ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois

CRITICAL STUDIES OF A
DILUTE OXIDE FAST REACTOR CORE
(ZPR-III Assembly 30)

by

P. I. Amundson, A. L. Hess, W. P. Keeney,
J. K. Long, and R. L. McVean

Idaho Division

May 1961

Operated by The University of Chicago
under
Contract W-31-109-eng-38

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

TABLE OF CONTENTS

	<u>Page</u>
ABSTRACT	5
I. INTRODUCTION	5
II. DESCRIPTION OF ASSEMBLY	7
III. PREDICTION OF CRITICAL MASS	10
IV. CRITICAL EXPERIMENT	11
V. FISSION RATIOS	12
VI. CENTRAL REACTIVITY COEFFICIENTS	14
VII. REACTIVITY COEFFICIENTS OF DISTRIBUTED MATERIALS	16
VIII. BUNCHING EXPERIMENTS	20
IX. AVERAGE NEUTRON LIFETIME	23
DISCUSSION	23
ACKNOWLEDGEMENTS	23
REFERENCES	23

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
1.	View of ZPR-III Critical Assembly	6
2.	View of Sample Drawers	6
3.	Top View of Core Drawer (Type 1)	7
4.	Top View of Core Drawer (Type 2)	7
5.	Top View of Core Drawer (Type 3)	8
6.	Top View of Core Drawer (Type 4)	8
7.	Face View of Core and Blanket (Both Halves)	9
8.	Drawer Sequence for Central Core Area.	9
9.	Subcritical Multiplication (Reciprocal Count Rate vs. Fuel Mass)	11
10.	Control Rod Calibration for Rod Number 10	11
11.	Reactivity-Period Curve	12
12.	Counter Drawer Loading Used for Measuring Fission Ratios	12
13.	Drawer Loading Used for 1 in. ³ (16.129 cm ³) Central Reactivity Coefficient Samples	15
14.	Drawer Loading Used for 8 in. ³ (129.032 cm ³) Central Reactivity Coefficient Samples	16
15.	Core Face Showing Section Utilized in Wedge Substitution Experiments	16
16.	Front View of Drawers Used for Measuring the Reactivity Worths of Aluminum, Oxygen, and Stainless Steel.	17
17.	Front View of Drawers Used for Measuring the Reactivity Worths of Distributed Depleted Uranium vs. Proximity to Fuel.	18
18.	Front View of Drawers Used for Measuring the Reactivity Worths of Distributed Aluminum vs. Proximity to Fuel	18
19.	Face View of Core Showing Drawer Positions for Measuring the Reactivity Worth of Distributed Depleted Uranium vs. Proximity to Fuel.	19

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
20.	Face View of Core Showing Drawer Positions for Measuring the Reactivity Worth of Distributed Aluminum vs. Proximity to Fuel.	20
21.	Typical Bunching and Unbunching Loadings (Front View of Drawers).	21
22.	Face View of Core Showing Wedge Area of Core and Columns Used for Bunching and Unbunching Experiments. . .	22
23.	Bunching and Unbunching Experiments [Change in Reactivity for Bunching and Unbunching $\frac{1}{8}$ -in. (0.3175 cm) Columns in a Wedge Section of the Core].	22

LIST OF TABLES

<u>No.</u>	<u>Title</u>	<u>Page</u>
I.	Core Compositions by Volume Per Cent	10
II.	Volume Per Cent Compositions of Calculated Core	10
III.	Delayed Neutron Data	12
IV.	Central SNG Spectrum.	13
V.	Cross Sections from SNG Spectrum	13
VI.	Fission Ratios	14
VII.	Central Reactivity Coefficients	15
VIII.	Reactivity Coefficients for Distributed Materials	17
IX.	Reactivity Coefficients Versus Proximity to Fuel	19

CRITICAL STUDIES OF A
DILUTE OXIDE FAST REACTOR CORE
(ZPR-III Assembly 30)

by

P. I. Amundson, A. L. Hess, W. P. Keeney,
J. K. Long, and R. L. McVean

ABSTRACT

The report describes critical studies of a fast reactor core containing a simulated oxide fuel having an oxygen-uranium atomic ratio of 1:1. Calculated and experimental critical masses are compared. Experimental results are given for fission ratio, central reactivity coefficient, fuel bunching, and distributed worth measurements.

