MASTER

Argonne National Laboratory

CRITICAL STUDIES FOR THE
FRENCH FAST REACTOR "RAPSDIE"
(ZPR-3 Assembly 44)

by

A. L. Hess, W. P. Keeney,
P. Caumette, and J. P. Boyer

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS

DISCLAIMER

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

DISCLAIMER

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

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

COPY PRICES

H.C. \$ 3.00 MN. 65

ANL-7044

Reactor Technology

(TID-4500)

AEC Research and
Development Report

ARGONNE NATIONAL LABORATORY

9700 South Cass Avenue

Argonne, Illinois 60439

RELEASED FOR ANNOUNCEMENT

IN NUCLEAR SCIENCE ABSTRACTS

CRITICAL STUDIES FOR THE
FRENCH FAST REACTOR "RAPSDIE"

(ZPR-3 Assembly 44)

by

A. L. Hess, W. P. Keeney,
P. Caumette,* and J. P. Boyer*

Idaho Division

*On assignment from Centre d'Etudes
Nucleaires de Cadarache, France

March 1966

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

Operated by The University of Chicago
under

Contract W-31-109-eng-38

with the

U. S. Atomic Energy Commission

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

TABLE OF CONTENTS

	Page
ABSTRACT	9
I. INTRODUCTION	10
II. DESCRIPTION OF RAPSODIE	10
III. DESCRIPTION OF ZPR-3	13
IV. OBJECTIVES OF THE STUDIES	16
A. General	16
B. Physics-core Program	16
C. Engineering-mockup Program	17
V. PHYSICS-CORE EXPERIMENTS	18
A. Construction of the Assembly	18
1. Core-drawer Loading	18
2. Blanket Loading	18
3. Approach to Critical	20
B. The Critical Experiment	24
1. Core Configuration	24
2. Fuel Worth at Edge of Core	24
3. Critical Core Volume	25
C. Operations with Plutonium in ZPR-3	25
1. Neutron Source from Spontaneous Fission in Pu^{240}	25
2. Control-rod Calibrations	27
3. Temperature Profile and Drift	27
D. Spectral Indices	29
1. Fission-chamber Experiments at Center of Physics Core	30
2. Fission-chamber Experiments at Radial Edge of Physics Core	31
3. Uranium-foil Irradiations	32
E. Central Reactivity Coefficients	32
1. Small Samples	33
2. Large Samples	34
F. Worths of Fuel versus Position in Core	35

TABLE OF CONTENTS

	<u>Page</u>
VI. ENGINEERING-MOCKUP EXPERIMENTS.	36
A. Critical Experiment with Engineering Core.	36
1. Change of Core Composition.	36
2. Approach to Critical.	37
3. Fuel Worth at Radial Edge of Core	39
4. Critical Volume of Engineering Core	39
5. Control-rod Calibrations.	40
B. Construction of Axial Gaps and Blankets.	41
1. Upper and Lower Axial Blankets	41
2. Upper and Lower End Gaps	41
C. RAPSODIE Safety-Control Rod Mockups	45
1. Mockup Description	45
2. Worth of Replacing Blanket with Rod Channel	46
3. Worth of Safety-Control Rods in "Up" Position	46
4. Worths of Safety-Control Rods in "Banked" Position	49
D. Critical Configuration of Four-rod Engineering Mockup of RAPSODIE.	50
E. Reactivity-coefficient Measurements	50
1. Distributed Worths in Core	50
2. Distributed Worths in End Gaps	52
3. Regional Worths of Sodium in Core	52
4. Material Worths at Core Center	53
5. Steel and Sodium Worths in Safety-Control Rod Channel	54
6. Axial Traverses of Reactivity Samples.	54
7. Radial Traverses of Reactivity Samples	57
F. Reaction-rate Distributions	58
1. Axial Counter Traverses	58
2. Radial Counter Traverses	61
G. Rossi-alpha Measurements	62
H. Six-rod Engineering Mockup of RAPSODIE	64
1. Construction of Six-rod Engineering Mockup.	64
2. Critical Configuration of Six-rod Engineering Mockup.	66
3. Worth of RAPSODIE Control Rod.	67
4. Flux Depression by Control Rod	68
5. Worth of RAPSODIE Safety Rods	70
ACKNOWLEDGMENTS	71
REFERENCES	72

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
1.	Vertical Section of RAPSODIE Reactor.	11
2.	Cross Section of RAPSODIE Reactor	11
3.	Axial Configuration of RAPSODIE Reactor	12
4.	View of ZPR-3 Critical Facility	14
5.	Typical Core and Axial Blanket Drawers	15
6.	Core-drawer Loading for Physics Core	19
7.	Fine, Medium, and Coarse Blanket Drawer Loadings, Assembly 44	21
8.	Radial Blanket Configurations in Assembly 44	22
9.	Axial Schematic Diagram of ZPR-3 Assembly 44 with Physics Core.	23
10.	Approach-to-critical Curves for Physics Core	23
11.	Reference Critical Configuration for Physics Core.	24
12.	Control-rod Calibrations with Physics Core	28
13.	Axial Temperature Profiles during Extended Run	29
14.	Engineering-core Drawer with Physics Axial Blankets	36
15.	Approach-to-critical Loadings for Engineering Core	38
16.	Reference Critical Loading of Engineering Core	39
17.	Control-rod Calibrations in the Engineering Core with Physics Blankets	40
18.	Engineering-core Drawers with Engineering Axial Regions	42
19.	Axial Schematic Diagram of Assembly 44 with Engineering Core and Engineering Axial Regions	45
20.	Core Configuration of Four-rod Engineering Mockup	46
21.	Details of Mockup Safety-Control Rod Used in Four-rod Engineering Mockup of RAPSODIE.	47
22.	Distribution of Substitution Drawers for Measurements of Material Worths in Core	51
23.	Core Zones for Regional Worths of Sodium Experiments	52
24.	Drawer Loadings for Traverse Experiments	55
25.	Worths of Pu, U^{235} , and U^{238} versus Axial Position in Engineering Mockup	56

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
26.	Worths of Pu, U^{235} and U^{238} versus Radius in Engineering Mockup	58
27.	Fission Rates of U^{235} and U^{238} versus Axial Position in Engineering Mockup	60
28.	Pu^{239} Fission and B^{10} Capture Rates versus Axial Position in Engineering Mockup	60
29.	Fission Rates of U^{235} and U^{238} versus Radius in Engineering Mockup.	62
30.	Pu^{239} Fission and B^{10} Capture Rates versus Radius in Engineering Mockup	62
31.	Core Configuration for Rossi-alpha Measurement	63
32.	Rossi-alpha Measurements in Assembly 44	63
33.	Axial Configuration for Mockup RAPSODIE Control Rods and Safety Rods in "Banked" Position.	64
34.	Cross Sections of Mockup RAPSODIE Safety Rod in "Banked" Position in M-17 Channel.	65
35.	Cross Sections of Mockup RAPSODIE Control Rod in "Banked" Position in O/P-12 Channel	65
36.	Core Configuration for Six-rod Engineering Mockup of RAPSODIE	67
37.	Radial Fission Rate Distributions through Mockup Control-rod Channels.	69
38.	Relative Depressions of Fission Rate Distributions through Mockup Control-rod Channels	70

LIST OF TABLES

<u>No.</u>	<u>Title</u>	<u>Page</u>
I.	Regional Compositions of RAPSODIE Reactor	13
II.	Composition of Physics-core Unit Cell.	20
III.	Blanket Loadings for Physics Core of Assembly 44	22
IV.	Specifications of Absolute Fission Chambers Used in Assembly 44	30
V.	Fission-ratio Measurements at Center of Physics Core . .	31
VI.	Fission-ratio Measurements at Radial Edge of Physics Core.	32
VII.	Results of Uranium Foil Irradiations in Physics Core. . . .	33
VIII.	Worths of Materials in Small Samples at Center of Physics Core.	34
IX.	Worths of Materials in 2 x 2 x 2 in. Samples at Center of Physics Core.	34
X.	Local Worths of Fuel in Physics Core	35
XI.	Composition of Engineering-core Unit Cell.	37
XII.	Compositions of Axial Regions of Engineering Mockup . . .	43
XIII.	Reactivity Worths of Engineering Axial Regions in RAPSODIE Mockup.	44
XIV.	Specifications of Rod Channel Loadings for Four-rod Mockup of RAPSODIE	47
XV.	Reactivity Worths of Mockup Safety-Control Rods in Four-rod Engineering Mockup of RAPSODIE.	48
XVI.	Distributed Worths of Materials in Engineering Core	51
XVII.	Worths of Steel and Sodium in Axial End Gaps.	52
XVIII.	Regional Worths of Sodium in Engineering Core.	53
XIX.	Worths of Materials at Center of Engineering Core	53
XX.	Axial Traverses of Uranium and Plutonium Reactivity Samples through Engineering Mockup.	56
XXI.	Radial Traverses of Uranium and Plutonium Reactivity Samples through Engineering Mockup.	57
XXII.	Axial Tranverses of Fission and BF_3 Counters through Engineering Mockup	59

LIST OF TABLES

<u>No.</u>	<u>Title</u>	<u>Page</u>
XXIII.	Radial Traverses of Fission and BF_3 Counters through Engineering Mockup	61
XXIV.	Specifications of Rod Channel Loadings for Six-rod Mockup of RAPSODIE	66
XXV.	Reactivity Worths of Simulated Control- and Safety-rod Motions in Six-rod Mockup of RAPSODIE	68
XXVI.	Foil Irradiations for Measuring Flux Depression in Mockup Control Rod	69

CRITICAL STUDIES FOR THE FRENCH FAST REACTOR "RAPSODIE"

(ZPR-3 Assembly 44)

by

A. L. Hess, W. P. Keeney,
P. Caumette, and J. P. Boyer

ABSTRACT

A program of critical studies was carried out on ZPR-3 with a mockup of the French fast reactor, RAPSODIE. Objectives of the program included the acquisition of critical data pertinent to the fuel specification for RAPSODIE, the evaluation of the RAPSODIE design control systems, and the collection of other data useful in verifying calculational techniques. The RAPSODIE core simulated in the mockup was approximately a 40-liter cylinder containing 40 v/o fuel (a $\text{PuO}_2\text{-UO}_2$ mixture), 34 v/o sodium, and 20 v/o steel. The experiments were done in two phases: a physics-core study, and an engineering-core study.

For the physics-core study, a simple core-blanket arrangement was constructed for ease of calculation, and spectral indices and local reactivity coefficients were measured. In the fuel for the physics-core mockup, the ratio of uranium to plutonium was about 5:1, with the uranium enriched to 60 a/o. Both radial and axial blankets had a composition equivalent to about 45 v/o uranium, 30 v/o sodium, and 15 v/o steel.

In the fuel for the engineering core, the uranium-to-plutonium ratio was reduced to 3.2:1, with the uranium 62 a/o enriched. Mockups of the RAPSODIE axial regions, steel-sodium end gaps, and different axial blankets were constructed with the engineering core, along with mockups of the RAPSODIE control and safety rods. Experiments in the full engineering mockup included rod-worth studies, material-reactivity-coefficient measurements, and traverses of neutron detectors and fuel-reactivity samples.

This was the first ZPR-3 critical assembly to contain a significant loading of plutonium, and the problems introduced thereby were investigated. Heat generation from alpha decay of the plutonium affected the accuracy of reactivity measurements, and the neutron source from spontaneous fission required different techniques for measuring reactor periods.

I. INTRODUCTION

Assembly No. 44 in Argonne National Laboratory's Zero Power Reactor 3 (ZPR-3) was a program of studies in support of the design of the French fast reactor RAPSODIE. A geometric and compositional mockup of the RAPSODIE reactor was constructed, and, as in similar ZPR-3 mockups of power reactors, the critical neutronic parameters of the system were investigated.

This mockup was the first ZPR-3 critical assembly to contain appreciable amounts of plutonium. In addition to the RAPSODIE experiments, the problems introduced into ZPR-3 operations by the plutonium loading were studied.

The experiments were conducted from September 20, 1963, to May 20, 1964, at the ZPR-3 facility located at the National Reactor Testing Station in Idaho.

II. DESCRIPTION OF RAPSODIE

RAPSODIE is a sodium-cooled, fast reactor under construction at Cadarache in southern France. Its purpose is to investigate the technological problems associated with the large, power-producing fast reactors envisaged for the future. Development and testing of fuels are among its objectives, and the PuO_2 concentration in the PuO_2 - UO_2 fuel mixture specified for RAPSODIE is approximately that expected in large, oxide-fueled reactors. The techniques and problems of liquid-metal cooling also will be studied. In RAPSODIE, the primary coolant system contains sodium and is intended to accommodate a design power level of 20 MWt. A secondary, NaK system, will cool to the atmosphere. The general construction of the nuclear plant of the system is shown in Fig. 1.

The description given herein is of the RAPSODIE design as visualized at the start of the mockup in ZPR-3; this formed the design criteria of the critical experiments.

The reactor, a cylinder with a height and diameter of about 130 cm contained within the primary vessel, is constructed from hexagonal subassemblies with center-to-center spacings of 5.08 cm. The cylindrical core, 34 cm long, is surrounded radially and axially by a breeder-type blanket containing natural uranium. Coolant provisions allow for a core loading of up to 53 fuel subassemblies, which gives an average core radius of 19.4 cm and a core volume of 40 liters. Figure 2 illustrates the arrangement of the core and radial blanket subassemblies.

ID-103-9613

Fig. 1. Vertical Section of RAPSODIE Reactor

ID-103-9614

Fig. 2. Cross Section of RAPSODIE Reactor

In the core section of each fuel subassembly are 37 stainless steel-clad fuel pins, 6.7 mm in diameter, containing $\text{PuO}_2\text{-UO}_2$ pellets to a length

Fig. 3. Axial Configuration of RAPSODIE Reactor

a radius of about 110 cm, with steel reflector subassemblies surrounding the blanket, as indicated in Fig. 2. Table I gives the compositions of the various RAPSODIE regions shown in Fig. 3.

Reactor control is provided by six boron carbide poison rods located in channels at the radial edge of the core (Fig. 2). Two rods contain natural boron carbide and are designated for control of reactivity during operations. The other four are safety rods, loaded with enriched (90 a/o B^{10}) boron carbide. During operation, the safety rods are positioned in the upper axial blanket and, for rapid shutdown, dropped into the core by spring-assisted gravity.

The objectives and further design details of the RAPSODIE reactor have been reported by Wustner¹ and by Vendryes.²

of 34 cm plus a void at the top for containment of fission gases. The fuel mixture will be approximately 20 v/o PuO_2 , with the possibility of varying the enrichment in the UO_2 according to the criticality demands. For the ZPR-3 experiment, an enrichment of 60 a/o U^{235} was chosen. Above and below the core section in a fuel subassembly are the axial blanket sections containing seven fertile pins, 14.5 mm in diameter. These fertile pins contain a uranium-molybdenum alloy, with sodium bonding to a steel cladding, for a length of 23.5 cm, and again space at the top for gas containment. Thus, an axial schematic diagram of the reactor depicts the core with end gaps (containing sodium and steel) above and below, and uranium blanket beyond the gaps. This is shown in Fig. 3.

The radial blanket subassemblies each contain seven, 16-mm-diam fertile pins containing 99-cm lengths of uranium-molybdenum alloy in steel cladding. Blanket subassemblies are loaded around the core to

TABLE I. Regional Compositions of RAPSODIE Reactor

Region	Composition, v/o at 250°C				
	Fuel	Fertile	Steel	Sodium	Helium
Core	42.9		23.0	33.2	42.2
Radial Blanket		58.1	15.7	26.2	
Axial Blanket		44.8	20.2	35.0	
Lower End Gap			35.9	49.9	
Upper End Gap A			24.8	33.0	
Upper End Gap B			52.5	47.5	
Top Gap			50	50	
Bottom Gap			35	65	
Steel Shield			90	10	

Fuel: 18 v/o PuO_2 , 82 v/o UO_2 .

