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DOPPLER EFFECT MEASUREMENTS, SODIUM VOID COEFFICIENTS,
AND CRITICAL STUDIES OF A 5,000-LITER FAST POWER
BREEDER REACTOR BY THE TWO-ZONE METHOD
(ZPR-III Assembly 43)

by

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Idaho Division

May 1964

Operated by The University of Chicago
under
Contract W-31-109-eng-38
with the
U. S. Atomic Energy Commission

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ABSTRACT

Assembly 43 is the second two-zone core built in the ZPR-III facility. The central zone and the low-density axial blanket were designed to simulate a large power-breeder reactor. Criticality was achieved by an annular driver, while an interposed buffer modified the spectrum of the neutrons diffusing from the driver to the central zone.

Experiments included sodium void coefficients for central substitutions and for full-length axial measurements. Other reactivity measurements included central reactivity coefficients and core-length column substitutions in the central zone and the annular driver. Spectral indices were measured, and reaction-rate traverses and activation experiments were performed. The reactivity change due to the Doppler effect caused by heating a large, depleted, uranium oxide sample was also determined in the experimental program.

The experimental results indicate that the measurements can be used for checking calculations of dilute fast cores, as well as evaluating the two-zone experimental technique.

I. INTRODUCTION

Assembly 43 is of interest because of the difficulty in simulating a 5,000-liter power reactor in small-size critical facilities. The driver-buffer concept reduces the mockup to a manageable size while maintaining the highly degraded spectrum in the center of the core region. This permits the measurement of spectral indices and reactivity effects due to material replacements.

The feasibility of the two-zone concept was evaluated in Assembly 42,⁽¹⁾ the first two-zone core assembled in the ZPR-III. The composition of the central zone was identical to that of a previous ZPR-III assembly; hence, calculational and experimental techniques could be evaluated by comparing the data from the two cores.

The composition of the central zone in Assembly 43 is typical of a projected fast power breeder reactor. The simulated fuel is uranium monocarbide with a high proportion of U^{238} to promote internal breeding. The basic structural material and coolant are stainless steel and sodium, respectively.

II. DESCRIPTION OF ASSEMBLY 43

A. The ZPR-III Facility

The ZPR-III is a critical assembly divided into two halves for ease of loading and low k_{eff} in the separated or shutdown state. The principal feature of each half is a matrix consisting of a 31×31 array of horizontal stainless steel tubes, each 2.175 in. wide, 2.183 in. high, and 33.5 in. long. The neutron and gamma detectors that drive the reactor instrumentation are located on top of the two matrices. Five fuel-bearing control-safety rods enter each matrix from the rear. Figure 1 is a photograph of the machine. A complete description of the ZPR-III facility has been given by Cerutti et al.⁽²⁾

Fig. 1. The ZPR-III Facility

Reactor cores are simulated in the ZPR-III by loading drawers, containing core and blanket material, into both matrices. Typical drawer loadings are shown in Fig. 2. Each drawer is loaded with $\frac{1}{8}$ -in.-thick plates of various materials in the desired portions, usually to a length of half the core. Then the core drawers are inserted into the matrices in the geometric pattern required to form the assembly. Each matrix contains half the core, usually in the form of half a horizontal cylinder; the halves are brought together to form the complete core assembly.

B. Design of the Core

CORE													LOW-DENSITY AXIAL BLANKET						
						SODIUM													
						SODIUM													
DEPLETED URANIUM																			
GRAPHITE																			
1/16-IN ENRICHED URANIUM						SODIUM													
						DEPLETED URANIUM							DEPLETED URANIUM STAINLESS STEEL						
1/16-IN STAINLESS STEEL						SODIUM													
GRAPHITE																			
DEPLETED URANIUM																			
						SODIUM													

* The calculations for composition and critical size were performed by George Fischer, Reactor Physics Division, Argonne, Illinois.

Fig. 4

Loading Master,
Buffer Region

Fig. 5
Loading Master,
Driver Region

The central zone is 86.51 cm long and roughly circular in cross section. The buffer is the same length as the central zone and was loaded to be at least one drawer thick in the radial direction. The driver is the same length as the central zone and buffer, and was loaded radially outward until the reactor became critical. A cross section of the assembly is shown in Fig. 6, and the dimensions are given in Table I.

Fig. 6

Cross Section of Core

Table I
COMPOSITION AND PHYSICAL PROPERTIES OF ASSEMBLY 43

		Central Zone	Low-density Axial Blanket	Buffer	Driver	High-density Blanket
Outer radius (cm)		24.40	24.40	31.39	45.26	~75.7
Length (cm)		86.51	38.1 ^a	86.51	86.51	38.1 ^b , 27.9 ^c
Volume (l)		161.72	71.35 ^a	106.05	288.98	
Volume Fractions						
	Assumed Density					
U ²³⁵	18.75	0.02376	0.0003186	0.000945	0.09506	0.0019
U ²³⁸	19.0	0.1523	0.1500	0.4518	0.00678	0.8330
C	1.67	0.09936	0.09908	0.3478	0.3935	-
Na	0.84	0.4820	0.4794 ^d	-	-	-
SS (304)	7.78	0.1955	0.2191	0.09166	0.4033	0.0731
Atomic Density (Atoms/cm ³ x 10 ⁻²⁴)						
	Atomic Mass					
U ²³⁵	235.12	0.001141	1.53 x 10 ⁻⁵	0.000045	0.004566	0.0000912
U ²³⁸	238.13	0.007318	0.007207	0.02171	0.000326	0.09998
C	12.011	0.008321	0.008298	0.02912	0.03295	-
Na	22.99	0.01061	0.01055 ^d	-	-	-
SS (304)	55.47	0.01651	0.01850	0.00774	0.03405	0.00619

^aValues are for each end only.

^bLength of the extended high-density axial blanket.

^cNormal length of high-density axial blanket.

^dThe quoted values are for the part of the axial blanket that does not contain aluminum.

The low-density axial blanket behind the central zone is identical in composition to the central zone except that the enriched uranium was replaced with stainless steel.

Since the low-density axial blanket is fairly transparent to neutrons, it was loaded to the end of the blanket drawers. However, the sodium inventory was insufficient to build up the entire volume; therefore, 45%-density aluminum was used to simulate sodium in the last 3 in. of the axial blanket in half No. 1. The buffer and driver are completely blanketed with depleted uranium. Since the low-density axial blanket was longer than the normal length of high-density blanket, an additional annular region of depleted uranium was extended to maintain continuity in the radial direction. This is illustrated in Fig. 7, a section taken along the cylindrical axis of the assembly, and Fig. 8, a cross section of the axial blankets. The composition and physical properties of the assembly are listed in Table I.

Fig. 7
Vertical Section of Assembly 43
through Column 16 of ZPR-III
Matrix

Fig. 8
Cross Section of Axial Blankets

C. Critical Parameters

The first loading in the approach to criticality of Assembly 43 consisted of the central zone, the buffer, and the 10 control-safety rods in the driver. The remainder of the driver was loaded with depleted uranium at this point. Each rod contained three columns of enriched uranium, one more than the loading per drawer in the remainder of the driver. This was done to provide a total reactivity control in the rods of at least 1.8% k.

The approach to criticality was performed by first building the driver radially outward to the expected critical volume with one column of depleted uranium per drawer replacing one column of enriched uranium. The depleted uranium was then replaced with enriched uranium radially outward to continue the approach to criticality. This loading technique was employed to prevent large changes in the worth of the inner portion of the driver as its overall thickness was increased. Such a change in worth could conceivably cause a reverse bend in the inverse count-rate curve during the approach to criticality. The reactor became critical with 72.062 kg of U^{235} in the central zone, 1.879 kg in the buffer, and 515.062 kg in the driver, with the control rod withdrawn 3.092 in.

The control rod was calibrated by the usual method of moving the rod and measuring the corresponding asymptotic period. The inhour curve used for converting period to reactivity and the resultant rod calibration are shown in Figs. 9 and 10, respectively.

The worth of edge driver material was then measured as 29.2 lh/kg of U^{235} . The just-critical mass, including the U^{235} in the buffer, was 588.1 kg of U^{235} .

