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FINAL EVALUATION REPORT
HALLAM NUCLEAR POWER FACILITY
VOLUME VI

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HALLAM NUCLEAR POWER FACILITY

Post Critical Test Procedure

AI-PC-710

Wet Excess Fuel Loading

Test Procedure Submitted for Approval

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

Test Procedure Reviewed

9/6/62 D. E. Simpson
Date Atomic Energy
Commission

Test Approved for Performance

8/25/62 James E. Rhone
Date Atomic International

Test Completed

10/14/62 James E. Rhone
Date Atomic International

Test Approved for Performance

9/25/62 R. C. Buntain
Date Atomic Energy
Commission

Test Completed

10/14/62 R. C. Buntain
Date Atomic Energy
Commission

AI-PC-710

J. R. ROHLFS

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May 25, 1963

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HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
WET EXCESS FUEL LOADING TEST, AI-PC-710

I. PURPOSE OF TEST

The purpose of this test was to present the sequence to be followed for installing an operational loading of fuel in the HNPF sodium filled reactor. As a result of the fuel loading experiment, values for the following reactor parameters could be established:

- 1) The excess reactivity composing the operational loading of fuel.
- 2) The shutdown margin available to the reactor.
- 3) The total worth of the control rods in the reactor.
- 4) The reactivity worth of the sodium in the reactor.
- 5) The reactivity effect of moving the source element from the normal core position to a peripheral channel in a partially loaded core, and inserting a fuel element in the original source position.

II. DISCUSSION OF TEST PERFORMANCE

A. SUMMARY

Fuel loading commenced with a 53 element core which had been loaded previously as part of the Wet Critical Test, AI-P-1158. A total of eighty-seven additional elements were installed in the core in ten increments consisting of 3, 6, 6, 6, 9, 9, 9, 9, 15, and 15 elements, respectively, making the total core loading 140 fuel elements.

After each fuel loading increment, it was determined that the reactor would not go critical with the two most reactive control rods withdrawn. This was accomplished by fully withdrawing the two rods estimated to have the greatest reactivity worth, CR-1 (gang 5) and CR-6, and observing the sub-critical behavior of the neutron flux.

The reactivity worth of each incremental loading of fuel was measured by calibrating control rods. The control rod travel necessary to compensate for each fuel addition was calibrated by the differential period technique. The total excess reactivity for each fuel loading was taken as the sum of the individual reactivity increments for previous loadings. A summary of the wet excess fuel loading data is presented in Table I.

It can be shown that:

$$1 - k_{\text{eff}} \approx M^2 B^2 - (K_{\infty} - 1),$$

where $B^2 = \left(\frac{\pi}{H}\right)^2 + \left(\frac{2.405}{R - \delta}\right)^2$ = geometric buckling,

H = height of reactor with axial reflector savings,

δ = radial reflector saving,

R = effective core radius = $\sqrt{\frac{NA}{\pi}}$,

where N = number of fuel elements loaded and

A = cross sectional area associated with each fuel element.

For N < 95, A = 870 cm²

For N > 95, $R = \sqrt{\frac{95 \times 870}{\pi} + \frac{(N - 95)A}{\pi}}$ where A = 724 cm²

The change in the expression for R at N = 95 is due to the fact that there is one control rod for every 5 elements up to element 95, but no additional control rods for fuel elements in excess of 95.

Therefore, a plot of $1 - k_{\text{eff}}$ vs B^2 should approximate a straight line, with a slope equal to the migration area, M^2 , and a y-intercept equal to $(k_{\infty} - 1)$. In such a plot of the dry critical loading, M^2 was within 8 percent of its calculated value.

The experimentally determined reactivity vs fuel loading is plotted in Figure 710-1 as $k_{\text{eff}} - 1$ vs B^2 , where

$$\rho = (k_{\text{eff}} - 1)/k_{\text{eff}}$$

This plot departs from a straight line relationship, particularly near the full core loading of 140 elements. Two factors contribute to this departure,

TABLE I
SUMMARY OF WET EXCESS FUEL LOADING DATA

No. of FE Incore	Critical Control Rod Gang Position (inches) ***					All Rods Banked (inches)	No. of FE Loaded	Excess Loaded (cents)	Excess Loaded (%)	Total Excess (cents)	Total Excess (%)	Shut- down Margin
	1	2	3	4	5							
53	out	out	out	out	129.1	139.8	0	14.909	0.103	14.909	0.103	
56	out	out	out	out	90.6		3	69.636	0.483	84.545	0.586	
62	out	out	out	out	49.95	97.7	6	138.848	0.962	223.393	1.548	
68	out	out	out	115.0	in	84.8	6	146.561	1.016	369.954	2.564	
69	out	out	66.7	66.7	out	74.6	1					
74	out	out	out	84.6	in	76.6	5	107.804	0.707	477.758	3.311	
83	out	out	out	57.3	in	68.6	9	150.719	1.044	628.477	4.355	
83*	33.95	out	33.95	33.95	33.95	38.7	0					
92	out	out	out	34.75	in	62.65	9	141.769	0.982	770.246	5.338	
101	out	out	111.3	in	in	58.45	9	134.03	0.929	904.276	6.267	
110	out	out	76.9	in	in	54.45	9	96.488	0.669	1000.764	6.935	
110*	15.1	100.0	15.1	15.1	15.1	32.0	0					3.64
125	out	out	32.0	in	in	50.6	15	191.299	1.326	1192.063	8.261	
125*	121.6	in	in	in	in	29.0	0					2.45
125+	out	in	64.0	in	in	41.75	0	-147.19	-1.02	1044.873	7.241	3.47
125++	out	out	95.5	in	in	77.8	0					
140++	out	120.5	in	in	in	65.95	15	212.703	1.474	1257.576	8.715	
140+	109.6	in	in	in	in	35.9	0					3.1
140**	69.45	out	12	12	12	45.05	0					2.1

* drained

+ drained with fixed rods

++ fixed rods installed

** fixed rods removed

*** in = 0 inches

out = 156 inches

Figure 710-1. K_{eff}^{-1} vs $B^2 \times 10^6$

(1) direct summation of incremental reactivities overestimates total reactivity due to decrease in worth of the early fuel increments as the core size increases, and (2) as core loading increased the control rods near the center of the core were inserted to compensate for the excess reactivity. This shifted the neutron flux radially outward and increased the apparent worth of peripheral fuel elements.

Also plotted on Figure 710-1 is a multigroup calculation of $1 - k_{\text{eff}}$ vs B^2 . This plot is essentially linear, and at the smaller core loadings, where the corrections to the measured values are small, agrees fairly well with the experimental points. At a fuel loading of 140 elements, the calculated k_{eff} is 1.069, giving an excess reactivity of 6.5%. Because of the good agreement of the calculated excess reactivity with the measurements for the first few loading increments, it is believed that the excess reactivity in the 140 element core is about 6.5%.

Core shutdown margin and sodium worth were measured by a number of core sodium draining tests. Such tests, in which the reactor was drained of sodium and various reactivity measurements were made, were performed at the 68, 83, 110, 125, and 140 element loadings. At the 125 element core loading it was estimated that additional control rods would be required in the 140 element core to maintain the desired shutdown margin of 2% $\Delta k/k$ in the drained reactor. (It was later shown that the shutdown margin was underestimated at the 125 element loading). Therefore, three fixed poison columns were added near the core center to increase shutdown margin.

After 140 elements had been loaded, the shutdown margin was measured by three methods;

- a) inter-comparison of control rods (dry core)
- b) calibration of "split" gang 1 rods (dry core), and
- c) period calibration of gang 1 as a bank in the wet core and extrapolating to dry core.

These methods gave the shutdown margin as 2.49%, 3.15%, and 3.20%, respectively.

Method (a) consisted of measuring the total worth of the control rods in gang 1 by comparing each rod individually with gang 3, which was calibrated by period measurements, and summing the results for the 6 rods of gang 1.

It was determined that this method of measuring the shutdown margin gave too low a value, due to the radial tilting of the flux when one rod of gang 1 was inserted. Method (b) consisted of calibrating 3 rods of gang 1 moving downward with the other three (every other one to maintain radial symmetry) moving upward. The two halves of gang 1 were not worth exactly the same, due to shadowing within the gang itself. The total gang worth was taken as twice that of the half-gang with lower worth, or less than the sum of the two components. Method (c) was accomplished with gangs 3, 4, and 5, near the core center, fully withdrawn to minimize radial flux peaking. Gang 2 was used to compensate for the changed position of gang 1 during the period measurements.

Methods (b) and (c) are considered to be more accurate determinations of the shutdown margin, which in the dry, 140 element core with three fixed poison columns is $3.2\% \Delta k/k$.

It was estimated that the three fixed control rods could be removed from the core with ample shutdown margin remaining. Therefore, the fixed rods were removed from the reactor, and their worth was measured to be 0.9% in a dry core (see discussion on rod calibrations, below). Therefore, the shutdown margin was reduced to 2.3% , still greater than the desired 2% at the test temperature of 350 F . Subsequent temperature coefficient measurements indicate that the shutdown margin at operating temperatures will be $\sim 2.0\%$ in the dry core.

Measurement of the worth of the three fixed poison columns was made both upon insertion in the 125 element, dry core, and upon removal from the 140 element, wet core. The fixed rods increased the shutdown margin in the 125 element dry core, by 1.02% , so this represents the worth of these rods in this core.

At the 140 fuel element loading, the poison column worth upon removal was measured by noting the change in critical position of gang 1 before and after the poison column removal. Gang 1 had been calibrated again after the poison columns were removed. As expected, the worth of gang 1 decreased when the poison columns were removed, due to the shifting of reactor flux. Two values were thus obtained for poison column worth; 0.87% and 0.75% .

It is to be expected that poison column worth would be less in the 140 element wet core than in the 125 element dry core for two reasons. First, control rod worth is approximately proportional to the lattice migration area. The dry core migration area has been experimentally determined to be 485 cm², and that for the wet core 435 cm². Second, control rod worth is inversely proportional to the square of core radius. Therefore, the predicted worth of poison columns in the 140 element wet core, based on the measured value in the 125 element dry core, is $1.02 \times \frac{435}{485} \times \frac{125}{140} = 0.82\%$. This agrees with the above measured poison column worth. Again correcting the poison column worth by the migration area ratio, their worth in the 140 element dry core is between 0.83% and 0.96%.

Critical rod configurations at the 140 element loading showed that the insertion of gang 2 rods, from full out to full in, is adequate to compensate for draining the reactor of sodium. Gang 2 is worth 4% in the wet core and 4.4% in the dry core.

Gang 3 was used to compensate for the actual draining of sodium during the test at the 140 element loading. A dry core calibration of gang 3 shows the sodium to be worth 4.13%. The calibration was performed in the least perturbed state that could be arranged; gangs 2, 4, and 5 inserted and gang 1 being inserted as gang 3 was withdrawn. These measurements of sodium worth are somewhat less than the calculated worth of 5.4% $\Delta k/k$.

The best approximation to total control rod worth is the summation of the other measured values; namely excess reactivity (6.5%), sodium worth (4.1%), and shutdown margin (2.3%), or 12.9%. The shutdown margin and sodium worth are for a dry core, so the estimate of rod worth in a wet core is 12.2%. This compares with the theoretically calculated total rod worth of 14% $\Delta k/k$ and a total rod worth of 11% $\Delta k/k$ calculated from previous dry core measurements.

The neutron source relocation experiment was performed at the 68 element loading. The source was moved from its normal position near the core center to the periphery of the then active core and a fuel element was placed in the normal source position. The reactivity change due to this movement was determined from differences in critical control rod positions.

A fuel element in the source position was determined to be worth 122 cents with gang 5 withdrawn, and 69 cents with gang 5 inserted. During measurements, gangs 1 and 2 were full out and gangs 3 and 4 were at approximately 70 inches. The source was returned to its normal position following these measurements.

The major objectives of the wet excess loading test have been achieved and the measured core parameters are in reasonable agreement with calculations made prior to the loading.

B. FUEL LOADING

The Operations Manager was notified of the intention to start the test and his permission obtained. During the testing the Shift Leader was cognizant of the test status at all times, and his approval was required for all plant operations. The test was performed round-the-clock, seven days a week, with the Responsible Test Engineer directing the overall performance. Shift coordination of the test was by the Shift Test Engineer and the Shift Leader.

Testing under AI-PC-710 began immediately after the wet criticality test had been completed. (Refer to HNPF Test Completion Report, AI-P-1158, Wet Critical Test, for a discussion of reactor conditions at the start of this test.) Fuel was loaded into the reactor according to the sequence shown in Table II. The table also lists the specific core channel into which each particular fuel element (as designated by fuel accountability numbers) was inserted. The first loading increment was three elements. Only 3.6 w/o U^{235} 10M elements were loaded into the reactor, except for one element. A 3.45 w/o U^{235} element was the 42nd element loaded as a part of the wet critical test.

After an increment of fuel had been loaded, the count rate of each of the three incore nuclear instrument (ICNI) channels were recorded with all control rods inserted. The numerical average of the three count rates was used to calculate the inverse of the average count rate ($\bar{I}$). This inverse

TABLE II
WET EXCESS FUEL LOADING SEQUENCE

Loading Sequence	Core Position	Fuel Element Number	Shield Plug Number	Type of Fuel
(First Increment)				
54 **	C- 49	MF- 60	RP- 46	R *
55	C- 61	MF-105	RP- 91	R
56	C- 73	MF- 32	RP-157	R
(Second Increment)				
57	C- 57	MF- 33	RP-167	R
58	C- 69	MF-162	RP-162	R
59	C- 45	MF-154	RP-208	R
60	C- 93	MF-155	RP-147	R
61	C-109	MF-156	RP-117	R
62	C- 77	MF- 84	RP-150	R
(Third Increment)				
63	C- 95	MF- 85	RP-171	R
64	C-111	MF- 86	RP- 69	R
65	C- 79	MF- 87	RP-140	R
66	C- 97	MF- 88	RP-176	R
67	C-113	MF- 83	RP- 89	R
68	C- 81	MF- 90	RP-187	R
(Fourth Increment)				
69	C-104	MF- 95	RP-113	R
70	C- 91	MF- 91	RP-161	R
71	C-107	MF- 92	RP-216	R
72	C- 75	MF- 93	RP- 58	R
73	C- 88	MF- 94	RP- 31	R
74	C-120	MF- 96	RP- 49	R

* Regular fuel element.

** First 53 elements were loaded as part of AI-P-1158, Wet Criticality Test.

TABLE II (continued)

Loading Sequence	Core Position	Fuel Element Number	Shield Plug Number	Type of Fuel
(Fifth Increment)				
75	C-100	MF-161	RP-139	R-P *
76	C-116	MF- 97	RP-131	R
77	C- 84	MF- 98	RP- 81	R
78	C-101	MF-108	RP-152	R
79	C-117	MF- 99	RP- 59	R
80	C- 85	MF-100	RP-173	R
81	C-103	MF-101	RP-172	R
82	C-119	MF-102	RP-169	R
83	C- 87	MF-103	RP-122	R
(Sixth Increment)				
84	C- 76	MF-151	RP-206	R-P
85	C- 92	MF-104	RP- 66	R
86	C-108	MF-106	RP-106	R
87	C-112	MF- 89	RP-125	R
88	C- 80	MF-107	RP-168	R
89	C- 96	MF- 19	RP- 17	R-P-T*
90	C-147	MF-120	RP- 90	R
91	C-167	MF-121	RP- 80	R
92	C-127	MF-121	RP- 99	R
(Seventh Increment)				
93	C- 90	MF-111	RP- 70	R
94	C-106	MF-112	RP- 84	R
95	C- 74	MF-113	RP- 88	R
96	C- 89	MF-114	RP- 82	R
97	C-105	MF-115	RP- 86	R
98	C-121	MF-116	RP- 61	R
99	C- 99	MF-117	RP- 94	R
100	C-115	MF-118	RP- 75	R
101	C- 83	MF-119	RP- 62	R
(Eighth Increment)				
102	C- 98	MF-109	RP- 86	R
103	C-114	MF-157	RP-170	R-P
104	C- 82	MF-110	RP-108	R
105	C-145	MF-122	RP- 95	R
106	C-165	MF-124	RP- 64	R
107	C-125	MF-125	RP- 68	R
108	C-149	MF-126	RP- 78	R
109	C-169	MF-127	RP- 38	R
110	C-129	MF-128	RP-136	R

* R - Regular fuel element

* T - Thermocoupled fuel element (in-fuel T/C's)

* P - Premeasured fuel slugs

TABLE II (continued)

Loading Sequence	Core Position	Fuel Element Number	Shield Plug Number	Type of Fuel
(Ninth Increment)				
111	C-138	MF-129	RP-105	R *
112	C-158	MF-130	RP-164	R
113	C-170	MF-131	RP- 65	R
114	C-156	MF-132	RP-153	R
115	C-176	MF-133	RP-111	R
116	C-136	MF-134	RP-143	R
117	C-146	MF-135	RP-175	R
118	C-166	MF-136	RP-185	R
119	C-126	MF-137	RP-120	R
120	C-148	MF-138	RP- 92	R
121	C-168	MF-139	RP-102	R
122	C-128	MF-140	RP- 76	R
123	C-143	MF-141	RP- 96	R
124	C-163	MF- 59	RP- 73	R
125 **	C-123	MF- 43	RP- 37	R

* R - Regular fuel element

** After the 125th element had been loaded, three fixed control rods were installed in the core, to ensure adequate shutdown margin would be available at the operational loading. To allow loading of the fixed control rods in the desired core positions, the following fuel element moves were required:

Fuel Element No.	From	To
a) MF- 4	C- 8	C- 10
b) MF- 71	C- 10	C-178
c) MF-153	C- 5	C- 13
d) MF- 69	C- 13	C-131
e) MF- 64	C- 11	C-154

TABLE II (continued)

Loading Sequence	Core Position	Fuel Element Number	Shield Plug Number	Type of Fuel
		(Tenth Increment)		
126	C-151	MF-160	RP-103	R-P *
127	C-171	MF-159	RP-126	R-P
128	C-174	MF- 57	RP-215	R
129	C-134	MF- 67	RP-212	R
130	C-140	MF- 56	RP- 43	R
131	C-160	MF- 75	RP- 63	R
132	C-180	MF- 79	RP- 55	R
133	C-150	MF- 82	RP-100	R
134	C-130	MF-150	RP-142	R-P
135	C-144	MF-147	RP-190	R
136	C-164	MF- 73	RP- 41	R
137	C-124	MF- 38	RP- 21	R
138	C-137	MF- 74	RP- 87	R
139	C-157	MF-148	RP-123	R
140 **	C-177	MF- 76	RP- 74	R

* R - Regular fuel element

* P - Premeasured fuel slugs

** Fifteen more fuel elements were loaded (loading sequence 126 to 140), bringing the reactor to an operational loading of 140 fuel elements.

Measurements made following completion of the operational loading determined that the three fixed control rods were not required to allow adequate shutdown margin. Basis for the decision to remove the fixed rods is discussed in section III. O, (Conclusions), of this report.

Following removal of the fixed rods, the fuel elements incore positions C-137, C-157 and C-177 were moved to core positions C-8, C-11 and C-5, respectively.

count rate was plotted on a graph of inverse count rate vs fuel elements loaded from which an extrapolated critical loading (with all rods inserted) was obtained by extrapolating to $\bar{I}$ equals zero.

After each fuel addition and prior to withdrawing control rods, the control rod positions for criticality were predicted with limits which required cessation, review, evaluation and approval by the Test Engineer.

To comply with technical specifications and prove that the core could not go critical by withdrawing any two rods, CR-1 (gang 5) and CR-6 were fully withdrawn first. For this control rod configuration, the inverse of the average count rate ($\bar{I}$) was determined and plotted vs N. An extrapolation of this plot to $\bar{I}$ equals zero indicated the loading at which the core might be expected to become critical with these two control rods withdrawn. Control rods CR-1 and CR-6 were then fully inserted and rod withdrawal followed the sequence shown in Table III, until the core was brought critical.

TABLE III
SEQUENCE OF ROD WITHDRAWAL

Step	0% Withdrawn	50% Withdrawn	100% Withdrawn
A	Gangs 1, 2, 3, 4, 5		
B	Gangs 1, 2, 3, CR-9 and CR-12		Gang 5, CR-6
C	Gangs 4, 5	Gangs 1, 2, 3	
D		Gangs 1, 2, 3, 4, 5	
E		Gangs 4, 5	Gangs 1, 2, 3
F		Gang 5	Gangs 1, 2, 3, 4
G	Gangs 1, 2, 3, 4, 5		

The reactivity worth of an addition of fuel was considered equal to the reactivity corresponding to the difference in the critical positions of the control rods before and after adding the fuel increment. The excess reactivity worth of the fuel increment loaded was determined by calibrating this section of control rod(s) by the differential period technique. After determining the reactivity worth of a fuel loading increment, total excess reactivity (ρ_{ex}) was plotted vs the total number of fuel elements loaded (N).

The excess reactivity loaded was determined by summing the incremental worths of each loading. The last two points on this plot were extrapolated linearly to determine how many fuel elements could be loaded (1% reactivity or less) in the next increment.

It was recognized that the summing of the incremental worths, to determine total excess in the reactor, was not proper and that corrections had to be made for core size and arrangement. However, summing the values was a conservative approach and the corrections, which were necessarily theoretical, could be made at any subsequent time. Therefore, the method of summation was utilized in developing a value for the total excess loading in the reactor during the performance of the test.

The criticality extrapolations were utilized to load ten increments of 3, 6, 6, 6, 9, 9, 9, 9, 15 and 15 fuel elements, respectively. Table II also lists the individual elements that were loaded during each increment, including the core channels occupied. Reactor symmetry was achieved with each loading increment, at least to the extent of loading an equal number of elements into each 120 degree sector of the reactor.

After the fuel loading had been completed, dummy elements were placed in the unoccupied core channels to complete the core loading. The reactivity change due to addition of the dummy elements was also measured.

In addition, sodium draining experiments were performed at fuel loadings of 68, 83, 110, 125 and 140 fuel elements, to experimentally determine reactor shutdown margin and worth of sodium.

A source relocation test was performed as part of AI-PC-710. Following the reactor sodium draining test at 68 elements, the neutron source was moved from its normal position (C-4) to the periphery of the then active core (C-104) and a fuel element was placed in core position C-4. The reactivity was determined from differences in critical control rod positions. Following this, the source was returned to C-4 and the fuel loading continued.

C. COUNTING PROCEDURE

Indicated in the detailed, step-by-step procedure were certain counting rate statistical accuracies, such as 0.5%, 1%, 2%. The statistical accuracy in percent was found from the following formula:

$$\frac{1}{\sqrt{\text{total counts}}} \times 100 = 1\% \text{ statistical accuracy}$$

Thus, for a statistical accuracy of:

0.5%, 4×10^4 counts are required.

1%, 10^4 counts are required.

2%, 2.5×10^3 counts are required.

5%, 400 counts are required.

A 1% statistical accuracy was obtained for most measurements used in obtaining the inverse count rate.

The counting rates on the incore and source range instruments were determined at specified times during the loading procedure.

1. Sub-critical

After each addition of fuel with all rods inserted, the inverse of the average count rate ($\bar{I}$) of the three fission chambers was determined and plotted vs the number of fuel elements (N). Extrapolations of $\bar{I}$ to zero indicated the number of fuel elements required to attain criticality with all rods in and, thus, provided an indication of the shutdown capabilities of the control rods.

2. Period

The reactor was made super-critical after each loading of excess fuel, to obtain positive periods and, consequently, to determine the excess reactivity of each loading of excess fuel. The periods were determined using scalers with automatic printouts.

With counting rates of 1.36×10^5 cpm or greater, resolution time errors (0.5%) are introduced and count rates must be corrected using:

$$C_a = \frac{C_m}{1 - \tau C_m}$$

where,

C_a = adjusted or corrected count rate

C_m = measured count rate

τ = resolution time of the circuitry
(3.67×10^{-8} minute)

Periods were initiated from equilibrium count rates of about 500 cpm. The fission chambers were positioned at approximately eight feet above the core centerline to allow sufficient range for flux increases to provide adequate information for accurately determining reactor periods. (Refer to Test Completion Report AI-P-1158, Wet Critical Test, for a discussion of the chamber re-positioning and of the incore nuclear instrumentation in general.) The ICNI printout recorded the total count for 10-sec intervals. The approximate period was found from the following expression:

$$T = \frac{t_d}{\ln 2} = \frac{t_d}{0.693} = 1.443 t_d \quad (1)$$

where,

T = reactor period

t_d = doubling time

The period is related to the reactivity change by the inhour equation:

$$\rho = \frac{\lambda}{(T) k_{eff}} + \sum_{i=1}^m \frac{\beta_i}{1 + \tau_i T} \quad (2)$$

where,

ρ = reactivity

k_{eff} = effective multiplication factor

λ = neutron lifetime

m = number of delayed neutron groups

β_i = delayed neutron fraction for i^{th} group

T = reactor period

λ_i = the decay constant for the i^{th} group

A tabulated list of reactivity (in cents vs period) is available in Part D, Volume IV of the Operations Manual. This list was used to convert ratio of total counts during an asymptotic period directly to reactivity without first calculating the period and without plotting. Reactivity in cents (ρ cents) is related to reactivity in % $\frac{\Delta k}{k}$ as follows:

$$\rho \text{ cents} \times (.00693) = \frac{\Delta k}{k} \text{ percent}$$

Data for calculating periods were obtained at flux levels where nuclear heating does not yet exist (point A).

D. CONTROL ROD CALIBRATION

Control rod increments were calibrated by the differential period technique, as the core was loaded toward the operational loading, to determine the worth of each incremental fuel loading.

An example best explains the technique. With 56 fuel elements in the core, the control rod positions at criticality, with the neutron flux at or above point B (the minimum critical flux level) were; gangs 1, 2, 3 and 4 fully withdrawn and gang 5 at 88.1 inches (X_1). With 62 fuel elements and the neutron flux at or above point B, the control rod positions at criticality were; gangs 1, 2, 3, and 4 fully withdrawn and gang 5 at 48.55 inches (X_2). Thus, the total reactivity change resulting from the addition of the 57th, 58th, 59th, 60th, 61st and 62nd fuel elements was equivalent to the reactivity worth of gang 5 from X_2 to X_1 (ΔX). To find the reactivity associated with these six fuel elements it was necessary to calibrate gang 5 from X_2 to X_1 (ΔX).

First gang 5 was withdrawn from 48.55 to 52.0 inches, which produced a period of approximately 80 seconds. The reactivity associated with an 80 second period is 11.4 cents. Thus, the reactivity worth of gang 5 from 48.55 to 52.0 inches is 11.4 cents. The flux level was then brought to approximately point B again with gang 5 about one-half inch below its position, (about 51.5

inches). To accomplish this, gang 2 had to be inserted. Gang 5 was then withdrawn to 52.0 inches again and a long (590 sec) period obtained. After this period was measured, the flux was dropped below point B again, and a short (41 sec) period obtained by pulling gang 5 to approximately 56.5 inches. The reactivity associated with a 590 second period is 2.0 cents; with a 41 second period 18.2 cents. Thus, the reactivity worth of gang 5 from 52.0 inches to 56.5 inches is 16.2 cents (18.2 minus 2.0). The reactivity worth of gang 5 from 48.55 inches to 56.5 inches is 29.6 cents (18.2 plus 11.4). This long and then short period technique was continued until the desired portion of gang 5 was calibrated (from 48.55 to 88.1 inches in this example), and the reactivity worth of the fuel increment determined. This is the differential period technique of determining reactivity. For a detailed explanation of how period data are reduced to reactivities by the "Ratio Method", see HNPF Operations Manual NAA-SR-7500, Volume IV, Section D.

E. CONTROL ROD MANIPULATION

The control rods are grouped in five gangs consisting of 6, 6, 3, 3, and 1 rod in gangs 1, 2, 3, 4, and 5, respectively. Table IV lists the respective gang core position. Figure 710-2 shows a cross-sectional view of the reactor indicating the positions of the core components and process channels.

As the reactor loading progressed to the operational loading, the number of control rods that could be withdrawn without making the reactor supercritical decreased. The required ICNI data which could not be attained under the prevailing conditions were therefore eliminated and the plots discontinued. The two-rod, "no-critical" requirement and associated graph were available throughout the test, however, and a minimum of this plot and the "rods in" plot were maintained.

The control rods were operated as shown in Table III (preceding). The inverse average count rate ($\bar{I}$) was determined and plotted for steps A, B, and C. One plot was maintained for each of these control rod configurations. $\bar{I}$ vs the total number of fuel elements loaded (N).

NORTH

NOTE: Core positions prefixed by "C" have removable plugs in loading face shield

M-3 is the instrumented moderator can

LEGEND

- Control Rod
- ⊙ Source
- ◻ Sodium Level Measuring Element
- ▲ Sodium Temperature Measuring Element
- ⊙ Reflector Filler
- ⊕ In-Core Fission Chambers

Figure 710-2. Reactor Loading Face Shield Core Component and Process Channel.

TABLE IV
CONTROL ROD POSITIONS

Gang No.	Rod No.	Core Position
1	14	C- 78
	15	C- 86
	16	C- 94
	17	C-102
	18	C-110
	19	C-118
2	5	C- 38
	7	C- 44
	8	C- 50
	10	C- 56
	11	C- 62
	13	C- 68
3	2	C- 20
	3	C- 28
	4	C- 36
4	6	C- 16
	9	C- 24
	12	C- 32
5	1	C- 1

III. ANALYSIS AND PRESENTATION OF RESULTS

A. GENERAL

Several aspects of the wet excess fuel loading test required consideration throughout the analysis of the test. Among these is the source separation utilized during the loading increment calibrations (mentioned previously as a part of II. B). This and other items are discussed here from a qualitative standpoint to alleviate repetition during the subsequent description of analyses performed and results determined. A summary of conclusions resulting from analysis of all the test data obtained in Part III. P, Conclusions.

1. Source Separation

The ICNI channels respond with a count rate of approximately 6×10^5 cpm at criticality, when the chambers are positioned to give adequate indication of neutron population when the reactor is shutdown (15 - 25 cpm). The upper limit of the instruments is 1×10^7 cpm with an automatic scram at 9×10^6 cpm and an alarm at 7.5×10^6 cpm. Thus, the allowable range for period measurements was just slightly over one decade on the instruments with the instruments positioned as described above. It was, therefore, desirable to withdraw the antimony capsule portion of the source from within the beryllium sleeve to decrease the count rate at criticality. With the source separated, periods could be run from approximately 500 cpm (Point "B", minimum critical flux level) to 120,000 cpm (Point "A" beginning of nuclear heating, as indicated on the ICNI). This represents approximately 2.5 decades of range, which is sufficient for period measurements. Reactor shutdown required re-insertion of the source, of course, to give indication of the neutron population with rods inserted.

As the source was separated (at criticality), it was necessary to withdraw control rods to compensate for the manipulation, prior to 110 fuel elements. Subsequent to the 110 element loading, it was necessary to insert control rods to compensate for source separation. This phenomenon is felt to be due to the relative importance of the source neutrons becoming less significant than the poison effect of the source holder element, as the loading progressed and/or due to the axial flux peaking which resulted from the manner in which the reactor was loaded. Certainly, the importance of the center region of the reactor was decreased as fuel elements were loaded. With a small loading of elements in the reactor, the positive reactivity due to the source neutrons was relatively significant. It was, therefore, necessary to withdraw rods to insert reactivity in order to hold the reactor level constant with the source separated. The relative contribution of the source neutrons decreased to being equal to the poison removal effect somewhere between the 101 and 110 fuel element loadings. Thereafter, their contribution relative to the total reactor was less and it became necessary to insert

rods to hold the reactor critical, thus compensating for the removal of the poison contained in the source holder element.

The determination of the worth of each loading increment was determined from the "source-in" to "source-in" critical position. Table V lists the rod movements required to maintain criticality at the various fuel loadings upon separation of the source.

TABLE V
ROD MOVEMENTS REQUIRED BY SOURCE SEPARATION

No. of Elements Loaded	Critical Rod Gang	Position With Source Together	Position With Source Separated	*Required Rod Movement
53	5	129.1	122.0	-7.1
56	5	90.6	88.1	-2.5
62	5	49.95	48.55	-1.40
68	4	115.0	113.3	-1.7
74	4	84.6	84.0	-0.6
83	4	57.3	57.2	-0.1
92	4	34.75	34.4	-0.35
101	3	111.3	110.85	-0.45
110	3	76.9	76.95	+0.05
125	3	32.0	33.7	+1.7
140 (1)	2	120.5	121.3	+0.8
140 (2)	1	73.85	73.95	+0.1

* Plus indicates rod withdrawal

Minus indicates rod insertion

(1) Fixed rods in core.

(2) Fixed rods out of core

2. Temperature Corrections

The bulk sodium temperature in the primary heat transfer system was maintained at $350\text{ F} \pm 5\text{ F}$ during the test performance. On limited occasions, the spread increased to $\pm 10\text{ F}$, but was always corrected shortly thereafter. No corrections were required of the counting data obtained to compensate for these variations because of the following considerations: The temperature coefficient associated with the clean, initially loaded core is calculated to be $0.34\text{ cents per degree F}$ ($2.13 \times 10^{-5} \frac{\Delta k}{k}$ per degree F

measured in the dry zero power experiments, plus 0.59×10^{-5} added for sodium). Hence, the temperature swings previously described could be worth only ± 3.4 cents in the worst case (± 10 F). Further, it is quite reasonable to assume that, due to the length of time involved in the loading operations, the positive and negative quantities of these small variations would be equal.

Subsequent to this test, the temperature coefficient was measured as a part of the post critical testing program and was determined to be only about 0.1 cent per degree F. This lessened the errors imposed by the temperature variations described above.

3. Summing of Measured Excess Reactivities

The original method for determining total excess reactivity in the reactor was to sum arithmetically the reactivity values determined for each loading increment. This method was utilized throughout the loading, but with increasingly less reliability as the test progressed. Analysis of the data verified that this method introduced two errors. One was due to flux perturbations created by the insertion of the central control rods as the loading progressed. This tended to peak the flux at the outside of the core, thus, the measured (apparent) worth of each additional excess loading increment was greater than expected. The second error resulted from the enlarging of the active core as fuel was loaded, thus reducing the actual worth of each loading increment as succeeding increments were loaded.

A plot of k_{eff} vs the factor proportional to radial buckling demonstrated that the errors were cumulative. The first six loading increments determined a straight line for this plot, as was expected. Figure 710-3 shows how the summed reactivities continually diverged from the expected straight line values after the sixth loading increment. Reading the k_{eff} for 140 fuel elements on the expected curve and transposing to excess reactivity gives the expected value as 7.55 percent. Figure 710-4 shows the originally calculated excess reactivity vs number of fuel elements curve and the summed excess reactivities that were measured.

Plots of the incremental rod calibrations performed to measure the excess in each loading increment were maintained throughout the test.

Figure 710-3. K_{eff}^{-1} vs Factor Proportional to Radial Buckling

Figure 710-4. Per Cent Excess Reactivity vs Number of Fuel Elements

These graphs had the familiar rod integral worth curve shape when the incremental rod calibrations were connected. To recall, a specific reactor loading had an associated critical rod position on one gang of rods. Addition of fuel resulted in a new critical position. The rods were then calibrated over the change in critical rod position and the excess reactivity determined was the worth of the fuel elements loaded. Thus, the total gang was calibrated from 156 inches (full out) to zero inches (full in). Making a single plot of the increments, such as in Figure 710-5, develops an approximate rod integral worth curve. It was known throughout the test that such plots were not accurate. It is improper to sum the rod incremental worths since they were obtained with different core configurations.

4. Flux Shapes and Rod Configurations

Data obtained by various techniques during the test, with respect to rod worths and sodium worth, were not consistent. As the reactor loading approached the operational loading of 140 fuel elements, it was obvious that the choice of rod configurations for determining sodium and rod worths was very crucial, and that slight variations had large effects on the measured values.

Shadowing of control rods was definitely occurring and affecting test results. The insertion of any three adjacent gang 1 and gang 2 rods resulted in a decreased worth of those rods and an increased worth of the rods across the reactor. The conclusion drawn from these results was that rod calibrations by intercomparison was not meaningful.

At the 83, 110, 125, and 140 element loadings, all rods were banked for measurement in an attempt to develop a two region reactor which was compatible with theoretical calculations. Resulting flux distortions required such significant corrections that no real validity could be attached to the values derived. A description of some of the analyses performed on data obtained under these conditions is contained in sections III. H, K, and L. The all rods banked configuration did, however, develop qualitative information with respect to shutdown margin. A plot of this configuration vs number of fuel elements in the drained reactor was an excellent indication

Figure 710-5. Approximate Gang 5 Integral Worth Curve Based on Worth of Successive Loading Increments.

of shutdown margin in terms of inches of rod yet to be utilized, and also of the trend the shutdown margin was following as the loading progressed. Figure 710-6 shows the banked rod configuration data obtained for both the drained and filled reactor.

