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Neutron Dose and Fluence Distributions in an Infinite Air Medium

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George L. Simmons** and Charles Eisenhauer

The moments method is applied to the problem of calculating the neutron dose and fluence distributions in an infinite medium of air. These calculations are compared with Monte Carlo and Discrete Ordinates (S_n) results. Simple parametric representations for the distributions are given which facilitate the calculation of dose and flux distributions in air with a different density.

Key words: Benchmark problems; dose distributions; moments method; neutron penetration; shielding; weapons radiation.

I. INTRODUCTION

The moments method for solving neutron and gamma-ray transport problems is ideally suited for obtaining simple, but accurate representations of neutron dose and fluence distributions. In this note, we present data on neutron transport in air for three point isotropic sources of interest in shielding applications. These sources are: 1) fission source, 2) 14-MeV source, and 3) a typical thermonuclear source. These results are compared with similar data obtained using the Monte Carlo and Discrete Ordinates (S_n) techniques. We also give simple parametric representations for the dose and fluence which may be used to calculate these quantities at various distances in air of arbitrary density.

The cross section data used in these calculations are taken from

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the ENDF/B-I compilation for oxygen [1][1] and nitrogen [2] and are the same as those used by Straker in Shielding Benchmark Problem 3.0 [3]. Air was assumed to be 79% nitrogen and 21% oxygen by volume, yielding nuclear densities of 4.25×10^{-5} and 1.13×10^{-5} nuclei/barm-cm, respectively, for an air density of 1.29 grams/liter. Our calculations require that some prescription be given for interpolating on the basic cross section data. We used the specification given on the ENDF/B data file, namely, linear log-log interpolation on the cross sections and linear interpolation on the Legendre expansion coefficients of the elastic scattering angular distribution.

Calculation of spatial distributions by the moments method involves two steps. First, spatial angular moments of the fluence are obtained. These moments are then used to reconstruct distributions at various depths. We used the computer code MOMENT-I [4] to calculate spatial-angular moments at 510 energies between 15 MeV and 10^{-4} eV. The solution grid used in these calculations is given in Table I. All of the source

Table I. Solution Energy Grid Specification

Upper Energy (MeV)	Intervals Lethargy	Number
15.0	0.0314	35
5.0	0.0196	200
0.1	0.0316	60
0.015	0.0769	125
1.0×10^{-6}	0.1023	90
1.0×10^{-10}		

spectra used in these calculations were normalized to one neutron emitted. The fission source energy distribution used has the form

[1] Figures in brackets indicate the literature references at the end of this paper.

$$S(E) = 0.453 \sinh(\sqrt{2.29E}) \exp(-E/0.965). \tag{1}$$

The 14 MeV source was assumed to be represented by a histogram between 15 and 12.2 MeV. The thermonuclear weapon spectrum is given by Straker [5] and was also represented in histogram form.

In obtaining the fluence from the moments, we could perform reconstructions at all 510 energies. However, this would produce more data than can be easily managed. Also, since many of the calculational tools used in shielding analysis produce multigroup results, it is desirable to give our results in a comparable form. For these reasons, we have found it convenient to define multigroup moments as

$$\phi_n^j = \frac{1}{(E_j - E_{j-1})} \int_{E_{j-1}}^{E_j} \phi_{n,o}(E)\,dE \,, \qquad n = 0,2,4,\ldots \tag{2}$$

where E_j and E_{j-1} are the group boundaries for the j^{th} energy group (see Table II). Consistent with this procedure, we can weight these multigroup moments by a dose response functions, C_j, and sum over all groups:

$$D_n = \sum_{i=1}^{22} \Delta E_j \, \phi_n^j \, C_j \tag{3}$$

thereby obtaining dose moments (See Table II for a set of dose response functions for several detectors of interest). The dose moments can then be used to reconstruct dose distributions.

We have applied two different procedures in reconstructing distributions from moments: 1) function fitting and 2) plural series approxi-

Group	Upper Energy (eV)	Henderson Tissue Dose+	Snyder-Neufeld Dose+	Tissue Kerma*	Mid-Phantom Dose+	Concrete Kerma*	Air Kerma*	Non-Ionizing Silicon Kerma*	Ionizing Silicon Kerma*	Snyder-Neufeld Rem$
1	1.50+07	5.46-09	7.00-09	6.36-07	4.90-09	1.58-07	2.66-07	7.50-09	8.60-08	5.64-08
2	1.22+07	5.13-09	7.00-09	5.74-07	4.50-09	1.17-07	1.93-07	6.60-09	9.00-08	4.58-08
3	1.00+07	4.84-09	7.08-09	5.17-07	4.20-09	8.20-08	1.41-07	6.40-09	8.10-08	4.16-08
4	8.18+06	4.61-09	6.72-09	4.87-07	3.50-09	7.05-08	1.11-07	6.60-09	5.00-08	4.11-08
5	6.36+06	4.44-09	6.03-09	4.50-07	2.80-09	5.75-08	1.05-07	5.50-09	1.60-08	3.88-08
6	4.96+06	4.13-09	5.43-09	4.21-07	2.30-09	5.40-08	1.20-07	5.10-09	9.00-09	3.72-08
7	4.06+06	4.01-09	4.83-09	3.98-07	1.75-09	5.80-08	1.06-07	4.80-09	5.40-09	3.61-08
8	3.01+06	3.39-09	4.48-09	3.43-07	1.25-09	4.10-08	5.38-08	4.80-09	3.60-09	3.51-08
9	2.46+06	3.15-09	4.33-09	3.15-07	1.15-09	3.20-08	3.09-08	4.70-09	3.00-09	3.50-08
10	2.35+06	3.09-09	4.23-09	3.05-07	7.00-10	3.50-08	3.37-08	4.20-09	2.70-09	3.55-08
11	1.83+06	2.64-09	3.96-09	2.63-07	5.30-10	3.12-08	2.95-08	3.40-09	2.10-09	3.67-08
12	1.11+06	1.97-09	3.30-09	2.05-07	2.80-10	2.61-08	1.61-08	3.10-09	1.70-09	3.14-08
13	5.50+05	1.12-09	1.73-09	1.27-07	2.00-10	1.48-08	9.84-09	2.00-09	1.40-09	1.59-08
14	1.11+05	2.29-10	7.00-10	4.00-08	1.20-10	3.55-09	2.67-09			3.55-09
15	3.35+03		6.07-10	1.96-09	1.05-10	1.58-10	3.61-10			1.22-09
16	5.83+02		6.72-10	3.67-10	1.10-10	2.85-11	5.62-10			1.31-09
17	1.01+02		5.35-10	1.17-10	1.15-10	7.10-12	1.28-09			1.34-09
18	2.90+01		3.88-10	1.11-10	1.10-10	5.00-12	2.26-09			1.27-09
19	1.07+01		3.42-10	1.62-10	1.00-10	6.35-12	3.70-09			1.22-09
20	3.06+00		3.27-10	2.65-10	8.50-11	1.02-11	6.71-09			1.19-09
21	1.12+00		3.22-10	4.26-10	7.90-11	1.63-11	1.13-08			1.16-09
22	4.14-01		3.20-10	9.36-10	5.50-11	3.62-11	2.43-08			1.04-09

Table II. Neutron Response Functions. (+) indicates units of rad/(neutron/cm^2); (*) indicates units of (ergs/gm)/neutron/cm^2, and ($) indicates units of rem/(neutron/cm^2).

mations [6]. The first procedure has the advantage that the representations obtained for the distributions are relatively simple and are therefore easily used in point-kernel type calculations. The plural series technique, on the other hand, has the advantage that it enables us to estimate the truncation error associated with using only a finite number of moments in reconstructing the distribution. The function fitting representations make use of the form

$$4\pi R^2 D(R) = D_o e^{-\Sigma_o R} + \sum_{i=1}^{N} \alpha_i (\Sigma_o R/\beta_i)^k e^{-\Sigma_o R/\beta_i} \tag{4}$$

while the plural series representation uses the product of exponentials and U-polynomials [6] and has the form

$$4\pi R^2 D(R) = D_o e^{-\Sigma_o R} + (\Sigma_o R)^k \sum_{j=1}^{3} A_j/X_j^{k+1} \sum_{i=1}^{10} \alpha_{ij} U_n^k(\Sigma_o R/X_j) e^{-\Sigma_o R/X_j} \tag{5}$$

Here Σ_o is the mean-free-path for neutrons at the highest energy considered, i.e. 15 MeV, and k is a parameter which is specified for the fit and is usually set equal to unity. In both cases, D_o is the value of $4\pi R^2$ times the unscattered flux or dose at R = 0. D_o is easily calculated from Equations (2) and (3), provided that the source energy distribution, S(E) is substituted for the moments, $\phi_{no}(E)$.

II. RESULTS

We first compare our results with existing calculations. Theoretically, in the reconstruction of distributions from moments, an infinite number of moments are required to obtain the correct distribution. Since only a finite number of moments (six to ten) are available to characterize the distribution, we estimate the truncation error due to neglect of remaining moments.

The apparent convergence of the distribution, using successively more moments in the fit, has been used in the past to assess the truncation error. That is to say, distributions are constructed using N moments and N+1 moments, and the difference between the two distributions is examined. Generally, as more moments are used, these differences become smaller. However, there is no theoretical justification for assuming that the distributions converge to the correct value.

Spencer [6] has developed independent procedures for estimating the truncation error when plural series approximations are used to represent the distribution. These procedures are based on a more rigorous theory which gives criteria guaranteeing convergence, and which gives rigorous bounds to the truncation error which involve the norm of the unknown function. We had no difficulty satisfying the convergence criteria; but as in Ref. 6, we rely on estimations rather than rigorous values for the norm and therefore also for the truncation error bounds. The truncation error bounds represent <u>maximum</u> errors, rather than probable errors. They are therefore a conservative estimate of the truncation error.

For the data presented in this report, we have relied heavily upon the convergence of the distribution as the method for selecting the best fit to the moments. We have then applied Spencer's approach to determine rigorous error bounds on the selected fit. In Table III, we give a typical distribution which illustrate both the convergence and error

DISTRIBUTIONS				$4\pi R^2$ x Dose(R) rad-cm^2 per fission neutron				
				Number of Moments Used in the Fit				
MFP	METERS	CONVERG	BOUND	6	7	8	9	10
.04	5.00	1.86+00	1.37+02	2.8478-09	2.8437-09	2.8381-09	2.8363-09	2.8341-09
.19	25.00	1.24+00	1.93+02	3.3192-09	3.3053-09	3.2874-09	3.2816-09	3.2746-09
.37	50.00	7.08-01	1.18+02	3.7775-09	3.7615-09	3.7417-09	3.7357-09	3.7286-09
.56	75.00	3.49-01	7.28+01	4.1114-09	4.0986-09	4.0839-09	4.0797-09	4.0751-09
.75	100.00	1.13-01	6.43+01	4.3412-09	4.3335-09	4.3258-09	4.3240-09	4.3224-09
1.12	150.00	1.28-01	6.66+01	4.5548-09	4.5565-09	4.5608-09	4.5627-09	4.5656-09
1.49	200.00	1.93-01	5.14+01	4.5255-09	4.5325-09	4.5426-09	4.5460-09	4.5502-09
1.87	250.00	1.76-01	3.20+01	4.3301-09	4.3386-09	4.3493-09	4.3526-09	4.3563-09
2.24	300.00	1.25-01	1.93+01	4.0267-09	4.0343-09	4.0429-09	4.0452-09	4.0476-09
2.99	400.00	1.22-02	1.71+01	3.2658-09	3.2692-09	3.2717-09	3.2721-09	3.2721-09
3.73	500.00	5.95-02	1.62+01	2.4870-09	2.4870-09	2.4859-09	2.4853-09	2.4845-09
4.48	600.00	7.86-02	1.12+01	1.8108-09	1.8094-09	1.8073-09	1.8066-09	1.8059-09
6.72	900.00	2.35-02	9.30+00	5.9392-10	5.9344-10	5.9325-10	5.9328-10	5.9339-10
8.96	1200.00	5.75-02	5.30+00	1.7227-10	1.7244-10	1.7267-10	1.7272-10	1.7277-10
11.20	1500.00	3.63-02	8.58+00	4.7665-11	4.7737-11	4.7766-11	4.7762-11	4.7749-11
13.44	1800.00	8.32-02	5.87+00	1.2957-11	1.2954-11	1.2933-11	1.2927-11	1.2923-11
15.68	2100.00	4.13-03	1.06+01	3.4954-12	3.4858-12	3.4775-12	3.4769-12	3.4777-12
17.92	2400.00	1.26-01	9.19+00	9.3929-13	9.3603-13	9.3593-13	9.3647-13	9.3710-13
20.16	2700.00	1.26-01	1.22+01	2.5232-13	2.5215-13	2.5312-13	2.5336-13	2.5344-13
22.40	3000.00	3.74-02	1.85+01	6.8075-14	6.8420-14	6.8845-14	6.8868-14	6.8819-14
24.64	3300.00	2.52-01	1.55+01	1.8530-14	1.8737-14	1.8810-14	1.8790-14	1.8763-14
26.88	3600.00	3.32-01	2.33+01	5.1029-15	5.1732-15	5.1543-15	5.1420-15	5.1372-15
28.38	3800.00	2.42-01	3.40+01	2.1743-15	2.1998-15	2.1766-15	2.1710-15	2.1713-15
29.87	4000.00	2.98-02	3.99+01	9.3065-16	9.3609-16	9.1975-16	9.1799-16	9.1948-16
31.37	4200.00	2.74-01	3.84+01	3.9972-16	3.9802-16	3.8896-16	3.8884-16	3.9003-16
32.86	4400.00	6.11-01	3.57+01	1.7204-16	1.6886-16	1.6467-16	1.6502-16	1.6568-16
34.35	4600.00	9.02-01	4.72+01	7.4088-17	7.1385-17	6.9817-17	7.0173-17	7.0452-17
35.85	4800.00	1.07+00	7.28+01	3.1882-17	3.0040-17	2.9657-17	2.9897-17	2.9977-17
37.34	5000.00	1.03+00	9.95+01	1.3692-17	1.2573-17	1.2627-17	1.2758-17	1.2758-17
38.83	5200.00	7.45-01	1.17+02	5.8619-18	5.2303-18	5.3890-18	5.4507-18	5.4295-18
40.33	5400.00	1.82-01	1.22+02	2.4998-18	2.1615-18	2.3057-18	2.3298-18	2.3099-18