Fig. 9
Inhour Curve for
Assembly 43

Fig. 10
Control Rod Calibration Curve

D. Assembly 43A

After the experimental program, including the Doppler measurements, was completed on Assembly 43, the central zone and buffer regions were modified to provide a softer spectrum for additional Doppler effect measurements. Two of the $\frac{1}{4}$ -in. columns of canned sodium were replaced with graphite in the central zone, as was the first 4 in. of the axial blanket. The remainder of the axial blanket was left unchanged. Three columns of depleted uranium were replaced with graphite in the buffer. The modified Assembly 43 was referred to as the Assembly 43A. The loading masters of Assembly 43A are shown in Figs. 11 and 12.

A small wedge section of the central zone was changed to the new loading to evaluate the worth of the change. The remainder of the core was then changed, and only a small increase in reactivity was noted. The buffer was then changed in a similar manner. The critical mass of Assembly 43A was 550.7 kg of U^{235} , and the volume of the driver was reduced

to 274 liters. The reactivity change was due principally to the modification of the buffer. A cross-sectional outline of Assembly 43A is shown in Fig. 13. The modified compositions are given in Table II.

Fig. 11

Loading Master,
Central Zone of
Assembly 43A

Fig. 12. Loading Master, Buffer Region of Assembly 43A

Fig. 13

Cross Section of Assembly 43A

Table II

COMPOSITION OF ASSEMBLY 43A

	Volume Fractions			Atoms/cm ³ x 10 ⁻²⁴		
	Central Zone	Axial Blanket (First 4 in.)	Buffer	Central Zone	Axial Blanket (First 4 in.)	Buffer
U ²³⁵	0.02376	0.00032	0.00063	0.001141	0.000015	0.000030
U ²³⁸	0.1523	0.1500	0.3012	0.007318	0.007207	0.01447
C	0.2981	0.2973	0.4848	0.02496	0.02489	0.04060
Na	0.2892	0.2876	-	0.006364	0.00633	-
Stainless Steel (Type 304)	0.1642	0.1880	0.09166	0.01386	0.01587	0.00774

III. REACTION-RATE TRAVERSES

For the radial traverses, the blanket and core drawers in half No. 2, Column 16 (see Fig. 6), were modified to accept a 1.27-cm-OD, vertical thimble, approximately 0.8 cm back of the midplane. The affected columns of materials in the high-density blanket, driver, buffer, and central zone drawers were simply pushed back to provide room for the thimble; i.e., materials at the back of the drawer were removed. The loading master for the central zone is shown in Fig. 14. Thus, as the detectors were driven down through the thimble, they passed between large parallel sheets of core material formed by the vertically loaded plates in each drawer lining up throughout the core.

Fig. 14. Central Zone Loading Master for
Radial Reaction-rate Traverses

The axial traverses were performed by driving the same detectors horizontally through a thimble loaded in the 1-P-16 and 2-P-16 matrix locations (see Fig. 6). In this case, the fuel column was moved to the side of the drawer. The loading of the central zone drawer is shown in Fig. 15. The low-density blanket was loaded in a similar manner.

Fig. 15. Front View of Central Zone Loading Master for Axial Traverses

The detectors are 1.1 cm in diameter, with active volumes 5 cm in length. The traverse mechanism was set so that the center of each traverse occurred when the center of the active volume of the counter was located at the center of the core. Thus, the data represent reaction rates integrated over 2.5 cm on either side of the indicated position.

The reaction-rate traverses were performed with a BF_3 proportional counter and four fission counters. Each fission counter contains several isotopes, as listed in Table III, but no correction has been applied to the data that are listed under the principal isotope of each counter. The normalized data are presented in Tables IV and V and as graphs in Figs. 16 through 25.

Table III

ISOTOPIC ANALYSES OF THE ZPR-III TRAVERSE COUNTERS

Principal Isotope	Composition
U^{235}	93.2% U^{235} , 5.8% U^{234} , 1.0% U^{238}
U^{234}	89.34% U^{234} , 7.72% U^{235} , 2.89% U^{238}
U^{238}	99.99% U^{238} , 80 ppm U^{235}
Pu^{239}	94.92% Pu^{239} , 4.75% Pu^{240} , 0.32% Pu^{241} , 0.01% Pu^{242}

Table IV

NORMALIZED RADIAL REACTION-RATE TRAVERSES IN ASSEMBLY 43

Position (cm)	U^{234}	U^{235}	U^{238}	Pu^{239}	BF_3	Position (cm)	U^{234}	U^{235}	U^{238}	Pu^{239}	BF_3
0	0.996 $\pm$ 1.3%	1.000 $\pm$ 0.6%	1.008 $\pm$ 0.8%	1.006 $\pm$ 0.8%	1.007 $\pm$ 0.7%	31.75	1.261 $\pm$ 1.4%	-	-	-	-
2.54	0.997 $\pm$ 1.3%	1.000 $\pm$ 0.6%	1.008 $\pm$ 0.8%	0.992 $\pm$ 0.8%	0.998 $\pm$ 0.7%	33.02	1.336 $\pm$ 1.3%	0.995 $\pm$ 0.9%	1.925 $\pm$ 0.9%	1.048 $\pm$ 1.2%	0.911 $\pm$ 1.0%
5.08	0.996 $\pm$ 1.5%	1.014 $\pm$ 0.9%	0.990 $\pm$ 1.0%	1.001 $\pm$ 1.2%	1.002 $\pm$ 1.0%	34.29	1.351 $\pm$ 1.3%	-	1.950 $\pm$ 0.9%	-	-
10.16	0.991 $\pm$ 1.5%	1.000 $\pm$ 0.9%	0.976 $\pm$ 1.1%	1.001 $\pm$ 1.2%	1.006 $\pm$ 1.0%	35.56	1.338 $\pm$ 1.3%	0.931 $\pm$ 0.9%	1.907 $\pm$ 0.8%	1.040 $\pm$ 1.2%	0.841 $\pm$ 1.0%
15.24	1.000 $\pm$ 1.5%	1.018 $\pm$ 0.9%	0.977 $\pm$ 1.1%	1.012 $\pm$ 1.2%	1.035 $\pm$ 1.0%	36.83	-	0.888 $\pm$ 0.9%	1.933 $\pm$ 0.9%	-	0.804 $\pm$ 1.0%
17.73	-	1.031 $\pm$ 0.9%	-	-	1.040 $\pm$ 1.0%	39.12	1.221 $\pm$ 1.4%	0.833 $\pm$ 0.9%	1.815 $\pm$ 0.9%	0.945 $\pm$ 1.2%	0.712 $\pm$ 1.0%
20.32	1.006 $\pm$ 1.5%	1.024 $\pm$ 0.9%	0.966 $\pm$ 1.1%	1.013 $\pm$ 1.2%	1.058 $\pm$ 1.0%	41.66	1.074 $\pm$ 1.4%	0.727 $\pm$ 0.9%	1.478 $\pm$ 1.0%	0.808 $\pm$ 1.2%	0.620 $\pm$ 1.0%
22.30	0.991 $\pm$ 1.5%	1.040 $\pm$ 0.9%	0.997 $\pm$ 1.1%	1.034 $\pm$ 1.2%	1.054 $\pm$ 1.0%	44.20 ^c	0.838 $\pm$ 1.6%	0.627 $\pm$ 0.9%	1.052 $\pm$ 1.0%	0.700 $\pm$ 1.3%	0.527 $\pm$ 1.0%
24.84 ^a	0.996 $\pm$ 1.5%	1.034 $\pm$ 0.9%	0.952 $\pm$ 1.1%	1.032 $\pm$ 1.2%	1.056 $\pm$ 1.0%	46.74	0.573 $\pm$ 1.7%	0.457 $\pm$ 0.9%	0.585 $\pm$ 1.4%	0.550 $\pm$ 1.3%	0.414 $\pm$ 1.0%
27.38	1.052 $\pm$ 1.5%	1.031 $\pm$ 0.9%	1.047 $\pm$ 1.1%	1.059 $\pm$ 1.2%	1.060 $\pm$ 1.0%	50.80	0.349 $\pm$ 2.1%	0.336 $\pm$ 1.0%	0.241 $\pm$ 2.8%	0.379 $\pm$ 1.5%	0.287 $\pm$ 1.0%
29.21	-	-	-	-	1.008 $\pm$ 1.0%	55.88	-	0.210 $\pm$ 1.0%	0.093 $\pm$ 4.5%	-	0.184 $\pm$ 1.0%
30.45 ^b	1.227 $\pm$ 1.4%	1.017 $\pm$ 0.9%	1.505 $\pm$ 1.0%	1.062 $\pm$ 1.2%	0.980 $\pm$ 1.0%	60.96	0.103 $\pm$ 3.7%	0.131 $\pm$ 1.1%	0.047 $\pm$ 6.2%	0.142 $\pm$ 2.0%	0.116 $\pm$ 1.1%

