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HIGH CONVERSION CRITICAL EXPERIMENTS

by

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January 1967

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ABSTRACT

A program of critical experiments has extended the range of measured nuclear parameters of light-water-moderated, slightly enriched, uranium oxide lattices further into the undermoderated region. The H-to- U^{238} atom ratios in the cores assembled ranged from about 5 to 0.5. The initial conversion ratio, U^{238} -to- U^{235} fission ratio, U^{238} capture cadmium ratio, thermal disadvantage factor, and cadmium ratios of various materials reported were measured in the full range of lattices; the bucklings, critical masses, and reflector savings reported were measured in the looser assemblies. The fuel used most frequently consisted of 3.04-w/o enriched UO_2 pellets of 0.935-cm diameter stacked to a height of 122 cm in either aluminum or stainless steel tubing. In addition, some experiments were performed with 4.95-w/o enriched, 0.871-cm-diam oxide fuel stacked to a height of 61 cm in stainless steel tubing.

While some cores were uniformly loaded, most stainless steel-clad cores were composed of a central zone of stainless steel-clad pins located inside an annulus of aluminum-clad pins with a uniform lattice throughout both zones. The tightest cores required an outer feeder zone of a higher H-to- U^{238} ratio for criticality. All cores were constructed in a 2-m-diam tank providing an infinite radial H_2O reflector. Information on temperature coefficient and control-element and various material worths is included. Although the experimental results are not compared with calculations in this report, these results are correlated with some results obtained at other laboratories with similar fuel at higher H-to- U^{238} ratios.

I. INTRODUCTION

A. The High Conversion Critical Experiments Program

The High Conversion Critical Experiments Program (Hi-C) was undertaken to extend the range of investigations with light-water-moderated, slightly enriched, uranium oxide cores. Although for a given initial core excess reactivity the dependence of the conversion ratio on H-to- U^{238} atom ratio and fuel enrichment in oxide cores is not very strong, a maximum

does occur for reactors that are somewhat drier than conventional designs.¹ The nominal enrichment selected for the fuel to be used in the experiment was 3%, since this was sufficient for criticality in the fairly dry loadings of greatest interest.

Reactors with high initial conversion ratios to allow high burnup are expected to become increasingly important in the utilization of nuclear energy. Thus, these investigations are of general interest, although no specific reactor design was planned. For example, the undermoderated condition is not unlike the reduced moderator density existing in a boiling-water reactor; therefore this information will aid in understanding the reactor statics of boiling systems.

Previously, very few studies of uranium-water systems had extended into the range of H-to- U^{238} atom ratios covered in this investigation, which involved H-to- U^{238} atom ratios from 5 to as low as 0.5. These H-to- U^{238} atom ratios correspond to water-to-fuel volume ratios ranging from 1.7 to 0.16. The several H-to- U^{238} atom ratios were obtained by using square and triangular grids of various pitches. In addition, some loadings were done with fuel pins removed in a uniform pattern, and the tightest loading was achieved without gridwork. At each lattice pitch, critical parameters (buckling, reflector savings, control-rod worths, etc.) and the lattice microparameters (initial conversion ratio, thermal disadvantage factor, cadmium ratios, etc.) were measured. With the exception of some cadmium ratios, and a single value of δ^{28} , the microparameters reported were measured in stainless steel-clad cores or in large zones of stainless steel-clad fuel located in driver cores of aluminum-clad fuel.

Early measurements of critical parameters were made in one of the looser lattices. This lattice produced a relatively high H-to- U^{238} atom ratio, and critical cores of both aluminum- and stainless steel-clad fuel were assembled. Initially, approximately 3200 aluminum-clad and 2100 stainless steel-clad fuel pins were obtained. Aluminum cladding tubes were provided for all the fuel, and eventually all but 400 stainless steel-clad fuel pins were dejacketed and the fuel pellets transferred to aluminum cladding. This transfer resulted in the availability of about 4900 aluminum-clad fuel pins, a quantity sufficient for some low H-to- U^{238} atom ratio critical loadings.

Experimentation started in the ZPR-7 facility in June 1961. However, from July 1961 to January 1962, BORAX-V boiling zone fuel was available and measurements were made with it at two lattice pitches. The BORAX-V boiling zone fuel was of higher enrichment and smaller diameter than the Hi-C fuel. The Hi-C program was again interrupted when from July 1962 to September 1962 the ZPR-7 facility was used for the BORAX-V superheater experiments.

In general, the Hi-C program proceeded from the looser lattices to the tighter ones. From October 1962 to July 1964, when measurements in the tightest loading were completed, most of the Hi-C data were collected. After all data had been analyzed, however, it seemed desirable to perform measurements in an even looser uniform lattice that would allow an overlap between Hi-C data and data from previous experiments with similar fuel at other laboratories. Hi-C measurements in the ZPR-7 facility began again in October 1965 and continued on this last lattice until January 1966.

B. Scope of the Report

This report is restricted to the experimental aspects of the Hi-C program. It was desired to present primarily the experimental methods and procedures, and resulting data. However, because of the great amount of foil-counting data resulting from some of the measurements, derived values are given to conserve space. In every case, an effort was made to report values that required minimal theoretical treatment. For example, the thermal disadvantage factor and U^{238} capture cadmium ratio are reported instead of the corresponding values of thermal utilization and resonance escape probability, since the latter depend primarily on the theoretical approach used to calculate them from the experimental values.

Some of the early Hi-C results have been presented at technical meetings,^{2,3} while later results have been reported in Reactor Physics Division Annual Reports.^{4,5} The behavior of the thermal disadvantage factor has been reported separately,⁶ as has an analysis of the cadmium ratios obtained in the Hi-C cores.⁷ The structure of the ZPR-7 facility has been described in the Hazards Summary Report,⁸ and the modifications prior to the Hi-C experiment are given in Addendum II⁹ to that report.

II. CRITICAL FACILITY AND FUEL DESCRIPTION

A. ZPR-7 (Hi-C Facility)

The ZPR-7 facility is a water critical facility of conventional design; the pertinent features of the system were described in detail in the Hazards Summary Report⁸ and a subsequent addendum⁹ dealing with the Hi-C experiment. In simple form, the ZPR-7 is a 2-m-diam reactor tank connected by fill and drain lines with a pump and a water storage tank. The water level in the reactor tank may be adjusted to any desired level through a variable-level overflow, and there is provision for electrical heating of the moderator in the dump tank. The control-element drives are connected to the control elements by pulleys and cables providing relatively easy access to the top of the core. Many of these features are noted in Fig. 1.

Fig. 1. ZPR-7 Core Structure and Control Mechanism

Two types of control elements were used: control blades and five-element finger control rods. The control blades consisted of 25.4-cm-wide, 0.635-cm-thick boral sheets (35-w/o boron), clad in 0.458-mm stainless steel, and were usually placed at the edge of the grid plates. Some grid plates had movable slots so that the control blades could be moved in towards the core. The finger control rods were composed of five 0.794-cm-OD stainless steel tubes packed with a B_4C (90%) and glass powder, which was fused in the tube, and were guided in tubes of the same diameter as the fuel cladding. Each of the five fingers occupied a lattice position. The guide tubes were perforated to allow a water column to follow each rod. The finger control rods were normally located either inside the core or at the core periphery.

At least two grid plates were used to support the fuel pins at each lattice pitch. The grid plates varied in size and shape to accommodate the expected cores, but were all of 0.318-cm-thick Type 6061-T6 aluminum. Sections of the top grid plate were made removable to facilitate foil activation measurements. These removable patches may be seen near the center of Fig. 2, which shows a Hi-C core loaded in the ZPR-7 facility. The variable-level overflow is seen in the bottom right corner of the photograph. The tubes attached to the control-blade guides are instrument thimbles.

112-3845

Fig. 2. Hi-C Core in ZPR-7 Critical Facility

When inserted into the grid plates, the fuel pins rested on a 15.25-cm-thick Lucite base, which functioned as a bottom reflector.

B. Fuel Pins

The Hi-C fuel was in the form of nominal 3-w/o enriched UO_2 pellets. These centerless-ground pellets were contained in either aluminum or stainless steel tubing to a column height of 122 cm. Early in the Hi-C program the fuel was removed from most of the stainless steel cladding and placed in aluminum tubes. This transfer was accompanied by about a 2% loss in the average fuel weight per tube. Measurements were also made with the BORAX-V boiling zone fuel. This fuel was in the form of nominally 5-w/o enriched UO_2 pellets loaded to a height of 61 cm in stainless steel tubes.

The major irregularity in these fuel pins was the voiding produced by chipping, for a chip that breaks from a pellet may separate the pair of pellets touching it.

1. Hi-C Fuel

Fuel pellets in the 5600 fuel pins had an average uranium content in the UO_2 of 88.166% and an average weight percent of U^{235} in the uranium of 3.0424 ± 0.018 . The exact stoichiometric composition of uranium in UO_2 (88.15%) was not expected because of the addition of binder material in making the pellets, and because the uranium may not all be present as the UO_2 compound. The uranium content in the UO_2 was determined by analysis of 122 pellet samples adjusted to the final accounting by the fabricator. The U^{235} enrichment was determined from the analysis of 81 pellet samples.

TABLE I. Isotopic Composition^a of Uranium in Hi-C Fuel

Isotope	Weight Percent
U^{234}	0.0176 ± 0.008
U^{235}	3.044 ± 0.055
U^{236}	0.0262 ± 0.0013
U^{238}	96.913 ± 0.058

^aAs determined by ANL Spectrochemical Laboratory.

The weight-percent composition of uranium isotopes in the Hi-C fuel was determined by the ANL Spectrochemical Laboratory from analysis of 10 pellet samples. The results are given in Table I. The previously given value of U^{235} enrichment is preferred since more samples were taken. The diameters of the pellets were uniform at 0.935 cm, and their density averaged 10.17 g/cm^3 . The average column length of pellets was $121.92 \pm 0.32 \text{ cm}$, the average weight of UO_2 was 853.33 g, and the average U^{235} content per fuel pin was 22.89 g. These two values were reduced to 836.26 and 22.43 g, respectively, for the fuel pins after transfer of the pellets from stainless steel to aluminum tubing.

The level of impurities in the Hi-C fuel appears satisfactory. Chemical analyses of samples from four different batches of fuel pellets gave 40.6 ± 47.4 ppm of carbon present. Ten samples from each of six batches of fuel were subjected to emission spectroscopic analysis. The resulting data in ppm, with an accuracy of a factor of two, are presented in Table II.

TABLE II. Spectroscopic Analysis of Hi-C Fuel Pellets

Element	PPM ^a	Element	PPM ^a
Ag	<1	Mg	5.8
Al	44.2	Mn	<1
As	<10	Mo	<20
B	<0.1	Na	<10
Be	<0.5	Ni	34.7
Bi	<1	P	<50
Ca	<20	Pb	<1
Co	<5	Sb	<1
Cr	16.7	Si	103.9
Cu	8.1	Sn	<5
Fe	273.6	Ti	<50
K	<50	Zn	<50
Li	<1		

^aAccuracy is a factor of two.

2. BORAX-V Fuel¹⁰

The BORAX-V fuel was not available for an extended period; no detailed chemical and spectrochemical analyses were performed. These fuel pellets were 0.872 cm in diameter and were loaded to a 61.0-cm height in stainless steel cladding tubes. The oxide had an average density of 10.2 g/cm^3 , and the uranium contained 4.95-w/o U^{235} . Average net weight of UO_2 was 374.15 g/pin, and U^{235} content averaged 16.269 g/pin.

3. Fuel Tubes

The Hi-C fuel pellets were loaded to a height of 121.9 cm in either aluminum or stainless steel cladding tubes as shown in Fig. 3. The Type 6061-T6 aluminum tubing had an outside diameter of 1.058 cm and a wall thickness of 0.048 cm. Tube weights were 49.3 g. The aluminum top plug and O rings weighed 2.3 g, and the Type 304 stainless steel bottom plug and O ring weighed 8.2 g.

112-7938

Fig. 3. Hi-C Fuel Pin

The Type 304 stainless steel tubing was 1.057-cm OD and had a wall thickness of 0.0496 cm. The tube weight was 155.3 g. The bottom and top plugs used on the aluminum tubes were identical to those used on the stainless steel tubes.

The BORAX-V fuel was contained in Type 304 stainless steel tubing of 0.955-cm OD and 0.0381-cm wall thickness. The bottom and top plugs were also of Type 304 stainless steel. Although the cladding tubes were 86.2 cm in length, the fuel section occupied only 60.96 cm as shown in Fig. 4. The large volume of stainless steel in the top of these fuel pins resulted in the BORAX-V cores having a large fraction of stainless steel in the top reflector.

Fig. 4

BORAX-V Fuel Pin

112-7939

SECTION THRU FINISHED ROD

The important details of the fuels and claddings are given in Table III.

TABLE III. Hi-C and BORAX-V Fuel and Cladding Description

	BORAX-V		Hi-C
<u>Fuel</u>			
Enrichment, w/o	4.95		3.04
Diameter, cm	0.871		0.935
Length, cm	60.96		121.92
Oxide density, g/cm ³	10.2		10.17
<u>Cladding</u>			
Type	304 SS	6061-T6 Al	304 SS
Outer diameter, cm	0.955	1.058	1.057
Wall thickness, cm	0.0381	0.048	0.0496

III. GENERAL EXPERIMENTAL METHODS

A. Core Loadings and Descriptions

In general, the critical cores were constructed with the outer fuel pins arranged to provide a boundary as nearly circular as possible. The boundary cannot be perfectly circular with each fuel pin on the periphery at the same distance from the center, but for the cores constructed in these experiments, the approximation to a cylinder was quite good.

The radius, R , of a cylindrical core containing n fuel pins and N finger control rods occupying fuel positions is computed from

$$R = r_0 \sqrt{n + N}, \quad (1)$$

where r_0 is the radius of the unit cell. The radius of the unit cell is computed from the cylindricized unit cell of the same area as the hexagonal unit cell in a triangular lattice or the square unit cell in a square lattice.

Thus, in a square lattice,

$$r_0 = p(1/\pi)^{1/2}, \quad (2)$$

and in a triangular lattice,

$$r_0 = p(\sqrt{3}/2\pi)^{1/2}, \quad (3)$$

where p is the lattice pitch.

Two BORAX-V grids and five Hi-C grids were used. To give more H-to-U²³⁸ atom ratios than obtainable by uniform loadings of the five different Hi-C grids, some loadings were done with fuel pins removed in a uniform pattern. The loading in which every ninth fuel pin was removed is referred to as the 8/9 loading; the loading in which every sixth fuel pin was removed is referred to as the 5/6 loading, etc. It was convenient to be able to compute the radii of these nonuniform cores from the number of fuel pins in the assembly rather than the total number of fuel-pin positions in the core. To be able to do this, the unit-cell radius, r_0 , must be adjusted to include the proper fraction of the unoccupied fuel-pin positions. Equations (2) and (3) then become

$$r_0 = p \left(\frac{x+1}{x} \cdot \frac{1}{\pi} \right)^{1/2} \quad (4)$$

and,

$$r_0 = p \left(\frac{x+1}{x} \cdot \frac{\sqrt{3}}{2\pi} \right)^{1/2}, \quad (5)$$

for a loading in which every $(x + 1)$ th fuel pin is removed [an $x/(x + 1)$ loading]. The H-to- U^{238} atom ratio assigned to this fictitious unit cell was computed from the water and fuel volume fractions of the unit cell of the uniformly loaded lattice by

$$\text{H-to-}U^{238} = (N_H/N_{28})[(w + 1/x)/f], \quad (6)$$

where w and f are the water and fuel volume fractions of the uniform unit cell, and N_H and N_{28} are the hydrogen and U^{238} atom densities in the water and fuel, respectively.

Table IV lists the radii of the effective unit cells and volume fractions of the various uniform and nonuniform loadings. At any given lattice pitch, the volume fractions are not the same for aluminum-clad and stainless steel-clad Hi-C fuel. This is due to the slight dimensional differences between the aluminum and stainless steel cladding tubes.

TABLE IV. Volume Fractions and Effective Unit-cell Radii of BORAX-V and Hi-C Lattices

Lattice Pitch, cm	Loading and Cladding	Volume Fractions				Unit-cell Radius, cm
		Fuel	Water	Cladding	Void	
<u>BORAX-V Fuel</u>						
1.27 square	Uniform SS	0.370	0.556	0.068	0.006	0.7165
1.27 triangular	Uniform SS	0.427	0.487	0.079	0.007	0.6668
<u>Hi-C Fuel</u>						
1.349 square	Uniform Al	0.377	0.517	0.084	0.022	0.7609
1.349 square	Uniform SS	0.377	0.518	0.086	0.019	0.7609
1.24 square	Uniform Al	0.446	0.428	0.099	0.027	0.6996
1.24 square	Uniform SS	0.446	0.429	0.102	0.023	0.6996
1.24 square	7/8 SS	-	-	-	-	0.7479
1.24 square	3/4 SS	-	-	-	-	0.8078
1.27 triangular	Uniform Al	0.491	0.371	0.109	0.029	0.6668
1.27 triangular	Uniform SS	0.491	0.372	0.112	0.025	0.6668
1.166 triangular	Uniform Al	0.583	0.253	0.129	0.035	0.6122
1.166 triangular	Uniform SS	0.583	0.255	0.133	0.029	0.6122
1.127 triangular	Uniform Al	0.624	0.201	0.138	0.037	0.5917
1.127 triangular	15/16 Al	-	-	-	-	0.6111
1.127 triangular	11/12 Al	-	-	-	-	0.6180
1.127 triangular	8/9 Al	-	-	-	-	0.6276
1.127 triangular	5/6 Al	-	-	-	-	0.6483
1.127 triangular	Uniform SS	0.624	0.202	0.142	0.032	0.5917
1.069 triangular	Uniform SS	0.693	0.114	0.158	0.035	0.5613

Table V describes the cores in which measurements were performed. Each core is numbered and will subsequently be referred to by this number. The H-to- U^{238} atom ratio is also given for each core. These ratios were

computed from the water and fuel volume ratios as given in Table IV and the H and U^{238} atom number densities in water and the oxide fuel, respectively. The number densities used were, in the Hi-C fuel, $N_{28} = 0.022020 \times 10^{24}$ atoms/cm³; in the BORAX-V fuel, $N_{28} = 0.021615 \times 10^{24}$ atoms/cm³, $N_H = 0.066761 \times 10^{24}$ atoms/cm³. The U^{238} number density in the Hi-C fuel reflects the actual density of the oxide fuel (10.17 g/cm³) and the actual rather than stoichiometric weight percent of uranium in the UO_2 (88.166 w/o rather than 88.15 w/o). The BORAX-V U^{238} number density was computed for stoichiometric oxide fuel of a 10.2-g/cm³ density. The hydrogen number density is for 20°C water at atmospheric pressure.

TABLE V. Hi-C and BORAX-V Core Descriptions

Core No.	Lattice Pitch, cm	Type and Cladding	H-to- U^{238} Atom Ratio
<u>BORAX-V Fuel</u>			
1	1.27 square	Uniform SS	4.65
2	1.27 triangular	Uniform SS	3.53
<u>Hi-C Fuel</u>			
3	1.349 square	Uniform Al	4.15
4	1.349 square	Uniform SS zone ^a	4.16
5	1.27 square	b	3.24
6	1.24 square	3/4 loading SS	5.18
7	1.24 square	7/8 loading SS	3.89
8	1.24 square	Uniform SS	2.92
9	1.24 square	Uniform SS zone ^a	2.92
10	1.24 square	Uniform Al	2.91
11	1.27 triangular	Uniform Al	2.29
12	1.27 triangular	Uniform SS zone ^a	2.29
13	1.166 triangular	Uniform Al	1.32
14	1.166 triangular	Uniform SS zone ^c	1.33
15	1.127 triangular	5/6 loading Al	1.95
16	1.127 triangular	8/9 loading Al	1.58
17	1.127 triangular	11/12 loading Al	1.42
18	1.127 triangular	15/16 loading Al ^d	1.30
19	1.127 triangular	Uniform SS zone ^c	0.98
20	1.127 triangular	Uniform Al zone ^c	0.98
21	1.069 triangular	Uniform SS zone ^c	0.50

^aThese cores contained an outer feeder zone of aluminum-clad fuel.

