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HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
DRY CRITICALITY
AI-P-1155

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HALLAM NUCLEAR POWER FACILITY
PREOPERATIONAL TEST COMPLETION REPORT
DRY CRITICALITY

AI-P-1155

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PREOPERATIONAL TEST COMPLETION REPORT

DRY CRITICALITY, AI-P-1155

I. PURPOSE OF TEST

The purpose of this test was to determine the minimum critical mass of the HNPF core without sodium. The data obtained will subsequently be used to determine the reactivity worth of sodium which is normally present in the core. It is important to know the worth of the sodium so that adequate shutdown margin can be assured for the unlikely event of complete loss of sodium. A subcritical calibration of the central control rod was performed and the relative reactivity worths of the inner ring of six control rods were determined. In addition, the data obtained will be used to adjust pre-critical physics calculations.

II. DISCUSSION OF TEST PERFORMANCE

There was no sodium in the reactor during this test and the core was at ambient temperature. Three fission chambers were placed in the core in positions C-191, C-215 and C-239. These chambers had three important functions; first, control; second, protection; and third, to obtain required data. These instruments were required because the normal nuclear instrumentation is insensitive to low neutron flux levels. Before criticality was obtained, however, the source range instruments (BF_3 chambers) were on scale. Figure 1155-1 shows how the three scalers, two automatic digital printers and the dual preset counter were arranged. Two of the startup channels recorded automatically, 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 was

set at 75% of full scale. In addition, each range switch had a physical stop on the 10^6 range. This was done so that saturation of the incore instruments could not occur and to maintain the count rate below the point where resolution time errors would be appreciable. The resolution time (τ) for these instruments was determined to be 3.67×10^{-8} min. Thus, saturation would occur at approximately 2.7×10^7 com. to correct for resolution time errors the following relationship was utilized:

$$c_a = \frac{c_m}{1 - \tau c_m}$$

where,

c_m is the measured count rate

c_a is the adjusted count rate

τ is the resolution time of the circuitry

Resolution corrections were applied for count rates greater than 1.36×10^5 cpm where c_a is approximately 0.5% greater than c_m .

Period alarm set points on the source range instruments were set at a 30 sec period. Period scram set points were set at 5 sec and setback setpoints were set at 20 seconds on the intermediate range channels.

In addition, a final backup high flux scram was set at approximately 2% of full power flux on three of the normal power range protective system instruments. This was a coincidence scram requiring two of the three instruments to be at the scram setpoint in order to cause a scram. The in-fuel thermocouples of the fuel element in core position C-3 were connected to temperature recorder TR-12. A high temperature alarm was set at 25 F above ambient. Upon the annunciation of this alarm, "High Temp Fuel Elements", the operator was required to manually scram the reactor.

The control rods must be disconnected from their respective drive mechanisms and the control rod support carriage removed from the loading face in order to load fuel into the reactor. Accordingly, the poison section of the control rods is in the fully inserted position during any fuel loading operation.

Fuel elements were loaded into the active core with the fuel handling machine (FHM). A licensed HNPf operator was stationed on the FHM. During the insertion of each fuel element, the control room operator watched the nuclear instruments and maintained continuous, direct communication with the fuel handling machine. The licensed operator wore a telephone head set which provided an audible indication of the neutron count rate so that he could make an independent determination of reactor conditions and if necessary stop or reverse the fuel handling machine.

After each incremental fuel loading the corresponding shield plugs in the loading face shield were helium leak checked to detect any reactor cover gas leakage. Then the control rod support carriage was placed on the loading face and bolted down. A standard operating procedure was performed (SOP 5003, Testing of the Control Rod Drive Operation) each time the control rod support carriage was reinstalled to verify the operability of the control rods.

A temporary SOP was written for the dry, zero power experiments to be used each time prior to startup in checking out the instruments required for reactor operation. A total of seven nuclear instruments was required throughout the tests. These were three incore fission chambers, two source range instruments (BF_3 's), and two intermediate range instruments (compensated ion chambers).

