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CRITICAL STUDIES OF A DILUTE CARBIDE FAST REACTOR CORE

ZPR-III Assembly 34

by

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Idaho Division

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DISCLAIMER

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ABSTRACT

Critical studies were made with a simulated, large, dilute power reactor having uranium carbide as fuel. The uranium in the core was 30.7% enriched, and the atomic ratio of uranium to carbon was 0.946. The critical mass was 503.01 kg U^{235} and the critical volume 574.47 liters. Central reactivity coefficients, effective fission cross-section ratios, heterogeneity effects, reactivity worth of distributed materials, foil irradiations, and the average prompt neutron lifetime were measured. Multigroup calculations using the Yiftah, Okrent, and Moldauer cross-section set overestimate k for the critical configuration by 4.7%.

INTRODUCTION

ZPR-III Assembly 34 was a critical experiment carried out to measure the nuclear properties of a simulated fast power reactor having uranium carbide as fuel. It was one of a series of similar experiments constructed to study the characteristics of large, dilute, fast power reactors with ceramic fuels. Other assemblies in this group have simulated uranium oxide fuels of varying atomic ratios of uranium to oxygen. Diluent materials in the core, with sodium replaced by aluminum, were those which would probably be used as structural or coolant materials in a power reactor. The properties measured included those bearing directly on the physics of the core, such as critical mass and volume, and fission ratios, along with information of importance to reactor design, such as the reactivity worth of the possible structural and coolant materials. Some effort was made to compare the experimental results with calculated values. The calculations were multigroup calculations using the sixteen-group cross-section set of Yiftah, Okrent, and Moldauer,[1]* and the SNG and DSN transport theory codes.[2,3]

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*Subsequently referred to as the "YOM" cross-section set.

DESCRIPTION OF REACTOR

The ZPR-III is a fast critical facility designed so that a great variety of cores of differing shapes, sizes, and compositions may be conveniently constructed. A description of the reactor has been given by Cerutti et al.[4]; however, a résumé follows:

The reactor (see Fig. 1) is constructed in two equal halves, one of which is moveable. Each half, which is equipped with four safety rods and one control rod, all of the fuel-removal type, is primarily a matrix for the insertion of long drawers which contain the core and blanket materials. The drawers have a rectangular cross section about 2 x 2 in. The core is built up in two equal parts, one in each half of the machine. The materials used are $\frac{1}{8}$-in.-thick plates, which are loaded into the drawers in any desired arrangement. The blanket, also loaded into the matrix tubes in each half of the reactor, is composed of larger pieces of depleted uranium. Figure 2 shows core drawers loaded with core and blanket materials. Criticality is achieved by remotely driving the halves of the machine together and inserting the eight safety rods and two control rods.

COMPOSITION OF ASSEMBLY 34

This assembly was intended to mockup a typical fast reactor fueled with uranium monocarbide. It contained highly enriched uranium, depleted uranium, graphite, stainless steel, and aluminum. Considering the core in its entirety, the atomic ratio of uranium to carbon was 0.946. The overall enrichment of the core was 30.7%. The composition of the critical core is given in Table I.

Table I

COMPOSITION OF CRITICAL CORE

Material	Quantity (kg)	Volume Per Cent (v/o)	Assumed Density of 100% v/o Material, gm/cm^3
U^{235}	503.01	4.67	18.75
U^{238}	1129.30	10.35	19.0
Stainless Steel	1111.27	24.64	7.85
Aluminum	394.90	25.46	2.70
Carbon	87.37	10.64	1.43

The approximately 25 volume per cent aluminum is used to simulate 50 volume per cent sodium.

The assembly in its final form was fully blanketed with a minimum of 8 in. of depleted uranium. The blanket region was composed of 83.3 v/o U^{238}, 7.31 v/o stainless steel, and 0.19 v/o U^{235}. The U^{235} was that remaining in the depleted uranium.

This composition was achieved by loading the individual drawers as shown in Fig. 3. Each drawer contained 17 in. of core material and 4 in. of blanket material, with an additional 10 in. of blanket loaded behind each drawer. The core material was mainly in the form of $\frac{1}{8}$-in.-thick rectangular plates, either 2 x 2 in. or 2 x 3 in. Reduced-density aluminum, made by perforating the individual plates, was used. As a result, the core contained considerable void. The drawers and matrix tubes were all of stainless steel.

Aside from the drawers used as control rods, all drawers in the assembly were loaded in the same way, except for a few drawers specially loaded for the particular experiments. For all experiments except the critical experiment the control drawers had twice the usual amount of 93% enriched uranium in order to increase their reactivity worth.

CRITICAL EXPERIMENT

Preliminary multigroup calculations and previous experience with large, dilute oxide cores[5] indicated that the critical mass would be greater than 450 kg U^{235}. Initially the space to be occupied by the core was filled with core material which had all enriched uranium replaced with depleted uranium. The critical approach was made by inserting the enriched uranium in each drawer (see Fig. 3), starting at the center, in incremental loadings. The initial loading of enriched material, consisting of the central 89 drawers in each half, contained 206.57 kg U^{235}, less than half of the expected critical mass. Successive loadings were increased by, at most, 46.4 kg of U^{235} and generally a quantity much less than the necessary amount to reach criticality. As the critical mass was approached, the loading increment was greatly reduced. Figure 4 is a plot of inverse count rate vs. mass for the last four subcritical loadings. The tenth loading, which contained 503.77 kg of U^{235}, was 13.6 Ih supercritical with both control rods fully inserted. This loading contained 434 drawers of core material. Figure 7 shows the cross section of this loading. The positions of the control and safety rods are also shown.

