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A PROBLEM IN THE CONVERSION OF NEUTRON DOSE TO FLUX

D. E. McMillan

June 29, 1953

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192 002

3

A PROBLEM IN THE CONVERSION OF NEUTRON DOSE TO FLUX

D. E. McMillan

ABSTRACT

A calculation is made of the flux entering the ORNL
Lid Tank and the water at the north face of the ORNL Bulk
Shielding Reactor. The values found by converting Dosimeter
measurements to flux are 3.51×10^6 neut/cm² sec and
 $9.01 \times 10^5 \frac{\text{neuts/cm}^2 \text{ sec}}{\text{watt}}$ respectively as compared with
 35.4×10^6 neut/cm² sec and $84.1 \times 10^5 \frac{\text{neuts/cm}^2 \text{ sec}}{\text{watt}}$
calculated from power leakage considerations. The un-
certainties in the calculations are such that the apparent
differences may be in error by a factor of 2 or 3.

100 224

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192 005

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I. Introduction

The problem of converting neutron dose to flux and vice versa is one that constantly arises in shielding calculations. Since most flux calculations are made for complex shielding configurations, it is of interest to perform such a calculation for a relatively simple configuration for which experimental data are available.

Thus, as a test of the methods used in the conversion of dose to flux, a calculation is made of the flux entering the ORNL Lid Tank and the water next to the north face of the ORNL Bulk Shielding Reactor. This is done in two ways. First, a conversion factor is calculated and used to convert the measured dose to flux and then the power leakage is converted to flux by assuming an energy release of 200 Mev/fission and an average production of 2.5 neutrons/fission.

The method of calculation is described in Section II and the application to the Lid Tank and Bulk Shielding Facility is made in Sections III and IV.

No similar comparison has been found in the literature.

192 006

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II. Calculation of Conversion Factors

The first step is to calculate the total flux which should be observed at the surface of a fission source when the power leakage is given and the assumption is made that the emergent neutrons are distributed in a fission spectrum. If we let this leakage in watts/cm² be P_L , then the number of fissions/sec taking place effectively on the surface is $3.12 \times 10^{10} \frac{\text{fissions/sec}}{\text{watt}}$ $P_L \frac{\text{watts}}{\text{cm}^2}$ where the constant factor embodies an energy release of

200 Mev/fission¹. Then using an average value of 2.5 neutrons of all energies per fission, the total neutron flux² at the interface (ϕ_0) is found to be:

$$\begin{aligned}\phi_0 &= 3.12 \times 10^{10} P_L \frac{\text{fissions}}{\text{cm}^2 \text{ sec}} 2.5 \text{ neutrons/fission} \\ &= 7.80 \times 10^{10} P_L \text{ neutrons/cm}^2 \text{ sec} \quad (1)\end{aligned}$$

The next problem is to calculate the appropriate factor for conversion of neutron dose to flux allowing for the fact that the conversion factor is a function of the neutron energy. The flux is again assumed to have energies distributed as in the fission spectrum. To do this, we first note that the fission spectrum may be approximated over the energy range from .1 to 10 Mev by 2, 3:

$$\begin{aligned}N(E) &= \sqrt{\frac{2}{\pi E}} e^{-E} \sinh \sqrt{2E} \\ &= .484 e^{-E} \sinh \sqrt{2E}\end{aligned}$$

where $N(E) dE$ is the fraction of all fission neutrons which will have energies in the range from E to $E + dE$. Now it is slightly more convenient to calculate the conversion factor from flux to dose than it is to calculate the inverse quantity so this will be done below and then the reciprocal will be taken to convert from dose to flux.

Let the conversion factor from flux to dose be $C(E) \frac{\text{mrep/hr}}{\text{neutrons/cm}^2 \text{ sec}}$.

