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ORNL 1027
Reactors-Research and Power
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DETERMINATION OF THE POWER OF THE SHIELD-TESTING REACTOR - I. NEUTRON FLUX MEASUREMENTS IN THE WATER-REFLECTED REACTOR

J. L. Meem
E. B. Johnson

OAK RIDGE NATIONAL LABORATORY
OPERATED BY
CARBIDE AND CARBON CHEMICALS COMPANY
A DIVISION OF UNION CARBIDE AND CARBON CORPORATION
UCC
POST OFFICE BOX P
OAK RIDGE, TENNESSEE

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ORNL-1027

This document consists of 32 pages. Copy 9 of 138. Series A.

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Contract No. W-7405, eng 26

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ORNL

PHYSICS DIVISION

DETERMINATION OF THE POWER OF THE SHIELD TESTING REACTOR -

I. NEUTRON FLUX MEASUREMENTS IN THE WATER REFLECTED REACTOR

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Date Issued: AUG 13 1951

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A Division of Union Carbide and Carbon Corporation
Post Office Box P
Oak Ridge, Tennessee

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DETERMINATION OF THE POWER OF THE SHIELD TESTING REACTOR - 1. NEUTRON FLUX MEASUREMENTS IN THE WATER REFLECTED REACTOR

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Introduction

The reactor at the bulk shielding facility[1] is being used as a source of neutrons and gamma rays for shielding experiments. The procedure is to erect shielding samples in front of the reactor and measure the attenuation of these neutrons and gamma rays through the shield. Since the reactor can be operated at various power levels, all shielding measurements are normalized to a power level of 1 watt. It is immediately obvious that some calibration of the power of the reactor is required. It is also necessary to know the flux leaking from the reactor when making shielding measurements. The leakage flux may be changed when a different shielding sample is placed next to the reactor depending on the characteristics of the shielding sample as a reflector. The determination of this leakage then becomes a part of the particular shielding experiment underway and is not considered further in this report.

The loading of the fuel elements in the reactor can be changed as described in Reference 1. When this is done, a new power calibration is necessary. This report describes the power calibration for the fuel elements as arranged in the second critical experiment[1]. Calibrations for other critical assemblies will appear in future reports.

On the basis of the activation of several gold foils in the reactor, the original power level was set at what was assumed to be approximately 1 watt. From that time on, all shielding measurements have been made with the reactor controls set to hold the reactor constant at this power of "1 watt" or at some multiple of this power level.

(1) Breazeale, W. M., "The Bulk Shielding Facility at ORNL", ORNL-991, May 8, 1951

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It remained to determine by careful measurement how close this nominal setting of "1 watt" was to the actual power of the reactor.

The Reactor

The arrangement of the fuel elements for this calibration is shown in Fig. 1. A permanent beryllium oxide reflector was mounted on the back (south) side of the reactor with the control instruments held directly above it. The other five sides of the reactor were surrounded by water, and since shielding measurements were made from the front (north) of the reactor, the shield samples were exposed to what is essentially a water reflected pile. As shown in Fig. 1, the reactor went critical in a symmetrical 5 x 5 array of fuel elements. Since each element is 3 in. on a side, the reactor assembly was 15 in. square. The fuel elements are 3 ft. long but only the central 24 in. contain fuel (Fig. 2). The dimensions of the active core were then 15 in. x 15 in. x 24 in. Twenty-two of the fuel elements contained about 138 gm. of fissionable material (exact amounts are given later). The other three elements contained about half this amount of uranium and held the control rods. The critical mass of the reactor was 3.2 kg. For further details of the reactor, see Reference 1.

Method of Determining the Neutron Flux

Flux determinations were made on the north and south faces of representative elements at the midplane, 6 in. above and below the midplane, and 12 in. above and below the midplane. Each fuel element could then be considered as being divided into four quarters, and for each quarter a known flux value existed on both sides at the top and on both sides at the bottom. The average of these four flux values was taken as the flux in that quarter element. Fig. 2 is a sketch of a fuel element showing the points of measurement.

Foils were not exposed at every point in the reactor at which a flux determination was made. Sufficient foil exposures were made so that a general knowledge of the flux patterns throughout the reactor was known and the fluxes at the points not measured could be determined graphically as described later. Advantage was taken of symmetry wherever possible. It is obvious from the loading diagram, Fig. 1, that the flux distribution should be symmetrical to the east and to the west of the centerline of the reactor. Also, as is shown in Table I, there was a vertical symmetry above and below the midplane of any element. No such north-south symmetry existed because of the beryllium oxide on the south side of the reactor, and to a lesser degree because of the location of the control rods.

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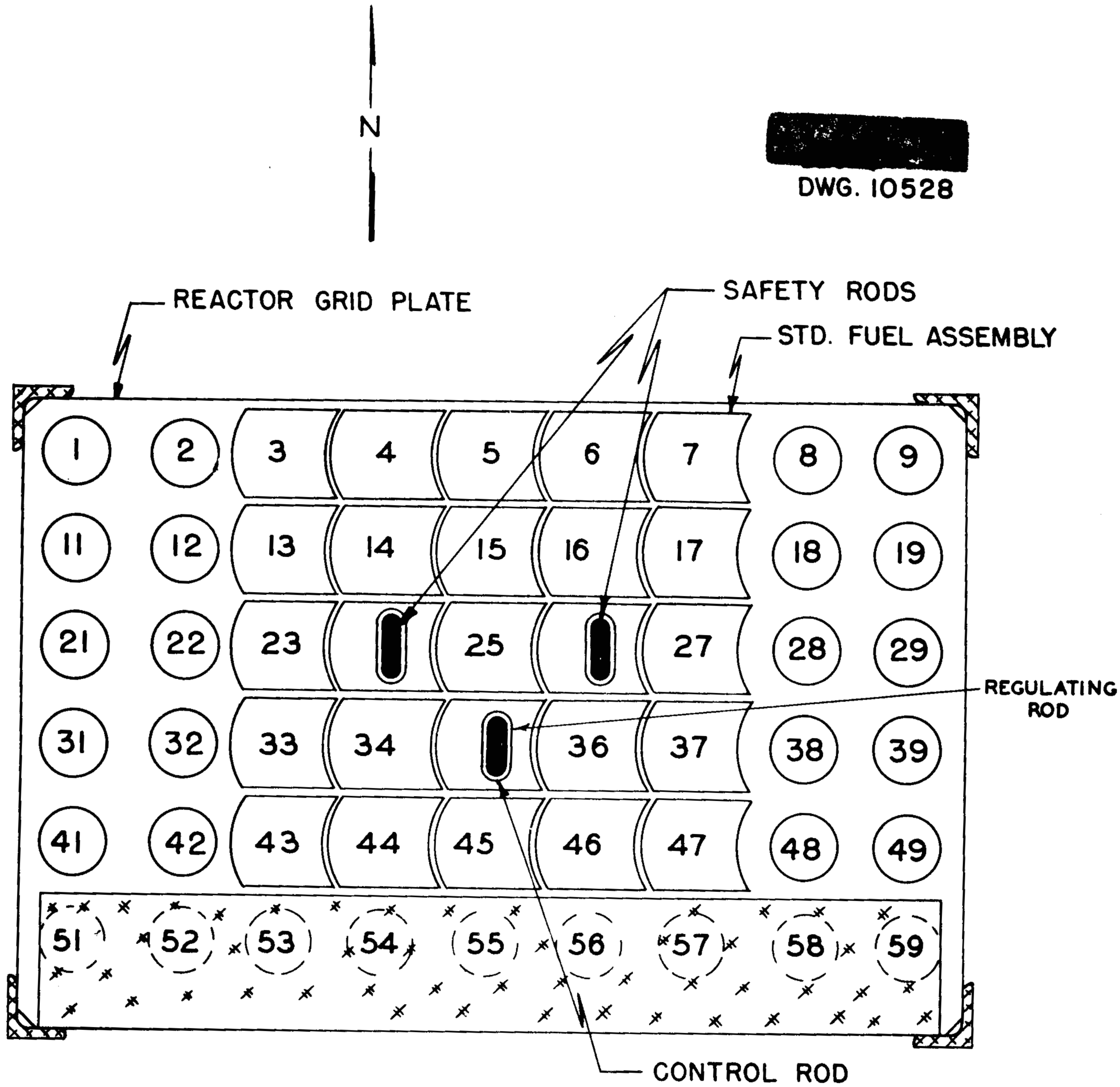

NOTE: BeO REFLECTOR ONE SIDE
WATER REFLECTOR THREE SIDES

FIG. 1

FUEL ASSEMBLY ARRANGEMENT — TEST No. 2

CRITICAL EXPERIMENTS - BULK SHIELDING FACILITY REACTOR

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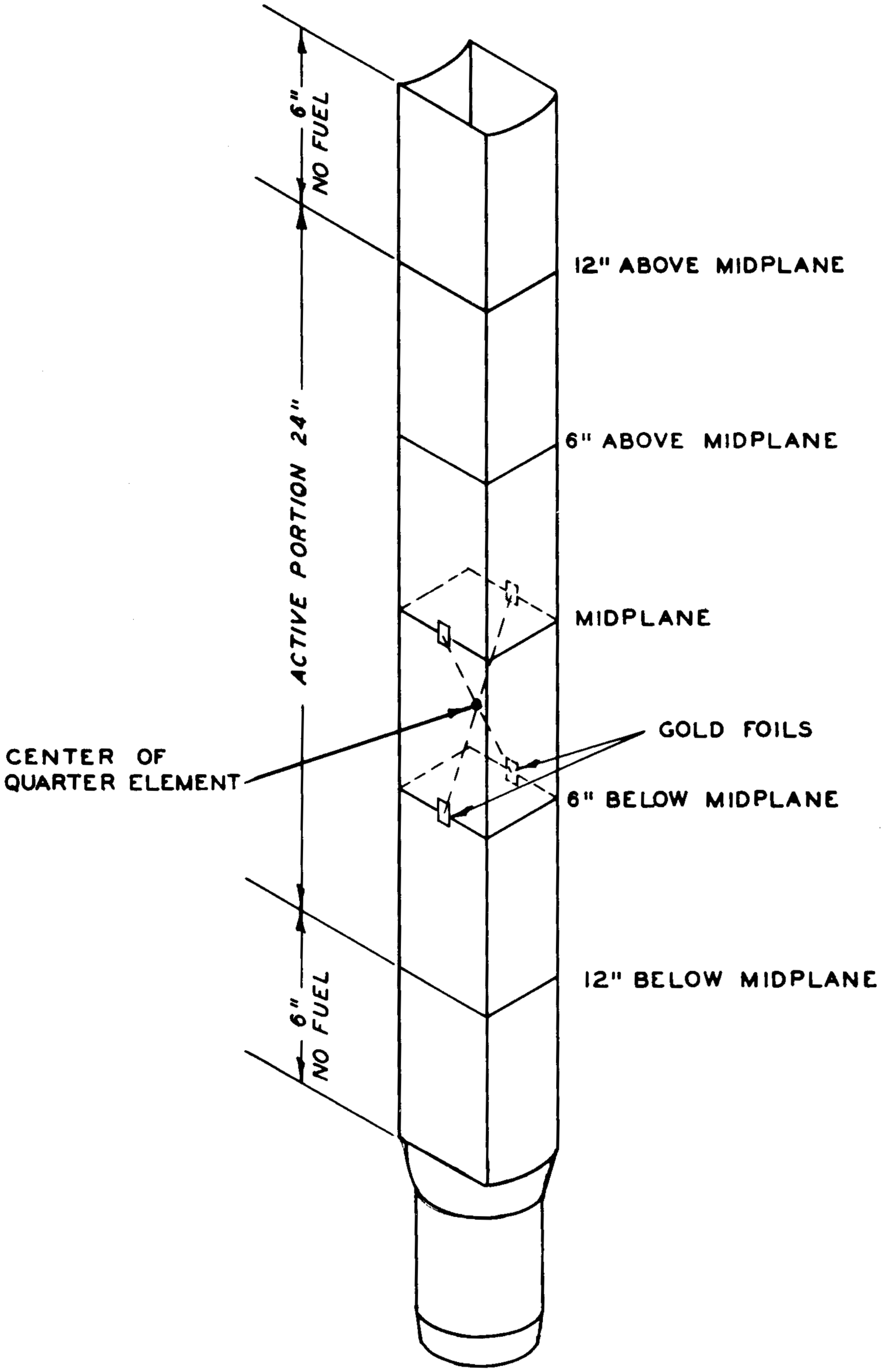