^bThis core refers to measurements in the peripheral zone of the BORAX-V superheater critical experiment.

^cThese cores required an outer feeder zone of higher H-to- U^{238} ratio for criticality.

^dThis core was rectangular.

Cores 4, 9, 12, 14, 19, 20, and 21 are listed as zoned cores. In these assemblies, a central region, or zone, was surrounded by a peripheral region of aluminum-clad fuel pins. If needed for criticality, an outer peripheral zone of higher H-to-U²³⁸ ratio was added. Core 21 was constructed by inserting an aluminum box of inside dimensions 19.44 x 20.31 cm and 0.64-cm thickness into the center of the 1.127-cm triangular grid. The box was then loaded with 388 stainless steel-clad Hi-C fuel pins in a hexagonal pattern. The loading in the box was snug; the average dimensions of the box and the number of pins it held indicated a 0.012-cm average gap between fuel pins. This spacing corresponds to a 1.069-cm triangular lattice pitch.

In all the zoned cores, the H-to-U²³⁸ ratio listed is for the central zone of the assembly. The slight difference in H-to-U²³⁸ ratio between aluminum- and stainless steel-clad cores is again due to the dimensional differences between the two types of cladding tubes.

Table VI lists the details of the zoned loadings. The loadings detailed there are cores that had an excess reactivity of 0.2 to 0.4%, and were, in general, those in which most of the parameter measurements were made. The outer peripheral zones were composed of alternating rings of water (unoccupied fuel-pin positions) and fuel pins. The innermost water ring would generally contain some fuel pins, and the core reactivity would be adjusted by adding and removing fuel pins from this ring.

TABLE VI. Zoned-loading Details

	Core Number						
	4	9	12	14	19	20	21
Central zone							
Radius, cm	15.2	14.0	13.3	11.9	11.5	28.0	11.0 ^a
No. of fuel pins	400	400	397	379	379	2234	388
First peripheral zone ^b							
Radial thickness, cm	6.7	10.9	14.8	27.4	19.9	None	20.4
No. of fuel pins	409	845	1364	3719	2418	None	2395
Outer peripheral zone							
Radial thickness, cm	None	None	None	4.5	8.5	8.0	8.5 ^c
No. of fuel pins				699	948	824	974
No. of water-filled positions				297	790	640	764
Loading number	418	383	283 & 343	373	342	119	391

^aEquivalent radius; the zone was nearly square.

^bThis zone held 20 finger-control-rod locations.

^cDifferent lattice pitch than central zone.

B. Foil Irradiations

The nature of the close-packed lattices under study made it necessary to perform most foil irradiations within the fuel pins rather than in the moderator volume between pins. To facilitate this, special slotted fuel pins were constructed. These pins were made by removing a 0.48-cm-wide band of cladding around half the pin diameter at the desired foil location. The fuel pellets were then loaded into these pins so that the faces of two pellets would meet at the slot and a foil could be inserted there. For most irradiations, care was taken to assure that the fuel-pellet faces adjacent to foils were smooth and chip-free.

Foils that were the same diameter as the oxide fuel pellets were usually covered with a 0.025-mm aluminum foil to prevent contamination from the fuel, and irradiated directly between fuel pellets. Many types of irradiations (activation traverses and cadmium ratio measurements) were done with foils slightly smaller in diameter than the fuel. These foils were positioned for irradiation in small "pillboxes" as shown in Fig. 5. These

"pillboxes," used only in the Hi-C cores, gave a covering of 0.051 cm of material over the foil. Two types of material were used, cadmium for cadmium-covered irradiations, and aluminum for "bare" irradiations.

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Fig. 5. Foil Cover for Irradiations in Hi-C Fuel

When these covers were used to obtain activation traverses, care was taken to assure that the presence of a cover, especially a cadmium cover, did not influence the activation of a foil in a neighboring fuel pin. In most cadmium-covered activation traverses, a separation of three fuel pins between foil pins was maintained. Even when a separation of only one fuel pin between foil pins existed, no interaction between foils and adjacent covers was noted. It was found that bare foils could be irradiated in adjacent fuel pins, although it was not common practice to do so.

The slots in the cladding of the fuel pins that held foils for irradiation were covered with polyethylene tape to prevent water leakage. This effort was not always successful. About every third or fourth pin would show signs of water leakage after irradiation. No difference was noted between "dry" and "wet" foils.

Most of the aluminum-clad slotted fuel pins had only one slot, placed at the center of the fuel. Most of the stainless steel-clad slotted pins had two slots, about 5 cm apart and symmetrically placed with respect to the central plane of the reactor fuel. In addition, foil pins containing 17 to 22 slots were used to make axial traverses.

For some of the cadmium ratio measurements in the moderator, the foils were placed in covers and taped to thin strips of steel or plastic. Each

strip was then inserted between rows of fuel pins. Usually an attempt was made to position each foil in the center of a moderator region rather than between two fuel pins, using a small plastic spacer to hold the foil and the surrounding fuel elements in exactly the intended positions.

The type of irradiation given the foils depended on the type of traverse or measurement desired. Radial traverses, in which it was not necessary to retain flux symmetry at the two positions in the double-slotted pins, were generally accomplished with a water-level critical configuration. In this configuration, all control elements were withdrawn from the core, and criticality was maintained by adjusting the height of the top water reflector. Axial traverses and radial activations, in which it was desired to preserve axial symmetry, were irradiated by a series of rising periods, each of which would be terminated by the insertion of one of the peripheral control blades. When this type of irradiation was performed, it was necessary to adjust the core loading so that the reactor would rise on a long period when it possessed a full top reflector and all control elements were withdrawn. For most of the microparameter determinations and other measurements involving only a few foils in the central volume of the assembly, irradiations were performed with a full top reflector and with one or more peripheral control blades partially inserted to maintain criticality.

C. Foil Counting and Data Reduction

Foil activity was assayed in automatic foil-counting equipment.¹¹ Gamma counting was accomplished with 5-cm-diam by 5-cm-thick NaI(Tl) scintillators; beta counting was done with a 0.08-cm Pilot-B plastic scintillator, covered by a 0.01-cm-thick aluminum foil cover. The gamma crystals were shielded from betas by an aluminum cover.

The raw counting data, punched automatically on IBM cards, were processed on the IBM-704 or CDC-3600 computer using the RE-202 foil data-reduction code. In general, each foil was counted at least three times. For each count of the foil, the data-reduction code computes the saturated activity corrected for background, counter deadtime, irradiation, decay, counting times, and weight (or intercalibration) differences.

The general counting and intercalibration methods used for the various foil materials (copper, dysprosium, gold, indium, lutecium, manganese, and uranium) were as follows:

1. Copper

Integral gamma counting was used. Intercalibration was by weight. Foils were nominal 0.05 mm thick of pure metal.

2. Dysprosium

Dysprosium was both beta- and gamma-counted. The beta-counting results are reported here since the gamma-counting rate was frequently too low to be useful. Intercalibration was by counting after irradiation on a rotating wheel in the Argonne Thermal Source Reactor (ATSR) thermal column. Foils were of nominal 4- and 5-w/o dysprosium in aluminum alloy. Foil thicknesses of 0.13-0.26 mm were used.

3. Gold

Integral gamma counting was used. Most foils were 0.03 mm thick of pure metal. The weight correction allowed for the fact that much of the episcadmium activation is strongly self-shielded, and corrected to a standard weight of foil. The weight correction for gold in the RE-202 code, instead of division by the foil weight, involves division by the expression¹²

$$w \left[1 - \frac{1}{CdR} \cdot \frac{2 F(0.44)}{2 + F(0.44)} \right], \quad (7)$$

and for cadmium-covered foils would divide by the expression

$$w \left[\frac{2 - F(0.44)}{2 + F(0.44)} \right], \quad (8)$$

where the factor F is defined as

$$F = 2 \left[\frac{w - w_0}{w + w_0} \right]. \quad (9)$$

Here w is the weight of the foil being counted and w_0 is the weight of the standard foil, taken as 11.50 mg/cm² for the gold foils. The constant 0.44 is semiempirical¹³ and is appropriate for both gold and indium foils of about 0.03 mm thickness.

4. Indium

Integral gamma counting was used. The pure metal foils were about 0.03 mm thick. Weight corrections were made identically to the gold weight corrections. The standard foil weight used was 53.9 mg/cm².

5. Lutecium

Both gamma and beta counting were used. Intercalibration was by counting after irradiation on a rotating wheel in the ATSR thermal column. Foils were of an alloy of a nominal 2½-w/o lutecium in aluminum. Foils of a 0.26-mm thickness were used.

6. Manganese

Both gamma and beta counting were used. In either case, intercalibration was by weight. The 0.06-mm-thick foils used were of an alloy of 90-w/o manganese and 10-w/o copper. Care was taken when gamma counting to be sure that there was no interference from copper.

7. Uranium

Fission-product counting was done on 0.13-mm-thick metallic foils of 0.21- and 3-w/o enrichment by integral gamma counting above 0.41 or 1.2 MeV. The higher discriminator level was used when counting soon after irradiation to eliminate pulses due to bremsstrahlung from U^{239} beta rays. Intercalibration was by weight. Uranium-aluminum alloy foils of 17.5-w/o uranium, which was enriched to 92.75-w/o U^{235} , were fission-counted also. Integral gamma counting above 0.411 MeV was used. Foil thicknesses were from 0.05 to 0.66 mm. Before any irradiation, the foils were intercalibrated either by weight or by differential gamma counting about the broad low-energy gamma peak from U^{235} decay. Counting for U^{238} capture is described in Section V-A.

D. Reactivity Measurements

The change in reactivity from one reactor condition to another was usually inferred from observing the change in critical position on a previously calibrated control element. When conditions permitted, the excess reactivity of each core configuration was obtained by measuring the rising period in the control-rod-free core.

Conversion from asymptotic rising period to reactivity was made with the usual inhour equation. Three sets of period-versus-reactivity tables were computed: one for the BORAX-V cores, one for the looser Hi-C cores, and one for the tighter Hi-C cores. The prompt-neutron lifetimes, effective delayed-neutron fractions, and decay constants for the delayed-neutron groups used for the three calculations are given in Table VII.

TABLE VII. Data for Period-to Reactivity Conversion in Hi-C and BORAX-V Cores

Decay Group	Core 1 ($\beta_i \times 10^3$)	Core 2 ($\beta_i \times 10^3$)	Cores 3-12 ($\beta_i \times 10^3$)	Cores 13-21 ($\beta_i \times 10^3$)	All Cores $\lambda_i, \text{sec}^{-1}$
i = 1	0.2845	0.2776	0.2702	0.2373	0.0127
i = 2	1.6233	1.5984	1.5730	1.4496	0.0317
i = 3	1.4752	1.4609	1.4459	1.3684	0.115
i = 4	3.2319	3.2079	3.1824	3.0433	0.311
i = 5	1.1226	1.1341	1.1452	1.1803	1.40
i = 6	2.583	2.66	0.2737	0.3032	3.87
$\sum_i \beta_i = \beta_{\text{eff}}$	7.9958	7.9449	7.890	7.582	-
Prompt-neutron Lifetime, μsec	13.47	11.16	17.24	9.48	-

IV. LATTICE MEASUREMENTS

A. Control-element and Fuel-pin Worths

One of the first things done each time a new core was assembled to critical was the calibration of control elements. This was done not only to assist in later measurements, but also to be sure that shutdown requirements could be met. An in-core finger control rod was customarily calibrated first. The e-folding time resulting from displacement of the rod by a known amount was observed for successive increments in the rod position. Between subsequent period measurements, the reactor was restored to critical by a compensating displacement of another finger control rod, first the one diametrically opposite the rod being calibrated and later an adjacent one. Only a very slight interaction between the finger control rods was noticed, and the effect was neglected in the results reported here. The finger-control-rod worths were insensitive to small changes in the core radius when located in in-core positions.

The control blades were generally calibrated by observations of the displacement of a calibrated finger control rod required to return the reactor to a critical condition after a displacement of a control blade. Since the control blades were located in the side reflector, their worth was strongly dependent on their distance from the core periphery.

When finger control rods are withdrawn, the volume vacated by the rods fills with water. To arrive at a value for the clean critical mass, it is necessary to determine the reactivity effect resulting when the rod guides are replaced by fuel pins. In most of the cores, the in-core fuel-pin worths were measured at the finger-control-rod locations.

To assist in estimating the clean critical core size, a knowledge of the reactivity worth of the peripheral fuel pins is also required. Such data were obtained from a number of small core loading changes. The results were computed from the changes in control-element positions required to maintain criticality and the known control-element worths.

The results of the control-element and fuel-pin worth measurements are given in Table VIII. No values are given for zoned cores.

B. Critical Mass

The control-element-free, fully reflected critical loading was determined for each core assembled except the zoned cores. With one exception (Core 18, which was rectangular), the values obtained were for cylindrical loadings. The results of these measurements appear in Table IX.

TABLE VIII. Control-element and Fuel-pin Worths in Hi-C and BORAX-V Cores

Core Number	Finger Control Rods		Control Blades		Peripheral Pin Worth		In-core Pin Worth	
	Total Worth, % $\Delta k/k$	Radial Location, cm	Total Worth, % $\Delta k/k$	Location	Worth, % $(\Delta k/k)/pin$	Core Radius, cm	Worth, % $(\Delta k/k)/pin$	Radial Location, cm
1	1.49	12.6	0.22	5 cm from core periphery	0.027	18.6	-0.0782 -0.0678 -0.0465 -0.0022	0 5.1 10.2 17.8
2	Not used in this core		Varied over wide limits		0.0141	20.7	-0.0811 -0.0578 -0.0102	0 10.2 19.0
3	0.67	At core periphery 18.8	0.525	Core periphery	0.0250	18.8	-	-
8	0.72	21.2	0.355	31.9 cm	0.0071	29.7	-0.020	21.2
10	1.70 0.54	12.5 22.0	0.71	24.8 cm from center	0.0168	21.2	-0.0489	12.5
11	0.589	21.6	0.035	33.0 cm from center	0.0112	25.1	-0.0198	21.6
13	0.54	21.0	0.133	45.5 cm from center	0.00161	42.6	-0.0292	21.0
15	0.675	20.0	Not used in this core		0.00858	27.7	-0.0406	20.0
17	0.38	28.0	0.13	41.7 cm from center	0.00275	37.4	-0.017	28.0
18 ^a	0.40	20.0	0.40	At edge of core	0.0011	Edge of core	-0.0137	20.0

^aRectangular core.

TABLE IX. Clean Critical Loadings for Cylindrical Hi-C and BORAX-V Cores

Core Number	Clean Critical Number of Fuel Pins	Clean Critical U ²³⁵ Mass, kg	Clean Critical Core Radius, cm
1	647 ± 2	10.45 ± 0.03	18.23 ± 0.03
2	967 ± 3	15.63 ± 0.05	20.74 ± 0.03
3	589 ± 2	13.48 ± 0.04	18.47 ± 0.03
6	769 ± 1	17.60 ± 0.02	22.40 ± 0.02
7	1093 ± 1	25.02 ± 0.02	24.73 ± 0.02
8	1766 ± 6	40.42 ± 0.14	29.40 ± 0.09
10	950 ± 1	21.75 ± 0.02	21.56 ± 0.02
11	1417 ± 3	32.44 ± 0.07	25.10 ± 0.03
13	5091 ± 13 ^a	115.69 ± 0.29	43.68 ± 0.01
15	1862 ± 3	42.62 ± 0.07	27.96 ± 0.02
16	2788 ± 15	63.82 ± 0.34	33.15 ± 0.09
17	3680 ± 18 ^a	83.92 ± 0.40	37.49 ± 0.09
18	4937 ± 10 ^{a,b} 4618 ± 118 ^c	112.24 ± 0.23 ^b 105.04 ± 2.65 ^c	- 41.53 ± 0.53 ^c

^aContained transferred fuel pins; they have been accounted for in the critical mass.^bRectangular core.^cComputed for a cylindrical core from rectangular-core data.

In most cores, the critical number of fuel pins was obtained by evaluating the reactivity worth of peripheral and in-core (at the finger-control-rod locations) fuel pins in a core of known loading and known excess reactivity. Some cores contained no finger control rods, and the critical number of fuel pins was found from the core loading, excess reactivity, and peripheral fuel-pin worths. In a few of the cores, the critical mass was determined by several methods including extrapolation of approach-to-critical data in a core with no finger control rods.

The larger cores contained some of the aluminum-clad fuel pins in which the fuel pellets had been transferred from stainless steel-clad pins. These transferred fuel pins contained an average of 2% less fuel than the normal fuel pins. No core contained more than 1665 of these pins. The critical mass, as given in Table IX, includes an adjustment for such pins present.

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Fig. 6. Clean Critical Masses for Cylindrical Hi-C Cores

The critical-mass results for the Hi-C cores are also presented in Fig. 6 as a plot of clean critical mass versus H-to- U^{238} atom ratio. The aluminum-clad Hi-C results reveal that the nonuniformly loaded cores are more reactive than the uniformly loaded cores of the same indicated H-to- U^{238} atom ratio. This means that in these undermoderated lattices the reactivity worth of water is less when introduced uniformly than when added as columns throughout the lattice.

C. Critical Bucklings and Reflector Savings

Critical bucklings were obtained from the critical core size and cosine- and Bessel-function fits of axial and radial activation traverses.

Axial traverses were made by placing foils in aluminum or cadmium pillboxes and placing the pillboxes between fuel pellets in a special slotted fuel pin. The slots were at 5.56-cm intervals in the stainless steel-clad Hi-C pins, 5.5-cm intervals in the aluminum-clad Hi-C pins, and 5.96-cm intervals in the BORAX-V fuel. Radial traverses were obtained in double-slotted fuel pins in the stainless steel-clad Hi-C cores. These slots were 5.08 cm apart and allowed two radial traverses to be taken at each irradiation. In the aluminum-clad Hi-C cores, radial traverses were taken in singly slotted fuel pins. Foils were attached to a steel or plastic tape and positioned at the core center plane in the BORAX-V cores to obtain a radial traverse.

Foil materials used were highly enriched uranium (uranium-aluminum alloy), 0.21-w/o depleted uranium, dysprosium (dysprosium-aluminum alloy), gold, manganese (manganese-copper alloy), and lutetium (lutetium-aluminum alloy). Both bare (aluminum-covered) and cadmium-covered traverses were made.

The general irradiation configurations in the Hi-C cores were water-level criticals for radial traverses and full top reflector with irradiation on a series of rising periods for axial traverses. In the BORAX-V cores, axial and radial traverses were generally taken simultaneously. Both water-level and rod criticals were used. Only those axial traverses taken with a full top reflector are presented here.

The axial activation distributions were fitted by a least-squares iteration technique to the cosine function

$$y = A_z \cos B_z(z - C_z), \quad (10)$$

where A_z is the amplitude, B_z is the square root of the axial buckling, and C_z is the displacement of the fitted curve from the center plane of the fuel region (taken as $z = 0$). The extrapolated height, H_e , was then found from the square root of the axial buckling by

$$H_e = \frac{\pi}{B_z}. \quad (11)$$

The radial activation traverses were fitted to the zero-order Bessel function

$$y = A_r J_0[B_r(r - C_r)], \quad (12)$$

where A_r is the amplitude, B_r is the square root of the radial buckling, and C_r is the displacement of the fitted curve from the centerline of the reactor lattice. The extrapolated radius, R_e , was then found from the square root of the radial buckling by

$$R_e = 2.405/B_r. \quad (13)$$

Each data point was weighted in the fitting by the inverse of the square of its estimated error. Although the error from counting statistics was always kept much below 1%, other counting errors and uncertainties in foil position indicated that an error of about 1.5 to 2% should be given each data point in the fitting. The data points in the outer portions of the traverses were in positions of greater flux gradient than the data points in the lattice interior and could be expected to have greater errors because of uncertainties in foil position.

