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CALCULATION OF FISSION-SOURCE THERMAL-NEUTRON DISTRIBUTION IN WATER BY THE TRANSFUSION METHOD

D. K. Trubey

OAK RIDGE NATIONAL LABORATORY
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

Printed in USA. Price: $1.25 Available from the
Office of Technical Services
U. S. Department of Commerce
Washington 25, D. C.

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ORNL-3487

Contract No. W-7405-eng-26

Neutron Physics Division

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AUGUST 1964

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TABLE OF CONTENTS

	Page
Abstract	1
Introduction	1
The Transfusion Method	5
Comparison of Calculated and Experimental Results	15
Conclusions	25
Acknowledgments	25
References	26
Appendix A: Hydrogen Total Cross Sections	28
Appendix B: Differential Number Spectra for Point Fission Source in Water	29
Appendix C: Fast-Neutron Dose-Rate Calculation	47

CALCULATION OF FISSION-SOURCE THERMAL-NEUTRON DISTRIBUTION IN WATER BY THE TRANSFUSION METHOD

D. K. Trubey

Abstract

A calculational method which combines transport theory and diffusion theory, called the "transfusion" method, has been used to obtain the thermal-neutron flux from a point isotropic fission source in water. The transport theory code RENUPAK, which utilizes the moments method, was used to calculate the energy-spatial neutron distribution above 1 keV, and the multigroup diffusion code MODRIC was used to continue the calculation down to thermal energies. The resulting thermal-neutron fluxes agree very well with experimental results for distances out to 70 cm from the source, but the calculated fluxes are as much as 40% higher for greater distances.

Introduction

There is a continuing interest in the accurate calculation of low-energy neutron distributions in reactor shields because of the large contribution to the shield leakage of the secondary gamma rays produced by these neutrons. With the development of large fast computers many numerical techniques and machine programs have been developed for solving the neutron slowing-down problem in reactor shields, the programs generally taking the form of either transport theory codes or multigroup diffusion theory codes. The best transport codes in use are designed to solve the problem by incorporating a physical model which is as exact as possible and by using the best detailed cross sections available. The accuracy of the neutron flux at high energy is likely to be satisfactory and often can be achieved with a small amount of machine time. To be at

low energy, however, a neutron must have had its energy degraded through many decades, which means that a calculation must account for many collisions. Consequently, the results are less accurate and a large amount of computer time is required.

In general, the diffusion codes are unsuitable for calculations at high energy but with proper sources are likely to give good results at low energy. For deep penetrations, however, even the diffusion codes are not well suited for calculating low-energy fluxes, because neutrons observed deep in a shield originally arrived there either unscattered or after having suffered only a few small-angle scatterings. Thus they arrived at high energies with a highly peaked angular distribution, which is not in accord with the assumptions of diffusion theory. (An angular distribution measured by Beissner and Western,[1] shown in Fig. 1, clearly shows the peaked component and the multiply scattered, or "built-up" component.) It is only at low energies that the neutron flux and the cross sections are nearly isotropic and the diffusion theory becomes applicable. It would appear, then, that with the proper combination of the transport and diffusion methods a technique could be developed for accurately predicting thermal-neutron flux distributions deep in a shield. The "transfusion" method,[2] which is described in this report, was developed for the express purpose of combining the two neutron slowing-down methods.

The central assumption of the model is that, at some energy, neutrons cross over from the transport part of the calculation to the diffusion part. The crossover energy for shifting from transport theory to diffusion theory was set in the vicinity of 1 keV. Essentially all fission neutrons

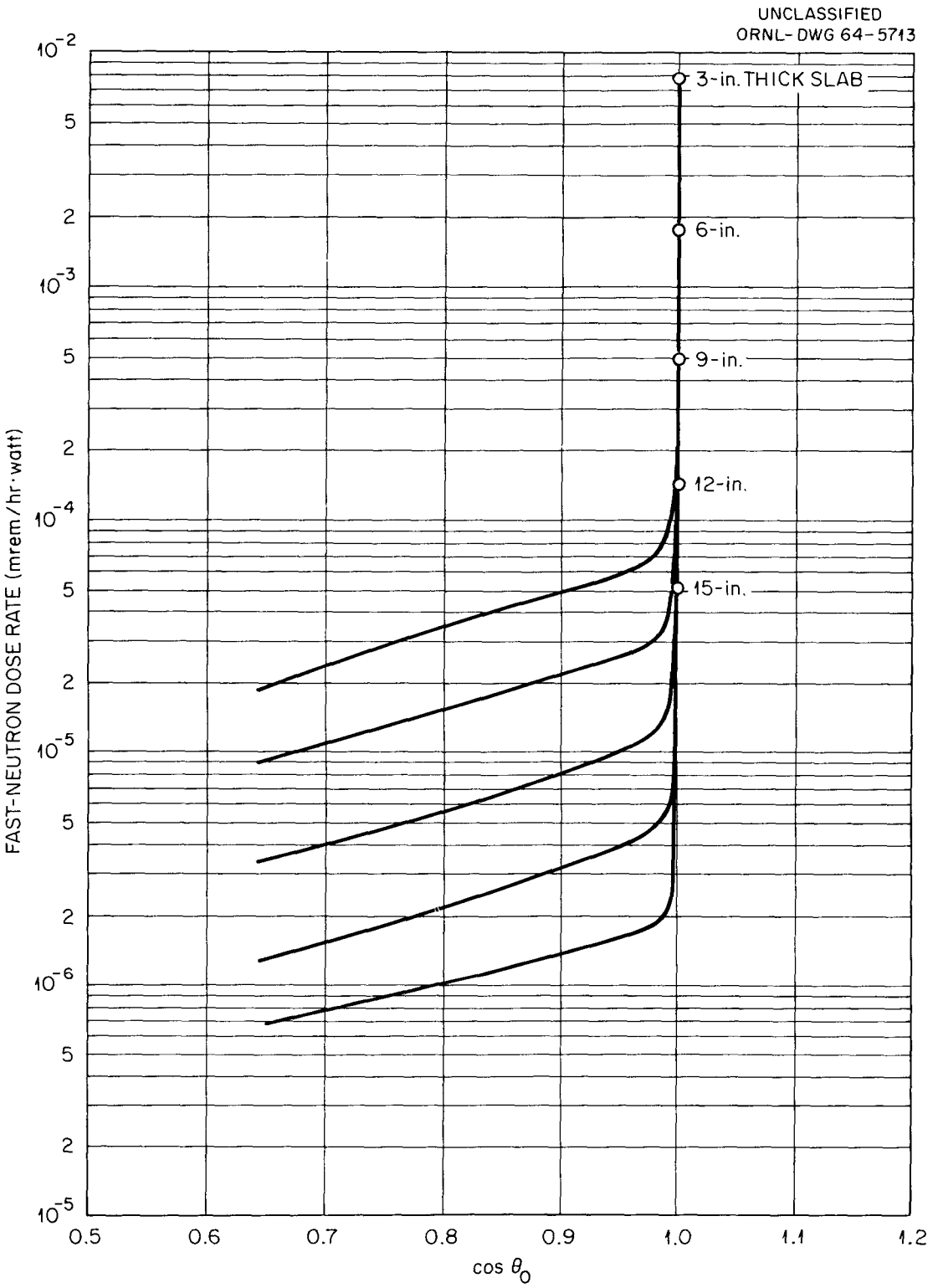

Fig. 1. Measured Fast-Neutron Angular Distribution Through Polyethylene Slabs. (Reprinted from ANS Transactions 5, No. 2, p 403, 1962, through the courtesy of the American Nuclear Society.)

are born above this energy, and in most shields essentially all absorption is below this energy. Hence the slowing-down density at the crossover energy is computed by means of the transport theory. Neutrons that slow down past the crossover energy, below which the scattering cross sections are essentially isotropic, are then used as source neutrons in the diffusion calculation. Actually, since most of the transport is accomplished above 1 keV, the treatment below this energy can be crude unless there are severe inhomogeneities in the shield.

There have been other calculational models used to separate the high- and low-energy components in a neutron transport calculation. As early as 1952 Blizard[3] suggested using the Gaussian slowing down from the first collision, thus predicting the thermal-neutron flux from the fast-neutron current and the average Fermi age to thermal after removal. And in 1960 Avery et al.[4] of the United Kingdom described a technique that they had developed for predicting thermal-neutron fluxes in concrete and other types of shields. They used removal theory for high-energy neutrons, calculating the flux by means of an "uncollided" type of kernel characterized by an energy-dependent removal cross section, and then applied a multigroup slowing-down process. In later work by Butler[5] their model was extended to use an overlapping energy structure of the removal and diffusion neutrons. This later model, as represented by the RASH E Code, yields improved results. This technique has been called the "Spinney method."

The transfusion method, however, does not resort to approximate models, but requires a complete transport calculation which in many cases will be

feasible with present-day computing machinery. This is demonstrated below by a transfusion calculation of the thermal-neutron flux distribution in water due to a fission source, which was chosen because good experimental data are available for comparison. The results are also compared with a calculation by the diffusion method.

The Transfusion Method

Examples of transport theory computer codes that can be used for the high-energy portion of the transfusion calculation are RENUPAK[6] (moments method), NIOBE[7] (numerical integration of the Boltzmann equation), and Monte Carlo codes. For the example given here, RENUPAK was chosen. The cross sections used with the code were supplied on punched cards by United Nuclear Corporation, who originated the code. The hydrogen cross-section values are shown in Appendix A, and the oxygen cross-section data can be found in earlier publications.[8] Inelastic scattering was treated as absorption.

First, the spatial moments were calculated for the case of a point source in water, with the source spectrum assumed to be a Watt-type fission spectrum with the parameters of Cranberg:[9]

$$0.453\ e^{-E/0.965}\ \sinh\sqrt{2.29E}\ .$$

Then moments were fitted to the expression

$$N(R,E) = 4\pi R^2\ \phi(R,E)$$

$$= R \sum_{i=1}^{M} \frac{A_i}{B_i}\ e^{-R/\sqrt{B_i}} \text{ neutrons sec}^{-1}\ \text{MeV}^{-1} \text{ per source neutron/sec,}$$

where R is the distance from the source, $\phi(R,E)$ is the flux at R per unit energy from the source, and A and B are fitting parameters. This fit is

called the "exponential fit" and for the case considered gives a more satisfactory spatial behavior of N(R,E) than the "Gaussian fit," which is also available in RENUPAK. The value of M was generally 3, but at some energies it was 2. Values of A_i, B_i, and N(R,E) were recorded on magnetic tape and a special code[10] was written to read the tape for further processing of the data.

The resulting values of N(R,E) for 26 space points and 101 energy points are given in Appendix B. The energy grid is spaced at a uniform lethargy step of 0.1, starting at approximately 18 MeV. The values of N(R,E) are quite similar, though more detailed, to moments method results published previously,[11] as is apparent from the comparison of a few selected points shown in Table 1. Plots of N(R,E) are presented in Figs. 2 through 4 as a function of energy for certain distances and in Fig. 5 as a function of distance for certain energies. Although not pertinent to the transfusion calculation as such, fast-neutron dose rates, which are readily obtainable from the N(R,E) values and any response function, were computed for several response functions (see Appendix C).

It is pointed out that the exponential fit used in the calculation requires that N(R,E) approach zero as R approaches zero, which is not correct for the energy range where there are uncollided neutrons. Therefore for this and other reasons[12] the data close to the source cannot be regarded as accurate.

With the high-energy spatial distribution available, the neutrons arriving at low energies at each space point could be computed. In this calculation it was assumed that the energy distribution below E_c (see Fig. 6) could be computed with the model of isotropic scattering in the center of mass. This should be a good model for water since it is accurate for hydrogen

Table 1. Comparison of Energy Spectra N(R,E) Calculated by RENUPAK and by Earlier Moments Method Calculation

Energy (MeV)	Energy Spectra (neutrons sec^{-1} MeV^{-1}) at R = 20 cm: Moments Method[a]	at R = 20 cm: RENUPAK	at R = 120 cm: Moments Method[a]	at R = 120 cm: RENUPAK
10.9	8.77×10^{-5}	7.98×10^{-5}	1.82×10^{-8}	2.00×10^{-8}
6.0	2.90×10^{-3}	2.55×10^{-3}	1.68×10^{-7}	1.42×10^{-7}
2.7	1.62×10^{-2}	1.60×10^{-2}	2.16×10^{-7}	2.45×10^{-7}
1.1	3.14×10^{-2}	2.46×10^{-2}	2.99×10^{-7}	2.96×10^{-7}
0.33	5.06×10^{-2}	5.51×10^{-2}	5.37×10^{-7}	6.17×10^{-7}

[a]See ref. 11.

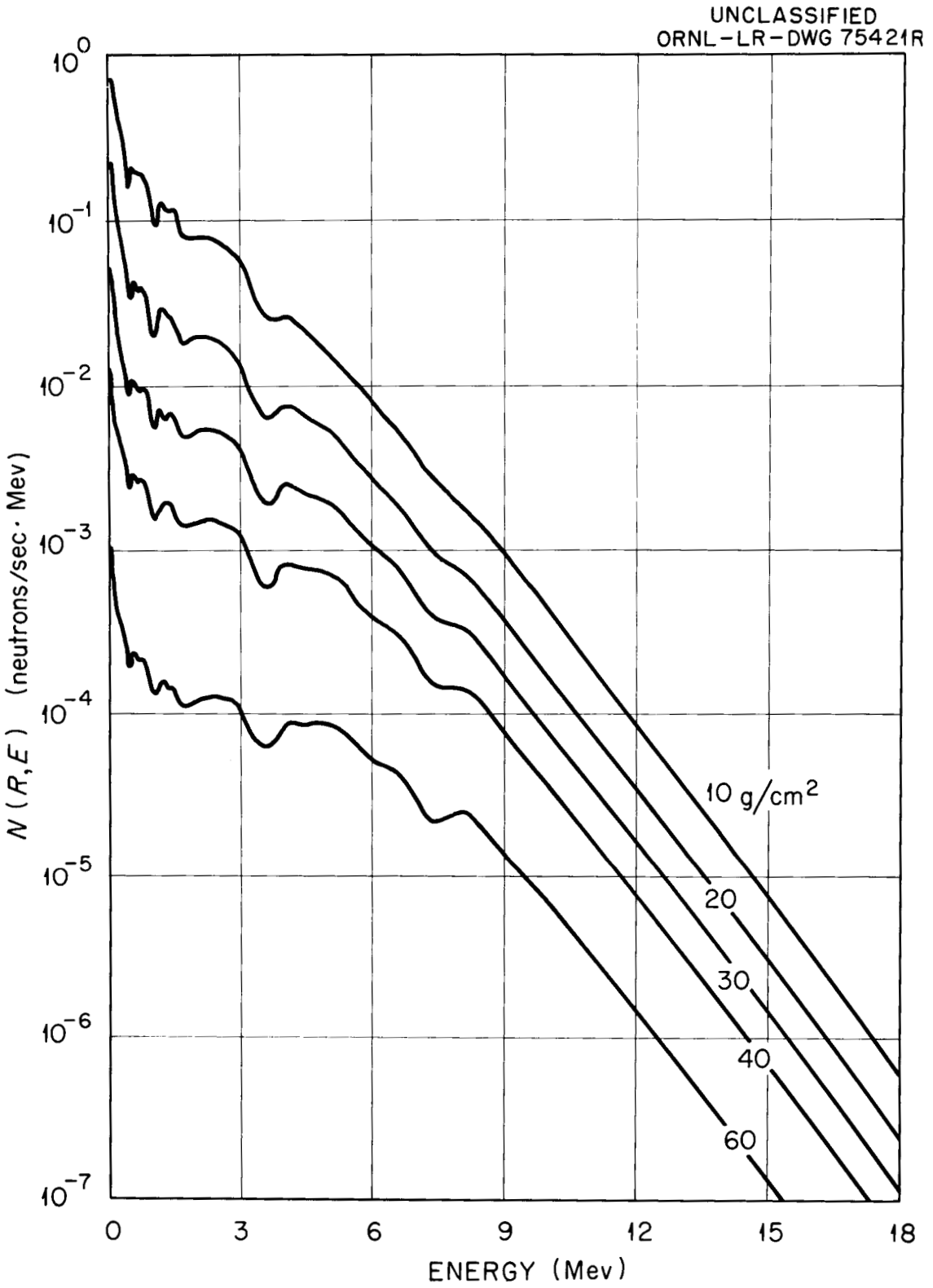

Fig. 2. Neutron Energy Spectra in Water for Distances 10 to 60 cm from a Point Source (Semilog Plot).

