American Nuclear Society
9th Annual Meeting
Salt Lake City, Utah
June 17-19, 1963

NSA 2C

14 P.

GD

FZM-2936
10 June 1963

CONF-39-19

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Paper to be Presented at the 1963 Annual
Meeting of the American Nuclear Society
Salt Lake City, Utah

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General Dynamics/Fort Worth
Nuclear Aerospace Research Facility

THE CALCULATION OF FAST-NEUTRON ATTENUATION PROBABILITIES THROUGH A NINE-INCH POLYETHYLENE SLAB AND COMPARISON WITH EXPERIMENTAL DATA

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Abstract

Calculations of neutron penetration probabilities were performed to evaluate the Monte Carlo Multilayer Slab Penetration Procedure developed at the Nuclear Aerospace Research Facility of General Dynamics/Fort Worth under contract with the Army Tank Automotive Center. A 9-in. polyethylene slab was chosen for the calculations and results were compared with experimental data measured during an Angular and Energy Distribution Experiment at NARF. The calculated and measured dose rates agree within 20% for all exit polar angles. The calculations indicate that incident neutrons with energies less than 2.5 Mev do not contribute significantly to the transmitted dose rate.

Introduction

This paper reports the results of calculations performed at NARF. Use was made of a Monte Carlo Multilayer Slab Penetration Procedure[1], designated C-18. Information for the source term input was derived from experimental data. Calculations were made of (1) the fast-neutron flux distribution in a 9-in. polyethylene slab, (2) the total fast-neutron leakage from the slab, and (3) the dose

[1]Wells, M. B., Radiation Resistant Combat Vehicle Investigation - Final Report, Volume III: Monte Carlo Multilayer Slab Geometry Shielding Code C-18. GD/FW Report FZK-134-3 (December 1961).

*Sponsor: H. R. Dvorak

rate as a function of polar angle θ on an arc 192 inches from the slab center. These data were compared with measurements made during an energy and angular distribution experiment[2] performed previously at NARF.

Included in this paper are explanations of the experimental geometry, a description of the calculation procedure, and presentation and comparison of measured and calculated data.

Experimental Geometry

The Aerospace Systems Test Reactor (ASTR)[3] was positioned within the Outside ASTR Tank (OAT) to serve as the source of radiation for the experiment. Shown in Figure 1 is the internal experimental geometry of the OAT.

Preliminary requirements of the experimental geometry were that an infinite-slab condition be closely approximated and that radiation fluxes of sufficient intensity to obtain reliable data be available. These requirements were accomplished by the 6-in.-diam, 76-in.-long collimator shown in Figure 2. Details and effectiveness of the experimental geometry are discussed in Reference 2.

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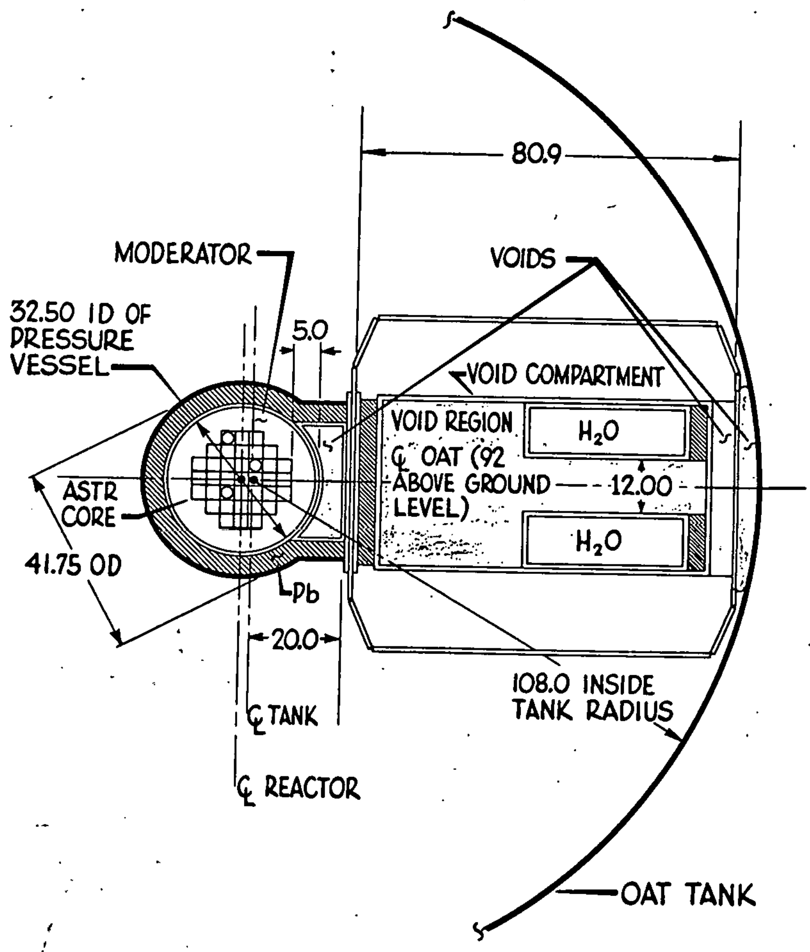
[2] Western, G. T., Energy and Angular Distribution Experiment, Vol. I: Angular Distribution of Reactor Radiation from Slabs and of Emergent Secondary Gamma Rays. GD/FW Report FZK-9-183-1 (31 December 1962).