It became evident as the test progressed that both radial and axial flux distortions were being created by the various rod configurations. The flux can best be described as "quite mobile" and defies accurate measurement. This does not create any problem for the operating reactor, as the control rods will be positioned to develop minimum flux distortions and undergo little movement. The flux peaking and depressions that result from rod manipulations are an unfortunate circumstance that arises when rods are necessarily manipulated in attempted experiments.

The major conclusions drawn from analysis of the above situations are that:

- 1) The control rods must be operated as gangs (split gangs in the case of gangs 1 and 2, as long as adjacent rods are oppositely positioned) for both normal power operation and experimental measurements.
- 2) Meaningful rod calibrations can be performed but only with concentrically adjacent banks serving as the compensating rods.
- 3) All rod worth determinations must include a description of all non-involved rod positions, to allow evaluation of control rod shadowing in the configuration utilized for calibration.

Sodium worth could not be accurately determined until the operational loading of 140 fuel elements was reached. Accurate rod gang calibrations could be performed at this time which were required for determining the incremental worth of the sodium as it was drained. The maximum reactivity worth that can be associated with the sodium is 4.2 percent, the minimum 3.6 percent. Attempts to measure sodium worth prior to the full core loading resulted in inconsistent results, as will be discussed in the 68, 83, 110 and 125 element loading descriptions.

Figure 710-6. All Rods Banked Critical Position vs Number of Fuel Elements

B. COUNT RATE DATA

One plot of average inverse count rate vs fuel elements loaded for each of the following control rod configurations was maintained during the loading, as required by the test procedure.

- 1) All rods in
- 2) CR-1 and CR-6 withdrawn
- 3) All rods 50% withdrawn

Plot number 3) had to be discontinued at the 68 element loading, as positioning all rods at 50% withdrawn resulted in a positive reactor period.

CR-1 and CR-6 were completely withdrawn following the loading of each fuel increment, verifying that the reactor could not be made critical on the two most reactive rods, thus complying with HNPF Technical Specifications. Figure 710-7 displays the control curves developed by the count rate data obtained. No analysis of this information was made, other than that required to utilize the curves to denote proper fuel loading progress.

C. 53 ELEMENT LOADING

Initial criticality was attained in the sodium filled reactor at the 53 element loading with all rods withdrawn, except gang 4, which was at 134.0 inches. (Refer to test completion report AI-P-1158, Wet Critical Test.) The critical rod configuration was shifted to gang 5 controlling at 129.1 inches with gang 4 withdrawn, to initiate wet excess fuel loading test measurements.

Gang 5 was withdrawn to full out (156.0 inches) and a reactor period of 55.16 seconds was measured. This period corresponds to a reactivity of 14.909 cents. It was thus established that the 53 element loading contained an excess reactivity of 14.909 cents or 0.1035 percent. This was the first point plotted on the following loading curves:

$K_{\text{eff}} - 1$ vs factor proportional to radial buckling (Figure 710-3)

Percent ρ_{ex} vs number of fuel elements (Figure 710-4)

Figure 710-7. Inverse Count Rate ($\bar{I}$) vs Number of Fuel Elements

K_{eff} was determined by the following relationship:

$$K_{\text{eff}} = \frac{1}{1 - \rho_{\text{ex}} (\%)}$$

The buckling factor was derived from the following:

$$\beta_n = \frac{10^5}{(R + S)^2}$$

where,

$$S = 22 \text{ cm}$$

$$R = \sqrt{NA/\pi}$$

N = Number of fuel elements

$$A = 870 \text{ cm}^2$$

NOTE

For loading above 95 fuel elements, the value for "A" changes to 725 cm^2 as discussed in Part II. A.

D. 56 ELEMENT LOADING

The loading of the 54th, 55th, and 56th elements resulted in a change in gang 5 critical position from 129.1 inches to 90.6 inches. The calibration of this increment of rod determined that these three fuel elements were worth 69.636 cents in reactivity. The addition of this amount and the previously measured 14.909 cents established the total excess reactivity in the reactor at the 56 element loading as 84.545 cents. This point was plotted on the excess loading curve and allowed the first extrapolation to determine the number of fuel elements that could be loaded in the next increment. The criterion of linearly extrapolating the last two measured points to one percent greater than the last point, to determine the maximum number of fuel elements to be loaded in the next increment, was used throughout the test. In this instance, the extrapolation showed six more fuel elements could be loaded (Figure 710-4).

Extrapolation of the two excess loading curve points in the negative direction (to the excess equals zero point) also gives a better definition of where minimum criticality would have occurred. This extrapolation indicated that criticality would have occurred with all rods withdrawn at 52.4 fuel elements.

The first approximation of sodium worth was made at this time. Excess reactivity in the reactor at this loading was 84.545 cents, as a result of loading 3.6 fuel elements in addition to those required for minimum criticality. This gives a worth of approximately 0.16 percent for each of the elements loaded. Wet criticality was achieved with 21 more fuel elements in the reactor than were required for dry criticality. The first excess loading was immediately adjacent to the wet criticality elements. Hence, the worths of the elements must be similar. Thus, the 21 additional elements represent approximately 3.4 percent reactivity. This value, it was found later, was somewhat low, but represented a good initial value for sodium worth. Final sodium worth determined at the 140 element loading was approximately four percent.

E. 62 ELEMENT LOADING

Calibration of gang 5 from 49.95 inches to 90.6 inches was required to determine the excess reactivity associated with the six elements loaded to bring total reactor loading to 62 fuel elements. The excess reactivity measured was 138.848 cents, or 1.548 percent. This point was plotted on Figure 710-4, the one percent extrapolation made, and the next loading determined to be six elements.

F. 68 ELEMENT LOADING

1. Excess Reactivity Determination

Critical rod positions for the 68 fuel element reactor loading were: gang 5 full in, gang 4 at 115.0 inches, and gangs 1, 2, and 3 full out. The loading of the six elements in the third increment was therefore equivalent in worth to gang 5 from 49.95 inches to full in and gang 4 from 156.0 inches to

115.0 inches. These sections of rods were calibrated by the differential period technique, again compensating with gang 2. The excess reactivity measured was 146.561 cents (1.016 percent). Total reactivity in the reactor was 369.954 cents. As shown in Figure 710-4, the summed excess reactivities were beginning to deviate slightly from the calculated loading curves. Some effort was expended at this time on determining the proper correction factors to apply to the excess reactivity determinations (as discussed previously in Part III. A. 4). The next loading was again extrapolated to be six fuel elements. It was decided that the 68 element loading was the appropriate place to perform the source relocation test and the first sodium draining experiment.

2. Sodium Draining Test

The first sodium draining test was initiated at the 68 element reactor loading. The first step of the procedure, which required a calibration of gang 5 in the wet core, was waived in favor of the approximate worth curve developed by summing the fuel loading increments (Figure 710-5). The final section of the curve had been supplied by this loading and it was felt that the information contained in the worth curve would not be significantly changed by the calibration data. This line of thinking was later proven to be in error.

Intercomparison of gangs 1 and 2, and of each rod in gangs 3 and 4, with the "calibrated" gang 5 in the wet core gave the rod worth results as shown in Table VI. The measured 1781 cents is equivalent to 12.34 percent.

The apparent discrepancy in the worth of the individual gang 3 and gang 4 rods can be attributed to the poison effect of the source element. The three rods which are lower than the average 187 ± 2 cents rod worth are that amount lower corresponding to their position relative to the source. CR-12 is the closest and sees a worth of 29 cents less. CR-3 and CR-4 are further away from the source element and see a lesser deviation (20 cents) from the average. Considering the source element, the measured rod worths are in excellent agreement with each other.

TABLE VI
ROD WORTHS AT THE 68 ELEMENT LOADING

Gang	Control Rod	Individual Worths	Gang Worths	Relative Worths
5	CR-1	323 cents	323 cents	1.0
4	CR-6 CR-9 CR-12	185 cents 186 cents 158 cents	529 cents	1.64
3	CR-2 CR-3 CR-4	189 cents 168 cents 166 cents	523 cents	1.62
2	All 6 rods	302 cents	302 cents	0.93
1	All 6 rods	104 cents	1104 cents	0.32
Total Rod Worth			1781 cents	

Table VII compares the relative worths of the rods at the 68 element loading and the predicted values for a wet critical reactor of 50 elements. The existing difference is probably due to the difference in reactor size. In the 68 element reactor, fuel is closer to gangs 1 and 2, thus, increasing the flux in that region. Thereby, the relative worth of gangs 1 and 2 is increased at the expense of gangs 3, 4, and 5.

TABLE VII
COMPARISON OF RELATIVE ROD WORTHS AT THE CALCULATED
CRITICAL LOADING OF 50 ELEMENTS AND THE 68 ELEMENT LOADING

Gang	Predicted Relative Worths at 50 Elements	Measured Relative Worths at 68 Elements
5	1.0	1.0
4	1.7	1.64
3	1.7	1.62
2	0.6	0.93
1	0.2	0.32

Critical rod positions at the start of the reactor draining were: gangs 1, 2, 4 and 5 full out, and gang 3 at 58.4 inches. As poison was removed from the reactor by the sodium draining, gang 5 was inserted to maintain criticality. The draining was stopped at numerous levels and the gang 5 critical position obtained precisely. The worth of the sodium drained at any of these levels was considered to be equivalent to the change in critical position of gang 5. Gang 5 incremental worth was obtained from Figure 710-5; the worth curve developed from the summation of the loading increments. Table VIII lists the sodium levels at which the critical rod position was obtained precisely, gang 5 position, incremental gang 5 and/or sodium worth, and total worth to that level.

TABLE VIII
SODIUM WORTH DATA - 68 ELEMENT LOADING

Sodium Level	Elev (ft. - in.)	Gang 5 Position (inches)	Increment Worth (cents)	Total Worth (cents)
6' above mod cans	1425' - 0"	156.0	0	0
3' above mod cans	1422' - 0"	152.7	1	1
Top of mod cans	1419' - 0"	140.7	4.5	5.5
Top of active core	1417' - 0"	135.0	4	9.5
Core $\mathbb{C}$ plus 5' - 4"	1415' - 7"	128.5	5.5	15
Core $\mathbb{C}$ plus 4' - 0"	1414' - 3"	109.0	26	41
Core $\mathbb{C}$ plus 2' - 8"	1412' - 11"	85.35	56	97
Core $\mathbb{C}$ plus 1' - 3"	1411' - 6"	55.3	107	204
Active core centerline	1410' - 3"	0 *	137	341

* Gang 4 also inserted to 133" to maintain critical

The test was prematurely terminated due to operational difficulties, when the sodium level was at the core centerline. Figure 710-8 is the sodium worth curve obtained using the results of this initial draining test.

3. Source Relocation Test

The following critical control rod configurations were obtained prior to relocating the source:

<u>Case I</u>		<u>Case II</u>	
Gang 1	out	Gang 1	out
Gang 2	out	Gang 2	out
Gang 3	77.7 inches	Gang 3	123.9 inches
Gang 4	77.7 inches	Gang 4	123.9 inches
Gang 5	out	Gang 5	in

The reactor was shutdown and the source moved to core position C-104. C-104 was a core position just outside the fueled part of the partially loaded reactor. A fuel element was inserted in the source core position (C-4), bringing the total reactor loading to 69 fuel elements. Criticality was then achieved with the following rod configurations:

<u>Case I</u>		<u>Case II</u>		<u>Case III</u>	
Gang 1	out	Gang 1	out	Gang 1	out
Gang 2	out	Gang 2	out	Gang 2	out
Gang 3	66.7 inches	Gang 3	104.7 inches	Gang 3	77.7 inches
Gang 4	66.7 inches	Gang 4	104.7 inches	Gang 4	77.7 inches
Gang 5	out	Gang 5	in	Gang 5	56.05 inches

Comparison of Case I before and after the source and fuel element exchange shows that with gang 5 out, the exchange required the insertion of gangs 3 and 4 a distance of 11 inches to maintain criticality. Case II demonstrates that the required rod movement was 19.2 inches with gang 5 in. The worth of gangs 3 and 4 at 120 inches is, of course, considerably less than at 70

Figure 710-8. Sodium Worth vs Elevation of Sodium

inches. Comparison of the "before" Case I and Case III shows that the manipulation required the insertion of gang 5 from 156.0 inches to 56.05 inches or a distance of 99.95 inches.

Gangs 3 and 4 were calibrated from 66.7 inches to 77.7 inches with gang 5 out and compensating with gang 1. The reactivity change associated with this calibration was 122.368 cents. Gangs 3 and 4 were calibrated from 104.7 inches to 123.9 inches with gang 5 in, again compensating with gang 1. The worth of the source-fuel element manipulation was determined to be 69.648 cents by this measurement. It was concluded that the decrease in apparent worth of the fuel element installed in C-4 was due to the shadowing effect of gang 5 when it is inserted.

Utilizing Figure 710-5 to obtain the worth of the gang 5 position change, as recorded in Case III and the "before" Case I, the manipulation was worth 201 cents. This was the first definite indication that the gang 5 worth curve obtained by summing the worth of the fuel loading increments was not valid. It was not unexpected, as core size had increased appreciably from the starting point of 53 fuel elements. Reviewing the dry excess testing data showed the gang 5 worth to be even greater in the smaller dry core (3.8 percent @ 42 fuel elements).

G. 74 ELEMENT LOADING

Loading of the reactor from 68 to 74 elements resulted in an excess reactivity contribution of 107.804 cents, measured by calibrating gang 4 from 84.6 inches to 115 inches, (the amount of rod insertion required to compensate for loading the six elements). Gang 5 was fully inserted during the calibration, gangs 1 and 3 withdrawn, and gang 2 compensating. Calibration of this loading increment allowed continuation of Figure 710-9, which was initiated the previous loading, when it was necessary to insert gang 4 to 115 inches to maintain criticality.

The excess reactivity determined for this loading was plotted on the excess loading graphs Figures 710-3 and -4. The procedural one percent extrapolation indicated nine fuel elements could be loaded as the next

Figure 710-9. Approximate Gang 4 Integral Worth Curve Based on Worth of Successive Loading Increments.

increment. This point caused the first significant "break" in the linear loading curve (Figure 710-4) and demonstrated that the loading was progressing similarly to that calculated. The buckling plot (Figure 710-3) continued on the same curve.

H. 83 ELEMENT LOADING

1. Excess Reactivity Measurements

Gang 4 critical position changed from 84.6 inches to 57.3 inches with the increase to 83 fuel elements in the reactor (gangs 1, 2, 3, out; gang 5 fully inserted). Calibration of this section of gang 4 determined the excess reactivity increase to be 150.719 cents, bringing the total excess reactivity to 628.471 cents (4.36 percent). Gang 2 was used to compensate during the calibration. The one percent extrapolation showed that nine more elements could be loaded (Figure 710-4).

2. Sodium Drain Test

Gang 5 was calibrated by the differential period technique and an intercomparison with all other rods was performed prior to draining the reactor at the 83 element loading. Total worth of gang 5 was 224.63 cents. All other rods were banked during the calibration and period compensation was accomplished by the total bank. Figure 710-10 shows the integral worth curve developed for gang 5.

The choice of all rods banked as the compensating gang was based on attempting to develop a two-region core which would lend itself to analytical calculations and determination of correction factors. It was subsequently determined that the all rods banked configuration resulted in significant axial flux peaking under the bank and depression above the bank. Also, the bank moved from 86.0 inches to 66.6 inches as compensation during the calibration of gang 5. Developing proper correction factors was thus made very difficult. Repeating the "measuring with banked rods" technique at the 110 element loading ultimately resulted in shifting the criteria for loading fuel. Dry core shutdown margin measurements were utilized to govern

Figure 710-10. Gang 5 Integral Worth Curve at 83 Elements (Calibration with all other rods banked; the bank moving from 86.0" to 66.6" wet and 43.9" to 34.9" dry)

loading progress, as opposed to determination of rod worths and excess reactivity in the reactor. This shift of emphasis will be discussed in more detail at the 110 fuel element loading.

Table IX lists the data from intercomparison of each of the rods with calibrated gang 5 prior to draining. The difference in the worths of the individual gang 3 and gang 4 rods is again attributed to the source element. The consistency in the deviation between the 68 and 83 element loadings substantiates the original theory of the source element creating the disturbance. The variation in gang 1 rod worths is due to CR-15, -17, and -19 being surrounded by more fuel than the partially covered rods CR-14, -16, and -18.

Total gang worths and relative worths are listed in Table X. The 68 element loading relative worths and predicted wet critical relative worths are also listed to allow comparison.

Sodium draining was started with the reactor critical and all rods at 65.0 inches, except gang 2, which was fully withdrawn to act as safety rods. The removal of poison (sodium) from the reactor was compensated for by the insertion of gang 5. When gang 5 reached nearly full in, the draining was stopped and gang 5 was again withdrawn to the all rods banked position (except gang 2), inserting the bank to maintain criticality. Sodium incremental worths were determined according to gang 5 position at the various sodium levels. Total worth was obtained by summing the increments. Final bank position was 33.95 inches, except gang 2 which was still full out. Table XI lists the draining results.

Following the draining of sodium, all rods (including the gang 2 safeties) were banked at 38.7 inches to maintain criticality. Gang 5 was then re-calibrated to determine what effect draining the sodium had on its worth. Similarly, intercomparisons were made with all the other rods. The all rods banked technique was employed in these measurements, with the bank being at 38.7 inches initially.

The worth of gang 5 in the dry core was determined to be 293.2 cents, appreciably higher than that of the wet core. Figure 710-10 shows both the wet and dry integral worth curves. Using the dry calibration integral worth curve for determining the incremental sodium worth during draining

TABLE IX
ROD WORTHS AT THE 83 ELEMENT LOADING

Gang	Control Rod	Worth Above Bank * (cents)	Worth Below Bank * (cents)	Total Worth (cents)
5	CR - 1	36	189	225
4	CR - 6	26	127	153
	CR - 9	26	127	153
	CR-12	23	122	145
3	CR - 2	28	131	159
	CR - 3	26	122	148
	CR - 4	26	121	147
2	CR - 5	7	41	48
	CR - 7	7	42	49
	CR - 8	8	44	52
	CR-10	7	40	47
	CR-11	7	38	45
	CR-13	7	38	45
1	CR-14	2	16.5	18.5
	CR-15	5	28	33
	CR-16	2	18	20
	CR-17	4	30	34
	CR-18	2	16	18
	CR-19	4	26	30
Total (cents)		253	1316.5	1569.5
(%)		1.75	9.12	10.87

* Bank position was 68.6 inches

TABLE X
RELATIVE ROD WORTHS AT WET CRITICAL,
68 ELEMENT, AND 83 ELEMENT LOADINGS

Gang	83 Element Worth	83 Element Relative Worth	68 Element Relative Worth	Predicted Wet Critical Relative Worth
5	225	1.0	1.0	1.0
4	451	2.0	1.64	1.7
3	454	2.0	1.62	1.7
2	286	1.3	0.93	0.6
1	153.5	0.7	0.3	0.2

TABLE XI
SODIUM WORTH DATA 83 ELEMENT LOADING

Sodium Level	Elev.	Gang 5 Position (inches)	Bank Position (inches)	Incremental Worth (cents)	Total Worth (cents)
6' above Mod Cans	1425' - 0"	65	65	0	0
3' above Mod Cans	1422' - 0"	64.9	65	0	0
Top of Mod Cans	1419' - 0"	64.7	65	1	1
Top of Active Core	1417' - 0"	64.1	65	2	3
Core $\mathbb{C}$ plus 5' 4"	1415' - 7"	63.9	65	4	7
Core $\mathbb{C}$ plus 4' 0"	1414' - 3"	63.4	65	5	12
Core $\mathbb{C}$ plus 2' 8"	1412' - 11"	61.0	65	11	23
Core $\mathbb{C}$ plus 1' 3"	1411' - 6"	55.4	65	29	52
Active Core $\mathbb{C}$	1410' - 3"	42.7	65	82	134
Core $\mathbb{C}$ minus 1' 5"	1408' - 10"	5.2	65	179	313
Core $\mathbb{C}$ minus 2' 4"	1407' - 11"	10.2	56.75	149	462
Core $\mathbb{C}$ minus 3' 2"	1407' - 1"	8.7	51.0	132	594
Core $\mathbb{C}$ minus 3' 11"	1406' - 4"	2.5	46.15	114	708
Core $\mathbb{C}$ minus 4' 9"	1405' - 6"	1.0	41.7	96.5	804.5
Core $\mathbb{C}$ minus 5' 6"	1404' - 9"	8.2	37.55	75	879.5
Bottom of Active Core	1403' - 6"	14.45	33.95	53	932.5
Bottom of Mod Cans	1401' - 0"	13.85	33.95	1	933.5

shows the sodium worth to be 8.45 percent. These values are, of course, much higher than expected and established that further analyses were required. Figure 710-11 depicts the sodium worth curve utilizing the calibration of gang 5 prior to draining.

Intercomparison of all the rods in the dry core with the calibrated gang 5 (dry) gave the results shown in Table XII.

It was concluded from the data obtained that the axial flux distortions due to the all rods banked configuration resulted in higher apparent sodium and rod worth values than really existed. The movement of the bank during the calibrations accented the flux peaking and drove the measured values still higher.

Figure 710-11. Sodium Worth vs Elevation of Sodium

TABLE XII
ROD WORTHS AT THE 83 ELEMENT LOADING, REACTOR DRAINED

Gang	Control Rod	Worth Above Bank	Worth Below Bank	Total Worth
5	CR- 1	135	158	293
	CR- 6	104	100.5	204.5
4	CR- 9	104	104.5	208.5
	CR-12	*102		202
	CR- 2	111	106	217
3	CR- 3	106.5	103	209.5
	CR- 4	105	100	205
	CR- 5	35	36	71
	CR- 7	34	37	71
2	CR- 8	36	37.5	73.5
	CR-10	35	36	71
	CR-11	* 35	40	75
	CR-13	* 35	38.5	73.5
	CR-14	* 12	19	31
	CR-15	* 22	28	50
1	CR-16	12	19	31
	CR-17	22	28	50
	CR-18	* 12	19	31
	CR-19	* 22	26	48
Totals		1079.5	1136.0	2215.5

* Not measured; values derived by extrapolating similar rod values according to previous relative worths.

The only information derived from the draining experiment was that the shutdown margin in the dry core was represented by all rods being at 38.7 inches at criticality. In the event of a scram in the dry core, all nineteen control rods would each fall 38.7 inches to shutdown the reactor.

In an attempt to interpret the data obtained from the wet and dry banked rods configuration, the following was done:

- 1) Differential worth data were developed from the gang 5 dry and wet calibration data.
- 2) The original gang 5 worth curve, as determined by the fuel loading instruments, was considered as most representative of the true flux shape in the reactor with all rods in or out (all other rods were out when the curve was being developed). A flux shape curve was developed

by plotting the square root of the gang 5 differential rod worths vs rod position (considering flux to be proportional to the square root of differential rod worth). This curve was considered as the flux shape in the reactor with all rods in.

- 3) Gang 5 dry and wet curves were similarly developed and superimposed over the "true" flux shape curve. Figure 710-12 displays the resulting curves. The dry and wet curves were normalized to the area under the true flux shape curve before they were plotted. The normalization assumes the areas (rod worth) under the "true" curve to be constant.
- 4) Ratios were then utilized to correct the measured values for shutdown margin. The rod worth under the 38.7 inch bank was measured to be 7.9 percent in the dry core. Thus, 7.9 percent was the measured shutdown margin. Figure 710-12 curves show that if this shutdown margin had been measured with all rods in (flux returned to true shape), the value determined would have been four percent.
- 5) Similarly, the rod worth under the 68.6 inch bank in the wet core represented sodium worth plus shutdown margin. This value was measured to be 9.12 percent. The value under the "rods in" curve shows that the 9.12 percent value would really be 8.8 percent with the reactor shutdown and drained. The difference in the two bank positions represents sodium worth in the dry core. This value can be seen to be 4.8 percent. At this time, the numbers thus established were considered crude but at least indicative of reactor condition.

It was concluded that adequate shutdown margin was available to load the reactor to two percent greater excess. These loadings would still allow the "no-less-than two percent" shutdown margin value to be maintained as required by the loading procedure.

I. 92 ELEMENT LOADING

Criticality was achieved with gangs 1, 2, and 3 full out, gang 5 full in, and gang 4 at 34.75 inch with the reactor loaded to 92 fuel elements. The excess reactivity added to the reactor by increasing the core loading from 83 to 92 elements was determined by calibrating gang 4 from 34.75 inch to 57.3 inch. Gang 4 position at 83 fuel elements was 57.3 inch, all other gangs positioned as described above. Gang 2 was used to compensate during the calibration.

Figure 710-12. Analysis of Rod Worths at the 83 Element Loading - Reactor Drained

The nine elements added to the 83 element reactor resulted in an excess reactivity increase of 141.769 cents. The plotting of this point on the linear excess loading curve (Figure 710-4) shows the curve to continue on the same slope, establishing the next loading as nine fuel elements. The k_{eff}^{-1} vs buckling factor graph (Figure 710-3) displays the continued divergence of total measured excess loaded from the theoretical value.

Figure 710-9 was continued with the data obtained at the 92 element loading. It was apparent that the next loading would complete this curve; gang 4, as well as gang 5, would, hence, be in during measurements made with subsequent fuel loadings.

J. 101 ELEMENT LOADING

Critical rod positions at the 101 fuel element loading were: gangs 1 and 2 full out, gangs 4 and 5 full in, and gang 3 at 111.3 inches. To determine the excess reactivity added by this seventh fuel loading increment, it was necessary to calibrate gang 4 from full in to 34.75 inches (the critical position with 92 elements), and gang 3 from 111.3 inches to full out. These calibrations were accomplished, compensating with gang 2, and an excess of 134.03 cents was obtained.

The linear plot of excess loading vs the number of fuel elements, Figure 710-4, shows this 101 fuel element point to make the curve "break" only slightly. The one percent extrapolation again showed that only nine fuel elements could be loaded in the next increment. The buckling plot continued to diverge, indicating that summation of incrementally determined excess reactivities was improper. This plot also supported the assumption that the linear plot was showing an incorrect result of loading the increment. Considering that the divergence was in the conservative direction (indicating more excess reactivity than really existed) the continuation of the loading, based on the plots as drawn, was safe.

The approximate integral worth curve for gang 4 (Figure 710-9) was completed at the 101 element loading. This plot was a reasonable approximation of gang 4 worth and was utilized in predicting control rod positions

for criticality and period measurements. An ultimate evaluation of the loading increment curves became available at the operational loading (140 fuel elements), following a complete calibration of gang 3. The results of that calibration and the comparison to the loading increment curve is discussed in detail under Section M., 140 Element Loading.

K. 110 ELEMENT LOADING

1. Excess Reactivity Measurements

Gang 3 was positioned at 76.9 inch to achieve criticality at the 110 element loading. Gangs 1 and 2 were full out and gangs 4 and 5 full in. The change of 34.4 inch in gang 3 position was calibrated to determine the amount of excess reactivity contributed by the nine elements contained in the eighth loading increment. Gang 2 was used to compensate during the calibration. The excess added was 96.488 cents.

Addition of the 110 element loading point to the excess loading curves (Figure 710-3 and -4) showed a "break" and approach to the shape of the calculated curve on the linear plot, as was expected. The buckling graph, on the other hand, showed the continued divergence.

The gang 3 approximate integral worth curve, initiated at the 101 element loading (Figure 710-13) was continued to 76.9 inch using the data obtained from the period measurements.

A sodium draining test was set up at the 110 element loading, to verify the required 2 percent shutdown margin. A measured (but uncorrected) 2.6 percent excess had been added since the measurement of 4 percent (minimum) shutdown margin at the 83 element loading.

2. Sodium Draining Test

Prior to performing the sodium draining test at the 110 element loading, the fuel loading criteria were re-evaluated. It was determined that the major requirement was to ensure a two percent shutdown margin. This decision was based on the fact that inconsistent data were being obtained (subsequently determined to be due to the extremely "mobile" quality of reactor

Figure 710-13. Approximate Gang 3 Integral Worth Curve Based on Worth of Successive Loading Increments.

flux) with respect to rod worths, sodium worth, and excess reactivity. Therefore, a direct measurement of shutdown margin was advisable. This was accomplished by calibrating the rods out of the reactor with the reactor drained.

In addition to performing the direct measurement of shutdown margin in the dry reactor, the technique of determining rod worths and sodium worth as used at the 83 element loading, was utilized in the 110 element reactor in an attempt to discover any possible correlations. Gang 5 was not re-calibrated in the wet core as it was at the 83 element loading, because it was felt the worth would not be changed appreciably.

Gang 2 was fully withdrawn (to serve as safety rods) and all remaining rods were banked at 48.7 inches, just prior to the start of sodium draining. The removal of poison by the draining of sodium was compensated for by the insertion of gang 5. The incremental sodium worths were then determined from the gang 5 calibration curve. When gang 5 reached full in, it was withdrawn to the bank position and the bank inserted to maintain criticality. The cycle was then repeated.

When the sodium had been drained to approximately three feet below the core centerline, gang 5 was full in and the bank was positioned at 15.0 inches. It was apparent that the fully withdrawn gang 2 worth was more than two percent. All six rods of gang 2 were then positioned at 100.0 inches, which returned all other rods banked to 29.8 inches to maintain criticality. Draining was then continued until the reactor was dry. Rod positions at that time were; all rods at 15.1 inches, except gang 2 which was at 100.0 inches. Table XIII lists the results of the draining.

The large number of critical positions in Table XIII is due to gang 5 worth below the bank being very small. Each time gang 5 was inserted, it had to be withdrawn and criticality regained with the bank. As a result, in many instances, gang 5 had to be fully inserted to compensate for draining as little as an inch of sodium. Subsequent draining procedures were modified to eliminate the requirement for determining so many critical positions.

Total measured worth of the sodium is higher than the real value, because the 83 element gang 5 calibration curve was used to measure the increment worths. The assumption that gang 5 worth would not be changed

TABLE XIII
SODIUM WORTH DATA 110 ELEMENT LOADING

Sodium Level	Elev. (ft. - in.)	Gang 5 Position (inches)	Bank Position (inches)	Incremental Worth (cents)	Total Worth (cents)
6' above mod. cans	1425' - 0"	48.55	48.7	0	0
3' above mod. cans	1422' - 0"	48.55	48.7	0	0
Top of mod. cans	1419' - 0"	48.4	48.7	1	1
Top of active core	1417' - 0"	48.2	48.7	1	2
Core $\mathbb{L}$ plus 5' - 4"	1415' - 7"	47.75	48.7	1.5	3.5
Core $\mathbb{L}$ plus 4' - 0"	1414' - 3"	46.5	48.7	4	7.5
Core $\mathbb{L}$ plus 2' - 8"	1412' - 11"	42.4	48.7	16.5	24
Core $\mathbb{L}$ plus 1' - 3"	1411' - 6"	31.25	48.7	44	68
Active core	1410' - 3"	16.5	46.3	97	165
Core $\mathbb{L}$ minus 1' - 1"	1409' - 2"	22.4 *	39.5	163	328
Core $\mathbb{L}$ minus 1' - 10"	1408' - 5"	50.8 **	20.0	88.5	416.5
Core $\mathbb{L}$ minus 2' - 10"	1407' - 5"	0	25.3	39	455.5
Core $\mathbb{L}$ minus 3' - 0"	1407' - 3"	8.0	23.4	30	485.5
Core $\mathbb{L}$ minus 3' - 1"	1407' - 2"	14.5	22.1	17	502.5
Core $\mathbb{L}$ minus 3' - 2"	1407' - 1"	4.0	21.5	28.5	531
Core $\mathbb{L}$ minus 3' - 3"	1407' - 0"	8.0	20.25	22	553
Core $\mathbb{L}$ minus 3' - 4"	1406' - 11"	3.0	19.3	23.5	576.5
Core $\mathbb{L}$ minus 3' - 5"	1406' - 10"	6.0	18.0	19	595.5
Core $\mathbb{L}$ minus 3' - 6"	1406' - 9"	3.0	17.0	19	614.5
Core $\mathbb{L}$ minus 3' - 7"	1406' - 8"	3.0	16.0	17	631.5
Core $\mathbb{L}$ minus 3' - 10"	1406' - 5"	3.0	15.0	15	646.5
Core $\mathbb{L}$ minus 4' - 1'***	1406' - 2"	14.2***	29.8	38.5	685
Core $\mathbb{L}$ minus 4' - 4"	1405' - 11"	5.0	28.3	46	731
Core $\mathbb{L}$ minus 4' - 7"	1405' - 8"	2.0	26.6	42.5	773.5
Core $\mathbb{L}$ minus 4' - 10"	1405' - 5"	4.5	24.85	36	809.5
Core $\mathbb{L}$ minus 5' - 1"	1405' - 8"	2.0	23.4	33	842.5
Core $\mathbb{L}$ minus 5' - 4"	1404' - 11"	2.9	21.9	30	872.5
Core $\mathbb{L}$ minus 5' - 7"	1404' - 8"	5.1	20.4	25.5	897
Core $\mathbb{L}$ minus 5' - 10"	1404' - 5"	5.0	19.3	23	920
Core $\mathbb{L}$ minus 6' - 6"	1403' - 9"	5.0	18.2	21	941

* Gang 5 at 71.3 prior to insertion

** Gang 5 at 111.0 prior to insertion

*** Gang 2 inserted to 100" from 156" at this point

significantly by the addition of 27 elements was in error. Gang 5 had decreased in worth appreciably, as was to be determined in the dry core. Figure 710-14 shows the sodium worth as derived from the 83 element wet calibration of gang 5.

Criticality was achieved in the dry core, with all rods banked at 32.0 inches. This value was plotted in Figure 710-6, and shows that shutdown margin was represented by all 19 rods at the 32.0 inches level. Figure 710-6 also shows that only 6 inches insertion of the all rods banked configuration was required to compensate for the measured 2.6 percent excess added since the 83 element loading.

Gang 5 was calibrated in the dry core, from full in to full out, compensating with all other rods banked. The bank was moved from 35.1 inches (with gang 5 at 0 inches) to 28.0 inches (with gang 5 at 156.0 inches). The total worth of gang 5 was thus determined to be 247.9 cents. Figure 710-15 shows the developed integral worth curve. The worth of gang 5 in the dry 83 element core was 293.2 cents. Ratios of these values, utilized to obtain a 110 element core worth for the wet gang 5 (multiply the ratio by the 83 element gang 5 wet worth), yield an approximate worth of 190 cents. Utilizing this curve in determining the worth of the sodium increments, as opposed to the 83 element wet curve (which was used), the seemingly high value of 6.5 percent for sodium worth would be decreased to a value of 5.5 percent.

Intercomparison of control rods with gang 5 calibrated in the dry core was performed with all remaining rods banked at 32.0 inches. Table XIV lists the results.

TABLE XIV
ROD WORTHS AT THE 110 ELEMENT LOADING - REACTOR DRAINED

Gang	Control Rod	Worth Below Bank	Worth Above Bank	Total Worth
5	CR- 1	100.4	147.5	247.9
4	CR- 6	79.7	140.4	220.1
3	CR- 3	77.8	135.4	213.2
2	CR- 5	37.6	59.4	97.0
1	CR-14	25.6	44.4	70.0
1	CR-15	24.1	38.4	62.5

Figure 710-14. Sodium Worth vs Elevation of Sodium

Figure 710-15. Gang 5 Worth Curve, 110 Element Loading ,
Dry Core.

Prior to the measurements in Table XIV, it had been demonstrated that individual rods within a gang were of similar worth, except gang 1, which demonstrated two rod worths at this time. The rod worths in Table XV were developed for the five gangs, by multiplying each of the measured values by the number of rods in the associated gang.

TABLE XV
GANG WORTHS AT THE 110 ELEMENT LOADING -
REACTOR DRAINED

Gang	Worth Below Bank	Worth Above Bank	Total Worth
5	100.4	147.5	247.9
4	239.1	421.2	660.3
3	233.4	406.2	639.6
2	225.6	356.4	582.0
1	149.1	248.4	397.5
Total (cents)	947.6	1579.7	2527.3
(%)	6.6	10.9	17.5

The above measured rod worths were considerably higher than anticipated. Non-validity of measurements made with the all rods banked configuration, which results in flux distortions and subsequent tilting was thus verified. These results also showed direct measurement of shutdown margin, in a configuration other than all rods banked, as a very desirable part of the test.