Practical density: PuO_2 , 10.7 g/cc; UO_2 , 10.3 g/cc.

Atoms U/atoms Pu: 4.43.

Uranium enrichment: 60%

III. DESCRIPTION OF ZPR-3

The ZPR-3 (Fig. 4) is a fast-reactor critical facility operated by the Idaho Division of the Argonne National Laboratory and located at the National Reactor Testing Station in Idaho. The machine consists of a fixed and a movable half, each composed of a 31×31 array of horizontal steel tubes 2.18 in. (5.54 cm) square and 33.5 in. (85 cm) long. Drawers containing various combinations of material are loaded according to a pattern into the matrix tubes to achieve the desired core-blanket configuration for criticality when the machine halves are brought together. Separability of the machine halves assures reactor safety during shutdown and provides ease in the changing of loadings.

Many materials are available for loading into the ZPR-3 drawers, including most of the fissile and structural materials contemplated for fast power reactors. They are principally in the form of thin plates, $1/16$, $1/8$, or $1/4$ in. thick, 2 in. wide, and $1/2$ to 3 in. long. Figure 5 shows typical loadings of a few drawers. Larger blocks of the structural and blanket materials are used in the regions remote from the core. The versatility available in the arrangements of the drawers allows the construction of a great variety of reactor compositions and enables the building of close geometric mockups of regions of power reactors.

Reactivity control and safety shutdown in the reactor is provided by one control rod and four safety rods in each assembly half. These are movable drawers normally loaded with core composition or reflector. In

ID-105-2157

Fig. 4. View of ZPR-3 Critical Facility

ID-103-2231

Fig. 5. Typical Core and Axial Blanket Drawers

the event of a scram they are ejected from the assembly. During start-up, two polonium-beryllium neutron sources are inserted in the assembly, the eight safety rods are inserted, and the halves are brought together. The control rods are then inserted, and after the reactor attains sufficient power, the sources are removed and "critical positions" found on the control rods. Normal operation with U^{235} -fueled cores is at approximately one watt. This provides a minimal fuel irradiation, so that after shut-down, drawers of the assembly may be handled without undue personnel exposure.

The construction, instrumentation, and operations of ZPR-3 have been described more completely elsewhere.³

IV. OBJECTIVES OF THE STUDIES

A. General

The experimental studies for RAPSODIE were instituted by Argonne National Laboratory under directives from the U. S. Atomic Energy Commission in accordance with an agreement between the USAEC and the French Commissariat A l'Énergie Atomique. The principal objectives of the studies were established by personnel at the Centre d'Études Nucléaires de Cadarache, where the RAPSODIE reactor will be built. Details of the experimental program were worked out jointly by the Cadarache group, F. W. Thalgott of ANL Idaho Division, and the ZPR-3 staff. Coauthors P. Caumette and J. Boyer were assigned to ZPR-3 from Cadarache for the duration of the studies to aid in the planning of experiments and analysis of the results.

A primary program objective was the collection of integral data, such as critical size, fuel reactivity coefficients, etc., for use in selecting fuel parameters for the final RAPSODIE design. Principal aspects of the $\text{PuO}_2\text{-UO}_2$ fuel specifications which could be varied are the fuel volume fraction, the Pu/U ratio, and the enrichment of the uranium. These parameters would have to be chosen so as to assure operational criticality within the core-volume limitation of 40.3 liters imposed by the RAPSODIE design. In the mockup it was possible to study a core composition approximating that of RAPSODIE and then to measure uranium, plutonium, and oxygen reactivity coefficients, which provide an evaluation of the effects of varying these fuel parameters. Furthermore, the experimental results could be used as empirical checks of the computational techniques used for RAPSODIE design studies.

Secondly, studies were required for the evaluation of the neutronic control and safety systems specified in the RAPSODIE design. Also, in regard to safety, measurements were requested of the reactivity effects of the sodium coolant of RAPSODIE in various regions of the core.

To these ends, the experimental program was divided into two phases. The first was a study of a physics core, a simple core-blanket configuration without features of the power reactor such as gaps and rod channels, amenable to calculation by computer codes. The second phase was the construction of an engineering mockup of RAPSODIE, including the various gaps, control rods, and channels, and other design details of the power reactor.

B. Physics-core Program

The basic purpose of the physics-core experiments was to obtain a critical core, with a composition approximately appropriate to RAPSODIE, and a simple radial and axial blanket configuration, so that comparison of results with calculations in simple geometry would have greater validity.

In this phase, both radial and axial blankets had the same composition, simulating that of the radial blanket in RAPSODIE. Besides the critical volume determination, several neutron spectral indices were obtained and the fuel reactivity effects were measured at various core positions. Included in the spectral indices experiments were measurements of fission ratios at the core center and core radial edge and measurements of reactivity coefficients for a number of materials at the core center.

As this was the first ZPR-3 critical assembly to contain a substantial amount of plutonium, additional investigations were required into the new problems encountered in reactor operation due to the heat generation and spontaneous neutron source from the plutonium.

C. Engineering-mockup Program

The second phase of the studies was to construct a detailed mockup of the RAPSODIE reactor. This would include the sodium-steel axial end gaps above and below the core, the different axial blankets, and simulations of the B_4C control- and safety-rod channels of RAPSODIE. It was possible also that, based on the results of the physics-core work, a fuel-composition adjustment would be necessary to keep the core volume within the RAPSODIE limitation. Beside determining the reactivity worths of the engineering features mentioned, the mockup program also called for extensive measurements of reactivity coefficients, measuring the reactivity effects of simulated RAPSODIE B_4C control- and safety-rod motion, and determinations of radial and axial reaction-rate distributions. The reactivity coefficients measured included the distributed worths of materials in the core and end gaps, to enable evaluation of reactivity effects due to composition discrepancies, and worths of sodium versus radial and axial positions in the core.

V. PHYSICS-CORE EXPERIMENTS

A. Construction of the Assembly

1. Core-drawer Loading

A core-drawer loading pattern was chosen which, on the basis of preliminary calculations, would yield a critical volume of 43 liters for a similar composition core in the RAPSODIE reactor. Several compromises had to be accepted in the loading as far as accurately mocking up a possible RAPSODIE core composition. First, since no oxides of uranium or plutonium were on hand, the RAPSODIE $\text{PuO}_2\text{-UO}_2$ fuel was simulated with a combination of enriched uranium, depleted uranium, and plutonium metal plates (Pu-1 w/o Al canned in steel) plus plates of Fe_2O_3 and Al_2O_3 . Then, with the inclusion of these oxide plates restricting the space available for sodium, the sodium was simulated with the aluminum in the Al_2O_3 and additional aluminum plates.

The core axial length was chosen as 14 in. (35.56 cm), consisting of 8 in. (20.32 cm) in the drawers of Half No. 1 and 6 in. (15.24 cm) in the opposing drawers of Half No. 2, the two sections together composing a unit core-cell with a volume of 1.094 liters (including 0.06 in. of drawer fronts). The sixteen 1/8-in. columns of each drawer were made up of $2\frac{1}{4}$ columns of enriched uranium, $1\frac{1}{2}$ columns of plutonium, $1\frac{1}{4}$ columns of depleted uranium, four columns of Fe_2O_3 , four columns of Al_2O_3 , and three columns of 45 v/o aluminum (perforated plates). These were arranged as shown in Fig. 6 for a drawer in Half No. 2. The three central columns of 45 v/o aluminum were placed adjacent to each other to provide a core portion for substitution experiments. The resulting core-cell composition is given in Table II.

In terms of simulating a RAPSODIE core composition, the physics-core unit cell composition was equivalent to approximately 40 v/o fuel ($\text{PuO}_2\text{-UO}_2$), 20 v/o steel, and 34 v/o sodium. In the fuel, the atom ratio of uranium to plutonium was 5.12, and the average uranium enrichment was 60 a/o U^{235} .

2. Blanket Loading

To simplify calculations, a uniform blanket, in both the radial and axial directions, was designed for the physics core. The composition approximated that of the RAPSODIE radial blanket. In the radial and axial regions adjacent to the core, a fine mockup blanket was used. This was constructed with 1/8-in.-thick plates and extended out about 5 in. from the core boundaries. The drawers of this fine blanket contained nine columns of depleted uranium, two columns of molybdenum, one column of aluminum, one column of 63 v/o aluminum, two columns of 45 v/o aluminum, and one column of stainless steel. Here the combination of aluminum was to

Fig. 6. Core-drawer Loading for Physics Core

TABLE II. Composition of Physics-core Unit Cell

Cell dimensions: 2.18 x 2.18 x 14.06 in.

Cell volume: 1.0944 liters

Material	Cell Composition		
	Density, g/cc	10^{22} atoms/cc	v/o ^a
Pu ²³⁹	0.6185	0.1558	3.24
Pu ²⁴⁰⁺²⁴²	0.0296	0.0074	0.15
Pu ²⁴¹	0.0027	0.0007	0.01
U ²³⁵	1.9651	0.5034	10.49
U ²³⁸	1.3013	0.3292	6.86
U ²³⁴⁺²³⁶	0.0283	0.0073	0.16
O	0.5198	1.9567	20.38
Al	0.4608	1.0309	17.10
Stainless steel ^b	0.8961	-	-
Fe	0.6695 ^c	1.4178 ^d	16.7 ^d
Cr	-	0.1868 ^d	2.33 ^d
Ni	-	0.0920 ^d	1.01 ^d

^aBased on 100% atomic densities listed for 16-group, fast cross-section set in ANL 5800,⁵ p. 559.

^bType 304, assumed composition 72 w/o Fe, 18 w/o Cr, 10 w/o Ni.

^cIron content from Fe₂O₃.

^dTotal contributions from steel and Fe₂O₃.

simulate sodium in coolant channels. The arrangement of these materials is shown in the fine axial blanket region behind the core section of Fig. 6 and also in the loading of the radial fine-blanket drawer shown in Fig. 7a. Table III contains the resulting compositions.

In the blanket regions further from the core, larger blocks of materials were used. A medium blanket surrounded the fine type, and beyond this was a coarse blanket. These outer regions contained no molybdenum, but had higher concentrations of uranium. The material arrangements are shown in Figs. 7b and 7c.

Before the loading of the core, the various blanket regions were constructed in the machine in the patterns illustrated by Figs. 8 and 9. Table III gives the compositions by region of the different blankets.

3. Approach to Critical

On the basis of preliminary calculations, the expected critical loading for the physics core was about 33 core unit cells, or 33 core drawers per machine half. In the initial loading, the core contained five unit cells,

Depleted U	
Depleted U	
Depleted U	
Mo	
Depleted U	1x1x5 in. Depleted U
63 v/o Al	over
Steel	1x1x5 in. Depleted U
45 v/o Al	
45 v/o Al	
Depleted U	1x1x5 in. Depleted U
100 v/o Al	over
Depleted U	
Mo	
Depleted U	1x1x5 in. 56 v/o Al
Depleted U	
Depleted U	

(a) Fine Radial Blanket Drawer for Half #2

1x1x5 in. 56 v/o Al over 1x1x5 in. Depleted U	1x1x5 in. Depleted U over 1x1x5 in. Depleted U	Pattern repeated to 25 in. in Half #1 to 24 in. in Half #2 →
1x1x5 in. Depleted U over 1x1x5 in. Depleted U	1x1x5 in. Depleted U over 1x1x5 in. 56 v/o Al	

(b) Medium Radial Blanket, loaded directly into matrix
(Medium Axial Blanket Similar)

1/4x2x2in 100v/o Al 1/4x2x2in 100v/o Al		
1x1x2 in Depleted U	2 x 2 x 5 in. Brick Depleted U	Pattern repeated to 28 in. in both halves →
1/4 x2x2in 100v/o Al 1/4 x2x2in 100v/o Al		

(c) Coarse Radial Blanket, loaded directly into matrix

Fig. 7. Fine, Medium, and Coarse Blanket Drawer Loadings, Assembly 44

TABLE III. Blanket Loadings for Physics Core of Assembly 44

Composition, g/cc	Blanket Type				
	Fine Axial (ABF)	Medium Axial (ABM)	Fine Radial (RBF)	Medium Radial (RBM)	Coarse Radial (RBC)
Depleted uranium ^a	8.566	11.794	8.590	11.791	12.427
Molybdenum	1.037	-	1.035	-	-
Stainless steel ^b	1.098	0.707	1.108	0.574	0.574
Aluminum	0.344	0.310	0.344	0.310	0.312
Composition, 10 ²² atoms/cc					
U ²³⁸	2.162	2.976	2.168	2.975	3.136
U ²³⁵	0.005	0.007	0.005	0.007	0.007
Mo	0.651	-	0.650	-	-
Al	0.768	0.692	0.768	0.692	0.696
Fe	0.852	0.549	0.860	0.446	0.446
Cr	0.229	0.147	0.231	0.120	0.120
Ni	0.113	0.072	0.114	0.059	0.059

^a0.216 w/o U²³⁵.^bType 304, nominally 72 w/o Fe, 18 w/o Cr, 10 w/o Ni.

Fig. 8. Radial Blanket Configurations in Assembly 44

Fig. 9. Axial Schematic Diagram of ZPR-3 Assembly 44 with Physics Core

Fig. 10. Approach-to-critical Curves for Physics Core

the other intended fuel drawers being temporarily loaded with blanket. The approach to critical was made by building the core out radially with successive replacements of radial blanket with core drawers. Criticality was attained with a loading of 35 core drawers in Half No. 1 and 36 core drawers in Half No. 2, or approximately 35.4 unit cells.

The variations of the sub-critical multiplication with loadings are indicated in the approach-to-critical curves of Fig. 10. Here the plots of reciprocal power versus loading show more curvature near criticality than usually observed in ZPR-3 assemblies. Presumably, the addition of a source along with the fuel accounts for the nonlinearity. The ratio of loading-to-power versus loading provides more

consistent extrapolations, although placement of detectors too near the external sources (Po-Be) can give overestimates of allowable loading with this method. On the graph, the extrapolations to just-critical average out to 35.33 unit cells.

B. The Critical Experiment

1. Core Configuration

Figure 11 shows interface views of the core region of the assembly with the reference critical loading for the physics core. This configuration had 35 Half No. 1 drawers and 36 Half No. 2 drawers loaded with fuel, or essentially 35-6/14 core unit cells. The total fuel loading was 25.24 kg of plutonium, 76.19 kg of U^{235} . On the basis of later control-rod calibrations, this reference configuration had an available excess reactivity of 40 lh if all ten ZPR-3 control rods were inserted.

Fig. 11. Reference Critical Configuration for Physics Core

2. Fuel Worth at Edge of Core

Measurements were next made of the reactivity worth of a core unit cell replacing blanket at the radial edge of the core. By a series of drawer replacements, the Half No. 2 core drawer in 2-Q-13, was found to be worth 179 ± 4 lh, and a Half No. 1 core drawer in 1-O-19, was worth 261 ± 5 lh. Thus the net worth of a core unit cell at this location on the core edge was 440 ± 7 lh. For an average core-edge worth, the uncertainty is probably ± 30 lh.

initiating the period and before accumulating data. Usually the necessary reactivity addition is completed at around 50 mW, which would allow data collection starting at 10 W, and would require going to 200 W to extend the measurement over three periods. This power is at about the upper limits desired for ZPR-3 operations. In a reactor without the spontaneous neutron source, as in uranium-fueled assemblies, the waiting time would be about 90 sec so that the measurement could be made between 0.2 and 4 W.