The Technical Specifications for the dry, zero power experiments specified the maximum operating conditions to be allowed during the tests. In general, the Technical Specifications defined the instrumentation required, the

operating equipment required, and the administrative controls required throughout the test. The loading of fuel was specified by the Technical Specifications as follows:

- a) The first two fuel loading increments will be 9 and no more than 6 fuel elements respectively.
- b) Each subsequent loading increment will not be greater than four fuel elements.
- c) Multiplication measurements will be made with control rods down after each element is inserted. When the multiplication as indicated by any of the three incore channels reaches a value of 10 during or following the insertion of any fuel element, this will be deemed a fuel loading increment. The increments, after reaching a multiplication of 10, will be limited to one element.
- d) After each incremental loading, curves of inverse multiplication will be plotted as functions of the number of fuel elements in the core. Extrapolation of the various curves to zero, or $k_{eff} = 1$, will be made prior to loading the next increment of fuel.
- e) The next increment will be no more than 1/2 the number of additional elements needed to reach the extrapolated critical loading. However, if the extrapolation indicates that less than two elements are required, one element will be loaded.
- e) Following the second incremental loading, the rods in and rods out inverse multiplication curves will be extrapolated to predict the worth of the control rods in terms of number of fuel elements and no loading increment thereafter will exceed 1/2 of this estimated worth.

In addition to the aforementioned specifications, the minimum allowable count rate was 300 on the incore instruments and the maximum scram level was no higher than 3,000 times source critical. Since the minimum count rate was 300 and the maximum scram level 9×10^5 counts per minute because of the physical stop on the range switches, the 3,000 times source critical scram requirement was inherent in the system.

Prior to loading any fuel, source count rate data was obtained in order to determine multiplication (M). Multiplication on any incore channel was defined as follows:

$$M = C_f / C_o$$

where,

C_o = count rate with no fuel

C_f = count rate with fuel

After each incremental loading, a point was plotted appropriately in each of the following inverse multiplication graphs:

Inverse of the average multiplication vs.

- 1) Number of fuel elements (N)
- 2) Square root of number of elements ($\sqrt{N}$)
- 3) Log of number of elements (Log N)
- 4) A factor proportional to radial buckling (B_r^2)

where,

$$B_r^2 \text{ is proportional to } \frac{10^5}{(R + S)^2}$$

and,

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

N = number of fuel elements

A = fuel cell area (870 cm^2)

S = reflector savings (approximately 22 cm)

These plots were maintained for the following control rod configurations:

- 1) all rods in
- 2) gangs 1, 2, 3, out; 4 and 5 in

- 3) gangs 1, 2, 3, in; 4 and 5 at 50%
- 4) all rods out

The number of control rods in each gang and the core position of each is given in Table I. Figure 1155-2 shows the location of the rods in the reactor.

TABLE I
CONTROL ROD POSITIONS

Gang No.	No. of Rods	Core Position
1	6	C-78, C-118, C-110, C-102, C-94, C-86
2	6	C-38, C-68, C-62, C-56, C-50, C-44
3	3	C-20, c-28, c-36
4	3	C-16, C-24, C-32
5	1	C-1 (central rod)

Control rod gangs 1, 2, and 3 were withdrawn as one bank, and the inverse of the average multiplication was determined and plotted. After determining that the core was sub-critical, control rod gangs 4 and 5 were withdrawn 50%, observing that the core remained sub-critical. Inverse of the average multiplication was again determined and plotted. Control rod gang 5 was then fully withdrawn. Observing that the core remained sub-critical, gang 4 was withdrawn the remaining 50% and the inverse of the average multiplication determined and plotted (all rods out). When it was thought that criticality would be reached on withdrawal of control rod gang 4, individual rods of this gang were withdrawn in 6-inch increments, until criticality was reached. The reactor was made critical on gang 4. This was desirable since it was expected that gang 4 would be worth less than gang 5 and thus criticality would be approached more slowly.

After criticality was reached with gangs 1, 2, 3 and 5 fully withdrawn and gang 4 in the critical position, gang 5 was inserted in 6-inch increments until fully inserted. Sub-critical multiplications were determined for each incremental insertion.

In turn, the relative reactivity worths of each of the six control rods in gangs 3 and 4 were determined. This was done by first bringing the reactor critical with the rod to be evaluated fully withdrawn and then fully inserting this rod and determining the multiplication. From this multiplication data, the relative reactivity worths of the central seven control rods were obtained.

The 31 fuel elements required for criticality were loaded in increments of 9, 2, and 3 elements, and seventeen one element loadings.

Table II shows the actual fuel loading sequence. The sequence for loading the twelfth through the nineteenth elements was rearranged from the original loading scheme.