The control rods were calibrated by making the reactor slightly supercritical and measuring the asymptotic period. Control rod increments were then converted to reactivity worths in inhours by the use of the inhour curve calculated from the data of Keepin.[6] Figure 5 shows the control rod calibration curve and Fig. 6 is a plot of the curve used to convert reactor period to inhours. For the purposes of measuring the critical mass, the control

rods contained a normal drawer loading. However, for subsequent experiments, the amount of 93% enriched uranium in the two control rods was doubled to provide better control.

The reactivity worth of core material substituted for blanket at the edge of the core was measured. This was 17.7 Ih/kg U^{235} in the core material. The critical mass and volume of this assembly were found by interpolation from the slightly supercritical tenth loading.

The critical mass was 503.01 kg of U^{235} and the critical volume 574.47 liters. The core was built in the form of a horizontal right cylinder, as nearly circular as possible, with the fully loaded 2 x 2-in. drawers of ZPR-III. The length was 86.36 cm and the radius of the equivalent circular cylinder was 46.02 cm. The ratio of length to diameter was 0.94.

After the initial criticality, it was necessary to remove a considerable portion of the radial blanket in order to reduce the weight of the moveable carriage. The outer 4 in. of depleted uranium blanket were removed everywhere except at the ends of the cylindrical core (see Fig. 7). This left a minimum of 8 in. of blanket. In all, about 12,000 kg depleted uranium were removed. The assembly lost 38.4 Ih.

PREDICTION OF CRITICAL MASS

A preliminary estimate of the critical volume of the core was based on a 16-group SNG calculation with the YOM cross-section set. The core for which the calculation was made contained 4.3 v/o more aluminum than the one actually built. The calculated critical spherical radius was 46.84 cm, corresponding to a volume of 430.5 liters, and the critical mass 376.9 kg U^{235}.

Assuming that the distributed worth of aluminum is the same in both cores (the one calculated above and the one actually built), it follows from the experimental measurements of the distributed worth of aluminum that the excess aluminum is worth about 230 Ih. Dividing this by the worth of fuel at the edge gives approximately the increase in critical mass needed to balance the effect of decreasing the aluminum content. This correction was 13 kg.

Since the core actually built was cylindrical instead of spherical, a small shape-factor[7] correction is required. For the length-to-diameter ratio of 0.94, the calculated critical mass should be multiplied by 1.04.

As indicated by the fuel-bunching experiments (described later), a homogeneous core of this composition is about 440 Ih less reactive than one in which the fuel is in $\frac{1}{8}$-in.-thick columns. The corresponding correction for heterogeneity is a decrease of about 26 kg in the calculated critical mass of U^{235}.

When the preceding corrections are applied, the estimate of critical mass obtained with the use of the YOM cross-section set and the SNG code was about 380 kg U^{235} for the cylindrical, heterogeneous core. The experimental critical mass was 503.01 kg.

CENTRAL FISSION RATIOS

Central fission ratios were determined from data obtained from absolute fission counters.[8] The fission ratio, as used here, is the ratio of the effective fission cross section of a particular isotope to the effective fission cross section of U^{235}. The fission ratios were measured by placing two counters adjacent to each other at the center of the reactor and comparing count rates. Since each counter contained differing amounts of several isotopes, fission ratios were calculated from the data obtained from several counters and a set of simultaneous linear equations..

Fission ratios were also calculated from the central neutron flux obtained from DSN multigroup calculations and the YOM cross-section set. In all cases the value calculated from the multigroup fluxes was larger than the experimental value. Table II lists both the calculated and experimental values.

Table II

CALCULATED AND EXPERIMENTAL CENTRAL FISSION RATIOS

Ratio	Measured Value	Calculated Value
$\sigma_{238}/\sigma_{235}$	0.0339	0.0396
$\sigma_{234}/\sigma_{235}$	0.247	0.3035
$\sigma_{233}/\sigma_{235}$	1.454	1.5384
$\sigma_{236}/\sigma_{235}$	0.0804	0.0968
$\sigma_{239}/\sigma_{235}$	1.067	1.1335
$\sigma_{240}/\sigma_{235}$	0.271	0.3027

FISSION TRAVERSE

Several drawers along a radial line at the midplane of the core were modified to accept a $\frac{1}{2}$-in.-diameter thimble. Small fission counters could be remotely placed at any radial position in the core or blanket. The relative fission rate as a function of radial position in the core was measured for both U^{235} and U^{238}. The results are plotted in Figs 8 and 9. Count rates have been normalized to unity at the center of the core.

Normalized fission rates for both U^{235} and U^{238} were calculated at several points by means of the neutron flux spectrum obtained from the multigroup calculations. The agreement seemed satisfactory, considering the fact that the multigroup calculation was carried out for a sphere whereas the core was cylindrical. The calculated values are plotted in Figs. 8 and 9 for comparison.

The counting statistics were best near the center of the core in both cases. For U^{238}, the standard deviation near the center was about 1% and for U^{235} about 2%.