Table III. Convergence and error bound data for the Henderson dose from a point isotropic fission source in air. Distances are given in both meters and mean free path (mfp) for an air density of 1.11 g/liter.

bound data for the distribution. Under the heading CONVERG, the percentage differences between an 8 and 10 moment fit is given. Note that the truncation error bound (BOUND) calculated as indicated in References [6] and [7], is larger than the convergence indicator (CONVERG). This gives us confidence that when convergence is good, the convergence indicator is a realistic estimate of the error that can be attributed to distribution construction. However, if this indicator is large, then we must rely solely upon error bound calculations to indicate the truncation error.

A. Benchmark Problem (Air density 1.29 g/liter)

We have calculated the multigroup fluence and Henderson dose at several distances for the fission and 14 MeV sources, and compared them with Monte Carlo calculations by Straker [3]. In Table IV, we present a comparison of the multigroup fluences for the two sources for various distances from the source point. Generally the two methods compare very favorably, except in regions where the statistical error of the Monte Carlo results is large. Table V shows a similar comparison for the Henderson dose.

B. Comparisons with S_n Calculations (Air density 1.11 g/liter)

A large tabulation of fluences and dose has been made by Straker [5] for the three source types of interest, as well as for other source distributions. These calculations were made using the ANISN code ($S_{16}P_5$) for a 5000 meter sphere. Figures 1, 2, and 3 give comparisons of the multigroup spectra for the three sources of interest, as calculated by the S_n and moments methods. The major discrepancies are in the groups

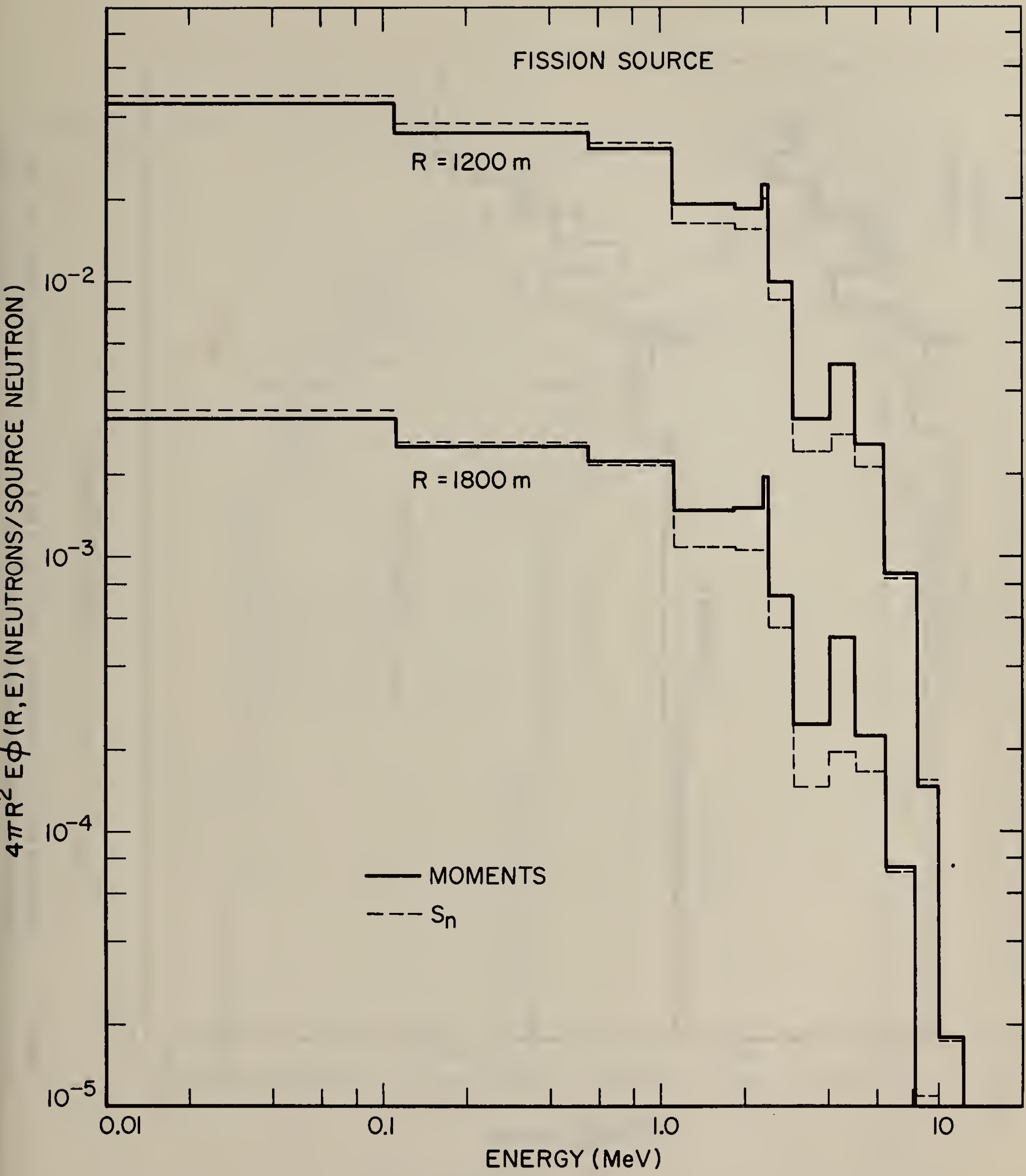

Figure 1. Comparison of moments and S_n (Ref. 5) calculations from a fission source at two depths in air.

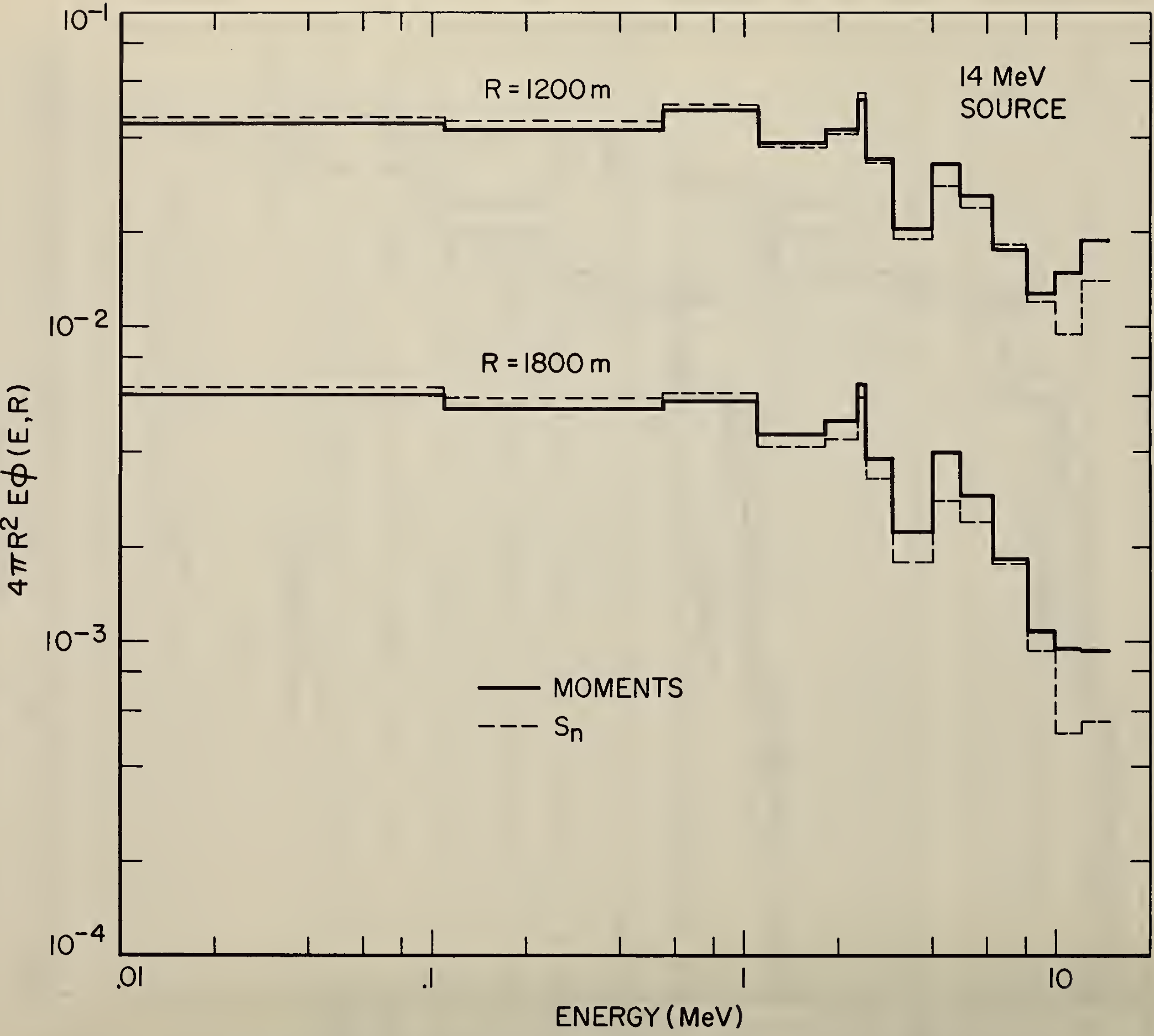

Figure 2. Comparison of moments and S_n (Ref. 5) calculations of the fluence spectrum from a 14 MeV source at two depths in air.

