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HALLAM NUCLEAR POWER FACILITY
VOLUME IV

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HALLAM NUCLEAR POWER FACILITY

Preoperational Test Procedure

AI-P-1158

Wet Critical Test

Test Procedure Submitted for Approval 3/1/61 James E. Owens
Date Atomic International

Test Procedure Reviewed 7/16/62 C. A. Purcell
Date Atomic Energy
Commission

Test Approved for Performance 8/7/62 James E. Owens
Date Atomic International

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Date Atomic International

Test Approved for Performance 8/7/62 R. B. Buntin
Date Atomic Energy
Commission

Test Completed 9/1/62 R. B. Buntin
Date Atomic Energy
Commission

HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
WET CRITICAL TEST

AI-P-1158

Test and Analysis

by

J. R. ROHLFS

Evaluation

by

O. G. JENKINS

J. M. ZETTERBAUM

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
PREOPERATIONAL TEST COMPLETION REPORT
WET CRITICAL TEST, AI-P-1158

I. PURPOSE OF TEST

The purpose of preoperational test AI-P-1158, Wet Critical, was to determine the minimum critical fuel element loading of the Hallam Nuclear Power Facility (HNPF) core when filled with sodium to the normal operating level. The data obtained will be utilized as the base information for AI-PC-710, Wet Excess Fuel Loading Test, and for verification of reactor analytical calculations including determination of sodium worth. The purpose of this test was to also determine minimum reactor criticality independent of neutron source activity (defined as Point "B") and to determine the flux level at which nuclear heating (a minimum of 2 F) is first observed (defined as Point "A").

II. DISCUSSION OF TEST PERFORMANCE

A. SUMMARY

Initial reactor criticality with sodium in the core was achieved at the HNPF on August 25, 1962, at 1306 hours, with 53 fuel elements in the reactor. Extrapolation of the wet excess loading curve (discussed in detail in the test completion report associated with AI-PC-710, Wet Excess Fuel Loading Test) indicates that 52.4 fuel elements were required for minimum criticality. Fuel loading and determination of the critical mass required approximately 11 days. The actual loading operation was accomplished in nine increments of 23, 7, 8, 3, 5, 3, 2, 1 and 1 fuel elements, respectively. Criticality was reached in three days less than originally scheduled, with the performance of all aspects of the procedure and equipment involved being equally satisfactory.

B. FUEL LOADING

The operations manager was notified of the intention to start the test and his permission was obtained to initiate loading. After fuel loading began, all plant operations required approval of the shift leader. Each of the shift leaders was cognizant of the test status at all times during his shift period. The test was performed on a 24-hours per day, seven-days-a-week basis, with the responsible test engineer directing the overall performance. Coordination of the test on the various shifts was by the shift test engineer and the shift leader.

Table I indicates the sequence for loading fuel, and the specific location for each fuel element according to its accountability numerical designation. Figure 1158-1 depicts the numbering arrangement used for the core process channels.

Following loading of the first ten fuel elements, and after each two subsequent elements were loaded, the count rate of each of the three incore nuclear instrumentation (ICNI) channels was recorded. The numerical average of the three count rates was used to determine the inverse of the average count rate ($\bar{I}$), which was plotted against number of fuel elements loaded (N) for extrapolating to criticality. By recording the count rates and extrapolating to "rods in" criticality after each two elements were loaded, a close instrument check and safe approach to criticality were maintained.

When an increment had been loaded, the control rod support carriage was placed on the loading face and the control rods were withdrawn according to the specified procedure. The count rates of the three ICNI channels were observed closely during all control rod withdrawals, and were recorded after steady state was achieved with the rods in four different configurations. These configurations were:

- 1) all rods inserted,
- 2) gangs 1 and 2 fully withdrawn with gangs 3, 4, and 5 fully inserted,

TABLE I
FUEL LOADING SEQUENCE

Loading Sequence	Core Position (C)	Fuel Element Number (MF)	Shield Plug Number (RP)	Type of Fuel
(First Increment)				
1	11	64	54	R
2	12	1	10	R-I
3	3	17	15	R-T-P
4	2	2	6	R-I
5	10	71	20	R
6	31	143	71	R
7	33	3	7	R-I
8	13	69	97	R
9	5	153	201	R-P
10	9	30	184	R-P
11	6	18	16	R-T-P
12	8	4	11	R-I
13	7	40	33	R
14	23	53	53	R
15	15	145	179	R
16	25	5	12	R-I
17	17	6	5	R-I
18	27	46	24	R
19	35	49	29	R
20	19	47	39	R
21	21	54	30	R
22	29	55	26	R
23	37	146	85	R
(Second Increment)				
24	26	31	166	R-P
25	34	152	133	R-P
26	18	158	93	R-P
27	22	61	83	R
28	30	62	98	R
29	14	63	42	R
30	53	42	213	R
R - Regular Fuel I - Instrumented - long probe T - Thermocoupled - meat T/C P - Premeasured				

TABLE I (Cont'd)

Loading Sequence	Core Position (C)	Fuel Element Number (MF)	Shield Plug Number (RP)	Type of Fuel
(Third Increment)				
31	65	72	34	R
32	41	34	214	R
33	51	35	47	R
34	63	36	48	R
35	39	37	40	R
36	55	39	211	R
37	67	41	56	R
38	43	66	45	R
(Fourth Increment)				
39	52	80	57	R
40	64	81	25	R
41	40	142	107	R
(Fifth Increment)				
42	54	20	159	S-P
43	66	51	51	R
44	42	65	22	R
45	58	68	67	R
46	70	70	44	R
(Sixth Increment)				
47	46	50	50	R
48	48	52	52	R
49	60	45	28	R
(Seventh Increment)				
50	72	48	27	R
51	47	44	35	R
(Eighth Increment)				
52	59	58	19	R
(Ninth Increment)				
53	71	144	82	R
R - Regular Fuel S - 3.45 Wt % U ²³⁵ P - Premeasured				

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
- ⊕ Incore Fission Chambers

Figure 1158-1. Reactor Cross-section

- 3) gangs 1, 2 and 3 fully withdrawn with gangs 4 and 5 withdrawn 50 percent; and
- 4) all rods fully withdrawn.

The inverse average count rates ($\bar{I}$) were determined for each control rod configuration and curves were plotted on three graphs in addition to the one mentioned in the preceding paragraph. These were average inverse count rate vs:

- 1) the square root of the total number of fuel elements loaded ($\sqrt{N}$),
- 2) the log of the total number of elements loaded ($\log N$), and
- 3) a number proportional to the radial buckling of the partially loaded core configuration (B_n).

Thus, at all times there were four graphs with four curves on each indicating fuel loading status and nearness to criticality.

The number of fuel elements in each fuel loading increment, after the first two, was determined by evaluation of the above mentioned curves. A specified fuel increment was loaded and the extrapolated critical loading was determined for each of the "all rods out" curves, by extrapolating each to $\bar{I}$ equals zero. Based on evaluation of these extrapolated critical loadings and the experience obtained during the dry criticality and dry excess tests, the official extrapolated critical loading was designated by the responsible test engineer. The number of fuel elements in the next loading increment was specified, using the procedural criterion that, "a succeeding loading increment will contain not more than one plus one-half of the number of fuel elements required to equal the designated extrapolated critical loading".

In addition, at each loading increment after 38 elements had been loaded, the worth of the control rods in terms of fuel elements was obtained. This was accomplished by determining the difference between the extrapolated critical loading as obtained from the "all rods in" and "all rods out" curves. This information was also considered in arriving at the number of elements in a succeeding loading increment.

The above procedure was repeated until the minimum number of fuel elements had been loaded to achieve criticality.

C. INCORE NUCLEAR INSTRUMENTATION

In addition to normal plant instrumentation, which was of insufficient sensitivity (with respect to low neutron flux levels) for monitoring the fuel loading activities, three incore nuclear instrumentation (ICNI) channels were employed. Three fission chambers were located in process channels C-191, C-215 and C-239. These chambers had three important functions; control, protection, and the acquisition of required data. Figure 1158-2 shows how the three ICNI channels, consisting of the three chambers, three scalars, two automatic digital printers and a dual preset counter, were arranged. Any two of these incore channels could be selected for automatic digital printout, while the third was recorded manually, as required. On each of the incore channels, a high level scram was set at 90% and an alarm at 75% of full scale. Each channel read out in cpm, up to a maximum of 1×10^7 cpm.

After 53 fuel elements had been loaded, and criticality was verified, the fission chambers were desensitized to prevent reaching the high level flux scram point when rods were withdrawn and periods established. This was accomplished by withdrawing the three chambers to approximately eight feet above the original chamber position at the core centerline.

D. COUNTING AND DATA PLOTTING METHODS

The count rates, indicative of the reactor neutron population, were obtained from the three ICNI channels and recorded from the digital printout tapes. They were considered as true count rates until the measured count rate reached 1.36×10^5 cpm. For measured count rates above 1.36×10^5 , the true count rate was obtained by using the following relationship:

Figure 1158-2. Arrangement of Nuclear Counting Instruments

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

where,

C_m = measured count rate

τ = resolution time of the circuitry

C_a = true count rate

Resolution time (τ) for the incore instruments, as determined during preoperational test AI-P-1166, Protective System Modifications for Fuel Loading, was 3.67×10^{-8} minutes. Therefore, C_a would not be appreciably greater (0.5%) than C_m until C_m reached approximately 1.36×10^5 cpm.

Indicated in the detailed step-by-step procedure were certain count rate statistical accuracies such as 0.5%, 1%, 2% and 5%. The statistical accuracy in percent is found from the following formula:

$$\text{Percent statistical accuracy: } \frac{1}{\sqrt{\text{total counts}}} \times 100$$

Thus, for a statistical accuracy of:

- 1) 0.5%, 5×10^4 counts were required
- 2) 1%, 10^4 counts were required
- 3) 2%, 2.5×10^3 counts were required
- 4) 5%, 400 counts were required

The true count rates of the three ICNI channels, with the specified statistical accuracy, were averaged and the inverse of this average was used as the inverse of the average count rate ($\bar{I}$). After each loading increment, the four $\bar{I}$'s were calculated corresponding to the four control rod configurations and the points were plotted on the appropriate graph.

There were four graphs plotted continuously throughout the test. Each graph had four curves on it corresponding to the four different control rod configurations for which steady state count rate data were recorded after each loading increment. The four graphs had inverse average count rate ($\bar{I}$) as the ordinate plotted vs:

- 1) total number of fuel elements loaded (N),
- 2) square root of the total number of fuel elements loaded ($\sqrt{N}$),
- 3) log of the total number of fuel elements loaded (log N), and
- 4) a number proportional to the radial buckling of the partially loaded core (B_n).

where,
$$B_n = \frac{10^5}{(R + S)^2}$$

and
$$R = \sqrt{\frac{NA}{\pi}}$$

N = number of fuel elements loaded

A = 870 cm² (cross sectional area of fuel element)

S = 22 cm (estimated radial reflector savings)

R = effective radius of core

In addition, the inverse count rate of each ICNI channel was individually plotted and evaluated for the rods full out configuration at each loading increment, for the purpose of checking instruments only.

E. CONTROL ROD MANIPULATION

The 19 control rods were operated as five gangs, consisting of 6, 6, 3 and 3 rods in gangs 1, 2, 3, 4 and 1 rod in Gang 5. Figure 1158-1

depicts control rod layout on the reactor loading face. The core locations of each gang are listed in Table II.

TABLE II
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

At the completion of each incremental fuel loading, the control rods were withdrawn by the following procedure:

- 1) Fully withdraw gangs 1 and 2 simultaneously.
- 2) Fully withdraw gang 3.
- 3) Withdraw gangs 4 and 5 to 50 percent out.
- 4) Fully withdraw gang 5.
- 5) Fully withdraw gang 4.

The nuclear instrumentation was observed closely whenever control rods were moved, and the count rates were allowed to reach steady state before proceeding, after each of the above steps. The count rates of the ICNI channels were recorded after steps 1, 3, and 5, in addition to the "all rods in" configuration previously discussed in B., "Fuel Loading".

When it was thought that criticality would be reached upon withdrawal of control rod gang 4 (53 fuel elements loaded), individual rods of this gang were withdrawn in 1/2 - foot increments, until criticality was reached.

F. DETERMINATION OF POINT "B" AND POINT "A"

To assure that true criticality had been attained, a plot of steady state count rate vs position of gang 4 was made. True criticality was defined as that range where any upward rod movement, however small, resulted in a change of flux, with respect to time, greater than zero or

$$\frac{d\phi}{dt} > 0, \text{ when } t \neq 0.$$

Any flux changes satisfying this condition indicate that the reactor is above the threshold of criticality. This threshold is termed "Point B". It can be shown that the source neutron effect is negligible at or above point "B". Thus, no source corrections are required when measuring reactor periods above this point.

Point "A" is the lowest flux level at which nuclear heating of not more than 2 F occurs in the fuel. As an assurance of obtaining satisfactory period measurements, it is important that Point "A" be 1 - 2 decades above Point "B"; thus obviating the need for reactivity corrections due to temperature effects when operating below point "A".

When Point "B" had been determined, the individual rods of gang 4 were used to maintain a period between 200 and 100 seconds. Nuclear heating occurred at approximately 10^5 cpm on source range channel II, as evidenced by a two degree fahrenheit temperature rise on the in-fuel thermocouples in the fuel element in core position C-3. It became apparent during testing to determine Point "B", that the range between point "B" and point "A" was limited (about 1 decade). To minimize nuclear heating and to reduce the critical flux level, the strength of the source was reduced by separating the antimony capsule from the beryllium sleeve. The antimony and beryllium sections are both approximately 10 inches long. It was found that the antimony capsule had to be withdrawn 30 inches in order to obtain approximately two decades of counting range on the incore instruments. A procedure was written and approved for accomplishing the source separation operation. Point "B" was re-determined with the source separated.

G. NUCLEAR PURITY

Nuclear counting statistical purity was determined with 23, 30 and 34 elements in the reactor. The purity check was made according to HNPF SOP No. 5902, "Procedure for the Investigation of Statistical Purity in Nuclear Counting". The purpose of the purity determinations was to:

- 1) compare the statistics of the ICNI counting to the theoretical statistics associated with neutron counting.
- 2) evaluate the deviation of actual counting data from theoretically pure neutron counting statistics, in order to ascertain the presence of systematic errors, double pulsing and overloading, and resolution time errors.

The method employed in the investigation is as described in the referenced standard operating procedure.

III. ANALYSIS AND PRESENTATION OF RESULTS

A. RODS-IN DATA

The average count rate of the three ICNI chambers for the "no fuel" condition was 6.7 cpm. The chambers were positioned 99 inches from the bottom of their respective thimbles (treated as the core centerline). Initial count rates with fuel in the core were obtained with 12 fuel elements in the reactor. An average count rate of 57.2 cpm was determined under this condition. The average count rate increased to 713.2 cpm with the 53 element critical loading in the reactor. Table III lists the average "rods in" count rate for each loading increment.

TABLE III

RODS-IN COUNT RATE DATA

Number of Fuel Elements in Core	Average Count Rate
0	6.7
23	135.97
30	216.86
38	410.0
41	468.2
46	644.4
49	685.9
51	692.2
52	706.1
53	713.2

The first plot initiated was "inverse count rate with rods in vs number of fuel elements" (Figure 1158-3). This plot was utilized for criticality extrapolation as specified in the test procedure. The extrapolated critical mass with rods "in" was determined by drawing a straight line through the last two points and extending it to the ordinate of the plot ($\bar{I}$ equals zero). An example of such an extrapolation is shown on Figure 1158-3.

Count rates were utilized for monitoring loading progress and the plotting of graphs, as opposed to multiplication, due to the uncertainty as to "no-fuel" data being available at the time the wet critical test procedure was developed. No corrections were made to count rates to compensate for temperature variations nor for decay of source strength. Sodium temperatures were held at $350\text{ F} \pm 5\text{ F}$ throughout the test. Source strength was sufficiently low and test performance time sufficiently short to make source decay effect insignificant.

B. INTERPRETATION OF $\bar{I}$ VS $\sqrt{N}$, $\text{LOG } N$, AND B_n PLOTS

Plots of inverse count rate vs the variables listed below were initiated after 23 fuel elements had been loaded. The linear plot initiated at 12 fuel elements was also continued.

- 1) Square root of the total number of elements loaded
- 2) Log of the total number of elements loaded
- 3) Number proportional to the radial buckling of the partially loaded core.

Data obtained during the fuel loading operations have been reduced and are presented in Figures 1158-4, -5, -6, -7 for the variables listed above. Each plot contains a curve for the "rods in" and "rods out" configurations as required by the procedure.

Each of the four plots was analyzed in the same manner with regard to determination of the number of fuel elements for the subsequent loading increment. A discussion as to the evaluation of the B_n graph is presented here and should be considered as representative of the methods used in interpreting the other three plots.

Figure 1158-3. Number of Fuel Elements (Rods In) vs Inverse Count Rate

Figure 1158-4. Number of Fuel Elements vs Inverse Count Rate

Figure 1158-5. Square Root of Number of Fuel Elements vs Inverse Count Rate

Figure 1158-6. Log of Number of Fuel Elements vs Inverse Count Rate

Figure 1158-7. Inverse Count Rate vs Factor Proportional To Radial Buckling

Considering only the B_n "rods out" curve on Figure 1158-8, a linear extrapolation was first made after 30 fuel elements had been loaded. As can be seen by the figure, the extrapolation indicated criticality would occur at a buckling value of 6.2, corresponding to 40.6 fuel elements. The greatest number of elements that could be loaded in the next increment was thus established by the procedure as six (one element plus one-half the difference between the actual and extrapolated loadings). However, this procedural limit was not imposed on the loading until after the thirty eighth element loading, as described in the HNPF Technical Specifications. Eight elements were therefore loaded to bring the reactor total loading to 38 elements.

The average rods out count rate associated with the 38 element loading was 4,902.65 cpm and the inverse of cpm was 0.000204. After plotting the 38 element point vs 0.000204, the 30 - 38 element buckling extrapolation indicated criticality would occur at 43.5 elements. It can be seen from comparing the two above extrapolations that the curve was beginning to break slightly and take the expected "approach to critical" shape. Experience gained during the dry critical loading showed the curve would break in this manner, with the fission chambers at the radii in which they were positioned. Three elements were loaded, per the extrapolation.

The characteristic breaking of the curve continued with the 41 element loading, more significantly than before, and the extrapolation indicated criticality would occur at 50 elements. The next increment was therefore five elements. The B_n plot was less conservative than any of the other three plots at this time, but experience gained from the dry critical loading showed that it was the most reliable.

After loading to 46 fuel elements, the B_n curve broke very little. The 41 and 46 element extrapolations indicated 50.3 elements were required for criticality, and agreed within 0.3 elements with the other three plots. This was the first positive indication of a specific critical point. The reactor was loaded to 49 elements.

Figure 1158-8. Factor Proportional to Radial Buckling vs Inverse Count Rate Showing Extrapolation

Once again the curve broke, this time somewhat unexpectedly, and extrapolated to criticality at 51.9 elements. The spread between the four curves also increased to 0.4 fuel elements with the log N plot being the most conservative for the first time. These extrapolations allowed the reactor to be loaded to 51 fuel elements.

With 51 elements in the reactor, the buckling curve indicated that criticality would occur exactly at 52 elements. Experience up to this time indicated that the curve would probably break at least slightly with the next loading, thus requiring perhaps slightly more than 52 elements for criticality. Criticality was predicted at 52.5 elements by the linear and log plots at the 52 element loading. The $\sqrt{N}$ curve indicated 52.6 elements and the buckling curve indicated 52.4 elements. Thus, the final element was recognized as being required and the reactor was brought to the critical loading of 53 fuel elements.

It is interesting to note that the 53 element wet critical loading was 21 elements more than that required for the minimum dry critical loading and that these 21 elements complete the first ring of fuel elements around the dry loading. Figure 1158-9 displays this relationship more prominently. It can be derived that the effect of filling the reactor with sodium at 350 F required these 21 elements to maintain criticality, and is the first indication of sodium worth.

Control rod worth in terms of a number of fuel elements was determined from each graph after the $\bar{I}$ value for each increment had been determined. This was done by subtracting the extrapolated number of "rods out" fuel elements from the extrapolated loading for "rods in" criticality. As can be seen from the linear plot in Figure 1158-7, the rod worth in terms of fuel elements at the 41 element loading was ~ 13 fuel elements. At criticality, the difference in the curves extrapolates to considerably more fuel elements.

Table IV lists the extrapolated value for criticality from the "rods out" configuration plots for each of the four plots maintained. The number of fuel elements loaded in each increment was derived from evaluation of these values.

Figure 1158-9. Comparison of Wet and Dry Critical Loadings

TABLE IV
EXTRAPOLATED CRITICAL LOADING - "RODS OUT"

No. of Elements Loaded	Linear Plot (N)	Square Root Plot ($\sqrt{N}$)	Log Plot (Log N)	Radial Buckling (B_n)
30	37.2	37.2	38.0	41
38	42.2	42.3	42.6	43.5
41	48.4	49.0	49.0	50.1
46	50.0	50.1	50.2	50.3
49	51.4	51.4	51.3	51.7
51	52.0	52.0	52.0	51.9
52	52.5	52.6	52.5	52.4

It should be noted that the linear plot is extremely conservative. The buckling plot appears to have merit because, even after a relatively few loadings of fuel, a critical loading close to actual criticality was extrapolated. In addition, the log plot and square root plot indicated the critical loading more closely than the linear plots. When near criticality, all four plots indicated similar values.

It was concluded that a shaping factor (square root, log, radial buckling) should be used to extrapolate criticality, especially during the first few fuel loading increments.

A general summarization of the data obtained in the critical test is presented in Table V.

C. POINT "B" AND POINT "A"

Following minimum criticality, the reactor flux level was increased to 20, 40, 60, 80 and 100 times the minimum value in an attempt to determine point "B". Analysis of the data obtained revealed that the procedure assumption that point "B" would fall somewhere between 20 and 100 times source critical was in error and that it actually occurred between 2, 3, 4, 5, 6, 8 and 10 times the minimum critical value. Figure 1158-10 displays the results and shows that establishing point "B" as about seven times minimum criticality is proper.

LEGEND: X + South, - North; Z + West, - East;
Y + Up, - Down
0 No movement or less than 1/8" observed or theoretically expected
X Reading is not valid and useless for analysis
— Measurement is not practical or does not represent pipe movement
NOTE: Movement is from ambient to temperature shown, except as noted.

TABLE I
PIPING DISPLACEMENT MEASUREMENTS

Measure- ment Point No.	Pipe Line No.	Area	Identi- fication or Hanger No.	Ambient				340°								Expected @ 350°				440°				Max°				Expected @ 540°				Movement 540° - 350°								
				Pipe Temp (F)	Base Reading (inches)			Pipe Temp (F)	Movement (inches)								Pipe Temp (F)	Movement (inches)				Pipe Temp (F)	Movement (inches)				Pipe Temp (F)	Movement (inches)				(Cool Down)								
					X	Z	Y		W/O Na	With Na	W/O Na	With Na	W/O Na	With Na	W/O Na	With Na		X	Z	Y	X		Z	Y	X	Z		Y	X	Z	Y	X	Z	Y	X	Z	Y	X	Z	Y
1			Bel lows Assy.	78°	0	-1/8		330°	342°			-1/2	-5/8			350°	-	-3/4	-	433°		-3/4		530°		-1 1/4		540°	-	-1 3/8	-		+3/8		-	+5/8	-			
2		Primary	504				64 7/8							+3/8	+3/8		-	-	+1/4				+3/8			+3/8		-	-	+1/2			-5/16	-	-	-1/4				
3	231	Pipe	Leak Det. BD-104		0	0						-1/2	-7/8				-	-7/8	-			X			X		-	-15/8	-		+3/8	-	-	+3/4	-					
4		Tunnel	505				98 1/8							+1/4	+3/8		-	-	+1/8				+3/8			+3/8		-	-	+1/4			+5/16	-	-	+1/8				
5		East	907	78°	-1/8	0		330°	342°			-1	-1			350°	-	-1/8	0	433°		-1 3/4		530°		-2 1/4		540°	-	-2 1/8	0		+5/8	-	-	+1	-			
6			Bel lows Assy.	78°	0	+1/8		345°	338°			+5/8	+5/8			350°	-	+3/4	-	433°		+1		530°		+1 3/8		540°	-	+1 3/8	-		-1/2	-	-	-5/8	-			
7			648				64 1/4							+1/8	+1/8		-	-	+1/4				+3/8			+3/8		-	-	+1/2			0	-	-	-1/4				
8	232		Leak Det. BD-105		-1/4	0						+3/4	+3/4				-	+7/8	-			X			X		-	+15/8	-		-1/16	-	-	-3/4	-					
9		Primary	649				105 1/8							0	0		-	-	+1/8				+1/8			0		-	-	+1/4			-1/16	-	-	-1/8				
10		Pipe	673		0	0		345°	338°			+1	+1/8				-	+1	0				+1 1/2			+2 1/8		-	+2	0		-13/16	-	-	-1	-				
11		Tunnel	Bel lows Assy.		0	0		330°	340°			+1/2	+3/4				-	+3/4	-				+7/8			+1 3/8		-	+1 3/8	-		-5/8	-	-	-5/8	-				
12		West	654				62 5/8							0	-1/4		-	-	+1/4				-3/8			-3/8		-	-	+1/2			-1/8	-	-	-1/4				
13	233		Leak Det. BD-106		0	0						+1 3/8	+1/4				-	+7/8	-			X			X		-	+15/8	-		-1/2	-	-	-3/4	-					
14			655				100							0	0		-	-	+1/8				-3/8			-1/4		-	-	+1/4			0	-	-	-1/8				
15			679	78°	0	0		330°	340°			+7/8	+1/8			350°	-	+1 1/8	-	433°		+1 1/2		530°		+2 1/8		540°	-	+2 1/8	-		-13/16	-	-	-1	-			
16	231		910	78°	0	0		330°	342°	-1/4	-1/8			0	0	350°	-1/4	-	0	433°	-1/4		0	530°	-1/4		0	540°	-1/2	-	0	+1/8			+1/4	-	-			
17		Primary	911		-3/8	-2 1/8		330°	342°			+1/2	+3/8				-	+3/8	0				+1/2	0		+5/8	0		-	+5/8	0		-1/4	-	-	-1/4	-			
18	232	Pipe	401														-	-3/8	0							-	-5/8	0		+3/8	-	-	+1/4	-	-					
19		Tunnel	911A		+1/8	0		345°	338°	0	+1/4			0	0		+3/8	-	0				+1/2	0		+5/8	0		+5/8	-	0	-1/8	-	-	-1/4	-				
20	233	North	683	78°	-4/4	+2 7/8		330°	340°			-1/2	-3/8				-	-3/8	0				-3/8	0		-5/8	0		-	-5/8	0		+1/4	-	-	+1/4	-			
21			682	76°	0	+1/8		330°	340°	-1/4	-1/8			0	0	350°	-1/4	-	0	433°	-1/8		0	530°	-1/4		0	540°	-1/2	-	0	+1/8			+1/4	-	-			
22			864A	74°	-1/8	+1/4		330°	342°	-1/2	-1/2					350°	-1/2	-	-1/8	433°	-1/2			530°	-5/8			540°	-7/8	-	-1/4	+1/4			+3/8	-	-			
23	231	IHX	865		0	+1/8						+3/8	0				-	+1/4	-1/4				+3/8			+1/2		-	+1/2	-1/2		-3/16	-	-	-1/4	-				
24		Cell	PI-100T		-1/8	0				+1/8	+1/8	+1/8	+1/4				0	+1/8	-				0	+1/8		+1/8	+1/8		0	+1/4	-	0	-1/8	0	-1/8	-				
25		No. 1	Pump Suct. Fl1		0	0				0	0	0	0				0	0	-3/8	433°	0	0	X		0	0	X		0	0	-5/8	0	0	0	0	-				
26	234		PI-101T		0	0				-1/8	-1/8	+1/8	+1/4				0	+1/8	-	435°	-1/8	+3/8			-1/8	+1/4		0	+1/4	-	0	-3/16	0	-1/8	-					
27			866	74°	0	0		330°	342°			+1/4	+1/2			350°	-	+1/4	-1/4	435°		+1/2		530°		+3/4		540°	-	+1/2	-1/2		-1/8	-	-	-1/4	-			
28			437A	72°	0	0		345°	338°	-3/8	-1/2					350°	-1/2	-	-1/8	433°	-3/4			530°	-7/8			540°	-7/8	-	-1/4	+3/8			+3/8	-	-			
29	232	IHX	436		0	0						+1/8	+1/4				-	-1/4	-1/4				-3/8			-3/8		-	-1/2	-1/2		+1/8	-	-	+1/4	-				
30		Cell	PI-200T		0	0				0	-1/4	0	-1/4				0	-1/8	-				-1/4	-1/4		-1/8	0		0	-1/4	-	0	+1/16	0	+1/8	-				
31		No. 2	Pump Suct. Fl1	72°	+1/8	0		345°	338°	0	0	-1/8	0			350°	0	0	-3/8	433°	0	0	X	530°	0	0	X	540°	0	0	-5/8	0	0	0	0	-				

LEGEND: X + South, - North; Z + West, - East;
Y + Up, - Down
O No movement or less than 1/8" observed or theoretically expected
X Reading is not valid and useless for analysis
— Measurement is not practical or does not represent pipe movement
NOTE: Movement is from ambient to temperature shown, except as noted.

TABLE I
PIPING DISPLACEMENT MEASUREMENTS

Measure- ment Point No.	Pipe Line No.	Area	Identif- ication or Hanger No.	Ambient				340°				Expected @ 350°				440°				Max.°				Expected @ 540°				Movement 540° - 350° (Cool Down)									
				Pipe Temp (F)	Base Reading (inches)			Pipe Temp (F)	Movement (inches)				Pipe Temp (F)	Movement (inches)				Pipe Temp (F)	Movement (inches)				Pipe Temp (F)	Movement (inches)				Actual				Expected					
					X	Z	Y		W/O	With	W/O	With		W/O	With	W/O	With		W/O	With	W/O	With		W/O	With	W/O	With	X	Z	Y	X	Z	Y	X	Z	Y	
32	235	IHX Cell No. 2	PI 201T	72°	0	0		345°	338°	-1/8	-1/8	-1/4	-3/8		350°	0	-1/8	—	427°	-1/4	-1/2		525°	-1/4	-5/8		540°	0	-1/4	—	0	0		0	+1/8	—	
33			438	72°	0	0		345°	338°			-1/4	0		350°	—	-1/4	-1/4		427°		-1/4		525°		-3/8		540°	—	-1/2	-1/2		+3/16		—	+1/4	
34	233	IHX Cell No. 3	452A	71°	0	-1/8		330°	340°	-5/8	-3/8			350°	-1/2	—	-1/8		433°	-7/8			530°	-1			540°	-7/8	—	-1/4	+7/16			+1/2	—	—	
35			451	71°	0	-1/8						0	-1/8			—	-1/4	-1/4				-1/8			-1/4			—	-1/2	-1/2		+3/16		—	+1/4	—	
36			PI 300T		+1/8	0					-1/8	-1/8	0	0			0	-1/8	—			-1/8	-1/8			-1/4	-1/8		0	-1/4	—	0	0		0	+1/8	—
37			Pump Suct. M11		0	0				0	-1/8	0	-1/8			0	0	-3/8		433°	0	0	X		0	0	X		0	0	-5/8	0	0		0	0	—
38	236		PI 301T		0	0				-1/4	-3/8	-1/8	-1/4			0	-1/8	—	429°	-3/8	-3/8			-3/8	-3/8			0	-1/4	—	0	+1/8		0	+1/8	—	
39			455	71°	0	0		330°	340°			-1/8	-1/4		350°		-1/4	-1/4		429°		-1/4		530°		-1/2		540°	—	-1/2	-1/2		+3/16		—	+1/4	—
40	265	IHX Cell No. 2	445A	72°	0	+1/4		335°	340°			-1/8	-1/8		350°	—	+1/8	+3/8		425°		0		520°		0		540°	—	+1/4	+5/8	0		—	-1/8	—	
41			446A		0	0						+1/8	+1/4			—	+1/8	+3/8				+1/8			+1/4			—	+1/4	+5/8		-1/8		—	-1/8	—	
42			449		0	0				+1/8	+1/8					+1/8	—	+3/8				+1/8			+1/4			—	+1/4	—	+5/8	-1/16		-1/8	—	—	
43			450		0	0						+1/4	+1/4			—	+1/4	—				+1/4			+3/8			—	—	+1/2	—		-1/4		—	-1/4	—
44			277A		0	0						0	+1/8			—	0	—				+1/8			+1/8			—	—	0	—		0	—	0	—	—
45			277B		0	0				0	0					0	—	+1/8		425°	0			520°	-1/8			0	—	+1/4	+1/8		0	—	—	—	
46			252			0	0					0	0				0	0		432°		0		530°		-1/8			—	0	0		0	—	0	—	—
47	274		253	72°	0	0		335°	340°	-1/8	0			350°	-1/8	—	0		432°	-1/8			530°	-1/8			540°	-1/4	—	0	+1/16			+1/8	—	—	
48			279	81°	0	0		335°	340°			-1/2	-1/2		350°	—	-1/4	-1/4		432°	0	-5/8		520°	0	-7/8		540°	—	-1/2	-1/2		+3/8		—	+1/4	—
49	2	IHX Cell	279A	81°	0	0		335°	340°	+1/8	+1/8			350°	0	—	-3/8		432°	+1/8	-1/4		520°	+1/8	-1/2		540°	0	—	-5/8	-1/8			0	—	—	
50			260	86°	-1/8	0		335°	340°	-1/4	-3/8				350°	-1/2	—	—		425°	-1/2			520°	-3/4			540°	-7/8	—	—	+7/16			+3/8	—	—
51	268	Steam Gen. Cell No. 2	260A		0	0						-1/4	-1/4			-1/2	—	—	425°	-1/2				-5/8			-7/8	—	—	+3/8			+3/8	—	—		
52			262A		0	0						-3/8	-1/8			-3/8	—	—	427°	-1/2				-5/8			-5/8	—	—	+1/4			+1/4	—	—		
53			PI 203T		0	0								+1/8	+1/8		—	-1/4		427°		X			X			—	-1/2	—		0		—	+1/4	—	
54			Pump Suct. M11		0	0					0	+1/4	-1/8	0			0	0	-1/2	427°	0	0	X	520°	0	0	X		0	0	-7/8	0	0		0	0	—
55	271		264A		0	+1/8				-1/4	-1/4				-3/8	—	-1/8		430°	-3/8			533°	-1/2			-5/8	—	-1/4	+3/16			+1/4	—	—		
56			208		-1/8	0						+3/8	+1/8			—	+3/8	0		430°		+3/8		533°		+5/8			+5/8	0		-3/8		—	-1/4	—	
57			263	86°	0	+1/8		335°	340°			+1/8	0		350°	—	+1/4	+5/8		430°		+1/8		533°		+1/4		540°	—	+1/2	+1/8		-1/8		—	-1/4	—

Figure 1158-10. Factor Times Initial Critical vs Control Rod Position In Inches (Gang 5)

During the initial test for point "B", when approximately 100 times source critical was reached, nuclear heating of 2 F was observed on in-fuel thermocouples. This event is defined as Point "A". Point "A" occurred at a count rate of $\sim 6 \times 10^5$ cpm, corresponding to a source range channel II reading of 1×10^5 cpm. Indications of heating occurred as low as 5×10^4 cpm, but a 2 F rise did not result until the 1×10^5 cpm level was reached.

It was apparent from the information available that point "A" and point "B" were too close together to allow adequate period runs to be made. As a remedial action the source was separated 30 inches, as it was in the dry reactor experiments, and the range between the low points expanded. Point "B" was re-determined and the band between source critical and true critical, with the source separated, was found to be practically non-existent. To ensure that adequate and proper data would be obtained in subsequent testing, the approximate limit of 5×10^4 cpm on source range channel II was decided as being the most desirable. This allowed adequate margin with respect to point "A". No lower limit was established as point "B" because it was readily apparent that with the source separated, minimum operational criticality was above Point "B". Any decrease in count rate below Point "B" resulted in insufficient indication of reactor condition, requiring a reactor shutdown.

D. NUCLEAR PURITY AND INSTRUMENTATION PERFORMANCE

Counting data were analyzed frequently to ensure the counting information was of adequate statistical purity. The method employed was as described in HNPF SOP No. 5902, "Procedure for the Investigation of Statistical Purity in Nuclear Counting", but will not be repeated here because of its length. However, Figure 1158-11 is a graphic result of such a purity check performed on one of the ICNI channels. The measured curve is within 1.0 ± 0.1 as required by the referenced SOP to indicate nuclear purity in counting.

