**TUBULAR FUEL ELEMENT
WITH INTEGRAL TARGET
PCTR EXPERIMENT**

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JUNE 1, 1968

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ABSTRACT

Two tubular fuel element with integral target lattices have been investigated in the Physical Constants Test Reactor (PCTR). Multiplication constants, neutron utilization ratios, fast fission factors, and initial conversion ratios have been inferred from null reactivity experiments and measurements of spatial distributions of neutron reaction rates. Both wet and dry experiments were carried out in a three-by-three array of lattice cells 37.5" long placed in the cavity of the PCTR.

Inferred lattice characteristics and measured spatial distributions are in reasonably good agreement with transport calculations performed with Program S and THERMOS. This report includes primary descriptions of the experiments and results, and some significant comparisons with theory.

INTRODUCTION AND DATA SUMMARY

A number of neutronic characteristics of a graphite lattice, loaded with two different tubular fuel elements with integral targets, have been inferred or measured directly in experiments conducted in Battelle-Northwest's Physical Constants Test Reactor (PCTR). Results of these experiments are in reasonable agreement with few group P-3 diffusion and transport calculations.

Experimental results include values of multiplication constants, with and without primary coolant; associated $1/v$ utilizations; fast fission factor and conversion ratios for lattices with primary coolant; and spatial distributions of neutron reaction rates in copper-63. Descriptions are presented of the experiments, some of the techniques (most are reported elsewhere, as referenced), the results, and a brief experiment-theory comparison.

The two integral target fuel elements consisted of a common annular fuel rod of uranium enriched to 2.1 wt% U-235 and different $\text{Li}_2\text{O} \cdot \text{Al}_2\text{O}_3$ target rods having 0.297 (base) and 0.475 (natural) wt% Li^6 . Geometry and isotopic comparison details are given in Tables I, IA, IB, and IC. Table I includes material densities and modified densities to account for endcaps.

Values of the neutronic characteristics of the lattices are presented in Table II. All results are completely corrected for end effects and known small systematic errors. All uncertainties are standard deviations accumulated by the law of propagation of statistical errors.

TABLE I

Composition and Geometry of the Wet Base Target Central Cell

(Low Density Li⁶)

Region			Radii	Density	Cell	
Name	No.	Isotope	Outside cm	g/cc	Mass, Grams	Volume cc
Target	1	Li ⁶	1.366	1.9388	649.0	334.76
		Li ⁷		0.00575	1.926	
		Al		0.1775	59.41	
		O		0.8155	273.0	
		O		0.938	313.9	
Clad	2	Al	1.620	2.720	289.73	139.83
Clad	3	Zr	1.695	6.5435	153.0	23.38
Water	4		2.182	1.000	365.5	365.46
		H				
		O				
Clad	5	Zr	2.247	6.400	316.0	49.375
Fuel	6		2.960	18.82(c)	13172.0	699.90
		U-235		0.3894	275.96	
		U-238		18.43	12896.0	
Clad	7	Zr	3.045	6.400	600.6	93.844
Water	8		3.442	1.000	484.3	484.33
		H				
		O				
Process						
Tube	9	Zr	4.135	6.400	6495.0	1014.84
Void	10		4.320			301.0
		N		9.932x10 ⁻⁴	0.299	
		O		2.990x10 ⁻⁴	0.101	
Graphite	11		8.578(b)			10588.0
		C		1.67	17673.0	
		N		2.57x10 ⁻⁴	2.725	
		O				
Poison	12	Cu	8.700		793.0(wet)	
Graphite	13		12.160(b)			8849.0
		C		1.67	14777.8	
		N		2.57x10 ⁻⁴	2.277	
		O				
Target		Zr		6.400	48.56	7.588
End caps		Al		2.710	33.95	12.529
Vents		N		9.932x10 ⁻⁴	5.27	5306.08
Fuel(a)		Zr		6.4	274.0	42.813
End Caps						

(a) The zirconium end cap mass includes the clad over the length of the end cap

(b) Cylindricized radii

(c) The reduced density, to account for end caps, is 18.43 g/cm³

TABLE IA

Natural and Base Target Compositions

Isotope	Density g/cc	Natural Mass	Weight %
$\text{Li}_2\text{O} \cdot \text{Al}_2\text{O}_3$	1.8654	624.41	
Li^6	0.00887	2.969	6.4 of total Li ^(a)
Li^7	0.1297	43.4	93.6 of total Li
Al	0.7899	264.44	
O	0.9368	313.61	
Li	0.13860	-----	7.43 Li in $\text{Li}_2\text{O} \cdot \text{Al}_2\text{O}_3$

Target Volume = 334.755 cm

(a) Natural Target = 0.475 wt% Li^6

Base

Li^6	3.14 wt% in Li ^(b)
Li	9.45 wt% in $\text{Li}_2\text{O} \cdot \text{Al}_2\text{O}_3$
$\text{Li}_2\text{O} \cdot \text{Al}_2\text{O}_3$ is	74% of Theoretical Density

(b) Base Target = 0.297 wt% Li^6

TABLE IB
End Cap Detail(d)
Target^(a)

Material	Diameter cm	Thickness cm	Mass per cap/g	Number in Central Cell
Al	2.837	0.576	9.87	4(c)
Zr	3.233	0.462	24.28	2(c)
	Fuel(b)			
Zr	ID = 6.884 OD = 8.270	0.249	137	2

- (a) Target end cap masses do not include the cladding.
- (b) Fuel end cap masses do include the cladding.
- (c) The target was two separate aluminum clad pieces that were in turn clad in zirconium.
- (d) The total target length includes 22.508 inches of target material, 1.271 inches of end caps and a 0.221 inch air gap.

TABLE IC

Mass Spectrometer Isotopic Analysis of Test Cell Fuel

	Sample No. 1	Sample No. 2
Isotope	Weight %	Weight %
U-238	97.878 ± 0.010	97.873 ± 0.010
U-236	0.0215 ± 0.0004	0.0219 ± 0.0004
U-235	2.091 ± 0.010	2.096 ± 0.010
U-234	0.0087 ± 0.0003	0.0088 ± 0.0003

Quoted uncertainties are 95% confidence intervals.

TABLE II

Summary of Neutronic Characteristics Measured and Calculated

W E T	Tubular Fuel Element, Integral Target		k* Air Atmos.	k* Vac. Atmos.	$f_{1/v}$	$t_{1/v}$	ϵ	δ^{28}	$C_o(U^{238})$	$C_o(Li^6)$	Li^6 Density (g/cc)	Li^6 wt% in Target
	Base Target	Experiment Cal. P-3	1.098 $\pm$.005 1.094	1.101 $\pm$.005 -----	0.734 $\pm$.010 0.729	0.178 $\pm$.004 -----	1.0293 $\pm$.0015 1.0292	0.0530 $\pm$.0027 -----	0.444 $\pm$.003 -----	0.264 $\pm$.0023 -----	0.00575 0.00560	0.297 0.280
	Natural Target	Experiment Cal. P-3	1.051 $\pm$.010 1.022 (a)	1.053 $\pm$.010 -----	0.709 $\pm$.010 0.681	0.205 $\pm$.004 -----	----- 1.0292	----- -----	----- -----	----- -----	0.00886 0.01354	0.475 0.677

D R Y	Tubular Fuel Element, Integral Target		k* Air Atmos.	k* Vac. Atmos.	$f_{1/v}$	$t_{1/v}$	ϵ	δ^{28}	$C_o(U^{238})$	$C_o(Li^6)$
	Base Target	Experimental	1.033 $\pm$.007	1.036 $\pm$.007	0.754 $\pm$.010	0.203 $\pm$.004	-----	-----	-----	-----
	Natural Target	Experimental (b)	0.971 $\pm$.020 (b)	0.973 $\pm$.020 (b)	0.718 $\pm$.010	0.242 $\pm$.004	-----	-----	-----	-----

(a) This value of k was inferred from a calculation for Li^6 density, 52% higher than that used in the experiment.

(b) This large uncertainty arises primarily because of the negative value of Δk , which requires extrapolation to a negative mass of poison.

Considerable data, beyond that summarized in Table II, have been collected and are presented in the body of the report. Specifically, spatial distributions of neutron reaction rates with copper-63 were measured both bare and cadmium covered, which agree reasonably well with transport theory calculations obtained with Program S⁽⁸⁾ and THERMOS.⁽⁹⁾ Additional theory-experiment comparisons are present in a companion report by C. L. Bennett.⁽¹⁰⁾ End cap flux peaking was measured for the fuel and targets including the inner end caps of the "twin" targets.

II. Physical Model and Discussion of Results

A. Multiplication Constants, k^*

Multiplication constants presented here are values for finite systems as opposed to the more common infinite medium multiplication constants. This arises from the fact that the experiments are performed in lattice samples having the characteristic leakage of the central region of a finite reactor. For low k systems (say ~ 1.03 or so) this effect or difference from k_∞ is normally 0.1% or so and normally ignored. However, in systems with a Δk of 10% ($k = 1.1$) the magnitude of the difference approaches 0.5 to 1.0% of k , which is of the order of the uncertainty in a multiplication constant measurement and is therefore significant. In principal this accounts, in part, for the frequent disagreement between measured multiplication constants and calculated infinite medium values.

