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EXPERIMENTS WITH A SERIES OF SMALL,
Pu-PLUS-U-FUELED FAST-REACTOR CRITICALS
(FARET Mockup--ZPR-3 Assembly 46)

by

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January 1970

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ABSTRACT

ZPR-3 Assembly 46 was a series of small Pu-plus-U-fueled fast-reactor cores studied in conjunction with design efforts for FARET, a conceptual fast-fuels test reactor. Six different cores, labeled A through F, were constructed as mockups of possible FARET loadings. Cores A and B simulated PuC-UC-fueled cores with volumes of 57 and 68 liters, respectively. Core C was an 87-liter system with an $\text{PuO}_2\cdot\text{UO}_2$ fuel representation. Cores D, E, and F, with volumes of 80, 108, and 95 liters, respectively, were each composed of four annular fuel zones. The different zones contained simulations of oxide, carbide, and alloy fuels as mockups of envisaged multifueled FARET cores. An additional three-zoned core, Core G, was used for fast-reactor physics experiments of a general nature. Principal experiments with the mockups were: evaluating the reactivity control afforded by FARET-design B_4C control rods, measuring worths of fuel interchanges, obtaining flux distributions, and measuring reactivity coefficients for core-constituent materials.

I. INTRODUCTION

A. General Intent of Experiments

The Zero Power Reactor 3 (ZPR-3) is a fast-reactor critical facility operated by Argonne in Idaho. Assembly 46 in this reactor was devoted to mockups of possible loadings of FARET¹ (Fast Reactor Experimental Test), a reactor then intended for irradiation testing of fast-reactor fuels. The FARET design has since been abandoned, however, in favor of a larger test-reactor design with greater possible power densities. The Assembly-46 experiments were planned in conjunction with the design development for the fuel loadings in FARET. Besides the evaluation of possible FARET loadings, the ZPR-3 mockups provided critical data on a series of plutonium cores in the range of 57 to 106 liters for use in checking computational techniques for systems of this size.

In all, seven core designs were constructed in Assembly 46 proper. These included two carbide-fuel simulations, an oxide-fuel simulation, and four multifuel-zoned cores containing oxide-, carbide-, and alloy-fuel simulation regions. The experiments were conducted between January 15 and October 1, 1965, by personnel of Argonne National Laboratory's Idaho Division at the ZPR-3 site at the National Reactor Testing Station in Idaho. The experimental program was established jointly by the Idaho personnel and by the FARET Project Group of the Argonne Reactor Physics Division in Illinois.

B. Presentation and Evaluation of the Results

This report describes the experiments carried out in Assembly 46 and the results thereof in as much detail as is considered necessary by the authors to give a proper understanding of the techniques used and to provide data in forms easily adaptable to computer codes for reactor analysis. Thus, the compositions for the principal drawer loadings are given in atoms/cc, and the major critical configurations of the drawers are diagrammed to enable two-dimensional calculations in X-Y, cell-pattern geometry. Each major core is also described in simplified, equivalent-cylinder geometry with average radial- and axial-zone compositions given in atoms/cc for R-Z cylindrical problems. Finally, for the substitution experiments, compositions in atoms/cc are given for regions, drawers, zones, etc., before and after whatever replacement took place. When applicable, extrapolations from the experiments to situations in FARET are presented.

Analysis of the data was not the responsibility of experimental personnel of the Idaho Division; therefore comparisons of calculated and experimental results are not included in this report. The data obtained were used principally during the design stages for FARET to evaluate computational techniques used in describing the system's physics. Some of the preliminary results are compared with calculations in a design report by Persiani et al.¹ A more thorough summary of analytical efforts and results has also been presented by Persiani at a conference on fast-reactor criticals.²

II. SPECIFIC OBJECTIVES OF THE ASSEMBLY-46 EXPERIMENTAL PROGRAM

A. FARET Design Criteria

The FARET design concept was that of a small reactor capable of testing compact cores at high temperatures and high power densities to assess the performance capabilities of fast-reactor fuels. The sodium coolant system proposed was to accommodate powers of 60 MWt, with outlet temperatures approaching 650°C. Features of the design included adaptability to a variety of core loadings, access for in-core experimental instrumentation, and operational flexibility. Pursuant to its purpose, the reactor cores would have contained annular zones or dispersed mixtures of different types of fuel subassemblies. Differences in fuel elements would have included variations in fuel type (oxide, carbide, metal, etc.), fissile makeup (relative proportions of plutonium, ^{235}U , and ^{238}U), cladding type, and pin diameter. The full scope of the then-planned FARET program has been presented in project reports by Persiani, Simmons, and others.^{1,3,4}

Basically, the reactor portion of FARET would have been constructed from hexagonal core and reflector subassemblies of the same reference design as the EBR-II subassemblies, with a lattice spacing of 5.89 cm. Figure 1 is a cross-sectional view of the FARET design with one of the postulated multizoned core loadings, a configuration that was mocked up in Assembly 46. In this plan, core subassemblies extend to an average outer radius of 24 cm, and reflector subassemblies (72 vol % steel, 18 vol % sodium) encircle the core for a thickness of 30 cm. Sodium then surrounds the reflector out to the primary vessel. In the core subassemblies for this design, the fuel sections were to be 50.8 cm long, with 50.8-cm axial reflector regions (56 vol % steel, 44 vol % sodium) above and below. Reactor control and burnup-reactivity adjustment would have been provided by 12 boron carbide control rods on the periphery of the core.

The overall objective of the Assembly-46 experiments was to obtain critical data on simplified mockups of postulated FARET loadings. Besides evaluation of the specific loading plans, the data were to be used for verification of the calculational models used in the design of these and similar systems. Among the essential experiments, other than critical size, were the determination of the reactivity control offered by the boron carbide control rods, measurements of fuel-interchange reactivity effects, and flux-mapping experiments. Material reactivity coefficients were also needed to assess the effects of compositional discrepancies between the mockups and the FARET designs. Studies were requested on three basic FARET core concepts: a 68-liter carbide-fuel core, a 93-liter oxide-fuel core, and a 93-liter multifuel-zoned core. In all the core mockups, the compositions were to effectively represent FARET core subassemblies of 36 vol % fuel, 44 vol % sodium, and 20 vol % steel.

CORE COMPOSITION	VOL-%	FUEL TYPE	U/Pu ATOM RATIO
FUEL	36	OXIDE	6/1
COOLANT	44	CARBIDE	6/1
STRUCTURE	20	METAL (30 at-% Ti)	4/1

ROW NO.:	1	2	3	4	5*	5*
NO. SUBASSEMBLIES:	1	6	12	18	12	12
TYPE OF FUEL:	(Pu,U)O ₂	(Pu,U)O ₂	(Pu,U)C	(Pu,U)O ₂	(Pu,U)O ₂	(Pu,U)Ti
U-235 ENRICH. (APPROX.):	0.400	0.400	0.489	0.400	0.400	0.114
ELEMENTS/SUBASSEMBLY:	37	37	37	37	37	61
ELEMENT DIAMETER, IN.:	0.29	0.29	0.29	0.29	0.29	0.22

Fig. 1. A Conceptual Loading of the FARET Reactor

B. Carbide Core Mockup

A mockup was requested of a FARET system containing 37 sub-assemblies of a PuC-UC fuel mixture. With a core-length specification of 24 in., the desired volume was 68 liters. The fuel simulation was to have a uranium-to-plutonium atom ratio of 6:1, and the uranium enrichment could be adjusted to achieve the desired critical size. It did develop that the first enrichment chosen for Core 46A was too high, and a lower enrichment was used for Core 46B. The experiments requested for the carbide core, and carried out on Core B, included B₄C control-rod studies, oxide-for-carbide fuel interchanges, flux mapping, and measurements of reactivity coefficients, spectral indices, and neutron lifetime.

C. Oxide Core Mockup

The second design study involved a mockup of a FARET loading with 61 oxide-fuel subassemblies, giving five complete hexagonal rings of core elements as shown in Fig. 1. A core length of 20 in. was specified. This gave 93 liters as the desired volume of the mockup core, which was to be a cylinder based on the ZPR-3 square-cell pattern. The fuel to be simulated was a 6:1 mixture of UO_2 and PuO_2 , with the uranium enrichment determining the critical size. The actual mockup attained was the 87-liter Core C of Assembly 46 wherein the uranium enrichment averaged 49%.

Studies with mockup FARET control rods in the oxide core were to be more extensive, including calibrations of rods with various enrichments in the B_4C . Because oxide fuel was contemplated as the basic loading of FARET, with test subassemblies of other fuel types interspersed, numerous exchange experiments with carbide- and alloy-fuel simulations were requested for the oxide core. Among the other desired experiments were the measurements of sodium worth according to regions in the core.

D. Multizoned-core Mockup

The third major objective in the Assembly 46 program was the mockup of a typical multizoned core in FARET. Figure 1 illustrates the subassembly loading pattern for FARET which was to be represented. As indicated, three types of fuel subassemblies constitute the core loading: a $\text{UO}_2\text{-PuO}_2$ ceramic fuel; a UC-PuC ceramic fuel; and a metallic-alloy fuel containing uranium, plutonium and some light metal (e.g., titanium, zirconium, vanadium). A total of 61 subassemblies of these various fuels are shown loaded into four radial zones, giving a total core volume of 93 liters for a length of 20 in.

Oxide fuel, considered the probable FARET driver fuel, is shown in the Fig. 1 scheme as the loading of the center, second, fourth, and half of the fifth hexagonal rings of subassemblies. The target composition for the mockup was a 6:1 mixture of UO_2 and PuO_2 , with the uranium 40% enriched in ^{235}U .

The third annular ring of subassemblies in the stipulated FARET core would have contained carbide fuel. For the mockup, a simulation was desired of a 6:1 mixture of UC and PuC with the uranium 50% enriched.

Alloy-fuel elements are shown in Fig. 1 as constituting half the subassemblies in the fifth hexagonal annulus in FARET, interspersed singly or in pairs with oxide-fuel subassemblies. Several possible types of alloy fuel could have been tested in FARET; the conceptual design indicates a mixture of 56 at. % uranium, 14 at. % plutonium, and 30 at. % titanium. However, a substitute material was to be used in the mockup for titanium, as no titanium plates were available at ZPR-3. A low uranium enrichment (about 10%) was

required in the alloy simulation, because in FARET the power densities in the oxide- and alloy-fuel subassemblies in the outer core zone were to be nearly the same.

The multizoned-core mockup in Assembly 46 was thus to contain four fuel zones: Zone I containing seven oxide fuel units, Zone II with 12 carbide fuel units, Zone III with 18 oxide-fuel units, and Zone IV containing 12 oxide-fuel and 12 alloy-fuel units. A full drawer unit, with a 10-in. axial fuel length, in both halves of ZPR-3 was to be equivalent to one FARET subassembly. A close representation of the FARET hexagonal geometry was desired. In the attempt to mock up this 61-subassembly configuration, three different multizoned cores (Cores D, E, and F) were actually built in Assembly 46. The variations in the three cases were only in the zone uranium enrichments. Core F was an acceptable mockup, and the experiments conducted with it included fuel exchanges and material-worth measurements in each core zone.

E. ZPR-3 Experimental Core

A fourth zoned-core system was constructed wherein the first three "subassembly" rings contained carbide-fuel drawers, the fourth ring had oxide-fuel drawers, and the fifth ring contained oxide and a few alloy-fuel drawers. This system was used for ZPR-3 experimental studies not involved with FARET. A technique for measuring fission ratios was tested at the center of the carbide zone of a core design called 46G. A slightly altered core was used for studies of uranium-blanket worths.

III. DESIGN OF ASSEMBLY

A. Description of ZPR-3

Zero Power Reactor 3, shown in Fig. 2, is a split-halves, critical-assembly machine used for the study of fast-reactor systems. The machine halves, one fixed and one movable, are each composed of an array of matrix tubes into which drawers are inserted. A variety of materials in plate or block form are available for loading into the drawers to achieve the desired representation of the core and blanket of a fast reactor with the machine halves together. Figure 3 illustrates typical loadings of drawers. The facility and its construction, instrumentation, and operation have been described by Cerutti *et al.*,⁵ Brittan *et al.*,⁶ and Long.⁷ Besides the study of the physics of simple fast-reactor systems, the ZPR-3 has been used for mockups of several other fast-power-reactor designs.

B. Radial Reflector Regions

The general design of Assembly 46 was a geometric simulation of the postulated FARET design. As seen in Fig. 1, the FARET system prescribed a section of core subassemblies surrounded by steel radial-reflector subassemblies, which would then have been contained within a sodium plenum. Figure 4 shows the Assembly-46 representation of this radial configuration. For FARET, the composition specified for the radial reflector was 82 vol % steel and 18 vol % sodium. In Assembly 46, a two-zone radial reflector was built, with an inner "fine" zone simulating the FARET reflector containing sodium, and an outer "coarse" zone composed of steel bricks. A "plenum" region surrounding the reflector in the assembly contained nickel, aluminum, and void as a representation of the FARET sodium plenum. The drawer loadings for these regions are shown in Figs. 5a through 5c, and the resulting compositions of the various reflector regions constructed from these drawers are given in Table I. Throughout the Assembly-46 experiments, the outlines of the radial reflector regions remained as illustrated in Fig. 4, except for the inner boundary of the fine reflector, which changed for each core. The average radial extensions for the regions will be included on the diagrams for each core.

C. Axial Reflector Regions

The sodium content in the axial reflectors of Assembly 46 was the same as in the cores, 39 vol %, simulating the continuation of the coolant channels above and below the core in the FARET design. The steel content was 54 vol %, which about equaled the combined content of the fuel and structural components in the cores. Figure 6 shows the loading of a typical 21-in. core drawer with a 12-in. section of core followed by 9 in. of axial reflector loading. The 11-in. backup drawers for the core region were loaded as shown in Fig. 5d.

Fig. 2. View of ZPR-3 Critical Facility. ANL Neg. No. ID-103-2157.

Fig. 3. Typical Core and Blanket Drawer Loadings. ANL Neg. No. ID-103-2231.

Fig. 4

Half No. 1 Interface View of Assembly 46

Fig. 5. Assembly-46 Reflector Region Loadings. ANL Neg. No. ID-103-2516.

TABLE I. Average Compositions of Assembly-46 Reflector Regions

Reflector Region	Material Densities, atoms/cm ³ x 10 ⁻²²						
	Fe	Cr	Ni	Mn	C	Na	Al
Axial reflector	3.425	0.920	0.452	-	-	0.849	-
Fine radial reflector	4.690	1.259	0.620	-	-	0.212	0.306
Coarse radial reflector I	7.413	0.119	0.058	0.057	0.065	-	-
Coarse radial reflector II	6.579	0.830	0.409	0.028	0.035	-	-
Plenum	0.445	0.119	1.923	-	-	-	1.384
Spring gap ^a	1.654	0.444	0.218	-	-	-	-

^aBetween 53.42 and 54.08 cm from interface in axial reflector regions of both assembly halves.

Fig. 6
Fuel-drawer Load-
ing for Core 46A

Appendix A includes the compositions for the individual types of drawer loading for the axial reflectors. The average atom densities of materials in the reflector regions of the assembly are listed in Table I. For each core configuration, an axial schematic diagram is presented which includes the axial extensions of the reflector regions. For example, Fig. 7 shows the various axial and radial boundaries for the assembly with Core 46A. The coarse radial reflector II (CRR-II) indicated in the diagram behind the fine radial reflector was constructed with steel bricks in backup drawers.

D. General Aspects of Core Designs

The drawer loadings for the various Assembly-46 cores were designed to simulate the FARET specifications of 36 vol % fuel, 44 vol % coolant, and 20 vol % steel as closely as possible, at least with regard to reactivity. For the representations of oxide, carbide, or alloy fuels, practical densities in FARET fuel pins were assumed in order to derive the target atomic densities for plutonium, ^{235}U , ^{238}U , carbon, oxygen, etc., in each case. These fuel specifications were simulated very well in the Assembly-46 cores. Columns of canned sodium were placed in the remaining core space, where possible, to achieve the coolant representation. However, the resulting density for sodium simulated only 39 vol % (at 500°C). The combinations of matrix tubes, drawers, cans, and claddings gave steel contents ranging around 18 vol %. A void content of about 7 vol % in all the cores was unavoidable, being an inherent feature of the ZPR-3 construction.

During the experiments, core drawer loadings simulating carbide (PuC-UC), oxide ($\text{PuO}_2\text{-UO}_2$), and alloy (Pu-U-Zr) fuels were made, with variations of the uranium enrichment in each. Specific details of the

material arrangements in the drawers are given below for each core, and the material densities for all the core drawer types used are listed in Appendix B. The compositions and the regional dimensions for the critical configurations are presented for each core.

Fig. 7. Axial Schematic Diagram of Assembly 46A

IV. CORE-A EXPERIMENTS

A. Composition of Core A

Core A in Assembly 46 was the initial attempt to construct the 24-in.-long, 68-liter mockup of a carbide-fueled FARET system. In the Core-A UC-PuC fuel simulation, the composition attained was a uranium-to-plutonium atom ratio of 5.5:1 and a uranium enrichment of 45.7% ^{235}U . The fuel representation was achieved by using plates of plutonium (canned in steel), 93%-enriched uranium, 31%-enriched uranium, depleted uranium, and graphite. Figure 6 shows the arrangement of the materials in the regular 46A core drawers, indicating the vertical and axial placements of the pieces. Some of the core drawers were used as reactor control rods, and these contained more steel and less sodium. The compositions of the regular drawers and the control drawers for Core 46A are given in Appendix B. For the critical configuration described below, the average composition for the core was that given in Table II.

TABLE II. Specifications of Core 46A

Material	Atoms/cm ³ x 10 ⁻²²	Material	Atoms/cm ³ x 10 ⁻²²
a. <u>Average Composition</u>		b. <u>Critical Loading</u>	
^{239}Pu	0.1558	Length	61.1 cm (24.1 in.)
$^{240+242}\text{Pu}$	0.0074	Average radius	17.2 cm (6.8 in.)
^{241}Pu	0.0007	Critical volume	56.8 ± 0.1 liters
^{235}U	0.4099	Total fuel:	
^{238}U	0.4819	Plutonium	36.99 kg
$^{234+236}\text{U}$	0.0058	^{235}U	90.97 kg
C	1.1096		
Na	0.8378		
Fe	1.1171		
Cr	0.2999		
Ni	0.1476		

B. Critical Experiment1. Approach to Critical

The initial loading of Core 46A contained nine core drawers in each assembly half. Approach-to-critical curves of inverse power versus volume and of volume/power versus volume for this experiment are shown in Fig. 8. The fuel additions were of from two to four drawers per half (replacing reflector) until the critical loading shown in Fig. 9 was achieved. These were derived from BF_3 proportional counters at the outer edge of the fine radial reflector.

Fig. 8. Approach-to-Critical Curves for Core 46A

icality of 17.2 cm, and Fig. 7 indicates this value along with the other Assembly-46A regional dimensions.

2. Critical Configuration

The critical loading of Core A, shown in Fig. 9, contained a total of 61 core drawers (57.1 liters) and had the average composition given in Table II. This loading had an available excess reactivity of approximately 0.05% k. No fuel worths were measured at the edge of Core A for assessment of the excess volume beyond just critical. However, the approach-to-critical curves indicate a just-critical Core-A volume of 56.8 ± 0.1 liters. As can be seen in Fig. 9, no attempt was made to obtain a smooth cylindrical outline. On the basis of experience with a similar, small, plutonium-fueled core (RAPSODIE mockup⁸), the irregular radial outline probably incurred an extra half liter for criticality over what would be required of a circular configuration. A 56.8-liter volume equates to an average radius at crit-

Fig. 9. Critical Configuration of Core 46A

The volume attained with Core A was smaller than desired for the studies planned on the carbide system. Consequently, the uranium enrichment in the core drawers was reduced to provide a less reactive composition, that of Core B.