Period measurements were done in Assembly 44 in the conventional manner, running to peak powers of about 150 W, but rod calibration methods were adopted that required a minimal number of periods. In the future, plutonium-fueled assemblies with greater Pu^{240} content and lower specific powers, the determination of reactivities from kinetic experiments will require methods accounting for source effects.

2. Control-rod Calibrations

In previous ZPR-3 assemblies, control rods had been calibrated by measuring periods for the reactivity worths of successive segments of the rods. Because of the high-power runs necessary for accurate period measurements with this assembly, the relative variation of rod worth versus position was determined by other methods, obtaining an absolute calibration with only a few period measurements.

In the physics core, the relative calibrations of the control rods were determined by subcritical methods. A rod was withdrawn in steps, steady power count rates being measured with neutron detectors for the successive rod positions. Two radial blanket-loaded rods (those designated as Nos. 3 and 8 in Fig. 11) and control rod No. 10 (which was loaded with fuel) were calibrated. For the overall calibrations, the method used was only fairly satisfactory in determining the shapes of relative-worth curves. Therefore, when the absolute calibrations were made, six period measurements were taken, over various sections of the rods, as a check on the subcritical worth measurements. The data from the period experiments were used to establish the absolute calibration of the rods for successive 2-in. intervals, the subcritical results providing details of the worth variations within the intervals. As discussed in Section VI, A-5, this method was modified in the engineering core to further reduce the number of period measurements necessary for the rod calibration. Figure 12 is a plot of the control-rod calibrations in the physics core.

3. Temperature Profile and Drift

Plutonium loading in a zero-power critical facility introduces a temperature-stability problem due to the heat generation from alpha decay. The particular isotopic mixture of the ZPR-3 plutonium plates has a heat output of 2.2 W/kg of plutonium, giving a total generation in the physics core

Fig. 12. Control-rod Calibrations with Physics Core

carried out in a repetitive pattern. Six thermocouples were positioned around the central matrix tubes (1- and 2-P-16) to indicate the axial temperature distribution in the core, and thermocouples placed at the midplane two and three matrix units away from the core center provided a radial temperature profile. The reactor room air temperature and the temperature of the frame around the matrix tubes were also measured. The reactivity effects due to temperature changes could be separated approximately into two components, interpreted as radial and axial expansion effects. While the axial expansion could be related to the changes in the axial temperatures, it appeared that the radial dimensions of the assembly were governed principally by the temperature of the frame. Also, the temperature coefficients seemed to depend on whether the changes occurred with the machine halves together or apart.

The reactivity and temperature differences among several repeated references in the physics core were studied. The correlations obtained were $-1.2 \pm 0.2 \text{ Ih}/^\circ\text{C}$ change of frame temperature and $-0.7 \pm 0.3 \text{ Ih}/^\circ\text{C}$ change in average axial temperature. These factors were used to calculate corrections for temperature effects in all the physics-core worth measurements, and a major part of the errors quoted for the measurements are due to the uncertainties in the corrections applied.

of about 57 W. An uneven temperature profile in the reactor results and, with the machine halves together, a temperature drift with an associated reactivity drift is observed. A special study had been made of these temperature distributions in a subcritical assembly, Assembly 37.⁶ In this study, with a loading of 24 kg of plutonium in one machine half and half-closure simulated with an insulating plate on the interface, at least four days were required before a temperature equilibrium was attained. Thus in day-to-day or daily operation of the facility, loading changes and opening and closing the reactor halves disturbs the temperature profile and introduce somewhat unpredictable reactivity effects.

Some semblance of a consistent relationship between temperatures and reactivity in the Assembly 44 experiments could be obtained when the operations were

Reactivity drift during extended operation, i.e., for traverses or sample changer experiments, was better correlated with temperature drift. The actual time variation of reactivity due to heating could be determined by periodically repeating "critical" rod-position measurements for a reference configuration. Such investigations during traverse experiments in the engineering core indicated an average core-temperature rise of about 0.22°C/hr . Depending on the variation of the frame temperature, the reference reactivity changed by 0 to -0.6 Ih/hr .

Fig. 13. Axial Temperature Profiles during Extended Run

coefficient of $-0.31 \text{ Ih}/^{\circ}\text{C}$, which is less than half that cited previously for axial expansion of the entire core.

The difficulties introduced into ZPR-3 experiments because of heat generation in a plutonium loading have been well exhibited during experiments with this assembly, and methods of counteraction are suggested thereby. Although studies have shown that forced air-cooling of a plutonium loading is not necessary to prevent heat damage to materials, such cooling could greatly minimize temperature drift and improve the accuracy of worth measurements. Persistence of a temperature variation then would require the altering of experimental techniques to enable evaluation of the associated reactivity drift. The use of more thermocouples distributed in the reactor and improved instrumentation for temperature readout could also reduce the uncertainties concerning temperature variations.

D. Spectral Indices

Ratios of fission and capture rates of the various plutonium and uranium isotopes were measured at a few positions in the physics core to provide some characterization of the neutron spectra. These spectral indices were obtained using fission chambers and foil irradiations.

A long run was made during the engineering-core experiments specifically to study temperature drift. Nine central drawers of each half were removed overnight and replaced before the run, and the variation of the axial temperature profile with time after half-closure is shown in Fig. 13. The average drift of the axial temperatures, principally for the nine central core cells was about $+1.4^{\circ}\text{C/hr}$, and this produced a drift of -0.44 Ih/hr in reactivity. This gives an axial temperature

Absolute fission counters of the design described by Kirn⁷ and Davey⁸ were used to measure fission ratios at the center and edge of the core. Basically, these counters consist of a cylindrical chamber, 2 in. in diameter by 1 in. long, with a thin, 1.5-in.-diam plating of fissile material on one inner face. A collector electrode is mounted parallel to and about $\frac{1}{4}$ in. from the plating, and the gas filling is an argon-methane mixture. Chambers with two variations in design were used in the experiments. In one type, the original counters constructed by Kirn, the average thickness of the chamber walls is about 0.10 in. and the chambers are sealed. In the other type, designated thin-walled gas-flow (TWGF), the thickness averages about 0.02 in. Specifications of the fissile platings of the counters used are given in Table IV.

TABLE IV. Specifications of Absolute Fission Chambers Used in Assembly 44

Counter or Foil	Fissile Plating Mass, ^a μg	Isotopic Composition, w/o						
		U ²³³	U ²³⁴	U ²³⁵	U ²³⁶	U ²³⁸	Pu ²³⁹	Pu ²⁴⁰
Kirn No. 16	498	98.33	0.13	0.01	-	1.53	-	-
Kirn No. 11	496	-	93.44	4.87	-	1.69	-	-
Kirn No. 4	802	-	1.07	93.41	-	5.52	-	-
Kirn No. 5	804	-	1.07	93.41	-	5.52	-	-
Kirn No. 24	777	-	0.06	2.85	96.48	0.62	-	-
Kirn No. 2	499	-	-	0.04	-	99.96	-	-
Kirn No. 19	493	-	-	-	-	-	99.97	0.03
Kirn No. 6	401	-	-	-	-	-	20.17	79.83
Kirn No. 12	398	-	-	-	-	-	20.17	79.83
TWGF No. 24	503.8	99.92	0.01	0.01	-	0.06	-	-
TWGF No. 39	528.8	-	97.56	1.86	0.01	0.58	-	-
TWGF No. 3	416	-	1.04	93.22	0.28	5.46	-	-
TWGF No. 27	397.2	-	0.06	2.85	96.48	0.62	-	-
TWGF No. 14B	493.2	-	0.01	0.72	-	99.27	-	-

^aPlating mass as determined from residue of electroplating process, estimated accuracy $\pm 5 \mu\text{g}$.

1. Fission-chamber Experiments at Center of Physics Core

For the central fission ratio measurements, a Half No. 1 core drawer, 1-P-16, was modified to enable the positioning of a fission chamber at the front. Various fission chambers in turn were set in this position, and their counting rates were determined relative to that of a standard counter during a run. The standard used was Kirn counter No. 5 (U²³⁵), which was permanently positioned at the front of drawer 1-Q-15. The first column of Table V lists the chambers used, and the third column gives the count rates obtained, each rate an average of several trials, the error quoted including counting statistics and uncertainties of dead-time corrections applied.

TABLE V. Fission-ratio Measurements at Center of Physics Core

Chamber	Principal Isotope	Counts per 10 ⁴ Standard Counts	Relative Central Count Rate	Derived Central Fission Ratios		
				Ratio	Experimental Value ($\pm 2\%$)	Adjusted Value ($\pm 3\%$)
Kirn No. 4	U ²³⁵	18567 $\pm$ 60	1.0000	-	-	-
Kirn No. 16	U ²³³	18282 $\pm$ 60	0.9846	$\sigma_f\text{-U}^{233}/\sigma_f\text{-U}^{235}$	1.506	1.506
Kirn No. 11	U ²³⁴	5534 $\pm$ 17	0.2981	$\sigma_f\text{-U}^{234}/\sigma_f\text{-U}^{235}$	0.431	0.448
Kirn No. 24	U ²³⁶	3419 $\pm$ 16	0.1841	$\sigma_f\text{-U}^{236}/\sigma_f\text{-U}^{235}$	0.156	0.165
Kirn No. 2	U ²³⁸	907 $\pm$ 4	0.0489	$\sigma_f\text{-U}^{238}/\sigma_f\text{-U}^{235}$	0.0746	0.0806
Kirn No. 19	Pu ²³⁹	14028 $\pm$ 53	0.7555	$\sigma_f\text{-Pu}^{239}/\sigma_f\text{-U}^{235}$	1.179	1.179
Kirn No. 6	Pu ²⁴⁰	6013 $\pm$ 20	0.3238	$\sigma_f\text{-Pu}^{240}/\sigma_f\text{-U}^{235}$	0.482	0.501
TWGF No. 3	U ²³⁵	9479 $\pm$ 35	1.0000	-	-	-
TWGF No. 39	U ²³⁴	5828 $\pm$ 23	0.6148	$\sigma_f\text{-U}^{234}/\sigma_f\text{-U}^{235}$	0.446	0.446
TWGF No. 27	U ²³⁶	1856 $\pm$ 9	0.1958	$\sigma_f\text{-U}^{236}/\sigma_f\text{-U}^{235}$	0.175	0.175
TWGF No. 14B	U ²³⁸	1025 $\pm$ 6	0.1081	$\sigma_f\text{-U}^{238}/\sigma_f\text{-U}^{235}$	0.0802	0.0802

Two sets of data were obtained, one with Kirn counters containing the uranium and plutonium isotopes, the other with the gas-flow counters (TWGF) containing only the uranium fissile foils. In column four of Table V, the relative central count rates of the Kirn counters were obtained by normalizing to the Kirn No. 4 (U²³⁵) count rate, and the relative rates of the gas-flow counters are with reference to the TWGF No. 3 (U²³⁵) count rate. The overall accuracy of these relative count rates is about $\pm 0.5\%$. Ratios of the fission cross sections of the principal isotopes in the counter to the U²³⁵ fission cross section were derived by solution of the simultaneous equations given by the data sets of both the Kirn counters and the TWGF counters. These derived fission ratios are listed in the sixth column of Table V and have an estimated accuracy of about $\pm 2\%$, reflecting uncertainties in the specifications of the fissile foil platings and in the relative counter efficiencies.

The last column in Table V (Adjusted Values) contains fission ratios after corrections for the effects of flux perturbations by inelastic scattering in the walls of the chambers. The spectral changes and corrections necessary have been investigated by Davey.⁸ The effects are primarily important in the heavy-walled Kirn counters containing threshold fissile isotopes. For the Kirn counter fission ratios, the adjustments applied were +8% for the U²³⁸/U²³⁵ ratio, +6% for the U²³⁶/U²³⁵ ratio, and +4% for the U²³⁴/U²³⁵ and Pu²⁴⁰/U²³⁵ ratios. For the other isotopes and for the thin-walled, gas-flow counters, the effects of the detectors on the spectral indices are considered unimportant. Overall, the limit of accuracy on the fission ratios reported is probably 3 to 4%, considering the calibration uncertainties and the possible differences between the neutron spectrum incident on the fissile platings and the average spectrum in the reactor region.

2. Fission-chamber Experiments at Radial Edge of Physics Core

In the same manner as for the central fission ratios, relative count rates of various fission chambers were measured at the radial edge

of the core. The counters were positioned at the front of 1-S-16. Thin-walled, gas-flow chambers were used with foils of the uranium isotopes, while for the plutonium ratios only Kirn chambers were available. Table VI lists the counters and the experimental results. For the relative core-edge count rates (column 4), all the data are normalized to the results with the TWGF No. 3 counter. The fission ratios derived are given in the sixth column, and only the $\text{Pu}^{240}/\text{U}^{235}$ fission ratio required a wall-thickness correction (+4%).

TABLE VI. Fission-ratio Measurements at Radial Edge of Physics Core

Chamber	Principal Isotope	Counts per 10^4 Standard Counts	Relative Edge Count Rate	Derived Core-edge Fission Ratios		
				Ratio	Experimental Value ($\pm 2\%$)	Adjusted Value ($\pm 3\%$)
TWGF No. 3	U^{235}	6228 ± 22	1.0000	-	-	-
TWGF No. 24	U^{233}	12199 ± 50	1.9586	$\sigma_f \text{U}^{233}/\sigma_f \text{U}^{235}$	1.508	1.508
TWGF No. 39	U^{234}	3158 ± 12	0.5070	$\sigma_f \text{U}^{234}/\sigma_f \text{U}^{235}$	0.363	0.363
TWGF No. 27	U^{236}	968 ± 9	0.1554	$\sigma_f \text{U}^{236}/\sigma_f \text{U}^{235}$	0.133	0.133
TWGF No. 14B	U^{238}	521 ± 3	0.0836	$\sigma_f \text{U}^{238}/\sigma_f \text{U}^{235}$	0.0604	0.0604
Kirn No. 19	Pu^{239}	8739 ± 20	1.4031	$\sigma_f \text{Pu}^{239}/\sigma_f \text{U}^{235}$	1.127	1.127
Kirn No. 12	Pu^{240}	3381 ± 10	0.5429	$\sigma_f \text{Pu}^{240}/\sigma_f \text{U}^{235}$	0.394	0.410

3. Uranium-foil Irradiations

Foils of enriched and natural uranium were irradiated at three radial positions near the reactor midplane to obtain relative fission and capture rates with respect to radial distance. These foils were then analyzed by radiochemical techniques, obtaining fissions per gram by detection of Mo^{99} and captures (in U^{238}) from the Np^{239} activity. Two separate sets of foils were involved, one set for analysis at radiochemical facilities of the Idaho Division, and the other for processing at Argonne, Illinois, where radiochemical analysis of ZPR-3 irradiated foils has usually been done. For the Illinois analysis, the foils were on top of drawers in the P row, and for the Idaho analysis, the foils were at corresponding radii on top of Q-row drawers. The two analyses should give similar results, because of symmetric locations about a radial plane. However, the fissions per gram measured by the two laboratories differ by about 15%, as seen in Table VII. This may be due to flux asymmetries or calibration errors. But the fission ratios derived, and listed in Table VII, would not reflect this error and are probably accurate to $\pm 5\%$.

E. Central Reactivity Coefficients

Substitution experiments were performed at the center of the physics core with samples of a number of materials. For most of the materials, a sample size of 8 cu in. (131 cc) was used; for the more reactive fissile and capture elements, the samples were 1/2 or 1 cu in. (8.2 or 16.4 cc).