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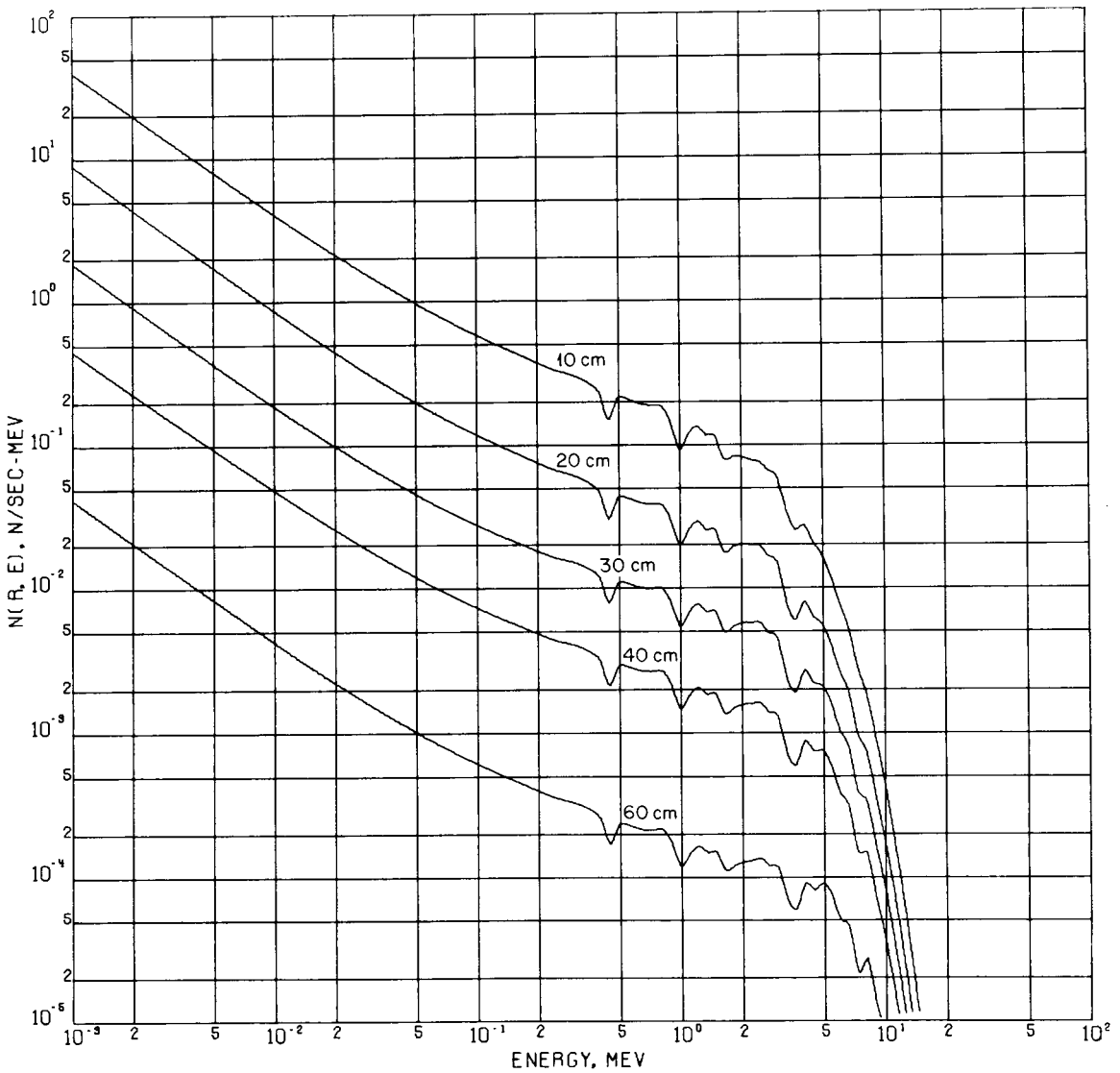

Fig. 3. Neutron Energy Spectra in Water for Distances 10 to 60 cm from a Point Source (Log-Log Plot).

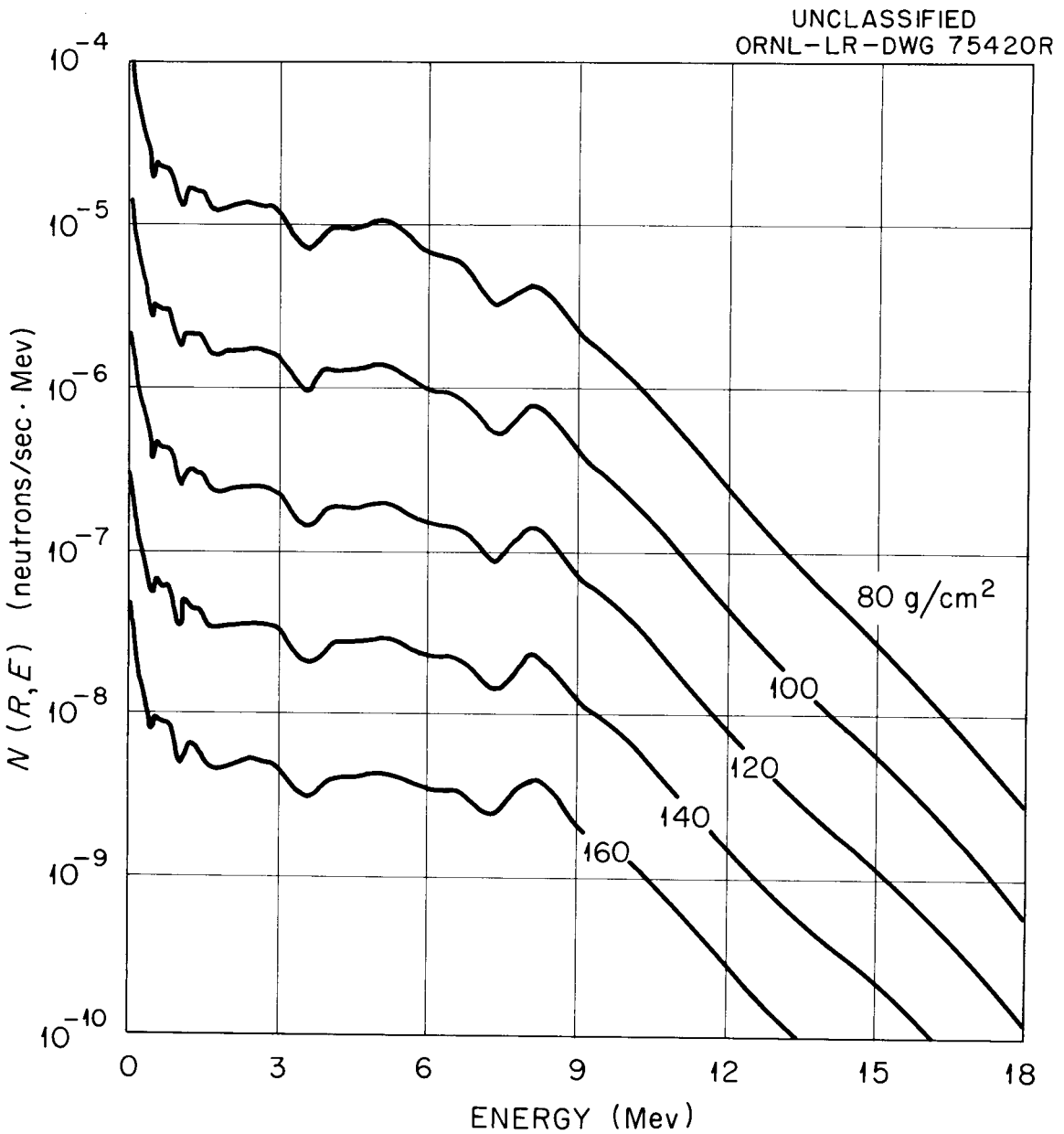

Fig. 4. Neutron Energy Spectra in Water for Distances 80 to 160 cm from a Point Source.

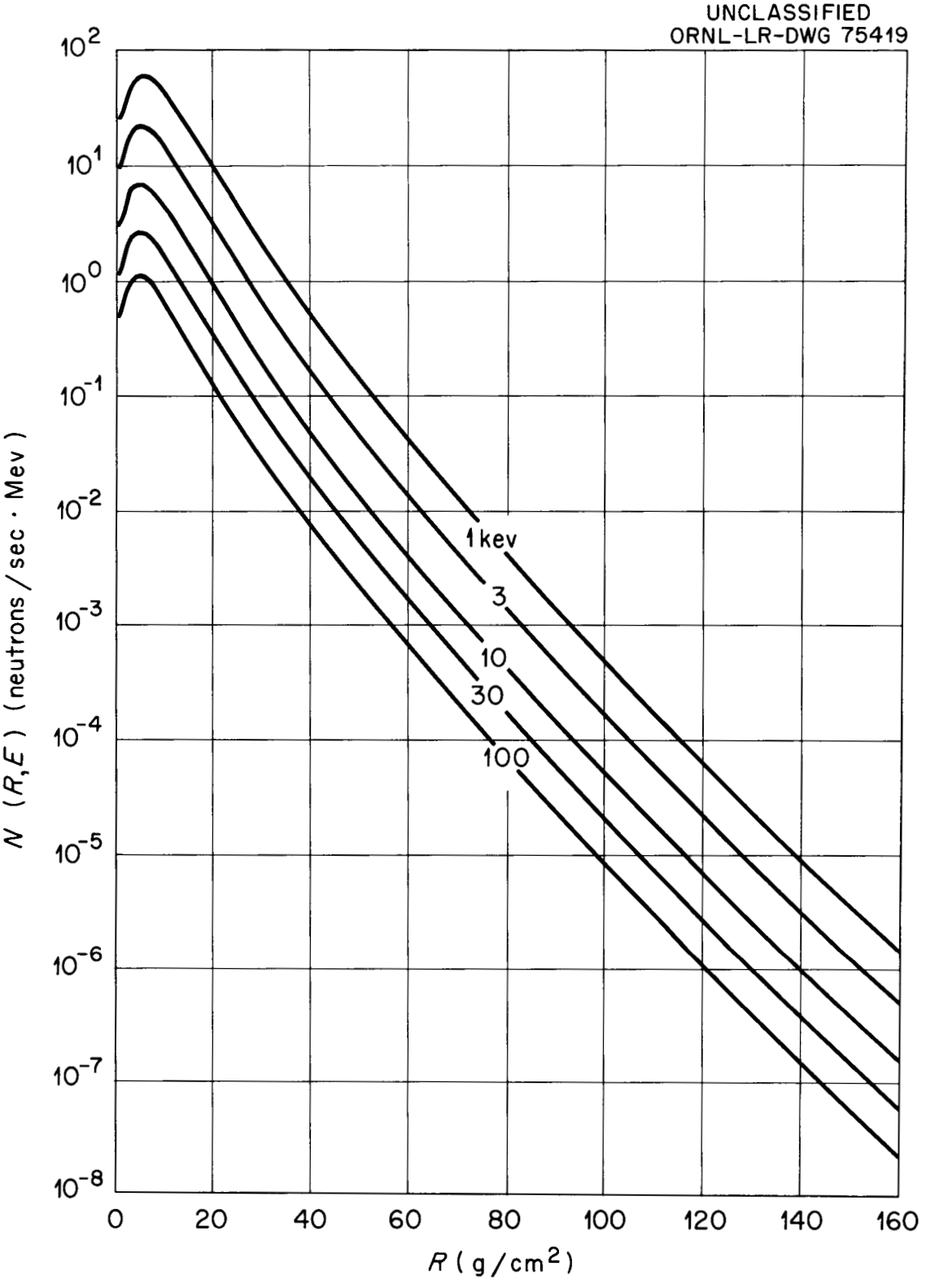

Fig. 5. Neutron Flux as a Function of Distance for Energies 1 to 100 keV.

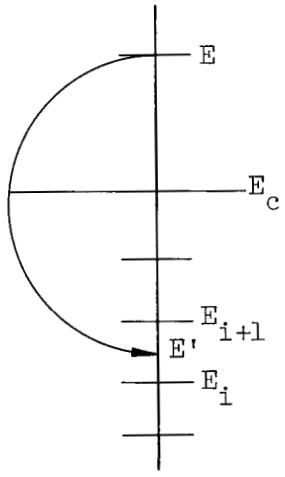

Fig. 6. Crossover Energy Diagram

at all energies and accurate for oxygen at energies below about 0.25 MeV. The angular distribution was assumed to be isotropic in the laboratory system since no angular flux information is available from the RENUPAK calculation.

The number of neutrons arriving at a space point R in the energy group defined by E_i and E_{i+1} was obtained (for one nuclide) from the relation

$$S_i(R) = \int_{E=E_c}^{E_c/\alpha} \frac{\phi(R,E)\ \Sigma_s(E)\ dE}{(1 - \alpha)\ E} \int_{E'=E_i}^{E_{i+1}} f(E',E)\ dE' ,$$

where

$\Sigma_s(E)$ = macroscopic scattering cross section (cm^{-1}),

$\alpha = (A - 1)^2 / (A + 1)^2$,

A = atomic weight,

$f(E',E) = 1$ for $E' \geq \alpha E$

$= 0$ for $E' < \alpha E$.

Integrating over E' gave

$$S_i(R) = \int_{E_c}^{E_c/\alpha} \frac{\phi(E,R)\ \Sigma_s(E)}{(1 - \alpha)\ E}\ g_i(E)\ dE,$$

where

$$\begin{aligned} g_i(E) &= E_{i+1} - E_i && \text{for } \alpha E < E_i \\ &= E_{i+1} - \alpha E && \text{for } E_i \leq \alpha E \leq E_{i+1} \\ &= 0 && \text{for } \alpha E > E_{i+1}\ . \end{aligned}$$

This single integration was performed numerically from E_c to 18 MeV by using 16-point Gaussian quadrature[13] in a series of arbitrary energy steps. The first step was from E_c to E_c/α, where α was evaluated for oxygen. Above this energy only hydrogen scattering makes a contribution. An E_c value of 0.001 MeV was used in one case, and, as a check on the results, a value of 0.01 MeV was used in a second case. In both cases the group structure used was that reported by Nestor[14] and shown in Table 2 (only groups 8-24 were used). Over the spatial range considered, the thermal-neutron fluxes obtained for the two cases differed from about 1 to 5%.

