporators)

- 1) Recorded data at each power level of the rise-to-power test are evaluated and an overall heat transfer coefficient is determined therefrom and compared to the predicted coefficient.
- 2) Evaporators are producing saturated steam at the flow and temperature and pressure conditions corresponding to the power level of the particular step of the rise-to-power test.

Remarks: Heat transfer coefficients were calculated and are plotted on Figures 530-1 and 530-2. The evaporators produced saturated steam in sufficient amounts and at the required conditions at all power levels.

c) Criterion: (Superheaters)

- 1) The recorded data at each power level of the rise-to-power test are evaluated and an overall heat transfer coefficient is determined therefrom and compared to the predicted coefficient.
- 2) The superheaters are producing superheated steam at the flow and temperature and pressure conditions corresponding to the power level of the particular step of the rise-to-power test.

Remarks: Heat transfer coefficients were calculated and are plotted on Figures 530-4 and 530-5. The superheaters produced superheated steam in sufficient amounts and at the required conditions at all power levels.

V. EVALUATION

A. GENERAL

Post critical test AI-PC-530 was performed to verify acceptable thermal and/or hydraulic performance of the evaporators, superheaters, and intermediate heat exchangers and to verify the accuracy of the reactor power as indicated on the nuclear instrumentation by comparison with the power calculated from a plant heat balance.

Plant heat balances were based on the steam flow which was assumed to be correct, and the secondary and primary sodium flow rates were then calculated from heat balances around the steam generators and IHX's, respectively. Steam flow was determined by measuring the feedwater flow with flowmeters and flow integrators. This method is justified since there was no evaporator blowdown during the test and the steam sampling flow was negligible. In this manner, it was not necessary to utilize readings from the individual loop steam and sodium flowmeters.

Use of a laboratory potentiometer to read out the millivolt signal from the thermocouples provided the best possible temperature data available from the plant thermocouples. Experience with the sodium pressure instrumentation demonstrated the lack of accuracy of these instruments, especially at low system flows and corresponding low system pressure drops. The practice of zeroing all gages at no-flow conditions during the test minimized errors associated with the pressure drop data.

B. EVAPORATORS AND SUPERHEATERS

As described in Section III A and B, direct comparison of the calculated and predicted heat transfer coefficients in the evaporators and superheaters is not justified due to the regenerative sodium-to-sodium heat transfer and the non-linear temperature profile along the length of the evaporator tube. Comparison can only be made at or very near full-power operation where the sodium-to-sodium heat transfer represents a small percentage of the total heat transferred, and the entire length of the evaporator tube is utilized for boiling heat transfer. As shown in Figure 530-1, at full power the calculated coefficients for all three evaporators fall between the coefficients predicted for clean and fouled conditions. Calculated coefficients for the superheaters (Figure 530-4) fall above those predicted for full-power operation.

Figure 530-24 shows the manufacturer's calculated steam generation rate and measured steam flow as a function of secondary sodium flow through the evaporators. Table V summarizes the design and measured superheated steam conditions at full-power operation. The good agreement between predicted and measured steam conditions shown in both Figure 530-24 and Table V, and the close agreement between predicted and measured heat transfer coefficients at full-power operation, demonstrates the capability of the steam generating equipment to produce steam of sufficient temperature and pressure and in sufficient quantity to generate the specified net electrical output of the plant.

Since, during the performance of this test, the plant was not operated beyond the nominal full-power rating, it is not possible to evaluate the performance of the evaporators and superheaters at their design base load rating of 256 Mwt or 251,000 lb/hr steam flow. However, calculations indicate that if the feedwater temperature can be maintained at the design value of 303°F, satisfactory performance will be achieved with the design sodium flow rate of 2.8×10^6 lb/hr and the design sodium ΔT .

The measured sodium side pressure drop for both the evaporators and superheaters are in excellent agreement with those predicted by the manufacturer. All deviations between expected and measured values are within the accuracy of the pressure instrumentation.

TABLE V
FULL POWER SUPERHEATER PERFORMANCE

Parameter	Design Value	Measured Value*
Steam flow (lb/hr)	710,000	717,000 [†]
Steam temp (°F)	825 [§]	870
Steam press (psig)	800 [§]	829**
Reactor power (Mwt)	240	237
Net electrical output (Mwe)	75	75

*Operational data taken 7/31/63.

[†]Average of individual loop and main header flow meters.

[§]Design steam conditions at turbine throttle.

**Measured on PRC-501 upstream of turbine.

C. INTERMEDIATE HEAT EXCHANGERS

Calculation of overall heat transfer coefficients for the intermediate heat exchangers showed good agreement between predicted and measured values. Discrepancies between the coefficients for the three loops and between the predicted and measured values are within $\pm 12\%$. This error is, in all probability, due to the unequal flow distribution between the three exchangers and errors in temperature measurements. In the case of the IHX, comparison of the overall heat transfer coefficients is a legitimate measure of performance as the temperature driving force is essentially constant over the entire tube length. The deviation between expected and measured pressure drops, for both the primary and secondary sodium sides, can be accounted for by the pressure drops in the piping between the points of pressure measurement and the IHX. Based on these comparisons, the test has demonstrated the satisfactory thermal and hydraulic performance of these units.

D. REACTOR THERMAL POWER

Measurement of heat gain of the plant cooling water and ventilation air in the steam generator room provided data for the heat losses from the heat transfer system. By including these losses in the heat balance calculations, the accuracy of calculated reactor thermal output is improved. The heat balance calculations

provide the basis for calibration of the nuclear instruments following changes in core configurations and/or repositioning of the nuclear channels. This method of calibration is justified only when a complete and accurate heat balance is made using a test procedure and temperature measurement system similar to that described in Sections II-B and III, respectively.

Figure 530-1. Evaporator Heat Transfer Coefficient

Figure 530-2. Evaporator Heat Transfer Coefficient

Figure 530-3. Evaporator Sodium Pressure Drop

Figure 530-4. Superheater Heat Transfer Coefficient

Figure 530-5. Superheater Heat Transfer Coefficient

Figure 530-6. Superheater Sodium Pressure Drop

Figure 530-7. IHX Heat Transfer Coefficient

Figure 530-8. IHX Heat Transfer Coefficient

Figure 530-9. Secondary IHX Pressure Drop

Figure 530-10. Primary IHX Pressure Drop

Figure 530-11. Secondary Sodium System Head Curve

Figure 530-12. Primary Sodium System Head Curve

Figure 530-13. HNPf Primary Sodium Flowmeter Calibration

Figure 530-14. HNPf Secondary Sodium Flowmeter Calibration

Figure 530-15. Sodium Pressure Instrumentation

Figure 530-16 Neutron Flux-Power Relationship

Figure 530-17. Superheater and Attemperator Outlet Temperature

Figure 530-18. 53.3 MWt Temperature Summary

Figure 530-19. 82.3 MWt Temperature Summary

Figure 530-20. 126 MWt Temperature Summary

Figure 530-21. 148.4 MWt Temperature Summary

Figure 530-22. 197.5 MWt Temperature Summary

Figure 530-23. 244.2 MWt Temperature Summary

Figure 530-24. Steam Generator Part Load Capacities

HALLAM NUCLEAR POWER FACILITY

Post Critical Test Procedure

AI-PC-552

Steam Generator Steam Purity

Test Procedure Submitted for Approval

6/30/61 James E. Owens
Date Atomic Energy International

Test Procedure Reviewed

12/14/62 J. E. Simpson
Date Atomic Energy
Commission

Test Approved for Performance

12/28/62 James E. Owens
Date Atomic Energy International

Test Completed

7/20/63 James E. Owens
Date Atomic Energy International

Test Approved for Performance

12/28/62 R. C. Buntain
Date Atomic Energy
Commission

Test Completed

7/20/63 R. C. Buntain
Date Atomic Energy
Commission

HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
STEAM GENERATOR – STEAM PURITY

AI-PC -552

Test and Analysis
by
L. E. KELLER
(Consumers Public Power District)

Evaluation
by
G. LANDLER

A T O M I C S I N T E R N A T I O N A L
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P. O. Box 309 Canoga Park, California

Contract: AT(11-1)-GEN-8

February 5, 1964

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HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
STEAM GENERATOR - STEAM PURITY, AI-PC-552

I. PURPOSE OF TEST

The purpose of this test was to evaluate the performance of the steam generators with respect to steam purity at incremental power levels from 0 to 100% power. Satisfactory performance at each power level was required before rising to the next higher power level.

II. DISCUSSION OF TEST PERFORMANCE

A. INTRODUCTION

Purity of the dry saturated steam was used to evaluate performance of each of the three steam generators. Satisfactory performance of a steam generator was defined as follows:

- 1) The amount of carryover must be less than 0.25% at all times.
- 2) The steam generator must deliver at least 251,000 lb/hr of steam.

Steam purity is an important characteristic of steam generator operation because of detrimental effects which evaporator water solids have on components downstream of the eliminator. Tests which were performed on a prototype steam generator^{*} showed that the superheater became fouled and partially plugged when an excessive amount of solids was allowed to carry over with the steam. It is also known that solids, carried by steam, can deposit on the turbine blades and significantly decrease turbine efficiency.

In the test results which follow, the steam purity is expressed in terms of "percent carryover," i.e., the amount of liquid water entrained in steam leaving the separator, expressed in percent. A quantitative test for sodium

^{*}L. J. Webster, "Operation and Analysis of a 3,000 kw Liquid Metal Model Steam Generator," NAA-SR-4884 (February 28, 1961)

was used to determine the amount of solids in both the evaporator water and the condensed steam samples. Thus,

$$\% \text{ carryover} = \frac{\text{ppm Na in the Steam}}{\text{ppm Na in the Evaporator Water}} \times 100$$

The sodium analyses were made using a modified Beckman DU Spectrophotometer and specially constructed polyethylene sample bottles.

B. SYSTEM DESCRIPTION

The steam generator units consist of three horizontal vessels (evaporator, superheater, and eliminator), stacked one above the other, with interconnecting piping. The superheaters are installed above the evaporators and below the eliminators. Sodium flows first through the superheaters and then through the evaporators. Feedwater flows to the evaporator where it is evaporated and exhausted to the eliminator. Entrained moisture is separated in the eliminator and returned to the evaporator. Dry saturated steam flows from the eliminator through the superheater and into the main steam line. A controlled quantity of dry saturated steam is fed into the superheater outlet to provide regulation of the temperature of the steam entering the steam header.

The superheaters are horizontal vessels about 5 ft OD by about 20 ft over-all length. Each superheater contains approximately 560 tubes, each about 10 ft long by 1-1/2 in. OD. Each tube consists of four distinct walls forming a bayonet-type arrangement. The outer two walls form the heat transfer tube with sodium inside and steam outside. The space between the two walls is filled with helium under pressure which serves as a leak monitor. The circumference of the inner wall is grooved axially and the two walls are expanded together for mechanical contact to maximize heat transfer. The inner two walls form the core tube. The space between walls is evacuated and filled to atmospheric pressure with argon to minimize sodium-to-sodium heat transfer. Sodium flow is into the annulus between core and heat transfer tube and out through the core tube. The heat transfer surface of each superheater is 2150 ft².

The evaporators are kettle-type evaporators approximately 32 ft over-all length by 7-1/2 ft diameter. Each evaporator contains approximately 680 tubes about 166 in. long by 1-1/2 in. OD. Each tube consists of four distinct walls

similar in construction to the superheater tubes. However, in the case of the evaporator, sodium flow is reversed so that sodium flows into the core tubes and returns through the annulus. Internal baffle boxes are installed beneath the steam outlet connections to decrease moisture entrainment in the steam.

The eliminators are horizontal vessels approximately 21 ft long by 2-1/2 ft diameter containing a specially designed baffling arrangement which removes entrained moisture from the flowing steam by centrifugal flow passage. This moisture is returned to the evaporator; essentially dry steam passes out of the top of the eliminators into the steam superheaters with a portion of the dry saturated steam being bypassed around the superheaters for superheat control.

Steam sampling nozzles are provided in each steam generator to test the quality of steam in two of the four risers from the evaporator to the eliminator, the line leaving the eliminator, and in the superheated steam line leaving the steam generator complex. Any one of the four steam samples can be sent to a Larson-Lane steam purity analyzer. An evaporator water sample was provided from the continuous blowdown line. This sample passes through a sample cooler, a conductivity cell, and a pH cell in series. A six-point conductivity recorder continuously records the conductivity of the evaporator water and of the Larson-Lane steam purity analyzer from each steam generator. A three point pH recorder continuously records the pH of the three evaporator samples. A flow diagram of the original sampling and monitoring system is shown in Figure 552-1.

Several modifications of the sampling system were necessary to obtain the required test data. A cooler for the steam samples was installed prior to the test program; other modifications were made later. By the end of the test program any one sample point could be monitored continuously for sodium. Figure 552-2 shows the final steam generator sampling system.

C. TEST PROGRAM

The test procedure was followed quite closely up to and including the 40 Mwe test. The first difficulty with steam purity was experienced at 50 Mwe; thereafter the test procedure was modified and tests were made as required to define the problem and to insure acceptable steam purity at all times. Special tests were made on both steam generator No. 1 and No. 3 to pinpoint the limiting steam flows. Steam generator No. 1 was then modified to provide exact

Figure 552-1. Steam Generator Sampling System As Originally Installed

Figure 552-2. Steam Generator Sampling System As Modified

measurement of the elevation of the steam-water interface in the evaporator. The results of these measurements made it possible to achieve full power by lowering the setpoint on the evaporator water level recorder controllers (LRC).

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

A. 0 TO 38 Mwt TESTING

During the first portion of the 2.5 Mwt test the sodium content in the steam was quite high in all three steam generators. The high values are attributed to the following two factors:

- 1) clean up of sample lines
- 2) extremely low steam flows.

During the remainder of the 0 to 38 Mwt testing the carryover from all three eliminators was between 0.01% and 0.10%. Since these power levels are below the normal operating range of HNPF, no attempt was made to correlate percent carryover with steam flow rate. The steam purity was considered to be acceptable if the percent carryover was less than 0.25% and did not indicate a definite upward trend. The results of these tests are shown in Table I.

B. TESTING BETWEEN 38 Mwt AND FULL LOAD

Carryover determinations were made at 13 Mwe, 19 Mwe, and 42 Mwe (see Table II). At 42 Mwe, it was noted that the steam leaving evaporators No. 1 and No. 2 contained more moisture than at lower power levels; however, the purity of the dry saturated steam was well below the 0.25% carryover limit.

Approximately 45 min after the reactor power level reached 50 Mwe for the first time, carryover determinations were made on all three steam generators (see Table III). The results indicated the existence of a serious carryover problem. The subsequent testing programs were organized, one at a time, in an attempt to define the problem. Each test program is explained in the following paragraphs.

1. Steam Generators No. 1, 2 and 3 (May 29, 1963)

Immediately after the gross carryover condition was recognized, the reactor power was reduced to 40 Mwe and evaporator water level was lowered

TABLE I

STEAM PURITY TEST RESULTS

Reactor Power (Mwt)	Percent Carryover From Eliminators		
	Steam Generator No. 1	Steam Generator No. 2	Steam Generator No. 3
2.5	0.970	1.65	0.120
	0.002	0.015	0.020
10	0.019	0.040	0.040
	0.034	0.010	0.010
	0.071	0.030	0.030
	0.028	0.038	0.020
	0.023	0.011	0.023
	0.090	0.012	0.012
	0.030	0.040	0.010
	0.010	0.007	0.010
	0.044	0.012	0.003
20	0.023	0.010	0.010
	0.022	0.097	0.025
	0.030	0.028	0.031
38	0.017	0.016	0.016
	0.015	0.029	0.028
	0.019	0.048	0.010

TABLE II
PERCENT CARRYOVER

Reactor Power (Mwe)	Steam Generator No. 1			Steam Generator No. 2			Steam Generator No. 3		
	Riser A	Riser B	Eliminator	Riser A	Riser B	Eliminator	Riser A	Riser B	Eliminator
13	0.37	0.10	0.10	0.47	0.14	0.11	0.059	<0.01	<0.01
19			0.16			<0.01			0.04
42	0.51	0.40	0.02	0.49	1.02	<0.01	0.03	0.02	<0.01

TABLE III
SPECIAL TEST OF STEAM GENERATOR NO. 1
(May 29, 1963)

Reactor Power (Mwe)	Percent Carryover From Eliminators		
	Steam Generator No. 1	Steam Generator No. 2	Steam Generator No. 3
50	23.2	4.18	0.03
40	0.12	0.01	0.02
	0.23	0.41	
	0.09	0.01	
42	0.14	0.05	0.05
44	0.01	0.01	0.01
	0.26		
46	0.12	0.02	0.09
	0.04		
	0.01		
	0.07		
	0.05		
	0.03		
	0.05		
	0.03		
	0.32		
	0.03		
	0.01		

two inches. Several carryover determinations were made as the reactor power was slowly increased to 46 Mwe (see Table III). At this time it was evident that carryover was erratic and that a single sample was not adequate to define existing conditions.

2. Steam Generator No. 1 (May 31 and June 1, 1963)

During this test period, loop No. 1 was placed on hand control and steam flow from this steam generator was raised incrementally. Two evaporator water levels were used, namely "0" inches and "-2.0" inches on the LRC. The eliminator steam sample was allowed to flow through a group of sample bottles in series. At the specified sampling time, the flow was stopped and each sample bottle was taken to the laboratory for analysis; the results are shown in Table IV.

This test proved that evaporator water level is an important parameter in controlling the amount of carryover and that a lower water level results in less carryover.

3. Steam Generator No. 3 (June 8 and 9, 1963)

The next test program was designed to find the limiting steam flow for steam generator No. 3 since it had produced less carryover than either No. 1 or No. 2 under identical conditions. The LRC setpoint was placed at -2.0 inches. Two of the four evaporator risers and also the eliminator effluent line were sampled simultaneously. A group of sample bottles in series was used for each sample point. The results (see Figure 552-3) show that the guarantee of 0.25% was exceeded at approximately 175,000 lb/hr steam flow as indicated on FR-351. It was also observed that evaporator riser A was carrying over more solids than riser B.

4. Steam Generator No. 1 (July 3 through July 19, 1963)

Previous tests had shown that the evaporator water level was an important factor in controlling carryover. In order to establish the exact elevation of the steam-water interface, a total of fourteen probes, or sample lines, were placed inside evaporator No. 1 (see Figure 552-4). These probes terminated at known locations; the solids content of the sample would determine whether the end of the probe was in the steam phase or water phase.

Evaporator No. 1 was filled with water prior to startup and the probes were used to establish the level column readings at several known water levels.

TABLE IV
SPECIAL TEST OF STEAM GENERATOR NO. 1
(May 31 and June 1, 1963)

Steam Flow FR-151 (lb/hr)	LRC (in.)	Percent Carryover from the Eliminator	
		Maximum	Minimum
111,000	-2.0	0.20	<0.01
115,800	-2.0	0.07	<0.01
122,400	-2.0	0.26	0.09
128,400	-2.0	0.15	<0.01
133,500	-2.0	0.35	0.01
140,400	-2.0	0.27	0.02
148,200	-2.0	0.36	0.01
151,500	-2.0	0.30	0.03
159,000	-2.0	0.16	0.02
171,000	-2.0	0.39	0.17
139,500	-2.0	0.16	0.01
139,500	0	25.2*	17.6*

*Note: These values were increasing with time.

Figure 552-3. Special Test of Steam Generator No. 3,
June 8-9, 1963

This correlation showed that the LRC had been reading approximately 5 in. lower than the actual water level. The LRC scale was adjusted to correct for this error; this adjustment made possible the use of LRC readings as low as -5 in. in succeeding power runs without exceeding any of the plant protective setpoints.

During the next power run the LRC setpoint on No. 1 steam generator was lowered as necessary to keep the water level approximately 2 in. above the top of the tubes at the tubesheet end of the evaporator. The LRC setpoints on steam generators No. 2 and No. 3 were also lowered by the same amount. At full power the LRC setpoint was approximately 10 in. lower than the original LRC setpoint. It was determined that the LRC setpoint must be programmed as a function of steam flow, to ensure that the tubes will remain covered, and that excessive carryover will not occur. Figure 552-5 shows the operating curve for programming the LRC setpoints.

After the post critical test program was completed, sufficient time was available to more accurately measure the carryover from all three steam generators. These results (see Table V) show that the percent carryover requirement is met at full power design steam flow but the results are marginal. The probes in steam generator No. 1 indicate that the water level is one inch above the tubes when the LRC reads -5.3 in. and the steam flow is 251,000 lb/hr.

C. SUMMARY AND CONCLUSIONS

From the results of these tests, it is evident that the basic problem causing the excessive carryover was attributable to the inability of the level devices to accurately indicate the evaporator water level.

It was possible to achieve full power by maintaining the evaporator water level at the proper elevation. The allowable deviation in evaporator water level at full power appears to be in the order of one inch. More testing is required to define the maximum water level which can be tolerated without exceeding the 0.25% carryover limit, and also the minimum water level which can be used and yet keep the sodium tubes covered. These limits must be clearly defined for successful operation of the steam generators.

Figure 552-5. HNPF Evaporator Water Level Control Point

TABLE V
HNPF STEAM GENERATOR PERFORMANCE AT FULL LOAD

	Steam Generator No. 1	Steam Generator No. 2	Steam Generator No. 3
Steam flow, FR-151, -251, -351 (lb/hr)	250,000	254,000	246,000
Evaporator water level, LRC (in.)	-4.5	-4.5	-4.9
Feedwater temperature (°F)	375	375	375
Sodium concentration in evaporator (ppm)	38.7	17.4	17.3
Sodium concentration in the steam from the eliminator (ppb)	57.0	42.0	44.0
Carryover (%)	0.15	0.24	0.25

IV. COMPLIANCE WITH TEST COMPLETION CRITERION

Following is the test completion criterion associated with post critical test AI-PC-552, Steam Generator – Steam Purity, as listed in NAA-SR-5700 (Supplement 4) pages XI-(c)-16, with remarks regarding the test to explain compliance with the criterion.

Criterion: Steam purity at full load conditions, and at anticipated transient conditions, is satisfactory (approximately 0.25% moisture and 1 ppm solids in the steam) for the superheater and the turbine.

Remarks: Steam purity at full load conditions was less than 0.25% moisture and less than 1 ppm solids in the steam. No testing was accomplished under anticipated transient conditions; this is scheduled for future tests.

V. EVALUATION

Post critical test AI-PC-552 has produced sufficient evidence that steam purity at full load conditions complied with the test completion criterion. However, compliance with this criterion at anticipated transient conditions must be proved by additional tests to be performed in the future.

The data from this test resulted in a program diagram (Figure 552-5) for the evaporator level control setpoints at varying steam generator steam flows. This diagram places the setpoints between minimum and maximum allowable indicated evaporator water levels defined by the following criteria: the minimum reading is that which maintained one inch of water above the top of the tubes at a distance of 27 in. from the helium-water tube sheet, and the maximum reading is that which resulted in a water level where the carryover from the eliminator is just under 0.25%.

The curve for programming the LRC setpoints was based on the minimum and maximum allowable level curves obtained by measurements performed only on steam generator No. 1. The minimum level measurements were not done for steam generators No. 2 and 3. The validity of the setpoint curve was verified from a carryover viewpoint for all three steam generators by performing carryover analyses up to and including essentially full power steam flows. The full power steam flow carryover numbers as given in Table V are shown in Figure 552-5, thus showing compliance with the test completion criterion and verifying the validity of the setpoint curve at these conditions.

The validity of the setpoint curve was proved for steam generator No. 1, but apparently only a narrow water level range permits compliance with the steam purity requirements at design flow rate. This range is decreasing with higher flow rates and sets the limit of maximum steam flow. Consequently, additional tests should be performed on the other two units to verify applicability of the curve to all three steam generators. These tests will also provide the data establishing the correlation between actual evaporator water levels and the readings on the LRC scale.

The problems which have arisen during the performance of the steam purity tests required modification of the steam generator sampling systems and these improvements are valuable for the future operation of the plant. The installation of water level probes contributes to the more accurate evaluation of the water level indicator.

HALLAM NUCLEAR POWER FACILITY

Post Critical Test Procedure

AI-PC-560

Cooling System Tests

Test Procedure Submitted for Approval

6/30/61 James E. Rhone
Date Atomic Energy International

Test Procedure Reviewed

12/14/62 W. E. Simpson
Date Atomic Energy
Commission

Test Approved for Performance

12/21/62 James E. Rhone
Date Atomic Energy International

Test Completed

7/20/63 James E. Rhone
Date Atomic Energy International

Test Approved for Performance

12/21/62 W. E. Buntain
Date Atomic Energy
Commission

Test Completed

7/20/63 W. E. Buntain
Date Atomic Energy
Commission

HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
COOLING SYSTEM
AI-PC-560

Test and Analysis

by

N. G. DUSANEK

Evaluation

by

E. J. FIRSICK

A T O M I C S I N T E R N A T I O N A L
A DIVISION OF NORTH AMERICAN AVIATION, INC.

P. O. Box 309

Canoga Park, California

Contract: AT(11-1)-GEN-8

January 15, 1964

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HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
COOLING SYSTEM, AI-PC-560

I. PURPOSE OF TEST

The purpose of this test was to determine whether the auxiliary cooling systems were adequate for cooling the reactor cavity, sodium pipe galleries, sodium containment cells, loading face shield, and the heating and ventilating equipment.

II. DISCUSSION OF TEST PERFORMANCE

Temperatures of all the areas and equipment cooled were recorded at least three times after each increase in reactor sodium temperature, and at several power levels ranging between 15 Mwe and 82 Mwe.