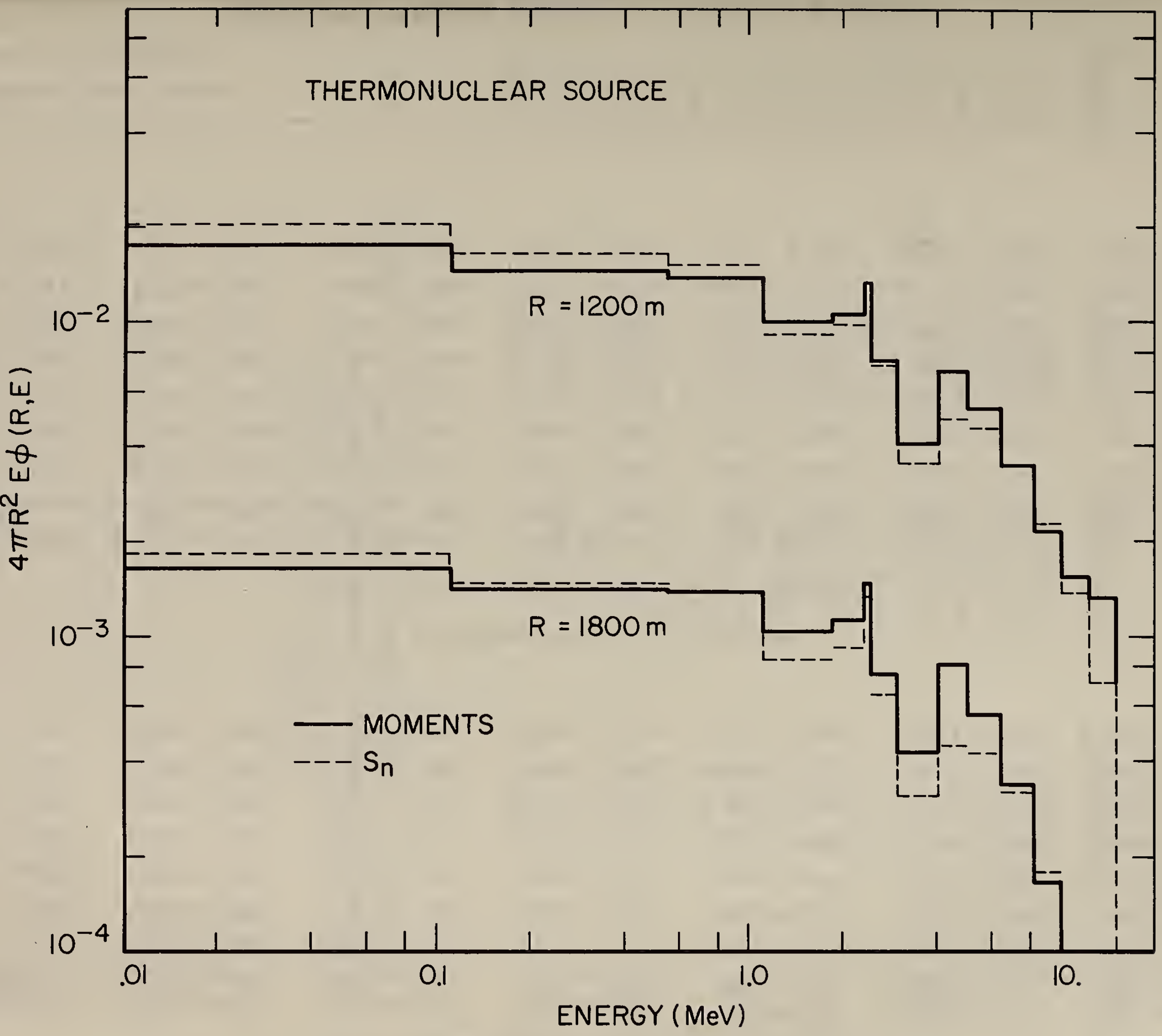

Figure 3. Comparison of moments and S_n (Ref. 5) calculations of the fluence spectrum from a thermonuclear source at two depths in air.

Table IV. FISSION SOURCE IN AIR (1.11 g/LITER)

$4\pi R^2$ x Fluence (Neutrons/MeV/Source Neutron)

Energy Bounds (MeV)	R= 71 Meters MOM*	MC+	142 Meters MOM	MC	284 Meters MOM	MC	567 Meters MOM	MC	1000 Meters MOM	MC	1420 Meters MOM	MC
15.0 - 12.2	3.85-5	3.60-5	2.64-5	2.50-5	1.21-5	1.03-5	2.47-6	1.73-6	2.00-7	1.63-7	1.64-8	7.07-9
12.2 - 8.18	9.53-3	8.09-4	7.58-3	6.11-4	4.05-4	3.24-4	1.06-4	8.41-5	1.22-5	9.40-6	1.38-6	9.90-7
8.18 - 6.36	6.34-3	6.38-3	5.13-3	4.96-3	3.10-3	2.91-3	9.69-4	8.70-4	1.39-4	1.21-4	1.89-5	1.69-5
6.36 - 4.06	3.22-2	3.25-2	2.66-2	2.63-2	1.64-2	1.61-2	5.34-3	5.23-3	8.18-4	7.77-4	1.22-4	1.05-4
4.06 - 2.36	1.46-1	1.39-1	1.22-1	1.18-1	7.52-2	7.21-2	2.20-2	1.98-2	2.72-3	2.44-3	3.40-4	2.76-4
2.36 - 1.11	3.89-1	3.79-1	3.96-1	3.94-1	2.90-1	2.85-1	9.85-2	1.02-1	1.32-2	1.26-2	1.68-3	1.38-3
1.11 - 0.11	9.77-1	7.93-1	1.34+0	1.32+0	1.34+0	1.31+0	5.71-1	5.18-1	7.90-2	6.55-2	9.41-3	7.20-3

14 MeV SOURCE IN AIR (1.11 g/LITER)

$4\pi R^2$ x Fluence (Neutrons/MeV/Source Neutron)

Energy Bounds (MeV)	R= 71 Meters MOM*	MC+	142 Meters MOM	MC	284 Meters MOM	MC	567 Meters MOM	MC	1000 Meters MOM	MC	1420 Meters MOM	MC
15.0 - 12.2	2.46-1	2.44-1	1.74-1	1.64-1	8.42-2	7.40-2	1.86-2	1.45-2	1.65-3	1.12-3	1.44-4	8.40-5
12.2 - 8.18	2.49-2	3.55-2	2.77-2	4.55-2	2.84-2	3.50-2	1.10-2	1.25-2	1.60-3	1.75-3	2.00-4	1.90-4
8.18 - 6.36	1.93-2	2.01-2	2.78-2	2.91-2	2.84-2	2.82-2	1.41-2	1.58-2	2.76-3	2.90-3	4.43-4	4.00-4
6.36 - 4.06	3.96-2	3.86-2	5.54-2	5.50-2	5.75-2	5.60-2	2.99-2	2.83-2	6.39-3	5.50-3	1.14-3	1.13-3
4.06 - 2.36	6.02-2	7.12-2	8.99-2	1.04-1	9.67-2	1.03-1	4.99-2	5.11-2	1.02-2	1.05-2	1.96-3	1.46-3
2.36 - 1.11	7.71-2	9.10-2	1.38-1	1.69-1	1.78-1	2.07-1	1.12-1	1.18-1	2.64-2	2.69-2	4.81-3	4.72-3
1.11 - 0.11	1.00-1	9.50-2	2.90-1	2.50-1	5.17-1	4.27-1	3.46-1	3.56-1	9.78-2	9.70-2	1.91-2	1.77-2

* Moments Method Results

\+ Monte Carlo Results

Table IV. Fission Source in Air (1.11 g/liter)

$4\pi R^2$ x Henderson Dose (Rad-cm^2/Source Neutron)

R	Fission Source		14 MeV Source	
(meters)	Moments	Monte Carlo	Moments	Monte Carlo
71	4.15-09	3.98-09	5.74-09	5.97-09
142	4.59-09	4.60-09	5.60-09	6.03-09
284	3.83-09	3.70-09	5.33-09	4.75-09
380	2.93-09	2.65-09	3.72-09	3.86-09
472	2.14-09	1.98-09	2.92-09	2.99-09
567	1.47-09	1.32-09	2.19-09	2.20-09
662	9.85-10	8.40-10	1.61-09	1.60-09
755	6.49-10	5.60-10	1.16-09	1.14-09
850	4.17-10	3.30-10	8.18-10	7.80-10
1000	2.03-10	1.75-10	4.58-10	4.40-10
1135	1.04-10	7.51-11	2.66-10	2.42-10
1275	5.17-11	4.25-11	1.48-10	1.38-10
1420	2.84-11	2.18-11	7.98-10	7.55-11

Table V. Comparison between Monte Carlo and moments methods calculations of Henderson dose for a point isotropic fission and 14 MeV source in air (density 1.29 g/liter).

6 and 9. At these energies there exist minima in the air cross section which may be difficult to represent in the multigroup procedure [5]. Consequently, the streaming through these minima will not be calculated correctly. This effect will become more pronounced as the source-detector separation is increased. In Table VI we show a comparison of the Henderson and Snyder-Neufeld dose for the three sources of interest. Generally the two calculations compare favorably, with the moments results tending to be higher than the S_n results. We feel that this is due to the increased streaming that is predicted by the moment calculations, as well as to the leakage effect that is associated with the finite sphere used in the S_n calculations.

C. Simple Representations for Fluence and Dose Distribution.

For point kernel calculations, it is useful to have a simple set of parameters which can be used to represent the dose and fluence for the three sources. Since we are interested in simple representations of these distributions, the results of the function fitting calculations are used. We note in passing that function fitting and plural series techniques give the same distribution over the range of interest. In making these tabulations, we restrict ourselves to representations of the distributions by 11 or fewer parameters. In Tables VII, VIII, and IX, we give coefficients for the dose quantities in Table II. The form used in calculating the distributions from the coefficients is

$$4\pi R^2 D(R) = D_o e^{-\Sigma_o R} + \sum_{i=1}^{5} \alpha_i (\Sigma_o R/\beta_i) e^{-\Sigma_o R/\beta_i} \quad (\text{rads-cm}^2) \qquad (6)$$

$4\pi R^2 \times$ Henderson Dose (Rads-cm^2/Source Neutron)

R (Meters)	Fission Source Moments	Fission Source S_n	14 MeV Source Moments	14 MeV Source S_n	Thermonuclear Weapon Source Moments	Thermonuclear Weapon Source S_n
150	4.50-09	4.49-09	5.69-09	6.07-09	2.41-09	2.49-09
300	4.05-09	3.93-09	4.88-09	5.06-09	1.95-09	2.03-09
600	1.81-09	1.76-09	2.59-09	2.60-09	8.55-10	8.85-10
900	5.92-10	5.82-10	1.09-09	1.08-09	3.08-10	3.13-10
1200	1.72-10	1.68-10	4.04-10	3.94-10	1.02-10	1.01-10
1800	1.29-11	1.16-11	4.48-11	4.24-11	9.91-12	9.12-12
2400	9.30-13	7.19-13	4.32-12	3.85-12		
3000	6.80-14	4.27-14	3.87-13	3.17-13		
3600	5.08-15	2.52-15	3.34-14	2.46-14		
4200	3.84-16	1.49-16	2.81-15	1.84-15		
4800	2.92-17	8.43-18	2.32-16	1.26-16		

$4\pi R^2 \times$ Snyder-Neufeld Dose (Rads-cm^2/Source Neutron)

R (Meters)	Fission Source Moments	Fission Source S_n	14 MeV Source Moments	14 MeV Source S_n	Thermonuclear Weapon Source Moments	Thermonuclear Weapon Source S_n
150	6.80-09	7.40-09	8.40-09	8.25-09	5.47-09	5.49-09
300	7.08-09	7.19-09	7.01-09	7.22-09	4.42-09	4.55-09
600	3.82-09	3.62-09	4.05-09	4.05-09	1.76-09	1.85-09
900	1.30-09	1.28-09	1.84-09	1.79-09	6.15-10	6.39-10
1200	3.86-10	3.82-10	7.10-10	6.88-10	2.01-10	2.02-10
1800	2.89-11	2.70-11	8.24-11	7.87-11	1.93-11	1.84-11
2400	2.02-12	1.66-12	8.11-12	7.39-12		
3000	1.41-13	9.78-14	7.35-13	6.20-13		
3600	1.03-14	5.68-15	6.36-14	4.86-14		
4200	8.10-16	3.31-16	5.35-14	3.65-14		
4800	6.84-17	1.83-17	4.40-16	2.48-16		

Table VI. Comparison between S_n calculations and moments method calculation of Henderson and Snyder-Neufeld dose for a point isotropic fission, 14 MeV, and thermonuclear weapon source in air (density 1.11 g/liter). S_n data taken from Reference 5.