CENTRAL REACTIVITY COEFFICIENTS

The reactivity changes caused by samples of both fissionable and nonfissionable materials placed at the center of the reactor were measured relative to void. The two center drawers were modified to provide a central 2-cu-inch void for the fissionable material and an 8-cu-inch void for the nonfissionable samples. Table III lists the results of these measurements. Some of the materials were in stainless steel containers. The number listed under "Reactivity Changes Due to Sample" includes the effect of stainless steel but the "Inhours per kilogram" value has been corrected for the effects of the stainless steel containers wherever necessary.

Table III

CENTRAL REACTIVITY COEFFICIENTS

Material	Sample Weight	Reactivity Change Due to Sample (Ih)	Ih/kg
U^{235}	268.42 g U^{235} 19.36 g U^{238}	28.6	107(a)
U^{233}	221.2 g U^{233} 5.3 g U^{238}	45.5	206(a)
Pu^{239}	186.0 g (95% Pu^{239})	32.2	182
U^{238}	2455 g U^{238} 4.92 g U^{235}	-13.7	-5.78(a)
Al	350.64	0.329	0.937
Al_2O_3	355.12	2.87	8.08
Ag	1366.05	-60.4	-44.2
Be	240.7	20.4	84.8
Bi	1276.2	-0.477	-0.374

Table III (Cont'd.)

Material	Sample Weight	Reactivity Change Due to Sample (Ih)	Ih/kg
B_4C (enriched)	16.087	-20.7	-1290
C	196.42	7.19	36.6
Cr	439.2 g Cr 102.4 g SS	-0.413	-0.603[b]
Fe	1028.0	-1.36	-1.32
Hg	1383.4 g Hg 197.3 g SS	-7.30	-5.07[b]
Li	54.9 g Li 195.1 g SS	-4.77	-81.8[b]
Mo	1279.8	-12.1	-9.49
Nb (Cb)	489.6 g Nb 102.4 g SS	-7.98	-16.0[b]
Na	91.2 g Na 195.0 g SS	0.37	7.17[b]
Ni	1150.4	-2.78	-2.42
Pb	1474.9	-0.138	-0.0936
Ph-I[d]	232.0 g Ph-I 168 g SS	-3.67	-14.8[b]
Ph-II	210 g Ph-II 275 g SS	-3.21	-13.4[b]
Stainless Steel	1017.2	-1.48	-1.45
Ta	1004.4 g Ta 102.4 g SS	-28.9	-28.7[b]
Th	1488.47	-18.9	-12.7
Y	582.1	-8.69	-14.9
Zr	846.0	-1.03	-1.21

[a] Reactivity of principal isotope only.

[b] Corrected for effects of stainless steel can.

[c] Stainless steel.

[d] Ph-I and Ph-II(9) are mixtures of elements used to simulate fission products.

The relative errors in the values listed above vary widely. Each measurement required opening and closing of the halves of the reactor, which introduced an error of about ±0.5 Ih. Therefore the values given for some samples, such as aluminum or sodium, have an uncertainty as large as the value itself.

REACTIVITY WORTHS OF DISTRIBUTED MATERIALS

The reactivity worths of uniformly distributed materials were measured for each material in the reactor. For these measurements, the quantity of a particular material was changed in a section approximating a wedge one-half of the length of the cylindrical core and the resulting change in reactivity was measured. The matrix structure of ZPR-III makes it difficult to change the composition of an actual wedge section. Figure 10 shows sketches of the cross section indicating the actual sections used in these experiments.

To measure the distributed worth of U^{235}, 0.010-in.-thick foils of enriched uranium, averaging 69.74 gm U^{235} per drawer, were added to each of the 15 drawers in the section shown in Fig. 10A. The distributed worth of U^{238} was measured by replacing one column of depleted uranium with 45% density aluminum in each of 28 drawers. The distributed worth of aluminum was measured by exchanging two columns of 45% density aluminum for two columns of 100% density aluminum in each of 28 drawers. To measure the worth of graphite, one column of graphite was replaced by 100% density aluminum in each of 15 drawers. The distributed worth of stainless steel was measured by replacing one column of stainless steel by 45% density aluminum in each of 53 drawers. Table IV gives the results.

Table IV

DISTRIBUTED WORTH OF MATERIALS

Material	Ih/kg
U^{238}	-0.846
U^{235}	44.1
Aluminum	4.65
Stainless Steel	1.66
Graphite	16.3

In this assembly, the uranium and graphite pieces were loaded next to each other to simulate a uranium carbide fuel element. The normal order of pieces was depleted uranium, graphite, and enriched uranium (see Fig. 3). The effect of the order of these pieces was measured by changing the relative position of the depleted uranium and graphite in a 20-drawer wedge

section of one of the reactor halves. The assembly gained 7.07 Ih reactivity. When extrapolated to the entire core, a gain of 154 Ih or 0.35% k* is indicated.

FUEL-BUNCHING EXPERIMENTS

The effect of the heterogeneity of the fuel was measured both by separating the normal $\frac{1}{8}$-in. column of enriched uranium into two, separated $\frac{1}{16}$-in. columns and by placing two $\frac{1}{8}$-in. columns side by side to form a single $\frac{1}{4}$-in. column in a 20-drawer wedge section of the core.