A. REACTOR CAVITY AND PIPE GALLERIES

The reactor cavity and pipe galleries are cooled by flowing water through cooling coils embedded in the concrete walls. Heat is removed from the water through a heat exchanger, using cooling water from the auxiliary cooling tower. The concrete temperatures are read from thermocouples embedded in the concrete. The return temperature of each cooling coil is measured in thermo wells with a contact pyrometer. The total flow rate and main inlet and outlet temperatures are read from the process instrumentation.

B. LOADING FACE SHIELD

The loading face shield (LFS) is cooled by pressurized nitrogen circulating in a closed loop through coils in the loading face shield. Circulation is accomplished by means of a reciprocating compressor operating at 235 psig design discharge pressure. Heat is removed from the nitrogen through a heat exchanger utilizing cooling water from the auxiliary cooling tower.

The loading face shield temperatures are read from thermocouples embedded in the concrete near the bottom of the shield, while the cooling coil

return temperature is measured in thermo wells by a contact pyrometer. Nitrogen flow rate, pressures, and temperatures are recorded from the associated process instrumentation.

C. SODIUM CONTAINMENT CELL COOLERS

The primary pipeway, the three intermediate heat exchanger (IHX) cells, the primary service cell, and primary storage tank cell have nitrogen atmospheres. The nitrogen in these cells is circulated through coolers to maintain in-cell temperatures within the desired limits. An additional nitrogen cooler is used to cool the primary cold traps. Nitrogen pressure in the system is nominally atmospheric. Water from the auxiliary cooling tower is used in the nitrogen coolers, which are similar in design to conventional heating and ventilating equipment. Cool nitrogen is ducted to the freeze traps and the sodium valve freeze seals which are within the cells, but the bulk of the nitrogen flows directly into the cells for general area cooling.

For the test, mercury thermometers were used to measure the IHX cell nitrogen cooler inlet and outlet temperatures. Process instrumentation was used to measure nitrogen cooler inlet temperature, cooling water outlet temperature, and cooling water flow rate. The freeze trap and sodium valve freeze seal temperatures were read from permanently installed thermocouples.

D. SECONDARY SODIUM SERVICE

Recirculating air coolers using water from the auxiliary cooling tower are used to cool the freeze traps, the sodium throttle valve freeze seals, the secondary sodium area, and the IHX cell nitrogen cooler room. The freeze trap and secondary sodium valve seal temperatures were recorded periodically from permanently installed thermocouples. A week-long series of checks at 4-hr intervals was performed in the IHX cell nitrogen cooler room and the secondary sodium service area after these areas became uncomfortably warm. A standard laboratory type mercury thermometer was used for the temperature checks.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

A. REACTOR CAVITY AND PIPE GALLERIES

One area in the reactor cavity biological shielding was not adequately cooled. Location of the thermocouples which read high and the geometry of this area are shown in Figure 560-1. The highest recorded temperature for each of the four thermocouples in the area was 197, 132, 189, and 167°F. Specified maximum allowable temperature is 175°F. Conceivably, the concrete in the loading face shield near the same area was also too hot, but there are no thermocouples installed there to allow verification of this condition. It is believed that the high temperatures do not extend beyond the bottom cooling coil because, as indicated by the cooling water return temperatures, the upper cooling coils do not remove as much heat.

All of the cooling water valves, except those serving the hot area, were throttled. The main supply throttle valve was then opened to obtain more flow in the unthrottled coils serving the hot area. If maintenance work requires valve manipulations, it is recommended that all of the throttle valves be set at one-half turn open from tightly closed, except valves V-6474, V-6475, V-6476, V-6477, and V-6185 which should be fully open (the last valve listed is discussed below).

Three cavity cooling water coils developed leaks and were, therefore, disconnected from the cooling system. These coils, shown in Figure 560-2, served the east pipe gallery. Actual concrete temperature in this area is not known, since there are no thermocouples in the immediate vicinity of the leaky coils. Data for the reactor cavity cooling system are tabulated in Tables I and II. The one remaining cooling coil in the area where the two coils leak was fully opened (V-6185). The concrete temperature in the pipe galleries where thermocouples are located was more than 50°F less than the specified maximum allowable operating limit. Therefore, the loss of these cooling coils is not expected to result in overheating of the concrete.

CROSS SECTION OF REACTOR LOADING FACE SHIELD
AND ADJACENT SUPPORT STRUCTURE

Figure 560-1. LFS Thermocouple and Cooling Water Line Location

Figure 560-2. Core Cavity Cooling Circuits

TABLE I
REACTOR CAVITY COOLING SYSTEM

Cooling water flow	FI-626	145 gpm
Cavity supply temperature	TI-619	80° F
Cavity return temperature	TI-648	90° F
Cooling coil return temperature nearest hot spot in concrete	TP-1721	99° F
Maximum cooling coil return temperature excluding the above		94° F

TABLE II
HEAT EXCHANGERS

Closed Loop Side		
Water flow	FI-618	198 gpm
Inlet temperature	TI-616	88° F
Outlet temperature	TI-619	80° F
Open Loop Side		
Water flow	FI-604	592 gpm
Inlet temperature	TI-609	80° F
Outlet temperature	TI-608	83° F

B. LOADING FACE SHIELD

The loading face shield compressor seals were leaking excessively at the beginning of the test.* The compressors are equipped with an auxiliary stuffing box housing which is penetrated by the compressor shaft. There are seals at each end of the housing. Nominal 100 psig service air was supplied to the inside of the stuffing box housing. The seal next to the cylinder was made of graphite, and the one next to the crankcase was a conventional packing seal. Thus the graphite seal did not have to hold the full 235 psig system pressure.

*N. G. Dusanek, "Reactor Loading Face Shield Cooling System Test,"
AI-P-1107 Atomics International (October 1963)

Recompression units were installed to pump nitrogen, which leaked through the graphite seal into the stuffing box housing, back into the system. Back-pressure on the graphite seal was reduced to a nominal 5-psig to prevent leakage through the packing seal. The graphite seals broke, due to high pressure; consequently, the graphite seals were replaced with Teflon seals. Nitrogen from the plant nitrogen system, instead of that from high pressure bottles, was admitted to the recompressors for make-up nitrogen.

The loading face shield cooled properly with the system operating at 230 psig compressor discharge pressure. During full power operation, the hottest concrete temperature was 142°F, eight degrees below specified maximum operating temperature. The system was operated at much lower pressures during the beginning of this test, however, it was necessary to increase the pressure as the sodium temperature was increased. It was not necessary to adjust the individual cooling coil flow rates. Data for the loading face shield cooling system are tabulated in Table III.

C. SODIUM CONTAINMENT CELL COOLERS

The nitrogen coolers have maintained the cell atmosphere temperatures below 150°F. Three of the primary throttle and block valve freeze seals exceeded 150°F (alarm point). Several of the sodium service valve freeze seals exceeded 150°F, also. None of the valve freeze seals exceed 176°F. Data for the nitrogen coolers and freeze seals are tabulated in Tables IV and V, respectively.

D. SECONDARY SODIUM SERVICE

The secondary sodium freeze trap and freeze seal temperatures were found to be well below the operating limits. These temperatures are tabulated in Table VI.

Average summertime temperature in the nitrogen cooler room was 115°F, and 106°F in the secondary sodium service area. Greater cooling capacity for these areas would be desirable.

TABLE III
LOADING FACE SHIELD COOLING SYSTEM

Compressor outlet pressure	PI-912	230 psig
Heat exchanger outlet pressure	PI-913	229 psig
Compressor inlet pressure	PI-914	211 psig
Nitrogen flow	FI-916	3670 scfm
Heat exchanger inlet temperature	TI-900	133° F
Heat exchanger outlet temperature	TI-901	91° F
Compressor inlet temperature	TI-902	115° F
Cooling water outlet temperature	TI-637	87° F
Cooling coil outlet temperatures	TP-900	113° F
	TP-901	113° F
	TP-902	113° F
	TP-903	112° F
	TP-904	111° F
	TP-905	113° F
	TP-906	117° F
	TP-907	115° F
	TI-18-9	128° F
	TI-18-10	136° F
	TI-18-11	134° F
	TI-18-12	133° F
	TI-18-13	not working
	TI-18-14	136° F
	TI-18-15	134° F
	TI-18-16	134° F
	TI-18-17	120° F
	TI-18-18	134° F
	*TI-18-19	142° F
	*TI-18-20	142° F
	*TI-18-21	142° F
	*TI-18-22	not working
	*TI-18-23	134° F
	*TI-18-24	not working

*These thermocouples are attached to the inside bottom steel and can be expected to read higher than the others, which are in the concrete mass at various locations.

TABLE IV
NITROGEN COOLERS

	Cooling Water Flow (gpm) (cooler 1)	Cooling Water Flow (gpm) (cooler 2)	Cooling Water Return temperature (F)	Nitrogen Supply temperature (F) (thermometer)	Nitrogen Return Temperature (F) (thermometer)	Nitrogen Return Temperature (F) (gage)
IHX Cell No. 1	FI-628	FI-601	TI-601			TIC-1500
	0	84	82	82	129	129
	80	88	79	84	135	134
IHX Cell No. 2	FI-629	FI-602	TI-602			TIC-1501
	84	0	84	84	142	140
IHX Cell No. 3	FI-630	FI-603	TI-603			TIC-1502
	0	83	83	85	131	125
Primary Pipe Tunnel East Cooler	FI-606	*	TI-613			TIC-1504
	176		81	86	110	110
Primary Pipe Tunnel West Cooler	FI-608	*	TI-615	†	†	TIC-1505
	178		74			100
Service Cell	FI-631	FI-605	TI-609	†	†	TIC-1503
	48	0	81			120
Primary Cold Trap	FI-645	*	TI-638	†	†	TIC-1508
	92		76			185
Primary Sodium Tank Cell	FI-632	FI-607	TI-614	†	†	TIC-1505
	68	0	83			142

*Only one cooler

†Not recorded

TABLE V
FREEZE SEAL AND FREEZE TRAP TEMPERATURES
FOR THE PRIMARY SYSTEM

Valve and Freeze Trap Number	Thermocouple No.	Temperature (F)
Primary Throttle and Block Valves		
V-101	TI-412-1	174
V-101	TR-1-1	154
V-103	TI-412-2	149
V-103	TR-1-2	150
V-201	TI-412-3	122
V-201	TR-1-3	120
V-203	TI-412-4	164
V-203	TR-1-4	155
V-301	TI-412-5	132
V-301	TR-1-5	133
V-303	TI-412-6	169
V-303	TR-1-6	166
Moderator Valves		
V-001	TI-412-30	118
V-002	TI-412-12	134
Primary Pump Drain Valves		
V-108	TI-412-7	150
V-208	TI-412-8	154
V-308	TI-412-9	146
V-109	TI-412-31	139
V-209	TI-412-32	154
V-309	TI-412-33	154
Primary Fill and Drain Valves		
V-453	TI-412-14	123
V-460	TI-412-15	138
V-454	TI-412-16	107

TABLE V
FREEZE SEAL AND FREEZE TRAP TEMPERATURES
FOR THE PRIMARY SYSTEM (Continued)

Valve and Freeze Trap Number	Thermocouple No.	Temperature (F)
V-461	TI-412-17	138
V-455	TI-412-18	105
V-462	TI-412-19	114
V-459	TI-412-20	113
V-463	TI-412-21	154
V-464	TI-412-22	146
V-465	TI-412-23	117
V-456	TI-412-24	125
V-457	TI-412-25	defective
V-458	TI-412-26	114
Moderator Coolant Line		
FzT-14	TI-400-4*	130
FzT-14	TI-400-5†	385
Heat Exchangers		
FzT-11	TI-400-14*	102
FzT-11	TI-400-15†	600
FzT-12	TI-400-16*	133
FzT-12	TI-400-17†	709
FzT-13	TI-400-18*	121
FzT-13	TI-400-19†	664
Primary Sodium Cold Traps		
FzT-2	TI-400-20	108
FzT-2	TI-400-21	106
FzT-3	TI-400-22	106
FzT-3	TI-400-23	106
FzT-4	TI-400-24	120
FzT-4	TI-400-25	149
FzT-5	TI-400-26	151
FzT-5	TI-400-27	255

*Vent side of freeze trap

†Sodium side of freeze trap

TABLE VI
FREEZE SEAL AND FREEZE TRAP TEMPERATURES
FOR THE SECONDARY SYSTEM

Valve and Freeze Trap Number	Thermocouple No.	Temperature (F)
Secondary Sodium Throttle Valves		
V-102	TI-412-27	115
V-202	TI-412-28	123
V-302	TI-412-29	118
Secondary Sodium Cold Trap		
FzT-6	TI-400-28	122
FzT-6	TI-400-29	150
FzT-7	TI-400-30	120
FzT-7	TI-400-31	141
Secondary Sodium Service Drain Tank		
FzT-10	TI-400-34	105
FzT-10	TI-400-35	160

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with post critical test AI-PC-560, Cooling System, as listed in NAA-SR-5700 (Suppl 4), pages XI(c)-18, -19 and -20, with remarks regarding the test to explain compliance with the criteria.

- a) Criterion: All components requiring cooling from the cooling water systems are within design temperature limits at all power levels up to full power.

Remarks: All components requiring cooling were properly cooled, except for one area in the reactor cavity biological shielding as explained under "Reactor Cavity and Pipe Galleries," Sections IIA and IIIA of this report.

- b) Criterion: Systems requiring balancing (e.g., Cavity Liner) are balanced to maintain essentially uniform temperature throughout the component.

Remarks: The loading face shield cooling system was already temperature balanced without any special throttling. The reactor cavity cooling system was adjusted to compensate for defects in the system as stated in the body of this report. Adjustments in cooling water flow were relatively ineffective, because the average rise in temperature of the cooling water was less than the variation in concrete temperature.

- c) Criterion: Automatic temperature control devices (where provided, as in the cell cooling systems) function properly and maintain essentially constant temperatures in the areas being cooled.

Remarks: The nitrogen coolers were operated at maximum capacity to maintain the cell temperatures. The automatic temperature control devices apparently serve no useful purpose, because the variation in cell temperature was small.

- d) Criterion: Sufficient operating data has been obtained to verify plant design parameters of heat generation, flow, pressure drop and heat transfer for future use in analyzing plant and equipment performance.

Remarks: Nearly all of the plant instrumentation on the cooling systems was read and the data recorded. These data should be sufficient for analyses of future abnormalities. The process instrumentation is not satisfactory for accurately analyzing plant heat losses through the cooling system; however, a rough calculation was made for AI-PC-530, Heat Transfer Test.*

- e) Criterion: Mechanical components in the loading face shield coolant system operate satisfactorily in accordance with system design parameters.

Remarks: The mechanical components in the loading face shield cooling system operate in accordance with system design parameters. A design parameter was not given for system leak rate.

*A. J. Divis, "HNPF Post Critical Heat Transfer Test," AI-PC-530, Atomics International (February 1964)

f) Criterion: The loading face shield bottom plate temperature can be maintained within design limits (~180°F maximum).

Remarks: The measured loading face shield temperatures were maintained well below 180°F.

g) Criterion: Cooling circuits are balanced to maintain uniform temperature gradients.

Remarks: The cooling circuits were balanced as installed, without the need for adjustments.

V. EVALUATION

The purpose of the cooling system post-critical test was to determine that various system components were adequately cooled.

The test purpose and the intent of the test completion criteria as noted in the HNPF Final Summary Safeguards Report, NAA-SR-5700, were met with the exception of an area in the reactor cavity biological shield in which concrete temperature exceeds design limits.

A. REACTOR CAVITY AND PIPE GALLERIES

Two thermocouples on the upper reactor cavity liner indicated temperatures that were 12 and 18°F above the design limit of 175°F. These two thermocouples are fixed directly to the liner as shown in Figures 560-1 and -3; heat conducted through the reactor cavity liner metal is causing the high readings. Cooling water line No. 1721, closest to the two thermocouples, has a return water temperature of 97°F. Line No. 1720 above the thermocouples has a return temperature of 87°F. Four lines forming loops on the same elevation, located 5 in. outward from line No. 1721, have temperatures of from 79 to 84°F. These indicate that the excessive temperature exists in a small triangular shaped localized area and does not penetrate the shielding concrete to any significant depth.

Failure of three cavity liner cooling water coils (see Figure 560-2 for location) has been previously noted in AI-P-1106, Cooling Water System. The evaluation of this situation, as made by L. L. Johnson was as follows:

Figure 560-3. Plan View - Upper Cavity Liner Cooling Coils

"Although the three leaking cooling coils are located in the same general region, there is still adequate cooling of the concrete. The original heat transfer analysis pertaining to coolant pipe spacing and maximum concrete temperatures specified that the pipe tunnels from the reactor cavity to the gallery seal shall have coolant pipes 1-in. or larger in diameter and spaced 10 in. on center. The pipe spacing is based on a heat load of 109 Btu/hr-ft² and gives a maximum concrete temperature of 135°F. With one cooling pipe out, the maximum concrete temperature rises to 159°F. Extrapolation shows that loss of two adjacent cooling coils will result in concrete temperatures of 180°F. Thus, concrete cooled by lines No. 1559 and 1560 will reach about 180°F while that cooled by line No. 1651 will reach about 160°F. Since the actual coil spacing is 8 in., instead of 10 in. as used in the above analysis, the concrete temperature will be somewhat less than that given above.

"The predicted concrete temperatures will have no detrimental effect on the strength or shielding properties of the concrete, or on the cavity liner. Concrete temperatures can approach the 500 to 600°F range before structural damage results. In terms of shielding, the only adverse effect of concrete temperatures near or above the boiling point of water is the degradation of fast neutron shielding capabilities. Since the temperature of the concrete in question should not approach the boiling point of water and since the region is one of low fast neutron flux, no detrimental effect on shielding should occur. The cavity liner proper is 1-in.-thick steel plate and should certainly not be adversely affected. Recent hydrostatic tests of the adjacent coils indicate that the coils are sound and therefore, it is doubtful that future failure in the same area will occur."

The above comments are still pertinent and it is to be noted that the last paragraph applies to the concrete in the area of the upper cavity liner as well as to the concrete in the outer liner area.

B. LOADING FACE SHIELD

Temperature readings of thermocouples imbedded in the body of the loading face shield indicate that the LFS nitrogen coolant system will maintain the LFS temperatures below the design criteria. The highest recorded temperatures of 142°F were from thermocouples attached to the interior bottom plate, whereas temperatures of thermocouples in the concrete mass ranged from 120 to 136°F.

Considerable difficulty was encountered in attempts to reduce excessive leakage from the nitrogen compressor shaft seals. To reduce the total system leakage to a minimum, a recompression system was designed and installed. Two positive displacement, diaphragm-type compressors (one operating and one spare) are utilized to return nitrogen that escapes through the seal, into the external stuffing box, into the LFS system.

Considerable difficulty was encountered in obtaining satisfactory operation from the recompression compressors. Failure of the mild steel diaphragms was the main cause of trouble. These have been replaced with stainless steel diaphragms and now are operating satisfactorily. The recompression system has reduced the total system nitrogen loss from about 2400 cf to about 400 cf for a 24 hr period.

C. SODIUM CONTAINMENT CELL COOLERS

The primary enclosures nitrogen cooling units performed satisfactorily. A routine maintenance check of the various nitrogen coolers was performed shortly after the test data was obtained. Some of the fans were found to be operating slightly below the design rpm. Adjustment of the variable pitch fan sheaves improved the performance of these units and at this time all units are operating at or above design capacity.

Temperatures of the three primary block and throttle valves exceeding the 150°F alarm point were subsequently found to be caused by partially closed dampers in the nitrogen ducting to the finned sections on the valves. The dampers were adjusted and secured in position to prevent future inadvertant throttling of nitrogen flow to these valves. The temperature of these valves has since remained below the 150°F alarm point.

D. SECONDARY SODIUM SERVICE AREA

The ambient temperature in the secondary fill tank operating area and the IHX nitrogen cooler room area was as high as 115°F during the hot summer months. To reduce the heat source, instead of maintaining all three secondary fill tanks at 500°F, one is held at this temperature and the other two are reduced to about 250°F.

E. CONCLUSION

All test completion criteria have been met with the exception of the one area of high thermocouple readings on the upper reactor cavity liner. As previously stated, this should have no major adverse affect on the structural or shielding characteristics of the concrete affected.

All systems function properly and are capable of adequately cooling the reactor cavity, sodium pipe galleries, sodium containment cells, loading face shield and the heating and ventilating equipment.

HALLAM NUCLEAR POWER FACILITY

Post Critical Test Procedure

AI-PC-620

Plant Protective System

Test Procedure Submitted for Approval

6/30/61 James E. Quinn
Date Atomic Energy International

Test Procedure Reviewed

12/14/62 D. E. Simpson
Date Atomic Energy
Commission

Test Approved for Performance

12/28/62 James E. Quinn
Date Atomic Energy International

Test Completed

7/29/63 James E. Quinn
Date Atomic Energy International

Test Approved for Performance

12/28/62 R. C. Buntain
Date Atomic Energy
Commission

Test Completed

7/20/63 R. C. Buntain
Date Atomic Energy
Commission

AI-PC -620

January 27, 1964

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HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
PLANT PROTECTIVE SYSTEM, AI-PC-620

I. PURPOSE OF TEST

The purpose of this test was to ensure proper performance of the plant protective system by a continuous surveillance of system operation and by matching system operating characteristics with measured process variables.

II. DISCUSSION OF TEST PERFORMANCE

A. GENERAL

The basic method of performing this test was to compare operational results with those predicted, making adjustments where necessary and verifying performance on the operating system. The test was essentially restricted to the protective system computers; the remainder of the system is primarily switching circuitry and was adequately investigated and reported during the preoperational test.*

Three types of computers are utilized in the protective system, as follows:

- 1) Measured rate of change of fuel channel outlet temperature (dT/dt).
- 2) Power to flow ratio (T_o). These are sometimes referred to as "fuel channel outlet temperature computers."
- 3) Primary to secondary flow ratio (W_p/W_s).

There are three each of the dT/dt and T_o computers and nine W_p/W_s computers (three for each of the three sodium circuits) in the operating system. In addition, there is a "hot spare" for each computer type, aligned and warmed up, ready for use on short notice. These 18 computers are installed in a single enclosure, along with power supplies (including one hot spare of each type), test fixtures, a monitoring station, and other auxiliary equipment. The protective panel layout is presented in Figure 620-1.

*W. H. Spalding, "Plant Protective System," AI-P-1121 (December 26, 1962).

Figure 620-1. Protective System

To provide redundancy, a two-out-of-three coincidental computer action is required by each computer type for initiation of protective action. For the flow ratio computers, the coincidence must occur on two of the three computers monitoring the same sodium circuit. A block diagram of the protective system is presented in Figure 620-2. The diagram indicates the type of protection possible from each protective system device and resultant effect on plant and control systems. The computers with which this test was primarily concerned are shown in solid lines on the figure; other equipment is indicated by dashed lines.

B. PROTECTIVE ACTION DURING TESTING PROGRAM

Twenty-six scrams occurred during the testing program. These are listed in Table I, with appropriate comments for reference purposes, and are discussed in some detail in Section III of this report.

C. MATCHING COMPUTER CHARACTERISTICS TO THE MONITORED SYSTEM

The dT/dt and T_o computers had a thermocouple loading effect which resulted in a depression of temperature indications. The input impedance and gain of each of the computers was increased to correct the condition. Temperature indications were then demonstrated to be sufficiently accurate to meet the requirements of the acceptance specification.* In addition, it was found necessary to offset the zero sodium flow signal adjustments within the T_o computers to match the computer characteristics to those of the main heat transfer system. Computer alignment techniques were refined to obtain better correlation with the actual system.

The W_p/W_s computers performed their computing function properly throughout the program. A change in alignment procedures was necessary to ensure that the low-flow cutouts were effective at the proper values of flow. These low-flow adjustments were found to be set too low, the result being that it was extremely difficult to start up the sodium pumps without incurring a flow-ratio scram.

After completion of the above, the computers were demonstrated to perform within the error limits specified in the acceptance specification.*

*"Acceptance Test and Checkout of the HNPF Plant Protective Panel," Atomic International Specification 7518-572015 (Rev. 4-19-61).

Figure 620-2. Protective System Block Diagram

TABLE I
SCRAM SUMMARY

No.	Date	Method	Power Before Scram	Comment
1	1-04-63	Manual	2.5 Mwt	Test (AI-PC-660, Scram and Loss of Load).
2	1-15-63	W_p/W_s	10 Mwt	Operator was throttling pri TV No. 3 to increase flow in service system for cold trapping.
3	1-15-63	W_p/W_s	10 Mwt	Occurred ~3 min after sodium flow established through cold trap.
4	1-24-63	Manual	10 Mwt	Test (AI-PC-660, Scram and Loss of Load).
5	1-24-63	W_p/W_s	4.3 Mwt	Calibration of low-flow cutouts required.
6	1-24-63	W_p/W_s	0.5 Mwt	Calibration of low-flow cutouts required.
7	1-29-63	Neg dT/dt	20 Mwt	Neg 5 Mwt ramp for test (AI-PC-770).
8	2-04-63	W_p/W_s	<100 Kwt	Instrument technicians taking readings from flow ratio computer terminals for Ckt 2 and accidentally touched Ckt 1.
9	2-04-63	Ckt 1 pumps off	<100 Mwt	100-amp breaker tripped in critical panel No. 1.
10	2-23-63	Manual	38 Mwt	Test (AI-PC-660, Scram and Loss of Load).
11	3-01-63	Manual	35 Mwt	Test (AI-PC-660, Scram and Loss of Load).
12	3-01-63	Neg dT/dt	Subcritical	Computer C in scram. Technician adjusted computer B the wrong way.
13	3-06-63	Manual	38 Mwt	Test (AI-PC-660, Scram and Loss of Load).
14	5-20-63	W_p/W_s	2.5 Mwt	Occurred immediately after completion of sodium transfer from reactor to primary fill tank No. 3. Apparently due to flow upset when service system flow was secured.
15	5-21-63	Pos. dT/dt	18 Mwt	Flux controller malfunctioned; internal short in a relay resulted in rod withdrawal. Period recorder on channel IV indicated (approximately) a 26-sec positive period.