The reactor was made critical with gang 1 full out, gangs 2, 3, 4, and 5 full in, except for one rod of gang 2 (CR-8), which was at 123.1 inches. Thus, shutdown margin was represented by the withdrawal of six complete rods and 123.1 inches of a seventh rod. One rod of gang 1 (CR-15) was run full in and one rod of gang 3 (CR-4) was withdrawn to 90.0 inches to compensate. CR-15 was then calibrated to full out, by the differential period technique, compensating with CR-4. CR-15 was determined to be worth 92 cents. Figure 710-16 shows the integral worth curve for CR-15. Each of the remaining gang 1 rods was then intercompared with CR-15 to obtain their worths. Following is a list of the results of the intercomparison:

Figure 710-16. CR-15 Worth Curve, 110 Element Loading, Dry Core.

<u>Control Rod No.</u>	<u>Worth (cents)</u>
14	25.5
15	92
16	139.5
17	66.5
18	37
19	20

The test was terminated before the worth of CR-8 could be determined by intercomparison with CR-15. Test termination was caused by operational limitations imposed on reactor vessel temperatures while in the drained condition. The worth of CR-8 was estimated by averaging the worths of the adjacent gang 1 rods and ratioing it up by the relationship of gang 2 to gang 1, as determined by the all rods banked intercomparison. It was felt that, although the actual values as determined by the all rods banked configuration were non-valid, the relative worths could be considered as reliable. CR-8 worth at 123.1 inches was thus estimated to be 150 cents.

Total shutdown margin (gang 1 plus CR-8 at 123.1 inches) was equal to 3.67 percent at the 110 element loading, according to these calculations. An excess reactivity of 1.67 percent could be added to the reactor and the required two percent shutdown margin would still be maintained. This was the criterion on which the next fifteen fuel elements were loaded.

Although not completely recognized at the time, a significant flux tilt was imposed on the reactor when the shutdown margin configuration of gang 1 full out and CR-8 at 123.1 inches was arranged. The wide variation in gang 1 individual rod worths was the first positive indication that flux tilting or control rod shadowing was having an effect. This effect was investigated in detail at the 125 and 140 fuel element loadings. It was concluded at the 110 element loading that the shadowing effect of the rods would result in a conservative determination of shutdown margin, when determined by the intercomparison method. This conclusion was verified by measurements made at the 140 fuel element (operational) loading.

L. 125 ELEMENT LOADING

1. Excess Reactivity Measurements

Gang 3 critical rod position with 125 fuel elements loaded into the reactor was 32.0 inches. Gangs 1 and 2 were full out and gangs 4 and 5 were full in. Gang 3 was calibrated from 32.0 inches to 76.9 inches (gang 3 critical position at 110 elements), to determine the excess reactivity added to the reactor by the ninth loading increment (15 elements). The calibration showed that 191.299 cents (1.33 percent) had been added. Gang 2 was used to compensate during the calibration.

The 191.299 cents measured by the differential period technique was considerably higher than anticipated. Figure 710-4 shows that not only did the linear excess loading curve not continue to bend as was expected, but the slope actually increased with the fifteen elements loaded. These results indicated that only 0.34 percent of the additional total allowable excess loading, as determined at 110 fuel elements (shutdown margin of 3.67 percent minus 2 percent), was now available for loading (1.67 percent minus the 1.33 percent measured at this loading). Thus, another sodium draining test was required, to measure the shutdown margin directly and determine if fixed control rods would have to be installed in the reactor prior to further fuel loading. The rods would be required to maintain the 2 percent shutdown margin (minimum) if 1.33 percent had been loaded and at least 12 more elements were to be installed in the reactor to reach the operational loading.

2. Sodium Draining Test

The sodium was drained at the 125 fuel element core loading, to obtain a direct measurement of shutdown margin. No calibration of control rods was performed prior to draining, as determination of sodium worth was not a prime reason for performing the draining at this point. However, critical rod positions were observed and recorded during the draining. This information was combined with a gang 5 calibration in the dry core to give an indication of maximum sodium worth. Critical rod positions during the

draining operations are shown in Table XVI. With the reactor drained, all rods were in, except gang 1 at 121.6 inches. Shutdown margin was thus represented by six gang 1 rods at 121.6 inches.

TABLE XVI
CONTROL ROD POSITIONS DURING 125 ELEMENT DRAINING TEST

Sodium Level+ (ft. - in.)	Elev.	Rod Positions				
		Gang 5 (inches)	Gang 4 *	Gang 3 *	Gang 2 (inches)	Gang 1 (inches)
25' - 0"	1415' - 6"	120	in	in	117.8	out *
26' - 9"	1413' - 9"	78	in	in	117.8	out *
27' - 6"	1413' - 0"	in *	in	in	117.8	out *
27' - 6"	1413' - 0"	150	in	in	96.8	out *
28' - 4"	1412' - 2"	78	in	in	96.8	out *
28' - 11"	1411' - 7"	19.5	in	in	96.8	out *
28' - 11"	1411' - 7"	150	in	in	79.65	out *
29' - 11"	1410' - 7"	78	in	in	79.65	out *
30' - 2"	1410' - 4"	in *	in	in	79.65	out *
30' - 2"	1410' - 4"	150	in	in	59.1	out *
31' - 1"	1409' - 5"	78	in	in	59.1	out *
31' - 7"	1408' - 11"	25	in	in	59.1	out *
31' - 7"	1408' - 11"	150	in	in	in	147.9
33' - 1"	1407' - 5"	781	in	in	in	147.9
34' - 6"	1406' - 0"	18.1	in	in	in	147.9
34' - 6"	1406' - 0"	150	in	in	in	97.0
36' - 6"	1404' - 0"	78	in	in	in	97.0
38' - 9"	1401' - 9"	75.6	in	in	in	97.0
38' - 9"	1401' - 9"	in *	in	in	in	121.6

* Fully inserted

+ Measured from top of reactor loading face shield.

Gang 5 was calibrated from full in to 80 inches withdrawn, compensating with gang 1 rods CR-15, -17, and -19 as a bank. The worth of gang 5 to this point was 95.767 cents. Figure 710-17 shows the partial integral worth curve developed for gang 5. Sodium worth was determined by extrapolating the dry gang 5 curve from 80 inches to full out (Figure 710-17), allowing the sodium worth curve as a function of sodium level to be plotted (Figure 710-18).

Each of the gang 1 rods was intercompared from 121.6 inches to full in, with the partially calibrated gang 5. The results were as follows:

<u>Control Rod No.</u>	<u>Worth (cents)</u>
14	55
15	58
16	63
17	62.5
18	67
19	<u>48</u>
Total	353.5 (2.45%)

This measurement of shutdown margin indicated that 1.22 percent excess had been loaded by increasing the reactor loading from 110 to 125 elements (shutdown margin 3.67 percent - 2.45 percent = 1.22 percent). Shutdown margin of 1.33 percent had been measured by the calibration method.

The variation in gang 1 rod worths again showed that control rod shadowing was affecting measurements. CR-15 worth had been determined to be 58 cents by intercomparison with gang 5. To verify that shadowing was occurring, CR-14 (gang 1 rod adjacent to CR-15) was inserted, and gangs 3 and 4 withdrawn to 20.4 inches to compensate. CR-15 was again compared to gang 5 and the worth determined to be 31 cents, or about one-half of the original value. CR-16 was also inserted (other gang 1 rod adjacent to CR-15), and gangs 3 and 4 withdrawn to 29.95 inches to compensate. A re-comparison of CR-15 with gang 5 showed its worth had decreased to only 13 cents, less than one-fourth of the original value.

Figure 710- 17. Gang 5 Integral Worth Curve to 80" at 125 Elements
Reactor Drained

Figure 710-18. Sodium Worth at 125 Element Loading

CR-14, -15, and -16 are separated by gang 2 control rods. The above test demonstrated that even adjacent gang 1 rods have significant shadowing effects on each other. The intercomparisons at 110 fuel elements, which utilized CR-8 (a gang 2 rod) at 123.1 inches therefore, undoubtedly contained large discrepancies due to shadowing effects.

The above results lessened the validity of the 3.67 percent shutdown margin measured at the 110 element loading, and removed some credibility from the measured 2.45 percent shutdown margin at this loading, even though the comparisons were made in a much more symmetrical pattern. The difference in CR-18 (67 cents) and CR-19 (48 cents) worths indicated that some effect was present in the reactor, causing the discrepancies.

Considering the above results, a decision was reached to install three fixed poison columns in the reactor. Criticality was achieved in the dry reactor with all rods at 29.0 inches, just prior to reactor shutdown for insertion of the fixed poison columns.

3. Fixed Control Rods Data

Three fixed control rods (control rods without drive mechanisms) were installed in core positions C-5, C-8, and C-11. The installations were made with the reactor drained at the 125 fuel element loading. The premeasured fuel element in C-5 was moved to core position C-13 and the regular fuel element in C-13 was placed in empty channel C-131. The instrumented (long probe) fuel element in C-8 was moved to C-10 and the regular fuel element in core position C-10 was placed in the previously empty C-178. The regular element in C-11 was placed in channel C-154.

Criticality was achieved in the dry, 125 element core (with fixed rods installed), with gang 3 at 64.0 inches, gang 1 out, and all other rods full in. The installation of the fixed rods thus resulted in the addition of that amount of negative reactivity corresponding to the amount of positive reactivity which would be added by withdrawing gang 1 from 126.1 inches to full out, plus pulling gang 3 from full in to 64.0 inches. (Recall that critical rod positions in the dry, 125 element core, without fixed rods, were all rods in except gang 1 at 126.1 inches.)

Gang 5 was calibrated from full in to full out, compensating with gang 1. Gangs 2 and 4 remained fully inserted and gang 3 at 66.4 inches (source separated position) during the calibration. Gang 5 worth was determined to be only 43.51 cents in the dry core. Figure 710-17 shows the integral worth of gang 5 under these conditions, as well as showing gang 5 worth prior to fixed rods.

The installation of the three fixed rods resulted in a decreased in worth of gang 5 from 80 inches to full in (only 80 inches had been calibrated prior to fixed rods) of 62.867 cents (95.767 cents - 32.9 cents). It was thus verified that the three fixed rods had depressed the flux significantly in the center of the reactor and undoubtedly peaked it near the outer ring of control rods.

Each of the gang 1 and gang 3 rods which were out at criticality were inter-compared with the calibrated gang 5. The results of the inter-comparisons are as listed in Table XVII.

TABLE XVII
ROD WORTHS AT THE 125 ELEMENT LOADING -
REACTOR DRAINED AND FIXED RODS INSTALLED

Gang	Control Rod No.	Worth (cents)
1	14	59.7
1	15	61.75
1	16	60.0
1	17	67.5
1	18	63.25
1	19	58.0
3	2	44.0 *
3	3	44.5 *
3	4	42.5 *
Total Worth		501.20

* From 66.4 inches to full in

The total worth of gang 1 was 370.2 cents, of gang 3 from 66.9 inches to full in, 131.0 cents. The measured shutdown margin was the sum of these values, namely, 501.2 cents, or 3.47 percent. When compared to the 2.45 percent shutdown margin measured before fixed rod installation, the insertion of the three fixed rods in the reactor had a combined effect of 1.02 percent negative reactivity worth.

Three critical positions (40 inches, 80 inches, 120 inches) were obtained on CR-15, -17, and -19, during the intercomparison with gang 5. The finding of the three critical positions allowed integral worth curves to be drawn for the three rods. Figure 710-19 displays the three integral worth curves. These three rods were then intercompared as a bank against gang 5 in an attempt to determine the significance of shadowing effects in symmetrical rod configurations. Control rods CR-15, -17, and -19 were pulled as a bank from 76.7 inches to 97.7 inches. Gang 5 had to be inserted from 103 inches to full in, to compensate. The worth of gang 5 from 103 inches to full in was 39.5 cents, corresponding to the worth of the three-rod bank from 76.2 inches to 97.7 inches. Summing the values from the three integral worth curves on Figure 710-18, from 76.2 inches to 97.7 inches, shows that a 42 cent change resulted from the maneuver. Thus, a discrepancy of 5 percent in the measured reactivity worth was noted in the symmetrical reactor, and attributed to the shadowing that occurs when intercomparing individual rods.

The rods banked critical position was 41.75 inches. This was 11.75 inches higher than prior to fixed rod installation, demonstrating that shutdown margin had been appreciably increased. This critical bank position would not be so high, however, if all rods were, in fact, banked. Since three of the rods extend through the bank to full in, with a 22 control rod reactor, improper comparison of the 41.75 inches bank position with the 38.7 inches bank position associated with the 83 element drained reactor should be avoided. Excess reactivity in the amount of 4.1 percent (as measured) had been added since that loading, drawing the all banked position down from 38.7 inches to 29.0 inches. The addition of 1.02 percent more shutdown margin did not move this same bank back up to 41.75 inches. It moved 19 rods of lesser worth to 41.75 inches, while holding three new rods in. The reactor was re-filled with

Figure 710-19. Integral Worth Curves for CR-15, -17, and -19 at 125 Element Loading, Reactor Drained and Fixed Rods Installed

sodium, following the above measurements, in preparation for loading 15 more fuel elements. Critical rod positions in the 125 element, fixed-rod, sodium filled reactor were; gangs 1 and 2 full out, gangs 4 and 5 full in, and gang 3 at 95.5 inches. Installation of the fixed rods, therefore, contributed negative reactivity corresponding to gang 3 from the prefixed rods critical position of 33.7 inches (source out) to the new position of 95.5 inches. Figure 710-13, which is a calibration of loading increments by gang 3, as well as an approximation of gang worth, shows this rod movement to be equivalent to 153 cents (1.06 percent). This value agrees reasonably well with the 1.02 percent measured in the dry reactor and shows that the 1.02 percent is conservative, probably due to shadowing during the intercomparisons. The new gang 3 critical position must be associated with the fixed-rods-installed calibration curve for gang 3 on Figure 710-13, not the prefixed rods curve. Using this curve gives a value of approximately 0.94 percent for the fixed rods. However, the original critical position would have been lower on this lesser worth rod. Therefore, the best value available for the fixed rods worth, from these data, is 1.00 percent $\pm$.06 percent, which does not refute the measured 1.02 percent.

The all rods banked position in the sodium filled, fixed-rod reactor was 77.8 inches, an increase of 27.2 inches over the prefixed-rod position. Note that three rods are still in, and that the 77.8 inches banked rods worth had diminished due to the insertion of the fixed rods.

M 140 ELEMENT LOADING

1. Excess Reactivity Measurements

The tenth excess loading increment contained 15 fuel elements, bringing the reactor to the final operational fuel loading of 140 elements. Criticality was achieved in the 140 element reactor with gang 2 at 122.6 inches, gang 1 full out, and gangs 3, 4, and 5 full in. The change in critical rod positions resulting from this loading increment was the value of gang 3 being fully inserted from 91.2 inches, plus gang 2 being run in to 122.6 inches from full out. These two sections of rods were calibrated by the differential period technique and determined to be worth 212.703 cents excess reactivity, bringing

the total reactor loading to 9.73 percent (1403.4 cents). With 1.02 percent being unavailable due to the fixed rods, the net excess loaded was 8.71 percent.

The loading of these fifteen fuel elements allowed the completion of Figure 710-13, Loading Increment Calibrations by gang 3. An excellent comparison of the fixed-rod effect on gang 3 worth is shown on Figure 710-13, also. Prior to installation of the fixed-rods, the worth of gang 3 was approaching 3.7 percent. After the fixed rods were installed, the worth decreased to approximately 1.5 percent, substantiating the conclusion that installation of the rods considerably depressed the core center flux. Subsequent measurements verified that the worths of gangs 1 and 2, further from the reactor center, were increased appreciably.

2. Sodium Draining Test

Gang 5 was calibrated prior to initiation of the sodium draining operation to allow compensation for sodium removal, by use of gang 5 and thus to determine the incremental worths of the sodium. The calibration showed gang 5 to be worth only 18.77 cents, making it impractical to compensate for sodium draining with gang 5. Therefore, gang 3 was calibrated for this purpose. Gang 1 was used as the compensating bank during the calibration, gang 2 was full out, and gangs 4 and 5 were full in. Figure 710-20 shows the integral worth curve developed for gang 3 with its associated total worth of 215.324 cents.

The reactor was then drained, compensating with gang 3. When gang 3 reached full in, it was withdrawn and gang 2 inserted to compensate. When gang 2 reached full in, gang 1 was used to maintain criticality when gang 3 was withdrawn. Gangs 4 and 5 were full in during the entire draining operation. Table XVIII shows the sodium draining results.

Criticality was achieved in the 140 element, dry reactor with all rods in, except gang 1, at 109.8 inches. Shutdown margin was thus represented by six gang 1 rods at 109.8 inches.

Gang 3 was calibrated in the dry core, for comparison with the previous wet calibration. The dry total worth was determined to be 244.09 cents as compared to the wet value of 215.2 cents. Thus, gang 3 showed an apparent increase of 28.9 cents due to draining the sodium. The dry

Figure 710-20. Gang 3 Worth Curve, 140 Fuel Elements

TABLE XVIII
SODIUM WORTH DATA 140 ELEMENT LOADING

Sodium Level * (ft. - in.)	Elev. (ft. - in.)	Gang 3 Pos (inches)	Gang 2 Pos (inches)	Gang 1 Pos (inches)	Increment Worth (%)	Total Worth (%)
19' - 0"	1421' - 6"	156 "	73.3"	100 "	0	0
25' - 11"	1414' - 7"	120 "	73.3"	100 "	10	10
27' - 5"	1413' - 1"	100 "	73.3"	100 "	19	29
29' - 4"	1411' - 2"	70 "	73.3"	100 "	56	85
30' - 3"	1410' - 3"	50 "	73.3"	100 "	59	144
31' - 1"	1409' - 5"	10 "	73.3"	100 "	71	215
31' - 1"	1409' - 5"	125 "	41.4"	100 "	--	---
31' - 6"	1409' - 0"	100 "	41.4"	100 "	23	238
32' - 3"	1408' - 3"	70 "	41.4"	100 "	56	294
32' - 11"	1407' - 7"	50 "	41.4"	100 "	59	353
33' - 9"	1406' - 9"	10 "	41.4"	100 "	71	424
33' - 9"	1406' - 9"	125 "	in	80.6"	--	---
34' - 2"	1406' - 4"	100 "	in	80.6"	23	447
35' - 3"	1405' - 3"	74.5 "	in	80.6"	45	492
36' - 0"	1404' - 6"	65 6 "	in	80.6"	23	515
36' - 6"	1404' - 0"	62 85"	in	80.6"	9	524
37' - 0"	1403' - 6"	62.85"	in	80.6"	0	524

* Measured from top of loading face shield.

calibration was made with all rods in except the compensating gang 1 rods. Wet calibration was performed similarly, except that gang 2 was full out. The dry calibration was therefore more shadowed than the wet calibration, nevertheless, resulting in an increase of 28.9 cents. Figure 710-20 shows the comparison of the two calibration curves.

Using the dry gang 3 curve for determining the incremental sodium worths gives a total sodium worth of 4.13 percent, as opposed to the 3.64 percent listed in Table XVIII. Analysis of reactor condition during the sodium draining operation shows that neither curve is appropriate for measuring the sodium worth directly. The wet curve is proper at the start of

the draining, but the dry curve should be used at the end of the draining. It was concluded that the best approximation was to use the dry curve when the sodium increment being measured was compensated for by gang 3 out of the sodium, and the wet curve when gang 3 was below sodium level while compensating. This approach gives a sodium worth of 4.0 percent.

A second measurement of sodium worth was made by critical rod positions. Prior to draining sodium, gang 1 was at 109.5 inches, gang 2 was out, and all other rods in. In the drained reactor, gang 1 was at 109.8 inches and all other rods were in, including gang 2. For all practical purposes, reactivity effect of draining the reactor of sodium was equal to the effect of withdrawing gang 2 to the full out position. Therefore, sodium worth is equal to gang 2 worth. A calibration of gang 2 in the dry core was started, but had to be terminated at 42.6 inches, due to operational limitations imposed on reactor vessel temperatures with the reactor in the drained condition. A calibration of gang 2 in the wet core was made following the refill of the reactor. The worth of gang 2, hence, the value for sodium worth, was determined to be 4.2 percent. An estimate of gang 2 worth in the dry core was made by scaling up the measured wet value of 4.2 percent by the ratio of gang 3 dry-to-wet worths, both of which were calibrated. This resulted in an estimate of 4.7 percent as gang 2 worth (and sodium worth) in the dry core.

In the drained reactor, the six gang 1 rods at a critical position of 109.8 inches were individually intercompared from that position to full in, with the dry-core calibrated gang 3. Results were as follows:

<u>Control Rod No.</u>	<u>Worth (cents)</u>
14	56.5
15	61.5
16	60.0
17	64.0
18	58.0
19	<u>59.0</u>
Total	359.0 (2.49%)

Reactor drained shutdown margin was thus measured as 2.49 percent.

Additional effort was expended to determine the complete effects of control rod shadowing in the dry core, because of the desirability of accuracy in shutdown margin measurement. As in the 125 element dry core, gang 1 control rods CR-14 and -16 were inserted, while compensating with gang 4. CR-15 (which had been determined to be worth 61.5 cents by the above inter-comparison with gang 3) was again compared with gang 3. CR-15 worth was determined to be 18.0 cents when shadowed by CR-14 and -16. These results agree with the 125 element worth determination. The test was extended at this operational loading to check the worth of CR-18. CR-18 was worth 77.0 cents with CR-14 and -16 inserted and CR-15 worth 18 cents. These measurements demonstrated that depressing the flux in one region of the reactor, by the insertion of control rods, results in increased rod worth in other parts of the reactor, indicating flux peaking in that region. Thus, a flux tilt is created when the reactor control rods are not operated symmetrically.

In the next measurement, the gang 1 rods representing shutdown margin were split into two symmetrical banks. The one bank was then calibrated by the differential period technique, while compensating with the other bank. It was felt this arrangement of symmetrical banks would eliminate the "mobile" flux problem described in the previous paragraph. Control rods CR-15, -17, and -19 were inserted full in and CR-14, -16, and -18 and gang 3 were withdrawn to compensate. CR-14, -16, and -18 were pulled to full out and gang 3 was positioned at 109.5 inches, to maintain criticality.

CR-15, -17, and -19 were calibrated from full in to full out, compensating with CR-14, -16, and -18, and then with gang 3. CR-15, -17, and -19 worth was determined to be approximately 330 cents. The compensating bank, CR-14, -16, and -18, was measured to be worth approximately 280 cents. The difference in the two bank values can be attributed to CR-15, -17, and -19 being uncovered by gang 3 during the calibration (gang 3 at 109.5 inches), thus giving a higher value than is real. CR-14, -16, and -18 were covered by all immediately adjacent rods during the calibration, and hence, represent the true worth of one-half of the gang 1 rods in the all rods in configuration. Figure 710-21 depicts the two measured curves, plus a curve equal to twice the worth curve for CR-14, -16, and -18. This curve is considered the dry core gang 1 worth curve.

Figure 710-21. "Split" Gang 1, Integral Rod Worths
at 140 Element Loading, Reactor Drained

Utilizing the Figure 710-21 curve as the gang 1 curve gives a value of 3.2 percent to that amount of rod under 109.8 inches, which represents shutdown margin. The partial gang 2 calibration curve developed in the dry core used gang 1 to compensate. Gang 1, as the compensating bank, moved from the critical position of 109.8 inches, down to 61.8 inches as gang 2 was withdrawn. Plotting the worth of gang 1, as determined by the compensating operation, on Figure 710-21, shows that the points do fall on the curve. The lower section of the gang 1 compensating curve would probably be of slightly higher value if the calibration had continued (due to the gangs passing into an unshadowed region). This would push the total worth curve up and place the measured points even more closely on the gang 1 curve, thus substantiating the curve as being actual.

Gang 1 worth as determined by the wet gang 2 calibration (gang 1 again compensating) was 2.9 percent (404 cents). The integral worth curves for both gangs 1 and 2 are shown on Figure 710-22. Scaling the gang 1 curve up by the same proportion determined from the difference in the gang 3 dry and wet curves, brings the total worth to 3.3 percent (468 cents). This curve agrees reasonably well with the gang 1 curve shown on Figure 710-21 and substantiates that a shutdown margin of approximately 3.0 percent did exist in the 140 fuel element loading.

3. Effect of Dummy Fuel Elements

Thirty dummy fuel elements (reflector fillers) were added to the reactor following the loading of the last fuel increment. These dummy elements completed the reactor operational loading, except for the three ICNI chambers incore positions C-191, -215, and -239. The chambers will be removed and three dummy elements inserted following the completion of reactor zero power testing. The thirty dummy elements presently installed are in the following core positions:

Figure 710-22. Gangs 1 and 2 Integral Worth Curves
at the 140 Element Loading, Dry Core

C-132	C-139	C-211	C-222	C-162	C-173	C-179
C-133	C-202	C-213	C-224	C-233	C-175	C-181
C-135	C-191	C-152	C-226	C-235	C-296	C-122
C-198	C-142	C-153	C-159	C-237	C-248	C-185
C-200	C-209	C-155	C-161	C-172	C-250	C-181
						C-189

Critical rod position prior to insertion of the dummy elements was all rods banked at 65.95 inches. With the dummies in place, criticality was achieved with all rods banked at 63.85 inches. Gangs 1 and 2 were calibrated individually and gangs 3, 4, and 5 as one bank from 63.85 inches to 65.95 inches, to determine the worth of the dummy elements. The results are as follows:

<u>Gang</u>	<u>Worth</u>
3, 4, and 5	9.88 cents
2	9.98 cents
1	<u>7.54 cents</u>
Total	27.40 cents (0.19%)

The effect of the 36 dummies was, therefore, the addition of 27.4 cents excess reactivity. This brings the total reactor excess fuel loading to 1432.166 cents (9.9 percent) positive with 1.02 percent negative contained in the fixed rods for a net excess of 1284.88 cents (8.9 percent).

N. REMOVAL OF FIXED RODS

A detailed analysis of the data obtained at the 140 element loading was made to determine if the fixed control rods could be removed. The results of the study indicated that such action would not reduce the shutdown margin below the 2 percent desired. The fixed rods were subsequently removed from the reactor.

The three fixed rods were installed at the 125 element loading to ensure that a 2 percent shutdown margin would be maintained in the event of loss of sodium. Control rod gang 5 was calibrated by the differential period technique prior to fixed rod insertion, compensating with gang 1. After draining sodium, but before inserting the three fixed rods, the gang 1 rods out of the reactor

were intercompared with gang 5 and a dry shutdown margin of 2.45 percent was determined. The calibration and intercomparisons were repeated after the fixed rod insertion, giving a shutdown margin measurement of 3.47 percent. Therefore, the fixed rods were considered to be worth 1.02 percent (negative).

Flux tilting during the intercomparison of the gang 1 rods with gang 5 was indicated by measurements made at the 125 element loading. Specific tests were performed at the 140 element loading to verify that the tilting did exist. It was determined that measuring rod worths by the intercomparison method resulted in flux tilting in the reactor (away from the inserted rods), and a subsequent low measured value of rod worth. The 1.02 percent worth of the fixed rods is believed to be reliable, however, because it is the difference of two measurements that were made in the same manner.

Shutdown margin was measured at the 140 element loading by three methods:

- 1) Intercomparison of rods
- 2) Calibration of "split" gang 1 rods, and
- 3) Calibration of gang 1 as a bank

Methods 2 and 3 were utilized for the measurement, in addition to 1, because Method 1 had been determined unreliable due to flux tilting (discussed previously). Method 1 gave a value of 2.49 percent for shutdown margin. The "split" gang 1 calibration (Method 2) was performed with gang 3 withdrawn 110 inches, thus decreasing radial flux peaking. This "split" gang configuration was also chosen in order to minimize flux tilt. Method 2 gave a value of 3.20 percent for shutdown margin. Method 3 was accomplished in the wet core with all center rods out, except the three fixed rods. This rod configuration also minimized flux tilting. Method 3 led to a value of 3.15 percent for the shutdown margin in the dry core, which was obtained by multiplying the wet value by a factor of 1:1. This factor had been determined by repeating specific measurements in both the dry and wet core. Moreover, the calculated value for increased rod worth in the dry core over that in the wet core (due to absence of sodium) was also 10 percent of the wet core value, or a factor of 1:1.

The "split" gang calibration was considered the most meaningful, and it was concluded that a conservative value for shutdown margin with the three fixed rods in place was 3.1 percent. The calibration of gang 1 as a bank, compensating with gang 2, substantiates this conclusion.

The HNPF Final Summary Safeguards Report (NAA-SR-5700) required that 1 percent shutdown margin be available in the drained reactor. Data obtained during the post-critical isothermal temperature coefficient test were used to determine the reactivity increase due to temperature increase from 350 F, to the operating temperature of 945 F. A resultant 0.36 percent reading was obtained. This value was determined by taking the measured change in reactivity created by increasing core temperature from 350 F to 540 F and extrapolating to the operating temperature of 945 F. The isothermal temperature coefficient was also measured during the dry zero power experiments, but only up to a maximum temperature of 350 F. Extrapolation of this information gives a reactivity increase of 0.57 percent as a result of increasing temperature from 350 F to 945 F. This value is considered less reliable than the wet core coefficient determination, due to the magnitude of the extrapolation required of the dry core data. Both values are high, of course, as the 945 F temperature is not the reactor isothermal temperature, but the maximum sodium temperature during normal operation.

Control rod calibrations at 350 F and 540 F indicate that change in rod worth with temperature is not significant and will not effect the shutdown margin as measured.

A conservative estimate of the shutdown margin for a loading of 140 fuel elements with the three fixed rods removed was obtained as follows:

Measured shutdown margin with 3 fixed rods present . . .	3.1%
Worth of 3 fixed rods	1.0%
Temperature coefficient	0.5%
Estimated shutdown margin at 945 F with 3 fixed rods removed	1.6%

The required shutdown margin of 1.0 percent is thus assured with a safety margin of 0.6 percent. This was ample to allow the safe removal of the three fixed control rods.

Test procedure AI-PC-710, Wet Excess Fuel Loading, was amended to include a determination of the reactivity change in the reactor due to removal of the fixed rods. The technique utilized to determine this change was to calibrate the change in critical control rod positions due to removal of the rods. The differential period method was employed in the calibration.

Critical rod positions before and after fixed rod removal were as follows:

	<u>Before</u>		<u>After</u>
Gang 5	in	Gang 5	in
Gang 4	in	Gang 4	in
Gang 3	in	Gang 3	in
Gang 2	out	Gang 2	out
Gang 1	109.5 inches	Gang 1	74.05 inches

The worth of the fixed rods was therefore equal to the worth of gang 1 from 109.5 inches to 74.05 inches.

Gang 1 was calibrated from 74.05 inches to full out, compensating with gang 2. The worth of gang 1 attributable to the fixed rods (74.05 inches to 109.5 inches) was 128 cents, or 0.89 percent in the wet core. Scaling this value up by the proportion determined previously, gives a value of 144.6 cents in the dry core, or 1.0 percent.

The fixed rod worth of 1.0 percent in the dry core for the 140 element reactor is in fine agreement with measurements made previously. The value of the fixed rods at the 125 element loading was 1.02 percent in the dry core.

In summary, the measured shutdown margin of 3.1 percent at the 140 element loading was decreased to 2.1 percent by the removal of the fixed rods.

O. 140 ELEMENT LOADING - FIXED RODS REMOVED

With the removal of the fixed control rods, the fuel loading operation may be considered as a continuation of the conditions described in Section L., 125 Element Loading, Part 1, Excess Reactivity Measurements, where fixed rods had not yet been installed. Gang 3 critical position with 125 fuel elements loaded was 32.0 inches, with gangs 1 and 2 full out and gangs 4 and 5

full in. The loading of the reactor to 140 fuel elements and the addition of the dummy fuel elements resulted in a critical position of 80.25 inches on gang 2, with gang 1 full out and gangs 3, 4, and 5 full in. Thus, the fuel and dummy elements were worth the bottom 32.0 inches of gang 3 and gang 2 from 80.25 inches to full out.

The change in critical rod positions was calibrated by the differential period technique, compensating with gang 1. The worth of the loading increment was determined to be 205.7 cents, or 1.43 percent. This value agrees with that determined with the fixed rods installed (1.47 percent). The slight decrease can be attributed to a lesser flux depression in the reactor center with the fixed rods removed.

It was previously determined that the dummy elements were worth 0.19 percent (for 36 dummy elements). Since three more dummy elements were loaded to fill the channels vacated by the fuel elements which replaced the fixed rods, a total value of $0.19 \text{ percent} \times \frac{39}{36}$ or about 0.21 percent is attributed to the dummy elements. This value must now be subtracted from the measured 1.43 percent to get the worth of the 15 fuel elements (1.22 percent), but must be retained when considering total excess reactivity in the reactor.

Calibration of the change in critical rod position due to the last loading increment was done in a peaked flux region (gangs 3, 4, and 5 inserted - except lower 32.0 inches of gang 3 during that measurement). A re-determination of excess loaded in the last increment was made after purposely peaking the flux away from the calibration area. Gang 1 was fully inserted, gang 5 fully withdrawn, gangs 3 and 4 positioned at 46.8 inches and gang 2 controlling criticality at 67.1 inches for the measurement. A similar configuration had been achieved at the 125 element loading (prior to fixed rods), except gang 2 at 117.8 inches. The actual comparison is as follows:

<u>Gang</u>	<u>125 FE</u>	<u>140 FE</u>
1	in	in
2	117.8 inches	67.1 inches
3	46.8 inches	46.8 inches
4	46.8 inches	46.8 inches
5	out	out

The 140 element configuration was modified as described, driving gang 1 to full in, and withdrawing gangs 3 and 4 to 46.8 inches, to compensate. The 140 element calibration was performed at 530 F rather than 350 F, and required a correction of 18 cents (addition) to the measured reactivity change.

A worth of 113.8 cents was determined for the last loading increment in the depressed flux region. Adding the temperature correction brings the total to 131.8 cents or 0.91 percent. Again, one must deduct the value of the dummy elements to obtain a value for the excess reactivity added by the fuel. It is proper to deduct the 0.21 percent value for dummy elements, as it was determined from a banked configuration. Thus, the excess loaded was determined to be only 0.7 percent for the last 15 elements. Averaging the depressed and the peaked flux region calibrations gives a value of 0.96 percent excess reactivity for the last 15 fuel elements.

P. CONCLUSIONS

Large flux perturbations are created in the HNPF core by utilizing particular control rod configurations. Banking the rods results in axial flux peaking under the bank and axial flux depression above the bank. Inserting the center seven rods (10 including the three fixed rods) peaks the radial flux at the outer control rod rings and depresses it in the reactor center. Positioning the rods of the outer control rod rings in an asymmetrical pattern results in radial flux tilting; i. e., a depression at the rods inserted portion of the pattern across the reactor.

The choice of rod configurations for reactivity measurements are, therefore, very crucial. Those measurements requiring considerable rod manipulation develop questionable results. The following paragraphs discuss the determination of shutdown margin, sodium worth, excess reactivity and control rod worth, in light of the above statements.

1. Shutdown Margin

The reactor shutdown margin was measured at the operational loading by three methods;

- a) intercomparison of rods,
- b) calibration of "split" gang 1 rods, and
- c) calibration of gang 1 as a bank in the wet core and extrapolating to the dry condition.

Method a) gave a value of 2.49 percent, and methods b) and c) a value of 3.15 percent and 3.20 percent, respectively. Inter-comparison of rods representing shutdown margin was performed in a somewhat radially peaked region (gang 2 was inserted, as well as the center rods and also created a flux tilt during the measurement). The "split" gang calibration was performed with gang 3 withdrawn 110 inches, thus decreasing the radial peaking, and did not create a flux tilt, as far as can be determined. The wet gang 1 calibration (as a single bank) was accomplished with all center rods out, except the three fixed rods. This configuration also minimized flux tilting.

The "split" gang calibration is considered the most meaningful, leading to the conclusion that a 3.1 percent shutdown margin existed in the reactor with the fixed rods installed. Measurement of the effect of removing the fixed rods showed the rods to be worth 1.0 percent in the dry core. Therefore, shutdown margin in the operational reactor is 2.1 percent.

2. Sodium Worth

The insertion of gang 2 rods from full out to full in is required to compensate for draining the reactor of sodium, as determined by critical rod configurations at the 140 element loading. Gang 2 is worth 4 percent in the wet core and 4.4 percent in the dry core.

Gang 3 was used to compensate for the draining of sodium at the 140 element loading and shows the sodium to be worth 4.13 percent. The gang 3 calibration was performed in the least perturbed state that could be arranged; gangs 2, 4, and 5 inserted, and gang 1 being inserted as gang 3 was withdrawn.