V. CORE-B EXPERIMENTS

A. Composition of Core B

A less reactive carbide-fuel simulation was obtained by replacing the 1/8-in. column of 31%-enriched uranium in the 46A core drawer design with a column of depleted uranium (compare front views in Figs. 6 and 10A). The material arrangements in the regular Core-B drawer were thus as shown in Fig. 10A, and the resulting material densities for this loading and for the core control drawers are listed in Appendix B. Compositions are given in Table VI (on p. 31), for the regular 46B core drawer, and in Table III, which describes the Core-B critical loading including control drawers. In this Core-B design, the resulting average uranium enrichment was 37.2 at. % ^{235}U , with the U:Pu ratio kept at 5.5:1.

A. REGULAR 46B CORE DRAWER, CARBIDE-FUEL SIMULATION

B. OXIDE-FUEL SIMULATION DRAWER FOR EXCHANGE EXPERIMENTS

Fig. 10. Fuel-drawer Loadings Used in Core 46B. ANL Neg. No. ID-103-2540.

B. Critical Experiment

1. Critical Configuration

Since the composition change entailed a reactivity loss, the approach to critical with Core B was started at a loading of 25 drawers per assembly half. Fuel-for-reflector replacements were then made until again attaining criticality. Figure 11 outlines the reference critical

loading of Core 46B with a total of 73 fuel drawers. The Table III compositions are for this configuration. With all control drawers inserted, the available excess reactivity of the loading was determined from later evaluations as 143 ± 2 lh.

TABLE III. Specifications of Core 46B

Material	Atoms/cm ³ x 10 ⁻²²	Material	Atoms/cm ³ x 10 ⁻²²
a. Average Composition		b. Critical Loading	
²³⁹ Pu	0.1558	Length	61.1 cm (24.1 in.)
²⁴⁰⁺²⁴² Pu	0.0074	Average radius	18.8 cm
²⁴¹ Pu	0.0007	Critical volume	67.8 ± 0.1 liters
²³⁵ U	0.3346	Total fuel:	
²³⁸ U	0.5598	Plutonium	44.10 kg
²³⁴⁺²³⁶ U	0.0048	²³⁵ U	88.48 kg
C	1.1096		
Na	0.8469		
Fe	1.0908		
Cr	0.2929		
Ni	0.1441

Fig. 11. Critical Configuration of Core 46B

2. Fuel Worth at Core Edge

The worth of replacement of radial reflector by fuel at the edge of Core B was measured in two steps. For the outer half of drawer 2-Q-14, the substitution worth was 122.8 lh. The replacement in a complementary half-drawer in 2-Q-20 (outer half) was measured twice, yielding values of

104.8 and 109.2 lh. Thus, the average core-edge fuel worth was 230 ± 5 lh per drawer for this location on the periphery. The uncertainty of this value as representing an average fuel worth around the core perimeter is probably ± 20 lh. In terms of reactivity change per differential volume change, this gives the relation

$$\Delta k/k = 0.31 \Delta V/V.$$

3. Critical Volume of Core B

The excess reactivity of the Fig. 11 core loading was equivalent to 0.62 edge-core drawers, giving a just-critical loading of 72.4 drawers, or a critical volume of 67.8 ± 0.1 liters for the 46B core composition. This volume represented as a 61.1-in.-long circular cylinder would provide a critical radius of 18.8 cm, which is shown in the schematic for Assembly 46B in Fig. 12. Again, no adjustment has been applied to the critical size to account for the irregular cylindrical boundary. Temperatures across the assembly ranged up to 15°C above room air, averaging about 27°C in the core.

C. Control-rod Calibrations

The ZPR-3 control rods consist of special movable drawers, and their calibrations in all the Assembly-46 cores were carried out by an equilibration method. This involved equating the worths of successive increments of a rod moved inward and a rod moved outward with the worth of a specific portion of a third rod. This defined the relative worth versus portion of the rods quite accurately. An absolute calibration of reactivity in inhours was then derived by period measurements for a few rod increments. In this assembly, the power was run up to 200 W during period measurements to minimize effects from the spontaneous-fission neutron source. Figure 13 shows the calibrations of rod No. 1 (a core drawer in 2-N-19 in Fig. 11) and of rod No. 10 (a reflector drawer in 1-N-14) for Core 46B.

Details of reactor calculations for the Assembly-46B delayed-neutron characteristics are given in Appendix C. The principal results used for reactivity conversions are:

$$\begin{aligned}\beta_{\text{eff}} &= 0.00535, \\ 1\% k &= 560 \text{ lh}, \\ 1\$ &= 300 \text{ lh},\end{aligned}$$

and

$$\text{prompt lifetime} = 2.91 \times 10^{-7} \text{ sec.}$$

Fig. 12
Axial Schematic Diagram of Assembly 46B. ANL Neg. No. ID-103-2518.

Fig. 13
Rod Calibrations in Assembly 46B

D. Spectral Indices at Core Center

1. Fission Ratios

Ratios of the average fission cross sections of the various uranium and plutonium isotopes were measured in the central neutron spectrum by using absolute fission chambers. The design details of the chambers involved have been reported by Kirn⁹ and Davey.¹⁰ A 2 x 2 x 1-in. recess was constructed at the front of drawer 1-P-17 to accommodate the chambers. The core materials in drawer 2-P-17 were also recessed an inch away from the fission plating of the counter to diminish the effects of composition heterogeneity. Relative count rates were established for the various counters placed in turn in 1-P-17, and the results are given in Table IV, column 4. The statistics involved in the experiments imply accuracies of 0.5 to 1% for the count ratios. Column 6 in the table lists the fission ratios derived from the data; these values have an uncertainty of $\pm 3\%$ because of uncertainties in calibrations and in corrections for the effects of chamber-wall scattering upon the spectrum.

2. ²³⁸U Capture Rate

The ratio of capture in ²³⁸U to fission in ²³⁵U was measured at the center of Core 46B by Davey.¹¹ A new technique was used which combined an activation analysis for capture in a depleted-uranium foil with absolute counting of fissions in an enriched-uranium foil. The referenced report describes the methods and includes the results for this and other assemblies. For Core 46B, the measurements gave

$$\sigma_c(^{238}\text{U})/\sigma_f(^{235}\text{U}) = 0.111 \pm 0.005.$$

TABLE IV. Fission Ratios at Center of Core 46B

Fission Chamber, No.--Type	Plating Mass, μg	Isotopic Composition, wt %	Relative Count Rate, ($\pm 0.5\%$)	Derived Central Fission Ratio	
				Isotopes	Value ($\pm 3\%$)
TWC No. 43-- ²³⁵ U	419	²³⁵ U: 93.17, ²³⁸ U: 5.50, ²³⁴ U: 1.04, ²³⁶ U: 0.29	1.0000	-	-
TWC No. 24-- ²³³ U	504	²³³ U: 99.92, ²³⁵ U: 0.02, ²³⁸ U: 0.06, ²³⁴ U: 0.02, ²³⁶ U: 0.02	2.0100	233/235	1.559
TWC No. 51-- ²³⁴ U	522	²³⁴ U: 97.56, ²³⁵ U: 1.85, ²³⁸ U: 0.59	0.5688	234/235	0.419
TWC No. 27-- ²³⁶ U	397	²³⁶ U: 96.48, ²³⁵ U: 2.85, ²³⁸ U: 0.62, ²³⁴ U: 0.06	0.1607	236/235	0.140
TWC No. 14A-- ²³⁸ U	493	²³⁸ U: 99.27, ²³⁵ U: 0.72, ²³⁴ U: 0.01	0.0914	238/235	0.067
Kirn No. 21-- ²³⁹ Pu	490	²³⁹ Pu: 99.97, ²⁴⁰ Pu: 0.03	1.4961	239/235	1.223
TWC No. 49-- ²⁴⁰ Pu	445	²⁴⁰ Pu: 99.62, ²³⁹ Pu: 0.38	0.5326	240/235	0.479

E. Distributed Worths of Materials

The core-average reactivity coefficients of the core materials, except for plutonium and ^{235}U , were determined from two series of material-substitution experiments. One series involved measuring the average worths in most of the core, excluding the central drawers, 1- and 2-P-17. Nine drawer locations in Half No. 1 were selected, which together represented a quadrant of the core, in half the assembly, with regard to their relative placement around the core axis. Figure 14 illustrates the distribution of these drawers. The other series of material replacement was made in the central drawers, 1- and 2-P-17. In both series, successive substitutions were made in the central columns of the drawers with sodium cans, empty cans, steel, graphite, U_3O_8 , and depleted uranium. Material worths relative to void were derived from the resulting reactivity changes. Column 2 of Table V lists the material masses substituted per drawer. In the table, the reactivity coefficients obtained (as lh/kg) are given in columns 3 and 4 for the distributed-quadrant and central-drawer experiments, respectively. The last column of Table V states the core-average reactivity coefficients derived from the quadrant values and central values.

F. Fuel-exchange Experiments

The experimental program considered for FARET included the loading of cores composed of several different types of fuel subassemblies. In a basically carbide core, the testing of a few oxide-type subassemblies might have been attempted. A core containing annular zones of different fuels was also possible, as indicated in the description of FARET given in

Fig. 14. Substitution Locations for Material Worths in Core 46B

TABLE V. Distributed Worths of Materials in Core 46B

Material	Mass Substituted per Drawer, g	Reactivity Coefficients, $\Delta k/k$		
		Nine-drawer Distributed Quadrant in Half No. 1	Central Drawers (1- and 2-P-17)	Core Average
Sodium	77	55.9 ± 0.7	38.6 ± 1.3	55.4 ± 0.7
Stainless steel (304)	657	9.9 ± 0.1	-1.25 ± 0.22	9.6 ± 0.1
Graphite	77	132 ± 1	114 ± 3	132 ± 1
U_3O_8	356	20.3 ± 0.2	16.2 ± 0.3	20.2 ± 0.2
Depleted uranium	893	7.4 ± 0.1	6.6 ± 0.2	7.4 ± 0.1
Oxygen ^a	-	92.3 ± 1.5	70 ± 2	91.7 ± 1.5

^aCoefficients derived from U_3O_8 and depleted-uranium values.

Section II. In the Assembly-46 experiments, therefore, core drawers of different fuel simulations were interchanged to evaluate the reactivity effects that could be expected in FARET upon exchange of fuel subassemblies. Substitutions of an oxide-fuel simulation were made for carbide fuel in Core-46B drawers at a number of locations. The loading of the special oxide-core drawers is shown in Fig. 10B, and its composition is given in Appendix B and Table VI.

TABLE VI. Oxide-for-Carbide Fuel-exchange Experiments in Core 46B

a. Composition Change in Substitution												
	Material Densities, $\text{atoms/cm}^3 \times 10^{-22}$											
	^{239}Pu	$^{240+242}\text{Pu}$	^{241}Pu	^{235}U	^{238}U	$^{234+236}\text{U}$	O	C	Na	Fe	Cr	Ni
From	0.1558	0.0074	0.0007	0.3346	0.5598	0.0048	-	1.1096	0.8585	1.0576	0.2840	0.1397
To	0.1062	0.0051	0.0005	0.3340	0.3406	0.0048	0.8586	0.5044	0.8531	1.0909	0.2928	0.1441
b. Substitution Results												
Experiment No.	Description of Substitution ^a					Volume of Substitution, liters	Approximate Radial Location, cm			Substitution Worth, $\Delta k/k$		
1	Single-drawer exchange in 2-P-17					0.9364	0 to 3.1			-86.7 ± 0.9		
2	Single-drawer exchange in 2-Q-17					0.9364	3.1 to 8.5			-80.6 ± 0.8		
3	Single-drawer exchange in 2-R-17					0.9364	8.5 to 14.0			-64.1 ± 0.6		
4	Single-drawer exchange in 2-S-17					0.9364	14.0 to 19.5			-42.5 ± 0.5		
5a	Four-drawer exchange: 2-P-15, half of 2-Q-15, half of 2-Q-16, 2-R-16, and 2-R-17					3.7456	8.5 to 14.0			-248.1 ± 1.2		
5b	Two-drawer exchange: 2-R-18 and 1-R-18					1.8728	8.7 to 14.3			-120.4 ± 0.5		
5c	Four-drawer exchange: 1-P-15, half of 1-Q-15, half of 1-Q-16, 1-R-16, and 1-R-17					3.7456	8.5 to 14.0			-249.4 ± 4		
5 (total)	Five-drawer annular sector (both halves)					9.3640	8.5 to 14.0			-618 ± 6		

^aSee Fig. 15 for geometric details.

1. Single-drawer Substitutions, Oxide for Carbide

The Core-B carbide fuel drawers were replaced at four locations in Half No. 2 with an oxide-fuel drawer. The drawer positions, shown in Fig. 15, were in rows P, Q, R, and S of column 17, and these are equivalent to positions in the first four rows of subassemblies in the FARET design. Specific details and results of the experiments are presented in Table VI.

2. Annular-sector Substitution, Oxide-for-Carbide

An oxide-for-carbide fuel exchange was also carried out in an annular sector of the core representing a five-subassembly sector of the third hexagonal ring of subassemblies in the FARET design. The five drawer units composing the sector in both halves of ZPR-3 are shown in Fig. 15. In this experiment, the substitutions were actually carried out in three steps, and these are described in Table VI along with the results. Of particular importance in this study was the determination of how well a single-drawer substitution could predict the worth of a fuel exchange in the whole annulus. The result for the annular sector, involving a total of 10 drawers in the third ring, was 9.64 times the result for a single drawer in the ring (in 2-P-17). The two-drawer substitution in the second step of the sector replacement (experiment 5b in Table VI), with the sector partially changed already, was worth 1.88 times the single substitution. Thus, considering the coupling and geometry effects, the single-drawer substitution can predict the annulus replacement to about $\pm 5\%$.

G. Experiments with Mockup B_4C Control Rods

The worths of B_4C control-rod simulations were measured in the radial reflector at a position representative of the control-rod locations

Fig. 15. Locations of Fuel Exchange Experiments in Core 46B

in the FARET design. First, the worth of replacing reflector material by sodium in a drawer near the core boundary was determined. Then mockup boron carbide rods of two ^{10}B enrichments were substituted for the sodium.

1. Sodium "Flooding" of Rod Channel

Matrix unit 1-P-12, in the second row of the radial reflector, was chosen as the "channel" location for the mockup rod studies. This channel represented the half-length of a subassembly in the sixth ring of elements in the FARET hexagonal pattern. The reflector material in 1-P-12, including the back drawer, for a total of 32 in. (81.3 cm) was replaced with canned sodium. This "flooding" of a half-channel gave an 18.0-lh reactivity loss. Table VII presents the details of the experiment. In the FARET reactor, the removal of a full reflector subassembly at a comparable location would have therefore been worth -36 lh, or about -0.07% k.

TABLE VII. Mockup B_4C Control-rod Experiments in Assembly 46B

a. <u>Compositions Involved</u>								
Channel Section Loading	Material Densities, atoms/cm ³ x 10 ⁻²²							
	^{10}B	^{11}B	C	Fe	Cr	Ni	Na	Al
Fine radial reflector	-	-	-	4.690	1.259	0.620	0.212	0.306
Coarse radial reflector	-	-	-	5.638	1.514	0.745	-	-
Sodium filled	-	-	-	1.310	0.352	0.173	1.695	-
Natural B_4C active rod	0.562	2.363	0.788	1.539	0.413	0.203	0.846	-
53% enriched B_4C active rod	1.436	1.300	0.868	1.539	0.413	0.203	0.846	-
Rod follower	-	-	-	3.440	0.924	0.454	0.848	-

b. Relative Worths of Channel Loadings

1-P-12 channel having 5.54 x 5.53-cm cross section (30.64 cm²), average radius from reactor axis = 27.7 cm.

1-P-12 Channel Loading Specification	Sectional Loading of Channel ^a		Reactivity Worth, lh (relative to reflector)
	Axial Extension from Reactor Midplane	Volume, liters	
Reflector	0 to 53.4 cm: fine radial reflector; 54.1 to 79.5 cm: coarse radial reflector	1.636 0.778	0
"Flooded"	0 to 53.4, 54.1 to 82.0 cm: sodium-filled	2.491	-18.0 ± 0.3
Natural B_4C mockup rod	0 to 45.8 cm: natural B_4C active rod; 45.8 to 53.4, 54.1 to 82.0 cm: rod follower	1.403 1.090	-99.8 ± 1.0
Enriched B_4C mockup rod	0 to 45.8 cm: 53% enriched B_4C active rod; 45.8 to 53.4, 54.1 to 82.0 cm: rod follower	1.403 1.090	-169.1 ± 1.5

^aIn all cases, spring gap from 53.4 to 54.1 cm remains unchanged.

2. Natural B_4C Mockup Control Rod

The FARET control-rod design simulated in the Core-B rod studies had a 36-in.-long poison section containing 30 vol % B_4C , 25 vol % steel, and 38 vol % sodium, with followers of 55 vol % steel and 38 vol %

Fig. 16. Loadings of Mockup B_4C Control Rods for Assembly-46B Experiments in Drawer 1-P-12. ANL Neg. No. ID-103-2517.

B_4C . The loading for this half-length mockup rod is shown in Fig. 16b, and its composition is included in Table VII.

Relative to the natural B_4C loading, the worth of the 53%-enriched rod in the 1-P-12 channel was more negative by 69.3 lh. Thus, compared with sodium, the enriched half-length mockup was worth -151.1 lh. A FARET control rod with a 36-in. active section and the same composition would therefore have provided about 300 lh, or 0.6% $\Delta k/k$, reactivity control.

H. Axial Traverses of Reactivity Samples

The drawers in 1- and 2-P-17 were reloaded according to the patterns shown in Fig. 17a to accommodate a 1/2-in.-diam traverse guide tube through the length of the reactor. Traverses were then made along the reactor axis with small reactivity samples of enriched uranium,

sodium above and below. Mockups of this design were made for half the active length and substituted in 1-P-12 to represent half of a control rod in FARET. First, a mockup rod using all natural B_4C as the poison in the 18-in. active section was constructed, as shown in Fig. 16a. Replacing sodium in 1-P-12 with this rod mockup produced an 81.8-lh loss of reactivity. Table VII gives the composition changes involved for the channel. A comparably loaded, natural B_4C rod in FARET would have provided about 160 lh, or 1/4% k , of reactivity control.

3. 53%-enriched B_4C Mockup Control Rod

A boron-carbide-loaded active section with an average boron enrichment of 53% ^{10}B was created by replacing about half the natural B_4C in the previous design with 90%-enriched

natural uranium, and plutonium. Specifications for these samples are listed in Table VIII. The reactivity variation induced by the change of position of the sample was compensated for automatically by a calibrated autorod, and monitoring the autorod position provided the reactivity worths versus sample position.

Fig. 17. Drawer Loadings for Assembly-46B Traverse Experiments

TABLE VIII. Specifications of Reactivity Samples Used in Traverse Experiments

Sample Material	Dimensions, cm		Material Mass, g	Material Composition
	Length	Diameter		
Enriched uranium	1.27	1.11	22.54	93.1 wt % ^{235}U , 6.9 wt % $^{238+236}\text{U}$
Natural uranium	5.08	1.11	64.39	0.7 wt % ^{235}U , 99.3 wt % ^{238}U
Plutonium	5.08 ^a	0.92 ^a	24.09	94.9 wt % ^{239}Pu , 4.8 wt % ^{240}Pu , 0.3 wt % ^{241}Pu
Enriched boron	5.08 ^a	0.92 ^a	3.25	93.5 wt % ^{10}B , 3.6 wt % ^{11}B , 2.9 wt % impurity

^aExterior dimensions of steel canning.

