Measurements of Fission Neutron Spectra Transmitted through Iron, Heavy Concrete, Polyethylene and Graphite Slabs

By

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(Received June 25, 1981)

Abstract

Measurements of transmitted neutron spectra through slabs of iron, heavy concrete, polyethylene and graphite are shown. The experiments are done using collimated fission neutrons from a ^{252}Cf source. To verify the results, multigroup Monte Carlo calculations are done. Fairly good agreement is shown between spectra obtained by experiments and calculations. Some discrepancies shown between them are thought to come from inaccuracies in the cross section data used in the calculations.

I. Introduction

The transmission of fast neutrons through materials is one of the most important phenomena for reactor shielding. The gun-source geometry, i. e. collimated neutrons incident to a slab, is a fundamental geometry to be studied precisely from a macroscopic standpoint. The fast neutron angular spectra obtained by this geometry contain much information concerning various kinds of cross section data such as total cross sections, scattering cross sections and angular distributions in scatterings. Therefore, the measured data by this geometry are effective for the integral testing of the various cross sections. This geometry is, however, multidimensional and the analysis is a little troublesome.

Bolshkov et al[1]. studied the penetration of collimated neutrons through slabs of several materials from a reactor, measured scattered neutrons as a function of the scattering angle, and described the scattering function and the neutron buildup factor. Maerker et al[2)-5)]. measured the deep penetration of collimated neutrons from the

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Tower Shielding Facility of Oak Ridge National Laboratory. They published the data as benchmark experiments of neutron transport in Fe, stainless steel and Na. Miller et al[6]. measured fast neutron angular penetration spectra for water slabs. Measurements of fast neutron spectra transmitted through Fe and Na slabs were carried out[7] by authors at fast-neutron-source reactor "YAYOI" of the University of Tokyo.

This paper describes measurements of ^{252}Cf neutrons transmitted through slabs of Fe, heavy concrete polyethylene and graphite by the gun-source geometry.

II. Experimental

Figure 1 shows the experimental arrangements of the gun-source geometry. Neutrons from a 0.72Ci ^{252}Cf source were collimated by a three stepped duct (15.7 cm radius × 100.5 cm length, 17.4 cm radius × 15 cm length and 19 cm radius × 11.7 cm length), and injected into a slab as shown in the figure. Slabs of iron, heavy concrete, polyethylene and graphite were placed perpendicular to the beam axis. The sizes of the slabs are shown in Table I.

A detector was set in the direction θ with respect to the neutron beam line at the slab surface. The distances between the slab and the detector and the direction θ are

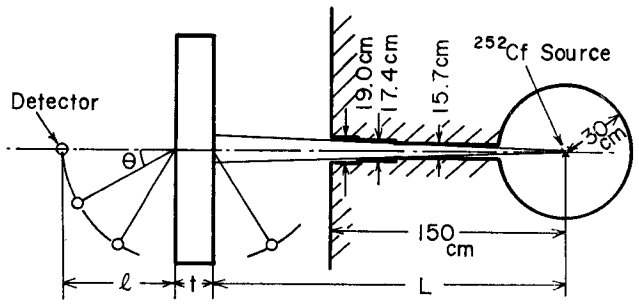

Fig. 1. Experimental arrangement for measurements of transmitted neutron spectra.

Table I. Slab Dimensions and Positons of the Slabs and NE-213 Detector.

Materials	Slab Size (cm)	Slab Position L* (cm)	Detector Position	
			Distance l** (cm)	Angle θ (deg.)
Iron	120×120×20	191.5	100	0°, 30°, 60°
Heavy Concrete	120×120×20	200.6	100	0°, 60°, 120°
Polyethylene	120×120×20	200.0	100	0°, 30°, 60°
Graphite	120×120×35	200.0	40	0°

* This distance is measured from the ^{252}Cf source to the slab.
** This distance is measured from the slab to the detector.

Table II. Atomic Composition of Heavy Concrete

Element	Weight %	Element	Weight %
S	47.5	Ca	1.66
Fe	41.4	Al	0.15
SiO_2	2.0	residual	7.29

shown in Table I. In the measurement at the backward direction ($\theta = 120°$), a heavy concrete shield of 1-m thickness was located between the detector and the collimator outlet in order to cut off scattered neutrons from the collimator wall. The background measurements were done by locating a 60 cm thick iron shield between the slabs and the detector.