Sodium is worth at least 4.13 percent at the operational reactor fuel loading (in the drained condition). Sodium worth may be slightly greater, but the exact value is not worth pursuing as it has no direct application to reactor operation, since shutdown margin is measured directly and has the greatest importance.

3. Excess Reactivity

Excess reactivities, loaded and measured in specific increments, are not additive. The measured increments cannot be summed, because they were determined in an increasingly perturbed radial flux pattern and the core size was changed with each added increment. Also, the method necessarily involves some axial peaking (although slight). Correction of the measured values is required, and must be theoretical. One method of analysis established that initial measurements were not involved in the error that accumulated due to the summing of the increments. Therefore, these values are extrapolated to give a reactor excess fuel loading of 7.5 percent. Figure 710-3, plot of $k_{\text{eff}} - 1$ vs a number proportional to radial buckling, displays the extrapolation.

4. Control Rod Worths

The calibration of all rods involves numerous rod patterns and manipulations. These configuration changes invalidate the measurements made, due to the axial and radial flux peaking and flux tilting described earlier.

The best approximation of rod worth, therefore, is the summation of other derived values; namely, excess reactivity (7.5 percent), shutdown margin (2.3 percent), and a sodium worth (4.1 percent) or 13.9 percent. The shutdown margin and sodium worth values are for the dry core. A better approximation of wet core rod worth is 13.3 percent.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with post critical test completion report AI-PC-710, Wet Excess Fuel Loading Test, as listed in NAA-SR-5700, Supplement 4, Page XI (c) - 40, with remarks regarding the test to explain compliance with the criteria.

a) Criterion: An operational fuel loading is in the core.

Remarks: The core loading has been completed. The reactor contains:

1) Fuel elements	140
2) Dummy elements.	39
3) Reflector filler elements	75
4) Control Rods	19
5) Source	1
6) Sodium level elements.	2
7) Sodium temperature elements.	3
8) Incore nuclear instrumentation thimbles . .	3

The three incore nuclear instrumentation thimbles will be replaced with dummy elements at the completion of zero power testing.

- b) Criterion: The shutdown margin in the core is equal to or greater than the worth of sodium plus 1 percent.
Remarks: The shutdown margin in the 350 F, wet core is 2.1 percent greater than the worth of the sodium. At operating temperatures the shutdown margin will be at least 1.6 percent greater than sodium worth.
- c) Criterion: The excess reactivity in the core is known.
Remarks: One method of analysis of test data indicated the core excess reactivity is 7.5 percent. The fuel loading increment excess measurements made by the test have been determined to be non-additive, and the total of 9.47 percent derived in this manner is not correct. The calculated excess reactivity for a 140 element loading is 6.4 percent. There is no method available for accurately measuring the excess reactivity in the core.
- d) Criterion: The reactor cannot be made critical by fully withdrawing any one control rod with the remaining control rods fully inserted.
Remarks: This criterion was verified as having been met throughout the loading operation and at the complete core loading (140 elements), by withdrawing the two rods of greatest worth, CR-1 and CR-6, and establishing that the reactor was not made critical.

V. EVALUATION

A. INTRODUCTION

The operational (or wet excess) loading of the HNPF was accomplished by loading fuel elements into the wet core in increments of approximately 1 percent $\frac{\Delta k}{k}$. At least 137 fuel elements were to be loaded, in order to provide sufficient heat transfer surface for achievement of design thermal power (256 Mwt). A maximum of 153 fuel elements was available. It was expected that the core excess reactivity at the operational loading would be in the range of 6 to 7 percent $\frac{\Delta k}{k}$, based on multigroup core calculations.

During the loading, it was desired to maintain a shutdown margin of 2 percent $\frac{\Delta k}{k}$ plus sodium worth, as a margin of safety, and to allow for the positive isothermal temperature coefficient. Of the 2 percent figure, 1 percent was allowed for safety, and 1 percent for the increase in reactivity expected to result from heating the core from 350 F to 850 F for power operation.

It was also desired to obtain a preliminary estimate of total control rod worth, sodium worth, and the reactivity change caused by replacing the source element with a fuel element.

B. DETERMINATION OF EXCESS REACTIVITY

After the first four fuel increments were loaded, it was noted that the measured excess reactivity obtained by summing the incremental reactivities was becoming larger than the calculated excess. At the operational loading of 140 fuel elements, the value of the measured k_{eff} was 1.109 as compared with an "analytical" value of 1.069. (Figure 710-23).

Two effects are postulated as to the cause of the discrepancy. They are; the change in reactor size with fuel loading, and distortion of flux due to control rod insertion.

As core size increases, the flux in the center of the core tends to get flatter. Thus, the reactivity value of a fuel element loaded toward the center of the core decreases as more fuel is loaded. The total excess

Figure 710-23. K_{eff} vs Number of Fuel Elements at 140 Fuel Element Loading.

reactivity was determined by summing the reactivities of the individual loading increments. The final value obtained by this procedure is therefore high.

A more important problem is the one caused by the insertion of control rods. As fuel was loaded, the excess reactivity was compensated for by insertion of the most centrally located control rods. As more and more fuel was loaded, more control rods were inserted into the core in an increasing radius from the core center. This depresses the flux in the central regions of the core, while peaking the flux in the uncontrolled outer portions of the core. Fuel elements were loaded in this high flux region, and thus given greater importance than they would have had assuming no flux perturbation. These two effects added together caused most of the discrepancy between the analytical and experimental values.

Three methods have been used to obtain a true estimate of the excess reactivity actually loaded. They are: (a) perturbation correction of measured excess; (b) two-region calculation of excess based on measured banked rod positions; and (c) extrapolation of excess loaded in the first two increments.

1. Perturbation Correction

From perturbation theory, the excess reactivity measured at each fuel loading is:

$$\rho_1 = \frac{\int_{V_0}^{V_1} \phi_0^* P'_1 \tilde{\phi}_1 dV}{\int_{V_0}^{V_n} \phi_{n-1}^* R'_1 \phi_1 dV} \quad \dots (1)$$

$$\rho_n = \frac{\int_{V_{n-1}}^{V_n} \phi_{n-1}^* P'_n \tilde{\phi}_n dV}{\int_0^V \phi_{n-1}^* R'_n \tilde{\phi}_n dV} \quad \dots (2)$$

where:

ρ_n is reactivity loaded during loading increment n

V_n is core volume after n loadings

V_{n-1} is core volume after n-1 loadings

ϕ^* is the adjoint flux matrix

$\tilde{\phi}$ is the perturbed flux matrix

R_n^1 and P_n^1 are cross section matrixes as shown below

$$P_n = \begin{pmatrix} P_{11} & P_{12} \\ P_{21} & P_{22} \end{pmatrix}$$

$$P_{11} = -\delta \sum a_1 + \delta \nu \sum f_1 + \delta \sum n_1 \quad \dots (3)$$

$$P_{12} = \delta \nu \sum f_z$$

$$P_{21} = \delta \sum n_1$$

$$P_{22} = -\delta \sum a_2$$

$$R_n = \begin{pmatrix} R_{11} & R_{12} \\ R_{21} & R_{22} \end{pmatrix}$$

$$R_{11} = \nu \sum f_1 + \delta (\nu \sum f_1) \quad \dots (4)$$

$$R_{12} = \nu \sum f_2 + \delta (\nu \sum f_2)$$

$$R_{21} = 0$$

$$R_{22} = 0$$

What we would like to be able to measure is shown in equation 5.

$$\rho_a = \frac{\int_{V_0}^{V_1} \phi_1^* P_1 \tilde{\phi} dv + \dots + \int_{V_{n-1}}^{V_n} \phi_1^* P_n \tilde{\phi} dv + \dots + \int_{V_{L-1}}^{V_L} \phi_1^* P_L \tilde{\phi} dv}{\int_0^{V_L} \phi_1 R_2 \tilde{\phi} dv}$$

where L is the final loading.

By substitution of equation 2 into equation 5, we obtain the correction factor, $K_n L$ for the individually measured reactivities.

$$\begin{aligned}
 \rho_a = & \rho_1 \frac{\int_0^{v_1} \phi_1^* R'_1 \tilde{\phi}_1 dv}{\int_0^{v_1} \phi_1^* R_L \tilde{\phi}_1 dv} \cdot \frac{\int_{v_0}^{v_1} \phi_1^* P_1 \tilde{\phi}_1 dv}{\int_{v_0}^{v_1} \phi_1^* P'_1 \tilde{\phi}_1 dv} + \\
 & \dots + \underbrace{\rho_n \frac{\int_0^{v_n} \phi_{n-1}^* R'_n \tilde{\phi}_n dv}{\int_0^{v_n} \phi_{n-1}^* R_L \tilde{\phi}_n dv} \cdot \frac{\int_{v_{n-1}}^{v_n} \phi_{n-1}^* P_n \tilde{\phi}_n dv}{\int_{v_{n-1}}^{v_n} \phi_{n-1}^* P'_n \tilde{\phi}_n dv}}_{K_n L} + \dots (6) \\
 & \dots + \rho_l \frac{\int_0^{v_l} \phi_{l-1}^* R'_l \tilde{\phi}_l dv}{\int_0^{v_l} \phi_{l-1}^* R_L \tilde{\phi}_l dv} \cdot \frac{\int_{v_{l-1}}^{v_l} \phi_{l-1}^* P_l \tilde{\phi}_l dv}{\int_{v_{l-1}}^{v_l} \phi_{l-1}^* P'_l \tilde{\phi}_l dv}
 \end{aligned}$$

Using the measured control rod positions to volume weight the core cross sections, the FOG * program was run. The adjoint and perturbed fluxes were thus obtained and used in equation 6 to determine the correction factors for the incremental loadings. The resulting corrected excess loading curve is shown in Figure 710-23 where it is compared with the measured (uncorrected) curve and the calculated curve.

* The FOG One-Dimensional Neutron Diffusion Equation Codes, by H. P. Flatt, January 1961. NAA-SR-6104

2. Two-Region Calculation of Excess Reactivity

Excess reactivity was also calculated using the axial banked control rod positions, recorded at each loading increment. A poison search was made using the FOG program to find the absorption cross section which would provide a shutdown k_{eff} of 0.94. This value was used throughout the whole study. Radial flux calculations were run for each fuel loading, both with and without the poison. Using the leakages calculated by the program, radial bucklings were determined for both groups. The cross sections in the core were flux, adjoint flux, and volume weighted to provide core cross sections. Axial FOG runs were then made using these bucklings and cross sections. The buckling from the radial calculation without poison was used for the lower reflector and lower uncontrolled core, and the buckling from the radial calculation with poison was used for the upper controlled core and upper reflector. This technique approximates the correct leakage. For loading X, the critical rod position for loading X and X-1 was run. The difference between the two reactivities is the excess loaded at that loading. Summing over all loadings provides the total excess reactivity.

The resulting excess loading curve is also shown in Figure 710-23. It follows the analytical curve quite closely.

3. Extrapolation of Initial Loading Increments

To obtain a "lower limit" estimate of excess reactivity loaded, the migration area M^2 of the lattice was experimentally determined as follows.

Following the method used by Dr. F. L. Fillmore in the HNPF dry critical calculations, k_{eff} was plotted as a function of geometrical buckling, as defined below:

$$B^2 = \left(\frac{\pi}{H}\right)^2 + \left(\frac{2.405}{R + \delta}\right)^2 *$$

* Glasstone and Edlund; The Elements of Nuclear Reactor Theory, 1952, page 214.

where:

H = Height of reactor with axial reflector savings

δ = Radial reflector saving

R = Effective core radius $\sqrt{\frac{NA}{\pi}}$

where:

N = number of fuel elements loaded and

A = cross sectional area associated with each fuel element

For $N < 95$, $A = 870 \text{ cm}^2$

For $N > 95$, $R = \sqrt{\frac{95.870}{\pi} + \frac{(N-95) A'}{\pi}}$ where $A = 724 \text{ cm}^2$

This change in the expression for R at $N = 95$ is due to the fact that there is one control rod for every 5 elements up to element 95, but no additional control rods for fuel elements in excess of 95.

The above expression strictly applies only to a bare homogeneous thermal reactor, but was found to yield a migration area for the dry critical loading within 8 percent of that calculated.

It can be shown that:

$$1 - k_{\text{eff}} \cong M^2 B^2 - (k_{\infty} - 1)$$

Therefore, a plot of $1 - k_{\text{eff}}$ vs B^2 should yield a curve which is approximately a straight line, with a slope equal to M^2 , and a y-intercept equal to $k_{\infty} - 1$. Figure 710-1 is the plot of $k_{\text{eff}} - 1$ vs $B^2 \times 10^6$; the deviation from a straight line is caused largely by the flux perturbations already discussed.

Plotted on the same graph is the multigroup calculation of k_{eff} vs B^2 . It can be seen that this curve agrees fairly well with the first four experimental points. At a fuel loading of 140 elements, k_{eff} was calculated to be 1.069, giving an excess reactivity of 6.5 percent.

The dashed curve is a straight line extrapolation of the first two experimental points. This curve extrapolates to a k_{eff} of 1.063 at the 140 element loading, or an excess reactivity of 5.9 percent.

The calculated value of M^2 for the "wet" lattice is 422 cm^2 at 350 F. The slope of the dashed curve is 435 cm^2 , within 4 percent of the calculated value. This agreement lends weight to the argument that the first few experimental measurements of excess reactivity are essentially correct, and that the extrapolation to the 140 element loading gives a valid estimate of the excess reactivity.

C. SUMMARY

The following table summarizes the various estimates of excess reactivity at the 140 element loading.

TABLE XIX
ESTIMATE OF EXCESS REACTIVITY - 140 ELEMENT LOADING

Method	k_{eff}	$\rho \left(\% \frac{\Delta k}{k} \right)$
Measured (by summing incremental excesses)	1.109	9.8
Perturbation correction	1.074	6.9
Axial two-region calculation	1.072	6.0
Extrapolation of first two loading increments	1.063	5.9
Calculated prior to loading	1.069	6.5

Examination of the numbers in the above table shows that the best estimate of excess reactivity actually loaded is between 5.9 and 6.9% $\frac{\Delta k}{k}$. The calculated value of 6.5% $\frac{\Delta k}{k}$ lies almost in the middle of the range.

As will be discussed in the report on control rod calibration, (AI-PC-750), the excess determined by control rod calibration curves is 983.1 cents, or about 6.8% $\frac{\Delta k}{k}$. Since this value also lies in the estimated range, it is suggested that the excess determined by control rod calibration be accepted as the "standard".

HALLAM NUCLEAR POWER FACILITY

Post Critical Test Procedure

AI-PC-740

Axial Power Distributions

Test Procedure Submitted for Approval 6/30/61 James E. Ours
Date Atomic Energy International

Test Procedure Reviewed 10/25/62 J. E. Simpson
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Commission

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AI-PC-740

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HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
AXIAL POWER DISTRIBUTIONS, AI-PC-740

I. PURPOSE OF TEST

The purpose of this test was to obtain axial power distributions in representative fuel channels throughout the reactor core and to determine the effect of different rod patterns on the axial power distribution.

II. DISCUSSION OF TEST PERFORMANCE

Originally, axial power distribution was to be determined by using the incore fission chambers as uncompensated ion chambers. The signal received from the chamber is proportional to the gamma and neutron flux. If the gamma activity is insignificant, the signal is proportional to the neutron flux which is directly proportional to the power produced at the position of the chamber. The measurement was to be done at specific axial positions in chosen fuel channels, with the reactor power level <100 Kwt. Core loading configuration is shown in Figure 740-1. Control rod core positions and gang designations are listed in Table I.

Prior to the test, several traverses were made for the purpose of adjusting instrumentation. During these trials several problems were uncovered. Several traverses of a given fuel channel indicated a power distribution which was completely unrealistic. Reevaluation of the method indicated that this anomaly was due to the fact that the gamma activity of Na^{24} is significant, and thus, the power profile was nothing more than a function of Na^{24} volume. To eliminate this error it was decided to use the fission chambers as neutron counters. Operation at a power level of about 100 Kwt yielded count rates of 100,000 to 200,000 cps. These count rates are considerably higher than the counting instrumentation can measure accurately. A period of about 100 sec was established and data were collected from the three fission chambers. The plot of count rate versus time indicated a straight line relationship up to a count rate of 10,000 cps after which irregularities in the counting system began to appear (see Figure 740-2). It was then decided that the reactor would be operated at a slightly subcritical level to obtain lower count rates.

TABLE I
CONTROL ROD CORE POSITIONS
AND GANG DESIGNATION

Gang 1	Gang 2
C-78; CR-14	C-38; CR-5
C-86; CR-15	C-44; CR-7
C-94; CR-16	C-50; CR-8
C-102; CR-17	C-56; CR-10
C-110; CR-18	C-62; CR-11
C-118; CR-19	C-68; CR-13
Gang 3	Gang 4
C-20; CR-2	C-16; CR-6
C-28; CR-3	C-24; CR-9
C-36; CR-4	C-32; CR-12
Gang 5	
C-1; CR-1	

C-Reactor Core Channel Designation
CR-Control Rod Designation

On the basis of the information obtained during the trial traverses, a new procedure for determining the axial power distribution was developed. The in-core fission chambers were used as neutron counters and were placed in selected fuel channels after the fuel elements had been relocated. The reactor was made critical with the three chambers at the expected position of maximum axial power. The selected controlling gang was inserted until the fission chamber in the channel to be traversed read 10,000 cps. The worth of the gang inserted was now determined and the subcritical k_{eff} calculated. All traverses were done at a "slightly subcritical" level with $k_{\text{eff}} \geq .995$. The chamber in the channel to be traversed was repositioned at predetermined axial levels as determined by the length of chamber cable inserted. These positions are accurate within ± 1 inch. At each position, the traversing channel was read (25 samples taken at each point) along with the other two, which remained in a fixed position. The fixed channels gave an indication of any changes in flux level during a traverse. This procedure was repeated for various core positions and rod patterns. In an effort to determine the reliability of the data, frequent checks of the statistical purity of the counting system were made in accordance with SOP No. 5902. If at any time the statistical purity deviated from 1 ± 0.1 , corrective maintenance was performed.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

To determine axial peak-to-average power ratios for the different axial distributions, the average relative power over the active core portion of a channel was obtained by graphically integrating the area under the distribution curve and setting this area equal to that of a rectangle fit over the length of the active core. Thus, the height of the rectangle was equal to the average relative power value. The axial peak-to-average ratio was then the ratio of the axial distribution peak value to that of the average.

Points at selected axial core positions for the distribution curves were obtained from the average of the 25 neutron count rates. This neutron count rate is proportional to power. Twenty-five counting samples were obtained at each data point giving a total data sample with a maximum statistical error of two

percent. The traversing fission chamber perturbed the neutron flux level sufficiently to require correction of the data. Since this perturbation was not to the extent of requiring the movement of control rods, correction of the data was accomplished by normalizing to the count rate readings from the fixed incore fission chambers at each data point. The resulting axial power distributions are given in Figures 740-3 through -26.

In order to investigate axial power distributions with the expected "normal operating pattern" (gangs 3, 4, and 5 - full in; gang 2 - full out; gang 1 - positioned for criticality) fission chamber traverses were made in channels C-9, C-71, C-46, C-134, and C-2. (See Table II and Figures 740-3, -4, -5, and -24, -25, and -26.) Additional traverses with control rod patterns D through O (Table II) were made to investigate axial power distributions for various positions of gang 1. The axial distribution curves and the resulting peak-to-average ratios indicated that no axial power distribution problems exist with gang 1 at different axial positions (see Figures 740-6 through 740-17). Positioning of the control rods at 12 inches, rather than full in, to facilitate the certainty of a "snubber up" condition during normal operations, was also checked. The results indicated no significant effect on axial power distribution. (See Figures 740-25 and 740-26.)

As anticipated, the poorest axial peak-to-average power condition occurred with the "all rods banked" configuration. With this configuration, the flux in the upper part of the reactor is depressed, resulting in an axial flux peak in the lower portion of the core and peak-to-average axial power ratios of 2.252, 2.625, and 2.363, for C-9, C-46, and C-71, respectively. (See Figures 740-18, -19, and -20.) Therefore, the "all rods banked" configuration should not be used in normal operation.

In the event it becomes necessary to control with the central control rods, axial traverses were performed in C-9, C-46, and C-71 with central rods controlling. (Control rod patterns G through O, S, T, and U in Table II.) Results indicated peak-to-average ratios and axial power distributions which were acceptable (see Figures 740-9, through -17, and -21, -22, -23).

Using the fission chamber, a traverse was made in C-2 to determine the axial thermal neutron distribution above the active core. The data were plotted and compared with the calculated curve. Since the calculated curve was based on an upper plenum sodium temperature of approximately 950°F, while the data

TABLE II
AXIAL POWER DISTRIBUTION PEAK-TO-AVERAGE RATIOS

Control Rod Pattern	Control Rod Configuration (inches out)	Fuel Channel Traversed	Peak to Average Ratio	Control Rod Pattern	Control Rod Configuration (inches out)	Fuel Channel Traversed	Peak to Average Ratio
A	Gang 1 78.9 Gang 2 out Gang 3 12.0 Gang 4 12.0 Gang 5 12.0	C-9	1.499	M	Gang 1 in Gang 2 out Gang 3 12.0 Gang 4 102.2 Gang 5 12.0	C-71	1.486
B	Gang 1 76.2 Gang 2 out Gang 3 12.0 Gang 4 12.0 Gang 5 12.0	C-71	1.565	N	Gang 1 in Gang 2 out Gang 3 12.0 Gang 4 100.7 Gang 5 12.0	C-9	1.586
C	Gang 1 78.8 Gang 2 out Gang 3 12.0 Gang 4 12.0 Gang 5 12.0	C-46	1.609	O	Gang 1 in Gang 2 out Gang 3 12.0 Gang 4 105.7 Gang 5 12.0	C-46	1.568
D	Gang 1 117.0 Gang 2: CR-5, CR-8, and CR-11 at 63.0 CR-7, CR-10, and CR-13 out Gang 3 12.0 Gang 4 12.0 Gang 5 12.0	C-71	1.544	P	Gang 1 48.4 Gang 2 48.4 Gang 3 48.4 Gang 4 48.4 Gang 5 45.7	C-71	2.363
E	Gang 1 117.0 Gang 2: CR-5, CR-8, and CR-11 at 68.7 CR-7, CR-10, and CR-13 out Gang 3 12.0 Gang 4 12.0 Gang 5 12.0	C-46	1.523	Q	Gang 1 48.4 Gang 2 48.4 Gang 3 48.4 Gang 4 48.4 Gang 5 48.0	C-9	2.252
F	Gang 1 117.0 Gang 2: CR-5, CR-8, and CR-11 at 67.5 CR-7, CR-10, and CR-13 out Gang 3 12.0 Gang 4 12.0 Gang 5 12.0	C-9	1.482	R	Gang 1 48.4 Gang 2 48.4 Gang 3 48.4 Gang 4 48.4 Gang 5 42.5	C-46	2.625
G	Gang 1 39.0 Gang 2 out Gang 3 12.0 Gang 4 68.1 Gang 5 12.0	C-71	1.625	S	Gang 1 out Gang 2 12.0 Gang 3 40.5 Gang 4 out Gang 5 in	C-71	1.400
H	Gang 1 39.0 Gang 2 out Gang 3 12.0 Gang 4 69.4 Gang 5 12.0	C-9	1.665	T	Gang 1 out Gang 2 12.0 Gang 3 43.7 Gang 4 out Gang 5 in	C-9	1.453
I	Gang 1 39.0 Gang 2 out Gang 3 12.0 Gang 4 71.5 Gang 5 12.0	C-46	1.684	U	Gang 1 out Gang 2 12.0 Gang 3 39.4 Gang 4 out Gang 5 in	C-46	1.494
J	Gang 1 12.0 Gang 2 out Gang 3 12.0 Gang 4 95.1 Gang 5 12.0	C-71	1.500	V	Gang 1 77.1 Gang 2 out Gang 3 12.0 Gang 4 12.0 Gang 5 12.0	C-134	1.612
K	Gang 1 12.0 Gang 2 out Gang 3 12.0 Gang 4 95.5 Gang 5 12.0	C-9	1.615	W	Gang 1 74.7 Gang 2 out Gang 3 12.0 Gang 4 12.0 Gang 5 12.0	C-2	1.488
L	Gang 1 12.0 Gang 2 out Gang 3 12.0 Gang 4 102.2 Gang 5 12.0	C-46	1.580	X	Gang 1 75.2 Gang 2 out Gang 3 12.0 Gang 4 12.0 Gang 5 in	C-2	1.554

were taken at $450 \pm 10^\circ\text{F}$, the necessary correction was made before comparing (see Figure 740-27). The experimental curve indicates the presence of noticeable reflector worth and compares favorably with the calculated, as does the axial distribution in the sodium.

The following conclusions were drawn on the basis of this test:

- a) Any operation of all the rods as a bank should be discouraged.
- b) The use of adjacent gangs in a banked configuration at the same axial position should be discouraged for positions between 39.0 inches and 117.0 inches. If operation in these configurations is necessary, the axial temperature distribution should be observed closely.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with post critical test AI-PC-740, Axial Power Distributions, as listed in NAA-SR-5700 (Supplement 4) page XI-(c)-41 with remarks regarding the test to explain compliance with the criteria.

- a) Criterion: The relative axial power distribution in the core has been determined in representative process channels.

Remarks: With the reactor slightly subcritical, axial traverses were completed at five different core radii. At each traverse position axial power distributions were obtained for numerous control rod patterns. Evaluation of the relative axial power distribution data yielded sufficient information to determine relative axial power distributions.

- b) Criterion: The attenuation of neutrons from the active core through the upper sodium pool has been determined.

Remarks: A traverse into the upper plenum from the top of the active core was performed with a resulting axial thermal neutron distribution that compared very favorably with the calculated distribution.

- c) Criterion: From data obtained, the range of vertical control rod travel that permits the minimum peak-to-average axial power distribution has been determined.

Remarks: For the cases examined, gang 1 at five different axial positions, peak-to-average axial power ratios in core channels C-9, C-71, and C-46 ranged from 1.400 to 1.684.

V. EVALUATION

Axial power distributions were measured in the HNPF core to obtain qualitative information for use in selection of the at-power control rod configuration. The variety of control rod patterns investigated was not all-inclusive, but was limited to those cases which were expected to give the worst result (all rods equally withdrawn) or were expected to give axial peak-to-average power ratios < 1.65 . In general, the latter class of patterns was selected so that reactor control would be effected with only one rod bank, with the remaining banks either full in (12 in. withdrawn) or fully withdrawn. There was no attempt to measure the axial power distribution for the rod pattern finally selected for operation; however, enough information was obtained in this test to indicate that the selected pattern might be feasible, and to justify its use until a better measurement could be obtained from the thermocoupled fuel elements.

Quantitative interpretation of the results of this test is difficult for two reasons: (1) the test was carried out in a perturbed core, since the fission chamber thimbles replaced fuel elements, and (2) the dimensional resolution of the chambers was too poor to show local flux peaking effects, which might be expected near the tips of control rods.

In addition to producing useful data for reactor operation, this test yielded verification of calculations concerning neutron attenuation in the reactor upper plenum.

In selecting an operating configuration for the control rods, the axial peak-to-average power is always a consideration, but rarely a limitation. Since the useful fuel lifetime is peak exposure limited, the lowest peak-to-average ratio obtainable under the operating conditions is the most desirable. Thus, if the reactor operates at a nearly constant load, it is possible to adjust the control rods to minimize the axial peak-to-average power ratio. However, if the reactor is a load-following device required to make frequent large percentage changes in load, the rod pattern must be such that the radial power distribution is not significantly affected by the rod motion required to make power changes and to

compensate for the changes in xenon poisoning. Load following is then best done when many rods move only a short distance, and in such a configuration there may be only minimum control over the axial peak-to-average power.

Figure 740-1. Core Loading Configuration for Determinations of Axial Power Distributions

Figure 740-2. Typical Plot of Incore Fission Chamber Count Rate for a Rising Reactor Period Versus Time

Figure 740-3. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern A)

Figure 740-4. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern B)

Figure 740-5. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern C)

Figure 740-6. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern D)

Figure 740-7. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern E)

Figure 740-8. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern F)

Figure 740-9. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern G)

Figure 740-10. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern H)

Figure 740-11. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern I)

Figure 740-12. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern J)

Figure 740-13. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern K)

Figure 740-14. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern L)

Figure 740-15. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern M)

Figure 740-16. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern N)

Figure 740-17. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern O)

Figure 740-18. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern P)

Figure 740-19. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern Q)

Figure 740-20. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern R)

Figure 740-21. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern S)

Figure 740-22. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern T)

Figure 740-23. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern U)

Figure 740-24. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern V)

Figure 740-25. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern W)

Figure 740-26. Axial Power Distribution for Selected Fuel Channel (Control Rod Pattern X)

Figure 740-27. Axial Distribution of Thermal Neutrons Above the Active Core

* Schiffries, W. P., "Axial Fast and Thermal Neutron Flux Above the HNPf Reference Core Design", TDR No. 2907, October 7, 1958

HALLAM NUCLEAR POWER FACILITY

Post Critical Test Procedure

AI-PC-750

Control Rod Calibration

Test Procedure Submitted for Approval

6/30/61
Date

James E. Owens
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Test Procedure Reviewed

10/25/62
Date

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Test Approved for Performance

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Test Completed

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Date

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Commission

HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
CONTROL ROD CALIBRATION

AI-PC-750

Test and Analysis
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P. T. BURNETT

Evaluation
by
O. G. JENKINS
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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

May 27, 1964

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HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
CONTROL ROD CALIBRATION, AI-PC-750

I. PURPOSE OF TEST

This test was performed to obtain data for use in future reactivity measurements. A secondary purpose was to obtain backup measurements of the operational excess reactivity and the shutdown margin.

II. DISCUSSION OF TEST PERFORMANCE

A. EXPERIMENTAL CONFIGURATION

The 19 control rods were calibrated at zero power (< 100 kw) in operational gangs and sub-gangs. The radial rod-gang designations are listed in Table I, and illustrated in Figure 1. Vertically, the zero position of the rods is three inches above the bottom of the fuel. Fully (156 in.) withdrawn the tip of a rod is even with the top of the fuel.

With the obvious exception of gang 5, no individual rod calibrations were performed, nor were any calibrations made of asymmetrical arrays of rods. (In actual operation at power, the rods are operated in an array that will maintain radial flux symmetry.)

The entire length of gang 5 was incrementally calibrated against the remaining 18 rods equally banked. Short vertical segments of each of the other gangs were calibrated on withdrawal from the "all gangs equally banked" critical configuration. The remaining calibrations were performed in configurations as close as possible to the operating array.*

B. TEST METHOD

Once the reactor was brought to criticality the startup source was separated; i. e., the antimony capsule was withdrawn from the beryllium sleeve.**

*

It was originally proposed that the at-power rod configuration be: gangs 3, 4, and 5 at 12 inches, gang 2 out and gang 1 positioned for criticality. Following the completion of the control rod test and phase I of the xenon test, it became obvious that: (1) the remaining reactivity associated with gang 1 would not compensate for the increase in xenon poisoning from zero to full power equilibrium, and (2) that changes in xenon poisoning during load following would be of such magnitude that compensation by one rod gang would cause gross changes in radial flux which would require many fuel element orifice adjustments. Therefore, the operating configuration was changed to: Gangs 3 and 5 at 12 inches, gang 2 out and gangs 1 and 4 equally withdrawn. Subsequently, an instrumented, immovable poison rod was installed in one position in gang 3. To prevent the inadvertent withdrawal of the remaining two rods in gang 3 and the concomitant radial flux tilt, all three drive units in gang 3 were removed and that gang was maintained fully inserted. The use of gangs 1 and 4 to compensate reactivity changes spreads the reactivity change across the core, and reduces, but does not completely eliminate, the change in radial flux with bank motion.

**

This was done for two reasons: (1) The flux at criticality is lower without the source, and, with an administratively fixed upper limit on flux level, this increased the range over which the period measurement could be made; and (2) the presence of source neutrons perturbs the period measurement. The method of data analysis used assumes that if the flux level is sampled at known time intervals the following equation can be solved:

$$\varphi(t + \Delta t) = \varphi(t)e^{\Delta t/\tau} \quad \text{or} \quad \tau = \Delta t \ln \frac{\varphi(t)}{\varphi(t + \Delta t)}$$

All of the terms on the right hand side of the equation are known from measurement. However, if there is a strong source present, one must be able to solve the following equation:

$$\varphi(t + \Delta t) = \varphi(t)e^{\Delta t/\tau} + \varphi_s$$

$$\tau = \Delta t \ln \left[\frac{\varphi(t)}{\varphi(t + \Delta t) - \varphi_s} \right]$$

where τ is the reactor period

φ_s is the source flux

It is quite simple to measure flux ratios in the first case, and in practice quite difficult to obtain the relative flux magnitudes required in the second case.

The differential reactivity worth of a segment of a gang was obtained from the measured change in reactor period caused by the incremental movement of the gang. Reactor periods and reactivity before and after gang movement were determined from previously prepared tables,¹ which converted the measured flux ratios to reactor periods. The period and reactivity tables had been calculated from the inhour equation and the best values of ℓ/β_{eff} and ϵ for the core. For each reactor period measurement an average from each of two period measuring systems was used to determine the period.*

After an increment of a gang had been calibrated it was necessary to reposition a compensating gang before the gang under test could be calibrated over the next increment. To obtain the integral worth of a gang, the curve described by the differential (incremental) worths was graphically integrated.

Fuel, moderator and coolant temperatures were continuously monitored during the test. The average core temperature remained between 535°F and 545°F throughout the test. Since it had been shown earlier (AI-PC-760, Zero Power Coefficients of Reactivity) that the control rods had no measurable temperature coefficient in the range from 350°F to 540°F, the only temperature drifts of concern were those that took place during the measurement of a reactivity increment. The core isothermal temperature coefficient at 534°F is 0.08¢/°F. Since the temperature change during the measurement of a reactivity increment rarely exceeded 1°F, only the data for the analysis shown in the appendix were temperature corrected.

¹ HNPF Operations Manual, Vol. IV, Section D "Ratio Period Reactivity Index".

* The two period measuring systems were identical, and each consisted of a high temperature fission counter, preamplifier, linear count rate meter and amplifier, scaler and printer. The system shared a time base and dual preset scaler, which gated the scalers on and off. The fission counters were located in dry thimbles on the perimeter of the core and were at an elevation slightly below core midplane. Figure 2 is a block diagram of a typical system.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

Curves of differential worth vs position are shown for each gang in figures 750-3 through 7. Note that only a typical curve is shown for gangs 3 and 4. These gangs were calibrated independently, but for ease of analysis, data from the two cases were merged. Merged data are good for either of the two gangs.

From observations of the differential worth curves it can be seen that there are three different axial flux regions. In the lower core region the differential worth curves follow a sine square distribution.* This is the active region of the core.

The central axial flux region appears to be one of enhanced worth.** In this region the compensating gang is some vertical distance from the gang being calibrated.

The upper region starts near the point at which the tips of the gang being calibrated and the tips of the compensating gang cross and continues to the top of the core. This is the exponential region.† It should be noted that gang 5 in either the standard configuration or the all rods banked configuration appears to have no enhanced region. The worth of gang 5 is so small in either case that the compensating gang(s) did not move enough to cause a measurable shift in flux distribution.

To determine the extent of the multiplying region, a digital computer code was used to obtain a weighted least squares fit to the data for each gang. The

* In the sine square distribution the differential worth follows the distribution:

$$\frac{dp}{d\ell} = A \sin^2 v(h-\ell). \text{ Where:}$$

$\ell \equiv$ height of the tip of the gang above the bottom of the core

$h \equiv$ the distance below the bottom of the core at which the flux extrapolates to zero.

** The insertion of a control rod causes a local flux depression and distant, compensating flux increase. The worth of a control rod operating in the region of the increased flux is increased or enhanced.

† In the exponential region the differential worth follows the distribution

$\frac{dp}{d\ell} = A \sinh^2 \gamma (h-\ell)$, where h is the distance from the bottom of the core to the point above the core at which the flux extrapolates to zero.

probable error of each datum was calculated assuming the uncertainty in rod movement was a constant 0.1 inch (the smallest increment of motion indicated by the rod digitizers), and that the uncertainty in reactivity change was a constant 0.2% of the reactivity difference (the accuracy of the tables) except in the cases when the initial or final reactivity was an estimate of criticality. Based on earlier experience, it was assumed that criticality cannot be measured more accurately than $\pm 1\%$.

The differential worth curves of the multiplying regions for all gangs were successfully fitted to the sine square distribution. The fit in the exponential region was successful for gangs 1, 3, and 4 and gang 5 vs the bank.