The resulting values of S_i for either 23 or 25 groups at the space points shown in Appendix B were interpolated to provide values at 226 space points between 0 and 160 cm. As a check, the distribution was integrated over space, which should have given a value of 1.0 since there is essentially no absorption above 1 keV. The values obtained for the two cases were 0.994 and 0.979. These were regarded as satisfactory, and

Table 2. MODRIC Group Structure

Group	Lethargy Width	Energy (eV)
1	0.916	$4.0 \times 10^6 - 1.0 \times 10^7$
2	0.693	$2.0 \times 10^6 - 4.0 \times 10^6$
3	0.693	$1.0 \times 10^6 - 2.0 \times 10^6$
4	1.204	$3.0 \times 10^5 - 1.0 \times 10^6$
5	1.099	$1.0 \times 10^5 - 3.0 \times 10^5$
6	1.204	$3.0 \times 10^4 - 1.0 \times 10^5$
7	1.099	$1.0 \times 10^4 - 3.0 \times 10^4$
8	1.204	$3.0 \times 10^3 - 1.0 \times 10^4$
9	1.099	$1.0 \times 10^3 - 3.0 \times 10^3$
10	0.916	$4.0 \times 10^2 - 1.0 \times 10^3$
11	0.981	$1.5 \times 10^2 - 4.0 \times 10^2$
12	0.405	$1.0 \times 10^2 - 1.5 \times 10^2$
13	0.105	$9.0 \times 10^1 - 1.0 \times 10^2$
14	0.118	$8.0 \times 10^1 - 9.0 \times 10^1$
15	0.208	$6.5 \times 10^1 - 8.0 \times 10^1$
16	0.262	$5.0 \times 10^1 - 6.5 \times 10^1$
17	0.105	$4.5 \times 10^1 - 5.0 \times 10^1$
18	0.195	$3.7 \times 10^1 - 4.5 \times 10^1$
19	0.114	$3.3 \times 10^1 - 3.7 \times 10^1$
20	0.095	$3.0 \times 10^1 - 3.3 \times 10^1$
21	0.182	$2.5 \times 10^1 - 3.0 \times 10^1$
22	0.223	$2.0 \times 10^1 - 2.5 \times 10^1$
23	0.163	$1.7 \times 10^1 - 2.0 \times 10^1$
24	0.231	$1.3 \times 10^1 - 1.7 \times 10^1$
25	0.300	$1.0 \times 10^1 - 1.3 \times 10^1$
26	0.288	$7.5 \times 10^0 - 1.0 \times 10^1$
27	0.310	$5.5 \times 10^0 - 7.5 \times 10^0$
28	0.318	$4.0 \times 10^0 - 5.5 \times 10^0$
29	0.470	$2.5 \times 10^0 - 4.0 \times 10^0$
30	0.580	$1.4 \times 10^0 - 2.5 \times 10^0$
31	0.560	$8.0 \times 10^{-1} - 1.4 \times 10^0$
32	0.288	$6.0 \times 10^{-1} - 8.0 \times 10^{-1}$
33	1.5	$1.4 \times 10^{-2} - 6.0 \times 10^{-1}$
34	Thermal	Thermal

the interpolated values were therefore written on magnetic tape to provide a source distribution for the low-energy portion of the transfusion calculation.

The low-energy portion of the calculation, that is, the calculation for neutrons with energies less than 1 keV, was performed with the MODRIC code,[15] a one-dimensional multigroup diffusion code written for the IBM-7090. The cross sections used were those generated by Nestor.[14] The resulting thermal-neutron fluxes, labeled "transfusion," are compared in Figs. 7 and 8 with fluxes obtained in a calculation performed entirely by MODRIC (labeled "diffusion"). For the latter calculation the entire group structure shown in Table 2 was used, and the source volume was assumed to be a sphere with a 0.5-cm radius. The diffusion calculation agrees fairly well with the transfusion calculation out to about 90 cm but falls below it at greater distances. The transfusion results are compared in Fig. 9 with the arbitrarily normalized 1-keV flux obtained from the RENUPAK calculation. The shapes of the 1-keV and thermal (transfusion) distributions are very similar beyond about 60 cm.

Comparison of Calculated and Experimental Results

The experimental data used for checking this application of the transfusion method were obtained[16] at the ORNL Lid Tank Shielding Facility. This facility, which has been described elsewhere,[17] has a 28-in.-diam (0.06-in. thick) uranium disk source with a strength of 1.62×10^{11} fissions/sec (ref. 18). Thermal-neutron fluxes measured with various detectors as a function of distance z in water from the source are given in Table 3.

In order to compare these experimental data with the transfusion calculation, they had to be normalized to a source strength of 1 neutron/sec

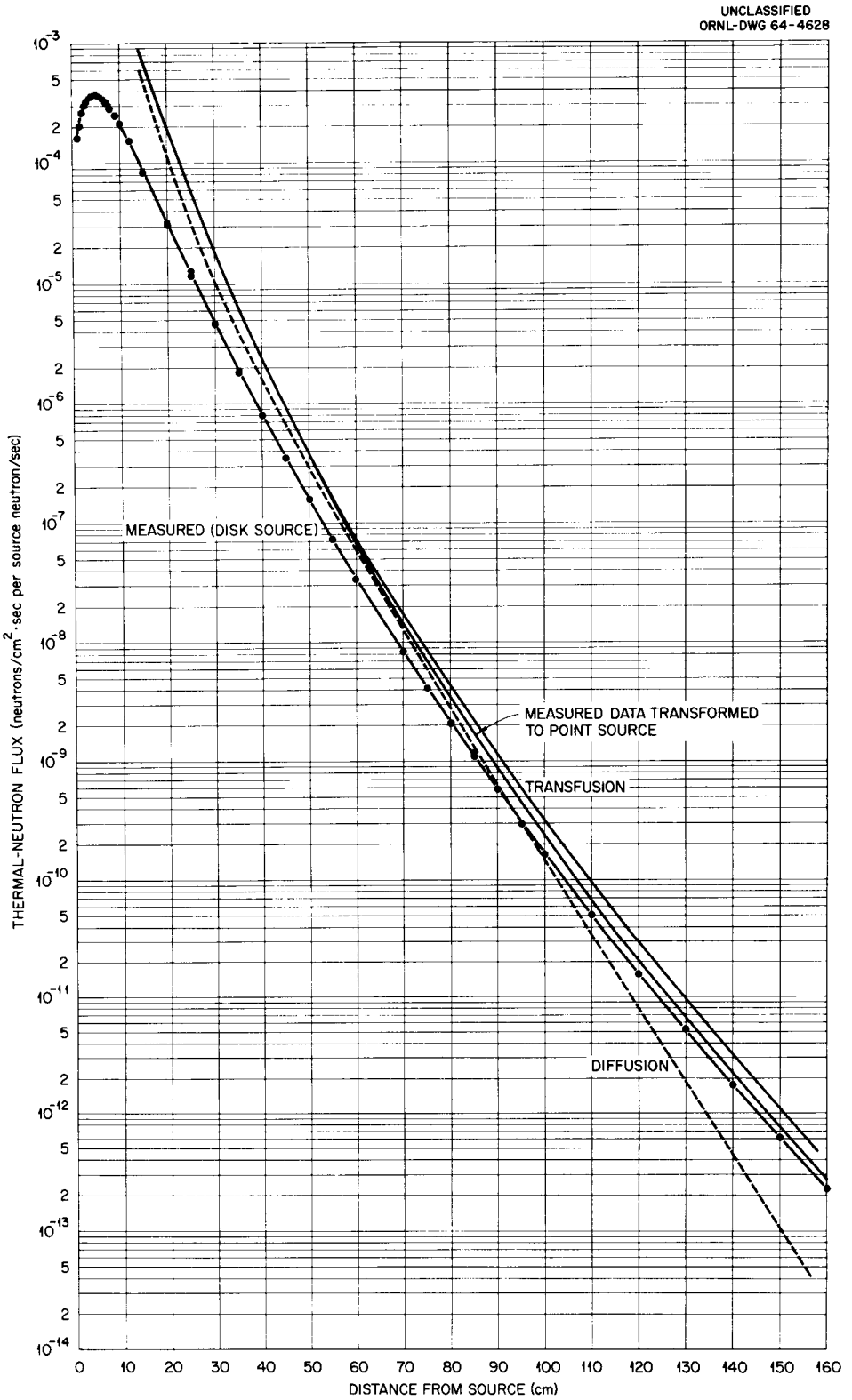

Fig. 7. Comparison of Measured and Calculated Thermal-Neutron Fluxes as a Function of Distance from the Source.

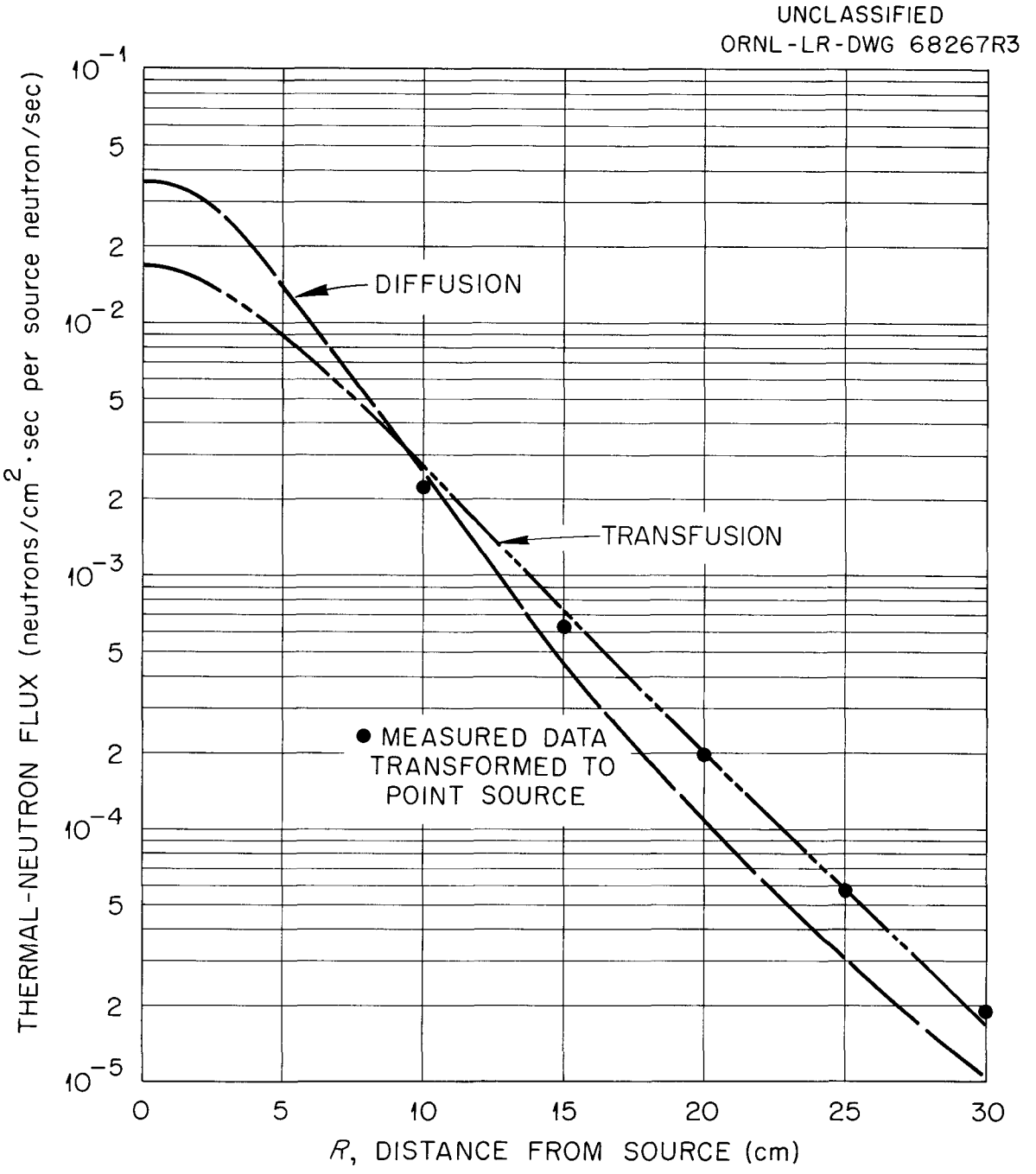

Fig. 8. Comparison of Measured and Calculated Thermal-Neutron Fluxes in the Region Close to the Source.

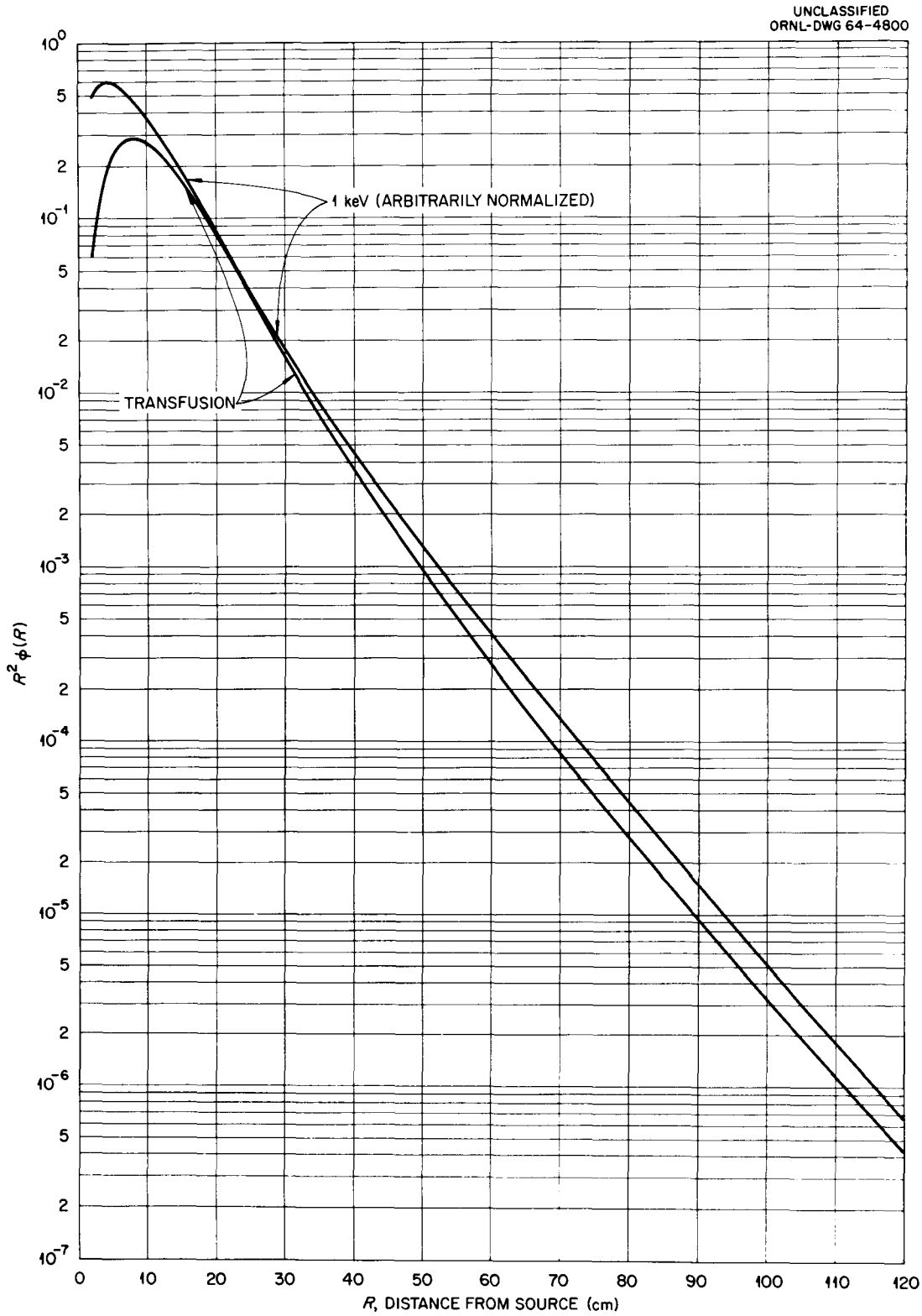

Fig. 9. Comparison of Transfusion Thermal-Neutron Flux with 1-keV Flux Obtained from RENUPAK Calculation.