TABLE II
FUEL LOADING SEQUENCE

1. C-11	9. C-5	17. C-15	25. C-34
2. C-12	10. C-9	18. C-17	26. C-18
3. C-3	11. C-6	19. C-27	27. C-22
4. C-2	12. C-25	20. C-19	28. C-30
5. C-10	13. C-35	21. C-21	29. C-14
6. C-31	14. C-8	22. C-29	30. C-53
7. C-33	15. C-7	23. C-37	31. C-65
8. C-13	16. C-23	24. C-26	

After 26 fuel elements were loaded, the fission chambers had to be moved to de-sensitize them in order to prevent reaching the scram point before all of the rods could be withdrawn. This was accomplished by withdrawing two of the chambers to the top of the moderator cans and inserting one to the bottom of the moderator cans.

At the critical loading of 31 fuel elements, it was necessary to move the chambers again in order to establish a period and achieve criticality. To indicate the relative positions of the incore chambers, the loading face shield (LFS) was taken as 0 inches. Then the top of the moderator cans is 258 in. below the LFS, the center line of the core 362 in. below the LFS, and the bottom of the moderator cans 465 in. below the LFS. Through the loading of 26 fuel elements the chamber positions of incore channels 11, 12, and 13 were 415, 410, and 410 in. respectively; from 27 to 31 fuel elements, the chamber positions were 263, 452, and 452 in. respectively. At criticality (31 fuel elements) the positions were 191, 452, and 268 in. respectively.

Some difficulties were encountered on the approach to criticality with 31 fuel elements, before criticality was attained. First, a high flux scram occurred on the incore channels because there was not enough counting range to reach criticality. Two fission chambers, channels 11 and 13, were then moved above the core. This was performed with control rod gangs 1, 2, 3, and 5 fully withdrawn and gang 4 withdrawn 50%. After the two were moved, channel 12 was deactivated since it was not working properly. The approach to criticality was attempted again and criticality was successfully attained with control rod gangs 1, 2, 3, and 5 fully withdrawn and gang 4 at 132.0 inches. This produced approximately a 370 sec period. Next the flux was reduced by inserting gang 4 and then withdrawing gang

4 to approximately 132.8 inches. This time a 260 sec period was attained. The flux rise was stopped by inserting gang 4 until a steady state flux level was achieved. This occurred with gang 4 at approximately 131 inches. From the data obtained in the dry excess loading, it was determined that the minimum critical mass was 30.6 fuel elements.

When the reactor was put on a period, nuclear heating occurred as evidenced by a two to three degree fahrenheit temperature rise of the in-fuel thermocouples in the fuel element in core position C-3. During the test which followed (Dry Excess Loading AI-P-1163), it was necessary to perform period measurements between point B and point A. Point B (true criticality) is the lowest flux level where the effect of the source is negligible. Point A is the flux level where nuclear heating occurs and was defined as a 2 F rise in fuel temperature. It was apparent from the approach to criticality that the range between point B and point A was limited. In order to minimize nuclear heating and to reduce the critical flux level, the strength of the source was reduced. This was accomplished by separating the antimony capsule from the beryllium sleeve. The antimony and beryllium sections are both approximately 10 in. long. However, it was found that the antimony capsule had to be withdrawn 29 in. in order to obtain about a decade and a half of counting range on the incore instruments. A procedure was written and approved for accomplishing the source separation operation.

Subcritical multiplication data was obtained in order to find the relative integral reactivity worth of the central control rod in the dry critical loading (Figure 1155-7).

The loading was performed around the clock on a 24-hour basis and criticality was achieved at 0322 on January 19, 1962 (on schedule). All operations were performed in accordance with master test procedure, Dry Criticality, AI-P-1155.

III. ANALYSIS AND PRESENTATION OF TEST RESULTS

Table III shows the extrapolated critical loading for the various plots after each incremental fuel loading with all rods out. Figures 1155-3, -3a, -3b, -3c, -4, -4a, -4b, -5, -5a, and -6 are plots of inverse average multiplication vs functions of fuel loading. Table IV is a tabulation of the multiplication data taken throughout the dry critical test.

