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ORNL - TM - 1965

COPY NO. - 167

DATE - Oct. 6, 1967

Neutron Physics Division

CONF-670932--3

**NEUTRON TRANSPORT STUDIES IN DUCTS AND SIMPLE SHIELDS:
COMPARISON OF MONTE CARLO AND DISCRETE ORDINATES CALCULATIONS WITH
EXPERIMENTAL RESULTS FROM THE TOWER SHIELDING FACILITY***

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ABSTRACT

A development program for neutron transport calculational techniques has included a series of experiments at the Tower Shielding Facility to measure neutron dose distributions in ducts and neutron energy spectra transmitted through simple shields. The experiments have been used to guide the development of a simplified Monte Carlo method for calculating neutron transmissions through ducts and to guide the development of the discrete ordinates techniques. Measurements of neutron spectra at specific emergent angles from slab shields of polyethylene and lead have been used to verify the discrete ordinates techniques and to aid in the selection of calculational parameters (space, angle and energy meshes, and order of Legendre expansion of the differential scattering cross sections) for use in a two-dimensional code that has recently been developed at Oak Ridge. Cross sections for use in the above neutron transport calculations have been selected on the basis of measurements of the uncollided neutron spectra through thick samples of shield materials.

NOTE:

This Work Partially Funded by
DEFENSE ATOMIC SUPPORT AGENCY
Under Orders EO-802-65 and EO-800-64

*Paper presented at the Institute of Physics and the Physical Society International Conference on the Physics Problems of Reactor Shielding, 26th to 29th September, 1967, Harwell, England.

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INTRODUCTION

Neutron transport studies in ducts and simple shields are being conducted at the Oak Ridge National Laboratory under the auspices of the U. S. Atomic Energy Commission and the Defense Atomic Support Agency as part of the reactor and weapons radiation shielding program of the Neutron Physics Division. Even though the specific shields of interest to these two agencies are very different, the programs for both require that techniques be developed for predicting the deep penetration of neutrons through various media having rather complex geometrical configurations. The approach taken in developing these methods relies heavily on experimental results obtained at the Tower Shielding Facility. The experiments are designed to test the accuracy of the calculational methods, to help evaluate the cross-section data used in the calculations, and to help in the selection of the spatial, energy, and angular meshes utilized in the discrete ordinates method. As a result of improvements in the experimental techniques used, it is now possible to design cleaner experiments which give a severe test of the calculational techniques. These improvements include the use of better fast-neutron spectrometers (the NE-213 proton-recoil spectrometer and the ^6Li -shielded diode spectrometer) and also the use of a boron-shielded BF_3 detector from which low-energy neutron spectra may be inferred.

The first study to be discussed is concerned with the prediction of neutron transport through concrete-lined ducts or entryways which are required in heavily shielded structures of interest to the Defense Atomic Support Agency. A Monte

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Carlo code has been written for predicting the transport of neutrons down a duct, and also the emission of secondary gamma rays from the duct walls. A neutron wall interaction is treated as a single event in which the distribution of the emerging particle as a function of the incident neutron energy and angle is obtained from completely differential albedo calculations for an infinite slab case. An experimental program was designed to demonstrate that the duct problem could be divided into two simpler problems: (1) determining the neutron albedo of the wall material, and (2) using the albedos in a Monte Carlo transport code which is dependent only on the duct geometry. In this code the wall material is replaced by a surface having previously determined reflection properties.

The second study to be discussed is concerned with an experiment which was designed to provide comprehensive data for comparison with a two-dimensional transport calculation using a discrete-ordinates code. The fundamental element or result of a two-dimensional transport calculation is the angular flux which is a function of two space dimensions, the energy, and two angles. The ideal experiment would provide a direct comparison with this five-dimensional flux. However, such an experiment is not feasible and therefore comparisons are usually made with integral quantities. In this case, the energy-dependent flux was measured at various points far removed from the exit face of a shield so that the neutrons emerging at particular angles from the entire back surface of the slab shield were detected. Lead and polyethylene were chosen as the shielding materials for this experiment because it was considered that the cross sections were reasonably well known for these materials.

The third and final study to be covered here is concerned with an experimental evaluation of total cross sections, particularly in the energy regions of the minima which are, of course, very important in shielding calculations. These experiments consisted of the measurement of the incident and transmitted neutron spectra for thick samples of shielding materials. The geometry was chosen so that no multiply scattered neutrons contributed to the measurement. This allowed a direct comparison of the experimental results with a simple exponential attenuation calculation using the various available total cross sections.