The χ^2 test was used as the measure of success or failure of a fit to a set of data. Fits having probabilities which were greater than 1% were arbitrarily assumed to be acceptable. Using this criterion, all but one of the eight cases in which the code converged were successful. The χ^2 test results are given in Table II. The set of individual error probabilities given by the χ^2 test indicates that the choice of statistics was reasonable.

After the test results had been analyzed to obtain integral worth curves (Figures 750-8 and -12) for each gang in the standard operating configuration, 11 different critical configurations were analyzed using the integral curves. The gang worths are summarized in Table III. The results of the analysis are given in Table IV. For the first 10 configurations listed, the agreement among values of shutdown margin and excess are very good, but the last configuration gives significantly different results. The first 10 cases are all variations of the so called standard operating configuration and the flux distributions within the reactor are quite similar to the flux distributions in which the gangs were calibrated. Therefore, the agreement among the first 10 configurations is not surprising. The eleventh case, all rods banked, gives values of shutdown margin and excess quite different from the 10 preceding cases. Thus the limit of applicability of the integral worth curves is made obvious. Any calibration curve can validly be applied only to configurations similar to the one that produced the curve.

The 10 critical configurations for which the calibration curves apply give a clean core average value of the excess at 540°F of 980¢ or 6.36% $\Delta k/k$, which

is in excellent agreement with the calculated value of 6.5% $\Delta k/k$.^{*} The 10 critical configurations are not independent measurements of the excess and shutdown margin, but are really 10 estimates of these values for one flux condition. For a reliable measure of excess and shutdown margin, these quantities must be shown to be invariant when measured in any flux distribution.

Only one other flux distribution was investigated, the "all rods equally withdrawn" configuration, and the investigation was too brief for complete analysis. The crude analysis that was performed yielded a value of clean core excess of $(5.28 \pm 1.3)\% \Delta k/k$. This analysis is described in the appendix.

The technical specifications for the reactor limit the reactivity insertion rate to 0.03% $\Delta k/\text{sec}$. Any of the proposed operating configurations has an "all rods moving" reactivity insertion from criticality well below this limit. Such was not the case of the "all gangs equally banked" critical configuration. Over the first 0.6 in. of withdrawal from criticality (all that could be calibrated), the average reactivity insertion rate was 0.0338% $\Delta k/\text{sec}$. Operation with banked rods was discontinued as a result of this measurement.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with post critical test AI-PC-750, Control Rod Calibration, as listed in NAA-SR-5700 (Supplement 4) page XI-(c)-49, with remarks regarding the test to explain compliance with the criteria:

- a) Criterion: Reactivity worths have been determined for each rod or group of rods specified in the procedure.
Remarks: The control rods have been calibrated in operational gangs. The reactivity worth of each gang has been measured for the flux distribution produced by the standard operating configurations.
- b) Criterion: The effect of shadowing on control rod worths has been determined.
Remarks: Since the control rods were calibrated in operational gangs no inter-rod effects were measured. Investigation of the interactions between gangs led to the qualitative determination of both shadowing and enhancement effects due respectively to short and long range interactions.

*Preoperational test report AI-PC-710, Wet Excess Fuel Loading, Evaluation section.

V. EVALUATION

The primary purpose of this test was to obtain control rod gang calibrations, in specified operating patterns, for application in future reactivity measurements. Adequate fulfillment of the purpose is evidenced by the success of later applications.

A secondary purpose was to obtain additional values of excess reactivity and shutdown margin for comparison with values obtained during the wet excess loading test (AI-PC-710). The value obtained, $6.36\% \frac{\Delta k}{k}$ at 540°F , was in excellent agreement with the $6.5\% \frac{\Delta k}{k}$ calculated value obtained at 540°F during wet excess loading.

To obtain the maximum possible rate of reactivity insertion, all rods were banked in a position to attain criticality, then withdrawn (as a bank) 0.6 inches to obtain a maximum reactivity insertion rate of $0.0338\% \Delta k/\text{sec}$. Elaboration on the means of obtaining this value is presented:

$$\frac{d}{dt} \Delta \rho = \frac{14.2\text{¢}}{0.6 \text{ in.}} \times 0.00693\% \frac{\Delta k}{\text{¢}} \times 12.4 \frac{\text{in.}}{\text{min}} \times \frac{1}{60} \frac{\text{min}}{\text{sec}} = 0.0338\% \Delta k/\text{sec}.$$

The banked configuration used in obtaining the above value was previously known to be unacceptable for power operations, because of the resulting large radial and axial peak power distributions. However, in the event that shutdown reactivity checks are made with all rods banked, an administrative limitation will be imposed, precluding simultaneous withdrawal of all rods. This will assure that the maximum rate of $0.03\% \frac{\Delta k}{k}$ designated in the technical specifications will not be exceeded.

Test completion criterion b) designates that the effects of shadowing on control rod worths has been determined. This criterion was not entirely satisfied in that quantitative values were not obtained. Interaction between gangs was investigated during calibration of the rods in the 10 operating patterns, however, providing a qualitative determination of both short range shadowing effects and long range enhancement effects. This qualitative information is all that is needed under the pattern limitations imposed by the present fuel loading, related radial and axial power distributions, and associated temperature gradient limitations.

Although such action was not required, the test engineer fitted the experimental differential weight data to theoretical function curves and used the χ^2 test to determine how well the experimental data fitted the curves as compared with assumed variances. This additional analysis, which indicated a reasonable choice of statistics, lent credibility to the accuracy of the calibration curves.

APPENDIX
ANALYSIS OF DATA OBTAINED IN THE ALL GANGS
EQUALLY BANKED CONFIGURATION

The critical position for the all rods equally withdrawn case was determined to be 45.05 in. at 536°F. Table V lists the reactivity change and differential worth for each gang when individually withdrawn a short distance from the critical position. Additionally, the entire length of gang 5 was calibrated against the remainder of the bank. The differential and integral worth curves for gang 5 vs the bank are shown in Figures 750-7 and -12. Note that no information about the shutdown margin is gained from the calibration of gang 5 from elevation zero to the bank critical position. This calibration was performed with gang 5 unshadowed by the other gangs, thus giving this gang a higher worth than it would have when operating as part of the bank. However, that part above the critical position was shadowed when calibrated, and may provide information about the excess at 536°F. The remaining worth in gang 5 when completely withdrawn from the critical position is 55.2¢, and the sum of the incremental worths for 2.45 inch withdrawal of each gang is 58.44¢. If it is assumed that the worth of gang 5 relative to the entire bank is a constant, independent of position, then the operational excess is 805¢. However, since each $\Delta\rho$ is measured from critical, there is an assumed probable error for each $\Delta\rho$ of $\pm 1\text{¢}$, and this estimate of operational excess becomes $805 \pm 203\text{¢}$ or $(5.28 \pm 1.3)\% \Delta k/k$. (The dominant source of error is the $\sim 25\%$ uncertainty in the $\Delta\rho$ for gang 5.)

TABLE I
RADIAL ROD GANG - DESIGNATIONS

Gang No.	Control Rod Nos.	Gang Radius
1	14-19 inclusive	55.5"
2	5, 7, 8, 10, 11, 13	48.0"
3	2, 3, 4	27.5"
4	6, 9, 12	27.5"
5	1	0.0"

TABLE II
 χ^2 TEST RESULTS

Gang No.	Region	Function	Probability of Worse Result ^a
1	Multiplying	Sine Square	93.8%
1	Exponential	Hyperbolic Sine Sq	6.1%
2	Multiplying	Sine Square	27%
3 & 4	Multiplying	Sine Square	4.1%
3 & 4	Exponential	Hyperbolic Sine Sq	<1%
5	Multiplying	Sine Square	10.2%
5 ^b	Multiplying	Sine Square	83%
5 ^b	Exponential	Hyperbolic Sine Sq	1.4%

a) This is the probability that a set of data with the assumed statistics would give a poorer fit to the function.

b) Calibrated against the remaining rods equally banked.

NOTE: $\chi^2 = \sum_{i=1}^N \frac{(x_i - \bar{x})^2}{\sigma_i^2}$

The χ^2 test is a method of comparing a set of measured variances with theoretically expected or assumed variances. If the measured and expected variances are identical for each datum, or if the average ratio of measured to expected variance is unity, the probability will be 50%. Thus probabilities >(<)50% indicate that the assumed statistics are larger (smaller) than the measured.

TABLE III
INTEGRAL WORTH OF CONTROL ROD GANGS

Gang No.	Integral Worth (¢)	Worth per Rod (¢)
1	430	71.6
2	560	93.3
3	379	126.3
4	379	126.3
5	94	94
Total	1842	

TABLE IV
CRITICAL CONFIGURATIONS
(Zero Power, 540°F)

Gang Position (inches)					Excess (cents)	Shutdown Margin (cents)	Configuration Number
1	2	3	4	5			
73.25	out	in	in	in	984.5	857.5	1
57.2	out	in	in	out	972.5	869.5	2
out	67.1	in	in	out	967.5	874.5	3
out	80.05	in	in	in	982.0	860	4
out	26.05	out	in	in	989.5	872.5	5
out	25.5	in	out	in	978.0	870	6
out	75.9	12	12	12	986	856	7
out	62.95	12	12	out	975.5	866.5	8
52.8	out	12	12	out	978.5	863.5	9
68.25	out	12	12	12	986.5	855.5	10
45.05	45.05	45.05	45.05	45.05	1222.5	619.5	11

TABLE V

"ALL RODS BANKED" CONFIGURATIONS*

Gang No.	Position (inches)		$\Delta\rho^\dagger$ (cents)	$\frac{\Delta\rho^\dagger}{\Delta\ell}$ (cents/in.)
	$\Delta\ell$	$\bar{\ell}$		
1	2.45	46.275	11.19	4.56
2	1.35	45.725	8.91	6.60
2	1.1	46.95	7.83	7.11
2	2.45	46.275	16.74	6.84
3	2.45	46.275	13.42	5.48
4	2.45	46.275	13.08	5.34
5	2.45	46.275	4.01	1.64
All	0.6	45.35	14.20	23.7

* Critical position at 536°F was all rods at 45.05 inches.

† Corrected for temperature changes between the time critical was established and the time of measurement.

NOTE: To indicate possible trends more than a significant number of figures are used in the above values.

- 265 REFLECTOR FILLER ELEMENT (GRAPHITE)
 212 DUMMY FUEL ELEMENT (GRAPHITE)
 99 FUEL ELEMENT, CORE POSITION 99 (-12% CHANGE IN RELATIVE POWER)
 94 CONTROL ROD (GANG 1)
 $\frac{C}{4}$ SOURCE

Figure 750-1. Core Map Showing Control Rod Locations

Figure 750-2. Typical Period Measuring System

Figure 750-3. Gang 1 Differential Worth

Figure 750-4. Gang 2 Differential Worth

Figure 750-5. Gangs 3-4 (Typical) Differential Worth

Figure 750-6. Gang 5 Differential Worth

Figure 570-7. Gang 5 (vs Bank) Differential Worth

Figure 750-8. Gang 1 Integral Worth

Figure 750-9. Gang 2 Integral Worth

Figure 750-10. Gangs 3-4 (Typical) Integral Worth

Figure 750-11. Gang 5 Integral Worth

Figure 750-12. Gang 5 (vs Bank) Integral Worth

HALLAM NUCLEAR POWER FACILITY

Post Critical Test Procedure

AI-PC-760

Zero Power Coefficients Of Reactivity

Test Procedure Submitted for Approval

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Date Atomic Energy International

Test Procedure Reviewed

10/5/62 W. E. Lingenfelter
Date Atomic Energy Commission

Test Approved for Performance

10/10/62 James E. Quinn
Date Atomic Energy International

Test Completed

10/20/62 James E. Quinn
Date Atomic Energy International

Test Approved for Performance

10/10/62 R. C. Buntman
Date Atomic Energy Commission

Test Completed

10/20/62 R. C. Buntman
Date Atomic Energy Commission

HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
ZERO POWER COEFFICIENTS OF REACTIVITY

AI-PC -760

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HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
ZERO POWER COEFFICIENTS OF REACTIVITY, AI-PC -760

I. PURPOSE OF TEST

The purpose of the zero power coefficients of reactivity test, AI-PC -760, was:

- a) To measure the change in reactivity associated with sodium flow variations in the isothermal core.
- b) To measure the change in reactivity associated with changes in the isothermal temperature of the core.
- c) To determine the variation in control rod worth with core temperature changes.

II. DISCUSSION OF TEST PERFORMANCE

A. ZERO POWER FLOW COEFFICIENT

With the reactor critical and low enough in power to render the core isothermal, the primary sodium flow rate was varied. Such a variation in flow at zero power produces no detectable temperature change. Thus, any effect on reactivity would be attributed to mechanical effects. The results of the test indicate that there is no zero power flow coefficient. It may thus be said that any mechanical changes produced by primary coolant flow variation have no effect on system reactivity.

Performance of this test at zero power makes it possible to better evaluate tests involving flow ramps at higher power levels; i.e., since the effect of flow ramps on reactivity at higher power levels might have involved both temperature and mechanical effects, mechanical effects have now been eliminated.

It was intended that this test be performed with fuel channel flow orifices full open, moderator coolant flow at 100%, and primary coolant flow ramped from zero to 100%. It was considered that these conditions would produce the maximum mechanical effects, thereby providing the maximum possible flow coefficient. In practice, it proved impossible to conduct the test under these extreme

conditions. With orifices and moderator coolant valve full open, the pressure drop across the core was so low as to produce low level alarms in the primary pumps when a flow ramp was attempted; nor could a flow ramp be satisfactorily initiated from zero flow. The conditions were modified by closing the fuel channel flow orifices to half open, closing the moderator coolant block valve, and ramping the flow from 100% down to 15%. During the flow ramp, the reactor was held level at Point "B" (minimum critical flux level) by manipulation of gang 5.

B. ISOTHERMAL TEMPERATURE COEFFICIENT OF REACTIVITY

Changes in the temperature of core components can be expected to produce changes in the available reactivity of the system. These changes are due to:

- 1) Changes in structure due to expansion and contraction, producing changes in neutron leakage.
- 2) Changes in material densities, producing changes in microscopic cross sections.
- 3) Changes in moderator temperature, producing shifts in the thermal neutron energy spectrum.
- 4) Changes in fuel temperature, producing Doppler broadening and increased resonance absorption in the fuel (U-238).

It is of importance to evaluate the reactivity changes associated with variation of system temperature. The most fundamental method by which this is done is to establish an isothermal temperature throughout the core and vary this temperature, taking care to keep all core materials at the same temperature. As the temperature is altered in this manner, some method must be exercised to measure concomitant changes in system reactivity. The ratio of change in reactivity to change in system temperature, $\left[\frac{\Delta \rho}{\Delta T}\right]$, is termed the isothermal temperature coefficient.

Preliminary to the measurement of the isothermal temperature coefficient, it was noted from the results of dry excess loading, (AI-P-1163),* that the isothermal temperature coefficient for the dry, 35 element core was about + 0.31 cent/F° in the range from 150 to 300°F. This information was used to estimate the sign and order of magnitude of the coefficient to be measured. The

*H. C. Kempt, "HNPF Dry Excess Loading," AI-P-1163, Atomics International (February 24, 1962).

addition of sodium should make the new coefficient slightly more positive; the addition of 105 more fuel elements should make the new coefficient more negative, more than off-setting the effect of sodium. The new coefficient was hence expected to be positive and slightly lower in magnitude than before. This information was used to establish an initial control rod configuration; i.e., to select a control gang for making measurements of reactivity changes and to provide a reasonable initial position of this gang. Gang 1, with a total worth of just over 200 cents (in the specified configuration) was selected as the "measuring" gang. Gang 1 was divided into two parts, gang 1A (CR-14, 16, and 18) and gang 1B (CR-15, 17, and 19). The two parts were initially positioned at about 100 in. (withdrawn). These two subgangs were alternately used during the test to measure changes in reactivity with temperature.

The method employed to measure reactivity changes with temperature was designed to minimize errors in positioning rods for criticality, as well as errors due to variation in rod worth with temperature changes. Initially, at 370°F, with gang 3, 4, and 5 full in, gang 2 full out, and gang 1A at 103.1 in., gang 1B was positioned to put the system on about a 150 sec period, (with gang 1B at 108.1 in., a period of 140 sec was measured). Then, gang 2 was inserted to level the flux during the ensuing temperature rise — care being taken not to change the position of any other gang. When the reactor temperature was leveled at 429°F, gang 2 was again fully withdrawn. This time, a 56.8 sec period was observed. The reactivity change due to the system temperature increase from 370 to 429°F was determined, from the inhour equation, to be 7.28 cents. Thus, this method assured that no error in rod position or rod temperature effect was introduced into the measurement.

This technique was used throughout the test in making reactivity measurements. When the predicted period at the next stable temperature level was estimated to be less than 50 sec, a new "base" period was established beforehand by repositioning gang 1A or 1B. For example, at 429°F, after the 56.8 sec period was measured, a new "base" period of 125.7 sec was established, by repositioning gang 1B to 104.0 in., again with gang 2 full out. Care was taken not to move any other gang (except gang 2) during the temperature rise.

Two instrument malfunction scrams occurred during these measurements, detracting from the accuracy the test personnel were attempting to preserve. In these instances, efforts were made to precisely reposition the rods, and a new "base" period was established.

Throughout the test, system temperatures were recorded periodically — at least hourly, and more often when stabilization of temperatures was sought. The thermocouples indicated in Table I were used.

TABLE I
TEST THERMOCOUPLE IDENTIFICATION

Thermocouple	Temperature
B - TI-14-9	Sodium (C-212 Middle T/C)
TRCp-1	Sodium Upper Plenum
A - TRC -2	Fuel channel Na exit T/C
TR-12-9	In-Fuel T/C (C-3)
C - TR-12-5	In-Fuel T/C (C-3)
TR-13-6	In-Fuel T/C (C-6)
T/C 1 of C-96	In-Fuel T/C (C-96)
TRC -3	Moderator Coolant
D - TI-9-2E	Moderator
TI-9-19E	Moderator
	Instrumented moderator can T/C's

The thermocouple signals marked A, B, C, and D were placed on a Sanborn recorder for more precise readings. The thermocouple marked B (sodium temperature) was used as the reference temperature. The thermocouple marked D (moderator temperature) was examined closely when system temperature was to be stabilized. The temperature was defined as stable when less than $1/3 F^{\circ}$ change in moderator temperature was observed in 1 hr. It is noted that determinations of reactivity variation with temperature were also made at intermediate, nonisothermal temperature conditions. These are considered to be less accurate than the stable measurements (Figure 760-1).

The system temperature at the start of the test was $370 F^{\circ}$. The preheat system and the sodium pumps were used to increase this temperature to the final value of $541 F^{\circ}$.

Figure 760-1. Change in Reactivity vs Average Core Temperature

C. CONTROL ROD WORTH VARIATION

Control rod calibrations involve a control rod or gang position change, which, in the critical system, adds or subtracts an increment of reactivity. The reactivity change is related to the rod movement and a calibration curve is obtained.

It is convenient to make control rod calibrations at zero power since, at that power level, a small change in system reactivity produces an insignificant temperature change. Thus, no power coefficient correction is required.

Rod calibration data obtained at zero power must be useful at other power levels — power levels at which the system temperatures have different values. It is of interest, therefore, to determine the effect of temperature on rod worth. As a part of the wet excess fuel loading (AI-PC-710),* gang 3 and gang 5 were calibrated over the entire lengths, using the differential period technique. At that time the core conditions and rod configurations were as follows:

- 1) Full operational core (140 elements)
- 2) Three fixed poison columns (C-5, C-8, and C-11)
- 3) Isothermal core temperature of 350°F
- 4) Gangs 3, 4, and 5 fully inserted
- 5) Gang 2 fully withdrawn
- 6) Gang 1 positioned for criticality
- 7) As gang 3 or gang 5 was withdrawn during calibration, compensation was made with gang 1.

To determine the effect of temperature on rod worth, gangs 3 and 5 were calibrated again, by exactly the same method and under identical conditions, with the exception that the isothermal core temperature was 541°F.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

A. ZERO POWER FLOW COEFFICIENT

With the fuel channel flow orifices half open and the moderator coolant block valve fully closed, the primary coolant flow was ramped down from 100 to 15%.

*J. R. Rohlf, "HNPF Excess Fuel Loadings," AI-PC-710, Atomics International (May 25, 1963).

During the flow ramp, the reactor was held level at Point "B" by manipulation of gang 5. Gang 5 was moved a total of 2.5 in. during the ramp. With gang 5 worth about 0.2 cent/in. in the range of movement, a total reactivity change of 1/2 cent is indicated. This is insignificant. It was concluded that no zero power flow coefficient exists for the HNPF.

B. ISOTHERMAL TEMPERATURE COEFFICIENT OF REACTIVITY

As the isothermal temperature of the core was increased from 370°F to 541°F, an increase of 17.53 cents in available reactivity was observed. This indicates an average isothermal temperature coefficient, in this temperature range, of + 0.103 cent/F°. The isothermal temperature coefficient in this temperature range is not precisely linear; at 400°F the coefficient has a value of + 0.123 cent/F°, and decreases to + 0.088 cent/F° at 500°F (Figure 760-2).

An extrapolation of the value of the isothermal temperature coefficient indicates that it may not be expected to reach zero before ~1500°F (Figure 760-2). This temperature is far above the average core temperature of 850°F at operating conditions. At 850°F, the extrapolated value is + 0.04 cent/F°. This supports the prediction of a positive steady-state power coefficient.

In Table II, a comparison is made of the previously calculated and the experimentally determined values for the isothermal temperature coefficient. The calculated values are for an isothermal temperature of 850°F; therefore, the experimental value is extrapolated to this temperature.

In increasing the system temperature from 370 to 541°F, temperatures were leveled at four points to obtain a reactivity measurement under isothermal conditions. These four stable temperature points are indicated in Table III with an asterisk. Intermediate reactivity measurements were also made with system temperatures increasing. The values measured at these intermediate points (Figure 760-1) are not as accurate as those obtained at the stable temperature points.

An extrapolation of these data is illustrated in Figure 760-3. It is seen that an increase of 40 cents, or about $0.3\% \frac{\Delta K}{K}$ in available reactivity is indicated as system temperatures increase from 350°F (normal cold startup) to 850°F (average operating temperature).

Figure 760-2. Isothermal Temperature Coefficient vs
Average Core Temperature Extrapolated

TABLE II
ISOTHERMAL TEMPERATURE COEFFICIENTS FOR HNPF CORE

Coefficient	Experimental [*]	NAA-SR-5700 [†]	A 81-710-1 [§]
Cents/F°	+0.04	+0.048	+0.036
$\times 10^{-5}/F^{\circ}$	+0.28	+0.33	+0.25

*Extrapolated to 850°F.

†"HNPF Summary Safeguards Report," April 15, 1961.

§C. L. Larson, "HNPF Reactor Core System Description," August 15, 1958.

TABLE III
SUMMARY OF REACTIVITY CHANGE WITH TEMPERATURE

System Temperature (°F)	Reactivity Change from 370°F $\Delta \rho$ (cents)
370*	0.0
385	1.92
403	4.01
429*	7.28
449	7.90
453	9.34
476	11.79
499*	14.06
507	14.88
515	15.41
522	16.28
529	17.04
538	17.61
541*	17.53

*Stable or isothermal temperatures.

Figure 760-3. Change in Reactivity vs Average Core Temperature Extrapolated

C. CONTROL ROD WORTH VARIATION

Gangs 3 and 5 were calibrated at 350°F (AI-PC-710, Wet Excess Fuel Loading) and again at 541°F, under otherwise identical conditions. The total worth of gang 5 was found to be 18.8 cents at both temperatures. The total worth of gang 3 apparently increased from 215 cents at 350°F to 218.5 cents at 541°F (Figure 760-4). This small change in total worth probably relates to an error in experimental determination. It is concluded that no change in actual rod worth is indicated as temperatures increase from 350°F to 541°F.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following is the test completion criterion associated with post critical test AI-PC-760, Zero Power Coefficient of Reactivity, as listed in NAA-SR-5700, Supplement 4, page XI(c) - 51, with the remark regarding the test to explain compliance with the criterion.

Criterion: The reactor isothermal temperature coefficient of reactivity has been determined in the range of 300°F to 500°F.

Remarks: The isothermal temperature coefficient of reactivity was determined in the range of 370°F to 540°F. The value of the isothermal coefficient of reactivity is + 0.123 cent/F° at 400°F and + 0.088 cent/F° at 500°F.

V. EVALUATION

A. ZERO POWER FLOW COEFFICIENT

The measurement of the zero power flow coefficient falls in the class of null experiments. Within the accuracy of the measuring technique, no coefficient was determined. The technique used can reasonably be expected to measure changes in reactivity as small as 4 cents; since experience has shown that a carefully adjusted critical position, one obtained by close observation of the flux trace on a chart, is usually less than $\pm 2\epsilon$ from zero excess. Thus, assuming that each critical position was in error by 2ϵ in opposite directions, the zero power flow coefficient is $\leq 4.71 \times 10^{-2} \epsilon / \% \text{ flow}$ or $3.26 \times 10^{-6} \Delta k / \% \text{ flow}$.

Figure 760-4. Gang 3 Worth Curve 140 Elements

The absence of a measurable flow coefficient verifies proper design and fabrication of the fuel elements with respect to the elimination of mechanical changes in fuel rod configuration with flow.

B. ISOTHERMAL TEMPERATURE COEFFICIENT OF REACTIVITY

The isothermal temperature coefficient was measured over the range of 370 to 541°F, and extrapolated from 541 to 850°F. This extrapolation was based on the assumption that the change in the coefficient with temperature is a constant, i.e., $\frac{d}{dT} \left(\frac{d\rho}{dT} \right) = \text{constant}$. Other than the reasonable expectation that the magnitude of the temperature coefficient would decrease monotonically with temperature, there is no theoretical justification for the slope or magnitude of the extrapolated curve of $\frac{d\rho}{dT}$ vs T_o .

It is difficult to compare the measured temperature coefficient with any of the calculated coefficients, because most of the calculations have been for essentially different systems and/or different reactor temperature. The coefficient given in NAA-SR-5700 was calculated to be "essentially independent of temperature," but is believed to be most valid for 850°F. That given in A81-710-1 (Reactor Core as described in NAA-SR-6502, Volume III) was for a core containing 3 w/o enriched fuel and for 850°F. A later calculation,* which has not been formally reported, was performed for the actual core. Temperatures ranged from 450 to 1120°F. At 450°F, the calculated coefficient was three times larger in magnitude than the measured. The deviations between the measured and calculated coefficients are much larger than any experimental error that would be assigned. An estimate of the precision of the experimental method can be obtained from the calibration of gang 3 at 350 and 541°F. Measurements made in other thermal assemblies† have shown that the temperature coefficient of black rods is essentially zero. Thus, the small difference between the values obtained in the two calibrations indicates that, at most, the error is a few percent.

With good precision, the only other sources of error in the measurement would be systematic errors in measuring the period and in converting the period to reactivity. Two instrument channels were used for each period measurement. Differences between periods were 2 to 3% of the mean. Except for the time base,

*M. P. Aronchick, "HNPF Temperature Coefficients," NAA-SR-TDR-6953, Atomics International (December 15, 1961).

†K. Einfeld, "Piqua OMR Critical Experiment," NAA-SR-TDR-5270, Atomics International (May 10, 1960).

which was checked periodically by stopwatch, the two period channels were independent. The statistical purity of each channel was checked periodically, using the standard operating procedure included in the HNPF Operations Manual (NAA-SR-6500). During this test no significant sources of systematic errors were observed in the period measuring channels.

Undoubtedly, the values of ℓ/β and E used in the calculation of reactivity from period are not exact. Here again, however, error would be in the order of a few percent of the measured reactivity. Even considering inaccuracies in the temperature measurements, the total error in the measurement of isothermal temperature coefficient is not greater than $\pm 10\%$.

The isothermal temperature coefficient is the sum of three partial coefficients: fuel, moderator, and coolant. The first is negative and the others are positive. Measurement of the isothermal temperature coefficient does not, by itself, give sufficient information to determine which of the partial coefficients are in error, but additional information which will be obtained from the measurements of prompt and steady-state power coefficients will allow better separation of these effects. It is desirable to identify the sources of error in the temperature coefficient, to obtain better calculations of temperature and power coefficients. This will allow prediction of reactor stability in burned and mixed cores and, consequently, allow for advance correction of any anticipated problems.

HALLAM NUCLEAR POWER FACILITY

Post Critical Test Procedure

AI-PC-770

Power Coefficient

Test Procedure Submitted for Approval 6/30/61 James E. Owens
Date Atomic Energy International

Test Procedure Reviewed 1/3/63 Q. E. Thompson
Date Atomic Energy Commission

Test Approved for Performance 1/3/63 James E. Owens
Date Atomic Energy International

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

Test Approved for Performance 1/3/63 R. E. Buntain
Date Atomic Energy Commission

Test Completed 7/20/63 R. E. Buntain
Date Atomic Energy Commission

HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
POWER COEFFICIENT

AI-PC-770

Test and Analysis
by
R. H. DAVIDSON

Evaluation
by
C. W. GRIFFIN

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
POWER COEFFICIENT, AI-PC-770

I. PURPOSE OF TEST

The power coefficient tests were made to determine the effect on reactivity of changes in certain core parameters. Changes in reactivity, along with associated time constants, were examined for changes in power level, coolant flow rate, system temperatures, and combination of these factors. Knowledge of these parameters and the relationships thereof are useful in predicting the dynamic behavior of the reactor.

II. DISCUSSION OF TEST PERFORMANCE

Reactivity changes effected by changes in reactor power output can be considered as composite effects. Changes in power level or coolant flow produce temperature changes which have the following effects:

- a) Temperature changes in the reactor structure induce expansion or contraction, thus changing neutron leakage.
- b) Thermally induced changes in density alter the number of nuclei per unit volume, with concomitant changes in macroscopic cross sections (Σ).
- c) Changes in moderator temperature result in shifts in the thermal neutron energy spectrum, with associated changes in the nuclear properties of the fuel, moderator, and other core materials.
- d) Increases in fuel temperature produce a broadening of the resonance absorption peaks, termed Doppler broadening, in the epithermal energy region for materials such as U-238. This results in a higher probability for nonfission capture in such materials.

The effects of the various coefficients studied in this test were examined from several standpoints, ranging from responses of control rods and flux to

changes in power level, flow, or other system variables. Careful interpretation of the magnitude and response time of the test signals was also included in the examination.

At reactor power levels ranging from 1 to 100%, the following types of tests were performed:

- a) A positive-negative power ramp at constant flow.
- b) A negative power change at constant flow.
- c) A positive flow ramp at constant power.
- d) Positive and negative changes of combined power and flow (normal changes in system electrical output).
- e) A negative ramp of reactor inlet temperature.

During these tests, the flux, flow, temperatures, and control rod position were monitored, to be later analyzed using an analog computer. With the exception of the inlet temperature ramp test (E), the inlet temperature was observed to remain constant during the several ramps. Details of each test are discussed in the following paragraphs.

A. POSITIVE-NEGATIVE POWER RAMP AT CONSTANT FLOW $\left(\frac{\partial \rho}{\partial P}\right) F$

This test was designed to evaluate the effect of those reactivity coefficients having short time constants, i.e., the coefficients of fuel and coolant. Since these coefficients have similar response time, they were treated as one and termed the "prompt" coefficient. The test for prompt coefficient involved a positive power ramp of small magnitude with an immediate return to the initial power level using a reversal of rod movement to end the positive ramp. The entire ramp required only 10 to 30 sec to complete (power dependent); thus, all effects with long time constants are removed. This P-N ramp was quite effective in evaluating the prompt (fuel-coolant) coefficient, which has a time constant of 1 to 3 sec. The test was performed at several different power levels.

B. NEGATIVE POWER CHANGE AT CONSTANT FLOW $\left(\frac{\partial \rho}{\partial P}\right) F$

To provide a means of evaluating the moderator and steady-state power coefficients, a negative power change at constant flow was performed. The test involved a negative power ramp of from 1 to 3 Mwt in magnitude with flow held constant. As with the P-N ramp, the actual ramp time was less than 30 sec. However, since equilibrium was sought at a new power level, a much longer temperature stabilization time was required. The coefficients with short time constants

were included in this measurement but since known, they can be separated. This permits evaluation of the moderator coefficient, possessing a time constant of about 200 sec. The steady-state power coefficient is also revealed. When carefully conducted, this test can be performed at power levels of up to about 15% power. During performance of the test, a small power change was essential to keep fuel channel outlet temperature changes out of scram condition ($\Delta T \pm 25 F^\circ$).

Above the 15% power level, so small a power change is difficult to assess using nuclear instrument indication. However, the test was conducted at one power level above 15% (at 65 Mwt). In this case, the power change was observed by watching the change in fuel channel outlet temperatures. With flows held constant, control rods were inserted to produce a 10 F° drop in fuel channel outlet temperature; the rods were subsequently repositioned continuously to maintain this new temperature. Correspondingly, rod position and flux were monitored. The test data resulting from this method of performance, though difficult to analyze precisely, did result in sufficient data to permit the evaluation of a moderator coefficient at the 65 Mwt level.

C. POSITIVE FLOW RAMP AT CONSTANT POWER $\left(\frac{\partial \rho}{\partial F}\right) P$

This test of flow coefficient involved a positive flow ramp with the power held constant. Performance of the test resulted in a core temperature reduction which, since produced by a flow change, was in effect a flow coefficient. It should be noted that the results of the zero-power flow coefficient test (AI-PC-760) preclude the possibility of mechanical effects being present in the flow coefficient. The flow coefficient test was performed at the 38 and 65 Mwt levels. At higher power levels, excessive fuel channel outlet temperature rate of change, as described above, prohibited the performance of this test.

D. POSITIVE AND NEGATIVE CHANGES OF COMBINED POWER AND FLOW $\left(\frac{\partial \rho}{\partial P}\right) T_i$

This test was designed to determine the net effect on the reactivity of normal changes in system electrical output. It was performed during plant load variations of 3 Mwe or larger, which resulted from normal plant operation. A combined (power and flow) steady-state coefficient was evaluated. The combined ramp provides the most practical demonstration of normal reactor behavior.

E. NEGATIVE RAMP OF REACTOR INLET TEMPERATURE $\left(\frac{\partial \rho}{\partial T_i}\right) P, F$

The final test consisted of varying reactor inlet temperature with reactor outlet temperature held constant. The usual parameters were monitored to check for evidence of system instability; no instability was observed.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

The signals of particular interest during a reactor power ramp are flux and control rod position. Special provisions were therefore taken to record these signals accurately and in a manner convenient for analysis.

The flux signal was obtained from an ion chamber in nuclear instrument channel X. During the period of actual test performance, this flux signal was recorded on a variplotter. The steady-state signal was nulled; thus, the signal recorded represents the change in flux observed and is referred to as $\Delta\phi$.

The control rod position signal was obtained from a 40-turn potentiometer connected to indicate control rod position. As with the flux signal, the steady-state signal is nulled and the change in rod position is recorded on a variplotter. The resulting scaled signal is referred to as $\Delta\rho$. The reactivity change due to any change in flow or power will correspond to the reactivity worth of the moved portion of the rod. The $\Delta\phi$ and $\Delta\rho$ data recorded during the test were analyzed on an analog computer and the power coefficients determined.

The zero power reactor kinetics equations with six delayed neutron groups are:

$$\frac{d\phi}{dt} = \frac{\delta k}{\ell} \phi - \frac{\beta}{\ell} \phi + \sum_{i=1}^6 \lambda_i C_i,$$

and

$$\frac{dC_i}{dt} = \frac{\beta_i}{\ell} \phi - \lambda_i C_i, \quad i = 1, 2, \dots, 6$$

where

φ = neutron flux,
 δk = reactivity (absolute units),
 ℓ = prompt neutron lifetime,
 β = delayed neutron fraction,
 λ = decay constant,
 C = precursor concentration.

Power ramps for coefficient determination began with the reactor at equilibrium power. It was then convenient to modify the kinetics equations for analog simulation * with the equations written in terms of a change in flux, $\Delta\varphi$, thus eliminating the difficulties associated with setting initial conditions. With the reactor equilibrium disturbed at time zero,

$$\begin{aligned}
 \varphi &= \varphi_o + \Delta\varphi \\
 C_i &= C_{io} + \Delta C_i \\
 \text{and } \delta k &= \delta k_o + \Delta\delta k .
 \end{aligned}$$

At $t = 0$, $\delta k_o = 0$.