^aCore-buffer Interface

^bBuffer-driver Interface

^cDriver-blanket Interface

Table V

NORMALIZED AXIAL REACTION-RATE TRAVERSES IN ASSEMBLY 43

Position (cm)	U ²³⁴	U ²³⁵	U ²³⁸	Pu ²³⁹	BF ₃	Position (cm)	U ²³⁴	U ²³⁵	U ²³⁸	Pu ²³⁹	BF ₃
0	1.005 ± 0.8%	1.005 ± 1.1%	1.000 ± 1.1%	1.008 ± 1.2%	1.000 ± 1.0%	38.10	0.527 ± 1.5%	0.565 ± 1.1%	0.470 ± 1.9%	0.576 ± 1.3%	0.590 ± 1.0%
2.54	0.991 ± 1.2%	1.007 ± 1.1%	0.990 ± 1.1%	0.994 ± 1.2%	1.000 ± 1.0%	40.64	0.456 ± 1.5%	0.546 ± 1.1%	0.427 ± 1.9%	0.537 ± 1.3%	0.553 ± 1.0%
5.08	0.984 ± 1.2%	0.982 ± 1.1%	0.996 ± 1.1%	0.998 ± 1.2%	1.003 ± 1.0%	43.18 ^a	0.400 ± 1.7%	0.515 ± 1.2%	0.320 ± 1.9%	0.498 ± 1.3%	0.538 ± 1.0%
7.62	0.973 ± 1.2%	0.980 ± 1.1%	0.985 ± 1.1%	0.983 ± 1.2%	0.972 ± 1.0%	45.72	0.347 ± 1.7%	0.501 ± 1.2%	0.227 ± 1.8%	0.463 ± 1.3%	0.522 ± 1.0%
17.78	0.871 ± 1.3%	0.889 ± 1.1%	0.880 ± 1.1%	0.904 ± 1.2%	0.880 ± 1.0%	48.26	0.294 ± 1.7%	0.454 ± 1.2%	0.171 ± 3.5%	0.434 ± 1.4%	0.501 ± 1.0%
27.94	0.732 ± 1.4%	0.744 ± 1.1%	0.727 ± 1.4%	0.752 ± 1.2%	0.736 ± 1.0%	53.34	-	-	-	-	0.438 ± 1.0%
33.02	0.640 ± 1.4%	0.651 ± 1.1%	-	0.658 ± 1.2%	0.659 ± 1.0%	58.42	0.166 ± 2.4%	0.305 ± 1.3%	0.063 ± 4.8%	0.285 ± 1.4%	0.353 ± 1.0%
35.92	-	-	-	-	0.618 ± 1.0%	68.58	0.092 ± 30.0%	-	-	0.160 ± 1.9%	0.202 ± 1.0%

^aCore-axial Blanket Interface

Fig. 16

U²³⁵ Radial Reaction-rate TraverseFig. 17
U²³⁴ Radial Reaction-rate Traverse

Fig. 18. U^{238} Radial Reaction-rate Traverse

Fig. 19. Pu^{239} Radial Reaction-rate Traverse

Fig. 20. BF_3 Radial Reaction-rate Traverse

Fig. 21. U^{235} Axial Reaction-rate Traverse

Fig. 22. U^{234} Axial Reaction-rate Traverse

Fig. 23. U^{238} Axial Reaction-rate Traverse

Fig. 24. Pu^{239} Axial Reaction-rate Traverse

Fig. 25. BF_3 Axial Reaction-rate Traverse

IV. SPECTRAL INDICES

A. Central Fission Ratios

The experimental fission ratios were determined from measurements made with absolute fission counters 2 in. in diameter by 1 in. long. Three types of counters, all of the same basic design,⁽³⁾ were used in the experiment. The Kirn counter is the design basis. The thick-walled, gas-flow counter differs from the Kirn counter only in that it is not sealed (requiring a steady flow of the counting gas), and the fissile foil is removable. The thin-walled, gas-flow counter is similar to the thick-walled counter, except that the counter body walls are very thin to reduce scattering effects, and the cable connection is moved an additional 2 in. from the counter body.

Measurements with the thick-walled counter were made with a single counter (containing an enriched U^{235} foil) loaded into one of the machine halves. Uranium foils of various isotopic concentrations were alternately placed in a second counter body loaded opposite the U^{235} counter. The measurements with the thin-walled counters were made using the same foils placed in counters loaded into the same positions. Kirn counters were used for the plutonium measurements.

The data were then reduced to isotopic values by a set of simultaneous equations involving the isotopic concentrations of the foils. Since no data were taken using a U^{234} foil, the experimental fission ratios were determined by using the calculated U^{234}/U^{235} value. A reduction of this ratio by 14%, as suggested by Davey,⁽³⁾ changed the experimental values by less than 0.05%. Therefore, the possible error in the reported fission ratios due to using the calculated U^{234}/U^{235} ratio is much less than the statistical deviations in the total counts. The data are presented in Table VI.

Table VI

CENTRAL FISSION RATIOS, ZPR-III ASSEMBLY 43

$\sigma_i/\sigma_{U^{235}}$	Thick-walled Counter	Thin-walled Counter	Calculated ^(a)
U^{233}	1.42 $\pm$ 1.2%	-	1.51
U^{236}	0.052 $\pm$ 1.2%	0.0537 $\pm$ 1.3%	0.0587
U^{238}	0.0200 $\pm$ 1.1%	0.0217 $\pm$ 1.2%	0.0254
Pu^{239}	0.99 $\pm$ 1.2%	-	1.05
Pu^{240}	0.183 $\pm$ 1.3%	-	0.201

(a) These values were determined from a fundamental mode calculation of the composition of the central zone using ANL cross-section Set 635 as derived by Davey.⁽⁴⁾

B. Uranium Foil and Sodium Activations

Natural and enriched uranium foils were activated with sodium and aluminum samples at various positions in the reactor. Three natural and three enriched uranium foils were each wrapped in packets of aluminum foil, so that when the packet was loaded across a drawer, the three foils were distributed across the unit cell. The sodium samples consisted of NaNO_3 powder loaded into 0.952-cm-diameter x 5.08-cm-long aluminum capsules. The actual dimensions of the NaNO_3 were 0.793 cm diameter by 4.61 cm long. The aluminum was irradiated in the form of strips 0.307 cm thick by 1.27 cm wide by 5.08 cm long.

The material arrangements in the central zone and low-density axial blanket drawers were changed to form 0.952-cm-thick voids for the full 5.08 cm width and height of the drawers at several axial positions. One of each type of sample or packet was placed in the void across the drawer and the sample location was measured from the midplane to the center of the void. The radial sample locations were similar, except that the void was loaded only at the face of the drawer. In the buffer and radial blankets, the sample void was 5.08 cm wide by 0.952 cm deep by only 1.27 cm high. The voids in the driver were 5.08 cm wide by 0.952 cm deep by 2.54 cm high. For the radial activations, the sample locations were measured to the center of the sodium capsules that were loaded across the bottom of the sample voids.

Radiochemical analysis of the natural and enriched uranium foils was performed at Argonne, Illinois. The total U^{235} and U^{238} fissions and captures at each location were determined from the total induced activity of Mo^{99} and Np^{239} in all three foils of each packet. The errors tabulated with the uranium foil results were determined at Argonne, Illinois.*

The sodium capsules and aluminum strips were counted for Na^{24} gamma activity in a scintillation well counter. Aluminum transmutes to Na^{24} by an (n,α) reaction above a threshold of about 6 MeV. Thus, aluminum acts as an additional high-energy threshold detector. The sodium activation was determined by subtracting the Na^{24} activity due to the aluminum in each capsule from the total measured activity of that capsule. The efficiency of the well counter had been determined previously by counting similar samples whose absolute activities had been determined by a $4\pi\beta\text{-}\gamma$ absolute counter and a $4\pi\beta$ liquid scintillator absolute counter.** The errors listed in Table VII, with the results on sodium and aluminum, represent statistics on the counting rates and the degree of confidence in the counting equipment.