FIG. 2

SKETCH OF A FUEL ASSEMBLY SHOWING TYPICAL POINTS AT WHICH FLUX DETERMINATIONS WERE MADE

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During the runs the two safety rods were completely withdrawn, and the regulating rod was withdrawn more than three-fourths of the way from the reactor. The reactivity was changed only slightly by the withdrawal of the regulating rod the rest of the way. Since only the flux in the immediate vicinity of this rod was depressed, the total flux in the reactor would be changed very little. Therefore, it was still assumed for the calculations that the flux patterns in the upper and lower halves of the reactor were similar.

Foil Exposures

The measurement of the thermal neutron flux in the reactor loaded in three different fuel configurations has been described by Hill[2]. Only those measurements in configuration #2 are pertinent to this calculation.

For these measurements, gold foils 1 cm. square and 5 mil thick were fastened with scotch tape at the desired positions on the outside of the fuel elements. Where epithermal measurements were required, the foils were enclosed in 20 mil Cd. boxes to eliminate the thermal flux. The activity of the foils was measured on two mica window G-M tubes and the saturated activities calculated. Since the presence of the foil, particularly in a cadmium cover, alters the neutron flux in its vicinity, it was necessary to do the exposures in several runs. Start-up corrections were applied to each run and all runs normalized together by means of duplicate positions of the foils, usually three positions on the element in hole 5 of the bottom grid.

The difference between the "bare" and the epithermal saturated activities is the thermal saturated activity. The thermal flux (nv_{th}) can be obtained from this number by multiplying it by 9. This factor is determined by the calibration of these gold foils in the standard graphite pile[3]. Table I is a summary of all measurements made in this lattice configuration. This table is identical to Table II in Reference 2.

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(2) Hill, J. W., "Neutron Flux Measurements in the Bulk Shielding Facility Reactor", ORNL C.F. 51-5-164.

(3) Arnette, T. and Jones, H., "Thermal Neutron Flux Distribution in a Standard Graphite Pile", CP-2804, April 30, 1945.

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Calculations and Results

Table I lists the saturated activities as described by Hill in Reference 2. The lattice positions correspond to the positions shown in Fig. 1. Since the foils were attached to the outside of the fuel elements, the point of measurement in general was between two adjacent elements. Thus, for example, a foil on the south side of position 5 would measure the same flux as a foil on the north side of position 15.

As has been stated, some vertical traverses were made by attaching foils to one face of a fuel element at the center, 6 in. above and below, and 12 in. above and below, giving five flux measurements for such a vertical traverse. From inspection of the saturated activities for such vertical traverses as listed in Table I, it can be seen that the points 6 in. above the centerline agree closely with the points 6 in. below. The same is true for the points 12 in. above and below the centerline. To simplify the calculations, points 6 in. above and below the centerline were averaged as were the points 12 in. above and below the centerline.

Given in Table I are all the measurements taken. The thermal saturated activities were of course obtained by subtracting the cadmium covered activities from the bare activities. Also, the cadmium ratios are listed wherever the information was available.

The saturated activities are tabulated in a different form in Table II. The values are tabulated according to the north side of the corresponding lattice positions. They are also tabulated according to the vertical position in the reactor. For example, position 05-0 is a point on the north side of the fuel element in lattice position 5 at the mid-point of the fuel element. Position 05-6 is a point on the north side of the same fuel element 6 in. away from the mid-point. The saturated activity listed at position 05-6 is an average of the values obtained 6 in. above and 6 in. below the mid-point. The positions listed in Table II are such that a complete mapping of the flux patterns throughout the reactor can be made taking advantage of the symmetry above and below the midplane and the symmetry east and west of the centerline. Accordingly, if measurements were available at all the positions listed, a complete mapping of one quarter of the reactor could be made. However, measurements were not made at every position listed. The values in parentheses were interpolated as follows.

The bare saturated activities were plotted in six different patterns, Figs. 3 - 8. Referring to the Reactor Diagram, Fig. 1, the saturated activities in Fig. 3 are plotted against the distance from the south face of the active lattice (the plane between the fuel elements and the beryllium oxide reflector). Three families of curves are shown, with 3 curves in each family.

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TABLE 1

SUMMARY OF FLUX MEASUREMENTS IN BSF LATTICE #2

Lattice Position	Assembly Face	Vertical Position on Assembly	A_s bare	A_s Cd covered	A_s Thermal	nv Thermal	Cd Ratio
(5)	North	CL	$1.007x10^{6}$	$1.958x10^{5}$	$8.112x10^{5}$	$7.301x10^{6}$	5.14
		6"above CL	$8.203x10^{5}$				
		12" " "	$4.371x10^{5}$	$5.480x10^{4}$	$3.823x10^{5}$	$3.441x10^{6}$	7.98
		6"below "	$8.435x10^{5}$				
		12" " "	$4.551x10^{5}$	$5.058x10^{4}$	$4.045x10^{5}$	$3.641x10^{6}$	9.00
	South	CL	$1.058x10^{6}$	$3.485x10^{5}$	$7.095x10^{5}$	$6.386x10^{6}$	3.04
(25)	North	CL	$1.328x10^{6}$	$4.128x10^{5}$	$9.152x10^{5}$	$8.237x10^{6}$	3.22
		6"above CL	$1.039x10^{6}$				
		12" " "	$6.821x10^{5}$				
		6"below "	$1.090x10^{6}$				
		12" " "	$6.866x10^{5}$				
	South	CL	$1.645x10^{6}$	$4.228x10^{5}$	$1.222x10^{6}$	$1.010x10^{7}$	3.89
(45)	North	CL	$1.480x10^{6}$	$3.715x10^{5}$	$1.108x10^{6}$	$9.972x10^{6}$	3.98
	South	CL	$1.140x10^{6}$	$3.160x10^{5}$	$8.240x10^{5}$	$7.416x10^{6}$	3.61
		6"above CL	$9.155x10^{5}$				
		12" " "	$5.083x10^{5}$				
		6"below "	$9.821x10^{5}$				
		12" " "	$5.726x10^{5}$				
(7)	North	CL	$6.594x10^{5}$	$1.271x10^{5}$	$5.323x10^{5}$	$4.791x10^{6}$	5.19
		6"above CL	$5.346x10^{5}$				
		12" " "	$2.925x10^{5}$	$3.505x10^{4}$	$2.574x10^{5}$	$2.317x10^{6}$	8.34
		6"below "	$5.420x10^{5}$				
		12" " "	$2.944x10^{5}$	$3.452x10^{4}$	$2.599x10^{5}$	$2.339x10^{6}$	8.53
	South	CL	$7.334x10^{5}$				
(27)	North	CL	$9.516x10^{5}$				
	South	CL	$9.964x10^{5}$	$2.797x10^{5}$	$7.167x10^{5}$	$6.452x10^{6}$	3.56
		6"above CL	$7.878x10^{5}$				
		12" " "	$5.235x10^{5}$	$7.496x10^{4}$	$4.485x10^{5}$	$4.037x10^{6}$	6.98
		6"below CL	$7.943x10^{5}$				
		12" " "	$5.084x10^{5}$	$8.706x10^{4}$	$4.213x10^{5}$	$3.792x10^{6}$	5.84

Continued

TABLE 1 (Continued)

Lattice Position	Assembly Face	Vertical Position on Assembly	A_s bare	A_s Cd covered	A_s Thermal	nv Thermal	Cd Ratio
(47)	North	CL	$8.500x10^5$				
	South	CL	$8.311x10^5$	$2.052x10^5$	$6.259x10^5$	$5.633x10^6$	4.05
		6"above CL	$6.493x10^5$				
		12" " "	$3.997x10^5$				
		6"below "	$7.696x10^5$				
		12" " "	$4.223x10^5$	$6.104x10^4$	$3.613x10^5$	$3.252x10^6$	6.92
16	North	CL	$1.037x10^6$				
	South	CL	$1.412x10^6$				
		6"above CL	$1.187x10^6$				
		12" " "	$6.706x10^5$				
		6"below "	$1.226x10^6$				
		12" " "	$6.688x10^5$				
46	North	CL	$1.079x10^6$				
		6"above CL	$9.164x10^5$				
		12"above "	$5.908x10^5$				
		6"below "	$9.446x10^5$				
		12" " "	$6.092x10^5$				
	South	CL	$1.072x10^6$	$2.876x10^5$	$7.844x10^5$	$7.060x10^6$	3.73