Two criteria must be met for a multiplication constant determination using a small lattice sample and the null reactivity technique⁽¹⁾. First the characteristic spectrum of the lattice must be established in a small test section of lattice sample and second the multiplication of test section must be reduced to unity by means of the addition of an absorber whose cross section is well known. In these typical experiments, three-by-three arrays of lattice cells 37.5 inches long were placed in the cavity of the PCIR. A 24 inch length of the central

lattice cell was used as the test cell. The array was surrounded by a graphite-natural uranium spectral buffer zone that, in turn, was surrounded by a graphite-U-235 driver zone and a graphite reflector.

Spectral equilibrium was established by adjusting the natural uranium-graphite composition until spectral index values in the test cell agree with those of the adjacent cells. Since the graphite lattice is asymmetric, the buffer zone was adjusted non-uniformly to yield a radius weighted spectral match between all eight adjacent cells and the central test cell. Further, since it is impractical to obtain a precise spectral match, measurements were made for various degrees of mismatch. Both too-fast and too-slow conditions were established and the correctly matched spectrum was established by interpolation. Figure 1 illustrates the experimental gold cadmium ratio spectral indices used to match the spectrum of the wet lattice (with process water) experiment. The quantities CdR_{ϕ} and $\overline{CdR}$ are the cadmium ratios at corresponding radial positions (outer surface of the process tube) in the central cell and the average for the eight surrounding cells, respectively.

Given a spectral match, the correct amount of absorber necessary to achieve unit multiplication is established. This is done by comparing PCTR excess reactivity with the test cell and its absorber removed. Nitrogen effects and additional detail is presented in Section IIA. Here again, the precisely correct mass of absorber to achieve unit multiplication is seldom employed as a matter of practicality. The correct mass of absorber for the spectral match conditions is interpolated from a plot of absorber mass vs cadmium ratio spectral index. Figure 2 illustrates the interpolation for the wet experiment. Needless to say, the achievement of a spectral match and correct absorber mass is by experimental iteration, since the absorber does influence the spectrum slightly.

FIGURE 1
Gold Cadmium Ratios for Three Neutron Spectral Conditions
Wet Base Target

FIGURE 2
Mass of Natural Copper to Poison Wet Base Target Cell

Differential measurements were made of the change in multiplication constant corresponding to replacement of the base target by the natural target. This measurement was done in an equilibrium spectral environment established for the base target lattice. Here the mass of absorber required to achieve unit multiplication was established by the null reactivity technique. No attempt was made to adjust or measure the spectrum match since the effect is considered small and natural target material was only available for the test cell.

Spatial distributions of the neutron reaction rates in each region of the test cell are required for determination of the $1/v$ utilization of the normal lattice cell and the cell with the correct mass of absorber. The ratio of these quantities yields the multiplication constant. Experimental data for the $1/v$ reaction rate in copper-63 for all wet and dry experiments used in these determinations are shown in Figures 3 through 10. In each, the transport calculation is shown as normalized to the data. Copper $1/v$ reaction rates are inferred by multiplying copper values by the ratio of medium average cross sections to that of copper. Detailed cross section data was obtained from Thermos, a 30-energy group transport model, and are presented in Table III for the Wet Poisoned Target Experiment. Detailed copper detector positions are shown in Figure 11 and 11a for all radial experiments. Copper foils of 1/2 inch diameter by 0.010 inch thick were used in the graphite. Sector foils were used in the fuel and disks were used in the targets. As shown in Figure 11, the disks in the target were sectioned. Sectioning was done after irradiation to yield spatial detail as well as aid in reducing geometrical corrections to counting due to excessive foil size. Pins of 0.0625 inch diameter by 1/2 inch long were used also in the target to obtain point measurements of the spatial distribution.

FIGURE 3

Case WUBT. Centerline Radial Distributions of Bare and Cadmium Covered Neutron Reaction Rates of Copper-63.

FIGURE 4

Case WPBT. Centerline Radial Distributions of Bare and Cadmium Covered Neutron Reaction Rates of Copper-63.

FIGURE 5

Case WUNT. Centerline Radial Distributions of Bare and Cadmium Covered Neutron Reaction Rates of Copper-63.

FIGURE 6

Case WPNT. Centerline Radial Distributions of Bare and Cadmium Covered Neutron Reaction Rates of Copper-63.

FIGURE 7

Case DUBT. Centerline Radial Distributions of Bare and Cadmium Covered Neutron Reaction Rates of Copper-63.

FIGURE 8

Case DPBT. Centerline Radial Distributions of Bare and Cadmium Covered Neutron Reaction Rates of Copper-63.

FIGURE 9

Case DUNT. Centerline Radial Distributions of Bare and Cadmium Covered Neutron Reaction Rates of Copper-63.

FIGURE 10

Case DPNT. Centerline Radial Distributions of Bare and Cadmium Covered Neutron Reaction Rates of Copper-63.

FIGURE 11

Graphite Moderated Experimental Tubular Fuel Element with Integral Target,
Cell Cross Section

FIGURE 11-A

PCTR Central Cell Graphite

TABLE III

Thermal Average Cross Sections from Thermos

Wet-Poisoned Base Target

Isotope	Reg.	$\bar{\sigma}_a$	$\bar{\sigma}_a 1/v^{(a)}$	$\hat{\sigma}_i = \frac{\bar{\sigma}_{ai}}{\bar{\sigma}_{ai} 1/v}$
Al	1	1.2916×10^{-4}	0.56158	0.23000
θ	1	1.1792×10^{-4}	0.56158	0.00021
Li ⁶	1	5.3065×10^2	0.56158	944.923
Li ⁷	1	2.0778×10^{-2}	0.56158	0.03700
Al	2	1.3325×10^{-1}	0.57934	0.23000
Zr	3	1.1054×10^{-1}	0.61412	0.18000
HW	4	2.0938×10^{-1}	0.63064	0.33201
θ	4	1.3242×10^{-4}	0.63064	2.0998×10^{-4}
Zr	5	1.1238×10^{-1}	0.62432	0.18000
U-235	6	3.1826×10^2	0.59788	637.6865
U-238	6	1.6594	0.59788	2.7755
Zr	7	1.2183×10^{-1}	0.67688	0.18000
HW	8	2.3286×10^{-1}	0.70135	0.33202
θ	8	1.4727×10^{-4}	0.70135	2.0998×10^{-4}
Zr	9	1.2746×10^{-1}	0.70814	0.18000
N	10	1.2391	0.70806	1.7500
θ	10	1.4868×10^{-4}	0.70806	2.1×10^{-4}
C	11	2.3981×10^{-3}	0.70533	3.400×10^{-3}
Cu	12	2.702777	0.70202	3.850
C	13	2.3948×10^{-3}	0.70436	3.40×10^{-3}
N	13	1.2327	0.70436	1.7500
θ	13	1.4790×10^{-4}	0.70436	2.10×10^{-4}

(a) $\sigma_{ao} = 1.000$

Axial measurements of reaction rates were made within the targets and on the surface of the target and fuel to determine end cap effects; and in the case of the targets, midcap effects; and on the surface of the process tube. Figures 12 through 16 show the experimental data and visually fitted curves. Corrections to the reaction rates in each region resulting from end effects were established by determining the average reaction rate axially relative to that at the radial position. These corrections were in all cases a few percent.

Given the corrected $1/v$ absorption rates in all test cell components, the multiplication constants k^* were inferred from the expression,

$$(1) \quad k^* - 1 = \frac{A'_p}{\sum_i A_i} + \frac{\sum_i (A'_i - A_i)}{\sum_i A_i}$$

where

A'_p = relative $1/v$ absorption rate in the poison.

A'_i = relative $1/v$ absorption rate in the i^{th} component of the poisoned cell.

A_i = relative $1/v$ absorption rate in the i^{th} component of the unpoisoned cell.

A condition on Equation 1 is that the absorption rates in fuel for the poisoned and unpoisoned conditions are normalized to be equal. Equation 1 has been derived^(1, 5, 6) from

$$(2) \quad k^* = \frac{f}{f'} \quad \text{where}$$

f = $1/v$ utilization of the unpoisoned cell.

f' = $1/v$ utilization of the poisoned cell.

FIGURE 12

WUBT Axial Distributions of Neutron Reaction Rates in Bare Copper-63

FIGURE 13

WPBT Axial Distributions of Neutron Reaction Rates in Bare Copper-63

FIGURE 14

WPNT and WUNT Axial Distributions of Neutron Reaction Rates in Bare Copper-63

FIGURE 15

DPBT and DUBT Axial Distributions of Neutron Reaction Rates in Bare Copper-63

FIGURE 16

DPBT and DUBT Axial Distributions of Neutron Reaction Rates in Cadmium Covered Copper-63

FIGURE 16

WPBT Axial Distributions of Neutron Reaction Rates in Cadmium Covered Copper-63

With regard to uncertainty calculations in k^* , Equation 1 is of particular interest. It illustrates first that k^*-1 is approximately equal to the ratio of the absorption in the poison to the absorption in all cell components of the unpoisoned cell; and secondly, it is therefore uncertain by ± 2 to $\pm 4\%$ due to counting statistics and cross section errors. For a k^* of approximately 1.1 this average is ± 2 to ± 4 mk. This uncertainty is approximately one tenth of what one might expect from casual inspection of Equation 2, which is a ratio of $1/v$ utilizations that are not known to better than $\pm 2\%$ (~ 15 mf).