The enriched uranium was removed from ten drawers and placed next to the enriched uranium in the other ten drawers. The assembly gained 21.8 Ih. A linear extrapolation to the entire core indicated that a gain of 475 Ih would result from placing the fuel in $\frac{1}{4}$-in. columns rather than $\frac{1}{8}$-in. columns. In the same 20-drawer section, the fuel was separated into two, $\frac{1}{16}$-in. columns per drawer. The assembly lost 9.83 Ih. A linear extrapolation for the entire core indicated a loss of 213 Ih by separating all fuel into $\frac{1}{16}$-in. columns. Figure 11 is a plot of the fuel-bunching data. From these data, it was estimated that the assembly would lose about 440 Ih if the fuel were homogeneous, i.e., of zero thickness.

RADIOCHEMICAL ANALYSIS OF IRRADIATED FOILS

Natural and enriched uranium foils were irradiated at the center and at the edge of the core. The results of radiochemical analysis of the foils, made by Roland J. Armani, were used to calculate $\sigma_f^{28}/\sigma_f^{25}$ and $\sigma_c^{28}/\sigma_f^{25}$ at the center and edge of the core. Table V gives the results.

Table V

FISSION AND CAPTURE CROSS-SECTION RATIOS

Core Position	$\sigma_f^{28}/\sigma_f^{25}$	$\sigma_c^{28}/\sigma_f^{25}$
Center	0.039 ± 30%	0.122 ± 25%
Edge	0.031 ± 30%	0.128 ± 25%

The comparatively large errors are caused by a long time delay between irradiation and analysis.

The measured central fission ratio at the center of the core was 0.034. This is within the 30% estimated error. For comparison, the fission and capture ratios were calculated from the multigroup fluxes obtained from the 16-group DSN calculations.

*1 inhour = 2.25×10^{-3} % k.

Table VI gives the calculated results.

Table VI

CALCULATED FISSION AND CAPTURE CROSS-SECTION RATIOS

Core Position	$\sigma_f^{28}/\sigma_f^{25}$	$\sigma_c^{28}/\sigma_f^{25}$
Center	0.0396	0.1483
Edge	0.0208	0.1317

PROMPT NEUTRON LIFETIME

The average prompt neutron lifetime was measured by the Rossi-alpha method, described by Brunson et al.[10] The Rossi alpha at delayed critical is $(2.92 \pm 0.08) \times 10^4$ sec^{-1}. Based on an effective value of β of 0.0073, the average prompt neutron lifetime is $(25.0 \pm 1) \times 10^{-8}$ sec.

CONCLUSIONS

The properties of the dilute carbide core studied in this experiment showed considerable similarity to ZPR-III Assembly 29, the dilute oxide core. The central reactivity coefficients measured in the two assemblies were similar in magnitude and sign. The central fission ratios of the two cores were essentially the same.

The inability to calculate accurately the critical mass of this assembly is a further indication that the multigroup cross-section set used is inaccurate for large dilute cores. The value of k for this carbide core is overestimated by about the same amount as for the other large, dilute cores recently built in ZPR-III. As indicated by the fuel-bunching experiments, the just-critical core would be some 440 Ih subcritical if the enriched uranium were uniformly distributed throughout the core, rather than in $\frac{1}{8}$-in. columns. If the core were made spherical, the critical mass would be reduced by about 30 kg. Considering the worth of core material at the edge, these two corrections indicate that a spherical, homogeneous, core having the same volume and composition as the "just-critical" assembly would be about 90 Ih or 0.2% supercritical. A "k" calculation for a spherical, homogeneous core having the same volume and composition as the "just-critical" configuration of Assembly 34, gave k = 1.0495. It follows that the YOM cross-section set overestimates k for this assembly by about 4.7%.

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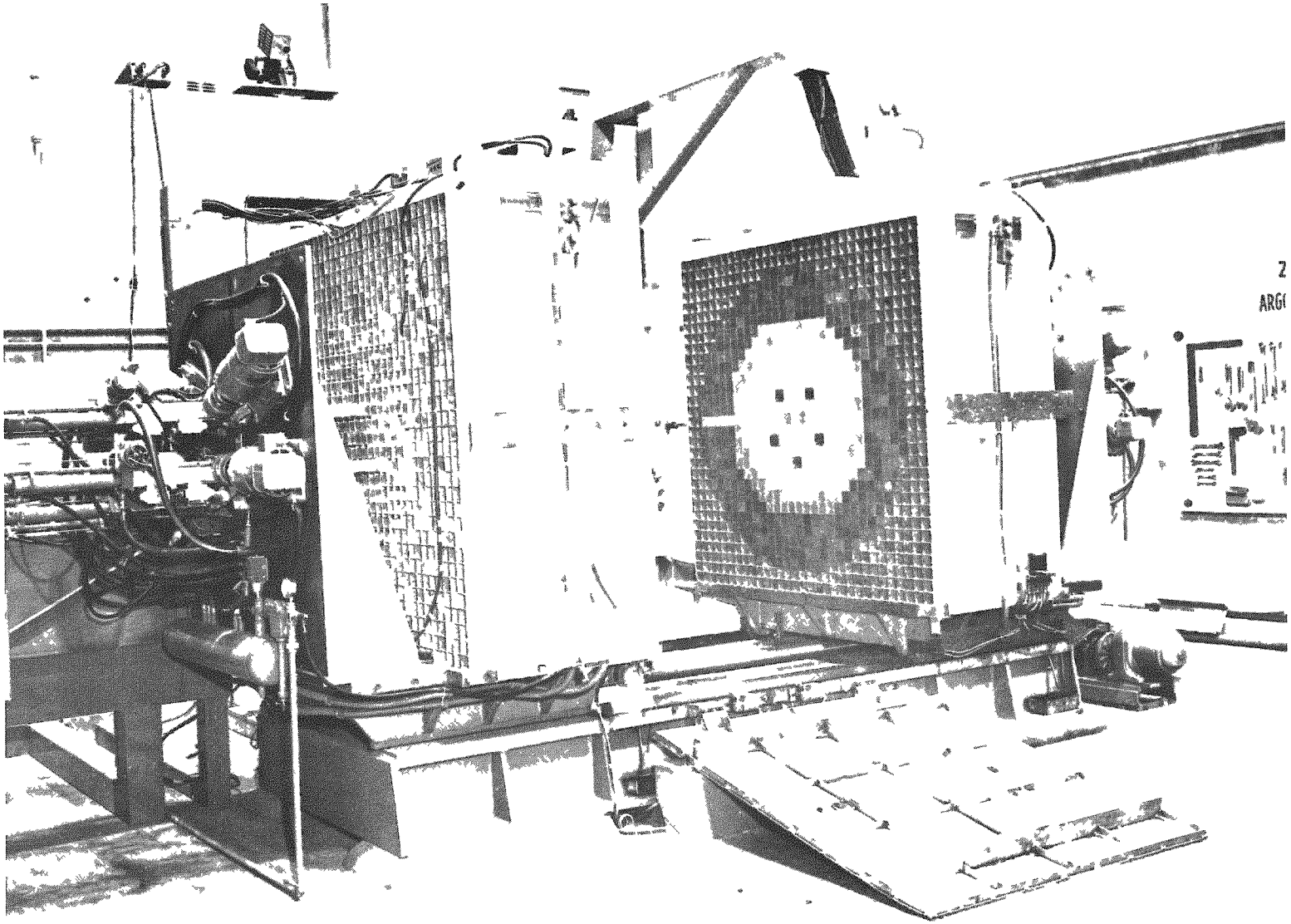


