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CRITICAL STUDIES OF URANIUM-STEEL AND
URANIUM-STEEL-SODIUM FAST REACTOR CORES
(ZPR-III Assemblies 32 and 33)

by

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ABSTRACT

The report describes critical studies of two fast reactor cores: one contains uranium and steel; the other contains uranium, steel, and sodium. Experimental results are given for fission ratio, central and edge reactivity coefficients, fuel bunching, average prompt neutron lifetime, and distributed worth measurements.

I. INTRODUCTION

Assemblies 32 and 33 of ZPR-III were constructed to study some of the overall reactivity effects of the contained core materials. The work was prompted by the inability to predict critical masses of previous dilute assemblies (ZPR-III Assemblies 29, 30, and 31).⁽¹⁻³⁾

Assembly 32 was constructed to study the reactivity effects of a simple combination of stainless steel and enriched uranium. Simple assemblies such as this were considered ideal for evaluating more accurately the reactivity properties of the contained materials and thus permitting to be made a more detailed and accurate analysis of possible errors in the cross-section sets for these materials.

Assembly 33 was a modification of Assembly 32. Part of the stainless steel in each drawer was replaced by sodium-filled, stainless steel cans. It was anticipated that this core could be used in conjunction with measured values in Assembly 32 to reveal some information on the adequacy of sodium and stainless steel cross-section values in the multi-group calculations.

The work to be described was performed during February and March, 1961, in the Zero Power Reactor III (ZPR-III) located at Argonne National Laboratory's Idaho Facility at the National Reactor Testing Station. The reactor, shown in Figure 1, is basically a platform on which two tables or carriages are mounted. Half of the assembly is built up on

each carriage by means of a matrix structure consisting of square (2.18 x 2.18 in.) cross-section stainless steel tubing, stacked in a square array (approximately 67 x 67 in.).

Fig. 1. The ZPR-III Critical Facility

One carriage is a stationary unit mounted at one end of the bed. The other carriage is movable and slides on ways machined for accurate alignment as the halves are brought together.

Each half of the assembly contains five safety-control rods of the fuel-removal type, consisting of stainless steel fuel drawers fastened to drive mechanisms. Four of the rods in each half function as safety rods, and the fifth is used as a control rod.

The basic building blocks for the core region are $\frac{1}{8}$ -in.-thick plates, 2 in. in height to match the drawer height. The plates vary in length from 1 to 5 in. The materials used to make up the assembly cores are placed in perforated stainless steel drawers and then into matrix tubes. A view of a typical core drawer is given in Figure 2. A complete description of the reactor and its operation has been compiled by Cerutti *et al.*⁽⁴⁾

Fig. 2. View of Typical Core Drawer

II. DESCRIPTION OF ASSEMBLY 32

Assembly 32 contained only uranium enriched to 93% and stainless steel. The core was a 61.1-cm x 52.4-cm-diameter cylinder containing 227 kg of U^{235} . It was completely blanketed by a 45-cm-thick depleted uranium blanket.

A preliminary calculation (DSN, S4, Cross Section Set 135) was used to specify the size of the assembly. It was found that an assembly with 9.3 v/o U^{235} and the remainder stainless steel would utilize 60% of the available steel. It was expected that the calculation would underestimate the critical mass; therefore, considerably more stainless steel may have been needed. This calculation was the basis of the assembly built, since it would allow a 40% calculation error before assembly size would be limited by lack of stainless steel.

Each core drawer contained two columns of enriched uranium and 14 columns of stainless steel. With this arrangement a one-drawer unit cell could be utilized. The top view of a typical core drawer is given in Figure 3. The volume-percent compositions of both core and blanket

Fig. 3. Typical Core Drawer of Assembly 32

along with the assumed densities used to calculate the volume fractions are given in Table I. A face view of the just-critical core is given in Figure 4.

Table I

CORE COMPOSITIONS OF ASSEMBLY 32

<u>Material</u>	<u>Volume % Composition</u>		<u>Assumed Density of Material, gm/cc</u>
	<u>Core</u>	<u>Blanket</u>	
U^{235}	9.26	0.19	18.75
U^{238}	0.66	83.30	19.00
Stainless Steel (Type 304)	81.0	7.31	7.85

Fig. 4. Face View of Just-critical Core (Assembly 32)

III. DESCRIPTION OF ASSEMBLY 33

Assembly 33 was a modification of the Assembly 32. Four $\frac{1}{8}$ -in. columns of stainless steel in each core drawer of Assembly 32 were replaced by two $\frac{1}{4}$ -in.-thick, sodium-filled stainless steel cans. The new

core contained two 12-in. columns of enriched uranium, ten columns of stainless steel, and two $\frac{1}{4}$ -in.-thick columns of sodium-filled stainless steel cans.