Plots of individual channel inverse count rates vs number of fuel elements were maintained (to verify proper channel functioning), in

Figure 1158-11. Plot of Sum of the Class Content (percent) vs the Upper Limit of Each Class
(Nuclear Counting Purity Check Plot)

addition to the four average inverse count rate curves utilized for fuel loading extrapolations. These individual plots are shown in Figure 1158-12, 13 and 14 and substantiate that ICNI performance during the test was satisfactory. The three plots, when compared, show that what appears to be improper instrument counting approximately every third point is in reality an excellent following of the fuel loading sequence. As an element was loaded near a specific chamber, its associated channel reading would show a definite increase and the other two channels would either remain constant or decrease. Analysis of the curves was thus based on the absence of the "apparent" malfunction of instruments, rather than on its presence. Review of the curves substantiates that ICNI performance during the wet critical test was quite satisfactory. Also, the criterion of averaging the three channels to obtain average inverse count rate is thus established as being well founded.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with pre-operational test AI-P-1158, Wet Criticality, as listed in NAA-SR-5700, Supplement 4, Page XI(a)-82, with remarks regarding the test to explain compliance with the criteria.

- a) Criterion: The reactor is made critical with a minimum number of fuel elements in the wet core at 350 F.
Remarks: The reactor was made critical with 53 fuel elements in the core and all rods out except Gang 4 which was at 138 inches withdrawn. At this time, the reactor contained sodium at ~ 350 F. Criticality was previously attempted with 52 fuel elements in the reactor, but could not be achieved although all rods were completely withdrawn.
- b) Criterion: A subcritical calibration of the central control rod has been made.
Remarks: This requirement of the wet critical test was eliminated prior to approval (for performance) of the final

Figure 1158-12. Number of Fuel Elements vs Inverse Count Rate - Channel XI

Figure 1158-13. Number of Fuel Elements vs Inverse Count Rate - Channel XII

Figure 1158-14. Number of Fuel Elements vs Inverse
Count Rate - Channel XIII

draft of the test procedure. Results of the dry critical test indicated that information derived from the calibration had no direct application. Also, a calibration of the rod by the period technique was preferable, and was initiated immediately following the wet critical test by beginning the wet excess fuel loading test. Finally, prior to the wet critical test it was not definitely known that a "no-fuel" condition would exist, hence it was assumed that the "no-fuel" count rates would not be available and multiplication data could not be developed. (Consideration was given to filling the reactor with sodium immediately following the dry excess test. If this had occurred, the reactor would have been only partially unloaded, down to some number of elements less than the dry critical loading, to eliminate unnecessary fuel loading operations.)

- c) Criterion: Initial sodium worth has been determined.
Remarks: The first determination of sodium worth is in terms of fuel elements. The difference between the dry critical loading at 70 F (32 elements) and the wet critical loading at 350 F (53 elements) indicates the sodium reactivity worth is equivalent to 21 fuel elements under these conditions. The four methods for determining sodium worth, as outlined in NAA-SR-5700 (Supplement 4), are not applicable to this test. Two of the methods involved were analytical calculations, the third assumed the dry excess test would load at least as many elements as would be required by wet criticality (this was not the case), and the fourth method required completion of AI-PC-710, Wet Excess Fuel Loading Test. However, to ensure an actual determination of sodium worth would be accomplished, the wet excess fuel loading test was modified to include a minimum of two sodium draining experiments to directly measure the worth of the sodium.

V. EVALUATION

A. TEST SUMMARY

The HNPF reactor went critical with sodium in the core at a core loading of 53 fuel elements. This was the minimum critical loading possible with the neutron source in core position C-4. Fuel was safely loaded into the core in increments of 23, 7, 8, 3, 5, 3, 2, 1 and 1 elements. The average inverse count rate of the three incore fission chambers was plotted against four parameters; the number of fuel elements, the square root of the number of fuel elements, the logarithm of the number of fuel elements, and a number proportional to the radial buckling of the core. Count rates were obtained and the inverses plotted for: (1) all control rods inserted, gangs 1 and 2 fully withdrawn and gangs 3, 4, and 5 inserted; (2) gangs 1, 2, and 3 fully withdrawn and gangs 4 and 5, 50% withdrawn; and (3) all rods fully withdrawn. Only the latter plot was used to determine fuel increments to be loaded.

B. LOADING METHOD

Only one of the four plots which were prepared to guide the loading has a theoretical basis. This is the plot of inverse count rate vs a number proportional to radial buckling. For the one-group homogeneous cylindrical reactor, $B_r^2 = \frac{2.405}{R + \delta}$, where R is the radius of the reactor and δ is the radial reflector savings. For the heterogeneous reactor actually loaded, $R = \sqrt{\frac{NA}{\pi}}$ where N is the number of fuel elements loaded and A is the reactor cross sectional area associated with each fuel element. It is evident that B_r^2 is approximately proportional to $1/N$, especially as NA becomes large as compared with δ .

In the practical case, it therefore appears adequate, for this type of reactor, to plot inverse count rate vs N and vs $1/N$. The former is simple, not prone to calculative errors, and is conservative. The latter is almost as simple, and has the added advantage of giving a more accurate prediction of criticality early in the loading.

C. CENTRAL CONTROL ROD WORTH

Although a subcritical calibration of the central control rod was not made, as required to meet one of the test completion criteria, a good estimate of the central control rod worth at the 53 element loading can be made.

An excess reactivity of 14.9 cents was measured at the 53 element loading by calibration of the upper 26.9 inches of gang 5. Using the gang 5 integral worth curve determined during the dry excess loading *, a normalized integral worth curve was drawn to place the worth of the upper 26.9 inches at exactly 14.9 cents. The resulting integral worth curve is shown in Figure 1158-15. Total reactivity worth of gang 5 at the critical loading is 424 cents, or $2.94\% \frac{\Delta K}{K}$.

D. SODIUM WORTH

Determination of the exact critical fuel loading of 52.4 elements permitted normalization of the K_{eff} vs loading curve which had previously been calculated. This curve, shown in Figure 1158-16, has been normalized so that the K_{eff} at 52.4 elements is 1.000. The normalization is accomplished by taking the analytically calculated values of K_{eff} vs number of elements, multiplying the number of elements by $\frac{52.4}{50}$, and calling this the new value for the number of elements.

Also shown in the same figure are the theoretical and experimental K_{eff} vs loading curves for the dry critical and dry excess loading. Using the extrapolation of the experimental dry excess loading curve, the K_{eff} of the dry core, at a loading of 52.4 fuel elements and at 70 F is seen to be 1.064. This value can be corrected to 350 F by using the fuel and moderator temperature coefficients of reactivity presented in NAA-SR-5700, Supplement 4. Doing this yields a dry core K_{eff} of 1.065 at 350 F, or $\rho = 6.1\% \frac{\Delta K}{K}$. The calculated value of sodium worth at 350 F in the 53 element core is $5.5\% \frac{\Delta K}{K}$. Therefore, the early experimental measurement of sodium worth is in good agreement with the calculated value.

* Test Report AI-P-1163, Dry Excess Loading

Figure 1158-15. Position of Gang 5 (inches) vs Total Reactivity Worth (cents)

Figure 1158-16. K_{eff} vs Number of Fuel Elements

E. CONCLUSION

The initial wet critical loading of the HNPF was accomplished in a safe manner, and early estimates of central control rod worth and of sodium worth were obtained. The actual critical loading of 52.4 fuel elements agreed well with the previously calculated value of 50 ± 5 fuel elements.

In future fuel loadings of the HNPF reactor, valuable operating time can be saved by eliminating the determination of the exact critical loading. The "all rods inserted" inverse multiplication curve can be used in conjunction with the "all rods withdrawn" curve to assure adequate shutdown margin at all times during the loading, and the number of fuel elements to be loaded per increment can be chosen based on both these curves. In this way, the major objective of loading the core to its operational configuration can be achieved more rapidly, with no decrease in safety.

February 11, 1964

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HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
SODIUM LEVEL INSTRUMENTATION, AI-P-1159

I. TEST PURPOSE

The purpose of this test was two-fold; verification of proper functioning of sodium level instrumentation and calibration of each instrument during system filling.

II. DISCUSSION OF TEST PERFORMANCE

The sodium level instrumentation at Hallam is comprised of induction-coil elements. Each element consists of a fiberglass-insulated constantan wire coil wound on a bobbin. These coils are mounted on a hanger rod, spaced at intervals as required.

There are three types of level instruments employed at HNPF: alarm only (Figure 1159-1), alarm and incremental level (exact level is not of importance) indicator (Figure 1159-2), and continuous level indicator (Figure 1159-3). Table I shows the various vessels or tanks and the type of indicator and/or alarm associated with each.

The principle of operation is that of an a-c impedance bridge. The sensing and reference coils are located on the same level probe to compensate for temperature, lead resistance and installation variables. Output signals from the bridge circuit actuate relays for alarms, lights for incremental indication, and indicators and recorders for continuous level monitoring.

The test consisted of two basic parts: (1) test the sodium level instruments (prior to sodium fill) to verify that all components were functioning and that the sensing coils were appropriately spaced, and (2) monitor the level instruments during the system fill operation and make adjustments as required to ensure proper calibration.

In the first step, correct wiring of the sodium level instruments was verified by removing the level sensing probes from the vessels and inserting each probe into an aluminum sleeve (simulating sodium). This was completed on all

probes with the exception of those in the main heat transfer pumps - to remove the probes from the pumps requires that the pumps be disassembled. However, while primary pump No. 3 was dismantled for maintenance, the level probe was pulled out and checked in the aluminum sleeve. Numerous minor problems were encountered during this "dry check." These were corrected, the probes were rechecked in the aluminum sleeve, and reinserted into the thimbles.

Assured that the instrumentation was functioning and the alarm coils were located in approximately the right positions, sodium filling of the service systems was initiated. At this time the sodium level indicators were monitored; however, the data obtained were not valid due to drift of the instrumentation. Investigation of this "drift" revealed that the level probe coils contained moisture. This prevented calibration of the fill tank level instruments at this time. Calibration was initiated after the moisture was baked out of the coils.

The level alarms were calibrated by filling and/or draining until the alarm actuated. When an alarm annunciated, flow was stopped and the alarm removed from the thimble; a dipstick was inserted into the thimble to obtain the actual sodium level elevation.

Continuous level indicators were calibrated in a similar manner; however, the dipstick was inserted into the alarm thimble. This allowed comparison of dipstick level indication and continuous level indication as a tank or loop was filled.

Inspection of primary pump No. 3 revealed that the coils had been installed incorrectly. The high level alarm had been connected with the low level alarm and the reference coils were wired together (making it impossible to actuate the alarms). This was corrected and a similar correction was initiated on the five remaining main heat transfer pumps.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

Many of the continuous level instruments were not calibrated during this test, due to such items as faulty tubes, unstable power supplies, and bad oscillators. Many of the above were calibrated at a later date and the results are included with the data obtained during the test. Other level instruments, such as the pump level alarms, were tested during plant operation.

Whereas, during the test and subsequent plant operation, the desired results were obtained, there are certain inadequacies to be corrected. The data as presented in this report indicate excellent accuracy. However, to maintain this level of precision requires continual recalibration of the instruments. Action is underway to eliminate or minimize these problems.

Table II and Figures 1159-4 through 12 are provided as graphic presentation of sodium level instrumentation data acquired to date.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with preoperational test AI-P-1159, Sodium Level Instrumentation as listed in NAA-SR-5700, Supplement 4, Page XI (a)-83 with remarks regarding the test to explain compliance with the criteria.

- a) Criterion: All sodium level indicators read correctly to ± 3 in. of sodium level.

Remarks: All sodium level indicators calibrated by varying sodium level read correctly to ± 3 in. except the lower two coils on the reactor continuous level indicators. The reactor level indicators show a deviation of as much as 6 in. on the 2- and 4-ft ranges; however, on the 6-, 8-, and 10-ft scales there is good agreement between the actual and indicated level. The deviation on the lower ranges is not considered a serious problem because: (1) the normal operating level is on the 10-ft scale, and (2) the low and low-low level alarms are on the 10- and 8-ft scales, respectively. All indicators not actually calibrated this way were calibrated using an aluminum sleeve to simulate sodium. These gave satisfactory response to the aluminum sleeve. The following indicators were calibrated against actual known sodium levels:

- LI-403, Primary Sodium Fill Tank No. 1
- LI-404, Primary Sodium Fill Tank No. 2
- LI-405, Primary Sodium Fill Tank No. 3
- LI-406, Primary Sodium Fill Tank No. 4
- LI-407, Primary Sodium Fill Tank No. 5
- LI-100, Secondary Sodium Expansion Tank No. 1
- LI-200, Secondary Sodium Expansion Tank No. 2
- LI-300, Secondary Sodium Expansion Tank No. 3
- LR-1, Reactor Level No. 1
- LR-2, Reactor Level No. 2

b) Criterion: All sodium level alarm units alarm within alarm setting.

Remarks: All sodium level alarms were checked out and ascertained to be within alarm setting.

c) Criterion: Proper annunciation occurs on simulated high and actual low sodium levels.

Remarks: All sodium level annunciators have been verified to annunciate at the specified levels.

V. EVALUATION

Proper functioning of the sodium level instrumentation was verified by this test. Calibration of each instrument by varying sodium level was not accomplished. Those instruments not calibrated by varying sodium level were calibrated using an aluminum sleeve to simulate sodium. This is a valid technique for this calibration. The test completion criteria for this test were adequately met.

Operating history with the sodium level instrumentation subsequent to this test indicates that the reliability of this equipment has been poor. Excessive maintenance has been required to keep these instruments operating satisfactorily. Replacement of critical components with high reliability equivalents has been initiated to alleviate this problem. Circuit modifications are also being made. These changes are expected to reduce maintenance to an acceptable level.

TABLE I

SODIUM LEVEL ALARM LOCATIONS AND FUNCTIONS

Unit	Location	Function
LA-1	Reactor	High, Low, Low-Low Alarm
LA-2	Reactor	High, Low, Low-Low Alarm
LA-3	Reactor	High, Low, Low-Low Alarm
LA-100	Secondary Expansion Tank No. 1	High and Low Level Alarm
LA-200	Secondary Expansion Tank No. 2	High and Low Level Alarm
LA-300	Secondary Expansion Tank No. 3	High and Low Level Alarm
LA-101	Primary Pump No. 1	High and Low Level Alarm
LA-103*	Primary Pump No. 1	High and Low Level Alarm
LA-201	Primary Pump No. 2	High and Low Level Alarm
LA-203*	Primary Pump No. 2	High and Low Level Alarm
LA-301	Primary Pump No. 3	High and Low Level Alarm
LA-303*	Primary Pump No. 3	High and Low Level Alarm
LA-102	Secondary Pump No. 1	High and Low Level Alarm
LA-104*	Secondary Pump No. 1	High and Low Level Alarm
LA-202	Secondary Pump No. 2	High and Low Level Alarm
LA-204*	Secondary Pump No. 2	High and Low Level Alarm
LA-302	Secondary Pump No. 3	High and Low Level Alarm
LA-304*	Secondary Pump No. 3	High and Low Level Alarm
LA-400	Secondary Fill Tank No. 1	High and Low Level Alarm
LA-401	Secondary Fill Tank No. 2	High and Low Level Alarm
LA-402	Secondary Fill Tank No. 3	High and Low Level Alarm
LA-403	Primary Fill Tank No. 1	High and Low Level Alarm
LA-404	Primary Fill Tank No. 2	High and Low Level Alarm
LA-405	Primary Fill Tank No. 3	High and Low Level Alarm
LA-406	Primary Fill Tank No. 4	High and Low Level Alarm
LA-407	Primary Fill Tank No. 5	High and Low Level Alarm
ALARM AND INCREMENTAL INDICATORS		
Unit	Location	
LA-408	Primary Sodium Service Drain Tank	
LA-409	Sodium Transfer Tank	
LA-410	Secondary Sodium Service Drain Tank	
LA-411	Containment Drain Tank	
LI-1	Reactor	
LI-2	Reactor	
LI-400	Secondary Fill Tank No. 1	
LI-401	Secondary Fill Tank No. 2	
LI-402	Secondary Fill Tank No. 3	
LI-403	Primary Fill Tank No. 1	
LI-404	Primary Fill Tank No. 2	
LI-405	Primary Fill Tank No. 3	
LI-406	Primary Fill Tank No. 4	
LI-407	Primary Fill Tank No. 5	
LI-100	Secondary Expansion Tank No. 1	
LI-200	Secondary Expansion Tank No. 2	
LI-300	Secondary Expansion Tank No. 3	

*Spare

TABLE II
SODIUM LEVEL ALARM ANNUNCIATION ELEVATION

Element Number	Element Location	Alarm Elevation per Spec.		Actual Alarm Elevation	
		Low	Low-Low (Scram)	Low	Low-Low
LA-1	Reactor Core	1428'0"	1427'6"	1427'10-3/4"	1427'5-1/16"
LA-2	Reactor Core	1428'0"	1427'6"	1427'10-3/8"	1427'4-3/4"
LA-3	Reactor Core	1428'0"	1427'6"	1427'10-1/2"	1427'4-1/2"
		<u>Low</u>	<u>High</u>	<u>Low</u>	<u>High</u>
LA-100	Sec Expan Tk No. 1	1465'6-1/2"	1469'1/2"	1465'8"	1469'2"
LA-200	Sec Expan Tk No. 2	1465'6-1/2"	1469'1/2"	1465'6-1/4"	†
LA-300	Sec Expan Tk No. 3	1465'6-1/2"	1469'1/2"	1465'11"	1469'6-1/8"
LA-102	Sec Pump No. 1	1466'5-3/8"	1468'3-3/4"	1466'7-1/8"	1468'3"
LA-104	Sec Pump No. 1	1466'5-3/8"	1468'3-3/4"	1466'7-1/8"*	1468'3"*
LA-202	Sec Pump No. 2	1466'5-3/8"	1468'3-3/4"	1466'5-3/8"	1468'3-7/8"
LA-204	Sec Pump No. 2	1466'5-3/8"	1468'3-3/4"	1466'5-3/8"*	1468'3-7/8"*
LA-302	Sec Pump No. 3	1466'5-3/8"	1468'3-3/4"	1466'4-3/4"	1468'3-1/4"
LA-304	Sec Pump No. 3	1466'5-3/8"	1468'3-3/4"	1466'4-3/4"*	1468'3-1/4"*
LA-101	Pri Pump No. 1	1428'5-3/4"	1429'10-3/4"	1428'4-3/4"	†
LA-103	Pri Pump No. 1	1428'5-3/4"	1429'10-3/4"	1428'4-3/4"	†
LA-201	Pri Pump No. 2	1428'5-3/4"	1429'10-3/4"	1428'4-3/4"	†
LA-203	Pri Pump No. 2	1428'5-3/4"	1429'10-3/4"	1428'4-3/4"	†
LA-301	Pri Pump No. 3	1428'5-3/4"	1429'10-3/4"	1428'4-3/4"	1429'10"
LA-303	Pri Pump No. 3	1428'5-3/4"	1429'10-3/4"	1428'4-3/4"	1429'10"
LA-403	Pri Fill Tank No. 1	1418'4-1/2"	1428'3-1/2"	†	1428'3"
LA-406	Pri Fill Tank No. 4	1418'4-1/2"	1428'3-1/2"	†	1428'3-1/2"

*LA-104, 204, 304, 103, 203, and 303 are spares. Due to physical construction it can be assumed that the alarm coils are at the same level.

†Data not obtained.

Ref: AI Dwg, 7508-D972416

Figure 1159-1. Sodium Level Alarm Diagram (Alarm Only)

Figure 1159-2. Sodium Level and Alarm Diagram for Step Level Monitor

Figure 1159-3. Sodium Level Diagram for Continuous Level Monitor

Figure 1159-4. Primary Fill Tank No. 1 Level Instrument (LI-403) Reading vs Dipstick Reading

Figure 1159-5. Primary Fill Tank No. 2 Level Instrument (LI-404) Reading vs Dipstick Reading

Figure 1159-6. Primary Fill Tank No. 3 Level Instrument (LI-405) Reading vs Dipstick Reading

Figure 1159-7. Primary Fill Tank No. 4 Level Instrument (LI-406) Reading vs Dipstick Reading

Figure 1159-8. Primary Fill Tank No. 5 Level Instrument (LI-407) Reading vs Dipstick Reading

Figure 1159-9. Secondary Expansion Tank No. 1 Level Instrument (LI-100)
Reading vs Dipstick Reading

Figure 1159-10. Secondary Expansion Tank No. 2 Level Instrument (LI-200)
Reading vs Dipstick Reading

Figure 1159-11. Secondary Expansion Tank No. 3 Level Instrument (LI-300)
Reading vs Dipstick Reading

Figure 1159-12. Reactor Level (LR-1 and LR-2) Reading vs Dipstick Reading

HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
SODIUM FLOW INSTRUMENTATION

AI-P-1160

Test and Analysis
by
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Evaluation
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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
PREOPERATIONAL TEST COMPLETION REPORT
SODIUM FLOW INSTRUMENTATION, AI-P-1160

I. PURPOSE OF TEST

The purpose of this test was to verify that all permanent-magnet sodium flowmeters and associated readouts operate correctly.

II. DISCUSSION OF TEST PERFORMANCE

The sodium flow instrumentation consists of six 14 in. , one 6 in. , three 3 in. , four 2 in. , and three 1 in. permanent-magnet flowmeters and the associated recorders and indicators. The 14 in. flowmeters measure flow in the primary and secondary coolant loops. The 6 in. flowmeter measures moderator coolant flow. The remaining flowmeters measure various flow rates in the sodium service system. Provisions for remote measurement of the fringe flux density are included in the 6 in. and 14 in. flowmeters.

Each flowmeter was visually inspected for correct installation. The electrodes were inspected for correct location and alignment. The permanent magnets were inspected to insure freedom from clinging magnetic material. Signal wiring was tested for continuity between readouts and flowmeters, and loop resistances of all signal leads were measured. The fringe flux densities of the 6 in. and 14 in. flowmeter magnets were measured by use of the magnet flip coils. All readouts were calibrated using flowmeter data obtained from tests performed at the Atomics International Santa Susana (California) Field Test Laboratory. These data are presented in Table I.

Upon examining the small flowmeters, it was found that the majority of flowmeter magnets had to be cleaned of metal debris left from construction. The electrode insulators of one flowmeter had to be cleaned of aluminum paint.

TABLE I
FLOWMETER CALIBRATION DATA *

Flowmeter	Flowmeter Size	Readout Full Scale Flow	Flowmeter Output For Full Scale	Sodium Temp
FRC-101E	14 in.	3×10^6 lb/hr	32.0 mv	600 F
FRC-201E	14 in.	3×10^6 lb/hr	32.0 mv	600 F
FRC-301E	14 in.	3×10^6 lb/hr	32.0 mv	600 F
FRC-102E	14 in.	3×10^6 lb/hr	34.0 mv	900 F
FRC-202E	14 in.	3×10^6 lb/hr	34.0 mv	900 F
FRC-302E	14 in.	3×10^6 lb/hr	34.0 mv	900 F
FR-3E	6 in.	500 gal/min	22.5 mv	600 F
FI-400E	3 in.	100 gal/min	11.0 mv	400 F
FI-401E	3 in.	100 gal/min	11.0 mv	400 F
FI-404E	3 in.	100 gal/min	11.0 mv	400 F
FI-402E	2 in.	30 gal/min	5.0 mv	400 F
FI-403-1E	2 in.	30 gal/min	5.0 mv	400 F
FI-403-2E	2 in.	30 gal/min	5.0 mv	400 F
FI-406E	2 in.	30 gal/min	5.0 mv	400 F
FR-400E	1 in.	1 gal/min	1.0 mv	400 F
FR-401-1E	1 in.	1 gal/min	1.0 mv	400 F
FR-401-2E	1 in.	1 gal/min	1.0 mv	400 F

* Magnetic flowmeter "flux decay vs time" factor was intentionally omitted in arriving at the calibration data because of the incompleteness of this information.

References used in obtaining the calibration data were as follows:

- 1) W. J. Fraser, "Calibration Test of HNPF Sodium Flowmeters", Atomics International TDR No. 5586, (August 15, 1960).
- 2) W. J. Fraser, "Calibration Test of HNPF 14-inch Sodium Flowmeters", Atomics International IOL to F. C. Grone-meyer, (August 25, 1960).

In testing the 6 in. and 14 in. flowmeters it was found that none of the flip coils for measuring the fringe flux of the flowmeters would operate. A variety of faults was found which included wiring errors, flip coil solenoid malfunction, lagging interference, and deterioration of flip coil high temperature insulating cement. The deterioration of the flip coil insulating cement was due to moisture from the lagging. The flip coils and flip coil solenoid motors were reworked and reinstalled, after which fringe flux measurements were obtained.

Fringe flux measurements were obtained first with the flip coils resting-position parallel with the magnet pole faces. Later measurements were obtained with the 14 in. flowmeter flip coils positioned 30 degrees off the plane parallel to the magnets pole face and in the direction of coil rotation.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

The fringe flux measurements which were taken with the flip coils on the 6 in. and 14 in. magnets are presented in Table II. The first fringe flux measurements which were taken at the site had the flip coils resting-position parallel with the magnet pole faces. This was thought to be the resting-position of the flip coil during the original fringe flux measurements at the Field Test Laboratory. The site measurements turned out to be lower than those taken previously at Santa Susana, by a factor of about 20%.

The 14 in. magnet flip coil resting-positions were then shifted 30 degrees in the direction of rotation as called out on the flip coil assembly drawing. New fringe flux measurements were made and these were found to be approximately 35% above those originally obtained in California.

Flux decay for these magnets was to be determined by intercomparison of the initial flux measurements made in California, with the fringe flux measurements performed at the site. With a known flux decay, the up-to-date flowmeter calibration could be obtained. In practice this was not successful, as is shown by the data. Unfortunately, several factors

TABLE II
FRINGE FLUX MEASUREMENTS

Flowmeter	Coil Circuit Resistance (ohms)	Fluxmeter Readings Taken at AI	Fluxmeter Readings (Coil Parallel with Pole Face)	Fluxmeter Readings (Coil 30 Degrees off Pole Face Plane)
FRC-101E	13.69	548 *	444 *	741 *
FRC-201E	13.46	558	449	849
FRC-301E	13.88	600	332	915
FRC-102E	15.27	570	520	813
FRC-202E	14.50	570	---	893
FRC-302E	15.71	615	432	834
FR-3E	14.37	---	1281	---
NOTE: Where possible, a resistance was added to the above coil circuit resistance so that the total circuit resistance was 15 ohms when flux readings were taken.				
* Divisions of Meter Deflection (Sensitive Instrument Research Corp., Model FM multirange fluxmeter).				

entered which combined to make the comparison of fringe flux measurements unacceptable. These factors are as follows:

- a) The flux measurements taken at Atomics International used one flip coil assembly which was transferred from one flowmeter to another.
- b) All flip coil assemblies were removed from the flowmeters at the Hallam Nuclear Power Facility (HNPF), repaired, and reinstalled.
- c) The final flowmeter installations at HNPF differed from the installations on which the original fringe flux measurements were made. Such things as surrounding electrical junction boxes, conduit, and pipe supports undoubtedly have some effect on the fringe flux of the magnet.
- d) Measurements were made with different fluxmeters.

It is therefore recommended that the fringe flux measurements taken in this procedure be used only in determining future magnet flux decay.

Flowmeter loop resistances are presented in Table III. All resistances were in agreement with the length of the signal leads.

TABLE III
FLOWMETER SIGNAL LOOP RESISTANCE

Flowmeter	Resistance (ohms)	Flowmeter	Resistance (ohms)
FRC-101E	3.5	FR-400E	0.92
FRC-201E	3.8	FR-401-1E	0.64
FRC-301E	3.2	FR-401-2E	0.73
FRC-102E	2.5	FI-400E	0.47
FRC-202E	1.2	FI-401E	0.95
FRC-302E	1.0	FI-402E	0.86
FR-3E	5.0	FI-403-1E	1.07
		FI-403-2E	0.62
		FI-404E	0.70
		FI-406E	1.07
NOTE: Flowmeter loop resistances were measured from the terminals in the board containing the readout instrument.			

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with pre-operational test AI-P-1160, Sodium Flow Instrumentation, as listed in NAA-SR-5700, Supplement 4, page XI(a)-84, with remarks regarding the test to explain compliance with the criteria.

- a) Criterion: All flowmeters are installed per installation drawings.

Remarks: Each flowmeter was visually inspected for correct installation.

- b) Criterion: All flowmeters are free of clinging material.
Remarks: All flowmeters which were found with clinging material were cleaned.
- c) Criterion: All flowmeters have correct signal circuitry. The signal loop resistances have been measured and recorded.
Remarks: A voltage was applied at the electrodes of each flowmeter, and to verify signal wiring, a signal was observed at the readout end of the corresponding signal leads. Table III presents the loop resistance of the flowmeter signal circuit.
- d) Criterion: All 6 and 14 inch flowmeters have freely operating and correctly wired flux test coils.
Remarks: The test or flip coil for each flowmeter was observed to operate freely. Measurements obtained when the flip coils were remotely actuated gave proof that the flip coils were wired correctly.
- e) Criterion: Calibration data for all flowmeters have been entered on data sheets.
Remarks: Tables I, II, and III present calibration data for all flowmeters.
- f) Criterion: All flowmeter readouts have been calibrated using the calibration data for its associated flowmeter.
Remarks: All flowmeter readouts were calibrated using data in Table I.

V. EVALUATION

Preoperational test AI-P-1160 was performed to verify proper installation, cleanliness, circuitry and flip coil action of the permanent-magnet sodium flowmeters, and to calibrate the associated flowmeter readouts. During the test, a number of installation errors were discovered and corrected. Objectives of the test were fulfilled as recorded in the master test procedure and discussed in the test completion report. The problem areas that require clarification, however, are:

- a) Flip Coils
- b) Noise
- c) Calibration

A. FLIP COILS

Magnet fringe flux measurements for the six 14 inch flowmeters were made at the Atomics International Santa Susana Laboratory, using one solenoid and coil assembly transferred from magnet to magnet. The fringe flux measurements performed after site installation of the flowmeters, were made under different circumstances from those at Santa Susana. Thus, correlation of flux readings is impossible. As suggested in the test report, use of field fringe flux measurements for future magnet flux decay predictions will be valuable as soon as the correlating "magnet flux density-to-fringe flux" function is defined. While this relationship has not yet been completely developed, it has been assumed to be linear. Additional effort will be required to prove the assumption.

B. NOISE

There are several factors to be considered to explain flow signal noise from magnetic flowmeters. Some of these are turbulent flow, thermal EMF, magnet temperature correction, and end effect currents. All of these causals do exist at the HNPF. The Reynolds number of the primary and secondary liquid sodium coolant flow will be in the turbulent region at all operating power levels. Classical methods for eliminating or minimizing thermals require using lead wires of material similar to the pipe, and special welding techniques for electrode-to-pipe welds. On the whole, however, these methods do not completely solve the problem. Sanborn recordings of flow signals during the main heat transfer tests at 350 F showed a maximum peak-to-peak noise of 6.4 percent. Noise of such magnitude can be expected from the factors mentioned above. Thermal generation at operating temperatures, however, will be different from that observed at 350 F and its effect upon the noise is unpredictable. Flow signal noise may or may not develop to be a problem, depending on how the control and protective systems accept the signals. If these systems are sufficiently damped, the noise will be unimportant. Operating

experience to date indicates that such is the situation. Behavior of the plant control and protective systems will be observed during the post critical testing program.

C. CALIBRATION

Many pages have been devoted to the subject of flowmeter calibrations. Data supporting the fine accuracy of the one inch, two inch and six inch meters are adequately documented. These meters can be calibrated theoretically to within three or four percent. Such accuracy, except of the six inch moderator coolant flowmeter, is not really required at HNPF. The state of the primary and secondary main heat transfer loop 14 inch flowmeters, however, is not as clear cut. Initial calibration of these meters was performed at Santa Susana. Subsequently, "suggested" calibrations were published which were based on an assumed three percent flux decay from the time of original calibration, interpolation from testing temperatures to actual operating temperatures, and a considered full power pump speed (675 rpm). The plant control and protective systems were then set using these suggested calibrations. Figures 1160-1, -2 and -3 show pump speed vs flow and flowmeter output vs flow for secondary loops No. 1, No. 2 and No. 3, respectively. Curves are presented for the theoretically predicted calibration for 895 F, the suggested calibration for 895 F, and the calibration using secondary loop characteristics at 350 F and Byron-Jackson test data. The pump speed vs flow relationship incorporates Byron-Jackson pump data. Magnet flux information for the theoretical prediction was taken from results of measurements at Santa Susana on August 15, 1960. "Points A" on Figures 1160-2 and -3 were also obtained from Byron-Jackson data and secondary loop characteristics, but after 36 days of operation at 540 F. In general, flowmeter sensitivity decreases with increasing temperature, but the 350 F data and "points A" exhibit the opposite trend. This can be attributed to wetting. The three percent flux decay assumed for the suggested calibration is not apparent, and pump speed at design flow is indicated to

Figure 1160-1. Secondary Loop No. 1 Flow Meter Calibration Comparison, (Element FRC-102E)

Figure 1160-2. Secondary Loop No. 2 Flow Meter Calibration Comparison. (Element FRC-202E)

Figure 1160-3. Secondary Loop No. 3 Flow Meter Calibration Comparison. (Element FRC-302E)

be between 730 to 750 rpm, instead of 675 rpm. Therefore, these curves and points suggest that it will be necessary to recalibrate the flowmeters, and to reset the related control and protective system computers. The recalibration should be based upon heat balance calculations. For the secondary systems, this can be backed up by loop characteristics and Byron-Jackson pump data calibration similar to that shown in Figures 1160-1, -2 and -3 (available primary pump data are not complete enough to be used). All necessary information for the recalibration will be obtained as part of the post critical heat transfer tests (AI-PC-530).

Assuming that the losses from the heat exchangers are negligible, the solution of mass and energy balance equations yields the following expressions for secondary and primary flows with no blowdown from the evaporators:

$$W_{2i} = W_{3i} \frac{H_{3i}^2 - H_{3i}^1}{H_{2i}^4 - H_{2i}^3}$$

and

$$W_{1i} = W_{2i} \frac{H_{2i}^2 - H_{2i}^1}{H_{1i}^2 - H_{1i}^1}$$

where: W_{3i} = feedwater flow per loop (lb/hr)

W_{2i} = secondary Na flow per loop (lb/hr)

W_{1i} = primary Na flow per loop (lb/hr)

H_{3i}^2 = steam enthalpy after attemperation (Btu/lb)

H_{3i}^1 = feedwater enthalpy (Btu/lb)

H_{2i}^4 = secondary Na enthalpy at superheater inlet (Btu/lb)

H_{2i}^3 = secondary Na enthalpy at evaporator outlet (Btu/lb)

H_{2i}^2 = secondary Na enthalpy at IHX outlet (Btu/lb)

H_{2i}^1 = secondary Na enthalpy at IHX inlet (Btu/lb)

H_{1i}^2 = primary Na enthalpy at IHX inlet (Btu/lb)

H_{1i}^1 = primary Na enthalpy at IHX outlet (Btu/lb)

i = 1, 2, 3 for loops No. 1, No. 2 and No. 3

The plot of calculated flow rate vs corresponding flowmeter millivolt signal, then is the desired calibration. Accuracy of the calculated flows will be in direct relation to the accuracies of the feedwater flows and enthalpies involved in the equations. With feedwater flow determined to $\pm 1\%$, sodium enthalpies to $\pm 1/2\%$, and steam enthalpies to $\pm 0.1\%$, the calculated secondary sodium flow will be accurate to $\pm 2.8\%$ and the primary flow to $\pm 6.2\%$. To insure good accuracy, it is recommended that data for the calibration be obtained at high flow-high power conditions, and that flow and temperature readings be taken in millivolt signals, from as close to the sensing elements as possible.

Based upon the above discussion and past magnetic flowmeter experience, it can be concluded that the HNPF flowmeter installation will perform reliably with minor maintenance and annual or biennial recalibration to compensate for magnet flux decay.

HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
SODIUM PRESSURE INSTRUMENTATION

AI-P-1161

Test and Analysis

by

D. J. SOBO

Evaluation

by

S. L. PFAHLER

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

April 16, 1962

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HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
SODIUM PRESSURE INSTRUMENTATION, AI-P-1161

I. PURPOSE OF TEST

The purpose of this test was to verify proper functioning of all sodium pressure instrumentation components when connected as complete sodium pressure instruments.