TABLE II

NEUTRON TRAVERSES THROUGH THE REACTOR

Values in Parentheses are Interpolated from the Appropriate Curves

North Side of Lattice Position	A_s Total	A_s Epicadmium	A_s Thermal	nv_{th}	Cd Ratio
05-0	1.007×10^6	1.958×10^5	8.112×10^5	7.301×10^6	5.14
15-0	1.058×10^6	3.485×10^5	7.095×10^5	6.385×10^6	3.04
25-0	1.328×10^6	4.128×10^5	9.152×10^5	8.237×10^6	3.22
35-0	1.645×10^6	4.228×10^5	1.222×10^6	1.100×10^7	3.89
45-0	1.480×10^6	3.715×10^5	1.108×10^6	9.972×10^6	3.98
55-0	1.140×10^6	3.160×10^5	8.240×10^5	7.416×10^6	3.61
05-6	8.319×10^5	(1.33×10^5)	(6.99×10^5)	(6.29×10^6)	(6.25)
15-6	(8.6×10^5)	(2.12×10^5)	(6.48×10^5)	(5.83×10^6)	(4.06)
25-6	1.065×10^6	(2.59×10^5)	(8.06×10^5)	(7.25×10^6)	(4.11)
35-6	(1.35×10^6)	(2.94×10^5)	(1.06×10^6)	(9.54×10^6)	(4.59)
45-6	(1.21×10^6)	(2.53×10^5)	(9.57×10^5)	(8.61×10^6)	(4.79)
55-6	9.488×10^5	(2.04×10^5)	(7.45×10^5)	(6.70×10^6)	(4.64)
05-12	4.461×10^5	5.269×10^4	3.934×10^5	3.537×10^6	8.47
15-12	(4.9×10^5)	(8.03×10^4)	(4.10×10^5)	(3.69×10^6)	(6.1)
25-12	6.844×10^5	(1.16×10^5)	(5.68×10^5)	(5.11×10^6)	(5.9)
35-12	(9.05×10^5)	(1.51×10^5)	(7.54×10^5)	(6.79×10^6)	(6.0)
45-12	(7.5×10^5)	(1.17×10^5)	(6.33×10^5)	(5.70×10^6)	(6.4)
55-12	5.405×10^5	(8.07×10^4)	(4.60×10^5)	(4.14×10^6)	(6.7)

Continued

TABLE II - Cont'd

NEUTRON TRAVERSES THROUGH THE REACTOR

North Side of Lattice Position	A_s Total	A_s Epicadmium	A_s Thermal	nv_{th}	Cd Ratio
06-0	(9.0×10^5)	(1.74×10^5)	(7.26×10^5)	(6.53×10^6)	(5.17)
16-0	1.037×10^6	(3.46×10^5)	(6.91×10^5)	(6.22×10^6)	(3.0)
26-0	1.412×10^6	(4.34×10^5)	(9.78×10^5)	(8.80×10^6)	(3.25)
36-0	(1.49×10^6)	(3.97×10^5)	(1.09×10^6)	(9.81×10^6)	(3.75)
46-0	1.079×10^6	(2.73×10^5)	(8.06×10^5)	(7.25×10^6)	(3.95)
56-0	1.072×10^6	2.876×10^5	7.844×10^5	7.060×10^6	3.73
06-6	(7.5×10^5)	(1.2×10^5)	(6.3×10^5)	(5.67×10^6)	(6.27)
16-6	(8.4×10^5)	(2.08×10^5)	(6.32×10^5)	(5.69×10^6)	(4.03)
26-6	1.206×10^6	(2.92×10^5)	(9.14×10^5)	(8.23×10^6)	(4.13)
36-6	(1.25×10^6)	(2.78×10^5)	(9.72×10^5)	(8.75×10^6)	(4.5)
46-6	9.305×10^5	(1.95×10^5)	(7.35×10^5)	(6.61×10^6)	(4.77)
52-6	(9.0×10^5)	(1.91×10^5)	(7.09×10^5)	(6.38×10^6)	(4.72)
06-12	(4.0×10^5)	(4.7×10^4)	(3.53×10^5)	(3.18×10^6)	(8.46)
16-12	(4.6×10^5)	(7.5×10^4)	(3.85×10^5)	(3.46×10^6)	(6.10)
26-12	$6.697 \times 10^5)$	(1.13×10^5)	(5.56×10^5)	(5.00×10^6)	(5.90)
36-12	(7.8×10^5)	(1.30×10^5)	(6.50×10^5)	(5.85×10^6)	(6.0)
46-12	6.000×10^5	(9.4×10^4)	(5.06×10^5)	(4.55×10^6)	(6.4)

Continued

TABLE II - Cont'd

NEUTRON TRAVERSES THROUGH THE REACTOR

North Side of Lattice Position	A_s Total	A_s Epicadmium	A_s Thermal	nv_{th}	Cd Ratio
07-0	6.594×10^5	1.271×10^5	5.323×10^5	4.791×10^6	5.19
17-0	7.334×10^5	(2.44×10^5)	(4.89×10^5)	(4.40×10^6)	(3.0)
27-0	9.516×10^5	(2.93×10^5)	(6.59×10^5)	(5.93×10^6)	(3.25)
37-0	9.964×10^5	2.797×10^5	7.167×10^5	6.450×10^6	3.56
47-0	8.500×10^5	(2.15×10^5)	(6.35×10^5)	(5.71×10^6)	(3.95)
57-0	8.311×10^5	2.052×10^5	6.259×10^5	5.633×10^6	4.05
07-6	5.383×10^5	(8.6×10^4)	(4.52×10^5)	(4.07×10^6)	(6.27)
17-6	(5.9×10^5)	(1.46×10^5)	(4.44×10^5)	(4.00×10^6)	(4.03)
27-6	(7.5×10^5)	(1.82×10^5)	(5.68×10^5)	(5.11×10^6)	(4.13)
37-6	7.910×10^5	(1.81×10^5)	(6.10×10^5)	(5.49×10^6)	(4.37)
47-6	(7.3×10^5)	(1.53×10^5)	(5.77×10^5)	(5.19×10^6)	(4.77)
57-6	7.094×10^5	(1.44×10^5)	(5.65×10^5)	(5.08×10^6)	(4.94)
07-12	2.934×10^5	3.478×10^4	2.586×10^5	2.327×10^6	8.44
17-12	(3.2×10^5)	(5.2×10^4)	(2.68×10^5)	(2.41×10^6)	(6.10)
27-12	(4.9×10^5)	(8.3×10^4)	(4.07×10^5)	(3.66×10^6)	(5.90)
37-12	5.159×10^5	8.601×10^4	4.299×10^5	3.869×10^6	6.00
47-12	(4.1×10^5)	(6.4×10^4)	(3.46×10^5)	(3.11×10^6)	(6.40)
57-12	4.110×10^5	6.104×10^4	3.500×10^5	3.15×10^6	6.73

[redacted]

Defining fuel elements 3, 4, 5, 6, and 7 to constitute a row, and elements 5, 15, 25, 35, and 45 to be a column, each family contains 3 curves plotted by columns. The three families of curves show the traverses according to the horizontal planes of the reactor.

In Fig. 4 the same curves are plotted, but in this case a family of curves is plotted by planes and each family represents a column.

In Figs. 5 and 6 the curves are east-west traverses. Saturated activities are plotted against the distance from the centerline of the reactor. In Fig. 5 the families are by rows while in Fig. 6 they are by planes. There are six curves by rows in each family in Fig. 6, and for ease in interpolation each family was divided into two sub-families as shown.

Figs. 7 and 8 are vertical traverses along the individual fuel elements. The families in Fig. 7 are by rows and in Fig. 8 by columns.

By plotting the saturated activities in six different ways, it was found that the values at the points not measured could be interpolated quite closely. It is felt that the interpolated values for the bare saturated activities (shown in parentheses in Table II) are good within a few percent.

Since not as many epicadmium measurements were taken as without cadmium, it was also necessary to interpolate cadmium ratios. The cadmium ratios obtained are also listed in Table II and are plotted in three different patterns in Figs. 9, 10, and 11. It can be seen that the cadmium ratios vary much more slowly with position than do the bare saturated activities and the missing points can be interpolated quite easily.

Having values for the saturated activities, bare and cadmium covered, the thermal saturated activities were then calculated and listed in Table II. While some of these values individually could be off as much as 10%, it is estimated that, on the average, the thermal saturated activities listed are good within 5%.

The thermal saturated activities were converted to thermal neutron fluxes which are also given in Table II.

Figs. 12a and 12b are plots of the thermal neutron flux in the midplane of the reactor.

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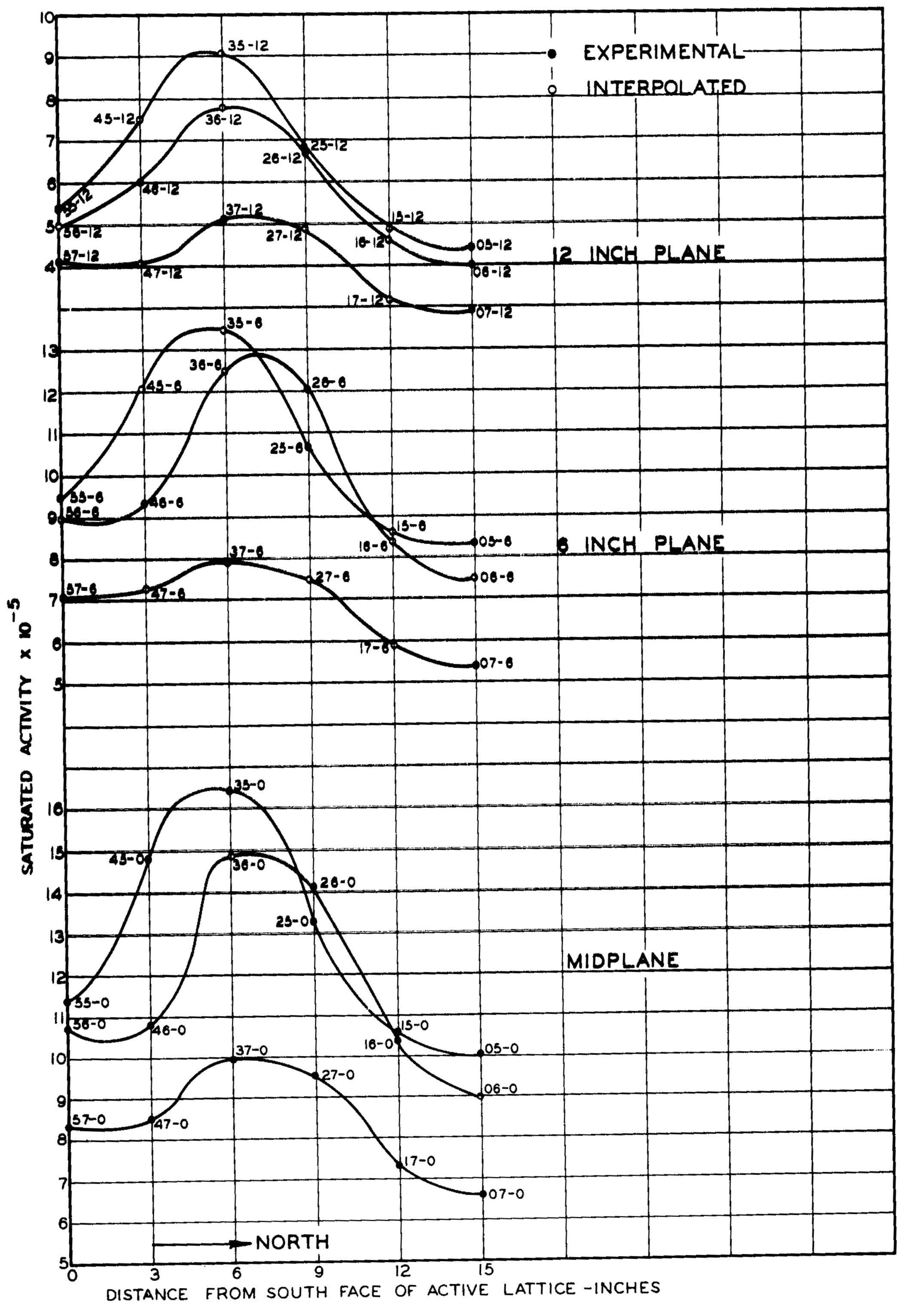