As in Assembly 32, a one-drawer unit cell was utilized. The top view of a typical core drawer is given in Figure 5.

Fig. 5. Typical Core Drawer of Assembly 33

The critical mass of this assembly was found to be 238 kg of U^{235} as compared with 227 kg for Assembly 32. The U^{235} critical mass and volume fraction and core length (61.1 cm) were used to calculate the equivalent critical diameter of 53.5 cm. A face view of one half of Assembly 33 is given in Figure 6.

Fig. 6. Face View of Just-critical Core (Assembly 33)

The volume-percent composition of Assembly 33, along with the assumed densities used to calculate the volume-percent compositions, is given in Table II.

Table II
CORE COMPOSITIONS OF ASSEMBLY 33

<u>Material</u>	<u>Volume-% Composition</u>		<u>Assumed Density of Material, gm/cc</u>
	<u>Core</u>	<u>Blanket</u>	
U ²³⁵	9.26	0.19	18.75
U ²³⁸	0.66	83.30	19.00
Stainless Steel (Type 304)	63.56	7.31	7.85
Sodium	18.2	-	0.84

Multigroup calculations indicated that Assembly 33 would be 8% larger than Assembly 32, whereas it was only 5% larger, indicating that the sodium cross-section values may be underreactive or steel could be overreactive.

IV. APPROACH TO CRITICAL

On the basis of the expected reactor size, an initial loading was made containing approximately half the critical mass. This subcritical core was completely blanketed both radially and axially out to a distance expected for the critical core and blanket.

The approach to critical consisted of stepwise radial replacements of the depleted uranium-loaded drawers with drawers containing enriched uranium. After each increment of fuel was loaded, the halves were brought together and the degree of subcriticality estimated. The latter was estimated by recording the inverse of the neutron count rate from two proportional counters situated at the axial core-blanket interface. The counters were positioned to minimize nonlinearities in the critical-approach curve, i.e., the plot of inverse count rate versus mass of U²³⁵. A plot of reciprocal count rates for both assemblies are given in Figures 7 and 8.

After achieving criticality, the control rods were calibrated by inserting one rod a measured amount and determining accurately the reactor period associated with this rod insertion. By adjusting two control rods alternately in opposite directions and by assuming that the control rods have no interaction, a calibration of both rods over their entire length is possible. The calibration curves for these control rods are given in Figures 9 and 10.

Fig. 7. Critical Approach (Assembly 32): Reciprocal Count Rate Versus Mass

Fig. 8. Critical Approach (Assembly 33): Reciprocal Count Rate Versus Mass

Fig. 9. Control Rod Calibration (Assembly 32, Rod #10)

Fig. 10. Control Rod Calibration (Assembly 33, Rod #10)

The reactivity worth of core material substituted for blanket material at the edge of the core was measured by replacing blanket material along the boundary with core material and measuring the change in critical-rod position. The control rod worth and core edge fuel worth were then used to calculate the just-critical mass of the assemblies. With the normal fuel configuration, the edge core-for-blanket worth was 45.8 lh per kg U^{235} in Assembly 32 and 46.4 in Assembly 33.

V. FISSION RATIOS

Measurements of fission rates of various fissile materials at the core centers were made with fission chambers of parallel plate-type construction.⁽⁵⁾ The coating of fissionable material in these counters was known to within $\pm 1\%$. Precise construction techniques resulted in all counters having nearly constant efficiency. These characteristics made possible accurate measurements of effective cross-section ratios.

Two opposing central drawers were modified in each assembly to allow for positioning of the counter at the front of the drawer. A U^{235} counter (No. 5), used as a standard and flux monitor, was placed in the front of one of the central drawers and maintained there throughout the experiments. The other counters were placed one at a time at the front of the opposing drawer. With the reactor running at a predetermined power, the count rates of both the standard U^{235} counter and the other fission counter were taken, and the data were normalized for all counters to the count rate data from the U^{235} standard counter.

The experimental data were corrected for isotopic content of the counters and are given in Table III. Theoretical fission ratios calculated from the DSN spectrum for both assemblies using cross-section Set 135 are also listed.