[3] The 3-Mw Aircraft Shield Test Reactor. GD/FW Report FZK-9-161 (NARF-61-19R, August 1961).



PLAN VIEW

Figure 1. Internal Geometry of the OAT

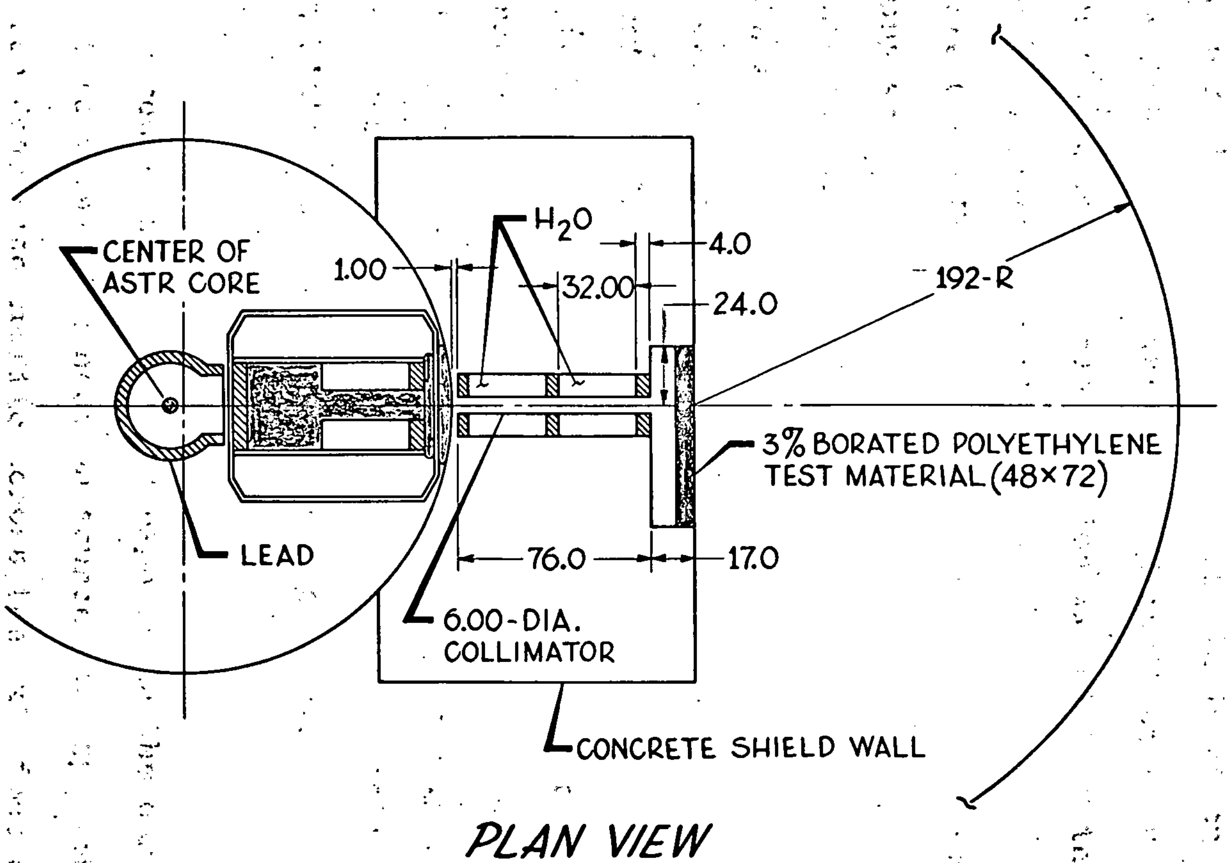


Figure 2. Total OAT Geometry

Calculation Procedure

A plane parallel beam of neutrons was incident on the slab normal to the slab surface. The energy distribution of the neutrons was continuous; however, the distribution was divided into four energy regions for calculational purposes.

The Monte Carlo Multilayer Slab Penetration Procedure may contain as many as five slabs of different composition and a total of ten elements in the five slabs. This, of course, far exceeds the requirements needed for the single 9-in. polyethylene shield. The experimental measurements were made through polyethylene containing 3% boron by weight. The calculations were made through unborated polyethylene (CH_2) having a density of 4.3×10^{22} carbon atoms and 8.6×10^{22} hydrogen atoms per cubic centimeter.