TABLE I (continued)

No.	Date	Method	Power Before Scram	Comment
16	5-21-63	W_p/W_s	36 Mwt	Position of sec throttle valve being adjusted. Air line carrying a control signal became disconnected; valve drove closed.
17	5-27-63	Neg dT/dt	15 Mwe	Operator error; reactor manually overcontrolled.
18	5-27-63	Neg dT/dt	<1%	Rate of cooldown from above scram initiated a second scram during performance of the hot startup SOP before adequate reactor power was available to reverse the temperature trend.
19	5-27-63	T_o computer	~38 Mwt	T_o cp B had been removed, establishing the first of a two-out-of-three coincidence scram condition. Cp A was disturbed when cp B was racked into position, thus establishing the two-out-of-three coincidence.
20	6-01-63	Manual	40 Mwe	Test (AI-PC-660, Scram and Loss of Load).
21	6-09-63	W_p/W_s	107 Mwt	Secondary sodium pump binding caused flow ratio upset in No. 2 sodium circuit.
22	6-09-63	LXL in steam gen. No. 2	Just critical	Initiated during recovery from No. 21, above. Feedwater pump 1B had been shutdown while the conventional plant crew was putting the boiler on the line. Due to a delay in the operation, the feedwater pressure dropped to ~750 psi by the time the HNPF needed water. The feedwater pump could not be started fast enough at this point to prevent the scram.
23	7-05-63	T_o computer	38 Mwt	Inadvertent and unnecessary; due to improper monitoring of flux signals at the plant protective system calibration unit.
24	7-15-63	W_p/W_s	70 Mwe	Due to a flow transient incurred while establishing service system flow for cold trapping.
25	7-19-63	Neg dT/dt	18 Mwe	Operator transferred neutron flux controller to manual when he thought rod withdrawal was excessive. It was apparent that the controller action was proper immediately afterward when the reactor scrammed on a negative rate of change of measured fuel channel outlet temperature.
26	7-21-63	Neg dT/dt	30 Mwe	TRC-2, fuel channel outlet temperature controller, failed; temperature decreased at a rate sufficient to exceed the scram trip.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

A. dT/dt COMPUTERS

Early in the testing program, it was determined by observation and verified by measurement that the thermocouple input signals to these computers was yielding a depressed temperature indication. An investigation revealed that this condition was caused by a loading effect, whereby the initial stage of amplification within each computer was drawing enough current to cause an appreciable input signal voltage drop due to the natural resistance of the thermocouple leads. (The reactor thermocouple resistances are in the order of 80 ohms.) Combined with the low impedance inputs to the computers, the net result was a comparatively high impedance voltage source applied to a low impedance device, resulting in signal depression. This condition was not apparent during the preoperational test because a very low impedance source was used to simulate the temperature signal. Similarly, the test fixture for the computer had a nominal zero source impedance, therefore, alignment on the test fixture ensured misalignment to the process system.

The dT/dt computer is designed to operate in the 500 to 950°F temperature range. The above loading affected only temperature indications and high temperature alarm, setback and scram, and not the more important temperature rate protection. The problem was corrected by increasing the computer input impedances and raising the current gains from about 6 to 25. This resulted in about a 0.1% degradation in linearity and limited the uncompensated temperature error to approximately 10°F in the range of interest.

Performance of the dT/dt computers after completion of the above work was excellent. Of the 26 scrams experienced during the testing program, seven were initiated by the temperature rate feature of the dT/dt computers (refer to Table I). The following comments are applicable to these seven scrams:

- 1) None of the seven were spurious.
- 2) Four are representative of protection against maloperation and operator error (No. 7, 12, 17, and 25).
- 3) Two were incurred following control system component failures (No. 15 and 26).

- 4) Two were not required for equipment protection (No. 12 and 18).

These were both incurred during hot startup operations subsequent to short or low power periods of operation followed by a scram. In each instance, there was insufficient afterglow heat generation in the core to overcome the combined effect of the initial negative temperature transient and subsequent cooldown rate. It was later demonstrated that the condition did not exist, following a short period of operation in the normal power range.

B. POWER TO FLOW RATIO (T_0) COMPUTERS

The same condition of thermocouple loading described in Section A, existed in the T_0 computers. The normal operating range of the temperature inputs to these computers is from 560 to 610°F. Incorporation of the same modification, as described for the dT/dt computers, corrected the loading effect.

The primary function of the T_0 computer is to predict fuel channel outlet sodium temperatures, in advance of the temperature change measured by the dT/dt computers, in the event a rapid power rise is experienced. This is accomplished by multiplying the normal full-power differential temperature (335°F) by the normalized power to normalized total primary sodium flow ratio and adding the resultant temperature increment to the measured reactor inlet temperature. By use of proper scale factors, the resulting output signal is a voltage which is proportional to temperature. The scram, setback and alarm trips operate on this voltage signal to initiate protective action.

From the foregoing, it may be seen that the computation depends on three variables; namely, measured reactor inlet temperature, measured primary sodium flow and measured power (neutron flux). Any calibration errors result in erroneous computations and indications. The greatest discrepancies were encountered in the sodium flowmeter calibrations.* Since final flowmeter calibrations were developed from heat balance calculations during power operation, apparent calibrations were incorporated into the T_0 computer alignment procedures to enable the extension of plant operation into the power range. The power (neutron flux signal) gain and computer span were adjusted, as necessary, to obtain proper correspondence between power and flow. High and low flux trip points (Figure 620-3)

*Refer to Test Completion Report AI-PC-830, "Master Schedule, Including Rise-to-Power," (November 5, 1963).

Figure 620-3. Power to Flow Ratio Computer Operating Curves

were adjusted proportionately to provide conservative high-power protection. These boot-strapping operations were performed several times during the program to permit the plant operating power levels required for accumulation of the data necessary for final calibration.

As soon as the revised sodium flowmeter calibration curves were developed, they were incorporated into the computer alignments. To match computer characteristics to the process system, it was necessary to offset the flowmeter zero input to the computers by approximately 3%. This was accomplished by use of internal zero adjustments in the computers. The full-flow sodium input signal was established at 26.2 mv, corresponding to a sodium flow of 2.8×10^6 lb/hr per loop. The low-flow cutout points were adjusted to 10, 12, and 14% of this figure for alarm, setback and scram, respectively.

After completion of the above recalibration, the computers were demonstrated to compute properly.

Two T_0 computer-initiated scrams occurred during the testing program (Table I; No. 19 and 23). These were both incurred as a result of activity at the protective panel and are not representative examples of computer function. These were random events that are not likely to recur.

C. FLOW RATIO (W_p / W_s) COMPUTERS

The uncertainties in sodium flowmeter calibrations affected operation of these computers only to the extent that the low-flow cutout points were in error by approximately 3%. The actual ratios obtained by use of the original recommended calibrations are theoretically on the order of 1.3% different from those obtained from the final calibration.

At the 10 Mwt test level, it was determined that the low-flow alarm and scram cutouts were well below the nominal 15 and 20% flow figures (see Figure 620-4). Investigation revealed that adjustments made by use of the test fixture were not valid on the process system. The alignment procedure was revised and realignment of the computers was verified on the process system to correct the condition. Operating characteristics subsequently measured are presented in Figure 620-4.

Figure 620-4. Primary-Secondary Flow Ratio Computer Operating Curves

Nine scrams were initiated by the flow ratio computers during the test program. Of these nine, four were directly related to initiation or cessation of sodium flow through the service systems (Table I; No. 2, 3, 14, and 24). Two scrams were due to the decreased cutout boxes discussed previously (No. 5 and 6). Another (No. 16) resulted from a main valve being driven closed during adjustments. Scram No. 8 was the result of a signal being shorted out, and No. 21 occurred when secondary pump binding during operation caused a quick reduction in secondary flow.

From these results, it is evident that:

- 1) Two of the scrams initiated by the flow ratio computers represented protection from equipment malfunction (No. 16 and 21).
- 2) Seven were unnecessary scrams, since the purpose of the computers is to protect against sharp temperature transients, and no such transients were observable prior to scram initiation (No. 2, 3, 5, 6, 8, 14, and 24). Three of these (5, 6, and 8) are of a unique nature that in all probability will not be experienced again. The other four represent a fast transient that has thus far been unpredictable. Further operational experience and investigation is required if these unnecessary scrams are to be avoided. The condition is being studied outside the cognizance of this test.

D. OTHER SCRAMS DURING THE PROGRAM

1. Manual

Six scrams were initiated by depressing the manual scram button during the performance of AI-PC -660, Scram and Loss-of-Load.

2. Loss of Electrical Power to Pumps

One scram was initiated from a low-power level by loss of the 4160 volt sodium pump supply voltage to circuit No. 1 when a 100-amp breaker tripped.

3. Low-Steam Generator Evaporator Water Level

A steam generator evaporator low-water level scram was initiated when steam from the HNPF was introduced to the turbine. A feedwater pump had been shutdown shortly before this and the pumping capacity remaining in service was not adequate to meet the increased requirements.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with post critical test AI-PC -620, Plant Protective System, as listed in NAA-SR-5700, Supplement 4, page XI (c)-24, with remarks regarding the test to explain compliance with the criteria.

- a) Criterion: Realignment of the protective panel and resetting (as necessary), all alarm, setback and scram trips.

Remarks: All computers were realigned to match process characteristics. Trip and cutout points were readjusted as necessary. Alignment procedures were correlated and corrected to the process systems.

- b) Criterion: All setback circuits operate correctly.

Remarks: Circuit verification was accomplished during the performance of the preoperational test;^{*} no further verification was deemed necessary. It was, however, verified that the inputs to the computers were correct and that the accuracies of computation were within acceptable limits.

- c) Criterion: All scram circuits operate correctly.

Remarks: The comment under (b), above, is applicable. However, it was verified that the computers loss of electrical power to the sodium pump motors, low water level in the steam generator evaporators, and depression of the manual scram button were all capable of initiating a scram.

- d) Criterion: It has been demonstrated that the following computers and remote indicators correctly compute and indicate at various sodium temperatures, sodium flows, and reactor power levels, according to measured values of inputs.

- 1) Flow to flow computers and indicators,
- 2) High Na temperature fuel channel outlet chassis indicators,
- 3) dT/dt temperature fuel channel outlet chassis and indicators,
- 4) T₀ computer and indicators.

^{*}W. H. Spalding, "Plant Protective System," AI-P-1121 (December 26, 1962).

Remarks: Compliance with this criterion has been demonstrated.

V. EVALUATION

The HNPF Plant Protective System is for the purpose of protecting the reactor and process system from thermal damage during all phases of plant operation. The purpose of the test was to verify that the system performed properly. This test purpose was adequately met. All test completion criteria, as listed in NAA-SR-5700 Supplement 4, were satisfactorily met. Observation of the operation of the plant protective system indicates the following:

a) Flow to flow computers

During the test program, nine scrams were initiated by action of the flow-to-flow computers. Seven of these scrams were caused by short flow transients not discernible on the plant process instruments. There were no temperature changes associated with the flow transients. The response time of the computers should be increased to nullify the effect of short flow transients.

b) Power to flow ratio computers

Since calibration of the power to flow ratio computers is dependent on sodium flowmeter and neutron flux calibrations, the final calibration could not be determined until the plant had been to full power. After the final calibration and correction of the thermocouple loading problem, it was demonstrated that the computers would compute within the accuracy limits of the acceptance specification (30°F).

c) Measured rate of change of fuel channel outlet temperature computers

A thermocouple loading problem existed with these computers. It was corrected by adding series resistance and increasing the current gain of the magnetic amplifiers. It was demonstrated during power operation that these computers perform in an excellent and dependable manner.

HALLAM NUCLEAR POWER FACILITY

Post Critical Test Procedure

AI-PC-630

Plant Control System

Test Procedure Submitted for Approval

6/27/61 James E. Rhans
Date Atomic International

Test Procedure Reviewed

12/14/62 W E Simpson
Date Atomic Energy
Commission

Test Approved for Performance

12/21/62 James E. Rhans
Date Atomic International

Test Completed

7/20/63 James E. Rhans
Date Atomic International

Test Approved for Performance

12/21/62 RE Buntam
Date Atomic Energy
Commission

Test Completed

7/20/63 RE Buntam
Date Atomic Energy
Commission

HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
PLANT CONTROL SYSTEM

AI-PC-630

Test and Analysis
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P. F. SHAW

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by
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A T O M I C S I N T E R N A T I O N A L
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Contract: AT(11-1)-GEN-8

January 16, 1964

ACKNOWLEDGEMENT

Successful completion of the work represented by this report is largely due to the capable assistance and cooperation of the following:

C. L. Dunsmore, Specialist, Atomics International

Merlin Mauderly, Instrumentation Foreman,
Consumer's Public Power District

D. J. Sobo, Instrumentation and Control Supervisor,
Atomics International

Their constructive suggestions and able assistance are hereby acknowledged.

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HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
PLANT CONTROL SYSTEM, AI-PC-630

I. PURPOSE OF TEST

The purpose of this test was to verify that the plant control subsystems are properly operable as a compatible, integrated system throughout the range of reactor operation.

II. DISCUSSION OF TEST PERFORMANCE

A. TEST PHASES AND SCOPES

The plant control system post critical test was divided into two parts: Phase II, which consisted of testing performed at less than 15% of design reactor power with the steam generated being condensed in the atmospheric boilers of the steam dump system, and Phase III which consisted of testing in the normal power operating range (15 to 100% of design power) with the steam produced being primarily directed to the turbine. During test performance, the following subsystems* were first tested individually and then as an integrated system:

- 1) Plant Power Control
- 2) Division of Load Computer
- 3) Secondary Sodium Flow Control
- 4) Primary Sodium Flow Control
- 5) Nuclear Power Computer
- 6) Neutron Flux Controller
- 7) Moderator Coolant Control
- 8) Convection Flow Control
- 9) Feedwater Control

*Refer to HNPF preoperational test completion report, AI-P-1120, Plant Control System by D. J. Sobo, Atomics International, (October 1962) for system descriptions and diagrams.

- 10) Attenuator Control
- 11) Power Relief Control
- 12) Primary Block Valve Control
- 13) Steam Dump Pressure Control

In the course of plant operation, the operating experience and application of the control system were such that the content of the formal test was far exceeded. This was beneficial in that it provided more information and opportunity for observing system performance than might otherwise have been available. Also, portions of the formal test were performed out of sequence as a matter of convenience and, at times, necessity. The work accomplished and observations made relative to the plant control system are reported with no attempt at distinction between test and nontest efforts. All documented observations and findings are presented in Section III of this report.

Controller adjustments at the end of the program are presented in Table I, along with those from the preoperational test of the system.

B. PHASE II TESTING

1. Control Tests with Reactor Shutdown

Prior to any appreciable rise in power with the reactor fully loaded, the primary and secondary sodium loops were placed on flow control and then slaved together from the division of load computer. By manually varying the power demand signal into the division of load computer, the cascaded system was checked for accuracy and stability.

2. Initial Power Operation (2.5 Mwt Reactor Power)

Power was increased to the 2.5 Mwt nominal test power level. At this point, power level was not sufficient to separate control system problems from noise on monitored process variables. Control system testing which was planned for this level was therefore deferred to the next planned power level of 10 Mwt. At this higher level, problems in the neutron flux controller became readily apparent. The work accomplished on the flux controller during this period is presented in Section III-A-5.

TABLE I
FINAL CONTROLLER ADJUSTMENTS

	AI-PC-630				AI-P-1120*			
	PB (%)	Reset (min-repeat)	Rate (min)	Deadband (in.)	PB (%)	Reset (min-repeat)	Rate (min)	Deadband (in.)
PRC-501 Plant Power Control	200	7	0.1	-	100	0.2	0.1	-
WaIC-501 Division Load Computer	-	-	-	-	150	0.02	-	-
FRC-102 Sec	100	0.2	-	-	-	-	-	-
202 Na	↓	↓	-	-	150	0.2	-	-
302 Flow Control	↓	↓	-	-	-	-	-	-
FRC-101 Pri	130	0.2	-	-	-	-	-	-
201 Na	130	0.2	-	-	-	-	-	-
301 Flow Control	130	0.15	-	-	-	-	-	-
TRC-101	200	7	0.01	-	200	1	0.01	-
201	↓	↓	↓	-	-	-	-	-
301	↓	↓	↓	-	-	-	-	-
TRC-2 Neutron Power Computer	250	3	OFF	-	150	2	0 (off)	-
NC-1 Flux Controller	3	0.01	OFF	0.20	20	2	0 (off)	0.20
TRC-3 Moderator Coolant	50	0.5	0.01	-	100	1	0.01	-
TRC-4 Convection Control	150	0.4	0.01	-	100	0.4	0.04	-
LRC-151 FW Level	100	2	-	-	50	0.2	-	-
251	↓	↓	-	-	-	-	-	-
351	↓	↓	-	-	-	-	-	-
FC-151	70	0.8	-	-	300	0.1	-	-
251	↓	↓	-	-	-	-	-	-
351	↓	↓	-	-	-	-	-	-
TRC-152 Attenuator	25	0.5	-	-	25	0.5	-	-
252	↓	↓	-	-	-	-	-	-
352	↓	↓	-	-	-	-	-	-
PIC-501 Steam Dump	20	0.2	-	-	20	0.2	-	-

*From Test Completion Report for AI-P-1120, Plant Control System, by D. J. Sobo

3. Semi-Automatic Control at 10 Mwt

After successful resolution of the flux controller problems, the HNPF was operated in a semi-automatic control mode. Starting on manual control, individual subsystems were aligned and transferred to automatic. The power demand signal was manually generated at HCS-504, steam demand hand control station; the steam being generated was condensed in the steam dump system atmospheric boilers. Control was demonstrated to be satisfactory.

4. Automatic Load Following at 20 Mwt

At the 20 Mwt reactor power test level, the control system was operated in the fully automatic, load following mode. Temporary connections were made to the steam dump system steam flow and pressure for inputs to the control system. Initial controller adjustments were performed at this time. Reactor temperature control was investigated; this is reported in Section III-B-3.

5. Hot Startup

After a planned scram from a reactor power level of 38 Mwt, a successful hot startup was accomplished in accordance with the associated standard operating procedure. A summary of the operation is presented in Section III-B-4.

C. OPERATION IN THE NORMAL POWER RANGE

1. Introduction of HNPF Steam to the Turbine

Introduction of HNPF steam to the turbine was accomplished in accordance with the test procedure, with the conventional plant operating on pressure control and the HNPF base-loaded from HCS-504, steam demand hand control station. Initially, HNPF steam was exhausted to the atmospheric boilers in the steam dump system. After warmup of the HNPF main steam header, and cautious opening of the header block valve, the steam dump valve was slowly closed and the conventional plant was backed down by the introduction of HNPF steam. When all steam being produced by the HNPF was going to the turbine, the steam dump was placed in normal standby service.

A subsequent rise in grid demand was met by increasing HNPF power to correspond to 15 Mwe turbine-generator power output. At this time, the HNPF was placed on automatic load-following control, and the conventional plant was

base-loaded on manual control. When the grid demand decreased, the conventional boilers were taken out of service and placed on hot standby, leaving the HNPF load following at 20 Mwe gross station generation. Further adjustments were made to the plant power control, nuclear power computer, and feedwater level controllers during and following the operations described above.

2. Parallel Operation

During the remainder of the testing program, the conventional and nuclear plants were frequently operated in parallel to provide the conditions required for testing while still meeting station power generation assignments. During the times when both plants were supplying steam to the turbine, at least one was maintained base-loaded on manual control to avoid interaction of control systems. No difficulties were encountered due to parallel operation.

3. Load-Following Capability Test

The planned load-following test was performed by making step changes in turbine-generator output with the HNPF supplying the full steaming requirement on automatic load-following control. From a base power level of 40 Mwe gross, steps of 1.0, 2.5, 5.5, and 6 Mwe were imposed first in the negative and then in the positive direction. Maximum rate of change observed was 41.4 Mwe/min for the 6 Mwe step change in demand. Load-following adequacy was verified.

4. Loss-of-Load Test

A loss-of-load test was performed by tripping the turbine from 25 Mwe with the HNPF supplying the full steam requirement on automatic load-following control. The plant control system successfully programmed the HNPF down to steam dump capacity without incurring any scram condition. Seven minutes after trip, the turbine was back on the line on HNPF steam. Seventeen minutes after trip, the HNPF was back on fully automatic load-following control.

5. Power Range Scrams

Three power range scrams of importance to this report were incurred. The first was a planned scram from a station power level of 40 Mwe with the HNPF supplying the required steam on automatic load-following control. The turbine was purposely left on the line down to the point where steam quality ceased to be acceptable, to determine if any deleterious pressure-temperature transients would occur in the steam generators. No such transients were incurred, and the

test resolved the question of whether or not immediate turbine trip on scram was necessary or desirable.

Following the above scram, the reactor was cooled down to 500°F isothermal on automatic convection flow control. This activity, along with the hot start-up previously performed, verified the alignment and adequacy of the convection flow control.

The other two scrams were initiated by negative rate of change of measured fuel channel outlet temperature (dT/dt) trips due to improper operator action. At the time of the first scram, the reactor was on manual flux control; the operator induced the scram in attempting to control on the dT/dt meter indications. The second occurred when an operator believed rod withdrawal at the end of a power decrease was excessive and transferred to manual control, only to find that the need for rod withdrawal was legitimate. Both of these scrams were of value in orienting the operators in relation to the control system. The scrams further served to emphasize the sensitivity of control in the power range of operation.

6. One-Loop Shutdown

A one-loop shutdown was successfully performed in accordance with the prepared test procedure. The operation proceeded smoothly and uneventfully. A partial one-loop shutdown was performed later to enable a maintenance activity to proceed. All of the steps of a normal one-loop shutdown were performed, except the three sodium block valves were left open and the one-circuit shutdown switch was maintained in the normal operating position. During the period when the pumps were out of service, primary and secondary sodium flows and circuit steam production continued in the circuit at a reduced rate on convection flow alone.

7. High Power Operation

After the initial operation of the turbine from HNPF steam, operation of the plant was more on a generation requirement basis than for test purposes. In keeping with the original program, however, no planned test power level was exceeded until it was demonstrated that operation at that level presented no information that could be construed as indicating any foreseeable unsatisfactory condition at the next level.

The final demonstration of control system performance was an 84 hour run at gross station generation rates of 80 to 83 Mwe with the HNPF supplying the required steam on automatic load-following control.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

Pertinent information on subsystems is presented in the following sections; information pertaining to system performance at low power (Phase II, <15% power) during initial checkout of the plant control system as a whole is presented in Section III-B, and information obtained during power operation (Phase III, 15% to full power) is presented in Section III-C. In phase II tests, the steam generated was exhausted to the atmospheric boilers of the steam dump system. In Phase III testing, the steam was supplied to the turbine.

A. SUBSYSTEM TEST RESULTS

1. Plant Power Control

The plant power control subsystem was initially placed into operation at the 20 Mwt test power level. At this time, steam was being supplied to the steam dump system atmospheric boilers. Since the normal plant control system main steam header, steam pressure, and flow sensing elements are on the turbine-side (downstream) of the HNPF main steam header blocking valve, temporary connections were made between the steam dump system, steam flow and pressure elements, and the plant power control subsystem. This permitted utilization of the steam dump as a variable load for initial checkout of the facility load-following feature.

Variations in steam demand were generated by manual adjustment of the steam dump control valve position. Load-following was demonstrated with the complete system in automatic control, and with the nuclear power computer at a fixed setting from HCS-6. This latter condition constituted reactor temperature control alone, and lacked the anticipatory action of the nuclear power computer. Under both conditions, the plant power control system was demonstrated to properly compute power requirements for achieving the demanded steam pressures and flows, and to transmit the resulting demands to the division of load computer.

The above check was repeated at 38 Mwt. However, it was not possible to obtain the same degree of stable facility operation that was demonstrated at the lower power level. This was attributed to pressure pulsation reflections from the steam dump atmospheric boilers and not to limitations within any individual subsystem. Demonstration of control adequacy in the power range of operation

(refer to following sections), with the generated steam going to the turbine, verified this assumption.

The final controller adjustments at the completion of the testing program were as follows:

PRC-501

a) Proportional band	200%
b) Reset time	7 min/ repeat
c) Rate time	0.1 min

2. Division-of-Load Computer

Prior to increasing reactor power to the level where measurable heating occurred, primary and secondary sodium loops No. 2 and No. 3 were placed on flow control and slaved together from the division-of-load computer.* The power demand signal into the division-of-load computer was manually varied from HCS-504, steam demand hand control station. HCS-504 was manually ramped up and down from 36% to 75% to 36%. Sodium circuit (primary and secondary loops) No. 1 was ramped from 20% to 30% to 20% by means of HCS-503, circuit sodium flow hand control station. In all instances, the systems smoothly followed the demands imposed, at the limited rate. Flow ratio alarms were encountered momentarily on circuits one and two at the approximate mid-range of the imposed ramps. These alarms cleared almost immediately, and no adjustments were required.

Division of power to the three circuits is affected by the somewhat different heat transfer characteristics of the three steam generators and, to a larger degree, by differences in the six sodium flowmeter characteristics. Distribution of load to the three circuits is on the basis of linear approximations of the flowmeter calibration curves, one for the three primary flowmeters and one for the three secondary flowmeters. In actual fact, the flowmeter calibrations are neither linear nor of the same shape, as is indicated in Figures 630-1 and -2. Actual deviation from the straight line approximation is up to 8% for the individual calibration curves. This appears as a difference in power extracted from the circuits. Determination of circuit power with respect to facility power was not accomplished, due to plant availability and station generation assignment

*One of the two intermediate heat exchangers had previously been removed from circuit one. The ramps imposed would have exceeded the capacity of the remaining IHX; this circuit was, therefore, checked out separately.