TABLE VII. Results of Uranium Foil Irradiations in Physics Core

Foil Set Position ^a	Approximate Radius, in. (cm)	Fissions per gram		Captures per gram in Depleted U ^c Foil	$\sigma_f\text{-U}^{238}/\sigma_f\text{-U}^{235}$	$\sigma_c\text{-U}^{238}/\sigma_f\text{-U}^{235}$
		Enriched U ^b Foil	Depleted U ^c Foil			
1-P-16 ^d	1.1 (2.8)	6.10×10^{10}	5.88×10^9	6.18×10^9	0.090	0.095
1-P-13 ^d	6.62 (16.8)	4.20×10^{10}	3.12×10^9	4.55×10^9	0.069	0.103
1-P-11 ^d	11.0 (27.9)	1.80×10^{10}	5.02×10^8	2.29×10^9	0.024	0.120
2-Q-16 ^{e,f}	1.2 (30.0)	5.35×10^{10}	4.96×10^9	-	0.086	-
1-Q-13 ^e	6.6 (16.8)	3.71×10^{10}	2.57×10^9	-	0.064	-
1-Q-11 ^e	11.0 (27.9)	1.46×10^{10}	-	-	-	-

^aFoils placed on top of drawer, at core midplane.

^b93.2 w/o U²³⁵, 6.8 w/o U²³⁸.

^c0.2 w/o U²³⁵.

^dActivation analysis performed in Illinois laboratory.

^eActivation analysis performed by Idaho Division laboratory.

^fSet about 3.8 cm from midplane.

1. Small Samples

Two series of experiments were done with small sample sizes. In one series, the substitution region was created by removing a $1/8 \times 2 \times 2$ -in. plate at the front of one or two of the central columns of 45 v/o aluminum (see Fig. 6) in drawer 1-P-16. Thus, the region was situated between the plutonium plates in the drawer and ran parallel to the core axis. Plates or cans of various materials were substituted into the region, including the fuel constituents, uranium and plutonium, enriched boron, and polyethelene. Samples of steel and aluminum were also tried, but their worths were of the order of the experimental uncertainties, principally because of temperature variations, and the results are not quoted. Table VIII gives the material reactivity coefficients of the materials as measured from the experiments.

In the other experiments with small samples, the substitution region was formed across the front of 1-P-16 by recessing the core materials in the drawer $1/4$ or $1/2$ in. The sample plates were therefore perpendicular to the core axis and about 1 in. from the center. In addition to remeasuring the uranium worths, we measured the worths of tantalum, carbon, and rhenium. Table VIII includes the results of these experiments.

A difference of 10% was found between the two measurements for the enriched uranium worth. The higher value, for the sample aligned parallel to the axis, is postulated to be due to the proximity of the sample to fuel columns, adding reactivity through a heterogeneity or fuel-bunching effect. As a comparison with computed reactivity coefficients, the results with the perpendicular orientation would be a better representation of the average worth in the core composition. A negative 10% adjustment on the plutonium value given would probably provide a better average worth.

TABLE VIII. Worths of Materials in Small Samples at Center of Physics Core

	Material	Sample Size, in.	Mass, ^(a) g	Worth, ^(b) lh	Reactivity ^(b) Coefficient, lh/kg
Samples aligned parallel to axis	Depleted U ^(c)	1/4 x 2 x 2	295.7	7.4	25.0 ± 2.3
	Enriched U ^(d)	1/8 x 2 x 2	143.9	69.2	481 ± 11
	Plutonium ^(e)	1/8 x 2 x 2	69.0	61.2	887 ± 21
	U ²³³ ^(f)	1/8 x 2 x 2	110.9	102.3	923 ± 16
	Enriched B ^(g)	1/4 x 2 x 2	7.8	-47.2	-6058 ± 140
	CH ₂	1/4 x 2 x 2	15.46	84.3	5453 ± 100
Samples aligned perpendicular to axis	Depleted U ^(c)	1/4 x 2 x 2	295.7	6.7	23 ± 4
	Enriched U ^(d)	1/8 x 2 x 2	143.9	63.0	438 ± 8
	Carbon	1/4 x 2 x 2	25.6	6.4	250 ± 40
	Tantalum	1/4 x 2 x 2	131.3	-11.0	-84 ± 7
	Rhenium	1/2 x 2 x 2	635.4	-76.5	-120.4 ± 2.3

(a) For material listed, excluding mass of canning or clad, if any.

(b) For material alone relative to void, corrected for worth of canning, if any.

(c) 0.22 w/o U²³⁵.

(d) 93.3 w/o U²³⁵.

(e) 95.00 w/o Pu²³⁹, 4.55 w/o Pu²⁴⁰, 39 w/o Pu²⁴¹.

(f) 97.40 a/o U²³³, 2.60 a/o U²³⁸.

(g) 86 w/o B¹⁰, 8 w/o B¹¹, 6 w/o impurities.

2. Large Samples

Worths of the less reactive materials were measured with 2 x 2 x 2-in. samples. The substitutions were made with a sample changer mechanism, which eliminated the opening and closing of the reactor halves for each measurement. A mobile drawer, with a partitioned section at the front for a 2 x 2 x 2-in. sample, is withdrawn by the mechanism to enable remote changing of the sample with the drawer out of the assembly. The drawer was loaded as a regular Half No. 1 core drawer (behind the 2-in. sample section) and positioned for insertion into matrix unit 1-P-16. Table IX lists the samples used along with the measured worths of the materials relative to void.

TABLE IX. Worths of Materials in 2 x 2 x 2-in. Samples at Center of Physics Core

Material	Mass, ^(a) g	Worth, ^(b) lh	Reactivity ^(b) Coefficient, lh/kg	Material	Mass, ^(a) g	Worth, ^(b) lh	Reactivity ^(b) Coefficient, lh/kg
Al	350.5	3.1 ± 0.6	8.8 ± 1.7	O (in Fe ₂ O ₃)			56 ± 3
Al ₂ O ₃	224.2	7.2 ± 0.3	32.0 ± 1.5	Physicum I ^(c)	232	-6.3 ± 0.3	-23.1 ± 1.5
C	196.4	30.7 ± 0.5	156.4 ± 2.5	Physicum II ^(c)	210	-6.9 ± 0.4	-25.7 ± 2.2
Fe	1028.0	-6.2 ± 0.5	-6.1 ± 0.5	Steel ^(d)	1016.5	-5.6 ± 0.4	-5.5 ± 0.4
Fe ₂ O ₃	341.0	4.1 ± 0.2	12.2 ± 0.5	Ta	1004.4	-69.0 ± 1.1	-68.7 ± 1.1
Mo	1279.8	-27.7 ± 0.6	-21.6 ± 0.5	Va	527.2	12.1 ± 0.5	23.1 ± 0.9
Na	90.2	+5.0 ± 0.2	55.4 ± 2.4	W	1867.2	-47.0 ± 0.8	-25.2 ± 0.4
Nb	489.6	-18.0 ± 0.5	-36.7 ± 1.1	Zr	846.0	0.66 ± 0.13	0.78 ± 0.15
O (in Al ₂ O ₃)			58 ± 4				

(a) Mass of material only, excluding mass of canning or clad, if any.

(b) For material alone relative to void, corrected for worth of canning, if any.

(c) Mixtures of fission-product oxides. Compositions given by Long et al.⁹

(d) Type 304 stainless steel, nominally 72% Fe, 18% Cr, 10% Ni.

Because the reactor halves remained together during the experiments, the possible closure error between measurements was eliminated, but reactivity drift was induced because of the gradual temperature escalation in the core. The reference runs, with the sample space void, were repeated from time to time to determine the drift rate and to provide adjustments for temperature effects. In Table IX, the errors quoted represent principally the uncertainties in the temperature corrections.

F. Worths of Fuel versus Position in Core

Substitution experiments were carried out with plates of the fuel constituents at different core locations to determine the variation of fuel worth with radial and axial position. Measurements were taken at two radial locations on the midplane, at approximately half the core radius and at the core radial edge, and at the core axial edge, in addition to the previously cited core-center measurements. For the worths at the midplane near the radial edge of the core, the substitutions were made at the front of one or more of the drawers 1-P-13, 1-P-17, 1-M-16, and 1-S-16. Worths at about half the core radius on the midplane were derived from experiments at the front of drawers such as 1-O-15, 1-O-17, and 1-Q-17. For the worths at the axial core edge, the substitutions were done in the last 2 in., from 6 to 8 in., of the core section in 1-P-16.

In the uranium substitutions, either enriched or depleted, the net worth of the material relative to void was obtained by replacing two $1/8 \times 2 \times 2$ -in. pieces of 45 v/o aluminum (in the central columns of the core drawer at the positions mentioned) with a sandwich of a $1/8 \times 2 \times 2$ -in. plate of the uranium between two $1/16 \times 2 \times 2$ -in. plates of 90 v/o aluminum. Worths of plutonium were found relative to void by substituting an empty can. The experimental results are presented in Table X.

TABLE X. Local Worths of Fuel in Physics Core

Core Location	Sample Position		Reactivity Coefficient, lh/kg (Relative to void)		
	Radial, ^a in.	Axial, ^b in.	Plutonium ^c	Enriched U ^d	Depleted U ^e
Core Center (0 to 2 in. in 1-P-16)	0	-1 to +1	887 ± 21	481 ± 11	25.0 ± 2.3
Half Core Radius at Midplane (0 to 2 in. in 1-O-15)	3.2	-1 to +1	692 ± 7	425 ± 6	22.5 ± 1.1
Radial Edge at Midplane (0 to 2 in. in 1-P-13)	6.6	-1 to +1	340 ± 9	245 ± 5	24.4 ± 1.0
Axial Edge (6 to 8 in. in 1-P-16)	0	+5 to +7	455 ± 6	258 ± 4	30.4 ± 2.7

^aFrom core axis to center of substitution column.

^bFrom core midplane (1 in. from interface of Half No. 1).

^c95.00 w/o Pu²³⁹, 4.51 w/o Pu²⁴⁰⁺²⁴², 0.49 w/o Pu²⁴¹.

^d93.26 w/o U²³⁵, 0.94 w/o U²³⁴, 5.80 w/o U²³⁸⁺²³⁶.

^e0.206 w/o U²³⁵.

VI. ENGINEERING-MOCKUP EXPERIMENTS

A. Critical Experiment with Engineering Core

1. Change of Core Composition

The critical volume of the experimental Assembly 44 physics-core composition was greater than values predicted by calculations used for RAPSODIE design studies. Using estimates of reactivity effects of the engineering features, such as end gaps, rod channels, etc., the expected volume of the same composition core in a detailed engineering mockup of the RAPSODIE reactor would have been in excess of 44 liters. A limit of 40.3 liters was established by the design specifications for RAPSODIE. Consequently, the decision was made to reduce the volume for the engineering core by increasing the fuel concentration from that contained in the physics-core composition. This was accomplished by replacing the mixed-fuel column in the physics-core unit cell loading (see Fig. 6), containing one-half column plutonium, one-quarter column enriched uranium, and one-quarter column depleted uranium, with a full column of plutonium.

Figure 14 shows a Half No. 2 core drawer containing a 6-in. section of the engineering core followed by the same axial blankets used for the physics core. The composition obtained for the unit cell of the engineering

Fig. 14

Engineering-core Drawer with
Physics Axial Blankets

core is given in Table XI. The change in the loadings of the core drawers provided a simulation of a different RAPSODIE composition. The Table XI compositions are equivalent to about 41 v/o fuel ($\text{PuO}_2\text{-UO}_2$), 21 v/o steel, and 34 v/o sodium, with 3.23 atoms of uranium per plutonium atom and 62 a/o enrichment of the uranium. Based on worth measurements of materials in the physics core, the new composition was expected to yield a critical core volume of 34.6 liters in a "clean" configuration (no gaps, channels, etc.). For the full engineering mockup, using estimates of the worths of the engineering features, a critical volume of 40.4 liters was predicted.

TABLE XI. Composition of Engineering-core Unit Cell

Dimensions of cell: 2.18 x 2.18 x 14.06 in.

Cell volume: 1.0944 liters

Material	Cell Composition		
	Material Density, g/cc	10^{22} atoms/cc	v/o ^(a)
Pu^{239}	0.8442	0.2126	4.43
$\text{Pu}^{240+242}$	0.0404	0.0101	0.21
Pu^{241}	0.0039	0.0010	0.02
U^{235}	1.7537	0.4493	9.36
U^{238}	1.0525	0.2662	5.54
$\text{U}^{234+236}$	0.0253	0.0065	0.14
O	0.5198	1.9567	19.79
Al	0.4635	1.0347	17.16
Stainless steel ^(b)	0.9306	-	-
Fe	0.6695 ^(c)	1.4446 ^(d)	17.06 ^(d)
Cr	-	0.1940 ^(d)	2.42 ^(d)
Ni	-	0.0955 ^(d)	1.05 ^(d)

(a) Based on 100% atomic densities listed for 16-group, fast cross-section set in ANL-5800, Ref. 5, p. 559.

(b) Type 304, assumed composition 72 w/o Fe, 18 w/o Cr, 10 w/o Ni.

(c) Iron content from Fe_2O_3 .

(d) Total contributions from steel and Fe_2O_3 .

2. Approach to Critical

Increasing the fuel composition of the core unit cells required a partial unloading of the core and a new approach to critical. In the initial loading of the approach, about 44% of the physics-core drawers were replaced with radial blanket, and the remainder were changed to the engineering-core composition. The initial and subsequent loadings of the

engineering core are shown in Fig. 15. Criticality was reached with $32\frac{1}{2}$ core drawers in each half of the reactor, and the available excess reactivity was determined as 260 lh.

Fig. 15. Approach-to-critical Loadings for Engineering Core

From later fuel-substitution experiments at the core edge, a unit cell (a Half No. 1 drawer with an 8-in. core section, plus a Half No. 2 drawer with a 6-in. core section) was found to be worth 523 lh. This would equate the available excess reactivity of the Fig. 15 loading to 0.50 unit cell, and the just-critical loading would be 32.0 unit cells or 35.0 liters. However, the core, as shown in Fig. 15, had only a crude cylindrical outline, and by smoothing out the irregularities, a smaller critical volume was attainable. Four full-core drawers were replaced by eight half-core drawers distributed around the periphery, and a half-core drawer was replaced with blanket in both halves of the assembly. This resulted in the reference critical loading of the engineering core shown in Fig. 16.

ZPR - 3 CONTROL ROD

Fig. 16. Reference Critical Loading of Engineering Core

3. Fuel Worth at Radial Edge of Core

A series of blanket-for-core substitutions was made at the radial edge of the core to obtain the reactivity worth of a unit cell. In each substitution, a drawer loaded as half-core and half-blanket was replaced by a full drawer of radial blanket. For half-core drawers in Half No. 1, substitution worths of -151.9 ± 2 and -160.9 ± 2 lh were obtained for matrix positions 1-O-13 and 1-Q-19, respectively (see Fig. 16). In Half No. 2, the replacements netted -101.0 ± 1.5 lh for position 2-O-13 and -109.5 ± 1.5 lh for position 2-Q-19. Together, these core sections comprise a unit cell with a total worth of 523 ± 5 lh (relative to blanket) at this edge location. The probable overall uncertainty for this value, as an average worth of fuel around the core periphery, is ± 30 lh.