FISSION SOURCE IN AIR

i	HENDERSON (RAD)		SNYDER-NEUFELD (RAD)		TISSUE KERMA (ERG/G)	
D_o	:2.711910-09	K= 1	3.843843-09	K= 1	2.752741-07	K= 1
i	α_i	β_i	α_i	β_i	α_i	β_i
1	7.024213-10	.21536	3.032895-08	.48881	9.387321-08	.21780
2	-4.313338-09	.89573	2.651148-08	1.46643	-5.611913-07	.90590
3	8.190579-09	1.38896	-3.700481-08	.63225	8.124505-06	1.40472
4	3.651962-09	1.30914	6.949531-10	1.85179	-6.808031-06	1.41258
5	2.513979-09	1.67617	0.000000	1.00000	3.164664-07	1.66805

D_o :	MID-PHANTOM (RAD)		CONCRETE KERMA (ERG/G)		AIR KERMA (ERG/G)	
	8.776196-10	K= 1	3.453603-08	K= 1	4.324899-08	K= 1
i	α_i	β_i	α_i	β_i	α_i	β_i
1	4.018530-09	.49140	9.839580-09	.21622	9.665720-08	.49046
2	4.205045-09	1.47419	-5.994808-08	.89933	1.955809-07	1.47138
3	-5.063073-09	.66646	3.863627-08	1.39454	-1.497707-07	.77937
4	1.109593-10	1.85908	1.162011-07	1.35447	6.125755-09	1.84231
5	0.000000	1.00000	3.363860-08	1.67387	0.000000	1.00000

i	NON-ION. SILICON KERMA		ION. SILICON KERMA		SNYDER-NEUFELD (REM)	
D_o	:3.710731-09	K= 1	4.329498-09	K= 1	3.210515-08	K= 1
i	α_i	β_i	α_i	β_i	α_i	β_i
1	9.003585-10	.69297	7.347080-10	.21288	-3.173191-07	1.13018
2	-1.335754-08	.98297	-4.782786-09	.88542	4.050881-07	1.25199
3	1.402235-08	1.15605	1.069465-08	1.37297	5.584109-08	1.64183
4	1.073952-08	1.44839	2.156281-09	1.00379	0.000000	1.00000
5	2.475567-09	1.69788	3.044102-09	1.66981	0.000000	1.00000

$$4\pi R^2 \times \mathrm{Dose}(R) = D_o \exp(-\Sigma_o R) + \sum_{i=1}^{5} \alpha_i (\Sigma_o R/\beta_i)^k \exp(-\Sigma_o R/\beta_i)$$

Table VII. Coefficients for calculating dose distributions in air for a point isotropic fission source. The dose distributions have units of (dose units-cm^2 per source neutron). Σ_o^{-1} is 133.059 meters for an air density of 1.11 g/liter.

14 MeV SOURCE IN AIR

i	HENDERSON (RAD) α_i	β_i	SNYDER-NEUFELD (RAD) α_i	β_i	TISSUE KERMA (ERG/G) α_i	β_i
D_o:	5.460019-09	K= 1	7.000024-09	K= 1	6.360022-07	K= 1
1	-6.645252-10	.25261	-8.955006-09	.15094	-7.657200-08	.25311
2	9.663185-09	1.05069	7.164796-09	.69768	1.172000-06	1.05276
3	6.827164-09	1.62923	4.898157-08	1.34196	8.068461-07	1.63244
4	-1.427677-08	1.14732	2.860304-08	1.70389	-1.694500-06	1.15221
5	9.676135-09	1.72521	-6.504560-08	1.31416	1.003330-06	1.72744

i	MID-PHANTOM (RAD) α_i	β_i	CONCRETE KERMA (ERG/G) α_i	β_i	AIR KERMA (ERG/G) α_i	β_i
D_o:	4.900017-09	K= 1	1.580005-07	K= 1	2.660009-07	K= 1
1	-4.743389-10	.23546	-1.517908-08	.23996	-2.829879-08	.23338
2	1.721564-09	.97933	6.707914-08	.99806	7.271133-08	.97071
3	3.092354-09	1.51858	8.164871-08	1.54762	1.485257-07	1.50521
4	-2.612731-09	1.26748	-9.334668-08	1.16231	-1.071440-07	1.31727
5	4.960820-09	1.71004	1.620361-07	1.71573	2.252018-07	1.71239

i	NON-ION. SILICON KERMA α_i	β_i	ION. SILICON KERMA α_i	β_i	SNYDER-NEUFELD (REM) α_i	β_i
D_o:	7.500026-09	K= 1	8.600030-08	K= 1	5.640019-08	K= 1
1	2.734338-08	1.63360	-6.418379-09	.21030	-9.282823-09	.26209
2	-1.368610-08	1.40440	7.669295-09	.87471	4.560220-07	1.09012
3	1.971405-09	1.86455	4.430919-08	1.35636	2.022771-07	1.69037
4	-5.864663-12	2.35862	3.612633-08	1.59763	-5.393797-07	1.13266
5	0.000000	1.00000	7.979537-09	1.74196	1.088691-08	1.61020

$$4\pi R^2 \times \text{Dose}(R) = D_o \exp(-\Sigma_o R) + \sum_{i=1}^{5} \alpha_i (\Sigma_o R/\beta_i)^k \exp(-\Sigma_o R/\beta_i)$$

Table VIII. Coefficients for calculating dose distributions in air for a point isotropic 14 MeV source. The dose distributions have units of (dose units-cm^2 per source neutron). Σ_o^{-1} is 133.059 meters for an air density of 1.11 g/liter.

THERMONUCLEAR SOURCE IN AIR

i	HENDERSON (RAD) α_i	β_i	SNYDER-NEUFELD (RAD) α_i	β_i	TISSUE KERMA (ERG/G) α_i	β_i
D_o:	1.598693-09	K= 1	2.462538-09	K= 1	1.772189-07	K= 1
1	2.235079-10	.53182	5.087558-09	1.37554	-3.418369-08	1.41642
2	2.229882-09	1.35445	-8.094390-09	.47979	5.063568-08	.60201
3	2.699256-09	1.69711	9.294106-09	.63088	2.976032-07	1.37311
4	2.951245-10	1.75437	9.040126-11	1.60195	3.183777-07	1.70691
5	0.000000	1.00000	5.399257-09	1.70926	0.000000	1.00000

i	MID-PHANTOM (RAD) α_i	β_i	CONCRETE KERMA (ERG/G) α_i	β_i	AIR KERMA (ERG/G) α_i	β_i
D_o:	8.905867-10	K= 1	2.863221-08	K= 1	4.131295-08	K= 1
1	-7.337294-10	.21165	4.211789-09	.61293	4.733615-08	1.39377
2	1.531144-09	.88030	3.432219-08	1.35017	-7.026025-08	.57382
3	1.160956-09	1.36502	3.735728-08	1.69671	7.407690-08	.82382
4	-8.230029-10	.99819	2.190751-09	1.77446	4.939778-08	1.70684
5	1.135242-09	1.70155	0.000000	1.00000	0.000000	1.00000

i	NON-ION. SILICON KERMA α_i	β_i	ION. SILICON KERMA α_i	β_i	SNYDER-NEUFELD (REM) α_i	β_i
D_o:	2.104689-09	K= 1	1.139916-08	K= 1	1.842464-08	K= 1
1	-2.849184-09	.48294	-1.102444-10	1.35895	-1.560135-08	.47091
2	1.578367-09	1.44882	-3.306599-12	.40243	2.265075-08	1.41273
3	2.592974-09	.43650	3.814736-09	1.17663	1.859109-08	.65473
4	1.771836-09	1.28652	7.002446-09	1.45429	1.240604-08	1.28815
5	3.855546-09	1.70903	4.738565-09	1.68985	3.849104-08	1.71016

$$4\pi R^2 \times \mathrm{Dose}(R) = D_o \exp(-\Sigma_o R) + \sum_{i=1}^{5} \alpha_i (\Sigma_o R/\beta_i)^k \exp(-\Sigma_o R/\beta_i)$$

Table IX. Coefficients for calculating dose distributions in air for a PTI thermonuclear weapon source. The dose distributions have units of (dose units-cm^2 per source neutron). Σ_o^{-1} is 133.059 meters for an air density of 1.11 g/liter.

Here $1/\Sigma_o$ = 133.905 meters which corresponds to an energy of 15 MeV and an air density of 1.11 grams/liter. These same coefficients may be applied to calculations at different air densities by using simple scaling procedures based on equivalent mean free paths. In Tables X, XI, and XII, the distributions obtained by using the coefficients in Tables VII, VIII, and IX are given.

In Tables XIII, XIV, and XV, the coefficients which represent the multigroup spectra for each source are given. The form for these distributions for the j^{th} energy group is

$$4\pi R^2 \phi^j(R) = \phi_o^j\, e^{-\Sigma_o R} + \sum_{i=1}^{5} \alpha_i (\Sigma_o R/\beta_i) e^{-\Sigma_o R/\beta_i} \quad \text{(neutrons/MeV/source neutron)} \tag{7}$$

Tables XVI, XVII, and XVIII give the distributions obtained using these coefficients. Note that for the thermonuclear weapon source, at very low energies and close to the source, the distributions are negative. This indicates that the coefficients for these particular energies should be used with caution at distances less than one mean-free-path. Indeed in this particular region, it is extremely difficult to fit the moments and obtain suitable distributions over the entire range.

D. Angular Distributions.

Angular distributions for dose or fluence can also be calculated by the moments method. Generally the procedure that is used for point sources is to calculate spatial distributions for Legendre harmonic coefficients $\phi_\ell(E,z)$ for values of ℓ up to about 8. In theory, the angular distributions at any distance can then be calculated by perform-

FISSION SOURCE IN AIR

$4\pi R^2 \times$ Dose (Dose Units $\times$ cm^2 / Source Neutron)

R (meters)	Henderson Dose (Rads)	Snyder-Neufeld Dose (Rads)	Tissue Kerma (Ergs/g)	Mid-Phantom Dose (Rads)	Concrete Kerma (Ergs/g)	Air Kerma (Ergs/g)	Non-Ionizing Silicon Kerma (Ergs/g)	Ionizing Silicon Kerma (Ergs/g)	Snyder-Neufeld Dose (Rems)
0	2.712-09	3.844-09	2.753-07	8.776-10	3.454-08	4.325-08	3.711-09	4.329-09	3.211-08
25	3.441-09	5.998-09	3.571-07	1.180-09	4.396-08	5.521-08	4.554-09	4.950-09	3.991-08
50	3.811-09	6.679-09	3.959-07	1.257-09	4.860-08	5.922-08	5.222-09	5.194-09	4.653-08
75	4.068-09	6.794-09	4.231-07	1.248-09	5.183-08	5.973-08	5.732-09	5.325-09	5.193-08
100	4.266-09	6.763-09	4.454-07	1.215-09	5.437-08	5.896-08	6.097-09	5.403-09	5.612-08
150	4.498-09	6.797-09	4.754-07	1.170-09	5.752-08	5.703-08	6.462-09	5.416-09	6.114-08
200	4.512-09	6.995-09	4.828-07	1.162-09	5.792-08	5.564-08	6.446-09	5.241-09	6.239-08
250	4.343-09	7.138-09	4.699-07	1.158-09	5.596-08	5.419-08	6.166-09	4.916-09	6.083-08
300	4.049-09	7.079-09	4.421-07	1.133-09	5.233-08	5.206-08	5.716-09	4.494-09	5.733-08
400	3.280-09	6.320-09	3.630-07	1.003-09	4.259-08	4.519-08	4.596-09	3.545-09	4.735-08
500	2.489-09	5.085-09	2.781-07	8.075-10	3.242-08	3.631-08	3.472-09	2.645-09	3.653-08
600	1.807-09	3.818-09	2.033-07	6.087-10	2.359-08	2.749-08	2.513-09	1.900-09	2.690-08
900	5.924-10	1.304-09	6.736-08	2.105-10	7.761-09	9.676-09	8.186-10	6.153-10	9.056-09
1200	1.724-10	3.861-10	1.965-08	6.288-11	2.259-09	2.915-09	2.371-10	1.796-10	2.655-09
1500	4.764-11	1.074-10	5.423-09	1.763-11	6.236-10	8.199-10	6.538-11	5.010-11	7.307-10
1800	1.288-11	2.894-11	1.462-09	4.785-12	1.684-10	2.228-10	1.768-11	1.371-11	1.959-10
2100	3.460-12	7.663-12	3.910-10	1.276-12	4.514-11	5.947-11	4.754-12	3.726-12	5.218-11
2400	9.298-13	2.015-12	1.047-10	3.378-13	1.211-11	1.576-11	1.279-12	1.012-12	1.393-11
2700	2.509-13	5.304-13	2.815-11	8.946-14	3.264-12	4.180-12	3.452-13	2.753-13	3.741-12
3000	6.803-14	1.407-13	7.613-12	2.385-14	8.842-13	1.116-12	9.357-14	7.514-14	1.010-12
3300	1.854-14	3.776-14	2.070-12	6.435-15	2.408-13	3.014-13	2.547-14	2.057-14	2.740-13
3600	5.075-15	1.030-14	5.657-13	1.763-15	6.587-14	8.256-14	6.963-15	5.645-15	7.449-14
3800	2.144-15	4.376-15	2.387-13	7.508-16	2.781-14	3.514-14	2.939-15	2.386-15	3.128-14
4000	9.064-16	1.875-15	1.008-13	3.224-16	1.175-14	1.507-14	1.243-15	1.009-15	1.313-14
4200	3.836-16	8.099-16	4.263-14	1.396-16	4.972-15	6.513-15	5.261-16	4.270-16	5.513-15
4400	1.624-16	3.527-16	1.803-14	6.091-17	2.105-15	2.835-15	2.230-16	1.807-16	2.313-15
4600	6.881-17	1.548-16	7.628-15	2.678-17	8.910-16	1.242-15	9.461-17	7.644-17	9.693-16
4800	2.915-17	6.839-17	3.227-15	1.186-17	3.773-16	5.479-16	4.018-17	3.234-17	4.060-16
5000	1.235-17	3.041-17	1.365-15	5.283-18	1.597-16	2.430-16	1.708-17	1.367-17	1.698-16
5200	5.232-18	1.359-17	5.772-16	2.367-18	6.762-17	1.083-16	7.261-18	5.780-18	7.099-17
5400	2.216-18	6.105-18	2.439-16	1.065-18	2.862-17	4.848-17	3.089-18	2.442-18	2.964-17