Then, since $\phi_0 N(E) dE$ is the total number of neutrons having energies in the range E to $E + dE$, the dose from these neutrons is given by:

* Note that in the particular cases of the Lid Tank and B.S.F., there is some question as to whether this power leakage calculation leads to a neutron current instead of flux.

$$\phi_0 C(E) N(E) dE.$$

Now if we have a fast neutron dosimeter which is biased such that no neutrons with energies less than E_0 are detected, then we wish to calculate the reading of this instrument when placed in a flux with fission energies. The average value of $C(E)$ over the energy range from E_0 to ∞ is given by:

$$\overline{C(E)}_{E_0 \leftrightarrow \infty} = \frac{\int_{E_0}^{\infty} C(E) N(E) dE}{\int_{E_0}^{\infty} N(E) dE} \quad \text{and the}$$

dose due to neutrons with energies in this range is:

$$D_{E_0 \leftrightarrow \infty} = \overline{C(E)}_{E_0 \leftrightarrow \infty} \phi_0 \times \int_{E_0}^{\infty} N(E) dE$$

or

$$D_{E_0 \leftrightarrow \infty} = \frac{\phi_0 \int_{E_0}^{\infty} C(E) N(E) dE}{\int_{E_0}^{\infty} N(E) dE} \int_{E_0}^{\infty} N(E) dE$$

$$D_{E_0 \leftrightarrow \infty} = \phi_0 \overline{C(E)}_{E_0 \leftrightarrow \infty} \quad (2)$$

where

$$\overline{C(E)}_{E_0 \leftrightarrow \infty} \equiv \frac{\int_{E_0}^{\infty} C(E) N(E) dE}{\int_{E_0}^{\infty} N(E) dE}$$

192 008

TABLE I

Neutron Energy	Neutron Fluxes Biologically Equivalent to 0.3 rep of Gammas per 40-hr Week	C(E) <u>mrep/hr</u> neut/cm ² sec
Thermal (.025 x 10 ⁻⁶ Mev)	1800 neut/cm ² sec	8.26 x 10 ⁻⁴
5 x 10 ⁻³ Mev	1640	9.14 x 10 ⁻⁴
.5 Mev	82	9.15 x 10 ⁻³
2.5 Mev	38	1.972 x 10 ⁻²
5 Mev	26	2.88 x 10 ⁻²
10 Mev	26	2.88 x 10 ⁻²

Now C(E) is obtained from measurements made by W. S. Snyder on neutron fluxes that are biologically equivalent to 0.3 rep of gamma rays per 40-hr week (See Table I). These data are taken from Reference 4*. To find C(E), first note that a dose of .3 rep of gammas/40 hrs = 7.5 mrep/hr of gammas and that a rep of fast neutrons is 10 times larger and a rep of thermal neutrons is five times larger than a rep of gammas. Thus, to calculate C(E), we must divide .75 mrep/hr by Snyder's fast flux data and divide 1.5 mrep/hr by Snyder's thermal flux data. Thus, the factor 1.5 will be used for Snyder's data at energies of 5 kv and thermal while .75 will be used for the energies of .5 Mev and higher. The data in Table I are plotted in Figure 1 and joined by straight lines. The integrations for C(E)[†] are then performed numerically (See Appendix I). The results are as follows:

$$\overline{C(E)}_{.1 \rightarrow \infty}^{\dagger} = 164.8 \times 10^{-4} \frac{\text{mrep/hr}}{\text{neut/cm}^2 \text{ sec}}$$

or

$$\frac{1}{\overline{C(E)}_{.1 \rightarrow \infty}^{\dagger}} = 60.7 \frac{\text{neut/cm}^2 \text{ sec}}{\text{mrep/hr}}$$

$$\overline{C(E)}_{.5 \rightarrow \infty}^{\dagger} = 156.6 \times 10^{-4} \frac{\text{mrep/hr}}{\text{neut/cm}^2 \text{ sec}}$$

*Blizard points out that Snyder's data are really for a collimated beam of neutrons incident normally on a 30 cm thick slab of meat and that radiation from other angles may alter the result. However, this is the best data available and will be used here.

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$C(E)$ vs. E

FIG 1.

or

$$\frac{1}{C(E)^{1.5 \rightarrow \infty}} = 63.9 \frac{\text{neut/cm}^2 \text{ sec}}{\text{mrep/hr}}$$

$$\overline{C(E)^{1.9 \rightarrow \infty}} = 97.4 \times 10^{-4} \frac{\text{mrep/hr}}{\text{neut/cm}^2 \text{ sec}}$$

or

$$\frac{1}{C(E)^{1.9 \rightarrow \infty}} = 102.6 \frac{\text{neut/cm}^2 \text{ sec}}{\text{mrep/hr}}$$

where it is to be remembered that $D_{E_0 \rightarrow \infty} = \phi_0 \overline{C(E)^{1.9 \rightarrow \infty}}$ and ϕ_0 is the total neutron flux.