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*Radiochemical analysis by R. J. Armani, Reactor Physics Division, Argonne, Illinois.

**Calibration by A. DeVolpi, Reactor Physics Division, Argonne, Illinois.

Table VII

ACTIVATION RESULTS RELATIVE TO U^{235} FISSIONS
AT CENTER OF ASSEMBLY 43

$$\frac{\bar{\sigma}_f(U^{238})}{\bar{\sigma}_f(U^{235})} = 0.0224 \pm 5\%$$

$$\frac{\bar{\sigma}_{n,\gamma}(U^{238})}{\bar{\sigma}_f(U^{235})} = 0.114 \pm 4\%$$

$$\frac{\bar{\sigma}_{n,\alpha}(Al)}{\bar{\sigma}_f(U^{235})} = 3.14 \times 10^{-5} \pm 10\%$$

$$\frac{\bar{\sigma}_{n,\gamma}(Na)^*}{\bar{\sigma}_f(U^{235})} = 5.8 \times 10^{-4} \pm 10\%$$

---

*The significance of this value is described in the text.

The sodium activation results have several uncertainties and should be regarded strictly as preliminary. The sodium sample is at least two mean-free-path lengths in diameter at the 2.95-keV resonance. Simple calculations indicate that the average 2.95-keV resonance flux in the sodium capsule may be only 40% of the resonance flux entering the surface. However, the resonance is sufficiently broad so that neutrons can be down-scattered in energy by the sample and still be within the sample volume and the sodium resonance. This implies that a careful analysis with a fine-energy group structure across the resonance is necessary. In addition, the sodium in the reactor will cause a general flux depression at the sodium resonance. Since the actual reactor is inhomogeneous, the atomic density of the sodium in the cans is greater than that for the homogeneous case, and the resonance flux depression is greater in the sodium cans. Therefore, the resonance depression may be less in the capsule, since it has a lower atomic density of sodium, than in the sodium cans. Thus, sodium activations in a core containing significant quantities of sodium require complex analysis.

The results of central activations relative to U^{235} fissions are listed in Table VII. The relative activations and normalized activations relative to the U^{235} fission rate are listed in Tables VIII and IX, respectively.

Table VIII

RELATIVE SODIUM, ALUMINUM, AND URANIUM ACTIVATIONS IN ASSEMBLY 43

Position (cm)	$U^{235} (\bar{\sigma}_f)$	$U^{238} (\bar{\sigma}_f)$	$U^{238} (\bar{\sigma}_{n,\gamma})$	Na ($\bar{\sigma}_{n,\gamma}$)	Al ($\bar{\sigma}_{n,\alpha}$)
Axial Measurements					
0.48	$1.00 \pm 3\%$	$1.00 \pm 4\%$	$1.00 \pm 3\%$	$1.00 \pm 3\%$	$1.00 \pm 3\%$
16.66	$0.91 \pm 4\%$	$0.88 \pm 5\%$	$0.91 \pm 4\%$	$0.94 \pm 4\%$	$0.90 \pm 4\%$
35.41	$0.63 \pm 4\%$	$0.55 \pm 5\%$	$0.66 \pm 4\%$	$0.67 \pm 4\%$	$0.59 \pm 4\%$
43.26	Core-blanket Interface				
43.97	$0.51 \pm 4\%$	$0.289 \pm 5\%$	$0.53 \pm 4\%$	$0.57 \pm 4\%$	$0.354 \pm 4\%$
53.34	$0.382 \pm 4\%$	$0.107 \pm 7\%$	$0.43 \pm 4\%$	$0.47 \pm 4\%$	$0.115 \pm 4\%$
69.70	$0.168 \pm 8\%$	$0.018 \pm 10\%$	$0.173 \pm 4\%$	$0.189 \pm 5\%$	$0.034 \pm 5\%$
Radial Measurements					
2.29	$1.00 \pm 3\%$	$1.00 \pm 4\%$	$1.00 \pm 3\%$	$1.00 \pm 3\%$	$1.00 \pm 3\%$
8.97	$1.01 \pm 4\%$	$0.82 \pm 5\%$	$1.03 \pm 4\%$	$1.09 \pm 4\%$	$0.95 \pm 4\%$
14.50	$0.99 \pm 5\%$	$0.85 \pm 5\%$	$1.09 \pm 4\%$	$1.10 \pm 4\%$	$0.98 \pm 4\%$
20.04	$0.98 \pm 5\%$	$0.88 \pm 5\%$	$1.07 \pm 4\%$	$1.22 \pm 4\%$	$1.00 \pm 4\%$
24.87	Core-buffer Interface				
28.14	$0.97 \pm 5\%$	$1.06 \pm 5\%$	$1.02 \pm 4\%$	$1.75 \pm 4\%$	$1.13 \pm 4\%$
30.40	Buffer-driver Interface				
33.66	$0.95 \pm 4\%$	$1.61 \pm 5\%$	$0.96 \pm 4\%$	$1.72 \pm 4\%$	$1.75 \pm 4\%$
39.19	$0.78 \pm 4\%$	$1.27 \pm 5\%$	$0.81 \pm 4\%$	$1.30 \pm 4\%$	$1.48 \pm 4\%$
44.25	Driver-blanket Interface				
46.00	$0.50 \pm 4\%$	$0.64 \pm 5\%$	$0.51 \pm 4\%$	$0.65 \pm 4\%$	$0.66 \pm 4\%$

Table IX

NORMALIZED ACTIVATION RESULTS RELATIVE TO
 U^{235} FISSION RATE IN ASSEMBLY 43

Position (cm)	$\frac{\bar{\sigma}_f (U^{238})}{\bar{\sigma}_f (U^{235})}$	$\frac{\bar{\sigma}_{n,\gamma} (U^{238})}{\bar{\sigma}_f (U^{235})}$	$\frac{\bar{\sigma}_{n,\gamma} (Na)}{\bar{\sigma}_f (U^{235})}$	$\frac{\bar{\sigma}_{n,\alpha} (Al)}{\bar{\sigma}_f (U^{235})}$
Axial Measurements(a)				
0.48	$1.00 \pm 4\%$	$1.00 \pm 3\%$	$1.00 \pm 3\%$	$1.00 \pm 3\%$
16.66	$0.97 \pm 5\%$	$1.01 \pm 4\%$	$1.04 \pm 4\%$	$1.00 \pm 4\%$
35.41	$0.90 \pm 5\%$	$1.08 \pm 4\%$	$1.10 \pm 4\%$	$0.96 \pm 4\%$
43.97	$0.56 \pm 5\%$	$1.03 \pm 4\%$	$1.11 \pm 4\%$	$0.69 \pm 4\%$
53.34	$0.284 \pm 7\%$	$1.13 \pm 4\%$	$1.25 \pm 4\%$	$0.305 \pm 4\%$
69.70	$0.114 \pm 10\%$	$1.10 \pm 4\%$	$1.20 \pm 5\%$	$0.215 \pm 5\%$
Radial Measurements(a)				
2.29	$1.00 \pm 4\%$	$1.00 \pm 3\%$	$1.20 \pm 3\%$	$1.00 \pm 3\%$
8.97	$0.82 \pm 5\%$	$1.02 \pm 4\%$	$1.09 \pm 4\%$	$0.95 \pm 4\%$
14.50	$0.84 \pm 5\%$	$1.08 \pm 4\%$	$1.09 \pm 4\%$	$0.97 \pm 4\%$
20.04	$0.86 \pm 5\%$	$1.05 \pm 4\%$	$1.19 \pm 4\%$	$0.98 \pm 4\%$
28.14	$1.04 \pm 5\%$	$0.99 \pm 4\%$	$1.70 \pm 4\%$	$1.09 \pm 4\%$
33.66	$1.65 \pm 5\%$	$0.98 \pm 4\%$	$1.75 \pm 4\%$	$1.79 \pm 4\%$
39.19	$1.53 \pm 5\%$	$0.97 \pm 4\%$	$1.57 \pm 4\%$	$1.78 \pm 4\%$
46.00	$1.20 \pm 5\%$	$0.97 \pm 4\%$	$1.23 \pm 4\%$	$1.25 \pm 4\%$

(a) The U^{235} fission-rate traverse curves were used for the relative $\bar{\sigma}_f (U^{235})$ values since the shape of these curves should be more accurate than the U^{235} activation results.