A gradient error assignment was used for most of the fittings from the Hi-C cores. This assignment gave an error of 1.8% for the outermost data points, 1.7% for the second outermost data points, 1.6% for the third outermost data points, and 1.5% for the interior data points. In the fitting of the BORAX-V data points, a constant error was assigned.

Only those data points that were thought to be within the asymptotic volume of the core were used for curve fitting. The terminology originated from the asymptotic approach of flux distribution in a reflected core to that of an equivalent bare core. Hence the asymptotic region is far enough removed from the reflector to yield consistent indications of flux spatial distributions, which are typically cosine in the axial direction and J_0 (Bessel) in the radial direction.

The extent of such regions was investigated in each of the uniformly loaded, nonzoned cores (1, 2, 3, 8, 10, 11, and 13). Gold and manganese foils, both cadmium-covered and bare, were irradiated to obtain radial activation traverses. A succession of J_0 fits was then made from each traverse, each fit including data points nearer the boundary. It was expected that if points too near the core-reflector interface were used, the fitted J_0 function would become increasingly in error. The radial region was then determined from this series of fits for which:

1. Both bare and cadmium-covered foil traverses gave the same value of radial reflector savings; and
2. The radial reflector savings were independent of fitting radius.

The reflector savings, λ , was used in these determinations rather than the extrapolated radius. The reflector savings is relatively independent of the core radius and can be averaged from all the determinations on a given lattice.

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Fig. 7. Radial Reflector Savings vs Fitting Radius in 1.24-cm Square Stainless Steel-clad Hi-C Lattice

Typical of the results obtained when these effects were investigated were the data from Core 8 (1.24-cm square stainless steel-clad Hi-C). These data are seen in Fig. 7, where the radial reflector savings are plotted against the radius of the outermost data points used in the J_0 fitting. As expected, when bare data points near the core-reflector interface were used in the J_0 fitting, the indicated reflector savings became increasingly larger.

The cadmium-covered data points, however, seem to give good results even when those close to the core-reflector interface were used in the fitting. In some cores, it was found that the cadmium-covered data points within one pin of the core boundary could be used in the J_0 fitting. Investigation in most cores also revealed a limitation on how few data points could be used successfully for Bessel-function fitting. When only a few points in the center of a core were used for the J_0 fit, the errors on the extrapolated radius became excessive, resulting in a practical limitation on how few data points could be used. In the uniform cores considered here, fitting was restricted to traverses for which outermost data points were within 40 to 65% of the extrapolated radius. Most of the Bessel-function fits used a diameter of data points extending to 55-60% of the extrapolated radius.

Table X presents the results of the radial activation traverse fittings. Only in the uniform cores were extensive data taken. Usually, only

TABLE X. Results of Radial Activation Traverse J_0 Fittings in Hi-C and BORAX-V Cores

Date	Run No.	Foil Type	Core Radius, cm	Extrap Radius, cm	Reflector Savings, cm	Best Center of Radial Distribution, C_r , cm	Radius of Fit, cm
<u>Core 1</u>							
7/28/61	21-63	Cd-covered Dy	18.14	25.94	7.80	+0.07	17.8
8/3/61	21-107	Bare Au	18.14	25.82	7.68	-0.47	12.7
8/3/61	21-108	Cd-covered Au	18.14	25.14	7.00	-0.13	15.2
8/29/61	30-1	Bare Dy	18.14	25.06	6.92	-1.08	12.7
<u>Core 2</u>							
10/20/61	58-3	Bare Dy	20.78	28.32	7.54	+0.08	15.2
11/9/61	71-9	Bare Au	21.03	28.49	7.46	+0.04	15.2
11/9/61	71-10	Cd-covered Au	21.03	28.61	7.58	+0.18	15.2
11/15/61	71-37	Bare Lu	21.03	28.34	7.31	+0.00	15.2
11/16/61	71-39	Cd-covered Lu	21.03	28.90	7.87	+0.02	15.2
<u>Core 3</u>							
10/7/65	399-5	Cd-covered Mn	18.57	25.89	7.32	+0.24	14.8
10/8/65	399-7	Cd-covered Mn	18.57	25.95	7.38	+0.08	13.5
10/8/65	399-9	Bare Au	18.57	26.42	7.85	-0.12	13.5
10/12/65	401-2	Cd-covered Au	18.57	25.70	7.13	+0.08	13.4
10/13/65	403-2	Bare Au	18.57	25.82	7.25	+0.34	13.5
10/13/65	403-5	Cd-covered Au	18.57	25.41	6.84	+0.22	13.5
10/14/65	403-8	Cd-covered Au	18.57	25.02	6.45	+0.12	13.5
10/15/65	403-10	Cd-covered Au	18.57	26.28	7.71	0.00	13.4
10/18/65	405-2	Cd-covered Mn	18.57	26.15	7.58	+0.06	13.4
10/19/65	407-2	Bare Au	18.57	26.56	7.99	+0.24	13.4
10/20/65	409-2	Cd-covered Mn	18.57	26.06	7.49	+0.04	13.4
10/21/65	411-2	Bare Au	18.57	26.16	7.59	+0.11	13.4
<u>Core 6</u>							
2/7/63	210-2	Cd-covered Mn	22.82	29.38	6.56	-0.29	18.6
2/7/63	210-2	Cd-covered Mn	22.82	29.39	6.57	-0.27	18.6
2/13/63	213-2	Cd-covered Mn	22.76	29.34	6.58	-0.07	21.1
2/13/63	213-2	Cd-covered Mn	22.76	29.44	6.68	-0.07	21.1
2/14/63	214-2	Cd-covered Au	22.82	29.35	6.53	-0.08	18.6
2/14/63	214-2	Cd-covered Au	22.82	29.38	6.56	-0.10	18.6

TABLE X. (Contd.)

Date	Run No.	Foil Type	Core Radius, cm	Extrap Radius, cm	Reflector Savings, cm	Best Center of Radial Distribution, C_r , cm	Radius of Fit, cm
<u>Core 7</u>							
1/23/63	204-2	Cd-covered Mn	24.91	32.17	7.26	-0.10	22.32
<u>Core 8</u>							
2/25/63	215-2	Cd-covered Mn	29.24	36.97	7.73	-0.14	24.8
2/25/63	215-2	Cd-covered Mn	29.24	37.05	7.80	-0.04	24.8
2/25/63	215-2	Cd-covered Mn	29.24	36.36	7.12	-	26.3
2/25/63	215-2	Cd-covered Mn	29.24	36.39	7.15	-	26.3
2/27/63	217-2	Cd-covered Au	29.34	36.90	7.56	-0.12	24.8
2/27/63	217-2	Cd-covered Au	29.34	37.34	8.00	-0.19	24.8
2/27/63	217-2	Cd-covered Au	29.34	36.57	7.23	-	26.3
2/27/63	217-2	Cd-covered Au	29.34	36.81	7.47	-	26.3
3/7/63	220-2	Bare Mn	29.17	36.79	7.62	+0.66	19.8
3/7/63	220-2	Bare Mn	29.17	36.81	7.64	+0.40	19.8
3/7/63	221-2	Bare Au	29.26	37.24	7.98	+0.46	19.8
3/7/63	221-2	Bare Au	29.26	37.12	7.86	+0.40	19.8
3/25/63	224-3	Bare Au	29.27	36.78	7.51	-	19.8
3/25/63	224-3	Bare Au	29.27	37.25	7.98	-	19.8
3/26/63	225-2	Cd-covered Au	29.36	37.35	7.99	-	24.8
3/26/63	225-2	Cd-covered Au	29.36	37.19	7.83	-	24.8
3/26/63	225-4	Cd-covered Mn	29.36	37.04	7.68	-	24.8
3/26/63	225-4	Cd-covered Mn	29.36	37.69	8.33	-	24.8
<u>Core 10</u>							
4/24/63	246-1	Cd-covered Mn	21.73	29.61	7.88	-0.08	17.4
4/25/63	246-2	Cd-covered Mn	21.73	29.82	8.09	-0.05	17.4
4/26/63	248-1	Cd-covered Mn	21.73	30.04	8.31	-0.10	17.4
4/29/63	250-1	Cd-covered Au	21.73	29.86	8.13	-0.14	17.4
5/1/63	253-1	Cd-covered Au	21.73	29.94	8.21	-0.11	17.4
5/1/63	253-2	Cd-covered Mn	21.73	29.98	8.25	+0.06	17.4
5/2/63	253-3	Cd-covered Au	21.73	29.76	8.03	-0.05	17.4
5/2/63	253-4	Cd-covered Au	21.73	29.86	8.13	-0.10	17.4
<u>Core 11</u>							
6/3/63	274-2	Bare Au	24.89	33.75	8.86	+0.07	17.8
6/5/63	274-5	Cd-covered Mn	24.89	33.45	8.56	+0.03	20.3
6/5/63	274-7	Bare Au	24.89	33.76	8.87	-0.06	17.8
6/6/63	274-9	Cd-covered Mn	24.89	33.67	8.78	0.00	20.3
6/6/63	274-10	Cd-covered Au	24.89	33.52	8.63	+0.03	20.3
6/7/63	274-11	Cd-covered Mn	24.89	33.11	8.22	-0.22	17.6
6/7/63	274-12	Cd-covered Au	24.89	33.59	8.70	+0.09	17.6
6/10/63	274-13	Cd-covered Mn	24.89	33.83	8.94	-0.05	17.6
6/10/63	274-15	Cd-covered Au	24.89	33.82	8.93	-0.03	17.6
6/13/63	276-1	Cd-covered Au	24.89	33.79	8.90	+0.01	22.0
6/13/63	276-2	Bare Au	24.89	33.95	9.06	-0.18	17.6
<u>Core 13</u>							
2/11/64	353-2	Cd-covered Mn	42.94	53.05	10.11	-0.06	35.0
2/11/64	353-2	Cd-covered Mn	42.94	53.94	11.00	+0.17	34.3
2/11/64	353-4	Cd-covered Au	42.94	53.98	11.04	-0.18	32.6
2/11/64	353-4	Cd-covered Au	42.94	52.40	9.46	+0.51	32.3
2/12/64	353-6	Bare Mn	42.94	52.83	9.89	-0.61	32.6
2/12/64	353-6	Cd-covered Mn	42.94	53.78	10.84	+0.11	32.3
2/12/64	353-8	Cd-covered Au	42.94	53.70	10.76	-0.30	32.6
2/12/64	353-8	Bare Au	42.94	54.25	11.31	+0.41	32.3
2/13/64	353-11	Cd-covered Mn	42.94	54.05	11.11	-0.02	32.6
2/13/64	353-11	Bare Mn	42.94	54.49	11.55	-0.25	32.3

TABLE X (Contd.)

Date	Run No.	Foil Type	Core Radius, cm	Extrap Radius, cm	Reflector Savings, cm	Best Center of Radial Distribution, C_r , cm	Radius of Fit, cm
<u>Core 13 (Contd.)</u>							
2/13/64	353-14	Bare Au	42.94	53.66	10.72	-0.09	32.6
2/13/64	353-14	Cd-covered Au	42.94	53.94	11.00	+0.15	32.3
2/14/64	354-1	Cd-covered Mn	42.33	53.77	11.44	-0.02	32.6
2/17/64	356-1	Cd-covered Au	42.29	53.22	10.93	-0.01	32.6
2/18/64	356-2	Cd-covered Au	42.29	53.24	10.95	-0.03	32.3
2/19/64	356-3	Cd-covered Mn	42.29	53.23	10.94	+0.05	32.3
<u>Core 15</u>							
10/16/63	323-3	Cd-covered Mn	27.60	36.55	8.95	+0.06	21.5
10/16/63	323-3	Cd-covered Mn	27.60	36.36	8.76	+0.05	22.5
<u>Core 17</u>							
10/3/63	307-2	Cd-covered Mn	37.31	48.06	10.75	+0.12	31.5
10/4/63	307-4	Cd-covered Mn	37.31	47.90	10.59	+0.11	31.5

a few cadmium-covered traverses were made in the nonuniform cores. Table X also gives C_r , the best center of radial distribution. The values of C_r obtained indicate reasonable radial symmetry about the core center. In some runs, foils were activated only along one radius of the core, and the results reflected about the core centerline to provide a diameter for J_0 fitting. The C_r obtained from these fits was of no meaning and is not presented in Table X. Two or more fits are given for the same run number in Cores 6, 8, and 13. Sometimes two diametrical traverses were made simultaneously, and in the stainless steel-clad cores, where the foil pins were double-slotted, four diametrical traverses could be obtained in each run.

In general, the cadmium-covered radial traverses were more satisfactory than the bare ones. Typical activation plots obtained from radial foil irradiations in Core 11 are shown in Figs. 8 and 9. Figure 8 is representative of the cadmium-covered foil data; there is very little scatter and

112-7967

Fig. 8. Typical Cadmium-covered Radial Activation Traverse in Hi-C Core

112-7953

Fig. 9. Typical Bare Radial Activation Traverse in Hi-C Core

no core-reflector interface effect. Typifying bare foil traverses, Fig. 9 shows more data scatter, and the effect of the core-reflector interface on the outermost points is pronounced.

The extent of the asymptotic region in the axial direction was also investigated and appeared to extend to about 85% of the extrapolated height. The results of the axial activation traverse fittings are given in Table XI.

TABLE XI. Results of Axial Activation Traverse Cosine Fittings in Hi-C and BORAX-V Cores

Date	Run No.	Foil Type	Core Height, cm	Extrap Height, cm	Reflector Savings, cm	Best Center of Axial Distribution, C_z , cm	Length of Fit, cm
<u>Core 1</u>							
7/28/61	21-64	Bare Dy	60.96	72.28	11.32	-0.46	47.0
<u>Core 2</u>							
10/20/61	58-3	Bare Dy	60.96	71.34	10.38	-0.31	46.7
10/23/61	59-1	Cd-covered Dy	60.96	72.06	11.10	-0.44	47.0
<u>Core 3</u>							
10/11/65	400-1	Bare Au	121.92	135.89	13.97	-0.10	93.8
10/12/65	402-2	Bare Au	121.92	135.06	13.14	-0.44	110.3
10/15/65	404-1	Bare Au	121.92	138.54	16.62	+0.50	93.8
10/18/65	406-1	Bare Au	121.92	137.41	15.49	-1.10	110.3
<u>Core 6</u>							
2/7/63	211-3	Bare Mn	121.92	135.21	13.29	-0.37	116.8
<u>Core 7</u>							
1/18/63	201-1	Bare Mn	121.92	134.30	12.38	+0.07	116.8
<u>Core 8</u>							
11/28/62	183-1	Bare Au	121.92	134.86	12.94	-0.04	116.8
11/30/62	183-2	Bare Dy	121.92	134.12	12.20	-0.17	116.8
2/26/63	216-3	Bare Au	121.92	134.89	12.97	-0.06	116.8
2/26/63	216-2	Bare Mn	121.92	135.24	13.32	-0.07	116.8
<u>Core 10</u>							
4/22/63	241-1	Bare Mn	121.92	138.21	16.29	-0.66	88.4
4/29/63	251-2	Bare Au	121.92	139.13	17.21	-0.65	88.4
4/30/63	251-3	Bare Mn	121.92	137.88	15.96	-0.50	88.4
4/30/63	251-4	Bare Au	121.92	138.55	16.63	-0.49	88.4
<u>Core 11</u>							
6/11/63	275-2	Bare Au	121.92	140.67	18.75	-0.78	46.0
6/11/63	275-3	Bare Au	121.92	140.87	18.95	-0.78	46.0
6/12/63	275-4	Bare Mn	121.92	141.52	19.60	-0.86	46.0
6/12/63	275-5	Bare Mn	121.92	141.06	19.14	-0.81	46.0
<u>Core 13</u>							
2/14/64	354-1	Bare Mn	121.92	142.69	20.77	-1.00	99.7
2/17/64	356-1	Bare Au	121.92	142.86	20.94	-0.83	99.7
2/18/64	356-2	Bare Au	121.92	143.28	21.36	-0.82	99.7
2/19/64	356-3	Bare Mn	121.92	143.52	21.60	-1.01	99.7
<u>Core 15</u>							
10/16/63	324-1	Bare Au	121.92	139.95	18.03	-0.41	40.4
<u>Core 17</u>							
10/4/63	309-1	Bare Au	121.92	144.62	22.70	-0.82	40.4

This table also lists C_z , the best center of axial distribution. The sign convention used was positive in the top half of the core. Almost without exception, the values of C_z obtained were negative, indicating that the bottom Lucite reflector was not quite equivalent to the infinite top water reflector used in all axial traverse activations. This displacement of the center of axial distribution from the center of the fuel zone is not large; in most cases it amounts to only 1/2% of the extrapolated length.

As shown in Table X, not all measurements on a given lattice were taken with the same core radius, R . These different radii reflect fuel-pin loading changes due to the presence of differing numbers of cadmium-covered foils, differing irradiation procedures, etc. The radial reflector savings, λ_r , which is the difference between the extrapolated radius obtained from the J_0 fitting and the core radius, is largely independent of the core radius and may be averaged from all the determinations on a given lattice. The critical radial buckling, B_r^2 , was calculated from the average reflector savings and the critical radius, R_c , in the control-rod-free, fully reflected core, as given in Table IX. The critical axial buckling, B_z^2 , was calculated from the average extrapolated height as obtained from the cosine fittings. The critical buckling was calculated from

$$B^2 = B_r^2 + B_z^2, \quad (14)$$

where

$$B_r^2 = \left(\frac{2.405}{R_c + \lambda_r} \right)^2, \quad (15)$$

and

$$B_z^2 = \left(\frac{\pi}{H_c + \lambda_z} \right)^2. \quad (16)$$

The critical buckling and reflector savings are given in Table XII. The error on the buckling is primarily a reflection of the uncertainty in the number of fuel pins required for criticality and the error in the reflector savings. The error indicated for the reflector savings is the standard deviation from the mean of repeated measurements. In those lattices in which the number of measurements was insufficient to compute a meaningful standard deviation, the error was estimated from that obtained in similar lattices. The critical bucklings obtained are plotted in Fig. 10. Figures 11 and 12 present the radial and axial reflector savings in the aluminum-clad Hi-C cores.

Again, as noted in the critical-mass results, the aluminum-clad Hi-C buckling results show that the nonuniformly loaded cores are more reactive than the uniformly loaded cores of the same indicated H-to- U^{238} atom ratio. The aluminum-clad Hi-C reflector savings, on the other hand, exhibit no differences between data from uniform and nonuniform loadings, although larger errors are indicated for the axial reflector savings results.