Other conversion factors can be found in the literature, an example of which⁷ is the value of $4.0 \times 10^{-6} \frac{\text{mrep/sec}}{\text{neut/cm}^2 \text{ sec}}$ which is equivalent to

$$1.44 \times 10^{-2} \frac{\text{mrep/hr}}{\text{neut/cm}^2 \text{ sec}} \text{ or } 69.5 \frac{\text{neut/cm}^2 \text{ sec}}{\text{mrep/hr}}.$$

This is for a polonium-beryllium source and so quite reasonably is somewhat different from those given above. However, it is a good check on the integrations carried out in this paper.

III. Application to the ORNL Lid Tank

There are many references which state that the power of the Lid Tank source plate is 6 watts under ordinary operating conditions and that there is an attenuation factor of 0.6 to be assigned to the several layers of material which are permanently in front of the source plate. However, different choices of effective source area have been made for the purpose of leakage calculations. Reference 5 divides by twice the area of one face of the source plate to obtain a value of 4.54×10^{-4} watts/cm² as the leakage into the lid tank while Blizard and Walton (Reference 1, p. 26) divide by 3970 cm² or the area of one face. The actual configuration is as follows: uranium cylinders which make up the source plate are approximately 2.7 cm in diameter; and interposed between these cylinders and the inside of the lid tank wall are 0.65 cm of Masonite, 0.238 cm of aluminum, 0.318 cm of B₂C powder in lacquer, and 1.27 cm of lead. Thus, taking approximate values for the removal cross sections and converting the uranium cylinders into an equivalent slab of equal volume yields an attenuation factor of approximately 0.6 (See Appendix III). Assuming that one-half the energy emerges from each face of the source plate assembly, the leakage is obtained by dividing the product of 6 watts and 0.6 by twice the area. Thus, the value of 4.54×10^{-4} watts/cm² will be used below.

Recent dosimeter data⁸ are given in Table II and Figure 2. At $Z = 0$, the

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⁸Private communication to H. E. Stone from G. T. Chapman, dated 11/16/52. Note that from 40 cm on out, these data are in agreement with published data in Reference 4.

TABLE II

DATA FOR LOW POWER RUN OF FAST NEUTRON
CURVE TO Z = 0 FOR ORNL LID TANKS*

<u>Z(cm)</u>	<u>Pile Power (KW)</u>	<u>Dose (mrep/hr) For 6W Source Plate</u>	<u>Z(cm)</u>	<u>Pile Power (KW)</u>	<u>Dose (mrep/hr)</u>
0	10	5.49×10^4	30	4000	2.52×10^2
2.5	10	3.88×10^4	30	2000	3.40×10^2
4.9	*	2.70×10^4	30	100	7.12×10^2
5.0	*	2.74×10^4	30	10	4.33×10^2
5.0	10	2.47×10^4	30	10	4.29×10^2
10	100	1.45×10^4	40	4000	8.54×10^1
10	10	1.16×10^4	40	10	9.82×10^1
15	10	5.40×10^3	50	4000	2.66×10^1
20	100	3.40×10^3	60	4000	7.24×10^0
20	10	2.16×10^3	70	4000	1.99×10^0
			80	4000	7.15×10^{-1}

* One of these was at 100 KW though the data are ambiguous as to which one.

** Private communication to H. E. Stone from G. T. Chapman dated 11/16/52.

192-013

dose is given as 5.49×10^4 mrep/hr. Assuming the dosimeter is sensitive to neutrons having energies of 0.5 Mev or greater, the proper conversion factor from dose to total neutron flux is $= \frac{1}{C(E) \cdot .5 \rightarrow \infty} = 63.9 \frac{\text{neuts/cm}^2 \text{ sec.}}{\text{mrep/hr}}$.

This gives:

$$\phi_0 = 5.49 \times 10^4 \times 63.9 = 3.51 \times 10^6 \text{ neut/cm}^2 \text{ sec.}$$

However, use of the source power relationship (Equation I) gives:

$$\phi_0 = 7.80 \times 10^{10} \times 4.54 \times 10^{-4} = 35.4 \times 10^6 \text{ neut/cm}^2 \text{ sec.}$$

This is an order of magnitude difference.