In these Assembly-46B axial-worth scans, a continuous-motion mode of traverse operation was used. The position of the autorod was indicated by the frequency of pulses emitted from a voltage-to-frequency converter. During a traverse, the pulses were input to a multichannel analyzer operating in the time mode. Two hundred time channels were used to scale the autorod readout when the traverse was run in, and another 200 channels were used when the traverse was run out again. The traverse cycle was repeated several times for better statistics, and then counts in the run-in

Fig. 18. Data from Axial Traverse of
Natural Uranium Reactivity
Sample in Assembly 46B

channels and run-out channels for corresponding traverse locations were summed. In this way, temperature drift effects should have been canceled out. Figure 18 is an example of the raw data obtained. This type of data was first run through a curve-smoothing routine before evaluating the reactivity worths versus traverse position.

In addition to the sample traverses, blank runs were made to provide corrections according to positions for the worths of the push rods and sample cannings. Table IX lists the resulting reactivity coefficients obtained from sample-material worths as a function of axial position. Columns 2 and 3 in the table are the worths per kilogram for the ^{235}U and ^{238}U isotopes as derived from the enriched-uranium and natural-uranium worths. The coefficients versus position for ^{235}U and plutonium are plotted in Fig. 19, and Fig. 20 shows

the axial worths for ^{238}U . It should be realized that the results given here, and also for all of the reactivity traverses in Assembly 46, are average values of material worths in the sample geometries specified. No adjustments have been made for whatever self-shielding effects are present or for the influence of fuel proximity to the traverse channels.

TABLE IX. Uranium and Plutonium Worths versus Axial Position in Assembly 46B

Axial Position, cm	Reactivity Coefficients, lh/kg			Axial Position, cm	Reactivity Coefficients, lh/kg		
	$^{235}\text{U}^a$ (± 2)	$^{238}\text{U}^a$ (± 0.25)	Plutonium ^b (± 3)		$^{235}\text{U}^a$ (± 2)	$^{238}\text{U}^a$ (± 0.25)	Plutonium ^b (± 3)
0.0	354	2.97	619	40.0	45	1.27	59
2.5	354	3.11	619	42.5	37	0.67	47
5.0	350	3.47	612	45.0	32	0.36	39
7.5	340	3.87	597	47.5	29	0.21	33
10.0	324	4.28	569	50.0	25	0.08	28
12.5	304	4.96	534	52.5	20	-	22
15.0	281	6.20	491	55.0	16	-	18
17.5	255	7.42	443	57.5	13	-	16
20.0	228	8.64	394	60.0	12	-	13
22.5	198	10.14	340	62.5	9	-	9
25.0	167	11.66	283	65.0	7	-	8
27.5	140	12.22	228	67.5	6	-	6
30.0	110	10.93	172	70.0	4.3	-	4.5
32.5	81	7.68	123	72.5	2.4	-	2.8
35.0	63	4.20	92	75.0	1.9	-	1.6
37.5	53	2.04	72	77.5	0.7	-	0.4

^aValues for isotope, derived from worths of enriched- and natural-uranium samples specified in Table VIII.

^bValue for plutonium mixture of sample specified in Table VIII.

Fig. 19. Axial Profiles of Plutonium and ^{235}U Worths in Assembly 46B

Fig. 20. Axial Profile of ^{238}U Worth in Assembly 46B

I. Radial Traverses of Reactivity Samples

The core and reflector drawers in the P row of Half No. 1 of the reactor were modified according to Fig. 17b to provide a traverse channel radially through the assembly near the midplane. Again, traverses were run with enriched-uranium, natural-uranium, and plutonium reactivity samples. In the radial traverses, however, a stepped mode of traversing was used wherein the autorod-position readout was averaged over 200-sec intervals with the sample at fixed locations. The temperature-drift effects were then evaluated by repeating reference locations. The results of these experiments are given in Table X. Included are the derived reactivity

TABLE X. Uranium and Plutonium Worths versus Radius near Midplane in Assembly 46B

Radius, cm	Reactivity Coefficient, Ih/kg			Radius, cm	Reactivity Coefficient, Ih/kg		
	$^{235}\text{U}^a$ (± 3)	$^{238}\text{U}^a$ (± 0.6)	Plutonium ^b (± 4)		$^{235}\text{U}^a$ (± 3)	$^{238}\text{U}^a$ (± 0.6)	Plutonium ^b (± 4)
0	359	2.4	616	25.40	96	-	4
2.54	352	2.4	607	27.94	-	2.3	99
5.08	340	3.9	585	30.48	71	-	-
7.62	-	5.8	542	33.02	-	1.8	68
10.16	290	8.0	-	35.56	50	-	-
12.70	-	10.6	421	38.10	-	1.2	44
15.24	221	12.3	356	43.18	25	0.5	26
17.78	-	12.9	280	48.26	-	0.1	14
18.03	180	-	-	50.80	10	-	-
19.30	149	-	-	58.42	-	-	3
20.32	144	10.4	212	60.96	2	-	-
22.86	-	5.8	156				

^aValues for isotope, derived from worths of enriched- and natural-uranium samples specified in Table VIII.

^bValue for plutonium mixture of sample specified in Table VIII.

coefficients versus radius for the isotopes ^{235}U and ^{238}U . Figure 21 is a graph of the values for ^{235}U and plutonium; the values for ^{238}U are plotted in Fig. 22.

Fig. 21. Radial Profiles of Plutonium and ^{235}U Worths in Assembly 46B

Fig. 22. Radial Profile of ^{238}U Worth in Assembly 46B

J. Axial Counter Traverses

While the traverse guide tube was installed axially through 1- and 2-P-17, axial scans were also made with small neutron detectors. The details of all the traverse counters utilized during the Assembly-46 program are given in Table XI. In the 46B axial traverses, the counters used were the BF_3 counter and the fission counters of enriched U (No. H), depleted uranium, plutonium and ^{234}U . Count rates were measured at points

TABLE XI. Specifications of Neutron Detectors Used in Traverse Experiments

Counter Label	Plating or Filling	Active Length, cm	Plating or Inner Diameter, cm	Isotopic Composition
F 2199	Enriched BF_3 , 40 cm Hg	5.08	0.95	Approximately 90 at. % ^{10}B in boron
J	5.1 mg Enriched uranium	2.06	0.87	93.2 wt % ^{235}U , 5.5 wt % ^{238}U , 1.3 wt % $^{234}+^{236}\text{U}$
H	0.94 mg Enriched uranium	2.06	0.87	93.2 wt % ^{235}U , 5.5 wt % ^{238}U , 1.3 wt % $^{234}+^{236}\text{U}$
IC24-4	2.8 mg Depleted uranium	2.06	0.87	80 ppm ^{235}U
K	2.8 mg ^{234}U mixture	2.06	0.87	89.3 wt % ^{234}U , 7.7 wt % ^{235}U , 3.0 wt % $^{238}+^{236}\text{U}$
S	2.04 mg plutonium	2.06	0.87	94.9 wt % ^{239}Pu , 4.8 wt % $^{240}+^{242}\text{Pu}$, 0.3 wt % ^{241}Pu

on both sides of the reactor midplane and then averaged to derive relative fission or reaction rates for the principal counter isotopes versus axial position. Table XII presents the results, expressing rates as percentages of core-center rates. To derive the ^{239}Pu relative rates from the plutonium-counter traverse data, corrections were applied for the ^{240}Pu fissions according to assumed fission ratio values. The fission-rate profiles for the

TABLE XII. Reaction Rates versus
Axial Position in Assembly 46B

Axial Position, cm	Relative Reaction Rate, ^a Percent of Rate at Core Center				
	^{235}U	^{238}U	^{234}U	Plutonium	^{10}B
	Fission ^b	Fission ^b	Fission ^b	Fission ^c	(n, α) ^b
0	100.0	100.0	100.0	100.0	100.0
2.54	99.6	100.0	100.5	99.7	100.0
5.08	99.4	99.8	99.9	98.8	98.9
7.62	-	-	-	-	97.5
10.16	95.2	96.8	96.0	95.7	95.2
15.24	89.0	88.5	87.8	88.5	89.5
17.78	-	-	-	-	85.8
20.32	79.7	77.7	77.3	79.0	82.0
22.86	74.6	71.3	70.8	73.6	79.0
25.40	69.6	63.0	63.4	67.7	78.7
27.94	65.9	53.2	53.8	62.3	82.0
30.48	65.6	38.8	42.2	60.0	98.0
33.02	69.1	25.4	30.5	61.0	122.5
35.56	71.1	17.0	23.3	62.1	-
38.10	72.9	12.0	17.6	62.9	159.4
40.64	73.0	8.5	13.2	63.0	-
43.18	72.1	6.0	9.8	62.1	176.5
45.72	70.2	4.3	7.8	60.8	-
50.80	63.1	2.3	4.6	55.4	167.0
58.42	51.1	0.94	2.4	45.2	146.0
63.50	-	-	-	-	125.6
66.04	36.6	0.34	1.06	32.0	-
68.58	-	-	-	-	97.2
73.66	21.4	0.125	0.58	19.2	68.0
78.74	12.4	0.070	0.36	11.7	41.2
83.82	6.0	0.037	0.21	6.4	22.5

^aEstimated accuracy of data is ± 2 in last significant figure, but no better than 0.7% of values.

^bValues for isotopes, derived from traverses with detectors specified in Table XI.

^cFor isotopic 239, as derived from count rates for plutonium mixture of chamber specified in Table XI with estimated corrections for ^{240}Pu fissions.

threshold isotopes-- ^{234}U and ^{238}U --are presented as semilog plots in Fig. 23. Normal graphs of the ^{239}Pu and ^{235}U fission rate and the ^{10}B reaction rate are shown in Fig. 24.

Fig. 23. ^{238}U and ^{234}U Fission-rate Profiles Axially through Assembly 46B

Fig. 24. ^{235}U Fission, ^{239}Pu Fission, and ^{10}B Reaction-rate Profiles Axially through Assembly 46B

K. Radial Counter Traverses

Radial scans were made with the four fission counters and the BF_3 counter when the traverse channel was aligned through the front of the P row of Half No. 1. Table XIII lists the results of these experiments, again reduced to relative rates for the principal counter isotopes. The

TABLE XIII. Reaction Rates versus Radius in Assembly 46B

Radius, cm	Relative Reaction Rate, ^a Percent of Rate at Core Center					Radius, cm	Relative Reaction Rate, ^a Percent of Rate at Core Center				
	^{235}U Fission ^b	^{238}U Fission ^b	^{234}U Fission ^b	^{239}Pu Fission ^c	^{10}B (n, α) ^b		^{235}U Fission ^b	^{238}U Fission ^b	^{234}U Fission ^b	^{239}Pu Fission ^c	^{10}B (n, α) ^b
0	100.0	100.0	100.0	100.0	100.0	33.02	85.9	4.3	9.7	76.1	235.6
2.54	99.4	99.2	99.4	99.4	98.9	35.56	-	-	-	-	229.9
5.08	98.0	96.7	96.7	97.1	97.5	38.10	76.2	2.0	5.1	67.1	222.0
7.62	94.1	92.4	92.7	93.2	97.0	43.18	63.4	0.88	2.8	57.5	-
10.16	90.6	86.0	86.8	88.7	95.0	45.72	-	-	-	-	181.6
12.70	-	-	-	83.4	95.5	48.26	49.7	0.39	1.6	48.0	-
15.24	81.6	67.8	70.4	78.1	97.5	50.80	-	-	-	-	142.4
16.51	79.8	-	-	76.0	-	53.34	36.5	0.21	1.1	33.8	-
17.78	79.6	52.6	58.2	74.3	115.0	55.88	-	-	-	-	100.3
19.05	81.7	43.4	49.4	74.5	127.0	60.96	21.9	0.09	0.7	19.9	-
20.32	85.2	33.4	42.7	76.0	138.0	66.04	-	-	-	-	54.0
22.86	90.6	21.7	30.9	79.4	180.5	68.58	14.5	0.026	0.2	13.0	-
25.40	93.2	-	-	80.9	206.5	71.12	11.1	0.016	0.09	10.1	39.4
27.94	93.3	9.5	17.1	80.3	223.2	77.47	7.5	0.004	0.03	7.1	-
30.48	90.0	-	-	79.0	233.9						

^aEstimated accuracy of data is ± 2 in last significant figure, but no better than 0.7% of values.

^bValues for isotopes, derived from traverses with detectors specified in Table XI.

^cFor isotopic ^{239}Pu , as derived from count rates for plutonium mixture of chamber specified in Table XI with estimated corrections for ^{240}Pu fissions.

^{238}U and ^{234}U fission rates versus radius are graphed in Fig. 25, and the results for ^{235}U fission, ^{239}Pu fission, and ^{10}B reaction are graphed in Fig. 26.

Fig. 25. ^{238}U and ^{234}U Fission-rate Profiles Radially through Assembly 46B

Fig. 26. ^{235}U Fission, ^{239}Pu Fission, and ^{10}B Reaction-rate Profiles Radially through Assembly 46B

L. Pulsed Neutron Experiments

Experiments on the prompt-neutron die-away characteristics of Assembly 46B were carried out by G. S. Brunson and R. J. Huber of the Idaho Division. The measurements were part of a series of studies with neutron pulsing of assemblies with light-element reflectors, and full details of the studies have been given in reports by the participants.^{12,13} Detectors of various neutron-spectral sensitivity were used at positions in the core and in the reflector to obtain time-response distributions after power bursts induced by a neutron generator. In all cases, the fittings of the die-away curves were best satisfied with two exponential terms plus a background; this was explained by means of a two-region transfer-function model for the reactor, whereby separate lifetimes and feedbacks for the core and the reflector contribute to the dual-decay character. The measurements thus yielded values for a core lifetime and a reflector decay constant, as well as an overall lifetime.

For the pulsed-neutron experiments in Assembly 46B, the generator target was situated at the midplane near the core radial edge (at the front of drawer 1-P-13). The detectors, which were similar to the traverse chambers specified in Table XI, were at various stages placed at the core center (front of 1-P-17) and in the reflector (at the front of 1-P-22). The measurements were made at several subcritical levels, which were established by withdrawal of fuel-loaded control drawers. The range of reactivity extended from 25 to 1210 lh below delayed critical. Figure 27 presents plots of the two exponential decay constants measured as a function of subcriticality. By the appropriate analysis with the two-region model, these decay parameters provided average lifetime values for Assembly 46B as follows:

Core lifetime	$2.65 \pm 0.20 \times 10^{-7}$ sec
Reflector lifetime ($1/\lambda_R$)	$140 \pm 20 \times 10^{-7}$ sec
Overall lifetime	$3.50 \pm 0.20 \times 10^{-7}$ sec

The referenced articles by the experimenters provide detailed explanations of the two-region model and analysis.

Fig. 27. Decay Constants Obtained by Fitting $N(t) = A_1 e^{-\alpha_1 t} + A_2 e^{-\alpha_2 t} +$ Background to Neutron Pulse Die-away Data in Assembly 46B

VI. CORE-C EXPERIMENTS

A. Composition of Core C

Core C in Assembly 46 was the study of a simulated oxide-fuel system as prescribed in the program objectives in Section II.C of this report. The FARET core to be represented was to have been a 20-in.-long hexagonal cylinder, 93 liters in volume. Throughout the assembly, UO_2 - PuO_2 fuel was represented by a combination of plates of plutonium, enriched uranium, depleted U_3O_8 , and graphite. Here the added carbon represented part of the stoichiometric requirements for oxygen. Figure 28 shows the arrangement of these constituents and the sodium in the Core-C fuel drawers. For this loading and for the drawers used as ZPR-3 control rods, the compositions are given in Appendix B. Atomic densities and the volume for the core section of a regular drawer are given in Table XVII (p. 50), and Table XIV contains the average densities in the critical loading, including the control drawers. The effective fuel simulation obtained was a 6.1:1 mixture of UO_2 and PuO_2 , with the uranium enrichment averaging 49% ^{235}U .

B. The Critical Experiment

1. Loading to Critical

The initial loading for the approach to critical with Core C contained 25 fuel drawers per assembly half. During the loading to critical, with successive replacements of reflector by Core-C fuel drawers, the pulsed-neutron generator and various detectors were left in the reactor from the Core-B lifetime experiments. This enabled the convenient continuation of such experiments in the oxide-fuel simulation core, and their

Fig. 28
Fuel-drawer Loading for
Core 46C. ANL Neg. No.
ID-103-2542.

TABLE XIV. Specifications of Core 46C

Material	Atoms/cm ³ x 10 ⁻²²
a. <u>Average Composition</u>	
²³⁹ Pu	0.1062
²⁴⁰ + ²⁴² Pu	0.0051
²⁴¹ Pu	0.0005
²³⁵ U	0.3343
²³⁸ U	0.3432
²³⁴ + ²³⁶ U	0.0048
Na	0.8269
O	0.8582
C	0.5104
Fe	1.1270
Cr	0.3026
Ni	0.1489
b. <u>Critical Loading</u>	
Length	50.95 cm
Average radius	23.3 cm
Critical volume	86.9 ± 0.1 liters
Total fuel:	
Plutonium	38.6 kg
²³⁵ U	113.4 kg

results are given in Section D below. The lifetime equipment, however, distorted the approach to critical. Later, a clean, reference-critical loading was established for Core C, and its configuration is shown in Fig. 29. This loading, with a total of 112 core drawers, had an estimated available excess reactivity of 88 lh, or about 0.17% Δk .

Fig. 29. Critical Configuration of Core 46C

2. Worth of Fuel at Core Edge

A replacement of fuel by reflector in the outer half of the core drawer in 1-O-21 produced a 62.0 ± 0.6 lh loss of reactivity, and a similar substitution in the complementary outer half of 1-Q-13 gave a 66.7 ± 1 lh loss. Together, these results imply a core-edge fuel worth of 129 ± 1 lh/drawer in the locations of the substitutions. On the basis of fuel-reflector exchanges made throughout the Core-C program at other locations, the accuracy of this edge-fuel worth as representing an average around the core periphery is about ± 8 lh. The relation of reactivity to differential volume change becomes

$$\Delta k/k = 0.29 \Delta V/V.$$

3. Critical Volume

Based on the cited core-edge fuel worth, the excess loading from critical for the reference configuration of Core C (Fig. 29) amounts to 0.68 drawer. The just-critical loading would then be 111.32 drawers, or 86.91 ± 0.08 liters, which gives an average core radius of 23.3 cm. No adjustments have been applied here for the reactivity effects of the irregular core boundary or the material heterogeneity. Figure 30 presents the axial schematic for Assembly 46C as represented in equivalent cylindrical form. For the reference critical loading, the average core temperature was about 27°C.

C. Control-rod Calibrations

Figure 31 shows the reactivity calibrations for the two fuel-loaded control drawers and the autorod used for worth measurements in Assembly 46C. Again, the curve shapes were determined in "standard unit" steps, and the absolute calibrations in inhours were found from period measurements. For Core C, the effective delayed-neutron fraction and reactivity conversion factors used were

$$\beta_{\text{eff}} = 0.00587,$$

$$1\$ = 295 \text{ lh},$$

$$1\% k = 503 \text{ lh},$$

and

$$\text{prompt lifetime} = 2.15 \times 10^{-7} \text{ sec.}$$

Details of the calculations for these parameters are included in Appendix C.

Fig. 30. Axial Schematic Diagram of Assembly 46C. ANL Neg. No. ID-103-2519.

Fig. 31
Rod Calibrations in Assembly 46C

D. Pulsed-neutron Experiments

Neutron lifetime measurements were carried out for Core C in the same way as for Core B. Details of these measurements are included in the papers by the experimenters Brunson and Huber.^{12,13} The generator target was again near the front of the drawer 1-P-13, and in this case the neutron detectors used were ²³⁸U fission chambers, plutonium fission chambers, and BF₃ counters stationed at the core center and in the fine radial reflector.