TABLE XII. Critical Bucklings and Reflector Savings for Cylindrical Loadings of Hi-C and BORAX-V Cores

Core Number	1	2	3	6	7	8	10	11	13	15	16	17	18 ^a
H-to-U ²³⁸ Atom Ratio	4.65	3.53	4.15	5.18	3.89	2.92	2.91	2.29	1.32	1.95	1.58	1.42	1.30
Critical Radius, cm	18.23 ±0.03	20.74 ±0.03	18.47 ±0.03	22.40 ±0.02	24.73 ±0.02	29.40 ±0.09	21.56 ±0.02	25.10 ±0.03	43.68 ±0.01	27.96 ±0.02	33.15 ±0.09	37.49 ±0.09	41.53 ±0.53
Radial Reflector Savings, cm	7.35 ±0.27	7.55 ±0.10	7.38 ±0.12	6.58 ±0.06	7.26 ±0.11	7.69 ±0.08	8.13 ±0.16	8.77 ±0.17	10.73 ±0.14	8.86 ±0.40	9.9 ^b ±0.5	10.67 ±0.40	10.78 ±0.50
Extrapolated Radius, cm	25.53 ±0.27	28.29 ±0.12	25.85 ±0.12	29.98 ±0.06	31.99 ±0.11	37.09 ±0.12	29.69 ±0.16	33.87 ±0.08	54.41 ±0.14	36.82 ±0.40	43.05 ±0.50	48.16 ±0.41	53.31 ±0.70
Radial Buckling, cm ⁻² x 10 ⁴	88.74 ±1.89	72.27 ±0.62	86.54 ±0.80	68.87 ±0.29	56.52 ±0.39	42.04 ±0.27	65.61 ±0.71	50.42 ±0.24	19.54 ±0.10	42.66 ±0.93	31.20 ±0.74	24.94 ±0.42	21.13 ±0.13
Axial Reflector Savings, cm	11.32 ±0.50	10.74 ±0.50	14.80 ±0.78	13.28 ±0.49	12.38 ±0.45	12.89 ±0.27	16.52 ±0.31	19.11 ±0.18	21.17 ±0.19	18.03 ±0.50	21.5 ^b ±1.0	22.70 ±0.50	21.78 ±0.50
Extrapolated Height, cm	72.28 ±0.50	71.70 ±0.50	136.72 ±0.78	135.20 ±0.49	134.30 ±0.45	134.81 ±0.27	138.44 ±0.31	141.03 ±0.18	143.09 ±0.19	139.95 ±0.50	143.42 ±1.00	144.62 ±0.50	143.70 ±0.50
Axial Buckling, cm ⁻² x 10 ⁴	18.89 ±0.26	19.20 ±0.27	5.28 ±0.06	5.40 ±0.04	5.47 ±0.04	5.43 ±0.02	5.15 ±0.02	4.96 ±0.01	4.82 ±0.01	5.04 ±0.04	4.80 ±0.07	4.72 ±0.03	4.78 ±0.03
Buckling, cm ⁻² x 10 ⁴	107.63 ±1.90	91.47 ±0.68	91.82 ±0.80	74.27 ±0.29	61.99 ±0.39	47.47 ±0.27	70.76 ±0.71	55.38 ±0.24	24.36 ±0.10	47.70 ±0.93	36.00 ±0.74	29.66 ±0.42	25.91 ±0.14

^aThese values were inferred from data obtained in a rectangular core.^bThese values were estimated from Figs. 11 and 12.

112-7952

Fig. 10. Critical Bucklings in Hi-C and BORAX-V Cores

112-7959

Fig. 11. Radial Reflector Savings in the Aluminum-clad Hi-C Cores

112-7968

Fig. 12. Axial Reflector Savings in the Aluminum-clad Hi-C Cores

D. Activation Traverses in Zoned Cores

Radial activation traverses were taken in several of the zoned loadings to determine if there was significant spectral mismatching between the stainless steel-clad central zone and the outer driver regions, and if there was a region of asymptotic spectrum in the central zone in which microparameter measurements could be made. In Cores 9 and 21, radial activation traverses were taken at the horizontal midplane of the fuel region. Core 9

typified the looser Hi-C lattices in which the central stainless steel-clad zone was surrounded by uniformly loaded aluminum-clad fuel at the same

lattice pitch; Core 21 represented the extreme case of a central zone surrounded by a uniformly loaded zone of aluminum-clad fuel at a looser lattice pitch, which in turn was surrounded by an outer driver zone of nonuniformly loaded aluminum-clad fuel at a still higher H-to-U²³⁸ atom ratio.

In Core 9, traverses were taken with three types of foils: (1) cadmium-covered and bare gold, (2) cadmium-covered and bare manganese, and (3) bare copper, dysprosium, and depleted uranium. A typical plot, that of bare gold, is shown in Fig. 13. Figure 14 presents plots of cadmium-covered gold, bare dysprosium, and bare depleted uranium all normalized to the bare-gold plot. This figure was constructed from smoothed plots of the data rather than from the data themselves. It appears that microparameter measurements may be made at almost any position within the stainless-steel zone and be in a position of asymptotic spectrum.

112-7947

Fig. 13. Bare-gold Activation Traverse in 1.24-cm Square Stainless Steel-clad Zoned Hi-C Core

112-7946

Fig. 14. Foil Activation Traverses Normalized to Bare Gold in 1.24-cm Square Stainless Steel-clad Zoned Hi-C Core

In Core 21, cadmium-covered and bare gold and manganese and bare depleted uranium foils were used for the activation traverses. Typical of the results is the bare-gold traverse shown in Fig. 15.

112-5674

Fig. 15. Bare-gold Activation Traverse in 1.069-cm Triangular Stainless Steel-clad Zoned Hi-C Core

Local irregularities in loading along the sides of the box containing the tight zone are probably responsible for the activation peak just outside the box. Figure 16, which was constructed from smoothed plots rather than from the data points, presents plots of cadmium-covered gold and bare depleted uranium normalized to the bare-gold traverse. No large volume of asymptotic spectrum is indicated

in the center of this zone as there was in the 1.24-cm square stainless steel-clad Hi-C zoned core. The tight zone was about an 11-cm equivalent radius, and the microparameter measurements made in it were obtained at radii of 3.4 to 4.8 cm. Judging by other studies of two-region cores in which comparisons were made with uniformly loaded critical cores,¹⁴ these

112-5673

Fig. 16. Foil Activation Traverses Normalized to Bare Gold in 1.069-cm Triangular Stainless Steel-clad Zoned Hi-C Core

of a corresponding change in temperature. In the second method, reactivity changes were detected with positive-period measurements; that is, in successive reactor runs at different temperatures, the critical control element was withdrawn to a fixed point. A change in control-blade worth with temperature would affect the results reported for those measurements. Consequently, with Core 13, excess reactivity was measured by rising periods with all control elements withdrawn, at each temperature. No significant difference was noted in the results obtained by the two methods.

The bulk water moderator temperature ranged from 20 to 46°C. Nine thermocouples were installed in the reactor vessel and storage tank; however, the average of four representative thermocouples was chosen for the calculations. These four were located as follows: at the approximate center of the core in a voided fuel tube, 2.5 cm below the top grid on a control-blade guide, 2.5 cm above the bottom grid on a control-blade guide, and at the midplane of the fuel zone on a control-blade guide. Readings were recorded only after a temperature equilibrium was reached, as indicated by a spread of less than 1.1°C among the four temperature readings.

Measurements were made during both the heating period, from room temperature to about 40-46°C, and the cooling period. The data were then least-squares fitted to yield the temperature coefficient. The results of these measurements are given in Table XIII.

All cores measured had negative coefficients, although no definite trend with H-to-U²³⁸ atom ratio was noted.

measurements are expected to be within a few percent of results that would have been obtained in a large region of asymptotic spectrum.

E. Temperature Coefficient of Reactivity

The temperature coefficient of reactivity was determined for Cores 1, 2, 3, 8, 11, and 13. Two methods were used. The first method compared the change of control-element position, with its associated reactivity change, to that

TABLE XIII. Temperature Coefficient of Reactivity in Hi-C and BORAX-V Cores

Core No.	Temperature Coefficient, % ($\Delta k/k$) per °C
1	-0.0128 $\pm$ 0.0005
2	-0.0101 $\pm$ 0.0005
3	-0.0076 $\pm$ 0.0008
8	-0.0106 $\pm$ 0.0004
11	-0.0101 $\pm$ 0.0007
13	-0.0095 $\pm$ 0.0005

F. Miscellaneous Measurements

During the Hi-C experiments, various material reactivity worths, void worths, top reflector worths, and grid plate worths were measured. These measurements and results are discussed below.

1. Material Reactivity Effects in BORAX-V Cores

Component worths were determined at various radii in both BORAX-V cores. Worths were established in the square lattice from the compensating changes in the positions of a finger control rod and represent reactivity differences between the core with the component and the core with the component position occupied by water. Not all measurements could be made in the same size core; the more effective absorbers such as boron-stainless steel required a larger core (up to 701 fuel pins instead of 621) than did the other measurements. The results obtained in Core 1 are given in Table XIV.

TABLE XIV. Reactivity Worths of Various Components in 1.27-cm Square BORAX-V Core

Component	Distance from Center, cm				Component Dimensions, cm
	0	5.08	10.16	17.78	
	Reactivity Loss per Component, % $\Delta k/k$				
Boron SS ^a in Cd Tube	0.858	-	-	-	Cd tube: 1.0795 OD, 1.0287 ID; SS tube: 0.955 OD, 0.659 ID; Boron SS: 0.635-OD solid rod
Water-filled Cd tube	0.760	-	-	-	1.0795 OD, 1.0287 ID
Fuel pin in Cd tube	0.689	-	-	-	1.0795 OD, 1.0287 ID
Void Cd tube	0.619	-	-	-	1.0795 OD, 1.0287 ID
Boron SS	0.465	-	0.310	0.131	0.955 OD, 0.659 ID
SS pin	0.180	0.159	0.108	0.385	0.955-OD solid rod
0.21 w/o UO ₂ in SS tube	0.139	0.128	0.087	-	SS tube: 0.955 OD, 0.8788 ID; UO ₂ : 0.871 OD
Void SS tube	0.099	0.090	0.067	-	0.955 OD, 0.8788 ID
Aluminum pin	0.081	-	0.051	-	0.955-OD solid rod
4.95 w/o UO ₂ in SS tube	0.078	0.068	0.046	0.0022	SS tube: 0.955 OD, 0.8788 ID; UO ₂ : 0.871 OD
Void Al tube	0.073	-	-	-	0.955 OD, 0.7881 ID
Teflon pin	0.067	0.060	0.043	0	0.955-OD solid rod
9.9 w/o UO ₂ in SS tube	0.042	0.036	0.025	-	SS tube: 0.955 OD, 0.8788 ID; UO ₂ : 0.871 OD
Water-filled Al tube	0.020	-	-	-	0.955 OD, 0.7881 ID

^a1.3 w/o natural boron in SS rod.

The effect on the core excess reactivity of evenly distributed stainless steel pins (0.955-cm OD) in the 701 fuel-pin core is given in

TABLE XV. Reactivity Worths of Distributed Stainless Steel Pins in 1.27-cm Square BORAX-V Core

No. of Pins	Reactivity Loss, % $\Delta k/k$
29	1.484
13	0.665
5	0.264
1	0.087

Table XV. Here reactivity differences are between the normal core and the core with fuel positions occupied by the stainless steel pins.

In the triangular lattice, worths were determined from the reactor period caused by the withdrawing of each component. Once again results are expressed as reactivity differences between the core with a component and the core with the component position occupied by water. The results in Core 2 are given in Table XVI.

TABLE XVI. Reactivity Worths of Various Components in 1.27-cm Triangular BORAX-V Core

Component	Distance from Center, cm			Component Dimensions, cm
	0	10.16	19.05	
	Reactivity Loss per Component, % $\Delta k/k$			
Cadmium Tube I	0.574	-	-	1.084 OD, 0.982 ID
Boron SS pin	0.395	-	-	0.955-OD solid rod
Cadmium Tube II	0.194	-	-	0.3454 OD, 0.2434 ID
0.21 w/o UO ₂ in SS tube	0.124	-	-	Same as fuel pin
Void SS tube	0.091	-	-	0.955 OD, 0.8788 ID
4.95 w/o UO ₂ in SS tube	0.081	0.058	0.010	Same as fuel pin
Void Al tube	0.080	-	-	0.955 OD, 0.783 ID
Al pin	0.078	-	-	0.955-OD solid rod
Teflon pin	0.066	0.048	0.011	0.955-OD solid rod
9.9 w/o UO ₂ in SS tube	0.058	-	-	Same as fuel pin
Water-filled Al tube	0.026	-	-	0.955 OD, 0.783 ID

2. Void Tube Worth and Stainless Steel-clad Pin Worths in Hi-C Cores

No detailed determination of component reactivity worths was made in the various Hi-C cores as was made in the BORAX-V cores. Only a few miscellaneous measurements were done.

In the 1.24-cm square aluminum-clad Hi-C core (Core 10), the worth of a voided aluminum cladding tube was $-0.0561\% \Delta k/k$. The measurement was made at 9.92 cm from the core center.

The substitution of stainless steel-clad fuel pins for aluminum-clad fuel pins at the center of Core 13 reduced the core reactivity by $0.0041\% \Delta k/k$ per pin. This value was obtained by observing the reactivity change from the substitution of 18 stainless steel-clad fuel pins about the core center. The central lattice position was unoccupied by a fuel pin both before and after the substitution.

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Fig. 17. Integral Worth of Water in 1.24-cm Square Stainless Steel-clad Hi-C Core

3. Top Reflector Worth

In several cores, the integral worth of the water as a function of height was determined. The integral water-worth curve was generated by noting control-element positions required to compensate for the indicated change in water level. Direct measurement of top reflector worth by rising periods was compared with control-blade interchange, in Core 1. The rising-period data indicated a top reflector worth of 0.347% , in comparison with 0.342% by the blade calibration. A typical water-worth curve is given in Fig. 17. The size of the core in which these measurements were made varied from 1749 to 1809 fuel pins. A reactivity worth of 0.12% is determined for the effectively infinite top reflector extending from the top of the fuel at 122 cm (48 in.). The top reflector worth for several of the Hi-C and BORAX-V cores is given in Table XVII.

4. Grid-plate Worth

A third positioning grid was installed in Cores 4 and 14 to reduce data scatter caused by the bowing of fuel pins. The third grid, identical to the top and bottom positioning grids, was placed about $7\frac{1}{2}$ cm below the fuel centerline and in Core 14 caused a 0.20% loss in reactivity because of its added absorption and water displacement.

TABLE XVII. Top Water Reflector Worths
in Some Hi-C and BORAX-V Cores

Lattice	Core Number	Top Water Reflector Worth, $\% \Delta k/k$
1.27-cm square BORAX-V	1	0.319
1.24-cm square Hi-C	8	0.117
1.24-cm square Hi-C	10	0.125
1.27-cm triangular Hi-C	11	0.157

V. INITIAL CONVERSION RATIOS AND OTHER MICROPARAMETER MEASUREMENTS

A. Initial Conversion Ratio

1. Theory and General Method

The initial conversion ratio (ICR) is defined for these assemblies as the ratio of the Pu^{239} production rate to the U^{235} destruction rate in a fresh core. The method of measuring the ICR involves determining the relative fission rates and the relative Pu^{239} production in uranium foils irradiated simultaneously in a fuel pin and in a pure thermal flux. These experimentally determined relative rates are then related to the ICR through the equation

$$\text{ICR} = \frac{N^{28}}{N^{25}} \frac{P}{R} \left[\frac{\sigma_{c,\text{thermal}}^{28}}{\sigma_{f,\text{thermal}}^{25}} \right] \left[\frac{1}{1 + \alpha^{25}} \right], \quad (17)$$

where N^{28} and N^{25} are the U^{238} and U^{235} atom densities for the fuel, the cross sections are the effective values for the thermal spectrum used, α^{25} is the average U^{235} capture-to-fission ratio in the fuel spectrum, P is the ratio of the relative Pu^{239} production in a foil irradiated in the fuel to the relative Pu^{239} production in a foil irradiated in the thermal flux, and R is the ratio of the relative U^{235} fission in a foil irradiated in the fuel to the U^{235} fission of a foil irradiated in the thermal flux. This method has been extensively developed.^{15,16}

Only P and R are experimentally determined; the other factors in Eq. (17) must be computed. The graphite thermal column of the ATSR was used as the reference thermal flux for these experiments. As part of each set of irradiations, the gold cadmium ratio (as determined by 0.025-mm-thick foils and 0.051-cm-thick cadmium covers) was measured at the reference location in the ATSR thermal column. The effective cross sections for the reference thermal flux were calculated¹⁷ for spectra characterized by these gold cadmium ratios. The α^{25} values were determined by a semi-empirical technique in which the subcadmium and epicadmium α 's were weighted by the U^{235} fission cadmium ratio by

$$\alpha^{25} = \alpha_s \frac{(\text{CR})_f^{25} - 1}{(\text{CR})_f^{25}} + \frac{\alpha_e}{(\text{CR})_f^{25}}. \quad (18)$$

The α_e and α_s values were obtained from the results of GAM-1 and THERMOS calculations.

Data were taken with 0.013-cm-thick uranium-metal foils of various enrichments (usually 0.21 and 3.23 w/o) and of the same diameter as the fuel pellets. The foils were covered with about 0.025 mm of aluminum foil to prevent contamination by the fuel and were irradiated between fuel pellets in flux-symmetric positions. Similar foils were placed in the reference position in the ATSR thermal column, and both sets of foils were irradiated simultaneously.

2. Counting Technique

The relative amount of Pu^{239} produced in the foils was determined by a gamma-gamma coincidence technique.¹⁸ About half the beta decays of Np^{239} result in an excited state of Pu^{239} , which emits a 106-keV gamma ray followed mainly by either a 228- or a 278-keV gamma ray. The two higher-energy gammas have large internal K-conversion coefficients, giving rise to 104-keV X rays of comparable intensity to the 106-keV gamma ray. A nominal 100-keV peak is then formed by the roughly equal mixture of unresolved 106-keV gamma rays and the Pu^{239} X rays. By looking for gamma-gamma or gamma-X-ray coincidences, we can obtain a high discrimination against the fission-product gamma activities that will be present in the irradiated uranium foils. Thus, a satisfactory measurement of Pu^{239} production is based on observing coincidences between the 106-keV gamma ray and the Pu^{239} X ray.

Since counting was not started until more than 4 hr after the termination of an irradiation, the U^{239} decay was completed and the observed gamma activity above 411 keV was a good measure of the fission-product activity. For foils in the ATSR thermal column, all the fission-product activity was considered the result of U^{235} fission; however, for foils in the fuel an appreciable part of this activity was the result of fast fission in the U^{238} , and it was necessary to irradiate two foils of different isotopic concentration in order to identify the component of the activity due to U^{235} fission. The ratio of the total fission-product activity of a foil irradiated in the fuel to the total fission-product activity of a foil irradiated in the ATSR thermal column, designated by r , is given by

$$r_1 = \frac{b_1 F_{\text{fuel}}^{25} + (1 - b_1) F_{\text{fuel}}^{28}}{b_1 F_{\text{T.C.}}^{25}}, \quad (19)$$

and for a foil of different enrichment by

$$r_2 = \frac{b_2 F_{\text{fuel}}^{25} + (1 - b_2) F_{\text{fuel}}^{28}}{b_2 F_{\text{T.C.}}^{25}}, \quad (20)$$

where indices 1 and 2 refer to two different degrees of enrichment, b , and fuel and T.C. refer to the fission-product activity of foils in the fuel

and the ATSR thermal column, respectively. The ratios r_1 and r_2 are the experimentally determined quantities. If Eqs. (19) and (20) are solved for the ratio $F_{\text{fuel}}^{28}/F_{\text{T.C.}}^{25}$ and combined, they give the following expression for R:

$$R = \frac{F_{\text{fuel}}^{28}}{F_{\text{T.C.}}^{25}} = \frac{r_1 - r_2(b_2/b_1)[(1-b_1)/(1-b_2)]}{1 - (b_2/b_1)[(1-b_1)/(1-b_2)]} \quad (21)$$

Here, it is assumed that the counting efficiency is the same for all foils. Note, however, that the above expression for R is independent of the ratio of activity per fission from fission products of U^{235} to the activity per fission from fission products of U^{238} .

If these factors are inserted into Eqs. (19) and (20), they will cancel out when expression (21) is formed.

*PULSE-HEIGHT ANALYZER

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Fig. 18. Coincidence Counting Circuit

lyzers followed by a coincidence circuit. The crystals were located inside a well-shielded sample changer, which is described along with its control system in ANL-6628.¹¹

The two crystals were each 5.08 x 5.08 cm and 1.5 cm apart. Two analyzers were centered on the nominal 100-keV peak from the Pu^{239} cascade. The channel width used was a compromise between rejection of fission-product activities and increased coincidence count rate and was usually 36 keV. The outputs from these differential selectors were fed into a coincidence circuit with a resolving time of 0.5 μ sec as well as being recorded individually. In addition, two other analyzers were used in the integral mode as discriminators to select pulses due to radiation greater than 411 keV, which was used as a measure of fission rate.