I. ALBEDO MONTE CARLO METHODS FOR DUCT TRANSPORT CALCULATIONS

1. Description of Calculations

Albedo methods have been applied to duct streaming problems as an obvious method of reducing the calculational effort required. In using an albedo method, the complex phenomenon of multiple scattering and absorption within a duct wall is replaced by a relatively simple reflection coefficient at a surface. The basic assumption made in the use of albedos is that the particles emerge at the point of incidence. This is the necessary simplifying assumption that permits the use of predetermined albedos in duct transmission calculations.

The albedos are always determined for an infinite plane and their application to finite geometry configurations such as rectangular concrete ducts is limited to ducts with cross-section dimensions that are large compared with the mean free path of neutrons (or gamma rays when dealing with capture gamma rays) in concrete. For this condition, the wall areas influenced by the corners comprise a relatively small portion of the total area and have small effect on the results. Similarly, the corner-lip in-scattering effect becomes rather unimportant. Depending on the particular neutron energy, the mean free path of fast and intermediate-energy neutrons will range from about 3 cm to about 10 cm, with about 5 cm being most probable for a fission spectrum. Consequently, the use of infinite plane albedos in streaming calculations for ducts with cross-section dimensions less than about 1 ft may give erroneous results.

In albedo Monte Carlo calculations the random-sampling or Monte Carlo process is used to follow a particle that is reflected down a duct. This method resulted from efforts to calculate low-energy neutrons at a duct exit, which in two- and three-legged ducts can contribute a large fraction of the total multicollision dose rate. Unlike the albedos for gamma rays, which are relatively low, the albedo for low-energy neutrons is high; consequently, good results can be obtained only by considering a large number of scatterings from wall to wall. With so many scatterings and the relatively complicated geometry of rectangular two-legged ducts, the analytic solutions of the integrals involved become so complex that the solutions are usually more efficient if a random-sampling technique is used.

An albedo Monte Carlo code for calculating neutron transport through ducts was developed by Cain [1] and modified by Maerker and Cain [2]. The modification

was major in that it replaced simple analytic albedo functions with doubly differential albedos, that is, albedos that are differential both in reflected solid angle and in energy. (These albedos had been determined by earlier Monte Carlo calculations for infinite slabs.) The modified code, designated as the AMC code, includes a provision for calculating the capture gamma-ray dose rate arising from neutron capture in the duct walls.

In the AMC code the walls are assumed to be reflecting and absorbing surfaces which reflect the incident neutrons with reduced weight into various energy groups and directions, the probabilities for which are predetermined from the doubly differential albedo data. For each wall interaction, statistical estimates based on the differential angular albedo data are made of the dose rate or flux contribution to any detector located within the duct that is visible from a scattering point. The neutrons are followed in a random-walk procedure down the duct until they either escape out the front or rear end of the duct or they are killed by Russian roulette when their weight falls below a predetermined value. For best accuracy the calculations are done separately for three energy ranges. Neutrons with energies from 200 keV to 8 MeV divided into six energy groups are sampled to obtain the fast-flux contribution to the dose rate. The dose from neutrons of intermediate energy (0.5 eV to 200 keV) and the thermal neutrons produced by slowing down from all nonthermal groups are determined with a 13-group calculation that involves sampling the whole energy range from thermal to 8 MeV, but scoring only those with energies below 200 keV. The dose rate from source thermal neutrons (subcadmium, $E_0 < 0.5$ eV) is then determined in a separate calculation which uses a single-velocity model. Different Monte Carlo data and correlations are used in each energy range. The albedos used for intermediate-energy neutrons were determined by Coleman et al. [3], and those used for fast neutrons and thermal neutrons were determined by Maerker and Muckenthaler [4,5]. They also developed the equivalent of an albedo for use in calculating the capture gamma-ray contribution to the dose.

2. Description of Duct Experiment

A series of measurements was completed at the Tower Shielding Facility to determine neutron intensities along the centerline of a 3-ft-square duct with concrete walls and having up to two 90° bends. Fast-neutron doses were measured with a

Hurst dosimeter and thermal-neutron fluxes were determined from difference measurements with bare and cadmium-covered $^{10}\text{BF}_3$ counters. In other measurements the BF_3 counter, which is spherical, was covered with large spheres of polyethylene. In this configuration the counter has a response function which approximates the Snyder-Neufeld [6] multicollision dose. These last measurements were made because they are more sensitive to low-energy neutrons which dominate the dose after one bend.

The source of neutrons for the experiment was a collimated beam from the Tower Shielding Reactor II (TSR-II). The beam collimator and duct arrangement are shown in Fig. 1. The beam strikes one wall of the duct at an angle of 45° and is collimated sufficiently well so that the number of neutrons hitting any other wall is negligible.