Subcritical levels ranging between 32 and 1020 lh below delayed critical were obtained for these experiments in two different manners for comparison: by withdrawal of fuel-loaded control drawers, and by vacating reflector drawers in an annulus two rows away from the core boundary. No discernable differences in results were obtained by the two methods. The lifetimes for Assembly 46C as derived from analysis with the two-region model are:

Core lifetime	$2.95 \pm 0.15 \times 10^{-7}$ sec;
Reflector lifetime ($1/\lambda_R$)	$150 \pm 20 \times 10^{-7}$ sec;
Overall lifetime	$3.90 \pm 0.20 \times 10^{-7}$ sec.

DRAWERS OF 14-DRAWER
DISTRIBUTED QUADRANT

CONTROL DRAWERS

Fig. 32. Substitution Locations for Material-worth Measurements in Core 46C

E. Distributed Worths of Materials

A series of material substitutions was conducted in a distribution of 14 core drawers to derive core-average reactivity coefficients. The drawers involved, all in Half No. 1 and located as indicated in Fig. 32, effectively represented all the core drawers as regards possible displacements from the axis of the assembly (except for the central drawers). In successive stages, the central columns (for the 10-in. core length) in these 14 drawers were loaded with steel, sodium cans, voided cans, depleted uranium, depleted U₃O₈, and graphite. The worths found for these material samples relative to void, along with the specific worths in lh/kg, are given in Table XV. Accepting these coefficients as core-average values does not introduce significant error from the omission of the central-drawer contributions. This can be argued by considering

the effects found in similar measurements in Core B (Table V) and considering that Core C is larger.

TABLE XV. Distributed Worths of Materials in Core 46C

Material	Total Mass in Distributed Substitution, ^a g	Net Worth of Material, Relative to Void, $1h$	Reactivity Coefficient, $1h/kg$
Sodium	870	48.3 ± 0.8	55.5 ± 0.9
Stainless steel (Type 304)	7,338	75.0 ± 1.0	10.2 ± 0.2
Depleted uranium ^b	10,372	72.9 ± 1.0	7.03 ± 0.10
U_3O_8 ^b	4,130	70.0 ± 1.0	17.4 ± 0.2
Carbon	902	106.2 ± 1.2	118.2 ± 1.3
Oxygen ^c			75 ± 1

^aSubstitutions made in 14 drawers shown in Fig. 32.

^b0.2 wt % ^{235}U in uranium.

^cDerived from uranium and U_3O_8 worths.

F. Regional Worths of Sodium

The reactivity coefficient of sodium in core-length axial columns was measured as a function of radius in Core C. In these experiments, sodium cans were replaced by empty cans in two columns each of drawers in three radial zones, thus halving the sodium density in the zones. Figure 33 shows the locations involved in the substitutions, and Table XVI presents the results. In all three zones, the average sodium reactivity coefficient for the full-core-length columns was surprisingly constant.

Fig. 33. Substitution Drawers for Sodium Regional Worth Experiments in Core 46C

TABLE XVI. Worths of Sodium in Columns at Various Radii in Core 46C

Radial Zone	Zone Radial Extension, cm	Axial Extension of Sodium Columns, cm from Midplane	Mass of Sodium Replaced by Void	Substitution Worth, lh	Sodium Reactivity Coefficient, lh/kg (relative to void)
I	0.0 to 3.84	-25.5 to +25.5	248.5	-14.3 ± 0.3	56.7 ± 1.1
I	0.0 to 3.84	0.0 to 82.0	795	-19.1 ± 0.4	24.0 ± 0.5
II	8.4 to 12.9	-25.5 to +25.5	497	-28.7 ± 0.5	57.7 ± 1.0
III	19.3 to 25.2	-25.5 to +25.5	497	-29.2 ± 0.5	58.8 ± 1.0

An additional experiment was done in Zone I, the central drawers of the assembly, where the length of the substitution columns extended from the midplane out through the axial reflector. All four sodium columns of the front and back drawers of 1-P-17 were replaced by empty cans, and this represented sodium voiding for a half-length of the central subassembly in FARET. From the results, the total coolant loss from a central oxide subassembly in FARET would have been worth -48 lh, or about -0.1% k.

G. Fuel-exchange Experiments

The fuel-exchange experiments in Assembly 46C were designed to obtain the worths of substituting typical carbide-fuel and metallic-alloy fuel subassemblies for regular core subassemblies at selected locations in an oxide-fueled FARET system. Also, the worths substituting all three fuel types for reflector in the next subassembly row around the core were desired. Figure 34 illustrates the loadings of simulated carbide- and alloy-fuel drawers used in the substitution experiments for Core C, and Fig. 35 shows the drawer locations involved.

Fig. 34. Carbide- and Alloy-fuel Drawer Loadings for Fuel-exchange Experiments in Core 46C. ANL Neg. No. ID-103-2520.

Fig. 35. Locations of Fuel-exchange Experiments in Core 46C

30 at. % titanium. However, no titanium was immediately available at ZPR-3, so stainless steel was used as an approximately equivalent substitute. Other inventory restrictions affected the attainable fuel ratios. Thus, the loading of the alloy-fuel drawer shown in Fig. 34b effectively simulated a subassembly with 36 vol % metallic fuel (U-Pu-Ti alloy), with a U-to-Pu ratio of 3.5 and a uranium enrichment of 14% ^{235}U . Its composition is included in Part a of Table XVII, and Part b of the table contains the worths of substitution of this alloy drawer for oxide core at three radial locations.

1. Carbide-for-Oxide Substitutions

The material arrangements shown for the core section in Fig. 34a provided a simulation of a carbide-fuel subassembly with a UC-to-PuC ratio of 5.8 and a uranium enrichment of 50%. Part a of Table XVII contains its composition along with that of the regular Core-C oxide-fuel drawers involved in the experiments. Oxide-fuel drawers at three radial locations in Core C were replaced by this carbide-fuel drawer and the results are given in Part b of Table XVII.

2. Alloy-for-Oxide Substitutions

The desired metallic-fuel simulation was an alloy of 56 at. % uranium, 14 at. % plutonium, and

TABLE XVII. Fuel-exchange Experiments in Core 46C

a. Compositions of Drawers Interchanged

Drawer Loading	Material Densities, atoms/cm ³ x 10 ⁻²²											
	²³⁹ Pu	²⁴⁰⁺²⁴² Pu	²⁴¹ Pu	²³⁵ U	²³⁸ U	²³⁴⁺²³⁶ U	O	C	Na	Fe	Cr	Ni
Oxide fuel	0.1062	0.0051	0.0005	0.3343	0.3432	0.0048	0.8582	0.5042	0.8525	1.0927	0.2933	0.1444
Carbide fuel	0.1481	0.0071	0.0006	0.4451	0.4465	0.0065	-	1.1061	0.8573	1.0631	0.2854	0.1404
Alloy fuel	0.2620	0.0125	0.0011	0.1309	0.8268	0.0019	-	-	0.8688	1.4802	0.4284	0.2109
Reflector	-	-	-	-	-	-	-	-	0.212	4.690	1.259	0.620
									+0.306 Al			

b. Worths of Drawer Exchanges (0.7807-liter fuel sections)

Drawer Location	Substitution	Reactivity Change, Δk
1-P-17	Carbide-for-oxide	132 $\pm$ 1
1-R-17	Carbide-for-oxide	105.3 $\pm$ 0.7
1-T-17	Carbide-for-oxide	56.9 $\pm$ 0.5
1-U-17	Carbide-for-reflector	130 $\pm$ 1
1-S-20	Carbide-for-reflector	177 $\pm$ 1
1-P-17	Alloy-for-oxide	44.4 $\pm$ 0.4
1-R-17	Alloy-for-oxide	33.0 $\pm$ 0.5
1-T-17	Alloy-for-oxide	12.2 $\pm$ 0.4
1-U-17	Alloy-for-reflector	95.9 $\pm$ 0.6
1-U-17	Oxide-for-reflector	92.1 $\pm$ 0.6

3. Fuel-for-Reflector Substitutions

Experimental situations in a FARET-type system might include additions of fuel in the reflector outside of the normal core boundary. For this reason, a few fuel-for-reflector replacements were carried out in Assembly 46C. The reflector locations involved are indicated in Fig. 35. Core drawers of all three fuel types--oxide, carbide, and metal--were substituted for the reflector drawer in 1-U-17, and the carbide-fuel drawer was also substituted in 1-S-20 in place of the reflector. Part b of Table XVII includes the results of these experiments. The reflector drawers replaced were the regular fine-radial-reflector type, and the composition for it (the last line in Part a of Table XVII) extended the full length of the 21-in. drawers. All the fuel drawers had an axial reflector loading from 10 to 21 in. (composition given in Appendix A). The drawer replacements therefore simulated the substitutions (for half-lengths) of core subassemblies for reflector subassemblies in FARET.

H. Mockup B₄C Control Rod Experiments

An extensive series of substitutions was carried out in Core 6C using mockup B₄C control rods to ascertain the effectiveness of the FARET-design rods. The locations of the 12 intended rods for FARET are shown in

Fig. 36. Locations of Mockup Control-rod Channels in Assembly 46C

Fig. 1. Four mockup-rod channels were constructed in the reflector of Assembly 46C at locations simulating those of four neighboring rods in the FARET configuration. The placements of the mockup channels in the reflector near Core C, shown in Fig. 36, involved segments of two or four reflector drawers per channel to effectively represent the rod channels in FARET with regard to cross-sectional area, separation, and distance from the core axis. Thus, the radii to the center of mockup channels No. 1 and 3 were approximately five FARET lattice spacings (29.5 cm), and to channels No. 2 and 4, the radii were about $4\frac{1}{2}$ times a subassembly diagonal (30.6 cm).

1. Loadings of Mockup Rod Channels

During the experiments, the loadings of the four mockup channels consisted of an "active rod" section (of either high or low boron enrichment) and sections of "rod follower." Cross-sectional views of the material

arrangement used for one channel, that in P-11 and -12, are given in Fig. 37. The same material concentrations were used for the other three channels,

only in slightly different patterns to accommodate the situation of the matrix walls between segments of the channels. In each of the two or four matrix units involved for every mockup channel, alterations and/or rearrangements were necessary in the reflector material of the drawer segments exclusive of the channel portion. The reactivity effects of these changes were considered negligible.

Fig. 37. Cross Sections of Loadings of P-11/12 Channel for Mockup B₄C Control-rod Experiments in Assembly 46C. ANL Neg. No. ID-103-2541.

The loadings for the active sections of the mockup rods, shown in Figs. 37A and 37B, were intended to simulate the dispersion of poison, coolant,

and structure constituents in a FARET control rod. Part a of Table XVIII contains the resulting compositions for both the high- and low-enrichment

TABLE XVIII. Specifications of Mockup FARET B₄C Control Rods Used with Cores 46C and 46F

a. Compositions of Channel Sectional Loadings								
Channel Section Loading	Material Densities, atoms/cm ³ × 10 ⁻²²							
	10B	11B	C	Fe	Cr	Ni	Na	Al
Fine radial reflector	-	-	-	4.690	1.259	0.620	0.212	0.306
Coarse radial reflector	-	-	-	5.638	1.514	0.745	-	-
32%-enrichment B ₄ C active rod	1.533	3.216	1.369	1.272	0.341	0.168	0.422	-
69%-enrichment B ₄ C active rod	3.337	1.493	1.665	1.052	0.282	0.138	0.422	-
Rod follower	-	-	-	4.482	1.203	0.592	0.418	-

b. Axial Placements of Channel Sectional Loadings for Various Rod Configurations	
Channel Loading Designation	Sectional Loadings in Channel, Axial Extensions in cm from Reactor Midplane ^a
Radial reflector in channel	Fine radial reflector: 53.4 to -53.4 Coarse radial reflector: 79.5 to 54.1, -54.1 to -79.5
Active rod in "up" position (+10 in.)	Rod follower: 82.0 to 76.9 Active rod: 76.9 to 54.1, 53.4 to 25.5 Rod follower: 25.5 to -53.4, -54.1 to -82.0
Active rod at +4 in.	Rod follower: 82.0 to 61.7 Active rod: 61.7 to 54.1, 53.4 to 10.2 Rod follower: 10.2 to -53.4, -54.1 to -82.0
Active rod at midplane (0 in.)	Rod follower: 82.0 to 54.1, 53.4 to 50.9 Active rod: 50.9 to 0.0 Rod follower: 0.0 to -53.4, -54.1 to -82.0
Active Rod at -4 in.	Rod follower: 82.0 to 54.1, 53.4 to 40.7 Active rod: 40.7 to -10.2 Rod follower: -10.2 to -53.4, -54.1 to -82.0
Active rod in "down" position (-10 in.)	Rod follower: 82.0 to 54.1, 53.4 to 25.5 Active rod: 25.5 to -25.5 Rod follower: -25.5 to -53.4, -54.1 to -82.0

^aIn all cases, spring gaps between 54.1 and 53.4 cm and between -53.4 and -54.1 cm remain the same.

poison sections used. The axial length of the mockup active section was 20 in., to match the core length, and Part b of Table XVIII describes the axial placement of the active section in the various experimental stages. The remaining axial sections of a channel with a mockup rod installed were loaded with "rod follower," as illustrated in Fig. 37C, to represent a typical connector or guide stem above and below the poison section of a FARET rod. Part a of Table XVIII includes the rod-follower composition, along with the composition of reflector sections replaced in the initial steps of the experiments. Together, Parts a and b of the table provide the full details of the replacement experiments, which simulated the introduction and stepwise insertion of the safety rods.

In the discussions to follow, the vertical extensions and movements of rods in FARET have been simulated by corresponding horizontal placements and movements in the axial direction in Assembly 46.

2. Worths of 32%-enriched B₄C Control Rods

The low-enrichment poison section, shown in Fig. 37A, represented a FARET control rod of about 55 vol % B₄C, 18 vol % sodium, and 19 vol % steel, with an average boron enrichment of 32% ¹⁰B. In the initial steps of the experiments, this type of mockup rod was introduced in the "up" position (bottom of active rod at top of core) in place of the reflector in the four mockup channels. Table XIX defines the steps involved and the

TABLE XIX. Worths of Low-enrichment (32%) B₄C Mockup FARET Control Rods in Assembly 46C

Experiment	Mockup Channel(s)	Description of Loading Change within Mockup Channel(s)	Change Worth, Δk
A	No. 1	Radial reflector replaced by mockup with 32% active rods in "up" position	-48.5 $\pm$ 0.5
B	No. 2	Radial reflector replaced by mockup with 32% active rods in "up" position	-41.3 $\pm$ 0.5
C	No. 3 + No. 4	Radial reflector in channels replaced by 32% active rod mockups in "up" position	-102 $\pm$ 1
D	No. 1	32% active rods moved from "up" position (+10 in.) to +4 in.	-70.0 $\pm$ 0.8
E	No. 1	32% active rods moved from +4 in. to midplane (0 in.)	-65.2 $\pm$ 0.7
F	No. 1	32% active rods moved from midplane (0 in.) to -4 in.	-59.7 $\pm$ 0.7
G	No. 1	32% active rods moved from -4 in. to "down" position (-10 in.)	-45.7 $\pm$ 0.7
H	No. 4	32% active rods moved from "up" position (+10 in.) to midplane (0 in.)	-132 $\pm$ 2
I	No. 4	32% active rods moved from midplane (0 in.) to "down" position (-10 in.)	-105 $\pm$ 2
J	No. 2 + No. 3	32% active rods moved from "up" position (+10 in.) to "down" position (-10 in.)	-374 $\pm$ 6

results. On the basis of the worths given by Steps A, B, and C, the inclusion of 12 low-enrichment control rods in FARET, in the "up" position, would have required 2.25 extra fuel subassemblies for criticality.

Next, the active section of mockup rod No. 1, that is the P-11/P-12 channel, was moved inward in steps to 30, 50, 70, and 100% of full insertion, to the "down" position (bottom of active rod at bottom of core). The worths of these simulated rod motions, all measured at criticality with ZPR-3 control rods, are listed under Steps D through G in Table XIX. From the total worths of these steps, a solitary FARET rod of similar composition and situation would have provided 241 lh, or about 0.48% k, of reactivity control.

The mockup rod in channel No. 4, situated in drawers J-17 and K-17, was then moved inward in two steps, first to the midplane and then to full insertion. Its total worth from "up" to "down" was -237 lh, the same worth within experimental error as that for rod No. 1. These steps, H and I in Table XIX, were also measured at criticality.

In the last experiment with the low-worth rods, Step J, the active sections in both the central mockup channels (No. 2 and 3 in Fig. 36) were moved together from the "up" position to the "down" position. The reactivity loss, determined by subcritical methods, was -374 lh. Compared with the -478-lh total worth for rods No. 1 and 4, a shadowing effect of about 22% is observed for the middle rods. Thus, for 12 similar, low-enrichment control rods in FARET, the total available reactivity control would have been about 2556 lh, or 5% $\Delta k/k$.

3. Worths of 69%-enriched B₄C Control Rods

Table XX presents the steps involved and the results of the experiments with the high-worth rods, with the active sections loaded as in Fig. 37B. The initial experiment was the substitution of the 69%-enrichment composition for the 32%-enrichment composition in the poison sections of the four rods as left in the "down" position at the end of Experiment J of Table XIX. A reactivity loss of 511 lh, measured subcritically, was thereby incurred. In the subsequent steps, the rod active sections were then moved "upward," the reverse of the steps with the low-enrichment rods. During most of these experiments, the reactor was subcritical and the worths of the steps were measured by subcritical techniques, with consequently higher probable errors.

Steps E through H in Table XX give the worth of a solitary rod of high enrichment as 379 lh for the No. 1 mockup channel. With that No. 1 rod full in, the rod in the No. 4 channel was worth 357 lh (Steps C plus D). The middle rods, No. 2 and 3, were together worth 556 lh with rods No. 1 and 4 fully inserted. Thus, with this high-enrichment composition, the

shadowing effect was only slightly higher than for the low-enrichment composition--a worth reduction of about 24%. The total worth for complete insertion of all four rods, -1292 lh, extrapolates to 3876 lh, or 7.7% $\Delta k/k$, of reactivity control available in 12 FARET rods of the same composition. The replacement of reflector in FARET by 12 high-enrichment rods in the "up" position would have required 3.3 extra fuel subassemblies for criticality.

TABLE XX. Worths of High-enrichment (69%) B₄C Mockup FARET Control Rods in Assembly 46C

Experiment	Mockup Channel(s)	Description of Loading Change within Mockup Channel(s)	Change Worth, lh
A	All 4	Replaced 32% active rods in "down" position (-10 in.) with 69% active rods in "down" position	-511 ± 10
B	No. 2 + No. 3	Moved 69% active rods from "down" position (-10 in.) to "up" position (+10 in.)	+556 ± 10
C	No. 4	Moved 69% active rod from "down" position (-10 in.) to midplane (0 in.)	+159 ± 3
D	No. 4	Moved 69% active rod from midplane (0 in.) to "up" position (+10 in.)	+198 ± 3
E	No. 1	Moved 69% active rod from "down" position (-10 in.) to -4 in.	+71 ± 2
F	No. 1	Moved 69% active rod from -4 in. to midplane (0 in.)	+100 ± 2
G	No. 1	Moved 69% active rod from midplane (0 in.) to +4 in.	+103 ± 2
H	No. 1	Moved 69% active rod from +4 in. to "up" position (+10 in.)	+108 ± 2

I. Axial Traverses of Reactivity Samples

Samples of enriched uranium, natural uranium, plutonium, and enriched boron were traversed axially through Assembly 46C in the step-wise manner cited for radial sample traverses in Core B. These samples were again about 3/8 in. in diameter, with the specifications listed in Table VIII. Figure 38a is a cross-sectional view of the core sections of the central drawers (1- and 2-P-17) as loaded for the experiment, showing the arrangement of materials around the 1/2-in.-diam traverse channel and the proximity of the fuel. This loading pattern was an attempt at presenting to the traverse channel an average "view" of material that was representative of the homogenized core composition. The loading pattern for the axial reflector sections in the matrix unit, 11 in. in the front drawers plus 11 in. in the backup drawers, is shown in Fig. 17a.

a. CROSS SECTION OF P-17 CORE SECTION FOR AXIAL TRAVERSE

b. SIDE VIEW OF FRONT OF P-ROW DRAWERS FOR RADIAL TRAVERSES

Fig. 38. Drawer Loadings for Assembly 46C Traverse Experiments

Table XXI lists the reactivity coefficients for the uranium isotopes ^{235}U and ^{238}U , as derived from the enriched-uranium and natural-uranium sample worths at the various axial positions. In deriving the coefficients for isotopic ^{10}B , we assumed the worths of the ^{11}B and impurity contents in the boron sample to be negligible. The coefficients for plutonium are for the isotopic mixture given in the sample specifications. In Table XXI, the results given at every axial position down through 38.1 cm are averages of values found on both sides of the assembly midplane. The probable errors that are given reflect principally the uncertainty in corrections for temperature-drift effects. The results listed in the table are also presented in graphic form in Figs. 39 and 40.