Table 3. Thermal-Neutron Flux Data Obtained at Lid Tank Shielding Facility with Detectors Placed at Various Distances from the Source

Gold Foils[a]		1/2-in. Fission Chamber A		3-in. Fission Chamber A		12-1/2-in. BF_3 Counter		3-in. Fission Chamber B	
z (cm)	ϕ (neutrons cm^{-2} sec^{-1})	z (cm)	ϕ (neutrons cm^{-2} sec^{-1})	z (cm)	ϕ (neutrons cm^{-2} sec^{-1})	z (cm)	ϕ (neutrons cm^{-2} sec^{-1})	z (cm)	ϕ (neutrons cm^{-2} sec^{-1})
5	1.30×10^8	1	5.75×10^7	30	1.70×10^6	75	1.16×10^3	30	1.72×10^6
10	7.72×10^7	2	9.46×10^7	25	4.56×10^6	70	2.99×10^3	31	1.42×10^6
15	2.99×10^7	3	1.17×10^8	35	6.83×10^5	85	3.90×10^2	35	6.61×10^5
20	1.12×10^7	4	1.30×10^8	40	2.83×10^5	90	2.06×10^2	40	2.84×10^5
25	4.21×10^6	5	1.32×10^8	45	1.23×10^5	80	7.48×10^2	45	1.34×10^5
30	1.68×10^6	6	1.26×10^8	50	5.53×10^4	100	5.91×10^1	50	5.49×10^4
35	6.78×10^5	7	1.16×10^8	55	2.54×10^4	110	1.79×10^1	55	2.54×10^4
40	2.85×10^5	8	1.02×10^8	60	1.21×10^4	120	5.56×10^0	60	1.22×10^4
45	1.25×10^5	9	9.01×10^7	85	3.96×10^2	130	1.86×10^0		
50	5.64×10^4	10	7.60×10^7	70	2.92×10^3	140	6.23×10^{-1}		
55	2.64×10^4	1.5	7.30×10^7	75	1.46×10^3	150	2.20×10^{-1}		
		2	9.47×10^7	80	7.65×10^2	160	8.20×10^{-2}		
		2.5	1.08×10^8	95	1.04×10^2	165	5.17×10^{-2}		
		3.5	1.26×10^8	90	2.06×10^2				
		4.5	1.33×10^8						
		5.5	1.30×10^8						
		6.5	1.22×10^8						
		7.5	1.09×10^8						
		12	5.41×10^7						
		15	3.07×10^7						
		20	1.14×10^7						
		25	4.19×10^6						
		30	1.65×10^6						
		35	6.57×10^5						
		40	2.89×10^5						
		45	1.29×10^5						
		50	5.55×10^4						
		55	2.59×10^4						

a. Measurements made in January 1961.

and transformed to represent a point source rather than a disk source. The normalization was performed by multiplying the experimental flux by

$$\frac{1.13}{(2.47 \text{ neutrons/fission})(1.62 \times 10^{11} \text{ fissions/sec})} = \frac{2.82 \times 10^{-12}}{\text{neutrons/sec}} ,$$

where the factor 1.13 was required to convert the measured flux units, which are in terms of the density multiplied by 2200 m/sec, to "average velocity" flux units. The transformation to a point source was effected by using the relationship derived by Blizard:[19]

$$\phi_{pt}(z) = -\frac{a^2}{2} \sum_{\nu=0}^{\infty} \frac{1}{z_\nu} \frac{d}{dz_\nu} \phi_{p\ell}(z_\nu) ,$$

where

$\phi_{pt}(z)$ = flux at a distance z due to a point source having a strength of 1 neutron/sec,

$\phi_{p\ell}(z)$ = flux at a distance z on the center line due to a disk source of radius a having a strength of 1 neutron/sec (called a smoothed measured kernel below),

$z_\nu = \sqrt{z^2 + \nu a^2}$,

a = radius of source plate.

The transformation was carried out on the computer at 1-cm intervals in the following way. The data were plotted, and interpolated values at 1-cm intervals were read into the computer. A program was written in which was incorporated the assumption that the curve between points was an exponential:

$$\phi(z) = \phi(z') \, e^{-b(z-z')} .$$

The derivative was then

$$\frac{d\ \phi(z)}{dz} = -\ b\ \phi(z)\ .$$

The procedure of reading points from a curve led to somewhat erratic values of b, but averaging over a 6-cm range yielded satisfactory values of the derivative. The series was cut off when a term contributed less than 1/2 of 1% to the sum. Since there were no experimental data beyond 165 cm, the remainder of the series was estimated from an expression derived by Blizard[19] and based on exponential extrapolation.

As a check on the accuracy of the point kernel obtained from the transformation, $\phi_{pt}(z)$ was integrated over the source to obtain a kernel which could be compared with the smoothed measured kernel. In general, the two kernels agreed to within 2% at distances less than 150 cm, but they differed by as much as 10% at greater distances. An iterative technique was then used to improve the agreement. The ratio of the integrated kernel to the point kernel was applied to the smoothed measured kernel to obtain a new point kernel, which in turn was integrated over the source. The ratio of the new integrated kernel to the new point kernel was then applied to the smoothed measured kernel, and so forth. When the final point kernel obtained in the iteration procedure was integrated over the source, the results agreed with the smoothed measured kernel to within 1% at distances less than 150 cm and to within 3% at greater distances.

Values of the smoothed measured kernel and the point kernel are given in Table 4, along with the ratio of the two kernels and the contribution of the extrapolated data. The fluctuations in the ratio are due to the numerical procedures.

Table 4. Experimental Thermal-Neutron Flux Data Smoothed and Transformed to a Point Source

z (cm)	$\phi_{pl}(z)$ (neutrons cm^{-2} source $neutron^{-1}$)	$\phi_{pt}(z)$ (neutrons cm^{-2} source $neutron^{-1}$)	ϕ_{pl}/ϕ_{pt}	Percent Remainder (Extrapolated Data)
10	2.14×10^{-4}	2.20×10^{-3}	0.097	0
15	8.50×10^{-5}	6.38×10^{-4}	0.13	0
20	3.19×10^{-5}	1.97×10^{-4}	0.16	0
25	1.20×10^{-5}	5.76×10^{-5}	0.21	0
30	4.70×10^{-6}	1.91×10^{-5}	0.25	0
35	1.91×10^{-6}	6.37×10^{-6}	0.30	0
40	8.04×10^{-7}	2.22×10^{-6}	0.36	0
45	3.60×10^{-7}	9.00×10^{-7}	0.40	0
50	1.59×10^{-7}	3.77×10^{-7}	0.42	0
55	7.40×10^{-8}	1.63×10^{-7}	0.45	0
60	3.42×10^{-8}	6.88×10^{-8}	0.50	0
65	1.65×10^{-8}	2.95×10^{-8}	0.56	0
70	8.23×10^{-9}	1.39×10^{-8}	0.59	0
75	4.12×10^{-9}	6.80×10^{-8}	0.61	0
80	2.16×10^{-9}	3.57×10^{-9}	0.60	0
85	1.11×10^{-9}	1.76×10^{-9}	0.63	0
90	5.81×10^{-10}	8.80×10^{-9}	0.66	0
95	2.93×10^{-10}	4.22×10^{-9}	0.69	0
100	1.67×10^{-10}	2.47×10^{-10}	0.68	0
105	9.00×10^{-11}	1.26×10^{-10}	0.71	0
110	5.05×10^{-11}	7.05×10^{-10}	0.72	0
115	2.78×10^{-11}	3.80×10^{-11}	0.73	0
120	1.57×10^{-11}	2.11×10^{-11}	0.74	0
125	8.90×10^{-12}	1.16×10^{-11}	0.77	0
130	5.20×10^{-12}	6.78×10^{-12}	0.77	0
135	3.00×10^{-12}	3.80×10^{-12}	0.79	10
140	1.76×10^{-12}	2.20×10^{-12}	0.80	17
145	1.04×10^{-12}	1.26×10^{-12}	0.83	27
150	6.20×10^{-13}	7.68×10^{-13}	0.81	45
155	3.71×10^{-13}	4.68×10^{-13}	0.79	67
160	2.31×10^{-13}	2.71×10^{-13}	0.85	85
165	1.45×10^{-13}	1.76×10^{-13}	0.82	100

In Fig. 7 the two experimental kernels are compared with results from the transfusion and diffusion calculations, and in Fig. 8 the point kernel alone is compared with the calculations. The point kernel results are 10 to 15% lower than those reported previously,[20] but this is probably because the geometric transformation procedure used in the earlier work was slightly in error. In general, the transfusion results are in agreement with the experimental results for distances out to 70 cm, but they rise about 40% above the experimental results beyond 70 cm. Factors in the calculation which might have contributed to this discrepancy are the following:

Cross Sections.--The cross sections most in doubt are the high-energy cross sections for oxygen, and they may have significantly affected the thermal-neutron flux at large distances since it is largely due to neutrons that leave the source at high energies. One source of error in the cross sections could be the treatment of inelastic scattering as absorption; however, this is consistent with removal theory, which assumes that the neutron is likely to disappear because the reaction is isotropic.

Fission Spectrum.--The statistical uncertainty in the fission-spectrum measurements on which Cranberg based his parameters is 30% or higher at 12 MeV and above, and this undoubtedly contributes to the discrepancy between the transfusion calculation and the LTSF measurements. If the original Watt formula[21] for the spectrum had been used, however, there probably would have been an even greater discrepancy since it predicts more neutrons at high energies than are obtained with the formula as modified by Cranberg.

Fitting Flux to Moments.--An uncertainty inherent in the moments method is the final step of constructing the flux from the moments. In the RENUPAK code an arbitrary form for the flux is assumed as described above.

Factors in the experiment which might have contributed to the discrepancy between the experimental and calculated data are:

Source Plate Power.--The uncertainty in the value of the source plate power (1.62×10^{11} fissions/sec) is estimated to be $\pm 5\%$,[18] but the difficulties of the measurement lead the author to believe this figure may be too small.

Transformation.--For the transformation to be valid the source plate must have a uniform power across its surface and the medium must be infinite and homogeneous. These conditions are probably sufficiently well met.

Detector Considerations.--The flux in the LTSF was measured with gold foils and correction factors were applied to their counting rates to compensate for effects such as flux depression. These factors introduce a small uncertainty. Beyond 55 cm the data were generated using proportional counters normalized to the foil data. There are some uncertainties regarding center of detection in addition to the statistical uncertainty.

A further check on the results is obtained by examining the spacial moments. The second moment of energy group 30 was computed yielding an age to indium of 25.6 cm^2 which is consistent with the value of 25.5 cm^2 reported by Goldstein[22] who used the moments method. The second moment of the thermal-neutron flux was computed yielding a migration area of 33.6 cm^2. This is consistent with an age from indium to thermal of 1.0 cm^2 and a diffusion length of 2.7 cm. A different transport theory calculation, such as Monte Carlo, would be more suitable for short distances than the technique used here. The experimental data of the LTSF, of course, does not yield good information close to the source because of leakage.

Conclusions

When all the sources of error are considered, it is not surprising to observe a 40% difference between the calculation and the experiment at large distances.

In future work inelastic scattering should be considered, the effect of the assumed spectrum should be investigated, and Monte Carlo should perhaps be used to do the transport phase of the calculation near the source.

Acknowledgments

The assistance of S. K. Penny, W. E. Kinney, V. E. Anderson, and H. E. Murchison in successfully completing the calculation is gratefully acknowledged.

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Appendix A

Hydrogen Total Cross Sections

E (MeV)	σ (barns)	E (keV)	σ (barns)	E (keV)	σ (barns)
18.02	0.536	734	5.03	29.9	16.96
16.30	0.592	665	5.31	27.1	17.20
14.75	0.657	601	5.59	24.5	17.45
13.35	0.723	544	5.87	22.2	17.59
12.08	0.792	492	6.18	20.1	17.91
10.93	0.865	445	6.53	18.2	18.12
9.888	0.942	403	6.87	16.4	18.30
8.947	1.026	365	7.23	14.9	18.47
8.096	1.116	330	7.58	13.5	18.63
7.325	1.213	299	7.95	12.2	18.77
6.628	1.315	270	8.31	11.0	18.89
5.997	1.420	244	8.71	9.97	19.00
5.427	1.530	221	9.11	9.02	19.09
4.910	1.630	200	9.51	8.16	19.17
4.443	1.760	181	9.91	7.38	19.25
4.020	1.880	164	10.31	6.68	19.32
3.638	2.010	148	10.77	6.04	19.40
3.291	2.140	134	11.27	5.47	19.46
2.978	2.28	121	11.71	4.94	19.52
2.694	2.43	109.8	12.13	4.48	19.58
2.438	2.58	99.4	12.58	4.05	19.65
2.206	2.73	89.9	13.05	3.67	19.71
1.996	2.88	81.4	13.5	3.32	19.77
1.806	3.06	73.6	13.9	3.00	19.83
1.634	3.24	66.6	14.3	2.72	19.87
1.479	3.42	60.3	14.7	2.46	19.91
1.338	3.61	54.5	15.1	2.22	19.94
1.211	3.81	49.4	15.4	2.02	19.97
1.096	4.03	44.7	15.7	1.82	19.99
0.991	4.26	40.4	16.08	1.65	19.99
0.897	4.52	36.6	16.37	1.49	19.99
0.812	4.77	33.1	16.68	0.03	19.99

Appendix B

Differential Number Spectra for Point Fission Source in Water

$$N(R,E) = 4\pi R^2 \phi(R,E)$$

DIFFERENTIAL NUMBER SPECTRA, FISSION POINT SOURCE

EXPONENTIAL FLUX IN WATER

N(R,E),N/SEC-MEV

R

E,MEV	1.	2.	3.	4.	5.
1.802E 01	5.979E-07	8.849E-07	9.894E-07	9.916E-07	9.407E-07
1.630E 01	2.608E-06	3.850E-06	4.293E-06	4.294E-06	4.066E-06
1.475E 01	9.768E-06	1.434E-05	1.591E-05	1.584E-05	1.493E-05
1.335E 01	3.189E-05	4.653E-05	5.133E-05	5.080E-05	4.765E-05
1.208E 01	9.234E-05	1.340E-04	1.471E-04	1.449E-04	1.354E-04
1.093E 01	2.378E-04	3.437E-04	3.760E-04	3.694E-04	3.444E-04
9.888E 00	5.500E-04	7.932E-04	8.659E-04	8.491E-04	7.902E-04
8.947E 00	1.155E-03	1.662E-03	1.810E-03	1.770E-03	1.643E-03
8.096E 00	2.259E-03	3.245E-03	3.529E-03	3.449E-03	3.199E-03
7.325E 00	4.070E-03	5.750E-03	6.147E-03	5.902E-03	5.375E-03
6.628E 00	6.855E-03	9.799E-03	1.060E-02	1.030E-02	9.487E-03
5.997E 00	1.098E-02	1.555E-02	1.666E-02	1.603E-02	1.463E-02
5.427E 00	1.668E-02	2.366E-02	2.540E-02	2.449E-02	2.241E-02
4.910E 00	2.415E-02	3.413E-02	3.650E-02	3.506E-02	3.196E-02
4.443E 00	3.338E-02	4.661E-02	4.924E-02	4.671E-02	4.204E-02
4.020E 00	4.480E-02	6.236E-02	6.568E-02	6.214E-02	5.578E-02
3.638E 00	5.618E-02	7.612E-02	7.787E-02	7.140E-02	6.198E-02
3.291E 00	7.146E-02	9.807E-02	1.016E-01	9.429E-02	8.281E-02
2.978E 00	9.520E-02	1.333E-01	1.411E-01	1.341E-01	1.207E-01
2.695E 00	1.135E-01	1.574E-01	1.652E-01	1.557E-01	1.390E-01
2.438E 00	1.270E-01	1.760E-01	1.846E-01	1.741E-01	1.558E-01
2.206E 00	1.403E-01	1.928E-01	2.005E-01	1.874E-01	1.663E-01
1.996E 00	1.546E-01	2.113E-01	2.186E-01	2.032E-01	1.793E-01
1.806E 00	1.690E-01	2.292E-01	2.352E-01	2.167E-01	1.894E-01
1.635E 00	1.839E-01	2.463E-01	2.495E-01	2.268E-01	1.955E-01
1.479E 00	2.114E-01	2.929E-01	3.065E-01	2.876E-01	2.554E-01
1.338E 00	2.264E-01	3.117E-01	3.241E-01	3.020E-01	2.664E-01
1.211E 00	2.367E-01	3.305E-01	3.485E-01	3.292E-01	2.940E-01
1.096E 00	2.306E-01	3.177E-01	3.305E-01	3.081E-01	2.718E-01
9.914E-01	2.294E-01	3.050E-01	3.064E-01	2.762E-01	2.361E-01
8.970E-01	2.799E-01	3.867E-01	4.032E-01	3.766E-01	3.326E-01
8.117E-01	3.334E-01	4.656E-01	4.906E-01	4.626E-01	4.122E-01
7.344E-01	3.144E-01	4.443E-01	4.735E-01	4.514E-01	4.063E-01