Figure 630-1. HNPF Primary Sodium Flowmeter Calibration
(obtained from AI-PC-530, Heat Transfer Test, Data)

Figure 630-2. HNPF Secondary Sodium Flowmeter Calibration
(obtained from AI-PC-530, Heat Transfer Test, Data)

factors. It is anticipated that a deviation of individual circuit power generation of up to 10 to 12% from the mean value will be obtained.

The final controller adjustments at the completion of the testing program were as follows:

WaIC-501

a) Proportional band	150%
b) Reset time	0.2 min/repeat

3. Sodium Flow Control

The primary and secondary sodium flow controllers were checked on the process channels in accordance with the prepared procedure. During the course of the testing program, different sodium flowmeter calibrations were incorporated into the control system spans.* The flow controllers were insensitive to the actual flowmeter characteristic, but the relationship between primary and secondary sodium flows affected system temperatures, which in turn affected the primary cold-leg temperature trim potential of TRC-101, -201, and -301.

From initiation of the post critical testing program until the 20 Mwt test level was reached, TRC-101, -201, and -301 were effectively deactivated by maintaining these recorder-controllers on manual and adjusting the temperature bias regulators to yield a 9 psi demand, corresponding to a +25°F temperature correction, or a 585°F primary return-leg temperature. At the 20 Mwt test level, TRC-101, -201, and -301 were placed into service long enough to allow a check-out under operating conditions, after which these recorder-controllers were again deactivated.

The expected primary cold-leg operating temperature range of 560 to 610°F from 0 to 100% power has not yet been verified to be an actual representation of the natural heat transfer characteristics of the HNPF. It is anticipated that the range will be more on the order of 580 to 610°F, due to the regenerative effect of the bayonet tube design in the steam generator evaporators.† Plant availability and operating conditions have not yet permitted determination of the actual characteristics.

*Refer to test completion report AI-PC-830, Master Schedule, Including Rise-to-Power, P. F. Shaw, Atomics International (November 1963) for further information.

†Refer to test completion report AI-PC-530, Heat Transfer Tests, A. J. Divis, Atomics International (February 1964).

Some difficulty was encountered in ensuring smooth startup of the pumps. Since this is essential for hot startup, it was necessary that the cause be investigated and corrected. The condition was determined to be due to a combination of "zero" and "span" adjustment on the pump speed controller, possibly aggravated to some unknown degree by pump binding.* The speed controllers were retuned, after which smooth startup could be accomplished.

During one period of high power operation, a normal grid system voltage drop resulted in decreased pump speeds and deviation transfers to manual control. The pump speed controller spans were increased so that the normal grid voltage drop would not decrease the maximum speed controls below the full power values.

Excessive temperatures in and around the six pump speed control cabinets are believed to have contributed substantially to both the controller drift and the decreased maximum speeds. Auxiliary cooling was recommended as a method of reducing speed controller cabinet temperatures and is under consideration.

Final controller adjustments at the completion of the testing program were as follows:

Primary sodium flow controllers

FRC-101 and -201

a) Proportional band	130%
b) Reset time	0.2 min/ repeat

FRC-301

a) Proportional band	130%
b) Reset time	0.15 min/ repeat

TRC-101, -201, and -301

a) Proportional band	200%
b) Reset time	7 min/ repeat
c) Rate time	0.01 min

*The pump binding condition subsequently interrupted the testing program. This condition was corrected by removing and cleaning the secondary pumps, filtering secondary system sodium, and installing auxiliary cooling to reduce pump case warpage.

Secondary sodium flow controllers

FRC-102, -202, and -302

a) Proportional band	100%
b) Reset time	0.2 min/repeat

4. Nuclear Power Computer

Checkout of the nuclear power computer was accomplished in accordance with the test procedure at 10 Mwt on semi-automatic control, with the demand generated from HCS-504, steam demand hand control station. Performance of the computer, together with peripheral temperature compensation equipment, was uniformly excellent and trouble free throughout the program. The different flowmeter calibrations used at various times during the program had no effect on the overall performance of the system.

Near the end of the testing program, failure of an amplifier in HCS-12, fuel outlet temperature flux compensation, while on fully automatic load-following control at a steady power of 35 Mwe, resulted in a negative rate dT/dt scram. Review of conditions following the scram indicated that the loss of the temperature compensation flux demand signal caused rods to be driven in, which was immediately followed by a decrease in fuel channel outlet temperatures and the scram. The amplifier was replaced and the HNPF was returned to power.

Final controller adjustments at the end of the testing program were as follows:

TRC-2

a) Proportional band	250%
b) Reset time	3 min/repeat
c) Rate time	0 min (OFF)

TRCp-1

Amplifier gain	2.5

5. Neutron Flux Control

Power was increased to the first planned test level of 2.5 Mwt nominal power. At this point, the power level was not sufficient to separate control system problems from noise on monitored process variables. Testing

planned for this level was deferred to the next planned power level (10 Mwt). At this higher level, previously observed problems in the flux controller became readily apparent. Part of these problems were a result of wiring errors and poor connections. The work accomplished on the flux controller during this period is as follows:

- a) Connections were changed from "as-shown" on Foxboro drawing WE-2802 (which is incorrect) to "as-shown" on Foxboro drawing WE-2390.
- b) An erratic deviation alarm and transfer to manual was traced to two loose connections on the flux deviation unit. Tightening the connections eliminated the problem.
- c) Two modifications to the flux controller were made, as follows:
 - 1) A bias potentiometer setting was made which modified the $\frac{1}{n} \left(\frac{1}{\text{number of rods on auto}} \right)$ compensation to 10:1 between 4% and 100% power, instead of the original 25:1 ratio.
 - 2) The "number of rods on control" gain variation was deleted. This was originally set up for one to 31 rods. The present critical rod configuration calls for three rods on control. A gain of two compensation is not critical, and from one to six rods may be placed on control without further adjustment.
- d) The Philbrick XP-7 dead-band amplifier was found to be defective and was replaced.
- e) A high noise level caused erratic automatic operation below 2.5% reactor power. The noise was determined to be a result of relay arcing and the nuclear instrument cable checking pulse (see f, below). Installation of simple R-C suppressors on the relays permitted automatic operation down to the design point of 1% of reactor power.
- f) In the General Electric nuclear instrumentation system, there is a periodic (about 1 cps) pulse of 20 volts magnitude and 70 microseconds duration that checks continuity of the system. Some five stages of high frequency filters contained in the flux controller are not adequate to reduce the pulse to a size that will not affect the

flux controller. It was clearly demonstrated that the pulse defeated the auctioneer function of the system and was, therefore, incompatible with operation of the flux controller.

As a permanent correction, a wiring modification was made to power-range control system nuclear channels VIII and IX. Now, the trouble pulse is off when the power range channels are in the OPERATE position. The pulse and alarms are in service when the flux amplifiers are switched to the CALIBRATE position. There is an auctioneer circuit for loss-of-signal protection in the flux controller and a flux comparator in the nuclear panel which compares the signal from channel VIII with that of channel IX. Performance and safety of the system are, therefore, not compromised by the modification.

During initial power operation, but before load-following operation, it was necessary to filter power supplies due to noise on the critical and noise-free sources. Some components were changed over from noise-free to critical-power (Table II). It was also obvious that the low frequency flowmeter oscillations were going to cause problems in the nuclear power computer. These were adequately corrected by filters.

At the start of the load-following operation, a relay in the flux controller failed. The failure was caused by two wires shorting inside the relay. The wires were in contact and continued motion of the relay parts resulted in an insulation failure. The relay was replaced.

Near the end of the testing program, the flux controller failed. The problems encountered and corrective actions taken were as follows:

- a) A transistor in the Philbrick XP-7 integrator power supply failed and was replaced.
- b) The XP-7 integrator, which probably had been slowly degrading, became nonfunctional. Since the XP-7 is no longer manufactured, it was replaced with an XP-2 integrator which proved to be a satisfactory substitution.
- c) A transistor in the new XP-2 output circuit failed and was replaced.

TABLE II

COMPONENTS TRANSFERRED FROM NOISE-FREE BUS 22-S
TO CRITICAL-POWER BUS 22-X

Instrument	Component	Date	Instrument	Component	Date
HCS-11	1 of 3	2-11-63	NC-1/HCS-1	3 of 23	2-11-63
HCS-10	1 of 3	↓	-	5 of 23	↓
HCS-10	2 of 3		-	6 of 23	
TI-1	1 of 2		-	7 of 23	
TI-2	1 of 2		-	9 of 23	
TI-3	1 of 2		-	11 of 23	
TI-1/2/3	2 of 2		-	12 of 23	
TRCp-1	1 of 7		-	-	
	2 of 7		-	-	
	3 of 7		FRC-101T	1 of 3	
	5 of 7		FRC-201T	1 of 3	
	6 of 7		FRC-301T	1 of 3	
HCS-7	1 of 3		FRC-102T	1 of 2	
HCS-7	2 of 3		FRC-202T	1 of 2	
TI-4	1 of 2		FRC-302T	1 of 2	
TI-5	1 of 2		TRC-3	2 of 6	
TI-4/5	1 of 1				↓
TRC-2	1 of 5				
	2 of 5				
	3 of 5				
	5 of 5				
HCS-12	1 of 2				
HCS-12	2 of 2				
NWaCp-1	1 of 7				
NWaCp-1	2 of 7				
HCS-6	1 of 1				
NLCp-1	1 of 4				
HCS-5	1 of 3				
HCS-5	2 of 3				
HCS-4	2 of 4				
HCS-4	3 of 4				
NLR-1/2	1 of 1				
NC-1/HCS-1	1 of 23				
NC-1/HCS-1	2 of 23				

This work was hampered by the physical arrangement and packaging of components within the flux controller. Down time was extended because a spare flux controller was not available. A recommendation was made that an improved flux controller be fabricated and installed, and the present one be used as a spare. An improved flux controller has been ordered.

Component failure in the flux controller and peripheral equipment is attributed to break-in and random failures. Ambient temperature inside the control system cabinet tends to run high. Since random failure of electronic components increases exponentially with temperature, a recommendation was made to study and rectify the temperature condition. The two recommendations were under consideration at the end of the testing program.

Final controller adjustments at the end of the program were as follows:

NC-1

a) Proportional band	3%
b) Reset time	0.01 min/repeat
c) Rate	0 min (OFF)
d) Deadband	0.2 in.

6. Moderator Coolant Control

HNPf moderator coolant temperatures have proven to be self-regulating. The control system was checked out with an artificial setpoint, after which it was maintained on manual control with only minor adjustments in valve position required. These adjustments were accomplished under the cognizance of AI-PC-510, Hydraulic Test-Main Heat Transfer System. Scaling of the recorder was adjusted to 60 mv for full-scale deflection. The low flow alarm was set at 5% of scale; the valve position alarm was set for 98% closed. Final controller adjustments at the end of the testing program were as follows:

TRC-3

a) Proportional band	50%
b) Reset time	0.5 min/repeat
c) Rate	0.01 min

7. Convection Flow Control

This system was demonstrated to control properly during the several scrams and hot startups. It was also demonstrated that the proper standby condition for the system is on "Manual" with a "99% closed" preset valve position demand. Since the initial temperature transient is negative and recovery is relatively slow, this mode of standby limits the magnitude of the potential transient. A short discussion of controller action after scram is presented below for clarification.

The initial temperature transient after scram results in a deviation transfer to "Manual" in HCS-8, convection flow control hand control station. HCS-8, however, has an "as is" failure feature, and transfers to manual with the output as demanded by TRC-4, convection flow controller. The demand from TRC-4 depends on the temperature trend just preceding the scram, and could be anything from full closed to full open. The valves are not affected by this demand until 30 seconds after initiation of a scram — in the interim the valves are programmed to a preset position, presently set at 99% closed. If HCS-8 were on automatic control at the time of scram initiation, the valves would move to somewhere between the full-open and full-closed position demanded by TRC-4 at the time of deviation transfer of HCS-8 to "Manual".

The initial temperature transient has been in the order of minus 50 F°; afterglow temperature recovery rate has not exceeded 50 F°/hr. Once the temperature has recovered to within the deviation transfer figure (plus or minus 25 F° difference between fuel channel outlet demand and measured temperature), HCS-8 can be placed on automatic control. After the transfer to "Auto", HCS-8 will control fuel channel outlet temperatures at TRCP-1 (plenum temperature) setpoint, within the limits imposed by afterglow heat generation, system losses, and operator programming (maintain temperature or cooldown).

At the end of the testing program, the final controller adjustments were as follows:

TRC-4

a) Proportional band	150%
b) Reset time	0.4 min/repeat
c) Rate	0.01 min

8. Feedwater Level Control

After operation at 10 Mwt and before power was increased to 20 Mwt, temporary connections were made from the feedwater level controller outputs to the 1-1/2 in. feedwater bypass valves to permit automatic single-element control of feedwater on the smaller valves. This was desirable, since leakage through the larger 4 in. valves was determined to be 6 to 8% in the full closed position; this exceeded makeup requirements.

Re-stroking of the larger valves during a shutdown after 15% power operation resulted in a leakage reduction. The feedwater level controllers were rescaled with a 45 in. offset to cover a 0 to 30 in. range, instead of the original 0 to 75 in. range. After completion of this, stable three-element control could be maintained down to approximately 8% of rated capacity on the 4 in. valves. Stable single-element control on the smaller valves permitted extension of automatic control to below 5%. In addition, bleed volume (surge) chambers were installed on the feedwater flow signal input to the controllers to dampen boiler feed pump pressure pulsations.

A later modification was incorporated into the feedwater level control system on loop one, which resulted in a split range control system operating both the 1-1/2 and 4 in. valves. The 1-1/2 in. valve operates from full closed to full open on a 3 to 9 psi air signal, and the 4 in. valve operates from full closed to full open on a 6 to 15 psi air signal. This modification extended the useful range of automatic three-element control down to near-zero power. Operating experience verified the efficacy of this modification, and the other two circuits have been similarly modified.

Excessive moisture carryover in the steam generators at high steaming rates is being investigated.* An acceptable program of water level was derived (Figure 630-3) which permitted operation up to and including full power without excessive carryover. Further investigation is required for optimization of the feedwater level programming with respect to circuit steaming rate.

*Refer to test completion report AI-PC-552, Steam Generator -- Steam Purity by L. E. Keller (CPPD) Atomics International (February 1964).

Figure 630-3. Permissible Water Level of HNPF Steam Generators
(from AI-PC-552, Steam Generators-Steam Purity)

At the end of the testing program, controller adjustments were as follows:

LRC-151, 251 and 351

a) Proportional band	100%
b) Reset time	2 min/repeat

FC-151, 251 and 351

a) Proportional band	70%
b) Reset	0.8 min/repeat

9. Attemperator Control

The attemperator control system was first placed into operation at the 35 Mwt test power level. Control was adequate, and no further testing effort other than surveillance was applied to this subsystem.

At the close of the formal testing program, the operator (Consumers' Public Power District) was giving some thought to discontinuing the use of these controllers. There is no known objection to this, since the purpose of the system is to condition HNPF steam to the turbine requirements, as desired by CPPD.

Controller adjustments at the end of the testing program were as follows:

TRC-152, 252 and 352

a) Proportional band	25%
b) Reset time	0.5 min/repeat

10. Power Relief Control

Proper performance of the three power relief controllers was demonstrated during checkout of the steam generator relief valves.

11. Primary Block Valve Control

No testing beyond that accomplished in the preoperational test was considered necessary.

12. Steam Dump Control

The steam dump control system was placed in service and checked out at the 2.5 Mwt test level, after which no further effort beyond minor fine tuning was required.

At the end of the testing program, controller adjustments were as follows:

PIC-501

a) Proportional band	20%
b) Reset time	0.2 min/repeat

B. LOW POWER (<15% REACTOR POWER) SYSTEM PERFORMANCE

1. Semi-Automatic Operation at 10 Mwt

After completion of the above work, the plant was operated on semi-automatic control. Starting on manual control, individual subsystems were aligned and transferred to "Automatic", with the demand generated from HCS-504, steam demand hand control system. Temperature control was noted to be somewhat sluggish, and a small limit cycle of about 2 F°, double amplitude and 45 second period, was observed when on temperature control alone.

2. Load-Following Operation at 20 Mwt

At the 20 Mwt reactor power test level, the control system was operated in the fully automatic, load-following mode. Temporary test connections were made to enable the use of the steam dump system flow and pressure (instead of the normal turbine steam flow and pressure) for the inputs to the plant power control system. Changes of power demand were generated by manually adjusting the position of the steam dump valve. The plant control system responded in a rapid and stable manner to the 2 to 5% changes which were made in power demand. Although temperature control alone was somewhat sluggish, no changes were noted in reactor outlet temperatures during load changes. This is due to the lead action of the nuclear power computer, which is necessary for load following operation.

3. Temperature Control at 20 Mwt

At a reactor power level of 20 Mwt, the temperature limit cycle reported in III-B-1 above was reexamined. With HCS-6 (nuclear power computer) and

HCS-7 (upper plenum temperature demand) in "Manual" and HCS-5 (flux controller) and HCS-12 (fuel outlet temperature) compensation in "Automatic", a well-defined limit cycle of 2°F double amplitude and about 50 sec period was measured. There was an apparent 75 degree phase shift between a measured fuel temperature (about middle height of core, as obtained from the instrumented fuel element in core position C-6) and the "long probe" thermocouple in the fuel element in core position C-2.

Decreasing the proportional band by a factor of two established a divergent oscillation which was allowed to run to a peak double amplitude of about 11°F in sodium outlet, 16°F in measured center height fuel temperature, 1.7% flux, and 1.5 inches of travel on rods 14, 16, and 18. A quick reduction by a factor of four in the proportional band resulted in a damped response in one-half cycle. During the period of divergent oscillation, there was:

- a) zero phase shift between rod position and neutron flux
- b) ninety degrees phase shift between flux and center height fuel temperature
- c) ninety degrees phase shift between center height fuel temperature and the measured long probe fuel channel outlet temperature.

4. Hot Startup^{*}

"On March 1, 1963, following a planned scram with a reactor ΔT of 220°F, a hot startup was made following HNPF SOP No. 1802 (Rev. 1). A careful review of plant process data indicates that the following took place (see Figure 630-4).

- "a) At the time the pumps were started, the throttle valves were 25 to 30% open. Convection flow control was on automatic at HCS-8 and the primary to secondary flow ratio relays were set at one. The reactor power was 2×10^{-6} amps ($\sim 7\%$) on channel III.
- "b) When the pump couplings were energized, with the pump speed hand control stations at zero output, no flow or temperature transients or changes occurred. This indicates that the pumps remained at zero speed.

^{*}This section is a copy of an Inter-Office Letter to J. E. Owens from R. W. Truitt, dated March 15, 1963, subject: Hot Startup.

Figure 630-4. Results of HNPf Hot Startup of March 1, 1963

- "c) The secondary pumps were started and the flows increased to approximately 5%. The primary pumps were then started and the flows increased to 5%. The flow recorders indicate that these increases were gradual, taking place over a period of several minutes. There was no evidence of any overshoot.
- "d) Reactor inlet temperature on loop No. 1 decreased from 590°F to 575°F in a period of 5 minutes and then leveled out at 570°F. Reactor inlet temperature on loop No. 2 decreased from 580°F to 550°F in 5 minutes and then leveled out at 570°F. Reactor inlet temperature on loop No. 3 decreased from 590°F to 550°F in 10 minutes and then leveled out at 560°F. Average reactor inlet temperature is shown on the attached graph for a two hour period.
- "e) Fuel outlet temperatures were level at 660°F prior to pump start. After pump start, they dropped 10 F° in a period of 2 or 3 minutes and then leveled out at 650°F. Fuel outlet temperature and reactor ΔT are shown for a two hour period on the attached graph.
- "f) There was no detectable change in reactor power as indicated on neutron flux channel No. III associated with the pump startup and flow increase.
- "g) Secondary sodium temperatures decreased after the pumps were started. Sodium to superheater temperatures dropped about 50°F in 45 minutes and then leveled out. Sodium from superheater temperatures dropped 50 F° in 30 minutes and then leveled out. Sodium from evaporator temperatures dropped 30 F° in 15 minutes and then leveled out.

"The hot startup was successfully made. No excessive temperature changes or transients were observed. It should be possible to make a successful hot startup after a full ΔT scram using the same procedure.

"The main area of concern during a hot startup is the introduction of large negative temperature changes. In addition to thermal shocks, this can cause the introduction of positive reactivity due to cooling of the fuel elements. In order to minimize such effects, the following precautions are observed:

- "a) All flow changes are made slowly (not to exceed 1% on hand control station at any one time).
- "b) The flux level is carefully monitored during flow changes. An operator is stationed at control rod bank switch when pump speed changes are made.
- "c) The operator is prepared to scram the reactor if flux should increase at a rate equal to a 10 second period or less."

C. OPERATION IN THE NORMAL POWER RANGE

1. Introduction of HNPF Steam to the Turbine

Introduction of HNPF steam to the Sheldon Station turbine was accomplished in accordance with the test procedure. The conventional plant was on pressure control; the HNPF was on manual control from HCS-504, steam demand, at less than 38 Mwt. Steam from the HNPF was being exhausted to the steam dump at a rate of about 89,000 lb/hr with the steam dump on manual valve position control.

At a power level of 20 Mwe gross generation, conventional plant steam conditions were matched to HNPF conditions. V-5164, HNPF main steam header block valve bypass, was opened slowly, after which V-5163, HNPF main steam header block, was slowly opened.

When the HNPF main steam header block valve was fully opened, the steam dump valve was slowly closed from HCS-505. As soon as the steam dump valve was fully closed, the setpoint of PIC-501, steam dump pressure controller, was adjusted approximately 40 psi above steam header pressure and HCS-505 was transferred to "Auto". At this time, the HNPF was supplying all generated steam to the turbine at a rate of 89,000 lb/hr.

Further controller adjustments were accomplished on the plant power control, nuclear power computer and feedwater level control subsystems during and following the "hot transfer" described above.

A subsequent rise in grid demand was met by increasing HNPF power output to correspond to 15 Mwe turbine power. At this time, the HNPF was transferred to automatic load-following control and the conventional plant was transferred to manual control. As grid demand decreased, the conventional

plant was backed down and the conventional boilers were dropped, leaving the HNPF load-following at 20 Mwe gross station generation, with the steam dump system in standby and the total nuclear output going to the turbine.

2. Parallel Operation

Following initial operation of the turbine from HNPF steam, the two plants (conventional and nuclear) were operated singly and in parallel through a wide range of power. During all stages of parallel operation, one plant was base-loaded and the other was load-following to meet generator requirement fluctuations. "Load-following" for the most part consisted of adjusting to small (approximately 1 Mwe) step changes at one to two minute intervals. The normal mode of station operation had thus far been as a base-load plant; power changes made at the dispatcher's request are normally accomplished in a series of small manual step changes.

3. Load-Following Test

No HNPF load-following limitation was approached during station generation changes made at the dispatcher's request. In accordance with the test procedure and with the dispatcher's cooperation, step changes in station generation rate were imposed by adjustment of the turbine throttle in order to create the transient conditions necessary for the fine tuning of controllers and to determine a figure for HNPF load-following capability.

Step changes were made from a base turbine generator power level of 40 Mwe gross. Each incremental step was made down from the base and, after apparently stable conditions were achieved, power was returned to the base level in similar positive steps. Steps imposed were 1.0, 2.5, 5.5, and 6 Mwe. Rate of the last load increase was 41.4 Mwe/min. The rate of change of measured fuel channel outlet temperature (dT/dt) meters moved to 30% scale on the last step; in keeping with the procedure, this step was defined as the HNPF load-following limit for this type and size of transient. It should be noted that this rate exceeded the minimum design objective of 5 Mwe/min by more than a factor of 8.

The loss of load test discussed in III-C-4, demonstrated the capability of the HNPF to ride out a dropped (approximately) 30% load and the resulting 15% power decrease to steam dump capacity in a one minute interval, without scrambling. Several 5 Mwe and smaller ramps were performed after the above to optimize controller adjustments.

4. Loss-of-Load Test

A loss-of-load test was performed from 25 Mwe. Conditions just prior to the test were as follows:

- a) reserve station transformer in service
- b) HNPF supplying full turbine requirements
- c) reactor outlet temperature 930°F
- d) steam dump system in normal standby
- e) HNPF on automatic load-following control, supplying the full turbine steam requirements.

From these conditions, the turbine was tripped manually. HCS-504, power demand hand control station, was transferred to manual and immediately reduced to 15%. As steam flow dropped and steam header pressure built up, the control system backed the reactor power down to 15%. The power relief valves each lifted and reseated six times. An operator transferred the steam dump valve position controller to "Manual" to accelerate opening of the valve; this was not necessary for the success of the test.

Approximately seven minutes after turbine trip, the generator was returned to service using steam from the HNPF. Ten minutes later, the HNPF was on fully automatic load-following control and the steam dump valve was being closed. Reactor flux, reactor period, sodium flows, and fuel channel outlet and in-fuel temperature for the transient period are presented in Figures 630-5 and 630-6.

After the test procedure was developed, and before performance, the automatic transfer from normal to reserve station transformer was disabled in the interest of protecting various large electrical motors. From the standpoint of the original purpose, the test was of academic interest only since without the automatic transfer feature, the reactor would scram on loss of load with a resultant trip of the normal station transformer. However, the test did serve to demonstrate control system adequacy by the introduction of transient conditions that were more severe than those the HNPF will normally experience.