Substituting into the kinetics equations yields:

$$\frac{d}{dt} (\varphi_o + \Delta\varphi) = \frac{\Delta\delta k}{\ell} (\varphi_o + \Delta\varphi) - \frac{\beta}{\ell} (\varphi_o + \Delta\varphi) + \sum_{i=1}^6 \lambda_i (C_{io} + \Delta C_i) ,$$

and

$$\frac{d}{dt} (C_{io} + \Delta C_i) = \frac{\beta_i}{\ell} (\varphi_o + \Delta\varphi) - \lambda_i (C_{io} + \Delta C_i) .$$

*A. L. Gunby and W. J. Carlson, "Power Ramp Analysis With The Analog Computer," NAA-SR-7929 (to be published).

The equilibrium conditions must also satisfy the kinetics equations; thus:

$$\frac{d}{dt} \varphi_o = -\frac{\beta}{\ell} \varphi_o + \sum_{i=1}^6 \lambda_i C_{io} ,$$

and

$$\frac{d}{dt} (C_{io}) = \frac{\beta_i}{\ell} (\varphi_o) - \lambda_i C_{io} ;$$

also

$$\frac{\Delta \delta k}{\beta} = \Delta \rho ,$$

where, $\Delta \rho$ represents the reactivity in dollars.

Now, if

$$\frac{\ell \lambda_i \Delta C_i}{\beta_i} \equiv \Delta k_i ,$$

subtracting and rearranging gives:

$$\frac{d}{dt} (\Delta \varphi) = \frac{\beta}{\ell} \left[(\varphi_o + \Delta \varphi) \Delta \rho - \Delta \varphi + \sum_{i=1}^6 \frac{\beta_i}{\beta} \Delta k_i \right] ,$$

and

$$\frac{d}{dt} (\Delta k_i) = \lambda_i \left[\Delta \varphi - \Delta k_i \right] .$$

The above two equations describe the zero power kinetic behavior of the system. The at-power response of the reactor includes a reactivity feedback resulting from temperature effects. Each of the coefficients (fuel, moderator, etc.) are described by a reactivity feedback of the form:

$$\Delta \rho_i = \frac{\partial \rho}{\partial T_i} \Delta T_i$$

where T_i is the average material temperature. The term $\frac{\Delta\rho_i}{\Delta T_i}$ is identified with the material temperature coefficient. If the heat balance equations are solved, and reduced to a form suitable for analog use, with the above relation substituted, there results,

$$\Delta\rho(s) = \frac{[K'_f + K'_{fc}] \alpha_f \Delta\varphi(s)}{\tau_f S + 1}$$

for the power ramp feedback (inlet temperature and flow constant); for the inlet temperature ramp (power and flow constant),

$$\Delta\rho(s) = \frac{\alpha_f T_{in}(s)}{\tau_f S + 1} ;$$

and for the flow ramp (flux and inlet temperature constant),

$$\Delta\rho(s) = \frac{\frac{\varphi_o \alpha_f}{K''_{fc} F_o [F_o + \Delta F(s)]} \Delta F(s)}{\tau_f S + 1} .$$

The above three equations are written in LaPlace notation. The subscript f refers to fuel and fc to fuel coolant.

The terms

$K' = \text{Constant}$

$K'' = \text{Constant}$

$\alpha = \text{Temperature coefficient}$

$\tau = \text{Time constant}$

$T_{in} = \text{Inlet temperature,}$

and $F = \text{Coolant flow.}$

Considering the first of the three feedback equations (for illustration), it may be more simply written

$$\Delta\rho(s) = \frac{K''' \Delta\varphi(s)}{\tau_f S + 1} ,$$

or

$$S\Delta\rho(s) = \frac{K_f}{\tau}\Delta\phi(s) - \frac{\Delta\rho(s)}{\tau} .$$

Here the constant K'' has been identified as K_f , the fuel-coolant (prompt) power coefficient. The last equation is electronically described on the analog by the following circuit:

In the power ramp analysis (of prompt power coefficient), this circuit is added in as a feedback to that depicted by Figure 770-1.

The analog computer electronically solves these reactor kinetics equations. The kinetics circuit takes $\Delta\rho$ as an input signal and electronically generates the $\Delta\phi$ signal to correspond. The computer circuitry may be adjusted to closely simulate the actual kinetic behavior of the HNPF core. The delayed neutron constants, neutron lifetime, and reactivity coefficients (in the feedback loops) may be adjusted by potentiometer manipulation; flux is described by the input voltage.

Figure 770-1. Reactor Kinetics Circuit

Power coefficient analysis is initiated with the premise that the delayed neutron constants and lifetimes are known to be correct.* The input signal to the kinetics circuit is fed through the function generator unit to the kinetics circuitry. By adjustment of the potentiometers in the feedback loop, the output signal from the kinetics is matched to the $\Delta\phi$ signal obtained during the ramp test. Upon matching these signals, the potentiometer settings will correspond to the magnitude and time constant associated with the particular coefficient. The diagram of the kinetics circuit used in the evaluation of reactivity coefficients is shown in Figure 770-1. The tests described have been performed at several power levels. Except for obvious changes introduced by the coefficient tests, the system was operating in a normal manner as test data were taken. No power oscillations or other stability problems were observed.

Results of the power coefficient tests are summarized and presented in Table I. The prompt power coefficient and the combined (power and flow) coefficient are graphically shown in Figures 770-2 and 770-3, respectively. The steady-state power coefficient is described by the algebraic sum of the prompt and slow power coefficients.

An immediate application of test results was in the verification of the kinetic behavior of the HNPF for the first time during the rise-to-power portion of the formal testing program. Information was provided at certain power levels and extrapolated to higher power levels before system power was actually increased.

In addition, certain of the tests performed as a part of the formal testing program were conducted periodically during normal reactor operation. The tests best adaptable to this purpose are:

- a) P-N ramp (prompt coefficient)
- b) Normal power change analyses (combined steady-state coefficient)

The original data from tests a) and b) (Figures 770-2 and 770-3) are used as base data. Through the regular periodic performance of these tests, changes in the kinetic behavior of the system can be evaluated. Expected changes in kinetic

*In practice, six groups of delayed neutron constants and prompt neutron lifetimes were calculated for the system prior to power operation. Through an analog "test circuit," these were verified as correct by comparing analog performance with zero power reactor behavior.

TABLE I
HNPF REACTIVITY COEFFICIENTS

Pri Na Flow (%)	Power Level (Mwt)	Power Coefficients					Flow Coefficient	Combined Coefficient
		Prompt		Slow		Steady-State	Steady-State K (¢/Mwt) ^{††}	Steady-State K (¢/Mwt)
		K (¢/Mwt) [*]	τ	K (¢/Mwt)	τ	K (¢/Mwt)		
20 [†]	2.5 [§]	-0.4		+0.4		zero	-	-
20	10 [§]	-0.3		+0.8		+0.5	-	-
20	20 [§]	-0.4	2.5 sec	+0.45	200 sec	+0.05	(+0.25) ^{**}	(-0.2)
15	38 ^{§§}	-0.30		+0.35		+0.05	+0.16	-0.11
20	50	-0.25		(+0.33)		(+0.08)	(+0.16)	-0.08
26	65	-0.20		(+0.30)		+0.10	+0.16	-0.06
32	80	-0.16		(+0.30)		(+0.12)	(+0.16)	-0.04
48	120	-0.10		(+0.26)		(+0.16)	(+0.16)	zero
96	240	-0.05	1.5 sec	(+0.26)		(+0.21)	(+0.16)	+0.05

*For the HNPF, reactivity in percent is equal to 0.00693 times reactivity in cents.

[†]Indicated flow.

[§]Coefficient values for power levels below 38 Mwt are less accurately known. These should not be taken to represent trends (see Figure 770-1).

**Values in parentheses are calculated; other values are measured.

^{††}Measured as negative ¢/% flow, multiplied by -0.4% flow/Mwt to convert to units of ¢/Mwt.

^{§§}Temperature conditions for power levels of 38 Mwt and higher are dictated by the requirement that power and flows be "matched" for normal operation.

Figure 770-2. Prompt Power Coefficient vs Thermal Power

Figure 770-3. Combined Coefficient (Power and Flow)
vs Reactor Thermal Power

behavior might be step changes, as with changes to the core makeup (refueling with uranium carbide fuel for example), or gradual changes (as with changes in fuel composition due to burnup).

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following is the test completion criterion associated with post critical test AI-PC-770, Power Coefficient, as listed in NAA-SR-5700 (Supplement 4), page XI (c)-54, with remarks regarding the test to explain compliance with the criteria.

Criterion: Steady-state power coefficients of reactivity have been determined over a wide power range for constant flow conditions and again for constant reactor ΔT conditions.

Remarks: A variety of tests were conducted at reactor power levels ranging from 1 to 100% power to determine power coefficients at various operating levels. Data obtained were analyzed and the effect on reactivity was noted as explained in this report, thus meeting the requirements of this criterion.

V. EVALUATION

The basic space-independent reactor kinetics equations, and those for the lumped parameter temperature feedback, used during these tests are well known and agree with equations published elsewhere. Similar analog computer determination of reactor power coefficients, through duplication of reactor input-output data, have been successfully applied at the Sodium Reactor Experiment (SRE).^{*} Since the determined power coefficients for this test agree with those obtained by other methods, the technique used at the HNPF is considered valid.

A general evaluation of test AI-PC-770, Power Coefficient, indicates it was conducted in such a manner that measureable changes in reactor inlet temperatures were avoided. Such changes would have caused a serious complication in the analysis. Although test results are reliable, the data obtained are so near the instrumentation error tolerance range that numerical accuracy is uncertain.

^{*}Results of this previous application are reported in NAA-SR-7929, "Analysis of Power Ramps with an Analog Computer"

This condition is due to protective system action which confines the transients to a rather limited span. With this in mind, each test was reviewed by comparison with the published predictions of Section 4.1.4, Reactor Kinetics, in NAA-SR-5700.

The prompt power coefficients which were measured are considerably less negative than those predicted for the clean core, as shown in Figure 770-4.* Multiplying the measured power coefficient (for the constant ΔT cases at 38 Mwt and higher) by the initial power and plotting the results indicates a scatter of plus or minus 8% from a theoretically obtainable straight line. This is very good and implies good data. However, the difference in amplitude needs to be analysed. The power coefficients are dominated by the individual isothermal temperature coefficients. The predicted total isothermal temperature coefficient from Table II* would be $+0.0475 \text{ } \epsilon / ^\circ\text{F}$; the value extrapolated to 850°F from measured data (reported in AI-PC-760) is about $0.044 \text{ } \epsilon / ^\circ\text{F}$. Considering the uncertainty in the measurement and extrapolation of reported data, and in the calculation of the coefficients, the observed power coefficient lies within the probable error of both.

Figure 770-4. Prompt Power Coefficient of Reactivity

*From Section 4.1.4 of NAA-SR-5700

TABLE II
HNPF TEMPERATURE COEFFICIENTS OF REACTIVITY

Case	Partial Coefficients of Reactivity* ($10^{-5}/^{\circ}\text{F}$)		
	Fuel	Graphite	Coolant
I	-1.556	1.33	0.556
II	-1.667	2.89	0.556
III	-1.556	1.33	0.556
IV	-1.667	2.89	0.556

*Depend on fuel burnup only.

In reviewing the procedure it was observed that zero power control rod worths were used, since no at-power control rod differential worths were known or could be obtained. That control rod worth increases with both power and temperature has been demonstrated at the SRE. Thus, this is in agreement with theory. If a similar increase is applied to the data reported at 120 and 240 Mwt, the coefficient would be even more negative, i. e., the data point scatter would be about $\pm 5\%$, instead of $\pm 8\%$ as stated on the preceding page.

The conclusion of the analysis for this portion of the test is that the prompt power coefficient is indeed negative and that the test was conducted in accordance with the best procedure available.

Evaluation of the steady-state power coefficient is accomplished by direct observation of equilibrium change in reactivity with respect to equilibrium change in power. Accuracy of measurement is dependent solely upon resolution of the measured quantities. In these tests, resolution was poor due to step change limitations which were set by the plant protective system. For this reason, it appears that the two values quoted for constant ΔT condition (38 and 65 Mwt) have magnitudes which are of the same order as the probable error. One conclusion, however, is that the contribution from the moderator is less positive than predicted. Fortunately, this is a more stable condition and is in agreement with recent calculations of the moderator temperature coefficient. Calculated values of this parameter are at best very uncertain with respect to probable error.

The measured flow coefficient value was essentially as could have been predicted from the apparent total power coefficient. Since no mechanical flow effects upon reactivity were observed in the zero power test reported in AI-PC-760,^{*} the only effect at power would be a coolant temperature change equivalent to that for the related power change accompanied by an almost identical fuel temperature change. In other words, it is the steady-state power coefficient minus the Doppler contribution of the average fuel driving temperature and the contribution due to the slight change in moderator driving temperatures. Thus, the converted flow coefficient should be slightly more positive than the steady-state, which it is.

Similar reasoning indicates that the combined coefficient is essentially that portion of the power coefficient not associated with a change in coolant temperature level. Thus, the converted flow coefficient added to the combined coefficient should equal the steady-state coefficient. Data given in Table I of this report indicates that, within the uncertainty of resolution, those points that were measurable appeared to add properly.

While the inlet temperature coefficient measurement value was not reported, due to insufficient resolution, the value must have been small. Since this was not required in the original test criteria, it does not affect the status of test completion.

Performance and results of the power coefficient test are considered satisfactory and fulfill the requirements set forth in the test completion criterion.

^{*}R. H. Davidson, "AI-PC-760, Zero Power Coefficients of Reactivity," *Atomics International* (November, 1963)

HALLAM NUCLEAR POWER FACILITY

Post Critical Test Procedure

AI-PC-800

Radial Power Distribution

Test Procedure Submitted for Approval 6/30/61 James E. Adams
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. Adams
Date Atomic Energy International

Test Completed 7/20/63 James E. Adams
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-800

by

by

February 12, 1964

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HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
RADIAL POWER DISTRIBUTION AND FUEL CHANNEL
ORIFICE ADJUSTMENT, AI-PC -800

I. PURPOSE OF TEST

The objective of this test was to determine reactor core radial power distributions and corresponding radial peak-to-average power ratios. This information will assist in selection of allowable full-power operating control rod patterns. As a part of the test, the fuel channel orifices were adjusted to produce uniform fuel channel sodium exit temperatures and the sensitivity of the fuel channel orifice and plug assembly was determined under reactor operating conditions.

II. DISCUSSION OF TEST PERFORMANCE

A. RADIAL POWER DISTRIBUTION

With all fuel channel orifice plugs positioned at 50% open, reactor power level was increased to 2.3 Mwt with reactor sodium flow at 20%. All fuel channel sodium exit temperatures were recorded and averaged. Since the sodium flows for each fuel channel should be identical under these conditions, radial peak-to-average ratios were determined by dividing the peak core ΔT (peak fuel channel outlet temperature minus reactor inlet temperature) by the average core ΔT (average fuel channel outlet temperature minus reactor inlet temperature). This method was also used at 10 Mwt.

The fuel channel sodium exit temperatures are obtained from iron-constantan thermocouples located at the tip of the orifice plug. These thermocouples give a temperature representative of that at the top of the fuel bundles in the center of the sodium stream. Temperature readout is obtained at the main control board using a direct reading potentiometer. Periodic checks at isothermal conditions were made to compare the readings of all the thermocouples. The inlet temperature, obtained by averaging the lowest thermocouple on each of the three temperature measuring elements, was representative of the lower plenum temperature.

At the 20 Mwt level it became necessary to adjust several fuel channel orifices to prevent creating a condition wherein the ΔT between two fuel channels adjacent to the same moderator can exceeds 50°F . Since the flow through all channels was no longer the same, the relative power for each channel had to be calculated from the product of channel ΔT and relative flow, where relative flow is a function of orifice plug position (see Table I and Figure 800-1). The peak-to-average ratio is then the quotient of the relative power for the highest power channel divided by the average relative power for all channels. This method was used from 20 Mwt to full power.

B. ORIFICE PLUG ASSEMBLY

A variable orifice (see Figure 800-2) is incorporated in the lower end of each fuel element hanger tube. The orifice enables control of sodium flow in the individual coolant channels to obtain uniform sodium outlet temperatures during reactor operation. The orifice drive mechanism is located in the shield plug and is operated manually from the loading face with a special tool. The tool is so designed as to permit only a 90° turn without repositioning, thus preventing inadvertent rapid orifice adjustment which could cause rapid changes in fuel coolant temperature. The minimum coolant flow through the fully-closed orifice is about 26% of the maximum flow. Orifice plug travel beyond the fully-closed position will result in a flow increase.

At 38 Mwt, initial temperature data were taken to determine the sensitivity of the plug and orifice assembly. The sensitivity of the plug and orifice assembly is defined as the change in fuel channel sodium exit temperature per revolution of the orifice plug drive assembly ($^{\circ}\text{F}/\text{rev}$). The sensitivity was determined from data obtained by adjusting the orifice of the fuel channel producing the greatest power and that producing the least power. The temperature range permitted by consideration of ΔT across the adjacent moderator can (50°F) limits the distance over which an orifice may be calibrated. Thus, it was necessary to use several orifices to determine the sensitivity over the full travel of the orifice plug.

Load-following operation from 40 Mwe to full power necessitated some adjustment of orifices to obtain a fuel channel outlet temperature distribution which allows plant operation over the full operating power range.

TABLE I
FUEL CHANNEL SODIUM EXIT TEMPERATURES

Reactor Operating Conditions (5/28/63)	Core Position	Sodium Exit Temp (°F)	Channel Δ T (F°)	Relative Flow	Core Position	Sodium Exit Temp (°F)	Channel Δ T (F°)	Relative Flow	Core Position	Sodium Exit Temp (°F)	Channel Δ T (F°)	Relative Flow	Core Position	Sodium Exit Temp (°F)	Channel Δ T (F°)	Relative Flow
Power level: ~125 Mwt	C-2	955	357	.465	C-46	931	333	.610	C-87	925	327	.544	C-127	940	342	.375
Avg. exit temp: 937°F	C-3	944	346	.500	C-47	937	339	.568	C-88	937	339	.480	C-128	939	341	.310
Peak/avg. ratio: 1.43	C-5	941	343	.620	C-48	936	338	.590	C-89	937	339	.398	C-129	934	336	.338
Control rod pattern	C-6	945	347	.550	C-49	928	330	.668	C-90	931	333	.430	C-130	940	342	.292
Gang 1 - 55.4 in.	C-7	962	364	.490	C-51	936	338	.662	C-91	926	328	.646	C-131	935	337	.322
Gang 2 - out	C-8	948	350	.550	C-52	937	339	.656	C-92	934	336	.535	C-134	948	350	.295
Gang 3 - in	C-9	943	345	.525	C-53	937	339	.668	C-93	936	338	.588	C-136	927	329	.321
Gang 4 - 55.0 in.	C-10	957	359	.485	C-54	932	334	.610	C-95	923	325	.610	C-138	932	334	.315
Gang 5 - 12.0 in.	C-11	956	358	.548	C-55	953	355	.688	C-96	922	324	.588	C-140	938	340	.292
Sodium flow (primary)	C-12	943	345	.605	C-57	930	332	.662	C-97	934	336	.668	C-143	917	319	.321
Loop No. 1 - 14.0%	C-13	944	346	.505	C-58	932	334	.626	C-98	932	334	.460	C-144	932	334	.292
Loop No. 2 - 14.0%	C-14	939	341	.602	C-59	937	339	.580	C-99	930	332	.388	C-145	926	328	.375
Loop No. 3 - 13.1%	C-15	941	343	.602	C-60	940	342	.610	C-100	920	322	.513	C-146	941	343	.322
Sodium outlet temp	C-17	942	344	.630	C-61	935	337	.662	C-101	931	333	.428	C-147	949	351	.375
C-188-2 - 922°F	C-18	940	342	.602	C-63	953	355	.680	C-103	931	333	.418	C-148	937	337	.318
C-212-2 - 922°F	C-19	949	351	.571	C-64	950	352	.618	C-104	929	331	.513	C-149	942	344	.350
C-236-2 - 921°F	C-21	953	355	.581	C-65	940	342	.658	C-105	917	319	.412	C-150	938	340	.310
Sodium inlet temp	C-22	943	345	.602	C-66	943	345	.630	C-106	926	328	.442	C-151	927	329	.330
C-188-16 - 598°F	C-23	945	347	.602	C-67	937	339	.700	C-107	941	343	.632	C-154	931	333	.312
C-212-16 - 598°F	C-25	948	350	.630	C-69	940	342	.670	C-108	925	327	.554	C-156	936	338	.332
C-236-16 - 598°F	C-26	941	343	.630	C-70	939	341	.568	C-109	942	344	.648	C-158	928	330	.302
	C-27	956	358	.581	C-71	935	337	.580	C-111	940	342	.580	C-160	934	336	.315
	C-29	957	359	.551	C-72	937	339	.600	C-112	942	344	.512	C-163	938	340	.329
	C-30	949	351	.602	C-73	941	343	.662	C-113	937	339	.602	C-164	935	337	.292
	C-31	944	346	.562	C-74	936	338	.428	C-114	930	332	.418	C-165	930	332	.375
	C-33	945	347	.610	C-75	929	331	.661	C-115	930	332	.382	C-166	927	329	.325
	C-34	942	344	.602	C-76	937	339	.520	C-116	944	346	.480	C-167	949	351	.358
	C-35	957	359	.548	C-77	945	347	.580	C-117	935	337	.430	C-168	950	352	.298
	C-37	946	348	.581	C-79	942	344	.580	C-119	927	329	.418	C-169	921	323	.375
	C-39	932	334	.728	C-80	943	345	.475	C-120	941	343	.480	C-170	925	327	.292
	C-40	942	344	.630	C-81	933	335	.700	C-121	934	336	.410	C-171	929	331	.315
	C-41	944	346	.658	C-82	932	334	.450	C-123	925	327	.340	C-174	933	335	.298
	C-42	948	350	.630	C-83	929	331	.410	C-124	920	322	.326	C-176	935	337	.310
	C-43	932	334	.728	C-84	938	340	.478	C-125	930	332	.356	C-178	923	325	.329
	C-45	942	344	.658	C-85	921	323	.415	C-126	928	330	.340	C-180	935	337	.303

Figure 800-1. Relative Sodium Flow Rate vs Orifice Plug Position

Figure 800-2. Fuel Channel Orifice Assembly

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

A. RADIAL POWER DISTRIBUTION

Results of testing from 0 to 80 Mwe indicate that the radial power peak-to-average ratios can be maintained within the established limit of 1.50. Table II lists the results of radial power distribution analyses.

The desirable limiting value of 1.50 for the peak-to-average ratio was established by consideration of burnup and fuel temperatures. The calculated peak fuel temperature of 1124°F at 240 Mwt is based on a radial peak-to-average power ratio of 1.47. Assuming a maximum possible error of 10% in this ratio, a maximum temperature of 1238°F is obtained, which is less than the 1250°F limit imposed by the fuel design.

During low power testing (0 to 38 Mwt), it was observed that the fuel channel outlet temperatures exhibited a self-regulating effect. The primary contributors to this effect are believed to be heat transfer and convection flow. However, efforts to determine the magnitude of these effects were unsuccessful. Operations at power levels above 38 Mwt and at flows above 20% did not exhibit any self-regulating effect.

Phase III testing (38 to 250 Mwt) was initiated by returning the reactor to 38 Mwt with the control rods in the "normal operating pattern." At this power level the normal operating pattern was changed from gang 1 banked (for control) to gang 1 and gang 4 banked for control (Table II). This change was required to enable plant operation at 15 to 100% power for at least 3 months with essentially the same control rod pattern. Concurrently, the fuel channel outlet temperatures were monitored to ensure that no undesirable radial temperature distributions occurred.

Upon completing the change in control rod configuration, the radial power distribution was determined. It was noted that a shift in the radial power distribution had occurred (Figures 800-3 and -4), resulting in a somewhat flatter distribution in the center of the core. The resulting radial peak-to-average power ratio was higher than that obtained with the previous normal operating pattern. However, the new ratio of 1.46 was less than the acceptable limit of 1.50.

With the "new" operating configuration, the plant was operated over the normal load range of 12 to 80 Mwe. For this range of operation the peak-to-average

TABLE II
RADIAL PEAK-TO-AVERAGE POWER RATIOS

Power Level	Reactor Conditions			Peak to Average Ratio
	Control Rod Pattern	Pri Na Flow (Indicated)	Orifice Position	
2.5 Mwt	"Normal" Operating*	20%	All at 50%	1.34
10 Mwt	Normal Operating	30%	All at 50%	1.34
10 Mwt	Normal Operating	20%	All at 50%	1.34
20 Mwt	Normal Operating	30%	Orifices had been adjusted	1.46
38 Mwt	Normal Operating	20%	Further adjustments completed	1.45
38 Mwt	Normal Operating	15%	All orifices had been adjusted	1.37
38 Mwt	"New" Operating†	16%	No further changes	1.46
20 Mwe	New Operating	24.1 %	No change	1.48
40 Mwe	New Operating	48.2 %	No change	1.43
65 Mwe	New Operating	83%	No change	1.47
80 Mwe	New Operating	100%	No change	1.45

*"Normal" Operating control rod configuration:

Gang 1 critical
Gang 2 full out
Gangs 3,4,5 at 12 in.

†"New" Operating control rod configuration:

Gangs 1 and 4 critical
Gang 2 full out
Gang 3 full in
Gang 5 at 12 in.

Figure 800-3. Typical Radial Power Distributions for HNPF at 38 Mwt with "Normal" Operating Control Rod Pattern

Figure 800-4. Typical Radial Power Distributions for HNPf with "New" Operating Control Rod Pattern at 40 Mwe

ratios were 1.45 ± 0.03 . However, data taken at 80 Mwe with 26 ϵ burnup indicated that the radial power distribution had changed. The resulting distribution indicated a distinct flux increase in the core center, resulting in a flatter distribution (Figure 800-5). This shift is primarily due to the withdrawal of the banked gangs 1 and 4 to compensate for fuel burnup.

A 3-day power run at 80 Mwe was performed to permit the buildup of xenon. Fuel channel exit temperature data were taken periodically during this run in an effort to determine the effect of xenon on the radial power distribution. No measurable effect on the radial power distribution was observed.

At higher power levels a discrepancy became apparent between the average fuel channel sodium exit temperature and the sodium outlet temperature measured at the reactor vessel. At full power this difference was as much as 15 to 20F°. An investigation to determine the reason for this discrepancy indicates that there is an uneven radial temperature distribution of the sodium in the vicinity of fuel channel exit thermocouples. As shown in Figure 800-2, the sodium from the center of the fuel cluster is channeled to the exit thermocouple by the fuel element support which is located just above the cluster. If the temperature of the sodium flowing through the center of the cluster is higher than the mixed mean outlet, it would explain this discrepancy.

A test was conducted to establish whether or not the thermocouples actually measure the true outlet temperature. Seven fuel channels were selected with orifices positioned as shown in Figure 800-6. Each fuel channel orifice was positioned incrementally within the temperature limits imposed by the plus or minus 50F° maximum moderator can head temperature difference. The exit temperature was measured at each position and a composite sensitivity curve was plotted for the orifice throttling range (Figure 800-7). The apparent sensitivity was then compared with the actual sensitivity as measured during testing of the prototype orifice.* As can be seen in Figure 800-7, there is a marked change in orifice sensitivity in the 2-1/2 to 3 in. range. This is attributed to the thermocouple being moved into a locality where the colder sodium from the peripheral region of the channel begins mixing with the sodium from the cluster center.

*C. P. Sutherland, "Hydraulic Characteristics of a Shield Plug Orifice," NAA-SR-TDR-7417, June 12, 1962.

Figure 800-5. Typical Radial Power Distributions for HNPf with "New" Operating Control Rod Pattern at 80 Mwe

Figure 800-6. HNPF Fuel Channel Orifice Throttling Range

Figure 800-7. Orifice Sensitivity at 80 Mwe and 100% Sodium Flow

The error introduced by this effect is only about 1.5 to 2% for a given orifice. The radial power distribution determination is virtually unaffected by this anomaly due to the small number of outlet temperatures which must be normalized.

B. ORIFICE PLUG ASSEMBLY

The orifice plug assembly performed satisfactorily throughout the testing program. However, after the plant had been operated with constant orifice positions for a period of about 30 days, some orifice plug sticking was observed. Upon adjusting certain orifice plugs, the drive could be operated up to two full turns without changing the plug position. At this point, turning became very difficult. If the plug drive cable was allowed to remain in tension for a short period of time, the plug moved. Observation of the sodium exit temperature for that orifice which had not changed for the initial two turns of movement then indicated a rapid temperature change of 10°F. If the tool was then reengaged in an attempt to change the orifice plug position, it was noted that the tool turned easily with each one-fourth turn indicating the anticipated temperature change.

Orifice plug sensitivities were determined at several power levels. The sensitivity at 38 Mwt is shown in Figure 800-8. It should be noted that this sensitivity determination was made at low flow (15%) where the self-regulating effect discussed previously appears to be significant and therefore masks the true sensitivity.

At 40 Mwe the sensitivity was determined from data obtained by moving C-170 and C-43 over the maximum permissible temperature range. Figure 800-9 contains the actual and calculated sensitivities for 50% primary sodium flow.

The orifice sensitivity at 80 Mwe was determined for the full range of travel by using seven selected fuel channels. Figure 800-6 shows the range covered by the individual orifices. Each fuel channel orifice was positioned incrementally within the temperature limits imposed by the $\pm 50^\circ\text{F}$ maximum moderator can head temperature difference. Using the data from these seven calibrations, a composite sensitivity curve (Figure 800-7) was determined for the full orifice throttling range.

Load-following operation from 40 Mwe to full power necessitated some adjustment of orifices. The orifice plugs were readjusted at 40 Mwe to obtain a more uniform fuel channel sodium outlet temperature distribution. A range

Figure 800-8. Orifice Sensitivity at 38 Mwt

Figure 800-9. Actual and Calculated Orifice Sensitivity at 50% Sodium Flow

of $\pm 7^{\circ}\text{F}$ was obtained with an average fuel channel outlet temperature of 955°F and a primary sodium flow of 50%. These adjustments resulted in a fuel channel outlet temperature distribution which allows plant operation over the full operating power range. A range of $\pm 23^{\circ}\text{F}$ was obtained at full power with an average fuel channel outlet temperature of 958°F and an indicated sodium flow of 100%.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with post critical test AI-PC-800, Radial Power Distribution and Fuel Channel Orifice Adjustment, as listed in NAA-SR-5700 (Supplement 4), pages XI(c)-57 and XI(c)-35, with remarks regarding the tests to explain compliance with the criteria.

- a) Criterion: Control rod positions or patterns that would produce either a peak-to-average axial power ratio greater than 1.65 or a peak-to-average radial power ratio greater than 1.5 in any fuel channel have been identified and are restricted for power operation.

Remarks: It has been established that any operation with all control rods banked should be restricted on the basis of axial power distribution test AI-PC-740. Because of the enormous number of possible rod patterns it was decided to concentrate the investigation to those chosen as possible operating rod patterns. These investigations indicated no undesirable ratios. In the event that a change in the operating rod patterns becomes necessary, it is suggested that the radial power distribution be determined.

- b) Criterion: An operating control rod pattern has been chosen for power operation which results in the best compromise between axial and radial power distributions.

Remarks: An operating configuration consisting of gangs 1 and 4 banked for control, gang 3 full in, gang 2 full out, and gang 5 at 12 in. has been established for the initial period of power operation. This configuration yields a radial peak-to-average of 1.46 and facilitates a change in rod pattern when burnup makes such necessary.

- c) Criterion: The shadowing of neutron detectors by control rods has been investigated.

Remarks: Extensive investigation of this effect was eliminated due to the possible excessive moderator can temperature differential which would result if attempts were made to excessively tilt the flux. Since the control rods are always operated in symmetrical patterns, there is no problem involving shadowing at neutron detectors. No shadowing was observed in the limited testing conducted.

- d) Criterion: A core radial power distribution has been determined for the preferred operating control rod pattern.

Remarks: A representative radial power distribution is contained in Figure 800-5.

- e) Criterion: The fuel channel orifice and plug assemblies of all fuel elements in the core operate under reactor operating conditions.

Remarks: The fuel channel orifice and plug assemblies operated satisfactorily over the full reactor operating range (12 to 80 Mwe).

- f) Criterion: All fuel channels are adjusted to obtain a uniform sodium exit temperature at full power.

Remarks: An operating sodium exit temperature distribution has been established which permits operation over the full power range without changing orifices.

- g) Criterion: The temperature sensitivity of the fuel coolant to orifice plug adjustment has been determined.

Remarks: A plot of sensitivity versus orifice plug position is shown in Figure 800-7.

- h) Criterion: The standard operating procedure for adjusting fuel channel orifices has been verified.

Remarks: The standard operating procedure for adjusting fuel channel orifices was used in all orifice adjustments and found to be satisfactory.

- i) Criterion: Position of all orifices has been measured and recorded.

Remarks: The position of all orifices has been measured and appropriate records are being maintained by operating personnel.

V. EVALUATION

A. SUMMARY OF TEST AND RESULTS

Radial power distribution across the core was determined for a number of different conditions of power, flow, control rod configuration, and orifice adjustment. At the same time, operability of the fuel channel variable orifices was verified under actual operating conditions.

The radial power distribution was measured in two manners; (1) at lower power with all orifices set at 50% flow, and (2) at higher power (above 10 Mwt) with the orifices adjusted to produce approximately equal channel outlet temperatures across the core. At 2.3 and 10 Mwt, with all the orifices set at 50% to produce equal flow through all channels, the outlet temperature distribution was theoretically representative of the radial power distribution. (This distribution may have been affected by the self-regulating effect.) A relatively flat power distribution with a peak-to-average ratio of 1.34 was measured. Above 10 Mwt, with the orifices adjusted, a more peaked power distribution was measured with a peak-to-average ratio of 1.43 to 1.48 (except for one measurement of 1.37).

In adjusting the orifices to produce uniform sodium outlet temperatures the orifice and the adjustment procedure were found to be satisfactory. Orifice sensitivity was measured and found to vary from 60 to 150F°/inch with full flow and ΔT across the core.

A control rod operating pattern that would permit plant operation from 15 to 100% power for at least 3 months was found to produce an acceptable peak-to-average power ratio of less than 1.50.

B. EVALUATION OF TEST PERFORMANCE

The test procedures and analytical methods which were followed are considered valid. A fairly accurate determination of radial power distribution can be obtained by measuring the relative power of each channel.

The following assumptions are necessarily made in using the above method of determining the radial power distribution:

- 1) The relative sodium flow through each channel is a function of the orifice position only. The relative flow is independent of the pressure drop across the core within the operating range of primary sodium flow (15 to 100% flow).

- 2) The experimentally determined relationship between relative sodium flow and orifice position must be fairly accurate.
- 3) Transverse heat loss from each channel is directly proportional to the relative power produced in the channel.
- 4) Measurement of the mixed sodium exit temperature of each channel is accurate.
- 5) Pressure variation radially across the inlet plenum, caused by the jets of sodium from the inlet nozzles, is no more than a small fraction of the pressure drop across the core.

All of the above assumptions are valid except the accurate measurement of the channel mixed sodium exit temperatures, which does introduce a small error in the calculation of relative power. Since relative rather than absolute powers are calculated, most small errors are reduced.

The test was conducted properly and in accordance with the test procedure. Although not specifically required by the test procedure, some additional testing would have provided valuable information. With the reactor at low power and (all orifices at 50% open), the sodium exit temperature distribution could have been recorded for several different control rod patterns to provide information on the change in radial power distribution with the change in control rod pattern. The radial power distribution was investigated for only the normal control rod pattern (gang 1 banked for control) while using the above test method. At higher power using the second test method an additional control rod pattern was used (gangs 1 and 4 banked for control). Information about the radial power distribution for other control rod patterns might be quite helpful in the future, when considering changing to a control rod pattern different than either of the two used during the test. Although the core characteristics will be somewhat different in the future with accumulated poisons, fuel exposure, control rod burnup, and additional fuel, information about how the radial power distribution changes with change in control rod pattern would be quite helpful. Very little change occurred in the magnitude of the radial peak-to-average power ratio when the control rod pattern was changed from the normal to the new control rod pattern. (The power distribution was shifted.) The magnitude of the radial peak-to-average power ratio should only be greatly effected by the buildup of poisons and fuel burnup. Therefore, it can be assumed that most other symmetrical control rod patterns which might be of future interest will not greatly affect the magnitude of the radial peak-to-average power ratio.

It is not apparent from the results presented whether the measured peak-to-average power ratio changed from 1.34 at low power (2.5 and 10 Mwt, tests No. 1, 2, and 3 Table II) to 1.46 at 20 Mwt (test No. 4, Table II) because a different test method was used or because the reactor power level and sodium flow rate were changed.