I. INTRODUCTION

An assembly was constructed to study some of the characteristics of a dilute fast reactor core fueled with a pseudo-uranium oxide containing one atom of oxygen per atom of uranium. It was the second of a series of assemblies designed to study the effects of varying the oxygen content of fast reactor fuels. These studies were prompted by the inability to predict accurately the critical mass of a previous dilute oxide core having a uranium to oxygen ratio of 1:2. (ZPR-III, Assembly 29).⁽¹⁾

The work to be described was performed in the Zero Power Reactor III, (ZPR-III) located at Argonne National Laboratory's Idaho Facilities at the National Reactor Testing Station. The reactor, shown in Fig. 1, is composed of two clusters of square, cross-sectional tubes forming a square matrix. The right half is movable and the left half stationary. Insertion of 8 safety rods, moving the halves together, and then inserting 2 control rods brings the assembly to critical. The control rods and safety rods are composed of normal core material.

The core material is assembled in thin, perforated stainless steel drawers (see Fig. 2), which are designed to slide into the square tubes of each matrix. A complete description of the reactor and its operation has been compiled by Cerutti et al.⁽²⁾

Fig 1 View of ZPR-III Critical Assembly

Fig. 2 View of Sample Drawers

II. DESCRIPTION OF ASSEMBLY

The core approximated a right circular cylinder, 26.06 in. (66.19 cm) long with an effective radius of 16.3 in. (41.4 cm) calculated from the critical mass, core composition, and length. A 4-drawer repetitive module was used, the arrangements of the drawer materials being as shown in Figs. 3 through 6.

Fig. 3 Top View of Core Drawer (Type 1)

Fig. 4 Top View of Core Drawer (Type 2)

Fig. 5 Top View of Core Drawer (Type 3)

Fig. 6 Top View of Core Drawer (Type 4)

The oxide fuel was simulated by employing uranium metal and alumina (Al_2O_3), the latter being the source of oxygen. The core material also contained stainless steel (SS) and metallic aluminum.

The core was constructed so that the outline of the cross section perpendicular to the axis of the cylinder was reasonably circular. The cross-sectional shape of the just-critical core is shown in Fig. 7 and the arrangement of drawers in Fig. 8.

A high-density blanket of depleted uranium completely surrounded the core to a thickness of approximately 12 in. (30.48 cm).

Fig 7
Face View of Core
and Blanket
(both halves)

3	4	1	2	3	4	1	2	3	4	1	K	3	4	1	2	3	4	1	2	3	4	1
3	4	1	2	3	4	1	2	3	4	1	L	3	4	1	2	3	4	1	2	3	4	1
3	4	1	2	3	4	1	2	3	4	1	M	3	4	1	2	3	4	1	2	3	4	1
3	4	1	CD	3	4	1	CD	3	4	1	N	3	4	1	CD	3	4	1	CD	3	4	1
3	4	1	2	3	4	1	2	3	4	1	O	3	4	1	2	3	4	1	2	3	4	1
3	4	1	2	3	+	1	2	3	4	1	P	3	4	1	2	3	+	1	2	3	4	1
3	4	1	2	3	4	1	2	3	4	1	Q	3	4	1	2	3	4	1	2	3	4	1
3	4	CD	2	3	4	1	2	CD	4	1	R	3	4	CD	2	3	4	1	2	CD	4	1
3	4	1	2	3	4	1	2	3	4	1	S	3	4	1	2	3	4	1	2	3	4	1
3	4	1	2	3	CD	1	2	3	4	1	T	3	4	1	2	3	CD	1	2	3	4	1
3	4	1	2	3	4	1	2	3	4	1	U	3	4	1	2	3	4	1	2	3	4	1
11	12	13	14	15	16	17	18	19	20	21		21	20	19	18	17	16	15	14	13	12	11
HALF #1 STATIONARY												HALF #2 MOVEABLE										
CD (Control Drawer)																						

Fig. 8 Drawer Sequence for Central Core Area

The compositions of the 4-drawer module, the just-critical core, and the blanket are shown in Table I.

