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Neutron Physics Division

MONTÉ CARLO CALCULATIONS OF THE NEUTRON FLUX WITHIN A $^{10}\text{B}_4\text{C}$ SHELL*

R. M. Freestone, Jr.

A Monte Carlo code, O5R, has been employed to study the transport of neutrons through a nearly spherical $^{10}\text{B}_4\text{C}$ shell. Such shells are often used to shield neutron detectors against the intense background of thermal neutrons associated with reactor or reactor-like neutron fluxes. An aluminum disk simulated a solid-state detector within the shell.

Calculations were made for isotropic monoenergetic neutrons having energies of 0.5, 1, 2, 4, 6, 8, 10, 12, and 14 MeV, for an isotropic fission spectrum source, and, for comparison purposes, for a 4-MeV normal beam source.

Results indicate that only a small distortion of the incident spectral shape is created by passage through the shell. The effect seems to be confined to incident neutron energies below 2 MeV. The degree of isotropy of the incident flux, however, importantly affects the absolute fraction of neutrons reaching the detector.

*Funded by Army Nuclear Defense Laboratory, Edgewood Arsenal, Maryland, Project Order No. N2242.

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Monte Carlo Calculations of the Neutron Flux Within a $^{10}\text{B}_4\text{C}$ Shell

R.M. Freestone, Jr.

The abscissa scales for both Fig. 3, p. 13, and Fig. 4, p. 14, are incorrect. The values shown (1, 2, . . . , 7 MeV) should be doubled (2, 4, . . . , 14 MeV). The right edge of the peak in Fig. 3 will then be at 500 keV and the right edge of the peak in Fig. 4 at 1 MeV.

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Introduction

An intense background of thermal neutrons is nearly always associated with reactor or reactor-like neutron fluxes. If a measurement of the epithermal or fast-neutron portion of such a flux is required, the detector must be shielded against the thermal neutrons to prevent saturation of the detector response. A time-honored and highly efficient material for such shields is boron carbide, the boron component of which often is natural boron, nominally 19.8% ^{10}B . Since, however, it is the large thermal absorption cross section of ^{10}B , 3837 ± 10 barns, which is responsible for the major part of the shielding, boron carbide utilizing highly ^{10}B -enriched boron makes a more efficient shield.

In the present work the Monte Carlo techniques of the O5R neutron transport code have been employed to study the transport of neutrons through a nearly spherical B_4C shell, in which the boron was assumed to consist entirely of ^{10}B ; a simple disk of aluminum was used to simulate the detector. The principal motivation for the study was the interest of J. H. McNeilly and J. W. Kinch of the Army Nuclear Defense Laboratory (ANDL) at Edgewood Arsenal, Maryland, who have used similar shells to shield a series of solid-state neutron detectors. Calculations were run on the ORNL CDC 1604-A computer.

Calculation

The O5R code has been rather completely described elsewhere.¹ Briefly, it is a general-purpose Monte Carlo neutron transport code which is distinguished by its highly detailed representation of neutron cross sections and which, as part of the package, contains an extremely flexible geometry routine. Neutrons are processed in batches through energy groups, with only the cross sections for a particular energy group being held in the machine memory at any given time. Some 34 parameters describing neutron collisions with nuclei are obtainable. In order to improve the statistical variance of results, neutrons are never permitted to be absorbed in a collision. Instead, the relative importance (the "weight" of

¹R. R. Coveyou et al., O5R, A General-Purpose Monte Carlo Neutron Transport Code, ORNL-3622 (1965).

a neutron) is reduced according to the probability of absorption. In order not to waste machine time in tracing neutrons having little importance, a Russian roulette option is available, by means of which a neutron whose weight falls below an arbitrary standard is subjected to a game of chance. If the neutron "wins" the gamble its weight is increased to an arbitrarily specified value; otherwise it is no longer followed. The "odds" of the game of chance are based on the values of the new weight and the original weight; thus no bias is introduced.

For the present calculation, cross sections were specified at 128 points per energy interval. Energy interval boundaries were a factor of 2 apart, from 2.5×10^{-2} eV to 19.28 MeV. Both elastic and inelastic scattering were permitted in the ^{10}B , and an anisotropic scattering distribution based upon a P_8 Legendre expansion was used. Inelastic scattering was handled by a simple evaporation model routine originally due to W. E. Kinney of ORNL. For carbon, only anisotropic elastic scattering was considered, with the scattering angle being picked from a so-called F_1 distribution given by $P(\mu) = (1 + 3f_1\mu)/2$. For the aluminum which simulated the detector, isotropic elastic scattering was postulated.

Model

The "experimental" configuration on which the calculation is based is shown schematically in Fig. 1. It is essentially identical to one of the ANDL shells. Only two compromises with reality were made: (1) The small amount of air that exists within the actual shell was ignored, being replaced by a perfect vacuum. This compromise obviously had little or no effect upon neutron transport but substantially reduced the amount of cross-section storage required. (2) The solid-state detector of the true experiment, which was of unknown (to us) and probably complicated makeup, was replaced by a simple disk of aluminum. Again, storage and running time were conserved but any effects due to neutron buildup at the surface of the detector were fairly well taken into account.