The fast-neutron spectrum of the beam was measured by Bokhari and Verbinski [7] using a shielded diode spectrometer. The measured spectra are shown in Fig. 2. The slow-neutron spectra were inferred from measurements using a boron-shielded BF_3 detector [8] having ^{10}B filters of up to 23 different thicknesses. The spectrum obtained from these measurements is shown in Fig. 3. Detailed measurements were made of the beam profile at the duct wall position with fast- and thermal-neutron detectors, and the total number of neutrons striking the wall was obtained by integration over these profiles. Typical distributions are shown in Fig. 4. All calculations were normalized to the measured number of neutrons striking the duct wall. This was obtained by assuming that the spectrum of the beam was invariant over the beam profile and that the dose per neutron calculated from the spectrum was accurate. Direct comparison of the dose rate calculated from the measured spectrum with dosimeter measurements in the same position was in good agreement. Figures 5, 6, and 7 are photographs of duct configurations for which measurements were obtained: a straight section (Fig. 5); a duct with one 90° bend (Fig. 6); and a duct with two 90° bends (Fig. 7). The reactor is contained in the spherical shield to the right of the duct array. The large cylinder in the foreground is a beam catcher and the many water-filled tanks surrounding the ducts are additional shielding, all of which were required to reduce the background from neutrons scattering externally to the duct (ducts).

3. Comparison of Calculated and Measured Results

The calculations and experiments were carried out for all three of the duct configurations. In all cases the duct opening was a 3-ft square. The straight

Fig. 1. Source Geometry for the Duct Experiments.

Fig. 2. Fast-Neutron Spectrum of the Neutron Beam from the TSR-II.

Fig. 3. Low-Energy Spectrum of the Neutron Beam from the TSR-II.

Fig. 4. Spatial Distributions of the Fast-Neutron Beam Incident on the Duct Wall.

Fig. 5. Photograph of the Straight Duct.

Fig. 6. Photograph of Duct with One Bend.

Fig. 7. Photograph of Duct with Two Bends.

duct was 45 ft long. For the duct with one bend the first leg was 15 ft long and the second leg was 30 ft long. For the duct with two bends, the first and second legs were 15 ft long and the third leg was 20 ft. The 45° angle of incidence of the neutron beam was chosen to serve as a rigorous test of the calculation, because the fluxes and doses for detector locations greater than a few feet from the duct mouth are due entirely to wall-scattered radiation.

Calculations of the fast-neutron dose rates by Maerker [9] were made for the straight duct and the two-legged duct only since the experimental results were limited to these cases. The experimentally measured absolute energy spectrum of the beam was used for the source spectrum in the calculations. Figure 8 shows a comparison of the calculated and measured results. In general, the agreement averages better than 20% through five orders of fast-neutron dose attenuation.

Figure 9 shows a comparison of the calculated and measured centerline subcadmium or thermal fluxes that result from an incident beam of subcadmium neutrons [10]. In general, the agreement is within 20% in the second and third legs and within 5% for most of the straight ducts. For the three-legged duct, it was necessary to follow approximately 18 wall backscatterings to obtain 90% of the flux contribution in the middle of the second leg, and 25 wall backscatterings in the middle of the third leg. Statistical errors in the second leg averaged approximately 10% for 20,000 source histories with a running time on a CDC-1604 of 2 hours; in the third leg the error averaged approximately 15% for 50,000 source histories with a running time of close to 3 hours.

In order to test the accuracy of the slowing-down model used in the code, calculations were made of the thermal flux resulting from all source neutrons with energies above cadmium cutoff. Comparison between calculations and experiment [11] are shown in Fig. 10. Agreement in general averaged about the same as that for incident subcadmium neutrons, and the statistical errors, number of wall backscatterings, and running times for the same number of source histories were comparable. It should be noted that the calculations were normalized in each case to the measured number of neutrons incident on the wall of the duct from the source beam.

The epicadmium or nonthermal spectra at several locations along the duct centerline were also calculated for a three-legged duct. Using the flux-to-multicollision-dose factors of Snyder and Neufeld [6], epicadmium multicollision dose rates were calculated at these locations [12] and are shown in Table 1 in comparison with measurements by a dosimeter designed to give a response proportional to the Snyder-Neufeld multicollision dose curve over the range of interest.

Fig. 8. Comparison of Measured and Calculated Fast-Neutron Dose Rates in Ducts.

Fig. 9. Comparison of Measured and Calculated Thermal-Neutron Fluxes from a Thermal-Neutron Source in Ducts.

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Fig. 10. Comparison of Measured and Calculated Thermal-Neutron Fluxes from an Epithermal Neutron Source in Ducts.