After the first nine elements were loaded, a linear extrapolation of the "rods-out" points indicated that criticality would occur at 10 1/2 elements, (See Figure 1155-3a). However, from a plot of inverse average multiplication with "rods-in", (Figure 1155-3), determined from data taken after each of the first nine elements was loaded, it can be seen that the curve is taking the characteristic bend. Therefore, the "rods out" curve should also assume this shape. For the second step, two fuel elements were loaded. The third loading was three elements after which a multiplication of 10 was reached on two incore instruments. Thus, from this loading until criticality was attained, only one element per increment was loaded. The inverse average multiplication plots and data compiled in Table IV have been corrected for source decay and normalized for changes in fission chamber location. It can be seen from the plots that there was a definite break in the extrapolated critical mass curves upon loading the 21st fuel

TABLE III

EXTRAPOLATED CRITICAL LOADING

No. of Elements Loaded	Linear Plot (N)	Square Root Plot (N)	Log Plot (Log N)	Radial Buckling (B_r^2)
9	10.5	12.3	----	----
11	16.8	19.0	20.0	25.5
14	19.5	20.3	21.5	25.5
17	22.0	22.0	23.5	25.0
18	22.2	22.0	23.2	23.5
19	24.5	25.0	25.3	27.0
20	22.8	23.0	23.7	24.0
21	26.8	27.0	27.5	28.5
22	26.5	27.0	27.5	29.0
23	26.8	27.0	27.5	27.5
24	27.4	27.5	27.5	28.0
25	28.0	28.5	29.0	29.0
26	30.0	30.5	31.0	31.0
27	30.0	30.2	30.5	30.7
28	31.0	31.2	31.0	31.5
29	31.3	31.4	31.5	31.4
30	30.5	30.5	30.5	30.5

element. It was concluded that the source neutrons, at approximately 50 Kev, have a much less probability of reaching the fission chambers than the high energy fission neutrons. Therefore, the count rates obtained without fuel were low relative to count rates obtained with fuel. Thus, the apparent multiplication (determined from the incore channels) was higher than the "true" multiplication.

It should be noted that the linear plot is extremely conservative. The buckling plot appears to have merit because even after two loadings of fuel (9 and 11) a critical loading of 25.5 elements was extrapolated. In addition, the log plot and square root plot indicated the critical loading more closely than the linear plots. When near criticality, (within five or six elements) all four plots were of equal value.

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.

As the loading approached 30 elements, it was obvious that the analytical calculation of 23 fuel elements for criticality, was in error. After criticality was attained, further analysis indicated that the effect of the source had been neglected in the original calculations. The importance of the source core position, C-4, was overlooked. Recalculation, accounting for the source and not a fuel element in C-4, indicated that criticality would be achieved with 32 ± 2 fuel elements. It was concluded that the difference in worth between the source and a fuel element in this location was approximately 4% Δk .

In order to find the relative integral reactivity worth of the central control rod in the dry critical loading, subcritical multiplication data were obtained. A plot of these data is shown in Figure 1155-7. It should