A 5 cm diam. by 5 cm thick NE-213 scintillator was used for fast neutron measurements. The pulse shape discrimination was done by a rise time to height converter[8] and a two dimensional multichannel analyzer. The obtained pulse height distributions were unfolded by the FERDO code[9]. The response functions of NE-213 were calculated by the Monte Carlo method[10]. A 30 mCi source was set in the source position (See Fig. 1.) in place of the 0.72 Ci source for the measurement of energy spectrum of source neutrons. This was because the 0.72 Ci source was so intense that the direct measurement of the source neutron spectrum by the NE-213 was impossible due to pile up events of output pulses. The NE-213 was located on the center line of the source neutron beam 200 cm away from the 30 mCi source, and the neutron spectrum was measured. A proton recoil proportional counter of 5 cm diameter filled with 4 atm H_2 gas was set at the same position as the NE-213, and the count rates of neutrons from the two neutron sources of 30 mCi and 0.72 Ci were measured. The spectrum obtained by the

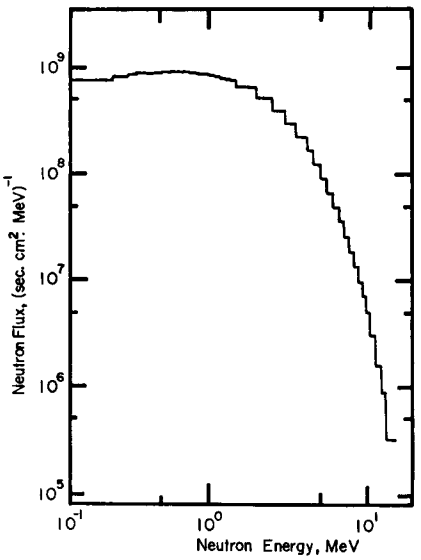

Fig. 2. Source neutron spectrum measured on the center line of the neutron beam at 200 cm from the ^{252}Cf source.

NE-213 with the 30 mCi source was renormalized using a ratio of the count rates measured by the proportional counter for the two sources. The final result of the source spectrum at a position 200 cm from the 0.72 Ci source is shown in Fig. 2.

For the purpose of determining the neutron beam profile, fast neutrons were measured by the NE-213 scintillator (the discrimination level being ~ 2 *MeV*) for the 30 mCi source at several points on the line perpendicular to the axis at 200 cm from the source. The ad hoc calculation method was used for the determnation of the beam profile, because the scintillator's diameter could not be considered small in comparison with the diameter of the neutron beam. At first, the following beam profile was assumed:

$$\phi(r)/\phi_0=\begin{cases}1.0 & (r\leq a_1),\\ \mathrm{Exp}\,[-\{(r-a_1)/b_1\}^2] & (a_1<r\leq a_2),\\ c_1\,[\mathrm{Exp}\{-b_2(r-a_2)\}+c_2\,\mathrm{Exp}\{-b_3(r-a_2)\}], & (a_2<r),\end{cases} \tag{1}$$

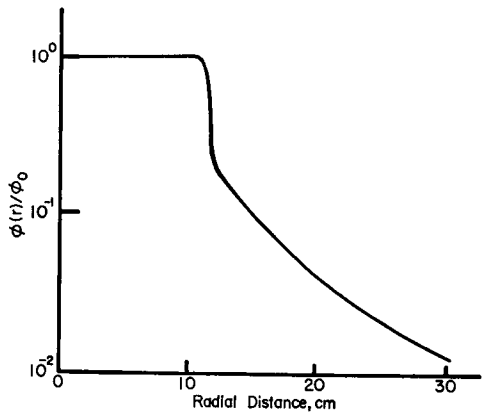

Fig. 3. Neutron beam profile measured at 200 cm from the ^{252}Cf source.

where r is the distance from the beam axis, and ϕ_0, a_1, a_2, b_1, b_2, b_3, c_1, and c_2 are constants to be determined empirically. A neutron counting rate was calculated at each detector position for each assumed beam profile. From the calculated count rates to the measured values the best fit was obtained when the following values were assumed for constants: $a_1=11.16$, $a_2=11.68$, $b_1=0.43$, $b_2=0.5$, $b_3=1.0$, $c_1=0.1219$, and $c_2=0.9$.