V. PROMPT NEUTRON LIFETIME

The prompt neutron lifetime was determined by measuring the Rossi- α of the assembly. The equipment and basic experimental technique have been described by Brunson *et al.*⁽⁵⁾

Two He³ counters, loaded in three different geometries, provided four sets of data. This was done in an attempt to separate the α 's of the central zone and the driver. Several runs were made with both counters in the central zone and both counters in the driver. The nature of the data and the results suggested that the transport of neutrons between the central zone and the driver was great enough to preclude separating the α 's of the two regions. Additional data were taken with one counter in the central zone and the second in the driver. The initiating and terminating roles of the counters were exchanged to determine whether there was a significant difference in the experiment.

The results not only were identical within the experimental uncertainties, but were also identical to the results with both counters in the driver. From these results, it was concluded that Assembly 43 behaved essentially as a single-region reactor with an average Rossi- α that would describe the entire reactor. The average Rossi- α at delayed critical was measured as $(2.42 \pm 0.1) \times 10^4 \text{ sec}^{-1}$.

The lifetime and β_{eff} for Assembly 43 were calculated at Argonne, Illinois,* using a spherical model in two cases. In the first case, the radius of the spherical central zone was set equal to the radius of the cylindrical central zone; in the second case, the two volumes were equated. The results were, respectively, $\beta_{\text{eff}} = 0.00709$, $\ell = 1.72 \times 10^{-7}$; and $\beta_{\text{eff}} = 0.00703$, $\ell = 1.88 \times 10^{-7}$.

The measured Rossi- α , with an average β_{eff} of 0.00705, gives an average prompt neutron lifetime of $(2.9 \pm 0.2) \times 10^{-7} \text{ sec}$.

VI. REACTIVITY MEASUREMENTS

A. Central Reactivity Worths

1. Measurements

The central reactivity measurements in ZPR-III normally fall into two groups: one of small-volume ($0.5 \times 2 \times 2 \text{ in.}$), fissionable samples, and the other of larger-volume ($2 \times 2 \times 2 \text{ in.}$), nonfissile samples.

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*Calculations performed by G. J. Fischer, Reactor Physics Division, Argonne, Illinois.

The fissionable samples were loaded across the front $\frac{1}{4}$ in. of the central drawer in each half of the core. To establish a reference, a run was made with two $\frac{1}{8} \times 2 \times 2$ -in. plates of aluminum loaded in each central drawer, one of 63% density and the other of 45% density. The enriched uranium sample in each central drawer consisted of one $\frac{1}{8} \times 2 \times 2$ -in. plate sandwiched between two $\frac{1}{16} \times 2 \times 2$ -in. plates of 100%-density aluminum. The Pu^{239} and U^{233} samples were contained in $\frac{1}{4} \times 2 \times 2$ -in. aluminum cans. The total mass of the aluminum in the sandwich and in the cans was nearly equal to that of the aluminum used in the reference. The measured values are given in Table X.

Table X

REACTIVITY COEFFICIENTS OF FISSIONABLE MATERIALS
AT CENTER OF ASSEMBLY 43

Material	Mass (kg)	Worth (Ih $\pm$ 0.5)	Reactivity Coefficients (Ih/kg)
93.1% Enriched U	0.2684	15.2	56.6 ± 1.8
99.8% Depleted U	2.439	-9.7	-4.0 ± 0.2
U^{233} Mixture ^(a)	0.221	22.4	101.1 ± 2.3
Pu^{239} Mixture ^(b)	0.186	14.7	79.3 ± 2.7
Isotopic Values			
	Ih/kg	Ih/gm-mole	$R = \frac{[(\nu - 1)\sigma_f - \sigma_c]}{\text{Ih/gm-mole}}$
U^{233}	104 ± 2.4	24.2 ± 0.6	0.164 ± 0.004
U^{235}	60.5 ± 1.9	14.2 ± 0.4	0.151 ± 0.004
U^{238}	-4.1 ± 0.2	-0.976 ± 0.048	0.170 ± 0.008
Pu^{239}	82.6 ± 2.8	19.7 ± 0.7	0.172 ± 0.006

(a) The mixture consisted of 97.40% U^{233} and 2.60% U^{238} .

(b) The mixture consisted of 94.51% Pu^{239} , 5.11% Pu^{240} , and 0.38% Pu^{241} .

The isotopic value was calculated by assuming that Pu^{241} had the same worth as Pu^{239} , and Pu^{240} had one-fifth the worth of Pu^{239} .

Special drawers were loaded into the central matrix tube in each half for the large sample measurements. These drawers allow $2 \times 2 \times 1$ -in.-thick samples to be inserted and removed in the first inch of the drawer without removing the core drawer. The reference run was made with the sample volume void, and all of the values reported are relative to void.

A large-volume sample of depleted uranium was included in this series of measurements due to the uncertainty in the worth of a small sample. The boron carbide samples contain 9.9 w/o chemical impurities, which are regarded as making no measureable contribution to the reactivity worth. The boron is enriched to 90.7 a/o in B^{10} , but the results are based on the total amount of boron carbide in the sample. The physicum samples, designated Ph-I and Ph-II, are mixtures of metals, oxides, and salts of stable, naturally occurring elements selected to simulate the nuclear effects of fission products according to criteria listed in a paper by Long et al.⁽⁶⁾ The formulas of these two mixtures are (Ph-I) $O_{1.519}$ and (Ph-II) $O_{1.146}$, with molecular weights of 142.2 and 136.1, respectively.

Sample size effects were determined for enriched B_4C and tantalum since the self-shielding of these materials is large enough to result in significant measurements. The results of all the nonfissile sample measurements are listed in Table XI.

Table XI

REACTIVITY COEFFICIENTS OF NONFISSILE SAMPLES
AT CENTER OF ASSEMBLY 43

Material	Mass (kg)	Inhours/kg	lh/gm-mole	σ_{eff} (mb/atom)
$B_4C^{*(a)}$	0.0294	-683 $\pm$ 17	-35.8 $\pm$ 0.9	-6160 $\pm$ 160
$B_4C^{*(b)}$	0.0593	-594 $\pm$ 12	-31.1 $\pm$ 0.6	-5350 $\pm$ 100
Type 304 SS	1 016	-2.6 $\pm$ 0.5	-	-
Na ^(d)	2.20	-1.5 $\pm$ 0.3	-0.034 $\pm$ 0.007	-5.8 $\pm$ 1.2
W	1.867	-5.6 $\pm$ 0.3	-1.030 $\pm$ 0.055	-177 $\pm$ 10
Ta ^(a)	0.262	-19.8 $\pm$ 1 9	-3.58 $\pm$ 0.34	-616 $\pm$ 58
Ta	1.004	-15.9 $\pm$ 0.5	-2.88 $\pm$ 0.09	-495 $\pm$ 15
Ph-I oxide*	0.232	-18.5 $\pm$ 2.2	-2.63 $\pm$ 0 31	-452 $\pm$ 53
Ph-II oxide ^(c)	0.210	-12.9 $\pm$ 2.4	-1 76 $\pm$ 0.33	-303 $\pm$ 57
Bi	1.276	-0.9 $\pm$ 0 4	-0.19 $\pm$ 0.08	-33 $\pm$ 14
Y	0.5821	-2.9 $\pm$ 0.9	-0.26 $\pm$ 0.08	-45 $\pm$ 14
Pb	1.475	-0 8 $\pm$ 0.3	-0.17 $\pm$ 0.06	-29 $\pm$ 10
Be	0.2407	+2.4 $\pm$ 2.1	+0.022 $\pm$ 0.019	+3.8 $\pm$ 3.3
C	0.200	-2.7 $\pm$ 2.5	-0.032 $\pm$ 0.030	-5.5 $\pm$ 5.2
Al	0 3506	-2.8 $\pm$ 1.4	-0.076 $\pm$ 0.038	-13.1 $\pm$ 6.5
Al_2O_3	0.355	-1.7 $\pm$ 1.4	-0.17 $\pm$ 0.14	-29 $\pm$ 24
O(by difference)	-	-0.4 $\pm$ 3.2	-0.013 $\pm$ 0.102	-2.2 $\pm$ 18

*These samples are described in the text.