Absorption rates and summations of absorption rates as required in Equation 1 are tabulated in Table IV for eight experiments. Each experiment is identified by a four-letter sequence defined as follows:

DPBT - Dry poisoned base target
 DUBT - Dry unpoisoned base target
 DPNT - Dry poisoned natural target
 DUNT - Dry unpoisoned natural target
 WPBT - Wet poisoned base target
 WUBT - Wet unpoisoned base target
 WPNT - Wet poisoned natural target
 WUNT - Wet unpoisoned natural target

(where wet refers to the presence of normal coolant water)

IIB. $1/v$ Utilization

The $1/v$ utilization of a cell component is defined as the ratio of the $1/v$ absorption rate in that component, A_i , to the sum of $1/v$ absorption rates in all cell components. Experimental values are inferred from measured spatial distributions of the specific activity of copper foils $SA(r)$ that are corrected for non- $1/v$ absorption rates. Non- $1/v$ corrections are inferred from cadmium ratio measurements and resonance integral data⁽²⁾. The $1/v$ absorption rate of Cu-63 at space point r is given by:

$$\overline{SA(r)}^{63} = c(N_o / A^{63}) \hat{\sigma}(r)^{63} \int \sqrt{E_o/E} \phi(E,r) dE$$

all E

TABLE IV

Experimental 1/v Absorption Rates

Page 37

#	Region	Isotope	WPBT % ABS ¹ /v		WUBT % ABS ¹ /v		WPNT % ABS ¹ /v		WUNT % ABS ¹ /v		DPBT % ABS ¹ /v		DUBT % ABS ¹ /v		DPNT % ABS ¹ /v		DUNT % ABS ¹ /v	
1	Target	Li ⁶	16.74	210.26	17.79	204.83	20.60	250.49	20.52	236.49	20.25	168.94	20.34	164.70	24.91	195.91	24.25	195.91
		Li ⁷	.02	.22	.02	.21	.01	.12	.01	.11	.02	.18	.02	.17	.01	.10	.01	.10
		Al	.13	1.62	.14	1.58	.10	1.21	.10	1.14	.16	1.30	.16	1.27	.12	.95	.12	.95
		O		.00		.00		.00		.00		.00		.00		.00		.00
2	Target Clad	Al	.17	2.14	.18	2.11	.15	1.82	.15	1.76	.18	1.54	.19	1.53	.17	1.37	.17	1.37
3	Target Clad	Zr	.02	.28	.02	.28	.02	.25	.02	.24	.02	.20	.02	.19	.02	.17	.02	.17
4	Water or Air	Composition	1.13	14.23	1.24	14.27	1.10	13.33	1.12	12.91		.03		.03		.03		.03
5	Inner Fuel Clad	Zr	.05	.58	.05	.57	.05	.55	.05	.56	.05	.38	.05	.37	.04	.34	.04	.34
6	Fuel	U-235	55.68	699.30	61.17	704.33	56.21	683.60	59.07	680.61	60.54	505.08	62.48	505.77	61.24	480.47	59.46	480.47
		U-238	11.18	140.44	12.24	140.92	11.27	137.08	11.82	136.22	12.52	104.42	12.88	104.31	12.66	99.38	12.30	99.38
7	Outer Fuel Clad	Zr	.12	1.48	.12	1.41	.12	1.43	.12	1.34	.13	1.11	.14	1.10	.13	1.01	.12	1.01
8	Water or Air	Composition	2.39	30.04	2.64	30.43	2.47	30.04	2.64	30.43	.01	.06	.01	.06	.01	.06	.01	.06
9	Process Tube	Zr	1.97	24.68	2.17	24.95	2.03	24.68	2.17	24.95	1.68	14.02	1.70	13.80	1.70	13.34	1.65	13.34
10	Air	N	.01	.08	.01	.08	.01	.08	.01	.08		.04		.04	.01	.04		.04
11	Graphite	C	.90	11.29	1.01	11.62	.93	11.29	1.01	11.62	.84	6.99	.88	7.09	.87	6.85	.85	6.85
		N	.06	.77	.07	.79	.06	.77	.06	.79	.06	.48	.06	.48	.06	.47	.06	.47
12	Absorber	Cu	8.42	105.74			3.78	45.98			2.52	21.05			-2.99	-23.446		
13	Graphite	C	.82	10.30	.92	10.56	.85	10.30	.92	10.56	.82	6.84	.87	7.02	.78	6.11	.76	6.11
		N	.06	.72	.06	.74	.06	.72	.06	.74	.06	.48	.06	.49	.05	.43	.05	.43
14	Vent	N	.13	1.66	.15	1.70	.14	1.66	.15	1.70	.13	1.10	.14	1.13	.13	.99	.12	.99
TOTAL FUEL			66.86	839.74	73.41	845.25	67.48	820.68	70.89	816.83	72.92	609.50	75.36	610.08	73.90	579.85	71.76	579.85
TOTAL CELL			100.00	1255.83	100.00	1151.38	99.96	1216.12	100.00	1152.25	99.99	834.24	100.00	809.55	102.97	784.574	100.00	808.02
k* Air Atmos.				1.09797				1.05053				1.03346			0.97104			
TOTAL FUEL			66.86	839.74	73.63	845.25	67.48	820.68	71.09	816.83	72.92	609.50	75.56	610.08	73.90	579.85		579.85
TOTAL CELL			100.00	1255.83	100.00	1148.04	99.96	1216.12	100.00	1149.02	99.99	834.24	100.00	807.41	102.97	784.574	100.00	806.09
k* Vac. Atmos.				1.10118				1.05306				1.03610			0.97331			

where c is a space independent normalizing constant.

$$4) \quad \text{In general } \phi(r) = \frac{\int_0^{E_{ec}} \sigma(E) \phi(E, r) dE}{\int_0^{E_{ec}} \sqrt{E_0/E} \phi(E, r) dE}$$

where
 E_{ec} = upper energy limit of the thermal neutron spectrum, taken as 0.683 ev
 E_0 = 0.0253 ev
 $\phi(E, r)$ - space-energy distribution of neutrons

A similar value of $\overline{SA^x}$ for the material, x , in any given zone, may be obtained from the simple ratio

$$5) \quad \overline{SA^x}(r) = \left[\frac{\phi^x(r) A^{63}}{\phi^{63}(r) A^x} \right] \overline{SA^{63}}(r)$$

where the A are atomic weights.

The total $1/v$ absorption in region or material x , $(Abs)^x$, may be obtained from the volume integral of $\overline{SA^x}(r)$

$$6) \quad (Abs)^x = M^x \int_{\text{all of region } x} \overline{SA^x}(r) dv \quad \Bigg/ \quad \int_{\text{all of region } x} dv \quad \text{where, } M^x \text{ is the mass of material } x \text{ in the region.}$$

Finally, then, the $1/v$ utilization of region x is

$$7) \quad f^x = (Abs)^x / \sum_i (Abs)^i$$

As stated earlier, values of the cross sections, $\phi(r)$, have been determined from transport calculations using the IBM code THERMOS (see Table III). Details of this calculation are given in a separate report by C. L. Bennett. This technique is considered to be a distinct improvement over the direct use of Equation 2, wherein one must, in practice, assume both the energy distribution of neutrons and its characteristic

spectral indices; e.g., a Maxwellian with an estimated neutron temperature. One may note that as defined in Equation (4), the values will be the 2200 m/sec values unless the cross section is non-1/v for $0 \leq E \leq 0.683$ eV., see U-235 in Table III.

The absorption rates tabulated in Table IV are normalized to a cell total of unity; hence, each is the region 1/v utilization.

IIC. Fast Fission Factor

The fast fission factor, ϵ , is the term used to include the contribution of neutrons produced by fast fissions of U^{238} in the calculation of the infinite medium neutron multiplication factor, k_{∞} . The measured quantity is actually δ^{28} , the ratio of total fission in U-238 to total fission in U-235. The relation between ϵ and δ^{28} is:

$$8) \quad \epsilon = 1 + \left[\frac{\bar{\nu}^{28} - 1 - \bar{\alpha}^{28}_{fast}}{\bar{\nu}^{25}} \right] \delta^{28}$$

The constants (averaged over the neutron spectrum in the fuel⁽¹¹⁾) are:

	<u>Unpoisoned</u>	<u>Poisoned</u>
$\bar{\nu}^{28} =$	2.7373	2.7373
$\bar{\nu}^{25} =$	2.4317	2.4317
$\bar{\alpha}^{28} =$	0.3928	0.3928
$\epsilon =$	$1 + 0.5529 \delta^{28}$	$1 + 0.5529 \delta^{28}$

Fission product gamma activities of 2.12^a/o U^{235} fuel and 20-fold depleted uranium foils are measured to determine δ^{28} ;

$$9) \quad \delta^{28} = \frac{N^{28} \int_{E_f}^{\infty} \sigma_f^{28}(E) \phi(E) dE}{N^{25} \int_0^{\infty} \sigma_f^{25}(E) \phi(E) dE}$$

Two small, thin (3 mil) uranium metal foils of differing U-238 to U-235 atom ratios were irradiated in the fuel at flux-equivalent positions. The fission product activity A of these two foils were measured. Let the subscripts D and F refer to the 20-fold depleted and 2.12^a/o U-235 fuel foils, respectively. The activities A_D and A_F were plotted against the source uranium foil U-235 content in atom percent U-235⁽⁴⁾, as in Figure 17.