Table X. Dose distributions in air (1.11 g/liter) for a point isotropic fission source.

14 MeV SOURCE IN AIR

$4\pi R^2 \times$ Dose (Dose Units $\times$ cm^2/Source Neutron

R (meters)	Henderson Dose (Rads)	Snyder-Neufeld Dose (Rads)	Tissue Kerma (Ergs/g)	Mid-Phantom Dose (Rads)	Concrete Kerma (Ergs/g)	Air Kerma (Ergs/g)	Non-Ionizing Silicon Kerma (Ergs/g)	Ionizing Silicon Kerma (Ergs/g)	Snyder-Neufeld Dose (Rems)
0	5.460-09	7.000-09	6.360-07	4.900-09	1.580-07	2.660-07	7.500-09	8.600-08	5.640-08
25	5.399-09	4.781-09	6.209-07	4.656-09	1.479-07	2.472-07	7.595-09	8.017-08	5.498-08
50	5.522-09	6.640-09	6.279-07	4.559-09	1.431-07	2.384-07	7.669-09	7.673-08	5.632-08
75	5.644-09	7.896-09	6.361-07	4.467-09	1.392-07	2.308-07	7.712-09	7.340-08	5.796-08
100	5.715-09	8.405-09	6.393-07	4.347-09	1.349-07	2.222-07	7.719-09	6.976-08	5.916-08
150	5.693-09	8.401-09	6.295-07	4.031-09	1.248-07	2.023-07	7.623-09	6.190-08	5.985-08
200	5.508-09	7.993-09	6.042-07	3.665-09	1.137-07	1.811-07	7.386-09	5.404-08	5.878-08
250	5.223-09	7.509-09	5.696-07	3.291-09	1.025-07	1.605-07	7.033-09	4.667-08	5.655-08
300	4.875-09	7.012-09	5.296-07	2.929-09	9.170-08	1.412-07	6.595-09	4.000-08	5.356-08
400	4.095-09	6.007-09	4.428-07	2.274-09	7.201-08	1.077-07	5.575-09	2.886-08	4.624-08
500	3.308-09	5.005-09	3.572-07	1.728-09	5.531-08	8.080-08	4.518-09	2.045-08	3.829-08
600	2.592-09	4.048-09	2.798-07	1.291-09	4.169-08	5.976-08	3.544-09	1.429-08	3.066-08
900	1.092-09	1.837-09	1.183-07	4.959-10	1.630-08	2.257-08	1.492-09	4.591-09	1.357-08
1200	4.040-10	7.101-10	4.392-08	1.742-10	5.780-09	7.874-09	5.519-10	1.391-09	5.179-09
1500	1.381-10	2.493-10	1.505-08	5.764-11	1.923-09	2.599-09	1.887-10	4.072-10	1.805-09
1800	4.484-11	8.245-11	4.895-09	1.830-11	6.135-10	8.254-10	6.133-11	1.168-10	5.941-10
2100	1.408-11	2.621-11	1.538-09	5.644-12	1.901-10	2.550-10	1.926-11	3.308-11	1.882-10
2400	4.316-12	8.106-12	4.719-10	1.706-12	5.774-11	7.724-11	5.900-12	9.302-12	5.808-11
2700	1.301-12	2.459-12	1.423-10	5.080-13	1.728-11	2.306-11	1.776-12	2.604-12	1.759-11
3000	3.873-13	7.348-13	4.235-11	1.496-13	5.116-12	6.811-12	5.282-13	7.272-13	5.253-12
3300	1.142-13	2.172-13	1.249-11	4.370-14	1.502-12	1.995-12	1.557-13	2.027-13	1.553-12
3600	3.339-14	6.360-14	3.652-12	1.268-14	4.379-13	5.805-13	4.558-14	5.648-14	4.552-13
3800	1.466-14	2.794-14	1.604-12	5.541-15	1.920-13	2.541-13	2.004-14	2.408-14	2.002-13
4000	6.424-15	1.224-14	7.026-13	2.416-15	8.397-14	1.110-13	8.799-15	1.027-14	8.780-14
4200	2.808-15	5.346-15	3.072-13	1.052-15	3.666-14	4.841-14	3.856-15	4.379-15	3.842-14
4400	1.225-15	2.330-15	1.340-13	4.570-16	1.598-14	2.107-14	1.688-15	1.868-15	1.679-14
4600	5.338-16	1.014-15	5.839-14	1.983-16	6.955-15	9.161-15	7.373-16	7.965-16	7.320-15
4800	2.322-16	4.401-16	2.540-14	8.594-17	3.022-15	3.977-15	3.215-16	3.398-16	3.188-15
5000	1.008-16	1.907-16	1.103-14	3.719-17	1.312-15	1.724-15	1.398-16	1.450-16	1.387-15
5200	4.374-17	8.254-17	4.787-15	1.608-17	5.687-16	7.466-16	6.064-17	6.187-17	6.024-16
5400	1.896-17	3.567-17	2.075-15	6.944-18	2.463-16	3.230-16	2.618-17	2.641-17	2.615-16

Table XI. Dose distributions in air (1.11 g/liter) for a point isotropic 14 MeV source.

14 MeV SOURCE IN AIR

$4\pi R^2 \times$ Dose (Dose Units $\times$ cm^2/Source Neutron

R (meters)	Henderson Dose (Rads)	Snyder-Neufeld Dose (Rads)	Tissue Kerma (Ergs/g)	Mid-Phantom Dose (Rads)	Concrete Kerma (Ergs/g)	Air Kerma (Ergs/g)	Non-Ionizing Silicon Kerma (Ergs/g)	Ionizing Silicon Kerma (Ergs/g)	Snyder-Neufeld Dose (Rems)
0	1.599-09	2.463-09	1.772-07	8.906-10	2.863-08	4.131-08	2.105-09	1.140-08	1.842-08
25	1.944-09	3.096-09	2.211-07	8.561-10	3.272-08	4.154-08	2.500-09	1.122-08	2.306-08
50	2.172-09	3.863-09	2.496-07	1.046-09	3.527-08	4.431-08	2.751-09	1.093-08	2.712-08
75	2.314-09	4.538-09	2.667-07	1.208-09	3.667-08	4.768-08	2.912-09	1.057-08	3.025-08
100	2.390-09	5.034-09	2.754-07	1.300-09	3.720-08	5.061-08	3.010-09	1.015-08	3.239-08
150	2.406-09	5.472-09	2.760-07	1.322-09	3.648-08	5.371-08	3.069-09	9.195-09	3.406-08
200	2.306-09	5.362-09	2.630-07	1.235-09	3.432-08	5.304-08	2.993-09	8.186-09	3.326-08
250	2.142-09	4.952-09	2.429-07	1.109-09	3.143-08	4.967-08	2.824-09	7.186-09	3.104-08
300	1.946-09	4.419-09	2.198-07	9.744-10	2.824-08	4.484-08	2.598-09	6.240-09	2.815-08
400	1.536-09	3.340-09	1.723-07	7.251-10	2.190-08	3.415-08	2.080-09	4.587-09	2.197-08
500	1.163-09	2.443-09	1.299-07	5.252-10	1.638-08	2.474-08	1.584-09	3.288-09	1.646-08
600	8.554-10	1.758-09	9.535-08	3.744-10	1.195-08	1.750-08	1.168-09	2.315-09	1.202-08
900	3.076-10	6.147-10	3.415-08	1.280-10	4.217-09	5.845-09	4.202-10	7.533-10	4.276-09
1200	1.016-10	2.010-10	1.125-08	4.136-11	1.376-09	1.872-09	1.388-10	2.308-10	1.406-09
1500	3.214-11	6.307-11	3.549-09	1.288-11	4.313-10	5.824-10	4.391-11	6.845-11	4.434-10
1800	9.913-12	1.932-11	1.092-09	3.920-12	1.320-10	1.775-10	1.355-11	1.993-11	1.364-10
2100	3.007-12	5.830-12	3.306-10	1.175-12	3.983-11	5.333-11	4.110-12	5.737-12	4.132-11
2400	9.017-13	1.741-12	9.897-11	3.485-13	1.189-11	1.586-11	1.232-12	1.639-12	1.237-11
2700	2.679-13	5.156-13	2.938-11	1.026-13	3.520-12	4.678-12	3.662-13	4.660-13	3.674-12
3000	7.900-14	1.517-13	8.657-12	3.001-14	1.035-12	1.372-12	1.080-13	1.320-13	1.083-12
3300	2.315-14	4.441-14	2.536-12	8.733-15	3.027-13	4.000-13	3.164-14	3.728-14	3.176-13
3600	6.745-15	1.293-14	7.388-13	2.529-15	8.804-14	1.161-13	9.219-15	1.050-14	9.261-14
3800	2.956-15	5.669-15	3.238-13	1.104-15	3.855-14	5.080-14	4.041-15	4.510-15	4.062-14
4000	1.293-15	2.480-15	1.416-13	4.813-16	1.685-14	2.218-14	1.768-15	1.935-15	1.778-14
4200	5.646-16	1.083-15	6.183-14	2.094-16	7.349-15	9.669-15	7.720-16	8.292-16	7.770-15
4400	2.461-16	4.722-16	2.695-14	9.096-17	3.201-15	4.208-15	3.365-16	3.552-16	3.389-15
4600	1.071-16	2.055-16	1.172-14	3.945-17	1.392-15	1.829-15	1.464-16	1.520-16	1.476-15
4800	4.652-17	8.933-17	5.093-15	1.708-17	6.044-16	7.937-16	6.364-17	6.502-17	6.420-16
5000	2.018-17	3.877-17	2.209-15	7.387-18	2.621-16	3.440-16	2.761-17	2.779-17	2.788-16
5200	8.746-18	1.681-17	9.569-16	3.190-18	1.135-16	1.489-16	1.197-17	1.187-17	1.209-16
5400	3.785-18	7.276-18	4.139-16	1.376-18	4.911-17	6.437-17	5.179-18	5.065-18	5.237-17

Table XII. Dose distributions in air (1.11 g/liter) for a point isotropic thermonuclear weapon source.