FIGURE 1

ZPR-III CRITICAL FACILITY

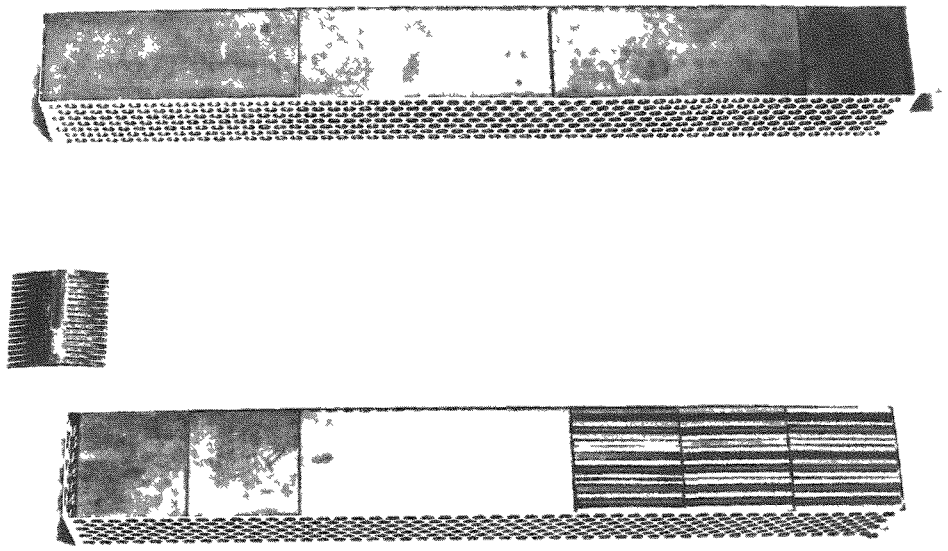
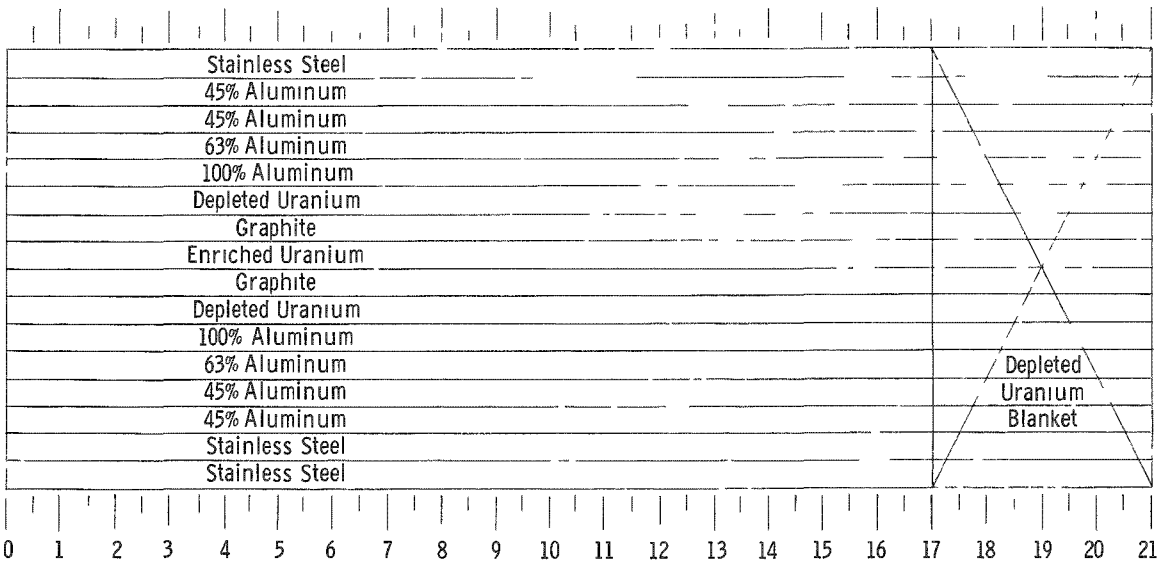