TABLE XXI. Uranium, Plutonium, and Boron Worths versus Axial Position in Assembly 46C

Axial Position, cm from Midplane	Reactivity Coefficient, Ih/kg			
	$^{235}\text{U}^{(a)} (\pm 3)$	$^{238}\text{U}^{(a)} (\pm 0.3)$	Plutonium ^(b) (± 5)	$^{10}\text{B}^{(c)} (\pm 30)$
0	308	4.74	511	-5198
2.54	306	4.98	509	-5175
7.62	288	5.42	480	-4874
12.70	249	6.93	-	-4285
13.97	-	-	399	-
17.78	200	8.68	331	-3426
20.32	175	9.40	277	-3019
22.86	148	9.92	228	-2621
25.40	120	8.20	176	-2317
27.94	95	4.79	133	-1996
30.48	76	2.58	108	-1690
33.02	66	1.32	87	-1353
38.10	50	0.43	60	-817
50.80	21	0.08	22	-121
63.50	2.3	0.06	7.5	-14

^aValues for isotope, derived from worths of enriched and natural-uranium samples specified in Table VIII.

^bFor plutonium mixture of sample specified in Table VIII.

^cFor isotope, derived from worth of enriched-boron sample specified in Table VIII assuming negligible worth for ^{11}B and impurities.

Fig. 39. Axial Profiles of Plutonium and ^{235}U Worths in Assembly 46C

Fig. 40. Axial Profiles of ^{238}U and ^{10}B Worths in Assembly 46C

J. Radial Traverses of Reactivity Samples

The four samples specified in Table VIII were also traversed radially through Assembly 46C. For these experiments, the P-row drawers in the core (P-13 through P-21) were rearranged according to the pattern in Fig. 38b to create the traverse channel near the midplane. The P-row drawers in the reflector on both sides of the core were loaded as shown on the right in Fig. 17b. In each traverse, the average autorod position was obtained for a number of radial sample positions. Sample worths versus radius were thereby obtained, after contributing worths of the pushrods and sample claddings were accounted for.

The reactivity coefficients derived from these radial traverses are given in Table XXII. The traverse setup allowed sample positioning between a radius of 73.66 cm on one side of the core (to drawer 1-P-4) and 10.16 cm on the other (to drawer 1-P-19). Therefore, the results given in the table for radii through 10.16 cm are averages for both sides of center. Figure 41 is a plot of the ^{235}U and plutonium coefficients obtained as a function of radius, and the results for ^{10}B and ^{238}U are graphed in Fig. 42.

K. Axial Counter Traverses

While the traverse channel was in place for the axial reactivity scans, with the central drawers 1- and 2-P-17 loaded as shown in Fig. 38a, traverses were also run to obtain reaction-rate profiles. The neutron detectors used, with their specifications as given in Table XI, were an enriched-uranium fission

TABLE XXII. Uranium, Plutonium, and Boron Worths
versus Radius near Midplane in Assembly 46C

Radius, cm	Reactivity Coefficients, lh/kg			
	^{235}U (a)	^{238}U (a)	Plutonium(b)	^{10}B (c)
0	300	3.61	503	-5086
2.54	293	3.77	493	-5038
5.08	287	4.11	480	-4898
7.62	272	5.14	454	-4645
10.16	253	6.17	422	-4351
12.70	231	7.35	383	-3960
15.24	207	8.01	340	-3551
17.78	178	9.57	291	-3116
20.32	148	9.30	241	-2705
25.40	96	5.69	141	-2040
30.48	70	2.04	86	-1204
35.56	50	0.86	57	-569
40.64	33	0.37	37	-230
45.72	22	0.30	23	-90
50.80	14	0.13	13	-33
58.42	4.9	0.04	4.4	-12
66.04	1.1	0.04	2.6	-4
73.66	0.2	0.01	0.4	-2

^aValues for isotope, derived from worths of enriched and natural-uranium samples specified in Table VIII.

^bFor plutonium mixture of sample specified in Table VIII.

^cFor isotope, derived from worth of enriched-boron sample specified in Table VIII, assuming negligible worth for ^{11}B and impurities.

Fig. 41. Radial Profiles of Plutonium and ^{235}U Worths in Assembly 46C

Fig. 42. Radial Profiles of ^{238}U and ^{10}B Worths in Assembly 46C

counter (No. H), a depleted-uranium fission counter (No. IC24-4), a plutonium fission counter (No. S), and a BF_3 proportional counter (No. F2199). The data obtained, count rates versus axial position, were subsequently reduced to fission or reaction rates relative to core-center rates for the principal counter isotopes. Estimates of $^{240}\text{Pu}/^{239}\text{Pu}$ fission ratios versus position were used to correct the data from the plutonium counter.

In Table XXIII, the results given through axial position 38.1 cm are averages of relative rates for corresponding positions on both sides of the assembly midplane. Figure 43 presents the axial profiles for ^{235}U fission, plutonium fission, and ^{10}B reaction rates, and Fig. 44 is a semilog plot of the results for ^{238}U fission.

TABLE XXIII. Reaction Rates versus Axial Position in Assembly 46C

Axial Position, cm	Relative Reaction Rates, ^a Percent of Rate at Core Center			
	^{235}U Fission ^b	^{238}U Fission ^b	Plutonium Fission ^c	$^{10}\text{B}(n,\alpha)^b$
0	100.0	100.0	100.0	100.0
2.54	100.2	99.2	100.1	99.9
5.08	99.2	98.4	99.1	-
7.62	96.8	96.8	97.2	97.2
10.16	94.4	94.1	94.4	-
12.70	91.3	89.9	91.2	90.5
15.24	87.4	84.2	86.1	86.8
17.78	82.5	77.7	81.4	83.4
20.32	78.9	69.8	75.8	80.4
22.86	74.7	59.9	71.7	79.8
25.40	75.0	45.0	69.6	85.8
27.94	80.2	30.0	71.6	96.9
30.48	83.6	20.9	73.5	108.0
33.02	85.4	15.0	75.0	116.4
35.56	87.0	10.8	75.7	123.0
38.10	86.2	7.6	75.1	126.4
40.64	84.8	5.4	74.0	128.5
43.18	82.8	3.96	71.7	128.9
45.72	79.7	2.91	68.6	126.2
50.80	70.0	1.49	61.5	-
53.34	64.4	1.13	57.1	107.5
58.42	53.6	0.66	48.8	-
60.96	49.7	0.50	44.3	84.8
66.04	39.2	0.28	35.3	-
68.58	33.9	0.21	30.3	59.2
73.66	23.2	0.115	21.1	41.5
78.74	13.3	0.066	12.4	25.5

^aEstimated accuracy of data is ± 2 in last significant figure, but no better than 0.7% of values.

^bValues for isotopes, derived from traverses with detectors specified in Table XI.

^cFor isotopic 239, as derived from count rates for plutonium mixture of chamber specified in Table XI with estimated corrections for ^{240}Pu fissions.

Fig. 43. ^{235}U Fission, ^{239}Pu Fission, and ^{10}B Reaction-rate Profiles Axially through Assembly 46C

Fig. 44. ^{238}U Fission-rate Profile Axially through Assembly 46C

L. Radial Counter Traverse

Traverses with neutron detectors were accomplished radially through Assembly 46C when the reactor loading was as stipulated in Section VI.J for radial worth scans. Count rates versus radius were obtained with the same four counters used in the axial direction (Section VI.K). The relative reaction rates, as a percentage of core-center rates, versus radius, for the principal counter isotopes are listed in Table XXIV. In the table, the values for radii through 10.16 cm are averages of rates on both sides of core center. The results given in the tables are also presented graphically in Figs. 45 and 46.

M. Spectral Indices at Core Center

The $^{238}\text{U}/^{235}\text{U}$ fission ratio was measured at the center of Core C with enriched-uranium and depleted-uranium fissile foils in an absolute fission chamber at the front of drawer 1-P-17. As in similar experiments in Core B, described in Section V.D, the foils were situated in the middle of a 2-in. cubical void during successive runs. The relative ratio of count rates between the foils gave a core-center spectral index of

$$\sigma_f(^{238}\text{U})/\sigma_f(^{235}\text{U}) = 0.068 \pm 0.002.$$

Here, the uncertainty given reflects counting statistics, probable calibration errors, and the possible error arising from minor, calculated corrections of the count ratio for the ^{234}U and ^{236}U content of the foils. It is interesting to find that this spectral index in Core C is the same as that measured in Core B.

TABLE XXIV. Reaction Rates versus Radius in Assembly 46C

Radius, cm	Relative Reaction Rates, ^a Percent of Rate at Core Center			
	²³⁵ U Fission ^b (±0.8)	²³⁸ U Fission ^b (±1.0)	Plutonium Fission ^c (±0.8)	¹⁰ B(n,α) ^b (±0.7)
0	100.0	100.0	100.0	100.0
1.27	99.5	98.4	99.8	-
2.54	98.5	96.1	98.8	100.3
5.08	97.9	97.5	96.9	98.7
7.62	95.1	91.7	94.5	96.6
10.16	91.9	88.6	91.2	95.5
12.70	88.2	82.7	86.5	92.4
15.24	84.1	76.5	81.6	-
17.78	79.4	69.2	76.8	90.2
20.32	75.1	58.5	72.1	-
21.59	74.8	51.8	69.9	-
22.86	75.1	45.0	68.9	104.9
24.13	77.1	38.2	69.5	-
25.40	80.4	31.6	71.1	131.0
26.67	83.0	25.9	72.6	-
27.94	85.3	20.7	73.8	162.2
30.48	86.4	12.3	75.4	183.1
33.02	85.8	7.9	74.7	-
35.56	82.1	5.4	72.6	195.5
40.64	72.6	2.47	65.0	195.1
45.72	60.7	1.15	55.4	174.8
50.80	47.0	0.50	43.0	139.1
58.42	26.9	0.18	24.4	79.7
66.04	18.7	0.075	17.1	54.4
73.66	10.8	0.036	10.2	32.8

^aEstimated accuracy of data is ±2 in last significant figure, but no better than 0.7% of values.

^bValues for isotopes, derived from traverses with detectors specified in Table XI.

^cFor isotopic ²³⁹Pu, as derived from count rates for plutonium mixture of chamber specified in Table XI with estimated corrections for ²⁴⁰Pu fissions.

Fig. 45. ²³⁵U Fission, ²³⁹Pu Fission, and ¹⁰B Reaction-rate Profiles Radially through Assembly 46C

Fig. 46. ²³⁸U Fission-rate Profile Radially through Assembly 46C

VII. CORE-D EXPERIMENTS

A. Composition of Core D

Core D in Assembly 46 was the first attempt at mocking up the multifuel-zoned loading pattern of FARET shown in Fig. 1. The fuel specifications for this conceptual FARET system and the design objectives for the mockup were discussed in Section II.D. In the core drawer loadings for Core D, the fuel-simulation achieved was as close as practical to that specified for the oxide-, carbide-, and alloy-type FARET fuel sub-assemblies. Obtaining a proper geometric simulation of the hexagonal FARET configurations involved staggered, half-drawer loadings, as will be shown for Cores D, E, and F. Each FARET core subassembly was represented in the multizone mockups with a full-drawer width of core, contained either in single drawer or in halves of adjacent drawers, for a 10-in. axial length in both halves of the assembly.

1. Oxide-fuel Simulation

The multizoned FARET loading called for oxide elements in the center, second, and fourth rings of subassemblies, and also in half of the fifth ring. Figure 47a shows the material arrangement for the basic oxide-fuel simulation drawer used in Core 46D. This loading was similar to that used as the regular Core-C fuel drawer, differing only in the uranium enrichment. Appendix B includes the material densities for the regular oxide-fuel drawers and the oxide-fuel control drawers used in Core D. The disposition of these drawers according to radial zone is shown in Fig. 48, the final critical configuration attained for Core D. On an idealized basis, the core was essentially composed of four concentric annular fuel zones, and an axial representation of this is shown in Fig. 49. Table XXV gives the average compositions of the Core-D zones. The individual oxide-fuel drawers (other than control drawers) in Zones I, III, and IV had the composition cited for Zone I in Table XXV. The oxide-fuel simulation had a U-to-Pu ratio of 6.1:1, and the uranium combination averaged out to 40.8% ^{235}U .

2. Carbide-fuel Simulation

The drawers and drawer sections of Core Zone II shown in Fig. 48 contained a carbide-fuel simulation to represent 12 carbide sub-assemblies in the third ring of the FARET loading diagrammed in Fig. 1. Figure 47b illustrates the arrangement of materials used for the carbide-fuel drawers in Core D, and the resulting composition is included in Appendix B. The Zone II atom densities, given in Table XXV, are the same as for each drawer, since there were no carbide-fuel control rods. The fuel simulation attained was a 5.5:1 mixture of UC and PuC with an average uranium enrichment of 49.7% ^{235}U . Table XXV includes the radial and axial extensions of the carbide-fuel zone represented as a smooth cylindrical annulus with an equivalent volume.

FRONT VIEWS

TOP VIEWS

a. OXIDE FUEL DRAWER

b. CARBIDE FUEL DRAWER

c. ALLOY - FUEL DRAWER

Fig. 47. Fuel-drawer Loadings for Core 46D

Fig. 48. Critical Configuration of Core 46D

Fig. 49
Axial Schematic Diagram
for Cores D, E, and F of
Assembly 46

TABLE XXV. Specifications of Core-46D Fuel Zones

	Zone I (Oxide)	Zone II (Carbide)	Zone III (Oxide)	Zone IV (Oxide-Alloy Mixture)
<u>Critical Geometry</u>				
Length, cm	50.95	50.95	50.95	50.95
Average outer radius, cm ^a	8.26	13.61	19.00	22.33
Volume, liters	10.93	18.74	28.11	22.06
<u>Average Composition, atoms/cm³ x 10⁻²²</u>				
²³⁹ Pu	0.1062	0.1558	0.1062	0.1653
²⁴⁰ + ²⁴² Pu	0.0051	0.0074	0.0051	0.0079
²⁴¹ Pu	0.0005	0.0007	0.0005	0.0007
²³⁵ U	0.2781	0.4475	0.2782	0.2152
²³⁸ U	0.3999	0.4464	0.3999	0.5696
²³⁴ + ²³⁶ U	0.0040	0.0065	0.0040	0.0031
Na ^b	0.8525	0.8577	0.8295	0.8441
O	0.8582		0.8582	0.5326
C	0.5042	1.1061	0.5042	0.3129
Fe	1.0927	1.0593	1.1608	1.1824
Cr	0.2933	0.2844	0.3116	0.3174
Ni	0.1444	0.1399	0.1534	0.1562
Zr				0.1056
<u>Total Fuel</u>				
kg plutonium	4.85	12.20	12.48	15.24
kg ²³⁵ U	11.87	32.74	30.52	18.53

^aAverage radii based on cross-sectional area.

^bIncludes aluminum in plutonium plates.

3. Alloy-fuel Simulation

In the postulated FARET zoned core, the fifth ring of subassemblies would contain single or paired titanium-alloy fuel subassemblies alternating with one or two oxide-fuel subassemblies. For the mockup, however, a simple one-for-one alteration was selected for convenience. Also, no titanium plates were available at ZPR-3, so a simulation of a zirconium-fuel alloy was adopted instead. Figure 47c shows the loading pattern used for the alloy-fuel drawers in Core D, and the resulting composition is given in Appendix B along with that for alloy-fuel control drawers. The fuel simulation achieved in the alloy-fuel drawers was a metallic mixture of about 62 at. % uranium, 18 at. % plutonium, and 20% zirconium, and for Core D the average enrichment of the uranium content was 11.7% ^{235}U .

The alloy-fuel drawers were added to the core in their appropriate Zone IV positions during the approach-to-critical procedure along with oxide-fuel drawers. Figure 48 indicates the eventual placements of alloy-fuel drawers in Core D. The dimensions and average compositions given for core Zone IV in Table XXV are for the mixture of oxide- and alloy-fuel drawers contained in the critical loading. Atomic densities for the individual Core-D alloy drawers can be found in Table XXVIII (p. 73). (The same composition was used for Core-F alloy-fuel drawers.)

B. The Critical Experiment

The initial loading for Core D contained all the required core drawers for the first three fuel zones plus five oxide-fuel drawers and one alloy-fuel drawer in the fourth zone. Combinations of alloy- and oxide-fuel drawers were then added in steps until criticality was attained with the loading shown in Fig. 48. This configuration had a total of 103 core drawers, but on the basis of the approach curves it was supercritical by about two-thirds of a drawer with all control rods inserted. The Fig. 48 loading therefore had a k value of roughly 1.002. No attempts were made to obtain edge-fuel worths or to correct the irregular, oblong shape of the loading, because this Core-D loading would not serve as the desired mockup of the multi-zoned FARET core. The fourth zone essentially represented 14.5 FARET subassemblies, whereas 24 were required.

In Fig. 49, the four fuel zones and the assembly reflector regions are represented as regular, concentric, annular cylinders. Core Zone I was composed of seven oxide elements, Zone II of 12 carbide elements, and Zone III of 18 oxide elements per assembly half. The volumes and the average radii (on an area basis) for these first three zones are given in Table XXV along with their average atomic densities. For Core Zone IV, the outer radius was calculated on the basis of a just-critical core loading of 102.33 drawers. The composition listed in Table XXV for Core Zone IV is the average obtained from the total combination of 18 oxide elements and

11 alloy elements indicated for the zone in Fig. 48, the reference critical configuration for Core 46D. The probable error in the Zone-IV volume for criticality is ± 0.5 liter.

Fig. 50. Radial Profiles of ^{235}U Fission Rate and Power Density in Assembly 46D

are marked off on the figure to indicate the effects of the fuel concentrations on the profiles for flux and power.

C. Foil Irradiations

A set of eight enriched-uranium foils were distributed radially in the Core-46D fuel zones and in the reflector to obtain an approximate flux profile. In reference to Fig. 48, a foil was centered at the front of each of the column-17 drawers designated K through P in Half No. 1 of the reactor, and on top, near the front, of drawers O-17 and N-17 in Half No. 2. The relative fission-product activities in these foils were then determined after irradiation. Figure 50 presents the results of this experiment as the relative fission rates of ^{235}U , normalized to 100% at the core center, versus radius. Also illustrated on the graph are the approximate power densities calculated from the irradiation data. The fuel-zone boundaries

VIII. CORE-E EXPERIMENTS

A. Composition of Core E.

Core 46E was the next attempt to achieve the mockup of the 61-subassembly, multizoned FARET core scheme shown in Fig. 1. The transition from the Core-D fuel-drawer simulations to those used with Core E involved only the uranium plates in the drawers; the total uranium content was kept the same, but the average enrichments were reduced to obtain less reactive fuel. This was done for all three fuel simulations--the oxide, the carbide, and the alloy--to an extent that later proved to be excessive. Compositions for all the fuel-drawer loadings used with Core E, including those used as ZPR-3 control rods, are listed in Appendix B.