FIGURE 17

Fission Activity Versus U-235 Content

The straight line is extrapolated to zero U-235 content, which is the fission activity due to U-238 fissions alone, A_O^{28} . Taking into account that the U-238 surface atom density is less in the 2.12^a/o U-235 enriched fuel foil than in pure U-238, $0.9788 A_O^{28}$ is subtracted from A_F to determine the activities due to U-235 fissions, A_F^{25} , in the fuel.

The slope of the straight line is given by:

$$10) \quad M = \frac{A_F - A_D}{a/o(F)^{U-235} - a/o(D)^{U-235}}$$

The ordinate and abscissa are taken as:

$$11) \quad Y = A_F$$

and

$$12) \quad X = 2.1197 \text{ } a/o \text{ } U-235$$

Thus, δ^{28} is inferred from experimental data using the equation:

$$13) \quad \delta^{28} = \frac{\text{fissions in U-238 in the fuel}}{\text{fissions in U-235 in the fuel}} = \frac{.9788 A_o^{28}}{A_F - .9788 A_o^{28}} P(t)$$

where,

$$P(t) = \frac{\text{count rate at time } t \text{ per } U^{235} \text{ fission}}{\text{count rate at time } t \text{ per } U^{238} \text{ fission}} \quad (\text{see Ref. 11})$$

Fission product gamma-rays were measured by counting integrally above 511 keV on a TMC 400 channel analyzer. Successive data taken from 20 to 60 hours after PCTR shutdown yielded values for δ^{28} with a standard error of $\pm 5\%$.

The fast fission factor, ϵ , and δ^{28} are tabulated below for the uranium fuel 2.12^a/o enriched in U-235 using the base target [5.753 mg/cc Li⁶ in (Li AlO₂)₂] with water between the fuel and target cans.

	Unpoisoned Lattice	Poisoned Lattice
ϵ	1.0293 $\pm$.0015	1.0301 $\pm$.0015
δ^{28}	0.0530 $\pm$.0027	0.0545 $\pm$.0027

IID. U-238 Conversion Ratio (Method 1)

The conversion ratio for U-238 at a given location, X, in the reactor core is defined as

$$14) \quad C_{28} = \frac{P_u^{239} \text{ atoms created in region X}}{U^{235} \text{ atoms destroyed in region X}}$$

which can be rewritten in terms of flux ϕ , cross section σ , and the number of atoms, N, as

$$15) \quad C_{28} = \frac{N^{28} \int_0^{\infty} \sigma_c^{28}(E) \phi(E) dE}{N^{25} \int_0^{\infty} \sigma_f^{25}(E) \phi(E) dE} \quad \frac{1}{(1 + \alpha^{25}_{\text{total}})}$$

The cross sections are considered to be averaged over the neutron spectrum in the fuel so that

$$16) \quad N^{28} \int_0^{\infty} \sigma_c^{28}(E) \phi(E) dE = N^{28} \bar{\sigma}_c^{28} \phi$$

$$17) \quad \text{and} \quad N^{25} \int_0^{\infty} \sigma_f^{25}(E) \phi(E) dE = N^{25} \bar{\sigma}_f^{25} \phi$$

$$18) \quad \text{where} \quad \bar{\sigma}_i = \frac{\int_0^{\infty} \sigma_i(E) \phi(E) dE}{\int_0^{\infty} \phi(E) dE}$$

Substituting the above equations into the conversion ratio equation (Equation 15), the following equation is obtained:

$$19) \quad C_{28} = \left(\frac{N^{28}}{N^{25}} \right) \left(\frac{\bar{\sigma}_c^{28}}{\bar{\sigma}_f^{25}} \right) \left(\frac{1}{1 + \alpha^{25}} \right)$$

The U-235 capture-to-fission ratio, $\alpha_{\text{total}}^{-25}$, is taken from transport theory (Program S) as the average over the neutron spectrum in the fuel. The N^{28}/N^{25} atom ratio is known from mass spectrometer analysis. This leaves the cross section ratio $\sigma_c^{-28}/\sigma_f^{-25}$ to be experimentally measured.

Identical sector foils were irradiated simultaneously both in the fuel and in the PCTR thermal column. Four types of uranium were used: foils enriched to 2.12% U-235, 20-fold depleted in U-235, foils 93% enriched in U-235, and copper foils.

The fuel-to-thermal column ratios of fission rates and capture rates are written as follows:

$$20) \quad \frac{\phi_f (\bar{\sigma}_f^{-25})_f}{\phi_c (\bar{\sigma}_f^{-25})_c} \qquad 21) \quad \frac{\phi_f (\bar{\sigma}_c^{-28})_f}{\phi_c (\bar{\sigma}_c^{-28})_c}$$

where ()_f refers to fuel and ()_c refers to thermal column. Multiplying by the corresponding average thermal cross section in the thermal column,

$$22) \quad \frac{\phi_f (\bar{\sigma}_f^{-25})_f}{\phi_c (\bar{\sigma}_f^{-25})_c} (\bar{\sigma}_f^{-25})_c = \frac{\phi_f}{\phi_c} (\bar{\sigma}_f^{-25})_f$$

$$23) \quad \frac{\phi_f (\bar{\sigma}_c^{-28})_f}{\phi_c (\bar{\sigma}_c^{-28})_c} (\bar{\sigma}_c^{-28})_c = \frac{\phi_f}{\phi_c} (\bar{\sigma}_c^{-28})_f$$

Solving for $(\bar{\sigma}_c^{28})_f / (\bar{\sigma}_f^{25})_f$ yields

$$24) \quad \frac{(\bar{\sigma}_c^{28})_f}{(\bar{\sigma}_f^{25})_f} = \frac{\frac{\phi_f (\bar{\sigma}_c^{28})_f}{\phi_c (\bar{\sigma}_c^{28})_c}}{\frac{\phi_f (\bar{\sigma}_f^{25})_f}{\phi_c (\bar{\sigma}_f^{25})_c}} \frac{(\bar{\sigma}_c^{28})_c}{(\bar{\sigma}_f^{25})_c}$$

Let R stand for the fuel-to-thermal column activity ratio, so that

$$25) \quad \frac{(\bar{\sigma}_c^{28})_f}{(\bar{\sigma}_f^{25})_f} = \frac{R^{28}}{R^{25}} \frac{(\bar{\sigma}_c^{28})_c}{(\bar{\sigma}_f^{25})_c}$$

The values for $\bar{\sigma}_c^{28}$ thermal and $\bar{\sigma}_f^{25}$ thermal are obtained from program THERMOS as the sub-cadmium average cross-sections over the thermal column neutron spectrum. The values for R^{28} and R^{25} are defined as follows:

$$26) \quad R^{28} = \frac{(A_f^{28})_{\text{bare}}}{(A_c^{28})_{\text{bare}} - (A_c^{28})_{\text{epi Cd}}}$$

$$27) \quad R^{25} = \frac{(A_f^{25 \text{ F.P.}})_{\text{bare}}}{(A_c^{25 \text{ F.P.}})_{\text{bare}} - (A_c^{25 \text{ F.P.}})_{\text{epi Cd}}}$$

where,

$(A_f^{28})_{\text{bare}}$ = The total U-238 capture activity of fuel in the core (measured by the Np-239 peak activity)

$(A_c^{28})_{\text{bare}}$ = The total U-238 capture activity of the fuel in the thermal column (measured by the Np-239 peak activity)

$(A_f^{25 \text{ F.P.}})_{\text{bare}}$ = The total U-235 fission product activity of the fuel in the core

$(A_c^{25 \text{ F.P.}})_{\text{bare}}$ = The total U-235 fission product activity of the fuel in the

Cadmium ratios of U-238_{capture} and U-235_{fission} in the thermal column are used to obtain the epi-Cd activities of U-238_{capture} and U-235_{fission} in the thermal column, respectively.

U-238 Conversion Ratio (Method 2)

Activity measurements were made with copper, fuel, depleted uranium, and enriched uranium sector foils in the 2.12^a/_o U-235 enriched fuel and in the thermal column simultaneously. The epi-Cd copper activity in the fuel was measured with Cd covered copper foils on the inner and outer surfaces of the fuel cladding. The bare activity of the copper in the fuel was obtained from sector foils placed in the fuel.