The beam profile calculated by Eq. (1) is shown in Fig. 3. The effective beam radius r at the position 200 cm from the source was

$$r=\left(\frac{1}{\phi_0}\int_0^{30}\phi(r)\,2r\,dr\right)^{1/2}=13.3\ \mathrm{cm}. \tag{2}$$

The effective radius r_z and the intensity ϕ_{0z} on the beam line at z cm from the source was calculated by Eqs. (3) and (4):

$$r_z=\frac{z}{200}\cdot r_{200}, \tag{3}$$

$$\text{and}\quad \phi_{0z}=\left(\frac{200}{z}\right)^2\cdot\phi_{0\,200}\quad . \tag{4}$$

III. Calculational

To test the accuracy of the measured results, the multigroup Monte Carlo calculation, neutron energy 1–15 MeV was divided into 26 groups, and group constants with P_8 approximations were generated by the code system RADHEAT-V3[11] from the pointwise cross section data in the ENDF/B-IV[12] file. Evaluated data for sulphur, however, were not available, so silicon data which seemed to approximate sulphur data were used for the sulphur data in generating group constants of heavy concrete.

The sampling of scattering angle in the Monte Carlo calculation was done by Coveyou's technique[12] for iron and heavy concrete shields. In the calculations for graphite and polyethylene, scattering angle was selected isotropically, and then a weight factor was multiplied to the neutron weight in order to express the anisotropy of the angular distribution.

IV. Results and Discussion

Neutron spectra for the iron, heavy concrete, polyethylene and graphite are

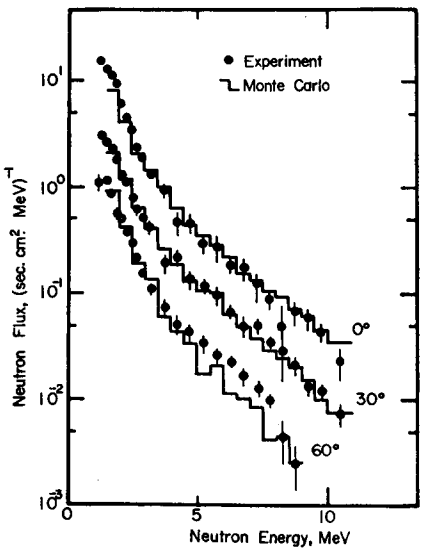

Fig. 4. Comparison of calculated and measured neutron spectra obtained for 20 cm iron slab at 0°, 30° and 60°.

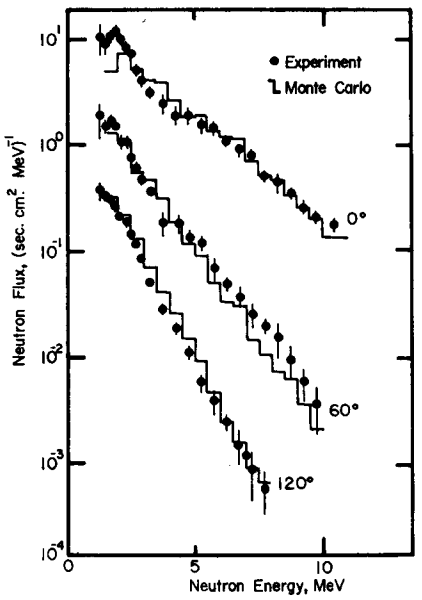

Fig. 5. Comparison of calculated and measured neutron spectra obtained for 20 cm heavy concrete slab at 0°, 60° and 120°.

shown in Figs. 4–7.

The comparison of measured and calculated spectra for the iron slab is shown in Fig. 4. The experiment agrees well with the calculation at directions 0° and 30°. In the 60° spectra, the measured result is larger than the calculated one in the energy range higher than 4 MeV. The convergence of the calculation is not so good in this direction, and the calculated spectrum shows small fluctuations. Therefore, nothing definite can be said about the discrepancies. However, we think that the differential cross section may have some inaccuracies in this region. Speaking in more detail, the measured spectrum is a little larger than the calculated one in the direction of 0° at the lower energies near 2 MeV. This may be explained by the overestimation of the total cross section data used in the calculation by about several percent. A fluctuation in the measured 0° spectrum at 8 MeV seems to be caused by the unfolding process, and does not have any physical meaning.