As another comparison, if the value of $\frac{1}{C(E) \cdot .1 \rightarrow \infty} = 60.7$ is used, the result is:

$$\phi_0 = 5.49 \times 10^4 \times 60.7 = 3.34 \times 10^6 \text{ neut/cm}^2 \text{ sec.}$$

It should be mentioned that the threshold of the dosimeters used in these measurements is not known exactly. Reference 8 mentions a value of .2 Mev for the dosimeters used in the B.S.F., so possibly the latter conversion made above is closer to the truth than the first.

Of all these values, the higher ones seem to be more consistent with those used by other workers in the past; an example of which is the value 28×10^6 fast neut/cm² sec as quoted in Reference 6. Of course, this value should be slightly lower than the ones calculated in this paper as not all the neutrons in the fission spectrum are fast (have energies greater than 0.5 Mev) though approximately 86.8% of them do (See Appendix II). Thus multiplying the value of $\phi_0 = 3.54 \times 10^7$ by this fraction, we get:

$\phi(E > .5 \text{ Mev}) = 30.7 \times 10^6 \text{ neut/cm}^2 \text{ sec}$ which agrees fairly well with Taylor's value.

It is realized that many simplifying assumptions are involved in the above calculations. For example, the shape of the neutron spectrum as it enters the lid tank is assumed to be the same as the birth spectrum which is certainly not exactly true. Furthermore, no attempt has been made to deal with the uneven power distribution which actually exists over the source plate. Thus, a discrepancy of a factor of 2 or 3 in the results would not be too surprising; but a factor of 10 is not explained on these grounds alone.

The calculation most similar to the above comparison that was found in the literature is that presented in ORNL-1133¹. However, the primary purpose of the latter was the comparison of lid tank data with B.S.F. results, and most of the work is applicable relatively far out in the water. Furthermore, the

experimental results were used to calculate a point-to-point attenuation kernel $G(2)$ which was in turn multiplied by $4\pi r^2$ to transform the result to an infinite plane source. Thus, the results do not lend themselves to predicting what the results would be at the interface.

IV. Application to ORNL Bulk Shielding Facility

Hungerford and Meem⁹ have calculated the leakage from the north face of the B.S.F. reactor to be 7.94×10^{-5} watts/cm². This calculation is based on data published in Reference 10 for the pure water case and a nominal reactor power level of one watt. However, as the data shows, the actual reactor power was .7378 watts so normalized to one watt the leakage is 1.08×10^{-4} watts/cm². This leakage was calculated on the basis of neutron attenuations so it is reasonable to use this value in conjunction with Equation 1 to get the total neutron flux at the interface. Thus

$$\phi_0 = 7.80 \times 10^{10} \times 1.08 \times 10^{-4} = 84.1 \times 10^5 \frac{\text{neutrons/cm}^2 \text{ sec}}{\text{watt}}$$

where the power unit in the denominator refers to the total reactor power.

A check on this value can be obtained by using the method indicated on page 27 of Reference 1. By this procedure, the surface strength is taken as being equal to the power at the center of the north face times the relaxation length for the core. Taking these values as 1.88×10^{-5} watt/cm³ (1) and 9.7 cm¹⁰ respectively gives a source strength of 1.82×10^{-4} $\frac{\text{watts/cm}^2 \text{ sec}}{\text{watt}}$. In terms of neutron flux, this is equivalent to $142 \times 10^5 \frac{\text{neutrons/cm}^2 \text{ sec}}{\text{watt}}$.

Fast dosimeter data for the B.S.F. in pure water is given in Reference 4, the graphical value at the interface being approximately 1.7×10^4 $\frac{\text{mrep/hr.}}{\text{watt}}$.

Now the B.S.F. reactor is definitely a volume distributed source so the neutron spectrum at its north face will probably differ markedly from that of the birth fission spectrum. Therefore, this is not as direct a comparison as was possible for the lid tank. However, if we assume a fission spectrum for the sake of comparison, we find that using the conversion factor $\frac{1}{0.5 \times 10^{-5}}$ this dose would

be equivalent to a ϕ_0 of $10.86 \times 10^5 \frac{\text{neutrons/cm}^2 \text{ sec}}{\text{watt}}$. Depending upon which

of the surface source strength values you prefer, there is a factor of 8 to 13 between the flux values calculated from power considerations and those resulting from converting dosimeter measurements.