The U-238 conversion ratio in the fuel is defined by:

$$28) \quad C_{28} = \frac{[\text{Total Absorption}]_{28} - [\text{Fast Fission}]_{28}}{[\text{Total Absorption}]_{25}} = \frac{\text{Total Np}^{239} \text{ Production}}{\text{Total Absorption in U}^{235}}$$

The U-235 fission cadmium ratio can be written as:

$$29) \quad \text{CdR}_{\text{fission}}^{25} = \frac{R_{\text{fission}}^{25}}{R_{\text{fission}}^{25} - R_{\text{Cu}} \left(1 - \frac{1}{\text{CdR}_{\text{f}}^{\text{Cu}}} \right) g_{\text{fission}}^{25}} \quad \text{for the fuel.}$$

Similarly, the U-238 capture Cd ratio in the fuel can be written as:

$$30) \quad \text{CdR}_{\text{f}}^{28} = \frac{R^{28}}{R^{28} - R_{\text{Cu}} \left(1 - \frac{1}{\text{CdR}_{\text{f}}^{\text{Cu}}} \right) g_{\text{f}}^{28}}$$

assuming that

$$31) \quad g_{\text{f}}^{25} \left(\frac{A_{\text{f}}^{\text{Cu}}}{A_{\text{c}}^{\text{Cu}}} \right)_{\text{sub Cd}} = \left(\frac{A_{\text{f}}^{25}}{A_{\text{c}}^{25}} \right)_{\text{sub Cd}}$$

and

$$32) \quad g_{\text{f}}^{28} \left(\frac{A_{\text{f}}^{\text{Cu}}}{A_{\text{c}}^{\text{Cu}}} \right)_{\text{sub Cd}} = \left(\frac{A_{\text{f}}^{28}}{A_{\text{c}}^{28}} \right)_{\text{sub Cd}}$$

Average cross sections in the fuel for thermal absorptions are available from THERMOS, Table IV. If we use the value $\left(\frac{\bar{\sigma}_c^{28}}{\bar{\sigma}_a^{25}} \right)_{f \text{ sub Cd}}$ from THERMOS, we can write

$$33) \quad C_{28} = \left[\frac{\frac{[A_f^{28}]_{\text{total}}}{[A_f^{28}]_{\text{sub Cd}}}}{1 + \frac{[A_f^{25}]_{\text{epi Cd}}}{[A_f^{25}]_{\text{sub Cd}}} \left(\frac{1 + \bar{\alpha}_{\text{epi Cd}}^{25}}{1 + \bar{\alpha}_{\text{sub Cd}}^{25}} \right)_f} \right] \left(\frac{\bar{\sigma}_c^{28}}{\bar{\sigma}_a^{25}} \right)_{f \text{ sub Cd}} \left(\frac{N^{28}}{N^{25}} \right)$$

The ratio of total activity to sub cadmium activity of a detector foil can be expressed in terms of the cadmium ratio: $\frac{A_{\text{total}}}{A_{\text{sub Cd}}} = 1 + \frac{1}{\text{CdR} - 1}$. Hence, the

U-238 conversion ratio is expressed as follows:

$$34) \quad C_{28} = \frac{\left(1 + \frac{1}{\text{CdR}_f^{28} - 1} \right)}{1 + \left(\frac{1}{\text{CdR}_f^{25} - 1} \right) \frac{[1 + \bar{\alpha}_{\text{epi Cd}}^{25}]_f}{[1 + \bar{\alpha}_{\text{sub Cd}}^{25}]_f}} \left(\frac{\bar{\sigma}_c^{25}}{\bar{\sigma}_a^{25}} \right)_{f \text{ sub Cd}} \left(\frac{N^{28}}{N^{25}} \right)$$

If we choose to use experimentally measured absorption data instead of THERMOS sub-cadmium cross sections, we can write:

$$35) \quad [(Abs)_f^{25}]_{\text{sub Cd}} = [(A_f^{25})]_{\text{sub Cd}} (1 + \bar{\alpha}_{\text{sub Cd}}^{25})_f$$

$$36) \quad [(Abs)_f^{25}]_{\text{epi Cd}} = \left(\frac{1}{\text{CdR}_f^{25} - 1} \right) [(Abs)_f^{25}]_{\text{sub Cd}} \frac{[1 + \bar{\alpha}_{\text{epi Cd}}^{25}]_f}{[1 + \bar{\alpha}_{\text{sub Cd}}^{25}]_f}$$

$$37) \quad [(Abs)_f^{28}]_{\text{epi Cd}} = \left(\frac{1}{\text{CdR}_f^{28} - 1} \right) [(Abs)_f^{28}]_{\text{sub Cd}}$$

Thus,

$$38) \quad C_{28} = \frac{\left(1 + \frac{1}{CdR_f^{28} - 1}\right) [(Abs)_f^{28}]_{sub\ Cd}}{[(Abs)_f^{25}]_{sub\ Cd} + [(Abs)_f^{25}]_{epi\ Cd}}$$

II E. Initial Conversion Ratio of Lithium-6 in the Base Target

The Li^6 conversion ratio is defined as the total number of neutrons captured by Li^6 in the target divided by the total number of neutrons absorbed in U-235 in the fuel. Sub-cadmium reaction rates of Li^6 in the target and U-235 in the fuel were used as the basis for the analysis. Corrections for the epi-cadmium contributions were made by use of cadmium ratios. Copper foil activation and cross sections were used to determine the sub-cadmium flux ratio ($\phi_{target}/\phi_{fuel}$). The cadmium ratio of Li^6 in the target, t , was inferred from copper foil data by the relationship:

$$39) \quad CdR_t^{Li^6} = \frac{\left(\frac{\sigma}{\sigma_t} Li^6\right)_{total}}{\left(\frac{\sigma}{\sigma_t} Li^6\right)_{total} - \left(\frac{\sigma}{\sigma_t} Cu\right)_{total} \left(1 - \frac{1}{CdR_t^{Cu}}\right) \left(\frac{\sigma}{\sigma_t} Cu\right)_{sub\ Cd\ THERMOS}}$$

The total cross sections were averaged over the neutron spectrum in the target using Program S.

The ratio of the sub-cadmium neutron flux in the target to that in the fuel was determined from copper thermal activation data using

$$40) \quad \left(\frac{\phi_t}{\phi_f}\right)_{sub\ Cd} = \left(\frac{A_t}{A_f}\right)_{sub\ Cd} \left(\frac{\frac{\sigma_f Cu}{\sigma_t Cu}}{\frac{\sigma_f Cu}{\sigma_t Cu}}\right)_{sub\ Cd\ THERMOS}$$

The sub-cadmium reaction rates can be written as:

$$41) \quad \frac{\text{Reaction rate in Li}^6 \text{ in the target}}{\text{Reaction rate in U-235 in the fuel}} = \left(\frac{\phi_t}{\phi_f} \right)_{\text{sub Cd}} \left(\frac{\left(\frac{\bar{\sigma}_c \text{ Li}^6}{\bar{\sigma}_a^{25}} \right)_t}{\left(\frac{\bar{\sigma}_a^{25}}{\bar{\sigma}_a^{25}} \right)_f} \right)_{\text{sub Cd}} \left(\frac{N_{\text{Li}^6}}{N_{25}} \right)_{\text{THERMOS}}$$

The initial conversion ratio for lithium-6 in the target can be written as:

$$42) \quad C_{\text{Li}^6} = \frac{\text{Neutrons captured by Li}^6 \text{ in the target}}{\text{Neutrons absorbed by U}^{235} \text{ in the fuel}}$$

Combining Equations (2) and (3), the Li⁶ initial conversion ratio equation used is as follows:

$$43) \quad C_{\text{Li}^6} = \left[\frac{1 + \frac{1}{\text{CdR}_t \text{ Li}^6 - 1}}{1 + \left(\frac{1}{\text{CdR}_f^{25} - 1} \right) \frac{(1 + \alpha_{\text{epi Cd}}^{-25})_f}{(1 + \alpha_{\text{sub Cd}}^{-25})_f}} \right] \left[\left(\frac{A_{\text{Cu}}}{A_{\text{f Cu}}} \right)_{\text{sub Cd}} \left(\frac{\bar{\sigma}_{\text{Cu}}}{\bar{\sigma}_t} \right)_{\text{sub Cd}} \right]_{\text{THERMOS}} \cdot \frac{\left(\frac{\bar{\sigma}_c \text{ Li}^6}{\bar{\sigma}_a^{25}} \right)_t}{\left(\frac{\bar{\sigma}_a^{25}}{\bar{\sigma}_a^{25}} \right)_{\text{sub Cd}}}_{\text{THERMOS}} \left(\frac{N_{\text{Li}^6}}{N_{25}} \right)$$

The first term is the correction for epi cadmium absorptions. The second term is the ratio of the sub cadmium neutron flux in the target to that in the fuel ($\phi_{\text{target}}/\phi_{\text{fuel}}$). The third term is the ratio of thermal absorption cross sections of Li⁶ to U-235, averaged over the target and fuel neutron spectra, respectively, using program THERMOS. The last term is the ratio of total nuclei of Li⁶ in the target to the total nuclei of U-235 in the fuel.

TABLE V

Calculated Data Used to Determine Initial Conversion Ratios

<u>Quantity</u>	<u>Source</u>	<u>Poisoned Lattice</u>	<u>Unpoisoned Lattice</u>
$\alpha^{25}_{\text{total}}$	Program S	.22104	.21908
$\alpha^{25}_{\text{epi Cd}}$	Program S	.5224	.5241
$\alpha^{25}_{\text{sub Cd}}$	THERMOS	.175	.174
N^{28}/N^{25}	Mass Spectrometer	46.1765	46.1765
Fuel Average Thermal Cross Sections, Barns			
$\bar{\sigma}_c^{28}$	THERMOS	1.6601	1.6984
$\bar{\sigma}_a^{25}$	THERMOS	381.43	391.69
$\bar{\sigma}_f^{25}$	THERMOS	324.68	333.56
$\bar{\sigma}_{1/v}$	THERMOS	.59812	.61243
$\bar{\sigma}_{\text{Cu}}$	THERMOS	2.30276	2.35786
Average Thermal Cross Sections in the Thermal Column, Barns			
$\bar{\sigma}_c^{28}$	THERMOS	2.4351	2.4351
$\bar{\sigma}_a^{25}$	THERMOS	595.26	595.26
$\bar{\sigma}_f^{25}$	THERMOS	508.84	508.84
$\bar{\sigma}_{1/v}$	THERMOS	.88352	.88352
Target Average Cross Sections, Barns			
$\bar{\sigma}_{\text{Li}^6}^{\text{total}}$	Program S	78.006	79.272
$\bar{\sigma}_{\text{Cu}}^{\text{total}}$	Program S	.357823	.362641
$\bar{\sigma}_{\text{Cu}}^{\text{thermal}}$	THERMOS	2.12828	2.1604
$\bar{\sigma}_{\text{Li}^6}^{\text{thermal}}$	THERMOS	522.36	530.25