Note. All sample sizes are 2 x 2 x 2 in except the following:

(a) 0.25 x 2 x 2 in. sample size.

(b) 0.50 x 2 x 2 in. sample size.

(c) 1.75 x 2 x 2 in. sample size.

(d) Described in sodium void coefficients, Section VI D.

The reactivity worths were determined from the difference in the critical position of the control rod for the sample and reference runs. In ZPR-III, the control rods (fuel-removal type) must be withdrawn and the halves of the reactor must be separated to insert the sample. Reproducibility in the control rod position is better than ± 0.005 in. (less than ± 0.05 lh), while the uncertainty in the closure of the halves is ± 0.5 lh. The uncertainties in the reactivity coefficients were calculated from the latter value.

2. Analysis of Results

Although Assembly 43 was a two-zone system, a simple analysis of reactivity worths can be made by using one-energy-group perturbation theory. In the first-order approximation, the reactivity effect of a fissile sample can be described by the equation

$$\frac{\Delta k}{k} = \frac{\int_{\text{sample}} [(\nu - 1)\sigma_f - \sigma_c] N \phi^2(r) dV}{\int_{\text{core}} \nu \sigma_f N \phi^2(r) dV}.$$

The denominator is a constant for a specific core, and, if the sample volume is in the center of the core where the spatial flux distribution can be normalized to one, the equation can be simplified to:

$$\frac{\Delta k}{k} = \frac{1}{A} \overline{[(\nu - 1)\sigma_f - \sigma_c]} N_s,$$

where

N_s = Total number of atoms in the sample;

$$A = \int_{\text{core}} \nu \sigma_f N \phi^2(r) dV;$$

$$\overline{[(\nu - 1)\sigma_f - \sigma_c]} = \sum_{j=1}^n [(\nu_j - 1)\sigma_{fj} - \sigma_{cj}] \phi_j.$$

For nonfissile samples, where ν and σ_f are 0, the equation reduces to

$$\frac{\Delta k}{k} = \frac{1}{A} (\sigma_{\text{eff}}) N_s,$$

where σ_{eff} is an effective cross section that includes the effects of energy downscattering. The central reactivity worth can be expressed in this form, since there is no contribution from the transport cross section in a flux with zero gradient.

The average value of $[(\nu-1)\sigma_f - \sigma_c]$ was calculated for each of the fissile samples from the fundamental mode results listed in Table V. The effective cross sections of the nonfissile samples were calculated by multiplying each reactivity coefficient by $[(\nu-1)\sigma_f - \sigma_c]/lh/\text{g-mole (Pu}^{239}\text{)}$. These results are also listed in Table XI.

B. Axial Column Substitutions

Reactivity worths were measured at three different effective radii in the central zone for core-length columns of various materials. A $\frac{1}{4}$ -in.-wide by 2-in.-high substitution region was loaded into the drawer by replacing one column of sodium-filled cans with empty cans and rearranging the remaining materials. The loading master is shown in Fig. 26. The stainless steel and graphite samples were $\frac{1}{4}$ -in. wide. Except for the enriched uranium substitutions, the remaining samples consisted of one $\frac{1}{8}$ -in. column of stainless steel and one $\frac{1}{8}$ -in. column of the substituted material. The worth of enriched uranium was determined by replacing the $\frac{1}{16}$ -in. column of enriched uranium and the $\frac{1}{16}$ -in. column of stainless steel in the normal central zone drawer with a $\frac{1}{8}$ -in. column of stainless steel. The substitution patterns for the measurements are shown in Fig. 27. The results are listed in Table XII.

Fig. 26. Loading Master for Axial Column Substitutions in Central Zone

Fig. 27

Substitution Patterns for Axial Column Measurements in Central Zone

Table XII

 REACTIVITY COEFFICIENTS FOR AXIAL COLUMN SUBSTITUTIONS
 IN CENTRAL ZONE OF ASSEMBLY 43

Sample	$R_{eff} = 2.04 \text{ cm}$		$R_{eff} = 12.4 \text{ cm}$		$R_{eff} = 17.53 \text{ cm}$	
	$\Delta \text{ Mass (kg)}$	Ih/kg	$\Delta \text{ Mass (kg)}$	Ih/kg	$\Delta \text{ Mass (kg)}$	Ih/kg
Enriched U	1.252	$+35.4 \pm 0.4$	0.626	$+37.5 \pm 0.8$	0.626	$+38.9 \pm 0.8$
Depleted U	12.67	-2.44 ± 0.05	10.14	-2.39 ± 0.05	10.14	-2.53 ± 0.05
C	2.20	$+1.9 \pm 0.3$	3.52	$+1.95 \pm 0.14$	3.52	$+0.74 \pm 0.14$
SS	9.27	-0.76 ± 0.05	14.44	-0.64 ± 0.03	14.44	-0.82 ± 0.03
Zr	4.305	-0.72 ± 0.14	6.888	-0.48 ± 0.08	6.888	-0.62 ± 0.08
Mo	6.872	-4.2 ± 0.1	5.498	-4.45 ± 0.11	5.498	-4.34 ± 0.11
Nb	5.687	-6.5 ± 0.1	4.550	-6.48 ± 0.14	4.550	-6.58 ± 0.14
Ia	5.24	-10.9 ± 0.1	2.096	-12.6 ± 0.2	2.096	-12.2 ± 0.3
Na(a)	6.014	$+1.09 \pm 0.08$	-	-	12.18	-0.49 ± 0.04

(a) These results were taken from Table XV.

Column substitutions were also performed in the driver for graphite and stainless steel. The loading master is shown in Fig. 28. The stainless sample was $\frac{1}{4}$ in. wide, while the graphite substitution consisted of one $\frac{1}{8}$ -in. column of graphite and one $\frac{1}{8}$ -in. column of stainless steel. The worth of enriched uranium was determined by reducing the 2-in.-high fuel column to 1 in. in height at each location. The total mass change in each measurement was 0.626 kg of 93.1% enriched uranium. The results of the measurements in the driver are listed in Table XIII.

Fig. 28

Loading Master for
Axial Column Substitutions in Driver

Table XIII

 REACTIVITY COEFFICIENTS FOR AXIAL COLUMN SUBSTITUTIONS
 IN DRIVER OF ASSEMBLY 43

Radius (cm)	Enriched U (Ib kg)	Stainless Steel (304)		C	
		Δ Mass (kg)	Ib/kg	Δ Mass (kg)	Ib/kg
24.74		Outer radius of core			
30.48		Outer radius of buffer			
31.75	+35.0 $\pm$ 0.8	10.83	-0.42 $\pm$ 0.05	1.32	+12.5 $\pm$ 0.5
33.02					
34.29	+31.5 $\pm$ 0.8				
37.08	+30.0 $\pm$ 0.8				
38.35		10.83	+1.34 $\pm$ 0.05	1.32	+16.4 $\pm$ 0.5
39.62	+27.0 $\pm$ 0.8				
42.42	+23.2 $\pm$ 0.8				
44.35					
		Outer radius of driver			

C. Boron Control Rod

The worth of an axial loading of enriched boron carbide was measured to provide a basis for poison rod calculations for dilute cores. The core and axial blanket drawers in the central matrix location of the central zone were loaded completely with sodium as a reference. Then, the core and axial blanket drawers in the central matrix location of Half No. 2 were changed to the loading shown in Fig. 29. Next, the core and first 4 in. of the axial blanket in Half No. 1 were changed. The results are summarized in Table XIV.

Fig. 29

Front View of Loading Master for
Boron Control and Experiment

Table XIV

BORON CONTROL ROD MEASUREMENTS IN ASSEMBLY 43

Location	Change in Mass (kg)			Measured Δk (Ih) ^(a)
	Enriched B ₄ C	SS (304)	Sodium	
Half No. 2	+0.1290	+2.856	-0.4009	-31.2 $\pm$ 0.5
Half No. 1	+0.0845	+1.875	-0.2648	-30.6 $\pm$ 0.5
Total	+0.2135	+4.731	-0.6657	-61.8 $\pm$ 0.7

(a) An uncertainty of ± 0.5 Ih is assigned to all reactivity measurements due to separation of the halves.