ORNL-DWG 67-3030

Fig. 1. Shell Configuration Used in Calculation.

Neutron Sources

Three types of neutron sources were used. For the first, the FISRNF subprogram of the O5R random-number package was used to provide a random selection of neutron energies from the Watt fission spectrum as modified by Cranberg et al.² A second group of computations used monoenergetic sources of 0.5, 1, 2, 4, 6, 8, 10, 12, and 14 MeV.

With both of the above sources, neutron flights were essentially isotropic. X, Y, and Z coordinates were selected at random over the "northwest" quadrant of the shell, and the source neutron was permitted any direction possible, the only condition being that the initial flight be through the shell.

In order to compare the effects of an isotropic flux with those of a beam current, a single calculation was made with a third type of source, which consisted of a 4-MeV beam having a diameter equal to that of the shell and directed along its major axis.

A typical problem consisted of three runs, each run containing four batches of 1500 neutrons per batch, or 6000 neutrons. With the relatively large detector area, this number of neutrons permitted the use of simple analog analysis techniques and gave good statistical accuracy in the energy regions of importance.

Results

Results are presented in both tabular and graphic form. Table 1 contains the results of the monoenergetic, isotropic source calculations. It is evident that the overwhelming majority of neutrons which eventually reached the detector did so with very nearly their original energies. For incident neutron energies of from 4 to 14 MeV the number of neutrons reaching the detector is very nearly the same; for energies of 2 MeV and below, a somewhat lower percentage of the incident neutrons was eventually counted.

Table 2 shows a comparison of the 4-MeV isotropic source data with the results of the 4-MeV beam calculation. It is clear that the

²L. Cranberg et al., Phys. Rev. 103, 662 (1956).

Table 1. Results of Monoenergetic, Isotropic Source Calculations

Energy Bin Limits (MeV)	Number of Neutrons Striking Detector, per Isotropically Incident Source Neutron								
	0.5 MeV	1.0 MeV	2.0 MeV	4.0 MeV	6.0 MeV	8.0 MeV	10.0 MeV	12.0 MeV	14.0 MeV
0-0.25	2.77(-3) ^a	7.45(-5)				1.59(-4)			
0.25-0.50	2.47(-1)	1.33(-2)	2.64(-4)			7.52(-4)	1.54(-4)	3.69(-5)	
0.50-0.75		6.75(-3)	2.44(-4)				1.84(-4)	8.66(-5)	1.74(-4)
0.75-1.0		2.81(-1)	5.06(-4)			3.63(-4)		1.30(-4)	
1.0-1.5			2.66(-3)		8.91(-5)	5.24(-4)	4.82(-4)	2.25(-4)	2.79(-4)
1.5-2.0			2.17(-1)	2.27(-4)	1.58(-4)	3.99(-4)	5.95(-4)	2.73(-4)	4.52(-4)
2.0-2.5				4.58(-4)	8.91(-5)	6.14(-4)	9.28(-4)	1.36(-4)	4.65(-4)
2.5-3.0				1.67(-3)	2.46(-4)	7.65(-4)	3.58(-4)	1.28(-4)	6.27(-4)
3.0-3.5				1.84(-3)	6.72(-4)	4.51(-4)	3.38(-4)	1.77(-4)	4.37(-4)
3.5-4.0				3.02(-1)	3.07(-4)	6.25(-4)	6.33(-4)	2.43(-4)	4.52(-4)
4.0-5.0					1.52(-3)	2.66(-4)	1.84(-4)	3.18(-4)	7.66(-4)
5.0-6.0					3.11(-1)	8.87(-4)	4.44(-4)	2.17(-4)	3.26(-4)
6.0-7.0						1.98(-3)	5.91(-4)	2.71(-4)	3.26(-4)
7.0-8.0						3.08(-1)	6.26(-4)	1.63(-4)	2.33(-4)
8.0-9.0							1.67(-3)	5.03(-4)	4.38(-4)
9.0-10.0							3.01(-1)	4.49(-4)	3.72(-4)
10.0-11.0								1.29(-3)	5.50(-4)
11.0-12.0								3.03(-1)	6.19(-4)
12.0-13.0									6.83(-4)
13.0-14.0									3.08(-1)

a. Numbers in parentheses indicate power-of-ten multipliers; e.g., 2.77(-3) = 0.00277.