II. DISCUSSION OF TEST PERFORMANCE

Actual performance of this test was initiated by the Atomics International Quality Control Department at the manufacturer's plant in Bridgeport, Connecticut. Prior to acceptance by Atomics International, and shipment to HNPF, the sodium pressure instruments were thoroughly inspected by an Atomics International Quality Control representative for conformance to Atomics International specification 7518-S72429, "Sodium Pressure Instruments". As part of this inspection, the pressure instruments for the heat transfer loops were calibrated at 800 F, 1000 F, and 1200 F. The pressure instruments for the primary fill tanks were calibrated at 400 F, 800 F, and 1000 F. At these various temperatures, all pressure transmitters were calibrated to $\pm 1\%$ of full scale accuracy and all vacuum transmitters were calibrated to $\pm 2\%$ of full scale accuracy. The test set-up used for calibration is shown in Figure 1161-1.

After arrival at the site, and subsequent installation, the pressure connections to sodium piping were X-rayed. Evaluation of the X-rays showed that four of the pressure transmitters had broken diaphragm seals. These four units were removed, returned to the manufacturer for repair, re-inspected, and were re-installed satisfactorily.

Upon arrival at the site, prior to filling the heat transfer system with sodium, and with all pressure sensing elements at ambient temperature and atmospheric pressure, the sodium pressure instruments were checked for proper interconnecting wiring and were zeroed. (After sodium fill, the pressure instruments will be re-zeroed by operations.) Some wiring errors existed between power supplies, transmitters, and readouts; this was corrected.

The high pressure alarms for the six sodium heat transfer loops were set using simulated pressure signals. The alarm units are relays which are actuated by the one-to-five milliamp signal developed by the pressure transmitter. The one-to-five ma signal corresponds to zero-to-full scale pressure. The simulated pressure signal was developed by using a precise current source (Foxboro Model 8131) to set the alarm relay. Proper operation of the annunciator drop associated with each alarm relay was verified.

It should be noted that the span adjustment, which was set at the factory, was not disturbed during these tests. Since the span and zero adjustments are independent, it is only necessary to re-zero the pressure transmitter after the sodium fill, and at operating temperatures.

Although it would have been more desirable to run an actual pressure calibration test on the pressure instruments after installation and at operating temperatures, this is not practicable due to the following reasons:

- 1) The pressure instrument diaphragm seals are welded to the sodium piping. Thus, it is impossible to pressurize the instrument without pressurizing the piping.

Figure 1161-1. TEST SETUP USED FOR CALIBRATING NaK FILLED SODIUM PRESSURE TRANSMITTERS

- 2) In the primary system, the reactor bellows will ~~take only~~ 15 psig. The throttle and block valves are not helium-tight, thus the pressure test cannot be conducted.
- 3) In the secondary system, the relief valves are set for 25 psig, thus the pressure test could not be conducted without making extensive piping modifications.
- 4) The pre-heat system will not take the piping to operating temperatures.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

Sodium pressure instrumentation calibration data are shown in Table I. As can be seen from the data, all pressure instruments were calibrated to within $\pm 1\%$ of full scale accuracy and all vacuum transmitters were calibrated to within $\pm 2\%$ of full scale accuracy. The data in Table I were extracted from Atomics International Quality Control Department inspection record "Sodium Pressure Instruments, HNPF, P. O. N941 GS-221083, Manning, Maxwell, and Moore".

All sodium high pressure alarms were set as noted in Table II. Proper operation of each annunciator drop, associated with its corresponding alarm relay, was verified.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

The following is a list of Test Completion Criteria taken from page XI (a)-85, NAA-SR-5700, Supplement 4, with remarks about the test to show compliance with the criteria.

- 1) Criterion: All sodium pressure transmitters calibrated to $\pm 1\%$ of full scale accuracy.
Remarks: Verified by Atomics International Quality Control Department.

TABLE I
CALIBRATION DATA FOR SODIUM PRESSURE INSTRUMENTS

Instrument Tag No.	Location	Range (psig)	Max Error % of full scale
PI-100T	Pri Na Pump No. 1 Suction	-15 to +30	+1.0
PI-200T	Pri Na Pump No. 2 Suction	-15 to +30	+0.7
PI-300T	Pri Na Pump No. 3 Suction	-15 to +30	-0.6
PI-101T	Pri Na Pump No. 1 Discharge	0 to 100	-0.6
PI-201T	Pri Na Pump No. 2 Discharge	0 to 100	+1.0
PI-301T	Pri Na Pump No. 3 Discharge	0 to 100	-0.6
PI-102T	Sec Na Pump No. 1 Discharge	0 to 100	-0.5
PI-202T	Sec Na Pump No. 2 Discharge	0 to 100	-0.8
PI-302T	Sec Na Pump No. 3 Discharge	0 to 100	-0.7
PI-103T	Sec Na Pump No. 1 Suction	0 to 50	-1.0
PI-203T	Sec Na Pump No. 2 Suction	0 to 50	-0.8
PI-303T	Sec Na Pump No. 3 Suction	0 to 50	+0.8
PI-109T	Pri Na Loop No. 1 Reactor Inlet	0 to 100	+0.9
PI-209T	Pri Na Loop No. 2 Reactor Inlet	0 to 100	+0.6
PI-309T	Pri Na Loop No. 3 Reactor Inlet	0 to 100	+0.8
PI-110T	Evaporator No. 1 Outlet	0 to 100	+0.4
PI-210T	Evaporator No. 2 Outlet	0 to 100	+0.6
PI-310T	Evaporator No. 3 Outlet	0 to 100	+0.9
PAH-100T	Pri Na Pump No. 1 Discharge	0 to 100	+0.7
PAH-200T	Pri Na Pump No. 2 Discharge	0 to 100	-0.7
PAH-300T	Pri Na Pump No. 3 Discharge	0 to 100	-1.0
PAH-101T	Evaporator No. 1 Inlet	0 to 100	+0.6
PAH-201T	Evaporator No. 2 Inlet	0 to 100	-0.5
PAH-301T	Evaporator No. 3 Inlet	0 to 100	-0.7
PI-400T	Pri Fill Tank No. 1	0 to 50	-1.0
PI-401T	Pri Fill Tank No. 2	0 to 50	-0.8
PI-402T	Pri Fill Tank No. 3	0 to 50	+0.8
PI-403T	Pri Fill Tank No. 4	0 to 50	-0.8
PI-404T	Pri Fill Tank No. 5	0 to 50	-0.9

TABLE II
ALARM SETTINGS

Alarm	Location	Setting (psig)
PAH-100	Pri Na Pump No. 1 Discharge	50
PAH-101	Evaporator No. 1 Na Inlet	75
PAH-200	Pri Na Pump No. 2 Discharge	50
PAH-201	Evaporator No. 2 Na Inlet	75
PAH-300	Pri Na Pump No. 3 Discharge	50
PAH-301	Evaporator No. 3 Na Inlet	75

- 2) Criterion: All sodium vacuum transmitters calibrated to $\pm 2\%$ of full scale accuracy.
Remarks: Verified by Atomics International Quality Control Department.
- 3) Criterion: Proper annunciation occurs on (simulated) high sodium pressures.
Remarks: Action verified (See Part III).

V. EVALUATION

The discovery of burst diaphragms, subsequent to installation of the pressure transducers, points out the desirability of more careful on site handling of the instruments. Since the transducers were intact prior to being shipped from the vendor's plant, as verified by AI representative, and the likelihood of damage during shipping is remote, it is felt that the diaphragms must have been damaged on site.

Satisfactory completion of the test procedure as outlined in AI-P-1161 together with calibration information as per AI specifications 7518-S72427 is felt to constitute an adequate check, as far as presently practicable, of the HNPF installed sodium pressure instrumentation. Additional confidence has been gained from accurate indications obtained during a recent pressurization of the No. 2 loop of the secondary sodium system.

HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST INTERIM REPORT
DRY EXCESS LOADING
AI-P-1163

Test and Analysis
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Evaluation
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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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ERRATA
FOR
HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
DRY EXCESS LOADING, AI-P-1163

<u>Page</u>	<u>Reads</u>	<u>Should read</u>
2 - Last paragraph first line	The addition	In addition
3 - Second paragraph second line	location (full)	location (full in)
5 - Last paragraph first line	488.983	489.017
6 - Last line above Table II	0.86% $\frac{\Delta k}{k}$	0.86% $\frac{\Delta k}{k}$ at 71 F
8 - Seventh line from top	31 F	313 F
Figure 1163-4	The $\square$ and the $\triangle$ should be interchanged on the legend.	

PREOPERATIONAL TEST INTERIM REPORT

DRY EXCESS LOADING, AI-P-1163

I. TEST PURPOSE

The purpose of this test was to obtain data to be used in determining the reactivity worth of the sodium in the reactor core, to obtain information on the dry temperature coefficient of reactivity and to train operators in reactor operation with a core loading of slightly greater than dry criticality.

II. TEST PERFORMANCE

The Dry Criticality Test, AI-P-1155, was completed prior to starting this test. The fuel was loaded in a specified sequence to maintain an approximate right cylinder core shape. A total of eleven fuel elements was loaded during the Dry Excess Loading. Table I shows the number and core positions of the individual loadings. Not more than 1% reactivity was loaded in one step. A plot of k_{eff} vs fuel loading, Fig. 1163-1, was maintained to assure this criterion. The excess reactivity worth of each fuel loading was determined from period measurements. The total excess reactivity loaded was determined by summing the incremental worths of each loading. Fuel loading was continued as long as adequate shutdown margin was assured. The minimum shutdown margin must be equal in magnitude to the core excess reactivity. A simple way to assure meeting this requirement was to not allow more fuel than that required to make

the reactor critical with all control rods withdrawn 50 percent.

TABLE I
FUEL LOADING SEQUENCE

Fuel Loading Increment	Number of Fuel Elements Loaded	Total Fuel Elements Loaded	Core Positions Loaded
1	1	32	C-41
2	2	34	C-51, C-63
3	2	36	C-39, C-55
4	2	38	C-67, C-43
5	2	40	C-52, C-64
6	2	42	C-40, C-54

Another precaution was that the core could not be loaded so that criticality could be achieved by full withdrawal of any one control rod with all other rods inserted. Since the central control rod has the greatest reactivity worth, it was fully withdrawn with all other control rods fully inserted after each incremental fuel loading to enforce this precaution. Inverse average multiplication vs functions of fuel loading were plotted (Fig. 1163-2, -3, -4) for this control rod configuration. By extrapolating these plots to $1/M = 0$, the fuel loading at which criticality would occur for this control rod configuration, could be determined.

The addition, inverse average multiplication vs number of fuel elements (N), Fig. 1163-2; vs square root of the number of fuel elements ($\sqrt{N}$), Fig. 1163-3; vs a factor proportional to radial buckling (B_r^2), Fig. 1163-4, was plotted after each

incremental fuel loading with all control rods fully inserted and with all control rods withdrawn 50%. From these plots, expected criticality was extrapolated for these control rod configurations.

A typical example of how the reactivity worth of each fuel increment was determined follows. With 34 fuel elements in the core, the control rod positions at criticality were gangs 1, 2, 3, and 4 fully withdrawn and gang 5 at 73.4 in. (X_1). With 36 fuel elements, the control rod positions at criticality were gangs 1, 2, 3, and 4 fully withdrawn and gang 5 at 57.75 in. (X_2). Thus, the reactivity change resulting from the addition of the 35th and 36th 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 two fuel elements it was only necessary to calibrate ΔX .

The aforementioned critical positions were with the source in its normal location (full). As pointed out in the report for the HNPF Dry Criticality Test, AI-P-1155, it was necessary to separate the source in order to minimize nuclear heating and reduce the critical flux level, thereby, giving enough range on the incore fission chambers to perform precise period measurements. With the source withdrawn 29 in. the desired range was obtained. In this particular example, 36 fuel elements, the critical rod positions with the source at 29 in. were gangs 1, 2, 3 and 4 fully withdrawn, and gang 5 at 57.45 in. Moving the source from full in to 29 inches resulted in a positive change equivalent to about 0.3 of an inch on gang 5 at this particular time. With the neutron flux at equilibrium or below point B (Point B is the minimum flux level at which the effect of the source neutrons is negligible), gang 5 was withdrawn to 57.75 in. and a long period, approximately 701 sec obtained. Next, gang 5 was inserted and the flux level dropped below point B again. Then gang 5 was withdrawn to 60.2 in. which produced a period

of approximately 48 sec. The reactivity associated with a 701 sec period is approximately 1.7 cents and with a 48 sec period 16.4 cents.

Thus from these two period measurements the reactivity worth of gang 5 from 57.75 to 60.2 in. was determined to be 14.7 cents (16.4 cents minus 1.7 cents). Now the flux level was brought to approximately point B again with gang 5 about one half in. below its last position or about 59.7 in. To accomplish this, gang 4 had to be inserted to 131 in. Now gang 5 was withdrawn to 60.2 in. again and a long period obtained. After this long period was measured the flux was dropped below point B again, and a short period obtained by pulling gang 5 to approximately 62.7 in. This technique was continued until the desired amount of gang 5 (ΔX) was calibrated and thus, 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 was reduced to reactivities by the "Ratio Method," see HNPF Operations Manual NAA-SR-6500 Vol. IV, section D.

After the reactivity of each fuel increment was determined, rod drop measurements were performed to estimate the total worth of the control rods. The reactor was run at a constant power level at criticality for a short time. Equilibrium was assured by watching the printouts for equal counts per unit counting interval. Total counts for a 15-sec interval were obtained at this power level. Right after a count had been printed, all rods were scrammed manually. The counter recycled (this was approximately the same time as was required for rods to drop) and counted again for 15 sec. The ratio of total counts before scramming, N_0 , and total counts after scramming, N_1 , is defined as R.

$$R = \frac{N_0}{N_1}$$

The change in k_{eff} (Δk) is obtained by using:

$$\frac{\Delta k}{\beta_{\text{eff}}} = C(R-1); \text{ or } \Delta k = \beta_{\text{eff}} (R-1)C$$

Where C is the Schulz constant, and

$$\beta_{\text{eff}} = 0.0069 \text{ for the } 3.6 \text{ } ^\text{w}/_0 \text{ U235} - 10 \text{ } ^\text{w}/_0 \text{ Mo fuel.}$$

The Schulz constant (C) was determined experimentally to be 0.673. The "book value" of C is from 0.367 to 0.417.

After 42 fuel elements were loaded and criticality attained, the loading of fuel was halted. The reactivity worth of the 41st and 42nd fuel elements was determined. Then seven (7) fuel elements were unloaded and the reactor heaters installed. To date the core has been preheated with 35 fuel elements from an ambient temperature of 71 F to 313 F. The reactor was brought critical approximately once each day and the core temperature recorded.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

The excess reactivity data taken during the Dry Excess Loading Test are summarized in Table II. A plot of the experimental data is presented in Fig. 1163-5. Also plotted are calculated $k_{\text{eff},s}$, (a) accounting for the source and, (b) not accounting for the source.

The total worth of the central control rod was 488.983 cents or 3.389 percent. A calibration plot of the central control rod is presented in Fig. 1163-6. After 42 fuel elements were loaded, criticality occurred with gangs 1, 2, 3, and 4 at 78 in. and gang 5 at 72.1 in. Since this compromised the minimum shutdown margin, the loading of fuel was halted after determining the reactivity worth of the 41st and 42nd fuel elements. Then seven fuel elements were unloaded

and the reactor heaters installed. With these heaters in place in the 35 element core, the excess reactivity was approximately 124 cents or $0.86\% \frac{\Delta k}{k}$.

TABLE II
SUMMARY OF EXCESS REACTIVITY DATA

Number of Fuel Elements (N)	Critical Rod Position (Gangs)	Incremental Worth (Cents)	Total Worth (Cents)	Reactivity (Percent)	K _{eff}
31	1,2,3,4 out, 5 at 123.9	29.062	29.062	.201	1.002
32	1,2,3,4 out, 5 at 95.4	69.391	98.423	.682	1.0068
34	1,2,3,4 out, 5 at 73.4	100.277	198.700	1.377	1.0139
36	1,2,3,4 out, 5 at 57.75	91.827	290.527	2.013	1.0205
38	1,2,3,4 out, 5 at 39.0	107.381	397.908	2.758	1.0283
40	1,2,3 out, 4 at 140.0, 5 at 23.1	69.435	467.343	3.239	1.0334
42	1,2,3,out,4 at 118.2, 5 at 1.9	74.668	542.011	3.756	1.0390

From Fig. 1163-6 it can be seen that the worth of the central control rod is asymmetrical. The first 50% of rod withdrawal is approximately 65% of the reactivity worth of the central control rod. When withdrawn 50%, only 35% of the reactivity worth of this rod remains in the core. The excess associated with 42 fuel elements is approximately 3.8%

and criticality occurred with the control rods withdrawn approximately 50%. If one assumes that all control rods had the same asymmetrical worth curve as the central control rod and, that only 35% of the reactivity worth of the control rods remained in the core upon 50% withdrawal, then it can be concluded that 35% of the reactivity worth of the rods equals $3.8\% \frac{\Delta k}{k}$ or the total worth of all control rods is $10.9\% \frac{\Delta k}{k}$.

The worth of gang 4 was calibrated from 118.2 to 156.0 inches and found to be 53¢. The same portion of gang 5 was found to be 37¢. If it is assumed that this ratio holds true over the entire length of gang 4, then gang 4 was worth 1.43 times gang 5. If it is also assumed that gang 3 equals gang 4 and that gangs 1 and 2 are worth nothing, then the total worth of the control rods is approximately $13.1\% \frac{\Delta k}{k}$.

From rod drop measurements and using the book value^k of 0.417 for Schulz's constant, the total reactivity worth of the control rods is approximately $14.5\% \frac{\Delta k}{k}$. If the experimental value of 0.673 for Schulz constant is used, the worth of the rods is approximately $20\% \frac{\Delta k}{k}$.

Very little can be concluded from these rod drop measurements. The outer 12 control rods of gangs 1 and 2 have little reactivity worth in the active, small, dry core but have a large effect on the number of neutrons which leak to the fission chambers. The control rods of gangs 1 and 2 shadow the detecting chambers so that a larger reduction in neutron flux occurs at the chambers than occurs in the active core. This in turn indicates that the apparent reactivity worth of the control rods is greater than is the case. The fission chambers are so located, radially, that a "representative sampling of the flux" can not be obtained when all control rods are dropped in the small core. Period data taken during this test are tabulated in Appendix A. As already mentioned

two periods (short and long) were required to calibrate each increment (ΔX) of the control rod. All other control rods were fully withdrawn (156.0 in.) during the period measurements except those where a position is specified (Appendix A). To date the core has been preheated (with 35 fuel elements) from an ambient temperature of 71 F to 313 F. A plot of reactivity change vs core temperature is presented in Fig. 1163-7.

The isothermal temperature coefficient was found to be positive throughout the range from 71 F to 313 F. In the dry core, only the fuel and moderator temperature coefficient can be considered. Apparently, the positive effect of the moderator is greater than the negative effect of the fuel in this temperature range. Table III lists the reactivity changes as a function of temperature increase.

TABLE III

REACTIVITY CHANGE VS TEMPERATURE

Average Core Temperature	Change in Temperature from Ambient (F)	Total Reactivity Change (Cents)
71	0	0
120	49	23
161	90	39
222	151	59
248	177	65
273	202	72
296	225	79
313	242	86

With 35 fuel elements in the core and at approximately 320 F, the 6 control rods of gangs 3 and 4 and gangs 1 and 2 have been intercalibrated against gang 5, the central rod. With the source separated 29 in., the control rod or gang to be intercalibrated fully inserted, and the remaining control rods except gang 5 fully withdrawn, the core was made critical with gang 5. This was done for each of the control rods of gangs 3 and 4 and for gang 1 and gang 2. With all rods fully withdrawn, criticality occurred with gang 5 at 75.025 in. Thus, when the rod, or gang, to be calibrated was fully inserted, criticality occurred with gang 5 at a new position. The reactivity worth of gang 5 from 75.025 in. to the new position is equal to the worth of the rod or gang to be calibrated as determined from data plotted in Fig. 1163-6. A summary of the intercalibration data follows in Table IV.

The ratio of reactivity worth of control rod 6 to gang 5 is .376. The first 38 in. of control rod 6 have been calibrated independently by period measurements and are worth 31.68 cents. The same 38 in. of gang 5 from the integral worth curve were worth 86.08 cents. This ratio of control rod 6 to gang 5 is .368, which indicates that the total reactivity worths listed are accurate to about 2%, not to be confused with 2% Δk over k .

Rod drop measurements have been performed on control rod 6 and gang 5. From these measurements the ratio of the reactivity worth of control rod 6 to gang 5 is .4015. This compares favorably with previously stated ratios. Assuming that gang 5 is worth 3.389% Δk over k , then Schultz constant is .4523 and the worth of all rods at the 35 fuel element loading is 12.1% Δk over k .

TABLE IV

CONTROL ROD REACTIVITY WORTHS

Control Rod or Gang Number	Core Position	Gang 5 Critical Position (Inches withdrawn)	Total Worth Cents	Total Worth Percent Delta k over k
CR6	C16	145.00	184	1.275
CR9	C24	141.10	182	1.261
CR12	C32	125.70	166	1.151
CR2	C20	139.50	181	1.254
CR3	C28	128.60	170	1.178
CR4	C36	127.70	169	1.171
Gang 2	C38, C44, C50, C56, C62, C68	105.10	124	.859
Gang 1	C78, C86, C94, C102, C110, C118	81.95	36	.249
Gang 5	C1	-----	489	3.389
Total worth of all rods		-----	1701	11.787

IV. TEST COMPLETION CRITERIA

The criteria as listed in NAA-SR-5700, Supplement 4 of the HNPF Safeguards Supplement Report are as follows:

- a) Fuel is loaded into the dry core as limited by Precaution "k". Precaution "k" states that the minimum shutdown margin is equal in magnitude to the core excess reactivity. A simple way to assure meeting this requirement is to not allow more fuel than that required to make the reactor critical with all control rods 50 percent withdrawn. Due to the slight asymmetry in control rod worth vs position, the lower 50 percent of rod travel is worth more than 50 percent of total rod worth. This asymmetry will be experimentally verified. In any event, not more than 75 fuel elements are loaded.
- b) From period measurements the total excess reactivity in the dry core is known.
- c) A plot of fuel loading vs excess reactivity for the dry core has been completed.
- d) The dry temperature coefficient of reactivity has been determined with the reactor core loaded to one fuel element above dry criticality, over the temperature range from ambient to 350 F. The data necessary to make this determination is obtained during the primary system helium purge and pre-heat. These operations afford valuable operator training in starting up and shutting down the reactor.

The above criteria a through c have been satisfied by the following:

- a) Forty two elements were loaded and criticality achieved with gangs 1, 2, 3 and 4 withdrawn 50% (78.0 inches) and gang 5 withdrawn 72.1 inches.
- b) The total excess reactivity in the dry core is known and is presented in Table II.
- c) A plot of fuel loading vs excess reactivity for the dry core has been completed (Figures 1163-1 and 1163-5).

Criterion "d" quoted above was modified by Technical Specifications allowing approximately 1% excess reactivity instead of one fuel element excess. Criterion "d" (as modified) has been partially met to date, since the core temperature has been raised to only 313 F. (Figures 1163-7 and Table III present data obtained to date).

V. EVALUATION

This report of the dry excess loading test is an interim report; and, as such, the evaluation can cover only the partial presentation of results which is contained in the previous section.

Thus far the dry excess loading test has been conducted in a very safe manner. Each of the loading increments added less than one percent reactivity to the core. The excess reactivity with the maximum loading of forty two elements was approximately 35% of the reactivity worth of the control rods.

Some of the expected results of this test have not been achieved. First a fuel loading equal to, or greater than, the expected wet critical loading of fifty elements was not achieved. Secondly, with the dry excess loading limited to forty two fuel elements it is impractical to

extrapolate the plot of inverse average multiplication, with rods down, to zero. Failure to obtain these expected results will require further reliance on theoretical calculations for determination of sodium worth.

However, the excess reactivity data obtained during the dry excess loading test correlate closely with previous calculations, thereby increasing confidence in method and assumption adopted for calculating this core.

The control rod worth measurements made during this test indicate the shut-down margins during the test were always within the limitations as specified in the technical specifications. No conclusions can be drawn at this time as to the adequacy of these control rods in an operationally loaded core.

The resolution of this question must await further testing in a more fully loaded core.

The isothermal temperature coefficient of the dry core was positive from 71 F to 313 F as anticipated. The data, as presented, do not indicate how control rod worth may have varied during rise in temperature. An indication of control rod worth variation will be made before this test is terminated.

Figure 1163-1. Percent Excess Reactivity vs Number of Fuel Elements

Figure 1163-2. Inverse Average Multiplication vs Number of Fuel Elements

Figure 1163-3. Inverse Average Multiplication vs Square Root of Number of Fuel Elements

Figure 1163-4. Inverse Average Multiplication vs Factor Proportional to Radial Buckling

Figure 1163-5. K_{eff} vs Number of Fuel Elements

Figure 1163-6. Integral Reactivity Worth of Central Rod vs Rod Position

Figure 1163-7. Reactivity Change vs Temperature

VI. A P P E N D I X A

PERIOD DATA
(Gang 5 Movement)

Number of Fuel Elements	Gang 5 (Inches)*	Gang 4 (Inches)	Period T (Sec)	ρ ** (Cents)	$\Delta \rho$ *** (Cents)	Total $\Delta \rho$ (Cents)
31	156.0	124.4	271.78	4.139	1.477	489.017
31	148.0	124.4	440.76	2.662		
31	148.0	126.0	172.71	6.123	3.007	487.540
31	141.0	126.0	370.16	3.116		
31	141.0	129.0	108.71	8.994	4.594	484.533
31	135.0	129.0	253.68	4.400		
31	135.0	133.5	78.43	11.551	7.318	479.939
31	129.0	133.5	264.96	4.233		
31	129.0	137.1	113.21	8.709	4.685	472.621
31	126.0	137.1	280.48	4.024		
31	126.0	156.0	61.86	13.757	13.757	467.936
31	119.3	156.0	∞	00.000		
32	119.3	107.0	105.74	9.176	6.246	454.179
32	115.2	107.0	397.39	2.930		
32	115.2	110.7	55.05	14.928	11.091	447.933
32	109.7	110.7	294.83	3.827		
32	109.7	115.0	52.82	15.364	10.146	436.842
32	105.7	115.0	208.87	5.218		
32	105.7	121.5	39.26	18.698	12.182	426.696
32	101.7	121.5	161.16	6.516		
32	101.7	128.9	43.92	17.396	12.686	414.514
32	98.2	128.9	234.81	4.760		
32	98.2	137.7	66.26	13.080	8.621	401.828

* Denotes inches withdrawn

** Reactivity

*** Change in reactivity

PERIOD DATA
(Gang 5 Movement Cont'd)

Number of Fuel Elements	Gang 5 (Inches)*	Gang 4 (Inches)	Period T (Sec)	ρ ** (Cents)	$\Delta \rho$ *** (Cents)	Total $\Delta \rho$ (Cents)
32	96.0	137.7	249.92	4.459		
32	96.0	156.0	82.40	11.146	2.569	393.207
32	95.4	156.0	115.17	8.577		
34	95.4	98.3	57.25	14.525	11.323	390.638
34	92.0	98.3	360.78	3.202		
34	92.0	101.0	68.84	12.726	8.224	379.315
34	90.0	101.0	247.58	4.502		
34	90.0	103.0	75.43	11.896	5.824	371.091
34	88.6	103.0	174.28	6.072		
34	88.6	106.0	56.31	14.694	10.637	365.267
34	86.0	106.0	277.76	4.057		
34	86.0	110.5	53.87	15.154	11.113	354.630
34	83.5	110.5	279.02	4.041		
34	83.5	116.1	51.57	15.608	12.007	343.517
34	81.1	116.1	317.31	3.601		
34	81.1	123.0	55.49	14.835	10.696	331.510
34	79.0	123.0	271.08	4.139		
34	79.0	133.2	51.21	15.691	13.365	320.814
34	76.5	133.2	509.22	2.326		
34	76.5	156.0	71.42	12.401	12.401	307.449
34	74.3	156.0	∞	0.000		
36	74.3	96.4	89.36	10.477	8.602	295.048
36	72.7	96.4	639.71	1.875		

* Denotes inches withdrawn

** Reactivity

*** Change in Reactivity

PERIOD DATA
(Gang 5 Movement Cont'd)

Number of Fuel Elements	Gang 5 (Inches)*	Gang 4 (Inches)	Period T (Sec)	ρ ** (Cents)	$\Delta\rho$ *** (Cents)	Total $\Delta\rho$ (Cents)
36	72.7	102.0	36.11	19.735	13.859	286.446
36	70.2	102.0	182.15	5.876		
36	70.2	107.0	37.46	19.266	14.273	272.587
36	67.7	107.0	219.74	4.993		
36	67.7	112.3	44.17	17.311	14.985	258.314
36	65.2	112.3	508.23	2.326		
36	65.2	121.5	37.20	19.369	14.925	243.329
36	62.7	121.5	250.32	4.444		
36	62.7	131.0	44.25	17.311	15.177	228.404
36	60.2	131.0	558.00	2.134		
36	60.2	156.0	48.03	16.411	14.693	213.221
36	57.8	156.0	701.31	1.718		
38	57.8	101.0	46.44	16.763	10.943	198.534
38	56.0	101.0	183.87	5.820		
38	56.0	104.7	43.49	17.501	11.852	187.591
38	54.2	104.7	189.52	5.649		
38	54.2	108.7	47.52	16.510	13.642	175.739
38	52.0	108.7	406.30	3.868		
38	52.0	114.6	44.17	17.324	14.438	162.097
38	49.6	114.6	403.29	2.886		
38	49.6	121.2	46.22	16.816	13.593	147.659
38	47.3	121.2	358.10	3.223		
38	47.3	129.5	44.52	17.242	14.236	134.066
38	44.7	129.5	386.37	3.006		

* Denotes inches withdrawn

** Reactivity

*** Change in reactivity

PERIOD DATA

(Gang 5 Movement Cont'd)

Number of Fuel Elements	Gang 5 (Inches)*	Gang 4 (Inches)	Period T (Sec)	ρ ** (Cents)	$\Delta\rho$ *** (Cents)	Total $\Delta\rho$ (Cents)
38	44.7	140.0	52.03	15.524	11.260	119.830
38	42.5	140.0	262.73	4.264		
38	42.5	156.0	71.88	12.336	12.336	108.570
38	40.0	156.0	∞	00.000		
40	40.0	115.7	37.63	19.232	16.094	96.234
40	36.8	115.7	367.35	3.138		
40	36.8	122.5	36.50	19.604	16.620	80.140
40	33.0	122.5	389.54	2.984		
40	33.0	130.5	38.90	18.843	15.926	63.520
40	28.6	130.5	399.12	2.917		
40	28.6	140.0	46.27	16.798	15.562	47.594
40	23.1	140.0	988.55	1.236		
42	23.1	115.6	30.87	21.754	16.570	32.032
42	16.0	115.6	210.74	5.184		
42	16.0	118.1	66.76	13.020	9.992	15.462
42	9.0	118.1	383.28	3.028		
42	9.0	119.9	111.35	8.837	5.470	5.470
42	0.0	119.9	340.29	3.367		

* Denotes inches withdrawn

** Reactivity

*** Change in reactivity

PERIOD DATA

(Gang 4 Movement)

Number of Fuel Elements	Gang 4 (Inches)*	Gang 3 (Inches)	Gang 5 (Inches)	Period T (Sec)	ρ ** (Cents)	$\Delta\rho$ *** (Cents)
40	156.0	156.0	20.0	215.81	5.078	3.394
40	148.0	156.0	20.0	713.96	1.684	
40	148.0	156.0	24.6	71.01	12.433	6.920
40	140.0	156.0	24.6	195.83	5.513	
42	140.0	123.0	0.0	98.34	9.716	4.699
42	135.5	123.0	0.0	218.52	5.017	
42	135.5	125.5	0.0	88.78	10.518	8.904
42	129.5	125.5	0.0	750.17	1.614	
42	129.5	132.5	0.0	66.34	13.080	9.517
42	125.0	132.5	0.0	320.81	3.563	
42	125.0	149.0	0.0	49.88	15.977	10.495
42	121.2	149.0	0.0	197.11	5.482	
42	121.2	156.0	12.0	49.10	16.124	9.021
42	118.2	156.0	12.0	145.85	7.103	

* Denotes inches withdrawn

** Reactivity

*** Change in reactivity

NOTE: To convert reactivity in cents (ρ_ϕ) to reactivity in terms of % $\frac{\Delta k}{k}$ ($\rho_{\Delta k}$) use the following relationship:

$$\rho_\phi \times (0.00693) = \rho_{\Delta k}$$

SUPPLEMENT I
TO
HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST INTERIM REPORT
DRY EXCESS LOADING, AI-P-1163

The interim report for Dry Excess Loading test was published prior to completion of preheating the reactor core and therefore did not contain all the isothermal temperature coefficient of reactivity data. In this supplement, the additional temperature coefficient and rod calibration data are reported. This supplement also contains a complete evaluation of the Dry Excess Loading test. The interim report, combined with this supplement, comprises the HNPF Preoperational Test Completion report for the Dry Excess Loading, AI-P-1163, test.

ANALYSIS AND PRESENTATION OF ADDITIONAL TEST RESULTS

The core with 35 fuel elements was preheated from an ambient temperature at 71 F to 369 F. The reactor was brought critical approximately once each day and the control rod positions at criticality and the core temperatures were recorded.

The isothermal temperature coefficient of reactivity was found to be positive throughout the range from 71 F to 369 F as expected. In the dry core, only the fuel and moderator temperature coefficients of reactivity can be considered. Apparently, the positive effect of the moderator is greater than the negative effect of the fuel in this temperature range. In Table III-a the reactivity changes as a function of core temperature increase are listed. In this table, two reactivities are listed,

(1) from a rod calibration at 71 F (See Fig. 1163-6 of Interim Report), and (2) from a rod calibration at approximately 365 F.

The position of gang 5 at criticality changed from 98.9 in. at 71 F to 78.05 in. at 347 F. From the calibration of this control rod at 71 F (Fig. 1163-6), this section of rod had a reactivity worth of 90 cents. From a recalibration of this section of rod at approximately 365 F, the reactivity worth was 88.6 cents. This indicates that the control rod reactivity worth has changed little, if any, from ambient to 365 F.

Test completion criterion "d" (modified as indicated in the Interim Report) has been met. In Table III-a the data obtained to satisfy this criterion are presented.

TABLE III-a

REACTIVITY CHANGE VS TEMPERATURE

Avg Core Temperature (F)	ΔT From Ambient (F)	$\Delta \rho$ Cents 71 F Calibration	$\Delta \rho$ Cents 365 F Calibration
71	0	0	0.0
120	49	23	20.1
161	90	39	34.1
222	151	59	54.8
248	177	65	60.7
273	202	72	68.6
296	225	79	76.3
313	242	86	84.4
329	258	89	86.3
347	276	90	88.6
369	298	95	93.5 *

* Estimated

EVALUATION

The dry, zero power tests were conducted in a very safe manner. Each of the fuel loading increments were within the limitation of the technical specifications.

Certain of the specific objectives of the dry, zero power tests have been achieved; a dry critical loading was determined; the central control rod was calibrated; the core was loaded to a maximum of 42 fuel elements; and the change in reactivity with isothermal temperature of the core has been determined for a 35 element core for a temperature range of 71 F to 369 F.

However, the premature termination of the dry excess loading, to comply with the test procedure restriction that the reactor would not be made critical with the rods less than 50% withdrawn, does not allow an experimental determination of sodium worth at any loading of fuel. This was the principal objective of the dry, zero power tests. Although close correlation between calculated and experimental reactivities during the wet excess tests should provide additional confidence in nuclear calculations, additional tests during wet excess are recommended to assure maintenance of an adequate shutdown margin at all times during the wet excess loading.

It was originally planned to utilize the experimental data obtained to correct pre-critical physics calculations. The large difference between calculations and experimental data has been explained by the presence of the neutron source in the central region of the core (HNPF Preoperational Test Completion Report, Dry Criticality AI-P-1155). This has not been verified experimentally.