The difference in composition of the 4-drawer module versus the just-critical core is due to nonuniform utilization of all drawers in the 4-drawer sequence

Table I

CORE COMPOSITIONS BY VOLUME PER CENT

	<u>U²³⁵</u>	<u>U²³⁸</u>	<u>Al</u>	<u>SS</u>	<u>O</u>
4-Drawer Sequence	5.81	9.135	23.348	24.502	7.259
Just-critical Core	5.909	9.038	23.35	24.50	7.26
Blanket	0.19	83.30	-	7.31	-
Assumed Density	18.75	19.0	2.7	7.85	*

* Assumed $N = 0.096 \times 10^{24}$ atoms/cc for 100 volume per cent oxygen, from cross-section set 58, ANL 5800, p.419

III. PREDICTION OF CRITICAL MASS

The critical mass was predicted from transport theory (SNG code, cross-section set 58, S4 approximation) for a homogeneous sphere. The input data were partially in error due to mistakes made in the initial calculation of the volume fractions of the 4-drawer module. The incorrect volume fractions are listed in Table II for reference.

Table II

VOLUME PER CENT COMPOSITIONS OF
CALCULATED CORE

	<u>U²³⁵</u>	<u>U²³⁸</u>	<u>Al</u>	<u>SS</u>	<u>O</u>
Volume Fraction	5.92	9.31	23.32	24.65	7.27

The calculated core had a critical radius of 40.14 cm (15.8 in.), giving a critical volume of 271.0 liters. Using the calculated 5.92 v/o of U²³⁵ in the 4-drawer sequence, the critical mass was computed to be 300.8 kg of U²³⁵ for a homogeneous sphere. Applying a -5.7% correction for heterogeneity (see Sect. VIII, Bunching Experiments, page 19 and a +3% correction for cylindrical shape, the expected critical mass becomes 292.16 kg of U²³⁵. This calculation was made only to assist in performing the critical experiment.

Although the input data were in error, it was estimated that the resulting critical mass would not be in error by more than 1-2% because of the volume fraction errors. The v/o of U²³⁵ in the incorrectly calculated 4-drawer sequence (5.92%) was nearly equal to that for the just-critical core (5.90%). The U²³⁸ and stainless steel compositions were in error by +2.9% and 0.6%, respectively.

IV. CRITICAL EXPERIMENT

The approach to criticality was initiated with approximately 140 kg of U^{235} , and was accomplished by adding radial increments of core, each increment replacing blanket material. Figure 9 shows the inverse count rate, after each incremental addition of core material, for one of the three counting channels used. Loading No. 13 (397.864 kg of U^{235}) was critical with control rod No. 1 at 0 in. and control rod No. 10 at 5.7 in. (14.48 cm).

Control rod No. 10 was calibrated by measuring the periods associated with incremental changes in control-rod position. The rod calibration for this core is shown in Fig. 10.

The worth of core material at the core-blanket interface was determined as 19.93 ih/kg U^{235} when replacing blanket material. The core material worth and rod calibrations were used to calculate a critical mass of 394.94 kg of U^{235} .

Fig. 9 Subcritical Multiplication (reciprocal count rate vs fuel mass)

Fig. 10 Control Rod Calibration for Rod Number 10

All periods measured in this assembly were related to reactivity changes in inhours by the use of an inhour curve plotted from information from Keepin *et al.* ⁽³⁾ It was assumed that 10% of all fissions occur in U^{238} , in which the estimated β_i/β for six groups of delayed neutrons are as shown in Table III. The inhour curve used is shown in Fig. 11.