FIG. 3
BARE FOILS
NORTH —— SOUTH TRAVERSES BY PLANES

DWG. NO. 11306

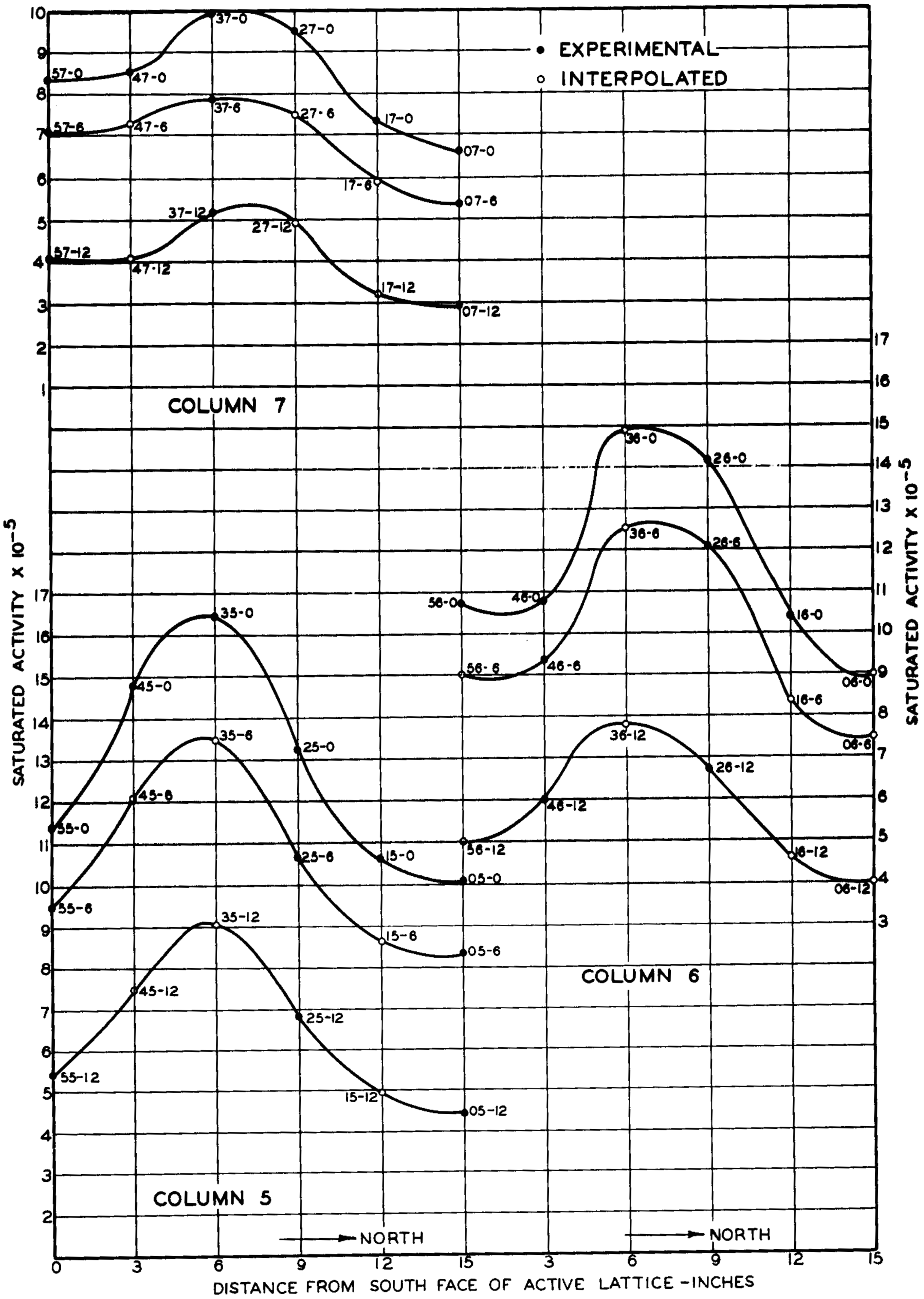

FIG. 4

BARE FOILS

NORTH - SOUTH TRAVERSES BY COLUMNS

DWG. NO. 11307

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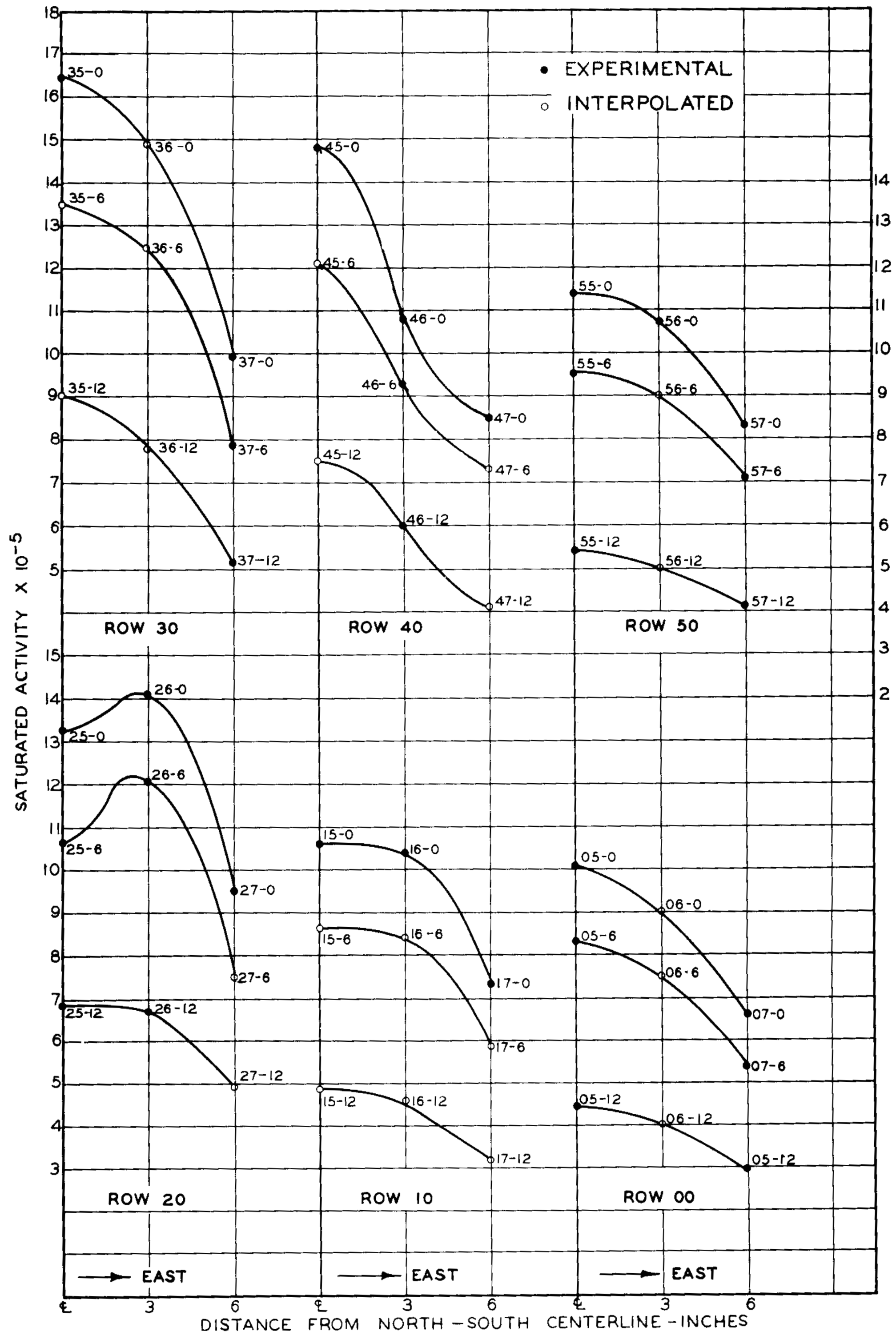

FIG. 5
BARE FOILS
EAST-WEST TRAVERSES BY ROWS

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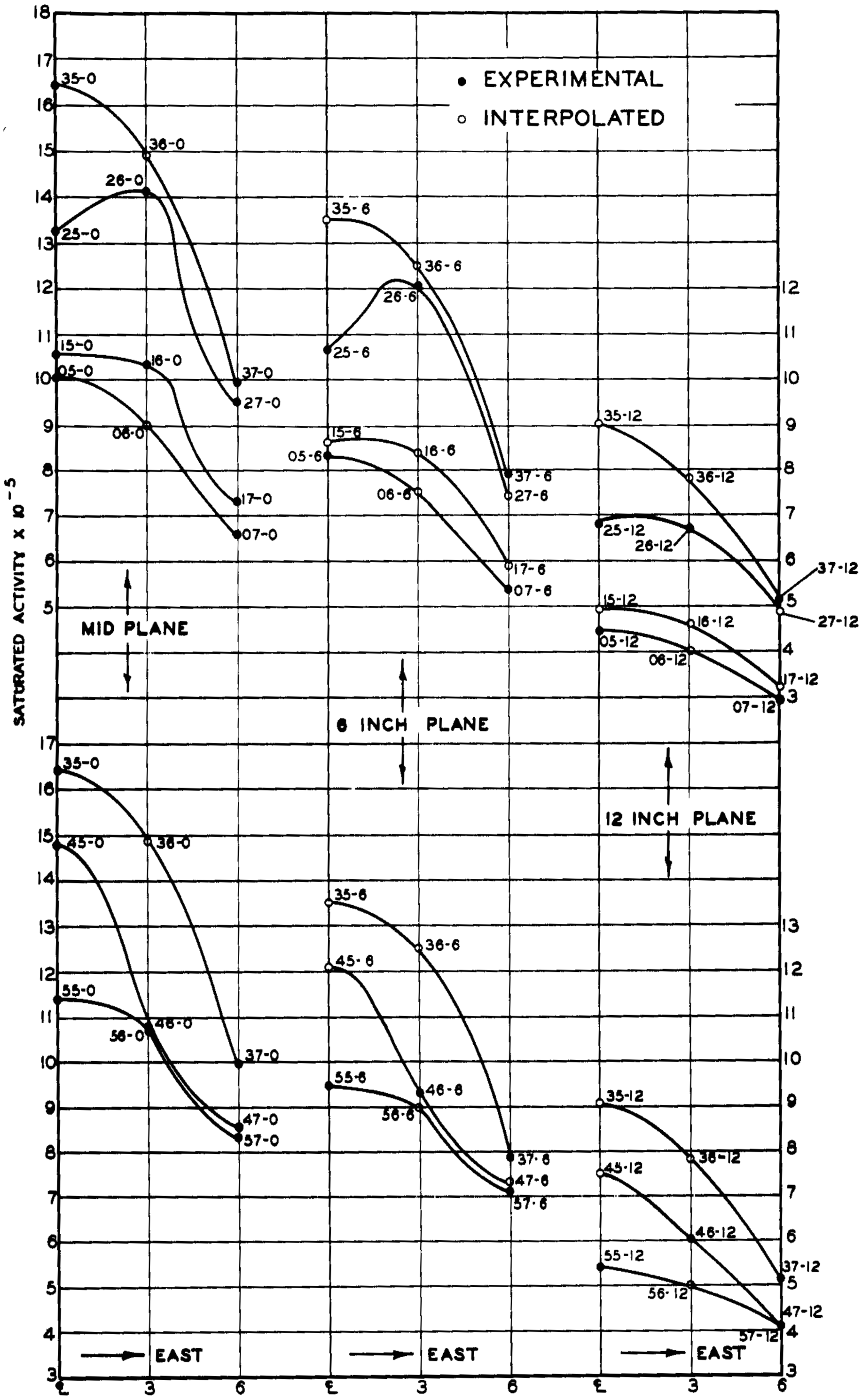