4. Critical Volume of Engineering Core

The reference critical loading for the engineering core (Fig. 16) contained 32 unit cells, and the available excess reactivity was 236 lh, equivalent to 0.45 ± 0.08 cell. Thus, the just-critical loading would be 31.55 ± 0.08 unit cells, or the critical volume would be 34.53 ± 0.09 liters. This is for the engineering core in a "clean" configuration, i.e., with none of the engineering features as gaps, channels, etc., with the same radial

and axial blankets as for the physics core (Table III and Figs. 8 and 9). For this reference, the average core temperature was about 30.6°C, the core center being at 34.2°C and the ZPR-3 frame at 23.3°C.

The critical volume for the reference critical loading (Fig. 16) differed from the value derived from the initial critical (Fig. 15) by about 0.5 liters, worth about 250 lh. This indicates the importance of the degree to which the radial core boundary approaches a circular cylinder in such small configurations in ZPR-3. The critical volume of 34.5 liters obtained compares well with the value of 34.6 predicted by extrapolation from the physics-core experiments.

5. Control-rod Calibrations

With the change of core composition and geometry, new control-rod calibrations were required. A different technique for calibration was

Fig. 17. Control-rod Calibrations in the Engineering Core with Physics Blankets

rod. Contributing to the additional reactivity would be the decrease in core size and the change in effective delayed-neutron fraction introduced by the change in core composition.

adopted than that used for the physics core. The relative shapes of calibrations of two rods were obtained by equating the worths of successive portions of the rods to the worth of a specific increment of a third rod. Period measurements then were made for a portion of one rod to obtain the absolute calibrations. This method provides more precision than the subcritical method employed for the physics core. After various stages in the subsequent construction of the engineering mockup, introduction of axial end gaps, etc., it was deemed necessary to recalibrate the control rods to account for shifting flux and worth patterns. Figure 17 is a graph of calibrations in the "clean" engineering core for a core-loaded rod (No. 10 in Fig. 16) and blanket-loaded rods Nos. 3 and 8.

Comparing rod calibrations in the engineering and physics cores reveals increased worths of about 6 and 12% for the blanket-loaded rods and about 16% for the fuel-loaded

B. Construction of Axial Gaps and Blankets

In the physics-core experiments and in the reference critical loading with the engineering core, both the radial and axial blankets had a composition simulating that of the radial blanket specified for RAPSODIE. In the axial regions above and below the core in the RAPSODIE reactor there are end gaps, containing steel and sodium only, and then axial blankets of a different composition than that of the radial blanket. The details of these regions of RAPSODIE were given in Fig. 3 and Table I. These engineering features were next to be constructed in the RAPSODIE mockup.

1. Upper and Lower Axial Blankets

The first step in the construction of the axial regions for the engineering mockup was to remove the axial blanket and substitute a composition simulating the design composition for the lower axial blanket in RAPSODIE. Later the proper end gaps were installed, replacing part of these blankets.

Figure 18a is a top view of a Half No. 2 core drawer with the engineering axial blanket in place behind the core section. Compared with the previous blanket loading (Fig. 14), there are fewer columns of depleted uranium and more "coolant-channel" columns. The engineering axial blanket was substituted for the physics-core blanket behind the core in 32 drawers of Half No. 2, and then in drawers of Half No. 1, although for Half No. 1 the RAPSODIE design called for a slightly different upper blanket. For Half No. 2, the length of the new blanket was 13 in. (9 in. in the front drawer and 4 in. in the back drawer), with 5 in. of 56 v/o aluminum beyond this simulating the bottom gap. The engineering axial blanket behind the core in Half No. 1 extended for 16 in., with a few inches of 56 v/o aluminum representing the top gap. The compositions of these engineering axial regions are presented in Table XII, and the regional replacements involved in their construction are given in Table XIII along with the associated reactivity effects.

2. Upper and Lower End Gaps

The fuel and blanket elements in the core subassemblies of the RAPSODIE reactor incorporate void spaces, above the slugs within the cladding, which allow for expansion and fission-product gas containment. This creates gap regions, containing only steel and sodium, immediately above and below the core. As seen in Fig. 3, the total upper end gap is about 6.7 in. (17 cm) long, and the lower end gap is 3.5 in. (8.7 cm) long. These axial regions were next to be mocked up in Assembly 44. For simplicity, the composition of both upper and lower mockup gaps was made the same, simulating that of the lower gap in RAPSODIE. The loading of a gap section behind core in a drawer was composed of six 1/4-in. columns of canned sodium and four 1/8-in. columns of steel. The resulting compositions for the upper and lower mockup gaps are presented in Table XII.

a) Half No. 2 Core Drawer with Engineering Axial Blanket

b) Half No. 2 Core Drawer with End Gap and Engineering Blanket

Fig. 18. Engineering-core Drawers with Engineering Axial Regions

TABLE XII. Compositions of Axial Regions of Engineering Mockup

	Axial Region				
	Upper End Gap	Lower End Gap	Engineering Axial Blankets		Top Gap and Bottom Gap
			Upper	Lower	
Axial length, in. (cm)	7 (17.78)	3 (7.62)	9 (22.86)	10 (25.4)	Top gap: 2 (5.08) Bottom gap: 5 (12.7)
<u>Compositions, g/cc</u>					
Depleted uranium	-	-	6.707	6.662	-
Stainless steel	3.018	3.050	1.490	1.491	0.706
Aluminum	-	-	0.379	0.377	1.237
Sodium	0.490	0.460	-	-	-
Molybdenum	-	-	1.038	1.037	-
<u>Compositions, 10²² atoms/cc</u>					
U ²³⁸	-	-	1.693	1.682	-
U ²³⁵	-	-	0.0035	0.0035	-
Fe	2.343	2.368	1.157	1.158	0.548
Cr	0.629	0.636	0.310	0.311	0.147
Ni	0.310	0.313	0.153	0.153	0.072
Al	-	-	0.846	0.842	2.761
Na	1.284	1.205	-	-	-
Mo	-	-	0.652	0.651	-

TABLE XIII. Reactivity Worths of Engineering Axial Regions in RAPSODIE Mockup

Exp. No.	Description of Substitution	Regional Changes of Composition, ^a Axial Distances (in cm) from Interface ^b		Substitution Worth, Δk
		From	To	
1	Replacement of physics axial blankets in Half No. 1 with engineering axial blanket	+20.40 to +30.56: ABF +30.56 to +64.24: ABM	+20.40 to +61.70: EAB +61.70 to +66.78: TG	-36.0 ± 1.1
2	Replacement of physics axial blankets in Half No. 2 with engineering axial blanket	-15.32 to -25.48: ABF -25.48 to -59.16: ABM	-15.32 to -49.00: EAB -49.00 to -61.70: BG	-34.7 ± 1.5
3	Replacement of 3 in. of engineering axial blanket in Half No. 2 with lower end gap	-15.32 to -49.00: EAB -49.00 to -61.70: BG	-15.32 to -22.94: LEG -22.94 to -49.00: EAB -49.00 to -61.70: BG	-302 ± 3
4	Replacement of 6 in. of engineering axial blanket in Half No. 1 with end gap	+20.40 to +61.70: EAB +61.70 to +66.78: TG	+20.40 to +35.64: UEG +35.64 to +61.70: EAB +61.70 to +66.78: TG	-386 ± 4
5	Replacement of 7th inch of engineering axial blanket in Half No. 1 with end gap	+20.40 to +35.64: UEG +35.64 to +61.70: EAB +61.70 to +66.78: TG	+20.40 to +38.18: UEG +38.18 to +61.70: EAB +61.70 to +66.78: TG	-20.4 ± 1.7
	Total reactivity change of replacing physics axial blankets with engineering axial regions (engineering core)			-779 ± 8

^aABF and ABM compositions in Table III; EAB, TG, BG, LEG, and UEG in Table XII.

^bFigures 9 and 19 illustrate axial configurations.

For the mockup of the lower end gap, a 3-in. length was chosen. Figure 18b shows this gap in place behind the core section, followed then by engineering axial blanket, in a Half No. 2 core drawer. The replacement of 3 in. of engineering axial blanket with end gap was done behind sectors of the core in Half No. 2 until 32 drawers contained the gap.

The mockup of the upper axial gap in Half No. 1 was first constructed to a length of 6 in. beyond the core, again by sectors of the core until contained in 32 drawers. This was then extended to a 7-in. mockup length, the worth of the last inch providing a possible basis for interpolation to the 6.7-in. (17.0-cm) gap in RAPSODIE.

The regional substitutions involved in the construction of the mockup end gaps are given in Table XIII along with the reactivity effects incurred. For the entire process of replacing the physics-core axial blankets with the engineering-mockup axial regions, a reactivity loss of $779 \pm 8 \Delta k$ was obtained. Maintaining a critical assembly required the total addition of approximately 1.5 core unit cells, thus producing a critical volume of 36.2 liters for the engineering core with engineering axial regions (no mockup rods). The axial configuration of the assembly at this stage, with the engineering axial regions, is shown in Fig. 19.

Table III Compositions

ABM	Medium Axial Blanket
RBF	Fine Radial Blanket
RBM	Medium Radial Blanket
RBC	Coarse Radial Blanket

Core Composition in Table XI

Table XII Compositions

TG	Top Gap
EAB	Engineering Axial Blanket
UEG	Upper End Gap
LEG	Lower End Gap
BG	Bottom Gap

Spring Gaps, 2.13 g/cc Steel

Fig. 19. Axial Schematic Diagram of Assembly 44 with Engineering Core and Engineering Axial Regions

C. RAPSODIE Safety-Control Rod Mockups

1. Mockup Description

In the RAPSODIE reactor (Fig. 2), four hexagonal subassemblies at the core radial edge are designated for enriched boron carbide-loaded safety rods and two for natural boron carbide control rods. This hexagonal arrangement is difficult to mock up in the ZPR-3 matrix, although a close approximation was achieved as the final phase of the Assembly 44 experiments (Section H below). For the majority of the experiments to be done in the engineering core, reaction-rate traverses, reactivity coefficients, etc., it was decided that a four-rod engineering mockup, with four mockup rods loaded to represent the total B^{10} content in the six RAPSODIE rods, would suffice. Figure 20 shows the orientation of these four channels of the mockup rods. This arrangement provides an average rod radius of 17.9 cm from the reactor axis, compared to 17.6 cm in RAPSODIE, while retaining the center of matrix position P-16 as the reactor axis, making subsequent experiments easier. In the six-rod engineering mockup later, it was necessary to shift the axis of the core loading away from the center of P-16.

Each of the four channels for the mockup rods consisted of a ZPR-3 matrix unit plus a 1/4-in. column in the drawer of an adjacent matrix unit. The total cross-sectional area of the four channels was 141.8 cm², compared to 134.8 cm² for the six rods of RAPSODIE. Construction of a channel in the assembly involved replacement of blanket in the main matrix

unit and 1/4-in. column of the adjacent unit with the rod mockup and some rearrangement of the other blanket materials in the drawers of the adjacent matrix unit. Thus, a 22-in. (55.9-cm) axial length of fine radial blanket and 26 in. (66.0 cm) axially of medium axial blanket were replaced in the channel (see Figs. 7 and 9). Also, the axial medium-blanket sections of the adjacent matrix unit, excluding the 1/4-in. channel column, had to be replaced by fine blanket. The net substitution in constructing four channels thus approximately simulated the removal of six radial-blanket subassemblies in RAPSODIE.

Fig. 20. Core Configuration of Four-rod Engineering Mockup

Figure 21 illustrates the loading of a channel of a mockup rod in matrix positions Q-19 and Q-20, with the poison section of the rod situated to represent a typical operational position in RAPSODIE, the "banked" position defined later in Section C-4. The active portion of the mockup control-safety rod was 14 in. (35.6 cm) long compared to the 18.2-in. (46.2-cm)-long RAPSODIE rods. However, for this "banked" position, the extra 4-in. length at the end of the rod would have little worth. The essential goal in the mockup was to attain a B^{10} loading per unit length equivalent to that of a safety rod plus half of a control rod in RAPSODIE. The mixture of enriched and natural boron carbide in the mockup provided 60.3 g/in. (23.7 g/cm) of B^{10} compared to a specified 61.1 g/in. (24.1 g/cm).

in the construction of the mockup rods will be described in subsequent sections, and the reactivity effects observed are given in Table XV. As the construction of these mockup rods proceeded, fuel was added at the radial core edge to maintain criticality.

TABLE XV. Reactivity Worths of Mockup Safety-Control Rods in Four-rod Engineering Mockup of RAPSODIE

Exp. No.	Mockup Channel(s)	Description of Experiment; Substitution Made	Regional Changes of Composition in Channel(s), ^a Axial Distances (in cm) from Interface		Substitution Worth, 1h
			From	To	
1	Q-19/20	Replaced radial blanket in channel with sodium cans (sodium-"flooding")	+64.24 to +30.56: ABM +30.56 to -25.32: RBF -25.32 to -59.16: ABM	+67.8 to -67.8: sodium channel	-142 ± 4
2	Q-19/20	Replaced sodium in channel below +10 in. with rod stem	+67.8 to -67.8: sodium channel	+67.8 to +25.48: sodium channel +25.48 to -67.8: rod stem	+113 ± 4
3	Q-19/20	Replaced sodium above +10 in. in channel with active rod in "up" position	+67.8 to +25.48: sodium channel +25.48 to -67.8: rod stem	+67.8 to +62.21: dummy rod +62.21 to +25.48: active rod +25.48 to -67.8: rod stem	-62 ± 1
4	O-13/12	Replaced radial blanket in channel with mockup rod in "up" position	+64.24 to +30.56: ABM +30.56 to -25.32: RBF -25.32 to -59.16: ABM	+67.8 to +62.71: dummy rod +62.71 to +25.48: active rod +25.48 to -67.8: rod stem	-101 ± 1
5	M/L-17 S/T-15	Replaced radial blanket in channels with mockup rods in "up" position	+64.24 to +30.56: ABM +30.56 to -25.32: RBF -25.32 to -59.16: ABM	+67.8 to +62.71: dummy rod +62.71 to +25.48: active rod +25.48 to -67.8: rod stem	-169 ± 2
6	O-13/12	Moved active-rod section inward 3 in., from "up" to "banked" position	+67.8 to +62.71: dummy rod +62.71 to +25.48: active rod +25.48 to -67.8: rod stem	+67.8 to +55.09: dummy rod +55.09 to +17.86: active rod +17.86 to -67.8: rod stem	-102 ± 1
7	O-19/20 M/L-17 S/T-15	Moved active-rod sections in channels inward 3 in., from "up" to "banked" position	+67.8 to +62.71: dummy rod +62.71 to +25.48: active rod +25.48 to -67.8: rod stem	+67.8 to +55.09: dummy rod +55.09 to +17.86: active rod +17.86 to -67.8: rod stem	-281 ± 4

^aChannel cross section = 6.40 x 5.54 cm (35.45 cm²). Blanket compositions in Table III, mockup rod compositions in Table XIV.

2. Worth of Replacing Blanket with Rod Channel

The first experiment for the rod mockups was to determine the reactivity effect of replacing radial blanket with a "void" rod channel, i.e., one containing sodium only. In the Q-19/20 channel in both assembly halves, nine 1/4-in. columns of canned sodium were substituted for blanket throughout the axial length of the reactor. The result was a reactivity loss of 142 ± 4 1h. Except for a minor adjustment for worth of steel, the reactivity change represents that due to a removal of 1½ blanket subassemblies at the core edge in RAPSODIE, and flooding with sodium.

3. Worth of Safety-Control rods in "Up" Position

The mockup rod for the channel in Q-19/20 was introduced in two steps. First, the rod guide or stem portion below the intended active region was constructed. In the "up" position, the bottom of the B₄C-loaded, active-rod section was to be situated 2 in. above the top of the core. The stem section below this was mocked up by replacing seven of the nine sodium columns in the channel with steel throughout Half No. 2 and up to 10 in. axially in Half No. 1. The net steel-for-sodium substitution was worth +113 ± 4 1h.