Control rod withdrawal speed tests were made as part of HNPF Preoperational Test AI-P-1148, Control Rod Operability. These rod speed tests and the preliminary control rod worth measurements made during the dry excess loading

indicate the maximum reactivity insertion rate may exceed the 0.03% ($\Delta k/k$)/second limitation in the technical specifications. Additional tests are recommended to assure compliance with technical specifications.

FIG. II63-7a PLOTS OF REACTIVITY CHANGE VS. TEMPERATURE

HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
REACTOR HEATERS

AI-P-1164

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Evaluation
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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 14, 1962

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HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
REACTOR HEATERS, AI-P-1164

I. PURPOSE OF THE TEST

The purpose of Section 1 of this test was to verify the proper rating, location, circuiting, switching, relaying, and operation of the reactor heaters. The temperature readout and recording instrumentation for thermocouples located throughout the reactor were also to be connected and verified.

Section 2 of this test was to safely and properly preheat the reactor core and adjacent primary piping and primary valves to 350 F.

II. DISCUSSION OF TEST PERFORMANCE

A. ELECTRICAL SYSTEM CHECKOUT (Section 1)

1. Instrumentation

The indicating and recording instruments were calibrated and the band widths adjusted as required. The main patch panel and preheat patch panel were circuited to provide the necessary read-out and print-out data. Thermocouples were connected to the instrument setpoint terminals.

2. Power Supply

All power circuitry to and from the control center and reactor heater pits (three) was checked and energized. Considerable rewiring was necessary to

achieve proper connections.

3. Switching and Circuit Control

The manual switches and automatic relaying systems for the power cabling were operated.

4. Reactor Heaters

The incore heaters were checked for continuity, and their resistances measured and numbered by class and core location, prior to insertion of the heaters into the reactor.

5. Portable Heater Cables

The portable heater cables were tested for continuity and identified for pit and core heater location. They were then connected for operation. Male and female socket inserts were available at the cable-heater connection, but the threaded portion of both connectors was on male-type connecting rings, making it impossible to make the connection. One of the rings was removed, and the connectors were plugged together and held with insulating tape.

6. Power Consumption

The reactor incore and bottom heaters were energized and the voltage and current for each heater was recorded. Heater-energized test periods did not exceed 15 min. One of the bottom heater breakers opened a few times upon energizing the heater, but the problem cleared up promptly, and the breaker stayed closed. One possible cause for the breaker operation could be moisture condensation accumulation of the preceding period.

B. HEATING THE REACTOR (Section 2)

Initial cold-start readings were recorded for the reactor, adjacent guard piping, and primary valves, after verifying controller setpoints of the instrumentation system.

The reactor atmosphere was noted to contain a minimum of 95% helium and the Helium System was in service, maintaining this minimum concentration.

Reactor preheating began on February 12th from an ambient reactor temperature of about 72 F and a pipe and valve temperature (previously preheated by others) of about 85 F. All incore reactor heaters and one circuit (C-219) of the containment vessel bottom heaters were energized.

The reactor and the piping continued to heat for several days at a rate of about 1 F per hour. The heating was controlled, and the relative temperature limitations were maintained, by the automatic control instrumentation. Many additional indicated temperatures were recorded and a vertical temperature traverse of the reactor vessel wall was frequently recorded and plotted.

Preheating of other components of the main heat transfer system forced a halt to the temperature rise of the reactor on February 20th. During the following days of steady state temperature conditions in the reactor, the other components were heated to temperatures consistent with the reactor and piping. On February 26th, the heaters were turned off for a day to determine the cooling trend of the reactor. On March 10th, the heater controls were again set to raise the temperature of the reactor. On March 15th, the reactor vessel attained the desired vessel wall temperature of 33 F. At this time, the core of the reactor varied

from 410 F near the outer moderator ring to 365 F at the center. After setpoint readjustment and a day of temperature stabilization, the preheated reactor was turned over to Operations.

III. ANALYSIS AND PRESENTATION OF RESULTS

A. ELECTRICAL SYSTEM CHECKOUT (Section 1)

The detailed planning and care used in the performance of Section 1 of this test greatly aided in the smooth operation of the reactor preheating. The numerous rewiring of power circuits and verification of thermocouple read-outs provided an appreciated reliability in the temperature measuring and control system.

B. HEATING THE REACTOR (Section 2)

The critical nature of the preheat of the reactor precluded any casual appraisal of tending the heater test. Volumes of data were taken. Recording periods ranged from every 15 min at first, to once a shift or once a day towards the end of the test. Critical data, however, were taken at least once an hour until steady state conditions were achieved.

The reactor preheat test was uneventful through its entirety. The inner and outer moderator ring of heaters were ON continuously until the setpoints of 330 F were reached. The 33 moderator heaters provided about 42.8 kw of power. It was not necessary to use any of the eight individual switches for sectionalized heating.

Initially, the piping and valves adjacent to the reactor were heated by manual operation of the switches. This did not provide adequate temperature control. Automatic switching, utilizing setpoints, provided a

much better control of the temperatures of the piping and valves. As a result, the piping and valves were preheated to a required temperature at rates comparable to the reactor preheat schedule.

Figure 1164-1 shows the temperature rate of rise for three points in a horizontal plane through the vertical midpoint of the reactor core. Curve A is a plot of a temperature point on the reactor vessel. Curve B is a plot of temperature in the middle of the core. Curve C shows a temperature in the outer ring of heaters in a channel between two heaters. At the end of 200 hrs, the following temperatures were achieved:

- 1) center of reactor 315 F
- 2) outer heater ring at center 337 F
- 3) reactor vessel wall at a core 250 F

Note the apparent heat transfer "lag" effect at the beginning of the test, and slower rate of rise the last 50 hrs, as shown in Figure 1164-1. Recall that there was a steady state condition for several days, to wait for component heating to catch up temperature-wise with the reactor.

Figure 1164-1 also shows the continuation of the heating test with final core temperatures pendent on the reactor vessel wall temperature of 330 F. Again, an apparent "lag effect" of heat transfer was evident at the start.

Moderator element temperature limitations required the temperature at the top of the core, at the outer ring of heaters, to be always above the temperature at the top of the core at the center of the reactor. Also, the maximum allowable temperature differential between the top of the core, at the outer ring of heaters, and the reactor vessel at the top of the core was 100 F. Neither of these limitations presented any problems.

Figure 1164-1. Reactor Vessel Wall, Center Core and Outer Heater Ring Temperature

The six upper plenum heaters, located above the outer ring of heaters, provided 29.4 kw of power. An upper plenum, reactor vessel wall temperature setpoint, maintained at 10 degrees above the reactor core centerline temperature, controlled the two circuits of the upper plenum heaters. With a band width of ± 5 F, the proper upper plenum temperature was easily maintained without manual control of the heaters. It should be noted that certain reactor vessel wall and bellows temperatures responded quite quickly to the output of the upper plenum heaters. The reactor loading face shield cooling system was used to maintain the loading face shield at less than 150 F.

The 15 lower plenum heaters, located below the core in both the inner and outer ring of heaters, provided 9.6 kw of power. Perhaps chimney and other circulation effects were greater than anticipated, but the lower plenum heaters seemed to lag behind the outer heaters in maintaining lower plenum area temperatures. The setpoint, as measured on the lower plenum reactor vessel wall, was maintained at +10 degrees with respect to the reactor vessel wall temperature.

To aid in maintaining the proper lower plenum temperature at a higher level, the reactor bottom heaters were manually turned on, when possible, even though the setpoint conditions did not call for additional heat energy. The setpoint was maintained 10 to 15 degrees above the indicated temperature of the lower plenum heaters and a permitted 35 degrees differential was easily maintained. The 30 kw provided by the bottom heaters (one circuit only, with a dry reactor) produced a temperature surge immediately noticeable on the recorders.

The containment vessel bottom temperature limitation of 50 F ΔT , between the outer edge and the average of four central temperatures, was rather critical and had to be observed occasionally; as often as several times an hour. The effect of the bottom heaters was quickly apparent on the TRC-P-10 recorder charts; both when the heaters were energized and again when they were de-energized.

The containment vessel bottom heater circuits malfunctioned on two occasions. During the steady state waiting period, about half way through the test, the bottom heaters were turned off by the automatic controller. Several days later, when they were required, the circuit breaker of the circuit used originally (C-219) would not hold, and so circuit C-220 was used. At a later date C-219 again operated satisfactorily and C-220 was no longer utilized. There was no explanation for this behavior. Perhaps moisture condensation again occurred. On another occasion, C-219 was found in a tripped position. It was reset and closed. There was no explanation for the tripping of this circuit. There was also no indication in the control room that this circuit was tripped.

The results of the cooling trend test of the reactor, on February 26th, indicated that the core and the wall temperature might lose 10 degrees in 24 hrs, and the area of the outer ring of heaters might lose 20 degrees in 24 hrs.

In summary, because of a well designed heater and automatic control system, and with the aid of close surveillance, the reactor was preheated electrically from an ambient temperature of 75 F to the desired 330-350 F condition, without incident.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

The following is a list of the Test Completion Criteria taken from pages XI (a)-91 and XI (a)-93, NAA-SR-5700, Supplement 4, with remarks about the test to show compliance with the criteria.

A. ELECTRICAL SYSTEM CHECKOUT (Section 1)

- a) Criterion: Heater control circuits operate properly.
Remarks: The heater control circuits were tested and operated properly.
- b) Criterion: Reactor heaters operate properly.
Remarks: The reactor heaters were energized and operated properly.
- c) Criterion: Associated thermocouples and their recorder-controller and indicating controllers are operable and are wired correctly.
Remarks: The associated thermocouples and circuitry were tested and checked for continuity. The recorder-controllers and indicating controllers were tested and calibrated and are operable. The circuitry to the controllers is correct and the controllers and indicators are operating and functioning properly.

B. HEATING THE REACTOR (Section 2)

- a) Criterion: The reactor core has been heated to 350 F without exceeding specified temperatures as measured by incore thermocouples.

Remarks: The reactor core was heated to 350 F. Specified temperature limitations and permissible temperature spreads between parts of the reactor were maintained within allowable ranges.

b) Criterion: The core temperature is automatically at, or near, 350 F.

Remarks: The core temperatures were maintained at, or near, 350 F by the recorder and indicating controllers.

V. EVALUATION

As specified in AI-P-1164, the allowable temperature gradients during pre-heat and the rate of pre-heating of the reactor, were found to be acceptable from the standpoint of thermal stress.

The reactor heaters and pre-heat instrumentation were adequate and operative. Pre-heating proceeded and was accomplished without major incidents.

HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
PROTECTIVE SYSTEM MODIFICATION FOR FUEL LOADING

AI-P-1166

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HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
PROTECTIVE SYSTEM MODIFICATION FOR FUEL LOADING AI-P-1166

I. PURPOSE OF TEST

The purpose of this test was to modify the plant protective system for dry critical fuel loading, and to verify the operability of the system with the modifications. Modification work included adding equipment to provide three incore nuclear instrumentation channels, incore flux level alarm and scram trips, and changing the protective system T_o computers 20% "Low Level Flux" trips to actuate at 2% of full power flux. (The equipment supplying the incore flux trips is referred to as incore nuclear instrumentation channels XI, XII, and XIII.)

Other modifications involved deactivating alarm, setback, and scram circuits and bypassing startup interlocks unnecessary for the dry critical fuel loading. In addition, the "High Temp Fuel Elements" annunciator contacts in temperature recorder TR-12 were set to actuate at 20 F above reactor ambient temperature.

II. DISCUSSION OF TEST PERFORMANCE

Prior to installation, the incore nuclear instrumentation system, which consists of three temporary startup channels, was submitted to comprehensive quality control tests at the manufacturer's plant, to demonstrate that the equipment met Atomics International purchase specification 7518-S72497. This testing consisted of checks on system accuracy, drift, and response. Simulated signals were substituted for actual chamber signals. All three chambers were tested to obtain integral bias curves at room temperature and at 700 F. Chamber voltage plateau curves were

also obtained at room temperature and at 700 F. Chamber test information is presented in Appendix A of this report.

After the incore nuclear instrumentation was installed, the equipment was again tested. Noise which plagued the source channels of the nuclear instrumentation also affected the incore nuclear instrumentation channels, but to an even greater degree. HNPF Preoperational Test Completion Report, "Nuclear Instrumentation System, AI-P-1119" presents the noise problem in detail and describes much of the corrective action taken in reducing the noise to an acceptable level. Additional action taken to correct the noise problem with the incore channels was as follows:

- a) The fission chamber preamplifiers were removed from the pipe and hanger trench and placed on the reactor loading face, close to the core position in which the associated fission chambers were located. Placing the preamplifier closer to the chamber reduced noise pickup.
- b) Two 24 V auxiliary relays and a transformer were installed in the main control board to switch the trip and alarm circuits of channels XI, XII and XIII. Making and breaking of the alarm and trip circuits without the auxiliary relays caused excessive noise interference within the linear count rate chassis. Figure 1166-1 delineates the change.
- c) The 24 V dc power supplying the preamplifiers from the control room "incore" nuclear equipment was rerouted in steel conduit, because of electrical power transients. Solution to this problem might have been accomplished by filtering, but, since the power cabling for the source range preamplifiers had to be rerouted, it was an easy matter to include the incore preamplifier power cabling in the same modification.

Figure 1166-1. Auxiliary Trip Alarm Relays for Incore Nuclear System

In addition to reducing noise, the following modifications and adjustments were found necessary and were made:

- a) The original high temperature cables which ran between the fission chambers and the reactor loading face were replaced with flexible cables made up of RG87A/U coaxial cable. These cables were connected directly to the pre-amplifiers, thereby eliminating unnecessary segments of cables and connectors. The original high temperature cables were replaced because of trouble encountered during initial efforts to install the cold-end connectors after the cables had been threaded through their shield plugs. The ends of three cables were broken in these attempts, and were replaced by the new cables mentioned above.
- b) Capacitor C-20, in the preamplifiers, was removed per the supplier's recommendation. This 0.01 mfd capacitor was across the high voltage input and caused "ringing" of the neutron pulse signals. The ringing was of greater magnitude than the neutron pulses and occurred due to the additional capacitance of the high voltage coaxial cable.
- c) The preamplifier for channel XI was repaired after three of its amplification stages were found to be inoperative. Cause of the trouble was traced to an error in initial cabling.
- d) The "Incore Flux" scram annunciator was rewired so that its field contacts came from the incore flux scram relay K57, in the protective panel, instead of the linear count rate chassis. The incore flux alarm contacts which were previously wired to the scram annunciator were re-wired to the "Incore Flux" alarm annunciator.

With the above adjustments and modifications made, the noise level on all coaxial cables was observed. Noise on cables routed

between the preamplifiers and the count rate meters, located in the control room, was observed on an oscilloscope and found to be less than 1 mv peak-to-peak when the system was hooked up, and 60 mv peak-to-peak when the cables were not connected to the count rate meters or preamplifiers.

Neutrons were detected on all channels from both a one curie plutonium-beryllium test source and from the antimony-beryllium source in the reactor. The shapes of the neutron pulse into the count rate meters, as viewed on an oscilloscope, are shown in Figures 1166-2, -3, -4, and -5. Alpha pulses from the fission chambers were lost in the noise of the system, since the system signal-to-noise ratio was approximately four-to-one.

The apparent resolution time of each channel was determined to be very close to 2.4 microseconds. Due to the high gain of the preamplifiers, which was required for the small signal pulses obtained from the fission chambers, feeding the double pulser signals through the preamplifiers was unsuccessful. As a result, the overall resolution time was determined in a less rigorous manner than asked for in the procedure. Pulses, the same width and height as neutron pulses out of the preamplifier, were fed into the preamplifier output cable. The pulse resolution time was measured as the distance between the leading edge of the closest pulses which could be counted. This time was approximately 2.2 microseconds.

Through the use of an oscilloscope on the output of the preamplifier, examination of occasional noise bursts originating from the input cable indicated that noise pulses of less than 0.2 microsecond width could get through the preamplifier. By adding this time to the measured resolution time of the remaining part of the system, an overall resolution time of not over 2.4 microseconds is evident.

The chambers were installed in core positions C-191, C-215, and C-239 and counts were taken with all control rods full in, one-half

Figure 1166-2. Oscilloscope Trace-Neutron Pulse, Channel XI
(0.5 usec/cm time scale, 0.1 V/cm voltage scale)

Figure 1166-3. Oscilloscope Trace-Neutron Pulse + Noise Interference, Channel XI (0.5 μ sec/cm time scale, 0.1 V/cm voltage scale)

Figure 1166-4. Oscilloscope Trace-Neutron Pulse, Channel XII
(0.5 usec/cm time scale, 0.1 V/cm voltage scale)

Figure 1166-5. Oscilloscope Trace-Neutron Pulse, Channel XIII
(0.5 μ sec/cm time scale, 0.1 V/cm voltage scale)

out and full out. For each channel, count rate vs discriminator curves were obtained, and from these curves discriminator settings were established. The incore fission chamber assembly is shown in Figure 1166-6.

To obtain a minimum of 300 cpm, as required by the dry zero power experiments, all three chambers were finally positioned 67 in. from the bottom of the respective thimbles.

In the remaining part of the procedure, channels XI, XII, and XIII level alarm and scram trips (not applicable to the dry zero power experiments) were installed as part of the protective system and then tested. Unnecessary alarms, setback and scram trips were bypassed. The "High Temp Fuel Elements" alarm setpoint on TR-12 was set and tested. All sodium valve position startup interlocks were bypassed.

The neutron source startup interlock from channels I and II was bypassed, and these trip contacts were temporarily wired as high level count rate scram trips for the initial loading of the reactor. These trip circuits were then set and tested to actuate a scram upon either channel counting 8×10^6 cpm.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

Test data obtained during this test are presented in the following figures:

Figure 1166-7 shows the actual count rate with no coincidence losses vs count rate with coincidence losses, using 2.4 microseconds as the resolution time of the channels. As indicated by the curve, when 1.6×10^7 neutron pulses per minute are detected by the fission chamber, approximately 10^7 pulses per minute find their way to the scaler output jack on the linear count rate chassis.

Figure 1166-6. Incore Fission Chamber Assembly

Figure 1166-7. Actual Count Rate vs Count Rate With Coincidence Losses

Figures 1166-8, -9, and -10 are count rate vs discriminator setting curves for channels XI, XII and XIII. To obtain the background count, with the chambers in the reactor, the source was separated; i. e., the antimony was removed from the beryllium sleeve. The difference between Figure 1166-10 curves and Figures 1166-8 and -9 curves was attributed to a lower gain in channel XIII preamplifier.

When the "source in" data for Figures 1166-8, -9, and -10 were taken, the chambers were located 18 in. from the bottom-most location. The final established discriminator setting was 6 for channel XI, 4.5 for channel XII, and 2.5 for channel XIII. These settings were selected between one and 1-1/2 divisions to the left of the knee of the curves on the basis of allowing all pulses within 25 % of maximum energy to be counted.

An investigation of statistical purity in nuclear counting was made for all three channels in accordance with HNPF SOP No. 5902 and is presented in Appendix B of this report.

The final data are shown in Tables I through III and plotted in Figures 1166-11 through 1166-13. For good statistics, the purity index of nuclear counting, ρ , must fall between 0.9 and 1.1. A satisfactory value of ρ was obtained for all three channels; * 1.08 for channel XI, 1.06 for channel XII, and 1.05 for channel XIII (Figures 1166-11, -12, and -13).

* A ρ of 1.59 was initially obtained for channel XIII, which indicated a narrowing Gaussian distribution, due to double pulsing or oscillation in the counting circuit. This was corrected by reducing the preamplifier gain.

Figure 1166-8. Count Rate vs Discriminator Setting For In-core Channel XI With No Fuel In Reactor

Figure 1166-9. Count Rate vs Discriminator Setting For In-core Channel XII With No Fuel In Reactor

Figure 1166-10. Count Rate vs Discriminator Setting For In-core Channel XIII With No Fuel In Reactor

TABLE I
NUCLEAR PURITY DATA FOR CHANNEL XI

Class No.	Class Limits		Class Contents Counts (%)	Sum of Class (%)
	Lower	Upper		
1	550	564	3	3
2	564	578	5	8
3	578	592	18	26
4	592	606	18	44
5	606	620	24	68
6	620	634	10	78
7	634	648	8	86
8	648	662	8	94
9	662	676	5	99
10	676	690	<u>1</u> 100	100

$$M_{\text{avg}} = 611.4 \quad \sigma = \sqrt{M_{\text{avg}}} = 24.7$$

$$M_{\text{avg}} - 2 \sigma = 562.0$$

$$M_{\text{avg}} + 2 \sigma = 660.8$$

$$M_L = 690 \quad M_e = 550$$

$$C = \frac{M_L - M_e}{10} = 14$$

TABLE II
NUCLEAR PURITY DATA FOR CHANNEL XII

Class No.	Class Limits		Class Contents Counts (%)	Sum of Class (%)
	Lower	Upper		
1	615.0	629.6	2	2
2	629.6	644.2	4	6
3	644.2	658.8	9	15
4	658.8	673.4	17	32
5	673.4	688.0	21	53
6	688.0	702.6	15	68
7	702.6	717.2	17	85
8	717.2	731.8	8	93
9	731.8	746.4	4	97
10	746.4	761.0	3	100
			100	

$$M_{avg} = 687.6 \sigma = \sqrt{M_{avg}} = 26.2$$

$$M_{avg} - 2 \sigma = 635.2$$

$$M_{avg} + 2 \sigma = 740.0$$

$$M_L = 761 \quad M_e = 615$$

$$C = \frac{M_L - M_e}{10} = 14.6$$

TABLE III
NUCLEAR PURITY DATA FOR CHANNEL XIII

Class No.	Class Limits		Class Contents Counts (%)	Sum of Class (%)
	Lower	Upper		
1	482.0	494.2	4	4
2	494.2	506.4	4	8
3	506.4	518.6	10	18
4	518.6	530.8	18	36
5	530.8	543.0	18	54
6	543.0	555.2	24	78
7	555.2	567.4	16	94
8	567.4	579.6	2	96
9	579.6	591.8	2	98
10	591.8	604.0	2	100
			<u>2</u> 100	

$$M_{\text{avg}} = 538.4 \sigma = \sqrt{M_{\text{avg}}} = 23.2$$

$$M_{\text{avg}} - 2 \sigma = 492$$

$$M_{\text{avg}} + 2 \sigma = 584.8$$

$$M_L = 604 \quad M_e = 482$$

$$C = \frac{M_L - M_e}{10} = 12.2$$

UPPER CLASS LIMIT (pulses)

Figure 1166-11. Nuclear Purity Curve for Channel XI

Figure 1166-12. Nuclear Purity Curve for Channel XII

Figure 1166-13. Nuclear Purity Curve for Channel XIII

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the test completion criteria associated with preoperational test AI-P-1166, Protective System Modification for Fuel Loading, as listed in NAA-SR-5700, Supplement 4, Page XI (a)-95 through XI (a)-97, with remarks regarding the test to explain compliance with the criteria.

- a) Criterion: Core loading and mapping cabinets 1 and 2 have been installed in the control room. Cables have been connected and checked for insulation resistance (10^{12} ohms minimum for low temperature coaxial cables, 10^{11} ohms minimum for high temperature signal cables, and 10^{10} ohms minimum for high temperature high voltage cables) and noise (no measurable noise with oscilloscope set at 0.005 V/cm).
Remarks: Cabinets 1 and 2 were installed in the control room. Cables and connectors were inspected for good workmanship. Insulation resistance was measured to be at least 10^{12} ohms for all cables except the high temperature (700 F) cables, which were replaced with the RG87A/ U cables. Measurement of the unamplified cable noise was found to be less than one millivolt when the cables were properly terminated. Upon amplification, this noise was not great enough to interfere with the counting of neutron pulses.
- b) Criterion: Each of the three channels XI, XII, XIII has been checked and calibrated in accordance with the manufacturer's instructions from the preamp through the Linear Count Rate Meter (LCRM) using a signal generator.

Remarks: A pulse generator was used at the preamp of each channel and pulses were fed through each channel to verify that the linear count rate meter reads out correctly. All power supplies were tested for correct voltages.

- c) Criterion: With each fission chamber connected to the channel, and using the Pu Be source for obtaining neutrons, the shape of the neutron and alpha pulses at the output of the LCRM has been observed and compared with predicted shapes. The countrate vs discriminator setting data have been recorded. The discriminator setting is locked at an alpha background of 20 cpm.

Remarks: Figures 1166-2, -3, -4, and -5 picture typical neutron pulses out of the preamplifiers of channels XI, XII, and XIII. Figures 1166-8, -9, and -10 are the count rate vs discriminator curves of the three channels. The final discriminator setting has been determined for each channel on the basis of the count rate vs discriminator curves and not on the basis of a 20 cpm alpha background.

- d) Criterion: The range selector physical stop on each LCRM has been set so that ranges greater than 10^6 cpm cannot be used.

Remarks: The physical stops were added to each linear count rate meter chassis.

- e) Criterion: The alarm and scram solid-state relays in each LCRM have been set at 75% and 90% full scale, respectively, using a simulated signal to verify the setting.

Remarks: The alarm and scram relays in each linear count rate meter were set and correct operation verified.

- f) Criterion: The three fission chambers have been installed in their thimbles in the reactor core and the insulation resistance of the high temperature coaxial cables verified to be greater than 10^{11} ohms for the signal cables and 10^{10} ohms for the high voltage cables. Remarks: The three fission chambers were installed in core positions C-191, C-215, and C-239. The original high temperature (700 F) cables (both signal and high voltage) were replaced with RG87A/U type cables, which "megged out" at above 10^{12} ohms.
- g) Criterion: Background countrate data have been recorded for each of the three channels, with the detector positioned near the core centerline. Remarks: The background count rate for each channel, with the chamber in the unloaded core and with the source separated, was measured to be less than one cpm.
- h) Criterion: With the reactor neutron source in the core, it has been verified that each chamber is counting neutrons. Countrate data have been obtained with each detector at the lower limit of travel, at core centerline, and at upper limit of travel. Remarks: Neutron counting by each channel was verified by observing a specific count rate with the reactor source intact and then observing near zero cpm when the source is separated. The count rate data obtained are reported in Figures 1166-8, -9, and -10. In addition, the neutron pulses were observed on an oscilloscope.
- i) Criterion: The following wiring connections have been made:
- 1) All three alarm contacts have been connected to the annunciator.

- 2) All three scram circuits in the three LCRM have been wired in series and connected to terminals TB-A-58 and TB-C-60 in the protective panel (reference drawings 7508-D72304 and 7518-D72315).
- 3) Relay contacts K-57-1, K-58-1, and K-59-1 have been wired in series. The lead from K-59-1B to K-1-1A has been removed.

Remarks:

- 1) Alarm contacts were connected and annunciator operation verified.
- 2) Scram contacts were connected in series with the incore flux protective relays K-57 and K-58. Correct operation was then verified.
- 3) Relay contacts were wired in series and correct operation was verified.

- j) Criterion: It has been demonstrated that any one of the three incore channels, or any one of the two period channels, or any one of two source range channels on high countrate will scram the reactor by putting in a simulated signal at each LCRM and period chassis, and verifying that control rods drop. (Calibration of the period channels is covered under AI-P-1119). Operation of the annunciators by any of the five channels has been verified.

Remarks: The three incore alarms and scrams were each tested and verified for correct operation. In addition, the source channel level scrams, which were temporarily wired from the source channel start-up interlocks, were each tested and verified for correct

operation. Finally, each period scram was tested and verified for correct operation.

- k) Criterion: A jumper has been installed in the main control board from lead 3X3 to 3X15 (Reference Drawing D794552), thereby bypassing all sodium valve interlocks and the channels I and II source interlocks. It has been verified that the remaining startup interlocks (key, gantry crane, control rods down, scram and set-back) are operative by opening each interlock in turn, and observing that rods cannot be withdrawn.

Remarks: The sodium valve and the source channel interlocks were bypassed. The remaining interlocks were proven operative in preoperational test AI-P-1121, "Plant Protective System". *

- l) Criterion: Jumpers have been installed in the nuclear panel to interrupt current to K-59 in the protective panel upon high source range count or short period on intermediate range channels (Reference Drawings 7508-D72304 and 104R189).

Remarks: K-59 has been wired to drop out only upon a short period signal from either of the intermediate range channels. A high count on any one of the source range channels or the incore channels will drop out K-57 and K-58.

* AI-P-1121, Plant Protective System, unpublished as of this report date.

V. EVALUATION

Modification of the protective system was affected, to make necessary temporary changes in the protective system, for fuel loading. These changes were the bypassing of certain scrams and interlocks, the adjusting of computer setpoints to correspond with anticipated nuclear conditions, and the provision of additional alarm and trip points.

The following scrams and interlocks were bypassed (wet critical loading):

- a) Steam generator low water level scrams.
- b) Source indication startup interlock.
- c) Moderator coolant valves startup interlock.
- d) Main head transfer system sodium valve startup interlocks.
- e) Loss of pump power scram.
- f) Flow ratio scrams.

Computer setpoints changed for fuel loading were:

- a) Scram trip at 2%, (normal 20%) full power flux, when primary sodium flow is less than 14%.
- b) Scram trip at neutron flux greater than 20% (normal 125%) full power flux.
- c) Setback trip at 2% (normal 16%) full power neutron flux, at primary sodium flow of less than 12% full flow.
- d) Setback trip at neutron flux greater than 18.5% (normal 115%) of full power flux.
- e) Alarm signal at 1% (normal 12%) full power neutron flux, when primary sodium flow is less than 10% of full flow.
- f) Alarm signal at neutron flux greater than 17% (normal 107%) full power flux.

Additional trips and alarms were provided for:

- a) Scram trip at 90% of full scale on any one of three incore nuclear instrumentation (ICNI) channels.
- b) Scram trip for count rate on source range channels, greater than 8×10^6 cpm.
- c) Scram trip for reactor period less than 5 seconds on intermediate range instrument channels.
- d) Alarm signal at 75% of full scale count rate on incore instruments.
- e) Alarm signal at measured fuel outlet sodium temperature of 50 F greater than bulk temperature. This alarm requires manual scram action.

Other interlocks, scrams, setbacks and alarms were operative and in the normally set condition.

The enumerated protective system conditions were ascertained from the master test procedure, direct evaluation checks and verbal communication with the cognizant protective system engineer.

The test completion report indicates that two of the criteria prescribed by the test procedure, and the prescribed method of determining resolution time could not be complied with. Therefore, substitute methods were introduced. The method used for determination of resolution time has been used many times before at other facilities, and is considered a good alternate procedure. Criterion a), which prescribes no measurable noise on the oscilloscope, is somewhat stringent. The less than one millivolt unamplified noise observed appears to have been small enough so as not to cause undue spurious counts. Criterion c), which calls for the discriminator setting to be locked at an alpha background of 20 cpm, was impossible to comply with, because the background noise and the alpha counts were similar in registered amplitude at the position of the counter. Hence, the discriminator settings had to be adjusted to discriminate against all of the line noise. This has subsequently been found to be adequate.

All comments applicable to the protective system (see test report and evaluation of AI-P-1121,*Protective System) are also pertinent to this evaluation. The modifications which were made insured general safety of the fuel loading operation. The protective system shall be returned to its normal operating status at the direction of the AI Field Superintendent.

* AI-P-1121, Plant Protective System, unpublished as of this report date.

APPENDIX A

STROMBERG-CARLSON

P. O. No. 19-105161

The enclosed curves, Figures 1166-A, -1 through 1166-A-12, were taken on three WX-4414's, No. 100, No. 101, and No. 102, supplied to Stromberg-Carlson.

The curves consist of three parts:

1. Integral bias curve at room temperature.
2. Integral bias curve at 700 F minimum.
3. Voltage plateau curve.

Each detector was placed in an oven for the integral bias curve and the source positioned along-side the oven. Therefore, the integral bias curves may be compared cold to hot for any one detector. It was not possible to position each tube precisely, therefore, tube to tube variation may exist.

Figure 1166-A-1. Integral Bias Curve at 68 F, Channel No. XI

Figure 1166-A-2. Integral Bias Curve at 68 F, Channel No. XII

Figure 1166-A-3. Integral Bias Curve at 68 F, Channel No. XIII

Figure 1166-A-4. Integral Bias Curve At 710 F Minimum, Channel No. XI

Figure 1166-A-5. Integral Bias Curve At 710 F Minimum, Channel No. XII

Figure 1166-A-6. Integral Bias Curve At 710 F Minimum, Channel No. XIII

Figure 1166-A-7. Voltage Plateau Curve At 68 F, Channel No. XI

Figure 1166-A-8. Voltage Plateau Curve At 68 F, Channel No. XII

Figure 1166-A-9. Voltage Plateau Curve At 68 F, Channel No. XIII

Figure 1166-A-10. Voltage Plateau Curve At 710 F, Channel No. XI

Figure 1166-A-11. Voltage Plateau Curve At 710 F, Channel No. XII

Figure 1166-A-12. Voltage Plateau Curve At 710 F, Channel No. XIII

APPENDIX B

Procedure for the Investigation
of Statistical Purity in Nuclear
Counting.

(HNPF SOP NO. 5902)

INITIATED BY _____ DATE _____ TIME _____

PROCEDURE FOR THE INVESTIGATION OF STATISTICAL PURITY IN NUCLEAR COUNTING¹

PURPOSE

To compare the statistics of counting as observed in counting experiments to the statistics theoretically associated with nuclear counting.

To evaluate the deviation of actual counting data from theoretically pure nuclear counting statistics in order to pinpoint noise effects and determine the validity of data analysis based on an assumed Poisson distribution.

METHOD

The statistical fluctuations in the counting of radioactive decay processes is governed by the Poisson distribution:

$$W_m = \frac{(Nt)^m}{m!} e^{-Nt} \quad (1)$$

where W_m is the probability that a certain number, m , counts will be recorded in a counting time, t , from a source whose average count rate is N . If m is larger than about 20 counts, equation (1) can be shown² to be well approximated by a Gaussian distribution:

$$W(\nu) = \frac{1}{\sqrt{2\pi Nt}} e^{-\frac{\nu^2}{2Nt}} \quad (2)$$

where $W(\nu)$ is the probability, (per unit ν) that a certain deviation value, ν , will occur. ' ν ' is the deviation of a count from the average:

¹ O. Riedel, "Statistical Purity in Nuclear Counting," Nucleonics, Vol. 12, No. 6, June 1954, p. 64.

² See for example, Y. Beers, Theory of Error, Addison-Wesley, 1957, pp 47-48.

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$$\nu = m - m_{\text{average}} \quad (3)$$

$$\text{and } m_{\text{average}} = Nt \quad (4)$$

Equation (2) is not only a Gaussian distribution, but a Gaussian distribution with a definite precision index (or width). The standard deviation³ of the Gaussian of equation (2) is:

$$\sigma = \sqrt{Nt} = \sqrt{m_{\text{average}}} \quad (5)$$

The standard deviation of any Gaussian is defined such that 95.4% of the counts obtained under the Gaussian error distribution fall in the range

$$m_{\text{average}} - 2\sigma \text{ to } m_{\text{average}} + 2\sigma .$$

It is possible for counting to be Gaussian, but not have the standard deviation predicted theoretically for pure nuclear counting; equations (2) and (5).

It is assumed that any change from pure statistics is introduced by the counting equipment.

In general, there are two types of noise of importance in connection with counting experiments: the large group of noise pulses which are readily obvious to the casual observer, and the more subtle, randomly distributed noise pulses which tend to broaden the count distribution. Thus, the problem is to distinguish between the particular Gaussian distribution associated with pure nuclear counting and a Gaussian distribution broadened by random noise pulses or narrowed by double-pulsing or oscillation in the counting system.

Graphical solution of the above problem can be accomplished through the use of "Probability Graph Paper." This graph paper has the property that a plot of the integral of any Gaussian

³ Op. cit., p. 48.

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$$\int_{-\infty}^v Wg(v) dv = f(m') \quad (6)$$

$$m' = m_{\text{average}} + v'$$

distribution as a function of the upper limit of the integration, is a straight line. $Wg(v)$ in equation (6) has the form:

$$Wg(v) = \frac{1}{\sqrt{2\pi}\sigma} e^{-\frac{v^2}{2\sigma^2}} \quad (7)$$

where σ is the standard deviation.

A step by step example of the application of this method is given in the Appendix.