Table III

DELAYED NEUTRON DATA

$$\beta = 0.0073$$

Group	Half-life (sec)	β_i/β
1	54.3	0.030
2	21.7	0.186
3	5.87	0.184
4	2.18	0.409
5	0.46	0.158
6	0.13	0.033

V. FISSION RATIOS

Fission ratios were calculated by means of counting data from absolute fission counters (4) placed at the center of the core. In each case the fission ratio was defined as the measured average fission cross section for some fissile material divided by the average measured fission cross section for U^{235} . For each run an absolute fission counter containing the desired fissile material was positioned at the core center opposite a U^{235} counter used both as a U^{235} standard and as a flux-monitoring reference. The U^{235} -counter position was held constant throughout the experiment. A diagram of the counter drawer loading is shown in Fig. 12.

Fig. 11. Reactivity-
Period Curve
(inhour curve)

Fig. 12. Counter Drawer Loading for
Measured Fission Ratios

The central SNG spectrum and the average cross sections for various materials, used to calculate the theoretical cross-section ratios, are listed for reference in Tables IV and V. Both the experimental and calculated (from SNG spectrum) cross-section ratios are given in Table VI.

Table IV

CENTRAL SNG SPECTRUM

Group	Lower Energy Limit, Mev	Neutron Fraction
1	2.25	0.0590
2	1.35	0.0762
3	0.825	0.1043
4	0.5	0.1495
5	0.3	0.1373
6	0.18	0.1338
7	0.11	0.1031
8	0.067	0.0904
9	0.025	0.0877
10	0.0091	0.0461
11	0	0.0126

Table V

FLUX-WEIGHTED EFFECTIVE
CROSS SECTIONS FROM
SNG SPECTRUM, BARN

$\sigma_f \text{ U}^{235} = 1.620$	$\sigma_c \text{ U}^{238} = 0.1995$	$\sigma_f \text{ U}^{236} = 0.162$
$\sigma_f \text{ U}^{238} = 0.0694$	$\sigma_f \text{ U}^{233} = 2.389$	$\sigma_f \text{ Pu}^{239} = 1.845$
$\sigma_c \text{ U}^{235} = 0.3312$	$\sigma_f \text{ U}^{234} = 0.506$	$\sigma_f \text{ Pu}^{240} = 0.483$

Table VI

FISSION RATIOS

Materials	Ratio From SNG Spectrum	Experimental Ratio* (fission counters)
$\sigma_f \text{U}^{238} / \sigma_f \text{U}^{235}$	0.0428	0.0427 ± 0.0004
$\sigma_f \text{U}^{233} / \sigma_f \text{U}^{235}$	1.475	1.488 ± 0.011
$\sigma_f \text{U}^{234} / \sigma_f \text{U}^{235}$	0.312	0.3005 ± 0.0021
$\sigma_f \text{U}^{236} / \sigma_f \text{U}^{235}$	0.100	0.0981 ± 0.0008
$\sigma_f \text{Pu}^{239} / \sigma_f \text{U}^{235}$	1.139	1.124 ± 0.008
$\sigma_f \text{Pu}^{240} / \sigma_f \text{U}^{235}$	0.298	
$\sigma_c \text{U}^{238} / \sigma_f \text{U}^{235}$	0.123	
$\sigma_c \text{U}^{235} / \sigma_f \text{U}^{235}$	0.204	

*The errors quoted are the 67% confidence limits calculated from the basic count rates only.

VI. CENTRAL REACTIVITY COEFFICIENTS

Reactivity coefficients for both fissile and nonfissile materials were determined for samples at center of the core. In the case of samples that were contained in stainless steel cans, the weight and reactivity worth of the stainless steel was accounted for.

The values obtained for the small (0.2965 kg) sample of depleted uranium and the aluminum sample may be in error by a factor of 2 because measured reactivity changes associated with these samples were near the sensitivity of the measurement (-0.85 and $+0.60$ ih as opposed to a measurement sensitivity of ± 0.5 ih).