Figure 630-5. Sodium Flow and Temperatures, 25 Mwe Loss of Load Test

CHANNELS III & IV LOG N FLUX RECORDER (amps)

CHANNELS III & IV PERIOD RECORDER (sec)

Figure 630-6. Period and Flux, 25 Mwe Loss of Load Test

5. Power Range Scram Experience

a. Planned Scram From 40 Mwe

A planned scram was performed from 40 Mwe with the HNPF supplying the full turbine steam requirements. The turbine was left on the line during the transient, to investigate the effect on the HNPF. The conventional plant was on hot standby to pick up the load. After scram, turbine power dropped to 2 Mwe, at which time steam temperatures indicated that the steam quality was poor, and the conventional plant was brought on the line. No severe pressure or temperature drops in the HNPF superheaters occurred during the period of power decay.

As the HNPF steam header pressure built up, the steam dump valve opened. At this time, the HNPF main steam header block valve was closed, isolating the HNPF from the turbine.

Prior to the scram, HCS-8, convection flow control, was on "Manual" with a 99% closed valve position demand present. The valves promptly closed to the initial position of 99% closed. At the end of the 30 sec time delay, control was properly transferred to HCS-8.

The indicated fuel channel outlet temperature just before scram was 945°F. This indication dropped rapidly to 855°F after scram, then recovered to 925°F. The subsequent rate of decrease was low enough to allow the fuel channel outlet demand temperature to overtake the measured temperature at the 50°F/hr limited rate of change (the fuel channel outlet temperature demand was reduced from HCS-12), at which time convection flow control was placed on "Automatic" from TRC-4, convection flow control, for rate-limited cooldown to 500°F isothermal.

b. Other Scrams of Significance to Test

Two more of the 11 scrams which occurred during the power range phase of the test program are of particular interest, in that these scrams yielded information regarding refinement of the control system.

The first of these two scrams was induced at a time when flux control was on "Manual" for the purpose of repositioning a feedback mechanism. Three rods were on manual control from HCS-1. The operator noticed that the rate of change of fuel channel outlet temperature (dT/dt) meters were drifting down-scale (negative). He withdrew rods to stop the drift and almost immediately

sent the dT/dt indications upscale. Insertion of rods to stop the positive temperature rate swing resulted in a negative deflection of the dT/dt meters, closely followed by a negative dT/dt scram. At the time of scram initiation, the HNPF was carrying the full station load of 15 Mwe gross.

The second of these two scrams occurred after the three day full power run. Power was being decreased from 80 Mwe to 15 Mwe at a rate of approximately 3 Mwe/min. At the end of the decrease to 18 Mwe, the demand flux increased sharply, causing the flux controller to pull rods further than the operator considered proper. The operator transferred the flux controller to "Manual". Immediately afterward, the reactor scrammed on a negative dT/dt scram; the demand was real, the control system was performing properly.

During the performance of power ramps for the power coefficient test it was determined that the maximum amount of rod withdrawal possible at a constant power extraction rate in the power range of operation is that amount corresponding to about six seconds with three rods on control, and that must be followed immediately by rod insertion. Any greater withdrawal, or any delay in insertion, would result in a dT/dt scram. This six second figure is for the lower end of the power range, and decreases to approximately three seconds at full power.

The above does not imply that the reactor is difficult to control manually. On the contrary, manual control is not difficult. The amount of permissible rod movement, however, is relatively small.

6. One-Loop Shutdown

During the performance of a test to determine steam generator carry-over, a special level probe fixture on steam generator No. 1 developed a high-pressure steam leak. The decision was made to shut down the single loop to permit cooldown of the steam generator and repair of the leaking fixture. This one-loop shutdown was performed in accordance with the plant control system test procedure.

Throughout the shutdown operation, the HNPF was base-loaded from HCS-504, power demand, and carrying approximately 35 of the 50 Mwe gross station load. Primary and secondary sodium loops No. 1 were transferred to manual control from HCS-501, circuit No. 1 sodium flow control. From HCS-501, circuit flow was reduced in steps to zero demand. Automatic three-element

feedwater level control was maintained until circuit power was near zero and was then switched to manual level control on the bypass valve. The circuit steam header block valve was closed and the superheater vent valves were opened slightly.

Simultaneously with the above operation, the circuit sodium block and two throttle valves were manually closed from the control room hand control stations, after which switch 69-1, circuit No. 1 one-loop shutdown switch, was switched to the bypass position. This successfully completed the one-loop shutdown operation; the portion of the test procedure that called for placing the circuit back into service was deleted as this is not an expected operation.

At a later date, both No. 3 sodium pumps were similarly shut down to permit the replacement of the seal oil pump on No. 3 secondary pump. All three sodium circuit valves were left open. Approximately 15% HNPF power was maintained throughout; process system temperatures remained essentially unchanged, and circuit No. 3 continued to contribute a (reduced) portion of the total steaming load. Temperatures throughout circuit No. 3 were maintained essentially constant by thermal convection flow.

7. Power Operation Demonstration

Early in the power-range phase of the testing program, facility operation took on most of the aspects of an operating power plant. Testing, operation activities, and power production were achieved by the close cooperation of site personnel (CPPD and AI) and the Nebraska Public Power System dispatcher. The many startups and shutdowns, parallel operation and loading-following operations, served to create the conditions required for performance of the various tests.

The final power run performed under the cognizance of the test program was an 84-hour run at gross station generation power levels of 80 to 83 Mwe, with the reactor supplying the full steam requirements in the automatic load-following control mode and the conventional boilers out of service. Performance of the plant control system was satisfactory throughout, with the exception of one short period when pump speed controllers had to be respanned.*

*Refer to Section III-A-3 of this report.

D. GENERAL COMMENTS AND OBSERVATIONS

As indicated in Section III-A, a substantial number of difficulties were encountered in the course of the test. However, this must not be construed as being indicative of system qualifications. On the contrary, the system performs extremely well. The apparent reliability indicated is misleading since the incidence of break-in failures is not known. More extensive operating history is necessary to arrive at a figure for reliability; this is beyond the scope of this test.

Unresolved areas at the conclusion of the test were as follows:

- 1) Provisions for increased control room and pump speed control cabinet cooling capacity
- 2) Resolution of noise-free and critical power supplies
- 3) Feedwater level programming
- 4) Determination of primary sodium cold leg temperature characteristics, and resolution of mode and range of operation of cold leg temperature controllers
- 5) Determination of circuit load division over the full normal range of power operation
- 6) Determination of component and system reliability.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with post critical test AI-PC -630, Plant Control System, as listed in NAA-SR-5700 (Supplement 4), pages XI (c)-26 and -27, with remarks regarding the test to explain compliance with the criteria:

- a) Criterion: Plant control system is stable at all power levels.

Remarks: After final tuning, the plant control system was demonstrated to be highly responsive and completely stable under all conditions of plant operation imposed during the testing program. These test conditions are believed to be more severe than those normally encountered in routine operation.

- b) Criterion: Bumpless transfer between automatic and manual control is possible.

Remarks: It was demonstrated many times over that bumpless transfer between manual and automatic control is easily accomplished by any qualified operator.

- c) Criterion: Plant control system will follow load changes of at least 5 Mwe/min.

Remarks: During the testing program, the plant control system followed load changes that were at a substantially higher rate than the above figure.

- d) Criterion: Feedwater control system controls evaporator level to ± 2 inches of normal water level at all power levels and during maximum load transients.

Remarks: It was demonstrated that the system was capable of controlling water level within closer limits than the above. However, due to limitations imposed on acceptable water levels by moisture carryover at higher steaming rates, further investigation of feedwater level is required for definition of "normal" water level. This effort will continue outside the cognizance of this test.

- e) Criterion: Plant power control subsystem controls steam pressure to turbine at 800 psig and load swings can be handled with pressure variations of less than ± 25 psig.

Remarks: This was demonstrated to be well within the capabilities of the system.

- f) Criterion: Attemperator control subsystem holds steam temperature at approximately 825 F.

Remarks: This was demonstrated to be within system capabilities. However, the operator (CPPD) has indicated that use of attemperation may be discontinued in order to gain the benefit of the increased superheat in the normal range of power operation.

- g) Criterion: Division of Load Computer correctly distributes and programs plant load to primary and secondary heat transfer systems, with primary ratio at 1.1, and secondary ratio at 1.0.

Remarks: The computer correctly distributes and programs plant load to the heat transfer loops. Distribution of the load to the three circuits will not result in equal portions at all power levels, due to the differences between the six sodium flowmeter calibrations, and the differences between each calibration and the straight-line approximation. The differences in loop power generation rates have no significance, provided no circuit is operated beyond 100% capacity, and provided steam conditions and sodium temperatures are near the average for the three circuits. The actual setting of the ratio relays depends on process variables and accuracy of the dial calibrations; the numbers have no importance relative to the importance of the controlled processes. Further operation throughout the power range is necessary to optimize these settings.

- h) Criterion: Nuclear Power Computer correctly calculates reactor power as a function of load and outlet temperature.

Remarks: This was verified. It should be noted that the peripheral temperature compensation equipment contributes substantially to uniform, stable reactor control.

- i) Criterion: Neutron Flux Controller is stable at all power levels and correctly controls neutron flux regardless of the number of rods on automatic control.

Remarks: The controller was demonstrated to render stable control from approximately 1 to 100% of full power. As presently set up, the controller can accommodate from 1 to 6 control rods on automatic control. The present rod configuration is for three on "Automatic"; it is not anticipated that more than six rods will ever be required.

Failures in this subsystem have prompted a review of maintainability and reliability. A recommendation to modify portions of the system was made. Improved equipment is on order.

V. EVALUATION

The test objective; i.e., verifying satisfactory operation of the plant control subsystems, was met. The subsystems operate properly throughout the range of reactor operation. All test completion criteria associated with post critical test AI-PC-630, as listed in the HNPF safeguards summary, NAA-SR-5700 Supplement 4, were adequately met. Comments pertinent to the individual subsystems are covered in the following paragraphs.

A. PLANT POWER CONTROL

System control adequacy was demonstrated during reactor load-following operation in the power range. Operation at all power levels indicated good operational stability.

B. DIVISION OF LOAD COMPUTER

Performance of the division of load computer is satisfactory even though some differences in loop power levels was in evidence. This condition is not deemed critical as adjustment of the ranges, to optimize the settings, will correct the condition. The necessary adjustments will be made as additional operational data are obtained.

C. SODIUM FLOW CONTROL

Satisfactory performance of the primary and secondary sodium flow controllers was demonstrated during the testing program. During this period the speed controllers were retuned, thus assuring smooth startup of the pumps. Primary cold leg temperature controllers were not placed in service during the testing program, but were subsequently placed in operation and have proved to be satisfactory.

D. NUCLEAR POWER COMPUTER

Performance of this subsystem was very good. Computer reliability appears to be excellent. No problem areas were apparent during the test and none are anticipated.

E. NEUTRON FLUX CONTROLLER

Design of the neutron flux controller appears to be more than adequate to provide satisfactory control. However, reliability of components in this unit is below average. Since completion of the testing program a new flux controller utilizing high reliability plug-in components was ordered. Upon installation of the new unit, the existing unit will be utilized as a spare.

F. MODERATOR COOLANT CONTROL

During the testing program the moderator coolant control was not placed in service because its use was not deemed necessary. Flat and scallop temperatures on the instrumented moderator can appear to be consistent with process channel temperatures thus indicating approximately the correct amount of moderator cooling. Control system adjustments made from a simulated setpoint should prove adequate for future system operation.

G. CONVECTION FLOW CONTROL

Observation of the several scrams which occurred during the testing program indicated that this system is working properly. Temperature decay rates can be held to 50 F°/hr or less, after a scram from any power level.

H. FEEDWATER LEVEL CONTROL

Satisfactory operation of the feedwater level control was attained following minor system changes. The system was modified to a split range control system operating both the 1-1/2 in. and 4 in. valves. As a result of this modification the system can be operated on three-element control down to very low power levels.

Level programming as a function of steam flow is being investigated in connection with the evaporator moisture carryover problems encountered during the testing program.

I. ATTEMPERATOR CONTROL

Adequate temperature control was demonstrated during the testing program. No problem areas were encountered.

J. POWER RELIEF CONTROL

Satisfactory operation of this system was demonstrated during the checkout of steam generator relief valves.

K. PRIMARY BLOCK VALVE CONTROL

Satisfactory operation of this control was verified during preoperational testing.

L. STEAM DUMP CONTROL

Operation of this system was verified early in the testing program. System control proved to be adequate following minor adjustments.

HALLAM NUCLEAR POWER FACILITY

Post Critical Test Procedure

AI-PC-640

Emergency Feedwater System Capacity

Test Procedure Submitted for Approval

6/30/61
Date

James E. Chene
Atomic Energy International

Test Procedure Reviewed

2/7/63
Date

E. L. Simpson
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Commission

Test Approved for Performance

2/29/63
Date

James E. Chene
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Test Completed

3/6/63
Date

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Test Approved for Performance

2/20/63
Date

R. E. Buntam
Atomic Energy
Commission

Test Completed

3/6/63
Date

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Atomic Energy
Commission

HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
EMERGENCY FEEDWATER SYSTEM

AI-PC -640

Test and Analysis
by
I. O. SUNDERMAN
(Consumers Public Power District)

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A T O M I C S I N T E R N A T I O N A L
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HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
EMERGENCY FEEDWATER SYSTEM, AI-PC -640

I. PURPOSE OF TEST

The purpose of this test was to prove both the operational capability, and the emergency standby operability, of the emergency steam turbine and the emergency feedwater pump.

II. DISCUSSION OF TEST PERFORMANCE

A. SYSTEM DESCRIPTION

In the event of a loss of housepower, the emergency feedwater system is intended to provide feedwater for the steam generators. Basically, the system is comprised of one 225 hp steam turbine driver, and one 250 gpm boiler feedwater pump with a shaft extension driving the condensate transfer pump, which has a 55 gpm (net) capacity.

The emergency steam turbine can draw steam from either the conventional or nuclear main steam header. Turbine exhaust may be directed to either the deaerator or to the atmosphere. The emergency feedwater pump draws suction from deaerator storage and delivers effluent to either the evaporator feedwater header or to deaerator storage. Condensate is transferred from storage to the deaerator by means of the emergency condensate pump, which also supplies clean water for the emergency feedwater pump.

An electric motor-driven standby emergency feedwater pump (rated at 12.7 gpm) provides feedwater to the evaporator as necessary to maintain the minimal requirement. This pump draws suction from the condensate storage tank and delivers effluent directly to the evaporator.

B. PREPARATION FOR TESTING

Prior to beginning the test, the steam piping and condensate piping were visually traced for continuity and type of service. Valves were checked for function, location, and proper number designation. Numbers were assigned to

untagged valves and the valves were subsequently tagged. P & I drawings were verified for correctness, and errors on the "As Built" drawing were corrected.

Gages and instruments were installed on lines and equipment as required to accomplish the test. Steam lines and the turbine housing (with associated traps) were cleaned out by the steam blowdown method. A defective gasket on the feedwater pump was replaced and a gland seal was repaired.

Circuit piping conditions prior to operation of the turbine were of a circulatory nature. The feedwater pump drew suction from and returned effluent to the deaerator storage. The emergency condensate pump output was limited to feedwater pump cooling because the deaerator was serving the conventional plant. Turbine steam supply was taken from the conventional main steam header and exhausted to the deaerator to avoid wasting condensate.

Preliminary operation of the turbine involved starting the emergency turbine according to SOP No. 5017, "Manual Operation of Emergency Turbine-Driven Feedwater and Condensate Transfer Pumps". Shaft alignments were checked and the apparatus was shimmed to (hot) alignment tolerance of 0.001 in. Governor and nozzle valve adjustments were made to permit turbine operation at 3600 rpm, after the overspeed trip was tested and found to be 4170 rpm. Vibrometer readings taken after alignment are shown in Table I.

TABLE I
VIBROMETER READING

Equipment Item	Location	Reading (in.)
Emergency Turbine	Outer Bearing	0.0005
	Inner Bearing	0.0008
Emergency Feedwater Pump	West Bearing	0.0005
	East Bearing	0.0005
Emergency Condensate Pump	Inner Bearing	0.0002
	Outer Bearing	0.0000
Governor	Governor	0.0022

C. CAPABILITY TESTING

The capability portion of the test began with the reactor operating at an estimated 19.4 Mwt; nuclear generated steam was routed to the steam dump system. Steam supply for the emergency turbine was taken from the conven-

tional plant steam header; turbine discharge was to the deaerator. The deaerator and conventional plant were on normal operation. The emergency feedwater pump drew suction from the deaerator storage and delivered feedwater to the evaporators. Suction to the emergency condensate pump was from the condensate storage tank, with effluent delivery to the feedwater pump cooling system. The level in the deaerator controlled flow from the emergency condensate pump. At steady operating conditions, the following pertinent data were taken for test and performance record purposes:

<u>In The Control Room</u>	<u>At The Emergency Turbine</u>	<u>Steam Generator Con- ventional Plant, et. al.</u>
Estimated Mwt	Steam in-valve position	Steam Generator (attemp outlet) Pressure and Temperature
Feedwater Pressure	Turbine rpm	
Feedwater Temperature	Feed Pump Pressure In and Out	Steam Dump Pressure and Temperature
Feedwater Flow		
Valve Position	Condensate Pump Pres- sure In and Out	Third Extraction Pres- sure
Steam Dump Flow	Turbine Metal Tempera- ture In and Out	Deaerator Heater Shell Pressure
Evaporator Level		

Data tabulated in Table II were recorded at five separate operating levels.

TABLE II
OPERATIONAL DATA

		Level 1	Level 2	Level 3	Level 4	Level 5
Estimated Mwt*		19.4	22.9	26.6	30.5	33.8
Turbine rpm		3600	3600	3600	3543	3543
Feedwater Pump Indicated Pressure		980	980	960	920	910
Feedwater Flow Indicated lb/hr	Loop 1	27,000	27,000	33,000	37,500	42,000
	Loop 2	27,000	33,000	36,000	42,000	43,500
	Loop 3	27,000	28,500	30,000	36,000	39,000
	Total	81,000	88,500	99,000	115,500	124,500
Steam Dump Flow Indicated lb/hr		64,000	73,800	86,100	100,000	112,000

* Indicated Mwt readings were later shown by other test data to be 10% to 12% low.

Evaporator levels were maintained within operating limits. The test was concluded when the capacity of the steam dump system (110,000 lb/hr) was exceeded.

D. OPERABILITY PORTION OF TEST

The operability portion of the test was conducted following a scram from 38 Mwt reactor power level. Similar data as noted in the capability portion of this report were recorded in the operability test. Feedwater pump B was serving the evaporators; nuclear generated steam output was routed to the steam dump system.

A simulated loss of housepower (simultaneously switching off electrical power to the conventional feedwater pumps) initiated a 30 sec time delay sequence which opened the in-valve to the emergency steam turbine. The valve opened during the interval from the 35th to the 38th sec after loss of power signal and scram.

Steam supply to the turbine was taken from the HNPF main steam header and exhaust was routed to the deaerator. The emergency feedwater pump drew suction from the deaerator storage and provided feedwater to the evaporators. The emergency condensate pump drew suction from the condensate storage and provided cooling water to the emergency feedwater pump. Deaerator level controlled any influent from the emergency condensate pump.

Initially, the major portion of the steam generated after scram was routed to the steam dump with the remainder to the emergency turbine. The steam dump flow meter indicated flows of about 11,000 to 27,000 lb/hr during the first hour period. In the second hour, flows were indicated at 9000 lb/hr.

As the test continued, the quantities of steam decreased until the in-valve to the steam dump system began to close as the result of a drop in steam pressure. Evaporator levels had been maintained within operating limits. After about 2-1/2 hr, the steam dump valve closed. Steam pressure had dropped to 770 psig from a pre-scram level of 838 psig. All steam generated was being delivered to the emergency turbine. It was estimated from previous test data that the turbine was using about 8800 lb/hr steam at full load.

Turbine speed was $\sim 3600^{+40}_{-10}$ rpm throughout the test, with the emergency feedwater pump indicating a discharge pressure of 1000 to 1020 psig.

At the end of the test, the steam generators were feeding steam only to the emergency turbine. When steam flow was within the capacity of the electric motor-driven emergency feedwater pump (12 gpm), the test was completed. This pump had been started many times previously and had been utilized to capacity on several occasions. The emergency turbine was tripped by closing the throttle valve, and satisfactorily shutdown. The feedwater system to the evaporators was returned to normal conditions by placing boiler feed pump B in service.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

A. CAPABILITY TEST

The emergency steam system adequately provided feedwater to the evaporators at a suitable pressure in the flow range of 64,000 through 112,000 lb/hr indicated (estimated at 126,900 lb/hr when corrected for orifice factor). Design rating of the emergency feedwater pump is 250 gpm (124,950 lb/hr).

Figure 640-1 shows the pressure-flow and speed-flow characteristics of the pump. Note that the design head curve is based on 3700 rpm, although pump design specification stipulates 3600 rpm. The two curves nevertheless are similar, and suitable pressure in the operational range is provided by the emergency feed pump. Speed-flow characteristics, as shown in Figure 640-1, indicate satisfactory governor control at near 3600 rpm.

Pressure from the emergency feedwater pump is considerably higher than that from conventional feedwater pump B, as read on the same recorder. Emergency feed pump pressure could be lowered, if desired, by adjusting the governor to a lower rpm setting.

Evaporator level was maintained within operational limits throughout the capability test, although the level was higher than normal (with conventional feedwater pump service).

The gland seal steam exhaust piping is not large enough to accommodate the flow. The 3/4-in. pipe which exhausts to atmosphere outside the walls of the plant should be increased to 1-1/2 in. size.

The emergency feedwater system provided steady service conditions in the power range of 20 Mwt to 38 Mwt. There were some discrepancies in low-scale

Figure 640-1. Emergency Feedwater Pump Performance Curve

meter readings as well as load levels. Corrected feedwater flow, steam flow, and steam dump flow readings did not agree. It was felt that the corrected feedwater values gave the more accurate flow readings, hence these flow values were used in plotting the curve. At higher scale readings, the three flows were in closer agreement.

Emergency feedwater pump pressure, as read at the pump effluent, was reading 20 lb low according to pressure gage calibration. The feedwater header pressure reading on the main board of the control room was reading 30 to 50 lb higher than the (corrected) gage reading at the pump.

Even though these discrepancies appeared in the test, the emergency feedwater system fulfilled the requirements of the capability portion of the test.

B. OPERABILITY TEST

The operability portion of the emergency system was satisfactorily tested in conjunction with a scram from 38 Mwt level.

Initiating simulated loss of reserve station load, which supplies the ultimate housepower service, triggered the emergency feedwater system time delay start relay. The function of the time delay is to permit immediate breaker reclosure when possible, in order to restore housepower. After 35 sec, the steam in-valve to the emergency turbine opened; in addition, 3 sec were required for the valve to reach a full open position.

In the 38 sec period, the feedwater pressure as indicated in the control room decreased and then increased. Evaporator level dropped about an inch and then increased. Significant data obtained during the feedwater pump changeover are tabulated in Table III.

TABLE III

FEEDWATER PUMP CHANGEOVER DATA

Time	Feed Pump Service	Feedwater Hdr. Press (psi)	Evaporator Level (Inches) Note: 0.0 = Control Level			Steam Press (psi)
			1	2	3	
Pre scram	B	900	+0.1	+0.2	+0.3	838
Scram + 1 sec	None	900	+0.1	+0.2	+0.3	838
Scram + ~35 sec	None	850(min)	-1.1(min)	-0.8(min)	-0.6(min)	835
Scram + ~1 min	Emergency	1100(max)	+1.0(max)	+1.1(max)	+1.2(max)	832
Scram + 5 min	Emergency	1080	+0.6	+0.8	+0.9	817(min)

Following the scram, the evaporator levels (with some manual setpoint adjustment) remained in the range of -1.0 in. to + 1.0 in. of the zero control level of 0 in. (This information was taken from the strip charts.)

If the reactor had been at a higher power level, the evaporator levels could be expected to drop following the scram and loss of feed pump. The pressure drop should aid in maintaining the evaporator level by closing off steam dump valves and relief valves.

As the test progressed for a 2-1/2 hr period, the emergency turbine operated at 3600 rpm, 1020 - 1040 psi feed pump discharge pressure, and 140 psi condensate pump discharge pressure.

Refer to Figure 640-2 for the emergency feedwater system characteristics under decreasing steam generator steaming conditions. Two hours and 18 min after scram, the steam dump valve had closed and, with the steam dump valve closed, all steam generated was being fed to the emergency turbine.

The test satisfactorily demonstrated the operability of the emergency feedwater system.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with post critical test AI-PC-640, Emergency Feedwater System, as listed in NAA-SR-5700 (Supplement 4) pages XI(c)-28 and XI(c)-29 with remarks regarding the test to explain compliance with the criteria.

- a) Criterion: The turbine driven emergency boiler feed pump supplies water to the steam generators to sustain a power level up to approximately 38 Mwt. This is adequate to remove afterglow heat immediately after shutdown from extended full power operation. The back-up motor driven emergency boiler feed pump supplies approximately 12 gpm of feedwater to the steam generator.

Remarks: The turbine driven emergency feed pump supplies feedwater exceeding the rating of the feedpump. This flow sustains a reactor power level of about 45 Mwt. The back-up motor-driven emergency boiler feed pump supplies 12 gpm of feedwater to the steam generator.

Figure 640-2. Emergency Feedwater Characteristics After Scram

- b) Criterion: The turbine driven emergency feedwater pump automatically starts following a loss of housepower.

Remarks: A loss of housepower initiates a 35 sec time delay, whereupon the emergency turbine steam-in valves open, starting the emergency feedwater system.

- c) Criterion: The standby emergency feedwater pump can be started when the turbine driven pump has an insufficient steam supply to keep it in operation. Also, the standby pump develops adequate pump discharge pressure.

Remarks: The standby emergency feedwater pump can be started at any time desired. The standby pump develops adequate pump discharge pressure and flow.