For calculation purposes each of the four energy regions was considered as a separate problem. To ensure good statistical results, 4000 histories were run for each problem.

Source Term

A nuclear research emulsion was exposed in the center of the collimator exit to obtain the neutron source-term spectrum. The measured neutron dose-rate profile across the exit face of the collimator (zero-thickness curve, Fig. 3) and the neutron flux of 1.38 neutrons/cm^2-sec-watt, as determined from measuring 1400 proton recoil tracks in the emulsion, were combined in an integration over the collimator exit to give a total neutron leakage of 370 neutrons/sec-watt.

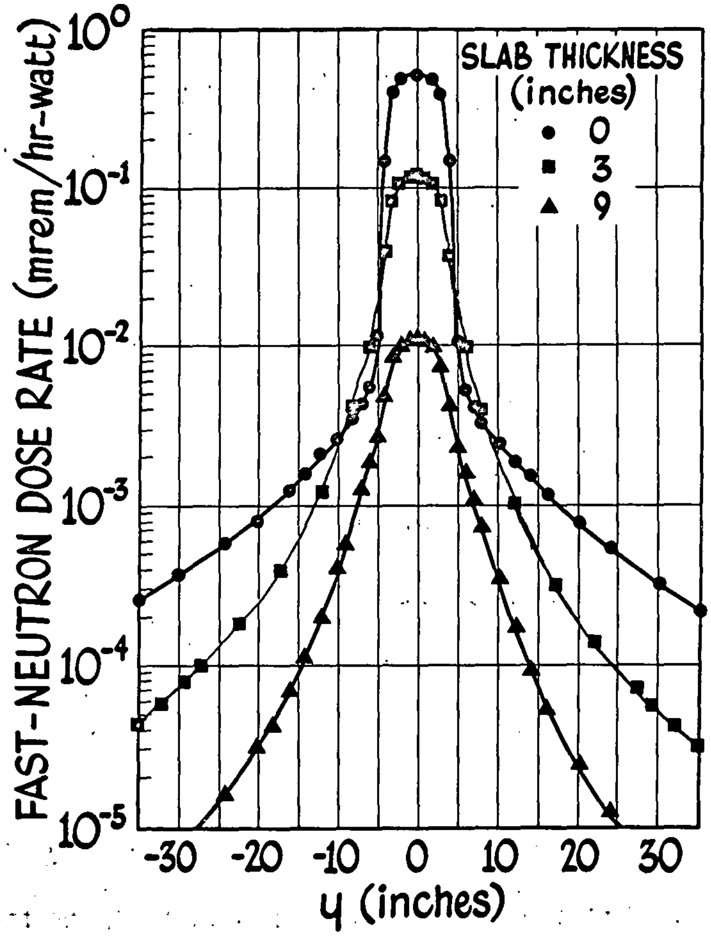


Figure 3. Fast-Neutron Dose-Rate Map across Exit Face of Particular Thicknesses of 3% Borated Polyethylene Slabs