D. Sodium Void Coefficients

Worths of sodium were measured for several geometries at two radial positions. The central value was determined only for 100% sodium removal, since the magnitude of the possible inaccuracies precluded detecting differences in worth with changing void fractions. Each of the five 6-in.-long sodium cans in the front of the central zone drawer was replaced with one 4-in.- and one 2-in.-long can for the reference. Then, the 4-in. sodium-filled cans were replaced with identical empty cans on both sides of the reactor midplane for the central measurements. The

drawer loading master is shown in Fig. 30. The experiment was performed for three samples of different sizes in an attempt to determine possible size effects due to the two-zone nature of the core. These geometries are shown in Fig. 31.

CORE										AXIAL BLANKET										
EMPTY CAN	SODIUM																			
EMPTY CAN	SODIUM																			
		DEPLETED URANIUM																		
		GRAPHITE																		
EMPTY CAN	SODIUM	1/16 in. ENRICHED URANIUM																		
		DEPLETED URANIUM																		
EMPTY CAN	SODIUM	1/16 in. STAINLESS STEEL																		
		GRAPHITE																		
		DEPLETED URANIUM																		
EMPTY CAN	SODIUM																			

Fig. 30

Loading Master for Central Sodium Void Measurements

Fig. 31
Substitution Patterns for
Sodium Void Measurements

The axial sodium void measurements were similar to the central experiment, except that the empty cans were loaded for the full length of the core and axial blanket in both halves. Substitutions were performed for two sodium-void fractions. The loading master for the 60% void measurement is shown in Fig. 32. The 100% void fraction measurements were similar, except that all of the sodium-filled cans were replaced with empty cans.

CORE										AXIAL BLANKET										
EMPTY CANS																				
SODIUM																				
		DEPLETED URANIUM																		
		GRAPHITE																		
EMPTY CANS		1/16 in. ENRICHED URANIUM																		
		DEPLETED URANIUM																		
SODIUM		1/16 in. STAINLESS STEEL																		
		GRAPHITE																		
		DEPLETED URANIUM																		
EMPTY CANS																				

Fig. 32

Loading Master for Full-length
Axial Sodium Void Measurements
with 60% Void Fraction

Sodium worths were also determined at an effective radius of 17.53 cm from the axial centerline. The substitution pattern was the same as that shown in Fig. 27. Instead of giving the radial variations in a large reactor, the data at this radius are a reflection on the two-zone nature of

the assembly. The results of all the measurements are summarized in Table XV. Note that although loss of sodium at the midplane has a positive reactivity effect, loss of sodium for the full length of the core and axial blanket results in a negative reactivity effect.

Table XV

SODIUM VOID COEFFICIENTS IN ASSEMBLY 43

	Sodium Mass Removed (kg)	Reactivity Change ($1\text{h} \pm 0.5$) ^a	Worth of Sodium ($1\text{h}/\text{kg}$)
<u>A. Central Measurements</u>			
<u>100% of Sodium Removed</u>			
2 Central Drawers	0.244	+0.3	-1.3 $\pm$ 2
10 Central Drawers	1.22	+1.6	-1.3 $\pm$ 0.4
18 Central Drawers	2.20	+3.3	-1.5 $\pm$ 0.3
<u>B. Axial Measurements</u>			
<u>60% of Sodium Removed</u>			
2 Central Drawers	1.203	-1.1	+0.9 $\pm$ 0.4
10 Central Drawers	6.014	-6.46	+1.09 $\pm$ 0.08
18 Central Drawers	10.825	-11.43	+1.06 $\pm$ 0.05
<u>100% of Sodium Removed</u>			
2 Central Drawers	2.005	-2.15	+1.1 $\pm$ 0.3
10 Central Drawers	10.02	-10.6	+1.06 $\pm$ 0.05
<u>C. Measurements at</u>			
<u>$R_{\text{eff}} = 17.53 \text{ cm}$</u>			
<u>100% of Sodium Removed</u>			
Central Value	1.955	+4.27	-2.2 $\pm$ 0.3
Axial Value	12.183	-6.0	+0.49 $\pm$ 0.04

^aA variation of $\pm 0.5 \text{ 1h}$ is assigned to all reactivity measurements due to uncertainty in the closure of the halves.

VII. DOPPLER EXPERIMENTS

Most of the information contained in this section has been presented previously in a paper by Fischer et al.⁽⁷⁾ Many sections of the paper are duplicated here with the permission of the authors. The reader is referred to the paper for a discussion in greater detail than given here, especially regarding details of the calculations and interpretations of the results.

The Doppler element and equipment were designed and constructed at Argonne, Illinois. The basic experimental procedure was also devised by Fischer et al. The information is presented here in the interest of a complete report on all of the experiments performed in Assemblies 43 and 43A.

A. The Doppler Element and Equipment

The design of the Doppler element that was used for these experiments is shown in Fig. 33. The uranium contained within the inner heater tube was in the form of UO_2 pellets of 70% theoretical density, sintered with a light bond of sodium meta-silicate. Each of the eleven pellets of an element was 3.7 cm in diameter and 2.5 cm high. Thermocouples located 2 cm from each end and at the center continuously measured the temperature along the center line of the cylinder defined by the UO_2 pellets during the experiments. The use of an external heater had the desirable consequence that the equilibrium temperature was constant across the diameter of the element. A vacuum between the heated element and the outer shell, plus a highly polished inner surface of the outer shell, made it possible to maintain an average temperature of 800°K throughout the UO_2 with less than a 100-Watt heat loss. A stream of air across the outer shell minimized heat exchange to the reactor matrix. However, there were temperature gradients along the length of the UO_2 pellets caused by heat loss through the end fittings on the heater. These temperature differences, though small compared to those between the heated and unheated conditions, were used to determine the average temperature of the heated sample during the experiments.

An important feature of the element design was that the heater cylinder which surrounded the UO_2 pellets was free to expand axially over the pellets because of the bellows at one end. Since the outer stainless steel shell of the Doppler element remained almost at room temperature, this shell and the internal porcelain supports acted to preserve the cold dimension of the UO_2 pellets. The intentional sintering of the pellets to 70% density provided a means of elastic intermotion of the U^{238}O_2 granules, with only a relatively small thrust on the porcelain supports. Heat was

Fig. 33. Design of Doppler Element

transferred from the heater tube to the UO_2 by argon gas which filled the inner chamber at low pressure when at room temperature. Repeated bench-test heat cycling, accompanied by X-ray pictures of the elements when both hot and cold, demonstrated the stability of the axial dimension. X-rays taken before and after the ZPR-III experiments gave no evidence of physical change during the experiments.

Each element was fastened to the end of a drive mechanism, which contained the thermocouple and heater leads as well as a vacuum line. The

Fig. 34. Location of Doppler Elements during an Experiment

drive mechanisms and elements were located in the matrix channels (from which the core and blanket drawers had been removed) above and below the reactor center line. The locations of the Doppler elements during an experiment are shown schematically in Figs. 34 and 37.

The location of the element when in the "out" position was chosen to be so far out that the reactivity importance in that position was negligible, particularly with respect to whether it was hot or cold. The element position was indicated at the reactor console both by selsyn

dial indicators, which could read a fraction of a millimeter, and by signal lights operating from microswitches. Tests of reproducibility of location of the "in" position demonstrated that this position could be repeated to within 0.025 mm with no difficulty. A similar precision was assumed for the "out" position location.

B. Experiments and Results

1. Procedure

A fine control rod was produced by reducing the fuel loading of a safety rod from three columns to one column of enriched uranium. All subsequent measurements were made with this rod, which had an uncertainty in reproducibility of less than ± 0.002 Δk .

The procedure of the experiment was repeatedly to exchange the position of the two elements while maintaining the power level of the reactor essentially constant at about 10 Watts. The reactivity worth of each exchange at the core center was determined from the difference in the critical position of the fine control rod. Nothing that could cause a

reactivity change was moved except the two samples and the fine control rod. The roles of hot element in the hot-vs-cold test were exchanged, and the cold-vs-cold measurements were repeated to improve confidence in the measurements. Figure 35 indicates the nature of the data for a typical run. The ordinates of the two exchange positions have been moved, relative to each other, a number of scale divisions to demonstrate the fit of the data points to the two curves. The curvature, which is identical within the accuracy of the data for the two exchange positions, represents the slow drift of the reactor during the course of the measurement.