- d) Criterion: The feedwater pump turbine can be manually tripped by closing the throttle valve.

Remarks: Closing the throttle valve to the turbine results in an oil pressure trip of the turbine valve.

V. EVALUATION

The emergency feedwater system provides feedwater to the nuclear facility steam generators in quantities sufficient to remove afterglow heat when the normal feedwater pumps are inoperative for any reason.

Emergency feedwater system post critical test required that the turbine-driven emergency feedwater pump automatically start following a loss of housepower and that it supply sufficient water to the steam generators to sustain a power level of 38 Mwt. This power level corresponds to the afterglow heat after extended full power operation. Other requirements are, that the standby emergency feedwater pump can be started and maintain 12 gpm when the turbine-driven pump has insufficient steam to remain in operation, and that the feedwater pump turbine can be manually tripped by closing the throttle valve.

During initial checkout of the system in preparation for the test, a large quantity of mud, sludge, and corrosion products was found in the steam lines leading to and from the feedwater pump turbine. It should be pointed out that the system had been unused for approximately one year prior to this test, and,

since the turbine location represents a low point in the steam lines, it is reasonable to expect some particulate settlement in this area. For this reason, it is recommended that the system be flushed monthly.

After the initial cleanup of the system, the test was conducted satisfactorily and without incident. After loss of housepower, the transfer to the emergency feedwater pump was accomplished smoothly and without any abnormal system changes. Pick-up of the standby emergency feedwater pump (12 gpm), when the steam supply became inadequate to drive the feedwater pump turbine, was not attempted due to the fact that this pump had been operated many times in the past without difficulty. All other components performed as per design criteria, with operation adequate for the assigned task.

HALLAM NUCLEAR POWER FACILITY

Post Critical Test Procedure

AI-PC-660

Scram and Loss of Load

Test Procedure Submitted for Approval

6/30/61
Date

James E. Paine
Atoms International

Test Procedure Reviewed

1/3/63
Date

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Test Approved for Performance

1/3/63
Date

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Test Completed

6/1/63
Date

James E. Paine
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Test Approved for Performance

1/3/63
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Test Completed

6/1/63
Date

R. C. Buntain
Atomic Energy
Commission

HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
SCRAM AND LOSS-OF-LOAD

AI-PC -660

Test and Analysis
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I. O. SUNDERMAN
(Consumers Public Power District)

Evaluation
by
R. W. TRUITT

A T O M I C S I N T E R N A T I O N A L
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P. O. Box 309 Canoga Park, California

Contract: AT-(11-1)-GEN-8

February 21, 1964

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HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
SCRAM AND LOSS-OF-LOAD, AI-PC-660

I. PURPOSE OF TEST

The purpose of this test was to verify that the Hallam Nuclear Power Facility (HNPF) will function properly and safely during the transient conditions of scram or loss-of-load.

II. DISCUSSION OF TEST PERFORMANCE

Scram and loss-of-load testing was coordinated with Phase II and Phase III testing activities conducted per AI-PC-830, Master Schedule, Including Rise-to-Power. The first five of the scram tests were conducted from reactor power levels ranging from 2.5 to 38 Mwt (Phase II). During Phase III testing, a loss-of-load test was conducted at 25 Mwe, and a scram test was performed at 40 Mwe.

The scram tests were scheduled at ascending reactor power levels and reactor differential temperatures (ΔT 's). As certain temperature, flow, flux, valve position, and power level values were expected to change quickly during the transient conditions, signals from the corresponding sensing elements were connected to high speed Sanborn recorders. Data thus obtained provided a basis for analysis in preparation for the next test. Prior to each scram test, steady-state plant operating data were recorded for comparison with post-scram data. All scram actions and related events were recorded immediately following each scram. In addition, plant operating data were recorded over periods ranging from 2 to 16 hr after the scram. Post-scram data were subsequently analyzed to ascertain the magnitude of temperature transients and to evaluate the adequacy of plant controls during the post-scram period. Transients throughout the system were controlled by manual manipulations when necessary, in addition to the action of the plant control system. The observed data are, therefore, the result of both manual and automatic control action. In all cases the tests were conducted with the plant control system functioning in a normal manner; i. e., maintaining the plant in a steady-state condition.

Steam was released to the steam dump following all of the scrams performed during the Phase II testing program. The steam dump system condenses the steam in a pair of atmospheric boilers to conserve feedwater during startup, testing, and post-scram conditions. During the Phase III testing period, steam was valved to the turbine-generator because of the higher loads. However, post-scram afterglow heat was dissipated in the steam dump system, the normal mode.

The plant protective system was monitored during all scram and loss-of-load tests. Records were made of all the alarm windows that were lighted prior to and immediately after the scram. Length of time of alarm conditions was also noted. The following scram actions were noted during the performance of each test:

- a) All 19 control rods dropped
- b) The scram plaque annunciator lighted
- c) The proper alarm window lighted red
- d) The convection flow controller activated
- e) Main pump magnetic couplers deenergized
- f) Control rod latches drove down
- g) CONTROL ROD SNUBBERS DOWN plaque lit
- h) INTERLOCKS NOT SATISFIED plaque lit
- i) REACTOR ON plaque off
- j) HCS-1 (Neutron Flux Control) transferred to manual

Six scrams and one loss-of-load were performed as part of this test. Table I shows the power levels and pre-test steady-state conditions for each of the tests. The scram tests were all initiated from a steady-state plant condition by depressing the "Manual Scram" button. The loss-of-load test, which simulated a loss of electric load, was initiated by depressing the "Manual Turbine Trip" button.

TABLE I
SUMMARY OF SCRAM TESTS

Power Level (Mwt)	Reactor ΔT^* (F°)	Indicated Primary Sodium Flow (%)	Top of Active Core-T/C C4-6 (°F)	Steam To:
2.3	2	36	530	Steam Dump
10.3	30	27	565	Steam Dump
47.3	235	20	810	Steam Dump
38.0	222	15	792	Steam Dump
38.0	325	10	940	Steam Dump
40 (Mwe)	321	50	938	Turbine
25 (Mwe) Loss-of-Load	338	29	940	Turbine

*Reactor ΔT here is defined as difference between top of active core temperature as indicated by T/C C4-6 and bottom of active core temperature as indicated by T/C C4-13.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

A. SCRAM TEST AT 2.5 Mwt

This test was considered the first scram from a significant power level. With the low ΔT (2 F°), no significant temperature transients were expected; only the nearly isothermal cooldown of the system. Thus, this was primarily a system scram checkout test or "dress rehearsal" for future higher power scrams. All normal scram actions were observed closely, especially the automatic action of the sodium throttle valves, which are controlled by the convection flow control subsystem.

Table II presents changes in some of the system variables 15 minutes after the scram. This low power test was inconclusive with respect to verifying the

TABLE II
SYSTEM CHANGES FOLLOWING SCRAM – 2.5 AND 10 Mwt SCRAM TESTS

Variable	2.5 Mwt Test Conditions		10 Mwt Test Conditions	
	Prescram	Scram + 15 min	Prescram	Scram + 27 min
Pri Na temps to IHX	520 to 540°F	no change	560 to 570°F	no change
Pri Na temps from IHX	524 to 527°F	-4F°	530 to 534°F	+4F°
Mod coolant out temp	490°F	+5F°	560°F	+6F°
Sec Na temps to IHX	520 to 530°F	no change	530 to 522°F	no change
Sec Na temps from IHX	530°F	no change	562 to 568°F	+1F°
Na temps to superhtr	530 to 550°F	-2F°	560 to 572°F	-2F°
Na temp to evap	540°F	-2F°	562°F	no change
Na temps from evap	502 to 520°F	no change	529 to 535°F	+8F°
Pri TV positions	open	95% closed	open	95% closed
Sec TV positions	open	92% closed	open	95% closed
Feedwater levels	0.5 to 1.1 in.	0 to +0.4 in.	-0.7 to +1.3 in.	+0.3 in.

ability of the plant control system to maintain all temperature transients within $50\text{F}^\circ/\text{hr}$. However, the test did show that all scram actions occurred per design and the plant could be expected to perform favorably after a higher power scram.

B. SCRAM TEST AT 10 Mwt

This test was also conducted with a relatively small ΔT (30F°). All scram actions occurred per design. The average of the fuel channel outlet temperatures dropped about 8F° immediately; thereafter it dropped at a rate of about $5\text{F}^\circ/\text{hr}$.

Upper plenum and bottom core temperatures did not change much, but core center temperature dropped 9F° in 20 minutes. This was a preview of things to come, for in later tests the center core temperatures changed drastically after a scram. Table II presents changes in some of the system variables 27 minutes after the scram. Once again the low power, low ΔT conditions did not result in a good test of the plant. However, post-scram data showed that the plant could be expected to perform favorably after higher power scrams.

C. SCRAM TEST NO. 1 AT 38 Mwt

Three scram tests were performed from the 38 Mwt power level. The initial scram test of this series, with an intermediate ΔT of 235F° and primary sodium flow of 20%, was performed on February 23, 1963. The power level at this time was calculated to be ~ 38 Mwt; however, later calculations showed that the reactor was actually operating at ~ 47 Mwt at the time of scram. This discrepancy was due to the lack of a good sodium flowmeter calibration, and to a condensate leak in the steam dump system. Thus, the steam flow, measured by observing the change in level of a condensate storage tank, was in error, resulting in the calculation of a low power level.

Main heat transfer circuit No. 1 was operating with only one intermediate heat exchanger (IHX) unit during this testing period. IHX-1A had been removed for repair of a leak and modification, leaving only unit 1B in the circuit. Absence of the IHX unit was evident but not detrimental to plant operation or test results.

This scram was the first scram test to show any significant temperature transients and trends. All scram actions occurred in a normal manner and the throttle valves closed to 95 - 98% closed positions. The average fuel channel outlet temperature, as indicated on TRC-2, dropped 40F° immediately, recovered about 10 degrees, and then decreased at a rate of about 80F°/hr. The initial drop occurred during the closing of the throttle valves; the 10 degree rise indicated the presence of afterglow heat; and the subsequent relatively high (design maximum 50F°/hr) cooldown rate showed that the convection sodium flow was too high (throttle valves not closed enough).

Figure 660-1 shows sodium temperatures in the core and upper plenum after the scram, as indicated by source element (located in process channel C4) thermocouples. As can be seen, the cooldown rate was the greatest at the center of the core, while the top of the upper plenum cooled very slowly, and the bottom of the active core held nearly constant. This figure shows clearly the temperature stratification that occurs in the reactor vessel after a scram, especially in the upper sodium plenum. Note that the upper plenum, having a large volume and very little circulation due to the low convection flows after the scram, cooled at a rate of only about 6F°/hr during the first three hours after the scram. Also of note is the fact that no sudden temperature drop and subsequent recovery, as was recorded for the fuel channel outlet temperatures, can be seen. This suggests that only fuel channels experience the initial temperature transient during the closing of the throttle valves.

Table III presents changes in some of the system variables 30 minutes after the scram. For comparison purposes the same variable changes are given for the other two 38 Mwt scram tests. The table shows that temperature changes throughout the sodium and steam systems were maintained at less than 50F°/hr.

The excessive rates of cooldown in the reactor vessel observed in this test, indicated the necessity for at least one more intermediate ΔT scram test prior to a full ΔT scram. Since the excessive core cooldown rates resulted from too much sodium flow, it was planned to manually control the throttle valves as required to maintain the cooldown rates at less than 50F°/hr. In addition, the throttle valves were restroked to assure reaching a 98% closed position immediately after the scram. This is per design; however, this test showed that some of the valves only closed to a 95% closed position.

TABLE III
SYSTEM CHANGES FOLLOWING SCRAM - 38 Mwt SCRAM TESTS

Variable	Test No. 1		Test No. 2		Test No. 3	
	Prescram	S + 30 min	Prescram	S + 45 min	Prescram	S + 30 min
IHX Pri Na inlet (°F)	800 to 809	-10*	780 to 790	-20	935	-10
IHX Pri Na outlet (°F)	565 to 571	- 2	561 to 569	+35	612	- 5
IHX Sec Na inlet (°F)	535 to 545	+45	535 to 542	+32	560	+45
IHX Sec Na outlet (°F)	788 to 793	-15	768 to 778	0	930	- 8
Mod coolant outlet (°F)	800	-30	778	-38	Not Recorded	
Na to superheater (°F)	781 to 790	- 6	762 to 780	-10	925	-10
Na to evaporator (°F)	749 to 760	+15	733 to 750	+15	865	+15
Na from evaporator (°F)	552 to 560	+35	555 to 560	+22	575	+43
Pri TV position (% closed)	open	98 to 95	open	90	open	100 [†]
Sec TV position (% closed)	open	96	open	90, 92, 96	open	99
Feedwater level (in.)	+0.1	+0.3	+0.2	-0.1 to +0.8	0.0	-0.2 to +1.2
Steam temp (°F)	775	-22	757 to 768	-35	824	0
Superheater outlet pres- sure (psig)	845 to 890	-15	850 to 880	0	860	-10

*Figures preceded by plus and minus signs denote net change from prescram value.

†As indicated on main board; valves were not actually closed.

D. SCRAM TEST NO. 2 AT 38 Mwt

Prior to initiation of the second 38 Mwt scram, the six sodium loop flow-meter signals were connected to Sanborn recorders. This indication of flow was used to show the effect of valve position change since the main board position indicators were not designed to show small changes. Thus, the actual change in sodium flow was observed as the throttle valves were manipulated.

The second scram from 38 Mwt was initiated with a reactor ΔT of 222F° and an indicated primary sodium flow of 15%. All scram actions occurred per design and the throttle valves closed to the 98% closed position. The average fuel channel outlet temperature (TRC-2) dropped 20 degrees immediately and continued to decrease at a rate of 65F°/hr. There was no temperature recovery after the initial drop, thus showing there was essentially no afterglow heat in the core. This was as expected, since the reactor had been raised to 38 Mwt in a minimum of time and scrammed immediately.

Figure 660-2 shows some sodium and in-fuel temperatures in the core and upper plenum after the scram. The results are similar to those observed after the 47 Mwt scram. However, the rate of sodium cooldown after the initial temperature drop was reduced from 88 to 58F°/hr at the top of the core, and 120 to 90F°/hr at the core center.

Fifteen minutes after scram the throttle valves were manually closed until the average fuel channel outlet temperature (TRC-2) was decreasing at a rate of 50F°/hr. The effect of this reduction in flow can be seen in Figure 660-2 on the in-fuel temperatures (C6-1 and C6-8) and the fuel channel outlet temperature (C10-2). However, no changes in the sodium temperatures (C4-1, C4-6, C4-9 and C4-13) can be detected.

Table III shows the changes in some of the system variables 45 minutes after the scram. Once again there was no indication of any problems due to excessive temperature transients in the sodium and steam systems.

This test showed that it is feasible and possible to manually control throttle valve position to regulate reactor cooldown rates. Nevertheless, the throttle valves were restroked prior to the next scram test to go to the 99% closed position immediately after a scram. It was concluded that with a combination of automatic and manual control of the plant systems a full ΔT scram could be conducted safely without undue strain on the systems.

E. SCRAM TEST NO. 3 AT 38 Mwt

The third scram from 38 Mwt was initiated with a reactor ΔT of 325F° and an indicated primary sodium flow of 11%. Again all scram actions occurred per design. Simultaneously with the closing of the throttle valves, the average fuel channel outlet temperature dropped 20F° and continued to decrease at a rate of 90F°/hr. As in the previous scram test, there was little afterglow heat; thus, it was again necessary to manually close the throttle valves beyond the 99% closed position to reduce the cooldown rate to an acceptable 50F°/hr rate. The throttle valves were closed cautiously and in small steps while observing both the sodium flow as indicated on the Sanborn recorders and the cooldown rate of the average fuel channel outlet temperature. After about 1-1/2 hours, and with an indicated primary sodium flow of 1.2% of full flow, the average fuel channel outlet temperature cooldown rate was reduced to 50F°/hr.

Figure 660-3 shows some sodium temperatures in the core and upper plenum after the scram. The curves are similar to those shown previously (Figures 660-1 and -2). However, the initial cooling rates at the top and center of the core are appreciably higher than in the intermediate ΔT scrams from 38 Mwt. The effect of slowly closing the throttle valves can be seen during the period of 20 to 90 minutes after the scram. The cooling rate was reduced from 100 to 38F°/hr at the top of the core, and from 204 to 20F°/hr at the center of the core.

Table III shows the changes in some of the system variables 30 minutes after the scram. These data indicate no potential problems in the sodium and steam systems.

It was concluded from the 38 Mwt scram tests that the HNPF could be expected to perform safely following a reactor scram. It was shown that it is feasible and possible to manually control the sodium system temperature transients after the initial closing of the throttle valves. In an attempt to eliminate the need for any manual flow control, it was decided to restroke the throttle valves to close to more than the 99% closed position immediately after a scram. However, it is anticipated that manual flow control will always be required to maintain acceptable temperature transients after scrams with no appreciable afterglow heat, such as the scrams reported to this point.

F. SCRAM TEST AT 40 Mwe

This test was initiated with the plant at steady-state conditions of 40 Mwe (approx 120 Mwt), reactor ΔT of 321°F , and indicated primary sodium flow of 50%. All scram actions occurred per design and the throttle valves closed to the more than 99% closed position. The average fuel channel outlet temperature immediately dropped 100°F , from 955 to 855°F , recovered 70°F in about a minute, and continued to decrease at a rate of $18^{\circ}\text{F}/\text{hr}$.

The initial temperature transient, also observed in all previous scrams but in lesser magnitude, indicated a trend toward larger transients for scrams at higher powers and consequently higher sodium flows. This is to be expected; since the throttle valves always close in about the same length of time, the higher initial sodium velocities and greater convection flow driving force will result in a longer flow decay time.

The 70°F temperature recovery and subsequent cooldown rate of only $18^{\circ}\text{F}/\text{hr}$ showed the presence of considerable afterglow heat. This test supported the conclusion drawn after the three 38 Mwt tests that reactor cooldown can be controlled when appreciable afterglow heat is present. This fact was demonstrated by using the convection flow controller to automatically reduce the main heat transfer system to isothermal conditions. After the average fuel channel outlet temperature had decreased for about an hour at the rate of $18^{\circ}\text{F}/\text{hr}$, the convection flow controller successfully programmed the average fuel channel outlet temperature down at a rate of $50^{\circ}\text{F}/\text{hr}$.

Figure 660-4 shows sodium temperatures in the core and upper plenum after the scram. Once again the curves are similar to those shown for the other scram tests. Comparing Figure 660-4 curves with those for the 38 Mwt, 325°F ΔT scram, Figure 660-3, some significant differences can be seen that are attributable to the afterglow heat. The initial cooldown rates of the sodium at the top and center of the active core are much less. The temperature of the sodium at the top of the upper plenum, however, decreases much faster than after the 38 Mwt scram. This is the result of higher convection flow causing greater circulation and mixing in the upper plenum. This greater mixing also reduces considerably the stratification in the upper plenum.

The change in slope of the curves about one hour after scram corresponds to the time the convection flow controller began to program the average fuel

channel outlet temperature down at a rate of 50°F/hr . Notice that the maximum rate of change for these sodium temperatures is greater than 50°F/hr . The sodium at the top of the upper plenum and the top of the active core cooled at a maximum rate of 64°F/hr , while at the center of the active core the maximum rate was 104°F/hr .

Table IV gives changes in some of the system variables up to two hours after the scram. As concluded previously, and shown in this table, there is no indication of any problems in the sodium and steam systems after a scram.

The 40 Mwe scram concluded the planned scram tests of the HNPF post-critical test program. The tests showed that the plant can be expected to perform safely and without undue strain on the systems or components following a scram. It was also shown that it is feasible, and possible, to manually control system temperatures if the need arises. The importance of afterglow heat at the time of scram (under normal operating conditions there will always be considerable afterglow heat in the core) was demonstrated by the slower cooling rates observed after the 40 Mwe scram. The initial fuel channel outlet temperature transient experienced after each scram undoubtedly only occurs in fueled process channels. This transient will become greater in magnitude at higher power scrams and can only be reduced by the faster reduction in sodium flow immediately after a scram. Because of the short duration of this transient, it is not believed that the fuel elements and moderator elements experience any detrimental effects.

G. LOSS-OF-LOAD TEST AT 25 Mwe

This test was initiated by manually tripping the turbine (closing turbine throttle valve) with the HNPF operating under steady-state conditions at a power level of 25 Mwe. The reactor ΔT was 338°F and the indicated primary sodium flow was 29%. Plant performance immediately after the turbine trip was quite different from that observed after a scram. This was not unexpected, since in a loss-of-load situation the demand for energy is removed, thus suddenly leaving the plant with an excess of energy which, if not released in a controlled manner, may result in potentially hazardous increasing temperature and pressure transients. In the case of a scram, the supply of energy is suddenly stopped without removing the load demand, resulting in the tendency for decreasing temperature transients as the turbine continues to try to meet the load requirements.

TABLE IV
SYSTEM CHANGES FOLLOWING 40 Mwe SCRAM TEST

Variable	Prescram	S + 2 min	S + 15 min	S + 30 min	S + 60 min	S + 120 min
IHX Pri Na inlet temp (°F)	920	918	910	890	865	852
IHX Pri Na outlet temp (°F)	588	585	556	570	593	606
IHX Sec Na inlet temp (°F)	540	540	558	575	602	602
IHX Sec Na outlet temp (°F)	900	900	908	903	875	850
Mod coolant outlet temp (°F)	910	910	910	910	900	840
Na to superheater temp (°F)	890	890	882	881	870	838
Na to evaporator temp (°F)	830	830	822	861	868	833
Na from evaporator temp (°F)	558	575	582	591	610	611
Pri TV position (% closed)	open	97	97	97	97	96
Sec TV position (% closed)	open	91	91	89	91	91
Feedwater level (in.)	0.0	+0.7	+1.7	-1.7	-1.8	-1.8
Steam temp (°F)	836	835	838	824	810	810
Superheater steam pressure (psia)	880	700	560	825	825	825

Immediately after the turbine trip the HNPF power demand hand control station (HCS-504) was manually transferred to manual control and turned down to a demand setting of 15% of full power, the steam dump capacity. This action caused the plant control system to begin power reduction in the normal, controlled manner, as for any normal change in load or demand. The output of HCS-504 is rate limited at 40%/min; however, the power decrease was limited by the 20%/min rate limit of the flow controller. Figure 660-5 shows the primary and secondary sodium flows after the loss-of-load. Notice the flow undershoot and stabilization at an indicated 10% after about 3 minutes when reactor power reached 15%.

Simultaneously with the programming down of power, steam header pressure began to increase, the steam dump valve opened, and the electromatic relief valves opened whenever the pressure reached 935 psig (valves setpoint). The steam dump and electromatic relief valves thus became the heat sinks for removal of the excess energy in the main heat transfer system.

Seven minutes after the turbine trip, when reactor power had stabilized at 15%, the turbine throttle valve was opened and the HNPF picked up the turbine load. The load had dropped to about 1 Mwe during the seven minute period. Figure 660-6 shows the neutron flux transient and the corresponding reactor periods during the time from loss-of-load to flux (power) stabilization.

The plant control system performed in an excellent manner throughout the test. Almost no operator action was required and very little variation in system temperatures and other variables was observed, as shown in Table V. Further evidence can be seen in Figure 660-7, which shows some core and upper plenum sodium temperatures as indicated by source element thermocouples.

Similar results can be expected for a loss-of-load from higher powers, since system performance is primarily governed by the plant control system. The possible exception is that one or more of the steam generator safety valves might open to supplement the steam dump and electromatic relief valves.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with post critical test AI-PC-660, Scram and Loss-of-Load, as listed in NAA-SR-5700 (Supplement 4), page XI(c)-33, with remarks regarding the test to explain the criteria.

TABLE V
SYSTEM CHANGES FOLLOWING 25 Mwe LOSS-OF-LOAD TEST

	Pretest	T + 1 min	T + 4 min
IHX Pri Na inlet temp (°F)	910	910	910
IHX Pri Na outlet temp (°F)	587	585	572
IHX Sec Na inlet temp (°F)	540	540	545
IHX Sec Na outlet temp (°F)	895	895	896
Mod coolant temp (°F)	900	900	900
Na to superheater temp (°F)	883	882	882
Na to evaporator temp (°F)	833	825	835
Na from evaporator temp (°F)	555	557	560
Pri TV position	open	open	open
Sec TV position	open	open	open
Feedwater level (in.)	+0.1	0.0	-0.2
Steam temp (°F)	838	838	838
Superheater outlet pressure (psig)	880	880	900

Criteria: The HNPF can be operated at full power with adequate equipment protection during the transient conditions of reactor scram or loss of load (turbine trip) as demonstrated by:

- a) The convection flow system operates automatically to maintain core temperature gradient following scram.

Remarks: The convection flow system demonstrated that it can operate automatically to maintain core temperature gradient following a scram.

- b) No excessive temperature transients (>50°F in one hour) occur in the Sodium Heat Transfer System or in the steam generators.

Remarks: No excessive temperature transients (>50°F° in one hour) occurred in the sodium heat transfer system or in the steam generators; however, temperature of the secondary sodium to the IHX increased 60°F°/hr on the 40 Mwe scram. This was not considered to be an excessive transient.

- c) Temperature transients in the fuel, in the moderator elements, of in the fuel channel exit sodium are not high enough to result in material damage.

Remarks: Test results indicate that the temperature transients in the fuel, in the moderator elements, or in the fuel channel exit sodium are not high enough to result in material damage.

- d) The HNPF has been scrammed from full power and the turbine has been tripped from full power but excessive thermal transients do not occur.