Energy Bounds:	1.50+01 -- 1.22+01		1.22+01 -- 1.00+01		1.00+01 -- 8.18+00		8.18+00 -- 6.36+00		6.36+00 -- 4.96+00	
ϕ_o:	5.632866-05	K= 1	4.018120-04	K= 1	1.909878-03	K= 1	7.892348-03	K= 1	2.443914-02	K= 1
i	α_i	β_i	α_{ii}	β_i	α_i	β_i	α_i	β_i	α_i	β_i
1	-1.301621-07	.16264	1.116488-05	.94829	1.053493-05	.42043	-3.228244-04	.45324	2.129615-03	.82521
2	7.854030-06	1.05943	6.850950-05	1.10952	9.441679-04	1.26129	1.926972-02	1.35972	1.071927-02	1.26895
3	2.027454-05	1.31825	-1.037932-04	1.30596	2.444758-04	1.02363	1.050721-03	.97965	1.689522-02	1.62182
4	-1.050993-06	1.37792	2.586967-04	1.33423	-6.833424-04	1.30292	-1.886199-02	1.37996	-3.996762-10	3.15789
5	4.433568-17	3.17237	0.000000	1.00000	9.990979-04	1.48645	6.485643-03	1.57620	3.363860-08	1.67387

Energy Bounds:	4.96+00 -- 4.06+00		4.06+00 -- 3.01+00		3.01+00 -- 2.46+00		2.46+00 -- 2.35+00		2.35+00 -- 1.83+00	
ϕ_o :	5.326964-02	K= 1	1.011647-01	K= 1	1.619726-01	K= 1	1.940693-01	K= 1	2.288763-01	K= 1
i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i
1	1.451025-02	1.37579	6.886339-02	1.15776	8.185966-02	1.24201	-9.971654-01	1.38149	2.089944-03	.20953
2	3.677606-03	.54627	8.635397-03	.37745	2.559570-02	.35365	2.280368-02	.46106	-2.339329-03	.87151
3	2.099414-02	1.07596	3.079467-02	1.02081	3.761886-02	.78850	5.799926-02	1.04819	3.087076-01	1.35139
4	-8.712698-03	1.55873	5.230919-03	1.45802	2.018002-01	1.38045	1.356330+00	1.41214	2.229334-01	1.54015
5	3.969473-02	1.69582	1.963104-02	1.69513	4.356923-02	1.69638	1.494185-01	1.69961	7.960137-02	1.71955

Energy Bounds:	1.83+00 -- 1.11+00		1.11+00 -- 5.50-01		5.50-01 -- 1.11-01		1.11-01 -- 3.35-03		3.35-03 -- 5.83-04	
ϕ_o :	3.005381-01	K= 1	3.526530-01	K= 1	3.056668-01	K= 1	1.435268-01	K= 2		K= 2
i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i
1	-4.842837-02	.20276	-9.555317-01	.83947	6.132516-01	.22897	4.421572-01	.34862	-1.248161+02	.80573
2	1.707540-02	.84332	1.763049+00	1.25420	-6.245990+00	.95233	-1.013522+01	.96197	3.173032+02	1.17099
3	6.835397-01	1.30767	1.319876+00	1.48668	1.342816+00	1.47671	2.336034+01	1.09706	5.564592+01	1.45920
4	3.700337-01	1.60277	2.773831-01	1.70971	7.903556+00	1.31998	5.380840+00	1.39057	3.324073+00	1.68896
5	9.385645-03	1.85939	-2.444138-08	2.78810	1.268266+00	1.67040	4.407096-01	1.64729	0.000000	1.00000

Energy Bounds:	5.83-04 -- 1.01-04		1.01-04 -- 2.90-05		2.90-05 -- 1.07-05		1.07-05 -- 3.06-06		3.06-06 -- 1.12-06	
		K= 2		K= 2		K= 2		K= 2		K= 2
i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i
1	1.107663+02	.19230	3.069106+03	.01423	2.467084+03	.19846	1.057624+04	.56679	2.687246+04	.20320
2	-1.037514+03	.79982	-6.520091+03	.88221	-1.626781+04	.82546	-6.558132+04	.85048	-1.423188+05	.84516
3	1.568298+03	1.24023	9.554401+03	1.13671	1.233288+04	1.27998	8.046995+04	1.15815	1.580482+03	1.31054
4	5.537764+02	1.00098	2.122711+03	1.41058	1.682138+04	1.15628	1.649077+04	1.42466	2.251181+05	1.21904
5	1.336390+02	1.57423	1.695596+02	1.65449	1.842126+03	1.57699	1.352962+03	1.65899	1.775146+04	1.56507

Energy Bounds:	1.12-06 -- 4.14-07		- T H E R M A L -	
		K= 2		K= 2
i	α_i	β_i	α_i	β_i
1	5.566967+04	.53052	1.939325+05	.20802
2	-1.126142+06	.96104	-8.555676+05	.86519
3	1.142417+06	1.06497	-1.087634+06	1.34159
4	1.731641+05	1.37674	2.244728+06	1.27665
5	1.229319+04	1.64245	1.886992+05	1.52995

Table XIII. Coefficients for representing the multigroup neutron spectra from a point isotropic fission source in air (1.11 g/liter). See Equation 93.

Energy Bounds:	1.50+01 -- 1.22+01		1.22+01 -- 1.00+01		1.00+01 -- 8.18+00		8.18+00 -- 6.36+00		6.36+00 -- 4.96+00	
ϕ_o:	3.571441-01	K= 1		K= 1		K= 1		K= 1		K= 1
i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i
1	-9.290789-03	.39675	4.126313-01	1.27011	3.446277-03	.39458	1.482274-04	.24285	2.259172-03	.53113
2	-7.230462-01	1.19025	8.645080-03	.29060	1.197162-02	1.28541	-5.892084-04	1.01009	-8.460720-01	1.59338
3	-2.015579-02	.22240	9.025999-03	.78844	-8.132236-02	1.36928	5.731073-01	1.56627	-1.234914-02	1.19401
4	6.742383-01	1.17148	-1.152540+00	1.30709	1.350053-01	1.49012	-1.225978-02	1.25898	9.897384-01	1.61535
5	2.417175-01	1.31021	8.507338-01	1.34109	1.003330-06	1.72744	-4.864726-01	1.54348	1.620361-07	1.71573

Energy Bounds:	4.96+00 -- 4.06+00		4.06+00 -- 3.01+00		3.01+00 -- 2.46+00		2.46+00 -- 2.35+00		2.35+00 -- 1.83+00	
		K= 1		K= 1		K= 1		K= 1		K= 1
i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i
1	-2.503868-03	.54704	-2.874851-03	.51240	-2.230654+00	1.63712	-2.778904-02	.56603	8.896306-01	1.75204
2	-6.524273-01	1.64112	-5.770746-01	1.53720	-2.658972-02	.63603	-4.140408-01	1.29345	-1.954077-02	.54798
3	-6.093340-02	1.26884	-3.186732-02	.92236	-2.510837-01	1.24598	8.639432-01	1.72201	-4.428944-01	1.26633
4	7.162002-02	1.33882	5.518757-01	1.50121	7.018354-01	1.53443	-4.083526-07	2.76049	2.456548-01	1.57690
5	8.340110-01	1.68345	2.657483-01	1.69004	2.138624+00	1.66509	1.088691-08	1.81020	-2.773128-01	1.78586

Energy Bounds:	1.83+00 -- 1.11+00		1.11+00 -- 5.50-01		5.50-01 -- 1.11-01		1.11-01 -- 3.35-03		3.35-03 -- 5.83-04	
		K= 1		K= 1		K= 1		K= 2		K= 2
i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i
1	-3.684829-02	.27655	-7.554149-02	.29430	6.240947-01	.68370	5.626931-01	.51674	-3.214752+02	.14196
2	-1.443094+00	1.15024	-2.182630+00	1.22407	4.372236-02	2.05110	8.038474+00	1.55022	6.435429+02	.65617
3	9.476437-02	1.78359	2.722457-02	1.89809	-7.278342+00	1.35607	-3.608272+00	1.01859	4.103831+01	1.26211
4	8.718076-01	1.09492	4.442757-01	.91685	7.824733+00	1.67135	2.065434-01	1.84072	1.060087+02	1.60250
5	9.685137-01	1.69798	2.507746+00	1.69979	0.000000	1.00000	0.000000	1.00000	-6.011826+02	.76446

Energy Bounds:	5.83-04 -- 1.01-04		1.01-04 -- 2.90-05		2.90-05 -- 1.07-05		1.07-05 -- 3.06-06		3.06-06 -- 1.12-06	
		K= 2		K= 2		K= 2		K= 2		K= 2
i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i
1	-1.773051+02	.14404	9.830564+01	.25489	4.343020+03	1.65652	1.371167+04	.91381	-8.018242+03	.56008
2	1.850012+02	.66579	3.009174+03	1.06017	1.995323+03	.82162	-3.903670+04	1.11368	2.236982+04	1.68025
3	1.846012+03	1.28061	1.719958+03	1.64393	-1.256467+04	1.16657	2.734534+04	1.45311	5.091614+04	.83639
4	4.625960+02	1.62599	-8.556520+03	1.17202	1.096577+04	1.42219	1.083221+04	1.66371	-1.047537+05	1.07425
5	-2.065347+03	1.19916	5.370048+03	1.35064	0.000000	1.00000	0.000000	1.00000	7.381961+04	1.49428

Energy Bounds:	1.12-06 -- 4.14-07		- T H E R M A L -	
		K= 2		K= 2
i	α_i	β_i	α_i	β_i
1	9.100818+04	1.69006	-4.147013+04	.26429
2	1.211004+05	.93040	-5.193724+05	1.09927
3	-2.820695+05	1.10420	8.177338+04	1.70456
4	1.799297+05	1.47118	2.005019+05	.78791
5	-3.266167+04	1.71194	4.417475+05	1.51774

Table XIV. Coefficients for representing the multigroup neutron spectra from a 14 MeV point isotropic source in air (1.11 g/liter). See Equation 93.

Energy Bounds:	1.50+01 -- 1.22+01		1.22+01 -- 1.00+01		1.00+01 -- 8.18+00		8.18+00 -- 6.36+00		6.36+00 -- 4.96+00	
ϕ_o :	2.520000-02	K= 1	1.160000-02	K= 1	7.740000-03	K= 1	8.080000-03	K= 1	1.290000-02	K= 1
i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i
1	-1.819502-02	1.17570	1.680330-04	.41098	-6.752077+00	1.33065	-1.723717-02	1.46258	3.511314-04	.50609
2	-7.605984-04	.29349	2.661907-02	1.23294	7.310342-04	.41317	6.700524-04	.69812	-2.830868-02	1.51827
3	1.812518-02	1.13817	8.722773-04	.88114	1.841819-03	.99683	6.026110-03	1.30139	7.852087-03	1.30393
4	1.391454-02	1.31890	-4.249716-02	1.27420	6.747640+00	1.33058	3.169744-02	1.58818	5.499784-02	1.63924
5	-1.355855-05	1.50748	3.001241-02	1.35373	1.871823-02	1.48943	1.135242-09	1.70155	0.000000	1.00000

Energy Bounds:	4.96+00 -- 4.06+00		4.06+00 -- 3.01+00		3.01+00 -- 2.46+00		2.46+00 -- 2.35+00		2.35+00 -- 1.83+00	
ϕ_o :	1.890000-02	K= 1	2.470000-02	K= 1	3.450000-02	K= 1	4.540000-02	K= 1	5.710000-02	K= 1
i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i
1	1.290782-03	.51611	3.125045-02	1.39452	6.913039-03	.22464	7.537701-01	1.56521	-4.276088-02	1.19856
2	2.279155-02	1.54832	3.430966-03	.16447	4.693604-03	.93433	-2.002625-03	.12616	8.364956-02	1.37536
3	8.435711-03	1.11848	1.797859-02	1.10505	6.452289-02	1.44880	-5.537297-02	1.33373	1.569961-01	1.71614
4	-2.362665-01	1.64267	-2.898112-02	1.52444	8.706519-02	1.75374	-8.347072-01	1.59310	0.000000	1.00000
5	2.584149-01	1.67339	4.558837-02	1.69547	-2.250457-02	1.80657	3.259750-01	1.69616	0.000000	1.00000