It should be mentioned that some correction factors for B.S.F. data have recently been published¹¹. Unfortunately, these are in the wrong direction to explain the results obtained above since the fast neutron data should be

¹¹ Note that if an average power density of 1.13×10^{-5} watts/cm³ as reported in Reference 4 is used, the surface source strength by this procedure becomes 1.095×10^{-4} $\frac{\text{watts/cm}^2 \text{ sec}}{\text{watt}}$ which is a closer check to the first value.

decreased by a factor of 0.8306 and/or the reactor power should be increased by the reciprocal of this factor, 1.204. The values of flux at the interface by the leakage calculations are not affected since the power correction would enter into both the numerator and the denominator. However, the corresponding value from the dosimeter data becomes:

$$\begin{aligned} & 0.8306 \times 1.086 \times 10^6 \frac{\text{neutrons/cm}^2 \text{ sec}}{\text{watt}} \\ & = 9.01 \times 10^5 \frac{\text{neutrons/cm}^2 \text{ sec}}{\text{watt}} \end{aligned}$$

Again we have very nearly an order of magnitude difference between this number and the value $84.1 \times 10^5 \frac{\text{neutrons/cm}^2 \text{ sec}}{\text{watt}}$ obtained above by what is probably the best of the leakage values.

In conclusion the discrepancy between the power measurement calculation of flux and the dosimeter measurements is one that requires further investigation.

APPENDIX I

$E(\text{MeV})$	e^{-E}	$\sqrt{2E}$	$\sinh \sqrt{2E}$	$N(E) = \frac{1}{484} e^{-E} \sinh \sqrt{2E}$	$C(E)$	$C(E) N(E)$
.1	.9048	.4472	.4622	2.02×10^{-1}	3.95×10^{-3}	7.97×10^{-4}
.2	.8187	.6324	.6754	$2.68 \times "$	$5.6 \times "$	$15.0 \times "$
.3	.7408	.7746	.8544	$3.06 \times "$	$6.8 \times "$	$20.8 \times "$
.4	.6703	.8944	1.018	$3.30 \times "$	$7.9 \times "$	$26.1 \times "$
.5	.6065	1.000	1.175	$3.44 \times "$	$8.9 \times "$	$30.6 \times "$
.6	.5488	1.095	1.327	$3.53 \times "$	$9.8 \times "$	$34.6 \times "$
.7	.4966	1.183	1.479	$3.56 \times "$	1.06×10^{-2}	$37.8 \times "$
.8	.4493	1.265	1.630	$3.54 \times "$	$1.12 \times "$	$39.6 \times "$
.9	.4066	1.342	1.783	$3.52 \times "$	$1.20 \times "$	$42.2 \times "$
1.0	.3679	1.414	1.935	$3.45 \times "$	$1.28 \times "$	$44.1 \times "$
1.1	.3329	1.483	2.090	$3.37 \times "$	$1.32 \times "$	$44.5 \times "$
1.2	.3012	1.549	2.247	$3.28 \times "$	$1.39 \times "$	$43.0 \times "$
1.3	.2725	1.612	2.407	$3.18 \times "$	$1.43 \times "$	$43.9 \times "$
1.4	.2466	1.673	2.570	$3.07 \times "$	$1.50 \times "$	$46.0 \times "$
1.5	.2231	1.732	2.738	$2.96 \times "$	$1.56 \times "$	$46.2 \times "$
1.6	.2019	1.789	2.908	$2.84 \times "$	$1.60 \times "$	$45.4 \times "$
1.7	.1827	1.844	3.082	$2.72 \times "$	$1.66 \times "$	$45.1 \times "$
1.8	.1653	1.897	3.258	$2.61 \times "$	$1.70 \times "$	$44.4 \times "$
1.9	.1496	1.949	3.440	$2.49 \times "$	$1.75 \times "$	$43.6 \times "$
2.0	.1353	2.000	3.627	$2.38 \times "$	$1.80 \times "$	$42.8 \times "$
2.1	.1225	2.049	3.816	$2.26 \times "$	$1.84 \times "$	$41.6 \times "$
2.2	.1108	2.098	4.014	$2.15 \times "$	$1.89 \times "$	$40.6 \times "$
2.3	.1003	2.145	4.212	$2.04 \times "$	$1.93 \times "$	$39.4 \times "$
2.4	.09072	2.191	4.416	$1.94 \times "$	$1.97 \times "$	$38.2 \times "$
2.5	.08209	2.236	4.624	$1.84 \times "$	$2.00 \times "$	$36.8 \times "$
2.6	.07427	2.280	4.837	$1.74 \times "$	$2.08 \times "$	$36.2 \times "$
2.7	.06721	2.324	5.059	$1.65 \times "$	$2.12 \times "$	$35.0 \times "$
2.8	.06081	2.366	5.280	$1.55 \times "$	$2.18 \times "$	$33.8 \times "$
2.9	.05502	2.408	5.511	$1.47 \times "$	$2.22 \times "$	$32.6 \times "$
3.0	.04979	2.449	5.745	$1.385 \times "$	$2.27 \times "$	$31.4 \times "$
3.1	.04505	2.490	5.989	$1.305 \times "$	$2.30 \times "$	$30.0 \times "$
3.3	.03688	2.569	6.488	$1.160 \times "$	$2.40 \times "$	$27.9 \times "$
3.5	.03020	2.646	7.013	$1.024 \times "$	$2.48 \times "$	$25.4 \times "$
3.7	.02472	2.720	7.557	9.04×10^{-2}	$2.53 \times "$	$22.9 \times "$
3.9	.02024	2.793	8.134	$7.96 \times "$	$2.60 \times "$	$20.7 \times "$
4.1	.01657	2.864	8.737	$7.00 \times "$	$2.66 \times "$	$18.6 \times "$
4.3	.01357	2.933	9.365	$6.15 \times "$	$2.70 \times "$	$16.6 \times "$
4.5	.01111	3.000	10.02	$5.39 \times "$	$2.74 \times "$	$14.8 \times "$
4.7	.009095	3.066	10.70	$4.71 \times "$	$2.79 \times "$	$13.1 \times "$
4.9	.007447	3.130	11.42	$4.12 \times "$	$2.84 \times "$	$11.7 \times "$
5.1	.006097	3.194	12.17	$3.59 \times "$	$2.88 \times "$	$10.3 \times "$
5.3	.004992	3.256	12.95	$3.125 \times "$	"	$9.00 \times "$
5.5	.004087	3.317	13.77	$2.72 \times "$	"	$7.84 \times "$
5.7	.003346	3.376	14.61	$2.37 \times "$	"	$6.83 \times "$
5.9	.002739	3.435	15.50	$2.055 \times "$	"	$5.92 \times "$
6.1	.002243	3.493	16.43	$1.785 \times "$	"	$5.14 \times "$