FIGURE 2

TYPICAL CORE DRAWERS AND LOADINGS



ASSEMBLY NO. 34
LOADING NO.
DRAWER NO. All

FIGURE 3

ASSEMBLY 34 DRAWER LOADING

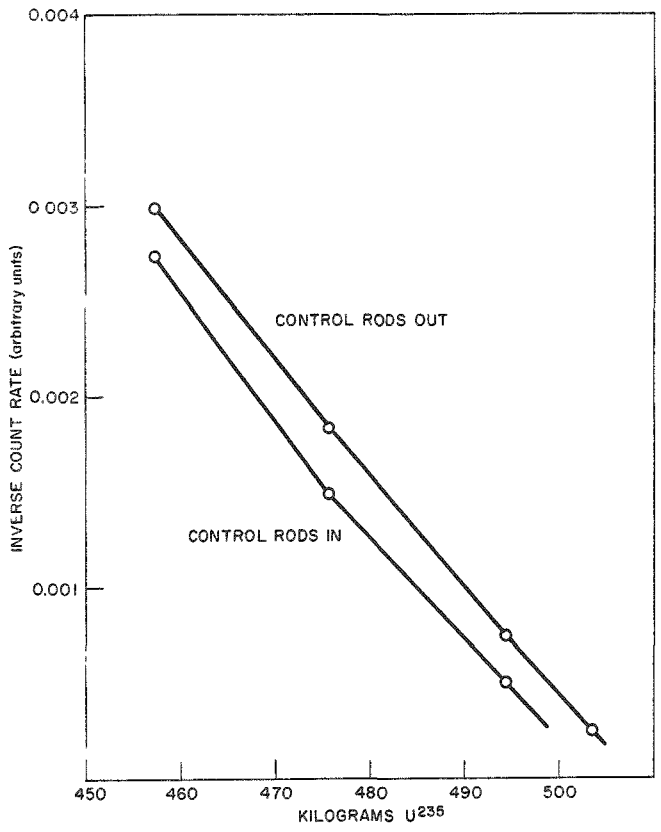


FIGURE 4
INVERSE COUNT RATE VERSUS
MASS FOR CRITICAL APPROACH

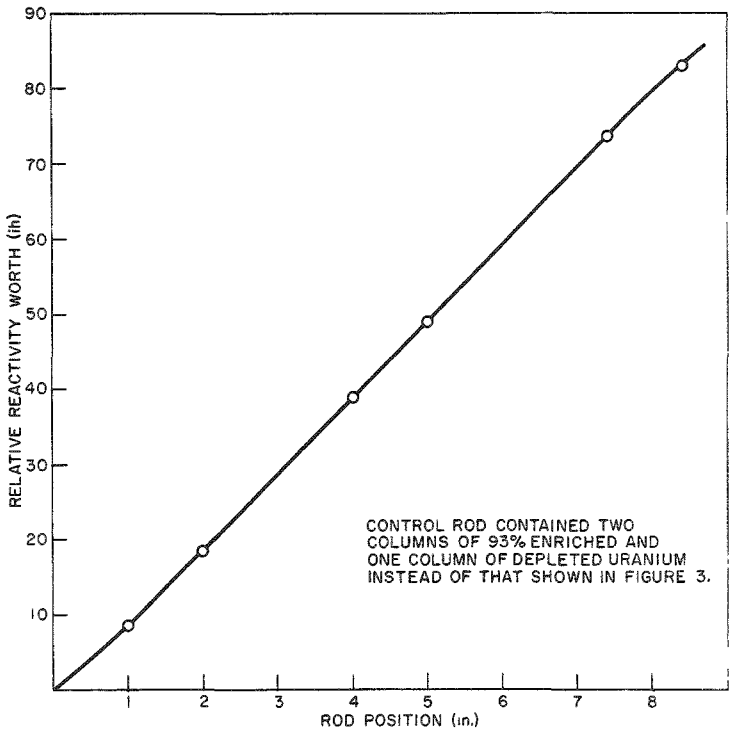


FIGURE 5
