CP-1084
(A-1475)

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SLOWING DOWN OF FISSION NEUTRONS IN GRAPHITE

E. Fermi, J. Marshall, L. Marshall

November 25, 1943

DECLASSIFIED Per Letter Instructions Of T/D-1055

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Abstract

The slowing down range of fission neutrons from natural uranium in graphite has been measured. Its value reduced to graphite density 1.6 is 35.6 cm. A calculation of the same range based on the Bloch Staub spectrum of fission neutrons and on a calculated relationship between energy and range gives instead $r_0 = 34.3$.

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SECRET

[redacted]

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Slowing Down of Fission Neutrons in Graphite

E. Fermi, J. Marshall, L. Marshall

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A 293 gram cube of Westinghouse metal was placed 121.1 cm below the top of the thermal purification column of the Argonne pile. This column consists of 17 layers of graphite of average density 1.62, each 10.47 cm thick and is built above the top of the pile which has three more layers of dead graphite. Indium foils encased in cadmium boxes 0.102 inches thick were placed at distances approximately 10, 30, 50, 70, 90 and 110 cm from the cube. Since the activity of the 6 foils bombarded simultaneously varied over a range too wide for measuring, 4 foils were activated at a single time with low pile intensity in positions 1, 2, 3, 4 or with high pile intensity in positions 3, 4, 5, 6. The last foil, position 6, was protected by laying three additional layers of graphite on top of the thermal column. For each activation the pile intensity was run up on a 13.9 second period and scrammed at a definite intensity. The kilowatt hours were measured by observing the rise int mperature of a thermocouple in a metal lump in the pile. This method allows successive irradiations to be duplicated quite accurately. If only thermal neutrons were originally present in the column the CdInCd activities would be due exclusively to the fission neutrons produced in the uranium cube. There is actually a background activity which is due to indium resonance neutrons not made thermal by the column. Its value was obtained by duplicating the measurement without the metal cube. The foils were read on two counters, Eeyore and Daleth, whose relative sensitivity is given by $\frac{\text{Eeyore}}{\text{Daleth}} = 1.248$. The activities were reduced to the same units and the background count was subtracted. The net activity was plotted against r^2 and interpolated values obtained. Using these, a numerical integration was made to obtain the mean square distance $\overline{r^2}$ for fission neutrons to become indium resonance neutrons.

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Data

On Counter Daleth CdInCd c/m

Foil position cm (distance from cube)	I	II	III	Background c/m	
10.2	38,241	38,520	37,652	6,030	
29.85	14,750	14,766	14,669	1,150	
50.00	3,319	3,301	3,286	283	
69.8	590	580	[illegible]	88	
Temperature rise of lump in pile	.46°	.46°	.45°	.45°	Average = .455°

On Counter Eeyore CdInCd c/m

Foil position cm (distance from cube)	I	II	III	Background c/m	
50.0		34,008	34,065	2,705	
69.8	5,950	6,006	5,862	807	
90.1	981	956	957	270	
109.8	191	232	218	114	
Temperature rise of lump in pile	3.85°	3.83°	3.85°	3.87°	Average = 3.85°

Interpolated Values

cm	activity net of background A CdInCd (Daleth)	$r^2A \times 10^{-4}$	$r^4A \times 10^{-7}$	r^6A
0	35,100	0	0	
10	31,700	317	32	
20	22,100	884	354	
30	13,100	1179	1061	
40	6,620	1059	1694	
50	2,970	742	1855	
60	1,210	436	1570	
70	480	235	1151	
80	176	113	723	
90	65.3	53	428	
100	24.0	24	240	
110	9.4	11	138	
120	(3.6)	(5)	75	
130	(1.35)	(2)	38	$38 \times 1.69 \times 10^{-4}$
140	(.50)	(1)	19	
150	(.18)		9	
160	(.07)		4	
170	(.025)		2	
			1	2.89×10^{-4}

Calculation of the Range

The column is sufficiently large so that the CdIn activities measured on the axis are not perturbed appreciably by the effect of the sides. We calculate, therefore, the mean square distance from the cube with the formula:

$$\overline{r^2} = \frac{\int_0^\infty Ar^4 dr}{\int_0^\infty Ar^2 dr} = \frac{94,053 \times 10^7}{50,667 \times 10^4} = 1856$$

We have further:

$$r_o^2 = \frac{2}{3}\overline{r^2} = 1237$$

$$r_o = 35.2 \text{ cm}$$

The average stacking density of the graphite used is 1.62. The preceding value of the range can be reduced, therefore, to the normal value of the density 1.6 by multiplying by the factor $\frac{1.62}{1.6}$. This gives the final result

$$r_o = 35.6 \text{ cm}$$

This value of the range corresponds to an age of 317 sq. cm.