APPENDIX I
(cont'd)

<u>E(Mev)</u>	<u>e^{-E}</u>	<u>$\sqrt{2E}$</u>	<u>$\sinh \sqrt{2E}$</u>	<u>$N(E) = \frac{.484 e^{-E}}{\sinh \sqrt{2E}}$</u>	<u>C(E)</u>	<u>C(E) N(E)</u>
6.3	.001836	3.550	17.41	1.55×10^{-2}	2.88×10^{-2}	4.46×10^{-4}
6.5	.001503	3.606	18.40	$1.34 \times "$	"	$3.86 \times "$
6.7	.001231	3.661	19.44	$1.16 \times "$	"	$3.34 \times "$
7.0	.0009119	3.742	21.08	9.30×10^{-3}	"	$2.66 \times "$
7.3	.0006755	3.821	22.82	$7.47 \times "$	"	$2.14 \times "$
7.6	.0005005	3.899	24.67	$5.97 \times "$	"	$1.71 \times "$
7.9	.0003707	3.975	26.62	$4.78 \times "$	"	$1.37 \times "$
8.2	.0002747	4.050	28.69	$3.82 \times "$	"	$1.10 \times "$
8.5	.0002035	4.123	30.86	$3.04 \times "$	"	$.875 \times "$
8.8	.0001507	4.195	33.17	$2.42 \times "$	"	$.696 \times "$
9.1	.0001117	4.266	35.61	$1.922 \times "$	"	$.554 \times "$
9.4	.0000827	4.336	38.16	$1.529 \times "$	"	$.440 \times "$
9.7	.0000613	4.405	40.93	$1.213 \times "$	"	$.349 \times "$
10.0	.0000454	4.472	43.76	9.61×10^{-4}	"	$.277 \times "$
10.3	.0000336	4.539	46.79	$7.61 \times "$	"	$.219 \times "$
12.3	4.55×10^{-6}	4.960	71.3	$1.57 \times "$	"	$.0452 \times "$
14.3	6.16×10^{-7}	5.349	105.2	3.14×10^{-5}	"	$.00905 \times 10^{-4}$
16.3	8.34×10^{-8}	5.710	151.0	6.09×10^{-6}	"	$.00175 \times "$
18.3	$1.129 \times "$	6.050	212.1	$1.159 \times "$	"	$.000334 \times "$