Table III

FISSION RATIOS

$\sigma_f(\text{isotope})/\sigma_f(U^{235})$	Assembly 32		Assembly 33	
	Experimental	Calculated	Experimental	Calculated
U^{233}	1.51	1.57	1.51	1.57
U^{234}	0.367	0.415	0.370	0.431
U^{236}	0.116	0.129	0.118	0.131
U^{238}	0.045	0.053	0.048	0.059
Pu^{239}	1.203	1.240	1.208	1.25
Pu^{240}	0.382	0.411	0.400	0.429

Fission ratios determined in this manner are consistent with previously reported ZPR-III experimental results. Recent studies by Davey⁽⁶⁾ have shown, however, that the threshold detector ratios are consistently in error on the low side by perhaps 5-8%, principally because of scattering in the stainless steel counter bodies.

VI. AVERAGE PROMPT-NEUTRON LIFETIME

Average neutron lifetimes for both assemblies have been measured by the Rossi- α method, which makes use of the experimentally determined exponential decay constant (the Rossi- α) of prompt neutron chains.⁽⁷⁾ The exponential decay constant is arrived at through observations of nonrandom variations of the neutron population while the reactor is in a delayed critical or subcritical condition. This is achieved experimentally by the use of a neutron counter or counters coupled to a time analyzer. The time analyzer is triggered by an incoming pulse from the neutron detector and is shut off by a succeeding pulse. The time interval between the initiating pulse and the stopping pulse determines in which of nine time channels the following pulse will be registered. A series of measurements are made with this equipment at or near delayed critical, with care being taken that the prompt fission chains are not excessively overlapping. The number of counts in each delayed channel are plotted against the delayed time observed after the expected number of random counts entering the delayed channels have been subtracted. The slope of the resulting exponential decay curve (the Rossi- α) is then used to calculate the average neutron lifetime in the assembly from the relationship

$$\tau_0 = \frac{1-K_p}{\alpha} = \frac{\beta}{\alpha} \quad .$$

Beta here is a calculated value, defined as the effective delayed-neutron fraction, and τ_0 is the average prompt-neutron lifetime in the assembly. The alphas, estimated betas, and resulting average prompt-neutron lifetimes are given in Table IV.

Table IV

ROSSI- α MEASUREMENTS

Assembly	Rossi- α ($\text{sec}^{-1} \times 10^{-4}$)	Estimated Effective β	Average Neutron Lifetime ($\text{sec} \times 10^8$)
32	5.37 ± 0.1	0.0068	12.6 ± 0.5
33	5.46 ± 0.1	0.0068	12.4 ± 0.5

VII. CENTRAL AND EDGE REACTIVITY MEASUREMENTS

Reactivity measurements were made in both Assemblies 32 and 33 for common reactor materials at the center of each core and for a few selected materials toward the edge of each core. The latter were taken to review and compare the transport properties of the materials with calculation and reactor experiments.

The reactivity changes in these experiments were measured by a difference in calibrated control rod position for the changed core over that of the reference core for each sample.

All fissile material samples were approximately $2 \times 2 \times \frac{1}{4}$ in. in size. The reference loadings for the small fissile samples contained one perpendicular column of 45% aluminum and one of 63% aluminum in the first quarter inch of two opposing central or edge drawers. Fissile materials were substituted either clad in or sandwiched between 100% aluminum such that the net aluminum change was negligible. A typical drawer used for fissile material reactivity worth measurements is shown in Figure 11.

Fig. 11. Drawer Arrangement for Fissile Reactivity Coefficient Measurements

To measure the central reactivity worth of nonfissile materials and depleted uranium, an 8-cu-in. void was created alternately at the core center and at $r = 16.6$ cm by recessing the core material one inch at the front of two drawers. The change in reactivity was observed for $2 \times 2 \times 1$ -in. samples inserted in these spaces. A top view of a typical drawer used for nonfissile material reactivity coefficient measurements is given in Figure 12. The reactivity worths of the materials examined are given in Table V.