All materials (listed by number in Table VII) were placed in the front of the opposing center drawers. A typical drawer loading diagram for materials 1 through 4 is shown in Fig. 13. The volume of material used was approximately 1 in.³ (16.39cm³, somewhat less for the canned Pu and U²³³). The reference was run with one piece of 45% aluminum and one of 63% aluminum, each $2 \times 2 \times \frac{1}{8}$ in. in size (5.08 x 5.08 x 0.3175 cm). The enriched uranium reactivity coefficient was measured in a sandwich of $\frac{1}{16}$ in. (0.1587-cm), 100% aluminum pieces.

Table VII

CENTRAL REACTIVITY COEFFICIENTS

No.	Material	Weight, kg	Sample Worth, Δh (relative to void)	Material Worth $\Delta h/\text{kg}$ (relative to void)
1	Enriched U ⁽¹⁾	0.2878	+ 33.14	+115.14
2	Depleted U	0.2965	- 0.85	- 2.87
3	Plutonium ⁽²⁾	0.1860	+ 39.35	+211.55
4	U ²³³ Mixture ⁽³⁾	0.2265	+ 52.55	+232.0
5	Depleted U	2.460	- 10.48	- 4.26
6	Aluminum	0.3506	+ 0.60	+ 1.71
7	Stainless Steel	1 017	- 1.92	- 1.89
8	Fe	1 028	- 2.06	- 2.00
9	C	0.1964	+ 8.45	+ 43.02
10	Na	0.0912	+ 1.098	+ 12.04
11	Ph I	0.232	- 3.17	- 13.66
12	Ph II	0.210	- 2.68	- 12.76

(1) The enriched U contained 93.41% U²³⁵

(2) The plutonium contained 95.0% Pu²³⁹

(3) The U²³³ mixture contained 97.6% U²³³

Fig. 13 Drawer Loading for 1-in.³ Central Reactivity Coefficient Samples

The drawer loading for materials 5 through 12 is shown in Fig. 14. The volume of material used was approximately 8 in.³ (129.032 cm³, somewhat less for canned sodium, Physicum I and II samples). Ph I and II are metal and metal oxide mixtures of stable elements simulating the nuclear properties of mixed fission products.⁽⁵⁾

Fig. 14 Drawer Loading Used for 8 in.³ (129.032 cm³)
Central Reactivity Coefficient Samples

VII. REACTIVITY COEFFICIENTS OF DISTRIBUTED MATERIALS

Material substitution experiments were performed in a wedge section (shown in Fig. 15) of the core to determine the reactivity worths of uniformly distributed aluminum, aluminum oxide, oxygen, and stainless steel.

Fig. 15
Core Face Showing Section Utilized in Wedge Substitution Experiments

The material substitutions of a typical wedge drawer are shown in Fig. 16. The experimental results are shown in Table VIII.

Fig. 16 Front View of Drawers Used for Measuring the Reactivity Worths of Aluminum, Oxygen, and Stainless Steel

Table VIII

REACTIVITY COEFFICIENTS FOR DISTRIBUTED MATERIALS			
Material	Mass (kg)	Sample Worth (1h)	Material Worth (1h/kg)
Aluminum	5 302	+ 36 60	+ 0 90
Aluminum Oxide	9 183	+114 40	+12 46
Oxygen*	4 321	+ 80 85	+18 71
Stainless Steel	9 277	+ 21 76	+ 2 35

*The reactivity worth of distributed oxygen is calculated from the aluminum oxide and aluminum measurements

The reactivity worths of depleted uranium and aluminum versus proximity to enriched uranium were determined for the materials uniformly distributed in one-half of the assembly. The material replacements are shown in Figs. 17 and 18. The drawer distributions are shown in Figs. 19 and 20. The experimental results are shown in Table IX.