The four energy regions into which the source spectrum was divided are shown in Figure 4. These regions are 1.5 - 2.5, 2.5 - 4.0, 4.0 - 6.0, and 6.0 - 10.0 Mev; the neutron flux in these regions was 91.0, 86.0, 87.0, and 106.0 neutrons/sec-watt, respectively. The energy widths of the regions were selected to compensate for the usual decrease in flux with increasing energy.

The solid curve in Figure 4 represents the experimentally measured data, while the dashed lines represent a fission curve adjusted in each region to encompass the same total flux as the measured curve.

To use the measured data as input would have restricted the use of the calculated penetration probabilities to problems of similar source-term spectra; therefore, the decision was made to substitute the fission spectrum within each region. The validity of the substitution was justified on the basis of a generally close agreement between the measured spectrum and the fission spectrum within each energy region. Neutron spectra measured at NARF have, within experimental error, the same curve shape above 6 Mev as that of a fission spectrum. The practice of substituting fission shapes in problems of this nature is considered a "standardization" procedure.

Calculation Results and Comparison with Measured Data

Data output from the C-18 procedure is in the form of (1) a flux-distribution table which gives the neutron flux as a function

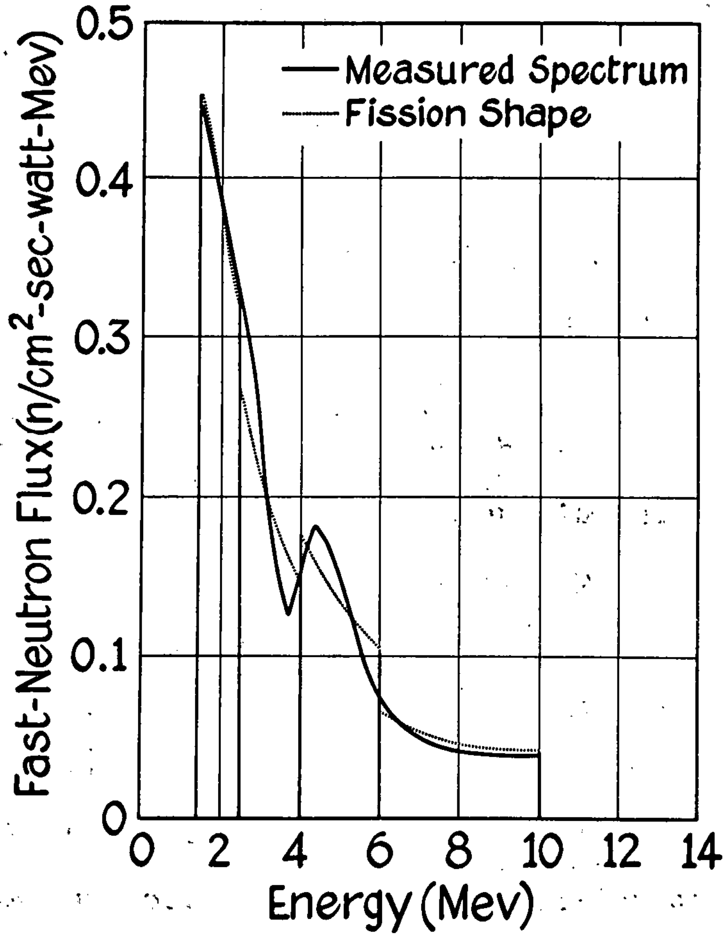


Figure 4. Fast-Neutron Source-Term Spectrum

of energy for 50 equal increments of slab thickness, (2) a dose-rate table which gives the leakage dose rate as a function of polar angle θ in 10-degree increments from 0 to 90 degrees, and (3) a total dose rate attributed to the uncollided neutrons which penetrate the slab.