TABLE VI

Initial Conversion Ratios

Wet Base Target

<u>Quantity</u>	<u>Method of Analysis Utilized</u>	<u>Poisoned</u>	<u>Unpoisoned</u>
C_{28}	Total activity ratios, $\alpha_{\text{total}}^{-25}$	$.4776 \pm .0038^*$	$.4510 \pm .0028^*$
C_{28}	Sub Cd activity ratios, $\alpha_{\text{thermal}}^{-25}$, $\alpha_{\text{epi Cd}}^{-25}$, U-238, U-235 Cd ratios	$.4690 \pm .0037^*$	$.4440 \pm .0028^*$
C_{28}	U-238, U-235 Cd ratios, $\alpha_{\text{thermal}}^{-25}$, $\alpha_{\text{epi Cd}}^{-25}$, THERMOS cross sections	$.4693 \pm .0037^*$	$.4443 \pm .0028^*$
C_{Li^6}	U-235, Li^6 Cd ratios, $\alpha_{\text{thermal}}^{-25}$, $\alpha_{\text{epi Cd}}^{-25}$, THERMOS cross sections	$.2703 \pm .0022^*$	$.2640 \pm .0023^*$
C_{total}	$C_{28} + C_{\text{Li}^6}$	$.7393 \pm .0084^*$	$.7080 \pm .0077^*$

* Standard error in repeated counting data quoted

III. Corrections

A number of systematic corrections are necessary in a practical experimental determination of neutronic characteristics. Null reactivity establishment of the correct mass of absorber for the unit multiplication is complicated somewhat by the presence of nitrogen in the voids, in the graphite, and in the vents. Copper is not an ideal $1/v$ absorber and must be corrected to obtain an equivalent $1/v$ mass. Copper foils placed in the target displace Li^6 creating a flux peaking effect, which, in turn, influences the $1/v$ utilization of the target and fuel.

A. Nitrogen Corrections

Nitrogen affects the null reactivity of the poisoned cell in two ways; first, air occupies the void spaces of graphite and the vent channels and with respect to a helium or void atmosphere acts as a poison; second, upon removal of the poisoned test cell ambient air enters. Consideration of the reactivity of the PCTR for both conditions, ρ_{CIA} (CIA = test cell in plus absorber) and ρ_{CO} (CO = test cell out), respectively, together with differential reactivities:

$$\Delta\rho_C = \text{worth of the test cell less nitrogen}$$

$$\Delta\rho_{NC} = \text{worth of the nitrogen in the test cell graphite and vents}$$

$$\Delta\rho_{NCO} = \text{worth of the nitrogen in the test cell volume with the test cell removed, and}$$

$$\Delta\rho_{Cu} = \text{worth of the copper in the test cell;}$$

yields the expression

$$44) \quad \Delta\rho = \Delta\rho_{NCO} - (\rho_{CIA} - \rho_{CO})$$

$$\Delta\rho = \Delta\rho_{NC} + \Delta\rho_{Cu} - \Delta\rho_C$$

where $\Delta\rho$ is the extent to which the cell is over ($\Delta\rho$ positive) or under ($\Delta\rho$ negative) poisoned. The quantities ρ_{CIA} and ρ_{CO} are measured directly while $\Delta\rho_{NCO}$ is inferred from a copper sensitivity, S_{CO} , measurement using the expression:

$$45) \quad \Delta \rho_{\text{NCO}} = DC \frac{P}{T} (M_{\text{NCO}})_{\text{STP}} S_{\text{CO}} (1-c)$$

where

D = thermal disadvantage factor for the thickness of copper used in the determination of S_{CO}

$$S_{\text{CO}} = \Delta \rho_{\text{Cu}} / M_{\text{Cu}}, \text{ } \phi/\text{g}$$

$\Delta \rho_{\text{Cu}}$ = worth of a mass of copper, M_{Cu} widely dispersed in the empty test cell, ϕ

P = atmospheric pressure, mb

T = ambient temperature $^{\circ}\text{K}$

$(M_{\text{NCO}})_{\text{STP}}$ = mass of nitrogen (at standard temperature and pressure) in the empty cell

$$C = \frac{\sigma_o^{\text{N}}}{\sigma_o^{\text{Cu}}} \frac{A^{\text{Cu}}}{A^{\text{N}}} \frac{T_o}{P_o}$$

σ_o^{N} = 2200 m/sec microscopic absorption cross section of nitrogen

σ_o^{Cu} = 2200 m/sec microscopic absorption cross section of natural copper

$A^{\text{Cu}}, A^{\text{N}}$ = copper and nitrogen atomic weights

T_o, P_o = standard temperature and pressure

The quantity $(1-c)$ is a correction for the non $1/v$ nature of the epi-thermal energy dependence of the absorption cross section of natural copper. It has been shown that c is given approximately by

$$46) \quad c = - \frac{m_1}{m_2} \frac{RI_{\text{ex}}^{\text{Cu}}}{PI_t^{\text{Cu}} (CdR - 1)}$$

where

m_1/m_2 = ratio of the fast-to-thermal adjoint fluxes and is approximately equal to the resonance escape probability of the test lattice⁽⁵⁾

$RI_{\text{ex}}^{\text{Cu}}$ = effective resonance integral of natural copper for the thickness of copper used in measuring S_{CO}

RI_t^{Cu} = the corresponding total resonance integral

CdR = cadmium ratio of copper in the empty test cell (approximately the test cell boundary value) for the thickness copper used in measuring S_{CO}

absorber masses.

Table VII contains values of reactivities employed in the determination of the correct

TABLE VII

EXPERIMENTAL PCTR REACTIVITY VALUES

		P13 6605F WPBT AlPT	P18 6605F4 WPBT AlPT	P19 6605 WPBT ZrPT	6605G WPNT ZrPT	6610A DPBT AlPT	6610A DPBT ZrPT	6610A DPNT ZrPT
$\rho(\text{CIA})$	ϕ	22.774	25.200	23.413	16.120	28.444	26.559	16.920
$\rho(\text{CO})$	ϕ	20.529	23.051	23.712	16.354	26.876	26.937	26.937
$\rho(\text{COSA})$	ϕ	15.223	17.751	18.404	11.13	22.644		
$\rho(\text{CIASA})$	ϕ	18.213	20.548	18.807	11.239	24.840		
$\Delta\rho(\text{CIA-CO})$	ϕ	2.245	2.149	- 0.299	- 0.234	1.568	- 0.378	- 10.017
$\Delta\rho(\text{C-COSA})$	ϕ	5.306	5.300	5.308	5.287	4.232	4.232	4.232
$\Delta\rho(\text{CIA-CIASA})$	ϕ	4.561	4.652	4.606	4.881	3.604		
M_{Co}	g	207.928	207.928	207.928	207.928	207.928		
S_{Co}	ϕ/g	0.02600	0.02597	0.02553	0.02591	0.02083		
S_{Cu}	ϕ/g	0.02194	~ 0.022	~ 0.022	0.02347	0.01733	~ 0.01733	~ 0.01733
$\Delta\rho_{\text{NC}}$	ϕ	1.411	1.392	1.368	1.380	~ 1.156(a)	~ 1.156(a)	~ 1.156
Δ	ϕ	+ 0.834	+ 0.757	- 1.667	- 1.614	- 0.4125	1.534 (over-poisoned)	11.73 (over-poisoned)
ΔM_{Cu}	g	38.02	34.4	- 75.77	- 68.77	23.4	- 88.5	- 578.02
M_{Cu}	g	840.08	842.12	+ 842.12	397.74	311.15	311.15	311.15
$M_{\text{Cu}}(k=1.000)$	g	878.10	876.5	+ 766.35	328.97	334.95	222.65	- 256.865
$1 + B$		1.0354	1.0354	1.0354	1.0377	1.0706	1.0706	1.0797
M_0	g	999.18	907.53	793.47	341.37	358.60	238.37	- 276.00

III B. 1/v Mass Equivalent of a Mass of Copper

Derivation of the 1/v mass equivalent M_o of a mass of copper absorber, M , requires simply that the reactivity coefficient of each be the same. Account is therefore taken of the worth of epithermal resonance captures in copper relative to the worth of 1/v captures. Using notation similar to that in the preceeding section it has been shown⁽⁶⁾ that

$$47) \quad M_o = M \left[1 + \frac{m_1}{m_2} \cdot \frac{RI_{ex}^{Cu}}{RI_t^{Cu} (CdP - 1)} \right]$$

Values of M_o are only a few percent greater than M and are included in Table VII.

III. C . Absorption in Copper Poison

Determination of the total absorption in the copper poison is made under the assumption that variation in the flux within the copper in the three Cartesian directions are independent. Copper foils are placed throughout the absorber to measure the x, y, z distributions. Table VIII coantains this data for a 0.040" thick layer of absorber. Curves of the reaction rates are shown in Figure 18. Cadmium covered copper data are measured also to allow determination of the 1/v absorber rates. As for other cell constituents, the 1/v absorber rate in the copper for each experiment is shown in Table IV.