FIG. 6
BARE FOILS
EAST—WEST TRAVERSES BY PLANES

DWG. NO. 11309

[redacted]

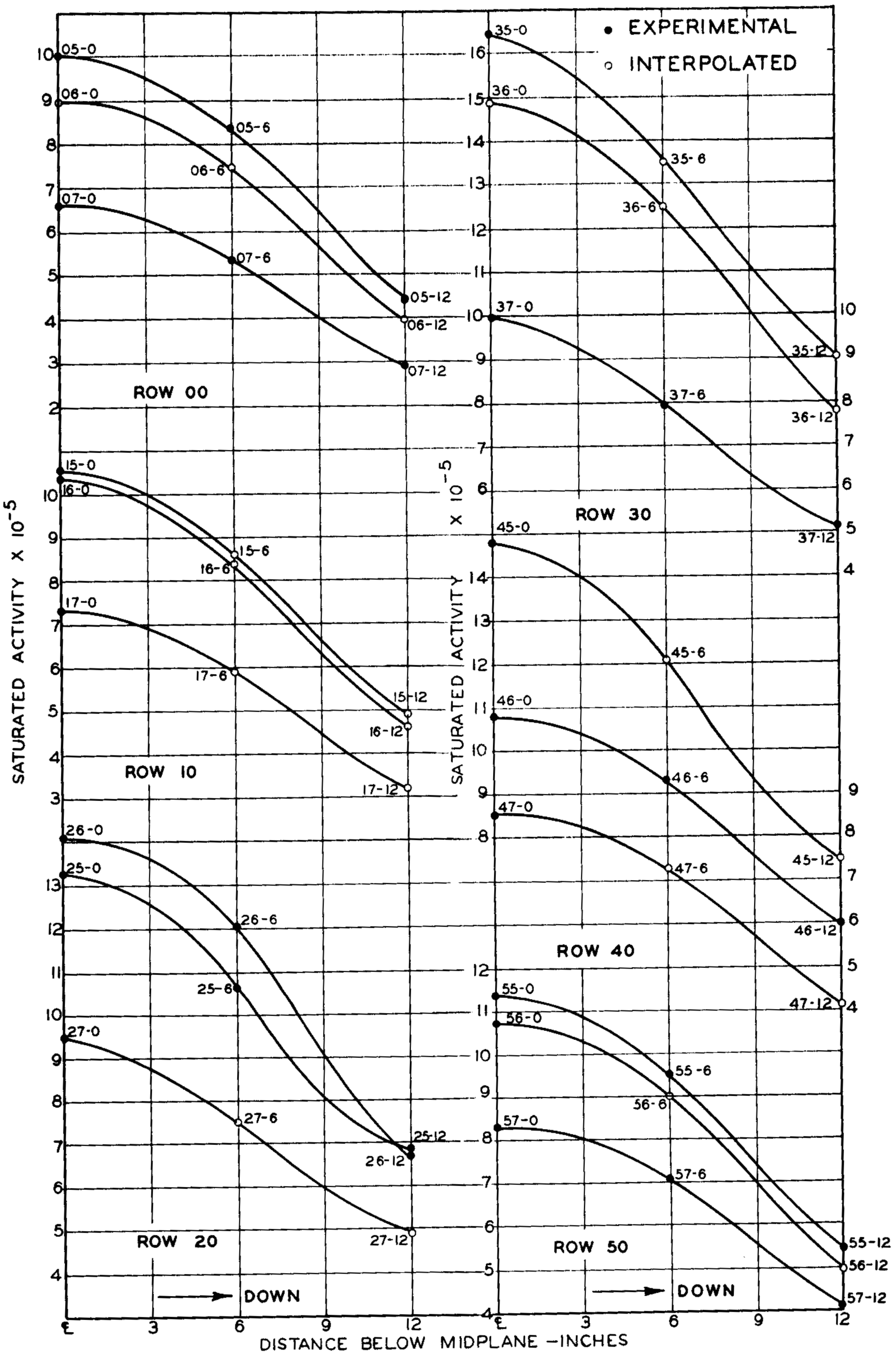

FIG. 7
BARE FOILS
VERTICAL TRAVERSES BY ROWS

DWG. NO. 11313
[redacted]

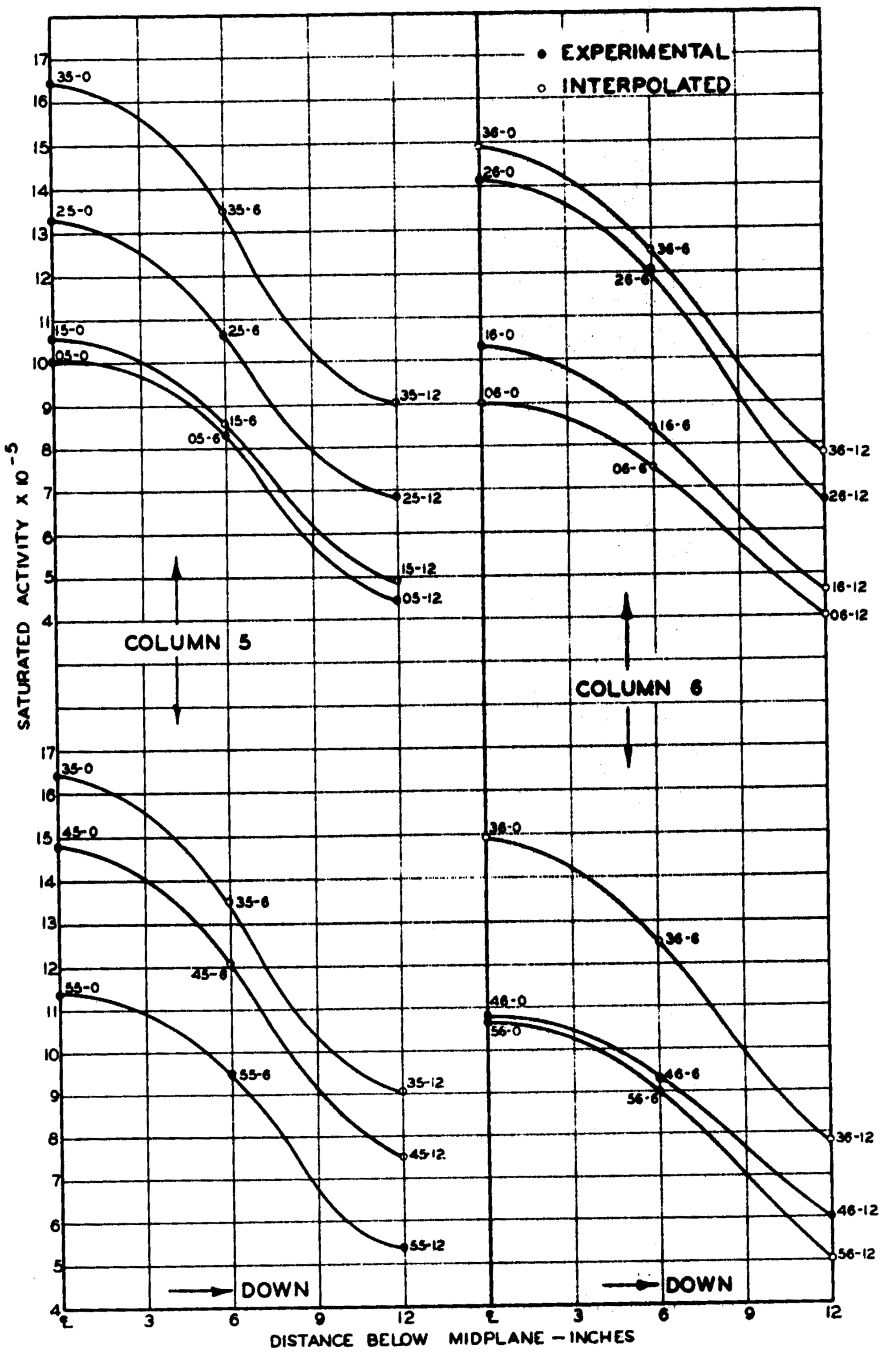

FIG. 8a
BARE FOILS
VERTICAL TRAVERSES BY COLUMNS

DWG. NO. 11312
[redacted]