Number of Neutrons emitted

The total number of neutrons emitted by the lump during the irradiation can be calculated from the volume integral of the activation measured in the foil at various positions. A calibration of the standard indium foils measured on counter Daleth gives the following relationship between the slowing down density in graphite and the activity A_{Cd} of the foil protected by Cd expressed in counts per minute at saturation

$$q = .00143 \times A_{Cd}$$

For a steady irradiation we could, therefore, obtain the number of neutrons emitted a second by the uranium cube by multiplying the volume integral of A_{Cd} by the factor .00143. Since our irradiation extends over a period of time very short compared to the life of indium, we obtain the total number of neutrons emitted during the burst by multiplying the volume integral of the

initial activity of the foil at various positions by the factor .00143 and by the life time of indium expressed in seconds, namely, 4,674.

The number of neutrons emitted is, therefore,

$$4 \quad A_{Cd}\, r^2 dr \times .00143 \times 4674 =$$

$$= 4 \quad \times 5.067 \times 10^8 \times .00143 \times 4674$$

$$= 4.26 \times 10^{10}$$

The burst irradiation produced a temperature rise of .455°. Since a 1 K.W.H. irradiation produces a temperature rise of 1.6°, the total energy of irradiation amounted to .28 K.W.H. We conclude that the number of neutrons emitted by the lump in an irradiation burst of 1 K.W.H. would have been 1.5×10^{11} or $\frac{1.5 \times 10^{11}}{3600} = 4.2 \times 10^7$ neutrons/second/ k.w. If we assume the fission cross-section of uranium 4.6×10^{-24}, and we assume further that two neutrons per fission are emitted, we find that the cross-section for neutron emission of our lump weighing 293 grams is

$$\frac{293 \times .602 \times 4.6 \times 2}{238} = 6.8 \text{ sq. cm}$$

If we neglect self-absorption, we find, then, the average flux of neutrons at the place of the lump by dividing the number of neutrons emitted per second by the cross-section of the lump, namely,

$$nv = \frac{4.2 \times 10^7}{6.8} = 6.2 \times 10^6 \text{ per k.w.}$$

This value of the flux is in agreement with values obtained by more direct estimates.

Comparison with the Theory

The range of the fission neutrons in graphite could be calculated if the energy distribution of fission neutrons and the relationship between the slowing down range and energy of the neutrons were known. We denote by $\overline{r^2}(E)$ the mean

square of the distance at which neutrons range emitted with energy E attain indium resonance energy by slowing down in graphite. Let, further, $\pi(E)dE$ give the number of fission neutrons emitted in the energy interval dE. We have, then,

$$\overline{r^2} = \int \overline{r^2}(E)\, \pi(E)dE \Big/ \int \pi(E)\, dE$$

This expression has been calculated numerically by using for $\pi(E)$ the energy spectrum measured by Bloch and Staub in (LA-17). The function $\overline{r^2}(E)$ has been recalculated using the carbon scattering cross-section measured by Williams. The following formula was used:

$$\overline{r^2}(E) = 2\lambda^2(E) + 2\lambda^2(1.44ev) + \frac{2}{.158 \times \frac{17}{18}} \int_{1.44eV}^{E} \lambda^2 \frac{dE}{E}$$

The values for these two functions are given in the table, where the first column gives the energy expressed in MeV. The second column gives function $\overline{r^2}(E)$, and the third column gives $\pi(E)$. This last column is taken from the data of Bloch and Staub up to the energy 3.2 MeV. The values beyond 3.2Mev were extrapolated.

By a numerical integration we find

$$\overline{r^2} = \frac{\int \overline{r^2}(E)\, \pi(E)dE}{\int \pi(E)\, dE} = \frac{3912}{2.213} = 1768$$

corresponding to a range

$$r_o = \sqrt{\frac{2}{3}\, 1768} \doteq 34.3$$

This value is somewhat smaller than the experimental value $r_o = 35.6$.

Table

E	$r^2(E)$	τ	$\pi(E)$
.1	1061	176.8	.45
.2	1157	192.8	.60
.3	1230	205.0	.73
.4	1290	215.0	.84
.5	1346	224.3	.92
.6	1398	233.0	.97
.7	1447	241.2	1.01
.8	1491	248.5	1.03
.9	1539	256.5	1.02
1.0	1580	263.3	1.00
1.2	1665	277.5	.93
1.4	1748	291.3	.84
1.6	1827	304.5	.75
1.8	1917	319.5	.65
2.0	2008	334.7	.56
2.2	2090	348.3	.47
2.4	2166	361.0	.40
2.6	2236	372.6	.33
2.8	2297	382.8	.28
3.0	2350	391.7	.23
3.2	2379	396.5	.19
3.4	2387	397.8	(.16)
3.6	2390	398.3	(.13)
3.8	2415	402.5	(.11)
4.0	2469	411.5	(.09)
4.2	2480	413.3	(.08)
4.4	2503	417.2	(.06)
4.6	2564	427.3	(.05)
4.8	2644	440.7	(.04)
5.0	2743	457.2	(.04)
5.2	2834	472.3	(.03)
5.4	2917	486.2	(.03)
5.6	3005	500.8	(.02)
5.8	3085	514.2	(.02)
6.0	3162	527.0	(.01)

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