If it is desired to check the counting of a particular counting system, proceed as follows:

1. Count a constant source. Obtain at least 50 data points of 500 to 5000 counts each. Record the counting time. The count rate should be less than 1000 counts per second so that resolution time effects may be neglected.
2. Calculate the average count, m_{avg} ; the theoretical standard deviation, $\sqrt{m_{\text{avg}}}$; and the Gaussian error distribution range,

$$m_{\text{avg}} - 2\sqrt{m_{\text{avg}}} \text{ to } m_{\text{avg}} + 2\sqrt{m_{\text{avg}}}.$$

3. Scan the 50 data points for the highest, m_L , and the lowest, m_e , count in the group.
4. Calculate the "class width," c :

$$C = \frac{m_L - m_e}{10} \quad (8)$$

5. Calculate the upper and lower class limits of the ten classes which span the 50 counts. The lower and upper limit of class No. 1 are:

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$$L = m_e, \quad U = m_e + C ;$$

Class No. 2:

$$L = m_e + C, \quad U = m_e + 2C ;$$

Class No. 10:

$$L = m_e + 9C, \quad U = m_e + 10C = m_L .$$

6. Find the number of counts out of the group of 50 counts which fall into each of the 10 classes. These are called the class contents.
7. Express the content of each class as a percent of 50.
8. Calculate the sum of class contents percentages at the upper limit of each class, i. e., at the upper limit of the third class the sum is the sum of the percentage contents of classes No. 1, 2, and 3; at the upper limit of the fourth class the sum is over the first four classes, etc.
9. Label the linear (ordinate) scale of a sheet of probability paper in convenient units so that m_L and m_e are at the upper and lower ends of the scale. The abscissa is pre-scaled in a nonlinear percentage scale.
10. Plot the sum of the class content percentages at the upper limit of each class.

Plot m_{avg} at 50%.

Plot $m_{avg} - 2 \sqrt{m_{avg}}$ at 2.3%.

Plot $m_{avg} + 2 \sqrt{m_{avg}}$ at 97.7%.

11. Draw a straight line through the $m_{avg} + 2 \sqrt{m_{avg}}$ and m_{avg} points. This is the line corresponding to pure nuclear counting. Draw the best straight line through the experimental data points.
12. An experimental slope less steep than the theoretical one indicates broadening of the Gaussian; a steeper slope indicates narrowing. Calculate the experiment standard deviation by reading the point where the experimental line crosses 2.3%:

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$$m_{\text{exp}} (2.3\%) = m_{\text{avg}} - 2 \sigma_{\text{exp}} \quad (9)$$

or,

$$\sigma_{\text{exp}} = \frac{m_{\text{avg}} - m_{\text{exp}} (2.3\%)}{2} \quad (10)$$

13. The ratio:

$$\rho = \frac{\sigma_{\text{exp}}}{\sqrt{m_{\text{avg}}}}$$

is called the purity index and should be 1 ± 0.1 for good statistics.

DATE _____

APPENDIX

EXAMPLE

Using subcritical counting data obtained during the sodium Reactor Experiment second core critical experiments, 50 points were available at an average count rate of 1.7 cps. Numbers in parentheses below correspond to procedural step numbers.

$$(2) \quad m_{\text{avg}} = 101.6 \text{ counts}$$

$$\sigma = \sqrt{m_{\text{avg}}} \approx 10 \text{ counts}$$

$$m_{\text{avg}} + 2\sigma = 121.6 \text{ counts}$$

$$m_{\text{avg}} - 2\sigma = 81.6 \text{ counts}$$

$$(3) \quad m_L = 132 \text{ counts}$$

$$m_e = 75 \text{ counts}$$

$$(4) \quad C = \frac{132-75}{10} = 5.7 \text{ counts}$$

(5), (6), (7), (8)

Class No.	Class Lower	Class Upper	Class Contents		Sum of	
			No. Counts	%	Class	%
1	75	80.7	1	2	2	
2	80.7	86.4	3	6	8	
3	86.4	92.1	6	12	20	
4	92.1	97.8	7	14	34	
5	97.8	103.5	11	22	56	
6	103.5	109.2	13	26	82	
7	109.2	114.9	5	10	92	
8	114.9	120.6	2	4	96	
9	120.6	126.3	1	2	98	
10	126.3	132.0	1	2	100	
			50			

DATE _____

$$(12) \quad m_{\text{avg}} - 2\sigma_{\text{exp}} = 80$$

$$\therefore \sigma_{\text{exp}} = 10.8 \text{ counts}$$

$$(13) \quad \rho = \frac{10.8}{10} = 1.08$$

HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
HOT SODIUM CIRCULATION TEST

AI-P-1167

Test and Analysis

by

P. F. SHAW

Evaluation

by

L. L. JOHNSON

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.

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SUMMARY

The purpose of this test was to verify the adequacy of the design, construction, and components of the main heat transfer system of the Hallam Nuclear Power Facility (HNPF) for elevated temperature and low power operation. To this end, certain tests were performed to:

- 1) Chart pipe displacements.
- 2) Determine system hydraulics.
- 3) Obtain an interrelationship between the main heat transfer system and plant control system.
- 4) Verify core alignment at elevated temperatures.
- 5) Determine the effect, if any, of elevated temperature on control rod operation.
- 6) Determine, by careful observation and inspection, that no adverse conditions of interference, motion, binding, or stress of system components occurred or could be expected to occur during power operation.

Item 1) was performed between 340 F and 540 F. Items 2), 3), and 6), above, were carefully investigated at system temperatures of 340 F, 440 F and 540 F. Items 4) and 5), above, were investigated only at 540 F, that being the highest temperature reached in the overall test.

Certain piping interferences existed, some hangers were found inoperable, some valve difficulties developed, and some of the recorded piping displacement measurements deviated from the predicted displacements. The interferences were cleared and the hangers were repaired. Valve difficulties were resolved as necessary to complete this test. Testing was extended to accomplish the following:

- 1) Investigation of the displacement discrepancies and recheck of data during cooling from 540 F to 350 F.
- 2) Re-determination and extension of system hydraulic characteristics.
- 3) Verification that corrective actions taken had indeed corrected the items listed and had not created any undesirable characteristics.

Testing results are exhaustively presented in an objective manner in Section III of this report.

HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
HOT SODIUM CIRCULATION TEST, AI-P-1167

I. TEST PURPOSE

The purpose of Hot Sodium Circulation Test, AI-P-1167, was to achieve the following:

- a) Verify that the piping displacements in the sodium heat transfer system, under simulated operating conditions and with circulating sodium at an elevated temperature, would not impair the integrity of the system. Verification of the integrity of the system at test conditions, and of the performance to be expected with power generation under actual operating conditions.
- b) Mechanically operate the main sodium pumps, flow control valves, and blocking valves in the sodium heat transfer system, and demonstrate their operability under the conditions of the test, which will simulate the plant operating conditions of an elevated temperature and a sodium environment.
- c) Operate control and safety rod drive mechanisms, in the reactor structure, with sodium at an elevated temperature flowing through the core. Simulate scram conditions to verify satisfactory operability of the control and safety rods under the conditions of the test.
- d) Verify core - loading face alignment at an elevated temperature, by demonstrating the capability of changing fuel with 540 F isothermal sodium in the reactor, through the removal and insertion of a dummy element and an actual fuel element assembly (without fuel cluster) in three peripheral core positions.

II. DISCUSSION OF TEST PERFORMANCE

A. PIPING DISPLACEMENT MEASUREMENTS

Sodium heat transfer system piping movement was monitored at 57 strategic points during the performance of AI-P-1167, "Hot Sodium Circulation Test". Measurement points No. 1 to 15 inclusive are in the vicinity of the reactor cavity diaphragm seal and are expected to give information on the movement of the primary pump suction lines inside the reactor cavity. Measurement points No. 24, 25, 26, 30, 31, 32, 36, 37, 38, and 54 are located near the sodium pumps and their displacements would be indicative of any deformation of the pump cases. The balance of the points were selected to provide indication of any obstructions at intermediate points which could prevent the free movement of the systems. Measurements cover all three primary loops and one of the three identical secondary loops.

The pipe movements at the selected strategic points are transmitted to plumb bobs or other pointers by means of pipe attachments and piano wire. The pointers are conveniently located for measuring the amount of movement with respect to a reference plate secured to the floor or other structure.

Piping displacement measurements were made with each of the following conditions existing:

- 1) Piping at ambient temperature and empty of sodium.
- 2) Piping at approximately 340 F and empty of sodium.

- 3) Piping at approximately 340 F and filled with sodium.
- 4) Piping at approximately 440 F and filled with sodium.
- 5) Piping at approximately 540 F and filled with sodium.

Additional measurements were taken at 440 F and 350 F (during and after the cooldown) in order to recheck and correct readings, and to demonstrate proper piping movement during cooldown.

B. PIPING INSPECTION

Concurrent with making piping displacement measurements at each of the conditions described above, a general inspection of system piping was performed. The inspection included investigation of the following items:

- 1) Adjustment of pipe hangers. All variable load spring hangers were checked and adjusted to assure that hanger loads at operating temperature will agree with the design loads.
- 2) Observation of movement of the constant load hanger supports.
- 3) Piping vibration with sodium flowing in the system (visual examination only). This investigation was performed by competent piping engineers.
- 4) Potential interference at operating temperatures between piping and adjacent structures or equipment. This investigation includes all material applied or attached to the piping, such as insulation, lagging, measuring devices, heaters and accessories.
- 5) Leakage of sodium at valve stems, freeze traps, or any other point.

C. RAISING SODIUM TEMPERATURE TO 440 F

Filling of the steam generators with water, blow-down and sampling activities, and the maintaining of evaporator level, were performed according to Standard Operating Procedures.

The electrical preheat system and the six main sodium pumps were utilized to increase sodium temperature. Temperatures were determined from the plant control system temperature recorder controllers and checked by obtaining superheater pressures and referring to a table of saturated steam properties for the corresponding temperatures.

Those items of equipment and piping that will be subjected to elevated temperatures during power operation were raised in temperature with the main heat transfer system. The equipment maintained at 350 F during this test is listed in Appendix A.

The test temperature of the system was maintained by controlling pump speed, thereby controlling the heat input capacity of the six main pumps.

A slight decreasing temperature trend occurred when the primary pump speeds were reduced below 20% for system testing. Testing was interrupted as necessary to maintain system temperatures within approximately 5 F of the desired test temperature.

D. TRANSFER OF LOOP HAND CONTROL STATIONS AT 440 F

Pump speeds were decreased from power input speeds to 20% when 440 F was reached.

Division of Load Computer hand control stations (steam demand) HCS-501, -502, and -503 were manually set at 20% steam (demand). Pump speed controllers

HCS-102, -202, -302, -104, -204, and -304 were then switched to automatic control to determine smoothness of transfer from manual control to automatic. A significant step change would result in an undesirable thermal transient at operating temperature conditions.

Loop flow (millivolt signal), pump speed, and pump differential pressure data were recorded for all six pumps at 20% speed in manual control and immediately after transferring to automatic control.

E. ESTABLISHING PRESSURE DROP

Primary throttle valves were positioned to simulate the calculated reactor pressure drop of 38.4 psi at rated flow.

F. PUMP SPEED INCREASES

The speed of circuit No. 1 pumps was increased to 30% steam demand by hand control station HCS-501. Both the primary and secondary pumps were observed to come up to 30% steam demand speed. Circuit No. 2, and then circuit No. 3, pumps were increased to 30% steam demand, employing HCS-502 and -503, respectively.

Circuit No. 3, pumps were increased to 50% steam demand speed. Flow in the remaining two circuits was similarly increased.

All pump speeds were then increased to 100% steam demand; circuit No. 1 first, followed by No. 2 and then No. 3.

Flow, speed, pressure in, pressure out, and pressure rise data were recorded for each of the six main sodium pumps with pump speed at 100% steam demand.

G. FLOW TRIMMING

With 60% scale settings on HCS-501, HCS-502, and HCS-503, load computer signals FCCp-501, -502, -503, -504, -505, and -506 were trimmed to equalize flows in all loops. Pump control was then transferred to HCS-504, total steam demand hand control station, by transferring HCS-501, -502, and -503 to automatic control. Flow control was thus established by the plant base load hand control station, HCS-504. The division of load computer ratio relays were adjusted as required to trim all flows to 60%.

H. DETERMINATION OF SYSTEM HYDRAULIC CHARACTERISTICS

The sodium heat transfer system was subjected to the following hydraulic manipulations to obtain data with which to appropriately evaluate system performance:

- 1) All three circuits set at approximately 600 rpm pump speed from HCS-504.
- 2) All three circuits set at approximately 550 rpm pump speed from HCS-504.
- 3) All three circuits set at approximately 440 rpm pump speed from HCS-504.
- 4) All three circuits set at approximately 330 rpm pump speed from HCS-504.
- 5) All three circuits set at approximately 220 rpm pump speed from HCS-504.
- 6) All three circuits set at approximately 110 rpm pump speed from HCS-504.
- 7) All three circuits set at 50% from HCS-504.
- 8) All three circuits set at 70% from HCS-504.
- 9) HCS-501 transferred to "Manual" and Circuit No. 1 adjusted to 70% flow.

- 10) Circuit 1 reduced to 60% flow from HCS-501; 2 and 3 compensating.
- 11) Circuit 1 increased to 75% flow from HCS-501; 2 and 3 compensating.
- 12) Control of Circuit No. 1 transferred to HCS-504 at 70% flow.
- 13) Items 9, 10, 11, and 12 repeated for Circuit No. 2; 1 and 3 compensating.
- 14) Items 9, 10, 11, and 12 repeated for Circuit No. 3; 1 and 2 compensating.
- 15) All three circuits at 50% flow, controlled from HCS-504.

Periodic inspections of the areas were conducted for possible noises, pipe motion, interferences or potential interferences, and any other indication of unsatisfactory performance.

I. RAISING SODIUM TEMPERATURE TO 540 F

Sodium heat transfer system temperature was increased to 540 F in the manner described in Section II. C of this report. Warm-up to 540 F was accomplished at a rate of 1.5 to 2.0 F per hour. Temperature was controlled by pump speed after 540 F had been attained,

J. TESTING AT MAXIMUM SODIUM TEMPERATURE

1. Sodium Pumps

The six main sodium pumps were operated at 20%, 50%, and 100% of design speed, at 540 F system temperature, to determine that proper operating characteristics exist at elevated temperatures. All pump instrumentation was observed closely during each of the three runs. The performance test was sufficiently long to allow stable temperature and flow conditions to be verified.

Each pump was stopped and started three times at 540 F. Each time a pump was stopped, it was shut down for 20 to 30 min. While the pump was in the shut-down condition, other 540 F testing was accomplished, such as main loops blocking valve manipulation and core alignment tests.

In addition to recording pump information as displayed by pump instrumentation for each of the six pumps, the following items were investigated:

- a) Internal noise or rumble.
- b) Vibration of pump assembly.
- c) Unstable speed or pulsation.
- d) Seal oil system operation.

2. Throttle and Block Valves

The primary throttle valves were operated to full open, to full closed (98%), and back to the simulated core pressure-drop position, with a system temperature of 540 F. Pump speeds were then increased to 100% and proper valve operation tested by simultaneously driving the throttle valves to full closed and de-energizing the pump coupling. During testing, the valves and adjacent piping were observed for stem leakage and movement, respectively. Other items investigated regarding valve operation:

- a) Operation of air motors.
- b) Operation of line shafts.
- c) Rigidity of bearing supports.

- d) Vibration.
- e) Adequate lubrication of components.
- f) Leaks on stem seals.
- g) Piping movement during quick-closure.
- h) Clearance of mechanisms.

During one of the three pump shutdowns, as discussed previously, the primary blocking valves were operated from the full open position to full closed (blocking) position and back to full open. The secondary throttling valves were subjected to the same quick-closure test described for the primary throttling valves. The valves were observed for the same eight items listed above for the primary throttling valves.

K. CONTROL ROD DRIVE TESTING

Control rods were operated at a system temperature of 540 F to verify proper operation at elevated temperatures. Each rod was individually withdrawn and then scrambled. Withdrawal time and drop time were measured and recorded. The control rod test cart was used to manipulate the rods.

Each control rod was operated as necessary to verify proper operation of the indicating lights and snubbers. The drive was then operated up and down sufficiently to obtain voltage and current measurements for each phase in each direction. Each rod was withdrawn until the drive-up limit switch stopped the drive. Proper position was verified and withdrawal time recorded. The rod was then scrambled and latch release time, drop time to the four ft. position (70% insertion), and time to the full insertion position were determined by the use of a Sanborn recording oscillograph.

L. TESTING CORE ALIGNMENT

Three dummy elements and one mock fuel element (a fuel element with the fuel cluster removed) were utilized to demonstrate proper core alignment at 540 F. The fuel handling machine was employed to accomplish the operation in order to maintain the helium atmosphere over the sodium pool in the reactor. The fuel handling machine was initially loaded with the mock fuel element. This element was then exchanged for each of the three dummy elements in the reactor; reinsertion of each dummy completed the testing. System primary pump speeds were at 20% during the operations.

The core positions utilized for this portion of the test are in the outer periphery of the core at approximately 120 degree intervals; specifically, core positions C-185, C-209, and C-233.

Fuel handling machine grapple load cells were used to determine if any misalignment of the elements was occurring during element moving operations. Load cell readings were observed and recorded at the following four points for both the dummy elements and the actual element at the three core locations:

- 1) First six inches of upward travel (breakaway).
- 2) Remainder of upward travel.
- 3) During downward travel.
- 4) During last six inches of downward travel (re-seating).

M. SYSTEM TESTING AT 540 F

All portions of main heat transfer system hydraulic and control testing performed at 440 F were repeated at 540 F, to verify proper operation at that temperature and

to detect any changes created by the 100 F rise. The portions of the test repeated at 540 F are listed below, with references to portions of this report which have already described the mode of test performance.

- 1) Transfer of Loop Hand Control Station at 540 F - Reference II. D.
- 2) Establishing Pressure Drop - Reference II. E.
- 3) Pump Speed Increases - Reference II. F.
- 4) Flow Trimming - Reference II. G.
- 5) Determination of System Hydraulic Characteristics - Reference II. H.

N. ADDITIONAL TESTING

Analysis and review of the overall test led to the addition of the following sections to the test:

- 1) Piping displacements measurements at 440 F, and 350 F (during cooldown).
- 2) Correcting assumptions made in originally establishing the primary pump 100% speeds and re-running the hydraulic tests on each of the six loops at each temperature during cooldown.

The first addition was for the purpose of determining the effect of repairing the reactor cavity pipe hangers, and to recheck readings where measured pipe displacement did not agree with predicted displacements.

The second addition was made to recheck and extend loop hydraulic characteristics.

O. COOLDOWN TO PREHEAT TEMPERATURE

Sodium system cooldown from 540 F to 350 F was accomplished at a rate of less than 50 F per hour.

Pump speeds on all six pumps were reduced to 20% speed and all preheat controller setpoints were estab-

lished at 350 F to initiate the cooldown. The equipment being maintained manually at 350 F was put on automatic control after the reduction of preheat controller setpoints to 350 F.

The cooldown rate was controlled by bleeding steam through the superheater stop check valve drain valves, the superheater vent valves, and the eliminator vent valves. The second drain valve on each of these blow-down lines was employed as a throttle valve.

III. ANALYSIS AND PRESENTATION OF RESULTS.

A. PIPING DISPLACEMENT MEASUREMENTS

Piping displacement measurements were made at the 57 locations in the sodium heat transfer system indicated in Figures 1167-1 through 1167-8.

The agreements between the calculated and measured displacements indicates the validity of the stress analysis performed in connection with the system design. This stress analysis was performed according to the following procedures and design objectives:

- a) Forces and movements due to the thermal expansion of each line were obtained by using the Kellogg General Analytical Method as programmed for the IBM 709. (Design of Piping Systems - The M. W. Kellogg Co. publication)
- b) Maximum stresses were computed from the movement and force data calculated by the IBM 709.
- c) According to the design objectives, the calculated pipe stresses in the sodium heat transfer system do not exceed 75% of the allowable values which are stipulated in the American Standard Code for Pressure Piping ASA B31.1 - 1955, except for secondary lines No. 264, 265, and 266 between the gas seal and the intermediate heat exchangers where the analysis showed a maximum stress of 100% of the Code allowable range.
- d) Forces and movements transmitted to equipment did not exceed the maximum values allowed by the equipment manufacturer.

Expected pipe movements were calculated prior to the performance of the test. These calculations were reviewed during the test performance and were modified according to more accurate computations required in the evaluation process.

Figure 1167-1 Plumb Bob Locations For Pipe Movement Test

Figure 1167-3 Plumb Bob Locations For Pipe Movement Test

Figure 1167-4 Plumb Bob Locations For Pipe Movement Test

Figure 1167-5 Plumb Bob Locations For Pipe Movement Test

Figure 1167-6 Plumb Bob Locations For Pipe Movement Test

INDICATOR ARM
PRIMARY SODIUM PUMPS

(3 REQ,D)

SEC SODIUM PUMP NO.2

(1 REQ,D)

Figure 1167-7 Plumb Bob Locations For Pipe Movement Test

TYPICAL SODIUM PI

CLAMP FOR PI HEAD 7 REQ,D

Figure 1167-8 Plumb Bob Locations For Pipe Movement Test

The following considerations were included in determining the expected piping movements.

- 1) Linear thermal expansion at test temperatures from 70 F to:
 - a) 350 F = 3.2 in./100ft
 - b) 540 F = 5.5 in./100 ft
- 2) Lines No. 231, 232, and 233.
 - a) Vertical movement of reactor outlet nozzle at design operating conditions equals 2.12 in., based on 820 F mean core tank temperature.
 - b) East-west movement of the junction point between sodium pipe and guard pipe results from the free expansion of the sodium pipe, less the restraint of the guard pipe (3.90 in. - 0.78 in. = 3.12 at 1000 F).
 - c) Guides at measurement points No. 2, 7, and 12 limit north-south movement.
 - d) Measurement points No. 5, 10, and 15 are provided with rigid pipe hangers, allowing zero vertical movement.

The listed vertical movements of measurement points No. 1 to 15 inclusive were calculated by assuming uniform slope between the reactor nozzle and points No. 5, 10, and 15. In the calculation of the movements for the balance of the measurement points, the following considerations apply:

- 1) North-south movements are required to establish the deflections of the individual east-west pipe members, by appraising the effect of adjacent elbows and the linear expansion of the perpendicular runs.
- 2) East-west movements are based on linear expansion and the deflections of the north-south members and adjacent elbows.

- 3) Vertical movements of the spring-supported points are figured by assuming uniform slope between the primary pump suction elbow and the next rigid rod hanger.
- 4) Measurement points No. 25, 31, and 37 are on the pump suction elbow. (Pump expansion at test temperatures was used for vertical displacement.)
- 3) Lines No. 234, 235, and 236 (Figure 1167-4). Measurement points are in the vicinity of the primary pump. Horizontal movements were established according to the linear expansion. Vertical movements are based on pump expansion.
- 4) Line No. 265 (Figure 1167-5). Measurement points No. 40 to 45, inclusive, are between the IHX and the seal plate in the IHX cell. Horizontal movements are calculated by the same procedures as used for lines No. 231, 232, and 233.

Vertical movements are based on the IHX expansion and uniform slope between the IHX and the vertical pipe leg.

Measurement points No. 50 and 51 are between the pipeway seal and the secondary expansion tank. North-south movement of these points follows the expansion of the tank and the adjoining radial pipe section. The restraint of the anchor at the seal is taken up by the long vertical riser. Vertical movement has no significance, since it depends rather on the method of support than on the expansion.

- 5) Line No. 268 (Figure 1167-6). North-south movement of this loop is free and the displacement of point No. 52 equals the expansion of legs in the same direction. East-west movement of point No. 53 is obtained by dividing linear expansion in this direction in proportion to the torsion of the vertical legs. Vertical movement of point No. 54 equals the sum of the pump and vertical suction pipe expansion.

- 6) Line No. 271 (Figure 1167-6)
Horizontal movements are calculated by the same procedures as used for lines No. 231, 232, and 233. Vertical movements are based on the downward displacement of the secondary pump discharge and the rise of the superheater inlet nozzle and adjacent radial pipe suction, assuming uniform slope between these points and point No. 56 (rigid hanger).
- 7) Line No. 274 (Figure 1167-5)
Measurement points No. 48 and 49 are between the evaporator outlet and seal plate. Horizontal movements are based on the free expansion of the straight pipe runs. Uniform slope and vertical movement, as caused by the expansion of the run adjacent to the evaporator, are assumed between this leg and the anchor seal. Measurement points No. 46 and 47 are in the IHX cell between the anchor seal and IHX inlet. Configuration and calculation method is similar to line No. 265 but without a vertical leg, and consequently no movement in that direction.

The results of the displacement measurements performed preceding and during the temperature rise are incorporated in Table I. This table includes the identification of all measurement points, the calculated movements at 350 F and 540 F and the actually measured displacements at or in the vicinity of 340, 440 and 540 F temperatures. The measurements were taken with the utmost care, but certain readings proved to be inconsistent with adjacent readings and not reliable for analysis purposes. These discrepancies are mainly due to the fact that a long time period elapsed between two readings at the same point and the measurement equipment had been disturbed by operations and maintenance activities. The behavior of the piping at these points was rechecked during the cooldown of the system; the resulting data are presented in the course of the analysis. However, some other inconsistent measurements which had negligible significance in this analysis have

TABLE IV
EXTRAPOLATED CRITICAL LOADING - "RODS OUT"

No. of Elements Loaded	Linear Plot (N)	Square Root Plot ($\sqrt{N}$)	Log Plot (Log N)	Radial Buckling (B_n)
30	37.2	37.2	38.0	41
38	42.2	42.3	42.6	43.5
41	48.4	49.0	49.0	50.1
46	50.0	50.1	50.2	50.3
49	51.4	51.4	51.3	51.7
51	52.0	52.0	52.0	51.9
52	52.5	52.6	52.5	52.4

It should be noted that the linear plot is extremely conservative. The buckling plot appears to have merit because, even after a relatively few loadings of fuel, a critical loading close to actual criticality was extrapolated. In addition, the log plot and square root plot indicated the critical loading more closely than the linear plots. When near criticality, all four plots indicated similar values.

It was concluded that a shaping factor (square root, log, radial buckling) should be used to extrapolate criticality, especially during the first few fuel loading increments.

A general summarization of the data obtained in the critical test is presented in Table V.

C. POINT "B" AND POINT "A"

Following minimum criticality, the reactor flux level was increased to 20, 40, 60, 80 and 100 times the minimum value in an attempt to determine point "B". Analysis of the data obtained revealed that the procedure assumption that point "B" would fall somewhere between 20 and 100 times source critical was in error and that it actually occurred between 2, 3, 4, 5, 6, 8 and 10 times the minimum critical value. Figure 1158-10 displays the results and shows that establishing point "B" as about seven times minimum criticality is proper.

TABLE V

SUMMARY OF WET CRITICAL TEST DATA

	Total number of fuel elements loaded (N)	23	30	38	41	46	49	51	52	53	
	Number of fuel elements loaded this step	23	7	8	3	5	3	2	1	1	
	Channel of last fuel element loaded	C-37	C-53	C-43	C-40	C-70	C-60	C-47	C-59	C-71	
	Square root of N	4.79	5.478	6.164	6.404	6.7823	7.000	7.1414	7.2111	7.2801	
	Log of N	1.3617	1.47712	1.57978	1.6128	1.66276	1.69020	1.70757	1.716	1.7243	
	Factor proportional buckling (B_n)	9.647	7.812	6.444	6.065	5.5163	5.2146	5.0413	4.9601	4.8799	
All Control Rods In	Count rate in channel XI (BF_3)	185.7	279.0	628	713.2	962.02	1045.3	1079	1076	1091.2	
	Count rate in channel XII (BF_3)	153.01	272.0	405.7	457.6	664.82	694.7	680	724	717.5	
	Count rate in channel XIII (BF_3)	69.2	99.6	198	217.5	306.34	317.6	317	318	331.0	
	Average count rate	135.9	216.9	410.6	462.8	644.39	685.9	692.3	706	713.2	
	Inverse of average count rate	.00735	.00461	.00244	.00216	.00155	.00146	.001445	.00142	.00140	
	Count rate source channel I	0	0	0	0	0	0	0	0	0	
	Count rate source channel II	0	0	0	0	0	0	0	0	0	
	Core temperature (F)	368	358	-	358	345	343	350	348	348	
	Time and date	⁰²⁰⁵ 8-17-62	¹⁵²⁸ 8-19-62	-	⁰¹⁰⁰ 8-23-62	¹⁵⁰⁰ 8-23-62	⁰⁶³⁰ 8-24-62	¹⁶¹⁵ 8-24-62	¹⁹²⁰ 8-24-62	⁰³³⁰ 8-25-62	
	Reactor Sodium Level (Elevation)	1426' 6-1/2"	1426' 10-1/2"	-	1426' 4-1/2"	1427' 2-1/4"	1426' 6-1/2"	1426' 6-1/2"	1426' 6-1/2"	1426' 7"	
Gangs 1, 2, 3 Out; 4 & 5 50 %	Count rate in channel XI (BF_3)	809	1440	4073.7	5034.2	8197.6	11476	14729	15270	16575	
	Count rate in channel XII (BF_3)	572	1214	2266.7	2784.6	4873.9	6348	7360	8278	8806	
	Count rate in channel XIII (BF_3)	180	302.1	523.7	879.4	1452.0	1721	2008	2155	2398	
	Average count rate	520	985.6	2288.0	2899.4	4841.2	6515	8032	8568	9260	
	Inverse of average count rate	.00192	.00101	.000437	.000345	.000207	.0001545	.0001245	.000117	.000108	
	Count rate source channel I	0	0	-	-	-	0	0	0	0	
	Count rate source channel II	0	0	-	-	-	0	0	0	0	
	Core temperature (F)	360	358	-	358	345	343	350	348	348	
	Time and date	²¹⁵⁰ 8-17-62	²²⁴⁰ 8-19-62	-	⁰⁴⁰⁰ 8-23-62	¹⁶¹⁰ 8-23-62	⁰⁶³⁰ 8-24-62	¹⁷⁰⁰ 8-24-62	²²²⁷ 8-24-62	⁰⁷³⁰ 8-25-62	
	Reactor Sodium Level (Elevation)	1427' 2-1/4"	1426' 10-1/2"	-	1426' 4-1/2"	1426' 5-1/2"	1426' 6-1/2"	1426' 6-1/2"	1426' 6-1/2"	1426' 7-1/4"	
All Control Rods Out	Count rate in channel XI (BF_3)	1246	2508	8951.5	12315.8	27363	65194	215285	747388	-	
	Count rate in channel XII (BF_3)	827	2015	4814.7	6449.6	15775	35224	103470	386429	-	
	Count rate in channel XIII (BF_3)	240	438.7	941.75	1762.7	4085.5	8031	24407	92958	-	
	Average count rate	771	1654	4902.65	6842.7	15741	36149	114387	408925	-	
	Inverse of average count rate	.00129	.000604	.000204	.000146	.0000635	.0000276	.000008742	.000002445	-	
	Count rate source channel I	0	0	0	-	0	0	0	0	-	
	Count rate source channel II	0	75 cpm	0	-	0	0	0	0	-	
	Core temperature (F)	360	358	366	358	345	343	350	348	-	
	Time and date	²²³⁰ 8-17-62	²³³⁰ 8-19-62	¹⁵¹² 8-22-62	⁰⁴¹⁵ 8-22-62	¹⁶³⁰ 8-23-62	⁰⁶³⁰ 8-24-62	²⁰⁰⁰ 8-24-62	⁰⁰⁴⁵ 8-25-62	-	
	Reactor Sodium Level (Elevation)	1427' 2-1/2"	1426' 10-1/2"	1426' 4-1/2"	1426' 4-1/2"	1426' 5-1/2"	1426' 6-1/2"	1426' 6-1/2"	1426' 6-1/2"	-	

been eliminated from the tabulation and marked X. In these cases, data which were obtained from the movements of adjacent points provided sufficient information.

The comparison of the computed and measured displacements took place at the values obtained at 540 F temperature. The figures for lower temperatures were too small as compared with the accuracy of the readings (approximately $\pm 1/8$ inch), and therefore these values were used for control purposes only, to ascertain the continuity of the movements. The analysis of the displacements is performed in three groups according to the areas covered and the information obtained from the readings.

- 1) Primary sodium pump suction lines in the vicinity of the reactor cavity diaphragm seals outside the cavity. These measurements were mainly taken to check the axial movement of the junction points between the sodium pipe and the guard pipe provided for the piping section between the reactor and the blocking valve. These readings should prove that the restraint on the stainless steel sodium pipe imposed by the carbon steel guard pipe is consistent with values used for the purpose of the original stress calculations, and thereby the validity of the stress analysis and the integrity of the system is verified.

All the east-west displacement measurements performed at these pipe sections are within one-eighth inch of the computed values. This amount of deviation is by all means acceptable, owing to the accuracy of this kind of measurement. The vertical movement of this piping was also within one-eighth inch of the expected displacement, with the exception of points No. 9, 12, and 14. These points were inspected after the test and the reference points on the insulation cladding had been obviously displaced by crushing of the sheet metal enclosure by maintenance or operation activities. Subsequent checking

of the movements during cooldown showed that the amount and direction of movement is in agreement with the expected movement at each of these points.

The axial movements of the junction points were also observed during the cooldown. These movements were slightly below the expected values. Although, due to the comparatively small temperature range, this difference is not significant, the observation of these movements by the already installed remote reading devices will give valuable information during future high temperature operation.

- 2) Piping adjacent to the primary and secondary sodium pumps. All pointers attached to the elbows below the pump suction at the center line of the pump indicated no horizontal movement with the rise of temperature. Consequently, no deformation of the pump casing resulted at any of the pumps observed. Some readings on the suction and discharge lines of the primary pumps in the immediate vicinity of the pump nozzle showed movements contradictory to the adjacent measuring points at the pump suction elbow and points located farther away from the pump along the same lines. A subsequent inspection of the measuring devices, which were attached to pressure transducers, indicated that the provisions were not sufficiently solid to offer acceptable readings. However, readings on adjacent points on the line confirmed the conclusion concerning the integrity of the pumps. Furthermore, the remote reading devices, which have already been installed, will record any possible movements of the pump casing at higher temperatures.
- 3) Intermediate measurement points. Observation of these points was mainly stipulated in order to obtain evidence that no obstructions exist which may restrain the anticipated free movement of piping and which could impose undue stress and strain to equipment and piping. All measurements taken at these points are within one-fourth inch of the expected values. These deviations are within the limits of accuracy of computation and measurement. Furthermore, any movement of this order of magnitude at

these points is at a considerable distance from any equipment, and does not have appreciable affect on the stresses.

Readings at point No. 53 were eliminated, since the inspection at the time of this analysis showed that the measuring devices were installed on an adjacent pipe line.

Readings of vertical movements of these points are not shown in the tabulation because these movements were affected by the adjustment of the variable load spring hangers and are discussed in the corresponding section of this report.

A remote pipe displacement transmitter-recorder system was completed and installed during the period the system was being maintained at 540 F. Six locations were instrumented for remote determination of displacement, as follows:

- 1) At each of the primary pumps to determine east-west and north-south displacement.
- 2) At each of the primary reactor outlet lines on the pump side of the outer bellows on the guard pipe to determine east-west and vertical displacement.

During cooldown, data were taken from the displacement recorder and compared with the measurements taken at the corresponding points. Displacements obtained from the remote transmitters related to the measured displacements within the accuracy of the measurement methods.

B. PIPING INSPECTION

The general inspection of sodium heat transfer system piping, performed concurrently with the taking of piping displacement measurements, determined the following:

- 1) Some of the variable load spring hangers require adjustments.
- 2) Some of the constant hanger supports for piping inside the reactor cavity did not appear to follow the expected pipe movement.
- 3) There was piping vibration in the secondary loops with high sodium flow rates.
- 4) There were piping interferences.

All defects mentioned above were corrected as required for continued performance of the hot circulation test. The discussions below describe the details of the problem areas and remedial action taken in each case.

1. Variable Load Spring Hangers

Figures 1167-B-1 through 1167-B-13 in Appendix B display the locations of sodium heat transfer system variable spring hangers.

The final adjustment of the spring hangers will be made at 350 F to show a hanger loading which will result in the design load when the anticipated pipe movement (obtained from an extrapolation of measured movement) is achieved at operating temperature. Appendix B shows the design data and the conditions of adjustment for each hanger in the primary and secondary heat transfer systems. These adjustments will be accomplished prior to initiation of the wet critical test.