DIFFERENTIAL NUMBER SPECTRA, FISSION POINT SOURCE

EXPONENTIAL FLUX IN WATER

N(R,E),N/SEC-MEV

R

E,MEV	1.	2.	3.	4.	5.
6.645E-01	2.996E-01	4.275E-01	4.598E-01	4.422E-01	4.013E-01
6.013E-01	3.009E-01	4.316E-01	4.667E-01	4.510E-01	4.112E-01
5.441E-01	3.065E-01	4.423E-01	4.808E-01	4.671E-01	4.279E-01
4.923E-01	3.085E-01	4.452E-01	4.840E-01	4.702E-01	4.307E-01
4.455E-01	2.600E-01	3.653E-01	3.871E-01	3.670E-01	3.286E-01
4.031E-01	3.424E-01	4.924E-01	5.337E-01	5.167E-01	4.718E-01
3.647E-01	3.802E-01	5.497E-01	5.985E-01	5.822E-01	5.338E-01
3.300E-01	3.991E-01	5.801E-01	6.350E-01	6.207E-01	5.717E-01
2.986E-01	4.065E-01	5.948E-01	6.552E-01	6.443E-01	5.968E-01
2.702E-01	4.088E-01	6.014E-01	6.660E-01	6.582E-01	6.125E-01
2.445E-01	4.136E-01	6.111E-01	6.795E-01	6.742E-01	6.296E-01
2.212E-01	4.261E-01	6.319E-01	7.051E-01	7.018E-01	6.574E-01
2.002E-01	4.427E-01	6.587E-01	7.373E-01	7.360E-01	6.914E-01
1.811E-01	4.620E-01	6.895E-01	7.740E-01	7.747E-01	7.296E-01
1.639E-01	4.835E-01	7.237E-01	8.146E-01	8.176E-01	7.719E-01
1.483E-01	5.055E-01	7.584E-01	8.557E-01	8.607E-01	8.143E-01
1.342E-01	5.284E-01	7.945E-01	8.982E-01	9.052E-01	8.580E-01
1.214E-01	5.559E-01	8.378E-01	9.493E-01	9.587E-01	9.103E-01
1.098E-01	5.860E-01	8.849E-01	1.005E 00	1.017E 00	9.670E-01
9.939E-02	6.183E-01	9.356E-01	1.064E 00	1.079E 00	1.028E 00
8.994E-02	6.529E-01	9.898E-01	1.128E 00	1.145E 00	1.093E 00
8.138E-02	6.914E-01	1.050E 00	1.198E 00	1.218E 00	1.164E 00
7.363E-02	7.354E-01	1.119E 00	1.279E 00	1.302E 00	1.247E 00
6.663E-02	7.833E-01	1.194E 00	1.367E 00	1.394E 00	1.336E 00
6.029E-02	8.356E-01	1.275E 00	1.462E 00	1.493E 00	1.433E 00
5.455E-02	8.924E-01	1.364E 00	1.566E 00	1.602E 00	1.539E 00
4.936E-02	9.590E-01	1.468E 00	1.688E 00	1.729E 00	1.664E 00
4.466E-02	1.032E 00	1.581E 00	1.821E 00	1.868E 00	1.799E 00
4.041E-02	1.106E 00	1.698E 00	1.958E 00	2.010E 00	1.939E 00
3.657E-02	1.193E 00	1.833E 00	2.116E 00	2.176E 00	2.101E 00
3.309E-02	1.285E 00	1.978E 00	2.286E 00	2.353E 00	2.275E 00
2.994E-02	1.388E 00	2.138E 00	2.475E 00	2.550E 00	2.468E 00
2.709E-02	1.503E 00	2.318E 00	2.686E 00	2.771E 00	2.684E 00

DIFFERENTIAL NUMBER SPECTRA, FISSION POINT SOURCE

EXPONENTIAL FLUX IN WATER

N(R,E),N/SEC-MEV

R

E,MEV	1.	2.	3.	4.	5.
2.451E-02	1.627E 00	2.513E 00	2.915E 00	3.010E 00	2.919E 00
2.218E-02	1.772E 00	2.740E 00	3.182E 00	3.290E 00	3.194E 00
2.007E-02	1.914E 00	2.963E 00	3.444E 00	3.564E 00	3.463E 00
1.816E-02	2.079E 00	3.221E 00	3.749E 00	3.884E 00	3.777E 00
1.643E-02	2.261E 00	3.509E 00	4.087E 00	4.238E 00	4.126E 00
1.487E-02	2.463E 00	3.825E 00	4.460E 00	4.629E 00	4.510E 00
1.345E-02	2.684E 00	4.172E 00	4.870E 00	5.059E 00	4.934E 00
1.217E-02	2.928E 00	4.556E 00	5.324E 00	5.536E 00	5.404E 00
1.101E-02	3.198E 00	4.982E 00	5.826E 00	6.064E 00	5.924E 00
9.965E-03	3.496E 00	5.451E 00	6.381E 00	6.647E 00	6.499E 00
9.017E-03	3.825E 00	5.970E 00	6.995E 00	7.293E 00	7.137E 00
8.159E-03	4.188E 00	6.542E 00	7.673E 00	8.007E 00	7.843E 00
7.382E-03	4.586E 00	7.171E 00	8.418E 00	8.792E 00	8.618E 00
6.680E-03	5.025E 00	7.865E 00	9.240E 00	9.658E 00	9.475E 00
6.044E-03	5.504E 00	8.622E 00	1.014E 01	1.061E 01	1.041E 01
5.469E-03	6.035E 00	9.462E 00	1.114E 01	1.166E 01	1.146E 01
4.949E-03	6.618E 00	1.038E 01	1.223E 01	1.282E 01	1.260E 01
4.478E-03	7.258E 00	1.140E 01	1.344E 01	1.409E 01	1.386E 01
4.052E-03	7.957E 00	1.251E 01	1.475E 01	1.548E 01	1.525E 01
3.666E-03	8.727E 00	1.373E 01	1.621E 01	1.702E 01	1.677E 01
3.317E-03	9.574E 00	1.507E 01	1.781E 01	1.871E 01	1.845E 01
3.001E-03	1.050E 01	1.655E 01	1.956E 01	2.058E 01	2.030E 01
2.716E-03	1.153E 01	1.818E 01	2.150E 01	2.263E 01	2.235E 01
2.457E-03	1.266E 01	1.997E 01	2.365E 01	2.491E 01	2.461E 01
2.224E-03	1.391E 01	2.196E 01	2.602E 01	2.743E 01	2.712E 01
2.012E-03	1.529E 01	2.415E 01	2.863E 01	3.020E 01	2.988E 01
1.820E-03	1.680E 01	2.657E 01	3.152E 01	3.327E 01	3.294E 01
1.647E-03	1.849E 01	2.926E 01	3.474E 01	3.669E 01	3.634E 01
1.490E-03	2.035E 01	3.222E 01	3.828E 01	4.045E 01	4.010E 01
1.349E-03	2.240E 01	3.549E 01	4.219E 01	4.461E 01	4.425E 01
1.220E-03	2.465E 01	3.909E 01	4.650E 01	4.920E 01	4.883E 01
1.104E-03	2.714E 01	4.305E 01	5.125E 01	5.426E 01	5.389E 01
9.991E-04	2.987E 01	4.742E 01	5.649E 01	5.984E 01	5.946E 01
9.040E-04	3.289E 01	5.224E 01	6.226E 01	6.600E 01	6.562E 01
8.180E-04	3.620E 01	5.754E 01	6.863E 01	7.279E 01	7.241E 01

DIFFERENTIAL NUMBER SPECTRA, FISSION POINT SOURCE

EXPONENTIAL FLUX IN WATER

N(R,E),N/SEC-MEV

R

E,MEV	6.	7.	8.	9.	10.
1.802E 01	8.659E-07	7.838E-07	7.037E-07	6.300E-07	5.642E-07
1.630E 01	3.737E-06	3.379E-06	3.032E-06	2.713E-06	2.430E-06
1.475E 01	1.367E-05	1.232E-05	1.102E-05	9.831E-06	8.785E-06
1.335E 01	4.341E-05	3.895E-05	3.470E-05	3.087E-05	2.751E-05
1.208E 01	1.230E-04	1.102E-04	9.803E-05	8.715E-05	7.767E-05
1.093E 01	3.123E-04	2.793E-04	2.484E-04	2.208E-04	1.968E-04
9.888E 00	7.154E-04	6.389E-04	5.676E-04	5.041E-04	4.490E-04
8.947E 00	1.483E-03	1.321E-03	1.170E-03	1.035E-03	9.195E-04
8.096E 00	2.889E-03	2.574E-03	2.282E-03	2.024E-03	1.800E-03
7.325E 00	4.762E-03	4.162E-03	3.619E-03	3.147E-03	2.747E-03
6.628E 00	8.503E-03	7.516E-03	6.606E-03	5.804E-03	5.114E-03
5.997E 00	1.299E-02	1.138E-02	9.908E-03	8.628E-03	7.538E-03
5.427E 00	1.995E-02	1.752E-02	1.530E-02	1.336E-02	1.171E-02
4.910E 00	2.834E-02	2.479E-02	2.157E-02	1.877E-02	1.638E-02
4.443E 00	3.680E-02	3.178E-02	2.730E-02	2.346E-02	2.023E-02
4.020E 00	4.871E-02	4.197E-02	3.598E-02	3.085E-02	2.655E-02
3.638E 00	5.225E-02	4.339E-02	3.583E-02	2.959E-02	2.455E-02
3.291E 00	7.056E-02	5.915E-02	4.923E-02	4.092E-02	3.410E-02
2.978E 00	1.055E-01	9.083E-02	7.762E-02	6.621E-02	5.655E-02
2.695E 00	1.207E-01	1.032E-01	8.771E-02	7.446E-02	6.334E-02
2.438E 00	1.356E-01	1.165E-01	9.948E-02	8.491E-02	7.267E-02
2.206E 00	1.437E-01	1.225E-01	1.039E-01	8.819E-02	7.508E-02
1.996E 00	1.540E-01	1.305E-01	1.101E-01	9.290E-02	7.868E-02
1.806E 00	1.611E-01	1.351E-01	1.128E-01	9.425E-02	7.906E-02
1.635E 00	1.640E-01	1.357E-01	1.117E-01	9.216E-02	7.641E-02
1.479E 00	2.200E-01	1.865E-01	1.569E-01	1.316E-01	1.106E-01
1.338E 00	2.281E-01	1.921E-01	1.605E-01	1.339E-01	1.118E-01
1.211E 00	2.546E-01	2.167E-01	1.827E-01	1.536E-01	1.290E-01
1.096E 00	2.327E-01	1.960E-01	1.637E-01	1.365E-01	1.139E-01
9.914E-01	1.963E-01	1.610E-01	1.315E-01	1.077E-01	8.861E-02
8.970E-01	2.848E-01	2.397E-01	2.000E-01	1.664E-01	1.386E-01
8.117E-01	3.558E-01	3.015E-01	2.530E-01	2.114E-01	1.765E-01
7.344E-01	3.539E-01	3.024E-01	2.555E-01	2.147E-01	1.801E-01

DIFFERENTIAL NUMBER SPECTRA, FISSION POINT SOURCE

EXPONENTIAL FLUX IN WATER

N(R,E),N/SEC-MEV

R

E,MEV	6.	7.	8.	9.	10.
6.645E-01	3.522E-01	3.029E-01	2.575E-01	2.174E-01	1.830E-01
6.013E-01	3.623E-01	3.128E-01	2.667E-01	2.258E-01	1.905E-01
5.441E-01	3.788E-01	3.283E-01	2.808E-01	2.384E-01	2.016E-01
4.923E-01	3.812E-01	3.302E-01	2.824E-01	2.397E-01	2.025E-01
4.455E-01	2.849E-01	2.423E-01	2.040E-01	1.708E-01	1.429E-01
4.031E-01	4.163E-01	3.597E-01	3.067E-01	2.597E-01	2.190E-01
3.647E-01	4.728E-01	4.099E-01	3.506E-01	2.975E-01	2.514E-01
3.300E-01	5.085E-01	4.424E-01	3.796E-01	3.230E-01	2.735E-01
2.986E-01	5.335E-01	4.663E-01	4.017E-01	3.429E-01	2.911E-01
2.702E-01	5.499E-01	4.825E-01	4.171E-01	3.571E-01	3.039E-01
2.445E-01	5.671E-01	4.991E-01	4.325E-01	3.711E-01	3.164E-01
2.212E-01	5.938E-01	5.239E-01	4.550E-01	3.912E-01	3.340E-01
2.002E-01	6.260E-01	5.536E-01	4.818E-01	4.149E-01	3.548E-01
1.811E-01	6.623E-01	5.869E-01	5.118E-01	4.414E-01	3.780E-01
1.639E-01	7.021E-01	6.235E-01	5.447E-01	4.706E-01	4.035E-01
1.483E-01	7.421E-01	6.601E-01	5.776E-01	4.997E-01	4.289E-01
1.342E-01	7.833E-01	6.978E-01	6.113E-01	5.295E-01	4.550E-01
1.214E-01	8.325E-01	7.428E-01	6.518E-01	5.652E-01	4.862E-01
1.098E-01	8.859E-01	7.916E-01	6.955E-01	6.039E-01	5.200E-01
9.939E-02	9.430E-01	8.439E-01	7.424E-01	6.453E-01	5.562E-01
8.994E-02	1.004E 00	8.995E-01	7.922E-01	6.893E-01	5.947E-01
8.138E-02	1.071E 00	9.611E-01	8.475E-01	7.381E-01	6.373E-01
7.363E-02	1.148E 00	1.032E 00	9.106E-01	7.939E-01	6.861E-01
6.663E-02	1.232E 00	1.108E 00	9.794E-01	8.547E-01	7.393E-01
6.029E-02	1.324E 00	1.192E 00	1.054E 00	9.210E-01	7.972E-01
5.455E-02	1.423E 00	1.283E 00	1.136E 00	9.930E-01	8.602E-01
4.936E-02	1.540E 00	1.390E 00	1.232E 00	1.078E 00	9.346E-01
4.466E-02	1.668E 00	1.507E 00	1.337E 00	1.171E 00	1.016E 00
4.041E-02	1.799E 00	1.627E 00	1.445E 00	1.266E 00	1.100E 00
3.657E-02	1.951E 00	1.766E 00	1.570E 00	1.377E 00	1.197E 00
3.309E-02	2.115E 00	1.916E 00	1.705E 00	1.497E 00	1.301E 00
2.994E-02	2.297E 00	2.083E 00	1.855E 00	1.630E 00	1.418E 00
2.709E-02	2.501E 00	2.270E 00	2.023E 00	1.779E 00	1.549E 00