TABLE IV - SUMMARY OF HNPf DRY CRITICALITY DATA

Total Number of Fuel Elements Loaded (N)		0	9	11	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31
Number of Fuel Elements Loaded This Step		0	9	2	3	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
Channel of Last Fuel Element Loaded		0	C5	C6	C8	C7	C23	C15	C17	C27	C19	C21	C29	C37	C26	C34	C18	C22	C30	C14	C53	C65
Square Root of N		0	3	3.317	3.741	3.873	4.00	4.123	4.242	4.359	4.472	4.583	4.690	4.796	4.899	5.00	5.099	5.196	5.291	5.385	5.477	5.568
Factor Proportional to B_r^2		---	19.33	16.79	14.09	13.39	12.75	12.18	11.66	11.19	10.76	10.36	9.99	9.65	9.332	9.036	8.76	8.50	8.26	8.028	7.813	7.608
All Control Rods Out																						
	Multiplication FC in C-191 (11)	1.0	6.876	9.209	10.68	11.83	12.548	16.06	21.072	20.836	25.126	26.704	27.46	30.008	30.398	31.369	38.23	41.412	42.581	46.16	48.30	48.808
	Multiplication FC in C-215 (12)	1.0	5.199	6.738	10.25	10.50	12.165	12.36	12.498	16.911	19.592	20.329	23.01	23.564	29.412	29.406	27.89	28.048	29.886	30.65	38.08	40.176
	Multiplication FC in C-239 (13)	1.0	6.276	5.067	6.18	6.974	8.126	7.84	7.777	8.455	9.383	8.955	9.15	10.549	10.339	13.249	13.14	14.658	15.994	17.72	17.33	23.989
	Average Multiplication	1.0	6.117	7.005	9.04	9.767	10.946	12.09	13.782	15.401	18.034	18.663	19.87	21.343	23.383	24.675	26.42	28.039	29.487	31.51	34.57	37.658
	Inverse of Average Multiplication	1.0	0.1635	0.1428	0.1106	0.1024	0.091358	0.0827	0.07256	0.06193	0.0555	0.0536	0.0503	0.0469	0.0428	0.04053	0.0379	0.03566	0.03391	0.0317	0.0289	0.066
Cages 1, 2, 3 Out; 4 and 5 In																						
	Multiplication FC in C-191 (11)	1.0	8.016	11.482	15.29	17.781	19.561	25.93	35.422	35.43	49.657	59.124	65.90	78.857	84.783	91.767	121.43	143.01	156.01	183.20	214.55	261.95
	Multiplication FC in C-215 (12)	1.0	6.077	8.276	14.43	15.334	18.815	20.13	21.775	31.39	39.026	44.823	55.13	60.913	79.651	83.073	90.11	103.11	120.61	130.50	216.86	275.39
	Multiplication FC in C-239 (13)	1.0	6.163	6.285	9.34	9.426	10.480	12.48	13.430	11.40	17.236	17.219	18.13	25.282	27.471	34.385	31.68	35.926	44.72	51.12	61.50	107.58
	Average Multiplication	1.0	6.752	8.681	13.02	14.180	16.285	19.51	23.542	26.07	35.306	40.389	46.39	55.017	63.968	69.742	81.07	94.02	107.11	121.61	164.30	214.97
	Inverse of Average Multiplication	1.0	0.1481	0.1152	0.0768	0.0705	0.06141	0.0513	0.0425	0.0384	0.0283	0.0248	0.0216	0.01818	0.01563	0.01434	0.0123	0.0106	0.009336	0.0082	0.00609	0.00465
All Control Rods In																						
	Multiplication FC in C-191 (11)	1.0	8.472	12.528	17.48	20.624	23.1308	31.36	44.385	46.21	66.220	83.210	98.37	124.201	141.151	161.105	22.40	312.14	401.76	595.24	843.47	
	Multiplication FC in C-215 (12)	1.0	6.678	9.393	17.15	18.633	25.029	26.05	28.692	42.15	53.956	64.389	84.64	98.540	138.313	152.917	178.50	21.45	310.70	421.00	641.41	
	Multiplication FC in C-239 (13)	1.0	6.792	7.522	11.02	11.483	12.645	15.72	17.277	18.38	23.456	22.417	27.08	40.138	46.070	61.514	63.84	81.38	116.83	166.84	473.69	
	Average Multiplication	1.0	7.314	9.814	15.22	16.913	20.268	24.38	30.118	35.58	47.877	56.672	70.03	87.626	108.845	125.178	158.25	208.32	276.43	394.36	1252.86	
	Inverse of Average Multiplication	1.0	0.1367	0.1019	0.0657	0.0591	0.04934	0.0410	0.0332	0.0281	0.02089	0.0177	0.0143	0.01141	0.009187	0.00799	0.00632	0.00480	0.003618	0.0025	0.000798	

be noted that the integral reactivity worth curve is asymmetrical. The first 50% of rod withdrawal is worth more than the last 50%. In addition, the relative worths of the control rods of gangs 3 and 4 were found from subcritical multiplication data and compared to the central control rod. These relative worths are tabulated in Table V. From these relative worths it was concluded that gangs 3 and 4 were worth approximately one and one-half times the central control rod in the dry critical loading. The delta k's tabulated in Table V were derived from the relationship $k_{\text{eff}} = 1 - 1/M$. It has already been pointed out that the measured or apparent multiplication was higher than the true multiplication. Therefore, there was no concern about the indicated low worths of the control rods.