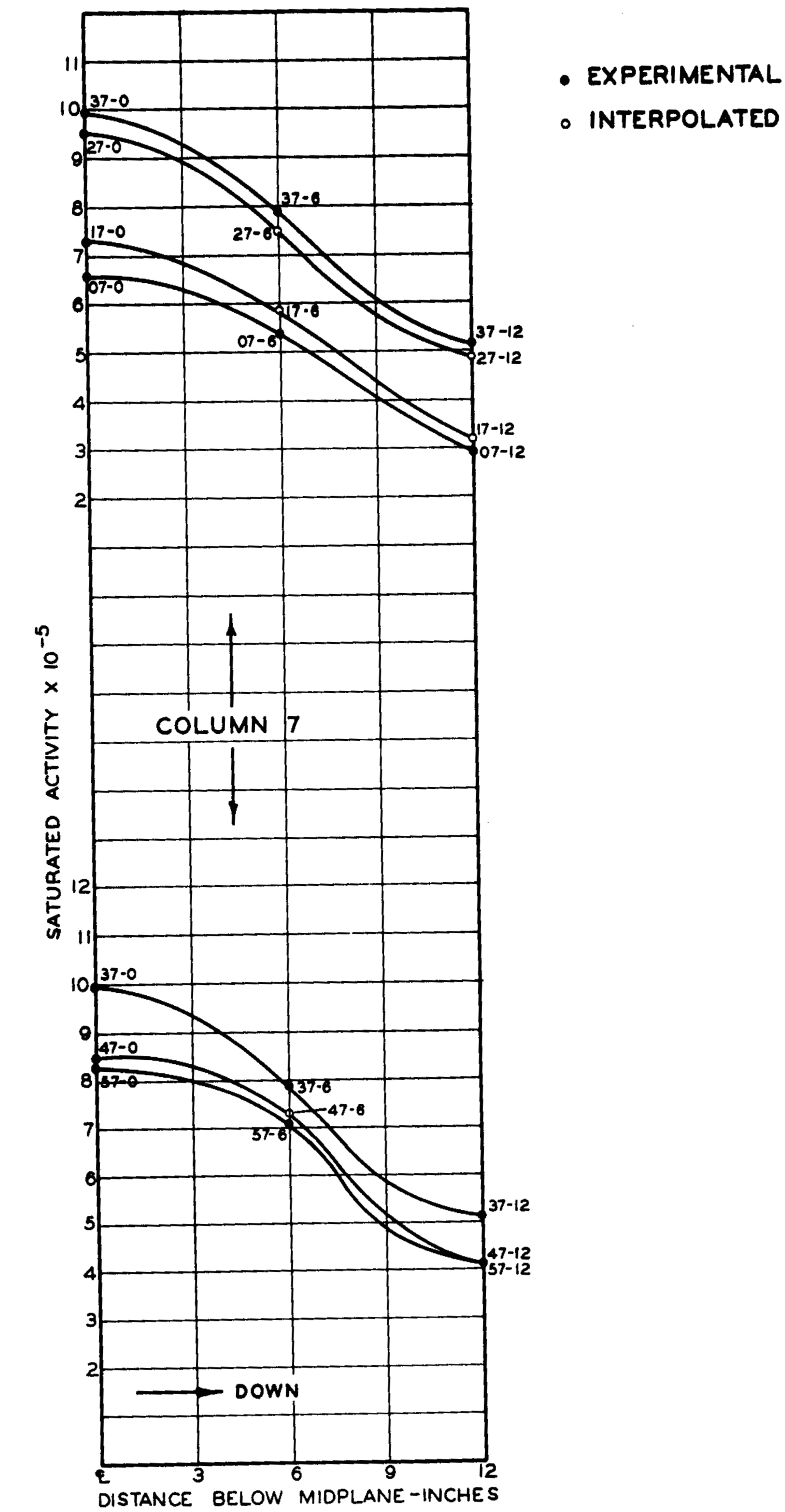

FIG. 8b
BARE FOILS
VERTICAL TRAVERSES BY COLUMNS

DWG. NO. 11304
[redacted]

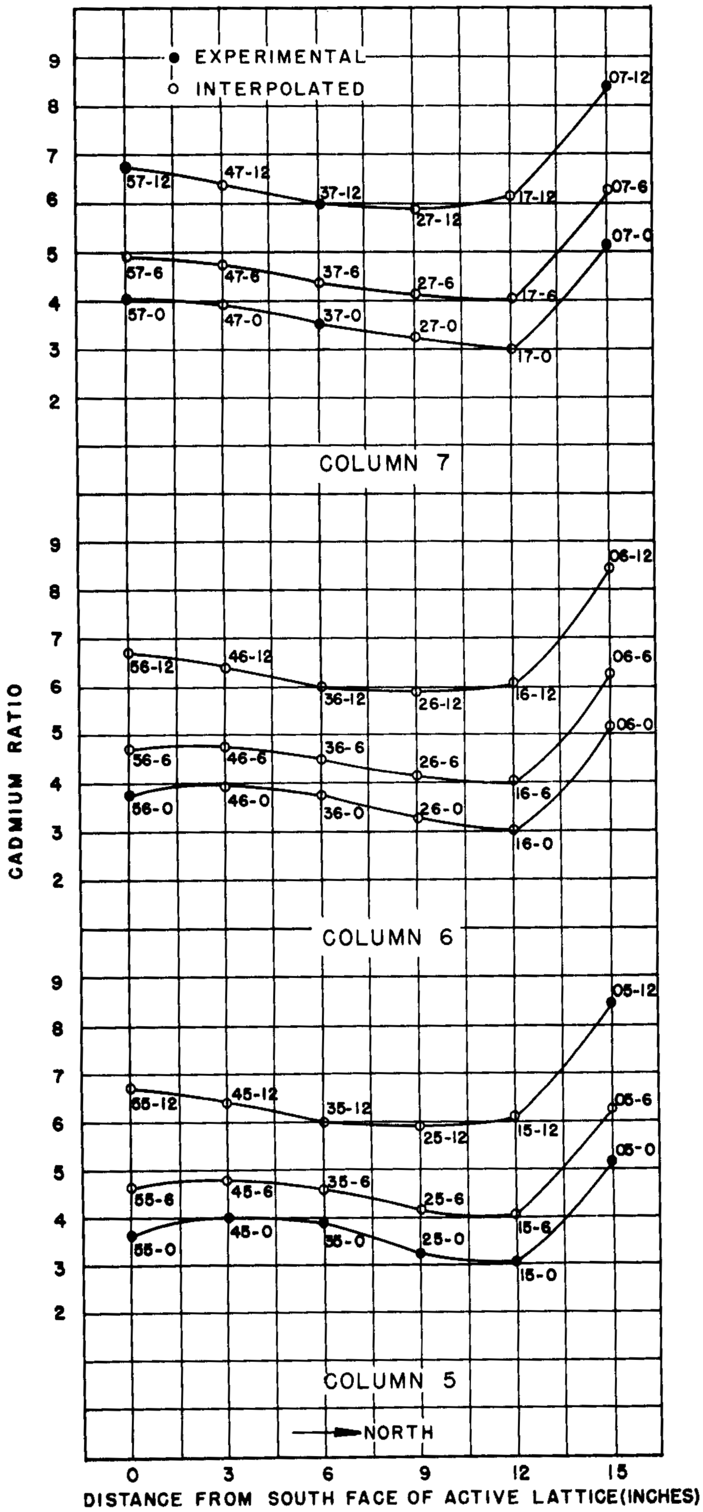

FIG. 9

CADMIUM RATIOS

NORTH-SOUTH TRAVERSES BY COLUMNS

[redacted]WG NO. 11310

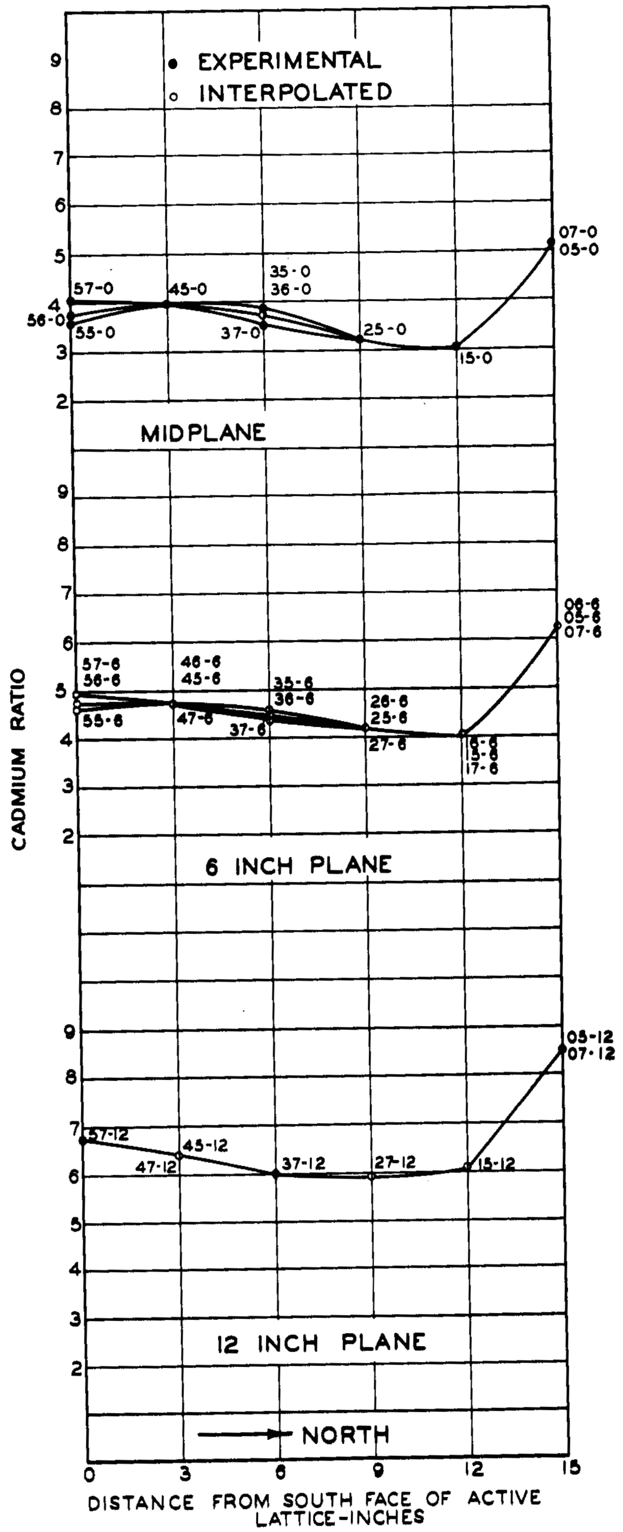

FIG 10

CADMIUM RATIOS

NORTH-SOUTH TRAVERSES BY PLANES

DWG. NO. 11311

[redacted]