Fig. 12. Drawer Arrangement for Nonfissile Reactivity Coefficient Measurements

Table V
CENTRAL AND EDGE REACTIVITY COEFFICIENTS
(in inours/kg)

Material	Weight (kg)	Assembly 32		Assembly 33	
		Center	Edge	Center	Edge
U ²³³	0.221	403 $\pm$ 2.5		367 $\pm$ 2.5	
U ²³⁵	0.268	212 $\pm$ 2		197 $\pm$ 2.0	
Pu ²³⁹	0.186	374 $\pm$ 2.5(a)		349 $\pm$ 2.5(a)	
C	0.196	165 $\pm$ 2.5	142 $\pm$ 2.5	143 $\pm$ 2.5	118 $\pm$ 2.5
Al	0.350	29.9 $\pm$ 1.5	38.5 $\pm$ 1.5	23.5 $\pm$ 1.5	32.0 $\pm$ 1.5
Fe	1.03	0.4 $\pm$ 0.5	6.8 $\pm$ 0.5	-0.2 $\pm$ 0.5	5.1 $\pm$ 0.5
Ni	1.15	3.6 $\pm$ 0.5	11.0 $\pm$ 0.5	1.4 $\pm$ 0.5	9.0 $\pm$ 0.5
Cr	0.439	8.2 $\pm$ 1.0(c)	16.3 $\pm$ 1.0(c)	4.4 $\pm$ 1.0(c)	11.5 $\pm$ 1.0(c)
Stainless Steel	1.02	1.8 $\pm$ 0.5	8.2 $\pm$ 0.5	1.1 $\pm$ 0.5	7.6 $\pm$ 0.5
Na	0.091	60 $\pm$ 5(c)	76.9 $\pm$ 5(c)	41 $\pm$ 5(c)	60.4 $\pm$ 5(c)
U ²³⁸	2.46	-2.8 $\pm$ 0.2	-2.4 $\pm$ 0.2	-0.8 $\pm$ 0.2	2.4 $\pm$ 0.2
Th ²³²	1.48	-15.7 $\pm$ 0.3	-4.3 $\pm$ 0.3	-14.0 $\pm$ 0.3	
Cu	1.16	2.6 $\pm$ 0.5			
Zr	0.846	7.7 $\pm$ 5.0			
Mo	1.28	-6.3 $\pm$ 0.4		-6.0 $\pm$ 0.4	
Y	0.582	10.0 $\pm$ 1.0			
Al ₂ O ₃	0.355	6.4 $\pm$ 1.5		46.4 $\pm$ 1.5	
B ₄ ¹⁰ C	0.032	-2160 $\pm$ 15(b)		-1960 $\pm$ 15(b)	
Pb	1.47			1.49 $\pm$ 0.3	5.4 $\pm$ 0.3
Ta	1.00			-30.5 $\pm$ 0.5(c)	
Nb	0.489			-11.6 $\pm$ 1.0(c)	
S	0.199			-1.8 $\pm$ 2.5(c)	
Ph I(d)	0.232	4.9 $\pm$ 2.0(c)		-0.1 $\pm$ 2.0(c)	
Ph II(d)	0.210	7.1 $\pm$ 2.5(c)		0.0 $\pm$ 2.5(c)	
V	0.527	24.8 $\pm$ 1.0(c)			

(a) Value is for plutonium metal containing 95% Pu²³⁹.

(b) Value is for total B₄C containing 69% boron enriched to 90.7% in B¹⁰.

(c) Corrected for stainless steel worth of material containers.

(d) These samples were canned in stainless steel. The sample mass is that of the named material only. The sample worth has been corrected for the effect of the stainless steel, and the reactivity coefficient is for the named material only. The Ph I and Ph II are mixtures of fission product metals and oxides with average compositions of Ph₁O_{1.519} and Ph₁O_{1.14}, and average molecular weights of 142.2 and 136.1, respectively. The samples are completely described by Long *et al.*(8)

A. Sources of Error in Reactivity Measurements

1. Consistency of the Closure of the Halves

It was shown that in repeated measurements of the position of delayed criticality, on opening and closing the halves without changing either the composition or the temperature of the assembly, that the critical position was reproducible to ± 0.35 Ih ($\sim 10^{-5} \Delta k/k$).

2. Temperature Changes

A uniform temperature change over the assembly and machine produced a reactivity change of -1.5 Ih/ $^{\circ}\text{C}$. This was relatively unimportant, as the temperature changed only slowly. Temperature-induced reactivity changes were avoided by repeated reference runs.

3. The Errors Associated with Control Rod Movement and Backlash

These were small compared with 1, but would have set a lower limit of ± 0.04 Ih ($10^{-6} \Delta k/k$) to possible measurements.