Next, the active-rod section, containing the mixture of enriched and natural B_4C , was loaded in place of sodium in the upper channel (drawer 1-Q-19), from 10 to 24.3 in. axially. The replacement produced a reactivity loss of 62 Ih. Combined with previous results, this gives the worth of a mockup rod in the "up" position of -91 ± 4 Ih when replacing blanket.

A direct replacement of blanket with a mockup rod in the "up" position was then made in the O-13/12 channel. Here the reactivity loss was 101 ± 1 Ih, the difference from the previous result probably due to a difference in core geometry near the channels. Finally, after a fuel addition at the core edge to obtain a new reference, the last two mockup rods were introduced into channels M/L-17 and S/T-15, the active sections being 2 in. behind the axial end of the core in Half No. 1. The worth of these rods together replacing blanket was -169 ± 2 Ih.

Overall, the introduction of the four mockup rods in the "up" position was worth -373 ± 7 Ih (replacing blanket). Compensating fuel additions totaling about 1 liter were made during the process. The final configuration contained approximately 34.7 unit cells with an available excess reactivity of 54 Ih. Using an estimated cell worth of 450 Ih gives a critical volume of 37.9 liters for the engineering core with mockup rods in the "up" position.

4. Worths of Safety-Control Rods in "Banked" Position

In RAPSODIE, during operation, the control and safety rods would be "banked" or moved in from the "up" position to achieve the proper reactivity balance. A typical "banked" position at 3 in. of insertion for all the rods was selected for the mockup configuration. This would place the bottoms of the active-rod sections 1 in. below the top of the core. Figure 21 illustrates the situation of a "banked" rod in the Q-19/20 channel. The active-rod mockup in channel 1-O-13 was the first to be reloaded and moved in 3 in. to the "banked" position, producing a reactivity loss of 102 ± 1 Ih.

The rod mockups in the other three channels (drawers 1-Q-19, 1-M-17, and 1-S-15) were "banked" simultaneously with an addition of fuel at the core radial edge.

Subsequent determinations of the worth of the added fuel indicated that these rods moving in 3 in. provided a reactivity loss of 281 Ih. Banking of the four mockup safety-control rods thus was worth -383 Ih. About another liter of fuel was required for compensation, and the final configuration obtained for the four-rod engineering mockup is that shown in Fig. 20.

D. Critical Configuration of Four-rod Engineering Mockup of RAPSODIE

The four-rod engineering mockup, with rods "banked," had an available excess reactivity of 123 ± 1 lh. The core contained a total of 36 Half No. 1 unit cell sections plus $35\frac{1}{2}$ Half No. 2 unit cell sections. An integral number of 36 unit cells would be provided by the addition of a half-core drawer in 2-S-18 with an estimated worth of 92 lh. From the worths of fuel additions during the construction of the rod mockups, the worth of a unit cell composed of half-core drawers in 1- and 2-M-14 and 1- and 2-S-18 was determined as 427 ± 30 lh (replacing blanket). A core of 36 unit cells with 215 lh available excess reactivity would indicate a just-critical loading of 35.50 ± 0.08 cells. This gives a critical volume of 38.85 ± 0.09 liters for the engineering core in the four-rod mockup of RAPSODIE. Temperatures for this loading were 36.4°C at the core center, 32.7°C for the core average, and 24.3°C for the frame.

E. Reactivity-coefficient Measurements

Design and material limitations of the ZPR-3 determined how precisely regions of the mockup represented corresponding regions of RAPSODIE in geometry and composition. Also, since the precise RAPSODIE core composition at the time was not determined, one purpose of the mockup was to aid in fuel design. Several material-worth measurements were therefore made in the engineering mockup to enable assessment of reactivity differences associated with differences in compositions. Besides finding distributed worths in the core, end gaps, and rod channels, local worths were measured and traverses of fuel samples were made.

1. Distributed Worths in Core

The material substitutions for the average worths in the core were done in two parts, one representing most of the core, excluding the central drawers, the other being in the central drawers. In the major part, a distribution of nine drawers in each assembly half was chosen so that together they represented a quadrant of the core in relation to their placement from the reactor axis. Figure 22 shows the drawer distributions. Substitutions were made in these drawers in place of one or two of the central columns of 45 v/o Al in the core section (8 in. in Half No. 1 and 6 in. in Half No. 2). The materials involved in the experiments included the nonfissile core constituents, depleted uranium, and sodium. Table XVI provides the details and results of the substitutions.

For the material worths in the central drawers, the substitutions were done in the central $1/8$ -in. columns of 1- and 2-P-16. The results of these experiments are included in Table XVI. Together with the worths in the nine distributed drawers, they provide the core-average reactivity coefficients listed in the table.

Fig. 22. Distribution of Substitution Drawers for Measurements of Material Worths in Core

TABLE XVI. Distributed Worths of Materials in Engineering Core

Material	Distributed Quadrant (nine drawers each half)			Central Drawers (1- and 2-P-16)			Core Average Coefficient, lh/kg
	Sample(a) Mass, g	Net(b) Worth, lh	Reactivity Coefficient, lh/kg	Sample(a) Mass, g	Net(b) Worth, lh	Reactivity Coefficient, lh/kg	
Aluminum	746	52.3 ± 1.0	70.1 ± 1.3	166	11.2 ± 0.6	68 ± 4	70 ± 1
Stainless steel (304)	3841	108.6 ± 2.5	28.3 ± 0.7	715	9.2 ± 0.4	12.9 ± 0.5	27.9 ± 0.7
Iron	3891	92.4 ± 2.8	23.7 ± 0.7	865	9.3 ± 0.7	10.8 ± 0.8	23.3 ± 0.7
Fe ₂ O ₃	2396	144.1 ± 3.5	60.2 ± 1.4	532	24.0 ± 0.5	45 ± 1	59.8 ± 1.4
Al ₂ O ₃	1268	135.6 ± 3.5	107 ± 3	282	29 ± 1	103 ± 3	107 ± 3
Sodium	790	97.4 ± 0.6	123 ± 1	83	10.1 ± 0.4	122 ± 5	123 ± 1
Depleted U	9391	231 ± 3	24.6 ± 0.3	2086	58.5 ± 0.7	28.0 ± 0.4	24.7 ± 0.3
Oxygen in Al ₂ O ₃ ^(c)			149 ± 7			143 ± 10	149 ± 7
Oxygen in Fe ₂ O ₃			152 ± 7			131 ± 10	151 ± 7

(a) Net for sample material, excluding canning.

(b) For sample mass relative to void.

(c) Derived from metal and metal oxide worths.

2. Distributed Worths in End Gaps

Since the upper and lower end gaps contained only steel and sodium, the worths of just these two materials were investigated there. In each assembly half, the same nine distributed-quadrant drawers involved in the average core-worth measurements (Fig. 22) were used for substitutions in the end gaps. In the lower gap section of each of the nine Half No. 2 drawers, a 1/4 x 2 x 3-in., sodium-filled can (one of the central columns from 6 to 9 in. in Fig. 18b) was replaced by an empty can and, subsequently, by steel. For the worth in the upper gap, the void can and steel substitutions were made for a 7-in. sodium-filled can behind the core in the nine Half No. 1 drawers. Table XVII gives the results of these experiments.

TABLE XVII. Worths of Steel and Sodium in Axial End Gaps

Region	Material	Net Substitution, g	Substitution Worth, I_h	Reactivity Coefficient, I_h/kg
3-in. lower gap	Sodium	-161	-13.8 ± 0.6	86 ± 4
3-in. lower gap	Steel ^a	1380	29.9 ± 0.8	21.7 ± 0.6
7-in. upper gap	Sodium	-401	-16.0 ± 0.4	40 ± 1
7-in. upper gap	Steel ^a	3242	41.9 ± 0.8	12.9 ± 0.3

^aType 304 stainless steel.

3. Regional Worths of Sodium in Core

The worth of sodium in the core was investigated in more detail to determine the worth variation according to radial and axial location. The information obtained enables the evaluation of reactivity effects in

RAPSODIE from possible localized voids in the coolant, density variation, etc., as well as providing a check on calculated worths. These experiments involved substitutions of sodium cans for empty cans in 12 regions of the engineering core composed of three axial regions in each of four radial zones. Figure 23 outlines the four radial zones. The axial regions were -1 to 1, 1 to 4, and 4 to 7 in. from the core mid-plane (at 1 in. in Half No. 1 drawers).

Fig. 23. Core Zones for Regional Worths of Sodium Experiments

In the drawers of the radial zones, a 1/4-in. column for a

substitution region was created by removing the two central 1/8-in. columns of 45 v/o aluminum of the core section. Experiments in Radial Zone I were done in 1-P-16 and 2-P-16, having 2-in., 3-in., and 3-in. cans loaded in sequence from the front. Table XVIII presents the results of these experiments. The values for Radial Zone I have poor accuracy because of the small perturbations involved. In the axial substitution from -1 to +1 in. in this zone, the net worth was -0.2 Ih, whereas the uncertainty involved with temperature corrections was about ± 0.6 Ih. The value found for the 2 x 2 x 2-in central sample (Table XIX) was therefore used for this region. In the table, the outer radii of the zones were determined as the radii of circles of equivalent areas.

TABLE XVIII. Regional Worths of Sodium in Engineering Core

Radial Zone			Sodium Reactivity Coefficient, Ih/kg (relative to void)		
No.	Inner Radius, ^a in. (cm)	Outer Radius, ^a in. (cm)	Axial Region No. 1: -1 to +1 in. ^b (± 2.54 cm)	Axial Region No. 2: 1 to 4 in. ^b (2.54 to 10.16 cm)	Axial Region No. 3: 4 to 7 in. ^b (10.16 to 17.78 cm)
I	0	1.23 (3.12)	(62 $\pm$ 3) ^c	89 $\pm$ 10	123 $\pm$ 20
II	1.23 (3.12)	3.70 (9.37)	80 $\pm$ 8	103 $\pm$ 5	176 $\pm$ 4
III	3.69 (9.37)	5.63 (14.31)	114 $\pm$ 3	119 $\pm$ 3	132 $\pm$ 3
IV	5.63 (14.31)	7.48 (19.00)	99 $\pm$ 6	132 $\pm$ 3	128 $\pm$ 3

^aAverage inner and outer radii determined on an area basis.

^bExtensions of axial regions given as distances from core midplane (at +1 in. from interface).

^cTaken from central worth measurements with 2 x 2 x 2-in. sample (Table XIX).

TABLE XIX. Worths of Materials at Center of Engineering Core

Material ^a	Mass, ^b g	Worth versus Void, ^b Ih, ± 0.2	Reactivity Coefficient, Ih/kg
Al, 56 v/o ^c	192	2.2	11.5 $\pm$ 1.0
Al, 100 v/o	350	4.1	11.7 $\pm$ 0.5
Steel, 56 v/o ^c	554	-3.1	-5.6 $\pm$ 0.4
Steel, 100 v/o	1016	-5.0	-4.9 $\pm$ 0.2
Al ₂ O ₃	224	7.8	35 $\pm$ 1
Fe ₂ O ₃	341	4.5	13.3 $\pm$ 0.6
Fe	1028	-5.6	-5.5 $\pm$ 0.2
Cr	439	1.4	4.4 $\pm$ 0.5
Ni	1150	-19.4	-16.9 $\pm$ 0.2
Na	91	5.7	62 $\pm$ 3
Oxygen (from Fe and Fe ₂ O ₃ results)			60 $\pm$ 1
Oxygen (from Al and Al ₂ O ₃ results)			61 $\pm$ 1

^aSubstitutions made in 2 x 2 x 2-in. sample space at center of reactor.

^bFor material only, excluding mass and worth of steel canning, if any.

^cPerforated samples for density-worth correlations.

4. Material Worths at Core Center

Material substitution experiments were made with 2 x 2 x 2-in. samples at the core center using the sample changer mechanism described for similar experiments in the physics core (Section V-E above). The core

section of the changer-drawer in matrix position 1-P-16 extended from 2 in., behind the sample region, to 10 in., thus extending beyond the normal axial boundary. Table XIX lists the materials substituted and their measured worths. Two different densities for steel and aluminum were investigated in an effort to detect possible self-shielding effects. However, to within the accuracy of the measurements, the worths per kg for the 56 v/o samples (2 x 2 x 2 in. with perforations) were the same as for the full-density samples.

5. Steel and Sodium Worths in Safety-Control Rod Channel

The reactivity coefficients for steel and sodium relative to void were measured in the "stem" portion of a rod-channel mockup. For steel in the S-15 channel, a worth of $+15.1 \pm 0.6$ lh/kg was obtained over the axial region from -25 to +6 in. (-63.5 to +15.2 cm) from the reactor midplane. Assuming a worth next to the core of 25 lh/kg (based on worth in 3-in. end gap) gives an average steel worth over 50 in. of channel of +12.8 lh/kg. Likewise for sodium, the rod stem-for-sodium replacement (Section VI-C above) and sodium worth in the 3-in. end gap (90 lh/kg), provide an average worth of +42 lh/kg for sodium in the rod channel from -25 to +25 in. (-64 to +64 cm).

Based on these material worths in the mockup channel, +42 lh/kg-sodium and +13 lh/kg-steel, and the sodium-for-blanket substitution, the interpolated worth of replacing a RAPSODIE blanket subassembly by sodium (0.84 g/cc at 500°C) would be about -130 ± 10 lh.

6. Axial Traverses of Reactivity Samples

Small samples of enriched uranium, natural uranium, and plutonium were traversed through the reactor, both axially and radially, to obtain a profile of their reactivity worths versus position. For traverses along the reactor axis, the front and back drawers in 1- and 2-P-16 were rearranged to accommodate a 1/2-in.-diam aluminum guide tube through the center. Figure 24a shows the core material arrangement around the tube in a front view of drawer 1-P-16. The traversing mechanism was mounted on the support plate behind Half No. 2. This permits traverses starting from outside the assembly, through the lower axial blanket, through the lower gap, through the core, and into the upper axial gap in Half No. 1.