The many fission products can give rise to true coincidences in the 100-keV region by several mechanisms. These include: actual cascade of two 100-keV gammas, degradation by Compton scattering of higher-energy cascade gammas, coincidence between bremsstrahlung and either original or scattered gammas, and any of the above coupled with photo-electric interaction of higher-energy gammas with uranium resulting in 97-keV X rays. In each of these, except for the first, a decrease may be achieved by shielding one crystal from the other. Since in the arrangement here the counting planchet afforded the best leakage path if simply made from aluminum, a special anti-Compton planchet (see Fig. 19) was designed

to reduce the effect of this scattering. The 0.025 cm of aluminum served as a structural base and foil support. The 0.051-cm thickness of lead stopped 90% of the gamma radiation in the 100-keV range. However, the characteristic X ray from the lead is not resolved from the 100-keV gammas,

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Fig. 19. Anti-Compton Counting Planchet

so the 0.076-cm cadmium discs were added to attenuate this radiation. The use of this planchet reduced the coincidences from fission-product gammas by about 60%. The small annulus of 0.21% depleted uranium was added to the planchet to equalize the self-attenuation experienced by radiation leaving the foil surfaces. These rings had little or no effect on the observed coincidence count rates, however.

3. Corrections to Data

The two observed quantities, gamma coincidences (nominal 100 keV) and fission-product gamma activity (above 411 keV) are subject to various corrections in order to

yield the desired information, namely P/R , the ratio of the relative Pu^{239} production in a foil irradiated in the fuel to the relative Pu^{239} production in a foil irradiated in a pure thermal flux, divided by the ratio of the U^{235} fission of a foil irradiated in the fuel to the U^{235} fission of a foil irradiated in a pure thermal flux.

The foil data-reduction program makes the calculations for normalized activities corrected for background, counter deadtime, counting time, and weight differences. Unirradiated foils are counted simultaneously with the activated foils to determine the background levels. For the relative fission-product activities, these corrections, plus a decay correction, are all that need be made. There is no problem with self-attenuation in the foils when counted for fission-product activities above 411 keV.

For the observed coincidence counts at the 100-keV setting, the following additional corrections are necessary:

- (a) Accidental coincidences.
- (b) Fission-product gamma coincidences.
- (c) Fuel-pellet foil gap.
- (d) Self-attenuation.
- (e) Thermal self-shielding in ATSR thermal column.
- (f) Epicadmium activation in ATSR thermal column.

These corrections are discussed in the paragraphs below.

a. Accidental Coincidences

Accidental coincidences were computed from the measured single-channel count rates and the resolving time of the coincidence circuit by

$$A.C. = 2\tau mn, \quad (22)$$

where τ is the resolving time of the coincidence circuit ($0.5 \mu\text{sec}$) and m and n are the single-channel count rates from the differential discriminators used as input to the coincidence circuit. With the levels of activity possible from a low-flux critical assembly, this correction was usually less than 1% of the total coincidence count rate.

b. Fission-product Gamma Coincidences

The coincidences arising from fission-product gamma activity were obtained as a time-dependent function from an auxiliary calibration in which the fission and coincidence activities of an irradiated U^{235} foil (93%) were compared at the same times. This experiment gives the factor, C_C , defined as the coincidence count rate divided by the single-

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Fig. 20. Fission-product Correction Factor, C_C , for Typical HI-C Core

channel count rate above 411 keV. This factor is plotted for a typical core in Fig. 20. In the determination of C_C , 0.025-mm-thick foils were irradiated and then counted while sandwiched between 0.051-mm-thick foils of unirradiated uranium to give the same self-attenuation as the 0.127-mm-thick foils for the 100-keV radiation. When applied to the 0.21-w/o enriched uranium foils, this correction was usually 1 to 6% of the total true coincidences. Foils of a nominal 3-w/o enrichment were also irradiated to evaluate the factor R as given by Eq. (21). It was hoped that these foils would serve a double purpose and help indicate relative Pu^{239} production from coincidence counting. However, the C_C correction for these foils ran from 4 to 63% of the total true coincidences. Because of the magnitude of this correction, the data obtained from the nominal 3-w/o enriched foils were not used in the determination of the relative Pu^{239} production. The anti-Compton planchet previously described reduced the C_C correction to less than one-third of the value obtained when the planchet was not used.

c. Fuel-pellet Foil Gap

When the uranium foils were sandwiched between two fuel pellets in an irradiation, the exposed surface of the foils was only a small

fraction of their total surface area, and any misalignment or gap between foil and pellet would cause an error in the contribution from surface resonance absorption. The metal foils used were covered with about 0.025 mm of aluminum foil to prevent contamination by the fuel pellets. (A gap of about 0.040 mm was created, however, because of folding over of the aluminum foil when wrapped about the uranium foil.) To evaluate the effect of this gap on the observed capture rate, a series of foils was loaded with gaps that varied from nominal zero to 0.013 cm. For these determinations, as for practically all the irradiations, specially prepared fuel slices were used adjacent to the foils to achieve surfaces as chip-free and flat as possible. This "gap" factor resulted in a correction of about 1.5% on the observed coincidence count rate. The "gap" measurement was performed in most of the cores, and the correction ranged from 1.2 to 1.75%. No trend of this correction with H-to- U^{238} ratio was noted. In those cores in which no "gap" measurement was made, a correction of 1.5% was used.

d. Self-attenuation

The coincidence counting of 100-keV radiation from 0.013-cm uranium foils necessitated self-attenuation corrections. Preferential overweighting of the surface activity at the edges of the foils was prevented by counting the foils within the unirradiated uranium annuli of the anti-Compton planchets. (It was previously mentioned, however, that the presence of these rings had little or no effect on the observed coincidence count rates.) An auxiliary self-attenuation experiment was performed by counting 0.025-mm-thick foils sandwiched in varying thicknesses

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Fig. 21. Self-attenuation Factor for Coincidence Counting of Uranium Foils

of unirradiated foils as well as by counting a stack of irradiated 0.025-mm-thick foils, which were previously counted separately. The results of the measurement of this self-attenuation factor are shown in Fig. 21. Here, w is the weight in grams of the 0.927-cm-diam nominal 0.127-mm-thick uranium foil. Since all foils were prepared with a precision punch-and-die set, it was assumed that any mass differences among the foils resulted from a thickness variation rather than an area difference when this correction factor was applied. The magnitude of the self-attenuation correction never exceeded 3%.

e. Thermal Self-shielding in ATSR Thermal Column

In several Hi-C cores, fuel pellets as well as metal foils were irradiated. These fuel pellets underwent radiochemical separation for neptunium, which was then counted as a check on the metal-foil

coincidence-counting technique.¹⁹ There was a significant difference between the results obtained with metal foils and the results obtained with the fuel pellets. This difference was traced to the thermal irradiation in the ATSR thermal column. Subsequent irradiations of both metal foils and various lengths of oxide pellets, followed by neptunium separation and counting of both, disclosed that the discrepancy was caused partially by thermal self-shielding.

All the coincidence activities of ATSR thermal-column foils were corrected for thermal self-shielding by dividing the measured foil activity by the self-shielding function, $G(\tau)$. $G(\tau)$, where τ is the foil thickness in absorption lengths, was obtained²⁰ from

$$G(\tau) = (1/\tau)[1/2 - E_3(\tau)], \quad (23)$$

where E_3 is the exponential integral. The oxide pellets were corrected for thermal self-shielding by an escape-probability method. The escape probability, P_0 , was computed from the rational approximation,

$$P_0 = [1 + \Sigma_a \bar{\ell}]^{-1}, \quad (24)$$

where

$$\bar{\ell} = 4V/A. \quad (25)$$

Here V is the volume and A the surface area of the pellet. After division by the escape probability, the specific coincidence activities of the oxide pellets were independent of pellet length and in agreement with foil data.

f. Epicadmium Activation in ATSR Thermal Column

In the thermal column location in which most of the thermal irradiations were made, the U^{238} capture cadmium ratio ranged from 24 for the metal foils to 56 for the 1.27-cm-long oxide-fuel pellets. The U^{238} capture cadmium ratio was determined in each thermal column location used and all data corrected for epicadmium activation. The gold cadmium ratio was also measured at each thermal column location.

4. Data and Results

The data obtained in the BORAX-V and Hi-C cores are listed in Table XVIII. Results for each irradiation are given in terms of a modified conversion ratio (MCR) as well as the initial conversion ratio. The MCR is the rate of plutonium production relative to the rate of U^{235} fission and is defined as

$$\text{MCR} = \frac{N^{28}}{N^{25}} \frac{P}{R} \left[\frac{\sigma_{c,\text{thermal}}^{28}}{\sigma_{f,\text{thermal}}^{25}} \right] \quad (26)$$

where the symbols are identical to those in Eq. (17). MCR differs from the ICR only in that it does not depend on the computed value of α^{25} .

TABLE XVIII. Initial Conversion-ratio Data in Hi-C and BORAX-V Cores

Date	Run No.	Type of P Measurement, Foil Enrichment, etc.	Ratio ^a P	Ratio ^a R	ATSR Gold Cd Ratio	Modified Conversion Ratio (MCR)	Initial Conversion Ratio (ICR)
<u>1.27-cm Square BORAX-V, Core 1</u>							
8/11/61	28-1	0.21% foil-count	2.64	0.690	90	0.378 (1) ^b	0.306 (1) ^b
10/4/61	38-7	0.21% foil-count	9.83	2.73	90	0.356 (1)	0.288 (1)
<u>1.27-cm Triangular BORAX-V, Core 2</u>							
1/8/62	71-93	0.21% foil-count	0.502	0.127	50	0.413 (1)	0.330 (1)
<u>1.349-cm Square Hi-C, Core 4</u>							
12/3/65	418-4	0.21% foil-count	2.34	0.643	35	0.656 (2)	0.536 (2)
12/7/65	418-5	0.21% foil-count	1.62	0.475	35	0.617 (2)	0.504 (2)
12/9/65	418-6	0.21% foil-count	1.77	0.506	35	0.631 (2)	0.515 (2)
<u>1.24-cm Square Hi-C, Cores 8 and 9</u>							
12/13/62	183-6	0.21% foil-count	2.32	0.584	115	0.634 (1)	0.510 (1)
3/18/63	224-1	6 ppm foil-count	1.17	0.298	115	0.627 (1)	0.504 (1)
3/18/63	224-1	0.21% foil-count	1.21	0.298	115	0.648 (1)	0.520 (1)
5/6/64	383-1	0.21% foil-count	4.96	1.25	105	0.637 (1)	0.512 (1)
5/6/64	383-1	1.27-cm pellet-radiochem	5.11	1.25	105	0.656 (2)	0.527 (2)
6/3/64	383-10	0.21% foil-count	14.8	3.79 ^c	105	0.627 (2)	0.504 (2)
6/3/64	383-10	1.27-cm pellet-radiochem	14.3	3.72	105	0.617 (2)	0.496 (2)
7/6/64	387-9	0.21% foil-count	11.4	3.04	105	0.602 (2)	0.484 (2)
7/6/64	387-9	0.21% foil-radiochem	11.9	3.04	105	0.629 (2)	0.507 (2)
7/6/64	387-9	1.27-cm pellet-radiochem	10.8	3.04	105	0.570 (2)	0.458 (2)
<u>1.27-cm Triangular Hi-C, Core 12</u>							
7/17/63	283-7	0.21% foil-count	16.1	3.35	115	0.767 (2)	0.610 (2)
8/13/63	285-6	0.21% foil-count	1.18	0.271	115	0.694 (2)	0.552 (2)
<u>1.166-cm Triangular Hi-C, Core 14</u>							
4/2/64	372-12	0.21% foil-count	2.65	0.399	105	1.067 (2)	0.823 (2)
4/3/64	372-13	0.21% foil-count	2.77	0.411	105	1.082 (2)	0.835 (2)
4/9/64	373-7	0.21% foil-count	2.49	0.364	105	1.098 (2)	0.847 (2)
<u>1.127-cm Triangular Hi-C, Core 19</u>							
12/3/63	342-10	0.21% foil-count	4.61	0.597	115	1.232 (2)	0.935 (2)
12/3/63	342-10	d -radiochem	4.09	0.597	115	1.093 (2)	0.830 (2)
12/5/63	342-11	0.21% foil-count	4.74	0.582	115	1.300 (2)	0.987 (2)
12/23/63	342-22	0.21% foil-count	1.64	0.217	115	1.206 (1)	0.916 (1)
12/23/63	342-22	e -radiochem	1.61	0.217	115	1.184 (2)	0.899 (2)
<u>1.069-cm Triangular Hi-C, Core 21</u>							
7/22/64	391-15	0.21% foil-count	2.20	0.192	32	2.094 (3)	1.537 (3)
7/23/64	391-16	0.21% foil-count	2.00	0.199	32	1.837 (3)	1.349 (3)
7/24/64	391-17	0.21% foil-count	2.21	0.189	32	2.138 (3)	1.570 (3)

^aThis ratio is defined on p. 42.

^bNumbers in parentheses denote number of independent determinations.

^cThis value of R was determined by radiochemical techniques; all other values of R were obtained from foil-counting data.

d1.27-cm pellets used in the lattice; 0.254-cm pellets used in the ATSR.

e1.27-cm pellets used in the lattice; 0.127-cm pellets used in the ATSR.

In the 1.24-cm square and the 1.127-cm triangular Hi-C cores, fuel pellets as well as metal foils were irradiated in both the fuel and the ATSR thermal column. After irradiation, the neptunium in the pellets was radiochemically separated from the uranium and fission products and counted. In the 1.24-cm square Hi-C core, a set of 0.21-w/o enriched uranium metal foils was first coincidence-counted and then processed radiochemically before the separated neptunium was counted. These data indicate that within the error limits of the experiment (2-3%), the radiochemical and coincidence-counting methods give identical results. These data also indicate that the use of metal foils to determine the capture rate in an oxide fuel rod introduced no significant error.

For each run, Table XVIII gives the gold cadmium ratio in the ATSR thermal column location used for the "thermal" irradiation. These gold cadmium ratios were obtained with 0.025-mm-thick, 0.818-cm-diam foils. The cadmium cover used was 0.051 cm thick. From these cadmium ratios, the Westcott g factor¹⁷ for each location was determined, and the average U^{238} capture and U^{235} fission thermal cross sections could be computed. The 2200-m U^{238} capture and U^{235} fission cross sections were taken to be 2.72 and 581.95 b, respectively.

Table XIX presents the average MCR and ICR for each core, computed from the data of Table XVIII. In the two BORAX-V cores, an error of 5% was assumed; in the Hi-C cores, for which numerous independent determinations were made, the standard deviation of the mean is quoted as the error. The α^{25} used for each core to compute the ICR is also given in Table XIX.

TABLE XIX. Initial Conversion Ratios in Hi-C and BORAX-V Cores

Lattice	Core No.	H-to- U^{238} Atom Ratio	Modified Conversion Ratio (MCR)	α^{25}	Initial Conversion Ratio (ICR)
1.27-cm square BORAX-V	1	4.65	0.367 ± 0.018	0.236	0.297 ± 0.015 (2) ^a
1.27-cm triangular BORAX-V	2	3.53	0.413 ± 0.021	0.252	0.330 ± 0.016 (1)
1.349-cm square Hi-C	4	4.16	0.635 ± 0.007	0.225	0.518 ± 0.006 (6)
1.24-cm square Hi-C	8 & 9	2.92	0.622 ± 0.006	0.244	0.500 ± 0.005 (16)
1.27-cm triangular Hi-C	12	2.29	0.730 ± 0.021	0.257	0.581 ± 0.017 (4)
1.166-cm triangular Hi-C	14	1.33	1.082 ± 0.006	0.296	0.835 ± 0.005 (6)
1.127-cm triangular Hi-C	19	0.98	1.203 ± 0.025	0.317	0.913 ± 0.019 (9)
1.069-cm triangular Hi-C	21	0.50	2.023 ± 0.047	0.362	1.485 ± 0.034 (9)

^aNumbers in parentheses denote number of independent determinations.

The ICR is plotted versus H-to- U^{238} atom ratio in Fig. 22. The value obtained in the loosest Hi-C core seems high. This value was obtained from the last data taken in the program, when the special pellets used to face foils were becoming slightly worn and chipped and may have caused high foil activations. This may also indicate that the foil positioning

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Fig. 22. Initial Conversion Ratios in Hi-C and BORAX-V Cores

both bare and cadmium-covered depleted uranium metal foils and computing the cadmium ratio from the measured Np^{239} activity was used.

Uranium metal foils, 0.21-w/o enriched, 0.013 cm thick, and the diameter of the fuel, were used for this measurement. The foils were irradiated between fuel pellets in double-slotted stainless steel-clad fuel pins as seen in Fig. 23. Only specially prepared, smooth-faced fuel pellets were used adjacent to foils. All foils were covered with 0.025-mm-thick aluminum foil as a protection against fission-product and uranium contamination. Cadmium-covered irradiations were obtained by placing the foil between 0.051-cm-thick cadmium discs and placing a 1.27-cm-long, 0.051-cm-thick cadmium sleeve around the fuel pin. It was necessary to place 0.254-cm-thick UO_2 spacers between the cadmium-disc covers and the foil for the cadmium-covered irradiations to prevent streaming of epithermal neutrons through the edges of the cadmium discs. The 0.254-cm thickness of UO_2 was found experimentally to be about optimum and has been used by other experimenters.^{21,22} In the Hi-C cores, five to eight irradiation positions (all the same distance from the core center) were used in each run to randomize the effects of local variations in the lattice. Usually, three runs were performed in each lattice.

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Fig. 23. Foil Placement for U^{238} Capture Cadmium Ratio Measurement

After activation, the foils were assayed for Np^{239} activity by the coincidence method previously described for the measurement of the ICR. The corrections necessary for the lattice data in that experiment were used for these data also, with the exception of the "gap" correction. In these lattices, the U^{238} capture cadmium ratio was very close to unity, and

and pellet-face chipping problem was more severe than anticipated, and that the results given in Table XIX should be assigned larger errors.

B. U^{238} Capture Cadmium Ratio

1. Methods of Measurement

The U^{238} capture cadmium ratio, C_{28} , was determined in most cores by two methods. In all but one core the straightforward method of irradiating

the "gap" corrections for bare and cadmium-covered foils were, within error limits, indistinguishable. Since the desired result was the ratio of bare-to-cadmium-covered activities, the "gap" correction was not necessary.

In Core 9 (1.24-cm square stainless steel zone), 0.254-cm-thick UO_2 fuel pellets, both bare and cadmium covered, were irradiated instead of the uranium metal foils. After irradiation, the pellets were dissolved and the neptunium separated and counted. This experiment served as a check on the foil technique.

This direct method of obtaining the U^{238} capture cadmium ratio presented some difficulties, however. The first was the effect of the cadmium on the measurement. The measurement of a cadmium-covered activation involved placing a cadmium sleeve around a fuel pin. Even though this sleeve was thin (0.051 cm), a considerable fraction of the moderator was removed from the vicinity of the measurement in the tight lattices used here. The effect of this may be large in the very tight lattices. In the extreme, it was not even possible to place a 0.051-cm-thick sleeve around a fuel rod in the tightest lattice (Core 21: 1.069-cm triangular stainless steel zone).

A second objection to the direct method is the large error in the quantity $(C_{28} - 1)$ as C_{28} approaches one. Of interest for theoretical comparisons is the quantity ρ_C^{28} , defined as the ratio of epicadmium to subcadmium captures in U^{238} , which is obtained from the measured capture cadmium ratio, C_{28} , by

$$\rho_C^{28} = (C_{28} - 1)^{-1}. \quad (27)$$

As may be seen, the error on ρ_C^{28} for a given error on C_{28} becomes very large as C_{28} approaches unity. For example, a 1% error in C_{28} produces a 25% error in ρ_C^{28} when the value of C_{28} is 1.04.