45% Aluminum	45% Aluminum
Enriched Uranium	Enriched Uranium
Aluminum Oxide	Aluminum Oxide
45% Aluminum	45% Aluminum
Stainless Steel	Stainless Steel
Stainless Steel	Stainless Steel
45% Aluminum	45% Aluminum
Aluminum Oxide	Aluminum Oxide
Depleted Uranium	Depleted Uranium
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
Stainless Steel	Stainless Steel
Aluminum Oxide	Aluminum Oxide
Enriched Uranium	Enriched Uranium
45% Aluminum	45% Aluminum

Reference for Worth of Depleted Uranium Away from Fuel

45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
Stainless Steel	Stainless Steel
Aluminum Oxide	Aluminum Oxide
Enriched Uranium	Enriched Uranium
Depleted Uranium	Depleted Uranium
Aluminum Oxide	Aluminum Oxide
Stainless Steel	Stainless Steel
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
Stainless Steel	Stainless Steel
Aluminum Oxide	Aluminum Oxide
Depleted Uranium	Depleted Uranium

Reference for Worth of Depleted Uranium Near Fuel

45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
Stainless Steel	Stainless Steel
Aluminum Oxide	Aluminum Oxide
Enriched Uranium	Enriched Uranium
45% Aluminum	45% Aluminum
Aluminum Oxide	Aluminum Oxide
Stainless Steel	Stainless Steel
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
Stainless Steel	Stainless Steel
Aluminum Oxide	Aluminum Oxide
Depleted Uranium	Depleted Uranium

Depleted Uranium Substitution Near Fuel

Fig. 17 Front View of Drawers Used for Measuring the Reactivity Worth of Distributed Depleted Uranium vs Proximity to Fuel

45% Aluminum	45% Aluminum
Enriched Uranium	Enriched Uranium
Aluminum Oxide	Aluminum Oxide
45% Aluminum	45% Aluminum
Stainless Steel	Stainless Steel
Stainless Steel	Stainless Steel
45% Aluminum	45% Aluminum
Aluminum Oxide	Aluminum Oxide
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
Stainless Steel	Stainless Steel
Aluminum Oxide	Aluminum Oxide
Enriched Uranium	Enriched Uranium
45% Aluminum	45% Aluminum

Reference for Worth of Aluminum Away from Fuel

45% Aluminum	45% Aluminum
Enriched Uranium	Enriched Uranium
Aluminum Oxide	Aluminum Oxide
45% Aluminum	45% Aluminum
Stainless Steel	Stainless Steel
Stainless Steel	Stainless Steel
45% Aluminum	45% Aluminum
Aluminum Oxide	Aluminum Oxide
100% Aluminum	100% Aluminum
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
Stainless Steel	Stainless Steel
Aluminum Oxide	Aluminum Oxide
Enriched Uranium	Enriched Uranium
45% Aluminum	45% Aluminum

Worth of Aluminum Away from Fuel

45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
Stainless Steel	Stainless Steel
Aluminum Oxide	Aluminum Oxide
Enriched Uranium	Enriched Uranium
45% Aluminum	45% Aluminum
Aluminum Oxide	Aluminum Oxide
Stainless Steel	Stainless Steel
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
Stainless Steel	Stainless Steel
Aluminum Oxide	Aluminum Oxide
Depleted Uranium	Depleted Uranium

Reference for Worth of Aluminum Near Fuel

45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
Stainless Steel	Stainless Steel
Aluminum Oxide	Aluminum Oxide
Enriched Uranium	Enriched Uranium
100% Aluminum	100% Aluminum
Aluminum Oxide	Aluminum Oxide
Stainless Steel	Stainless Steel
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
45% Aluminum	45% Aluminum
Stainless Steel	Stainless Steel
Aluminum Oxide	Aluminum Oxide
Depleted Uranium	Depleted Uranium

Worth of Aluminum Near Fuel

Fig. 18 Front View of Drawers Used for Measuring the Reactivity Worth of Distributed Aluminum vs Proximity to Fuel

Table IX

REACTIVITY COEFFICIENTS VERSUS
PROXIMITY TO FUEL

Material	Mass (kg)	Sample Worth (ih)	Material Worth (ih/kg)
U^{238} near enriched U	41.29	+20.47	+0.496
U^{238} away from enriched U	44.21	+14.24	+0.322
Aluminum near enriched U	3.307	+24.27	+7.34
Aluminum away from enriched U	3.533	+24.57	+6.95

Fig. 19 Face View of Core Showing Drawer Positions for Measuring the Reactivity Worth of Distributed Depleted Uranium vs Proximity to Fuel.