By Simpson's Rule:

$$\text{Area} = \frac{h}{3} (f_0 + f_{2n} + 4 \sum f_{\text{odd}} + 2 \sum f_{\text{even}})$$

$$\begin{aligned} \int_{.1 \text{ Mev}}^{3.1 \text{ Mev}} C(E) N(E) dE &= \frac{.1}{3} \left[7.97 + 30.0 + 4 (561.2) + 2(540.1) \right] 10^{-4} \\ &= \frac{.1}{3} [3363.0] 10^{-4} \\ &= 112.1 \times 10^{-4} \end{aligned}$$

$$\begin{aligned} \int_{3.1 \text{ Mev}}^{6.7 \text{ Mev}} C(E) N(E) dE &= \frac{.2}{3} \left[30.0 + 3.34 + 4 (120.73) + 2 (104.32) \right] 10^{-4} \\ &= \frac{.2}{3} [724.9] 10^{-4} = 48.3 \times 10^{-4} \end{aligned}$$

$$\begin{aligned} \int_{6.7}^{10.3} C(E) N(E) dE &= \frac{.3}{3} \left[3.34 + .219 + 4 (6.883) + 2 (5.288) \right] 10^{-4} \\ &= \frac{.3}{3} [41.67] 10^{-4} = 4.17 \times 10^{-4} \end{aligned}$$

$$\begin{aligned} \int_{10.3}^{18.3} C(E) N(E) dE &= \frac{2.0}{3} \left[.219 + .000334 + 4 (.0470) + 2(.00905) \right] 10^{-4} \\ &= \frac{2.0}{3} [.425] 10^{-4} = .283 \times 10^{-4} \end{aligned}$$

* Note that the integral over the interval from 10.3 Mev to 18.3 Mev is so small that we can neglect the value of the integral from 18.3 Mev to ∞ .

THUS :

$$\overline{C(E)}_{.1 \leftrightarrow \infty} = \int_{.1}^{18.3} C(E) N(E) dE = [112.1 + 48.3 + 4.17 + .28] 10^{-4}$$

$$\overline{C(E)}_{.1 \leftrightarrow \infty} = 164.8 \times 10^{-4} \frac{\text{milli rep/hr}}{\text{neut/cm}^2 \text{ sec}}$$

or

$$\frac{1}{\overline{C(E)}_{.1 \leftrightarrow \infty}} = 60.7 \frac{\text{neut/cm}^2 \text{ sec}}{\text{milli rep/hr}}$$

Now if we just take an average of C(E) from 1/2 Mev on up, we get:

$$\int_{.5}^{3.1} C(E) N(E) dE = \frac{1}{3} [30.6 + 30 + 4 (520.1) + 2 (488.7)] 10^{-4}$$

$$= \frac{1}{3} [3118.4] 10^{-4} = 103.9 \times 10^{-4}$$

So

$$\int_{.5 \text{ Mev}}^{18.3 \text{ Mev}} C(E) N(E) dE = [103.9 + 48.3 + 4.17 + .28] 10^{-4} = 156.6 \times 10^{-4} \frac{\text{milli rep/hr}}{\text{neut/cm}^2 \text{ sec}}$$

or

$$\frac{1}{\overline{C(E)}_{.5 \leftrightarrow \infty}} = 63.9 \frac{\text{neut/cm}^2 \text{ sec}}{\text{milli rep/hr}}$$