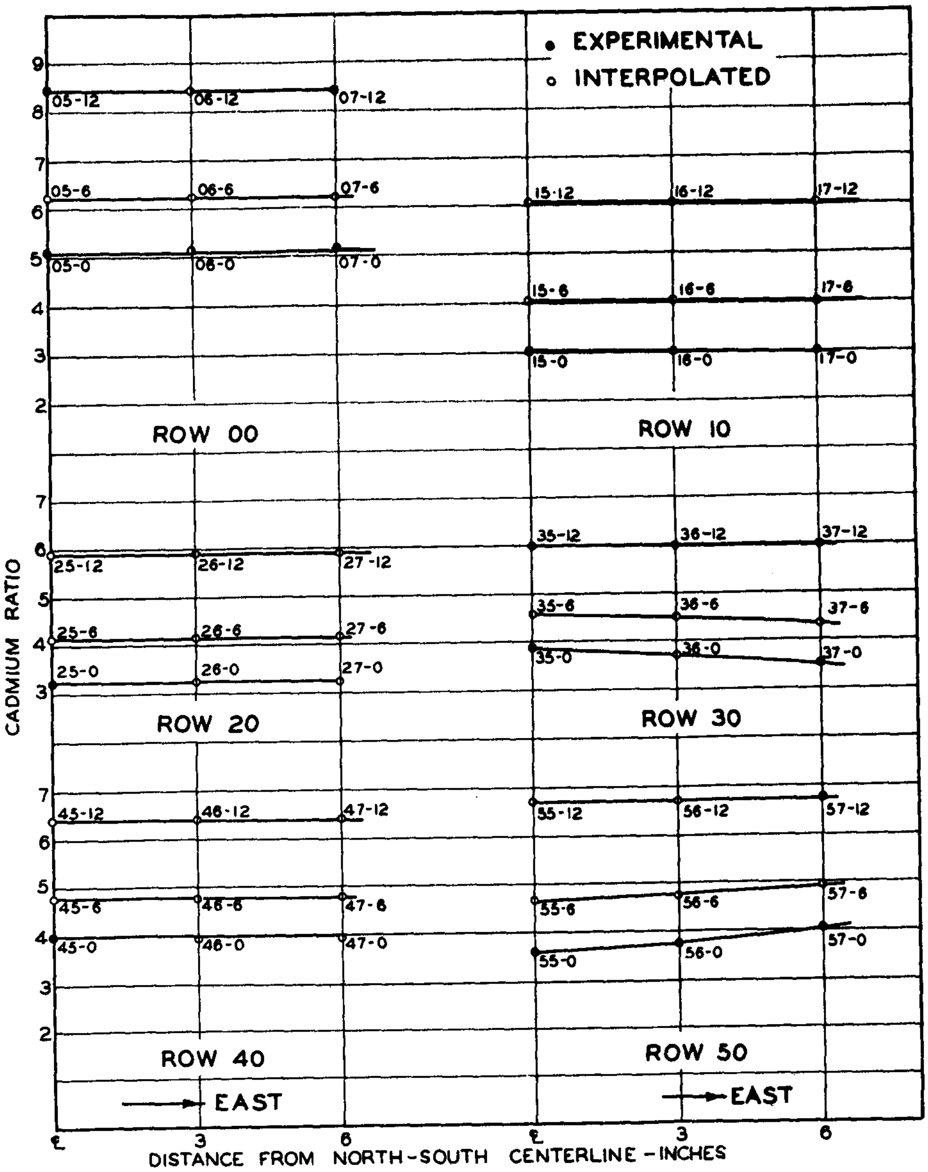

FIG. 11
CADMIUM RATIOS
EAST-WEST TRAVERSES BY ROWS

DWG. NO. 11303

[redacted]

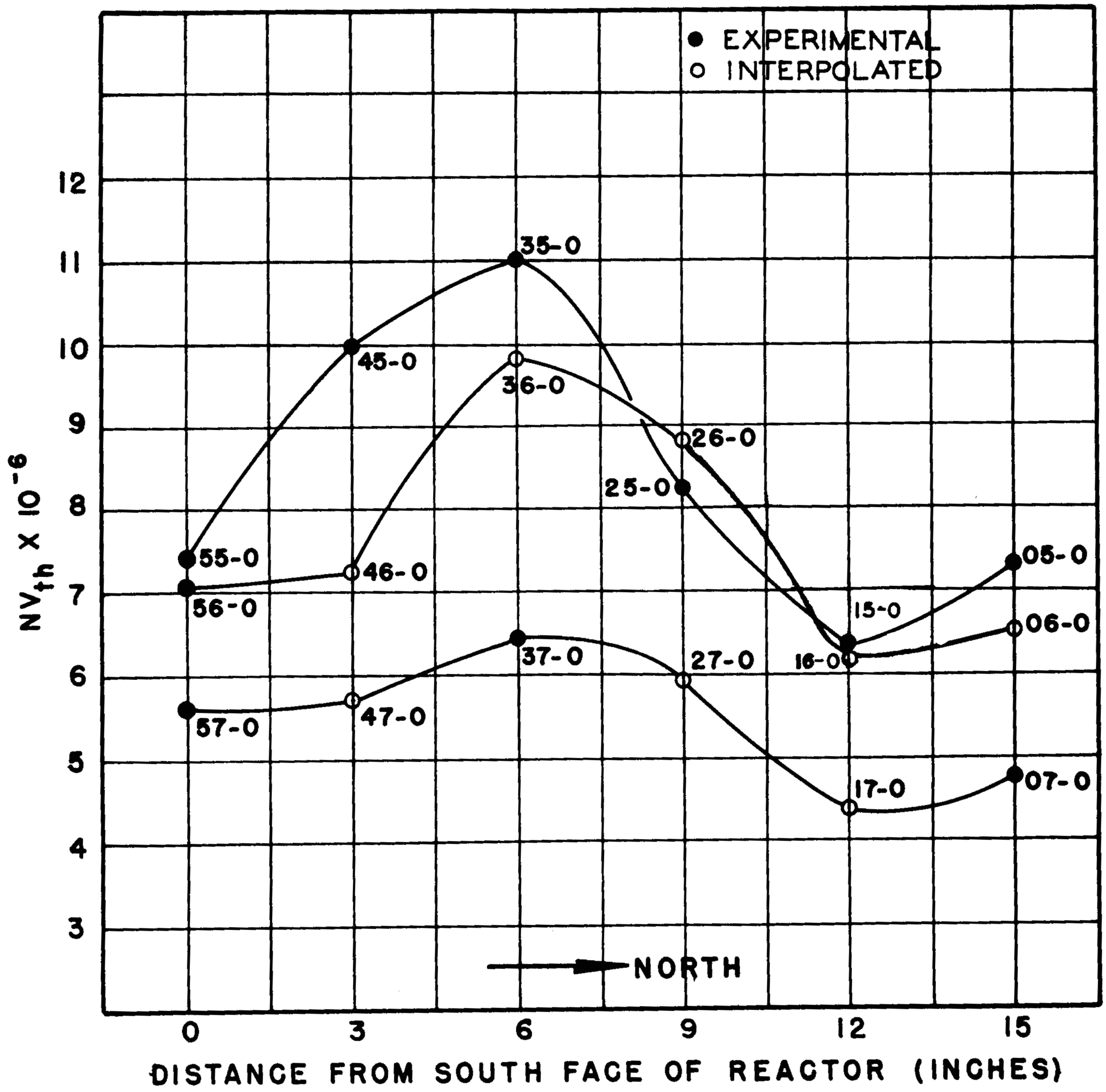

FIG. 12a

THERMAL NEUTRON FLUX

NORTH-SOUTH TRAVERSES AT MIDPLANE

DWG. NO. 11302

[redacted]

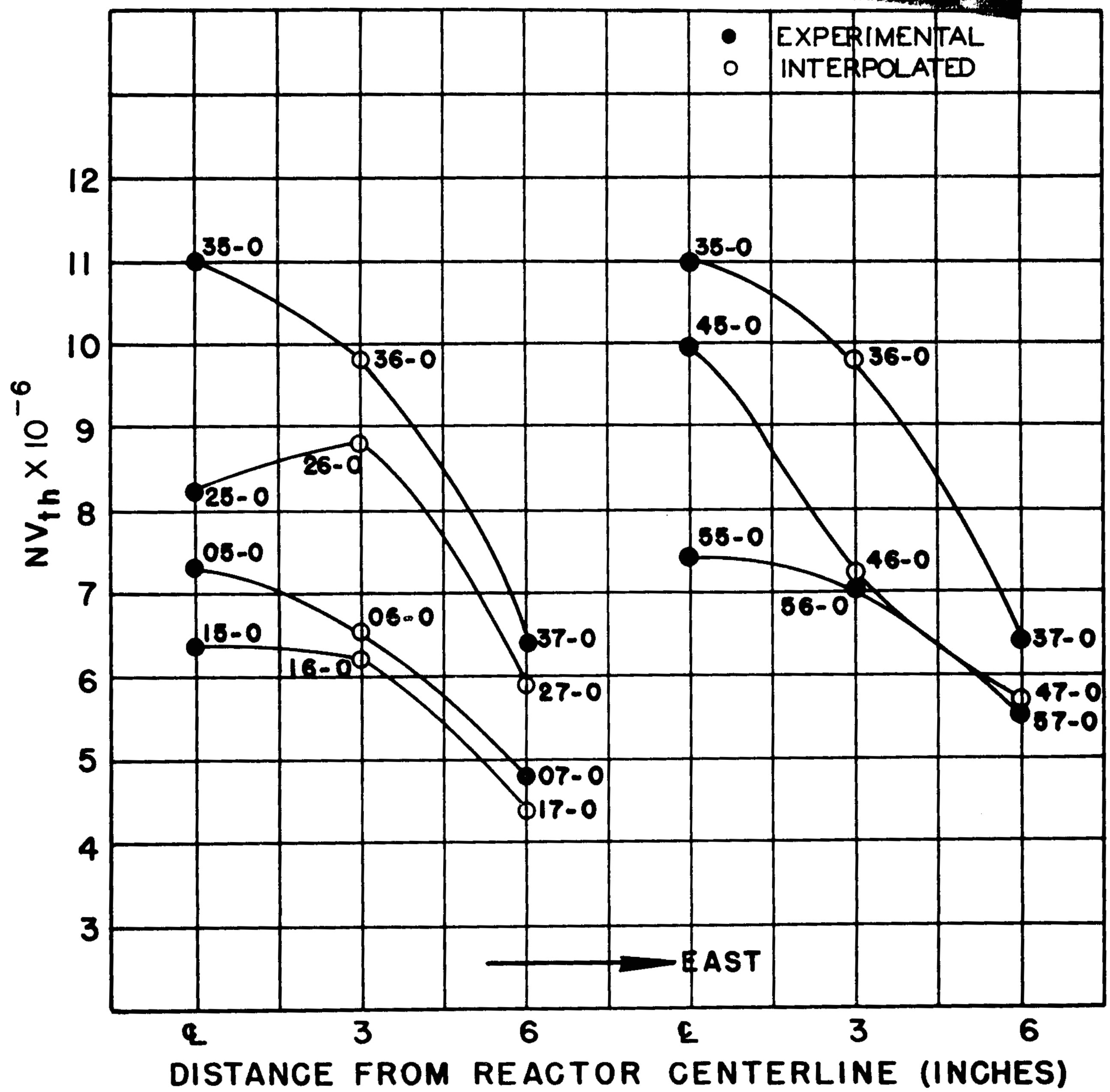

FIG. 12b.