It can be seen that for practical purposes the limitation of the measurements was item 1. This gives the lower limit of the experimental reactivities.

For all the measurements the reactor was operated at a steady power level of 0.3 w. It was found possible to make one measurement per hour. Great care was taken in establishing the critical position of the control rod. All measurements are the difference between control rod positions.

B. A Comparison of Measured and Calculated Central Reactivity Coefficients

In order to compare the central reactivity measurements of Assembly 32 with published cross sections, the experimental data were normalized to equivalent effective cross sections calculated for the reactor central flux from the available microscopic data.

A multigroup code for the IBM 1620 compiled by A. Hess was used to calculate the effective central cross sections from the microscopic data in the YOM Set 135 and the fluxes and adjoints from the Assembly 32 (DSN)* critical-size calculation.

In the code used, a perturbation function Q_{ij} is defined such that for special cases of group j (base group) and i (inscattering),

$$Q_{ij} = -\sigma_{fj} + \sigma_{fj} \nu_j \chi_j - \sigma_{n\gamma j} - \sigma_{erj} - \sum_{k=1}^6 \sigma_{j,j+k}, \quad \text{for } i = j ;$$

$$Q_{ij} = \sigma_{fj} \nu_j \chi_i, \quad \text{for } i < j ;$$

*Neutron transport code; see B. Carlson et al, LAMS-2346 and LAMS-2346, Appendix I.

$$Q_{ij} = \sigma_{fj} \nu_j \chi_i + \sigma_{j,j+1} + \sigma_{erj}, \quad \text{for } i = j + 1 \quad ;$$

$$Q_{ij} = \sigma_{fj} \nu_j \chi_i + \sigma_{j,i}, \quad \text{for } i > j + 1 \quad .$$

Matrix multiplication of the ϕ_i^* line matrix, the 16×16 Q_{ij} matrix, and the ϕ_j column matrix gives the general perturbation term

$$\sum_i \sum_j \phi_i^* Q_{ij} \phi_j \quad .$$

The perturbation term is normalized by the average center neutron importance term:

$$\sum_k \phi_k^* \phi_k \quad .$$

A comparison of the calculated and normalized experimental reactivity coefficients is given in Table VI.

Table VI

CENTRAL CALCULATED AND EXPERIMENTAL EFFECTIVE
REACTIVITY CROSS SECTIONS (Assembly 32)

Material	$\bar{\sigma}_{\text{eff}}$ (calculated)	$\bar{\sigma}_{\text{eff}}$ Experimental (normalized to Pu ²³⁹)
	$\sum_i \sum_j \phi_i^* Q_{ij} \phi_j$	
	$\sum_i \phi_i^* \phi_i$	
U ²³³	2974.0	2937.0
U ²³⁵	1681.0	1526.0
U ²³⁸	-11.98	-16.7
Pu ²³⁹	2912.0	2912.0
B ¹⁰	-994.6	-1048.0
C	57.9	63.5
O	42.6	48.7
Na	35.7	43.5
Al	24.8	25.8
Fe	6.72	0.0

Material	$\bar{\sigma}_{\text{eff}}$ (calculated)	$\bar{\sigma}_{\text{eff}}$ Experimental (normalized to Pu ²³⁹)
	$\sum_i \sum_j \phi_i^* Q_{ij} \phi_j$	
	$\sum_i \phi_i^* \phi_i$	
Cr	1.15	13.7
Ni	7.91	6.0
Zr	8.58	1.98
Mo	-19.6	-2.07
Set #145 (YOM corrected by Yiftah)*		
Al	23.09	25.8
Fe	6.31	0.0
Cr	5.2	13.7
Ni	6.06	6.0

Central Energy Flux and Adjoints Used to Calculate $\bar{\sigma}_{eff}$

YOM Group	ϕ	ϕ^*	YOM Group	ϕ	ϕ^*
1	0.019	0.0505	9	0.052	0.0684
2	0.043	0.0508	10	0.039	0.0691
3	0.084	0.0497	11	0.010	0.0738
4	0.123	0.0497	12	0.015	0.0730
5	0.199	0.0479	13	0.003	0.0749
6	0.162	0.0523	14	0.001	0.0747
7	0.150	0.0551	15	0.001	0.0732
8	0.099	0.0627	16	0.001	0.0742

*YOM refers to cross section sets devised by S. Yiftah et al., Fast Reactor Cross Sections, Pergamon Press (1960).