1. Oxide-fuel Simulation

Figure 51a shows the loading pattern used for the Core-E oxide-fuel simulation. Atomic densities for a drawer loaded accordingly

Fig. 51. Fuel-drawer Loadings for Core 46E

are as listed for Zone I of the Core-E specifications given in Table XXVI. The Zone-III densities given include the extra steel concentration due to oxide-fuel control drawers. In the transition from the Core-D oxide simulation, replacing enriched uranium with depleted uranium in one-fourth column per drawer, the average uranium enrichment dropped from 40.8 to 32.7% ^{235}U in Core E.

TABLE XXVI. Specifications of Core-46E Fuel Zones

	Zone I (Oxide)	Zone II (Carbide)	Zone III (Oxide)	Zone IV (Oxide-Alloy Mixture)
<u>Critical Geometry</u>				
Length, cm	50.95	50.95	50.95	50.95
Average outer radius, cm ^a	8.26	13.61	19.00	25.94 ^b
Volume, liters	10.93	18.74	28.11	49.97 ^b
<u>Average Composition, atoms/cm³ x 10⁻²²</u>				
^{239}Pu	0.1062	0.1558	0.1062	0.1841
$^{240}+^{242}\text{Pu}$	0.0051	0.0074	0.0051	0.0088
^{241}Pu	0.0005	0.0007	0.0005	0.0008
^{235}U	0.2243	0.3348	0.2243	0.1400
^{238}U	0.4575	0.5606	0.4575	0.6808
$^{234}+^{236}\text{U}$	0.0032	0.0048	0.0032	0.0020
Na ^c	0.8525	0.8577	0.8295	0.8606
O	0.8582	-	0.8582	0.4291
C	0.5042	1.1061	0.5042	0.2522
Fe	1.0927	1.0593	1.1602	1.1172
Cr	0.2933	0.2844	0.3115	0.2999
Ni	0.1444	0.1399	0.1533	0.1477
Zr	-	-	-	0.1548
<u>Total Fuel</u>				
kg plutonium	4.85	12.20	12.48	38.45 ^b
kg ^{235}U	9.57	24.49	24.61	27.31 ^b

^aRadii determined on area basis.

^bBased on extrapolated critical loading.

^cIncludes aluminum in plutonium plates.

2. Carbide-fuel Simulation

The loading pattern of the carbide-fuel elements for Zone II of Core E is illustrated in Fig. 51b. This pattern gave the composition for the carbide drawers as specified under Zone II in Table XXVI. The fuel simulation attained was a 5.5:1 mixture of UC and PuC, with an average uranium enrichment of 37.2%.

3. Alloy-fuel Simulation

Figure 51c is the loading pattern used for the alloy-fuel drawers in Core E. Drawers and drawer sections based on this pattern were alternated with oxide-fuel drawers in the construction of the fourth radial core zone during the critical experiment. The metallic mixture represented in this alloy simulation was about 62 at. % uranium, 18 at. % plutonium, and 20 at. % zirconium, as in Core D, but with the uranium enrichment reduced to 5.7% ^{235}U . For the individual alloy-fuel drawers in Zone IV of Core E (excluding the alloy control drawers) the composition in atoms/cc was the same as that of the Core-F alloy drawer, listed in Table XXVIII (p. 73), with these exceptions: ^{235}U --0.0558; ^{238}U --0.9040; and $^{234}+^{236}\text{U}$ --0.0008 (10^{22} atoms/cc units).

B. The Critical Experiment

1. Approach to Critical

For the initial loading for the multizoned Core E, all of the first three radial zones were fully loaded with their designated fuel. In the fourth zone, which was to mock up the fifth ring of FARET subassemblies containing 12 oxide and 12 alloy subassemblies, the initial configuration contained four alloy and six oxide drawers per assembly half. In subsequent loading stages, combinations of oxide plus alloy-fuel drawers were added to the fourth zone. The approach to critical was discontinued when the fourth zone contained not only the required 12 oxide and 12 alloy drawers per assembly half but one each additional oxide and alloy drawer. At this time, the extrapolations of the approach curves indicated that an additional 14 fuel drawers were needed for criticality. Figure 52 presents these curves, expressing reciprocal power and mass times reciprocal power versus the total fuel loading of the core (kg of ^{235}U plus plutonium). No further experiments were then done with Core E since it was obvious that the compositions selected were underreactive.

Fig. 52. Approach-to-Critical Curves for Core 46E

2. Extrapolated Critical Loading

In order to extract as much valuable information from the Core-E work as possible, a probable critical Core-E configuration was contrived on the basis of the approach-to-critical data. This hypothetical loading is shown in Fig. 53. Half drawers of oxide and alloy fuel were included around Core Zone IV to provide the total of 14 extra drawers, which were

estimated as necessary for criticality. Thus, Zone IV for a critical Core E would most likely have contained a total of 16 oxide and 16 alloy drawers per assembly, as Fig. 53 shows. The composition given in Table XXVI for Zone IV is therefore the average of the oxide- and alloy-fuel simulations, with the control drawers excluded for simplicity from the postulated critical loading. The dimensions for Zone IV in the table are based on a total of 64 drawers. All the essential geometric details, on a simplified cylindrical model, for the core and reflector regions of Assembly 46E are given in Fig. 49 and Table XXVI.

Fig. 53. Probable Critical Configuration of Core 46E

As a mockup of a possible loading for FARET, the configuration shown in Fig. 53 represents a multifuel-zoned core with a total of 69 fuel subassemblies. Achievement of the desired mockup of a 61-element core required increasing the enrichments in the fuel simulations partially back to the values used with Core D.

IX. CORE-F EXPERIMENTS

A. Composition of Core F

A close geometric mockup of the 61-subassembly, multifuel-zoned FARET system shown in Fig. 1 was finally attained with Core F in Assembly 46 by increasing the uranium enrichments of the carbide- and alloy-fuel simulations from the values in Core E. The oxide-fuel simulation used with Core E was retained in Core F. The compositions for all the Core-F fuel drawers, including control rods, are listed in Appendix B.

1. Oxide-fuel Simulation

The Core-F oxide-fuel drawers were loaded as shown in Fig. 54A (the same as in Core E), and this simulated a UO_2 - PuO_2 fuel with

A. OXIDE - FUEL DRAWER

B. CARBIDE - FUEL DRAWER

C. ALLOY - FUEL DRAWER

Fig. 54. Fuel-drawer Loadings for Core 46F. ANL Neg. No. ID-103-2537.

a U-to-Pu ratio of 6.1:1 and the uranium 32.7% enriched. Figure 55 is a diagram of the final four-zone mockup core, showing the placements of oxide drawers or drawer units in three of the core zones. Oxide elements filled the seven-drawer-unit central zone, the 18-drawer-unit third zone, and 12 alternate drawer units of the fourth zone in both halves of the assembly. The composition for the regular Core-F oxide-fuel drawer is the same as that given in Table XXVII for Core Zone I, which contained no control drawers.

Fig. 55
Critical Configuration
of Core 46F

TABLE XXVII. Specifications of Core-46F Fuel Zones

	Zone I (Oxide)	Zone II (Carbide)	Zone III (Oxide)	Zone IV (Oxide-Alloy Mixture)
Critical Geometry				
Length, cm	50.95	50.95	50.95	50.95
Average outer radius, cm	8.26	13.61	19.00	24.32
Volume, liters	10.93	18.74	28.11	36.89
Average Composition, atoms/cm³ x 10⁻²²				
²³⁹ Pu	0.1062	0.1558	0.1062	0.1841
²⁴⁰ Pu	0.0051	0.0074	0.0051	0.0088
²⁴¹ + ²⁴² Pu	0.0005	0.0007	0.0005	0.0008
²³⁵ U	0.2243	0.4103	0.2243	0.1682
²³⁸ U	0.4575	0.4819	0.4575	0.6206
²³⁴ + ²³⁶ U	0.0032	0.0059	0.0032	0.0024
Na	0.8525	0.8577	0.8295	0.8519
O	0.8582	-	0.8582	0.4291
C	0.5042	1.1061	0.5042	0.2522
Fe	1.0927	1.0593	1.1602	1.1594
Cr	0.2933	0.2844	0.3115	0.3113
Ni	0.1444	0.1399	0.1533	0.1532
Zr	-	-	-	0.1497
Total Fuel				
kg plutonium	4.85	12.20	12.48	28.38
kg ²³⁵ U	9.57	30.02	24.61	24.22

2. Carbide-fuel Simulation

Depleted uranium was replaced by enriched uranium for a half-height column in each of the 12 drawers per half that comprised Zone II of Core 46E. The new loading pattern, shown in Fig. 54B, provided the more reactive Core-F carbide-fuel simulation with the same 5.5:1 mixture of UC and PuC but with the average uranium enrichment raised to 45.7% ^{235}U . Average atomic densities for the drawer loading were the same as those listed for the carbide zone in Table XXVII.

3. Alloy Fuel Simulation

Increasing the uranium enrichment of the alloy-fuel drawers from the value used in Core E required returning to the alloy simulation used with Core D. Thus, the Core-46F alloy-fuel drawers, loaded as in Fig. 54C, again represented a metallic mixture of 62 at. % uranium, 18 at. % plutonium, and 20 at. % zirconium, with an average uranium enrichment of 11.7%. Figure 55 indicates the placements of these alloy drawers in Zone IV of the reference Core-F critical configuration. The average composition for the alloy drawer loading is given in Table XXVIII. (The Table XXVII composition for Zone IV is for the oxide-alloy mixture.)

TABLE XXVIII. Fuel-exchange Experiments in Fuel Zones of Core 46F

a. Compositions of Drawers Interchanged													
Drawer Loading	Material Densities, atoms/cm ³ x 10 ⁻²²												
	²³⁹ Pu	²⁴⁰ + ²⁴² Pu	²⁴¹ Pu	²³⁵ U	²³⁸ U	²³⁴ + ²³⁶ U	O	C	Na	Fe	Cr	Ni	Other
Oxide-fuel	0.1062	0.0051	0.0005	0.2243	0.4575	0.0032	0.8582	0.5042	0.8525	1.0927	0.2933	0.1444	-
Carbide-fuel	0.1557	0.0074	0.0007	0.4103	0.4819	0.0059	-	1.1061	0.8577	1.0593	0.2844	0.1400	-
Alloy-fuel	0.2620	0.0124	0.0012	0.1121	0.8474	0.0016	-	-	0.8688	1.1417	0.3065	0.1509	0.3170 Zr
Reflector	-	-	-	-	-	-	-	-	0.212+	4.690	1.259	0.620	0.306 Al

b. Results of Substitution Experiments		
Drawer Location	Description of Substitution, Change of Fuel Types in 0.7807-liter Core Section of Drawer	Substitution Worth, 1h
1-P-17	Carbide-for-oxide	+198 ± 2
1-P-17	Alloy-for-oxide	+107 ± 1
1-Q-17	Carbide-for-oxide	+191 ± 2
1-R-17	Alloy-for-carbide	-74.8 ± 0.8
1-S-17	Carbide-for-oxide	+132 ± 1
1-S-17	Alloy-for-oxide	+69.6 ± 0.8
1-T-17	Oxide-for-alloy	-42.4 ± 0.5
1-T-18	Alloy-for-oxide	+40.7 ± 0.5
1-U-17	Oxide-for-reflector	+75.3 ± 0.8
1-U-17	Carbide-for-reflector	+135 ± 1
1-S-14	Reflector-for-oxide	-106 ± 1
1-Q-21	Reflector-for-oxide	-102 ± 1
Inside halves of 1-Q-21 and 2-Q-21	Reflector-for-oxide	-106 ± 1
Inside halves of 1-S-14 and 2-S-14	Reflector-for-oxide	-106 ± 1

B. The Critical Experiment

1. Construction of Critical Configuration

In the initial loading of the critical approach for Core F, the first three fuel zones were fully loaded and the fourth zone contained seven oxide- and seven alloy-fuel drawers in each machine half. Subsequent loadings of two to four each of oxide- and alloy-fuel drawers were made until achieving criticality with the loading shown in Fig. 55. This configuration, with 12 oxide and 12 alloy elements in Zone IV of each half of the assembly, was the desired representation of the multizoned, hexagonal FARET system shown in Fig. 1. The compositions given in Table XXVII for the four Core-F fuel zones are based on the Fig. 55 loading, averaging the oxide and alloy elements in Zone IV and including the control rods in Zones III and IV. With 61 full fuel drawers per assembly half, the available excess reactivity was 97 ± 2 lh, giving a k value of about 1.001 for the configuration.

2. Just-critical Dimensions

In the axial schematic representation for Assembly 46F in Fig. 49, the hexagonal annuli of the core zones have been converted on an equal-volume basis to smooth cylindrical annuli. The radii expressed in Table XXVII for Core Zones I, II, and III were derived in this manner. However, the outer radius for Core Zone IV given in the table is for a just-critical loading of 23.62 core drawers per assembly half in that zone; the adjustment of 0.38 fuel drawer per half accounts for excess reactivity of the Fig. 55 configuration. Later experiments on edge-fuel worths defined the correction needed. The Zone IV specifications of volume and fuel mass in Table XXVII are for the 23.62 drawers per half critical loading. The error in the volume is probably about ± 0.2 liter, reflecting uncertainty in the worth of edge fuel. However, the errors due to the simplified, cylindrical representation and to the material heterogeneity effects may be considerably larger.

C. Control-rod Calibrations

The reactivity calibrations of the control rods used for worth measurements in Core-F experiments are shown in Fig. 56. The relative calibration shapes were determined by the equilibration technique cited for Core B, and absolute correlations to reactivity were found by period measurements. Details and results of the calculations to determine the

Fig. 56. Rod Calibrations in Assembly 46F

delayed-neutron characteristics for this multizoned core system are included in Appendix C. The principal parameters obtained are the following:

$$\begin{aligned}\beta_{\text{eff}} &= 0.00542; \\ l\$ &= 301 \text{ Ih}; \\ 1\% k &= 555 \text{ Ih}; \\ \text{Prompt lifetime} &= 2.44 \times 10^{-7} \text{ sec.}\end{aligned}$$

D. Fuel-exchange Experiments

Fuel-exchange experiments were conducted in Assembly 46F to evaluate the reactivity changes that would occur in a multizoned FARET core from insertion of fuel subassemblies into a dissimilar fuel zone or into the reflector. Thus, a number of fuel exchanges were made in each of the four 46F core zones, and each of the three fuel types was also substituted for reflector. Table XXVIII contains the pertinent details and results of all the experiments; Part a of the table lists the compositions for the three fuel types and radial reflector involved in the interchanges; Part b summarizes the locations of the exchanges and the reactivity effects measured. The locations involved are also illustrated in Fig. 57. The fuel-drawer loading patterns were the basic Core-F arrangements shown in Fig. 54.

Fig. 57. Locations of Fuel-exchange Experiments in Core 46F

1. Substitutions in Oxide Fuel Zones

Fuel exchanges in Core Zone I were carried out in Drawers 1-P-17 and 1-Q-17. First, an alloy-fuel drawer was substituted for the central oxide drawer, 1-P-17, and this alloy fuel was then replaced with carbide fuel. The net results, expressed as alloy-for-oxide and carbide-for-oxide exchange

worths in Table XXVIII, Part b, show twice the effect for the carbide fuel as for the alloy fuel. These worths represent half the reactivity effects that would have occurred in FARET from comparable replacements of the central subassembly; the effect would have been nearly 1% k with a carbide subassembly. A carbide-for-oxide replacement in Drawer 1-Q-17, in the second ring of Zone I, had almost the same worth as for the central replacement.

Experiments in Core Zone III involved carbide-for-oxide and alloy-for-oxide replacements in Drawer 1-S-17. As reported in Table XXVIII, the measured worths were again in a ratio of about 2 to 1.

2. Substitutions in Carbide-fuel Zone

In Core Zone II, only one substitution was made, the replacement of a carbide-fuel simulation drawer by an alloy-simulation drawer in 1-R-17. The result indicates that the insertion of an equivalent alloy-fuel subassembly into the carbide zone of a comparable FARET system would have decreased the reactivity by approximately $1/4\%$ k. Interpolation between the results from the carbide-for-oxide worths in Zones I and III suggest that an oxide-for-carbide subassembly replacement in Zone II of FARET would have reduced reactivity by about 0.4% k.

3. Substitutions in Mixed-fuel Zone

Only oxide-for-alloy and alloy-for-oxide replacements were carried out in Core Zone IV. These were the experiments at drawer positions 1-T-17 and 1-T-18 listed in Table XXVIII, and the worths obtained were reasonably consistent. From these measurements, it can be argued that the alternation of individual oxide and alloy elements as opposed to pairs of the elements does not affect the critical volume significantly.

4. Substitutions for Radial Reflector

Fuel-for-reflector replacements were made as part of the fuel-interchange studies and also for the determination of core-edge fuel worths and for adjustments of reactivity during the Core-F program. In all cases, the fuel drawers had 10-in.-long core sections followed by 11 in. of axial reflector (see Table I), and the reflector drawers contained 21 in. of fine radial reflector (see Table XXVIII).

The exchanges nearest the core were the joint replacements at half-drawer units in the inside halves of 1- and 2-S-14 and similarly in 1- and 2-Q-21. In each of these two experiments, the half-drawer units of oxide fuel in both halves of the assembly together comprised a regular oxide-fuel drawer, and the reflector-for-oxide substitution worths agreed closely with the results obtained with single-drawer replacements in

1-S-14 and 1-Q-21. The Q-21 and S-14 unit locations do not represent placements of FARET subassembly units in the hexagonal mockup scheme. However, the exchange results provide a core-edge worth of oxide fuel, 106 ± 1 lh/drawer. A core-edge worth of 148 lh/drawer for alloy fuel is derived thereby, with the assumption that that worth obtained in the alloy-for-oxide replacement in 1-T-17 applies also to the Q-21 location. The average of the oxide- and alloy-fuel worths at the core edge is thus 127 lh/drawer, and this value was used to calculate the mass correction for the excess reactivity in the critical loading.

Relating to the loading of FARET, the Table XXVIII results indicate that the substitution of reflector material for fuel in the outer half of a fifth-ring subassembly would be worth -106 lh for oxide fuel and -148 lh for alloy fuel. The replacement of reflector in the sixth subassembly ring with oxide fuel would be worth +150 lh, and with carbide fuel, +270 lh.

E. Mockup B₄C Control-rod Experiments

Experiments were conducted with mockups of FARET control rods at four locations in the reflector around Core 46F. These mockup rods

represented four neighboring rods of the 12-rod configuration shown for FARET in Fig. 1. The placements of the mockup channels around Zone IV of Core F, as illustrated in Fig. 58, were in keeping with the staggered-drawer, hexagonal arrangement adopted with Core F to represent the Fig. 1 FARET loading. Substitutions in the channel drawers were carried out as with Core C to simulate movements of the control rods in FARET.

Fig. 58. Locations of Mockup Control-rod Channels in Assembly 46F

1. Locations and Loadings of Mockup-rod Channels

Each of the four mockup-rod channels occupied one solitary matrix unit or adjacent halves of two matrix units in both ZPR-3 machine halves, as indicated in Fig. 58. Their placements within

the Core-F hexagonal pattern produced separations and distances to the core center in accordance with the ZPR-3 matrix unit and half-unit dimensions, rather than obtaining a close representation of the FARET dimensions. When loaded with a mockup rod, a channel contained a 20-in.-long poison or

"active" section with rod-follower sections before and after it. Figure 59 gives cross-sectional views of the loadings of active and follower sections for the one-matrix-unit channels, those in Drawers J-17 and P-22. In the split-matrix channels, K-19/20 and M-21/22, the same materials were

Fig. 59. Cross Sections of Mockup Control Rod for Core 46F. ANL Neg. No. ID-103-2568.

channel loadings during the different stages of the simulated rod motions with Core F as well as Core C. It must be recalled that the axial directions in the assembly correspond to the vertical directions in the FARET design. An additional axial placement was used with Core F, designated the "remote" position, which simulated the FARET rod positioned with the bottom of the active section 5 in. above the upper boundary of the core. This gave the active rod situation between +38.2 and +89.6 cm.