Table 1. Comparison of Calculated and Measured
Multicollision Dose Rate

Duct Leg	Distance From Mouth to Detector (ft)	Dose Rate (ergs g ⁻¹ hr ⁻¹ W ⁻¹)		
		Measured 10-in. Ball	Calculated	Measured 12-in. Ball
2	19	$2.5 \times 10^{-6} (2.1 \times 10^{-6})^*$	1.9×10^{-6}	$1.35 \times 10^{-6} (1.1 \times 10^{-6})$
2	23	$5.5 \times 10^{-6} (4.9 \times 10^{-7})$	4.5×10^{-7}	$2.85 \times 10^{-7} (2.45 \times 10^{-7})$
2	29	$1.28 \times 10^{-7} (1.2 \times 10^{-7})$	1.05×10^{-7}	$6.6 \times 10^{-7} (6.1 \times 10^{-8})$
3	33	$1.45 \times 10^{-8} (1.2 \times 10^{-8})$	9.2×10^{-9}	$6.4 \times 10^{-9} (4.8 \times 10^{-9})$
3	40	$1.08 \times 10^{-9} (9.0 \times 10^{-10})$	6.5×10^{-10}	$6.0 \times 10^{-10} (5.1 \times 10^{-10})$

*The dose rates in parentheses are those obtained when the center of detection was assumed to be the center of the polyethylene ball rather than the leading edge. The dosimeter was a spherical BF₃ counter placed inside a polyethylene ball covered with cadmium.

The excellent agreement shown between the measured and calculated results yields a high degree of confidence that the albedo Monte Carlo approach to the calculation of neutron transport in large ducts is quite adequate. The code is now being used to generate parametric data for more realistic problems of interest to the Defense Atomic Support Agency.

II. DISCRETE ORDINATES METHOD (S_n)

1. Description of Calculations

The discrete ordinates or Carlson's S_n method is a means of approximating the solution of the energy-dependent linear Boltzmann transport equation with anisotropic scattering. The method is therefore suitable for the calculation of neutron and gamma-ray transport in shields. Since the method is fundamentally formulated as a finite difference equation (rather than finite differencing of an analytic approximation), it is well suited for use on high speed digital computers. A minimum of limiting assumptions is required, and the solutions will approach the exact solution of the Boltzmann equation with increasing orders of approximation as the space, angle, and energy mesh approaches differential size. The method can solve either the homogeneous or the inhomogeneous Boltzmann equation, for single or multilaminar shields, and for a variety of source configurations, including both surface and volume-distributed sources.

In the derivation of the discrete ordinates equations the Boltzmann equation is first transformed such that, under multiplication by the volume element, the coefficients of the partial derivatives do not contain the variable of differentiation. The difference equations are derived by integrating the transformed equation, which is known as the conservative Boltzmann equation, over a finite (difference) phase space cell having dimensions of volume, angle interval, and energy group. For each term in the transformed equation the result of the integration is a two-point difference in the variable of differentiation. A mean value approximation is used for the other variables. The difference equations resulting from this technique do require definitions relating the mean value of the unknown function to its end point values. A linear interpolation scheme known as "diamond difference" is the most widely used technique in the S_n method.

The major development which has made the discrete ordinates method suitable for deep-penetration transport calculations is the inclusion of general anisotropic

scattering. The fundamental idea is the use of a single-direction set of angles and weights to evaluate the Boltzmann equation integrals by quadrature and the convection term by the mean-value-difference equation, and the use of multigroup constants and multigroup polynomial coefficients to describe the energy variation and angle-energy correlation. Two of the available one-dimensional codes, ANISN [13] and DTF IV [14], and the two-dimensional code, DOT [15], allow general anisotropic scattering. Of these codes ANISN and DOT were developed primarily for shielding problems where large numbers of mesh points, energy groups, and angles are usually necessary.

In order to check out the codes and gain some experience in the problems associated with the application of the $S_n P_n$ method to deep-penetration problems, a series of calculations were performed to investigate the effect of mesh size. The calculations were performed to compare with the ORNL Lid Tank experimental results in water. The effects of the space angle and energy mesh were determined, as well as the effects of the orders of polynomial expansion of the scattering cross section. When properly weighted 27-group cross sections were used in various regions of the shield, it was determined that for transport in water an $S_8 P_2$ approximation was adequate in slab geometry, but that an $S_{16} P_3$ approximation was required in spherical geometry and an $S_{10} P_3$ approximation was required for the two-dimensional calculation.

The Lid Tank problem may be solved in one dimension by performing a slab calculation with an isotropic plane fission source or a spherical calculation with a point fission source. The results of these problems may be transformed geometrically to the 35.56-cm-diam Lid Tank isotropic disk source. A two-dimensional calculation may also be performed, but except for the source centerline data there are no experimental checks.

Transformed results from one-dimensional calculations, including moments method results, are compared with Lid Tank centerline data in Fig. 11. Results from the two-dimensional calculations are shown in Fig. 12. Since these results could not be compared with experiment, the transformed one-dimensional results were extended to obtain off-centerline data, also shown in Fig. 12.