VIII. RADIAL FISSION RATE TRAVERSES

Relative fission rates of various fissile isotopes were measured along radii extending through the core centers. In Assembly 32 the relative fission rates of U^{235} , U^{234} , U^{238} , and Pu^{239} were measured. In Assembly 33, only U^{235} and U^{238} were used.

The measurements are made by traversing miniature cylindrical fission counters through a guide tube radially from the top of the core down through the core center. The guide tube, which stands in a vertical direction, is recessed into the 16th row of the movable half (Half No. 2) by inserting the tube in holes drilled through the tops and bottoms of the matrix tubes in this row. The counters are thus displaced axially $\frac{1}{4}$ in. from the core center. The $\frac{1}{4}$ -in. axial displacement normally will not alter the fission rate results in fast reactors of this size.

The $\frac{1}{2}$ -in.-diameter cylindrical counters have a coating of fissionable material on the inside of the cylinder wall. The collecting electrode is a wire supported on the cylindrical axis. The counters are operated in the ion chamber region. Pulses from the counters are amplified and fed to a scaler on the reactor console. Counters are moved in the guide tube by means of a motor-driven rod. The relative counter position is monitored by a servo system whose indicating device is located on the reactor console.

The number of counts recorded ranged from 3000 for the U^{238} counters to 17,000 for the plutonium counters at the core centers. The counter results were standardized for power fluctuations by referring them to the count rate of a standard U^{235} counter positioned near the edge of the core.

The count rate data were all normalized to one at the center of the core and are given in Figures 13-18. The error limits used are the rms values of the standard deviations of the counts from the sample counter and the U^{235} standard counter.

Fig. 13. Radial Fission Rate Distribution (U^{235}) Assembly 32

Fig. 14. Radial Fission Rate Distribution (Pu^{239}) Assembly 32

Fig. 15. Radial Fission Rate Distribution (U^{234}) Assembly 32

Fig. 16. Radial Fission Rate Distribution (U^{238}) Assembly 32

Fig. 17. Radial Fission Rate Distribution (U^{235}) Assembly 33

Fig 18
Radial Fission Rate Distribution
(U^{238}) Assembly 33

IX. BUNCHING EXPERIMENTS IN ASSEMBLY 32

Bunching experiments were performed in a wedge section (see Figure 19) of Half No. 1 of Assembly 32. The referenced core contained all $\frac{1}{8}$ -in. (0.3175-cm) columns of enriched uranium.

Fig 19
Wedge Section Used for
Bunching Experiments
(Assembly 32)

The first experiment consisted of unbunching 24 full and 2 half-high columns, $\frac{1}{8}$ in. (0.3175 cm) thick, to 48 full and 4 half-high columns, $\frac{1}{16}$ in. (0.1587 cm) thick, in 13 drawers of the wedge section. The partially unbunched core lost 20.3 lh of reactivity. The mass of uranium involved in this substitution was 20.375 kg. Since the full reference core contained 227.5 kg, the total expected change in reactivity for unbunching the entire core would be -227 lh.

The second experiment consisted of bunching 24 full and 2 half-high columns, $\frac{1}{8}$ in. (0.3175 cm) thick, to 12 full and 1 half-high column, $\frac{1}{4}$ in. (0.635 cm) thick, in 13 drawers of the same section of the core. The partially bunched core gained 59.4 lh of reactivity. The mass of uranium involved in this substitution was 20.375 kg. The total expected change in reactivity for bunching the entire 227.5 kg core would be +663 lh.

Typical reference, unbunched, and bunched drawers are shown in Figure 20. The experiment was planned to minimize any net radial shifting of fuel.

Fig. 20. Typical Bunching Experiment Drawers

The bunching data were used to plot a curve (Figure 21) of relative reactivity versus fuel thickness. Extrapolation to zero fuel thickness gave a net loss of reactivity of 400 lh. By use of the measured worth of core material at the edge of the core, a homogeneity correction of 3.7% $\Delta M/M$ was calculated.

Fig. 21. Relative Reactivity Versus Fuel Thickness (Assembly 32)

worth was 46.6 lh per kg of U^{235} removed, and the core-for-depleted worth was 47.9 lh per kg of U^{235} removed. These numbers are not greatly different from those measured in the normal core configuration.