2. Worths of 69%-enriched B_4C Mockup Control Rods

Table XXIX lists the experimental steps in the FARET control-rod-worth studies with Core F. In the initial experiment, the fine and coarse radial reflectors in one of the designated rod channels (P-22 drawers) were replaced by a mockup rod in the "remote" position. The results indicate that the inclusion of 12 similar rods with the same axial situation in FARET, in place of 12 reflector subassemblies, would have required one additional fuel subassembly in the core for reactivity compensation.

In the next two experimental steps, the three other mockup rods were introduced to Assembly 46F in the "up" position. Then, the active section of the first rod was inserted 5 in. from the "remote" to the "up" position. The sum of the worths for Steps A through D in Table XXIX suggests that 12 rods situated in the "up" position in a FARET system would require $2\frac{1}{2}$ more fuel subassemblies for criticality. However, the mockup-channel placements do not represent the actual FARET rod locations closely enough for a direct extrapolation.

loaded, only with the matrix walls located down the middle instead of at the sides. The composition achieved in the poison section was the same as that for the high-enrichment rods used with Core C, amounting to about 55 vol % B_4C , 18 vol % sodium, and 19 vol % steel, with the boron enrichment averaging 69% ^{10}B . A detailed breakdown of the material atom densities is given in Part a of Table XVIII for this 69% active section and also for the follower and reflector loadings involved in the experiments.

Part b of Table XVIII defines the axial extensions of the various

TABLE XXIX. Worths of High-enrichment (69%) B₄C Mockup FARET Control Rods in Assembly 46F

Experiment	Mockup Channel(s)	Description of Loading Change within Mockup Channel(s) ^a	Change Worth, lh
A	P-22	Replaced radial reflector in channel with mockup rod in "remote" position (+15 in.)	-21.2 ± 0.4
B	J-17	Replaced radial reflector in channel with mockup rod in "up" position (+10 in.)	-45.1 ± 0.5
C	K-19/20 and M-21/22	Replaced radial reflector in channels with mockup rods in "up" position (+10 in.)	-109 ± 1
D	P-22	Moved 69%-active rod from "remote" position (+15 in.) to "up" position (+10 in.)	-41.8 ± 0.5
E	P-22	Moved 69%-active rod from "up" position (+10 in.) to midplane (0 in.)	-242 ± 5
F	P-22	Moved 69%-active rod from midplane (0 in.) to "down" position (-10 in.)	-191 ± 10
G	J-17	Moved 69%-active rod from "up" position (+10 in.) to "down" position (-10 in.)	-270 ± 13
H	K-19/20 and M-21/22	Moved 69%-active rods from "up" position (+10 in.) to "down" position (-10 in.)	-540 ± 30

^aSpecifications of loadings given in Table XVIII.

The active section in mockup channel P-22 was then moved inward in two steps to the "down" position, giving the worths cited for Experiments E and F in Table XXIX. On the basis of these worths, a solitary FARET rod of the same makeup and radial position would provide a reactivity control of 433 lh or about 0.8% $\Delta k/k$. The mockup rod in the J-17 channel, with a reflector drawer between it and the core, was found to have the full travel worth stated for Step G in the table, -270 lh. Thus a similarly situated rod in FARET, in the second ring of subassemblies away from the core, would have given only about 0.5% reactivity control.

In the final experiment, Step H, the middle two mockup rods in channels K-19/20 and M-21/22 were moved from the "up" to the "down" position while both of the outside mockup rods were fully inserted. The net worth of the middle rods, 540 lh, differed from the combined worth of the outside channel rods, 703 lh, by -23%. This "shadowing" effect is the same as found in the Core-C mockup-rod experiments. The total worths measured for the insertion of all four mockup rods extrapolates to a worth of 3730 lh, or 6.7% $\Delta k/k$ of reactivity control, in 12 control rods of the same composition and situation in FARET.

F. Regional Worths of Materials

A series of material-substitution experiments was carried out in each Core-F radial fuel zone to obtain reactivity coefficients for the basic materials in each zone, excluding plutonium and ^{235}U . Successive replace-

Fig. 60. Substitution Locations for Material-worth Measurements in Fuel Zones of Core 46F

ments of material plates or cans were made in the central two 1/8-in. columns for the 10-in. core length in a number of drawer units in each particular zone. The total region involved for each zone consisted of all or half the drawers in Assembly Half No. 2 for that zone, and the drawer positions are indicated in Fig. 60. Table XXX is a summary of the measurements in all the zones.

1. Material Substitutions in Core Zone I

Reactivity coefficients in the central oxide-fuel zone were obtained from substitutions in all seven of the core Zone-I drawer units in Half No. 2. The materials involved provided the average zone worths for ^{238}U , oxygen, carbon, sodium, and steel, the constituents of the oxide-fuel simulation exclusive of plutonium and ^{235}U . Sodium was actually measured in two axial segments, from 0 to 4 in. and from 4 to 10 in. in the drawers.

TABLE XXX. Average Worths of Materials in Core Zones of Assembly 46F

Material	Reactivity Coefficients in Zones, 1h/kg				
	Radial Zone I (Oxide)	Radial Zone II (Carbide)	Radial Zone III (Oxide)	Radial Zone IV	
				Oxide Drawers	Alloy Drawers
Sodium, 0 to 4 in. axially	24.6 ± 1	44.6 ± 2.0	62.5 ± 1.3	63 ± 1 ^a	63 ± 1 ^a
Sodium, 4 to 10 in. axially	52.5 ± 1.2	58.2 ± 1.7	47.3 ± 0.9	45.8 ± 1.1 ^a	45.8 ± 1.1 ^a
Type 304 ^b stainless steel	2.26 ± 0.15	4.83 ± 0.17	9.71 ± 0.13	13.6 ± 2	10.6 ± 0.2
Depleted uranium ^c	3.24 ± 0.12	6.01 ± 0.11	4.88 ± 0.12	5.51 ± 0.15 ^a	5.51 ± 0.15 ^a
Graphite	107 ± 2	116.2 ± 1.6	108.8 ± 1.5	-	-
U ₃ O ₈ ^c	12.3 ± 0.3	-	15.6 ± 0.2	17.2 ± 0.4	-
Zirconium	-	-	-	-	17.2 ± 0.2
Oxygen ^d	66 ± 2	-	76 ± 2	84 ± 2	-

^aAverage worths in oxide and alloy drawers combined.

^bNominal composition 72 wt % iron, 18 wt % chromium, and 10 wt % nickel.

^c0.2 wt % ^{235}U .

^dDerived from uranium and U₃O₈ worths.

2. Material Substitutions in Core Zone II

The material replacements to obtain average worths in the carbide-fuel zone were carried out in six of the 12 Zone-II drawer units of Half No. 2. Figure 60 shows the outline of Region II which these drawer units formed, constituting an integral half of the Zone-II hexagonal annulus. The materials substituted were the nonfissile constituents of the carbide-fuel simulations: depleted uranium, graphite, sodium, and steel. Table XXX lists the reactivity coefficients derived, including the breakdown of the sodium coefficient into two axial segments.

3. Material Substitutions in Core Zone III

The average worths of materials in the second oxide-fuel zone were found from substitutions in nine of the 18 Zone-III drawer units in Half No. 2. Figure 60 also indicates the positions of these drawers as forming the outlined Region III, which was half of the Zone-III hexagonal annulus. Again, the constituent materials of the oxide-fuel loading were substituted at various stages into the central columns of the drawer units, and their worths were determined relative to void. Table XXX includes the Zone-III average reactivity coefficients.

4. Material Substitutions in Core Zone IV

The material-worth measurements in the mixed oxide-alloy fuel zone were made in the six oxide and six alloy drawer units that made up the Region IV outlined in Fig. 60. These constituted half of the total Zone-IV annulus in Half No. 2. Actually, three distinct measurement stages were involved with this zone:

(1) The materials contained in both the alloy- and the oxide-fuel simulations, sodium, depleted uranium, and steel, were substituted in all 12 Region-IV drawers simultaneously.

(2) The worth of U_3O_8 was measured only in the six oxide-fuel drawers to determine the oxygen reactivity coefficient.

(3) The worth of zirconium was measured only in the six zirconium-containing alloy-fuel simulation drawers.

The results from these specific measurements are identified in the last two columns of Table XXX.

G. Radial Traverses of Reactivity Samples

Radial profiles of the worths of fuel materials were measured in the same manner as in Cores B and C, by traversing small reactivity samples

and balancing their worth with an autorod. Again, a 1/2-in.-diam traverse channel was created through the front of the P-row drawers in Half No. 1

SIDE VIEWS AT FRONT OF P - ROW CORE DRAWERS

a. OXIDE CORE DRAWER

b. CARBIDE CORE DRAWER

c. ALLOY CORE DRAWER

Fig. 61. Drawer Loadings for Assembly-46F Radial Traverses

of the reactor. Figure 61 illustrates the loadings of the three different fuel-type drawers to accommodate the traverse guide tube through the Core-F fuel zones. The reflector drawers involved were reloaded as specified in Fig. 17b. The enriched-uranium, natural-uranium, plutonium, and enriched-boron samples described in Table VIII were traversed through the radial channel thus constructed.

Table XXXI presents the results of the Core-F reactivity traverses, again reduced, where plausible, to coefficients for the principal isotopes of the samples. In the table, the values listed for radial positions through 12.70 cm are averages of values on both sides of the reactor axis. Figure 62 gives the radial profiles of the plutonium and isotopic ^{235}U worths, and Fig. 63 shows the worth profiles for isotopic ^{238}U and ^{10}B . Assuming a probable worth of ^{11}B on the order of that for carbon, it can be argued that the error induced by neglecting the ^{11}B contribution to the enriched-boron worth is less than 1%.

TABLE XXXI. Uranium, Plutonium, and Boron Worths versus Radius near Midplane in Assembly 46F

Radius, cm	Reactivity Coefficient, 1h/kg				Radius, cm	Reactivity Coefficient, 1h/kg			
	$^{235}\text{U}^{(a)}$ (± 3)	$^{238}\text{U}^{(a)}$ (± 0.3)	Plutonium ^(b) (± 5)	$^{10}\text{B}^{(c)}$ (± 30)		$^{235}\text{U}^{(a)}$ (± 3)	$^{238}\text{U}^{(a)}$ (± 0.3)	Plutonium ^(b) (± 5)	$^{10}\text{B}^{(c)}$ (± 30)
0	306	-0.89	520	-5430	22.86	136	8.40	192	-2376
2.54	302	-0.44	513	-5034	27.94	86	3.84	110	-1622
5.08	296	0.59	505	-	33.02	63	1.58	68	-897
7.62	286	2.12	486	-4972	38.10	43	1.10	45	-424
10.16	270	4.07	459	-	43.18	27	0.74	27	-169
12.70	248	7.35	412	-4192	48.26	16	0.19	17	-59
15.24	213	8.55	345	-3712	53.34	7	0.03	9	-7
17.78	187	8.65	292	-3232	60.96	2	0	2	-2
20.32	160	8.69	239	-					

^aValues for isotope, derived from worths of enriched- and natural-uranium samples specified in Table VIII.

^bFor plutonium mixture of sample specified in Table VIII.

^cFor isotope, derived from worth of enriched-boron sample specified in Table VIII, assuming negligible worth for ^{11}B and impurities.

Fig. 62. Radial Profiles of Plutonium and ^{235}U Worths in Assembly 46F

Fig. 63. Radial Profiles of ^{238}U and ^{10}B Worths in Assembly 46F

H. Radial Traverses of Neutron Detectors

Traverses of neutron detectors were also made in the radial channel through Assembly 46F described in the preceding section. The fission counters of enriched uranium (No. J), depleted uranium, and plutonium, and also the BF_3 proportional counter specified in Table XI, were used in these experiments. Table XXXII lists the relative fission or reaction rates obtained for the principal isotopes of the counters as a function of radius.

TABLE XXXII. Reaction Rates versus Radius in Assembly 46F

Radius, cm	Relative Reaction Rate, ^a Percent of Core Center Rate				Radius, cm	Relative Reaction Rate, ^a Percent of Core Center Rate			
	^{235}U Fission ^b	^{238}U Fission ^b	^{239}Pu Fission ^c	^{10}B Capture ^b		^{235}U Fission ^b	^{238}U Fission ^b	^{239}Pu Fission ^c	^{10}B Capture ^b
0	100.0	100.0	100.0	100.0	27.94	85.4	21.4	74.9	168.7
2.54	100.3	99.7	99.6	99.7	30.48	87.6	14.4	76.6	188.1
5.08	99.4	99.9	98.6	98.4	33.02	86.3	-	76.1	-
7.62	96.7	99.4	96.5	96.2	35.56	83.5	7.4	73.3	204.7
10.16	93.4	99.2	93.7	93.8	38.10	79.7	4.8	71.4	204.3
11.43	-	96.0	-	-	40.64	74.2	3.0	66.8	199.9
12.70	89.9	90.8	88.6	92.0	43.18	-	2.2	-	-
13.97	-	83.5	-	-	45.72	61.8	-	57.1	178.7
15.24	84.9	78.3	82.6	91.6	48.26	55.5	1.1	-	-
16.51	-	73.9	-	-	50.80	48.8	-	44.8	143.0
17.78	80.5	68.1	76.7	92.1	53.34	-	0.56	-	-
19.05	-	63.1	-	-	55.88	34.0	-	-	-
20.32	76.9	58.8	72.2	97.0	58.42	27.8	0.27	25.3	83.2
21.59	75.5	54.4	69.8	-	66.04	19.5	-	17.4	-
22.86	75.8	48.9	68.6	108.5	68.58	16.5	-	-	48.8
24.13	76.4	41.5	69.2	-	73.66	11.2	-	10.3	-
25.40	79.6	33.5	71.2	133.5					

^aEstimated accuracy of data is ± 2 in last significant figures, but no better than 0.7% of values.

^bValues for isotopes, derived from traverses with detectors specified in Table XI.

^cFor isotopic ^{239}Pu , as derived from count rates for plutonium mixture of chamber specified in Table XI with estimated corrections for ^{240}Pu fissions.

Figure 64 contains plots of the results for the ^{235}U and plutonium fission rates and the ^{10}B reaction rate. The ^{238}U fission-rate profile is presented as a semilog graph in Fig. 65. In both figures, the drawer boundaries for the four fuel zones are delineated. A definite flattening is observed in the rate of ^{238}U fission, from the core center out into the carbide fuel zone. This would logically be expected from the higher power density in the carbide zone.

Fig. 64. ^{235}U Fission, ^{239}Pu Fission, and ^{10}B Reaction-rate Profiles Radially through Assembly 46F

Fig. 65. ^{238}U Fission-rate Profile Radially through Assembly 46F

X. CORE-G EXPERIMENTS

The final core arrangement constructed in Assembly 46 was not actually part of the requested experimental program for the FARET mockup. However, the details are included here to complete the set of critical-configuration data derived from the basic core-fuel simulations and reflector pattern used throughout the work with the assembly. Basically, Core G was used for special shell-detector and blanket-reflector experiments pertaining to fast reactors.

A. Composition and Geometry of Core G

1. Construction of Core-G Critical Configuration

The construction of Core G was essentially the transformation of the four-fuel-zoned Core F into a three-fuel-zoned core by converting all the central oxide-simulation zone to carbide fuel. Thus, the same fuel-drawer loadings were used as those illustrated in Fig. 54 for Core F. The transformation was made while remaining critical by successive replacements of central-zone oxide-fuel drawers by carbide-fuel drawers simultaneous with the replacements of oxide- and alloy-fuel drawers by reflector at the outer radial boundary of the core. A central carbide-fuel zone was thereby obtained which, when united with the previous carbide zone (Zone II in Core F), contained a total of 19 carbide drawer units per assembly half. Figure 66 shows the basic critical loading achieved for Core G at this stage. Some initially scheduled Core-G experiments involved fission counters at the core center, and for this reason the central drawer of

Fig. 66. Critical Configuration of Core 46G

Half No. 1 in Fig. 66 (1-P-17) was essentially void for the 10-in. length of the core. Atomic densities for the core sections of the three fuel-type drawers are given in Table XXVIII for Core F. The available excess reactivity for the configuration indicated in Fig. 66, with 37 carbide-, $58\frac{1}{2}$ oxide-, and eight alloy-fuel drawers, was about $0.15\% \Delta k/k$.

2. Critical Size of Core G in Idealized Geometry

Representing the Core-G critical system of Fig. 66 as a simplified set of equivalent cylindrical annuli, as was done for Cores D, E, and F, entails a number of adjustments and approximations because of the crude core-reflector boundary and the central void. The approach taken to derive the idealized geometry shown in Fig. 67 was to consider the Core-G system as a set of hexagonal annuli of drawer units like the Core-F configuration. Thus, the first three annuli, containing 17 carbide-fuel drawer units per half, constituted Core Zone I. The fourth annular ring of drawer units, all loaded with oxide fuel, became Core Zone II in Fig. 67. Finally, the fifth ring of drawer units, with the same geometric pattern as Zone IV of Core F, contained oxide fuel, alloy fuel and radial reflector, and this mixture comprises Zone III of Core G in Fig. 67.

Including a correction to the core-edge fuel loading which would compensate for filling the 1-P-17 void with carbide fuel, the just-critical loading of Zone III for Core G was estimated at 10 oxide- and three alloy-fuel drawer units plus 11 reflector drawers per assembly half. The average composition resulting from this mixture of fuels and reflector is listed under Zone III in Table XXXIII. In the table, the composition for Zone II is the same as that for oxide-fueled Zone III in Core F, and the Core-G central carbide zone had the same composition as Zone II of Core F.

B. Shell-detector Experiments at Core Center

The purpose of constructing the uniform, extensive, central carbide-fuel zone was to obtain a region for conducting fission-ratio measurements with heavy-shell detectors. A sufficiently large zone radius was desired to ensure that the neutron spectrum at the core center was principally representative of the carbide-fuel composition and could be reasonably predicted with multigroup calculations. An extensive series of experiments with fission counters at the core center was then conducted by Davey and Amundson¹⁴ of the Idaho Division. They investigated the effects of thick spherical shells of materials upon the fission ratios of $^{238}\text{U}/^{235}\text{U}$, $^{236}\text{U}/^{235}\text{U}$, and $^{234}\text{U}/^{235}\text{U}$ within the shells. The objective was to obtain the percentage changes in the ratios with the shells, compared to without the shells, for numerous shell materials. These changes were then compared with calculated changes to evaluate the propriety of the inelastic scattering cross sections used for the shell materials. The report by the experimenters indicates good-to-poor correlations and the need for further similar investigations.

Fig. 67. Axial Schematic Diagram of Assembly 46G

TABLE XXXIII. Specifications of Core-46G Fuel Zones

	Zone I (Carbide)	Zone II (Oxide)	Zone III (Oxide-Alloy- Reflector Mixture) ^a
<u>Critical Geometry</u>			
Length, cm	50.95	50.95	50.95
Average outer radius, cm	13.61	19.00	24.32
Volume, liters	29.67	28.11	36.89
<u>Average Composition, atoms/cm³ x 10⁻²²</u>			
²³⁹ Pu	0.1558	0.1062	0.0771
²⁴⁰ + ²⁴² Pu	0.0074	0.0051	0.0037
²⁴¹ Pu	0.0007	0.0005	0.0003
²³⁵ U	0.4103	0.2243	0.1075
²³⁸ U	0.4819	0.4575	0.2967
²³⁴ + ²³⁶ U	0.0059	0.0032	0.0016
Na	0.8577	0.8295	0.5354
O	-	0.8582	0.3576
C	1.1061	0.5042	0.2101
Fe	1.0593	1.1062	2.7867
Cr	0.2844	0.3115	0.7483
Ni	0.1399	0.1533	0.3682
Zr	-	-	0.0308
Al	-	-	0.1474

^aZone III for Core G equivalent to fifth ring of subassemblies in FARET design (same as Zone IV in Core F). Just-critical loading without central counter void estimated at three alloy, 10 oxide, and 11 reflector drawers per half.