2. Constant Load Spring Hangers

Movement of the constant support hangers for piping inside the reactor cavity was of special significance during warm-up to 540 F sodium temperature, because the piping could not be visually observed to determine proper movement due to thermal expansion. Eleven of the seventeen hangers for the cavity piping

could not be definitely verified as having moved upward during the increase to 540 F. Since only the spring assembly is accessible when the reactor cavity is sealed the reason of malfunction could not be established. The cavity access opening covers were removed and the cavity was entered in order to inspect the piping, the hanger structures, and the hanger rod penetrations through the concrete shielding. The inspection revealed that two hangers were obviously not carrying any piping load, while others appeared to be operating inadequately due to the lack of free movement of the hanger rods in the shielding penetration sleeves. Disassembly of the malfunctioning hangers indicated that the binding in the sleeves was due to one or both of the following reasons:

- a) The key which was provided to prevent rotation of the rod and thereby damage to the bellows seals, was made of square stock material and the sharp corners dug into the key-way in the sleeve.
- b) Rust accumulation in the sleeves caused by water penetration during construction.

In order to assure proper operation of the constant load hanger supports, all 17 assemblies were disassembled, repaired, cleaned and reinstalled. A detailed list of work performed on subject hangers is contained in Appendix D.

Operation of the hangers was observed during the cooldown period; the results obtained are presented in Appendix B. Comparison of data indicates that all hangers moved in the direction anticipated, but varied somewhat from the listed "expected" values. Measured movements are taken from hangers on the guard pipe, and are developed by the combined effects of the sodium line and guard pipe. Results may be expected to vary from line to line, depending on initial conditions established during installation.

3. Piping Vibration

Some piping vibration was observed in the secondary sodium loop piping at elevated flows (above 500 rpm pump speed). The vibration was investigated and determined to be the result of helium gas entrainment in the sodium. The loop expansion tank served as the source of the helium. As pump speed was increased past 500 rpm, the expansion tank level was observed to increase and tank pressure decrease, indicating the entraining of gas. The piping will be modified following completion of hot circulation testing. Namely, an expansion tank bypass line will be installed which will bypass approximately 96% of the flow around the tank, thus decreasing the turbulence therein and eliminating the gas entraining process.

The secondary pumps were limited to a speed of 500 rpm except when testing operations were being performed, or unless hourly inspection tours of IHX cells and steam generator rooms were performed.

Gas entrainment and the resulting pipe vibrations will be verified and examined in detail as a part of preoperational test AI-P-1103, "Main Heat Transfer System". This test will be completed following the planned modifications.

During the period when rework of the reactor cavity pipe hangers was being performed, motion of the reactor return lines was observed. Analysis of the condition was delayed until after completion of the hanger rework. In primary loops No. 1 and 2, with 70% flow and the throttle valves approximately 52% closed (to simulate core drop), an erratic motion of the reactor return line occurred. Peak-to-peak displacements of less than one-sixteenth inch were measured in the east-west runs of the pipes. This was accompanied by

a high level hydraulic noise. Upon opening the throttle valve in loop No. 1 and re-establishing 70% flow, the hydraulic noise level and line motion decreased to an insignificant level. In loop No. 2, approximately 30% flow was established with the throttle valve open with the same observed decrease in hydraulic noise and pipe motion (it was not possible to operate the loop at higher flow without obtaining a low pump case sodium level alarm due to lack of back pressure in the loop).

It was concluded from the observations noted above that the motion originally observed was due to turbulence in the reactor return lines, caused by use of the throttling valves to simulate a loaded core pressure drop.

4. Piping Interferences

Certain piping interferences were determined to exist by the investigation of the system piping as a part of preoperational test AI-P-1103, "Main Heat Transfer System". The interferences observed are listed in Appendix C. The problem areas were corrected, or are being corrected as required. No additional interferences developed during the performance of the test, nor are further interferences expected to occur at operating temperatures.

C. RAISING SODIUM TEMPERATURE TO 440 F

Increase in sodium temperature from 350 F to 440 F was accomplished in a total of 31 hours, for an average rate of 2.9 F per hour. A combination of the electrical preheaters and the heating capacity of the six main heat transfer system sodium pumps was utilized for the operation. Heat input from the primary pumps was approximately 90 kw each, from the secondary pumps

approximately 55 kw each, and approximately 300 kw from the preheat system electrical heaters. Figure 1167-9 depicts the sodium temperature with respect to time for the duration of the heatup to 440 F.

Upon attaining 440 F, pump speeds were decreased to 20% to start hand control station auto-to-manual transfer testing. It was determined that system temperature decreased rapidly (~ 2 F per hr) when the pumps were not operating at higher speeds. Thus, it was not necessary to employ the Electromatic relief valves to maintain temperature. This was desirable, as the relief valves temperature control deadband was expected to be such that close temperature control would be difficult.

Evaporator and superheater shell temperatures were monitored during heat-up to avoid temperature differences greater than 50 F between sodium and evaporator or superheater. Nearly isothermal temperatures existed throughout the test. Evaporator feedwater make-up was required for sampling losses only.

D. TRANSFER OF LOOP HAND CONTROL STATIONS AT 440 F

The six main sodium pumps were switched from manual control to automatic control at 15% speed, with a system temperature of 440 F. The switchover was accomplished smoothly. Table II presents the operating conditions immediately prior to transfer from manual to automatic control and also the conditions immediately following the transfer. The values listed indicate that a minor "bump" occurred during the transfer. Subsequent review of the recorded data and the control system revealed that the bump was due to the combined effect of the primary sodium flow control system temperature trim and mismatch between primary and secondary flow control systems.

Figure 1167-9 Raising Sodium Temperature to 440F

TABLE II

HCS TRANSFER TEST DATA AT 440 F

Pump No.	Variable	Units	Before Transfer	After Transfer
1 Primary	Flow	millivolts	5	4
	Speed	rpm	125	107
	Delta P	psig	0.5	0.2
2 Primary	Flow	millivolts	5	4
	Speed	rpm	127	113
	Delta P	psig	1.5	1.5
3 Primary	Flow	millivolts	5	4
	Speed	rpm	133	127
	Delta P	psig	2	2
1 Secondary	Flow	millivolts	5	5
	Speed	rpm	147	130
	Delta P	psig	2	1
2 Secondary	Flow	millivolts	5	5
	Speed	rpm	150	135
	Delta P	psig	2.5	2.3
3 Secondary	Flow	millivolts	4	5
	Speed	rpm	145	150
	Delta P	psig	1	1.3

Minor tuning of the flow controllers plus elimination of the temperature trim effect (by placing TRC-101, -201, and -301 on "Manual") eliminated the bump (see Table XIV).

E. ESTABLISHING PRESSURE DROP AT 440 F

A pressure drop of 38.4 psig was established across each primary pump to simulate expected core pressure drop in preparation for system hydraulic testing. In all three loops, the throttle valve position required to develop the necessary pressure drop was 60% - closed. Primary pump speeds were as follows when the throttle valve positions were established (these speeds were calculated to be the speeds necessary for 100% plant design flow conditions. Corrected assumptions based on a better understanding of the system are discussed in Section III. R of this report):

- 1) Primary Pump No. 1 - 675 rpm
- 2) Primary Pump No. 2 - 641 rpm
- 3) Primary Pump No. 3 - 641 rpm

Data shown in Table III were obtained with the six main sodium pumps at 100% speed. This information established the base data for system testing at 440 F.

TABLE III
440 F BASE DATA

Pump	Flow (mv)	Speed (rpm)	Inlet Press (psig)	Outlet Press (psig)	ΔP (psi)
Primary No. 1	28	678	-2.0	36.5	38.5
2	29	641	-2.0	36.5	38.5
3	27	641	-2.8	35.8	38.6
Secondary No. 1	29	675	16.5	61.0	45.0
2	28	675	16.0	59.0	43.0
3	26	675	17.0	61.0	44.0

F. PUMP SPEED INCREASES AT 440 F

The following facts concerning interaction of the pumps during speed increases were verified by observation of the main control board instruments:

- 1) With all primary pumps operating at the speed corresponding to 15% steam demand, any one pump can be increased to the speed corresponding to 30% of steam demand without affecting the flow through the other two loops.
- 2) The same results were obtained when increasing each loop to the speed corresponding to 50% steam demand with the other two loops at 30% steam demand.
- 3) The same results were obtained when increasing each loop to that speed corresponding to 100% steam demand (from 50%) with the remaining loops at the speed corresponding to 50% steam demand.

G. FLOW TRIMMING AT 440 F

The output signals from load computers FCCp-501, -502, -503, -504, -505, and -506 were trimmed to equalize all six flows when the loop steam demand hand control stations (HCS-501, -502, and -503) were set at 60% scale. Flow control was then transferred to base load controller HCS-504, and it was visually verified that the six loop flows equalized at the same value for various settings of HCS-504 (steam demand hand control station).

H. DETERMINATION OF SYSTEM HYDRAULIC CHARACTERISTICS AT 440 F

The system hydraulic characteristics determination portion of the hot sodium circulation test consisted of two areas of testing:

- 1) Operating all six loops as a unit (from plant steam demand controller, HCS-504) from 100% of plant design flow down to approximately 15% flow.

- 2) Operating each circuit on manual control, individually, with the remaining two circuits on automatic control, to verify proper automatic flow compensation when flow in the circuit on manual control was varied.

The results of these tests verify that circuit flows and pump speeds are stable over the entire operating range and that circuits on automatic control compensate for variations imposed on the system by operation of one loop on manual control.

Typical plots of pump speeds and developed sodium pressures as a function of loop flow at 440 F for one primary and one secondary loop are presented in figures 1167-10 and -11. Flow ramps were imposed to verify that all circuits changed flows smoothly.

The compensating ability of circuits on automatic control from 60% steam demand to 75% steam demand (HCS -501, -502, and -503), was demonstrated by establishing two circuits at 70% steam demand on automatic control and varying the third circuit from 70% to 60% to 75% and back to 70%. The two circuits on automatic control came to the position required for compensation. Table IV shows the manner in which the manipulations were accomplished and the results obtained. Errors in control system tuning are reflected in the table. These were corrected prior to performance of the corresponding section at 540 F (see Table XVI).

I. RAISING SODIUM TEMPERATURE TO 540 F

Increase in sodium temperature from 440 F to 540 F was accomplished in a manner similar to that discussed in section III. C of this report. The temperature increase required a total of 58 hours, corresponding to an average rate of 1.7 F per hour. Figure 1167-12

TABLE IV
AUTOMATIC FLOW COMPENSATION
AT 440 F

Circuit De- viating (on Manual)	Manual HCS Setting	PRIMARY FLOWS			Total Pri Flow (mv)	SECONDARY FLOWS			Total Sec Flow (mv)
		Ckt 1 Flow (mv)	Ckt 2 Flow (mv)	Ckt 3 Flow (mv)		Ckt 1 Flow (mv)	Ckt 2 Flow (mv)	Ckt 3 Flow (mv)	
No. 1, HCS-501	60%	20.2	26.2	26.0	72.4	21.6	27.2	28.5	77.3
No. 1, HCS-501	75%	25.8	27.4	23.6	76.8	27.8	24.4	25.8	78.0
No. 2, HCS-502	60%	25.5	20.8	26.9	72.2	27.4	21.6	28.5	77.5
No. 2, HCS-502	75%	23.4	26.3	24.3	74.0	24.5	27.3	25.6	77.4
No. 3, HCS-503	60%	26.8	27	19.8	73.6	28.0	28.0	21.4	77.4
No. 3, HCS-503	75%	23.6	24.2	25.3	73.1	25.3	25.4	26.8	77.5

Figure 1167-10 Hydraulic Characteristics at 440 F
Primary Loop No. 2

Figure 1167-11 Hydraulic Characteristics at 440 F
Secondary Loop No. 2

depicts the sodium temperature with respect to time for the duration of the heat-up.

Upon attaining 540 F, pump speeds were decreased to 20% to start hand control station auto-to-manual transfer testing. System temperature decreased rapidly (~ 2 F/hr) when the pumps were not operating at the higher speeds. Thus, as at 440 F, it was not necessary to employ the Electromatic relief valves to maintain temperature. It was necessary to interrupt the test occasionally, to operate the main pumps at higher speeds in order to maintain the system at 540 F.

Evaporator and superheater shell temperatures were monitored during heat-up to avoid excessive temperature differences. Essentially isothermal conditions were maintained throughout the test. Evaporator feedwater make-up was required for sampling losses only.

J. TESTING AT MAXIMUM SODIUM TEMPERATURE

Piping displacement measurements were made at 540 F sodium temperature and 20% flow. Results were previously discussed in III. A and B of this report.

Performance runs were made at 20%, 50% and 100% design speeds on each of the six main sodium pumps. Secondary pump No. 3 could not be operated at 100% design speed (675 rpm) because of a pump sodium high level alarm which actuated above 660 rpm until the sodium level was lowered. (Note: This is attributed to the expansion tank arrangement problem discussed earlier and will be corrected by the planned modification.) The following items were observed during the performance runs:

- 1) No adverse internal noise or rumble in the pumps. A high pitched squeal was observed in the vicinity of all pumps under certain condi-

Figure 1167-12. Raising Sodium Temperature to 540°F

tions of flow and speed. The sound appears to be speed and flow dependent; that is, it exists only at certain combinations of speed and flow. The sound does not translate to the single pump thrust bearing and it is concluded, therefore, that it does not originate in the fluid bearing. In one instance, the sound was due to motion of the pump preheat furnace. Careful consideration was given to this area by site personnel and specialists from Atomics International headquarters in Canoga Park, California, and it is concluded that no deleterious condition exists.

- 2) No vibration of pump assemblies.
- 3) A 10 rpm speed oscillation was observed in the pumps. This is covered in Section II. O of this report.
- 4) Satisfactory stopping and starting of pumps.
- 5) The thrust bearing oil circuit lower seal in primary pump No. 2 and secondary pump No. 1 developed oil leaks which permitted oil to be discharged into the first (of two) oil sump below the seal housing. The condition in the secondary pump was erratic and has since ceased. The primary pump seal will be disassembled for inspection and maintenance.

No further malfunctions were observed in the oil system operation. Periodic observations were made of each system including the oil pump, cooling fan, seals and piping.

Each pump was started and stopped three times at 540 F. There was no difficulty encountered in the stopping or starting operations, indicating proper functioning. Pump and drive data recorded at 540 F had also been recorded, under the same flow conditions, at 340 F. Tables V through X show the comparison of the values at the two temperatures.

The primary throttle valves (three main loop and one moderator loop valve) were operated from the full-open to the 95% closed position with sodium temperature at

TABLE V
PRI Na PUMP NO. 1 PERFORMANCE TESTS AT 340F AND 540F

Items	20% Speed		50% Speed		100% Speed	
	340F	540F	340F	540F	340F	540F
<u>PUMP DATA:</u>						
Na Temp (°F) TR-2-4	340	535	340	530	341	528
Suction Pressure (psig)	+0.7	+ .8	0	+ .2	-2.5	-2
Disch Pressure (psig)	1.5	2.2	8.5	+9	36	+36
TI-116						
(Internal Pump -1.	225	390	230	385	239	390
Thermal Barrier -2.	63	75	65	70	65	75
Temperature- °F) -3.	63	79	65	72	66	75
-4.	215	355	215	355	215	355
-5.	210	352	215	350	213	355
-6.	65	76	67	72	69	75
-7.	65	75	67	76	67	75
-8.	225	379	230	375	233	380
High Level Alarms	NONE	NONE	NONE	NONE	NONE	NONE
FIN Temps (°F) 1.	135	170	137	172	135	172
2.	118	146	118	147	118	147
3.	130	167	130	168	130	168
4.	102	116	102	116	101	116
5.	217	327	217	333	216	329
6.	158	269	158	221	159	221
7.	187	278	188	278	188	278
8.	137	205	137	206	137	206

TABLE V - Continued

Item	20% Speed		50% Speed		100% Speed	
	340F	540F	340F	540F	340F	540F
Seal Oil Temp (° F)	103	107	105	116	109	119
Throttle Valve Pos % Closed	50	60	50	60	50	60
Loop Flow * (lb/hr)	16x3x10 ⁴	17x3x10 ⁴	43x3x10 ⁴	40x3x10 ⁴	81x3x10 ⁴	78x3x10 ⁴
<u>DRIVE DATA:</u>						
Pump Speed (rpm)	135	160	335	340	675	675
Coupling Coil Temp (° F)	58	80	58	72	65	77
Motor Input Volts Amps	4040 18.7	4220 21	4080 20.6	4200 16	4080 36.5	4200 35.5
TI-108						
(Motor Coil -1.	85	108	89	91	96	113
Temperature-°F) -2.	26	20	30	21	43	21
-3.	82	100	86	87	93	110
-4.	103	116	108	100	114	123
-5.	79	100	84	86	91	107
-6.	78	99	84	83	93	106
Coupling Current (amps)	0.4	0.5	1.15	1.2	2.65	2.6

* As obtained from FRC-101, Primary Loop No. 1 Sodium Flow Recorder-Controller.

TABLE VI
PRI Na PUMP NO. 2 PERFORMANCE TESTS AT 340F AND 540F

Items	20% Speed		50% Speed		100% Speed	
	340F	540F	340F	540F	340F	540F
<u>PUMP DATA:</u>						
Na Temp (°F) TR-2-4	340	536	340	530	341	528
Suction Pressure (psig)	+0.3	+.4	-0.2	-.2	-2.3	-2
Disch Pressure (psig)	2.0	3.8	10	10.6	36	+37
TI-116						
(Internal Pump -1.	235	400	237	390	238	395
Thermal Barrier -2.	60	73	63	65	63	70
Temperature- °F) -3.	60	75	63	70	63	70
-4.	200	340	207	335	201	335
-5.	195	323	196	315	199	320
-6.	60	75	63	70	64	75
-7.	60	73	60	65	62	70
-8.	233	400	235	395	237	400
High Level Alarms	NONE	NONE	NONE	NONE	NONE	NONE
FIN Temps (°F) 1.	133	181	134	181	133	182
2.	117	149	117	148	117	148
3.	135	176	135	177	136	177
4.	114	141	114	140	114	140
5.	205	331	204	329	204	328
6.	159	232	160	232	160	230
7.	210	321	209	319	208	318
8.	156	259	156	218	155	218

TABLE VI - Continued

Items	20% Speed		50% Speed		100% Speed	
	340F	540F	340F	540F	340F	540F
Seal Oil Temp (° F)	95	102	95	107	94	104
Throttle Valve Pos % Closed	54	60	54	60	54	60
Loop Flow* (lb/hr)	$15 \times 3 \times 10^4$	$19 \times 3 \times 10^4$	$43.5 \times 3 \times 10^4$	$42 \times 3 \times 10^4$	$80 \times 3 \times 10^4$	$78 \times 3 \times 10^4$
DRIVE DATA:						
Pump Speed (rpm)	135	168	335	340	640	640
Coupling Coil Temp (° F)	60	85	65	77	68	82
Motor Input Volts Amps	4040 18.6	4220 21	4080 21	4200 21.5	4080 36.5	4200 36
TI-108 (Motor Coil -1. Temperature- ° F) -2. -3. -4. -5. -6.	80 75 83 76 80 75	99 91 102 95 99 94	84 80 90 83 85 83	85 80 90 82 87 83	94 88 100 92 96 93	113 100 116 104 110 107
Coupling Current (amps)	0.55	0.6	1.45	1.4	2.9	2.8

* As obtained from FRC-201, Primary Loop No. 2 Sodium Flow Recorder-Controller.

TABLE VII
PRI Na PUMP NO. 3 PERFORMANCE TESTS AT 340F AND 540F

Items	20% Speed		50% Speed		100% Speed	
	340F	540F	340F	540F	340F	540F
<u>PUMP DATA:</u>						
Na Temp (°F) TR-2-4	340	536	340	530	341	528
Suction Pressure (psig)	-0.2	0	-0.9	-.8	-3.8	-2.5
Disch Pressure (psig)	2.0	3	9	8.5	37.8	+35.5
TI-116						
(Internal Pump -1.	235	401	240	372	240	385
Thermal Barrier -2.	70	70	70	65	70	70
Temperature- °F)-3.	70	70	70	65	70	70
-4.	100	168	101	155	113	150
-5.	105	139	105	135	110	135
-6.	95	112	95	105	93	110
-7.	75	77	75	75	75	75
-8.	113	137	114	135	114	135
High Level Alarms	NONE	NONE	NONE	NONE	NONE	NONE
FIN Temps (°F) 1.	117	188	117	188	116	188
2.	100	127	100	127	100	126
3.	115	161	114	162	115	162
4.	113	140	113	140	112	141
5.	213	330	213	329	213	329
6.	132	214	135	215	134	212
7.	213	330	213	329	215	329
8.	155	226	154	226	154	227

TABLE VII - Continued

Items	20% Speed		50% Speed		100% Speed	
	340F	540F	340F	540F	340F	540F
Seal Oil Temp (°F)	100	106	103	114	107	117
Throttle Valve Pos % Closed	52	65	52	65	52	65
Loop Flow * (lb/hr)	$14 \times 3 \times 10^4$	$20 \times 3 \times 10^4$	44	$40 \times 3 \times 10^4$	$83 \times 3 \times 10^4$	$78 \times 3 \times 10^4$
DRIVE DATA:						
Pump Speed (rpm)	135	190	338	340	673	640
Coupling Coil Temp (°F)	67	98	75	83	77	90
Motor Input Volts Amps	4040 18.9	4220 23	4080 22.5	4200 23	4080 43	4200 39
TI-108 (Motor Coil -1. Temperature- °F)-2. -3. -4. -5. -6.	112 110 107 122 110 115	126 126 123 138 127 129	117 117 113 128 117 121	118 117 113 128 117 121	131 133 128 146 133 136	134 135 132 147 137 138
Coupling Current (amps)	0.25	0.5	1.05	1.0	2.70	2.4

* As obtained from FRC-301, Primary Loop No. 3 Sodium Flow Recorder-Controller.

TABLE VIII
SEC Na PUMP NO. 1 PERFORMANCE TESTS AT 340F AND 540F

Items	20% Speed		50% Speed		100% Speed	
	340F	540F	340F	540F	340F	540F
<u>PUMP DATA:</u>						
Na Temp (°F) TR-2-4	340	535	340	530	340	528
Suction Pressure (psig)	20.5	22	19.7	19.5	16.5	16.5
Disch Pressure (psig)	22	24.5	30.2	31	61	60
TI-116						
(Internal Pump -1.	145	235	151	215	157	220
Thermal Barrier -2.	175	290	187	275	192	285
Temperature- °F)-3.	260	445	265	420	271	420
-4.	270	465	280	440	282	440
-5.	270	455	275	430	278	430
-6.	245	412	250	380	250	380
-7.	155	210	160	208	160	208
-8.	130	175	140	175	140	173
High Level Alarms	NONE	NONE	NONE	NONE	NONE	NONE
FIN Temps (°F)1.	141	161	140	165	135	163
2.	140	186	138	185	136	187
3.	111	121	110	127	107	127
4.	127	138	123	146	121	143
5.	199	241	193	257	190	240
6.	191	298	192	295	193	298
7.	124	156	122	148	121	146
8.	143	171	138	165	139	163

TABLE VIII - Continued

Items	20% Speed		50% Speed		100% Speed	
	340F	540F	340F	540F	340F	540F
Seal Oil Temp (°F)	107	113	110	124	114	127
Throttle Valve Pos % Closed	0	0	0	0	0	0
Loop Flow * (lb/hr)	$15 \times 3 \times 10^4$	$21 \times 3 \times 10^4$	$46 \times 3 \times 10^4$	$43 \times 3 \times 10^4$	$88 \times 3 \times 10^4$	$86 \times 3 \times 10^4$
DRIVE DATA:						
Pump Speed (rpm)	130	175	335	340	675	675
Coupling Coil Temp (°F)	93	138	105	126	108	132
Motor Input Volts Amps	4040 18.5	4200 20.5	4080 21	4200 22	4020 40.5	4200 40.5
TI-108						
(Motor Coil -1.	122	163	134	155	148	183
Temperature- °F)-2.	135	177	148	170	160	200
-3.	125	170	138	160	150	188
-4.	130	173	142	163	155	192
-5.	132	176	146	167	157	193
-6.	127	172	139	160	150	188
Coupling Current (amps)	0.5	0.7	1.45	1.4	3.35	3

* As obtained from FRC-102, Secondary Loop No. 1 Sodium Flow Recorder-Controller.

TABLE IX
SEC Na PUMP NO. 2 PERFORMANCE TESTS AT 340F AND 540F

Items	20% Speed		50% Speed		100% Speed	
	340F	540F	340F	540F	340F	540F
<u>PUMP DATA:</u>						
Na Temp (°F) TR-2-4	330	527	330	530	330	528
Suction Pressure (psig)	20	20.8	19	19	16	15.8
Disch Pressure (psig)	22	24.5	30.2	30	60	59
TI-116						
(Internal Pump -1.	130	215	135	215	140	217
Thermal Barrier -2.	150	270	160	273	165	275
Temperature- °F)-3.	275	485	280	470	285	470
-4.	275	485	280	465	239	465
-5.	278	480	285	460	240	460
-6.	275	412	280	400	235	400
-7.	215	240	220	240	222	240
-8.	170	210	175	190	183	190
High Level Alarms	NONE	NONE	NONE	NONE	NONE	NONE
FIN Temps (°F) 1.	119	147	118	148	117	148
2.	131	185	130	192	130	191
3.	122	132	120	134	119	132
4.	156	167	155	167	153	164
5.	185	293	185	303	186	300
6.	132	217	129	220	127	214
7.	132	202	128	186	127	129
8.	207	233	206	245	205	241

TABLE IX - Continued

Items	20% Speed		50% Speed		100% Speed	
	340F	540F	340F	540F	340F	540F
Seal Oil Temp ($^{\circ}$ F)	105	114	110	124	115	127
Throttle Valve Pos % Closed	0	0	0	0	0	0
Loop Flow * (lb/hr)	$15 \times 3 \times 10^4$	$18 \times 3 \times 10^4$	$46.2 \times 3 \times 10^4$	$43 \times 3 \times 10^4$	$91 \times 3 \times 10^4$	$87 \times 3 \times 10^4$
DRIVE DATA:						
Pump Speed (rpm)	130	175	335	340	678	675
Coupling Coil Temp ($^{\circ}$ F)	82	140	93	125	100	132
Motor Input Volts Amps	4040 18	4220 20.7	4080 20.7	4200 21.5	4020 41	4200 40.5
TI-108 (Motor Coil -1. Temperature- $^{\circ}$ F)-2. -3. -4. -5. -6.	120 124 117 122 117 124	165 172 165 169 164 171	130 135 127 131 127 135	156 159 150 156 150 160	147 150 144 152 143 155	180 197 177 183 175 184
Coupling Current (amps)	0.45	0.6	1.4	1.4	3.2	2.6

* As obtained from FRC-202, Secondary Loop No. 3 Sodium Flow Recorder-Controller.

TABLE X
SEC Na PUMP NO. 3 PERFORMANCE TESTS AT 340F AND 540F

Items	20% Speed		50% Speed		100% Speed	
	340F	540F	340F	540F	340F	540F
<u>PUMP DATA:</u>						
Na Temp (°F) TR-2-4	340	530	340	530	340	528
Suction Pressure (psig)	19.8	20.1	19.2	19.2	16.5	16.5
Disch Pressure (psig)	20	22.5	28	29	60.2	57.5
TI-116						
(Internal Pump -1.	120	245	128	250	132	250
Thermal Barrier -2.	150	290	165	290	170	290
Temperature- °F)-3.	250	450	261	435	267	440
-4.	270	450	278	430	280	430
-5.	275	440	278	420	280	425
-6.	280	440	282	420	280	425
-7.	195	230	200	230	195	230
-8.	150	215	185	220	155	215
High Level Alarms	NONE	NONE	NONE	NONE	NONE	NONE
FIN Temps (°F) 1.	123	148	120	149	120	148
2.	129	212	129	219	131	218
3.	128	148	122	151	120	151
4.	151	192	150	192	149	192
5.	130	152	127	156	126	154
6.	163	300	170	306	180	298
7.	169	152	160	151	153	151
8.	205	290	202	239	202	234

TABLE X - Continued

Items	20% Speed		50% Speed		100% Speed	
	340F	540F	340F	540F	340F	540F
Seal Oil Temp (°F)	108	114	113	124	117	127
Throttle Valve Pos % Closed	0	0	0	0	0	0
Loop Flow * (lb/hr)	$13.3 \times 3 \times 10^4$	$21 \times 3 \times 10^4$	$42 \times 3 \times 10^4$	$43 \times 3 \times 10^4$	$85 \times 3 \times 10^4$	$86 \times 3 \times 10^4$
DRIVE DATA:						
Pump Speed (rpm)	130	190	335	340	675	660
Coupling Coil Temp (°F)	65		68		70	
Motor Input Volts Amps	4040 18	4200 20.3	4080 20.2	4200 21	4020 40.5	4200 39.5
TI-108						
(Motor Coil -1.	98	132	106	119	118	135
Temperature- °F)-2.	103	142	112	126	127	142
-3.	103	142	111	129	125	140
-4.	105	140	113	127	125	143
-5.	102	138	113	125	126	142
-6.	100	136	108	123	120	137
Coupling Current (amps)	0.6	0.7	1.55	1.50	3.55	2.2

* As obtained from FRC-302, Secondary Loop No. 3 Sodium Flow Recorder-Controller.

540 F and loop flows at 20%. The following operational discrepancies were observed:

- 1) No. 1 primary throttle valve became inoperable due to a sheared key on the jackshaft on one side of the operating yoke.
- 2) No. 3 primary throttle valve stroked only three-quarters of its travel and it was subsequently determined that binding within the valve was occurring.
- 3) The moderator throttle valve operator was observed to discharge oil from the air motor exhaust.

It was determined that the discrepancy noted in 1) above was caused by a maladjustment in the valve positioner. A faulty oil seal in the operator air motor was determined to be the cause of the oil discharge described in 3). Maintenance procedures are being developed for the purpose of investigating and repairing primary throttle valve No. 3.

With the exception of primary throttle valve No. 3, the above conditions were remedied by rechecking and (where necessary) re-aligning all valve positioners and by replacing the sheared key and oil seal after which one quick-closure cycle was performed on each valve, as described in part II., "Discussion of Test Performance". There was no excessive movement of valves or adjacent piping during the cycle. The following items were observed to be satisfactory during operation of the valves;

- 1) Air motor operation.
- 2) Line shaft operation.
- 3) Bearing support rigidity.
- 4) Component vibration.
- 5) Component lubrication.
- 6) Stem seals leakage.

7) Movement during quick-closure.

8) Mechanism clearances.

Each primary blocking valve was closed and re-opened one time at 540 F sodium temperature, with the pumps shutdown. Proper operation was determined by observing the same items as listed previously for the primary throttling valves.

Similarly, by observation of the same items, it was determined that the secondary throttling valves function satisfactorily.

K. CONTROL ROD DRIVE TESTING

Individual operation of control rods from the test cart revealed no significant effect on rod operability. Each rod was operated from the fully inserted position to 15 inches out, through various short up and down cycles (to obtain current and voltage data for each phase in each direction), back to the fully inserted position, and then to the fully withdrawn position. From this position, each rod was scrambled and drop times were measured.

Data obtained during control rod testing are presented in Table XI. Data obtained at ambient temperature during performance of preoperational test AI-P-1148 "Control Rod Operability" are included in the table under the heading "AMB" for comparison purposes. Of particular significance is the close correlation between the ambient and 540 F full insertion times, since these are timed free falls and reflect all changes in friction or binding.

There were no apparent differences in motor currents or voltages at 540 F, as compared to those obtained at ambient temperatures.

TABLE XI

CONTROL ROD OPERATION AT AMBIENT
AND AT 540 F SYSTEM TEMPERATURE

Rod No.	Withdrawal (min)		Average Speed (in./min)		Latch Release Time (msec)		Scram Time (msec) to:			
	AMB	540F	AMB	540F	AMB	540F	4 ft. Pos.	Full Insert.	AMB	540F
1.	12.4	12.2	12.52	12.79	51	50	768	765	960	960
2.	12.6	12.7	12.35	12.28	52	55	781	778	963	966
3.	12.5	12.35	12.52	12.63	51	50	770	779	960	955
4.	12.6	12.5	12.35	12.48	51	53	760	771	950	961
5.	12.8	12.6	12.20	12.38	50	55	783	783	971	975
6.	12.4	12.25	12.58	12.73	53	54	770.2	774	962	962
7.	12.3	12.75	12.70	12.24	52	53	769	772	967	969
8.	12.2	12.1	12.80	12.89	51	57	767	778	956	955
9.	12.5	12.55	12.44	12.43	52	56	771	794	978	969
10.	12.7	12.14	12.28	12.14	51	54	770	775	952	948
11.	12.4	12.55	12.57	12.43	50	52	768	768	956	960
12.	13.4	12.58	12.62	12.58	55	54	784	791	991	993
13.	12.7	12.4	12.24	12.58	51	58	772	772	955	964
14.	12.2	12.6	12.88	12.38	51	53	783	793	970	975
15.	12.7	12.65	12.24	12.33	51	54	773	775	946	953
16.	12.4	12.34	12.58	12.63	53	53	778.2	786	960	960
17.	12.7	12.45	12.30	12.53	53	53	774	778	947	943
18.	12.8	12.97	12.19	12.03	53	56	769	773	968	966
19.	12.4	12.65	12.60	12.33	52	52	767	768	957	961

L. TESTING CORE ALIGNMENT

A review of the test method concerning the testing of core alignment at 540 F sodium temperature, determined that the removal and re-insertion of a dummy element in the reactor was not a stringent enough test. This conclusion was based on the fact that a dummy element is 0.4 inch less in diameter than an actual fuel element. To supplement the check of core alignment, a process tube, hanger rod, and shield plug assembly (an actual fuel element with the fuel cluster removed) were inserted and removed from each of the three core positions which contained a dummy element. The test thus included:

- 1) Removal of a dummy element (previously installed)
- 2) Insertion of a mock fuel element
- 3) Removal of the mock fuel element
- 4) Insertion of the dummy element.

Observation of the fuel handling machine grapple load cells during element removal and insertion operations, determined that there was no binding nor hang-up of the dummy element, over the entire range of travel, from the fully inserted position to the travel position in the fuel handling machine.

The mock fuel element hung up during withdrawal from core positions C-185 and C-233, at a reading of about 353 on the fuel handling machine grapple position indicator. In each case, the element was lowered slightly and was withdrawn without further difficulty.

The position of "hang up" corresponds to the point previously experienced following the dry excess loading. At that time the "hang up" was due to a "flat" on the slip ring used to assemble the fuel cluster to the hanger rod. The slip rings were modified and further tests indicated the problem had been alleviated. The mock fuel

element used in the present test was equipped with a modified slip ring. Since this element did not contain a fuel cluster it was substantially lighter than a standard fuel element. Prior to its use, some consideration was given to the fact that the light process tube might not have sufficient weight to hang straight and hence would provide a more stringent test than a normal fuel element. The element used will be washed and inspected to determine the cause of the hang up.

When re-inserting the mock fuel element in core position C-233, the load cell momentarily indicated a minimum load of -1600 lbs during the last inch of travel. This is attributed to quad-ring drag; this is substantiated by the fact that the breakaway load upon removal of the element is comparable to that obtained in the other positions. Otherwise, all element breakaway forces were as anticipated and travel loads represented element weights plus fuel handling machine friction losses. Table XII presents load cell readings obtained over a storage cell for comparison purposes.

It is concluded from the observations made and the data obtained that core alignment was maintained at the elevated temperature. Table XIII presents load cell readings and the actual element weights.

A helium leak check verified that the helium gas integrity of the reactor was maintained following the insertion of each element.

M. TRANSFER OF LOOP HAND CONTROL STATIONS AT 540 F

The six main sodium pumps were switched from manual control to automatic control at 15% speed without difficulty. The switchover was accomplished smoothly and without creating a step-change ("bump") in system operating conditions.