The flux-distribution table may be used as library input for the calculation of secondary radiation production. It is of interest to compare the calculated flux distribution with measured data. Figure 5 shows such a comparison for neutrons above 2.9 Mev. Agreement is fair in both shape and magnitude. The total flux at the three material depths of 0, 3, and 9 inches was obtained from a combination of centerline flux data and the three dose-rate profiles shown in Figure 3. An assumption made here is that the flux profile is similar to the dose-rate profile, an assumption that should be reasonably valid for total flux.

The dose rate from the uncollided neutrons and that from the scattered neutrons are compared with experimental data in Figure 6. An examination of the experimental curve in the θ region of 0 - 5 degrees indicates that portions of the uncollided dose are distributed from $\theta = 0$ to $\theta = 4$ degrees. To facilitate the comparison of measured and calculated data in this region, the total calculated dose rate was distributed according to the measured curve.

A close examination of the data in Figure 6 reveals the excellent agreement in both shape and magnitude. Confidence limits for the calculated data vary from $\pm 7\%$ at the low θ angles to $\pm 13\%$

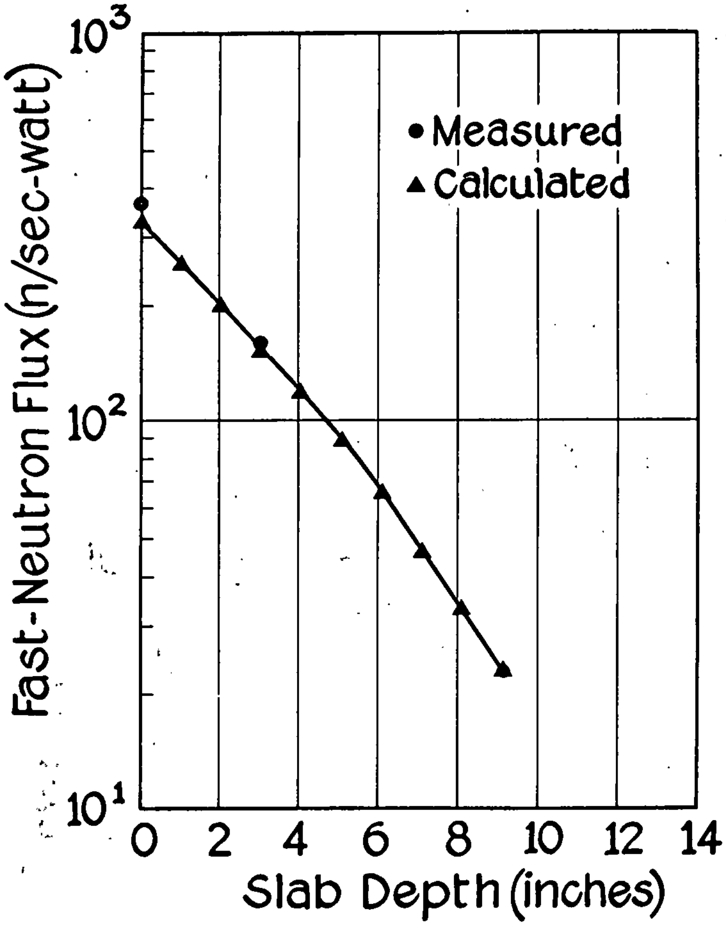


Figure 5. Comparison of Calculated and Measured Flux Distribution through Nine-Inch Polyethylene Slab