Fig. 20 Face View of Core Showing Drawer Positions for Measuring the Reactivity Worth of Distributed Aluminum vs Proximity to Fuel.

VIII. BUNCHING EXPERIMENTS

Bunching experiments were performed in a wedge section (see Fig. 15) of half No. 2 of the core. The reference core contained all $\frac{1}{8}$ -in. (0.3175-cm) columns of enriched uranium.

The first experiment consisted of unbunching twenty-five $\frac{1}{8}$ -in. (0.3175-cm) columns to fifty $\frac{1}{16}$ -in. (0.1587-cm) columns in 21 drawers of a wedge section. The partially unbunched core lost 13 ih of reactivity. Since the full reference core contained four hundred-fifty $\frac{1}{8}$ -in. (0.3175-cm) columns, the total expected change in reactivity for unbunching the entire core would be -234 ih.

The second experiment consisted of bunching twenty-four $\frac{1}{8}$ -in. (0.3175-cm) columns to twelve $\frac{1}{4}$ -in. (0.635-cm) columns in 20 drawers of the same section of the core. The partially bunched core gained 34.55 ih. With four hundred-fifty $\frac{1}{8}$ -in. (0.3175-cm) columns in the reference core, the total expected change in reactivity for bunching the entire core would be +648 ih.

Typical reference, unbunched, and bunched drawers are shown in Fig. 21. The experiment was planned to minimize any net radial shifting of fuel. The wedge patterns for the unbunched and bunched cores are shown in Fig. 22.

Fig. 21 Typical Bunching and Unbunching Loadings (front view of drawers)

The bunching data were used to plot a curve (Fig. 23) of relative reactivity versus fuel thickness. Extrapolation to zero fuel thickness would result in a net loss of reactivity of 447 ih. Using the measured worth of core material at the edge of the core, a homogeneity correction of 5.7% Δ mass/mass was calculated.

Fig. 22 Face View of Core Showing Wedge Area of Core and Columns Used for Bunching and Unbunching Experiments.

Fig. 23

Bunching and Unbunching Experiments (Change in Reactivity for Bunching and Unbunching $\frac{1}{8}$ -in. [0.3175-cm] Columns in a Wedge Section of the Core.)

IX. AVERAGE NEUTRON LIFETIME

The average neutron lifetime was determined by the Rossi- α method. The experimentally determined α was $-4.10 \pm 0.10 \times 10^4$. For an assumed β of 0.0073, the average neutron lifetime was calculated as $(17.8 \pm 0.7) \times 10^{-8}$ sec. A description of this technique may be found in G. S. Brunson et al.⁽⁶⁾

DISCUSSION

The critical mass of the calculated core (292.6 kg of U^{235}) compared with the experimental critical mass (394.94 kg of U^{235}) exhibits approximately the same relative difference as the calculated and experimental masses for Assembly 29. This information tends to give credence to the thesis that the inability to predict accurately the critical masses of Assembly 29 and 30 is probably not associated with the oxygen content.

ACKNOWLEDGEMENTS

The authors are indebted to F. W. Thalgott for guidance of the experimental program of which this assembly described is a part, and W. P. Rosenthal for his assistance in the editing of this report.

REFERENCES

1. A. L. Hess et al., Critical Studies of a 450 Liter Uranium Oxide Fast Reactor Core (ZPR-III Assembly 29) ANL 6336, to be published.
2. B. C. Cerutti et al., ZPR-III, Argonne's Fast Critical Facility, Nuclear Science and Engineering, 1, 126 (1956).
3. G. R. Keepin et al., Delayed Neutrons, A Review as of October, 1955, LA-1970 (Oct 1955).
4. F. S. Kirn, An Absolute Fission Counter, American Nuclear Society, 2nd Winter Meeting (1957).
5. J. K. Long et al., Fast Neutron Power Reactor Studies with ZPR-III, Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva (1958), Vol. 12, p. 119.
6. G. S. Brunson et al., Measuring the Prompt Period of a Reactor, Nucleonics 15 (11) 132 (1957).