In order to obtain experimental data for checking the two-dimensional code (DOT), a two-dimensional shielding experiment was planned as part of the DASA-supported shielding program. In the experiment a collimated beam of neutrons was incident on a slab of shielding material.

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Fig. 11. Comparison of One-Dimensional Calculations and Lid Tank Measurements of Fast Neutron Dose in Water.

Fig. 12. Comparison of Lid Tank Two-Dimensional Results (DOT) with Transformed One-Dimensional Results (Kernel).

A two-dimensional discrete ordinates transport calculation was performed for the shield represented as a cylinder, with the source distribution input as a boundary condition on the angular flux at one end. These calculations, which contain 30 neutron energy groups between 1 and 15 MeV, 675 mesh points, 70 angles, and a P_3 representation of the differential scattering cross sections, stored the angular fluxes on the exterior surface on tape. A surface integration code, SPACETRAN, then examined the exit surface mesh, located the angular flux closest to the vector from the surface to detection point by minimum dot product, and, assuming each surface interval to be represented by a point source, calculated and summed the contribution to the flux at the detection point. Figure 13 shows a graphical representation of the surface integration used in the calculation. Due to symmetry, only half of the cylindrical surface need be calculated, and, for this problem, only the exit end is important.

2. Description of the Experiment

Figure 14 shows the experimental configuration at the Tower Shielding Facility for the two-dimensional experiment. The TSR-II is placed in a blockhouse open at one end for access. The reactor vessel is surrounded by 25 in. of 50% lead shot and 50% water, followed by 23 in. of water. The collimator is approximately 8 ft long and extends from the aluminum pressure vessel surrounding the reactor. A 6-in. lead insert is placed in the collimator entrance to attenuate core gammas. The collimated neutrons impinge upon large slab shield samples which are backed by an 8-in. lead collar so that double albedo effects may be calculated if necessary.

The primary measurements were made with a 2 in. by 2 in. NE-213 scintillator in the spectrometer housing as shown. The most interesting data were taken when the spectrometer was rotated in an arc of radius 27.4 ft from the back surface of the shield. Since the spectrometer views the entire surface of the slab, the data at a point on the arc at an angle α from the normal corresponds to the scalar flux at that point in the radiation field. Various measurements of the radiation field in the 2π half space exposed to the shield exit face were also made with the Hurst fast-neutron dosimeter and Hornyak button.

An important aspect of this experiment was that the absolute spatial and energy distribution of neutrons incident on the shield sample were determined. The values obtained were then used as a source in the DOT calculations and as other normalization was applied. The space, angle, and energy distributions were

$$\varphi_{\text{DET}} = 2 \int_0^\pi \int_0^R \frac{\varphi(\vec{\Omega}) |\eta|}{|\vec{r}|^2} r dr d\alpha$$

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DETECTOR

 $\vec{\Omega}$ = UNIT VECTOR ALONG $\vec{r}$

$$\varphi_{\text{DET}} = 2 \int_0^\alpha \text{MAX} \int_0^H \frac{\varphi(\vec{\Omega}) |\mu|}{|\vec{r}|^2} R dZ d\alpha$$

$$\alpha \text{ MAX} \leq \pi/2$$

Fig. 13. The Surface Mesh and Detector Geometry for Spacetrans Surface Integration.

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Fig. 14. Experimental Configuration at Tower Shielding Facility.

determined primarily by measurements with the NE-213 scintillator and were augmented by measurements with the fast-neutron dosimeter and a 1/2-in.-diam Hornyak button.

A photograph of the experimental assembly is shown in Fig. 15. The reactor is under the black plastic raincoat in the blockhouse. The spectrometer assembly is shown connected to a pivot point at the back surface of the shield and is positioned at an angle of about 60° . A stack of lithiated paraffin blocks were placed on the side of the shield viewed by the spectrometer to prevent the measurement of neutrons streaming from the side of the slab. Side leakage was possible because the sample consisted of six 1-in.-thick pieces with small gaps between them.

Figure 16 shows a closeup of the collimator exit and shield support stand in the bare beam configuration used for mapping the source.

3. Comparison of Results

Figure 17 shows results for the neutron spectra emerging from the 6 in. of polyethylene at 0° . This is a special case as all of the flux was found to be uncollided. The error bars for the NE-213 curve include counting statistics, errors in response functions, and numerical errors in the unfolding. The uncollided flux was obtained from a hand calculation; however, it does demonstrate the relative accuracy of the multigroup total cross sections. The reason the uncollided flux is calculated by hand is due to the assumption in SPACETRAN that the surface flux may be represented by anisotropic point sources, which gives an incorrect R^2 geometry correction for the uncollided flux.