TABLE XII

BASE DATA
CORE ALIGNMENT TEST, FHM LOAD CELL DATA

Grapple No. 1, Storage Cell S-30 Dummy Element No. DE-31		Grapple No. 2, Storage Cell S-49 Mock Fuel Element RP-207	
Weight of Element	495 lbs	Weight of Element	430 lbs
Max. load first 6 inches travel up (breakaway)	800 lbs	Max. load first 6 inches travel up (breakaway)	700 lbs
Load during up -travel	600 lbs	Load during up -travel	510 lbs
Load during down -travel	500 lbs	Load during down -travel	400 lbs
Min. load last 6 inches travel down (reseat)	-350 lbs	Min. load last 6 inches travel down (reseat)	-500 lbs

TABLE XIII
CORE ALIGNMENT TEST, FHM LOAD CELL DATA

GRAPPLE NO. 1		GRAPPLE NO. 2	
CORE POSITION C-185			
<u>Dummy Element</u>		<u>Mock Fuel Element</u>	
Weight of Element	495 lbs	Weight of Element	430 lbs
Max. load first 6 inches travel up (breakaway)	600 lbs	Max. load first 6 inches travel up (breakaway)	850 lbs
Load during up -travel	550 lbs	Load during up -travel	550 lbs
Load during down -travel	400 lbs	Load during down -travel	400 lbs
Min. load last 6 inches travel down (reseat)	-200 lbs	Min. load last 6 inches travel down (reseat)	-200 lbs
CORE POSITION C-209			
<u>Dummy Element</u>		<u>Mock Fuel Element</u>	
Weight of Element	495 lbs	Weight of Element	430 lbs
Max. load first 6 inches travel up (breakaway)	800 lbs	Max. load first 6 inches travel up (breakaway)	600 lbs
Load during up -travel	550 lbs	Load during up -travel	500 lbs
Load during down -travel	400 lbs	Load during down -travel	400 lbs
Min. load last 6 inches travel down (reseat)	-200 lbs	Min. load last 6 inches travel down (reseat)	-400 lbs
CORE POSITION C-233			
<u>Dummy Element</u>		<u>Mock Fuel Element</u>	
Weight of Element	495 lbs	Weight of Element	430 lbs
Max. load first 6 inches travel up (breakaway)	900 lbs	Max. load first 6 inches travel up (breakaway)	650 lbs
Load during up -travel	600 lbs	Load during up -travel	500 lbs
Load during down -travel	400 lbs	Load during down -travel	400 lbs
Min. load last 6 inches travel down (reseat)	-300 lbs	Min. load last 6 inches travel down (reseat)	-1600 lbs

After transferring to automatic control on all six pumps, it was determined that:

- 1) the three primary pumps were operating at approximately the same speed and flow;
- 2) the three secondary pumps were operating at approximately the same speed and flow; and
- 3) no evidence of pump malfunction existed.

Table XIV presents the operating conditions immediately prior to transfer from manual to automatic control and also the conditions immediately following the transfer. The values listed substantiate that no significant bump occurred due to the transfer.

N. ESTABLISHING PRESSURE DROP AT 540 F

A simulated core pressure drop of 38.4 psig was established across each primary loop by proper positioning of the primary throttle valves. The throttle valve HCS setting required to develop the necessary pressure drop was 60% closed for each loop. Primary pump speeds were established at 100% of design speed, when the throttle valve positions were established. 100% design pump speed was calculated to be as follows:

- 1) Primary Pump No. 1 - 675 rpm
- 2) Primary Pump No. 2 - 641 rpm
- 3) Primary Pump No. 3 - 641 rpm

Data shown in Table XV were obtained for each of the six main sodium pumps at the assumed 100% design speed. This information establishes the base data for system testing at 540 F.

TABLE XIV

HCS TRANSFER TEST DATA AT 540 F

Pump No.	Variable	Units	Before Transfer	After Transfer
1 Primary	Flow	millivolts	3.4	3.5
	Speed	rpm	105	100
	Delta P	psig	1.0	1.0
2 Primary	Flow	millivolts	4.3	4.3
	Speed	rpm	125	125
	Delta P	psig	1.5	1.0
3 Primary	Flow	millivolts	4.5	4.5
	Speed	rpm	140	137
	Delta P	psig	2.0	2.0
1 Secondary	Flow	millivolts	4.4	4.3
	Speed	rpm	120	120
	Delta P	psig	0.5	0.5
2 Secondary	Flow	millivolts	4.8	4.8
	Speed	rpm	130	130
	Delta P	psig	1.5	2.0
3 Secondary	Flow	millivolts	5.4	5.5
	Speed	rpm	145	145
	Delta P	psig	0.5	0.5

TABLE XV

540 F BASE DATA

Pump	Flow (mv)	Speed (rpm)	Inlet Press (psig)	Outlet Press (psig)	ΔP (psig)
Primary No. 1	27.0	675	-1.8	36.6	38.4
	26.9	640	-1.9	37.5	39.4
	25.9	640	-2.4	36	38.4
Secondary No. 1	30.1	675	16.5	59	42.5
	30.8	675	16	57	41.0
	30.9	675	17	59	42.0

O. PUMP SPEED INCREASES AT 540 F

The following facts concerning interaction of the pumps during speed increases were verified by observation of the main control board instruments:

- 1) With all primary pumps operating at the speed corresponding to 15% steam demand, any one pump can be increased to the speed corresponding to 30% of steam demand without affecting the flow through the other two loops.
- 2) The same results were obtained when increasing each loop to the speed corresponding to 50% steam demand with the other two loops at 30% steam demand.
- 3) Increasing each loop to that speed corresponding to 100% steam demand (from 50%) with the remaining loops at the speed corresponding to 50% steam demand, yielded the same results.

P. FLOW TRIMMING AT 540 F

The output signals from load computers FCCp-501, -502, -503, -504, -505, and -506 were trimmed to equalize all six flows when the loop steam demand hand control stations (HCS-501, -502, and -503) were set at 60% scale. Flow control was then transferred to base load controller HCS-504 and it was visually verified that the six loop flows equalized at the same value for various settings of

HCS-504 (steam demand hand control station).

Q. DETERMINATION OF SYSTEM HYDRAULIC CHARACTERISTICS

The six main sodium pumps were operated simultaneously from steam demand hand control station HCS-504, over a range of approximately 110 rpm to 600 rpm, to determine significant system hydraulic characteristics. Results of the testing demonstrated the similarity between the three primary loops and between the three secondary loops.

Pump motor winding temperatures and coupling temperatures were closely observed during circulating operations, to detect any off normal conditions. At 100% speed on secondary pump No. 3, a high winding temperature alarm was actuated with an indicated 200 F. This occurred simultaneously with a pump high sodium level alarm (Refer to Part III. J) and the pump was shutdown. It was determined that the temperature alarm setpoint was excessively stringent according to pump manufacturer's information. Hence, the temperature was not considered an alarm condition. There had been no alarm prior to this time, nor was this winding temperature reached again during subsequent hot circulation testing. All coupling temperatures remained within alarm limits during all phases of testing.

During all pump operation, surveillance was maintained over the pumps for indications of malfunctioning, as indicated by unusual noise, vibration, over-heating, or similar off-normal conditions.

The main pumps were particularly scrutinized during operation at low flows, to detect signs of pump and flow instability. No drifting was observed. A speed oscillation occurred, which was subsequently analyzed and reported

in the Test Completion Report for Preoperational Test
AI-P-1120* - Plant Control System - as follows:

"When on manual speed control from HCS-102, -202, -302, -104, -204 and -304, a 10 rpm oscillation in pump rpm was observed. This oscillation was not present when operating from manual speed control at the coupling control cabinet. It was noted that the coupling is powered from critical control power, while the last amplifier in HCS-102, -202, -302, -104, -204 and -304 is powered by noise-sensitive power. By changing the frequency of the noise-sensitive M-G sets, it was possible to change the period of the pump oscillation. This problem will be corrected by changing the power feed to the last amplifier in HCS-102, -202, -302, -104, -204 and -304 to critical power."

Pump speed indicators were re-checked (employing a Stobotac) prior to 540 F, to firmly establish that the pump speed values recorded would be accurate; no adjustments were required.

As at 440 F, the system hydraulic characteristics determination portion of the hot sodium circulation test at 540 F consisted of two areas of testing:

- 1) Operating all six loops as a unit (from base load controller, HCS-504) from 100% of plant design flow down to approximately 15% flow.
- 2) Operating each circuit on manual control, individually, with the remaining two circuits on automatic control, to verify proper automatic flow compensation when flow in the circuit on manual control was varied.

The results of these tests verified that circuit flows and pump speeds are stable over the operating range and that circuits on automatic control compensate for

* To be published.

variations imposed on the system by operation of one loop on manual control.

Loop flow and pressure changes as a function of pump speed were as anticipated at 540 F. Figures 1167-13 through -15 display the information obtained at 540 F for these variations. Figure 1167-13 shows the close relationship of the three primary pumps and the three secondary pumps, with respect to pump speed as a function of flow. This information verifies the similarity of the loops under operating conditions. Figures 1167-14 and 1167-15 display a similar relationship with respect to loop pressure drop and power demand hand control station HCS-504, respectively.

Flow ramps were imposed to verify that all circuits changed flows smoothly at the correct rate-limited value of 20% per minute.

The compensating ability of circuits on automatic control from 60% steam demand to 75% steam demand (HCS -501, -502, and -503) was demonstrated by establishing two circuits at 70% steam demand on automatic control, and varying the third circuit from 70% to 60% to 75% and back to 70%. The two circuits on automatic control came to that position required for compensation. Table XVI shows both the manner in which the manipulations were accomplished and the results obtained.

The instrumentation balancing and tune-up accomplished following 440 F testing corrected the undesirable traits that were demonstrated at 440 F. The loop hand control stations were also transferred from automatic to manual as a re-check of the manual-to-auto operation to confirm that the transfer could be accomplished smoothly.

TABLE XVI
AUTOMATIC FLOW COMPENSATION
AT 540 F

Circuit De- viating (on Manual)	Manual HCS Setting	PRIMARY FLOWS			Total Pri Flow (mv)	SECONDARY FLOWS			Total Sec Flow (mv)
		Ckt 1 Flow (mv)	Ckt 2 Flow (mv)	Ckt 3 Flow (mv)		Ckt 1 Flow (mv)	Ckt 2 Flow (mv)	Ckt 3 Flow (mv)	
No. 1, HCS-501	60%	20	26	26	72	22	28	27.9	77.9
No. 1, HCS-501	75%	24.9	22.8	23.9	71.7	26.9	26.0	24.9	77.8
No. 2, HCS-502	60%	25.9	20	25.9	71.8	27.9	21.9	27.9	77.7
No. 2, HCS-502	75%	23.9	24.9	22.9	71.7	24.9	27.9	24.9	77.7
No. 3, HCS-503	60%	25.9	25.9	20	71.8	27.9	27.9	21.9	77.7
No. 3, HCS-503	75%	22.9	24	24.8	71.7	25.9	25.9	27	77.5

Figure 1167-13 Loop Flow (mv) vs Pump Speed (RPM)

Figure 1167-14 Loop Flow (mv) vs Pump Delta Pressure (psig)

Figure 1167-15 Loop Flow (mv) vs HCS-504 (Percent Scale)

R. EXTENDED HYDRAULIC TESTING

Analysis of the hydraulic data obtained led to the correction of basic assumptions made in establishing the 100% primary pump speeds. The 100% primary pump speeds were corrected to 675 rpm for pumps No. 2 and 3 and to 685 rpm for pump No. 1. Based on better understanding of the main heat transfer system gained in the test, it is believed that these values more adequately simulate the system performance with a loaded core.

The simulated core drop of 38.4 psig was established in each of the primary loops at the corrected 100% speed values at 540 F. Hydraulic data were obtained for 540, 440, and 350 F. for each of the six loops, without changing the primary throttling valve positions.

Expected values for the primary and secondary pumps are depicted in Figure 1167-16. Developed differential pressures and pump speeds for each loop at each temperature are presented in Figures 1167-17 through 1167-22.

S. COOLDOWN TO 350 F SODIUM TEMPERATURE

Cooldown of the main heat transfer system from 540 to 440 F and from 440 to 350 F was accomplished at a rate of approximately 25 F per hour. A limitation of 25 F steam-to-sodium temperature was established and was at no time exceeded during the cooldown.

Careful control was maintained during cooldown to ensure that the operation was accomplished safely and that sodium temperatures were accurately controlled.

Hydraulic tests were performed at 540, 440 and 350 F as part of the extended testing. Piping displacement data and hanger settings were repeated, as previously reported, at each of the three temperatures.

Figure 1167-16. Expected Hydraulic Characteristics

Figure 1167-17. Hydraulic Characteristics
Primary Loop No. 1

Figure 1167-18. Hydraulic Characteristics
Primary Loop No. 2

Figure 1167-19. Hydraulic Characteristics
Primary Loop No. 3

Figure 1167-20. Hydraulic Characteristics
Secondary Loop No. 1

Figure 1167-21. Hydraulic Characteristics
Secondary Loop No. 2

Figure 1167-22. Hydraulic Characteristics
Secondary Loop No. 3

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

Following are the Test Completion Criteria associated with preoperational test AI-P-1167, "Hot Sodium Circulation Test", as listed in NAA-SR-5700, Supplement 4, Page XI(a)-101, with remarks regarding the test to explain compliance with the criteria.

- 1) Criterion: Primary and secondary sodium heat transfer systems have been filled and sodium circulation has been accomplished at the temperatures specified in the test.

Remarks: Circulation of sodium was accomplished in the primary and secondary sodium systems from 340 F to 540 F, which was the maximum sodium temperature specified in the test procedure. The systems were filled prior to the test, in accordance with sodium filling preoperational test procedures AI-P-1156 and AI-P-1157.

- 2) Criterion: Operability of the system mechanical components has been demonstrated at the various temperatures specified.

Remarks: Two main pump oil seals developed leaks. One of the seals subsequently stopped leaking. The remaining one will be repaired by disassembly of the seal housing and replacing the faulty components.

Primary throttle valve No. 3 became inoperable due to internal binding. Maintenance procedures are being developed for investigation and repair of the valve.

All other system mechanical components were determined to be operable as described in detail in this report. No relationship between the above two discrepancies and system temperature are believed to exist.

- 3) Criterion: Piping movements have been charted at the temperature levels called for in the detailed test procedure. Piping movements have not caused failure, damage, or distress and consequent leakage in the systems.

Remarks: Piping movements were observed and recorded at the specified temperatures of ambient, 340 F, 440 F, and 540 F according to the test procedure. The information obtained is presented within the body of this report.

It was demonstrated that piping movements due to thermal expansion caused by sodium heat-up to 540 F created no failures, damage, or distress and consequent leakage in the systems, as described in detail within this report.

- 4) Criterion: Control rods function properly and without binding over the range of temperatures tested.

Remarks: Proper control rod operation, specifically without binding, was verified at 540 F system temperature. Data obtained regarding control rod operation at 540 F established that no change occurred in rod functioning at 540 F as compared to functioning at ambient temperature.

- 5) Criterion: Alignment of the core and loading face shield has been preserved within design limits.

Remarks: Proper alignment of the core and loading face shield was demonstrated at the hot sodium circulation test temperature of 540 F.

V. EVALUATION

The hot circulation test proved to be a very extensive and rigorous test of the sodium heat transfer system at temperatures up to 540 F.

Except for primary throttle valve malfunctions and minor sodium pump seal problems the test completion criteria as stated in the HNPF Safeguards Report, NAA-SR-5700 (Supplement 4), were met.

Although problems were encountered (as presented in Part III, "Analysis and Presentation", of this report), the majority were not directly related to the elevated temperature achieved during the test. The required action to correct these problems has been or will be completed before wet critical testing. Such corrective action will insure the satisfactory operation of the system at design operating conditions.

A. PIPING

The test demonstrated that the heat transfer piping and associated components will not be subjected to detrimental stress and strain resulting from thermal expansion over the full operating temperature range of the system. The actual measured piping movements compared well with the calculated predicted movements, thus confirming the validity of the piping stress analysis performed during the system design. In addition, the free movement of the system will be assured by the elimination of all interferences which could conceivably restrict movement during the operation of the system at design conditions.

During the cooldown from 540 F to 350 F there was evidence of some restraint on pump suction line 231 by the pipe guide located just outside the core cavity. The

measured axial movement of the pipe lagged behind the expected axial movement associated with the temperature change. Shortly after the system reached 350 F the total measured displacement from 540 F was only 1/8 in. as opposed to a calculated movement of 5/8 in. Approximately two days later the displacement had increased to 5/16 in. This condition was observed during heat-up to 540 F, but at that time it was attributed to erroneous measurements. It is theorized that this is a result of excessive friction in the guide, necessitating a build-up of sufficient force to overcome the friction before movement results. When the piping reaches a temperature equilibrium, there is still residual restraining force equal to the friction of the guide. The normal vibration due to the high fluid velocity in the system caused the line to gradually move, and after a period of time it reached an unrestrained condition. The restraint observed during the test did not endanger the integrity of the system since the calculated pipe stresses in this line amount to only approximately 50 percent of the code allowable stress at design temperature. However, to prevent any problems which may occur at elevated temperatures the restraint will be eliminated by modification of the guide contact surfaces.

B. CORE CAVITY HANGERS

The complete restoration of all core cavity hanger rod shield penetrations should insure continued freedom of movement and satisfactory operation of the support assemblies. During the cooldown of the system from 540 F to 350 F the piping moved as expected and the hangers moved accordingly. However, the position indicators on the cavity constant load hangers did not provide an accurate indication of movement and it was necessary to

enter the cavity to measure the movement of the hanger rods. A simple, direct reading position indicator will be installed on each constant load spring hanger assembly prior to the approach to operating temperatures. This will provide an accessible and accurate means of determining proper operation of the hangers and cavity piping after the cavity is sealed.

C. VALVES

During the hot circulation test the heat transfer system block and throttle valves were operated extensively for the first time. In this period three valve malfunctions occurred of which two were in the operating mechanism and cannot be attributed to the elevated temperature achieved during the test. Primary throttle valve No. 3 did however become inoperable as a result of internal binding at 540 F. The cause of this malfunction has not been determined at this time. However, during construction a shield block was inadvertently set on the valve actuator. The actuator was damaged but there did not appear to be any damage to the stem or the valve internals. It is conceivable that damage did result but did not appear until the valve was operated at elevated temperature. The difficulties encountered to date have been or will be corrected. Due to the limited operating experience, there is no assurance that additional valve or valve actuator malfunction will not occur, and it was therefore recommended that all valves be subjected to a thorough functional checkout consisting of cycling the valve over its full travel 25 to 50 times. This will be accomplished before power operation.

D. PUMPS

The operability of the six sodium pumps was demonstrated in the temperature range attained during the test. The hydraulic performance of all the pumps is comparable and is essentially unaffected at the higher test temperatures. With the exception of a few probably minor problems (discussed below), the operation of the pumps during the test indicates that dependable, satisfactory performance can be expected at design operating conditions.

Shaft seal oil leaks developed in two of the sodium pumps during the test. In one case, secondary pump No. 1, the leak ceased after a period of time, but the leak in primary pump No. 2 still exists.

The cause of this leakage cannot be determined until the seal is disassembled and inspected. Prior to this test an upper seal on primary pump No. 3 developed a leak. The seal was disassembled and it was found that the sealing surfaces were rough due to improper machining during the seal fabrication. New O-rings were installed and the seal returned to service. Since that time no leakage has resulted from this seal.

Based on this experience, it is speculated that this is also the problem with the other two pumps and that secondary pump No. 1 sealed as a result of extensive run-in. Primary pump No. 2 will undoubtedly require disassembly to correct the problem, but it is expected that a permanent repair will be accomplished.

Excessively low sodium levels in the primary pump cases have resulted when the pumps have been operated at the higher flow rates (60 to 100 percent of full flow) with less than design pressure differential across the pump. Under normal operating conditions, the sodium leakage across the wear rings to the upper side of the impeller is greater than the flow back through the impeller pressure balancing

holes. This provides an upward flow through the hydrodynamic bearing into the pump case and a constant operating level is maintained by means of the case overflow. Apparently, under certain conditions this flow relationship across the impeller is upset and sodium is drawn from the case, resulting in a low level condition. This condition was considered to be outside the normal operating range and was not checked out to determine the system conditions where the anomaly occurs. It is conceivable that the actual pressure drop across the loaded core will be less than the calculated value. (In the case of two-loop operation, the pressure drop will certainly be less.) In either or both of these cases, it is possible that the primary throttle valve will have to be throttled during normal operation to provide the required head conditions. It is therefore recommended that after the core is loaded, a hydraulic test be performed which will define this condition and provide a basis for a change in system operation if required.

An independent investigation of the high pitched squeal detected in the pumps indicates that this squeal is speed dependent and is not affected by flow or pressure differential. At speeds of 400 rpm in the secondary pumps and 500 rpm in the primary pumps, the noise reached a maximum level. At speeds higher and lower than those noted above, the noise becomes intermittent and finally ceases approximately 50 rpm above and below this point.

There are several theories as to the cause of the squeal, such as:

- 1) Hydraulic whistle
- 2) Bearing or seal face rubbing
- 3) Preheat furnace vibrating and rubbing on the outside of the pump case.

This condition has been investigated extensively, but at this time there is nothing to substantiate any of these

theories. The noise has existed since the pumps were first placed in operation and has not worsened with time or increased temperature. Since no obvious detrimental effects have resulted after extended operation, it must be assumed that this condition is not serious, unless further developments prove otherwise.

E. CORE ALIGNMENT

The successful insertion and removal of a mock fuel element in three peripheral core channels is considered a satisfactory test of core alignment. An investigation of the hang up of the mock fuel element in two of the three channels indicates that this condition is apparently not a result of core misalignment. A check of the test data indicates the point of hang up occurred at 169-1/2 inches out of the reactor. At this point, the top of the element process tube disconnect sleeve is at the same elevation as the bottom of the top shield. This corresponds to the point of hang up experienced during the removal of the fuel elements following the dry critical experiment.

Upon completion of the core alignment test, the mock fuel element was visually inspected and it was found that the sleeve had been modified (by chamfering the top edge of the sleeve). The top of the sleeve was well chamfered, but there was sufficient diametric clearance to allow the sleeve to slip to one side and project beyond the hanger tube. A drawing check of the sleeve and associated components indicates that under the most adverse combination of tolerances, the sleeve could project 0.047 inches. During future removal of these elements some hang up will undoubtedly be experienced, regardless of the reactor temperature conditions. It

does not appear that any element damage could result from such a hang up. To relieve this condition simply requires cycling the grapple up and down once or twice. At worst, this condition is only a minor inconvenience. Nevertheless, for future core loadings it is recommended that the disconnect mechanism be redesigned to eliminate this inconvenience.

F. CONTROL RODS

All 19 control rods were subjected to the same mechanical check out as was performed during the control rod preoperational test AI-P-1145. Rod drop times, latch release time, etc. compare well with the preoperational test data and are well within the limits set forth in HNPF Technical Specifications.

APPENDIX A
EQUIPMENT MAINTAINED AT 350F DURING
HOT CIRCULATION TESTING

Preheat System	Switch	Equipment or Line	Circuit Breaker
"B"	120	H-401, Carbon Trap Heater	C-304-A
"F"	1,2, and 3	Line 211, fill & drain line, secondary loop No. 3	C-307-A
"F"	4,5,6, and 7	Drain line, secondary loop No. 3	C-307-A
"F"	8 thru 24	Line 183 & V-499 - Secondary sodium line drain header	C-307-A
"F"	25,26, and 27	Line 108 & V-490 and Secondary Na fill tank No. 1 inlet line from sodium supply header	C-507-A
"F"	28,29, and 30	Line 109 & V-492 and Secondary Na fill tank No. 2 inlet line from sodium supply header	C-507-A
"F"	31 thru 34	Line 110 & V-494 - Secondary Na fill tank No. 3 inlet line from Na supply header	C-507-A
"F"	35	T-410 - Secondary Na fill tank No. 1	42-A-205
"F"	36	T-411 - Secondary Na fill tank No. 2	42-A-210
"F"	37	T-412 - Secondary Na fill tank No. 3	42-A-215
"F"	38,39, and 40	Line 178 & V-491 - Secondary Na fill tank No. 1 drain line	C-507-A

APPENDIX A - Continued

Preheat System	Switch	Equipment or Line	Circuit Breaker
"F"	41, 42, and 43	Line 179 & V-493 - Secondary Na fill tank No. 2 drain line	C-507-A
"F"	44, 45, and 46	Line 180 & V-495 - Secondary Na fill tank No. 3 drain line	C-507-A
"F"	47, 48, 49, and 50	Line 181 - Drain header. Secondary Na fill tank to secondary Na service pump	C-507-A
"F"	51	P-403-Secondary Na service pump header	C-507-A
"F"	52 thru 57	Secondary Na fill & drain header (cold trap supply)	C-507-A C-507-B
"F"	58 thru 61	Line 192 & V-4104 - Fill & drain line - secondary loop No. 1	C-307-B
"F"	62 thru 65	Line 198 & V-4105 - Fill & drain line - secondary loop No. 2	C-307-B
"F"	66 thru 72	Line 204 & V-4106 - Fill & drain line - secondary loop No. 3	C-307-B
"F"	73	Line 184 & V-498 - Cross connect between Na fill & drain header	C-307-B
"F"	74, 75, and 76	Line 185 & V-498 - Cross connect between secondary Na fill tank inlet & outlet headers	C-307-B

APPENDIX A - Continued

Preheat System	Switch	Equipment or Line	Circuit Breaker
"F"	77, 78, and 79	Line 187 & V-496 & V-497 - Drain to secondary Na service drain tank from Na fill tank outlet header	C-307-B
"F"	80	Secondary Na service drain tank	42-A-211
"F"	81 thru 84	Secondary Na line to plugging meter	C-307-B
"F"	85	X-406 - Secondary Na plugging meter economizer	C-307-B
"F"	86	X-403 - Secondary Na plugging meter cooler	C-307-B
"F"	87	P.M. 403 - Secondary sodium plugging meter	C-307-B
"F"	88, 89, and 90	Secondary sodium return line from plugging meter	C-307-B
"F"	91 thru 95	Secondary sodium line to cold trap	C-307-B
"F"	96	CT-403 - Secondary sodium cold trap	52-A-201-A
"F"	97 thru 100	Line 214 & V-452 - Secondary sodium line from cold trap	C-507-B
"J"	1 thru 6	Line 177 & V-107 - Primary Na pump No. 1 vent	C-502-A
"J"	8	Spare	Spare
"J"	13	Spare	Spare

APPENDIX A - Continued

Preheat System	Switch	Equipment or Line	Circuit Breaker
"J"	14 thru 22	Line 186 & V-207 - vent line from pri. pump No. 2	C-502-A
"J"	24	Spare	Spare
"J"	29	Spare	Spare
"J"	30 thru 40	Vent line from primary pump No. 3 to reactor, line 188 & V-307	C-302-B
"J"	42	Spare	Spare
"J"	47	Spare	Spare
"J"	48 thru 52	Line 217 - Vent line, secondary pump No. 1 to expansion tank	C-502-B
"J"	56 thru 60	Line 303 - Vent line, secondary pump No. 2 to expansion tank	C-502-B
"J"	64 thru 68	Line 304 - Vent line, secondary pump No. 3 to expansion tank	C-303-A C-502-B
"J"	72	T-404 - Primary sodium fill tank No. 4	42-B-209
"J"	73	P-401 - Primary Na service pump No. 1 - headers	C-303-A
"J"	74	P-402 - Primary Na service pump No. 2 - headers	C-303-A

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Hanger Number (Test)	Pipe Line Number	Area	Hanger Number (Constr)	Nature of Pipe	Operating Temp. (F)	Load @ 340° F With Na	Load @ 540° F With Na	Displacement From 340° To 540° F	Linearly Extrapolated Load @ Operating Temperature	Desired Calculated Load @ Operating Temperature	Load	Corrected Load @ 340° F (Final Setting)	LEGEND:
						ins lbs	ins lbs	ins lbs	lbs	lbs	lbs	lbs	+ Down - Up 0 No Movement Or Less Than 1/8" Observed Or Theoretically Expected. -- Measurement Is Not Practical Or Does Not Represent Pipe Movement
1000	270		157A	HS	900°	2 2900	1 7/8 2800	-1/8 -100	2620	2900	-280	3180	
1001	270		157	HS	900°	3 2600	2 5/8 2500	-3/8 -100	2320	2600	-280	2880	
1003	312		20"Line#1	20"Line#1	830°	1 3/4 2700	1 1/2 2600	-1/4 -100	2455	2700	-245	2945	
1004	209		155B	SS	340°	1 7/8 690	1 7/8 690	0 0	—	690	—	—	
1005	312		20"Line#1	20"Line#1	830°	2 3/8 3150	2 1/4 3000	-1/8 -150	2783	3150	-367	3517	
1006	209		155A	SS	340°	2 1/2 210	2 1/2 210	0 0	—	210	—	—	
1007	209		155C	SS	340°	2 180	2 1/4 190	+1/4 +10	—	180	—	—	
1008	267		156B	HS	900°	2 3/4 800	2 3/4 800	0 0	800	800	0	800	
1009	267		156B	HS	900°	2 3/4 800	2 3/4 800	0 0	800	800	0	800	
1010	270		158A	HS	900°	1 3/4 2070	1 3/4 2070	0 0	2070	2070	0	2070	
1011	267		156A	HS	900°	2 3/4 800	2 7/8 820	+1/8 +20	856	800	+56	744	NOTES:
1012	267	Steam	156A	HS	900°	2 3/4 800	2 7/8 825	+1/8 +25	870	800	+70	730	1. Load Correction =
1013	264	Generator	156	HS	900°	2 3/8 1630	2 3/8 1630	0 0	1630	1630	0	1630	Extrapolated Load At
1014	264	Room #1	155	HS	900°	5 1/4 3250	4 7/8 3150	-3/8 -100	2970	3250	-280	3530	Operating Temperature
1015	264		154B	HS	900°	1 1/4 850	1 800	-1/4 -50	710	850	-140	990	Minus (-) Desired
1016	276		159	AS	340°	2 140	2 140	0 0	—	140	—	—	Calculated Load At
1017	217		188	AS	340°	1 1/2 90	1 1/4 85	-1/4 -5	—	90	—	—	Operating Temperature
1018	217		188	AS	340°	1 1/2 90	1 1/4 85	-1/4 -5	—	90	—	—	
1019	217		187	AS	340°	2 170	1 7/8 160	-1/8 -10	—	170	—	—	2. Corrected Load At 340° =
1020	306		160C	AS	340°	2 375	2 375	0 0	—	375	—	—	Load At 340° - Load
1021	306		160C	AS	340°	2 375	2 375	0 0	—	375	—	—	Correction.
1022	306		160C	AS	340°	2 3/8 205	2 1/2 210	+1/8 +5	—	205	—	—	
1023	306		160B	AS	340°	2 3/8 205	2 1/2 210	+1/8 +5	—	205	—	—	3. Nature:
1024	293		160	AS	340°	2 1/4 60	2 1/4 60	0 0	—	60	—	—	AP Auxiliary Primary
1025	293		160	AS	340°	2 1/4 60	2 1/2 65	+1/4 +5	—	60	—	—	AS Auxiliary Secondary
1026	290		159B	AS	340°	5/8 50	3/8 47	-1/4 -3	—	50	—	—	HP Hot Primary
1027	290		159B	AS	340°	5/8 50	1/8 45	-1/2 -5	—	50	—	—	HS Hot Secondary
1028	192		174	SS	340°	3 230	3 3/8 240	+3/8 +10	—	230	—	—	CP Cold Primary
1029	273		182	CS	557°	1 7/8 4100	2 4200	+1/8 +100	4208	4100	+108	3992	CS Cold Secondary
1030	273		181	CS	557°	1 510	1 1/8 535	+1/8 +25	537	510	+27	483	SP Service Primary
1031	194	Nitrogen	175	SS	340°	1 1/2 240	1 1/2 240	0 0	—	240	—	—	SS Service Secondary
1032	264	Cooling	180D	HS	900°	2 1/4 1600	2 3/8 1620	+1/8 +20	1656	1600	+56	1544	
1033	264	Area #1	180D	HS	900°	2 3/4 1760	2 7/8 1800	+1/8 +40	1872	1760	+112	1648	
1034	264		180C	HS	900°	1 1/2 1210	1 1/2 1210	0 0	1210	1210	0	1210	
1035	209		176	SS	340°	1 1/2 330	1 3/8 340	+1/8 +10	—	330	—	—	
1036	192		174A	SS	340°	3 230	3 1/4 235	+1/4 +5	—	230	—	—	

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Hanger	Pipe		Hanger	Nature	Operating	Load @	Load @	Displacement	Linearly	Extrape-	Desired	Calcu-	Load	Corrected	Load @			
Number	Line	Area	Number	of	Temp.	340° F	540° F	From 340° To 540°	lated Load @ Oper.	lated Load @	lated Load @	Correction	340°F (Final	LEGEND:				
(Test)	Number		(Constr)	Pipe	(F)	With Na	With Na		Temperature	Oper. Temperature	Temperature		Setting)		+ Down	- Up		
						ins lbs	ins lbs	ins lbs	lbs		lbs	lbs	lbs	lbs				
1037	318		878	SP	340°	2 1/8 195	2 1/8 195	0 0	—		195	—	—			0	No Movement Or	
1038	234		866	HP	945°	2 1/4 1540	2 1/4 1575	0 +35	1643		1540	+103	1437			Less Than 1/8"		
1039	231		865	HP	945°	2 3/4 1710	2 7/8 1775	+1/8 +65	1904		1710	+196	1514			Observed Or		
1040	196		881	AP	340°	1 1/2 300	1 5/8 325	+1/8 +25	—		300	—	—			Theoretically		
1041	207		883	AP	340°	3 1/4 460	3 3/8 470	+1/8 +10	—		460	—	—			Expected.		
1042	177		880	AP	340°	2 1/4 60	2 1/4 60	0 0	—		60	—	—		--	Measurement Is Not		
1043	177		880	AP	340°	2 1/4 60	2 1/4 60	0 0	—		60	—	—			Practical Or Does		
1044	196		882	AP	340°	1 5/8 330	1 5/8 330	0 0	—		330	—	—			Not Represent		
1045	237		868A	CP	602°	2 3/8 850	2 3/8 850	0 0	850		850	0	850			Pipe Movement		
1046	237		384A	CP	602°	1 1000	1 1000	0 0	1000		1000	0	1000					
1047	231		379	HP	945°	1 1000	1 1000	0 0	1000		1000	0	1000	NOTES:				
1048	207		884	AP	340°	3 1/4 460	3 1/4 460	0 0	—		460	—	—		1. Load Correction =			
1049	231		864	HP	945°	2 1470	2 1470	0 0	1470		1470	0	1470			Extrapolated Load At		
1050	231		864A	HP	945°	2 2800	2 2800	0 0	2800		2800	0	2800			Operating Temperature		
1051	234		866A	HP	945°	3 1200	3 1200	0 0	1200		1200	0	1200			Minus (-) Desired		
1052	243	IHX	863	CP	602°	7/8 1400	7/8 1425	0 +25	1433		1400	+33	1367			Calculated Load At		
1053	237	Cell	868	CP	602°	2 1/2 2900	2 1/8 2900	0 0	2900		2900	0	2900			Operating Temperature		
1054	255	#1	862	CS	557°	7/8 960	3/4 900	-1/8 -60	895		960	-65	1025					
1055	254		873	CS	557°	1 7/8 960	1 3/4 950	-1/8 -10	949		960	-11	971		2. Corrected Load At 340° =			
1056	242		871	CP	602°	7/8 1400	7/8 1400	0 0	1400		1400	0	1400			Load at 340° - Load		
1057	273		869	CS	557°	2 2300	2 2200	0 -100	2192		2300	-108	2408			Correction		
1058	254		872	CS	557°	2 1/8 1840	2 1800	-1/8 -40	1793		1840	-43	1883					
1059	255		861	CS	557°	2 1/8 1840	2 1/8 1840	0 0	1840		1840	0	1840		3. Nature:			
1060	264		869	HS	900°	2 1/4 2300	2 2200	-1/4 -100	2020		2300	-280	2580		AP Auxiliary Primary			
1061	253		861	HS	900°	1 1/4 1840	1 1725	-1/4 -115	1518		1840	-322	2162		AS Auxiliary Secondary			
1062	253		862	HS	900°	1 7/8 960	1 1/2 900	-3/8 -60	792		960	-168	1128		HP Hot Primary			
1063	264		875	HS	900°	1 1/4 1810	1 1775	-1/4 -35	1714		1810	-96	1906		HS Hot Secondary			
1064	252		873	HS	900°	1 3/8 960	1 1/2 900	-3/8 -60	792		960	-168	1128		CP Cold Primary			
1065	252		872	HS	900°	1 1/4 1840	1 1700	-1/4 -140	1449		1840	-391	2231		CS Cold Secondary			
1066	240		874	HP	945°	5 1/2 1700	5 1640	-1/2 -60	1518		1700	-182	1882		SP Service Primary			
1067	234		867A	HP	945°	3 7/8 2830	3 1/2 2750	-3/8 -80	2588		2830	-242	3072		SS Service Secondary			
1068	241		860	HP	945°	1 3/4 1700	1 1/4 1575	-1/2 -125	1323		1700	-377	2077					
1069	234		867	HP	945°	2 1/4 2400	1 3/4 2250	-1/2 -150	1946		2400	-454	2854					
1070	297		877	AS	340°	2 1/4 200	1 7/8 185	-3/8 -15	—		200	—	—					
1071	293		877A	AS	340°	2 140	1 3/4 135	-1/4 -5	—		140	—	—					
1072	264		180B	HS	900°	2 3/8 1625	2 1/4 1600	-1/8 -25	1555		1625	-70	1695					
1073	264		180A	HS	900°	2 1/2 780	2 3/8 750	-1/8 -30	696		780	-84	864					