CONTROL ROD CALIBRATION

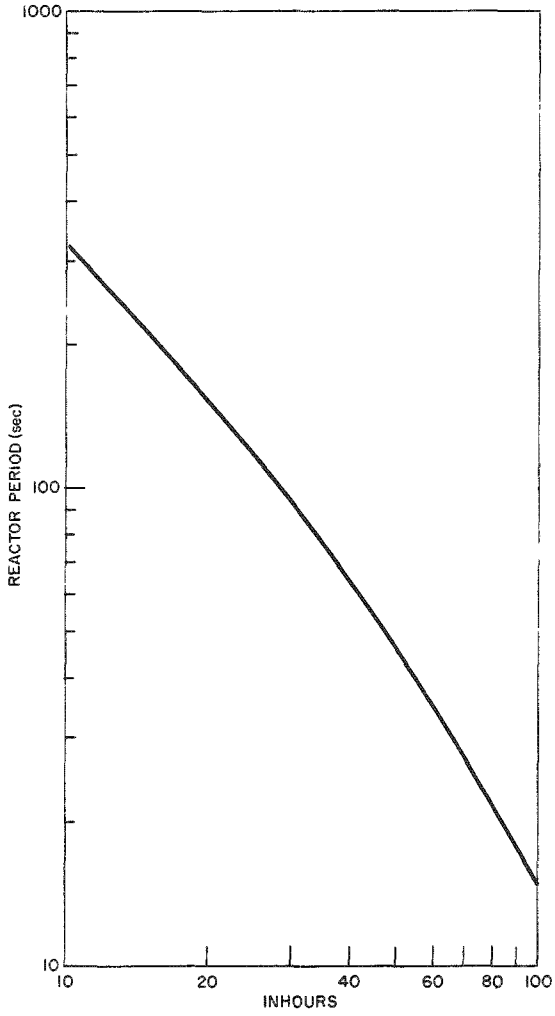


FIGURE 6
REACTOR PERIOD VERSUS INHOURS

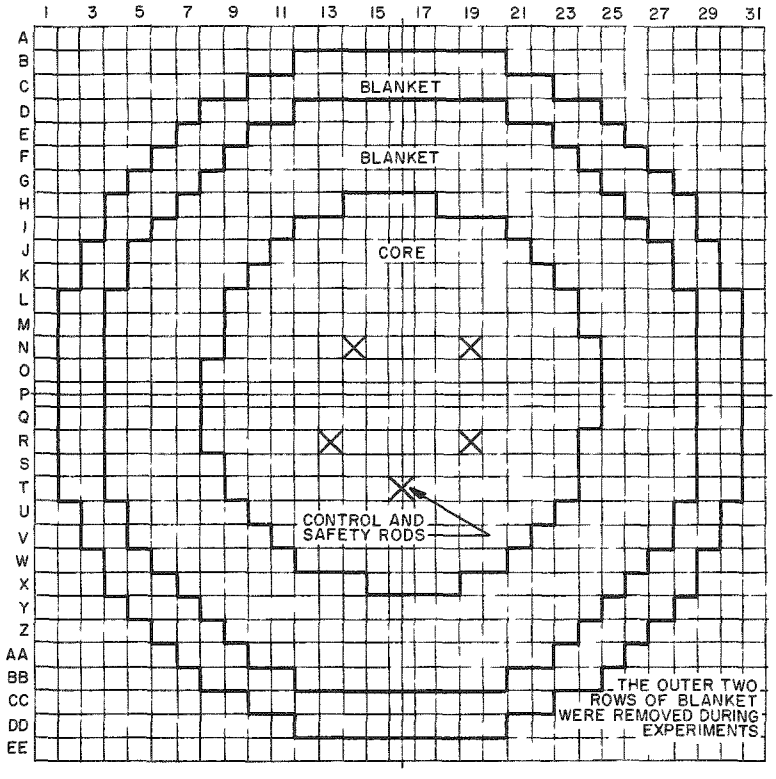


FIGURE 7
CROSS SECTION OF ASSEMBLY 34

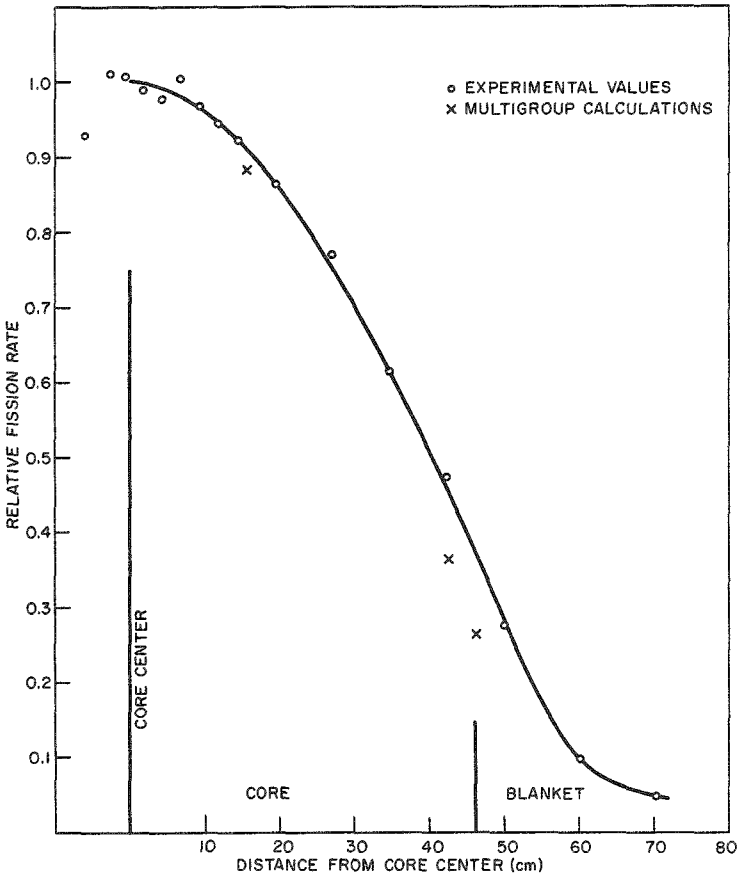


FIGURE 8
U^{235} FISSION TRAVERSE

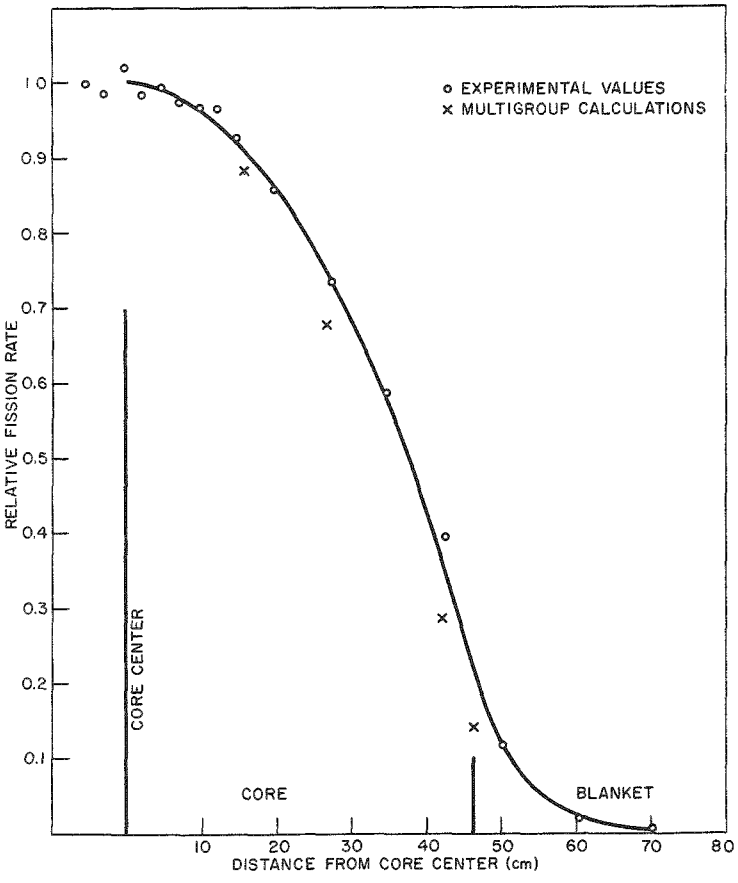