TABLE VIII

SPATIAL DISTRIBUTIONS OF RELATIVE SPECIFIC ACTIVITIES IN COPPER ABSORBER SHEETS; AS MEASURED WITH 0.443 INCH SQUARE X0.010 INCH THICK COPPER FOILS. (WET POISONED BASE TARGET)

<u>X (Inches)⁽¹⁾</u>	<u>Y (Inches)⁽¹⁾</u>	<u>Z = 0.005 Inch⁽¹⁾</u>	<u>Z = 0.015 Inch</u>	<u>Z = 0.025 Inch</u>	<u>Z = 0.035 Inch</u>
		<u>SA</u>	<u>SA</u>	<u>SA</u>	<u>SA</u>
0.22	0.22	2.3533	2.3279	2.3231	2.3329
0.22	1.50	2.2924	2.2474	2.2650	2.2920
0.22	2.78	2.3611	2.3257	2.3455	2.3612
3.97	0.22	2.3314	2.2978	2.2763	2.3154
3.97	0.86	2.2703	2.2204	2.2351	2.2526
3.97	1.50	2.2542	2.2288	2.2322	2.2621
12.00	0.22	2.3390	2.2982	2.2973	2.3201
12.00	0.86	2.2839	2.2506	2.2613	2.2750
12.00	1.50	2.2554	2.2193	2.2248	2.2477
20.03	1.50	2.2948	2.2504	2.2410	2.2677
23.78	1.50	2.3160	2.3062	2.2953	2.3257

⁽¹⁾ Values of X, Y, Z are the Coordinates of the Centroids of the Square Copper Foils in a Cartesian System Whose Origin is at the Lower Inside Corner of the Front End of the Inner Sheet as Shown in Figure 18.

FIGURE 18
Relative Reaction Rate Distributions of Bare Cu^{63} in Absorber, WPBT Cell

III. D. Flux Enhancement by Detectors

Copper has a cross section considerably smaller than that of the target. Hence, placing a foil in the target produces a flux peak and excessive activity in the foil. Experiments performed in earlier work on this program⁽⁷⁾ yielded an evaluation of this effect. Foils of diminishing thickness were activated in a single location in the target material and the resulting activity versus thickness was extrapolated to zero thickness. The effect for .010" thick foil used in these experiments was 5%. Hence, the reaction rate in the target is 5% lower than inferred directly from foil activation. This correction influences the value of $1/v$ target utilization to first order but only to second order in fuel utilization and multiplication constant. Corrected values of target absorption rate appear in Table IV.

IV. Appendicies

A. U-238 and Copper Cd Ratios

Since the cadmium ratio of U-238 was measured in the thermal column, (c), and the cadmium ratio of copper was measured in the fuel, (f), it should be possible to infer the U-238 Cd ratio in the core, and the copper Cd ratio in the thermal column.

The bare activities relative to the thermal column, $\frac{A_f}{A_c}$, were measured experimentally for both U-238 and copper.

The bare U-238 activity in the fuel relative to the sub-cadmium U-238 activity in the thermal column, R_{28} , is defined by:

$$R_{28} \equiv \frac{(A_f^{28})_{\text{bare}}}{(A_c^{28})_{\text{bare}} - (A_c^{28})_{\text{epi Cd}}} \quad (1)$$

$$R_{28} = \frac{(A_f^{28})_{\text{bare}}}{(A_c^{28})_{\text{bare}} \left[1 - \frac{1}{CdR_c^{28}} \right]} \quad (2)$$

$$R_{28} = \frac{A_f^{28}}{A_c^{28}} \left[\frac{CdR_c^{28}}{CdR_c^{28} - 1} \right] \quad (3)$$

Similarly, the bare copper activity in the core fuel relative to the sub-cadmium copper activity in the thermal column, R^{Cu} , is defined by:

$$R^{Cu} \equiv \frac{(A_f^{Cu})_{bare}}{(A_c^{Cu})_{bare} - (A_c^{Cu})_{epi\ Cd}} \quad (4)$$

$$R^{Cu} = \frac{(A_f^{Cu})_{bare}}{(A_c^{Cu})_{bare} \left[1 - \frac{1}{CdR_c^{Cu}} \right]} \quad (5)$$

$$R^{Cu} = \left(\frac{A_f^{Cu}}{A_c^{Cu}} \right)_{bare} \left[\frac{CdR_c^{Cu}}{CdR_c^{Cu} - 1} \right] \quad (6)$$

Solving for CdR_c^{Cu} column, we get:

$$CdR_c^{Cu} = \frac{R^{Cu}}{R^{Cu} - \left(\frac{A_f^{Cu}}{A_c^{Cu}} \right)_{bare}} \quad (7)$$

Let us assume that the sub-cadmium activities of the copper to the U-238 in the thermal column and the core have a constant relationship:

$$\frac{(A_c^{Cu})_{sub\ Cd}}{(A_c^{28})_{sub\ Cd} (g_c^{28})} = \frac{(A_f^{Cu})_{sub\ Cd}}{(A_f^{28})_{sub\ Cd} (g_f^{28})} \quad (8)$$

An equivalent expression in terms of Cd ratios may be written:

$$\frac{\left(\frac{1}{CdR_c^{Cu} - 1} \right)}{\left(\frac{1}{CdR_c^{28} - 1} \right) g_c^{28}} = \frac{\left(\frac{1}{CdR_f^{Cu} - 1} \right)}{\left(\frac{1}{CdR_f^{28} - 1} \right) g_f^{28}} \quad (9)$$

where

$$g_i^{28} = \frac{(\bar{\sigma}_c^{28})_i (\text{thermal})}{\left(\frac{\bar{\sigma}_{1/v}}{\sigma_{o1/v i}} \right) (\sigma_{oc}^{28})}$$

and the averages are taken from transport theory (THERMOS) as thermal spectrum averages over the region i in question (i.e., fuel region or thermal column). The U-238 Cd ratio in the fuel can be written as:

$$CdP^{28}_{core} = \frac{R^{28}}{R^{28} - R^{Cu} \left(1 - \frac{1}{CdR_f^{Cu}} \right) g_f^{28}} \quad (10)$$

Solving equation (7) for $1/(CdR_c^{Cu}-1)$, we get:

$$\frac{1}{CdR_c^{Cu} - 1} = \frac{R^{Cu} - \left(\frac{A_f^{Cu}}{A_c^{Cu}} \right)_{bare}}{\left(\frac{A_f^{Cu}}{A_c^{Cu}} \right)_{bare}} \quad (11)$$

Solving equation (10) for $1/(CdR_f^{28}-1)$, we get:

$$\frac{1}{CdR_f^{28} - 1} = \frac{R^{28} - R^{Cu} \left(1 - \frac{1}{CdR_f^{Cu}} \right) g_f^{28}}{R^{Cu} \left(1 - \frac{1}{CdR_c^{Cu}} \right) g_f^{28}} \quad (12)$$

Substituting equations (11) and (12) into equation (9), we get:

$$\frac{R^{Cu} - \left(\frac{A_f^{Cu}}{A_c^{Cu}} \right)_{bare}}{\left(\frac{A_f^{Cu}}{A_c^{Cu}} \right)_{bare}} = \frac{\left(\frac{1}{CdR_f^{Cu} - 1} \right)}{\left(\frac{1}{CdR_c^{28} - 1} \right) g_c^{28}} \left[\frac{R^{28} - R^{Cu} \left(1 - \frac{1}{CdR_f^{Cu}} \right) g_f^{28}}{R^{Cu} \left(1 - \frac{1}{CdR_f^{Cu}} \right)} \right] \quad (13)$$

All quantities in equation (13) are known except R^{Cu} . Substituting the numerical values for known quantities yield a value for R^{Cu} . Once R^{Cu} is found, the cadmium ratio of U-238 in the core fuel can be determined from equation (10). The copper cadmium ratio in the thermal column can also be determined from equation (7).

B. U-235 Cd Ratio in the Fuel by Extrapolation of Copper Cd Ratio Information

The cadmium ratio of U-235 fission product gamma activity (25 FP) in the fuel is defined by:

$$CdR_f^{25FP} \equiv \frac{[A_f^{25FP}]_{bare}}{[A_f^{25FP}]_{bare} - [A_f^{25FP}]_{sub\ Cd}} \quad (1)$$

Defining equations can be written:

$$\left[\frac{A_f^{25FP}}{A_c^{25FP}} \right]_{bare} \equiv \frac{[A_f^{25FP}]_{sub\ Cd} + [A_f^{25FP}]_{epi\ Cd}}{[A_c^{25FP}]_{sub\ Cd} + [A_c^{25FP}]_{epi\ Cd}} = \frac{[A_f^{25FP}]_{sub\ Cd}}{[A_c^{25FP}]_{sub\ Cd}} \left[\frac{\left\{ \frac{CdR_f^{25FP}}{CdR_f^{25FP} - 1} \right\}}{\left\{ \frac{CdR_c^{25FP}}{CdR_c^{25FP} - 1} \right\}} \right] \quad (2)$$

$$\left[\begin{array}{c} A_{f,Cu} \\ A_{c,Cu} \end{array} \right]_{bare} \equiv \frac{\left[A_{f,Cu} \right]_{sub Cd} + \left[A_{f,Cu} \right]_{epi Cd}}{\left[A_{c,Cu} \right]_{sub Cd} + \left[A_{c,Cu} \right]_{epi Cd}} = \frac{\left[A_{f,Cu} \right]_{sub Cd}}{\left[A_{c,Cu} \right]_{sub Cd}} \left[\begin{array}{c} \left(\frac{CdR_{f,Cu}}{CdR_{f,Cu} - 1} \right) \\ \left(\frac{CdR_{c,Cu}}{CdR_{c,Cu} - 1} \right) \end{array} \right] \quad (3)$$

for U-235 fission product activity ratios and copper foil activity ratios. Sub-cadmium activities are related by the equation:

$$\left[\begin{array}{c} A_{f,25FP} \\ A_{c,25FP} \end{array} \right]_{sub Cd} = \left[\begin{array}{c} A_{f,Cu} \\ A_{c,Cu} \end{array} \right]_{sub Cd} \left[\begin{array}{c} g_{f,25} \\ g_{c,25} \end{array} \right] \quad (4)$$

where,

$$g_{25} = \frac{\frac{\sigma_{-25 fission}}{\sigma_{thermal}}}{\left(\frac{\sigma_{1/v}}{\sigma_{o_{1/v}}} \right)^{\sigma_{25 fission}}}$$