DIFFERENTIAL NUMBER SPECTRA, FISSION POINT SOURCE

EXPONENTIAL FLUX IN WATER

N(R,E),N/SEC-MEV

R

E,MEV	6.	7.	8.	9.	10.
2.451E-02	2.723E 00	2.474E 00	2.206E 00	1.941E 00	1.691E 00
2.218E-02	2.982E 00	2.711E 00	2.420E 00	2.131E 00	1.858E 00
2.007E-02	3.236E 00	2.945E 00	2.631E 00	2.318E 00	2.023E 00
1.816E-02	3.532E 00	3.217E 00	2.876E 00	2.536E 00	2.214E 00
1.643E-02	3.862E 00	3.521E 00	3.150E 00	2.779E 00	2.428E 00
1.487E-02	4.226E 00	3.855E 00	3.452E 00	3.048E 00	2.664E 00
1.345E-02	4.626E 00	4.224E 00	3.785E 00	3.344E 00	2.925E 00
1.217E-02	5.071E 00	4.634E 00	4.155E 00	3.674E 00	3.215E 00
1.101E-02	5.564E 00	5.088E 00	4.565E 00	4.040E 00	3.537E 00
9.965E-03	6.109E 00	5.591E 00	5.020E 00	4.445E 00	3.894E 00
9.017E-03	6.714E 00	6.149E 00	5.525E 00	4.895E 00	4.291E 00
8.159E-03	7.383E 00	6.767E 00	6.085E 00	5.394E 00	4.731E 00
7.382E-03	8.120E 00	7.447E 00	6.701E 00	5.944E 00	5.217E 00
6.680E-03	8.934E 00	8.200E 00	7.383E 00	6.553E 00	5.755E 00
6.044E-03	9.826E 00	9.025E 00	8.131E 00	7.222E 00	6.345E 00
5.469E-03	1.082E 01	9.942E 00	8.963E 00	7.966E 00	7.002E 00
4.949E-03	1.191E 01	1.095E 01	9.880E 00	8.786E 00	7.728E 00
4.478E-03	1.311E 01	1.207E 01	1.089E 01	9.691E 00	8.528E 00
4.052E-03	1.443E 01	1.329E 01	1.200E 01	1.068E 01	9.407E 00
3.666E-03	1.588E 01	1.464E 01	1.323E 01	1.178E 01	1.038E 01
3.317E-03	1.749E 01	1.612E 01	1.458E 01	1.300E 01	1.145E 01
3.001E-03	1.925E 01	1.776E 01	1.607E 01	1.433E 01	1.264E 01
2.716E-03	2.120E 01	1.958E 01	1.772E 01	1.581E 01	1.395E 01
2.457E-03	2.337E 01	2.159E 01	1.955E 01	1.746E 01	1.541E 01
2.224E-03	2.576E 01	2.381E 01	2.158E 01	1.928E 01	1.703E 01
2.012E-03	2.840E 01	2.627E 01	2.382E 01	2.129E 01	1.881E 01
1.820E-03	3.133E 01	2.899E 01	2.631E 01	2.352E 01	2.080E 01
1.647E-03	3.459E 01	3.203E 01	2.908E 01	2.602E 01	2.301E 01
1.490E-03	3.819E 01	3.538E 01	3.214E 01	2.877E 01	2.546E 01
1.349E-03	4.217E 01	3.909E 01	3.553E 01	3.182E 01	2.817E 01
1.220E-03	4.656E 01	4.319E 01	3.927E 01	3.519E 01	3.117E 01
1.104E-03	5.141E 01	4.771E 01	4.341E 01	3.891E 01	3.449E 01
9.991E-04	5.676E 01	5.271E 01	4.798E 01	4.303E 01	3.815E 01
9.040E-04	6.267E 01	5.823E 01	5.303E 01	4.759E 01	4.221E 01
8.180E-04	6.919E 01	6.432E 01	5.862E 01	5.262E 01	4.670E 01

DIFFERENTIAL NUMBER SPECTRA, FISSION POINT SOURCE

EXPONENTIAL FLUX IN WATER

N(R,E),N/SEC-MEV

R

E,MEV	15.	20.	30.	40.	50.
1.802E 01	3.450E-07	2.326E-07	1.144E-07	5.506E-08	2.583E-08
1.630E 01	1.490E-06	1.006E-06	4.929E-07	2.354E-07	1.095E-07
1.475E 01	5.342E-06	3.581E-06	1.722E-06	8.043E-07	3.659E-07
1.335E 01	1.657E-05	1.101E-05	5.168E-06	2.350E-06	1.041E-06
1.208E 01	4.699E-05	3.131E-05	1.462E-05	6.586E-06	2.890E-06
1.093E 01	1.195E-04	7.984E-05	3.719E-05	1.669E-05	7.303E-06
9.888E 00	2.723E-04	1.815E-04	8.408E-05	3.752E-05	1.633E-05
8.947E 00	5.494E-04	3.618E-04	1.639E-04	7.155E-05	3.046E-05
8.096E 00	1.087E-03	7.208E-04	3.287E-04	1.441E-04	6.163E-05
7.325E 00	1.520E-03	9.379E-04	3.753E-04	1.448E-04	5.482E-05
6.628E 00	2.934E-03	1.852E-03	7.645E-04	3.006E-04	1.144E-04
5.997E 00	4.168E-03	2.552E-03	9.961E-04	3.711E-04	1.344E-04
5.427E 00	6.538E-03	4.014E-03	1.561E-03	5.759E-04	2.057E-04
4.910E 00	8.993E-03	5.434E-03	2.049E-03	7.342E-04	2.550E-04
4.443E 00	1.053E-02	6.102E-03	2.144E-03	7.220E-04	2.381E-04
4.020E 00	1.365E-02	7.776E-03	2.627E-03	8.513E-04	2.713E-04
3.638E 00	1.094E-02	5.801E-03	1.849E-03	5.810E-04	1.823E-04
3.291E 00	1.509E-02	7.766E-03	2.340E-03	7.046E-04	2.142E-04
2.978E 00	2.738E-02	1.452E-02	4.323E-03	1.260E-03	3.685E-04
2.695E 00	3.036E-02	1.603E-02	4.707E-03	1.348E-03	3.881E-04
2.438E 00	3.574E-02	1.908E-02	5.573E-03	1.571E-03	4.439E-04
2.206E 00	3.634E-02	1.924E-02	5.546E-03	1.543E-03	4.313E-04
1.996E 00	3.733E-02	1.954E-02	5.555E-03	1.529E-03	4.244E-04
1.806E 00	3.620E-02	1.866E-02	5.248E-03	1.439E-03	3.992E-04
1.635E 00	3.369E-02	1.717E-02	4.824E-03	1.327E-03	3.697E-04
1.479E 00	4.951E-02	2.471E-02	6.724E-03	1.819E-03	5.003E-04
1.338E 00	4.913E-02	2.435E-02	6.614E-03	1.789E-03	4.927E-04
1.211E 00	5.716E-02	2.807E-02	7.497E-03	2.010E-03	5.499E-04
1.096E 00	4.981E-02	2.455E-02	6.605E-03	1.774E-03	4.860E-04
9.914E-01	3.806E-02	1.913E-02	5.280E-03	1.430E-03	3.945E-04
8.970E-01	5.953E-02	2.897E-02	7.748E-03	2.086E-03	5.733E-04
8.117E-01	7.572E-02	3.640E-02	9.619E-03	2.586E-03	7.105E-04
7.344E-01	7.769E-02	3.701E-02	9.653E-03	2.587E-03	7.102E-04

DIFFERENTIAL NUMBER SPECTRA, FISSION POINT SOURCE

EXPONENTIAL FLUX IN WATER

N(R,E),N/SEC-MEV

R

E,MEV	15.	20.	30.	40.	50.
6.645E-01	7.937E-02	3.755E-02	9.685E-03	2.590E-03	7.108E-04
6.013E-01	8.280E-02	3.900E-02	9.992E-03	2.670E-03	7.331E-04
5.441E-01	8.785E-02	4.118E-02	1.047E-02	2.798E-03	7.690E-04
4.923E-01	8.809E-02	4.123E-02	1.049E-02	2.805E-03	7.720E-04
4.455E-01	6.134E-02	2.931E-02	7.688E-03	2.064E-03	5.671E-04
4.031E-01	9.461E-02	4.426E-02	1.131E-02	3.037E-03	8.383E-04
3.647E-01	1.089E-01	5.072E-02	1.287E-02	3.456E-03	9.550E-04
3.300E-01	1.188E-01	5.509E-02	1.388E-02	3.727E-03	1.032E-03
2.986E-01	1.270E-01	5.862E-02	1.463E-02	3.927E-03	1.089E-03
2.702E-01	1.332E-01	6.121E-02	1.515E-02	4.065E-03	1.130E-03
2.445E-01	1.392E-01	6.379E-02	1.568E-02	4.207E-03	1.171E-03
2.212E-01	1.474E-01	6.739E-02	1.647E-02	4.417E-03	1.232E-03
2.002E-01	1.570E-01	7.165E-02	1.741E-02	4.668E-03	1.304E-03
1.811E-01	1.678E-01	7.641E-02	1.847E-02	4.949E-03	1.385E-03
1.639E-01	1.796E-01	8.165E-02	1.963E-02	5.259E-03	1.474E-03
1.483E-01	1.914E-01	8.691E-02	2.080E-02	5.568E-03	1.563E-03
1.342E-01	2.036E-01	9.229E-02	2.199E-02	5.885E-03	1.654E-03
1.214E-01	2.181E-01	9.876E-02	2.343E-02	6.265E-03	1.764E-03
1.098E-01	2.338E-01	1.058E-01	2.498E-02	6.676E-03	1.882E-03
9.939E-02	2.507E-01	1.133E-01	2.665E-02	7.116E-03	2.009E-03
8.994E-02	2.686E-01	1.213E-01	2.842E-02	7.583E-03	2.144E-03
8.138E-02	2.885E-01	1.302E-01	3.038E-02	8.102E-03	2.293E-03
7.363E-02	3.113E-01	1.403E-01	3.263E-02	8.693E-03	2.464E-03
6.663E-02	3.361E-01	1.515E-01	3.508E-02	9.337E-03	2.650E-03
6.029E-02	3.632E-01	1.636E-01	3.775E-02	1.004E-02	2.852E-03
5.455E-02	3.928E-01	1.768E-01	4.065E-02	1.080E-02	3.072E-03
4.936E-02	4.277E-01	1.924E-01	4.408E-02	1.170E-02	3.332E-03
4.466E-02	4.659E-01	2.095E-01	4.782E-02	1.268E-02	3.615E-03
4.041E-02	5.053E-01	2.272E-01	5.168E-02	1.368E-02	3.906E-03
3.657E-02	5.511E-01	2.477E-01	5.615E-02	1.485E-02	4.244E-03
3.309E-02	6.005E-01	2.698E-01	6.096E-02	1.610E-02	4.607E-03
2.994E-02	6.556E-01	2.945E-01	6.633E-02	1.749E-02	5.011E-03
2.709E-02	7.176E-01	3.223E-01	7.235E-02	1.906E-02	5.465E-03

DIFFERENTIAL NUMBER SPECTRA, FISSION POINT SOURCE

EXPONENTIAL FLUX IN WATER

N(R,E),N/SEC-MEV

R

E,MEV	15.	20.	30.	40.	50.
2.451E-02	7.851E-01	3.526E-01	7.890E-02	2.075E-02	5.957E-03
2.218E-02	8.645E-01	3.882E-01	8.658E-02	2.273E-02	6.534E-03
2.007E-02	9.426E-01	4.233E-01	9.413E-02	2.468E-02	7.100E-03
1.816E-02	1.034E 00	4.643E-01	1.029E-01	2.695E-02	7.760E-03
1.643E-02	1.136E 00	5.101E-01	1.128E-01	2.947E-02	8.494E-03
1.487E-02	1.249E 00	5.608E-01	1.236E-01	3.226E-02	9.305E-03
1.345E-02	1.374E 00	6.170E-01	1.356E-01	3.533E-02	1.020E-02
1.217E-02	1.513E 00	6.796E-01	1.489E-01	3.873E-02	1.119E-02
1.101E-02	1.668E 00	7.493E-01	1.637E-01	4.250E-02	1.229E-02
9.965E-03	1.839E 00	8.267E-01	1.801E-01	4.667E-02	1.351E-02
9.017E-03	2.031E 00	9.130E-01	1.984E-01	5.131E-02	1.486E-02
8.159E-03	2.244E 00	1.009E 00	2.186E-01	5.643E-02	1.636E-02
7.382E-03	2.479E 00	1.115E 00	2.410E-01	6.206E-02	1.800E-02
6.680E-03	2.740E 00	1.233E 00	2.657E-01	6.830E-02	1.983E-02
6.044E-03	3.027E 00	1.362E 00	2.929E-01	7.512E-02	2.182E-02
5.469E-03	3.347E 00	1.507E 00	3.232E-01	8.271E-02	2.404E-02
4.949E-03	3.700E 00	1.667E 00	3.566E-01	9.106E-02	2.648E-02
4.478E-03	4.091E 00	1.844E 00	3.936E-01	1.003E-01	2.917E-02
4.052E-03	4.521E 00	2.039E 00	4.341E-01	1.103E-01	3.211E-02
3.666E-03	4.999E 00	2.255E 00	4.791E-01	1.215E-01	3.537E-02
3.317E-03	5.526E 00	2.495E 00	5.288E-01	1.337E-01	3.895E-02
3.001E-03	6.109E 00	2.759E 00	5.836E-01	1.473E-01	4.290E-02
2.716E-03	6.757E 00	3.053E 00	6.445E-01	1.622E-01	4.726E-02
2.457E-03	7.476E 00	3.380E 00	7.121E-01	1.787E-01	5.209E-02
2.224E-03	8.276E 00	3.744E 00	7.872E-01	1.971E-01	5.744E-02
2.012E-03	9.161E 00	4.148E 00	8.702E-01	2.173E-01	6.333E-02
1.820E-03	1.015E 01	4.596E 00	9.626E-01	2.397E-01	6.985E-02
1.647E-03	1.125E 01	5.099E 00	1.066E 00	2.647E-01	7.711E-02
1.490E-03	1.247E 01	5.656E 00	1.180E 00	2.922E-01	8.513E-02
1.349E-03	1.382E 01	6.275E 00	1.307E 00	3.227E-01	9.397E-02
1.220E-03	1.532E 01	6.961E 00	1.448E 00	3.563E-01	1.037E-01
1.104E-03	1.698E 01	7.722E 00	1.603E 00	3.935E-01	1.145E-01
9.991E-04	1.882E 01	8.567E 00	1.776E 00	4.345E-01	1.263E-01
9.040E-04	2.086E 01	9.503E 00	1.967E 00	4.798E-01	1.394E-01
8.180E-04	2.312E 01	1.054E 01	2.179E 00	5.299E-01	1.539E-01

DIFFERENTIAL NUMBER SPECTRA, FISSION POINT SOURCE

EXPONENTIAL FLUX IN WATER

N(R,E),N/SEC-MEV

R

E,MEV	60.	70.	80.	90.	100.
1.802E 01	1.199E-08	5.549E-09	2.568E-09	1.189E-09	5.503E-10
1.630E 01	5.039E-08	2.311E-08	1.059E-08	4.851E-09	2.221E-09
1.475E 01	1.647E-07	7.388E-08	3.313E-08	1.486E-08	6.656E-09
1.335E 01	4.560E-07	1.992E-07	8.704E-08	3.803E-08	1.661E-08
1.208E 01	1.256E-06	5.436E-07	2.352E-07	1.017E-07	4.389E-08
1.093E 01	3.164E-06	1.367E-06	5.898E-07	2.541E-07	1.093E-07
9.888E 00	7.046E-06	3.032E-06	1.303E-06	5.597E-07	2.399E-07
8.947E 00	1.286E-05	5.420E-06	2.286E-06	9.640E-07	4.063E-07
8.096E 00	2.611E-05	1.103E-05	4.656E-06	1.963E-06	8.259E-07
7.325E 00	2.084E-05	8.039E-06	3.155E-06	1.257E-06	5.068E-07
6.628E 00	4.315E-05	1.632E-05	6.226E-06	2.401E-06	9.353E-07
5.997E 00	4.845E-05	1.764E-05	6.526E-06	2.457E-06	9.397E-07
5.427E 00	7.288E-05	2.599E-05	9.400E-06	3.457E-06	1.292E-06
4.910E 00	8.802E-05	3.067E-05	1.088E-05	3.936E-06	1.451E-06
4.443E 00	7.907E-05	2.689E-05	9.426E-06	3.403E-06	1.259E-06
4.020E 00	8.769E-05	2.925E-05	1.013E-05	3.635E-06	1.341E-06
3.638E 00	5.890E-05	1.990E-05	7.029E-06	2.575E-06	9.666E-07
3.291E 00	6.790E-05	2.275E-05	8.035E-06	2.951E-06	1.111E-06
2.978E 00	1.122E-04	3.628E-05	1.246E-05	4.485E-06	1.667E-06
2.695E 00	1.167E-04	3.739E-05	1.277E-05	4.588E-06	1.703E-06
2.438E 00	1.309E-04	4.124E-05	1.390E-05	4.947E-06	1.825E-06
2.206E 00	1.264E-04	3.970E-05	1.339E-05	4.771E-06	1.762E-06
1.996E 00	1.239E-04	3.897E-05	1.318E-05	4.709E-06	1.743E-06
1.806E 00	1.170E-04	3.699E-05	1.258E-05	4.518E-06	1.677E-06
1.635E 00	1.090E-04	3.468E-05	1.186E-05	4.276E-06	1.591E-06
1.479E 00	1.459E-04	4.610E-05	1.570E-05	5.645E-06	2.098E-06
1.338E 00	1.440E-04	4.563E-05	1.558E-05	5.613E-06	2.089E-06
1.211E 00	1.600E-04	5.053E-05	1.722E-05	6.200E-06	2.306E-06
1.096E 00	1.417E-04	4.488E-05	1.533E-05	5.531E-06	2.059E-06
9.914E-01	1.158E-04	3.689E-05	1.266E-05	4.580E-06	1.708E-06
8.970E-01	1.675E-04	5.311E-05	1.815E-05	6.548E-06	2.438E-06
8.117E-01	2.075E-04	6.572E-05	2.244E-05	8.090E-06	3.011E-06
7.344E-01	2.072E-04	6.550E-05	2.233E-05	8.043E-06	2.992E-06