Figure 18 shows the results for the 6 in. of polyethylene at 30° . In this case, all of the flux is collided. Since normalization is based on the neutron intensity incident on the slab, the measured and computed spectra should compare directly. As shown, the agreement with experiment is excellent.

Figure 19 shows the results for 11 in. of lead at 0° . The comparison here is with a hand calculation of the uncollided flux (about 50% of the total) added to the discrete ordinates calculation of the collided flux. There is considerable disagreement below 2 MeV, probably indicating significant inaccuracies in the total cross sections for lead. Figure 20 shows the comparison for 11 in. of lead at 13° . In this case the flux is all collided and the agreement is quite good in intensity

Fig. 15. Photograph of Experimental Assembly.

Fig. 16. Photograph of Collimator and Shield Support Stand.

Fig. 17. Comparison of Calculated and Measured Neutron Spectra Through 6 in. of Polyethylene at 0°.

Fig. 18. Comparison of Calculated and Measured Neutron Spectra Through 6 in. of Polyethylene at 30°.

Fig. 19. Comparison of Calculated and Measured Neutron Spectra Through 11 in. of Lead at 0°.

Fig. 20. Comparison of Calculated and Measured Neutron Spectra Through 11 in. of Lead at 13°.

and shape. Figure 21 is for the 11-in. lead at an angle of 64° , the largest angle measured. The intensity and shape are still quite good in the 1- to 5-MeV range. All experimental data were taken at least twice to check reproducibility of the data, which was good in all cases.

The calculational results presented here are considered to be preliminary in that the calculations used the readily available GAM-II cross sections [16]. The materials chosen exemplify both strong angle-energy correlation in polyethylene and highly anisotropic scattering in lead.

Comparisons with experimental results such as these will be used to evaluate future development work on the two-dimensional code. Since the basic methods appear to be accurate, the major effort will be toward increasing efficiency while maintaining or improving accuracy. The experimental work has already been extended to include laminated slabs of two materials and will be extended to include the measurement and calculation of capture gamma-ray spectra.

III. EXPERIMENTAL EVALUATION OF MINIMA IN TOTAL CROSS SECTIONS

A good geometry experiment was performed at the Tower Shielding Facility in which the spectra of the uncollided flux of neutrons transmitted through thick samples of shielding materials was measured [17]. Because the samples were quite thick, the transmitted spectra were influenced most strongly by the minima in the total cross sections. The spectrum incident on the sample, as well as the transmitted spectra, was measured using the NE-213 neutron spectrometer covering the energy range from 0.8 - 11 MeV. The total cross sections were evaluated by a direct comparison of calculated and experimental data. The uncollided spectra were calculated using the most detailed cross-section data available and then folded with the response function of the detector. Although the energy resolution of the spectrometer is rather broad, this technique still allows a determination of the errors in the very narrow minima in the total cross section, since the total number of transmitted neutrons are measured to within a few percent. Because the geometry of the experiment was very good, no correction for multiple scattering was required; therefore, the calculations were quite simple and straightforward, involving only the exponential attenuation of the incident spectra using the fine-mesh data for the total cross section. Experiments were done to demonstrate that multiple scattering was indeed negligible. The sample diameter was varied by a factor of 2 with no measurable change in the transmitted spectra.

Fig. 21. Comparison of Calculated and Measured Neutron Spectra Through 11 in. of Lead at 64°.

Good agreement between the measured and calculated spectra indicates that the valleys are adequately represented in a cross-section set. Where disagreement exists, detailed information about the specific region of disagreement is not possible, unless the minima are well separated, because of the poor energy resolution of the spectral measurements. Where a number of minima are close together only the general energy range in which the disagreement exists can be determined. Because of this, these measurements can in no way be described as a measurement of the total cross section but simply as an experimental evaluation of the existing cross-section data.

The capability for making spectral measurements such as those required in this experiment has only recently been acquired. It results from the fact that a high-sensitivity proton-recoil neutron spectrometer has been developed to the point that it can now be used in experiments involving large neutron attenuations. The spectrometer utilizes a 2- by 2-in. liquid organic scintillator (NE-213) and has an efficiency that ranges from 0.15 to 0.4 for neutrons in the MeV energy range. This allows it to be used in fluxes that are several orders of magnitude lower in intensity than those measurable with other spectrometers, all of which have been handicapped by low efficiencies (10^{-4} to 10^{-6}). The validity of the experimental technique was checked by measurements on thick samples of carbon, lead, and uranium that could be easily compared with calculations since the total cross sections for these elements are well known.