Remarks: The HNPF was scrammed from 40 Mwe and a loss-of-load test was initiated at 25 Mwe with full ΔT conditions. No excessive thermal transients were observed. Extrapolation of the results of these two tests indicates that the plant will perform safely after a scram or loss-of-load from full power.

V. EVALUATION

When the first low power, full ΔT scram was performed, it was found that the rate of decay of reactor outlet temperature could not be maintained at a rate of 50F°/hr or less. Under low power operating conditions very little afterglow heat is generated. Analysis of the data obtained from the scram test indicated that this problem could be alleviated by restroking the throttle valves. Later scrams verified that this problem was solved and a temperature decay rate of 50F°/hr or less could be maintained after scram from any power level.

Difficulties were also encountered with the throttle valve operating mechanism. There were several instances of the mechanism jamming following a scram. A temporary repair was made thus allowing safe operation until a new mechanism can be designed. A new mechanism is being installed to permanently correct the condition.

The data obtained from the scram tests has been carefully reviewed to determine the effect on system components. With the exception of the large temperature undershoot of short duration immediately following a scram, no problem areas have been uncovered. Installation of the new valve operating mechanisms will materially improve this condition as the valves will go closed 1-1/2 times faster.

The progressive-step method used in the scram test program was very useful in that it pointed out the areas where actual and potential problems existed.

Steps were then taken to correct these conditions before high power levels and large temperature differentials were encountered. The scram tests were also helpful from an operator training standpoint.

A loss-of-turbine-load test was performed from a power level of 25 Mwe. The response of the system to this transient was excellent. Although no such test was made from full power, a similar result can be expected.

Figure 660-1. Reactor Core and Upper Plenum Sodium Temperatures - 38 Mwt Scram Test No. 1

Figure 660-2. Reactor Sodium and In-fuel Temperatures - 38 Mwt Scram Test No. 2

Figure 660-3. Reactor Core and Upper Plenum Sodium Temperatures - 38 Mwt Scram Test No. 3

Figure 660-4. Reactor Core and Upper Plenum Sodium Temperatures - 40 Mwe Scram Test

Figure 660-5. Sodium Flow and Temperatures - 25 Mwe Loss-of-Load Test

CHANNELS III & IV LOG N FLUX RECORDER (amps)

CHANNELS III & IV PERIOD RECORDER (sec)

Figure 660-6. Period and Flux - 25 Mwe Loss-of-Load Test

Figure 660-7. Reactor Core and Upper Plenum Sodium Temperatures - 25 Mwe Loss-of-Load Test

HALLAM NUCLEAR POWER FACILITY

Post Critical Test Procedure

AI-PC-690

Radiation Shielding

Test Procedure Submitted for Approval

6/30/61 James E. Chens
Date Atomic Energy
Commission

Test Procedure Reviewed

12/14/62 DE Simpson
Date Atomic Energy
Commission

Test Approved for Performance

12/28/62 James E. Chens
Date Atomic Energy
Commission

Test Completed

7/20/63 James E. Chens
Date Atomic Energy
Commission

Test Approved for Performance

12/28/62 RC Buntain
Date Atomic Energy
Commission

Test Completed

7/20/63 RC Buntain
Date Atomic Energy
Commission

HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
RADIATION SHIELDING

AI-PC-690

Test and Analysis
by
R. L. ROBB
(Consumers Public Power District)

Evaluation
by
S. BERGER

A T O M I C S I N T E R N A T I O N A L
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P.O. Box 309 Canoga Park, California

Contract: AT(11-1)-GEN-8

January 30, 1964

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HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
RADIATION SHIELDING, AI-PC-690

I. PURPOSE OF TEST

The primary purpose of post critical test AI-PC -690 was to determine the adequacy of the radiation shielding used to shield the reactor and all process systems used at the Hallam Nuclear Power Facility (HNPF). In addition, the test was used to determine how reactor operation effected area radiation levels.

II. DISCUSSION OF TEST PERFORMANCE

Radiation shielding tests were conducted in conjunction with HNPF rise-to-power testing. In general, the test method consisted of measuring gamma and neutron levels above the reactor top shield and over primary coolant loops during operation of the reactor at various power levels and at full power operation. Detailed gamma and neutron surveys were conducted following each increase in reactor power level and the results recorded in the radiation survey log. Two phases of data collection, described in the following paragraphs, were incorporated in the collection of testing data.

A. RADIATION DETECTION AND MONITORING SYSTEM DATA

Readings were recorded from the radiation detection and monitoring (RD&M) system and plant constant air monitors (CAM) each time the reactor power was increased. Data taking was started when the power level was changed. Data taking continued at approximately eight hour intervals until equilibrium sodium activity was reached, or for the duration of operation at that power level if operation was shorter than the time required to reach equilibrium. Since some of the data to be collected was not recorded by the RD&M equipment, all readings were taken from instrument indicators to get a more consistent set of data. Readings for this phase of data taking were obtained at the following approximate reactor power levels:

2.5 Mwt	50 Mwt	150 Mwt
10 Mwt	80 Mwt	195 Mwt
20 Mwt	125 Mwt	100% (~244 Mwt)
38 Mwt		

B. PLANT SURVEY DATA

Detailed plant surveys (gamma and neutron) were taken at five different reactor power levels to cover the range of 1% to 100% power. These five surveys were in addition to routine surveys conducted by Health and Safety personnel. After the proper test power level was reached, a waiting period of two to four days was allowed before surveys were begun. The waiting period was necessary to allow the primary system sodium to closely approach equilibrium activity.

Data for the surveys were measured with the assistance of Health and Safety personnel and were recorded on data sheets used by Health and Safety for this purpose. The data sheets are, for the most part, portions of reactor building drawings and provide a convenient means of recording survey locations and values of survey readings. The completed data sheets have been placed in the Health and Safety survey files as part of plant radiation level information. The instruments used in taking the plant surveys are listed in Table I.

Survey readings for this phase of the test were taken at the following approximate reactor power levels:

2.5 Mwt	125 Mwt
10 Mwt	100% (~244 Mwt)
35 Mwt	

TABLE I
HNPf PLANT SURVEY INSTRUMENTS* -
RADIATION SHIELDING TESTING

Instrument Manufacturer	Model Number and Type	Number of Ranges	Range
Nuclear Chicago	2612 GM meter	3	0 to 0.2 mr/hr 0 to 2.0 mr/hr 0 to 20 mr/hr
Eberline Instrument Corp.	E 500-A GM meter	4	0 to 0.2 mr/hr 0 to 2.0 mr/hr 0 to 20 mr/hr 0 to 200 mr/hr
Technical Associates	SRJ-7 (Standard range Juno) Ionization chamber	3	0 to 50 mr/hr 0 to 500 mr/hr 0 to 5000 mr/hr
	HRJ-7 (High range Juno)	3	0 to 250 mr/hr 0 to 2500 mr/hr 0 to 25,000 mr/hr
RCL Fast-Slow Neutron	Survey meter	1	0 to 1000 cps Meter calibration: Slow neutron - reading X.06 mrem/hr Fast neutron - reading X1.50 mrem/hr

*Further description of the above instruments may be found in HNPf Operations Manual, Volume Ia (NAA-SR-6500).

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

A. RADIATION DETECTION AND MONITORING SYSTEM (RD&M) AND CONTINUOUS AIRBORNE PARTICULATE MONITOR (CAM) DATA

As stated in Part II of this report, these data were taken at ten different reactor power levels ranging from 2.5 Mwt to 244 Mwt (100%). Table II gives a summary of representative readings for each of the test power levels. Where possible, these data were taken at eight hour intervals until sufficient time had passed for the primary system sodium to reach equilibrium activity. In some cases, however, short durations of test run at the particular power level made this impossible. Very little error is present however since few of these readings display any prominent change with reactor power.

Most of the detectors in the RD&M system and in all of the CAM units should be relatively independent of reactor power if shielding and containment of radioactivity about the plant is adequate. This independence is shown in the data. RD-1, the reactor atmosphere monitor, should show some relation to reactor power but should display some maximum which may be considered normal full power level. The carbon and cold trap cell monitors (RD-1400-2, 1400-3, and 1400-4) will be dependent on whether the traps are in service. The stack and prestack monitors will be somewhat dependent upon whether there is any release being made from the radioactive gas decay tanks. All other detectors covered in this group of data should show low and relatively stable readings at any reactor power when plant conditions are normal.

Two irregularities are shown in the RD&M data:

- 1) During the 50 and 80 Mwt tests, RD-1 was reading about a decade higher than at previous levels. The high readings of RD-1 were found to be due to carryover of radioactive sodium into the reactor vent system and hence into the detector filter and piping. During the 125 Mwt test, a helium leak developed in a control rod thimble thereby dumping extra helium into the reactor. The increased venting of the reactor resulting from the thimble leak caused accelerated sodium carryover and RD-1 readings increased another decade. The leaking thimble was subsequently replaced. During the last three test runs, sodium carryover in the reactor vent system was slight. It can be assumed that the vent system sodium carryover problem will

TABLE II
AREA RADIATION LEVELS
(Data from RD&M System
and CAM units)

Channel	Units	Reactor Power Level (Mwt)										Location
		2.5	10	20	38	50	80	125	150	195	244 (100%)	
RD-1	CPM	9×10^3	1×10^4	1.2×10^4	1.1×10^4	$1 \times 10^5^{**}$	$1.2 \times 10^5^{+}$	$2.5 \times 10^6^{***}$	2×10^3	2×10^4	6×10^4	Reactor atmosphere
RD-100	CPM	15	15	15	15	18	15	15	15	15	20	Secondary sodium loop No. 1
RD-200	CPM	15	20	15	20	47	35	25	10	20	55	Secondary sodium loop No. 2
RD-300	CPM	15	25	20	25	12	10	15	10	15	10	Secondary sodium loop No. 3
RD-900	uc/cc	2.5×10^{-5}	1.3×10^{-5}	1.2×10^{-5}	1.3×10^{-5}	1.6×10^{-5}	1.6×10^{-5}	2.5×10^{-4}	1.7×10^{-5}	1.8×10^{-5}	2×10^{-5}	N ₂ vents
RD-1000	uc/cc	1.5×10^{-5}	2.7×10^{-5}	2.6×10^{-5}	2.6×10^{-5}	3.1×10^{-5}	3.3×10^{-5}	1.2×10^{-4}	3.2×10^{-5}	3.5×10^{-5}	4×10^{-5}	Non-R/A vents
RD-1001	CPM	300	300	300	340	390	340	340	300	350	300	Reactor building stack
RD-1002	uc/cc	1×10^{-4}	1.1×10^{-4}	8×10^{-5}	1.1×10^{-4}	6×10^{-5}	6×10^{-5}	6.1×10^{-5}	8.5×10^{-5}	8.5×10^{-5}	1×10^{-4}	Prestack line
RD-1400-1	mr/hr	0.02	0.018	0.015	0.013	0.02	0.02	0.018	0.023	0.028	0.028	Control room
RD-1400-2	r/hr	*	500	0.001	<0.001	0.185	3.5	50	0.002	0.0025	0.0033	Carbon trap cell
RD-1400-3	r/hr	*	*	0.001	0.001	0.026	0.006	0.032	0.0027	0.0028	0.0040	Primary cold trap cell No. 1
RD-1400-4	r/hr	*	500	3.2	91	0.195	1.5	800	4.0	1.2	>1000	Primary cold trap cell No. 2
RD-1400-5	mr/hr	0.025	Not on scale	Not on scale	*	0.015	0.015	0.016	0.015	0.014	0.015	Maintenance cell operating area
RD-1400-6	mr/hr	0.030	0.03	0.022	0.048	0.015	0.015	0.015	0.016	0.017	0.017	Equipment decontamination room
RD-1400-7	mr/hr	0.030	0.45	0.27	0.048	0.016	0.017	0.014	0.010	0.013	0.021	Control rod carriage
RD-1400-8	mr/hr	0.160	0.1	0.049	0.085	0.060	0.060	0.130	0.065	0.070	0.070	Local boards 29 and 30
RD-1401-1	CPM	10	15	20	25	30	40	65	35	60	105+	IHX cell No. 1 atmosphere
RD-1401-2	CPM	10	10	15	20	25	25	55	25	50	90+	IHX cell No. 2 atmosphere
RD-1401-3	CPM	10	10	10	15	23	25	75	35	55	130+	IHX cell No. 3 atmosphere
RD-1401-4	CPM	15	15	20	35	20	20	65	15	20	100+	Cold and carbon trap cells atmosphere
RD-1401-5	CPM	10	10	10	10	10	10	25	5	15	10	Primary fill tank cell atmosphere
RD-1401-6	CPM	15	20	20	20				15	30		Primary pipe tunnel atmosphere (east side)
RD-1401-7	CPM	10	15	10	15	15	10	15	10	20	15	Primary pipe tunnel atmosphere (west side)
RD-1402	CPM	250	300	280	210	250	420	400	325	500	350	Constant air monitor (CAM), H ₁ Bay
RD-1403	CPM	60	500	95	200	100	150	110	120	400	65	CAM, over IHX cells
RD-1404	CPM	*		*	*	200	380	200	225	400	75	CAM, auxiliary bay
RD-1405	CPM		*	*	*	*	*	*	*	*	*	CAM, maintenance cell operator area
RD-1406	CPM	*	*			350	650	80			*	CAM, decontamination room
RD-1407	CPM	130	250	33	180	40	15	40	75	200	24	CAM, reactor building stack

Not in service, being repaired.
 *Detector seeing sodium carryover in reactor vent system.
 ***Sodium carryover increased due to increase of reactor venting caused by control rod thimble helium leak.
 +Detector receiving beam from sodium source inside the cell.

reoccur unless adequate corrective action is accomplished. A close check will be maintained to detect the onset of any further sodium carryover.

- 2) Elevated readings were recorded during later testing runs for RD-1401-1, -2, and -3 (IHX cell atmosphere monitors) and RD-1401-4 (cold trap and carbon trap cell atmosphere monitor). Information obtained in the detailed plant surveys explain these elevated readings. The detectors are located on nitrogen cooling system ducts exterior to the cells being cooled and will normally see only low levels. However, due to the location, these detectors are receiving gamma streaming from sodium sources within the cells being cooled. It may be desirable to surround the detectors with shielding so that the cell atmosphere readings will not be masked by the beam and so that detector alarm and control action setpoints will not require changes with reactor power level.

B. DETAILED PLANT SURVEYS

As stated in Part II of this report, surveys were made at five different reactor power levels ranging from 2.5 Mwt to 244 Mwt (100%). The survey data were taken after a two to four day waiting period had been allowed to permit the primary system sodium to approach equilibrium activity. The percent of equilibrium activity in the sodium, based on estimates (see Appendix A) at the start and finish of each survey, are presented in Table III.

TABLE III
ESTIMATED SODIUM EQUILIBRIUM ACTIVITY

Reactor Power (Mwt)	Percent Equilibrium at Start	Percent Equilibrium at Finish
2.5	90.0	98.0
10	98.7	99.6
35	95.8	98.9
125	89.7	97.0
244 (100% power)	90.2	97.0

The error introduced by starting the surveys at $\sim 90\%$ of equilibrium, in some cases, is felt to be negligible. Spot checks of initial readings were made at the end of some of the surveys and no appreciable differences were noted. Another point to consider in this regard is that the accuracy of instruments used is, in general, in the order of 10% so that the magnitude of difference may well be less than instrument accuracy.

From the first few test runs the surveys have indicated that, under normal plant operating conditions, there existed five basic areas in which radiation levels approached or exceeded "maximum permissible level" (MPL). Data taken in these areas are tabulated for each test power level in Table IV. After the 35 Mwt survey, shielding was added to a pipe chase in the east primary pipe tunnel. This addition of shielding eliminated Area No. 1 as an area of concern and reduced the levels in Area No. 2 by a factor of approximately 100. The areas described under Nos. 2, 3, 4, and 5 remain without further shielding being added but have been properly posted and identified (Area No. 3 has three physical locations in the plant, i.e., in each steam generator room. Area No. 4 has three physical locations in the plant, i.e., by each IHX cell cooler). Posting in these areas has been located so the radiation level at the signs will be well below MPL and entry beyond the signs will be controlled by special work permits (SWP) issued by Health and Safety. Area No. 2 is a limited access area and is posted at the locked entry door where levels are background. Area 3 is posted at a location beyond (in a lower radiation level region) the positions shown as 3a in Table IV, and Area 4 is posted at a location beyond the positions shown as 4b in Table IV. Area 5 is an area which is so filled with ductwork and piping that entry is fairly difficult. Area 5 is posted at a point considerably away from where the readings were obtained; radiation level at the posting point is nearly background at full power. Spots in the plant where radiation levels are above MPL, such as Areas No. 2, 3b, 4a, and 5 of Table IV, would not normally be areas in which work is done. Work could be performed in these areas during an emergency by limiting occupancy times and perhaps also by reducing reactor power. It may be desirable to add additional shielding in certain of these areas at a later date.

The areas numbered 6 and 7 on Table IV were included in the areas approaching or exceeding MPL following occurrence of the control rod thimble leak which produced increased sodium carryover in the reactor vent system and thereby

TABLE IV
DETAILED PLANT RADIATION SUR-
VEY (AREAS WHERE LEVELSHAVE
APPROACHED OR EXCEEDED MPL)
Dose Rates Are for Gamma Radiation

Area Number	Reactor Power Level					Area Description
	2.5 Mwt	10 Mwt	35 Mwt	125 Mwt	100% (244 Mwt)	
1a	0.14 mr/hr	0.65 mr/hr	1.8 mr/hr	+0.015 mr/hr	0.015 mr/hr	North shielding wall of operating area adjacent to valve and gage bd No 30, against north wall and approximately 3 ft west of east wall.
1b	*	0.2 mr/hr	0.65 mr/hr	+0.015 mr/hr	0.015 mr/hr	North shielding wall of operating area adjacent to valve and gage bd No. 30, 3 ft south of north wall and 3 ft west of east wall.
2a	400 mr/hr	1.8 r/hr	7.5 r/hr	+250 mr/hr	650 mr/hr	Pipe seal between primary pipe tunnel and pipe gallery behind N ₂ valve and gage bd No. 30, far north end. Contact with seal, near floor (this is the source of radiation causing 1a and 1b above).
2b	48 mr/hr	300 mr/hr	1.3 r/hr	+11.0 mr/hr	21.5 mr/hr	Pipe seal between primary pipe tunnel and pipe gallery behind N ₂ valve and gage bd No. 30, far north end. Approximately 6 ft south of pipe seal, near floor.
3a - Loop 1	0.035 mr/hr	0.070 mr/hr	0.25 mr/hr	0.80 mr/hr	2.0 mr/hr	Above shadow shielding for the vent lines from the IHX to sec Na expansion tanks (in steam generator rooms). At surface of south wall of the room, around the vent pipe.
3a - Loop 2	0.040 mr/hr	0.090 mr/hr	0.40 mr/hr	1.55 mr/hr	2.5 mr/hr	Above shadow shielding for the vent lines from the IHX to sec Na expansion tanks (in steam generator rooms). At surface of south wall of the room, around the vent pipe.
3a - Loop 3	0.050 mr/hr	0.100 mr/hr	0.35 mr/hr	1.20 mr/hr	2.3 mr/hr	Above shadow shielding for the vent lines from the IHX to sec Na expansion tanks (in steam generator rooms). At surface of south wall of the room, around the vent pipe.
3b - Loop 1	0.06 mr/hr	0.15 mr/hr	0.80 mr/hr	1.95 mr/hr	3.5 mr/hr	Above shadow shielding for the vent lines from the IHX to sec Na expansion tanks (in steam generator rooms). Behind shadow shield (inside the opening in the south wall).
3b - Loop 2	0.10 mr/hr	0.21 mr/hr	1.25 mr/hr	5.00 mr/hr	6.5 mr/hr	Above shadow shielding for the vent lines from the IHX to sec Na expansion tanks (in steam generator rooms). Behind shadow shield (inside the opening in the south wall).
3b - Loop 3	0.10 mr/hr	0.25 mr/hr	1.00 mr/hr	3.80 mr/hr	5.5 mr/hr	Above shadow shielding for the vent lines from the IHX to sec Na expansion tanks (in steam generator rooms). Behind shadow shield (inside the opening in the south wall).
4a - Cell 1	0.5 mr/hr	4.0 mr/hr	18.0 mr/hr	97.0 mr/hr	185 mr/hr	IHX cell to nitrogen cooler area seal plates. At contact with the weld of the nitrogen duct to the end plates (west side of lower N ₂ duct).
4a - Cell 2	1.7 mr/hr	6.5 mr/hr	27.0 mr/hr	125.0 mr/hr	240 mr/hr	IHX cell to nitrogen cooler area seal plates. At contact with the weld of the nitrogen duct to the end plates (west side of lower N ₂ duct).
4a - Cell 3	1.5 mr/hr	7.5 mr/hr	31.7 mr/hr	142.0 mr/hr	275 mr/hr	IHX cell to nitrogen cooler area seal plates. At contact with the weld of the nitrogen duct to the end plates (west side of lower N ₂ duct).
4b - Cell 1	0.025 mr/hr	0.05 mr/hr	0.15 mr/hr	0.50 mr/hr	1.10 mr/hr	IHX cell to nitrogen cooler area seal plates. Even with south wall of cooler operating area (approximately 6 ft north of seal)
4b - Cell 2	0.015 mr/hr	0.06 mr/hr	0.20 mr/hr	0.60 mr/hr	1 00 mr/hr	IHX cell to nitrogen cooler area seal plates. Even with south wall of cooler operating area (approximately 6 ft north of seal).
4b - Cell 3	*	*	0.30 mr/hr	0.75 mr/hr	1.30 mr/hr	IHX cell to nitrogen cooler area seal plates. Even with south wall of cooler operating area (approximately 6 ft north of seal).
5a	0.10 mr/hr	1.0 mr/hr	5.0 mr/hr	7.5 mr/hr	30 mr/hr	Pipe seal north of and behind nitrogen cooler No. 5 (at top of lower nitrogen duct). Approximately 2 ft south of pipe seal.
5b	0.05 mr/hr	0.12 mr/hr	0.8 mr/hr	2.0 mr/hr	10 mr/hr	Pipe seal north of and behind nitrogen cooler No. 5 (at top of lower nitrogen duct). Approximately 6 ft south of pipe seal.
6	*	*		2.5 mr/hr	0.55 mr/hr	Pipe gallery behind local boards 29 and 30. Reactor vent valve CV-1004 (went up to 2.5 r/hr during CR thumble leak on 6-1-63)
7	*	*		0.015 mr/hr	0.025 mr/hr	Catwalk in operating area behind He valve and gage board (went up to 9 mr/hr when CV-1004 was 1.5 r/hr).

NOTE: See Figures 1 through 8 for further identification of area.
*Data not recorded
+Change due to shielding which was added in primary pipe tunnel adjacent to this area.

high radiation levels in the vent system (see Item (1) under irregularities in RD&M data, Part III-A of this report). The extreme sodium carryover at this time caused very high radiation levels in such components of the reactor vent system as the reactor vent control valve CV-1004. It has been seen that under such an abnormal carryover condition in the vent system, shielding between the control valve and LB29 operating area is inadequate. It is felt that the above situation is a problem in the reactor vent system which must be corrected, thus further shielding is unnecessary. Design of the entire vent system and associated shielding was not based upon such high radiation levels.

The gamma radiation levels observed on the reactor loading face shield (LFS), at control rod housings and over the shield in general, at the various test power levels are tabulated in Table V. Table V gives the highest reading obtained on the actuator housings from the surface of the loading face to waist level. Also given in Table V is the maximum level measured, under the normal operating condition, between control rods and around the shield periphery. Not shown in the table (and not originally checked as part of this test) is the increase in radiation level on the LFS and control rod actuators that results after a reactor scram. Following the 125 Mwt tests, at which time a scram test was performed, the highest reading of activity after scram was observed. Following the scram test from 125 Mwt, a reading of 36 mr/hr was obtained at the top of control rod No. 6 actuator housing ~40 minutes after the scram. The prescram reading was 0.020 mr/hr. Both readings were taken with the instrument probe in contact with the actuator housing. Since no further scram tests were performed at higher power levels, no further data of this sort was obtained. From the data it can be seen that, during normal operation, the gamma radiation on the LFS represents no real problem as it stays well below MPL. These gamma radiation levels also display no general trend relative to reactor power level. The increased radiation levels following scram will be taken into account by Health and Safety in regard to issue of special work permits for work on the LFS after the reactor has been scrammed. The loading face shield is an area requiring Special Work Permit (SWP) for entry at all times during plant operation and immediately after shutdown.

During all five of the surveys performed, background radiation level in the plant was measured at from 0.010 to 0.025 mr/hr. Taking the larger value (0.025 mr/hr) as a general figure for plant background, then twice background

TABLE V
DETAILED PLANT RADIATION SURVEY
REACTOR LOADING FACE
(Highest reading obtained on control rod actuator housings from
loading face surface to waist level).
(mr/hr, Gamma)

Area Number 8		Reactor Power Level				
Control Rod Actuator Housing Number	Core Position	2.5 Mwt	10 Mwt	35 Mwt	125 Mwt	100% (244 Mwt)
1	C-1	+	0.070	0.050	0.450	0.140
2	C-20	+	0.130	0.075	0.150*	1.200*
3	C-28	+	0.120	0.090	0.055*	0.200*
4	C-36	+	0.110	0.050	0.500*	0.300*
5	C-38	+	0.600	0.090	0.350	0.200
6	C-16	+	0.170	0.150	0.090	0.100
7	C-44	0.250	0.500	0.135	0.650	0.450
8	C-50	0.250	1.600	0.300	0.120	0.550
9	C-24	+	0.080	0.150	0.075	0.600
10	C-56	+	0.060	0.100	0.125	0.300
11	C-62	0.300	0.750	0.150	0.165	0.600
12	C-32	+	0.080	0.170	0.060	0.150
13	C-68	+	0.050	0.050	0.125	0.180
14	C-78	+	0.450	0.250	0.125	0.110
15	C-86	+	0.550	0.550	0.090	0.140
16	C-94	+	0.130	0.130	0.040	0.150
17	C-102	+	0.300	0.600	0.045	0.130
18	C-110	+	0.050	0.200	0.075	0.070
19	C-118	+	0.150	0.040	0.060	0.060
		0.025	+	0.200	0.100	0.140**

NOTE: See Figures 1 through 8 for further identification of area.