Define N so that

$$N \equiv \left[\begin{array}{c} \left\{ \frac{CdR_{f,25FP}}{CdR_{f,25FP} - 1} \right\} \\ \left\{ \frac{CdR_{c,25FP}}{CdR_{c,25FP} - 1} \right\} \end{array} \right] = \left[\begin{array}{c} A_{f,25FP} \\ A_{c,25FP} \end{array} \right]_{bare} \left[\begin{array}{c} A_{c,Cu} \\ A_{f,Cu} \end{array} \right]_{sub Cd} \left[\begin{array}{c} g_{c,25} \\ g_{f,25} \end{array} \right] \quad (5)$$

Let $X \equiv CdR - 1$ (6)

Then,

$$N = \frac{1 + \frac{1}{X_{f,25}}}{1 + \frac{1}{X_{c,25}}} ; \quad N + \frac{N}{X_{c,25}} = 1 + \frac{1}{X_{f,25}} \quad (7)$$

thus,

$$N-1 = \frac{1}{X_{f,25}} - \frac{N}{X_{c,25}} \quad (8)$$

Taking X_c^{25} as the variable and assuming a linear relationship exists, one may plot

so that $(N-1) = \frac{1}{X_f^{25}}$, when $\frac{1}{X^{Cu}} \rightarrow 0$

Expressed mathematically,

$$\frac{(N_c - 1) - 0}{\frac{1}{X_c^{Cu}} - \frac{1}{X_f^{Cu}}} = \frac{(N_{\text{correct}} - 1) - 0}{0 - \frac{1}{X_f^{Cu}}} = - \frac{\frac{1}{X_f^{25}}}{1/X_f^{Cu}} \quad (10)$$

Reversing the sign and substituting for $(N_c - 1)$ we get:

$$\left(\frac{1}{X_f^{25}} - \frac{N_c}{X_c^{25}} \right) = \frac{1}{X_f^{25}} \quad (11)$$

$$\frac{1}{X_c^{Cu}} + \frac{1}{X_f^{Cu}} = \frac{1}{X_f^{Cu}}$$

Hence,

$$X_f^{25} \equiv \text{CdR}_f^{25\text{FP}} - 1 = \frac{1 - \frac{X_f^{Cu}}{X_c^{Cu}}}{N_c - 1} = \frac{1 - \frac{(\text{CdR} - 1)_f^{Cu}}{(\text{CdR} - 1)_c^{Cu}}}{\left[\frac{A_c^{Cu}}{A_f^{Cu}} \right]_{\text{sub Cd}} \left(\frac{A_f^{25\text{FP}}}{A_c^{25\text{FP}}} \right) \left(\frac{g_c^{25}}{g_f^{25}} \right) - 1} \quad (12)$$

Simplifying the above equation

$$\text{CdR}_f^{25\text{FP}} = 1 + \frac{1 - \frac{(\text{CdR} - 1)_f^{\text{Cu}}}{(\text{CdR} - 1)_c^{\text{Cu}}}}{\left(\frac{A_c^{\text{Cu}}}{A_f^{\text{Cu}}}\right)_{\text{sub Cd}} \left(\frac{A_f^{25\text{FP}}}{A_c^{25\text{FP}}}\right) \left(\frac{g_c^{25}}{g_f^{25}}\right) - 1} \quad (13)$$

From equation (5) we can solve for the cadmium ratio of U-235 in the thermal column.

$$\frac{\text{CdR}_c^{25\text{FP}}}{\text{CdR}_c^{25\text{FP}} - 1} = \frac{\frac{\text{CdR}_f^{25\text{FP}}}{\text{CdR}_f^{25\text{FP}} - 1}}{\left(\frac{A_f^{25\text{FP}}}{A_c^{25\text{FP}}}\right) \left(\frac{A_c^{\text{Cu}}}{A_f^{\text{Cu}}}\right)_{\text{sub Cd}} \left(\frac{g_c^{25}}{g_f^{25}}\right)} \quad (15)$$

$$\text{CdR}_c^{25\text{FP}} = \frac{\left\{ \frac{\text{CdR}_f^{25\text{FP}}}{\text{CdR}_f^{25\text{FP}} - 1} \right\}}{- \left(\frac{A_f^{25\text{FP}}}{A_c^{25\text{FP}}}\right) \left(\frac{A_c^{\text{Cu}}}{A_f^{\text{Cu}}}\right)_{\text{sub Cd}} \left(\frac{g_c^{25}}{g_f^{25}}\right) + \left\{ \frac{\text{CdR}_f^{25\text{FP}}}{\text{CdR}_f^{25\text{FP}} - 1} \right\}} \quad (16)$$

U-235 Fission Product Activity in the Fuel

Since the fuel is only 2.12% U-235, the fission product activity due to U-238 fissions cannot be neglected. Two types of foils were used for the determination of the U-238 fission product activity: (1) the 2.12% U-235 fuel foils, and (2) the 20-fold depleted U-235 foils. The mass normalized fission product activity integral (above the N_p^{239} "pile up" peaks) of each foil is plotted against the U-235 content of the foil for identical decay times. The straight line is extrapolation to zero U-235 content, which is the fission activity due to U-238 fissions alone, A_o^{28FP} . Taking into account that the U-238 surface atom density is less in the fuel than in pure U-238 (a factor at 0.9788 for 2.12 a/o U-235 fuel) $0.9788 A_o^{28FP}$ is subtracted from the fuel fission product activity, at the corresponding decay time, to determine the fission product activity due to U-235 fissions, A^{25FP} . The true U-235 fission product activity was determined for both the fuel foils in the fuel and the fuel foils in the thermal column.

TABLE IX

Cadmium Ratio Data for Wet Base Target Measurements

<u>Quantity</u>	<u>Location</u>	<u>Poisoned Lattice</u>	<u>Unpoisoned Lattice</u>
CdR U^{238} _{capture}	Fuel	1.535 $\pm$.015	1.160 $\pm$.016
CdR U^{238} _{capture}	Thermal Column	5.967 $\pm$.042	5.967 $\pm$.042
CdR U^{235} _{fission}	Fuel	6.791 $\pm$.298	7.967 $\pm$.300
CdR U^{235} _{fission}	Thermal Column	63.10 $\pm$ 2.77	65.25 $\pm$ 2.48
CdR Li^6	Target	5.34 $\pm$.06	5.56 $\pm$.07
CdR _{Cu}	Fuel	4.770 $\pm$.033	5.134 $\pm$.036
CdR _{Cu}	Target	3.596 $\pm$.025	3.710 $\pm$.026
CdR _{Cu}	Thermal Column	36.02 $\pm$.35	34.86 $\pm$.35

D. ^{238}U Capture Activity

To determine the magnitude of the contribution of fission products activity, the J foil fission product spectral data was used. At identical decay times (after reactor scram) the gamma spectra of the J foil and the fuel foil or depleted foil is shown above. Note that a depression is observed in the Np^{239} window when counting a J-foil. This indicates that a small amount of Np^{239} produced in the 93% enriched U^{235} foil (J foil) can be neglected. The quantities F_J and F_u are measured integral fission product activities above the Np^{239} peak region. What is to be determined is f_u , the fission activity in the Np^{239} peak.

If bremsstrahlung radiation and the uranium scattering higher energy gammas into the Np^{239} peak is small, we can write:

$$\frac{f_u}{F_u} = \frac{f_j}{F_j}$$

which can be rewritten as:

$$f_u = \frac{f_j}{F_j} F_u$$

The unknown, f_u , is given in terms of directly measurable quantities. The ratio f_j/F_j is time-dependent. Substitution of $r(t)$ for this ratio yields

$$f_u = r(t)F_u$$

With the magnitude of f_u thus known, the true Np^{239} activity of a fuel foil at decay time, t , can be written as:

$$A^{\text{Np}}(t) = N'(t) - r(t)F_u(t) \quad (1)$$

where, $N'(t)$ = the gross measured activity in the Np^{239} peak region at time (t) minus the activity of background and natural decay.

The resulting true Np^{239} activity was corrected to a common decay time using the known Np^{239} half-life decay.

$$A_o^{28} = \frac{A^{\text{Np}}(t)}{e^{-\frac{0.693}{t_{1/2}} t}} = \frac{A^{\text{Np}}(t)}{e^{-\frac{0.693}{55.92 \text{ hr}} t}} = \frac{A^{\text{Np}}(t)}{e^{-(0.01239 \text{ hr}^{-1})t}} \quad (2)$$

A statistical analysis of the values thus obtained for A_o^{28} was done to determine the standard error in the mean.

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