1. Description of the Experiment

In order to maximize the ratio of the uncollided flux at the detector to the collided flux, it was necessary to have a good-geometry experiment. This was achieved by locating the detector at a distance of 100 ft from the shield of the reactor and placing the sample at the approximate midpoint between the detector and the reactor shield. The arrangement is shown in Fig. 22.

The majority of the samples were cylinders positioned with their axes coincident with the axis of the neutron beam. Oxygen and nitrogen samples were in liquid form contained in either Dewar vessels or styrofoam-insulated brass cans.

Because neutrons scattered by the air and ground become important when the scattering sample is thick, both the detector and the neutron beam were shielded and very tightly collimated. The neutron beam incident on the sample passed

NOT TO SCALE. ALL DIMENSIONS IN INCHES EXCEPT AS NOTED.
BEAM CENTERLINE ~78 INCHES ABOVE CONCRETE PAD.

Fig. 22. Schematic Diagram of Uncollided Flux Experiment.

through two collimators: one which was adjacent to the reactor shield and another which was adjacent to the sample position and limited the beam to $3\frac{1}{2}$ in. in diameter. These collimators also served to reduce the air-scattered background at the detector without producing any observable change in the measured spectra incident on the samples.

Discrimination between proton- and electron-induced light pulses is based on differences in the pulse shapes. The discriminator circuit is an improvement of a design of Forté et al. [18]. It can label pulses from recoil protons having energies from 0.2 MeV (an approximately 0.025-MeV electron pulse) to 20 MeV (12-MeV electrons).

The neutron pulse-height distributions obtained were processed by a number of computer programs which reduced the data to normalized neutron spectra. The major computer program used is the unscrambling code (FERDoR) developed by Burrus [19]. This code requires accurate response functions (pulse-height distributions for monoenergetic neutrons) that were obtained from Monte Carlo calculations based on and checked by a number of calibration experiments [20]. Using these response functions, an a priori assumption of nonnegativity in the neutron spectrum, and otherwise classical extremal formulation, the FERDoR code gives upper and lower confidence limits which will encompass the true answer with a 68% probability. The error in the response functions must be estimated and the statistical uncertainties in the experimental results must be determined as inputs to the program.

An example of the structure obtainable by unfolding a pulse-height distribution is shown in Fig. 23 and typical results obtained for nearly monoenergetic neutrons are shown in Fig. 24.

The lowest energy at which neutron spectra may be obtained depends on the lower energy limit at which the pulse-shape gamma-ray discriminator will satisfactorily reject gamma-ray pulses. The efficiency of rejecting gamma rays depends on the ratio of gamma-ray counts to neutron counts. For a 10:1 ratio, satisfactory data can be obtained down to about 0.8 MeV. This ratio or less was obtained in the TSF experiment by placing from 2 to 4 in. of lead in the neutron beam at the sample position.

The spectrometer has an efficiency that ranges from 0.15 to 0.4 for neutrons in the MeV energy range. Its energy resolution varies from 32% at 1 MeV to 9% at 14 MeV. An integration of an unfolded neutron spectrum from a Po-Be source agreed within 5% with the absolute number of neutrons as determined from a United States National Bureau of Standards calibration.

Fig. 23. Typical Pulse-Height Curve and Corresponding Unfolded Spectrum.

Fig. 24. Neutron Spectra for Nearly Monoenergetic Neutrons.

2. Evaluation of Results

Table 2 summarizes the comparison on an absolute basis of measurements and calculations. The material and thicknesses for which data were obtained are listed in columns 1 and 2. The remainder of the table lists the source of the total cross sections which were used to calculate the uncollided spectra and comments on the comparison. The limitations of the evaluation as discussed above should serve as a basis for interpreting and understanding the comments in the last column.

For notation purposes "slight or small disagreements" between the calculated and measured spectra is used to denote differences between the measured and calculated spectra of less than 20% (generally less than 5% effect in the total cross section). This notation is used only if there is a disagreement in one energy range but agreement for higher and lower energies.

Where possible, comparisons have been made with ENDF/B cross sections (only tape 102 was available at this writing). Comparisons with ENDF/B cross sections for the other materials will be made when additional tapes are provided; however, cross sections for Ca, Si, K, and Pb will not be provided in the ENDF/B file in the near future [21].

In general there is good agreement between the measurement and the calculations. Typical results are shown in Figs. 25 through 34. For every material except calcium (see remarks in Table 2) there is a sizable energy range over which there is good agreement. In general the region of disagreement is below 3 MeV, where there is considerable structure in the cross sections. The calculated spectra are low for every case in which there is disagreement (calcium excluded); this indicates that minima in the cross section are not adequately represented in the cross-section set, due to either lack of energy detail or lack of resolution of the valleys.