A method has been developed in the past few years to overcome these difficulties.²³ Called the thermal-activation method, the technique allows $(C_{28} - 1)$ to be determined directly without performing a cadmium-covered uranium activation in the experimental lattice. (When a cadmium-covered uranium activation is not made in the experimental lattice, the quantity actually determined is $(C_{28} - 1)/C_{28}$. However, when C_{28} is close to unity, the technique may be said to determine $(C_{28} - 1)$ directly.) Briefly stated, the method compares total absorptions in U^{238} to thermal absorptions in an intermediate material (usually dysprosium) having a much higher probability of thermal absorptions. Then, the thermal absorptions in the intermediate material and in the uranium are normalized in a thermal irradiation. The U^{238} capture cadmium ratio in the experimental lattice is given by

$$\left(\frac{C_{28}-1}{C_{28}}\right)_{\ell} = \left(\frac{A_x}{A_{28}}\right)_{\ell} \left(\frac{C_x-1}{C_x}\right)_{\ell} \left(\frac{A_{28}}{A_x}\right)_{\eta} \left(\frac{C_{28}-1}{C_{28}}\right)_{\eta} \left(\frac{C_x}{C_x-1}\right)_{\eta} K, \quad (28)$$

where A is the bare irradiation activity of the product isotope (Np^{239} in the depleted uranium foils) at some convenient time, the subscript 28 refers to U^{238} , the subscript x refers to the intermediate material, ℓ denotes an irradiation in the critical assembly lattice, η denotes an irradiation in another, more thermal neutron flux, C is the cadmium ratio, and the factor K is defined as

$$K = \frac{\left[\frac{(\sigma\phi)_{28}^{\text{th}}}{(\sigma\phi)_x^{\text{th}}} \right]_{\ell}}{\left[\frac{(\sigma\phi)_x^{\text{th}}}{(\sigma\phi)_{28}^{\text{th}}} \right]_{\eta}}. \quad (29)$$

All the ratios in Eq. (28) are measurable except K . K will depart from unity when the intermediate material departs from a $1/v$ behavior in the ℓ - and η -irradiations. When copper is used as the intermediate material, K is unity. When dysprosium is used, however, K must account for its non- $1/v$ behavior. Details of the computation of K for dysprosium are given in Ref. 23. The computation involves averaging a $1/v$ cross section and the Dy^{164} cross section over a variety of thermal-neutron distributions. For each spectrum, Westcott¹⁷ g -factors were calculated and a non- $1/v$ factor determined from the ratio of the Dy^{164} g -factor to the $1/v$ g -factor. The correction factor K was then found as the ratio of the non- $1/v$ factor in the normalizing thermal flux to the non- $1/v$ factor in the experimental lattice. In the Hi-C cores, K ranged from 1.016 to 1.079. The uncertainty in K is estimated to be about $\pm 20\%$ of $K - 1$.

Equation (28) indicates that the thermal or n -irradiation need only be performed once as long as subsequent lattice irradiations are counted with the same efficiency and time-corrected identically. Since many of the runs were weeks apart and the counting equipment was recalibrated periodically, a thermal irradiation was performed simultaneously with a lattice irradiation for each run.

The 0.21-w/o enriched uranium foils were counted for Np^{239} activity by the coincidence method as in the direct determination. It was necessary to apply the gap correction to these data, however, since a uranium lattice activation appears only once in the expression for C_{28} (Eq. 28). Some attention must also be given to thermal self-shielding, especially in the ATSR thermal column irradiations. For the foils used in these experiments (0.21-w/o enriched, 0.127-mm-thick uranium; 5-w/o, 0.127-mm-thick dysprosium-aluminum alloy; 0.051-mm-thick copper) the optical thicknesses ($\Sigma_a t$) were well enough matched with each other and the fuel that the net effect of thermal self-shielding on the U^{238} capture cadmium ratio was less than 1% and was neglected.

Both copper and dysprosium were used as intermediate materials. While dysprosium was used in all cores, copper was used only in one (Core 19), primarily as a check on the non-1/v correction for dysprosium.

The relatively long copper half-life made it convenient to count simultaneously with the uranium foils. Because of the relatively short half-life (140 min), dysprosium activity was counted during the first few hours after shutdown. The dysprosium foils irradiated in the experimental lattice contained appreciable amounts (5-10 cps) of an activity with about a 15-hr half-life. It was suspected that this activity was Na^{24} formed in an n, α reaction by fast neutrons on the aluminum in the dysprosium foils. The presence of this activity prevented data reduction with the RE-202 code when counting continued much beyond 5 or 6 hr after shutdown. It became the practice then to count dysprosium and uranium separately; dysprosium was counted within a few hours of shutdown to allow computer data reduction, and uranium counting was begun about 5 hr after shutdown.

2. Data and Results

The results of both the direct and thermal-activation method measurements of the U^{238} capture cadmium ratio are given in Table XX.

TABLE XX. U^{238} Capture Cadmium Ratio Results in Hi-C and BORAX-V Cores

Core No.	H-to- U^{238} Atom Ratio	Direct Measurement C_{28}	Thermal-activation Measurement			
			Intermediate Material	K	C_{28}	ρ_c^{28}
1	4.65	1.182 ± 0.041 (1) ^a	-	-	-	5.49 ± 1.24 ^b
2	3.53	1.188 ± 0.083 (1)	-	-	-	5.32 ± 2.35 ^b
4	4.16	1.279 ± 0.017 (9)	Dy	1.016	1.328 ± 0.004 (3) ^a	3.05 ± 0.04
9	2.92	1.257 ± 0.008 (19) (counting)	Dy	1.022	1.243 ± 0.002 (3)	4.12 ± 0.03
		1.236 ± 0.013 (3) (radiochemical)				
12	2.29	1.204 ± 0.012 (15) (counting)	Dy	1.026	1.185 ± 0.003 (3)	5.40 ± 0.09
		1.178 ± 0.035 (1) (radiochemical)				
14	1.33	1.105 ± 0.009 (9)	Dy	1.039	1.100 ± 0.001 (3)	10.0 ± 0.1
19	0.98	1.107 ± 0.022 (7)	Cu	1.000	1.068 ± 0.003 (2)	14.5 ± 0.4
			Dy	1.049	1.070 ± 0.001 (2)	
				ave	1.069 ± 0.002 (4)	
21	0.50	-	Dy	1.079	1.035 ± 0.001 (3)	28.6 ± 0.8

^aNumbers in parentheses denote number of independent determinations.

^bThese values were computed from the direct measurement C_{28} .

The results from the BORAX-V cores have large error assignments for several reasons. In each core, only one set of foils was irradiated. In Core 2, drifts in the counting equipment injected an additional

uncertainty. Analysis of these data is further complicated because, although the cadmium sleeves in the cadmium-covered irradiations were 0.051 cm thick, the discs that were used were only 0.038 cm thick. The Hi-C data, however, are beset by none of these difficulties.

The number of independent determinations given for the values of C_{28} determined by the thermal-activation method in Table XX is somewhat deceptive. These numbers given in Table XX are actually the number of separate reactor runs performed. In each reactor run, three to five foils of each kind necessary were irradiated. However, the nature of the data reduction made it desirable to consider the result of each reactor run the independent determination to be averaged to obtain C_{28} .

Several features of the Hi-C results deserve comment. First, in Core 19 both copper and dysprosium were used as intermediate materials in the thermal-activation method measurement. Within the error limits, the two materials give the same result. Since copper has no non- $1/v$ correction, this result indicates that the non- $1/v$ correction for dysprosium was satisfactory. Secondly, the direct measurements give, in general, values of C_{28} slightly larger than those obtained from the thermal-activation measurements. The error limits, given here as the standard deviation from the mean, on the thermal-activation method C_{28} values, however, are much smaller, in some cases by an order of magnitude, than those computed from the direct-measurement data. Some of the direct-measurement data in Cores 9 and 12 were taken in the early phases of the Hi-C measurements. For an unexplained reason, the data obtained then tend to give larger values of C_{28} than those obtained from later data. If these early data are neglected, the direct C_{28} values are then only slightly larger than the thermal-activation values. The error limits remain almost unchanged, however.

The directly determined C_{28} values are expected to be somewhat larger than the values obtained by the thermal-activation method. The cadmium sleeve required in the direct method removes a considerable fraction of the moderator from the vicinity of the measurement, and the epithermal U^{238} activation is thus decreased because of the depression of the source of fast neutrons in the immediate neighborhood of the measurement.

The limited amount of radiochemical data obtained in Cores 9 and 12 agrees well with the foil and thermal-activation methods of obtaining C_{28} . Since the values obtained from the thermal-activation method are more consistent, they are used whenever available to compute the episcadmium-to-subcadmium U^{238} capture ratio, ρ_C^{28} . These values of ρ_C^{28} are plotted against H-to- U^{238} atom ratio in Fig. 24.

Fig. 24
Epicadmium-to-Subcadmium U²³⁸ Capture
Ratio in HI-C and BORAX-V Cores

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C. U²³⁸-to-U²³⁵ Fission Ratio (δ^{28})

1. Method of Measurement

The U²³⁸-to-U²³⁵ fission ratio, δ^{28} , is of interest in these lattices since it is related to the fast fission factor, ϵ , through

$$\epsilon = [(\nu^{28} - 1)/\nu^{25}] \delta^{28} + 1, \quad (30)$$

where ν^{25} and ν^{28} are the average number of neutrons released per fission in U²³⁵ and U²³⁸, respectively.

The U²³⁸-to-U²³⁵ fission ratio may be written as

$$\delta^{28} = \frac{N^{28} \int_0^\infty \sigma_f^{28} \phi(E) dE}{N^{25} \int_0^\infty \sigma_f^{25} \phi(E) dE}, \quad (31)$$

where N^{28} and N^{25} are the U²³⁸ and U²³⁵ number densities in the fuel, and the σ_f 's are the fission cross sections. A common technique of evaluating δ^{28} (Ref. 22) is to irradiate both a depleted and an enriched uranium foil in the fuel. If $\gamma(t)$ is defined as the ratio of the measured fission-product activity per uranium atom of the depleted foil to the measured fission-product activity per uranium atom of the enriched foil, it may be written

$$\gamma(t) = \frac{\mu^{28}(t) N_D^{28} \int_0^\infty \sigma_f^{28} \phi(E) dE + \mu^{25}(t) N_D^{25} \int_0^\infty \sigma_f^{25} \phi(E) dE}{\mu^{28}(t) N_E^{28} \int_0^\infty \sigma_f^{28} \phi(E) dE + \mu^{25}(t) N_E^{25} \int_0^\infty \sigma_f^{25} \phi(E) dE}, \quad (32)$$

where the subscripts D and E designate the depleted and enriched foils, and the time-dependent quantities, $\mu^{28}(t)$ and $\mu^{25}(t)$, represent the activities

per fission from the fission products of U^{238} and U^{235} . The easily measured quantity, $\gamma(t)$, is then related to δ^{28} through Eqs. (31) and (32), to give

$$\delta^{28} = P(t) \left[\frac{\gamma(t) - (N_D^{25}/N_E^{25})}{(N_D^{28}/N_E^{28}) - \gamma(t)} \right]. \quad (33)$$

The quantity $P(t)$ is the ratio $\mu^{25}(t)/\mu^{28}(t)$ and has been evaluated as a function of time after scram.^{22,24,25} Equation (33) must be adjusted if the enriched foil being used for the determination has a different enrichment than the fuel and also if the foils used are not uranium metal but some alloy. These adjustments result in the rewriting of Eq. (33) as

$$\delta^{28} = P(t) \left(N_F^{28} N_E^{25} / N_F^{25} N_E^{28} \right) \left[\frac{\gamma(t) - (N_E^U N_D^{25} / N_D^U N_E^{25})}{(N_E^U N_D^{28} / N_D^U N_E^{28}) - \gamma(t)} \right], \quad (34)$$

where the subscript F refers to the fuel, and the superscript U refers to total uranium atom density. The activity ratio, $\gamma(t)$, should be weight-normalized to the mass of uranium in the foil.

2. Counting Technique

The quantity $\gamma(t)$ was measured by irradiating enriched uranium foils and depleted foils in flux-symmetric positions in fuel pins. The foils were 0.127 mm thick and the same diameter as the fuel pellets in the lattice being measured. Enriched foils of 2.97-, 3.23-, and 93-w/o U^{235} and depleted foils of 0.21-w/o and 6-ppm U^{235} were used. The 93-w/o U^{235} foils were not uranium metal but an alloy of 17.5-w/o uranium-aluminum.

After irradiation, the foils were assayed for fission-product activity with a NaI scintillation counter biased to reject pulses due to gammas with energies below 1.2 MeV. This bias level was used to eliminate pulses due to bremsstrahlung from the U^{239} beta rays and to allow the use of the $P(t)$ function as measured by Futch,²⁵ since he had also used a bias of 1.2 MeV. In the Hi-C cores, counting began about 40 to 50 min after shutdown and continued until 140 min after the end of irradiation since Futch's $P(t)$ determination was only valid between 30 and 140 min after irradiation. Most irradiations were of a 30- to 45-min duration.

In the BORAX-V cores, the time-dependent $P(t)$ factor determined by Klein *et al.*²⁴ was used. However, especially for the square lattice determination of $\gamma(t)$ (Core 2), the foils were counted much longer after the end of irradiation than the range of experimental values for $P(t)$. Considerable extrapolation of the function was required in order to convert $\gamma(t)$ to δ^{28} . The BORAX-V results are therefore under some suspicion because of this extrapolation of $P(t)$.

The choice of Futch's $P(t)$ determination for the Hi-C data was not entirely arbitrary. Grob *et al.*²² compared three measurements of $P(t)$: a then recent measurement at the Westinghouse Reactor Evaluation Center, the measurement of Futch, and the measurement of Klein *et al.*²⁴ With the exception of short times (less than about 50 min) after shutdown, the three measurements agreed within the experimental error. Futch's results had the lowest overall error (2%) and were determined over a longer shutdown time than the other two.

Unirradiated foils were counted with the irradiated foils for a background correction. Each irradiated foil was counted (usually a 1-min count) eight to ten times during the first $2\frac{1}{2}$ hr after shutdown, and the activity plotted as a function of time. The $\gamma(t)$ ratios were then taken from these curves at several different times. For each set of enriched and depleted foils, δ^{28} was computed at several times after shutdown. In each case, δ^{28} was found to be time-independent, and the average of the several values was reported.

3. Data and Results

The measurement for δ^{28} was performed in Cores 1, 2, 4, 8, 12, 14, 19, 20, and 21. The results are summarized in Table XXI. In

TABLE XXI. Results of δ^{28} Measurements in Hi-C and BORAX-V Cores

Core No.	Method	Foil Enrichment, w/o ^a		Radius, cm	δ^{28}	Average Core δ^{28}
		Depleted	Enriched			
1	Counting	0.21	3.23	2.6	0.065 (1) ^b	0.065 $\pm$ 0.006
2	Counting	0.21	3.23	3.8	0.119 (1)	0.119 $\pm$ 0.010
4	Counting	0.21	2.97	5.4	0.0742 (12)	0.0742 $\pm$ 0.0011
8	Counting	0.21	3.23	0	0.0981 (2)	0.1011 $\pm$ 0.0004
				5.0	0.0996 (4)	
				9.9	0.1026 (3)	
				14.9	0.1023 (3)	
				19.8	0.1017 (3)	
				24.8	0.1019 (2)	
12	Counting	0.21	3.23	4.0	0.1237 (2)	0.1160 $\pm$ 0.0017
				7.6	0.1138 (2)	
				8.1	0.1146 (2)	
				10.2	0.1120 (2)	
14	Counting	0.21	2.97	4.7	0.1495 (7)	0.1539 $\pm$ 0.0039
	Counting	0.21	3.23		0.1673 (4)	
	Counting	6 ppm	93.0		0.1565 (4)	
	Radiochemical	6 ppm	93.0		0.1466 (4)	
19	Counting	0.21	3.23	4.5	0.1852 (4)	0.1877 $\pm$ 0.0020
				6.8	0.1902 (4)	
20	Counting	0.21	3.23	4.5	0.1889 (2)	0.1889 $\pm$ 0.0131
21	Counting	0.21	2.97	3.8	0.2576 (4)	0.2622 $\pm$ 0.0036
	Radiochemical	0.21	2.97		0.2566 (2)	
	Counting	0.21	3.23		0.2832 (2)	
	Counting	0.21	93.0		0.2808 (2)	
	Radiochemical	0.21	93.0		0.2607 (2)	
	Counting	6 ppm	2.97		0.2486 (2)	
	Radiochemical	6 ppm	2.97		0.2516 (2)	
	Counting	6 ppm	93.0		0.2666 (2)	
	Radiochemical	6 ppm	93.0		0.2590 (2)	

^aExcept where otherwise noted.

^bNumbers in parentheses denote number of independent determinations.

Cores 8 and 12, data were taken at several radial positions in order to detect any noticeable trend of δ^{28} with position of measurement. Core 8 was a uniform stainless steel-clad core in which data were taken out to within 4 cm of the core boundary. The data from this core indicate that δ^{28} is independent of the position of the measurement in the region extending from the core center out to within a few centimeters of the core-reflector interface.

In Cores 14 and 21, some of the sets of foils underwent radiochemical fission analysis after counting. In this technique, the foils are dissolved and the Mo^{99} separated and counted.¹⁹ Equation (34) is easily used to obtain δ^{28} because $\gamma(t)$ is time-independent and $P(t) = 1$ since only Mo^{99} is counted and its fission yield is relatively independent of the atom fissioned (U^{235} or U^{238}). As seen in Table XXI, no systematic discrepancy exists between counting and radiochemical results. The counting rates from the separated Mo^{99} were only slightly above background and in some cases were too low to be used. Because of this, the results from foil counting are considered superior to those from the radiochemical separation. The general agreement between the two methods indicates that appreciable systematic errors in either method are unlikely. In particular, the agreement shows that $P(t)$, the time-dependent factor that is used in the counting method, corrects for the difference in gamma activity between the fission products of U^{235} and U^{238} and is reasonably accurate.

Fig. 25. U^{238} -to- U^{235} Fission Ratio in Hi-C and BORAX-V Cores

The results for the BORAX-V and Hi-C cores are plotted against H-to- U^{238} atom ratio in Fig. 25.

D. Thermal Disadvantage Factor

1. General Method

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The thermal disadvantage factor, ζ , obtained in these experiments is defined as the ratio of the average subcadmium U^{235} fission-product activity per unit weight of a highly enriched foil in the moderator to that of a highly enriched foil in the fuel. Since the thermal-energy fission cross section is nearly $1/v$, the results obtained are essentially neutron number-density disadvantage factors. The so-called integral technique requires the measurement of the U^{235} fission rate in representative portions of the fuel and moderator volumes of a unit cell as well as the U^{235} fission cadmium ratios in both fuel and moderator. The thermal disadvantage factor, ζ , is then computed from

$$\zeta = \frac{A_m}{A_f} \left(\frac{C_m - 1}{C_m} \right) \left(\frac{C_f}{C_f - 1} \right), \quad (35)$$

where A represents the U^{235} fission-product activity per unit weight in a bare foil, C is the U^{235} fission-cadmium ratio, and the subscripts m and f denote moderator and fuel, respectively.

The foil material used was a uranium-aluminum alloy of 17.5-w/o uranium, which was enriched to 93-w/o U^{235} . The fission-product activity of the irradiated foils was assayed by gamma counting in a scintillation counter biased to reject pulses from gammas whose energies were below 411 keV. In each lattice, foils were irradiated in six to eight different positions, all equidistant from the center of the core. In two of these positions, bundles of three fuel pins in the triangular lattices and four fuel pins in the square lattice could be removed and fitted with foils for the moderator region.

The fuel foils were the same diameter as the fuel and were irradiated between fuel pellets in the slotted fuel pins. The cadmium-covered irradiation was obtained by placing 0.051-cm-thick cadmium discs between the foil and the fuel pellets, and placing a 0.051-cm-thick cadmium sleeve over the fuel pin. Three lengths of cadmium sleeve were used: 0.635, 0.952, and 1.27 cm.