THERMAL NEUTRON FLUX

EAST–WEST TRAVERSES AT MIDPLANE

[redacted]

Referring to Fig. 2, it can be seen that values of the thermal flux were now available at four points around each quarter fuel element. The values at the center of each quarter element were obtained by averaging the values at the four points and are listed in Table III. The flux at the center of the interior quarter elements might be a little higher than such an averaging gives, but the flux at the center of the end quarter elements is probably lower for the same reason. The true shape of the vertical flux traverses was not known well enough to warrant a better procedure for determining the center fluxes of the quarter elements. It is felt that the errors introduced are compensating.

The average flux per fuel element (Table III) was obtained by averaging the fluxes of the two quarters of each element.

Since the reactor was symmetrical about a vertical plane through the north-south principal axis, the average flux for each fuel element in the reactor was now available. These are listed in Table IV.

The number of grams of U^{235} and the average thermal flux were known for each fuel element in the reactor (Table IV). Measurements in a graphite column at a point where the cadmium ratio was 3.7, using first a bare and then a cadmium covered fission chamber, have been reported.[4] These measurements indicated that 93% of the fissions occurred at energies below the cadmium cut-off. Since inspection of Table II shows that the cadmium ratio in the interior of the reactor was in the neighborhood of 3.7, the number of thermal fissions per fuel element was multiplied by 1.07 to get the total number of fissions. It was assumed that the fission cross-section was 545 barns[5] and that 165 Mev of heat energy was generated in the reactor for each fission[6].

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(4) Mann, M. M. and A. B. Martin, "Further Critical Experiments on a Small Reactor of Enriched U-235 with Al-H_2O Moderator and Beryllium Reflector", ORNL-79, p 17, Sept. 16, 1948

(5) "Nuclear Data for Low-Power Research Reactors", TID-235 (1950)

(6) Weinberg, A. M., Private Communication

[redacted]

TABLE III

THERMAL NEUTRON FLUX

Center of Lattice Position	Quarter Elements Bounded by Midplane and 6" Plane	Quarter Elements Bounded by 6" Plane and 12" Plane	Average nv_{th} per Element
05	6.45×10^6	4.84×10^6	5.645×10^6
15	6.93×10^6	5.47×10^6	6.20×10^6
25	9.01×10^6	7.17×10^6	8.09×10^6
35	9.78×10^6	7.66×10^6	8.72×10^6
45	8.17×10^6	6.29×10^6	7.23×10^6
06	6.03×10^6	4.50×10^6	5.265×10^6
16	7.24×10^6	5.60×10^6	6.42×10^6
26	8.90×10^6	6.96×10^6	7.93×10^6
36	8.10×10^6	6.44×10^6	7.27×10^6
46	6.82×10^6	5.34×10^6	6.08×10^6
07	4.31×10^6	3.20×10^6	3.755×10^6
17	4.86×10^6	3.79×10^6	4.325×10^6
27	5.74×10^6	4.53×10^6	5.135×10^6
37	5.71×10^6	4.41×10^6	5.06×10^6
47	5.40×10^6	4.13×10^6	4.765×10^6

TABLE IV

Lattice Position	G Grams of U^{235}	Average nv_{th} per Element	P Power in Watts
03	137.37	$3.755x10^6$	0.02034
04	137.74	$5.265x10^6$	0.02860
05	137.39	$5.645x10^6$	0.03059
06	136.48	$5.265x10^6$	0.02834
07	135.02	$3.755x10^6$	0.02000
13	138.54	$4.325x10^6$	0.02363
14	139.98	$6.42x10^6$	0.03544
15	138.54	$6.20x10^6$	0.03388
16	139.95	$6.42x10^6$	0.03544
17	138.41	$4.325x10^6$	0.02361
23	134.90	$5.135x10^6$	0.02732
24	69.70	$7.93x10^6$	0.02180
25	139.98	$8.09x10^6$	0.04466
26	69.08	$7.93x10^6$	0.02160
27	136.39	$5.135x10^6$	0.02762
33	139.95	$5.06x10^6$	0.02792
34	137.19	$7.27x10^6$	0.03934
35	69.14	$8.72x10^6$	0.02378
36	137.19	$7.27x10^6$	0.03934
37	138.10	$5.06x10^6$	0.02756
43	134.90	$4.765x10^6$	0.02535
44	137.53	$6.08x10^6$	0.03298
45	138.54	$7.23x10^6$	0.03950
46	138.54	$6.08x10^6$	0.03322
47	138.22	$4.765x10^6$	0.02598
			0.7378

[redacted]

Let F = the number of fissions per second in each element

N = the number of atoms per cc of U^{235}

σ = the thermal neutron cross-section of U^{235}

V = the volume of a fuel element

1.07 = total fissions per thermal fission

$$\text{Then } F = 1.07\, N \sigma V\, (nv_{th}) \tag{1}$$

$$\text{But } N = G/V \cdot \frac{6.02 \times 10^{23}}{235} \tag{2}$$

Where G = the number of gm. of U^{235} in the volume V of a fuel element.

If P = the power in watts per fuel element,

$$P = F \text{ fissions/sec} \times 165 \text{ Mev/fission} \times 1.60 \times 10^{-13} \text{ joules/Mev} \tag{3}$$

$$= 1.07 \frac{G}{V} \cdot \frac{6.02 \times 10^{23}}{235} \cdot 545 \times 10^{-24} \cdot V \cdot nv_{th} \cdot 165 \cdot 1.60 \times 10^{-13}$$

$$= 3.944 \times 10^{-11}\, G \cdot nv_{th}$$

The power in watts for each fuel element is given in Table IV. The total power for the 25 elements in this reactor is the sum of the powers in the individual elements, 0.74 watts. The power distribution through the different fuel elements is plotted in Fig. 13. The points in Fig. 13 are not to be compared directly with the corresponding points in Fig. 12a. The points for the power distribution represent the power at the center of the particular fuel elements, while in Fig. 12a the thermal neutron fluxes are plotted with respect to the north side of each fuel element.

It is interesting to observe that the power in positions 24, 26, and 35 was quite low relative to that in the surrounding elements. This was not caused by the presence of the control rods as these rods were withdrawn from

[redacted]

DWG. NO. 11301
[redacted]

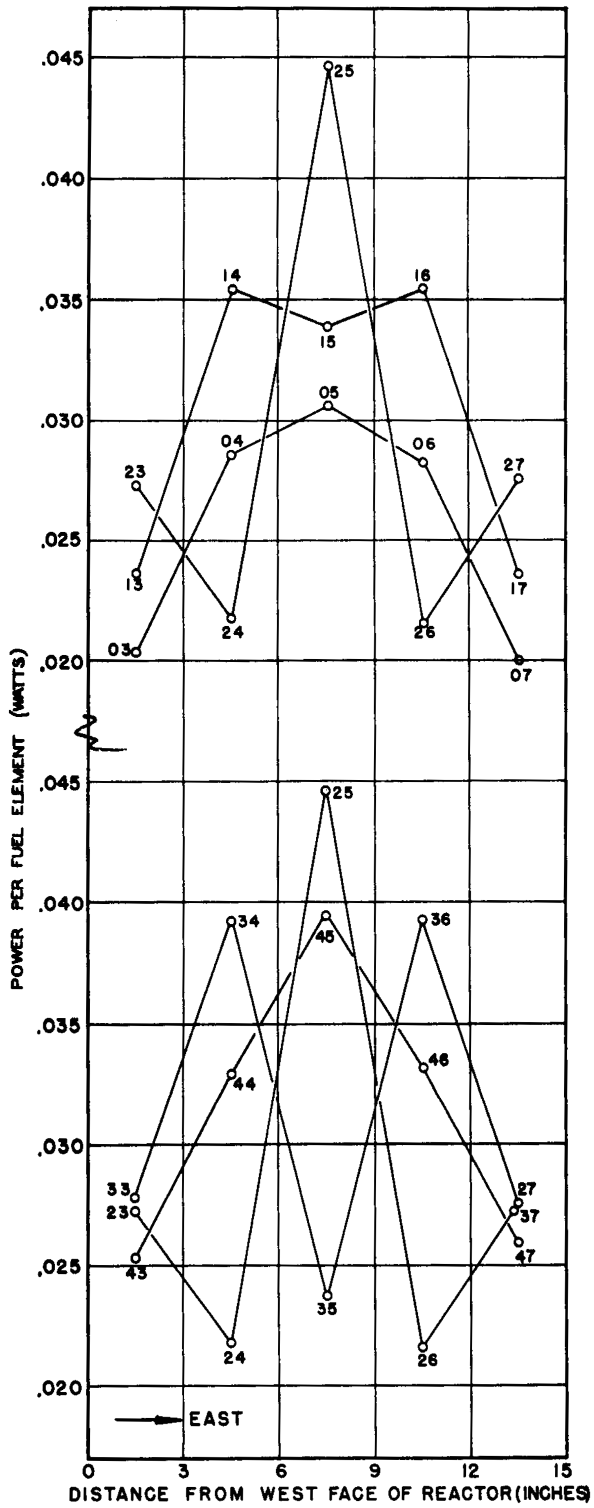

FIG. 13
POWER DISTRIBUTION
EAST-WEST TRAVERSES BY ROWS

[redacted]

the reactor, Rather, to make room for the rods, half the fuel has been removed from these three elements, and with the rods withdrawn this fuel was replaced with water causing the drop in power production.

Discussion

The result of the calculations shows that the approximate "1 watt" power level of the reactor was actually 0.74 watts. All shielding data that have been reported to date have been normalized to 1 watt by this factor.

By suspending a U^{235} fission chamber out in the water at a fixed distance in front of the reactor, a check on the reproducibility of the reactor power was made. It was found that the reactor power could be duplicated from day to day within a few percent. Also, it was found that the automatic servo control could be set at factors of ten above the original power of 0.74 watts within 1%. The reactor has been operated at power levels up to 7.4 kw.

It is estimated that the probable error in determining the thermal neutron flux distribution throughout the reactor is within 5%, assuming the standard graphite pile at ORNL gives a correct calibration of the foils. Present indications are that this standard pile may be off by as much as 20%. Dr. E. D. Klema of the Physics Division is now recalibrating that pile[6]. The value of 165 Mev per fission is also the subject of some controversy at this time. There may be an error in the value 1.07 total fissions per thermal fission although this error is probably small compared to the others.

An independent check on the power of this reactor is underway. Water will be forced through some of the fuel elements at a given rate and the temperature rise of the water measured with thermocouples. From these measurements, it is hoped that the power per fuel element can be calculated directly. If by that time the re-standardization of the standard graphite pile is complete so that the absolute value of the thermal neutron flux in the reactor is known, then an independent check on the amount of heat released per fission will be available.

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(6) "Physics Quarterly Progress Report for Period Ending March 20, 1951", ORNL-1005, July, 1951

[redacted]