This technique of evaluating minima in total cross sections has been very useful in deciding which evaluated set of total cross sections is adequate for shielding calculations and thereby permitting an efficient updating of cross-section libraries by updating only those cross sections that are not adequate. It also provided information about the energy regions in which new cross-section measurements should be made.

Table 2. Comparison of Calculated and Measured Spectra - Evaluation of Total Cross Sections

Sample	Thickness (cm)	Total Cross Sections Used in Calculations	Best Cross Section	Comments on Comparison of Spectra (Calculation Based on Best Cross Section)
Be	27.28	05R, ^a BNL, ^b ENDF/B ^c		Good agreement with all three cross sections; although there are some small differences, no one cross section is best
H ₂ O	20.32, 30.48	H: 05R; O: UNC, ^d 05R, KAPL ^e	O: KAPL	Good agreement for all energies; primarily another test of the oxygen cross sections (see oxygen sample)
LiH	10.16, 20.32, 15.24, 25.4	H: 05R; Li: 05R, ENDF/B		Good agreement for all energies; the Li cross sections in both 05R and ENDF/B are from the same source
Al	30.48	05R, ENDF/B	ENDF/B	Slight disagreement for energies between 1.6 and 3 MeV; cross section appears to be too high
Fe	20.32	05R, KFK ^f and BNL, ^g ENDF/B	KFK and BNL	Good agreement above 2.0 MeV but all cross sections are too high for lower energies
Ni	15.24	05R, BNL ^g and BNL, ^h FZK ⁱ	BNL ^g and BNL ^h	Good agreement above 2.5 MeV but cross section is too high for lower energies
Na	60.96	05R, ENDF/B, BNL ^b	ENDF/B	Slight disagreement in peaks at 1.7, 3.0, and 4.5 MeV
Ca	55.56 (chips)	05R, BNL ^b and ORNL ^j	BNL and ORNL	Sample was not "thick" because of voids; therefore the comparison is not very sensitive to cross section; approximately 20% disagreement in magnitude for all energies
K	76.2	05R, BNL ^b	BNL	Good agreement except for slight differences between 1 and 2 MeV
Mg	60.96	05R, BNL, ^b ENDF/B	BNL, ENDF/B	Good agreement for energies greater than 4 MeV; for lower energies the cross sections are too high
Zr	19.526, 39.0	05R, BNL ^g		Good agreement with both cross sections
Concrete	15.24, 30.48	O: KAPL; Others: 05R	Only set tried	Good agreement except for slight difference near 2.4 MeV, the valley regions of oxygen and carbon; requires further study
W	10.63	BNL ^h	Only one available	Good agreement except for energies less than 2.0 MeV
Sand (SiO ₂)	55.56	Si: 05R; O: KAPL	Only set tried	Slight disagreement for energies around 2.4 MeV, the valley region of oxygen; requires further study
U ²³⁸	8.69	05R, ENDF/B		Very good agreement for both cross sections for all energies

Sample	Thickness (cm)	Total Cross Sections Used in Calculations	Best Cross Section	Comments on Comparison of Spectra (Calculation Based on Best Cross Section)
Lead	2.45, 5.08, 7.62, 10.16, 20.32, 30.48	BNL, ^h BNL ^h and ORNL ^k	BNL and ORNL	Very good agreement at all energies when Pb ²⁰⁸ cross sections ^k are used for energies less than 4.3 MeV; disagreement (see ref. 17 in text) for energies less than 2.0 MeV was eliminated
C	10.64, 20.32, 30.48	05R, ENDF/B	ENDF/B	Good agreement for both cross sections, with ENDF/B slightly better around 8 MeV (see ref. 17 in text)
N	60.96, 91.44	05R, UNC, ORNL ⁱ	ORNL	See ref. 17 in text
O	60.96, 91.44, 152.4, 182.88	05R, UNC, KAPL	KAPL	See ref. 17 in text; good agreement even for 5- and 6-ft samples
Ni + H ₂ O	Ni: 10.16; H ₂ O: 15.24	Ni: 05R, BNL ^g and BNL, ^h H: 05R; O: KAPL	Ni: BNL ^g and BNL, ^h O: KAPL	Primarily another test of nickel cross sec- tion; disagreement for energies less than 2.5 MeV

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Fig. 25. Neutron Spectra for Carbon.

Fig. 26. Neutron Spectra for Lead.

Fig. 27. Neutron Spectra for Uranium.

Fig. 28. Neutron Spectra for Tungsten.

Fig. 29. Neutron Spectra for Lithium Hydride.

Fig. 30. Neutron Spectra for Magnesium.

Fig. 31. Neutron Spectra for Iron.

Fig. 32. Neutron Spectra for Oxygen.

Fig. 33. Neutron Spectra for Concrete.

Fig. 34. Neutron Spectra for Sand.

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