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Fig. 26. Foil Placement in Fuel for Measurement of U^{235} Fission Cadmium Ratio and Thermal Disadvantage Factor

Uranium oxide fuel spacers were not used between the foil and cadmium discs as was done in the irradiation geometry for the U^{238} capture cadmium ratio. There, they were necessary to prevent the streaming of epithermal neutrons through the edges of the cadmium covers into a strong resonance absorber. The inclusion of the oxide fuel spacers does introduce some moderation of epithermal neutrons, and for the U^{235} fission-cadmium ratio will decrease the observed cadmium ratio. Those foils in contact with the fuel pellets were covered with 0.025-mm-thick aluminum foil as a protection against fission-product and uranium contamination. The foils in the cadmium-covered irradiations were not covered with the aluminum foil.

The moderator foils for the bare irradiations in the Hi-C cores were shaped to fit either the roughly triangular moderator area enclosed by three fuel pins in the triangular lattices or the cross-like moderator area enclosed by four fuel pins in the square-pitch lattice. The foils were held in position between 0.318-cm-thick Lucite pieces, which were shaped to match the foils. These pieces were taped to the bundles of three or four pins and prevented the bare moderator foil from slipping out of position during irradiation or pin placement. The typical arrangement of moderator foil placement (shown in the 1.166-cm triangular Hi-C core) in one of the fuel bundles is shown in Fig. 27. The cadmium-covered moderator foils

did not have to be matched in shape to the cross section of the moderator as the variation in epicadmium flux over the region is small.²² Only in the

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Fig. 27. Foil Placement in Moderator for Measurement of U^{235} Fission Cadmium Ratio and Thermal Disadvantage Factor

looser two Hi-C lattices was it possible to have shaped cadmium-covered moderator foils. In all of the other Hi-C cores, small foil strips, placed in 0.051-cm-thick cadmium tubing and flattened to fit in the moderator volume, were used. Various strip sizes and weights of cadmium tubing were used.

In the BORAX-V cores, the fuel foils were irradiated in the same arrangement as in the Hi-C lattices. The moderator foils were constructed to sample one-fourth of the moderator unit cell in the 1.27-cm square BORAX-V lattice. Both the bare foil and the shaped cadmium-covered foil were taped directly to a fuel pin for irradiation. For the 1.27-cm triangular BORAX-V lattice, the moderator foils were hexagonal in shape with a hole in the center to fit the outside diameter of the fuel pin. Both bare and cadmium-covered foils were taped directly to a fuel pin for irradiation.

Before irradiation, all foils were intercalibrated by counting the natural gamma activity. Intercalibration on a mass basis gave identical results.

The relative activities of the foils after irradiation were determined using the concurrent normalization feature of the RE-202 code.¹¹ Each foil was counted several times.

2. Data Corrections

It was necessary to apply two corrections to the foil data beyond normal counting corrections.

First, a correction for thermal self-shielding was necessary after an auxiliary experiment was performed in the 1.27-cm triangular Hi-C core to determine if the U^{235} fission cadmium ratio was dependent upon foil thickness. The U^{235} fission cadmium ratio in the fuel was measured in this core with five thicknesses of foils: 0.051, 0.127, 0.254, 0.381, and 0.660 mm. The processed data revealed that the cadmium ratio increased with decreasing foil thickness, an effect indicative of thermal self-shielding. The bare foil data were then corrected for thermal self-shielding by standard techniques.²⁰ The cross sections used for this correction were those obtained from the output of Nelkin kernel multigroup THERMOS problems. The correction was not large; for a 0.25-mm-thick foil,

the cadmium ratio minus one is increased by about 4%. The calculated flux perturbation outside typical foils was nearly negligible (i.e., ~0.5% or less) compared to the self-shielding within the foils. No resonance self-shielding was considered; the simple thermal self-shielding correction produced cadmium ratios that were independent of foil thickness. This correction was applied to the U^{235} fission cadmium ratio data in both the fuel and moderator.

Second, these corrected cadmium ratios were examined for a dependence upon the mass of cadmium used in the cadmium-covered irradiation. This was done in each Hi-C core. Figure 28 shows the cadmium ratios obtained in the fuel and moderator plotted against the mass of cadmium used in the measurement for a typical core--the 1.27-cm triangular Hi-C lattice. Four reactor runs were made; in each one, three different masses of cadmium were used in the cadmium-covered fuel irradiation, and two different masses of cadmium were used in the cadmium-covered moderator irradiation. The three different masses of cadmium for the cadmium-covered fuel irradiations were obtained by using three lengths of cadmium

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Fig. 28. Effect of Cadmium Mass in Measurement on Fuel and Moderator U^{235} Fission Cadmium Ratios in Typical Hi-C Core

sleeve: 0.635, 0.952, and 1.270 cm. The mass of the cadmium discs was the same in each irradiation. The lines shown in Fig. 28 are least-squares fits to the data points. The trend in the fuel cadmium ratio is as might be expected; the greater the mass of cadmium in the measurement, the greater the cadmium ratio. This suggests that the cadmium mass may depress the fission source in the neighborhood of the foil and thus decrease its epithermal cadmium activation. Because of the constant contribution of the cadmium discs covering the foils, measurements could not be made with a total weight of cadmium less than about 1.5 g. A value of the cadmium ratio corrected for the mass of cadmium in the measurement was obtained by extrapolating the straight-line fit through the data points to zero cadmium mass. The correction is not large; average fuel cadmium ratios were lowered about 3 to 4%.

Various masses of cadmium for the cadmium-covered moderator irradiations were obtained by using several sizes of foil strips. As seen in Fig. 28, the effect of the cadmium mass on the fission cadmium ratio in the moderator was opposite to that in the fuel. In the very tight lattices considered here, the placing of a cadmium-covered foil in the moderator volume displaces most of the moderator, thus decreasing

moderation and increasing epicadmium activity in the moderator volume as more moderator is removed. In the very tight lattices, this effect appears to be stronger than the depression of the fast-neutron source in the surrounding fuel pins. In the loosest Hi-C lattice, the effect of the cadmium mass on the U^{235} fission cadmium ratio in the moderator was the same as in the fuel; the cadmium ratio increased with increasing cadmium mass. This suggests that, as expected, the effect of displacing the moderator with the cadmium-covered foil volumes is much less in the looser cores, where a much smaller fraction of the moderator volume is displaced.

In each core, the corrected value of the moderator cadmium ratio was determined by extrapolating the straight-line fit through the data points to zero cadmium mass. As found with the fuel cadmium ratios, the correction was not large; corrected values of the moderator cadmium ratio differed by 1-2% from the uncorrected values.

3. Data and Results

The data obtained in the various cores are listed in Table XXII. These data are uncorrected for self-shielding and cadmium mass. Results obtained from these data are tabulated in Table XXIII.

TABLE XXII. Data from Thermal Disadvantage Factor and U^{235} Fission Cadmium Ratio Measurements in Hi-C and BORAX-V Cores

Date	Run No.	Foil Thickness, mm	Fuel Cd Ratio	Cd Mass, g	Moderator Cd Ratio	Cd Mass, g	Ratio of Bare Moderator to Bare Fuel ^a
<u>1.27-cm Square BORAX-V Core 1</u>							
8-1-61	21-85	0.66	4.94	1.91	5.04	0.90	1.159
<u>1.27-cm Triangular BORAX-V Core 2</u>							
12-19-61	71-81	0.66	3.90	1.91	3.44	0.75	1.197
<u>1.349-cm Square Hi-C Core 4</u>							
12-16-65	418-10	0.254	6.23	1.58	6.64	0.67	1.108 ^b
		0.254	6.27	2.06	6.78	1.45	1.121 ^b
		0.254	6.50	2.55	-	-	-
12-17-65	418-11	0.254	6.62	1.58	6.05	0.62	1.145
		0.254	6.18	2.06	-	-	1.125
		0.254	6.37	2.55	-	-	-
12-20-65	418-12	0.254	6.17	1.58	6.18	0.67	1.143 ^c
		0.254	6.29	2.06	6.76	1.45	1.137 ^c
		0.254	6.17	2.55	-	-	-
12-21-65	418-13	0.254	6.08	1.58	6.34	0.41	1.138 ^d
		0.254	6.23	2.06	6.21	1.40	1.151 ^d
		0.254	6.24	2.55	-	-	-
12-22-65	418-14	0.127	6.22	1.58	6.62	0.45	1.158
		0.127	6.37	2.06	6.91	1.40	1.153
		0.127	6.44	2.55	-	-	-
12-27-65	418-15	0.127	6.25	1.58	6.96	0.67	1.170 ^c
		0.127	6.47	2.06	7.06	1.45	1.153 ^c
		0.127	6.45	2.55	-	-	-

TABLE XXII (Contd.)

Date	Run No.	Foil Thickness, mm	Fuel Cd Ratio	Cd Mass, g	Moderator Cd Ratio	Cd Mass, g	Ratio of Bare Moderator to Bare Fuel ^a
1.349-cm Square Hi-C Core 4 (Contd.)							
12-28-65	418-16	0.254	6.24	1.58	6.32	0.58	1.146 ^e
		0.254	6.22	2.06	-	-	1.159 ^e
		0.254	6.27	2.55	-	-	-
12-29-65	418-17	0.127	6.14	1.58	6.86	0.65	1.182 ^d
		0.127	6.28	2.06	7.12	1.45	1.182 ^d
		0.127	6.17	2.55	-	-	-
1-3-66	418-18	0.127	6.25	1.58	7.05	0.67	1.190 ^e
		0.127	6.35	2.06	7.09	1.40	1.187 ^e
		0.127	6.30	2.55	-	-	-
1-4-66	418-19	0.127	6.20	1.58	6.81	0.67	1.148 ^b
		0.127	6.32	2.06	6.73	1.45	1.148 ^b
		0.127	6.34	2.55	-	-	-
1.24-cm Square Hi-C Core 9							
5-12-64	383-3	0.381	4.61	1.58	4.82	0.66	1.114
		0.381	4.50	2.06	4.75	0.73	1.103
		0.381	4.69	2.55	4.72	0.90	-
5-13-64	383-4	0.254	4.59	1.58	4.94	0.66	1.122
		0.254	4.69	2.06	5.02	0.73	1.098
		0.254	4.82	2.55	5.79	0.76	-
5-14-64	383-5	0.254	4.73	1.58	5.02	0.66	1.104
		0.254	4.88	2.06	5.01	0.69	1.089
		0.254	4.94	2.55	4.91	0.84	-
5-15-64	383-6	0.127	4.65	1.58	5.16	0.62	1.164
		0.127	4.81	2.06	5.10	0.66	1.150
		0.127	4.80	2.55	4.91	0.73	-
1.27-cm Triangular Hi-C Core 12							
7-10-63	283-4	0.66	3.90	2.55	3.69	0.6	1.012
	283-5	0.66	3.91	2.55	3.77	0.6	1.065
8-1-63	283-13	0.381	3.99	2.55	3.82	0.6	1.062
8-2-63	283-15	0.254	4.06	2.55	3.85	0.6	1.111
8-8-63	285-1	0.127	4.05	2.55	3.93	0.6	1.058
8-9-63	285-2	0.051	4.23	2.55	4.01	0.6	1.123
8-23-63	289-1	0.254	3.78	2.55	3.77	0.6	1.067
8-26-63	289-2	0.127	4.06	2.55	4.04	0.6	1.104
1-10-64	343-5	0.381	3.77	1.58	3.88	0.39	1.122
		0.381	3.80	2.06	3.97	0.72	1.026
		0.381	3.87	2.55	-	-	-
1-13-64	343-6	0.381	3.81	1.58	3.96	0.45	1.038
		0.381	3.89	2.06	3.92	0.69	1.107
		0.381	3.89	2.55	-	-	-
1-14-64	343-7	0.254	3.73	1.58	4.00	0.41	1.109
		0.254	3.81	2.06	3.92	0.81	1.077
		0.254	3.79	2.55	-	-	-

TABLE XXII (Contd.)

Date	Run No.	Foil Thickness, mm	Fuel Cd Ratio	Cd Mass, g	Moderator Cd Ratio	Cd Mass, g	Ratio of Bare Moderator to Bare Fuel ^a
1.27-cm Triangular Hi-C Core 12 (Contd.)							
1-15-64	343-8	0.381	3.76	1.58	3.89	0.70	1.101
		0.381	3.73	2.06	-	-	1.022
		0.381	3.84	2.55	-	-	-
1.166-cm Triangular Hi-C Core 14							
3-18-64	372-3	0.381	2.54	1.58	2.57	0.28	1.042
		0.381	2.54	2.06	2.59	0.38	1.044
		0.381	2.58	2.55	-	-	-
3-19-64	372-4	0.381	2.59	1.58	2.63	0.31	1.050
		0.381	2.58	2.06	2.62	0.55	1.052
		0.381	2.62	2.55	-	-	-
3-20-64	372-5	0.254	2.58	1.58	2.66	0.33	1.044
		0.254	2.60	2.06	2.65	0.50	1.058
		0.254	2.58	2.55	-	-	-
3-23-64	372-6	0.127	2.56	1.58	2.67	0.36	1.063
		0.127	2.63	2.06	2.68	0.51	1.078
		0.127	2.64	2.55	-	-	-
1.127-cm Triangular Hi-C Core 19							
12-9-63	342-13	0.127	2.08	1.58	2.31	0.54	1.040
		0.127	2.18	2.06	2.32	0.54	1.095
		0.127	2.18	2.55	-	-	-
12-10-63	342-14	0.254	2.21	1.58	2.26	0.59	1.029
		0.254	2.16	2.06	2.22	0.88	1.056
		0.254	2.28	2.55	-	-	-
12-11-63	342-18	0.127	2.24	1.58	2.28	0.60	1.061
		0.127	2.19	2.06	2.26	0.96	1.039
		0.127	2.17	2.55	-	-	-
12-17-63	342-19	0.254	2.20	1.58	2.23	0.35	1.046
		0.254	2.24	2.06	2.27	0.95	1.034
		0.254	2.20	2.55	-	-	-
1.069-cm Triangular Hi-C Core 21							
7-27-64	391-18	0.254	1.58 ^f	0	1.61	0.24	1.108
		-	-	-	1.62	0.39	0.993
7-28-64	391-19	0.254	1.57 ^f	0	1.66	0.29	0.970
		-	-	-	1.75	0.41	1.180
7-29-64	391-20	0.127	1.58 ^f	0	1.73	0.27	1.219
		-	-	-	1.65	0.29	1.023

^aBare moderator foils were supported by 0.318-cm Lucite pieces unless otherwise noted.

^bBare moderator foils were supported by 0.635-cm Lucite pieces.

^cBare moderator foils were supported by 0.159-cm Lucite pieces.

^dBare moderator foils were supported by 0.079-cm Lucite pieces.

^eBare moderator foils were not supported by Lucite.

^fThese values were obtained by the thermal-activation technique.

TABLE XXIII. Thermal Disadvantage Factors and U^{235} Fission Cadmium Ratios in Hi-C and BORAX-V Cores

Core No.	1	2	4	9	12	14	19	21
H-to- U^{238} atom ratio	4.65	3.53	4.16	2.92	2.29	1.33	0.98	0.50
Fission cadmium ratio in fuel, C_f	5.11 ± 0.20	3.98 ± 0.16	6.13 ± 0.20	4.49 ± 0.11	3.75 ± 0.08	2.55 ± 0.06	2.15 ± 0.05	1.59 ± 0.03
Epicadmium-to-subcadmium U^{235} fission ratio, p_f^{25}	0.243 ± 0.012	0.336 ± 0.018	0.195 ± 0.008	0.286 ± 0.009	0.364 ± 0.011	0.645 ± 0.025	0.870 ± 0.038	1.695 ± 0.086
Fission cadmium ratio in moderator, C_m	5.54 ± 0.22	3.72 ± 0.15	6.55 ± 0.20	5.09 ± 0.15	4.14 ± 0.10	2.72 ± 0.08	2.33 ± 0.08	1.69 ± 0.09
Bare moderator-to-bare fuel fission ratio, A_m/A_f	1.159 ± 0.049	1.197 ± 0.051	1.152 ± 0.006	1.118 ± 0.009	1.075 ± 0.009	1.054 ± 0.004	1.050 ± 0.008	1.081 ± 0.041
Thermal disadvantage factor, ζ	1.180 ± 0.052	1.170 ± 0.055	1.166 ± 0.011	1.156 ± 0.014	1.112 ± 0.015	1.096 ± 0.026	1.120 ± 0.038	1.189 ± 0.109

Results obtained from the BORAX-V data have been assigned large error limits. As is seen in Table XXII, only one independent determination was made in each of the two BORAX-V lattices. In addition, foil placement was not as precise as in the later Hi-C cores, and 0.371-mm cadmium discs were used as covers on the cadmium-covered fuel foil irradiations rather than the 0.508-mm-thick cadmium that was used for the sleeves. The correction to zero cadmium mass was of necessity estimated from the Hi-C data. A reduction of 4% in the fuel cadmium ratio and an increase of 2% in the moderator cadmium ratio were assumed.

Several exceptions to the data corrections as previously outlined should be noted. In both the 1.24-cm square and 1.069-cm triangular Hi-C cores (Cores 7 and 19), the plots of moderator U^{235} fission cadmium ratio versus mass of cadmium used in the measurement were bunched to prevent a reasonable least-squares fitting. For these cores, this correction was estimated from the other Hi-C cores; the average moderator cadmium ratio was unchanged in the 1.24-cm square lattice and increased 3/4% in the 1.069-cm triangular core. It was not physically possible to place a 0.051-cm-thick cadmium sleeve about a fuel pin in the 1.069-cm triangular Hi-C core. Here, it was necessary to employ the thermal-activation technique²³ to obtain the fuel U^{235} fission cadmium ratio. Dysprosium was used as the intermediate material since an accurate value of the fuel dysprosium cadmium ratio was obtained with nonrepresentative fuel foils irradiated in aluminum and cadmium pillboxes. The value of the U^{235} fission cadmium ratio obtained by this method need not be corrected for cadmium mass since no cadmium is used on the U^{235} fission foils and the dysprosium cadmium ratio is rather large.

The results of Table XXIII are given in Figs. 29 and 30 as plots of epicadmium-to-subcadmium U^{235} fission ratios and thermal disadvantage factors. Of interest is the indication of the data that the disadvantage factor does not decrease monotonically with decreasing H-to- U^{238} atom ratio, but possesses a minimum and will increase as the lattice pitch decreases beyond a certain value. Although this behavior is difficult to understand from the physical point of view, calculations of the disadvantage factor for

small reactor cells have established a theoretical basis for the existence of the minimum.²⁶⁻²⁸ These experimental results have previously been compared⁶ with some current calculations.

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Fig. 29. Epicadmium-to-Subcadmium U^{235} Fission Ratio in Hi-C and BORAX-V Cores

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Fig. 30. Thermal Disadvantage Factors in Hi-C Cores

The results of an auxiliary experiment performed in the 1.349-cm square Hi-C lattice are given in Table XXIV and Fig. 31. In this experiment, the disadvantage factor was measured using several thicknesses of Lucite foil supports. In the other Hi-C cores, the bare moderator foils were supported by 0.318-cm-thick Lucite pieces. Some investigators have found that the inclusion of these supports perturbs the disadvantage-factor measurements;^{29,30} however, others have noted no effect.^{27,30} In the 1.349-cm square Hi-C lattice, Lucite supports as thick as 0.635 cm were used, and several measurements were made using no Lucite spacers. These runs may not have used any Lucite, but large amounts of waterproof tape were necessary to insure that the foils were not displaced during irradiation. Two bare moderator foils were irradiated in each of two reactor runs with each thickness of Lucite. Figure 31 illustrates that the disadvantage factor does seem to be depressed by the presence of the Lucite. For

TABLE XXIV. Thermal Disadvantage Factor in 1.349-cm Square Hi-C Core

Lucite Foil Support Thickness, cm	Ratio of Bare Moderator to Bare Fuel	ζ_{25}
0	1.1706 ± 0.0107	1.185 ± 0.015
0.079	1.1632 ± 0.0110	1.178 ± 0.015
0.159	1.1509 ± 0.0074	1.165 ± 0.013
0.318	1.1452 ± 0.0071	1.160 ± 0.012
0.635	1.1311 ± 0.0107	1.145 ± 0.015

the 0.318-cm Lucite pieces used in these experiments, the disadvantage factor is reduced by about 2%. In addition, there is some evidence that

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Fig. 31. Thermal Disadvantage Factor in 1.349-cm Square Hi-C Core

this effect increases as the H-to- U^{238} atom ratio of the lattice decreases²⁹ and could make a larger error in the very tight lattices. The thermal disadvantage factors given here may thus be low by as much as 2 or 3%.