Table 2. Comparison of 4-MeV Isotropic Source Results
with 4-MeV Beam Source Results

Energy Bin Limits (MeV)	Number of Neutrons Striking Detector per Incident Neutron	
	Isotropic Source	Beam Source
0-0.25		1.46(-3) ^a
0.25-0.50		3.90(-5)
0.50-0.75		2.48(-3)
0.75-1.0		3.49(-5)
1.0-1.5		1.66(-4)
1.5-2.0	2.27(-4)	4.84(-3)
2.0-2.5	4.58(-4)	6.45(-4)
2.5-3.0	1.67(-3)	5.69(-3)
3.0-3.5	1.84(-3)	9.47(-4)
3.5-4.0	3.02(-1)	6.56(-1)

a. Numbers in parentheses indicate power-of-ten
multipliers.

directional characteristics of the source bear importantly on the apparent efficiency of the detector. In the use or calibration of boron carbide shells it is apparently most important to know the degree of isotropy of the flux in which the shell is to be used.

The results contained in Tables 1 and 2 are shown graphically in Figs. 2-12, in somewhat different fashion. These plots were prepared as the data were accumulated, and the ordinate for each group therefore represents the neutrons per MeV energy interval striking the detector, for an incident flux of 6000 neutrons. Figure 2 compares the results of the fission source calculation with the Cranberg input spectrum. It is clear that, within statistics, no modification of the spectral shape has occurred.

Figures 3-12 display the results of the monoenergetic calculations. Each graph shows a strong peak at the input energy, with an erratic and inconsiderable scattering of degraded neutrons over the low-energy region.

At the suggestion of C. E. Clifford (ORNL) a detailed study was made of some of the monoenergetic peaks. Neutrons falling within the highest 1-MeV energy interval of the 1-, 2-, 4-, 6-, 8-, and 14-MeV runs were re-analyzed into 0.10-MeV bins. These results are shown in Figs. 13-19. Apparently most of the monoenergetic neutrons which reach the detector do so without collision, particularly in the higher energies. The 1-MeV analysis shows that there is some tendency toward a smooth smearing into lower energies.

Some of the figures show a value for "outscatter." This represents the number of neutrons which were scattered out of the shell before penetrating to the detector. It is not given in some cases because the tallying device was incorporated only after some data had been accumulated.

Errors

An examination of the statistical range of the monoenergetic results indicates that an error of 1 to 2% is appropriate for the peak regions. So few counts were recorded in the lower energy ranges that accuracy in these regions is dubious or nonexistent. For the fission results, accuracy in the peak region is again of the order of 2%, becoming steadily poorer as higher energies are approached. At 5 1/2 MeV the accuracy is about +12%.

Conclusions

The assembled results indicate that only a small distortion of the incident spectrum shape results from the use of a $^{10}\text{B}_4\text{C}$ shell. This effect seems to be apparent only for neutrons having energies of about 2 MeV or lower. The results also show that the degree of isotropy in the incident flux importantly affects the number of neutrons reaching the detector, which in turn suggests that calibrations intended to result in absolute values of flux may be valid only in situations where the degree of isotropy is well known.

Acknowledgments

Cross sections used in this calculation are from the recent compilations of D. C. Irving, ORNL Neutron Physics Division, who made a complete updating especially for this work.

Fig. 2. Comparison of Calculated Transmitted Spectrum with Source Spectrum.

Fig. 3. Calculated Transmission of 500-keV Isotropic Neutrons Through a B_4C Shell.

Fig. 4. Calculated Transmission of 1-MeV Isotropic Neutrons Through a B_4C Shell.

Fig. 5. Calculated Transmission of 2-MeV Isotropic Neutrons Through a B_4C Shell.

Fig. 6. Calculated Transmission of 4-MeV Isotropic Neutrons Through a B_4C Shell.

Fig. 7. Calculated Transmission of 4-MeV Neutrons (Monoenergetic Beam Parallel to X Axis) Through a B_4C Shell.

Fig. 8. Calculated Transmission of 6-MeV Isotropic Neutrons Through a B₄C Shell.

Fig. 9. Calculated Transmission of 8-MeV Isotropic Neutrons Through a B_4C Shell.

Fig. 10. Calculated Transmission of 10-MeV Isotropic Neutrons Through a B_4C Shell.

Fig. 11. Calculated Transmission of 12-MeV Isotropic Neutrons Through a B₄C Shell.

Fig. 12. Calculated Transmission of 14-MeV Isotropic Neutrons Through a B₄C Shell.

Fig. 13. Fine-Mesh Analysis of the Peak from 1-MeV Isotropic Source Neutrons.

Fig. 14. Fine-Mesh Analysis of the Peak from 2-MeV Isotropic Source Neutrons.

Fig. 15. Fine-Mesh Analysis of the Peak from 4-MeV Isotropic Source Neutrons.

Fig. 16. Fine-Mesh Analysis of the Peak from 4-MeV Beam Source Neutrons.

Fig. 17. Fine-Mesh Analysis of the Peak from 6-MeV Isotropic Source Neutrons.

Fig. 18. Fine-Mesh Analysis of the Peak from 8-MeV Isotropic Source Neutrons.

Fig. 19. Fine-Mesh Analysis of the Peak from 14-MeV Isotropic Source Neutrons.

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