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Hanger Number	Pipe Line	Area	Hanger Number	Nature of Pipe	Operating Temp. (F)	Load @ 340° F With Na	Load @ 540° F With Na	Displacement From 340° to 540°	Linearly Extrapolated Load @ Operating Temperature	Desired Calculated Load @ Operating Temperature	Load Correction	Corrected Load @ 340° F (Final Setting)	LEGEND:	
(Test)	Number		(Constr)			ins lbs	ins lbs	ins lbs	lbs	lbs	lbs	lbs		↓ Down - Up
1074	273	Nitrogen Cooling	182A	CS	557°	2 1/8 1550	2 1/2 1625	+3/8 +75	1631	1550	+81	1469	0	No Movement Or
1075	273	Area #1	182A	CS	557°	2 1/8 1550	2 3/8 1625	+1/4 +75	1631	1550	+81	1469		Less Than 1/8"
														Observed Or
1076	224		506	SP	340°	2 1/4 180	2 1/4 180	0 0	180	180	0	180		Theoretically
1077	231		505	HP	945°	2 3/8 2330	2 3/8 2330	0 0	—	2330	—	—		Expected
1078	231	Primary	504	HP	945°	3 1/8 2660	3 1/8 2660	-1/4 0	2660	2660	0	2660	--	Measurement Is NOT
1079	136	Pipeway	501A	SP	340°	3 1/4 330	3 1/4 330	0 0	330	330	0	330		Practical Or Does
1080	136		501B	SP	340°	1 3/4 310	1 7/8 350	+1/8 +40	—	310	—	—		Not Represent
1081	136		500C	SP	340°	2 1/4 185	2 1/4 185	0 0	—	185	—	—		Pipe Movement
2000	201		470	AP	340°	1 1/2 300	1 1/2 300	0 0	—	300	—	—	NOTES:	
2001	319		479	SP	340°	2 1/4 200	2 1/4 200	0 0	—	200	—	—	1. Load Correction =	
2002	201		471	AP	340°	1 5/8 330	1 5/8 330	0 0	—	330	—	—	Extrapolated Load At	
2003	238		465A	CP	602°	2 3/8 850	2 3/8 850	0 0	850	850	0	850	Operating Temperature	
2004	238		416B	CP	602°	1 1000	7/8 975	-1/8 -25	967	1000	-33	1033	Minus (-) Desired	
2005	232		403	HP	945°	1 1000	1 1000	0 0	—	1000	—	—	Calculated Load At	
2006	208		474	AP	340°	3 1/8 460	3 1/4 465	+1/8 +5	—	460	—	—	Operating Temperature	
2007	208		475	AP	340°	3 1/8 460	3 1/4 465	+1/8 +5	—	460	—	—		
2008	232		437A	HP	945°	2 1/4 2990	2 1/4 2990	0 0	2990	2990	0	2990	2. Corrected Load At 340° =	
2009	232		437	HP	945°	1 7/8 1470	2 1490	+1/8 +20	1530	1470	+60	1410	Load at 340° - Load	
2010	232		436	HP	945°	2 3/4 1710	2 3/4 1710	0 0	1710	1710	0	1710	Correction	
2011	235		438	HP	945°	2 1/4 1540	2 1/4 1540	0 0	1540	1540	0	1540		
2012	235	IHX	438A	HP	945°	3 1200	3 1/8 1220	+1/8 +20	1260	1200	+60	1140	3. Nature:	
2013	186	Cell	468	AP	340°	2 1/4 60	2 1/4 60	0 0	—	60	—	—	AP Auxiliary Primary	
2014	186	#2	468	AP	340°	2 1/4 60	2 1/4 60	0 0	—	60	—	—	AS Auxiliary Secondary	
2015	247		444	CP	602°	3/4 1400	3/4 1400	0 0	1400	1400	0	1400	HP Hot Primary	
2016	238		449	CP	602°	2 1/2 2900	2 1/2 2900	0 0	2900	2900	0	2900	HS Hot Secondary	
2017	246		443	CP	602°	3/4 1400	3/4 1400	0 0	—	1400	—	—	CP Cold Primary	
2018	258		445	CS	557°	2 1/8 1840	2 1/8 1840	0 0	1840	1840	0	1840	CS Cold Secondary	
2019	274		449	CS	557°	2 2300	2 2300	0 0	2300	2300	0	2300	SP Service Primary	
2020	258		445A	CS	557°	2 1/4 960	2 1/8 950	-1/8 -10	949	960	-11	971	SS Service Secondary	
2021	259		446A	CS	557°	2 1/4 960	2 1/8 950	-1/8 -10	949	960	-11	971		
2022	259		446	CS	557°	2 1/8 1840	2 1800	-1/8 -40	1797	1840	-43	1883		
2023	257		446	HS	900°	6 1/4 1840	6 1800	-1/4 -40	1728	1840	-112	1952		
2024	257		446A	HS	900°	1 3/4 960	1 5/8 900	-1/8 -60	792	960	-168	1128		
2025	245		442	HP	945°	5 3/8 1700	4 7/8 1625	-1/2 -75	1473	1700	-227	1927		
2026	265		449	HS	900°	2 1/4 2300	2 2200	-1/4 -100	2020	2300	-280	2580		

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Hanger Number (Test)	Pipe Line Number	Area	Hanger Number (Constr)	Nature of Pipe	Operating Temp. (F)	Load @ 340° F With Na	Load @ 540° F With Na	Displacement From 340° to 540°	Linearly Extrapolated Load @ Oper. Temperature	Desired Calculated Load @ Oper. Temperature	Load Correction	Corrected Load @ 340° F (Final Setting)	LEGEND:
						ins lbs	ins lbs	ins lbs	lbs	lbs	lbs	lbs	+ Down - Up
2027	265		450	HS	900°	3 1810	2 7/8 1780	-1/8 -30	1726	1810	-84	1894	0 No Movement Or
2028	256		445A	HS	900°	1 7/8 960	1 1/2 900	-3/8 -60	792	960	-168	1128	Less Than 1/8"
2029	244		441	HP	945°	5 3/8 1700	4 7/8 1625	-1/2 -75	1473	1700	-227	1927	Observed Or
2030	256		445	HS	900°	6 1/4 1840	6 1800	-1/4 -40	1728	1840	-112	1952	Theoretically
2031	235		439	HP	945°	5 1/8 2400	4 5/8 2310	-1/2 -90	2128	2400	-272	2672	Expected.
2032	235		440	HP	945°	3 7/8 2830	3 1/2 2750	-3/8 -80	2588	2830	-242	3072	-- Measurement Is Not
2033	299	IHX	478	AS	340°	2 1/4 200	2 190	-1/4 -10	—	200	—	—	Practical Or Does
2034	294	Cell	466	AS	340°	2 140	1 7/8 135	-1/8 -5	—	140	—	—	Not Represent
2035	264	#2	277A	HS	900°	2 1/4 780	2 1/4 780	0 0	780	780	0	780	Pipe Movement
2036	265		277B	HS	900°	2 1/2 1625	2 1/2 1625	0 0	1625	1625	0	1625	
													NOTES:
2037	198		272	SS	340°	1 7/8 260	2 1/8 270	+1/4 +10	—	260	—	—	1. Load Correction =
2038	274		279A	CS	557°	2 1/4 1550	2 1/2 1650	+1/4 +100	1658	1550	+108	1442	Extrapolated Load At
2039	274		279A	CS	557°	2 1/4 1550	2 1/2 1650	+1/4 +100	1658	1550	+108	1442	Operating Temperature
2040	274		279	CS	557°	1 7/8 4100	2 4200	+1/8 +100	4210	4100	+110	3990	Minus (-) Desired
2041	198	Nitrogen	271	SS	340°	2 1/2 220	2 1/2 220	0 0	—	220	—	—	Calculated Load At
2042	274	Cooling	278	CS	557°	1 510	1 1/8 525	+1/8 +15	526	510	+16	494	Operating Temperature
2043	265	Area #2	277C	HS	900°	2 1/2 1670	2 3/4 1700	+1/4 +30	1754	1670	+84	1754	
2044	265		277C	HS	900°	2 5/8 1690	2 5/8 1690	0 0	1690	1690	0	1690	2. Corrected Load At 340° =
2045	200		274	SS	340°	1 3/4 250	1 3/4 250	0 0	—	250	—	—	Load at 340° - Load
2046	265		277H	HS	900°	1 1/2 1210	1 1/2 1210	0 0	—	1210	—	—	Correction
2047	210		276C	SS	340°	3 1/2 350	3 1/2 350	0 0	—	350	—	—	
													3. Nature:
2048	271		263A	HS	900°	2 2900	1 3/4 2800	-1/4 -100	2620	2900	-280	3180	AP Auxiliary Primary
2049	271		263	HS	900°	3 2600	2 5/8 2500	-3/8 -100	2320	2600	-280	2880	AS Auxiliary Secondary
2050	313	Steam	20" Line #2	20" Line #2	830°	1 5/8 2700	1 5/8 2700	0 0	2700	2700	0	2700	HP Hot Primary
2051	313	Generator	20" Line #2	20" Line #2	830°	2 3/8 3150	2 1/4 2900	-1/8 -250	2540	3150	-610	3760	HS Hot Secondary
2052	210	Room #2	268A	SS	340°	1 7/8 690	1 3/4 650	-1/8 -40	—	690	—	—	CP Cold Primary
2053	210		268B	SS	340°	2 1/8 190	2 1/8 190	0 0	—	190	—	—	CS Cold Secondary
2054	210		268C	SS	340°	1 7/8 160	2 170	+1/8 +10	—	160	—	—	SP Service Primary
2055	294		269	AS	340°	2 1/4 60	2 1/4 60	0 0	—	60	—	—	SS Service Secondary
2056	294		269	AS	340°	2 1/4 60	2 1/4 60	0 0	—	60	—	—	
2057	307		273H	AS	340°	2 375	2 375	0 0	—	375	—	—	
2058	307		273H	AS	340°	2 325	2 1/8 390	+1/8 +15	—	375	—	—	
2059	268		262B	HS	900°	2 7/8 840	3 850	+1/8 +10	868	800	+28	772	
2060	268		262B	HS	900°	2 1/2 770	2 5/8 780	+1/8 +10	798	800	+28	772	
2061	265		260A	HS	900°	2 1/2 1650	2 3/8 1600	-1/8 -50	1510	1650	-140	1790	

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Hanger Number (Test)	Pipe Line Number	Area	Hanger Number (Constr)	Nature of Pipe	Operating Temp. (F)	Load @ 340° F With Na	Load @ 540° F With Na	Displacement From 340° to 540°	Linearly Extrapolated Load @ Operating Temperature	Desired Calculated Load @ Operating Temperature	Load Correction	Corrected Load @ 340° F (Final Setting)	LEGEND:
						ins lbs	ins lbs	ins lbs	lbs	lbs	lbs	lbs	+ Down - Up 0 No Movement Or Less Than 1/8" Observed Or Theoretically Expected. -- Measurement Is Not Practical Or Does Not Represent Pipe Movement
2062	265		260	HS	900°	5 1/4 3250	4 3/4 3150	-1/2 -100	2970	3250	-280	3530	
2063	265		260B	HS	900°	2 1000	1 5/8 930	-3/8 -70	808	1000	-192	1192	
2064	271		264A	HS	900°	1 3/4 2070	1 3/4 2070	0 0	2070	2070	0	2070	
2065	268		262A	HS	900°	2 5/8 800	2 3/4 810	+1/8 +10	814	800	+14	786	
2066	268		262A	HS	900°	2 5/8 800	2 3/4 810	+1/8 +10	814	800	+14	786	
2067	277	Steam	267	AS	340°	2 140	1 7/8 138	-1/8 -2	—	140	—	—	
2068	303	Generator	282	AS	340°	1 1/2 90	1 1/4 86	-1/4 -4	—	90	—	—	
2069	303	Room #2	282	AS	340°	1 1/2 90	1 1/4 86	-1/4 -4	—	90	—	—	
2070	303		283	AS	340°	2 170	1 7/8 160	-1/8 -10	—	170	—	—	
2071	291		268H	AS	340°	2 70	2 70	0 0	—	70	—	—	
2072	291		268H	AS	340°	2 70	2 70	0 0	—	70	—	—	NOTES:
2073	232		649	HP	945°	1 5/8 2730	1 5/8 2730	0 0	2730	2730	0	2730	1. Load Correction = Extrapolated Load At
2074	232	Primary	648	HP	945°	1 3/8 2570	1 1/4 2500	-1/8 -70	2358	2570	-212	2782	Operating Temperature
2075	225	Pipeway	509	SP	340°	2 1/4 190	2 1/4 190	0 0	—	190	—	—	Minus (-) Desired
2076	186		500D	AP	340°	2 3/8 200	2 3/8 200	0 0	—	200	—	—	Calculated Load At Operating Temperature
2077	307	Steam	273G	AS	340°	2 3/8 205	2 1/2 205	+1/8 0	—	205	—	—	
2078	307	Rm. #2	273G	AS	340°	2 3/8 205	2 1/2 205	+1/8 0	—	205	—	—	2. Corrected Load At 340° = Load at 340° - Load Correction
3000	272		265A	HS	900°	2 2900	1 3/4 2800	-1/4 -100	2620	2900	-280	3180	
3001	272		265	HS	900°	3 2600	2 3/4 2550	-1/4 -50	2460	2600	-140	2740	3.
3002	211	Steam	268E	SS	340°	1 1/2 240	1 1/2 240	0 0	—	240	—	—	Nature:
3003	211	Generator	268D	SS	340°	2 3/4 310	2 3/4 310	0 0	—	310	—	—	AP Auxiliary Primary
3004	314	Room #3	20"Line#3	20"Line#3	830°	1 1/2 2700	1 1/2 2600	0 -100	2455	2700	-245	2945	AS Auxiliary Secondary
3005	211		268F	SS	340°	2 3/8 210	2 1/4 200	-1/8 -10	—	210	—	—	HP Hot Primary
3006	211		268G	SS	340°	2 3/4 250	2 7/8 250	0 0	—	250	—	—	HS Hot Secondary
3007	295		270	AS	340°	2 1/4 60	2 1/4 60	0 0	—	60	—	—	CP Cold Primary
3008	295		270	AS	340°	2 1/4 60	2 1/4 60	0 0	—	60	—	—	CS Cold Secondary
3009	314		20"Line#3	20"Line#3	830°	2 3/8 3150	2 1/4 3050	-1/8 -100	2905	3150	-245	3395	SP Service Primary
3010	269		262D	HS	900°	2 3/4 800	2 7/8 810	+1/8 +10	828	800	+28	772	SS Service Secondary
3011	269		262D	HS	900°	2 1/2 780	2 3/4 800	+1/4 +20	836	800	+36	744	
3012	266		262	HS	900°	2 1/2 1650	2 3/8 1600	-1/8 -50	1510	1650	-140	1790	
3013	266		261	HS	900°	5 1/4 3250	4 3/4 3125	-1/2 -125	—	3250	—	—	
3014	266		261A	HS	900°	2 1000	1 3/4 950	-1/4 -50	860	1000	-140	1140	
3015	272		266A	HS	900°	1 3/4 2070	1 3/4 2100	0 +30	2154	2070	+84	1986	
3016	269		262C	HS	900°	2 3/4 800	2 7/8 830	+1/8 +30	884	800	+84	716	

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Hanger Number (Test)	Pipe Line Number	Area	Hanger Number (Constr)	Nature of Pipe	Operating Temp. (F)	Load @ 340° F With Na	Load @ 540° F With Na	Displacement From 340° to 540°	Linearly Extrapolated Load @ Operating Temperature	Desired Calculated Load @ Operating Temperature	Load Correction	Corrected Load @ 340° F (Final Setting)	LEGEND:	
						ins lbs	ins lbs	ins lbs	lbs	lbs	lbs	lbs		+ Down
														- Up
3017	269		262C	HS	900°	2 3/4 800	2 7/8 825	+ 1/8 +25	870	800	+70	730	0	No Movement Or
3018	278		268	AS	340°	2 140	1 3/4 130	- 1/4 -7	—	140	—	—		Less Than 1/8"
3019	304	Steam	284	AS	340°	1 1/2 90	1 1/4 85	- 1/4 -5	—	90	—	—		Observed Or
3020	304	Generator	284	AS	340°	1 1/2 90	1 1/4 85	- 1/4 -5	—	90	—	—		Theoretically
3021	304	Room #3	285	AS	340°	2 170	1 3/4 160	- 1/4 -10	—	170	—	—		Expected
3022	292		268K	AS	340°	1/2 50	1/2 48	0 -2	—	50	—	—		-- Measurement Is Not
3023	292		268K	AS	340°	1/2 50	3/8 47	- 1/8 -3	—	50	—	—		Practical Or Does
														Not Represent
3024	204		276	SS	340°	1 3/4 250	1 7/8 260	+ 1/8 +10	—	250	—	—		Pipe Movement
3025	275		281A	CS	557°	2 1/4 1550	2 1/2 1650	+ 1/4 +100	1658	1550	+108	1442		
3026	275		281A	CS	557°	2 1/4 1550	2 1/2 1650	+ 1/4 +100	1658	1550	+108	1442	NOTES:	
3027	211		276D	SS	340°	3 3/4 500	3 7/8 510	+ 1/8 +10	—	500	—	—	1. Load Correction =	
3028	266		277F	HS	900°	1 1/2 1210	1 1/2 1200	0 -10	1182	1210	-28	1238	Extrapolated Load At	
3029	206	Nitrogen	277	SS	340°	1 3/4 250	1 7/8 260	+ 1/8 +10	—	250	—	—	Operating Temperature	
3030	275	Cooling	280	CS	557°	1 510	1 1/4 530	+ 1/4 +20	531	510	+21	489	Minus (-) Desired	
3031	266	Area #3	277G	HS	900°	2 3/8 1625	2 1/2 1650	+ 1/8 +25	1691	1680	+66	1614	Calculated Load At	
3032	266		277G	HS	900°	2 3/8 1625	2 3/4 1725	+ 3/8 +100	1900	1680	+275	1405	Operating Temperature	
3033	275		281	CS	557°	1 7/8 4100	2 4200	+ 1/8 +100	4208	4100	+108	3992		
3034	204		275	SS	340°	3 250	3 260	0 +10	—	250	—	—	2. Corrected Load At 340° =	
													Load at 340° -- Load	
3035	202		472	AP	340°	1 5/8 330	1 5/8 330	0 0	—	330	—	—	Correction	
3036	320		480	SP	340°	2 1/8 195	2 1/8 195	0 0	—	195	—	—		
3037	202		473	AP	340°	1 5/8 330	1 5/8 330	0 0	—	330	—	—	3. Nature:	
3038	239		465B	CP	602°	2 3/8 815	2 3/8 900	0 +25	905	850	+30	820	AP Auxiliary Primary	
3039	239	IHX	421B	CP	602°	1 1000	1 1000	0 0	1000	1000	0	1000	AS Auxiliary Secondary	
3040	233	Cell	408	HP	945°	1 1000	1 1000	0 0	1000	1000	0	1000	HP Hot Primary	
3041	233	#3	452A	HP	945°	2 1/4 2990	2 1/4 2990	0 0	2990	2990	0	2990	HS Hot Secondary	
3042	233		452	HP	945°	1 3/4 1420	1 7/8 1475	+ 1/8 +55	1552	1470	+132	1338	CP Cold Primary	
3043	233		451	HP	945°	2 5/8 1700	2 7/8 1750	+ 1/4 +50	1850	1710	+150	1560	CS Cold Secondary	
3044	212		476	AP	340°	3 1/4 460	3 1/2 475	+ 1/4 +15	—	460	—	—	SP Service Primary	
3045	212		477	AP	340°	3 1/4 460	3 1/4 460	0 0	—	460	—	—	SS Service Secondary	
3046	188		469	AP	340°	2 5/8 70	2 5/8 70	0 0	—	70	—	—		
3047	188		469	AP	340°	2 5/8 70	2 5/8 70	0 0	—	70	—	—		
3048	236		455	HP	945°	2 1/8 1540	2 1/4 1560	+ 1/8 +20	1600	1540	+60	1480		
3049	236		455A	HP	945°	3 1200	3 1/8 1225	+ 1/8 +25	1275	1200	+75	1125		
3050	251		459	CP	602°	7/4 1400	7/8 1425	+ 1/8 +25	1433	1400	+33	1367		
3051	239		464	CP	602°	2 2300	2 2300	0 0	2300	2300	0	2300		

PREPARED BY	ATOMICS INTERNATIONAL A DIVISION OF NORTH AMERICAN AVIATION INC Appendix B SPRING HANGER ADJUSTMENT DATA	PAGE NO 7 OF 8
CHECKED BY		REPORT NO
DATE 6-26-62		MODEL NO

Hanger Number	Pipe Line	Area	Hanger Number	Nature of Pipe	Operating Temp. (F)	Load @ 340°F With Na	Load @ 540°F With Na	Displacement From 340° to 540°	Linearly Extrapolated Load @ Oper. Temperature	Desired Calculated Load @ Oper. Temperature	Load Correction	Corrected Load @ 340°F (Final Setting)	LEGEND:	
(Test)	Number		(Constr)			ins lbs	ins lbs	ins lbs	lbs	lbs	lbs	lbs		
3052	250		458	CP	602°	3/4 1400	3/4 1400	0 0	1400	1400	0	1400	0	No Movement Or
3053	262		460	CS	557°	2 1/8 1840	2 1/8 1800	-1/8 -40	1797	1840	-43	1883		Less Than 1/8"
3054	262		460A	CS	557°	2 1/4 960	2 1/8 950	-1/8 -10	949	960	-11	971		Observed Or
3055	275		464	CS	557°	2 2300	2 2300	0 0	2300	2300	0	2300		Theoretically
3056	263		461	CS	557°	2 1/8 1840	2 1/8 1840	0 0	1840	1840	0	1840		Expected
3057	263		461A	CS	557°	2 1/4 960	2 1/8 920	-1/8 -40	917	960	-43	1003	--	Measurement Is Not
3058	261		461	HS	900°	6 1/4 1840	6 1800	-1/4 -40	1728	1840	-112	1952		Practical Or Does
3059	249	IHX	457	HP	945°	5 3/8 1700	5 1650	-3/8 -50	1549	1700	-151	1851		Not Represent
3060	261	Cell	461A	HS	900°	1 7/8 960	1 1/2 900	-3/8 -60	792	960	-168	1128		Pipe Movement
3061	236	#3	453	HP	945°	3 3/4 2830	3 3/8 2750	-3/8 -80	2588	2830	-242	3072		
3062	223		431	AS	340°	2 1/4 200	1 3/4 180	-1/2 -20	—	200	—	—	NOTES:	
3063	295		467	AS	340°	2 140	1 5/8 135	-3/8 -5	—	140	—	—	1. Load Correction =	
3064	266		464	HS	900°	2 1/4 2300	2 2200	-1/4 -100	2020	2300	-280	2580		Extrapolated Load At
3065	266		465	HS	900°	3 1810	2 7/8 1790	-1/8 -20	1754	1810	-56	1866		Operating Temperature
3066	260		460A	HS	900°	1 3/4 960	1 3/8 900	-3/8 -60	792	960	-168	1128		Minus (-) Desired
3067	260		460	HS	900°	6 1/4 1840	6 1800	-1/4 -40	1728	1840	-112	1952		Calculated Load At
3068	236		454	HP	945°	5 2400	4 1/2 2300	-1/2 -100	2095	2400	-305	2705		Operating Temperature
3069	248		456	HP	945°	5 3/8 1700	5 1650	-3/8 -50	1549	1700	-151	1851		
3070	266		277E	HS	900°	2 3/4 1625	2 1/4 1610	-1/4 -15	1583	1625	-42	1667	2. Corrected Load At 340° =	
3071	266		277D	HS	900°	3 3/4 810	3 5/8 800	-1/8 -10	782	800	-28	828		Load at 340° - Load Correction
3072	233		654	HP	945°	1 5/8 2650	1 5/8 2650	-1/8 0	2650	2650	0	2650		
3073	233	Primary	655	HP	945°	2 1/4 2300	2 1/4 2300	0 0	2300	2300	0	2300	3. Nature:	
3074	145	Pipeway	641	SP	340°	1 1/4 100	1 3/8 110	+1/8 +10	—	100	—	—	AP Auxiliary Primary	
3075	308	Steam	273N	AS	340°	2 375	2 375	0 0	—	375	—	—	AS Auxiliary Secondary	
3076	308	Generator	273N	AS	340°	2 375	2 375	0 0	—	375	—	—	HP Hot Primary	
3077	308	Room #3	273K	AS	340°	2 1/4 205	2 1/2 210	+1/8 +5	—	205	—	—	HS Hot Secondary	
3078	308		273K	AS	340°	2 1/4 205	2 1/2 210	+1/8 +5	—	205	—	—	CP Cold Primary	
5000	159		645A	SP	340°	4 110	4 110	0 0	—	110	—	—	CS Cold Secondary	
5001	284	Primary	512	SP	602°	1 5/8 300	1 5/8 300	0 0	300	300	0	300	SP Service Primary	
5002	305	Pipeway	519	SP	602°	2 3/8 200	2 5/8 230	+1/4 +30	239	200	+39	161	SS Service Secondary	
5003	189		500B	SP	340°	7/8 80	3/4 70	-1/8 -10	—	80	—	—		

[illegible]

Figure 1167-B-1. Reactor Cavity - Hanger Locations

SPRING HANGER LOCATION

● SPRING HANGER

TEST
CONSTR.

 HANGER NUMBERS

Figure 1167-B-2. Primary Pipe Tunnel West - Hanger Locations

Figure 1167-B-3. Primary Pipe Tunnel East - Hanger Locations

Figure 1167-B-4. Primary Pipe Tunnel North - Hanger Locations

Figure 1167-B-5. IHX Cells No. 1 South and No. 1 Center - Hanger Locations

Figure 1167-B-6. IHX Cell No. 1 Center and N₂ Cooling Area No. 1 - Hanger Locations

Figure 1167-B-7. Steam Generator Cell No. 1 - Hanger Locations

Figure 1167-B-8. IHX Cells No. 2 South and No. 2 Center - Hanger Locations

Figure 1167-B-9. IHX Cell No. 2 Center and N₂ Cooling Area No. 2
- Hanger Locations

Figure 1167-B-10. Steam Generator Cell No. 2 - Hanger Locations

Figure 1167-B-11. IHX Cells No. 3 South and No. 3 Center - Hanger Locations

Figure 1167-B-12. IHX Cell No. 3 Center and N₂ Cooling Area No. 3 - Hanger Locations

Figure 1167-B-13. Steam Generator Cell No. 3 - Hanger Locations

APPENDIX C
PIPING INTERFERENCES

The following piping interferences were determined by the investigation of the system piping and were corrected:

- 1) IHX Cell No. 1 - Hanger No. 148 on line No. 273 bearing against steel brace.
- 2) IHX Cell No. 2 - Two insulation boxes on line No. 186 bearing against west wall.
- 3) IHX Cell No. 3 - Hanger No. 421 on line No. 239 bearing against line 223.
- 4) Nitrogen Cooling Area No. 1
 - a) Line No. 192 bearing against nitrogen cooling housing.
 - b) Air duct tearing insulation box from line No. 273 at hanger No. 182.
 - c) Line No. 273 bearing against cable tray.
- 5) Nitrogen Cooling Area No. 2 - Line No. 198 bearing against nitrogen cooling housing.
- 6) Nitrogen Cooling Area No. 3
 - a) Line No. 204 bearing against nitrogen cooling housing.
 - b) Air duct bearing against secondary throttle valve No. 3.
- 7) Steam Generator Room No. 1
 - a) Line 209 bearing against service lines.
 - b) Line 267 one-quarter in. from kick plate.
 - c) Vertical support on personnel platform penetrates insulation on line No. 303.
 - d) Shield on insulation off at hanger No. 154B on line No. 264.

- e) Hanger No. 156B bearing against line No. 270.
- f) Thermocouple tubing on line No. 270 bearing against room divider.
- g) Building wall penetrates insulation on line 217.

8) Steam Generator Room No. 2

- a) Hanger No. 260B on line No. 265 bearing against air duct.
- b) Line No. 325 bearing against building joist.
- c) Line No. 268 1/4 in. from kick plate.
- d) Line No. 210 bearing against service lines.
- e) Vertical support on personnel platform penetrates insulation of line No. 307.

9) Steam Generator Room No. 3

- a) Line No. 211 bearing against service lines and west wall.
- b) Line No. 326 bearing against building joist.
- c) Line No. 269 bearing against kick plate.
- d) Vertical support on personnel platform penetrates insulation on line No. 269.
- e) Insulation on line No. 266, at expansion tank, sagging.
- f) Hanger 262D bearing against line No. 272.
- g) Hanger 261A bearing against air duct.
- h) Thermocouple terminal box and tubing on line No. 272 bearing against room divider.
- i) Loose thermocouple tubing on line No. 272.
- j) Line No. 278 bearing against west wall.
- k) Line No. 295 bearing against west wall.

10) Primary Pipe Tunnel (West)

- a) Hanger No. 671A on line No. 168 bearing against thermocouple terminal on line No. 239.
- b) Hanger No. 608A on line No. 141 bent loose and bearing against line No. 232.
- c) Line No. 145 bearing against south wall.

11) Primary Pipe Tunnel (North)

- a) Two insulation boxes on line No. 159 bearing against south wall. One torn from pipe.
- b) Hanger No. 506 on line No. 224 bearing against line 123.

12) Primary Pipe Tunnel (East)

- a) Valve No. V-463 bearing against line No. 237.
- b) Valve No. V-461 bearing against building column. Insulation is cut away.
- c) Line No. 413 bearing against steel beam.

APPENDIX D

REWORK OF THE PIPE HANGERS IN THE CAVITY LINER

I. SUMMARY OF REWORK PERFORMED

- a. Spring Hanger No. 503 was checked for proper travel. It was found to be free to move full travel, up and down. When moved downward, approximately half a cup of clear water dripped out of the opening through the ceiling of the cavity liner.
- b. Hanger No. 514 was disconnected and jacked up and down the full length of travel. The hanger rod would not move freely and had to be jacked in both directions. The bellows assembly was cut free of the shaft and unscrewed from the threaded sleeve. While unscrewing the bellows, three convolutions at the bottom of the bellows were dented. It is felt that this bellows would still be useable; but since a spare was available, it has not been reused. When the hanger rod was pulled out of the sleeve, the key which prevented rotation of the rod was observed to dig into the shaft. It required up to 5000 lbs. force to remove this shaft. A close examination of the key revealed that it had been made from rectangular stock. No effort had been made to break the sharp corners. This key could readily prevent free motion of the hanger rod within the sleeve. The key will be replaced with a standard radius Woodruff-type key. Some rust was found on the shaft. Only a small portion of the shaft had any evidence of lubrication. The shaft was cleaned with trichlorethylene, wire brush, emery cloth and grinder. The sleeve penetrating through the floor was cleaned with trichlorethylene and the bearing surfaces were honed with an automobile cylinder hone. The shaft was lubricated with a film of radiation resistant grease. It was placed in the sleeve and found to be free to travel up and down and to rotate 360 degrees. The new key was then installed and the bellows was set at the proper position, welded in place, and leak tested.

- c. Hanger No. 511 was checked for proper up and down travel. It was found to be free to move as required. The hanger rod at the bottom of the penetration through the shielding was found to be free to rotate. This indicated that no tack welds had been made inside the penetration sleeve. The bellows was cut from this hanger and the rod removed. The key was again found to interfere with proper removal of the shaft. This rod was reworked, re-installed, and helium leak tested.
- d. Hanger No. 645 was found to be not supporting the pipe. That is, the pipe had moved up off the hanger support. Jacks were used to move this hanger rod through the up and down travel range. Considerable force was required to move the rod in either direction. The rod was then removed and observed to contain a coating of rust. The key again interfered with removal of the rod. This rod has been cleaned and re-installed. The lower flange of the bellows assembly came loose from the bellows when the solder joint between these two pieces failed. Since no permanent damage was done to the bellows, the joint was resoldered.
- e. Hanger No. 501 was also observed to be not supporting the pipe, and the hanger rod also required jacking force to travel either up or down. This rod was removed and found to contain a rust film. The key again interfered with proper removal of the shaft. The hanger was reworked and reinstalled.

It is felt that the proper travel of the hanger rods was being prevented by either or both of the problems noted above (that is: rust film on the shaft and improper key). All spring hangers in the cavity liner (17) were removed and reworked.

II. PROCEDURE EMPLOYED FOR HANGER REPAIR

A. To Remove Rods

- 1. Work Required Inside Cavity Liner.
 - a. Shore Piping.

- b. Remove tack weld or welds, and disconnect rod at upper stop and remove nut and washers. When the stop is at the down stop position.

2. Work Required At Spring Hanger

- a. Jack down spring hanger until position indicator reads midscale. Lock in position by welding keeper on both sides of hanger.
- b. Remove nut at top of rod.
- c. Remove spring hanger. Caution: Check to see that rod does not move up as hanger is being removed.
- d. Remove two lead shields.
- e. Cut out seal weld between rod and top of bellows.
- f. Unscrew and remove bellows. Apply torque at bottom nipple. Make sure no torque is applied to the bellows convolutions.
- g. Remove rod. Watch for key as rod is removed.

B. Clean Hangers

- 1. Inspect for rust.
- 2. Remove all moisture from bellows, rod and hole.
- 3. Clean up rod with emery paper and wire brush or automobile cylinder hone.
- 4. Clean rust from hole using wire brush or automobile cylinder hone.
- 5. Lubricate rod with radiation resistant grease.
- 6. Grind off remaining weld at washer at top.

7. Check to see that all segments of rod are tack welded together per drawing requirements.
8. Grind bevel on top washer of bellows as required for welding.
9. Reinstall rod and check to see that it is free to move up and down.

C. Rework Hanger Rods At Pipes

1. Trapeze Hangers

- a. Cut off 3/4" up 12" from bottom of rod.
- b. Cut nuts off rod and throw away nuts and segments of rod.
- c. Install new rod segment and sleeve in position. Do not weld at this time.

2. Clevis Hangers

- a. Cut off bolt through clevis and eye on pipe and replace with new bolt and nut.
- b. Cut tack weld off of nut at clevis.

D. Reinstall Hangers

1. Reinstall rod and check to see that it is free to move up and down. Check to see that key is in place and rod will not rotate.
2. Screw bellows in place. Make sure no torque is applied to convolutions of the bellows.
3. Weld bellows to rod.
Caution: Protect bellows convolutions from weld splatter. Weld per A.W.S. Code. Weld to be leak tight and will be helium leak checked at a later date by AI. Dye penetrant inspect completed weld.
4. Reinstall two lead shields.
5. Replace spring hanger.

6. Reinstall nuts and washers at top and bottom of rod.
7. Connect rod at upper stop.
8. Adjust nuts at bottom of rod to approximately mid range of adjustment threads.
9. Wedge bellows cover cylinder to proper dimension.
10. Shim spring hanger until spring arm is just resting freely on top of bellows cover cylinder.
11. Screw top nut down to top of spring hanger arm.
12. Adjust stop at ceiling to proper clearance dimension.
13. Weld new rod segment in place for a trapeze hanger. Tighten adjustment bolts in lower hanger rods so that rod is supporting a portion of the weight of the pipe.
14. Remove wedges from bellows cover cylinder.
15. Set jack on top of spring hanger and take the load. Then cut keepers off the spring hanger.
16. Slowly remove jack load and check for motion of hanger rod.
17. Check clearances at top and bottom of rod for proper dimension.
18. If clearances off more than 1/8", readjust nuts at top and bottom as required.
19. Align the spring hanger so that the level hinge pins are centered in the slotted holes.
20. Disconnect rod at pipe and check to see that spring hanger is free to travel from full up to full down positions.

21. Reconnect rod and tighten to take a portion of the weight of the pipe.
22. Tack weld the spring hanger to the floor.
23. Replace tack welds on stop at ceiling in the cavity liner.