An anomaly had been noticed in Assemblies 22 and 23 concerning a particular fuel rearrangement.⁽⁹⁾ When the two columns of fuel in each drawer were moved to the sides of the drawer, a gain in reactivity was observed for Assembly 23, but in the case of Assembly 22 a loss was recorded. The apparent contradiction has not been explained. To obtain more information on this phenomenon, a similar rearrangement was made in the upper half of Assembly 32. The resulting reactivity change, when extrapolated to the full core, represented a gain of 276 lh. This is a higher figure than that of Assembly 23, in line with the higher worth of bunching in Assembly 32 (663 lh compared to 467).

Two further experiments evaluated the worth of enriched uranium for depleted and core material for depleted at the edge of the core with the fuel loaded in the side of the drawers. The enriched-for-depleted worth was 46.6 lh per kg of U^{235} removed, and the core-for-depleted worth was 47.9 lh per kg of U^{235} removed. These numbers are not greatly different from those measured in the normal core configuration.

X. DISTRIBUTED REACTIVITY WORTHS IN ASSEMBLY 32

An empirical approach to the calculation of the critical-mass property of reactors, whose composition differs only slightly from the reactor being studied, is the use of measured reactivity changes associated with uniform compositional changes of the core.

The distributed worths of stainless steel, sodium, and aluminum were determined by substitutions of these materials over one octant of the core (shown in Figure 22). The substitutions consisted of replacing the steel in six columns of each of 18 drawers alternately with empty stainless steel cans, sodium-filled stainless steel cans, 100% aluminum, and 45% aluminum. A typical drawer is shown in Figure 23. The reactivity

change was observed as a difference in control rod position at delayed critical. Measurements made with these materials allow calculation of a reference core that has six columns in each of 18 drawers as void. The results of these measurements are given in Table VII.

Fig. 22. Segment of Core Utilized in Distributed Worth Measurements

Fig. 23. Typical Core Drawer for Distributed Worth Measurements

Table VII

DISTRIBUTED REACTIVITY WORTHS
(relative to void)

<u>Material</u>	<u>Worth Averaged over Core</u>
Stainless Steel (Type 304)*	+0.40 lh/g-mole
Na	+0.98 lh/g-mole
Al	+0.72 lh/g-mole

*It is assumed that stainless steel has an average gram molecular weight of 57.

XI. SODIUM REACTIVITY WORTH VERSUS RADIUS

Considerable interest has been aroused in the elastic scattering resonant structure of light elements such as sodium.⁽¹⁰⁾ A few preliminary measurements were made of the influence of the mass of sodium present on the reactivity worth of sodium and its variation as a function of position. A major influence in such measurements is the resonant structure of the sodium scattering cross section. To obtain a measurable reactivity change, four symmetric positions at a given radius were utilized. The positions utilized are shown in Figures 24 and 25. Each position could

Fig. 24

Face View Showing Position
of Sample for Measurements
of Sodium Worth Versus
Radius for Assembly 32

Fig. 25. Face View Showing Sample Positions for Measurements of Sodium Worth Versus Radius for Assembly 33

accommodate 43.8 gm of sodium. This arrangement consisted of having space for two cans ($5.1 \times 5.1 \times 1.26$ cm) of sodium in each of 8 drawers such that, when the halves were closed, the sodium was located at four symmetrical positions. Typical substitution drawers for Assemblies 32 and 33 are shown in Figures 26 and 27. The volume at each radial point was thus 59 cm^3 . The mass of sodium at a given radius was varied by changing the ratio of full cans to empty cans from 0 through 1. The results of the measurements are given in Table VIII. The results suggest that the reactivity of sodium is influenced by the mass of sodium present but the uncertainties in the results are such that more refined accurate experiments are needed.

Fig. 26

Typical Core Drawer for Measurements of Sodium Worth Versus Radius and Mass (Assembly 32)

Fig. 27. Typical Core Drawer for Measurements of Sodium Worth Versus Radius and Mass (Assembly 33)

Table VIII

RADIAL REACTIVITY DISTRIBUTION (1h) OF SODIUM

Assembly 32					Assembly 33			
Mass (g)	0	87.6	131.4	175.2	Mass (g)	0	175.2	350.4
Radius, in.					Radius, in.			
3.08	0	6.2	8.0	11.3	2.18	0	7.87	16.0
6.53	0	7.3	9.8	12.1	6.53	0	9.27	18.9
7.85	0	6.4	8.9	10.9	10.88	0	6.9	13.1