The uranium samples traversed were 3/8-in.-diam metal cylinders, 2 in. long for the natural uranium and 1/2 in. long for the enriched uranium. A "blank" run was also made, determining the contribution from the traverse push-rod to the worths versus position with the sample traverses. Differences in steady-power control-rod settings for corresponding traverse positions with and without the samples gave the sample worths versus axial position. Reactivity drift due to temperature effects was also accounted for by repeating references with the samples withdrawn. The

plutonium sample was contained in a 2- x 1/4-in.-diam cylindrical steel can, and this was traversed within an aluminum capsule. For the plutonium-blank run, an empty steel can within the aluminum capsule was traversed. Table XX lists the material worths for the samples at the various axial positions. Also in the table are the reactivity coefficients for the isotopes U^{235} and U^{238} , derived from the enriched and natural worths. The U^{235} , U^{238} , and plutonium worths are plotted versus axial position in Fig. 25.

a) Front View of 1-P-16 with Loading for Axial Traverse

b) Top View of Front of Half No. 1 Core Drawer with Tube Channel for Radial Traverse

c) Top View of Front of Radial Blanket Drawer with Tube Channel for Radial Traverse

Fig. 24. Drawer Loadings for Traverse Experiments

TABLE XX. Axial Traverses of Uranium and Plutonium Reactivity Samples through Engineering Mockup

Axial Position, in. from Midplane	Reactivity Coefficient, lh/kg				
	Plutonium(a) (± 6)	Enriched U(b) (± 6)	Natural U(c) (± 1.6)	U ²³⁵ (d) (± 6)	U ²³⁸ (d) (± 1.6)
-23	0	0	0.0	0	0.0
-19	5	10	0.8	11	0.7
-16	15	11	1.9	12	1.8
-13	49	33	1.7	35	1.5
-10	128	87	6.7	93	6.1
-8.5	188	118	13.5	126	12.7
-7	318	192	29.6	204	28.4
-6	452	266	37.3	282	35.6
-5	586	-	-	-	-
-4	704	374	32.3	400	29.7
-3	804	-	28.8	-	25.8
-2	878	477	28.3	510	24.9
-1	914	483	26.5	517	23.1
0	934	497	25.3	532	21.7
1	915	493	25.4	528	21.8
2	869	466	27.9	499	24.6
3	789	-	28.1	-	25.1
4	681	374	33.4	399	30.8
5	560	-	37.3	-	35.2
6	430	251	43.0	266	41.4
7	298	160	37.9	169	37.0
8	196	-	-	-	-
9	-	-	15.4	-	14.9
10	98	66	-	71	-
11	-	-	7.4	-	7.1
12	54	-	-	-	-
13	-	27	-	29	-
14	29	-	1.3	-	1.2
16	-	16	-	17	-
17	15	-	-	-	-
20	-	9	2.0	9	1.9
21	9	-	-	-	-
24	-	9	1.7	9	1.6
24.8	9	-	-	-	-

(a) Measured with 24.09-g sample, 94.9 w/o Pu²³⁹, 4.8 w/o Pu²⁴⁰, 0.3 w/o Pu²⁴¹.(b) Measured with 22.93-g sample, 93.1 w/o U²³⁵, 6.9 w/o U²³⁸.(c) Measured with 64.39-g sample, 0.7 w/o U²³⁵.

(d) Derived from enriched uranium and natural uranium worths.

Fig. 25. Worths of Pu, U²³⁵ and U²³⁸ versus Axial Position in Engineering Mockup

7. Radial Traverses of Reactivity Samples

For the radial-traverse experiments, the traverse guide tube was installed vertically through the drawers in column 16 in Half No. 1, about 3/4 in. in from the reactor interface (about 1/4 in. from core mid-plane). Figures 24b and 24c show the material arrangement around the traverse channel in a core drawer and in a radial blanket drawer. With the drive mechanism mounted on top of the ZPR-3, traverses were possible from the outer edge of the coarse radial blanket, through the radial blankets, through the core, down to drawer 1-S-16.

In the same manner as for the axial sample traverses, the worths of enriched uranium, natural uranium, and plutonium were measured at several radial positions, and the results are given in Table XXI. The derived worths of U^{235} and U^{238} and the plutonium worths are plotted versus radius in Fig. 26.

TABLE XXI. Radial Traverses of Uranium and Plutonium Reactivity Samples through Engineering Mockup

Radius, in. from Core Center	Reactivity Coefficient, $\Delta k/k$				
	Plutonium ^(a) (± 6)	Enriched U ^(b) (± 6)	Natural U ^(c) (± 1.6)	U^{235} ^(d) (± 6)	U^{238} ^(d) (± 1.6)
24	0	0	-	0	-
23	-	-	0.0	-	0.0
22	-	2	-	2	-
20	0	2	-	2	-
19	-	-	0.0	-	0.0
18	-	3	-	4	-
16	2	5	-	5	-
15	-	-	1.3	-	1.3
14	23	15	-	16	-
13	-	-	1.6	-	1.4
12	50	34	-	36	-
11	-	-	6.9	-	6.6
10	108	-	-	-	-
9.6	-	85	-	90	-
9.0	-	-	15.9	-	15.2
8.5	198	-	-	-	-
7.6	278	178	-	189	-
7.5	-	-	28.1	-	26.9
6.6	-	233	-	248	-
6.5	406	-	-	-	-
6.0	-	-	32.1	-	30.3
5.0	571	328	32.1	349	30.0
4.0	685	-	31.0	-	28.4
3.5	-	399	-	427	-
3.0	781	-	29.0	-	26.1
2.0	865	469	28.1	502	24.8
1.0	904	481	27.3	514	23.9
0.0	929	496	27.0	531	23.5
1.0	903	479	25.5	513	22.0
2.0	859	-	27.8	-	24.6
3.0	774	-	29.7	-	26.8
3.5	-	389	-	415	-
4.0	685	-	31.3	-	28.7
5.0	573	315	-	336	-
5.5	-	-	31.8	-	29.9
6.0	-	-	31.6	-	29.8
6.5	409	-	-	-	-
7.6	-	166	-	177	-

(a) 24.09-g plutonium sample, 94.9 w/o Pu^{239} , 4.8 w/o Pu^{240} , 0.3 w/o Pu^{241} .

(b) 22.93-g enriched uranium sample, 93.1 w/o U^{235} , 6.9 w/o U^{238} .

(c) 64.39-g natural uranium sample, 0.7 w/o U^{235} .

(d) Derived from enriched and natural uranium worths.

Fig. 26. Worths of Pu, U²³⁵ and U²³⁸ versus Radius in Engineering Mockup

F. Reaction-rate Distributions

The variations in fission rates of the fuel isotopes throughout the assembly were measured to provide means of assessing the power distributions, burnup rates, etc., of the various regions of RAPSODIE. These fission profiles were obtained from traverses of small fission chambers, with the traversing mechanism set up as described for the radial and axial traverses of reactivity samples (Section V-E above). In addition to three fission chambers, with platings of enriched uranium, depleted uranium, and plutonium, a small BF₃ proportional counter was also traversed.

1. Axial Counter Traverses

For the axial traverses, the four chambers were each run through 1- and 2-P-16 and count rates were established at a number of axial positions relative to rates on standard counters. The results of these experiments are listed in Table XXII, where for each chamber the values are normalized to 10³ at the core center. The probable errors quoted include the statistical standard deviation of the counts obtained and the uncertainties of dead-time corrections applied.

TABLE XXII. Axial Traverses of Fission and BF₃ Counters through Engineering Mockup

Axial Position, in. from Core Center	Relative Count Rates of Chambers			
	Enriched Uranium ($\pm 1.5\%$)	Depleted Uranium ($\pm 2\%$)	Plutonium ($\pm 1.5\%$)	B ¹⁰ F ₃ ($\pm 2\%$)
20	61	5	51	96
17	115	15	97	173
15	179	-	150	265
14	211	44	177	318
13	245	60	207	366
12	-	-	241	-
11	322	109	-	-
10.5	-	-	299	473
9	416	206	380	-
8	486	297	447	575
7	554	442	546	628
6	657	618	666	701
4	848	827	857	862
2	962	947	969	957
1	1004	979	999	979
0	1000	1000	1000	1000
-1	995	984	994	995
-2	958	953	983	992
-3	-	899	-	-
-4	856	826	857	884
-6	701	588	666	756
-7	602	393	553	705
-8	520	-	461	640
-8.5	-	218	-	-
-9	465	-	403	592
-10	407	127	345	525
-11	340	-	289	458
-12	287	-	237	393
-13	-	41	-	-
-16	123	-	99	188
-18	-	7	-	-
-20	57	-	43	88
-23	-	1	-	-
-24	-	-	18	41
Chamber Specifications				
Diameter, in.	7/16	7/16	7/16	7/16
Active length, in.	2	2	2	2
Plating mass, mg	5.1	-	1.96	-
Composition	93.2% U ²³⁵ 6.8% U ²³⁸	80 ppm U ²³⁵	95% Pu ²³⁹ 5% Pu ²⁴⁰	~90 a/o B ¹⁰ in B

Assuming that the relative counting rates versus axial position of the enriched uranium counter is that of only U²³⁵ fission introduces a small error, a maximum of 0.6% at the core center ($\sigma_f\text{-U}^{238}/\sigma_f\text{-U}^{235} = 0.08$; $0.08 \times 6.9 \text{ w/o} = 0.0055$). Likewise, with the depleted uranium counter the fission of the U²³⁵ content (80 ppm) contributes a maximum of about 1% to the count in the blankets. In Fig. 27 the relative U²³⁵ fission rate is plotted versus axial position, normalized to unity at the core center. For the U²³⁸ traverse, the values have been normalized to the central fission ratio of U²³⁸ to U²³⁵ (0.0802) measured in the physics core.

Figure 28 shows the plutonium fission rate and the B¹⁰ capture rate plotted versus axial position. Here the normalizing value for the plutonium at the center is 1.179, the measured Pu²³⁹/U²³⁵ fission ratio at the

Fig. 27. Fission Rates of U^{235} and U^{238} versus Axial Position in Engineering Mockup

Fig. 28. Pu^{239} Fission and B^{10} Capture Rates versus Axial Position in Engineering Mockup

center of the physics core. A calculated central B^{10} capture-to- U^{235} fission ratio of 0.601 was used to normalize the BF_3 counter traverse plot. Thus, the four graphs in Figs. 27 and 28 refer relative reaction rates to that of fission of U^{235} at the core center.

2. Radial Counter Traverses

Table XXIII gives the results of the counter traverses radially, done through the drawers in column 16 at the front in Half No. 1. The average fission and reaction rates, relative to fission of U^{235} at the core

TABLE XXIII. Radial Traverses of Fission and BF_3 Counters through Engineering Mockup

Radius, in.	Relative Enriched U(a) Count Rate ($\pm 1.5\%$)	Radius, in.	Relative Count Rates of Chambers		
			Depleted U(a) ($\pm 2\%$)	Plutonium(a) ($\pm 1.5\%$)	$B^{10}F_3$ (a) ($\pm 2\%$)
25.44	9	26	0	8	19
24.44	13	25	0	10	20
23.44	18	24	0	12	23
22.44	23	23	1	-	29
21.44	29	22	1	20	36
19.44	47	20	2	32	56
17.44	75	18	4	52	87
15.44	113	16	9	83	132
13.44	177	14	19	129	197
11.44	265	12	43	203	285
9.44	390	10	104	309	405
8.04	511	8.6	199	412	509
7.04	612	7.6	340	517	600
5.44	756	6.6	-	622	693
4.44	866	6	594	682	-
3.44	890	5	706	756	819
2.44	944	4	820	859	875
1.44	983	3	886	910	923
0.44	1002	2	957	961	976
(0)(b)	(1000)(b)	1	966	987	993
0.56	985	0	1000	1000	1000
1.56	972	1	974	989	992
2.56	925	2	956	966	950
3.56	875	3	888	925	917
4.56	813	4	814	867	866
5.56	727	5	705	778	805
6.56	651	6	592	690	-
8.16	500	7	-	588	-
		7.6	340	524	-

(a) Chambers specifications as given at bottom of Table XXII.

(b) By interpolation, not actually measured because of traverse indication misalignment.

center, are plotted versus radius in Figs. 29 and 30. Again the contributions from fissions of the minor constituents in the fission chambers have been ignored in the graphs. In the plutonium counter, the Pu^{240} contributes about 2% of the fissions. This fraction, however, should not vary much with radius, and the relative Pu^{239} fission rate should be the same as the plutonium chamber count rates. Some of the radial fission rate distributions of Table XXIII exhibit slight asymmetries around the core center, with variations of the same order of magnitude as the quoted accuracies. The apparent shifts of the curve centers from the core center range up to 0.1 in. It is quite possible for the assumed fissile-plating centers to be misjudged by 0.1 in., or that other misalignments are responsible. It is also possible that the asymmetries are real, owing to disproportionate fuel loadings, or ZPR-3 cell floor and wall reflection effects. For the purpose of comparing with calculations, a minor realignment of the data, assuming the curve center as the core center, would be justified.

Fig. 29. Fission Rates of U^{235} and U^{238} versus Radius in Engineering Mockup

Fig. 30. Pu^{239} Fission and B^{10} Capture Rates versus Radius in Engineering Mockup

G. Rossi-alpha Measurements

Since this was the first ZPR-3 critical assembly to contain extensive amounts of plutonium, it was of interest to determine the neutron lifetime in the reactor. Therefore, experiments were undertaken to measure Rossi-alpha ($\alpha = -\beta/\ell$, where β is the effective delayed-neutron fraction and ℓ is the prompt-neutron lifetime). The measurements were carried out by R. J. Huber of the Idaho Division using a standard technique.¹⁰

Because a simple core configuration was desired for the experiments, the four boron carbide mockup rods and channels were replaced with radial blanket and fuel removed at the core edge. The end gaps, however, were left in place. The detectors used were installed in blanket drawers at the core edge, in 1-O-13 and 1-Q-19. Figure 31 shows the configuration of the assembly during the alpha measurements. Initially the detectors were two BF_3 proportional counters, but after a few runs these were replaced by two He^3 counters for increased sensitivity. Using a 20-channel time analyzer, with $2\text{-}\mu\text{sec}$ channel widths, the distribution in time separations between initial and secondary counts of pulse pairs from the detectors was determined at a number of subcritical reactor power levels. The power levels involved correspond to reactivities ranging from about 110 to 1175 lh subcritical. Alpha values were derived from the analyzer data for each subcritical level, and the results are shown in Fig. 32. Extrapolation of the graph to delayed critical gives a value of $\alpha = -7.7 \times 10^4 \text{ sec}^{-1}$.

Fig. 31. Core Configuration for Rossi-alpha Measurement

Fig. 32
Rossi-alpha Measure-
ments in Assembly 44

Values of β and ℓ for this assembly arrangement were not calculated. However, values were calculated for a system with the engineering core and physics axial blanket; the calculated lifetime was 8.2×10^{-8} sec, and the calculated β_{eff} was 0.0055. Assuming this β value to be reasonably correct would indicate a measured lifetime of 7.1×10^{-8} sec.

H. Six-rod Engineering Mockup of RAPSODIE

The final phase of the Assembly 44 experiments was the construction of a more realistic, six-rod mockup of RAPSODIE, with the four safety rods and two control rods, and an evaluation of the effectiveness of these rods.

1. Construction of Six-rod Engineering Mockup

To construct a mockup of the six-rod, hexagonal configuration of RAPSODIE (Fig. 2) with the proper rod positioning and for convenience in the rod-worth experiments, it was necessary to translate the axis of the core a half matrix unit horizontally and one-quarter unit vertically in the assembly. This can be seen in the final core configuration of Fig. 36, where the center of the rod pattern lies on the division between the 15th and 16th matrix columns and $1/2$ in. down from the O-P row division. Each of the six mockup rod channels were constructed as partial loadings of one or two matrix cells, whereby each occupied a total of three-fourths of a matrix unit for the length of the assembly. The arrangements provided cross-sectional areas of 22.35 cm^2 for the mockup safety channels and 22.99 cm^2 for the control channels, compared with the RAPSODIE subassembly cross section of 22.46 cm^2 . Average axis-to-rod center radii were 17.68 cm for the mockup safety rods and 18.01 cm for the control-rod mockups, compared to 17.64 cm for the rods in RAPSODIE.

Figure 33 illustrates the axial configuration for the "banked" condition of the control-rod mockups in the O/P-12 and O/P-19 channels

Fig. 33. Axial Configuration for Mockup RAPSODIE Control Rods and Safety Rods in "Banked" Position

and for the safety-rod mockups in the M-14 and M-17 channels. These contained 18-in. active rod lengths. The safety-rod mockups in the R/S-14 and R/S-17 channels had 14-in. active rod sections, extending from +7 to +21 in. from the interface in the "banked" position. Cross-section views for the enriched B_4C safety-rod mockup in the M-17 channel are shown in Fig. 34, illustrating the loadings of the active rod and rod stem sections of the channel and also of the gap, core, or blanket loading in the remainder of the matrix unit. The loadings of matrix units O-12 and P-12 are shown in the sectional views of Fig. 35, illustrating the construction of the natural B_4C active rod and rod stem for a control-rod mockup. Table XXIV gives the compositions for the various channel sections of all the mockups. For the control rods, an axial loading of 17.30 g/cm of natural boron carbide in the active rod section was achieved, compared to 16.24 g/cm specified for RAPSODIE. The active rod section of a mockup safety rod contained 21.76 g of B^{10} per cm length, whereas 22.89 g of B^{10} per cm was specified for RAPSODIE.