DIFFERENTIAL NUMBER SPECTRA, FISSION POINT SOURCE

EXPONENTIAL FLUX IN WATER

N(R,E),N/SEC-MEV

R

E,MEV	60.	70.	80.	90.	100.
6.645E-01	2.072E-04	6.543E-05	2.228E-05	8.014E-06	2.979E-06
6.013E-01	2.137E-04	6.744E-05	2.294E-05	8.248E-06	3.065E-06
5.441E-01	2.242E-04	7.071E-05	2.403E-05	8.631E-06	3.206E-06
4.923E-01	2.253E-04	7.106E-05	2.415E-05	8.672E-06	3.221E-06
4.455E-01	1.657E-04	5.248E-05	1.792E-05	6.462E-06	2.406E-06
4.031E-01	2.450E-04	7.736E-05	2.629E-05	9.444E-06	3.508E-06
3.647E-01	2.793E-04	8.812E-05	2.992E-05	1.074E-05	3.987E-06
3.300E-01	3.019E-04	9.519E-05	3.228E-05	1.157E-05	4.293E-06
2.986E-01	3.190E-04	1.005E-04	3.403E-05	1.218E-05	4.514E-06
2.702E-01	3.312E-04	1.043E-04	3.525E-05	1.260E-05	4.665E-06
2.445E-01	3.436E-04	1.081E-04	3.651E-05	1.303E-05	4.822E-06
2.212E-01	3.618E-04	1.138E-04	3.838E-05	1.368E-05	5.058E-06
2.002E-01	3.834E-04	1.206E-04	4.061E-05	1.446E-05	5.342E-06
1.811E-01	4.076E-04	1.281E-04	4.311E-05	1.533E-05	5.660E-06
1.639E-01	4.342E-04	1.365E-04	4.587E-05	1.629E-05	6.009E-06
1.483E-01	4.609E-04	1.449E-04	4.864E-05	1.726E-05	6.360E-06
1.342E-01	4.884E-04	1.535E-04	5.148E-05	1.825E-05	6.720E-06
1.214E-01	5.212E-04	1.638E-04	5.488E-05	1.943E-05	7.150E-06
1.098E-01	5.568E-04	1.749E-04	5.857E-05	2.071E-05	7.615E-06
9.939E-02	5.950E-04	1.869E-04	6.253E-05	2.209E-05	8.113E-06
8.994E-02	6.356E-04	1.997E-04	6.674E-05	2.355E-05	8.643E-06
8.138E-02	6.806E-04	2.138E-04	7.141E-05	2.517E-05	9.231E-06
7.363E-02	7.321E-04	2.300E-04	7.675E-05	2.702E-05	9.901E-06
6.663E-02	7.883E-04	2.477E-04	8.258E-05	2.904E-05	1.063E-05
6.029E-02	8.495E-04	2.670E-04	8.894E-05	3.124E-05	1.143E-05
5.455E-02	9.161E-04	2.880E-04	9.586E-05	3.363E-05	1.229E-05
4.936E-02	9.947E-04	3.127E-04	1.040E-04	3.645E-05	1.330E-05
4.466E-02	1.081E-03	3.399E-04	1.130E-04	3.953E-05	1.441E-05
4.041E-02	1.169E-03	3.678E-04	1.222E-04	4.270E-05	1.555E-05
3.657E-02	1.272E-03	4.002E-04	1.328E-04	4.638E-05	1.688E-05
3.309E-02	1.383E-03	4.352E-04	1.443E-04	5.034E-05	1.830E-05
2.994E-02	1.506E-03	4.742E-04	1.572E-04	5.475E-05	1.988E-05
2.709E-02	1.645E-03	5.181E-04	1.716E-04	5.970E-05	2.165E-05

DIFFERENTIAL NUMBER SPECTRA, FISSION POINT SOURCE

EXPONENTIAL FLUX IN WATER

N(R,E),N/SEC-MEV

R

E,MEV	60.	70.	80.	90.	100.
2.451E-02	1.795E-03	5.658E-04	1.873E-04	6.509E-05	2.357E-05
2.218E-02	1.972E-03	6.218E-04	2.057E-04	7.139E-05	2.582E-05
2.007E-02	2.146E-03	6.770E-04	2.238E-04	7.760E-05	2.803E-05
1.816E-02	2.349E-03	7.414E-04	2.450E-04	8.483E-05	3.060E-05
1.643E-02	2.575E-03	8.133E-04	2.687E-04	9.290E-05	3.347E-05
1.487E-02	2.825E-03	8.929E-04	2.948E-04	1.018E-04	3.663E-05
1.345E-02	3.101E-03	9.809E-04	3.237E-04	1.117E-04	4.011E-05
1.217E-02	3.408E-03	1.079E-03	3.559E-04	1.226E-04	4.398E-05
1.101E-02	3.749E-03	1.188E-03	3.917E-04	1.348E-04	4.828E-05
9.965E-03	4.127E-03	1.309E-03	4.315E-04	1.483E-04	5.302E-05
9.017E-03	4.548E-03	1.444E-03	4.758E-04	1.633E-04	5.829E-05
8.159E-03	5.014E-03	1.593E-03	5.249E-04	1.799E-04	6.413E-05
7.382E-03	5.528E-03	1.759E-03	5.793E-04	1.983E-04	7.056E-05
6.680E-03	6.098E-03	1.942E-03	6.396E-04	2.187E-04	7.768E-05
6.044E-03	6.723E-03	2.144E-03	7.060E-04	2.411E-04	8.548E-05
5.469E-03	7.419E-03	2.368E-03	7.800E-04	2.660E-04	9.416E-05
4.949E-03	8.188E-03	2.617E-03	8.619E-04	2.936E-04	1.037E-04
4.478E-03	9.035E-03	2.892E-03	9.525E-04	3.241E-04	1.143E-04
4.052E-03	9.965E-03	3.194E-03	1.052E-03	3.576E-04	1.259E-04
3.666E-03	1.100E-02	3.529E-03	1.163E-03	3.948E-04	1.387E-04
3.317E-03	1.213E-02	3.900E-03	1.285E-03	4.360E-04	1.528E-04
3.001E-03	1.338E-02	4.309E-03	1.421E-03	4.815E-04	1.685E-04
2.716E-03	1.477E-02	4.764E-03	1.572E-03	5.320E-04	1.858E-04
2.457E-03	1.631E-02	5.270E-03	1.739E-03	5.882E-04	2.050E-04
2.224E-03	1.802E-02	5.831E-03	1.926E-03	6.508E-04	2.264E-04
2.012E-03	1.990E-02	6.453E-03	2.133E-03	7.202E-04	2.500E-04
1.820E-03	2.200E-02	7.145E-03	2.363E-03	7.975E-04	2.762E-04
1.647E-03	2.433E-02	7.918E-03	2.621E-03	8.840E-04	3.056E-04
1.490E-03	2.691E-02	8.775E-03	2.908E-03	9.803E-04	3.382E-04
1.349E-03	2.976E-02	9.725E-03	3.226E-03	1.087E-03	3.744E-04
1.220E-03	3.290E-02	1.078E-02	3.580E-03	1.206E-03	4.145E-04
1.104E-03	3.638E-02	1.194E-02	3.973E-03	1.338E-03	4.591E-04
9.991E-04	4.022E-02	1.323E-02	4.408E-03	1.485E-03	5.085E-04
9.040E-04	4.446E-02	1.466E-02	4.892E-03	1.648E-03	5.635E-04
8.180E-04	4.915E-02	1.625E-02	5.430E-03	1.829E-03	6.246E-04

DIFFERENTIAL NUMBER SPECTRA, FISSION POINT SOURCE

EXPONENTIAL FLUX IN WATER

N(R,E),N/SEC-MEV

R

E,MEV	110.	120.	130.	140.	150.
1.802E 01	2.544E-10	1.174E-10	5.403E-11	2.480E-11	1.135E-11
1.630E 01	1.015E-09	4.629E-10	2.106E-10	9.551E-11	4.319E-11
1.475E 01	2.978E-09	1.330E-09	5.921E-10	2.629E-10	1.164E-10
1.335E 01	7.246E-09	3.155E-09	1.370E-09	5.933E-10	2.562E-10
1.208E 01	1.890E-08	8.120E-09	3.477E-09	1.484E-09	6.316E-10
1.093E 01	4.687E-08	2.004E-08	8.537E-09	3.625E-09	1.534E-09
9.888E 00	1.025E-07	4.369E-08	1.855E-08	7.849E-09	3.310E-09
8.947E 00	1.709E-07	7.174E-08	3.002E-08	1.252E-08	5.207E-09
8.096E 00	3.465E-07	1.450E-07	6.042E-08	2.510E-08	1.039E-08
7.325E 00	2.059E-07	8.396E-08	3.430E-08	1.401E-08	5.718E-09
6.628E 00	3.674E-07	1.452E-07	5.761E-08	2.291E-08	9.116E-09
5.997E 00	3.639E-07	1.421E-07	5.584E-08	2.201E-08	8.685E-09
5.427E 00	4.896E-07	1.874E-07	7.229E-08	2.801E-08	1.088E-08
4.910E 00	5.435E-07	2.060E-07	7.867E-08	3.020E-08	1.163E-08
4.443E 00	4.742E-07	1.806E-07	6.928E-08	2.666E-08	1.026E-08
4.020E 00	5.041E-07	1.918E-07	7.344E-08	2.819E-08	1.082E-08
3.638E 00	3.682E-07	1.413E-07	5.435E-08	2.090E-08	8.024E-09
3.291E 00	4.240E-07	1.627E-07	6.249E-08	2.397E-08	9.173E-09
2.978E 00	6.306E-07	2.405E-07	9.200E-08	3.517E-08	1.342E-08
2.695E 00	6.436E-07	2.453E-07	9.372E-08	3.578E-08	1.363E-08
2.438E 00	6.868E-07	2.609E-07	9.943E-08	3.788E-08	1.440E-08
2.206E 00	6.635E-07	2.520E-07	9.598E-08	3.652E-08	1.386E-08
1.996E 00	6.569E-07	2.496E-07	9.499E-08	3.611E-08	1.369E-08
1.806E 00	6.334E-07	2.408E-07	9.168E-08	3.484E-08	1.321E-08
1.635E 00	6.019E-07	2.291E-07	8.726E-08	3.318E-08	1.258E-08
1.479E 00	7.925E-07	3.012E-07	1.146E-07	4.351E-08	1.647E-08
1.338E 00	7.895E-07	3.002E-07	1.142E-07	4.337E-08	1.642E-08
1.211E 00	8.711E-07	3.310E-07	1.258E-07	4.775E-08	1.806E-08
1.096E 00	7.783E-07	2.958E-07	1.125E-07	4.267E-08	1.614E-08
9.914E-01	6.465E-07	2.459E-07	9.357E-08	3.552E-08	1.344E-08
8.970E-01	9.221E-07	3.506E-07	1.334E-07	5.064E-08	1.917E-08
8.117E-01	1.139E-06	4.331E-07	1.648E-07	6.258E-08	2.369E-08
7.344E-01	1.131E-06	4.301E-07	1.637E-07	6.216E-08	2.353E-08

43

DIFFERENTIAL NUMBER SPECTRA, FISSION POINT SOURCE

EXPONENTIAL FLUX IN WATER

N(R,E),N/SEC-MEV

R

E,MEV	110.	120.	130.	140.	150.
6.645E-01	1.126E-06	4.281E-07	1.629E-07	6.189E-08	2.344E-08
6.013E-01	1.158E-06	4.404E-07	1.677E-07	6.371E-08	2.414E-08
5.441E-01	1.211E-06	4.606E-07	1.754E-07	6.668E-08	2.528E-08
4.923E-01	1.217E-06	4.630E-07	1.763E-07	6.705E-08	2.543E-08
4.455E-01	9.098E-07	3.460E-07	1.317E-07	5.002E-08	1.894E-08
4.031E-01	1.326E-06	5.046E-07	1.923E-07	7.315E-08	2.776E-08
3.647E-01	1.5[illegible]7E-06	5.735E-07	2.186E-07	8.322E-08	3.159E-08
3.300E-01	1.622E-06	6.175E-07	2.355E-07	8.971E-08	3.408E-08
2.986E-01	1.705E-06	6.493E-07	2.478E-07	9.445E-08	3.592E-08
2.702E-01	1.761E-06	6.709E-07	2.562E-07	9.773E-08	3.720E-08
2.445E-01	1.820E-06	6.934E-07	2.649E-07	1.011E-07	3.852E-08
2.212E-01	1.909E-06	7.273E-07	2.780E-07	1.062E-07	4.049E-08
2.002E-01	2.016E-06	7.681E-07	2.937E-07	1.123E-07	4.284E-08
1.811E-01	2.135E-06	8.137E-07	3.113E-07	1.191E-07	4.547E-08
1.639E-01	2.266E-06	8.638E-07	3.306E-07	1.266E-07	4.837E-08
1.483E-01	2.397E-06	9.140E-07	3.500E-07	1.341E-07	5.128E-08
1.342E-01	2.532E-06	9.655E-07	3.698E-07	1.418E-07	5.426E-08
1.214E-01	2.693E-06	1.027E-06	3.936E-07	1.510E-07	5.783E-08
1.098E-01	2.868E-06	1.094E-06	4.193E-07	1.609E-07	6.170E-08
9.939E-02	3.054E-06	1.165E-06	4.467E-07	1.716E-07	6.584E-08
8.994E-02	3.252E-06	1.240E-06	4.759E-07	1.829E-07	7.024E-08
8.138E-02	3.472E-06	1.324E-06	5.083E-07	1.955E-07	7.512E-08
7.363E-02	3.722E-06	1.420E-06	5.452E-07	2.098E-07	8.070E-08
6.663E-02	3.995E-06	1.524E-06	5.853E-07	2.254E-07	8.678E-08
6.029E-02	4.291E-06	1.637E-06	6.290E-07	2.424E-07	9.339E-08
5.455E-02	4.613E-06	1.760E-06	6.764E-07	2.608E-07	1.006E-07
4.936E-02	4.992E-06	1.904E-06	7.322E-07	2.825E-07	1.091E-07
4.466E-02	5.405E-06	2.061E-06	7.930E-07	3.062E-07	1.183E-07
4.041E-02	5.829E-06	2.223E-06	8.554E-07	3.306E-07	1.278E-07
3.657E-02	6.320E-06	2.410E-06	9.277E-07	3.587E-07	1.389E-07
3.309E-02	6.847E-06	2.610E-06	1.005E-06	3.890E-07	1.507E-07
2.994E-02	7.434E-06	2.833E-06	1.092E-06	4.227E-07	1.640E-07
2.709E-02	8.089E-06	3.083E-06	1.188E-06	4.604E-07	1.788E-07