TABLE V

RELATIVE REACTIVITY WORTHS OF THE 7 CENTRAL CONTROL RODS

Core Position	Gang Number	Relative Worth	Delta K
C-1	5	1.000	.00944
C-16	4	.289	.00273
C-24	4	.265	.00260
C-32	4	.239	.00226
C-20	3	.248	.00234
C-28	3	.259	.00245
C-36	3	.219	.00207
			Σ .02389

With the source strength reduced, point B was determined. Figure 1155-8 is a plot of critical control rod position vs neutron flux level. The count rate at which the control rod position became constant as new steady flux

levels were attained was defined as point B. Point B was marked on the source range count rate meters on the main board at approximately 300 cpm.

Figure 1155-9 is a plot of k_{eff} vs the number of fuel elements derived from subcritical multiplication data as in the case of the control rod worths. Figure 1155-10 shows the apparent inverse multiplication plot for the individual chambers.

The experience gained from this test indicates that restrictions on loading of fuel could be minimized. A restriction, such as: "No more than one half the number of additional fuel elements needed to reach extrapolated critical loading, plus one, may be loaded per fuel increment," may suffice. Also, it appears that it would be more advantageous to derive the extrapolated critical loading from plots of inverse average multiplication vs several functions of fuel loaded (N , $\sqrt{N}$, $\log N$, B_r^2). Further, it was found that the most meaningful plots of inverse multiplication were based on the average rather than the individual multiplications of the three incore instruments.

IV. COMPLIANCE WITH TEST COMPLETION CRITERIA

In Supplement 4 of NAA-SR-5700, Safeguards Report for the HNPF, the test completion criteria for the Dry Criticality Test, AI-P-1155, are: (a) the reactor is made critical with a minimum number of fuel elements in the dry core, and (b) a subcritical calibration of the central control rod has been made. These criteria were satisfactorily met.

V. EVALUATION

The dry critical experiment was conducted in a very safe, conservative manner. The loading procedure followed was slow but yielded additional data for analysis and evaluation.

The test completion criteria as stated in the test procedure have been met. The reactor was made critical, data was taken which will be used in determination of sodium worth, the pre-critical physics calculations have been adjusted and the sub-critical calibration of the central seven control rods has been made.

The performance of the test and evaluation of the results have indicated two areas which will require further investigation:

- a) Predicted Critical Mass
- b) Control Rod Worth

A. PREDICTED CRITICAL MASS

The difference between predicted and actual critical loading required re-evaluation of calculations made to predict "dry" and "wet" critical mass. Prior to the dry critical experiment, the predicted dry critical loading was 23 ± 2 fuel elements, and the predicted wet critical loading was approximately 37 elements (Safequards Report on Dry, Zero-Power Experiments in HNPF, Supplement 1, NAA-SR-5700). When it became apparent that criticality would not be achieved with a 23 fuel element loading the calculations were reviewed to evaluate discrepancies between actual and assumed core conditions. The neutron source, which replaces a fuel element in a position of maximum worth has an appreciable negative reactivity effect, particularly in the small core. Calculations were made accounting for the presence of the source. The results are presented in Figure 1155-11. The calculated critical loading, with the source, is close in agreement with actual critical loading. The new prediction for wet critical loading, with the source, is 50 ± 5 fuel elements. Although calculated

and actual critical mass are now in close agreement, further evaluation of calculated and actual fuel element worths will be made during Dry Excess Loading.

B. CONTROL ROD WORTH

A preliminary evaluation of "rods-down" apparent multiplication in the subcritical loading indicates the calculated control rod worths may be in error. The test data, thus far, are inconclusive. Further evaluation will be made when data are available from the Dry Excess Loading, AI-P-1163.

FIG. 1155-1

ARRANGMENT OF NUCLEAR
COUNTING INSTRUMENTS

FIG. 1155-2 HNPf DRY CRITICAL LOADING

FIG. 1155-3

HNPF DRY CRITICALITY
TEST
AT - P - 1155

INVERSE AVG. MULTIPLICATION
NUMBER OF FUEL ELEMENTS

○ - ALL RODS IN

U2U U25

FIG. 1155-3b
 UNDE DRY CRITICALITY
 TEST
 A1-P-1155

INVERSE AVERAGE MULTIPLICATION
 VS
 NUMBER OF FUEL ELEMENTS

O-ALL RODS IN
 B-GANGS 1,2,3 OUT
 1/5 50% OUT
 A-ALL RODS OUT

180 079

FIG. 1155-11 PLOT OF PREDICTED K_{eff} VS NO. OF FUEL ELEMENTS

U1U U33