Energy Bounds:	1.83+00 -- 1.11+00		1.11+00 -- 5.50-01		5.50-01 -- 1.11-01		1.11-01 -- 3.35-03		3.35-03 -- 5.83-04	
ϕ_o :	8.270000-02	K= 1	1.520000-01	K= 1	2.320000-01	K= 1	3.300000+00	K= 2	4.400000+01	K= 2
i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i
1	-4.260591-03	.48581	3.202569-01	1.44818	2.481250-02	.22659	1.633216+01	.39894	6.469371+00	.86052
2	1.097630-01	1.45742	-1.688541-01	.77916	-9.357995-01	.94244	3.160942+00	.76376	3.474972+02	.53503
3	3.817417-02	1.03128	2.467395-01	1.08930	7.925017-01	1.46137	4.449308+00	1.12828	8.152529+01	1.10653
4	1.909968-01	1.71857	4.449058-01	1.71694	1.168446+00	1.17832	1.780121+00	1.43891	3.341817+01	1.42203
5	-2.540856-10	3.97973	0.000000	1.00000	1.105195+00	1.71604	5.624430-01	1.66411	1.004797+01	1.66089

Energy Bounds:	5.83-04 -- 1.01-04		1.01-04 -- 2.90-05		2.90-05 -- 1.07-05		1.07-05 -- 3.06-06		3.06-06 -- 1.12-06	
ϕ_o :	4.980000+01	K= 2	2.780000+01	K= 2		K= 2		K= 2		K= 2
i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i	α_i	β_i
1	-2.201101+02	.85898	-2.534614+03	.31528	-1.626527+04	.30212	-8.325080+04	.46178	-1.916836+05	.14711
2	2.179980+03	.60208	1.026585+04	.61345	8.176312+03	.90636	1.333027+05	.61197	1.604321+05	.61185
3	5.463406+02	1.10595	2.187380+03	1.16433	2.979218+04	.56861	1.918773+04	1.17149	9.059385+04	.94876
4	1.790347+02	1.43773	6.705701+02	1.47317	5.218138+03	1.31478	5.796845+03	1.47429	3.363941+04	1.39998
5	5.268318+01	1.66505	2.133694+02	1.67153	1.134977+03	1.64093	1.880950+03	1.67122	5.238106+03	1.68132

Energy Bounds:	1.12-06 -- 4.14-07		- T H E R M A L -	
		K= 2		K= 2
i	α_i	β_i	α_i	β_i
1	-4.502714+05	.14978	-1.091711+06	.15368
2	3.304314+05	.62298	6.729935+05	.63920
3	2.251973+05	.96601	6.114461+05	.99117
4	7.199143+04	1.43803	1.586183+05	1.49708
5	7.529495+03	1.71417	3.828477+03	1.83242

Table XV. Coefficients for representing the multigroup neutron spectra from a point isotropic thermonuclear weapon source in air (1.11 g/liter). See Equation 93.

FISSION SOURCE IN AIR

$4\pi R^2$ × Flux (Neutrons/MeV per Source Neutron

Energy Bounds	R(meters): 0	25	50	75	100	150	200	250
1.50+01 -- 1.22+01	5.633-05	5.022-05	4.480-05	3.995-05	3.561-05	2.825-05	2.237-05	1.768-05
1.22+01 -- 1.00+01	4.018-04	3.635-04	3.285-04	2.964-04	2.672-04	2.164-04	1.746-04	1.404-04
1.00+01 -- 8.18+00	1.910-03	1.771-03	1.637-03	1.508-03	1.386-03	1.165-03	9.736-04	8.096-04
8.18+00 -- 6.36+00	7.892-03	7.385-03	6.929-03	6.502-03	6.094-03	5.320-03	4.604-03	3.953-03
6.36+00 -- 4.96+00	2.444-02	2.376-02	2.288-02	2.187-02	2.077-02	1.848-02	1.620-02	1.405-02
4.96+00 -- 4.06+00	5.327-02	5.286-02	5.146-02	4.944-02	4.705-02	4.179-02	3.650-02	3.152-02
4.06+00 -- 3.01+00	1.012-01	1.032-01	1.012-01	9.701-02	9.162-02	7.948-02	6.735-02	5.618-02
3.01+00 -- 2.46+00	1.620-01	1.881-01	1.996-01	2.024-01	2.000-01	1.867-01	1.682-01	1.480-01
2.46+00 -- 2.35+00	1.941-01	2.299-01	2.523-01	2.649-01	2.706-01	2.677-01	2.534-01	2.329-01
2.35+00 -- 1.83+00	2.289-01	2.591-01	2.786-01	2.899-01	2.949-01	2.910-01	2.746-01	2.515-01
1.83+00 -- 1.11+00	3.005-01	3.584-01	4.141-01	4.531-01	4.754-01	4.833-01	4.607-01	4.225-01
1.11+00 -- 5.50-01	3.527-01	5.220-01	6.667-01	7.847-01	8.762-01	9.864-01	1.017+00	9.907-01
5.50-01 -- 1.11-01	3.057-01	7.152-01	9.269-01	1.111+00	1.297+00	1.639+00	1.875+00	1.985+00
1.11-01 -- 3.35-03	1.435-01	5.394-01	1.477+00	2.666+00	3.941+00	6.378+00	8.336+00	9.653+00
3.35-03 -- 5.83-04	0.000	2.400+00	9.542+00	2.075+01	3.492+01	6.746+01	9.901+01	1.244+02
5.83-04 -- 1.01-04	0.000	4.301+01	8.233+01	1.130+02	1.551+02	2.877+02	4.498+02	5.975+02
1.01-04 -- 2.90-05	0.000	1.796+01	9.840+01	2.632+02	5.064+02	1.155+03	1.877+03	2.529+03
2.90-05 -- 1.07-05	0.000	8.114+02	1.349+03	1.488+03	1.701+03	2.952+03	4.936+03	6.967+03
1.07-05 -- 3.06-06	0.000	3.316+02	1.212+03	2.557+03	4.318+03	8.820+03	1.399+04	1.898+04
3.06-06 -- 1.12-06	0.000	8.266+03	1.303+04	1.255+04	1.186+04	1.764+04	3.115+04	4.657+04
1.12-06 -- 4.14-07	0.000	2.239+03	7.496+03	1.487+04	2.421+04	4.828+04	7.721+04	1.067+05
- T H E R M A L -	0.000	5.720+04	8.866+04	7.881+04	6.247+04	7.116+04	1.325+05	2.129+05

Energy Bounds	R(meters): 300	400	500	600	900	1200	1500	1800
1.50+01 -- 1.22+01	1.396-05	8.666-06	5.353-06	3.291-06	7.475-07	1.649-07	3.551-08	7.502-09
1.22+01 -- 1.00+01	1.126-04	7.188-05	4.550-05	2.860-05	6.870-06	1.590-06	3.578-07	7.884-08
1.00+01 -- 8.18+00	6.704-04	4.552-04	3.057-04	2.035-04	5.780-05	1.575-05	4.171-06	1.081-06
8.18+00 -- 6.36+00	3.371-03	2.413-03	1.700-03	1.184-03	3.809-04	1.163-04	3.430-05	9.852-06
6.36+00 -- 4.96+00	1.208-02	8.769-03	6.242-03	4.381-03	1.429-03	4.403-04	1.310-04	3.809-05
4.96+00 -- 4.06+00	2.702-02	1.955-02	1.394-02	9.844-03	3.339-03	1.093-03	3.490-04	1.092-04
4.06+00 -- 3.01+00	4.632-02	3.074-02	1.997-02	1.282-02	3.323-03	8.802-04	2.425-04	6.896-05
3.01+00 -- 2.46+00	1.281-01	9.256-02	6.473-02	4.427-02	1.307-02	3.615-03	9.718-04	2.588-04
2.46+00 -- 2.35+00	2.097-01	1.628-01	1.213-01	8.787-02	3.008-02	9.403-03	2.796-03	8.077-04
2.35+00 -- 1.83+00	2.253-01	1.726-01	1.267-01	9.036-02	2.952-02	8.844-03	2.539-03	7.131-04
1.83+00 -- 1.11+00	3.775-01	2.859-01	2.068-01	1.452-01	4.533-02	1.307-02	3.640-03	9.979-04
1.11+00 -- 5.50-01	9.272-01	7.489-01	5.632-01	4.046-01	1.286-01	3.662-02	9.971-03	2.669-03
5.50-01 -- 1.11-01	1.984+00	1.764+00	1.417+00	1.066+00	3.655-01	1.063-01	2.884-02	7.602-03
1.11-01 -- 3.35-03	1.033+01	1.014+01	8.736+00	6.918+00	2.583+00	7.743-01	2.110-01	5.528-02
3.35-03 -- 5.83-04	1.414+02	1.508+02	1.369+02	1.124+02	4.432+01	1.357+01	3.724+00	9.753-01
5.83-04 -- 1.01-04	7.073+02	7.943+02	7.429+02	6.211+02	2.520+02	7.814+01	2.157+01	5.655+00
1.01-04 -- 2.90-05	3.027+03	3.472+03	3.309+03	2.811+03	1.176+03	3.690+02	1.020+02	2.673+01
2.90-05 -- 1.07-05	8.622+03	1.027+04	9.976+03	8.566+03	3.639+03	1.151+03	3.200+02	8.392+01
1.07-05 -- 3.06-06	2.313+04	2.761+04	2.721+04	2.371+04	1.034+04	3.296+03	9.170+02	2.406+02
3.06-06 -- 1.12-06	6.001+04	7.516+04	7.528+04	6.603+04	2.914+04	9.387+03	2.629+03	6.901+02
1.12-06 -- 4.14-07	1.326+05	1.638+05	1.658+05	1.475+05	6.684+04	2.168+04	6.070+03	1.593+03
- T H E R M A L -	2.879+05	3.814+05	3.948+05	3.541+05	1.628+05	5.354+04	1.514+04	3.983+03

Table XVI. Multigroup spectra in air for a point isotropic fission source. 1.11 g/1.

FISSION SOURCE IN AIR

$4\pi R^2$ × Flux (Neutrons/MeV per Source Neutron)

Energy Bounds	R(meters): 2100	2400	2700	3000	3300	3600	3800	4000
1.50+01 -- 1.22+01	1.559-09	3.195-10	6.470-11	1.297-11	2.576-12	5.076-13	1.712-13	5.759-14
1.22+01 -- 1.00+01	1.708-08	3.647-09	7.697-10	1.608-10	3.331-11	6.848-12	2.376-12	8.225-13
1.00+01 -- 8.18+00	2.751-07	6.907-08	1.714-08	4.212-09	1.026-09	2.482-10	9.597-11	3.702-11
8.18+00 -- 6.36+00	2.774-06	7.687-07	2.102-07	5.684-08	1.522-08	4.042-09	1.663-09	6.823-10
6.36+00 -- 4.96+00	1.089-05	3.074-06	8.587-07	2.378-07	6.535-08	1.784-08	7.485-09	3.133-09
4.96+00 -- 4.06+00	3.362-05	1.020-05	3.060-06	9.083-07	2.673-07	7.806-08	3.424-08	1.497-08
4.06+00 -- 3.01+00	1.996-05	5.822-06	1.699-06	4.948-07	1.434-07	4.139-08	1.802-08	7.834-09
3.01+00 -- 2.46+00	6.903-05	1.853-05	5.020-06	1.372-06	3.781-07	1.048-07	4.470-08	1.909-08
2.46+00 -- 2.35+00	2.293-04	6.442-05	1.800-05	5.019-06	1.398-06	3.897-07	1.663-07	7.102-08
2.35+00 -- 1.83+00	1.980-04	5.470-05	1.509-05	4.162-06	1.149-06	3.180-07	1.352-07	5.750-08
1.83+00 -- 1.11+00	2.718-04	7.392-05	2.011-05	5.482-06	1.497-06	4.100-07	1.732-07	7.328-08
1.11+00 -- 5.50-01	7.110-04	1.896-04	5.071-05	1.363-05	3.681-06	9.992-07	4.200-07	1.769-07
5.50-01 -- 1.11-01	1.987-03	5.203-04	1.371-04	3.640-05	9.739-06	2.622-06	1.096-06	4.594-07
1.11-01 -- 3.35-03	1.432-02	3.712-03	9.682-04	2.544-04	6.738-05	1.797-05	7.476-06	3.119-06
3.35-03 -- 5.83-04	2.515-01	6.482-02	1.682-02	4.402-03	1.162-03	3.090-04	1.283-04	5.339-05
5.83-04 -- 1.01-04	1.454+00	3.731-01	9.645-02	2.521-02	6.664-03	1.777-03	7.390-04	3.079-04
1.01-04 -- 2.90-05	6.873+00	1.765+00	4.561-01	1.188-01	3.125-02	8.283-03	3.432-03	1.427-03
2.90-05 -- 1.07-05	2.153+01	5.510+00	1.420+00	3.703-01	9.759-02	2.596-02	1.078-02	4.483-03
1.07-05 -- 3.06-06	6.178+01	1.583+01	4.077+00	1.060+00	2.780-01	7.358-02	3.047-02	1.265-02
3.06-06 -- 1.12-06	1.767+02	4.508+01	1.159+01	3.018+00	7.948-01	2.113-01	8.763-02	3.641-02
1.12-06 -- 4.14-07	4.088+02	1.046+02	2.689+01	6.975+00	1.825+00	4.819-01	1.993-01	8.268-02
- T H E R M A L -	1.018+03	2.588+02	6.635+01	1.725+01	4.541+00	1.207+00	5.007-01	2.079-01