If interested in the range from 1.9 Mev to ∞ get:

$$\int_{1.9}^{3.1} C(E) N(E) dE = \frac{.1}{3} [43.6 + 30 + 4 (223.0) + 2 (185.4)] 10^{-4}$$

$$= \frac{.1}{3} [1336.4] 10^{-4} = 44.8 \times 10^{-4}$$

$$\int_{1.9}^{18.3} C(E) N(E) dE = [44.6 + 48.3 + 4.17 + .28] 10^{-4} = 97.4 \times 10^{-4} \frac{\text{milli rep/hr}}{\text{neuts/cm}^2 \text{ sec}}$$

or

$$\frac{1}{C(E)}_{1.9 \rightarrow \infty} = 102.6 \frac{\text{neuts/cm}^2 \text{ sec}}{\text{milli rep/hr}}$$

APPENDIX II

A check to see how closely the energy range from .1 to 18.3 Mev comes to including all the neutrons in a fission spectrum is made below by evaluating

$$\int_{.1}^{18.3} N(E) dE$$

$$\int_{.1}^{3.1} N(E) dE = \frac{.1}{3} [2.02 + 1.305 + 4(39.44) + 2(37.56)] 10^{-1}$$

$$= \frac{.01}{3} [236.2] = .787$$

$$\int_{3.1}^{6.7} = \frac{.2}{3} [1.30 + .116 + 4(4.58) + 2(3.90)] 10^{-1}$$

$$= .066 [27.54] 10^{-1} = .1836$$

$$\int_{6.7}^{10.3} = \frac{.3}{3} [11.6 + .761 + 4(24.0) + 2(18.4)] 10^{-3}$$

$$= .1 [115.2] 10^{-3} = .0115$$

$$\int_{10.3}^{18.3} = \frac{1}{3} [7.61 + .0116 + 4(1.63) + 2(.314)] 10^{-4}$$

$$= \frac{1}{3} [14.77] 10^{-4} = .000492$$

Thus

$$\int_{.1 \text{ Mev}}^{18.3} N(E) dE = .787 + .1836 + .0145 + .0005$$

$$= .986$$

or 98.6% of the neutrons have energies in the range from .1 to 18.3 Mev.

The fraction of fission neutrons with energies greater than .5 Mev is calculated below.

$$\begin{aligned} \int_{.5}^{3.1} N(E) dE &= \frac{1}{3} [3.44 + 1.31 + 4(33.46) + 2(31.06)] 10^{-1} \\ &= \frac{.01}{3} [200.71] \\ &= .669 \end{aligned}$$

Using results for first integration of this section get

$$\begin{aligned} \int_{.5}^{18.3} N(E) dE &= .669 + .1836 + .0145 + .0005 \\ &= .868 \end{aligned}$$

APPENDIX III

A rough approximation is made below of the attenuation suffered by the neutron flux coming from the lid tank source plate (For a diagram of the source plate see Page 52 of Reference 1).

If d is the diameter of the uranium slugs which make up the source plate and t is the thickness of a volume equivalent slab, then

$$td = \frac{\pi d^2}{4}$$

or

$$t = \frac{\pi d}{4} = \frac{3.14 \times 2.7 \text{ cm}}{4} = 2.12 \text{ cm}$$

or

$$t/2 = 1.06 \text{ cm}$$

The necessary macroscopic removal cross sections are:

$$\begin{aligned} \sum_{B_4C} &= .0883/\text{cm} \quad (\text{Reference 5}) \\ \sum_U &= .2/\text{cm} \quad (\text{Reference 9}) \\ \sum_{\text{Masonite}} \approx \sum_{H_2O} &= .115/\text{cm} \quad (\text{just a guess}) \\ \sum_{Al} &= .075/\text{cm} \quad (\text{Reference 12}) \\ \sum_{Pb} &= .112/\text{cm} \quad (\text{Reference 3}) \end{aligned}$$

and using these, we get:

$$\begin{aligned} \sum x_{B_4C} &= .0883/\text{cm} \times .313 \text{ cm} = .0281 \\ \sum x_U &= .20/\text{cm} \times 1.06 \text{ cm} = .212 \\ \sum x_{Al} &= .075/\text{cm} \times .238 \text{ cm} = .0178 \end{aligned}$$

$$\sum \kappa_{\text{Pd}} = .112/\text{cm} \times 1.27 \text{ cm} = .142$$

$$\sum \kappa_{\text{Masonite}} = .115/\text{cm} \times .65 \text{ cm} = .075$$

$$.475 = \sum_1 \sum_1 \kappa_1$$

Thus, the neutron attenuation $\approx e^{-.475} = .622$

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