Fig. 35

Typical Experimental
Data for a Hot-vs-cold
Element Exchange

2. Experiments

Before the first experiment began, the central zone drawers immediately surrounding the matrix tubes through which the Doppler elements were driven were modified to minimize resonance-flux depression at the Doppler element due to the U^{238} in the central zone. All the sodium was moved to the side of the drawers nearest to the Doppler element. The remaining materials were maintained in their original order as they were reloaded in the far side of the drawer. This geometry provided approximately one mean free path for collision of neutrons in the energy range pertinent to Doppler reactivity changes. Two of the three columns of depleted uranium in the central drawer (directly between the Doppler element positions) were replaced with sodium. The loading masters for the modified drawer loadings are shown in Fig. 36. Figure 37 is a schematic representation of the core loading for the minimized resonance-depression measurement.

The first step in the experiment consisted of measuring the change in reactivity when Doppler element No. 1, with its $U^{238}O_2$ heated to an average temperature of $800^\circ K$, replaced Doppler element No. 2, with its $U^{238}O_2$ cold ($300^\circ K$), at the axial center of Assembly 43 as represented in Fig. 34. The second step consisted of repeating the above measurement with both Doppler elements cold ($300^\circ K$). The net reactivity difference of

these two measurements represents the reactivity change at the center of the assembly due to heating the $U^{238}O_2$ of element No. 1 from 300°K to an average temperature of 800°K

Fig. 36. Front View of Modified Loading Masters for Central Zone

Fig. 37. Schematic Representation of Minimized Resonance-depression Effect Experiment

The first two lines of Table XVI give these results and show that a reactivity loss of 0.402 inhour occurred as a result of this temperature change. When the role of hot element was reversed and element No. 2 was heated to an average temperature of 800°K, a net reactivity loss of 0.398 inhour occurred, corrected for a 1% mass difference in the Doppler elements. The agreement of the two results to better than 1% is probably fortuitous and should not be taken too seriously, although Table XVII

demonstrates the same kind of agreement for the U^{238} Doppler experiment with the second zoned loading. Until further experiments justify such optimism, a precision of approximately 5% would seem more reasonable for the present results. A third cold-vs-cold result of -0.138 inhour represents a considerably less reliable measurement and has been arbitrarily deleted from the present analysis. The average of the remaining reactivity losses was $-0.90 \times 10^{-5} \Delta k/k$ for the 1.96 kg of U^{238} of the Doppler element

Table XVI

RESULTS OF DOPPLER EFFECT MEASUREMENTS
IN ASSEMBLY 43

<u>Configuration</u>		<u>Measured</u> <u>Reactivity</u> <u>Change</u> <u>(In hours)</u>
1. <u>Minimized Secondary Flux Depression</u>		
1 Hot vs 2 Cold		-0.251
1 Cold vs 2 Cold		-0.151
	Difference	-0.402
2 Hot vs 1 Cold		-0.542
2 Cold vs 1 Cold		-0.151
	Difference	-0.391
1 Cold vs 2 Cold		-0.155
-0.402 - (-0.155) = -0.247		-0.247
2. <u>Measurements of Possible Doppler Effects</u>		
2 Hot vs 1 Cold (net)		-0.382
<u>Extrapolated</u>		
1 Hot vs 2 Cold (net)		-0.077

Table XVII

RESULTS OF DOPPLER EFFECT MEASUREMENTS
IN ASSEMBLY 43A

<u>Configuration</u>		<u>Measured</u> <u>Reactivity</u> <u>Change</u> <u>(In hours)</u>
1. <u>High-temperature Measurements</u>		
1 Hot vs 2 Cold ($T = 800^\circ K$)		-0.592
1 Cold vs 2 Cold		-0.090
	Difference	-0.682
2 Hot vs 1 Cold		-0.765
2 Cold vs 1 Cold		-0.088
	Difference	-0.677
2. <u>Reduced-temperature Measurements</u>		
2 Hot vs 1 Cold ($T = 552^\circ K$)		-0.457
2 Cold vs 1 Cold		-0.084
	Difference	-0.373
3. <u>Measurements of Possible on-Doppler Effects</u>		
Axial Expansion of U^{238}		-0.0086 In. min
Element Core Center		-0.003 In. min
Element Out Position		-0.005 In. min
Dummy Element at Core Center		-0.002 In. min

A second experiment measured the importance of the resonance-depression effects in the central zone. In this experiment, the drawer loadings were reversed from those shown in Fig. 36 to place the bunched fuel materials next to the Doppler elements and the sodium on the far side of the drawer. This configuration is shown schematically in Fig. 38. The results, indicating approximately 5% difference from the previous measurements, are summarized in Table XVI.

Fig. 38. Schematic Representation of Maximized Resonance-depression Effect Experiment

A third experiment, described as the Boron Test, added to the confidence that the experimental results were largely of true Doppler origin. The idea of this test was to use boron to surround the Doppler element which was to be heated during the course of an experiment. The boron should greatly perturb the flux and importance of neutrons in the energy region of a few kilovolts, which is the major source of the Doppler effect, while leaving largely unperturbed the remainder of the spectrum. Thus the non-Doppler reactivity effects due to heating the Doppler element would be largely unperturbed, while the Doppler reactivity change should be greatly reduced.

A $\frac{1}{2}$ -in.-thick layer of natural B_4C was loaded around sample No. 1 as shown in the loading masters, Fig. 39, and the core loading, Fig. 40. A residual Doppler effect of -0.066 lh was calculated for this experiment, whereas the measured reactivity change was -0.077 lh, as shown in Table XVI. While the difference between the two values could represent non-Doppler effects, it is only about half as large as the estimated uncertainty in the measurement.

Fig. 39. Front View of Loading Masters for Boron Test Experiment

Fig. 40. Schematic Representation of Boron Test Experiment

C. Measurements in a Softened Spectrum

After the Doppler effect measurements were completed in Assembly 43, the loading of the central zone was modified, as described in Section II-D (p. 12), to provide a softer spectrum for additional Doppler effect experiments. These experiments were performed with the minimized resonance-depression effects loading as shown schematically in Fig. 37. The drawer loading masters are shown in Fig. 41.

Measurements were made at two different temperatures. The high temperature corresponds to the previous experiments, while the reduced-temperature experiment was performed to provide an additional check for calculations. The results are summarized in Table XVII.

Fig. 41. Front View of Loading Masters for Measurements in Assembly 43A

Additional experiments were performed to evaluate the size and importance of possible non-Doppler reactivity effects which could contribute to the measured results. One type of experiment evaluated the significance of uncertainties or errors in location of the Doppler element during the reactivity measurements. The last three listings of Table XVII indicate that errors of 1 mm for these three types of positional error lead to reactivity changes which are of the order of 1% of the total reactivity changes measured during the experiments. The "dummy" element was complete in every respect except that the U^{238}O_2 was removed. Since measurements have shown that positional locations were reproducible within a fraction of a millimeter, this source of experimental uncertainty must be small.

Although the design and construction of the Doppler element used for these experiments indicate that considerable efforts were expended to eliminate as far as possible any axial expansion of the U^{238}O_2 of the elements, it remained important to evaluate the worth of this expansion if it could occur. This was done by replacing one of the Doppler elements with a "clean geometry" experiment. Standard ZPR-III plates of depleted uranium were mounted in a fixed position in one of the halves of the reactor, to the same depth as that of the Doppler element when it was located at the reactor midplane. An equal number of plates was mounted in a mirror position in the other half within a thin aluminum tube, which passed through the entire axis of the central zone. The thin aluminum tube was then connected to the drive mechanism built for the Doppler elements.

With this type of very simple geometry, it was then possible to say that a carefully measured worth of the displacement of the aluminum tube from its proper position measured only the reactivity effect of this "expansion" of the U^{238} . This experiment, on Assembly 43A where the total Doppler reactivity change measured was -0.680 inhour, showed that a 1-mm displacement represented a reactivity change of only 0.0086 inhour.

If an average temperature coefficient of expansion of 10×10^{-6} is assumed, the temperature rise of the UO_2 elements would have caused a motion of the center of mass of the UO_2 somewhat more than 1 mm. Thus, if the design of the Doppler element failed so that the UO_2 was perfectly free to expand, and if a crude equivalence was assumed between the two reactivity worths per mm of each type of expansion, only a little more than a 1% reactivity difference would have occurred per mm of expansion.

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