C. EVALUATION OF TEST RESULTS AND ANALYSIS

1. Radial Power Distribution

The results obtained for the radial power distribution are reasonable and fairly consistent with the theory of thermal flux distribution in large, thermal reflected reactors. As shown in Figures 800-3, -4, and -5 the power peaks in the vicinities of the gang 2 control rods which are fully withdrawn. The power is depressed in the central area of the core by the control gang 5 control rod and the gang 3 control rods. Gang 5 is essentially fully inserted (12 in. withdrawn). Gang 3 is fully inserted. Power drops off rapidly at the edges of the core to about 0.6 of the average power. Since xenon and samarium and other fission product poison buildup-and-burnup occurs in direct proportion to the power level in the fuel, the areas of high power in the core achieve fuel burnup and poison buildup more rapidly than areas of lower power. This causes a gradual flattening effect on the radial and axial power distributions. Figure 800-10 shows the relationship of radial power peak-to-average ratio versus negative reactivity caused by xenon, samarium and fuel burnup. The points shown in this figure were calculated from HNPF operating data during October, November, and December 1963. From the beginning of October the core contained 10 UC elements in addition to the 140 U - Mo elements which were in the core when the AI-PC-800 radial power tests were conducted. This effect of decreasing radial peak-to-average power ratio was not measurable during the three day full power run made to permit the buildup of full power equilibrium xenon. Possibly it was not measurable because the levels of xenon at which the radial power distributions were measured were not widely different. The

Figure 800-10. HNPF Radial Peak-to-Average Power Ratio vs Poison Reactivity

three day full power run was not immediately preceded by a three or four day shutdown period to permit xenon concentration to become very small. Therefore, by the time the reactor reached full power steady state temperature conditions at the beginning of the full power run, a large percentage of full power equilibrium xenon had already built up.

2. Orifice Plug Assembly

Using the relationship of relative flow through a channel, versus orifice position (experimentally determined with a prototype assembly), the orifice sensitivity was calculated with the Equation,

$$S = \frac{(f_1 - f_2) \Delta T_{\text{core}}}{\frac{(f_1 + f_2)}{2} X}$$

where

S = orifice sensitivity, F°/inch

f_1 = relative flow before orifice change

f_2 = relative flow after orifice change

ΔT_{core} = the average ΔT across the core

X = the distance of orifice plug movement, inches

Measured orifice sensitivities were determined by adjusting the orifice plug over a small increment, then dividing the resulting change in exit temperature by the increment. The measured orifice sensitivities were found to be in the approximate range expected from the results of the previous sensitivity calculations. The sensitivity measured allows safe adjustment of the channel outlet temperatures to a fairly uniform level. Examination of Figures 800-7 and -9 shows that from the closed orifice position (0 inches) to 2-1/2 inches open, the variation in measured sensitivity resembled the variation in calculated sensitivity. The measured sensitivity was about 20 to 30% less than the calculated sensitivity. At an orifice position of around three inches, the measured sensitivity rapidly became about 60% greater than the calculated sensitivity. This rapid change in sensitivity apparently resulted from incorrect measurement of

the channel average sodium outlet temperature caused by incomplete mixing of the sodium in the vicinity of the thermocouple. The orifice position at which mixing of the sodium occurs, as shown by the rapid change in orifice sensitivity, is a function of the primary sodium flow rate. At a primary sodium flow of 100% the rapid change occurs from an orifice position of 2-1/2 to 3 inches. At 50% primary sodium flow the rapid change occurs from 3 to 3-1/2 inches. Incorrect measurement of channel outlet temperature causes a small error in measurement of the radial power distribution. The dependence of orifice sensitivity on primary sodium flow rate should cause a small change in measured power distribution, with a change in primary flow.

HALLAM NUCLEAR POWER FACILITY

Post Critical Test Procedure

AI-PC-830

Post Critical Master Schedule Including Rise To Power

Test Procedure Submitted for Approval 6/20/61

Date James E. Chene
Atoms International

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Date J. E. Simpson
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Commission

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Date R. C. Buntain
Atomic Energy
Commission

HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
MASTER SCHEDULE, INCLUDING RISE-TO-POWER

AI-PC -830

Test and Analysis
by
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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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HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
MASTER SCHEDULE, INCLUDING RISE-TO-POWER, AI-PC-830

I. PURPOSE OF TEST

The purposes of this test were to coordinate testing activities during Phases II and III* of the post critical test program, and to serve as the instrument for increasing power. Addenda to the test were utilized as a means of documenting certain aspects of the HNPF startup program that did not come under the cognizance of any specific test, but were not of a nature to justify the development of additional formal tests. Included in this category are core alignment, control rod inspection and rod drop time determinations, sodium flowmeter calibrations, and sodium pipe displacement surveillance.

II. DISCUSSION OF TEST PERFORMANCE

The 12 tests listed below were coordinated through the performance of this test procedure.

AI-PC-510, Hydraulic Test - Main Heat Transfer System
AI-PC-530, Heat Transfer Tests
AI-PC-552, Steam Generator - Steam Purity
AI-PC-560, Cooling System Tests
AI-PC-620, Plant Protective System
AI-PC-630, Plant Control System
AI-PC-640, Emergency Feedwater System
AI-PC-660, Scram and Loss-of-Load
AI-PC-690, Radiation Shielding
AI-PC-770, Power Coefficient

*Phase I, reactor power <100 Kwt ("Zero" power)
Phase II, reactor power < 15% of full power
Phase III, reactor power 15 to 100%

AI-PC-800, Radial Power Distribution and Fuel Channel Orifice Adjustment

AI-PC-840, Xenon Poisoning

Two of the above tests, AI-PC-552 and AI-PC-620, were of a continuing observational nature and were, in effect, continuously in progress at all times when the reactor was operating. One of the above tests, AI-PC-640, was completed during the Phase II (0 to 15% of full power) portion of the program. Specific portions of each test were performed at each planned power level. Prior to each power increase, data reduction was accomplished to the extent necessary for the responsible test engineers to determine that the plant could be expected to operate satisfactorily and safely at the new power level.

Standard scheduling methods were employed for laying out the program. Coordination between test engineers was accomplished by daily meetings and personal contact. Precise advancement along the planned schedules was not universally possible because of plant availability and the press of other activities. However, good cooperation between Atomics International and Consumers Public Power District contributed to favorable utilization of facility time.

The planned nominal testing power levels were as follows:

Phase II	2.5 Mwt	(1% of full reactor power)
	5 Mwt	(2% of full reactor power)
	10 Mwt	(4% of full reactor power)
	20 Mwt	(8% of full reactor power)
	38 Mwt	(15% of full reactor power)
Phase III	38 Mwt	
	50 Mwt	
	120 Mwt	
	150 Mwt	
	180 Mwt	
	210 Mwt	
	100% of full power	

Early in the Phase III portion of the program, it became clear that redefinition of test power levels in terms of Mwe gross station power generation was preferable, since determination of this quantity was more readily accomplished. The Phase III nominal testing power levels were subsequently established at 15 to 20, 25, 40, 50, and 65 Mwe gross and 75 Mwe net station power. The highest power level achieved was 250 Mwt reactor power, (defined as 100% of full power) as determined from steam power calculations.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

A. PHASE II TESTING ("ZERO" TO 15% OF FULL POWER)

1. General

Testing and related activities described in the following took place between December 26, 1962 and March 10, 1963. The emphasis in this phase was placed on a safe and orderly approach to the lower limit of the normal power operation range (defined as 15% of full power), and on initial investigation of plant operating characteristics. Throughout this phase of testing, generated steam was condensed in the atmospheric boilers of the steam dump system.

The sequence of testing activity, reactor operation for testing purposes, and reactor upper and lower plenum temperatures for this phase of the testing program are presented in Figure 830-1.

2. Reactor Power Increases

a. Initial Increase to 2.5 Mwt

Prior to increasing power to the 2.5 Mwt test level, the amplifier gains on the three protective system nuclear instrument channels were advanced so that 15% reactor power would result in 100% flux indication, according to the best available correlation between power and nuclear indication. This resulted in conservative protection; i.e., a flux level scram trip would occur at 125% of 15% power.

The increase to the 2.5 Mwt nominal test level was accomplished in three steps with primary system sodium flow established at approximately 43% of full flow, according to the best flowmeter calibration available. The final power level achieved was determined to be 2.3 Mwt, based on reactor differential temperature (defined as the difference in upper and lower plenum temperatures), and the indicated primary system sodium flow.

b. Power Increase to 10 Mwt

After testing was completed at the 2.5 Mwt level, power was increased to 10 Mwt in four approximately equal steps with an indicated primary system sodium flow of 30%. At each step, indications on each nuclear instrument channel were compared to power level as determined from sodium flow and reactor temperatures. Nuclear instrument indications were extrapolated to the next level

NOTES:

- ▽1 Planned scram
- ▽2 Flow ratio scrams (2) while setting up for cold trapping
- ▽3 Flow ratio scrams (2) while starting pumps during hot startup
- ▽4 Temperature rate (dT/dt) scram during power ramp for power coefficient test
- ▽5 Low power dT/dt computer scram (operator error while adjusting computer)
- Specific testing activity
- Continuous surveillance

Reactor power levels are nominal test levels

Figure 830-1. Phase II Testing Schedule

prior to each power increase. The final power achieved was 9.5 Mwt, determined in the same manner as before. Power calculated from the steam dump flow and enthalpy gain at this time was 8.7 Mwt, including 0.5 Mwt assumed losses. The five power range nuclear instrument channel amplifier gains were adjusted to yield more accurate (though still slightly conservative) indications of power level.

c. Subsequent Power Increases

The remaining power increases (to 20 and 38 Mwt) were accomplished in a manner similar to that discussed above. Reactor response to operator action proved to be completely predictable and control was easily maintained on manual or semi-automatic control.

3. Development of Full Operating Temperatures

During reactor operation at 38 Mwt, reactor lower and upper plenum temperatures of 610 and 945°F, respectively, were developed by reducing sodium flows from HCS-504 (steam demand hand control station) while maintaining a constant power level on automatic flux control. System temperatures proved to be extremely sensitive to sodium flow changes and considerable care was required to avoid dT/dt computer protective action. However, no problem areas were revealed, and the temperatures proved to be easily maintained.

4. Power Level Determination

Power level determination through most of this phase of the testing program was accomplished by use of reactor differential temperature and indicated primary system sodium flows. Sodium flows were determined by obtaining the flowmeter output in millivolts and referring to a flowmeter calibration curve. Steam and feedwater calculations below 15% reactor power were believed to result in power determinations of questionable accuracy, although reasonable agreement was obtained between these calculations and those using indicated primary sodium flows and temperatures at 2.5, 10 and 20 Mwt. These data are presented in columns 1 and 3 of Table I.

Subsequent to the initial achievement of 38 Mwt, and prior to completion of the planned testing, it became necessary to reduce reactor power to a low level when a leak developed in the steam dump atmospheric boilers, resulting in a loss of condensate in excess of plant make-up capabilities. Reactor power level was kept below 100 Kwt for training purposes for the following 6 days, and was shutdown completely for an additional 13 days for maintenance purposes.

TABLE I
COMPARISON OF POWER LEVEL DETERMINATIONS

Date	Time	Power Mwt		
		No. 1	No. 2	No. 3
1-2-63	1200	2.3	3.1	-
1-3-63	1000	-	-	2.3
1-5-63	1135	2.6	3.3	-
1-5-63	1600	9.5	11.9	-
1-7-63	1308	8.7	10.8	-
1-7-63	1900	-	-	10.3
1-7-63	2313	8.7	10.8	-
1-28-63	1430	18.5	23.0	-
1-29-63	0930	21.3	26.5	-
1-29-63	1100	18.0	-	17.05*
1-30-63	0245	16.7	21.7	-
2-1-63	1500	-	-	44.9
2-1-63	1700	36.1	46.4	-
2-1-63	2015	37.4	48.2	-
2-23-63	1300	-	-	47.3
3-1-63	0510	-	34.1	35.4
3-4-63	1735	-	38.1	38.0
3-5-63	-	-	38.2	-
3-5-63	-	-	37.9	34.7
3-5-63	-	-	39.8	-
3-5-63	-	-	39.4	-
3-6-63	-	-	39.7	-

Column No. 1 Calculated from recommended flowmeter calibration curves presented in IOL from W. H. Spalding to J. E. Owens "HNPF 14 in. Na Flowmeter Calibration Curves," Atomics International, October 9, 1962

Column No. 2 Recalculation from flowmeter calibration curves presented in IOL from P. F. Shaw "HNPF Na Flowmeter Calibration," Atomics International, March 9, 1962

Column No. 3 Calculated from steam power considerations.

*Originally reported as 17.05. A leak in the steam dump system was found, and power was recalculated to be 21.1 Mwt

After completion of training and maintenance, the reactor was again taken to a calculated power level of 38 Mwt (with a few hours interruption due to further problems in the steam dump atmospheric boiler). Upon reaching steady-state, the first set of reliable steam power data was obtained. These data yielded a power level determination of 47.3 Mwt. Analysis and comparison of the sodium and steam power data revealed that the discrepancies in power determinations were due to flowmeter calibration error in excess of that expected. Accordingly, a test was developed to accomplish a flowmeter calibration based on steam power. Data obtained were reduced and used to develop a corrected flowmeter calibration for flows up to approximately 50% of full flow. Powers calculated for the Phase II portion of the program, based on the corrected flowmeter calibrations are presented in column 2 of Table I.

5. Unplanned Scrams and Other Program Interruptions

Eleven scrams occurred during the Phase II testing program. Of these, three were originally planned in the testing program and two were added to obtain additional information. Two unplanned scrams from the 10 Mwt test level were initiated by the flow-ratio computers when sodium flows in one circuit were inadvertently upset during preparation for primary system cold trapping.

During recovery after the planned 10 Mwt scram, two flow-ratio scrams were initiated at low power levels. It was determined at this time that computer action was taking place well below the 20% flow level where protective action is supposed to become effective. The computer chassis were realigned and checked with the reactor subcritical.

During performance of a small negative power ramp for AI-PC-770, Power Coefficients, at the 20 Mwt test level, the rate of change trip of the measured fuel channel outlet temperature computer (dT/dt) was exceeded, initiating a scram. A review of after-scram conditions led to the substitution of this scram for the one planned from 20 Mwt.

One scram was originally planned from 38 Mwt. However, it was decided that two were desirable, one at less than full temperature gradient and one at full gradient. It was planned to make a hot startup from the first of these two scrams; however, throttle valve problems prevented successful completion of the hot startup. Therefore, a second scram at less than full temperature gradient was performed, and a hot startup was again attempted. A low power scram was initiated

during manual resetting of the dT/dt computers.* Following recovery from this scram, a completely successful hot startup was performed.

Three post-scram secondary throttle valve operator failures occurred during the program. These were related to a later corrected design deficiency in the secondary valve operators. The primary valve operators are of similar design, and were similarly modified during the next shutdown.

6. Additional Testing (Performed as Addenda to this Test)

a. Core Alignment

Subsequent to completion of Phase II testing at 38 Mwt, a reactor core alignment test and control rod operability test were performed. The core alignment test consisted of removing and reinserting three fuel elements with the reactor at an isothermal temperature of about 400°F. The three elements were easily and successfully handled with the fuel handling machine. No excess binding or loading was indicated by the fuel handling machine load cell, thus indicating that there is no apparent misalignment in the core at this time. Data obtained in this test are presented in Appendix I.

b. Control Rod Operability

The control rod operability test consisted of individually operating the rods from the control rod test cart and recording the following data: withdrawal and insertion times, latch release time, and drop times for the first 9 ft and for full insertion. All of the above times were found to be within the limits specified in the technical specifications, thus verifying operability. Details of this test and the data obtained are presented in Appendix II.

c. Control Rod Measurement and Inspection Following 38 Mwt Testing

An inspection and measurement of control rods was performed at the completion of Phase II testing. Results of the inspection indicated that satisfactory future performance could be expected. Information obtained from the inspection and measurements is contained in Appendix III.

*The initial temperature drop from the scram plus the subsequent cooling rate placed the dT/dt computers in scram condition. During the convection flow period, the cooling rate was reduced sufficiently to allow the dT/dt computers to just clear the scram condition. It was desired to move the computer indication farther from the trip level to avoid obtaining a dT/dt scram on a negative rate prior to reaching a reactor power level sufficient to reduce the rate of cool-down. This was being accomplished manually when the technician adjusted one computer in the wrong direction, thus initiating the scram.

d. Piping Displacements

During the initial rise-in-power to 38 Mwt, line 231, primary loop No. 1 reactor outlet, did not move as anticipated with the temperature increase. The primary pipeway was entered during the maintenance shutdown to inspect for interferences. Inadequate clearances were determined to exist between main heat transfer system and service system lines. Clearances were adequately increased by trimming insulation. A general inspection of the primary pipeway revealed no further areas of interference.

During the development of full operating temperatures, lines 231 and 233, primary loops No. 1 and No. 3 reactor outlet, moved in a manner that indicated an axial restraint existed. During the next shutdown, the primary pipeway was reentered and another inspection was performed. Small bits of debris (sheet metal screws, an electrical fitting, a small hexagonal wrench, and insulation) were found between the rubbing faces of the restraint saddles, which are installed on each of the three reactor outlet lines just outside of the guard pipe expansion bellows at the reactor cavity diaphragm seals. The purpose of these saddles is to prevent north-south movement of the three lines. The result of the debris in the contact faces was to increase the axial frictional restraint.

After all other maintenance in the immediate areas of the saddles was completed, the contact faces were thoroughly cleaned, inspected and coated with a dry lubricant. The three lines were observed to move satisfactorily, (within 0.5 in. of expected) as may be seen in Figure 830-2.

B. PHASE III (15 to 100% OF FULL POWER) TESTING

1. General

Included in this section is pertinent information regarding power range operation of the HNPF for testing purposes from the beginning of Phase III testing (HNPF steam first introduced to the turbine) to full design power. This phase of testing encompasses the period from May 24, 1963 through July 21, 1963. The report is chronologically oriented and includes information on testing and other activities affecting scheduling and plant availability. The sequence of testing activity, reactor operation, and reactor upper and lower plenum temperatures for this phase of the testing program are presented in Figure 830-3.

In this section of testing, several modes of operation were utilized to establish the conditions required for performance of the various tests and still meet (as nearly as possible) the grid dispatcher's requests for station generation.

Figure 830-2. Pipe Displacement vs Sodium Temperature

Figure 830-3. Phase III Testing Schedule

These are recognizable in the following sections as being one form or another of the following:

- a) Parallel operation of the conventional boilers and the HNPF, with the generated steam being supplied either to the turbine or to the turbine and steam dump system. Under this mode of operation, one or the other (or both) of the complexes must be base-loaded to avoid interaction of pressure controllers.
- b) Operation of the HNPF in the fully automatic load-following mode with the generated steam being supplied either to the turbine, the steam dump, or both, with either the turbine or steam dump base-loaded.
- c) Hot crossover, where the full or a part of the steam load is transferred from one source to the other. Again, this requires that at least one "side" (conventional or nuclear plant) be base-loaded.
- d) One plant supplying the full steam load and the other on "hot" standby. With the HNPF on standby, the generated steam was directed to the steam dump system.

2. Initial Power Operation to 25 Mwe

With the turbine operating at 20 Mwe on steam supplied from the conventional plant, the HNPF power level was increased to approximately 15%. Steam produced by the HNPF was supplied to the steam dump; manual load control was maintained from HCS-504, plant power control hand control station. Conventional plant steam conditions were matched to HNPF conditions and the HNPF steam header block valve was opened. The HNPF and conventional plant were operated in parallel. Initially, the HNPF was supplying approximately 37,000 lb/hr of steam to the turbine and approximately 52,000 lb/hr to the steam dump with HCS-505, steam dump pressure control, on manual. HCS-505 was programmed closed, and the HNPF was base-loaded from HCS-504 and supplying 89,000 lb/hr of steam to the turbine.

A subsequent rise in grid requirements resulted in an increase in HNPF output to 15 Mwe, at which time the HNPF was placed on fully automatic load-following control. As grid requirements decreased, the conventional boiler was secured; the HNPF was load-following at 20 Mwe with the total steam generated being supplied to the turbine.

During the following 48 hours, gross station electrical generation varied between 15 and 25 Mwe in keeping with grid requirements. Data were obtained at 15 and 25 Mwe for all tests.

On the fourth day of power operation, three scrams occurred. The first of these was initiated when the reactor was over-controlled manually during a period when manual control was being maintained for repositioning a control system feedback mechanism. The scram trip was a dT/dt computer negative temperature ramp.

The second scram occurred during a hot startup. The dT/dt computers were deflected into the negative trip range due to the decreasing core temperatures before sufficient reactor power was achieved to reverse the temperature trend.

The third scram was initiated by the power-to-flow ratio system (T_o computers). Computer B had been removed from the protective panel for an instrument check. Computer A was disturbed when computer B was racked out of position, thus initiating the two-out-of-three trips required for scram to occur.

3. Operation and Testing at Intermediate Power Levels (to 50 Mwe)

Upon successful startup of the reactor, power was increased to 40 Mwe gross to satisfy grid requirements. Steady-state data were recorded for all pertinent tests, and portions of AI-PC-630, Plant Control System, and AI-PC-770, Power Coefficients, were performed. A loss-of-load test was then performed from 25 Mwe in accordance with AI-PC-630, Plant Control System, and AI-PC-660, Scram and Loss of Load. The test was performed with the reserve station transformer in service; initiation was by manually tripping the turbine. The turbine was back on the line in approximately 7 min. Power was subsequently increased to 15 Mwe with the HNPF supplying the required steam on automatic load-following control.

Station power was increased to 50 Mwe gross with the HNPF supplying the full steam requirements. The load was reduced to 40 Mwe when it was determined that excessive carryover of water existed in the steam generators at the 50 Mwe load. Station power was maintained at a gross electrical generation of 40 Mwe for approximately 3-1/2 days. During this period, several activities took place, as follows:

- a) The uncompensated ion chambers in channels V, VII, VIII, and IX were withdrawn 11-1/4 in., 10 in., 7-3/4 in., and 6-3/4 in. from the bottom position, respectively, to equalize chamber output currents to that of the chamber in channel VI (the lowest reading). Upon completion of this effort, the five power range chamber currents were equal to 6.5×10^{-5} amp at 40 Mwe gross station power generation.
- b) Flux amplifier gains for the three protective system nuclear channels (V, VI, and VII) were adjusted to provide an indication of 42.1% at 40 Mwe. A similar adjustment was made on the plant control system channels (VIII and IX) to provide an indication of 50% at this power level.
- c) Adjustment of 70 fuel channel orifices was completed under AI-PC-800, Radial Power Distribution and Fuel Channel Orifice Adjustment.
- d) The "50% power" radiation survey was completed under AI-PC-690, Radiation Shielding.

Upon completion of the above, a planned scram was conducted in accordance with AI-PC-630, Plant Control System, and AI-PC-660, Scram and Loss of Load. The scram was manually initiated by depressing the SCRAM button from a station power level of 40 Mwe gross with all of the HNPF-produced steam directed to the turbine. Turbine power dropped to about 2 Mwe, at which point the conventional boilers picked up the load.

Prior to performance of the scram test, it had been determined that one control rod thimble was leaking helium into the reactor and another was suspected of leaking. These thimbles were identified as TZ-12 and TZ-13, respectively. After the scram, the reactor was cooled down, partially drained, and the thimbles removed to a storage cell. Final placement of the affected control rods was as follows:

<u>Core Position</u>	<u>Thimble No.</u>	<u>Poison Column No.</u>
C-1	TS-6 (stainless steel)	SN-13
C-32	TZ-1	SN-1
C-68	TZ-2	SN-12

Since the three poison columns were handled, control rod drive travel time tests were performed using the test cart. Results of the test were as follows:

TRAVEL TIME

<u>CR No.</u>	<u>Up</u>	<u>Down</u>
CR-1	12 min 11 sec	11 min 56 sec
CR-12	12 min 10 sec	12 min 01 sec
CR-13	12 min 26 sec	12 min 14 sec

The above travel times are all within the limits implied by a "down" travel speed of 12.6 ± 0.6 in./min and an "up" travel speed of 12.4 ± 0.6 in./min.

Following the planned scram from 40 Mwe, the reactor was shut down for 7 days for removal of the two leaking Zircaloy-2 control rod thimbles. At 0130 on June 8, 1963, the reactor was taken critical for the purpose of performing definitive tests on the steam generator water carryover condition. By late evening, the HNPF was carrying the full turbine load of approximately 15 Mwe gross. By the end of the day, power had been increased to 20 Mwe gross, 92.5 Mwt (as determined from channel X). The steam generator tests continued until the following morning, when a flow ratio scram occurred. The reactor was taken critical, but scrammed almost immediately on a low water level in steam generator No. 2. The reactor was again taken critical by performance of the hot startup standard operating procedure, but was shutdown soon afterwards when difficulties in starting the secondary sodium pumps were encountered. All three secondary pumps were determined to be bound up during the day. This was subsequently determined to be due to the combined effects of foreign material in the small clearances within the pumps, and pump case warpage due to uneven convection cooling.

4. Site Activities Affecting Scheduled Testing

The reactor was shutdown from June 9 until July 3, 1963. During this period, the following major activities took place:

- a) The secondary loops were drained and the three secondary sodium pumps removed for inspection and repair. It was determined that no permanent damage had been incurred. The pumps were cleaned, wear ring clearances increased slightly, and the pumps reinstalled.
- b) Sodium filters were installed in a fill and drain line (line 183) in the secondary sodium service system. Sodium was circulated from the

fill tanks through the filters and back to the fill tanks for the purpose of removing any foreign material that might be carried in the sodium. After the secondary pumps were reinstalled, a stream of sodium was tapped off the loops (one at a time) and passed through the filters to remove foreign material remaining in the loops.

- c) Auxiliary blowers and ducting were installed in each steam generator room to provide forced-draft cooling for the pump cases and, thus, to alleviate the pump case warpage condition.
- d) The primary pipeway was entered to inspect the primary system sodium valve operators.
- e) The secondary pump speed controllers were retuned.
- f) Special water level probes were installed in steam generator No. 1 evaporator for use in the water carryover testing.

The reactor was taken critical for a short time on July 3, 1963, to verify critical rod position. On July 5, the reactor was again taken critical and brought to a higher power level. Shortly before noon, the reactor was at a power level of approximately 38 Mwt with the steam produced being directed to the steam dump. At this time, the reactor was inadvertently scrammed by incorrect manipulation of the protective panel pushbuttons. The reactor was returned to criticality and by the end of the day, preparations for putting steam to the turbine were underway.

On July 6, reactor power was increased to carry about 35 Mwe of the 50 Mwe gross station load. A leak developed in the special evaporator water level probe fixture in steam generator No. 1. A one-circuit shutdown of circuit No. 1 was performed in accordance with AI-PC-630, Plant Control System. The steam generator carryover test was continued using steam generator No. 2. Circuits No. 2 and No. 3 carried the 35 Mwe load until late in the day, at which time reactor power was reduced and the reactor was shut down by manual scram.

In the ensuing 2 days, the reactor remained shutdown while the evaporator water level probe fixture was repaired and reinstalled in steam generator No. 1. The following day, the reactor was taken critical and increased in power to approximately 15%. The following day, HNPF steam was supplied to the turbine and steam generator carryover tests were continued. A reactor power level corresponding to 50 Mwe gross generation was maintained for the purpose of recording test data. All testing planned for the 50 Mwe test level was completed. Eight

fuel channel orifices were adjusted under AI-PC-800, Radial Power Distribution and Fuel Channel Orifice Adjustment, which reduced the measured fuel channel outlet temperature spread to 20°F.

5. Testing and Related Activities at High Power (Above 50 Mwe)

The following day, station power was increased to 65 Mwe gross with the reactor providing the required steam on fully automatic control. Power was maintained at a steady 65 Mwe gross while data were obtained for all concerned tests. Further investigation of the steam generator carryover problem was also accomplished.

Power was increased to a maximum of 81 Mwe gross with the HNPF supplying the required steam. A flow ratio scram occurred late in the evening after the peak generation requirements were past. The scram occurred coincidentally with initiation of flow in the secondary sodium service system for cold trapping, and was due to a flow upset created by the operation. Recovery was accomplished prior to the end of the day.

The planned full-power run was initiated the next day. Power was maintained between 75 Mwe and 83 Mwe gross station generation for the next 3-1/2 days. The lower power was necessary due to flow ratio alarms initiated by a decrease in maximum pump speeds, which in turn was triggered by a normal night-time system voltage drop, during the first night of the run. The pump speed controllers transferred from automatic to manual on deviation transfers. Pump speed control was maintained manually for a short period. After the condition repeated on the following night, the pump speed controller spans were increased to prevent further recurrence of the conditions.

Data were obtained for all tests during the full-power run. While at a power level calculated to be 242 Mwt, the five power range nuclear instrument chamber output currents were obtained. These were as follows:

<u>Channel No.</u>	<u>Chamber Current (microamps)</u>
V	12.8
VI	12.7
VII	12.7
VIII	12.7
IX	13.8

The chamber in channel IX was withdrawn 1-1/32 in., which reduced the chamber current to 12.7 microamp. Channels VIII and IX (power range control system channels) recorders and indicators yielded a reading of 100% after adjustment of the flux amplifier gains to the proper (10 volt) output at this power. The 100% indications were later readjusted to correspond to a reactor power level of 250 Mwt, as determined by calculated steam power.

Preparations to end the full power run were begun on the evening of the fourth day. It was planned to drop the load to 15 Mwe and hold this load to follow the xenon buildup and decay. The load decrease was initiated at 3 Mwe/min. When power reached 18 Mwe, an operator interpreted the amount of rod withdrawal by the neutron flux controller as being excessive and transferred flux control to manual. This resulted in a negative rate of change of measured fuel channel outlet temperature (dT/dt) scram.

Recovery from the above scram was accomplished early in the morning of the following day. During the day, both sodium pumps in circuit 3 were shut down for approximately 1 hr to permit replacement of the No. 3 secondary pump seal oil pump. No valve action was required to avoid system temperature drifts. Sodium flows in the circuit continued during the shutdown at a reduced rate due to convection head alone.

Power was increased to meet grid requirements after completion of the final portion of AI-PC-840, Xenon Poisoning. The next day, failure of TRC-2, fuel channel outlet temperature controller, resulted in a negative rate dT/dt scram. The controller was replaced and recovery was accomplished via the hot startup procedure.

Sodium flowmeter calibration curves were developed from information obtained from AI-PC-530, Heat Transfer Tests. The data obtained are presented in Figures 830-4 and -5 for reference purposes. Plant control system sodium flow transmitters were respanned using these curves. Work remaining includes re-alignment and verification of flux and flow signal spans and trip points in the computed fuel channel outlet chassis in the plant protective system.

Figure 830-4. HNPf Primary Sodium Flowmeter Calibration
(obtained from AI-PC-530, Heat Transfer Test, data)

Figure 830-5. HNPF Secondary Sodium Flowmeter Calibration
(obtained from AI-PC-530, Heat Transfer Test, data)

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with post-critical test AI-PC-830, Master Schedule - Including Rise to Power, as listed in NAA-SR-5700, Supplement 4, Page XI(c)-73, with remarks regarding the test to explain compliance with the criteria.

a) Criterion: The post critical tests have been completed satisfactorily.

Remarks: Status of all tests coordinated under this test procedure is as follows:

Completed -

AI-PC-510, Hydraulic Test - Main Heat Transfer System

AI-PC-530, Heat Transfer Tests

AI-PC-552, Steam Generator - Steam Purity

AI-PC-560, Cooling System Tests

AI-PC-640, Emergency Feedwater System

AI-PC-660, Scram and Loss-of-Load

AI-PC-690, Radiation Shielding

AI-PC-770, Power Coefficient

AI-PC-800, Radial Power Distribution and Fuel Channel
Orifice Adjustment

AI-PC-830, Master Schedule - Including Rise to Power

AI-PC-840, Xenon Poisoning

AI-PC-620, Plant Protective System — Formal prepared test completed. Verification of proper T_o computer performance over the full operating range after incorporation of flow calibration and correction of flux inputs remains. Component reliability evaluation under normal operating conditions is desirable. These two efforts, particularly the latter, are of a continuing nature and fall under the category of operational analysis.

AI-PC-630, Plant Control System — Effort remaining on this system is the verification of proper system performance over the operating power range after incorporation of the new flow calibration.

- b) Criterion: Safe operation of the plant at power levels up to and including nominal full power (240 Mwt) has been demonstrated.

Remarks: The plant has been safely operated at a maximum gross reactor power level of 250 Mwt. "Full power" has been defined as being 250 Mwt for purposes of establishing the parameters necessary for use in aligning the plant control and plant protective systems. Numerous hardware problems have been encountered; however, operation of the plant has not resulted in any unsafe conditions.

- c) Criterion: All components performance tested during the rise-to-power tests have performed satisfactorily in accordance with plant design parameters. Otherwise, actual performance has been measured and deemed acceptable, or suitable rectifying action has been taken.

Remarks: Actual performance of the various systems and components is reported in the individual test completion reports for the systems concerned. The major remaining problem; i.e., the Zircaloy-2 thimble failures, was resolved outside the cognizance of the test program, after the test was terminated, by replacing these with thimbles fabricated from type 304 stainless steel. (Also, see "Remarks" under Criterion a) above.)

V. EVALUATION

The objective of this test was to coordinate the testing activities during the rise-to-power portion of the post critical test program. A test procedure of this nature proved to be an excellent means for maintaining control of and assuring smooth performance of the rise-to-power testing. The responsible test engineer for this test was the center of control and information and, thus, was able to effectively coordinate and schedule the required testing activities.

The 12 formal post critical, rise-to-power tests resulted in a functional plant startup program. The program provided surveillance of the operating systems, collection of operating data, and determination of at least the minimum steady-state and transient operating parameters during the initial rise to and operation at full power. The data obtained at each of the 12 power level steps were sufficient to indicate possible system modifications and/or adjustments, and to predict plant conditions at the next planned power level. Plant safety could thus be evaluated and system adjustments accomplished prior to the next rise to higher power level. The selection of 12 steps in the rise to full power appears to be a good choice for accomplishing the overall objectives of the start-up program; i.e., to attain full power safely, obtain plant operating characteristics as a function of power, and train operating personnel.

Difficulties were encountered in making power level determinations at low power. There were several reasons for the difficulties. Uncertainty existed in sodium flowmeter calibrations; thermocouple errors at low ΔT had a large effect. The process instruments used for temperature readouts were, at best, accurate to only 1%. This will continue to be a problem when testing large power reactors at low power levels. However, the problem disappears at higher power levels as the error is no longer an appreciable part of the total measurement.

APPENDIX I

HNPF CORE ALIGNMENT TEST FOLLOWING 15% POWER OPERATION

The purpose of this test was to verify the alignment of the HNPF core after completion of the Phase II testing (0-38 Mwt) by removal and reinsertion of three selected fuel elements. The test was performed with the reactor isothermal at about 400°F and with no sodium flow in the primary system.

The alignment of the core was verified by withdrawing and reinserting fuel elements C-144, C-124, and C-164. These elements are located in the outer periphery of the core and are approximately 120° apart.

The elements were withdrawn and observed. No excess binding or load (see Table I) was observed on the fuel handling machine load cell for the three elements. Therefore, at this time, there is no apparent indication of misalignment. However, during the removal of C-144, a temporary hang-up occurred. This anomaly was investigated and attributed to a condition previously encountered. A position of 353 in. on the fuel handling machine position indicator corresponds to a position of the fuel assembly where the sleeve surrounding the hanger rod connection to the fuel element comes in contact with the bottom of the loading face shield.

TABLE I
FUEL ELEMENT LOAD

Position	Load (lb)			
	C-144	C-124	C-164	Acceptable Range
First 6 in. of travel up	1480	1450	1330	1100 - 2000
Maximum on remainder of travel	1180	1180	1130	1100 - 1400
Minimum on downward free travel	1000	1000	980	1000 - 1300
During last 6 in. of downward travel	-200	-180	-120	-1200 - 0

APPENDIX II

CONTROL ROD OPERABILITY

A. PURPOSE OF TEST

The purpose of the test was to check the performance of the control rod drive, actuator, and poison column assemblies after completion of testing at 38 Mwt. The test thus satisfies the operability check specified in Supplement 3 of the Safe-guards Report, NAA-SR-5700.

B. TEST METHOD

The operability of the control rod assemblies was verified by observing that various parameters were within technical specifications, and that the various indicators were operating satisfactorily. The data were analyzed and checked to verify that there were no trends, with respect to preoperational test completion reports AI-P-1145* and AI-P-1148,[†] that indicated possible future component malfunction.