DIFFERENTIAL NUMBER SPECTRA, FISSION POINT SOURCE

EXPONENTIAL FLUX IN WATER

N(R,E),N/SEC-MEV

R

E,MEV	110.	120.	130.	140.	150.
2.451E-02	8.801E-06	3.353E-06	1.293E-06	5.014E-07	1.949E-07
2.218E-02	9.632E-06	3.669E-06	1.415E-06	5.492E-07	2.137E-07
2.007E-02	1.045E-05	3.979E-06	1.535E-06	5.961E-07	2.322E-07
1.816E-02	1.140E-05	4.339E-06	1.674E-06	6.507E-07	2.538E-07
1.643E-02	1.245E-05	4.738E-06	1.829E-06	7.113E-07	2.777E-07
1.487E-02	1.361E-05	5.178E-06	1.999E-06	7.782E-07	3.042E-07
1.345E-02	1.489E-05	5.662E-06	2.186E-06	8.517E-07	3.333E-07
1.217E-02	1.631E-05	6.198E-06	2.393E-06	9.332E-07	3.657E-07
1.101E-02	1.788E-05	6.791E-06	2.623E-06	1.023E-06	4.015E-07
9.965E-03	1.961E-05	7.444E-06	2.875E-06	1.123E-06	4.411E-07
9.017E-03	2.154E-05	8.168E-06	3.155E-06	1.233E-06	4.851E-07
8.159E-03	2.366E-05	8.966E-06	3.463E-06	1.354E-06	5.336E-07
7.382E-03	2.599E-05	9.841E-06	3.801E-06	1.488E-06	5.869E-07
6.680E-03	2.857E-05	1.081E-05	4.174E-06	1.635E-06	6.458E-07
6.044E-03	3.138E-05	1.186E-05	4.580E-06	1.795E-06	7.102E-07
5.469E-03	3.451E-05	1.303E-05	5.029E-06	1.973E-06	7.816E-07
4.949E-03	3.795E-05	1.431E-05	5.522E-06	2.167E-06	8.599E-07
4.478E-03	4.173E-05	1.571E-05	6.062E-06	2.380E-06	9.460E-07
4.052E-03	4.587E-05	1.725E-05	6.650E-06	2.613E-06	1.040E-06
3.666E-03	5.044E-05	1.893E-05	7.297E-06	2.868E-06	1.143E-06
3.317E-03	5.547E-05	2.079E-05	8.005E-06	3.148E-06	1.257E-06
3.001E-03	6.100E-05	2.282E-05	8.780E-06	3.454E-06	1.381E-06
2.716E-03	6.711E-05	2.506E-05	9.632E-06	3.790E-06	1.517E-06
2.457E-03	7.388E-05	2.753E-05	1.057E-05	4.159E-06	1.667E-06
2.224E-03	8.136E-05	3.025E-05	1.160E-05	4.564E-06	1.832E-06
2.012E-03	8.962E-05	3.323E-05	1.272E-05	5.006E-06	2.012E-06
1.820E-03	9.877E-05	3.653E-05	1.396E-05	5.491E-06	2.210E-06
1.647E-03	1.090E-04	4.019E-05	1.533E-05	6.025E-06	2.427E-06
1.490E-03	1.202E-04	4.420E-05	1.682E-05	6.607E-06	2.665E-06
1.349E-03	1.327E-04	4.862E-05	1.846E-05	7.241E-06	2.923E-06
1.220E-03	1.464E-04	5.348E-05	2.024E-05	7.931E-06	3.203E-06
1.104E-03	1.617E-04	5.883E-05	2.220E-05	8.681E-06	3.507E-06
9.991E-04	1.785E-04	6.471E-05	2.433E-05	9.494E-06	3.835E-06
9.040E-04	1.972E-04	7.119E-05	2.666E-05	1.038E-05	4.190E-06
8.180E-04	2.179E-04	7.832E-05	2.920E-05	1.133E-05	4.571E-06

DIFFERENTIAL NUMBER SPECTRA, FISSION POINT SOURCE

EXPONENTIAL FLUX IN WATER

N(R,E),N/SEC-MEV

E,MEV	R 160.	E,MEV	R 160.
1.802E 01	5.182E-12	6.645E-01	8.849E-09
1.630E 01	1.947E-11	6.013E-01	9.114E-09
1.475E 01	5.139E-11	5.441E-01	9.549E-09
1.335E 01	1.103E-10	4.923E-01	9.610E-09
1.208E 01	2.679E-10	4.455E-01	7.146E-09
1.093E 01	6.468E-10	4.031E-01	1.050E-08
9.888E 00	1.391E-09	3.647E-01	1.195E-08
8.947E 00	2.158E-09	3.300E-01	1.291E-08
8.096E 00	4.285E-09	2.986E-01	1.362E-08
7.325E 00	2.329E-09	2.702E-01	1.412E-08
6.628E 00	3.627E-09	2.445E-01	1.463E-08
5.997E 00	3.427E-09	2.212E-01	1.539E-08
5.427E 00	4.229E-09	2.002E-01	1.630E-08
4.910E 00	4.479E-09	1.811E-01	1.732E-08
4.443E 00	3.950E-09	1.639E-01	1.844E-08
4.020E 00	4.150E-09	1.483E-01	1.956E-08
3.638E 00	3.073E-09	1.342E-01	2.072E-08
3.291E 00	3.500E-09	1.214E-01	2.210E-08
2.978E 00	5.105E-09	1.098E-01	2.360E-08
2.695E 00	5.178E-09	9.939E-02	2.521E-08
2.438E 00	5.458E-09	8.994E-02	2.692E-08
2.206E 00	5.247E-09	8.138E-02	2.882E-08
1.996E 00	5.176E-09	7.363E-02	3.099E-08
1.806E 00	4.989E-09	6.663E-02	3.335E-08
1.635E 00	4.753E-09	6.029E-02	3.593E-08
1.479E 00	6.214E-09	5.455E-02	3.874E-08
1.338E 00	6.193E-09	4.936E-02	4.205E-08
1.211E 00	6.807E-09	4.466E-02	4.567E-08
1.096E 00	6.081E-09	4.041E-02	4.939E-08
9.914E-01	5.069E-09	3.657E-02	5.372E-08
8.970E-01	7.228E-09	3.309E-02	5.838E-08
8.117E-01	8.939E-09	2.994E-02	6.358E-08
7.344E-01	8.881E-09	2.709E-02	6.941E-08

DIFFERENTIAL NUMBER SPECTRA, FISSION POINT SOURCE

EXPONENTIAL FLUX IN WATER

N(R,E),N/SEC-MEV

E,MEV	R 160.
2.451E-02	7.575E-08
2.218E-02	8.319E-08
2.007E-02	9.050E-08
1.816E-02	9.904E-08
1.643E-02	1.086E-07
1.487E-02	1.191E-07
1.345E-02	1.307E-07
1.217E-02	1.436E-07
1.101E-02	1.579E-07
9.965E-03	1.738E-07
9.017E-03	1.914E-07
8.159E-03	2.110E-07
7.382E-03	2.325E-07
6.680E-03	2.564E-07
6.044E-03	2.825E-07
5.469E-03	3.115E-07
4.949E-03	3.435E-07
4.478E-03	3.787E-07
4.052E-03	4.172E-07
3.666E-03	4.598E-07
3.317E-03	5.066E-07

E,MEV	R 160.
3.001E-03	5.580E-07
2.716E-03	6.147E-07
2.457E-03	6.773E-07
2.224E-03	7.461E-07
2.012E-03	8.217E-07
1.820E-03	9.048E-07
1.647E-03	9.966E-07
1.490E-03	1.097E-06
1.349E-03	1.206E-06
1.220E-03	1.326E-06
1.104E-03	1.455E-06
9.991E-04	1.595E-06
9.040E-04	1.746E-06
8.180E-04	1.908E-06

Appendix C

Fast-Neutron Dose-Rate Calculation

The computer code which reads the RENUPAK tape also computes the integral

$$D(R) = \int N(R,E)\ F(E)\ dE$$

for any response function F(E). The integral was actually performed over lethargy u using Simpson's rule. Since dE = E du,

$$D(R) \cong \sum_i N(R,E_i)\ F(E_i)\ E_i\ W_i\ \Delta u,$$

where W_i is the weight given by Simpson's rule and Δu is the lethargy spacing (0.1).

The proper response function to use in computing the dose rate is open to question. The simplest procedure is to assume isotropic scattering in the center of mass and compute energy deposition from elastic collisions. Values of the response function for this procedure given by Henderson[1] for absorbed dose in tissue and also for biological dose (which included an arbitrary biological effect function) were used to compute two sets of dose-rate values shown in Table C-1 and Fig. C-1.

Table C-1 also shows dose-rate values obtained from multicollision flux-to-dose conversion factors (resulting from taking into account many processes) generated by Snyder and Neufeld,[2] as well as dose-rate values obtained by

---

[1] B. J. Henderson, Conversion of Neutron or Gamma-Ray Flux to Absorbed Dose Rate, XDC-59-8-179 (Aug. 14, 1959).

[2] H. Goldstein, Fundamental Aspects of Reactor Shielding, pp 19 and 30 Addison-Wesley, Reading, Mass., 1959.

Table C-1. Fast-Neutron Dose Rates (Multiplied by $4\pi R^2$)

Due to Point Fission Source in Water

R (cm)	Absorbed Tissue Dose Rate [cm^2 (rads/hr)/(neutron/sec)]			Biological Dose Rate [cm^2 (rems/hr)/(neutron/sec)]	
	Henderson	Snyder	NDA	Henderson	Snyder
5	8.88×10^{-6}	1.32×10^{-5}		7.63×10^{-5}	1.08×10^{-4}
10	4.04×10^{-6}	6.09×10^{-6}	3.78×10^{-6}	3.45×10^{-5}	4.90×10^{-5}
15	1.90×10^{-6}	2.86×10^{-6}		1.60×10^{-5}	2.27×10^{-5}
20	9.91×10^{-7}	1.47×10^{-6}	8.27×10^{-7}	8.26×10^{-6}	1.17×10^{-5}
30	2.98×10^{-7}	4.36×10^{-7}	2.55×10^{-7}	2.43×10^{-6}	3.41×10^{-6}
40	9.08×10^{-8}	1.32×10^{-7}		7.27×10^{-7}	1.01×10^{-6}
50	2.82×10^{-8}	4.07×10^{-8}		2.21×10^{-7}	3.07×10^{-7}
60	9.13×10^{-9}	1.32×10^{-8}	8.73×10^{-9}	7.08×10^{-8}	9.77×10^{-8}
70	3.12×10^{-9}	4.49×10^{-9}		2.39×10^{-8}	3.30×10^{-8}
80	1.12×10^{-9}	1.61×10^{-9}		8.51×10^{-9}	1.17×10^{-8}
90	4.16×10^{-10}	5.98×10^{-10}	3.56×10^{-10}	3.15×10^{-9}	4.33×10^{-9}
100	1.59×10^{-10}	2.28×10^{-10}		1.20×10^{-9}	1.64×10^{-9}
110	6.13×10^{-11}	8.82×10^{-11}		4.62×10^{-10}	6.34×10^{-10}
120	2.39×10^{-11}	3.44×10^{-11}	1.99×10^{-11}	1.80×10^{-10}	2.46×10^{-10}
130	9.38×10^{-12}	1.35×10^{-11}		7.02×10^{-11}	9.62×10^{-11}
140	3.68×10^{-12}	5.30×10^{-12}		2.74×10^{-11}	3.76×10^{-11}
150	1.45×10^{-12}	2.08×10^{-12}		1.07×10^{-11}	1.47×10^{-11}
160	5.67×10^{-13}	8.17×10^{-13}		4.19×10^{-12}	5.74×10^{-12}

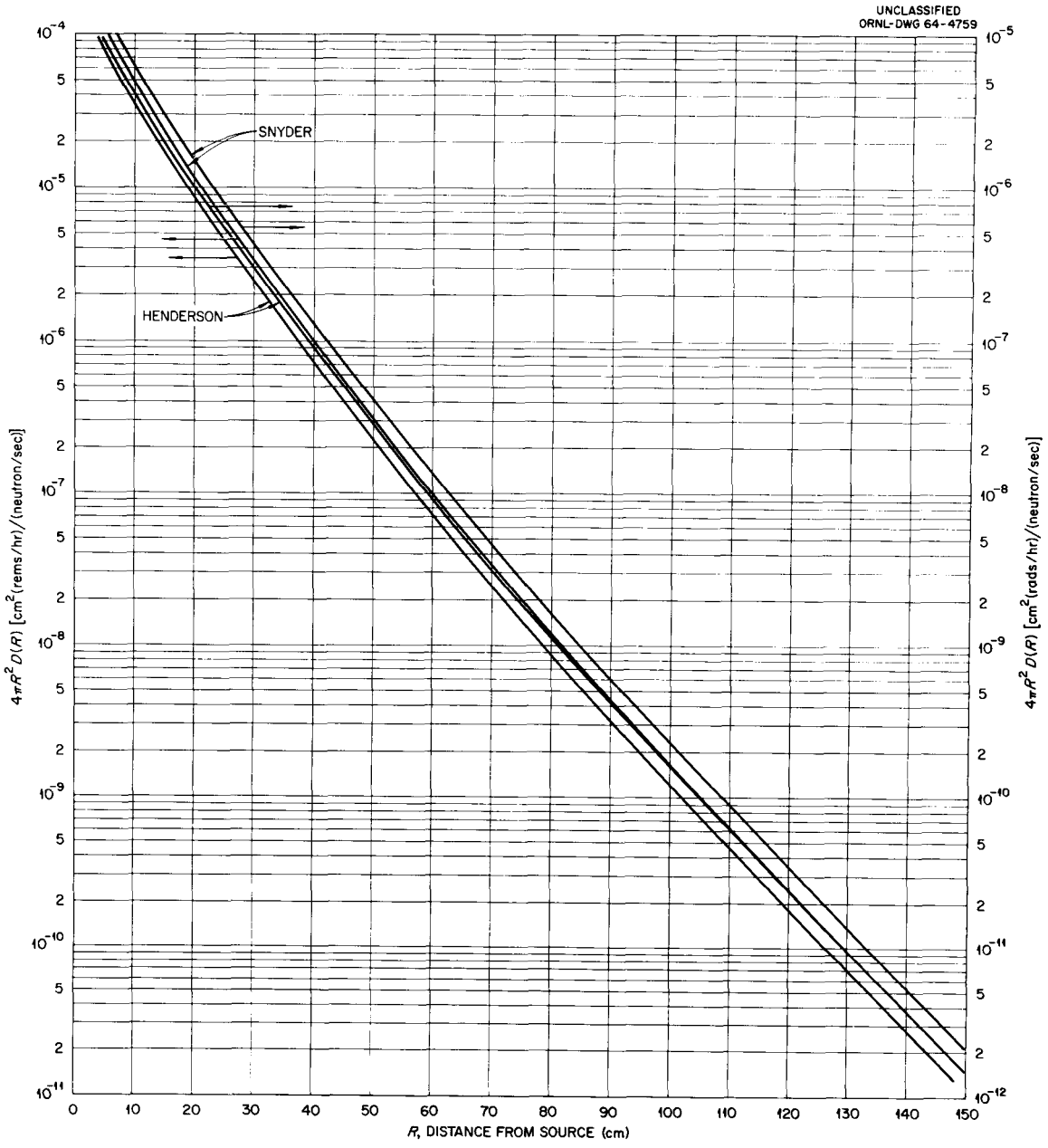

Fig. C-1. Fast-Neutron Dose Rate in Water as a Function of Distance from the Source for Various Response Functions.

previous moments method calculations[3] (divided by 0.93 to convert reps to rads). It should be noted that the previous calculation included only neutrons above 0.33 MeV. The contribution between 0.001 and 0.33 MeV adds approximately 5 to 10% to the dose-rate value.

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