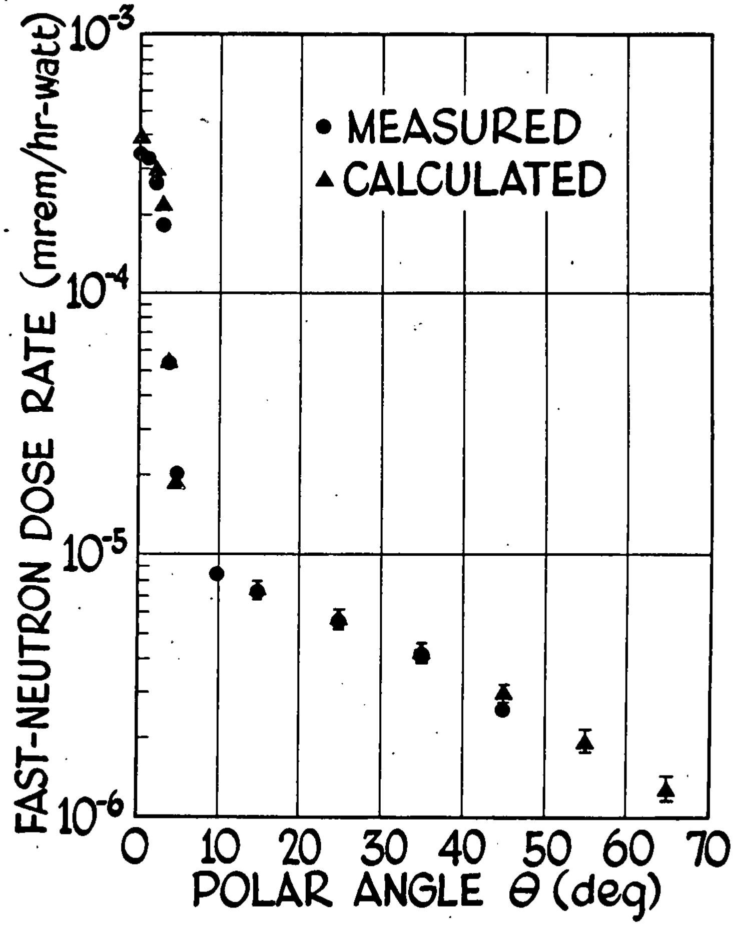


Figure 6. Comparison of Calculated and Measured Fast-Neutron Dose-Rate as a Function of θ

at the high θ values. An examination of the calculated data shows the uncollided dose rate to be approximately 13% of the total. An integration over both curves in Figure 6 gives the calculated total dose rate to be approximately 17% higher than the measured total dose rate.

Table 1 gives the tabulated values of the calculated and measured dose rates as a function of θ. Also included are the confidence limits assigned to the calculated data.

Table 2 shows the contribution to the transmitted dose rate attributed to neutrons in each of the four input energy regions. Approximately 99% of the emergent dose rate from the polyethylene slab is due to input neutrons of energy greater than 2.5 Mev. From this it is reasonable to assume that no serious error would result in deleting input neutrons of energies less than 2.5 Mev.

Table 1

Calculated and Measured Dose Rates as a Function of Θ (mrem/hr-watt)*

Θ (deg)	Calculated		Measured Dose Rate
	Dose Rate	Confidence Limits	
0	3.95(-4)**		3.4(-4)
1	3.85(-4)		3.3(-4)
2	3.00(-4)		2.6(-4)
3	2.20(-4)		1.8(-4)
4	5.40(-5)		5.4(-5)
5	1.83(-5)		2.0(-5)
10	-		8.5(-6)
15	7.40(-6)	±6.4%	7.3(-6)
25	5.67(-6)	±6.8%	5.6(-6)
35	4.30(-6)	±6.8%	4.0(-6)
45	2.94(-6)	±7.1%	2.7(-6)
55	2.00(-6)	±9.7%	-
65	1.29(-6)	±11.0%	-
75	9.90(-7)	±14.4%	-
85	8.10(-7)	±40.5%	-

Table 2

Dose-Rate Contribution from the Various Energy Groups (mrem/hr-watt)

Incident Neutron Energy E (Mev)	Scattered Dose Rate	Direct Dose Rate	Total Dose Rate at Energy E	Percent of Total Dose Rate
1.5 - 2.5	4.48(-2)	4.91(-3)	4.97(-2)	1.10
2.5 - 4.0	1.91(-1)	2.11(-2)	2.12(-1)	4.70
4.0 - 6.0	8.29(-1)	1.39(-1)	9.68(-1)	21.4
6.0 - 10.0	2.86(0)	4.42(-1)	3.30(0)	72.8
		Total Dose Rate	4.53(0)	

*Neutron RBE of 10

**Read 3.95(-4) as 3.95×10^{-4}