E. Cadmium Ratios

1. Method of Measurement

The cadmium ratios of dysprosium, lutetium, indium, gold, manganese, copper, U^{235} fission, and U^{238} capture have been obtained for most of

the cores assembled. Because of the large spectral variations in the loadings in which every nth fuel pin is removed, cadmium-ratio measurements were confined to the uniform lattices. In the zoned lattices, the measurements were made well inside the zone boundary.

Measurements were made both within the fuel pins and in the moderator. Cadmium ratios in the moderator were not measured in the tighter lattices because of the difficulty of foil placement.

In the BORAX-V cores, both bare and cadmium-covered foils were the same diameter as the fuel pellets and were irradiated between fuel pellets in special slotted fuel pins. The bare foils were wrapped in 0.025-mm aluminum foil, the cadmium-covered foils were sandwiched between 0.038-cm-thick cadmium discs, and 1.27-cm-long 0.051-cm-thick cadmium sleeves were placed around the fuel pin at the foil location. U^{238} capture and U^{235} fission in both BORAX-V and Hi-C cores were irradiated as described in Sections B and D of this chapter. Moderator cadmium ratios in the BORAX-V cores were made with foils taped to a Lucite strip.

In the Hi-C cores, fuel foils were 0.818 cm in diameter (slightly smaller than the fuel-pellet diameter) and were placed in aluminum and cadmium pillboxes for irradiation as described in Section B of Chapter III. For moderator measurements, the foils in their covers were taped to thin strips of steel or plastic, and each strip was inserted horizontally (with the foils vertical) between two rows of fuel pins.

Foil counting, intercalibration, and data reduction for measurements in both Hi-C and BORAX-V cores were done as previously described in Section C of Chapter III.

2. Data and Results

The cadmium ratios obtained are presented in Table XXV. In general, the data have been subjected only to normal counting corrections, and effects such as thermal transmission or epicadmium absorption by the cadmium covers have been ignored. The U^{235} fission and U^{238} capture data have been corrected for mass of cadmium used in the measurement and thermal self-shielding. These corrections are described in Sections B and D of this chapter.

TABLE XXV. Cadmium Ratios in Hi-C and BORAX-V Cores

Core	H-to- U^{238} Atom Ratio	Core Moderator Ratio	Fuel or Moderator	Cadmium Ratios									U^{235} fission	U^{238} capture
				Dy	Lu ¹⁷⁶	Lu ¹⁷⁵	In	Au	Mn	Cu				
1	4.65	2.287	Fuel	17.7 ± 0.7	-	-	1.470 ± 0.020	1.413 ± 0.025	3.19 ± 0.06	3.09 ± 0.07	4.94 ± 0.20	1.182 ± 0.041		
			Moderator	18.5 ± 1.4	-	-	1.558 ± 0.040	1.520 ± 0.045	3.58 ± 0.06	3.51 ± 0.07	5.04 ± 0.20	-		
2	3.53	1.758	Fuel	13.4 ± 1.0	7.4 ± 1.0	1.056 ± 0.030	1.382 ± 0.030	1.288 ± 0.030	2.61 ± 0.08	2.43 ± 0.10	3.90 ± 0.16	1.188 ± 0.083		
			Moderator	14.9 ± 1.1	9.4 ± 0.4	1.079 ± 0.025	1.412 ± 0.040	1.354 ± 0.040	2.79 ± 0.08	2.65 ± 0.15	3.44 ± 0.15	-		
3	4.15	3.332	Fuel	23.66 ± 0.80	-	-	1.61 ± 0.03	1.62 ± 0.02	4.20 ± 0.09	3.83 ± 0.04	-	-		
4	4.16	3.067	Fuel	24.58 ± 0.47	-	-	1.58 ± 0.03	1.58 ± 0.02	4.14 ± 0.04	3.65 ± 0.04	6.13 ± 0.20	1.328 ± 0.004		
			Moderator	-	-	-	-	-	-	-	6.55 ± 0.20	-		
5	3.24	2.664	Fuel	20.9 ± 0.6	9.5 ± 0.8	1.083 ± 0.020	1.551 ± 0.030	1.522 ± 0.030	3.515 ± 0.030	3.352 ± 0.050	-	-		
8 & 9	2.92	2.181	Fuel	17.4 ± 0.7	11.1 ± 1.6	1.065 ± 0.015	1.469 ± 0.008	1.435 ± 0.008	3.00 ± 0.03	2.750 ± 0.020	4.49 ± 0.11	1.243 ± 0.002		
			Moderator	-	-	-	1.483 ± 0.020	1.442 ± 0.008	3.19 ± 0.05	3.02 ± 0.03	5.09 ± 0.15	-		
10	2.91	2.409	Fuel	17.2 ± 0.5	-	-	1.482 ± 0.015	1.455 ± 0.010	3.118 ± 0.030	2.950 ± 0.030	-	-		
			Moderator	18.9 ± 0.5	-	-	1.518 ± 0.020	1.503 ± 0.015	3.367 ± 0.020	3.160 ± 0.020	-	-		
11	2.29	1.916	Fuel	14.1 ± 0.8	-	-	1.398 ± 0.015	1.361 ± 0.015	2.58 ± 0.06	2.49 ± 0.05	-	-		
12	2.29	1.736	Fuel	13.8 ± 1.5	-	-	1.373 ± 0.007	1.346 ± 0.015	2.63 ± 0.10	2.40 ± 0.04	3.75 ± 0.08	1.185 ± 0.003		
			Moderator	-	-	-	-	-	-	-	4.08 ± 0.10	-		
13	1.32	1.132	Fuel	9.18 ± 0.30	-	-	1.260 ± 0.008	1.221 ± 0.008	1.863 ± 0.015	1.801 ± 0.030	-	-		
14	1.33	1.024	Fuel	8.73 ± 0.12	-	-	1.234 ± 0.005	1.201 ± 0.005	1.80 ± 0.02	1.67 ± 0.04	2.55 ± 0.06	1.100 ± 0.001		
			Moderator	-	-	-	-	-	-	-	2.72 ± 0.08	-		
19	0.98	0.773	Fuel	7.6 ± 0.3	-	-	-	1.152 ± 0.015	1.587 ± 0.020	1.500 ± 0.030	2.15 ± 0.05	1.069 ± 0.002		
			Moderator	-	-	-	-	-	-	-	2.30 ± 0.08	-		
20	0.98	0.855	Fuel	7.1 ± 0.3	4.6 ± 0.7	1.027 ± 0.015	1.204 ± 0.020	1.177 ± 0.010	1.574 ± 0.030	1.526 ± 0.030	-	-		
21	0.50	0.406	Fuel	4.31 ± 0.20	-	-	1.135 ± 0.008	1.099 ± 0.008	1.322 ± 0.030	1.210 ± 0.030	1.59 ± 0.03	1.035 ± 0.001		
			Moderator	-	-	-	-	-	-	-	1.69 ± 0.09	-		

In the BORAX-V fuel, the bare foils were wrapped in 0.025-mm aluminum; in the Hi-C fuel, the bare foils were irradiated in pillboxes as shown in Fig. 5. The Hi-C arrangement produces a larger separation between fuel pellets and thus gives higher values for the cadmium ratios resulting from thermal-neutron streaming. The difference in the two irradiation techniques was experimentally determined to be about 5% in the value of the cadmium ratio minus one. The increased corrected values for the BORAX-V cores are those listed in Table XXV.

When the data were reduced with the RE-202 code, the weight corrections for indium and gold were made with consideration for the fact that much of the epicadmium activation is strongly self-shielded. However, it was necessary to reduce the cadmium ratio for foils of thickness t to what would have been observed for some common thickness, t_0 . Since the foils were prepared with a precision punch-and-die set, it was assumed that any variations in weight among foils resulted from thickness variations. The common thickness, t_0 , was chosen to be 21.9 mg/cm² for indium and

53.9 mg/cm² for gold, corresponding to weights of 11.50 and 28.31 mg, respectively, for the Hi-C foils. The correction to gold and indium cadmium ratio was made by

$$CR(t_0) - 1 = [CR(t) - 1] \left[1 - 0.44 \frac{t - t_0}{t} \right], \quad (36)$$

where 0.44 is a semiempirical constant¹³ appropriate for both gold and indium foils of about 0.03-mm thickness.

With the exception of the values reported for Core 5, all cadmium ratios were based on measurements using more than one pair of foils. The errors given are in general the standard deviation of the mean obtained from a number of foil pairs.

3. Linear Relationship

When the cadmium ratios given in Table XXV from the Hi-C cores were plotted as a function of H-to-U²³⁸ atom ratio, a linear relationship was noted. Since the H-to-U²³⁸ atom ratio is very nearly a measure of the moderating ratio, it was felt that data from BORAX-V cores could be included in these plots if the core moderating ratio instead of the H-to-U²³⁸ atom ratio were chosen as the abscissa. The moderating ratios were calculated assuming homogenized cores from

$$MR = \frac{[N\xi\sigma_S]_H + [N\xi\sigma_S]_O}{\Sigma_a}, \quad (37)$$

where the N's are the atom densities in the homogenized core, ξ is the average logarithmic energy decrement, σ_S is the scattering cross section, and Σ_a is the macroscopic absorption cross section of the core. Scattering

112-7961

Fig. 32. Dysprosium Cadmium Ratio in Fuel in BORAX-V and Hi-C Cores

by oxygen in the oxide fuel was included, and the $\xi\sigma_S$'s for hydrogen and oxygen were taken as 20.4 and 0.45 b, respectively. The absorption cross sections used were those for 2200-m/sec neutrons. The moderating ratio thus computed for each core is given in Table XXV.

Figures 32 through 38 present the cadmium ratios of the various materials plotted against core moderating ratios. The data were fitted to a least-squares line; each data point was weighted by the inverse square of its relative error.

The lutecium and BORAX-V U^{238} capture data were not plotted since they were somewhat fragmentary and of low precision.

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Fig. 33. Indium Cadmium Ratio in Fuel in BORAX-V and Hi-C Cores

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Fig. 34. Gold Cadmium Ratio in Fuel in BORAX-V and Hi-C Cores

112-7963

Fig. 35. Manganese Cadmium Ratio in Fuel in BORAX-V and Hi-C Cores

112-7949

Fig. 36. Copper Cadmium Ratio in Fuel in BORAX-V and Hi-C Cores

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Fig. 37. U^{235} Fission Cadmium Ratio in Fuel in BORAX-V and Hi-C Cores

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Fig. 38. U^{238} Capture Cadmium Ratio in Hi-C Cores

The linearity of the results plotted in Figs. 32-38 is consistent with two-group theory.⁷ The utility of these plots is enhanced by their simplicity--the fact that the data, uncorrected for effects that are tedious to calculate (i.e., self-shielding and local flux depression), are empirically linear. The use of the moderating ratio instead of the more common H-to-U²³⁸ atom ratio as the abscissa makes it possible to plot data from diverse reactor cores on the same basis. The linear plots of Figs. 32-38 come from cores with two different cladding materials (aluminum and stainless steel), two fuel diameters (0.87 and 0.935 cm), and two fuel compositions (3- and 5-w/o enriched UO₂).

VI. COMPARISON WITH OTHER DATA AND CONCLUSIONS

The importance that light-water-moderated reactors have assumed in the development of civilian power reactors is reflected in the large amount of theoretical and experimental attention they have received. Most work has been done with either uranium metal or uranium oxide fuel of enrichments up to about 5 w/o in U^{235} . Recently, the experimental work in this area has been gathered into a definitive review.³² It is thus easy to compare these data with those obtained in the Hi-C experiments. The comparison will be restricted to those data obtained with oxide fuel in the range of enrichment from 2.7- to 3.7-w/o U^{235} . Four sets of experiments fall into this category. They are as follows:

1 & 2. Westinghouse Atomic Power Division. Two series of experiments have been performed as part of the work on the Yankee reactor project. One series used 2.70-w/o enriched UO_2 fuel of density 10.2 g/cm^3 . The fuel pellet diameter was 0.762 cm, and the stainless steel cladding was 0.860 cm in OD and 0.041 cm thick. Only square lattice pitches were used. A second series of experiments employed 3.70-w/o enriched, 10.40-g/cm^3 oxide fuel of 0.754-cm diameter in the same size stainless steel cladding. These data will be referred to as WAPD-2.7 and WAPD-3.7, respectively.

3. Brookhaven National Laboratory. As part of a continuing program of lattice measurements, buckling and microparameter measurements³³ have been performed with 3.01-w/o enriched 9.57-g/cm^3 oxide rods of 1.120-cm diameter in 1.267-cm-OD, 0.074-cm-thick, stainless steel cladding. Triangular lattice pitches were used. These data will be referred to as BNL-3.0.

4. Babcock & Wilcox. Buckling measurements have been performed with 3.01-w/o enriched 9.28-g/cm^3 oxide fuel of 1.125-cm diameter loaded into stainless steel cladding of 1.270-cm OD and 0.071-cm thickness. Square lattice pitches were used. These data will be referred to as B&W-3.0.

These four sets of experiments cover the range of H-to- U^{238} atom ratios of about 3 to 15. Since the Hi-C lattices extend from H-to- U^{238} atom ratios of 0.5 to 4 (the nonuniformly loaded cores extend to H-to- U^{238} of 5), the combination of Hi-C data with the WAPD, BNL, and B&W data provides a rather complete picture of slightly enriched oxide cores.

Figure 39 presents the critical buckling data from the Hi-C and the previous experiments. A theoretical study of undermoderated, slightly enriched, uranium and uranium oxide lattices has been made¹ which will help in the interpretation of the data insofar as making allowances for the slight differences in enrichment and rod diameter among the various series of experiments. For the same H-to- U^{238} atom ratio, the WAPD-3.7 data give

higher bucklings than the Hi-C stainless steel-clad data, as expected because of the higher fuel enrichment. The WAPD-2.7, BNL-3.0, and B&W-3.0 data fall rather close to one another below the Hi-C stainless steel-clad data. The WAPD-2.7 data would be expected to fall low because of lower enrichment. The BNL-3.0 and B&W-3.0 fuel had cladding of about twice the thickness of the Hi-C and WAPD-2.7 fuel, thus producing the lower bucklings.

112-7960

Fig. 39. Critical Bucklings in Slightly Enriched Uranium Oxide Cores

Figure 40 presents the modified conversion ratio for the Hi-C stainless steel cores and the WAPD-2.7 and WAPD-3.7 experiments. The data are rather limited but do indicate a reasonable progression from the looser cores to the Hi-C data.

112-7948

Fig. 40. Modified Conversion Ratio in Slightly Enriched Uranium Oxide Cores

The U^{238} -to- U^{235} fission ratios for the WAPD-2.7, WAPD-3.7, BNL-3.0, and the stainless steel-clad Hi-C core are plotted in Fig. 41. Here, as expected, the lower enrichment fuels give, in general, higher values of δ^{28} than the higher enrichment fuels. The Hi-C data seem to fit well with the other data and give a reasonable extension into the very dry core region.

Figure 42 gives the U^{238} capture cadmium ratio for the WAPD-2.7, BNL-3.0, and stainless steel-clad Hi-C cores. Here, the BNL-3.0 and Hi-C data match well. The WAPD-2.7 data give slightly higher cadmium ratios, as expected.

112-7954

Fig. 41. U^{238} -to- U^{235} Fission Ratio in Slightly Enriched Uranium Oxide Cores

112-7971

Fig. 42. U^{238} Capture Cadmium Ratio in Slightly Enriched Uranium Oxide Cores

Thermal disadvantage factors are plotted in Fig. 43. Here again, the BNL-3.0 and Hi-C stainless steel-clad data match rather well. The WAPD-2.7 data had larger errors than the other data, but still gave smaller values of the thermal disadvantage factor, as expected because of the lower enrichment.

112-7964

Fig. 43. Thermal Disadvantage Factors in Slightly Enriched Uranium Oxide Cores

a reasonable extension into the very dry lattice region from previous slightly enriched uranium oxide-water reactor data.

With assistance from the computations of Bengston and Hellens¹ and some recent Hi-C calculations,³⁴ extrapolation of the critical buckling curves of Figs. 10 and 39 indicates that the aluminum-clad Hi-C cores have zero buckling at about a 0.7 H-to- U^{238} atom ratio, while the stainless steel-clad Hi-C cores have zero buckling at about a 1.1 H-to- U^{238} atom ratio. Thus, from Fig. 22, the largest ICR to be expected from a just critical (zero buckling) Hi-C core would be about 1.15 for aluminum-clad fuel and about 0.9 for stainless steel-clad fuel. (This assumes that the value of the ICR in aluminum cores is about the same as in the stainless steel-clad cores.)

However, in practical reactor cores (those with effective multiplication constants of about 1.1), these values of the ICR would be reduced to about 0.9 and 0.7 for aluminum-clad and stainless steel-clad fuel, respectively. While these values are higher than the values of the ICR encountered in conventional light water-uranium oxide reactors, only a detailed study could reveal whether these higher conversion ratios would be worthwhile, in view of the additional difficulties associated with the control, mechanical, and heat-transfer requirements of such tightly packed lattices.

Since the ICR appears to increase dramatically at H-to- U^{238} atom ratios below 1.0, it may prove worthwhile to investigate these dry lattices as blanket zones surrounding (or surrounded by) driver cores of a more conventional design. In an arrangement such as this, the high conversion ratio of the dry lattice blanket may tend to overcome the lowered conversion ratio of the driver region, for some configurations, leading to a net increase in the conversion ratio of the system.

APPENDIX

Seasonal Variation of Moderator Temperature

The cell that contained the ZPR-7 facility was not kept at constant temperature, but was subject to seasonal fluctuations in temperature. The light-water moderator, however, was stored in a dump tank below floor level and was pumped into the reactor tank only during operation. During each reactor operation, the moderator temperature was measured with thermocouples at several locations in the tank. The thermocouple at the midplane of the fuel zone on a control-blade guide was representative of the bulk moderator temperature and is used here. The range of temperatures observed during each month of the Hi-C program is shown in Fig. 44. Data taken during temperature coefficient measurements are not shown; these data are given to reflect only the normal seasonal changes. The range of moderator temperature for the entire period of operation was 13.1°C (from 16.4 to 29.5°C).

112-7976

Fig. 44. Seasonal Variation of Moderator Temperature in ZPR-7 Facility

The H-to- U^{238} atom ratios were computed for water at 20°C . The variations in water temperature change these ratios very little. For example, the H-to- U^{238} atom ratio in the 1.349-cm square Hi-C core is reduced less than 0.3% if the water temperature is raised from 20 to 30°C , neglecting changes in fuel and cladding density.

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The ZPR-7 facility was operated and maintained by J. W. Armstrong, K. Freeze, T. W. Johnson, F. Kodrick, and R. A. Schultz. The counting rooms were maintained during the program by K. Freeze, R. E. Grajeck, and T. Jensen. L. R. Dates assisted in preparing the system for the critical experiments, and theoretical assistance for the program was provided by E. M. Pennington. R. J. Armani, assisted by J. G. Hamilton and F. Lyon, provided the radiochemical results. The ATSR crew of G. Lowe and R. Rutkowski, under the supervision of J. Haugsnes and R. C. Doerner, helped with foil intercalibrations and simultaneous runs with ZPR-7. I. K. Olson handled significant amounts of necessary data processing, W. Knapp participated in some of the critical loadings and blade calibrations, and J. M. Christenson contributed to the solution of data-processing problems and the coincidence counting technique.

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