Fig. 5 shows the spectra for heavy concrete slabs. The agreement between the calculation and the measurement is fairly good, though silicon cross section data were used in the calculation for the sulphur data. Speaking in detail, however, the measured spectrum is larger than the calculated one near 2 MeV in the 0° spectrum, and at energies 3–4 MeV the calculated results are larger than the measured ones in all directions. The structure shown in the 0° spectrum is different with regard to the calculation and the measurement. In the 60° spectrum, disagreement is shown between the calculation and the measurement in the energy range higher than 5 MeV. All these discrepancies seemed to be caused from differences in the cross section data between sulphur and silicon,

Fig. 6 shows the results for polyethylene slabs. The Monte Carlo calculation underestimates the neutron flux in the forward direction at energies lower than 5 MeV. This may be caused from the isotropic sampling method used in the calculation. This method has a tendency to underestimate the scattering neutron flux in the forward direction. At energies higher than 5 MeV, however, the agreement between the calculation and the measurement is good because almost all the neutrons are uncollided ones at these energies. In the directions 30° and 60°, the agreement between the measurement and the calculation is fairly good. At the resonance region around 4 MeV, the calculated spectra becomes much larger than the measured ones. This may be caused by the inaccuracies in the cross section data of carbon in the ENEF/B-IV file.

Fig. 7 shows graphite spectra at the 0° direction. The calculated and measured spectra agree well with each other. The discrepancy near 4 MeV, which was pointed out above for polyethylene spectra, is not shown here. Therefore, the inaccuracy is attributed to differential scattering cross sections of carbon in the directions of 30°

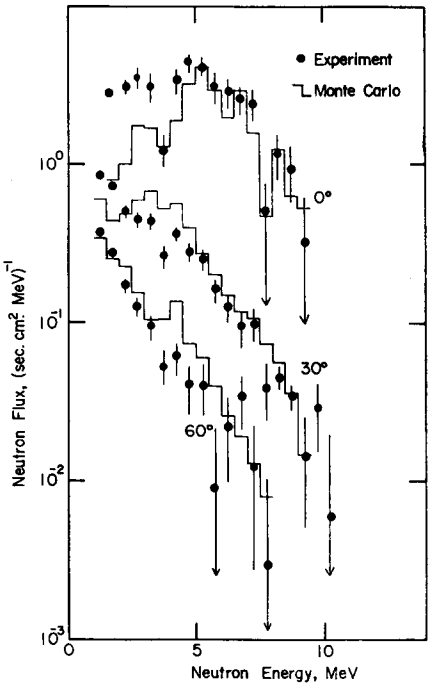

Fig. 6. Comparison of calculated and measured neutron spectra obtained for 20 cm polyethylene slab at 0°, 30° and 60°

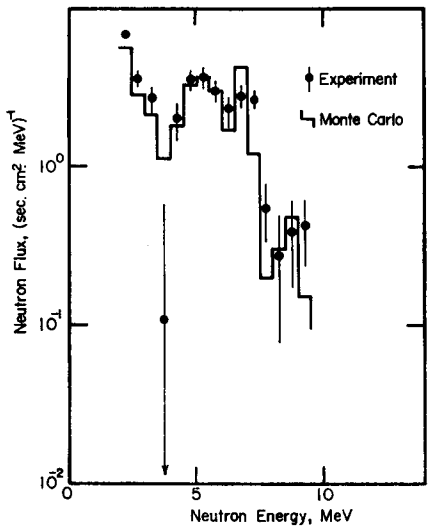

Fig. 7. Comparison of calculated and measured neutron spectra obtained for 40 cm graphite slab at 0°.

and 60°.

V. Conclusions

The measurements of transmitted neutron spectra through the slabs of iron, heavy concrete, polyethylene and graphite were done using collimated fission neutrons from a ^{252}Cf source. These results were compared with results by the multigroup Monte Carlo calculations, and a fairly good agreement was obtained between the spectra obtained by the experiments and the calculations. Excluded were some discrepancies which seemed to come from the inaccuracies in the cross section data used in the calculation.

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