C. Uranium-blanket Experiments

Near the completion of the experimental program for the FARET mockup, an interest had developed in the relative worth of stainless steel reflector compared to depleted uranium blanket in small fast-reactor systems. This arose out of the operational plans for Experimental Breeder Reactor II (EBR-II), wherein the replacement of uranium radial blanket by steel reflector would eliminate fuel-cycle processing for blanket elements and speed up the fuels-testing program. Such reflector-blanket experiments had been done on larger fast-reactor systems,¹⁵ but it was not certain what effects would be experienced in EBR-II. The similar size and nature of the FARET mockup made it an ideal system for comparable measurements.

The blanket experiments in Assembly 46G were carried out in a quadrant of the radial reflector in both halves of the reactor. First, a smoother core boundary was created next to the reflector region, and this is illustrated in Fig. 68 for Half No. 1. The figure also shows the orientation of the reflector quadrant. In the area shown, all the fine and coarse steel reflector material was replaced by depleted-uranium blanket in a series of substitutions. The numbers labeled on each quadrant drawer in the diagram are indices for the locations of drawers involved together in particular substitution steps.

Fig. 68. Half No. 1 Interface View of Assembly 46G Showing Blanket Experiment Locations

The drawer and matrix loadings for the uranium blanket compositions used in the experiments are shown in Fig. 69. As shown, the substitution region extended axially to 10 in. from the reactor midplane, the same as the core length. The substitutions of blanket for fine radial reflector, in the locations with indices 1 through 7 in Fig. 68, consisted of replacing stainless steel by depleted uranium in twelve 1/8-in. columns per drawer (about 60 vol. % in each drawer). In the coarse radial reflector, with the No. 8 index specification in Fig. 68, the substitution was the replacement of 2 x 2 x 5-in. carbon-steel bricks by depleted-uranium bricks of the same size, amounting to 84 vol % in each drawer. All the compositions involved in these experiments are given in Table XXXIV.

Table XXXV lists the sequence of the experiments and the results. For each step, the table gives the number of drawers changed and specifies their locations according to the index numbers indicated in Fig. 68. The worths given express the reactivity effect from the blanket-for-reflector

exchange in the specified drawers in both assembly halves. The sum of worths for Steps A through H, with the entire quadrant then changed to uranium blanket, was -263 ± 2 lh. On a direct extrapolation, the replacement of the entire steel radial reflector by a depleted-uranium blanket in Assembly 46G would be worth about -1052 lh, or -1.9% k. Approximately the same effect could be expected with other fast-reactor cores of around 95 liters from a similar replacement of steel reflector by uranium.

A. FINE RADIAL BLANKET

B. COARSE RADIAL BLANKET
(LOADED DIRECTLY INTO MATRIX)

Fig. 69. Loadings of Depleted Uranium Blanket Used
in Assembly-46G Reflector Experiments

TABLE XXXIV. Compositions Involved in Assembly-46G Blanket-Reflector Experiments

Drawer Loading Designation ^a	Symbol	Material Densities, atoms/cm ³ x 10 ⁻²²						
		²³⁵ U	²³⁸ U	Na	Fe	Cr	Ni	Other
Fine radial reflector	FRR	-	-	0.212	4.690	1.259	0.620	0.306 Al
Coarse radial reflector ^b	CRR	-	-	-	7.413	0.119	0.058	0.057 Mn 0.065 C
Fine radial blanket	FRB	0.006	2.887	0.212	0.954	0.256	0.126	0.306 Al
Coarse radial blanket ^b	CRB	0.008	3.998	-	0.445	0.119	0.058	-
Fine iron reflector	FIR	-	-	0.212	6.072	0.256	0.126	0.306 Al

^aDimensions per drawer section 25.5 x 5.54 x 5.54 cm; volume = 0.7807 liter.

^bThese coarse types composed from 2 x 2 x 5-in. bricks loaded in matrix without drawers.

TABLE XXXV. Uranium Blanket-for-Steel-Reflector Replacement Worths in Assembly 46G

Experiment Step	Index of Exchange Locations ^a	Number of Drawers Exchanged per Assembly Half	Symbolic Exchange To/From	Description of Experiment ^b	Exchange Worth, lh (± 0.8)
A	1	2	FRB/FRR	Depleted uranium substituted for 60 vol % stainless steel	-47.3
B	2	2	FRB/FRR	As above	-36.1
C	3	2	FRB/FRR	As above	-37.3
D	4	4	FRB/FRR	As above	-41.8
E	5	3	FRB/FRR	As above	-39.9
F	6	8	FRB/FRR	As above	-41.6
G	7	9	FRB/FRR	As above	-18.0
H	8	28	CRB/CRR	Depleted uranium substituted for 84 vol % carbon steel	-0.8
I	1	2	FRR/FRB	Stainless steel substituted for 60 vol % depleted uranium	+27.3
J	1	2	FIR/FRR	Iron substituted for 60 vol % stainless steel	-24.9
K	2	2	FRR/FRB	Stainless steel substituted for 60 vol % depleted uranium	+27.5

^aAs illustrated in Fig. 68 for Half No. 1, corresponding locations in Half No. 2 also used.

^bSubstitutions made only in front 25.5 cm. Section as defined in Table XXXIV.

In Table XXXV, Steps I and K were the return of a few blanket drawers next to the core to their original reflector composition. The relative worth of steel versus uranium is substantially different in the uranium blanket than in the steel-reflector environment. An iron-for-steel replacement also was carried out in a few drawers in experimental Step J. The result of that substitution suggests that iron and uranium have about equal worth as reflectors for a fast reactor. This has been borne out in experiments in past assemblies with steel, iron, and nickel reflectors; these studies indicate that the nickel content of the steel constitutes the major advantage over uranium with regard to reactivity.

APPENDIX A

Material Loadings of Reflector Drawers Used in Assembly 46

Reflector Drawer or Section Designation	Section Length, cm	Section Volume, liters	Material Density, g/cm ³				
			304 Stainless Steel ^a	Carbon Steel ^b	Aluminum	Sodium	Other
Axial reflector behind 12-in. core section of 21-in. front drawer	22.86	0.700	4.406	-	-	0.3263	-
Axial reflector behind 10-in. core section of 21-in. front drawer	27.94	0.856	4.400	-	-	0.3270	-
Spring gap	0.66	0.020	2.129	-	-	-	-
Axial reflector in 11-in. back drawer	27.94	0.856	4.415	-	-	0.3225	-
Fine radial reflector	53.42	1.637	6.040	-	0.137	0.0809	-
Coarse radial reflector I	76.20	1.946	0.573	6.525	-	-	-
Coarse radial reflector II-A (outer drawers)	25.40	0.778	0.707	6.525	-	-	-
Coarse radial reflector II-B (inner drawers)	25.40	0.778	7.263	-	-	-	-
Radial plenum	63.50	1.946	0.573	-	0.621	-	Ni: 1.816
Uranium fine radial blanket	53.42	1.637	1.234	-	0.137	0.0809	U ^c : 11.434
Uranium coarse radial blanket	76.20	1.946	0.573	-	-	-	U ^c : 15.833
Iron fine radial reflector	53.42	1.637	1.234	-	0.137	0.0809	Fe: 4.745

^aNominal composition of 72 wt % iron, 18 wt % chromium, and 10 wt % nickel used to derive atom densities in tables. Actual compositions ranged between 69.7 to 72.0 wt % iron, 16.9 to 17.6 wt % chromium, 7.8 to 10.4 wt % nickel, 0.3 to 1.5 wt % manganese, and 0.20 to 0.60 wt % silicon.

^bComposition of 99 wt % iron, 0.2 wt % manganese, and 0.8 wt % carbon.

^cDepleted uranium: 0.20 wt ²³⁵U.

APPENDIX B

Material Loadings of Core Drawers Used in Assembly 46

Core Drawer Designation	Fuel Length, ^a in.	Material Density, g/cm ³							
		Pu-Al Alloy ^b	²³⁵ U	²³⁸ U	²³⁴ + ²³⁶ U	Stainless Steel ^c	Sodium	Carbon	Other
A - Carbide Regular	12	0.658	1.600	1.905	0.023	1.362	0.322	0.221	-
A - Carbide Control	12	0.658	1.600	1.905	0.023	2.143	0.241	0.221	-
B - Carbide Regular	12	0.658	1.3062	2.214	0.019	1.362	0.322	0.221	-
B - Carbide Control	12	0.658	1.3062	2.214	0.019	2.143	0.241	0.221	-
B - Oxide Substitution	12	0.449	1.304	1.347	0.019	1.405	0.322	0.101	O: 0.228
C - Oxide Regular	10	0.449	1.305	1.357	0.019	1.402	0.321	0.101	O: 0.228
C - Oxide Control	10	0.449	1.305	1.357	0.019	2.095	0.241	0.101	O: 0.228
C - Carbide Substitution	10	0.628	1.737	1.765	0.025	1.369	0.321	0.221	-
C - Alloy Substitution	10	1.107	0.511	3.402	0.007	1.948	0.321	-	-
D - Oxide Regular	10	0.449	1.085	1.581	0.016	1.407	0.321	0.101	O: 0.228
D - Oxide Control	10	0.449	1.086	1.577	0.016	2.194	0.241	0.101	O: 0.228
D - Carbide Regular	10	0.658	1.747	1.765	0.025	1.364	0.321	0.221	-
D, F, & G - Alloy Regular	10	1.107	0.438	3.350	0.006	1.471	0.321	-	Zr: 0.480
D, F, & G - Alloy Control	10	1.107	0.438	3.350	0.006	1.982	0.321	-	Zr: 0.158
E, F, & G - Oxide Regular	10	0.449	0.876	1.809	0.013	1.408	0.321	0.101	O: 0.228
E, F, & G - Oxide Control	10	0.449	0.876	1.809	0.013	2.194	0.241	0.101	O: 0.228
E - Carbide Regular	10	0.658	1.307	2.217	0.019	1.364	0.321	0.221	-
E - Alloy Regular	10	1.107	0.218	3.575	0.003	1.471	0.321	-	Zr: 0.480
E - Alloy Control	10	1.107	0.218	3.575	0.003	1.982	0.321	-	Zr: 0.158
F & G - Carbide Regular	10	0.658	1.602	1.905	0.023	1.364	0.321	0.221	-

^aDimensions of core sections: 12-in. Length--5.53 x 5.54 x 30.56 cm, volume = 0.9364 liter.

10-in. Length--5.53 x 5.54 x 25.48 cm, volume = 0.7807 liter.

^bComposition of plutonium fuel slugs (excluding canning) = 98.9 wt % plutonium + 1.10 wt % aluminum. Aluminum content has been included with sodium in all atom density listings in text tables. Plutonium isotopic breakdown = 95.04 wt % ²³⁹, 4.55 wt % ²⁴⁰, 0.41 wt % ²⁴¹.

^cType 304: Nominal composition of 72 wt % iron, 18 wt % chromium, and 10 wt % nickel was used to derive atomic density breakdown for all text tables. Actual compositions ranged from 69.7 to 72.0 wt % iron, 16.9 to 17.6 wt % chromium, 7.8 to 10.4 wt % nickel, 1.3 to 1.5 wt % manganese, and 0.20 to 0.60 wt % silicon.

APPENDIX C

Delayed-neutron Parameters for Assembly 46, Cores B, C, and F

Neutron lifetime and delayed-neutron characteristics for the three basic core studies with Assembly 46, Cores B, C, and F, were calculated using the BAILIFF routine in the Argonne, MACH-I diffusion-theory code. For these computations, a 22-group cross-section set, ANL No. 224, was used, and this set incorporates delayed-neutron parameters as listed in ANL-5800,¹⁶ page 18. The BAILIFF results are reproduced below:

Core 46B: BAILIFF Results

Isotope	Delayed Worth	Prompt Worth
238U	2.29731×10^{-4}	2.26225×10^{-4}
235U	2.25717×10^{-4}	2.22285×10^{-4}
239Pu	2.26499×10^{-4}	2.23050×10^{-4}
240Pu	2.28822×10^{-4}	2.25378×10^{-4}

Effective Delayed-neutron Fractions

Isotope	BEFK(M,1)	BEFK(M,2)	BEFK(M,3)	BEFK(M,4)	BEFK(M,5)	BEFK(M,6)	BEFM(M)
238U	1.20713×10^{-5}	1.27213×10^{-4}	1.50427×10^{-4}	3.60282×10^{-4}	2.08926×10^{-4}	6.96421×10^{-5}	9.28561×10^{-4}
235U	1.38998×10^{-4}	7.79121×10^{-4}	6.87675×10^{-4}	1.48874×10^{-3}	4.68204×10^{-4}	9.51040×10^{-5}	3.65785×10^{-3}
239Pu	2.85385×10^{-5}	2.10284×10^{-4}	1.62219×10^{-4}	2.46332×10^{-4}	7.73543×10^{-5}	2.62855×10^{-5}	7.51013×10^{-4}
240Pu	4.06300×10^{-7}	3.96142×10^{-6}	2.78605×10^{-6}	5.07875×10^{-6}	1.85737×10^{-6}	4.20810×10^{-7}	1.45107×10^{-5}
Sums	1.80014×10^{-4}	1.12058×10^{-3}	1.00311×10^{-3}	2.10044×10^{-3}	7.56342×10^{-4}	1.91452×10^{-4}	5.35193×10^{-3}

The total delayed-neutron fraction = 5.35193×10^{-3} .

The prompt-neutron lifetime = $(3.54574 \times 10^{-9}) / (1.21882 \times 10^{-2}) = 2.90915 \times 10^{-7}$ sec.

Inhours per percent reactivity = 559,526.

Core 46C: BAILIFF ResultsWorths of Delayed and Prompt Fission Neutrons in Region N

Isotope	Delayed Worth	Prompt Worth
238U	1.84330×10^{-4}	1.76326×10^{-4}
235U	1.80880×10^{-4}	1.73015×10^{-4}
239Pu	1.81571×10^{-4}	1.73653×10^{-4}
240Pu	1.83616×10^{-4}	1.75642×10^{-4}

Effective Delayed-neutron Fractions

Isotope	BEFK(M,1)	BEFK(M,2)	BEFK(M,3)	BEFK(M,4)	BEFK(M,5)	BEFK(M,6)	BEFM(M)
238U	9.71443×10^{-6}	1.02430×10^{-4}	1.21097×10^{-4}	2.90100×10^{-4}	1.68217×10^{-4}	5.60723×10^{-5}	7.47630×10^{-4}
235U	1.69699×10^{-4}	9.51208×10^{-4}	8.39564×10^{-4}	1.81757×10^{-3}	5.71618×10^{-4}	1.16110×10^{-4}	4.46576×10^{-3}
239Pu	2.44092×10^{-5}	1.79857×10^{-4}	1.38747×10^{-4}	2.10690×10^{-4}	6.61617×10^{-5}	2.24821×10^{-5}	6.42347×10^{-4}
240Pu	3.41311×10^{-7}	3.32778×10^{-6}	2.34042×10^{-6}	4.26639×10^{-6}	1.56028×10^{-6}	3.53501×10^{-7}	1.21897×10^{-5}
Sums	2.04164×10^{-4}	1.23682×10^{-3}	1.10175×10^{-3}	2.3262×10^{-3}	8.07557×10^{-4}	1.95018×10^{-4}	5.86793×10^{-3}

The total delayed-neutron fraction = 5.86793×10^{-3} .

The prompt-neutron lifetime = $(1.77333 \times 10^{-9}) / (8.25533 \times 10^{-3}) = 2.14810 \times 10^{-7}$ sec.

Inhours per percent reactivity = 503.110.

Core 46F: BAILIFF Results

Worths of Delayed and Prompt Fission Neutrons in Region N

Isotope	Delayed Worth	Prompt Worth
^{238}U	2.33673×10^{-4}	2.26827×10^{-4}
^{235}U	2.33927×10^{-4}	2.26827×10^{-4}
^{239}Pu	2.33899×10^{-4}	2.26827×10^{-4}
^{240}Pu	2.33707×10^{-4}	2.26827×10^{-4}
^{238}U	2.28407×10^{-4}	2.21914×10^{-4}
^{235}U	2.28673×10^{-4}	2.21931×10^{-4}
^{239}Pu	2.28642×10^{-4}	2.21929×10^{-4}
^{240}Pu	2.28443×10^{-4}	2.21919×10^{-4}
^{238}U	2.12735×10^{-4}	2.07388×10^{-4}
^{235}U	2.12826×10^{-4}	2.07262×10^{-4}
^{239}Pu	2.12822×10^{-4}	2.07282×10^{-4}
^{240}Pu	2.12729×10^{-4}	2.07355×10^{-4}
^{238}U	1.80385×10^{-4}	1.77944×10^{-4}
^{235}U	1.79964×10^{-4}	1.77405×10^{-4}
^{239}Pu	1.80043×10^{-4}	1.77493×10^{-4}
^{240}Pu	1.80261×10^{-4}	1.77813×10^{-4}
^{238}U	1.45710×10^{-4}	1.46313×10^{-4}
^{235}U	1.44929×10^{-4}	1.45471×10^{-4}
^{239}Pu	1.45074×10^{-4}	1.45620×10^{-4}
^{240}Pu	1.45489×10^{-4}	1.46099×10^{-4}

Effective Delayed-neutron Fractions

Isotope	BEFK(M,1)	BEFK(M,2)	BEFK(M,3)	BEFK(M,4)	BEFK(M,5)	BEFK(M,6)	BEFM(M)
^{238}U	1.35125×10^{-5}	1.42477×10^{-4}	1.68442×10^{-4}	4.03519×10^{-4}	2.33984×10^{-4}	7.79947×10^{-5}	1.03993×10^{-3}
^{235}U	1.33659×10^{-5}	7.49194×10^{-4}	6.61261×10^{-4}	1.43156×10^{-3}	4.50220×10^{-4}	9.14509×10^{-5}	3.51734×10^{-3}
^{239}Pu	3.21953×10^{-5}	2.37228×10^{-4}	1.83005×10^{-4}	2.77896×10^{-4}	8.72662×10^{-5}	2.96536×10^{-5}	8.47244×10^{-4}
^{240}Pu	4.32759×10^{-7}	4.21940×10^{-6}	2.96749×10^{-6}	5.40948×10^{-6}	1.97833×10^{-6}	4.48214×10^{-7}	1.54557×10^{-5}
Sums	1.79800×10^{-4}	1.13312×10^{-3}	1.01568×10^{-3}	2.11838×10^{-3}	7.73449×10^{-4}	1.99547×10^{-4}	5.41997×10^{-3}

The total delayed-neutron fraction = 5.41997×10^{-3}

The prompt-neutron lifetime = $(2.35877 \times 10^{-9}) / (9.65636 \times 10^{-3}) = 2.44271 \times 10^{-7}$ sec.

Inhours per percent reactivity = 555.365.

ACKNOWLEDGMENTS

This report is the culmination of efforts by several Argonne personnel, both in Idaho and in Illinois. The authors were the principals involved with conduct of the ZPR-3 experiments. We here gratefully acknowledge the assistance of the remainder of the ZPR-3 staff, especially J. K. Long and J. M. Gasidlo. The mockup program was established by P. J. Persiani on behalf of the FARET project. The contributions to the Assembly-46 work by the FARET project group and by the ZPR-3 technicians are appreciated. Computational assistance was provided by Mrs. E. Slansky and D. A. Kucera.

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