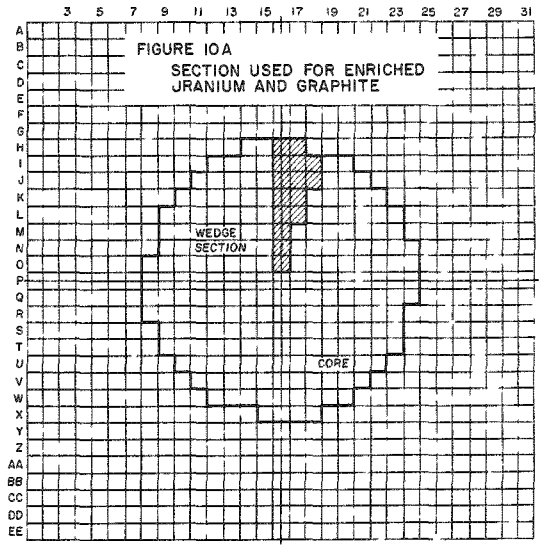
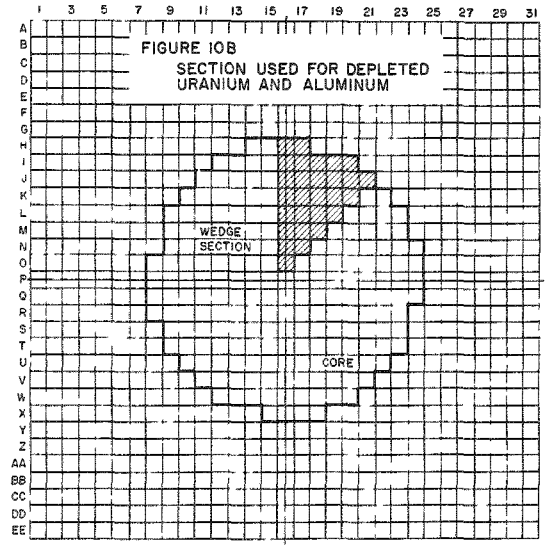


FIGURE 9
U^{238} FISSION TRAVERSE

FIGURE 10

WEDGE SECTION OF CORE USED FOR DISTRIBUTED WORTH MEASUREMENTS





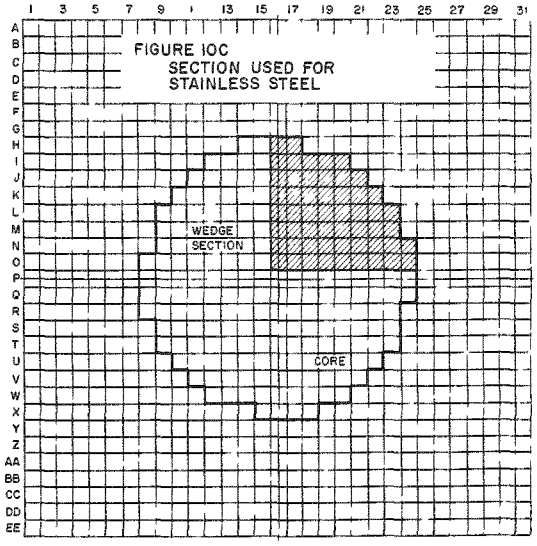


FIGURE 11
REACTIVITY VERSUS FUEL THICKNESS