The measured parameters included motor currents and voltages, withdrawal and insertion drive times, magnet latch release times (t_m), nine foot drop times (T_{al}), and full-drop times (T_{sl}). The motor currents and voltages, the drive times, and operation of the indicators were observed on the control rod test cart which was used to operate the control rod drives. After the forementioned data were taken for both rod insertion and withdrawal, the rod was driven to full-out position and dropped by manual scram from the test cart. The scram, "nine-foot-in," and "full-in" times were recorded as blips on a Sanborn recording oscillograph which was operating at a chart speed of 100 mm/sec.

The drive times were defined to be the times for the poison columns to travel over all but the tenth of an inch travel next to the "drive-in limit" position. These drive times were measured with a stop watch used in conjunction with the digital position indicator on the test cart. The drop times were recorded on the Sanborn recorder with the use of electrical signals.

The signals of the scram, release, and "full-in" positions were actuated by the respective indicator switches. The signal of the "nine-foot-in" position was generated when a magnet, which is on the falling part of the control rod assembly,

*Control Rod Drive and Actuator Assemblies

†Control Rod Operability

passed through a coil which was wound around the actuator housing. The circuits used are shown in Figure 830A-1.

C. PRESENTATION AND ANALYSIS OF TEST RESULTS

The data taken during the performance of the test are presented with the technical specifications in Tables I through IV. Results of the analysis are presented in Table V, and the methods of analysis are presented in Appendix A. In all cases, the results of this test are listed next to the results published in the preoperational test completion reports mentioned previously.

As can be seen from Table I, all of the drive times, except those of control rod 3, are within technical specifications. There are no apparent trends between this test and the preoperational tests. The drive times of control rod 3 were greater than technical specifications, but do not now affect the operability of the system since this control rod was replaced by a fixed poison column, and the actuator (SN-3) and drive (SN-20) were removed.

The motor currents and voltages are presented in Tables II and III. The motor currents were smaller, in general, than those observed in the preoperational tests. Since a decrease in current is a change away from an overload condition, and since the drive times were not affected, it was concluded that there is no trend toward drive malfunction.

The indicators performed satisfactorily except for a "Drive-In Limit" indicator and a "Snubber Up" indicator. The "Drive-In Limit" switch of control rod 15 was found to be faulty and was easily repaired. The "Snubber Up" indication for control rod 8 remained positive after the rod was driven to its drive-in limit. This malfunction resulted from the pickup latch orientation which caused an arm of the latch to close the snubber "up" switch. The repair was accomplished by replacement of the actuator and drive assembly with the assembly that was on control rod 2 before it was replaced with a fixed poison column.

The data taken during the rod drops are presented in Table IV and can be seen to be within the technical specifications. The magnet latch release times appeared to be consistently longer than those presented in AI-P-1145. The majority of the times for the first 9 ft of the drop were longer, and the majority of the times for the full drop were shorter than the results of AI-P-1145. Initial comparison of these results would indicate either faster fall, faster snubber closure, or a combination of these effects. It was recognized, as shown in Appendix B, that the

experimental error associated with the precision of the reading of the Sanborn chart is not negligible and affects the significance of the forementioned trends. Furthermore, the response of the magnet latch release time circuit is comparable to the precision of observation and is therefore a source of possible systematic error between observers.

In order to determine the extent of the changes in rod acceleration and snubber closure, the analysis of the drop was divided into three parts: the actual latch release, the free fall, and the snubber closure. The times associated with these parts of the drops were calculated as shown in Appendix A, and are presented in Table V. The calculations showed that the free fall times compared favorably. The changes which were larger than the precision of the test ($\pm 0.3\%$) are listed below.

<u>Control Rod</u>	<u>Change in Calculated Free Fall Time</u>
1	-0.8%
2	-1.8%
9	+1.5%
11	-0.9%
14	-0.9%

The calculated snubber closure times were found to be essentially unchanged. The changes observed were less than the precision of the calculation. All of the times were longer than the time necessary for stopping the falling rod with a uniform deceleration over the snubber closure distance (Appendix A).

D. CONCLUSION

The compliance of the drive and drop times with technical specifications and the verification of satisfactory indicator performance established the operability of the control rod system. Furthermore, the analysis indicated that any changes were small, and that no component malfunction should be expected during operation similar to, or less strenuous than that experienced during the period which included the completion of dry criticality, wet criticality, excess loading, and rise to the temperature conditions of full power operation at 38 Mwt.

TABLE I
DRIVE TIMES

Control Rod	Withdrawal Time (min) 12.6 ± 0.6		Insertion Time (min) 12.4 ± 0.6	
	P-1148	This test	P-1148	This test
1	12.434	12.305	12.200	12.276
2	12.618	12.881	12.267	12.886
3	12.450	13.500	12.250	12.969
4	12.618	12.579	12.216	12.092
5	12.783	12.735	12.467	12.485
6	12.400	12.691	12.217	12.566
7	12.292	12.309	12.167	12.043
8	12.184	12.290	11.985	12.097
9	12.535	12.893	12.342	12.513
10	12.710	12.851	12.500	12.816
11	12.417	12.739	12.167	12.298
12	12.362	12.308	12.167	12.187
13	12.735	12.766	12.300	12.318
14	12.117	12.414	11.892	12.150
15	12.735	12.647	12.517	12.574
16	12.400	12.507	12.133	12.304
17	12.684	12.830	12.467	12.592
18	12.800	12.490	12.492	12.118
19	12.384	12.526	12.116	12.170
3	12.450	13.667	12.250	13.157

TABLE II
WITHDRAWAL POWER

Control Rod	Current (amps)				Voltage (volts)			
	Phase A - B		Phase C - N		Phase A - B		Phase C - N	
	P-1145	this test	P-1145	this test	P-1145	this test	P-1145	this test
1	0.90	0.81	1.45	1.32	206	208	113	117
2	0.99	0.80	1.68	1.30	208	208	117	117
3	0.90	0.95	1.58	1.50	203	208	113	117
4	0.82	0.85	1.22	1.37	210	209	118	118
5	0.85	0.80	1.42	1.30	203	210	114	117
6	0.92	0.86	1.45	1.42	219	210	118	117
7	0.92	0.82	1.55	1.38	210	210	116	117
8	0.90	0.84	1.55	1.38	208	209	117	117
9	0.97	0.90	1.62	1.54	201	209	112	116
10	0.90	0.90	1.72	1.65	202	210	112	116
11	0.99	0.82	1.64	1.35	208	209	117	116
12	0.95	0.80	1.64	1.34	207	206	116	115
13	0.97	0.82	1.64	1.38	207	203	117	113
14	0.91	0.78	1.50	1.23	208	206	116	114
15	1.00	0.85	1.75	1.35	210	207	118	115
16	0.90	0.82	1.48	1.35	204	207	115	116
17	0.84	0.73	1.44	1.22	208	208	117	116
18	0.99	0.78	1.68	1.25	208	207	117	116
19	0.95	0.85	1.62	1.35	208	207	117	116

TABLE III
INSERTION POWER

Control Rod	Current (amps)				Voltage (volts)			
	Phase A - B		Phase C - N		Phase A - B		Phase C - N	
	P-1145	this test	P-1145	this test	P-1145	this test	P-1145	this test
1	1.45	1.45	0.90	0.63	200	208	115	117
2	1.61	1.38	0.98	0.58	208	208	118	118
3	1.58	1.62	0.90	0.70	202	208	115	118
4	1.19	1.40	0.82	0.60	210	208	120	118
5	1.43	1.35	0.88	0.58	202	210	116	119
6	1.22	1.48	0.93	0.65	218	210	118	119
7	1.60	1.42	0.95	0.58	218	210	119	119
8	1.55	1.42	0.95	0.62	208	209	117	119
9	1.62	1.58	0.98	0.75	201	210	115	119
10	1.70	1.68	0.95	0.65	201	210	116	119
11	1.73	1.38	1.02	0.60	208	205	119	115
12	1.61	1.42	0.96	0.58	208	204	119	116
13	1.65	1.36	0.98	0.58	207	204	118	114
14	1.40	1.32	0.93	0.58	208	207	119	117
15	1.71	1.45	1.00	0.65	210	207	120	117
16	1.50	1.42	0.90	0.60	204	207	117	117
17	1.43	1.33	0.87	0.55	208	208	119	117
18	1.66	1.28	1.00	0.50	208	207	118	117
19	1.58	1.43	0.97	0.59	208	207	118	117

TABLE IV
DROP TIMES

	Magnet Latch Release time (t_m) (msec)		Nine Foot Drop time (T_{al}) (msec)		Full Drop Time (T_{sl}) (msec)	
Tech specs	45 - 60		803		900 - 1015	
Control Rod	P-1145	this test	P-1145	this test	P-1145	this test
1	51	59	768	770	960	957
2	52	57	781	772	963	954
3	51	56	770	771	960	957
4	51	60	760	768	950	948
5	50	56	783	788	971	973
6	53	58	770	779	962	963
7	52	56	769	772	967	962
8	51	54	767	770	956	952
9	52	55	771	785	969	975
10	51	56	770	774	956	952
11	50	57	768	768	952	953
12	55	55	784	788	991	995
13	51	55	772	778	955	954
14	51	58	783	783	970	965
15	51	53	773	773	951	945
16	53	54	778	784	960	965
17	53	53	774	775	947	949
18	53	52	769	764	968	961
19	52	55	767	766	957	952

TABLE V
CALCULATED FACTORS

Control Rod	Latch Release* Time (t_r)		Free Fall Time* ($t_s - t_r$)		Snubber Closure* Time (t_c)	
	P-1145	this test	P-1145	this test	P-1145	this test
1	26	34	868	861	66	62
2	27	32	882	866	54	56
3	26	31	870	866	64	60
4	26	35	859	858	65	65
5	25	31	887	886	59	56
6	28	33	868	873	66	57
7	27	31	868	867	72	64
8	26	29	867	867	63	56
9	27	30	870	883	72	62
10	26	31	870	869	60	52
11	25	32	869	861	58	60
12	30	30	882	887	79	78
13	26	30	873	875	56	49
14	26	33	886	878	58	54
15	26	28	874	872	51	45
16	28	29	878	883	54	53
17	28	28	873	874	46	47
18	28	27	867	862	73	72
19	27	30	866	861	64	61

*All times are given in milliseconds.

A. SCRAM

B. LATCH RELEASE

C. INDUCTION COIL

D. ROD DOWN

Figure 830-A-1. Circuits Used For Drop Time Signals

APPENDIX A

A. MAGNET LATCH RELEASE TIME

There is about 0.1 in. free fall after the magnet releases the rod and before the release time is indicated. The time for this fall is calculated as follows:

for free fall,

$$S = 1/2 g t^2$$

where,

S = distance,

g = acceleration of gravity = 32.2 ft/sec²,

t = time for fall.

Using these numbers, $t = 22.8$ msec, $25 \text{ msec} \pm 3$ is the approximation used.

Then actual release time (t_r) is related to magnet latch release time (t_m) by

$$t_r = t_m - 25$$

B. FREE FALL TIME TO THE TOP OF THE SNUBBER

1) Free fall time through the first 9 ft is $T_{al} - t_r$.

where,

t_{al} is the measured 9 ft drop time, and t_r is actual release time.

2) Then by using the basic equation of free fall, $S = 1/2 at^2$ (where a is the acceleration);

$$\text{time for free fall to snubber} = t_s - t_r = (T_{al} - t_r) \left(\frac{\text{distance to snubber}}{9} \right)^{1/2}$$

C. SNUBBER CLOSURE TIME (T_c)

$$T_c = T_{sl} - (t_s - t_r) - t_r$$

where,

T_{sl} = full drop time

APPENDIX B

EXPERIMENTAL ERROR

The various drop times are observed by measuring the distance between recorded indicator movements on a Sanborn recorder chart. The precision of the reading is about 1 msec, and thus since any measurement of a time involves measurements of start and stop positions, no time can be known to be better than ± 2 msec.

- 1) Precision of t_m is ± 2 msec
- 2) Precision of $(T_{al} - t_r) = T_{al} - t_m + 25$ is ± 4 msec
- 3) Since $T_{al} - t_r > 730$ msec, precision is to $< \pm 0.28\%$
- 4) $T_s - t_r$ is proportional to $T_{al} - t_r$ and therefore is precise to $< \pm 0.28\%$
- 5) $t_s - t_r < 890$ msec therefore precision is to $< \pm 2.5$ msec
- 6) Precision of T_{sl} is ± 2 msec
- 7) Therefore precision of $T_c = T_{sl} - (t_s - t_r) - t_r$ is ± 6.5 msec

t_m = magnet latch release time,

T_{al} = 9 ft drop time,

t_r = actual latch release time,

t_s = time to fall to top of the snubber,

T_{sl} = time for complete drop,

T_c = time to close the snubber.

APPENDIX III

HNPf CONTROL ROD MEASUREMENT AND INSPECTION FOLLOWING 38 Mwt TESTING

To satisfy the requirements of NAA-SR-5700, four control rods have been inspected following 38 Mwt testing. Preceding the first inspection, the control rods had experienced a maximum temperature of 540°F and had no radiation history. During the 38 Mwt testing, the control rods were exposed to full operating temperature and to a neutron flux corresponding to about 15% of full power.

Nineteen control rods were measured for length using the fuel handling machine. While in the maintenance cell, the maximum and minimum diameters of the poison cans of four control rods were measured using go, no-go gauges. Also, in the maintenance cell, five control rods were examined for abnormalities such as scratches, marks, and dents. A description of the method used in making control rod measurements is given in TDR-8294.

Before wet criticality, and with the reactor isothermal, all the necessary information for comparing control rods was collected and recorded.* The information was reported in the appendix of TDR-8294. The information was again collected following 38 Mwt testing but with the reactor isothermal temperature greater than when the information was recorded prior to wet criticality. In order to compare the lengths of the control rods, a correction must be made for the temperature difference. The correction was determined to be 4.376×10^{-3} in./F°. The control rod lengths, as measured following 38 Mwt testing, are presented in column 5 of Table I and the corrected measurement for control rod lengths are presented in column 8. The measurements recorded prior to wet criticality are presented in column 9. A comparison of the measurements found in column 8 to those in column 9 indicates that no control rod has experienced an appreciable change in length.

Since control rod SN-16 was measured at ambient temperature prior to wet criticality, it will not have any temperature correction. The reason being that the upper plenum temperature was used as the temperature of the control rods and to determine the correction factor. After the control rods enter the fuel handling machine, they experience a temperature loss. How the temperature loss was accounted for was explained in TDR-8294. Due to this temperature loss, the temperature correction was not applicable to SN-16. However, control rod

*J. L. Mowrey, "Inspection of the HNPf Control Rods," TDR-8294, March 22, 1963.

SN-16 was considered not to have changed in length as it compares favorably with the other 18 control rods measured following 38 Mwt testing.

Following 38 Mwt testing, the poison cans were measured in the maintenance cell and allowed to cool to ambient. The poison can diameters following 38 Mwt testing were diminished since they were measured at an elevated temperature prior to wet criticality. Table II contains the poison can diameters as measured prior to wet criticality and following 38 Mwt testing. From the information shown in Table II, it was evident that none of the poison cans had swollen or deformed.

A visual observation revealed that two control rods had been scratched.

Serial No.	Core Position	Gang
SN-4	C-36	3
SN-15	C-86	1

Control rod SN-4 had three groups of short scratches on the bottom of poison can 13 and on the poison column hanger rod. The scratches were believed to be caused by the control rod rubbing the maintenance cell shield plug as the control rod was lowered into the cell for examination. Control rod SN-15 had a series of scratches from the bottom of poison can 13 to the bottom of the reactor shield plug. The scratches were all in the same area of the control rod and appeared to have little depth. The scratches that were found were believed to have been caused by an obstruction near the top of the control rod thimble. The scratches will have no effect upon safe operation.

The control rods now have experienced at least 50 drops. This is an unusually high number of drops when compared to the expected normal of 4 drops per year with a 3 or 4 year life expectancy. Thus, the control rods have shown their ability to withstand more than normal operation without failure. Furthermore, all the control rods that were measured and examined have shown no deformation and no abnormalities.

TABLE I
CONTROL ROD LENGTHS

Core Position	Serial Number	Pull Rods (in.)	Poison Column (in.)	Total Length following 38 Mwt Testing (in.)	Temperature (°F)	Temp Diff (°F)	Corrected* Total Length (38 Mwt-in.)	Total Length Prior To Wet Criticality (in.)
C-118	SN-19	278.00	156.00	434.00	390	68	433.70	433.45
C-38	SN-5	278.00	155.95	433.95	390	87	433.60	433.70
C-78	SN-14	278.05	155.90	433.95	390	85	433.60	433.60
C-44	SN-7	278.00	155.95	433.95	390	85	433.60	433.70
C-16	SN-6	278.05	155.95	434.00	390	85	433.65	433.55
C-36	SN-4	278.70	155.90	434.60	390	76	434.25	434.25
C-68	SN-13	278.05	156.00	434.05	390	87	433.65	433.60
C-110	SN-18	278.10	155.90	434.00	390	80	433.65	433.60
C-32	SN-12	278.05	155.90	433.95	388	78	433.60	433.65
C-1	SN-1	278.05	155.95	434.00	385	75	433.65	433.70
C-20	SN-2	278.00	155.90	433.90	385	80	433.55	433.50
C-86	SN-15	278.05	155.95	434.00	390	85	433.65	433.60
C-50	SN-8	277.95	155.90	433.85	390	85	433.50	433.55
C-24	SN-9	278.15	155.80	433.95	390	85	433.60	433.65
C-28	SN-3	278.00	155.95	433.95	390	80	433.60	433.60
C-62	SN-11	278.10	155.95	434.05	390	80	433.70	433.75
C-102	SN-17	278.15	155.80	433.95	390	80	433.60	433.60
C-56	SN-10	278.00	156.00	434.00	390	86	433.65	433.70
C-94	SN-16	278.10	155.85	433.95	390	-	-	433.35

*Correction Factor 4.376×10^{-3} in./F°

TABLE II
CONTROL ROD MEASUREMENT FOLLOWING 38 Mwt TESTING

Control Rod Poison Assembly CR-12				SN-12		Core Position C-32	
Can No.	Measurements Prior to Wet Criticality*			Measurements Following 38 Mwt Testing†			
	Temp (°F)	Max Diameter (in.)	Min Diameter (in.)	Temp (°F)	Max Diameter (in.)	Min Diameter (in.)	
1	210	3.430	3.428	116	3.424	3.423	
2	210	3.428	3.423	105	3.421	3.417	
3	185	3.429	3.427	102	3.425	3.422	
4	190	3.431	3.424	100	3.426	3.425	
5	180	3.428	3.422	98	3.424	3.423	
6	180	3.430	3.426	94	3.427	3.423	
7	175	3.429	3.426	94	3.424	3.421	
8	185	3.427	3.425	91	3.422	3.420	
9	170	3.430	3.425	90	3.425	3.415	
10	170	3.429	3.426	90	3.426	3.421	
11	170	3.431	3.426	90	3.425	3.417	
12	170	3.430	3.424	89	3.425	3.420	
13	170	3.430	3.426	89	3.424	3.421	

*Measurements taken 7-6-62.

†Measurements taken 3-13-63.

Control Rod Poison Assembly CR-15				SN-2		Core Position C-86	
Can No.	Temp (°F)	Max Diameter (in.)	Min Diameter (in.)	Temp (°F)	Max Diameter (in.)	Min Diameter (in.)	
1	208	3.430	3.429	101	3.424	3.423	
2	215	3.436	3.431	97	3.429	3.425	
3	202	3.433	3.428	96	3.426	3.423	
4	205	3.429	3.425	95	3.423	3.419	
5	200	3.435	3.430	94	3.429	3.423	
6	205	3.431	3.423	91	3.425	3.421	
7	195	3.431	3.426	90	3.425	3.420	
8	190	3.431	3.421	90	3.421	3.419	
9	185	3.428	3.425	90	3.421	3.419	
10	186	3.427	3.426	89	3.423	3.421	
11	185	3.428	3.426	87	3.423	3.421	
12	185	3.426	3.424	87	3.422	3.421	
13	185	3.429	3.425	85	3.422	3.420	

*Measurements taken 7-6-62.

†Measurements taken 3-14-63.

TABLE II (Continued)

Control Rod Poison Can Assembly CR-1				SN-4		Core Position C-1	
Can No.	Measurements Prior to Wet Criticality*			Measurements Following 38 Mwt Testing†			
	Temp (°F)	Max Diameter (in.)	Min Diameter (in.)	Temp (°F)	Max Diameter (in.)	Min Diameter (in.)	
1	220	3.432	3.424	116	3.428	3.420	
2	210	3.429	3.427	121	3.425	3.422	
3	210	3.429	3.423	115	3.425	3.421	
4	215	3.430	3.424	115	3.432	3.425	
5	190	3.436	3.428	114	3.425	3.421	
6	185	3.430	3.421	110	3.430	3.427	
7	190	3.431	3.426	111	3.428	3.421	
8	190	3.436	3.428	108	3.429	3.425	
9	190	3.429	3.426	103	3.426	3.423	
10	195	3.425	3.421	103	3.424	3.420	
11	190	3.436	3.428	98	3.425	3.423	
12	190	3.432	3.427	105	3.426	3.425	
13	190	3.441	3.423	94	3.436	3.417	

*Measurements taken 7-6-62.

†Measurements taken 3-14-63.

Control Rod Poison Can Assembly CR-2				SN-19		Core Position C-20	
Can No.	Measurements Prior to Wet Criticality*			Measurements Following 38 Mwt Testing†			
	Temp (°F)	Max Diameter (in.)	Min Diameter (in.)	Temp (°F)	Max Diameter (in.)	Min Diameter (in.)	
1	210	3.429	3.427	125	3.426	3.425	
2	215	3.431	3.426	127	3.428	3.421	
3	195	3.425	3.422	107	3.422	3.420	
4	200	3.430	3.427	123	3.425	3.421	
5	200	3.431	3.429	108	3.428	3.425	
6	200	3.436	3.426	118	3.431	3.421	
7	205	3.426	3.423	114	3.421	3.419	
8	190	3.435	3.429	107	3.429	3.424	
9	200	3.429	3.425	114	3.426	3.423	
10	230	3.437	3.432	105	3.430	3.425	
11	230	3.434	3.430	106	3.430	3.428	
12	230	3.433	3.429	95	3.425	3.422	
13	230	3.431	3.425	96	3.425	3.424	

*Measurements taken 7-6-62.

†Measurements taken 3-15-63.

HALLUM NUCLEAR POWER FACILITY

Post Critical Test Procedure

AI-PC-840

Xenon Poisoning

Test Procedure Submitted for Approval

6/30/61
Date

James E. Owens
Atomic Energy
Commission

Test Procedure Reviewed

2/1/63
Date

James E. Owens
Atomic Energy
Commission

Test Approved for Performance

2/20/63
Date

James E. Owens
Atomic Energy
Commission

Test Completed

7/20/63
Date

James E. Owens
Atomic Energy
Commission

Test Approved for Performance

2/20/63
Date

RE Buntin
Atomic Energy
Commission

Test Completed

7/20/63
Date

RE Buntin
Atomic Energy
Commission

HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
XENON POISONING

AI-PC-840

Test and Analysis

by

R. H. DAVIDSON

(Consumers Public Power District)

Evaluation

by

P. T. BURNETT

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

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HALLAM NUCLEAR POWER FACILITY
POST CRITICAL TEST COMPLETION REPORT
XENON POISONING, AI-PC -840

I. PURPOSE OF TEST

In essence, this test was the evaluation of xenon characteristics of the Hallam Nuclear Power Facility (HNPF) as a function of power history. The test was conducted first at 15 percent power as a part of Phase II testing. From this, xenon data were available to aid in reactivity analyses during the Phase III (15% to 100% power) portion of HNPF testing program. With the system at full power, the xenon test was performed again. This was done to verify earlier data, as well as to provide a more accurate description of full power xenon behavior. Specific test objectives included the evaluation of the following:

- a) Time required to reach equilibrium xenon after reaching a specified power level
- b) Reactivity worth of equilibrium xenon (at any power level)
- c) Reactivity worth of xenon buildup
- d) Reactivity worth of xenon buildup after shutdown
- e) Peak xenon reactivity worth
- f) Time required to reach peak xenon after shutdown.

II. DISCUSSION OF TEST PERFORMANCE

In keeping with the post critical testing program, this test was broken into two phases: (a) Phase II (0-15% of full power) and (b) Phase III (15-100% of full power).

Different test methods were used for Phase III testing than those employed for Phase II. For this reason, the two tests are considered separately.

A. PHASE II TEST

After a period of power operation at 38 Mwt, a period sufficient in length to achieve xenon equilibrium conditions, the reactor was scrammed. A hot

startup followed, and in a period of about 1-3/4 hours the reactor was raised to a low power level of ~ 0.1 Kwt.

Changes in core reactivity with time, due to xenon and temperature effects, were determined by a differential period technique. This method is designed to minimize errors in positioning rods for criticality as well as to minimize errors due to unknowns in rod worth. Reactivity determinations were made about once per hour, beginning soon after scram, but at a lesser frequency as the xenon decayed and reactor temperatures approached isothermal. Periods were measured using the output of a compensated ion chamber (Channel X).

During the first 26 hours after scram, the convection flow control system was employed to collapse reactor ΔT and reduce system temperatures. The sodium pumps were started in the 26th hour, helping to bring the core isothermal 33 hours after shutdown. Coolant, fuel, and moderator temperatures were monitored throughout the test.

The end point (no xenon) on the reactivity curve (Figure 840-1) was determined prior to bringing the reactor to power — at a time when no measureable xenon was present. It was necessary to make a small correction for fuel burn-up, as well as samarium and other fission product generation for the intervening period; however, by obtaining this base point in advance, it was possible to shorten the period of reactivity following by more than 30 hours. Figure 840-1 presents a graphic plot of reactivity change (temperature plus xenon) vs time, as determined by the differential period technique.

B. PHASE III TEST

To achieve test conditions, the reactor was operated at full power until equilibrium xenon was attained. Plans were to quickly reduce reactor power to 40 Mwt and follow xenon level by noting control rod position for 15 hours after power reduction. This method was chosen in preference to scrambling the system, since operating temperatures could be maintained. The correction for power coefficient effect between these two power levels is conveniently zero;* thus, for this effect no correction to measured data is required.

*Note integral of steady-state power coefficient curve, AI-PC-770, Power Coefficient.

Figure 840-1. Reactivity Change vs Time Following Scram From 38 Mwt

In practice, the reactor scrammed when the power had been reduced to about 50 Mwt. A hot startup followed, and the reactor was again operating at the 40 Mwt level with normal temperatures in just over four hours. From this point, no temperature correction to data was required. A calculated correction was made for samarium buildup after power reduction. This correction amounted to only 4 cents after 15 hours. The test was ended 15 hours after shutdown.

Changes in core reactivity with time were determined through control rod position and rod calibration data. Reactivity determinations were made every half hour during the test. Reactivity change vs time as determined is presented in Figure 840-2.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

A. PHASE II TEST

The reactivity data taken in the period following the scram from 38 Mwt equilibrium, show the effects of changes in both temperature and xenon poisoning. Samarium buildup and other miscellaneous fission product effects are negligible at this point. It is, therefore, necessary to first correct the reactivity data for the effects of temperature changes. The fuel coefficient and the moderator coefficient at the temperatures involved are approximately equal in magnitude, but opposite in sign (Refer to AI-PC-760, Zero Power Coefficients of Reactivity; AI-PC-770, Power Coefficient). Thus, with the fuel and moderator temperatures approximately equal, it follows that the overall temperature coefficient relates to the average sodium temperature in the core; this was the case throughout most of the period in which reactivity data were taken.

Figure 840-2 is the xenon curve obtained from measured data with temperature correction. The lower portion of this curve (core isothermal) was fitted with the theoretical xenon worth curve. The result is an exceptionally good fit, except during periods of great temperature transients. The first of these periods, ~0-6 hours after scram, is the initial core cooling; the second period, ~11-16 hours after scram, represents an adjustment in cooling rate as the core ΔT had been nearly collapsed; the third period, ~26-33 hours after scram, coincides with the startup of the primary sodium pumps. It becomes evident that a proper correction for temperature coefficient effects during transients is not known.

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page 3,5,1s

Figure 840-2. Xenon Poisoning vs Time After Scram From 38 Mwt Equilibrium

B. PHASE III TEST

Following the power reduction from full power, no reactivity data were available during the first two hours, due to an inadvertent scram which occurred. From two to five hours following shutdown, reactivity data were once more available, but a correction for temperature was necessary. In the fifth hour normal operating temperatures were attained; hence, no further temperature correction was needed. Figure 840-3 is a plot of corrected and uncorrected data. Except for a small correction for samarium buildup, no further correction to the measured reactivity data was necessary in separating the xenon data. The samarium correction reached a maximum of about four cents after 15 hours. The corrected data were fitted with a theoretical xenon worth curve; the results were favorable with a maximum deviation of about 5 cents.

An analog computer was used to aid in curve fitting and extending results to other power levels. The equations describing the production and loss of Iodine-135 and Xenon-135 were programmed on an analog computer (see Figure 840-4). Curve fitting was then accomplished by varying the magnitude of the parameters in these equations. The results of fitting experimental and theoretical data set the values in the following iodine and xenon equations:

$$\frac{dI}{dt} = \dot{I} = \gamma_i \Sigma_f \phi - \lambda_i I - \sigma_i \phi I$$

and,

$$\frac{dX}{dt} = \dot{X} = (\gamma_x \Sigma_f - \sigma_x X) \phi + \lambda_i I - \lambda_x X .$$

The subscripts i and x relate to iodine and xenon respectively, and

- $\dot{I}$ = concentration of Iodine-135
- $\dot{X}$ = concentration of Xenon-135
- σ = microscopic thermal absorption cross-section
- ϕ = thermal flux
- λ = decay constant
- γ = fractional yield from fission process
- Σ_f = macroscopic thermal fuel fission cross-section

Those parameters for the HNPF as determined are summarized in Table I.

Figure 840- 3. Xenon Worth vs Time After Shutdown From Full Power to 40 Mwt

Figure 840-4. Xenon Analog Computer Circuit

TABLE I
XENON AND IODINE PARAMETERS

Parameter	Symbol	HNPF Value
(Xenon c/s) (thermal flux) at 100% power	$\sigma_x \phi$	$1.84 \times 10^{-5} \frac{\text{neutrons}}{\text{sec}}$
Xenon decay constant	λ_x	$2.1 \times 10^{-5} \text{ sec}^{-1}$
Iodine decay constant	λ_i	$2.4 \times 10^{-5} \text{ sec}^{-1}$
Xenon yield — fractional	γ_x	0.003
Iodine yield — fractional	γ_i	0.056
Fission c/s of fuel (macroscopic)	Σ_f	0.42 cm^{-1}

Comparing the results obtained in this test with those predicted reveals excellent agreement. Furthermore, indications are that data obtained at one power level can be extrapolated with reasonable accuracy to higher power levels. A summary of the xenon worth curve fitting is presented in Table II; Table II also presents a comparison of Phase II and III data with predicted xenon worth data (pre-testing). Curves of negative reactivity due to xenon poison buildup vs time for 15, 50, and 100 percent power are given in Figure 840-5.

Through the use of the analog computer in the HNPF control room, xenon reactivity worth can be evaluated with an accuracy of better than ± 2 percent. To date, these data now used in reactivity history studies have suggested that the true accuracy is probably within ± 1 percent.

C. APPLICATION OF TEST RESULTS

Reference has already been made to the primary application of the results of the xenon test — that is, in reactivity history calculations. Through the use of the analog computer along with these test results, the xenon worth in the core at any time can be predicted quite accurately; the only additional requirement is that the power history for the preceding 60 hour period be known. This application is of importance since it makes possible the daily prediction of available reactivity with reasonable assurance.

A second application of the xenon test results is, in a sense, related to that already discussed. Through the results of the xenon test, an insight is provided for evaluating the reactivity worth of another important fission product poison — Samarium-149.

Samarium is like xenon in that:

- 1) It has a rather large cross section for absorption of thermal neutrons ($\sigma = 5.05 \times 10^4$ barns).
- 2) It builds up to an equilibrium value.
- 3) It builds up (to a higher equilibrium) upon shutdown.

Samarium is unlike xenon in that:

- 1) The equilibrium value is independent of flux.
- 2) The new equilibrium after shutdown is maintained until restart — there is no decay as with xenon.

TABLE II
XENON CHARACTERISTICS OF THE HNPF

Power Level	Source of Data	Equilibrium Xenon		Peak Xenon after Shutdown	
		Time	Magnitude	Time	Magnitude
15%	Measured Xenon	~3 days	-59¢*	1.5 hours	- 69¢†
	Theo. Fit to Meas. Xe		-61¢		- 62¢
	KINDLE		-62¢		- 63¢
50%	Extrapolation of Theo. Fit		-158¢	3 hours	- 163¢
	KINDLE		-159¢		- 164¢
100%	Measured Xenon		-239¢	4.8 hours	- 263¢
	KINDLE		-241¢		- 266¢
	NAA-SR-5700		-204¢§		- 228¢

*For the HNPF reactor, reactivity (ρ) in percent is approximately equal to 0.00693 times reactivity in cents.

† Contains improper temperature correction.

§ Equilibrium xenon plus samarium is reported as -2.3 percent. A breakdown results in -204 cents for equilibrium xenon.

Figure 840-5. Negative Reactivity - Xenon vs Time

An estimate is made of the value of equilibrium samarium through the simple ratio:

$$\frac{\sigma_x X}{\rho_{xeq}} = \frac{\sigma_s S}{\rho_{seq}}$$

Solving for equilibrium samarium reactivity (ρ_{seq}), and substituting for previously evaluated parameters yields,

$$\rho_{seq} = 122 \text{ cents} .$$

Using this value for equilibrium samarium and programming on the analog computer the promethium and samarium equations (Figure 840-6):

$$\frac{dP}{dt} = \dot{P} = \gamma_p \Sigma_f \phi - \lambda P$$

and,

$$\frac{dS}{dt} = \dot{S} = \lambda_p P - \sigma_s S \phi$$

produces the samarium buildup curve (Figure 840-7).

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with AI-PC-840, Xenon Poisoning, as listed in NAA-SR-5700 (Supplement 4), Page XI(c)-75, with remarks regarding the test to explain compliance with the criteria.

- a) Criterion: The time required to reach equilibrium xenon after reaching full power has been determined.

Remarks: As plotted on Figure 840-5, equilibrium xenon is reached approximately 63 hours after reaching full power.

- b) Criterion: The equilibrium xenon reactivity has been determined.

Remarks: Equilibrium xenon reactivity has been plotted for three power levels (15%, 50%, and 100%) as shown on Figure 840-5

- c) Criterion: An accurate graph of negative reactivity due to xenon poison buildup vs time has been made.

Remarks: Xenon poison buildup vs time was plotted as described in the report, and as depicted on Figure 840-5.

$\phi = 0-25 \text{ volts} \equiv 0-250 \text{ Mwt}$

Figure 840-6. Samarium Analog Computer Circuit

Figure 840-7. Samarium Poisoning

- d) Criterion: An accurate graph of negative reactivity due to xenon buildup after reactor shutdown vs time has been made.

Remarks: Xenon worth vs time after shutdown from full power is shown on Figure 840-3.

- e) Criterion: Peak xenon reactivity has been determined.

Remarks: Peak xenon reactivity has been determined as noted in Table II.

- f) Criterion: The time required to reach peak xenon after reactor shutdown has been determined.

Remarks: The time required to reach peak xenon after shutdown has been determined and is listed in Table II. Determination was over a three day period and relates to three power levels.

V. EVALUATION

Determination of the HNPF reactor xenon characteristics, during and after operation, was accomplished within the scope of post critical test AI-PC-840, Xenon Poisoning. Data were obtained and plotted for xenon buildup, equilibrium, and decay times, as set forth in NAA-SR-5700, Final Summary Safeguards Report for the Hallam Nuclear Power Facility.

Analysis of the test results yielded a quantitative understanding of the HNPF core xenon characteristics. The results are directly applicable to use in reactivity history calculations. There is excellent agreement between measured and calculated values of reactivity, although temperature related reactivity changes hampered the reduction of test data.

Test data divergences from the "best fit" curve are attributed to temperature transients, since the deviations unfailingly correspond to times when system temperatures were known to be changing. Inability to accurately determine actual reactor temperatures with the system in a nonisothermal transient temperature condition, and lack of knowledge of the partial temperature coefficients, makes determination of precise temperature corrections an improbable task.

The deviations between measured and calculated xenon reactivity effects are so slight, as to obviate the need for reexamination of the methods for calculating

these effects when predicting excess reactivity. Also, with an isothermal system, temperature corrections can be made within the accuracy of data plotted for AI-PC-760, Zero Power Coefficients of Reactivity.