Fig. 34. Cross Sections of Mockup RAPSODIE Safety Rod in "Banked" Position in M-17 Channel

Fig. 35. Cross Sections of Mockup RAPSODIE Control Rod in "Banked" Position in O/P-12 Channel

TABLE XXIV. Specifications of Rod Channel Loadings for Six-rod Mockup of RAPSODIE

	Safety-rod Channels in M-14 and M-17		Safety-rod Channels in R/S-14 and R/S-17		Control-rod Channels in O/P-12 and O/P-19	
	Active Rod	Rod Stem	Active Rod	Rod Stem	Active Rod	Rod Stem
Dimensions, in. (cm)						
Length	18.0 (45.72)	33.0 (83.82)	14.0 (35.56)	32.0 (81.28)	18.0 (45.72)	33.0 (83.82)
Height	2.18 (5.54)	2.18 (5.54)	2.18 (5.54)	2.18 (5.54)	3.27 (8.30)	3.26 (8.30)
Width	1.59 (4.04)	1.59 (4.04)	1.59 (4.04)	1.59 (4.04)	1.09 (2.77)	1.09 (2.77)
Axial extension ^a for rods in "banked" position (distances from ZPR Interface), in. (cm)	+25.3 to +7 (64.3 to 17.8)	+7.0 to -26.3 (17.8 to -66.8)	+21.3 to +7 (54.1 to 17.8)	+7.0 to -25.3 (17.8 to -64.3)	+25.3 to +7 (64.3 to 17.8)	+7.0 to -26.3 (17.8 to -66.8)
Composition, g/cc						
B ¹⁰	0.973	-	0.985	-	0.103	-
B ¹¹	0.110	-	0.111	-	0.477	-
Carbon	0.478	-	0.484	-	0.173	-
Sodium	-	0.221	-	-	-	0.215
Aluminum	-	-	-	0.236	0.996	-
Steel ^b	1.152	5.706	1.141	5.516	1.516	5.354
Composition, 10 ²² atoms/cc						
B ¹⁰	5.848	-	5.921	-	0.619	-
B ¹¹	0.610	-	0.618	-	2.656	-
C	2.398	-	2.428	-	0.868	-
Na	-	0.579	-	-	-	0.563
Al	-	-	-	0.526	2.223	-
Fe	0.894	4.430	0.886	4.283	1.177	4.157
Cr	0.240	1.189	0.238	1.150	0.316	1.115
Ni	0.118	0.585	0.117	0.566	0.156	0.549

^aIncluding spring gaps.^bType 304, 72 w/o Fe, 18 w/o Cr, 10 w/o Ni.

The loading of partial core drawers, necessary with the offset rod mockups and to obtain a smooth periphery, reduces the uniformity of core composition. The differences, however, are slight. Quarter-core drawer loadings can be seen with the rod mockups in Figs. 34b and 35b.

2. Critical Configuration of Six-rod Engineering Mockup

After completion of the Rossi-alpha experiments, part of the fuel at the radial edge of the engineering core was removed and the six mockup rods were constructed. Core material was added at the edge in small steps until criticality was achieved. Figure 36 shows the interface views of the core for the six-rod engineering mockup.

For the mockup rods in the "banked" positions, the bottom of the active sections 1 in. below the top of the core, the critical loading contained 36 core unit cells with 120 ± 1 lh excess reactivity available. From later additions of fuel, the worth of a unit core cell replacing blanket at the core edge was determined as 387 ± 3 lh. The just-critical loading would thus be 35.69 unit cells, giving a critical volume of 39.06 ± 0.01 liters for the engineering core in the six-rod mockup of RAPSODIE. For the reference, the average core temperature was $33.5 \pm 1^\circ\text{C}$, the central temperature was 38.5°C , and the ZPR-3 frame was at 27°C .

Fig. 36. Core Configuration for Six-rod Engineering Mockup of RAPSODIE

Compared with the four-rod mockup (Section VI-D above), the core with the six-rod arrangement was larger by about 0.2 liters. The difference could probably be due to a loss in reactivity from a decrease in "self-shielding" within the smaller boron concentrations of the six-rod arrangement as compared to the four-rod arrangement. However, it is also possible that with the various fractional core drawer loadings involved in the six-rod arrangement, the average fuel concentration near the edge was slightly lower than in the composition of the unit cell of the engineering core.

3. Worth of RAPSODIE Control Rod

The effective reactivity control offered by the control rods in RAPSODIE were measured by a simulated motion of a mockup control rod in the assembly. The active section, containing natural B_4C , steel and aluminum, of the mockup rod in the O/P-12 channel was reloaded to extend from 1 to 19 in. axially in Half No. 1. The change represented the motion of the mockup rod inward 6 in. from the banked position, the bottom of the active section moving from 1 in. below the top of the core to the core mid-plane. This produced a reactivity loss of 69.6 ± 0.7 lh.

Next, the mockup rod was moved in another 8 in., placing the active section from 1 in. below the core to 3 in. above the core, with "rod stem" composition in the remainder of the channel. The rod "motion," with the bottom of the active section going from the midplane of the core to "full in" (1 in. below core), was worth -78.7 ± 0.6 lh. Overall, for 14 in. of travel, the rod worth was thus -148.3 ± 1 lh. This gives an average rod calibration, from 1 in. below the top to 1 in. below the bottom of the core, of -10.6 ± 0.7 lh/in. (4.17 lh/cm). Details of these experiments are presented in Table XXV.

TABLE XXV. Reactivity Worths of Simulated Control- and Safety-rod Motions in Six-rod Mockup of RAPSODIE

Exp. No.	Mockup Rod Involved	Description of Experiment, Motion Simulated	Change of Position of B ₄ C Active Rod Section, ^a Distances from ZPR Interface in cm		Worth of Motion, lh
			Extension Before	Extension After	
1	Control rod in O/P-12	Moved control rod from "banked" position (1 in. below top of core) to core midplane	+64.24 ^b to +17.86	+48.98 ^b to +2.60	-70 ± 1
2	Control rod in O/P-12	Moved control rod from midplane to "in" position (1 in. below bottom of core)	+48.98 ^b to +2.60	+28.02 to -17.86	-79 ± 1
3	Safety rod in M-17	Moved safety rod No. 2 from "banked" position (1 in. below top of core) to 1 in. below core midplane	+64.24 ^b to +17.86	+46.46 ^b to +0.08	-493 ± 11
4	Safety rod in M-17	Moved safety rod No. 2 from 1 in. below midplane to "in" position (1 in. below bottom of core)	+46.46 ^b to +0.08	+28.02 to -17.86	-390 ± 18
5	Safety rod in M-14	Moved safety rod No. 1 from "banked" position to "in" position	+64.24 ^b to +17.86	+28.02 to -17.86	-695 ± 16

^aRemainder of channel contains "rod stem," compositions in Table XXIV.

^bIncludes 0.66 cm of spring gap.

4. Flux Depression by Control Rod

Indications of the flux depression by a B₄C control rod were obtained by measuring fission rates in U²³⁵ and U²³⁸ within and near the mockup rod and comparing to rates measured without the rod present. The rates were found from the activations of enriched and depleted uranium foils, which were placed at various radial positions near the midplane and irradiated during a 35 W-hr run. The foils, in pairs of one enriched uranium and one depleted uranium foil, were situated near the front of Half No. 1 drawers in the P-row at 11 radial positions, including the center, core edge, and blanket edge of the natural B₄C-loaded mockup control rod in P-12, the center of the rod stem in P-12, and other core and blanket locations. Table XXVI gives details of the foil placements and the relative activations obtained, normalizing to the foil pair placed in 1-P-15.

TABLE XXVI. Foil Irradiations for Measuring Flux Depression in Mockup Control Rod

Foil Set No.	Placement of Foils ^a	Radius, in. (cm) from Core Axis	Relative Activation ^b	
			Enriched U Foil	Depleted U Foil
1	Top, center of 1-P-15	1.03 (2.6)	100.0	100.0
2	Top, center of 1-P-14	3.22 (8.2)	91.1	89.0
3	Top, center of 1-P-13	5.40 (13.7)	74.6	65.6
4	Core-edge of B ₄ C control rod in 1-P-12	6.64 (16.9)	62.1	46.1
5	Middle of B ₄ C control rod in 1-P-12	7.14 (18.1)	55.0	35.2
6	Blanket-edge of B ₄ C control rod in 1-P-12	7.64 (19.4)	51.7	28.1
7	Top, left side of 1-P-12	8.14 (20.7)	48.3	22.8
8	Top, center of 1-P-17	3.34 (8.5)	90.5	87.7
9	Top, center of 1-P-18	5.5 (14.0)	74.4	64.1
10	Middle of rod stem in 1-P-19	7.14 (18.1)	58.1	34.6
11	Top, right side of 1-P-19	8.14 (20.7)	49.5	22.4

^aLocated axially about 0.5 in. from drawer front.

^bFoil activation per gram as percentage of activation per gram of foil on 1-P-15 (activity includes gammas from products of fission and capture).

Fig. 37. Radial Fission Rate Distributions through Mockup Control-rod Channels

The results of the irradiation have been plotted in Fig. 37 as the fission rates of U^{235} and U^{238} versus radius through the mockup control rod and rod stem, neglecting the contributions of minor isotope fissions and the captures to the activations of the foils. For comparison, the fission rates versus radius obtained from the radial counter traverses of Section VI-F-2 above, which were run through a regular core-blanket boundary, are also graphed. The rates are normalized at the core center to a U^{235} fission rate of unity, using 0.0802 as the U^{238}/U^{235} fission ratio there.

The relative depression of the fission rates by the mockup rod and stem are shown in Fig. 38.

Fig. 38. Relative Depressions of Fission Rate Distributions through Mock-up Control-rod Channels

"rod," the U^{235} fission rate, approximately representing the average flux, is depressed 10% compared with the unperturbed case, 5% compared with the rod stem.

5. Worth of RAPSODIE Safety Rods

The shutdown reactivity provided by the designed safety rods in RAPSODIE was determined from simulated rod-motion experiments with the mockup rods in the M-14 and M-17 channels. Included in the investigation was a measurement of the influence of a rod's axial position upon the worth of an adjacent rod, or the "shadowing" effect. The worths of the various substitutions involved were obtained by the subcritical method without increasing the core volume. The accuracies quoted are the weighted standard deviations from the mean of the several measurements of subcriticality for each experiment.

First, the enriched B_4C , active section of the mockup safety rod in the M-17 channel was moved inward 7 in. This represented about half the travel for a RAPSODIE rod in typical operation, the bottom of the poison section going from 1 in. below the upper end of the core to 1 in. below the core midplane. The resulting reactivity loss was 493 ± 11 lh.

Next, the mockup rod in M-17 was moved to the "full-in" position, the bottom of the B_4C section extending 1 in. beyond the end of the core in Half No. 2. Thus, the 18-in. active section was situated between -8 and +10 in. axially from the core midplane. This additional 7 in. of travel produced a reactivity loss of 390 ± 18 lh. The total worth of the mockup safety rod, inserted from the "banked" position to the "full-in" position was, therefore, -883 ± 25 lh. This value pertains to the 14-in. motion of a solitary safety rod, the other three safety rods and one control rod remaining in the "banked" position, the other control rod being inserted. The essential detail of the mockup-rod composition in the experiment is the poison loading per unit length, 55.25 g of B^{10} per in. (21.75 g/cm).

These were obtained from the ratios of the rates with the mockups to the rates on the unperturbed curve at the positions concerned. For the U^{238} fission rates, approximately representing the flux of neutrons with energies above 1 MeV, the depressions in the B_4C rod and in the SS-Na stem are about the same. The U^{235} fission rate is also seen to be depressed by the rod stem when compared to the unperturbed rates, presumably because of the differences between the core boundary in the two cases. In the natural B_4C

The final experiment was the simulated insertion of a safety rod in the M-14 mockup channel. For this, the mockup safety rod in the M-17 channel and the mockup control rod in the O/P-12 channel were left in the "full-in" position. Moving the enriched, B_4C -loaded active section of the M-14 mockup rod inward 14 in., to the "full-in" position, gave a reactivity loss of 695 ± 16 lh. This value is about 25% less than the previously cited worth for a solitary safety rod, indicating a substantial interaction or shadowing effect between the rods for the manner in which these experiments were done. The principal cause of the decreased worth of the second rod "inserted" was probably the shift of the effective reactivity axis of the reactor produced by the insertion of the first rod. The axial regional substitutions involved in all the foregoing experiments of simulated safety- and control-rod motions are presented in detail in Table XXV.

Based on the average results of the two mockup safety-rod experiments, the total reactivity shutdown provided by four rods in RAPSODIE, containing 221 g of B^{10} per in. total and traveling 14 in. in from the "banked" position, would be 3156 ± 40 lh, or about 5.7% $\Delta k/k$. Assuming no shadowing effect or reactive axis shift with insertion of four rods would indicate up to 6.4% $\Delta k/k$ available for shutdown.

ACKNOWLEDGMENTS

The successful conduct of the RAPSODIE critical studies required the assistance of several people of Argonne National Laboratory and of the French fast reactor group, for which the authors are grateful. Calculations for the systems were carried out by W. G. Davey and D. A. Meneley of Argonne and by J. M. Chaumont and others at Cadarache. Foil irradiations were analyzed under the direction of E. R. Ebersole and R. J. Armani. Members of the ZPR-3 staff assisting with the experiments included J. K. Long, P. I. Amundson, R. L. McVean, and J. M. Gasidlo, and credit is due also to the ZPR-3 technicians. The Rossi-alpha experiments were conducted by R. J. Huber. Particular acknowledgment is due to F. W. Thalgott for the overall organization of the project and for guidance during the program.

REFERENCES

1. P. R. Wustner, *RAPSODIE, Reacteur Experimental α' Neutrons Rapides*, Atompraxis 9, 493 (1963).
2. G. Vendryes, *RAPSODIE, Energie Nucléaire* 3, 25 (1961).
3. B. C. Cerutti *et al.*, *ZPR-3, Argonne's Fast Critical Facility*, Nucl. Sci. Eng. 1, 126 (1956).
4. B. J. Toppel, *Sources of Error in Reactivity Determinations by Means of Asymptotic Period Measurements*, Nucl. Sci. Eng. 5, 88 (1959).
5. *Reactor Physics Constants*, ANL-5800, Second Edition (July 1963).
6. P. I. Amundson, R. Jiacoletti, J. K. Long, and R. L. McVean, *A Sub-critical Plutonium-fueled Fast Reactor Core (ZPR-3 Assembly 37)*, ANL-6595 (July 1962).
7. F. S. Kirn, *An Absolute Fission Counter*, American Nuclear Society, 2nd Winter Meeting (1957).
8. W. G. Davey, *A Critical Comparison of Measured and Calculated Fission Ratios for ZPR-3 Assemblies*, ANL-6617 (Sept 1962).
9. J. K. Long *et al.*, *Fast Neutron Power Reactor Studies with ZPR-3*, Second U. N. International Conference on Peaceful Uses of Atomic Energy, Geneva, Switzerland 12, 119 (1958).
10. G. S. Brunson *et al.*, *Measuring the Prompt Period of a Reactor*, Nucleonics 15(11), 32 (1957).

o

u

p

t

y

r