*Control rod actuators removed, reading taken over opening in shield plug dust cover.

**Maximum reading elsewhere on the loading face

+Readings not recorded

would be 0.050 mr/hr. Listed in Table VI is a summary of all other areas in the plant, which have not been previously covered in this report, where survey readings at some time during the testing equaled or exceeded this general twice background radiation level of 0.05 mr/hr. The data for 100% power operation shows that none of these areas reached so much as one fourth of MPL. The 100% power level data for these areas also shows that none exceeded the general design value for a full time occupancy area (0.75 mr/hr). In general, the surveys cover all full-time occupancy areas* in the plant and one limited access area (Area No. 2 of Table IV).

The results of neutron surveys which were conducted simultaneously with the gamma surveys are summarized in Table VII. Calibration of the instrument used is given in Part II of this report. It should be pointed out that these readings are so low on the instrument scale (log scale 1 to 10^3) that accuracy is quite poor; the readings merely indicate the possibility of some neutrons present. It is felt that the readings (observed) times the calibration constant for the meter are not a true indication of the dose rate that a person would receive. It should also be pointed out that no neutron indication was obtained at a distance of a few feet from the points indicated in the table (with exception of N-2 which is a general level across and through most of the length of the pipe gallery). Of the six general areas where neutrons were detected, only two areas have significant gamma radiation as previously discussed. Area 2 (and N-2) is a limited access area to which the entrance door is kept locked and which requires an SWP for work in the area. Area 8 (and N-8) is the loading face shield which is an SWP area at all times during and immediately after reactor operation.

C. EFFECT OF REACTOR OPERATION ON AREA RADIATION LEVELS

From a review of data and discussion presented in Parts III-A and III-B of this report, it is evident that most areas in the plant display no change during normal plant operation. This is to say that, for the most part, areas are sufficiently shielded so that the radiation levels remain at essentially background values.

*Final Summary Safeguards Report for the Hallam Nuclear Power Facility, NAA-SR-5700, p 4-21, Table 4.2.

TABLE VI
 DETAILED PLANT RADIATION SURVEY
 AREAS HAVING RADIATION LEVELS GREATER THAN 0.05 mr/hr BUT LESS THAN MPL
 (mr/hr, Gamma)

Area Number	Reactor Power Level					Area Description
	2.5 Mwt	10 Mwt	35 Mwt	125 Mwt	100% (244 Mwt)	
9a - Cell 1	X	X	0.020	0.070	0.120	Passageway in IHX cell N ₂ cooler area (in line with seal plates mentioned under Area No. 4)
9b - Cell 2	X	X	0.035	0.060	0.075	Passageway in IHX cell N ₂ cooler area (in line with seal plates mentioned under Area No. 4)
9c - Cell 3	X	X	0.030	0.100	0.090	Passageway in IHX cell N ₂ cooler area (in line with seal plates mentioned under Area No. 4)
10a - Loop 1	X	X	X	0.025	0.090	Primary sodium pumps, maximum readings (taken around pump, inside shaft housing, etc.)
10b - Loop 2	X	X	X	0.050	0.250	Primary sodium pumps, maximum readings (taken around pump, inside shaft housing, etc.)
10c - Loop 3	X	X	X	0.020	0.050	Primary sodium pumps, maximum readings (taken around pump, inside shaft housing, etc.)
11a	X	0.030	0.125	0.300	0.500	Primary pipe tunnel access plug No. 8, at contact with joint around plug.
11b	X	X	0.050	0.120	0.175	Primary pipe tunnel access plug No. 9A, at contact with joint around plug.
11c	X	X	0.050	0.150	0.150	Primary pipe tunnel access plug No. 9B, at contact with joint around plug.
12a - Cell 1	X	X	0.035	0.075	0.120	Small IHX cell access plugs at elevation 1440 ft - 6 in., at contact with joint around plug.
12b - Cell 2	X	X	0.075	0.125	0.200	Small IHX cell access plugs at elevation 1440 ft - 6 in., at contact with joint around plug.
12c - Cell 3	X	X	0.075	0.150	0.300	Small IHX cell access plugs at elevation 1440 ft - 6 in., at contact with joint around plug.
13a - Cell 1	X	X	0.030	0.030	0.050	Large access plugs over IHX cells at elevation 1460 ft - 0 in., at contact with joint around plug.
13b - Cell 2	X	X	0.035	0.030	0.060	Large access plugs over IHX cells at elevation 1460 ft - 0 in., at contact with joint around plug.
13c - Cell 3	X	X	0.050	0.045	0.055	Large access plugs over IHX cells at elevation 1460 ft - 0 in., at contact with joint around plug.
14	+	+	+	+	0.16	South wall at top of N ₂ cooler room, housing coolers 4 and 9, at cable seal plate.

NOTE: See Figures 1 through 8 for further identification of area.
 X Background radiation level
 + Readings not recorded

TABLE VII
DETAILED PLANT RADIATION SURVEY
NEUTRON SURVEYS
(Readings are in Counts/sec)

Area Number	Reactor Power Level					Area Description	
	2.5 Mwt	10 Mwt	35 Mwt	125 Mwt	100% (244 Mwt)		
N-2	s	None	None	≤1.5	2 to 4	<1.2	Pipe gallery behind LB-29 and LB-30 (general level through gallery).
	f			None	≤1.5	<2	
N-8	s			≤1.2	≤1.2	<2	Reactor loading face (measured 2 in. off surface of loading face).
	f			None	≤1.2	<2	
N-11a	s				≤1.2	None	Primary pipe tunnel access plug No. 8, in contact with joint around plug.
	f				≤1.2		
N-11b	s				≤1.2		Primary pipe tunnel access plug No. 9A, in contact with joint around plug.
	f				None		
N-11c	s				≤1.5		Primary pipe tunnel access plug No. 9B, in contact with joint around plug.
	f				≤1.2		
N-12	s				≤1.2		Small IHX cell access plugs (elev. 1440 ft - 6 in.), in contact with joint around plug.
	f				None		
N-14	s						South and southwest wall* by stair No. 1 (elev. 1420 ft) (stair loading down from top of N ₂ cooler 4 and 9).
	f					≤1.3	
N-15	s					≤1.2	North and northwest wall* of area around spent fuel storage pit gage board No. 5.
	f					≤1.5	

NOTE: See Figures 1 through 8 for further identification of area.

*Indicates readings obtained with detector in contact with wall.

$\leq$: Less than or equal to

$<$: Less than

f: fast neutrons

s: slow neutrons

From the discussion in Part III-B and data in Table IV of this report, it can be seen that there are only four general areas (Areas 2, 3b, 4a, and 5 of Table IV) in which radiation varies appreciably with reactor power and where, at some locations, full time occupancy limits are exceeded. These areas are not normally occupied and have been posted at locations where the variation of radiation with power is considerably less and where maximum permissible levels will not be exceeded even at full power.

To summarize, all areas of the plant which are normally occupied during plant operation (including areas up to the position where radiation signs are posted) are seen to vary by only relatively small amounts in relation to reactor power and remain below MPL at all times. It would be well to make note here that radiation levels in some areas of the plant will be dependent on operations that take place, other than reactor operation alone. When special operations such as maintenance and repair work are being performed, levels of radiation in areas such as the maintenance cell operating area and equipment and personnel decontamination rooms can be expected to differ from those measured in this test. This test shows these areas to be within full time occupancy limits when the reactor is operating and when other special work is not in progress. The determination of radiation levels and access controls during special operations, such as maintenance and fuel changing, is an established function of Health and Safety.

D. ADEQUACY OF SHIELDING AROUND THE REACTOR AND PROCESS SYSTEMS

From previous discussions and data presented in Part III of this report, it is felt that the following statement is true: "The shielding around the reactor and all process systems at HNPF is adequate to maintain less than maximum permissible level of radiation in all normally occupied areas of the plant for all power levels of reactor operation."

Shielding was added in the primary pipe tunnel to insure that one particular operating area would remain below MPL (see Area No. 1, Table IV). There are four general areas (eight physical locations) in the plant where radiation exceeds MPL. These four general areas (Nos. 2, 3b, 4a, and 5 of Table IV) are relatively inaccessible and have been posted and identified. The posting is established at locations with less than MPL radiation and excludes the areas beyond

as full time occupancy areas. Work beyond the posted boundaries will require SWP coverage and the restrictions and safety precautions which are part of the SWP. Additional shielding may prove desirable in some of these areas as a matter of convenience but is not necessary for safe plant operation.

For aid in identifying the location of areas listed in Tables IV through VII, Figures 690-1 through -8 are included. These figures are reproductions of reactor building drawings. Areas indicated by the numbered call-outs correspond to the number-letter area designations listed in Tables IV through VII.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with post critical test AI-PC-690, Radiation Shielding, as listed in NAA-SR-5700 (Supplement 4), page XI (c) - 36, with remarks regarding the test to explain compliance with the criteria.

- a) Criterion: Detailed gamma and neutron surveys have been made for each steady state power level from one kilowatt to full power.

Remarks: As covered in Part III-B of this report, surveys were made at five power levels ranging from 2.5 Mwt to 244 Mwt (100% power). During these surveys, sodium activity was, for all practical purposes, at steady state. Data below 2.5 Mwt is of little value as it can show nothing but lower levels. Radiation levels at any power may be closely determined by interpolation and extrapolation of the data taken.

- b) Criterion: All sources of neutron or gamma streaming have been attenuated to MPL by appropriate shielding, or access to these areas has been restricted or limited.

Remarks: This criterion has been met. All locations where MPL is exceeded have been made restricted access areas by posting and require special work permits for entry.

- c) Criterion: All areas that contain detectable radiation levels are identified with appropriate posting.

Remarks: This criterion has been satisfied to the practical extent. At the position of posting for the four basic areas which have been

restricted (Part III-D), the radiation level is, in general, less than one-half MPL. Access to the plant proper (wherein all survey data were taken) is denied to the general public. Occupancy of any of the areas covered in this report is restricted to personnel whose radiation history is kept by film badge and other methods as particular situations require. To post areas of acceptable, but yet detectable radiation levels (such as given in Table VI) would lead to confusion, lack of respect for posted areas, and, in general, defeat the purpose intended for posting of areas.

V. EVALUATION

A. INTRODUCTION

A substantial amount of data was collected during the radiation shielding post critical test. Evaluating these experimental data, in light of previously predicted analytical values, leaves one with a feeling of both satisfaction and frustration. The reactor is adequately shielded and the shielding does in fact meet the design criteria.¹ These criteria, stated briefly, are as follows:

- 1) To limit the radiation level in full-time occupancy areas to 0.75 mrem/hr, except where shield penetrations occur such as gaps between shield plugs. Over these localized areas the radiation level may be increased to 7.5 mrem/hr.
- 2) To restrict the radiation level in limited access areas to 7.5 mrem/hr.

There are several areas, and the loading face shield in particular, which may well be overdesigned. A radiation survey of the shield shows that at full power the neutron flux (either fast or slow) is negligible, and that the gamma-ray dose rate is approximately 0.5 mrem/hr. Unfortunately this dose rate is not due to gamma-ray penetration of the bulk shield or of the numerous fuel plugs. If it were it would meet the design objective of the shield almost perfectly.² Instead, this radiation level is due to the induced argon activity present in the helium as an impurity. This helium is contained within the control rod thimbles which extend above the loading face shield. Thus, it is difficult to quantitatively assess the degree of overdesign.

B. RADIATION DETECTION AND MONITORING SYSTEM DATA

With but two exceptions the radiation detection and monitoring system and the continuous air monitors gave readings which were independent of reactor power level. This independence of reactor power level is as it should be and indicates that even at full reactor power the activity, during normal operation, is below the sensitivity of the instrument. Thus, in the unlikely event of an incident where fission product activity may be released, detection of the incident becomes much more positive.

Specifically the two exceptions are: (1) reactor atmosphere and, (2) the IHX and cold and carbon trap cell atmospheres. The excessively high reactor atmosphere activity was due to carryover of radioactive sodium into the reactor vent

system and then into the detector filter and piping. The sodium carryover was in turn due to helium leaks in several control rod thimbles, thus causing extra helium and higher pressures in the reactor atmosphere. Once the defective control rod thimbles were replaced the problem was greatly alleviated.

The high readings recorded for the IHX cell atmosphere and for the cold and carbon trap cells atmosphere were not due to high specific activity of the nitrogen. Instead they were due to location of the detectors on the nitrogen cooling ducts. In this position the detectors were receiving some gamma-ray streaming up the cooling ducts. This problem can be eliminated by surrounding the detectors with sufficient shielding. This will prevent any gamma-ray streaming from masking any cell atmosphere activity. As noted from the data, this gamma-ray streaming causes the detectors to read about 10 times background at full reactor power. The equivalent of 1 in. of lead around these detectors will suffice to reduce the scattered photons to a negligible contributor.

C. DETAILED PLANT SURVEY

A detailed plant survey was conducted during five different power levels, after allowing Na-24 activity to essentially reach equilibrium. Again the shielding design criteria¹ are met in all cases. Extremely high gamma radiation levels were noted in several areas during the 35 Mwt power run. This was due to a void in the east primary pipe tunnel shield wall. This void (3 ft wide and 5 ft high) was subsequently filled in with bricks following shutdown from the 35 Mwt run. No further problems in the way of excessive radiation levels were noted.

During the plant survey at full reactor power (244 Mwt) only four areas were noted as reading above MPL, i.e., >2.5 mrem/hr. These areas are noted in Table IV as area No. 2, 3b, 4a, and 5. All of these areas are stated in reference 1 as being inaccessible during reactor operation. Following reactor operation, however, and after allowing a reasonable period for Na-24 decay, these areas do become accessible on a limited basis.

In reviewing the myriad of shielding calculations performed during design of the HNPF, it is encouraging to note how close many of the experimental values agreed with the analytical ones. For example, in areas 4a and 4b the measured values during full power were about 250 mrem/hr and 1.2 mrem/hr, respectively. The corresponding analytical values (unpublished) are 270 mrem/hr and 3.8 mrem/hr.

The measured radiation levels on the loading face shield, taken during full power operation, are in the range of 0.1 to 0.6 mrem/hr (Table V). Although this range agrees quite well with the predicted value² of 0.32 mrem/hr the similarity is fortuitous. As discussed in the introduction, the analytical value was calculated on the basis of gamma-ray penetration of the shield. The measured value is to a large extent due to the induced argon activity present in the helium as an impurity. Each of the control rod thimbles which extend above the loading face shield contains this helium, thus masking any radiation levels due to direct penetration. This helium activity clearly explains the unprecedented increase in radiation level at the actuator housing 40 min after scram. It was merely due to the displacement of the helium, located in the core (high flux) region during reactor operation, by the poison rods.

The activation of argon in helium had never been calculated during the HNPF design stages. If the calculations are performed now, using measured values for the specific activity of the helium and a reasonable assumption for the average thermal neutron flux in the helium region, the agreement with experimental values are extremely good. The measured dose rate at the control rod thimble was 3.5 to 4.0 mrem/hr; the calculated value was 4.0 mrem/hr. The calculated volume percent of argon in helium was 4.1×10^{-4} *; the estimated experimental value was 5×10^{-4} . The complete calculations are included in Appendix B.

An evaluation of recent measurements[†] is of sufficient interest to be included at this point. The data show that when the control rod housing gas has a specific activity of $2.7 \times 10^{-2} \mu\text{c/cc}$ the radiation level at the housing surface is approximately 6 mr/hr. When the specific activity is decreased to $2.6 \times 10^{-5} \mu\text{c/cc}$, the corresponding radiation level at the housing surface is about 0.15 mr/hr (five times background). We then see that for a thousand-fold decrease in specific gas activity the radiation level drops by a factor of only forty. Thus, the 0.15 mr/hr is due to radiation which penetrates the loading face shield, rather than due to any gas activity. Similar radiation levels were determined at a point 1 in. above several of the fuel plugs, and at the lower gas activity. These radiation levels were between 0.03 and 0.09 mr/hr for an average of 0.06 mr/hr. Since the above measurements were taken at 57% of full power they should be multiplied by 1.75 to determine the values at full power.

*Values reported in HNPF field operations weekly report of October 23, 1963.

†Taken from radiation surveys conducted on November 21 and November 29, 1963, by CPPD personnel.

Thus, the radiation levels at the control rod housing and at the fuel plugs at full power are 0.26 mr/hr and 0.11 mr/hr, respectively. The calculated value for the dose rate over the fuel plugs, as stated above, is 0.32 mr/hr. This represents a conservativeness factor of three in the fuel plug calculation, or only about 3.5 in. of extra concrete.

D. SUMMARY

To reiterate, the biological shielding around the Hallam reactor does meet all design criteria. This must not imply, however, that all radiation measurements should cease. On the contrary they should be continued and, if possible, extended. There are two important reasons for this continued effort, namely:

- 1) To determine the degree of overdesign in those areas, such as above the numerous annuli in the loading face shield, which are felt to be overdesigned. Only in this way can existing calculational methods be modified. Only in this way can the results be quantitatively applied to future sodium reactors, making them more economical and more competitive with fossil systems.
- 2) To make measurements beyond the mere biological dose rate surveys. The field of shielding includes more than just determining thicknesses of concrete. Extensive effort was expended in determining neutron flux distributions at large distances from the core in both axial and radial directions. In addition, heat generation values were calculated in both the loading face shield and in the radial concrete surrounding the core. The induced activity of the primary sodium (Na-22 and Na-24), as well as the main primary sodium valves, has been determined. Much of this calculated information has been formally reported,³ but not yet experimentally verified. Only by good experimental data can confidence, or lack of confidence, be placed on existing calculational techniques.

REFERENCES

1. "Shielding Design Criteria and Radiation Areas for Hallam Nuclear Power Facility," Bechtel Corporation, Lab No. 2933 (January 6, 1959)
2. S. Berger, "Shielding Analysis for the HNPF Fuel Element Shield Plugs," TDR-5182 (April 14, 1960)
3. S. Berger, "Gamma Ray Heat Generation in the HNPF Biological Shields," NAA-SR-6244

Figure 690-1. North Half of Reactor Plant, Plan Elevation 1464 ft-0 in.

Figure 690-2. North Half of Reactor Plant, Plan Elevation 1440 ft-6in.

Figure 690-3. South Half of Reactor Plant, Plan Elevation 1440 ft-6in.

Figure 690-4. North Half of Reactor Plant, Plan Below Elevation 1440 ft-6 in.

Figure 690-5. South Half of Reactor Plant, Plan Below Elevation 1440 ft-6 in.

Figure 690-6. South Half of Reactor Plant, Plan Below Elevation 1426 ft-0 in.

Figure 690-7. South Half of Reactor Plant, East-West Elevation

Figure 690-8. North Half of Reactor Plant, North-South Elevation

APPENDIX A ESTIMATE OF PERCENTAGE EQUILIBRIUM ACTIVITY IN PRIMARY SODIUM

In the production of a relatively short half-life radioactive nuclide, decay of the radioisotope during irradiation causes the activity to reach a limiting value at which the rates of formation and rates of decay are equal.

Based upon the above, the following approach was used to obtain percentage equilibrium activity at any time (t) from start of constant power level.*

Sodium 24 is used since it is the principle source of activity. Number of atoms of Na²⁴ present at any time (t):

$$Q = \frac{I\Sigma}{\lambda} (1 - e^{-\lambda t}) \quad \dots(1)$$

where

Q = Number of atoms of Na²⁴

I = Neutron intensity

Σ = Macroscopic cross section for neutron sodium reaction

λ = Radioactive decay constant of Na²⁴

It is seen that as time (t) approaches infinity in Equation 1, e^{-λt} approaches zero. The equilibrium activity then is:

$$Q_e = \frac{I\Sigma}{\lambda} \quad \dots(2)$$

To obtain the fraction of equilibrium at any time (t), divide both sides of Equation 1 by Equation 2

$$\frac{Q}{Q_e} = \frac{\frac{I\Sigma}{\lambda}(1 - e^{-\lambda t})}{\frac{I\Sigma}{\lambda}} = 1 - e^{-\lambda t} \quad \dots(3)$$

*Richard Stephenson, "Introduction to Nuclear Engineering," McGraw-Hill, 1958 p 39.

Let $E = Q/Q_e$ = fraction of equilibrium activity and insert the basic relation between decay constant and half-life: $\lambda = 0.693/T$, where T = half-life

$$E = 1 - e^{-\left(\frac{0.693}{T}\right)t} \quad \dots(4)$$

Using Equation 4 and the half-life of Na^{24} , $T \approx 15$ hours, the following values are obtained:

$t = 24$ hours (1 day)	$E = 0.673 = 67.3\%$
$t = 48$ hours (2 days)	$E = 89.3\%$
$t = 72$ hours (3 days)	$E = 96.5\%$
$t = 96$ hours (4 days)	$E = 98.9\%$
$t = 120$ hours (5 days)	$E = 99.6\%$

This information, then, is plotted in Figure 690-A-1.

The times involved at each survey power level are as follows:

Δt_1 = time at start of power run - time at start of survey

Δt_2 = time at start of power run - time at finish of survey.

Reactor Power Level	Δt_1	Δt_2
2.5 Mwt	50 hours	83 hours
10 Mwt	91 hours - 33 minutes	117 hours - 18 minutes
35 Mwt	67 hours - 40 minutes	95 hours - 40 minutes
125 Mwt	49 hours - 40 minutes	75 hours - 40 minutes
244 Mwt	50 hours - 10 minutes	74 hours - 45 minutes

Based on the curve in Figure 690-A-1 and the above times, the estimated percent equilibrium for each survey was determined and is presented in Part III-B of this report (Table III).

Figure 690-A-1. Percent Equilibrium Sodium Activity vs Time

APPENDIX B

ARGON ACTIVITY FROM He GAS IN CONTROL ROD THIMBLES

Basis:

$$S_v = \text{Specific activity of He} = 5 \times 10^{-3} \mu\text{c/cc}^* \text{ (at } \sim 25\% \text{ power)}$$

Assume activity is due entirely to A^{41} :

$$T_{1/2} = 1.8 \text{ hr}$$

$$\text{energy} = 1.3 \text{ Mev}$$

$$\sigma(n, \gamma) = 0.53 \text{ barns}; A^{40} + n \rightarrow A^{41} + \gamma$$

Assume that the activated He extends above the loading face shield in a long cylinder of 4 in. radius (10 cm); neglect self absorption.

Calculations:

In a long, nonabsorbing cylinder, the γ energy flux at the surface is represented by the following equation:

$$\phi = 0.8 S_v R$$

where:

$$S_v = \frac{\text{Mev}}{\text{cc-sec}} = 5 \times 10^{-3} \frac{\mu\text{c}}{\text{cc}} \times 3.7 \times 10^4 \frac{\text{d}}{\text{sec-}\mu\text{c}} \times \frac{1.3 \text{ mev}}{\text{d}}$$

$$\therefore S_v = 2.4 \times 10^2 \frac{\text{Mev}}{\text{cc-sec}}$$

$$R = 10 \text{ cm}$$

$$\therefore \phi = 0.8 \times 2.4 \times 10^2 \times 10 = 1.9 \times 10^3 \text{ Mev/cm}^2\text{-sec}$$

$$\text{mr/hr} = \frac{1.9 \times 10^3}{5 \times 10^2} \simeq 4 \quad \text{This value agrees well with the measured value of} \\ 3.5 \text{ to } 4 \text{ mr/hr.}^*$$

*HNPF field operations weekly report, 10/23/63 to 10/27/63.

The vol % of argon in He may now be estimated if we assume that the He within the thimble is irradiated at an overall average thermal neutron flux of $1 \times 10^{13} \text{ n/cm}^2 - \text{sec}$.

$$S_v \left(\frac{\mu\text{c}}{\text{cc}} \right) = \frac{\text{atoms of A}}{\text{cc}} \times \sigma \times \phi \times \frac{1}{3.7 \times 10^4}$$

where:

$$S_v = 5 \times 10^{-3}$$

$$\sigma = 0.53 \times 10^{-24} \text{ cm}^2$$

$$\phi = 3 \times 10^{12} \text{ n/cm}^2 - \text{sec (at 25\% power)}$$

$$\begin{aligned} \therefore \frac{\text{atoms of A}}{\text{cc}} &= 5 \times 10^{-3} \times 3.7 \times 10^4 \times \frac{1}{0.53 \times 10^{-24}} \times \frac{1}{3 \times 10^{12}} \\ &= 1.1 \times 10^{14} \end{aligned}$$

The concentration of He atoms per cc of He at STP is merely:

$$\frac{6.02 \times 10^{23} \text{ atoms}}{2.24 \times 10^4 \text{ cc}} = 2.7 \times 10^{19}$$

Since atom % is equal to vol % for mono atomic gases the % of argon in He is:

$$\% = \frac{1.1 \times 10^{14}}{2.7 \times 10^{19}} \times 100 = 4.1 \times 10^{-4}$$

NAA helium specifications states that grade "A" He shall have impurities no greater than $5 \times 10^{-3} \%$ (by vol). Of these impurities about one-half shall consist of O_2 , H_2O , and hydrocarbons. The remainder of the impurities will consist of gases such as H_2 , N_2 , Ne, and argon. From this we might expect an argon concentration of about $5 \times 10^{-4} \text{ vol } \%$; again a fair agreement with the above calculated value of $4.1 \times 10^{-4} \text{ vol } \%$.