Energy Bounds	R(meters): 4200	4400	4600	4800	5000	5200	5400
1.50+01 -- 1.22+01	1.931-14	6.461-15	2.156-15	7.178-16	2.384-16	7.903-17	2.614-17
1.22+01 -- 1.00+01	2.839-13	9.779-14	3.361-14	1.153-14	3.947-15	1.349-15	4.602-16
1.00+01 -- 8.18+00	1.425-11	5.470-12	2.096-12	8.015-13	3.060-13	1.166-13	4.437-14
8.18+00 -- 6.36+00	2.791-10	1.139-10	4.636-11	1.883-11	7.632-12	3.088-12	1.247-12
6.36+00 -- 4.96+00	1.308-09	5.451-10	2.267-10	9.412-11	3.901-11	1.614-11	6.666-12
4.96+00 -- 4.06+00	6.532-09	2.843-09	1.235-09	5.354-10	2.317-10	1.001-10	4.316-11
4.06+00 -- 3.01+00	3.399-09	1.472-09	6.363-10	2.746-10	1.184-10	5.094-11	2.189-11
3.01+00 -- 2.46+00	8.163-09	3.493-09	1.496-09	6.407-10	2.744-10	1.175-10	5.033-11
2.46+00 -- 2.35+00	3.034-08	1.296-08	5.540-09	2.368-09	1.012-09	4.328-10	1.850-10
2.35+00 -- 1.83+00	2.448-08	1.043-08	4.446-09	1.897-09	8.096-10	3.457-10	1.477-10
1.83+00 -- 1.11+00	3.106-08	1.319-08	5.611-09	2.393-09	1.023-09	4.385-10	1.885-10
1.11+00 -- 5.50-01	7.461-08	3.153-08	1.334-08	5.654-09	2.399-09	1.019-09	4.330-10
5.50-01 -- 1.11-01	1.927-07	8.095-08	3.403-08	1.431-08	6.022-09	2.534-09	1.066-09
1.11-01 -- 3.35-03	1.305-06	5.472-07	2.301-07	9.692-08	4.090-08	1.729-08	7.319-09
3.35-03 -- 5.83-04	2.228-05	9.320-06	3.908-06	1.643-06	6.920-07	2.922-07	1.236-07
5.83-04 -- 1.01-04	1.284-04	5.356-05	2.234-05	9.311-06	3.877-06	1.613-06	6.697-07
1.01-04 -- 2.90-05	5.950-04	2.487-04	1.043-04	4.380-05	1.844-05	7.778-06	3.286-06
2.90-05 -- 1.07-05	1.867-03	7.780-04	3.242-04	1.351-04	5.622-05	2.338-05	9.710-06
1.07-05 -- 3.06-06	5.272-03	2.202-03	9.222-04	3.871-04	1.628-04	6.863-05	2.897-05
3.06-06 -- 1.12-06	1.514-02	6.296-03	2.617-03	1.087-03	4.506-04	1.866-04	7.714-05
1.12-06 -- 4.14-07	3.442-02	1.437-02	6.019-03	2.527-03	1.063-03	4.482-04	1.892-04
- T H E R M A L -	8.627-02	3.577-02	1.481-02	6.116-03	2.520-03	1.035-03	4.242-04

Table XVI. - continued.

14 MeV SOURCE IN AIR

$4\pi R^2 \times$ Flux (Neutrons/MeV per Source Neutron)

Energy Bounds	R(meters): 0	25	50	75	100	150	200	250
1.50+01 -- 1.22+01	3.571-01	3.108-01	2.784-01	2.514-01	2.271-01	1.845-01	1.489-01	1.196-01
1.22+01 -- 1.00+01	0.000	1.730-02	2.802-02	3.491-02	3.934-02	4.346-02	4.356-02	4.131-02
1.00+01 -- 8.18+00	0.000	7.768-03	1.332-02	1.731-02	2.016-02	2.344-02	2.451-02	2.417-02
8.18+00 -- 6.36+00	0.000	6.893-03	1.245-02	1.687-02	2.034-02	2.492-02	2.709-02	2.757-02
6.36+00 -- 4.96+00	0.000	1.264-02	2.268-02	3.057-02	3.667-02	4.463-02	4.836-02	4.916-02
4.96+00 -- 4.06+00	0.000	1.688-02	3.084-02	4.217-02	5.116-02	6.330-02	6.936-02	7.108-02
4.06+00 -- 3.01+00	0.000	1.883-02	3.455-02	4.736-02	5.750-02	7.094-02	7.719-02	7.831-02
3.01+00 -- 2.46+00	0.000	2.480-02	4.769-02	6.798-02	8.535-02	1.112-01	1.264-01	1.329-01
2.46+00 -- 2.35+00	0.000	2.572-02	5.178-02	7.648-02	9.883-02	1.348-01	1.589-01	1.723-01
2.35+00 -- 1.83+00	0.000	2.386-02	4.809-02	7.113-02	9.208-02	1.260-01	1.490-01	1.620-01
1.83+00 -- 1.11+00	0.000	1.788-02	4.693-02	7.718-02	1.049-01	1.494-01	1.795-01	1.973-01
1.11+00 -- 5.50-01	0.000	1.180-02	4.934-02	9.424-02	1.389-01	2.176-01	2.784-01	3.213-01
5.50-01 -- 1.11-01	0.000	4.184-02	8.044-02	1.210-01	1.652-01	2.634-01	3.652-01	4.583-01
1.11-01 -- 3.35-03	0.000	5.554-02	1.800-01	3.393-01	5.202-01	9.289-01	1.378+00	1.833+00
3.35-03 -- 5.83-04	0.000	-1.361+02	-1.231+02	-3.789+01	2.731+01	7.356+01	6.997+01	5.618+01
5.83-04 -- 1.01-04	0.000	-7.400+01	-6.600+01	-1.396+01	3.143+01	8.392+01	1.141+02	1.393+02
1.01-04 -- 2.90-05	0.000	2.762+01	6.168+01	8.821+01	1.161+02	2.015+02	3.296+02	4.808+02
2.90-05 -- 1.07-05	0.000	2.288+01	8.442+01	1.782+02	3.004+02	6.186+02	1.011+03	1.439+03
1.07-05 -- 3.06-06	0.000	5.776+01	2.157+02	4.595+02	7.808+02	1.625+03	2.674+03	3.831+03
3.06-06 -- 1.12-06	0.000	-4.069+00	1.987+02	7.420+02	1.622+03	4.149+03	7.254+03	1.052+04
1.12-06 -- 4.14-07	0.000	3.787+02	1.362+03	2.813+03	4.670+03	9.457+03	1.541+04	2.208+04
- T H E R M A L -	0.000	-7.180+03	-1.073+04	-5.658+03	4.186+03	2.571+04	4.322+04	5.756+04

Energy Bounds	R(meters): 300	400	500	600	900	1200	1500	1800
1.50+01 -- 1.22+01	9.585-02	6.108-02	3.863-02	2.427-02	5.830-03	1.347-03	3.019-04	6.607-05
1.22+01 -- 1.00+01	3.778-02	2.947-02	2.166-02	1.533-02	4.754-03	1.314-03	3.393-04	8.375-05
1.00+01 -- 8.18+00	2.294-02	1.917-02	1.501-02	1.126-02	4.083-03	1.298-03	3.825-04	1.073-04
8.18+00 -- 6.36+00	2.689-02	2.358-02	1.929-02	1.508-02	6.128-03	2.166-03	7.072-04	2.192-04
6.36+00 -- 4.96+00	4.796-02	4.221-02	3.473-02	2.735-02	1.142-02	4.168-03	1.409-03	4.535-04
4.96+00 -- 4.06+00	6.979-02	6.207-02	5.156-02	4.101-02	1.772-02	6.731-03	2.378-03	8.019-04
4.06+00 -- 3.01+00	7.595-02	6.562-02	5.278-02	4.059-02	1.589-02	5.529-03	1.816-03	5.768-04
3.01+00 -- 2.46+00	1.329-01	1.207-01	1.011-01	8.044-02	3.384-02	1.227-02	4.114-03	1.316-03
2.46+00 -- 2.35+00	1.770-01	1.687-01	1.474-01	1.217-01	5.613-02	2.189-02	7.785-03	2.614-03
2.35+00 -- 1.83+00	1.669-01	1.594-01	1.393-01	1.149-01	5.253-02	2.023-02	7.099-03	2.354-03
1.83+00 -- 1.11+00	2.051-01	1.993-01	1.765-01	1.472-01	6.855-02	2.656-02	9.323-03	3.084-03
1.11+00 -- 5.50-01	3.479-01	3.606-01	3.355-01	2.906-01	1.456-01	5.868-02	2.105-02	7.049-03
5.50-01 -- 1.11-01	5.339-01	6.192-01	6.227-01	5.703-01	3.160-01	1.344-01	4.988-02	1.706-02
1.11-01 -- 3.35-03	2.251+00	2.858+00	3.087+00	2.987+00	1.836+00	8.245-01	3.154-01	1.099-01
3.35-03 -- 5.83-04	4.438+01	3.485+01	3.654+01	3.912+01	2.913+01	1.353+01	5.173+00	1.803+00
5.83-04 -- 1.01-04	1.632+02	2.038+02	2.271+02	2.292+02	1.559+02	7.390+01	2.900+01	1.022+01
1.01-04 -- 2.90-05	6.329+02	8.815+02	1.014+03	1.030+03	7.035+02	3.366+02	1.335+02	4.739+01
2.90-05 -- 1.07-05	1.866+03	2.579+03	2.984+03	3.060+03	2.132+03	1.029+03	4.101+02	1.460+02
1.07-05 -- 3.06-06	4.989+03	6.953+03	8.102+03	8.364+03	5.916+03	2.883+03	1.155+03	4.124+02
3.06-06 -- 1.12-06	1.366+04	1.885+04	2.193+04	2.273+04	1.637+04	8.064+03	3.244+03	1.162+03
1.12-06 -- 4.14-07	2.890+04	4.090+04	4.842+04	5.066+04	3.686+04	1.826+04	7.385+03	2.653+03
- T H E R M A L -	7.062+04	9.428+04	1.110+05	1.174+05	8.798+04	4.413+04	1.794+04	6.469+03

Table XVII. Multigroup spectra in air for a 14 MeV point isotropic source. 1.11 g/l.

14 MeV SOURCE IN AIR

$4\pi R^2$ x Flux (Neutrons/MeV per Source Neutron)

Energy Bounds	R(meters): 2100	2400	2700	3000	3300	3600	3800	4000
1.50+01 -- 1.22+01	1.417-05	2.990-06	6.219-07	1.277-07	2.595-08	5.221-09	1.784-09	6.075-10
1.22+01 -- 1.00+01	2.000-05	4.654-06	1.061-06	2.378-07	5.254-08	1.146-08	4.130-09	1.481-09
1.00+01 -- 8.18+00	2.903-05	7.649-06	1.974-06	5.010-07	1.254-07	3.104-08	1.217-08	4.752-09
8.18+00 -- 6.36+00	6.551-05	1.905-05	5.423-06	1.518-06	4.191-07	1.143-07	4.784-08	1.993-08
6.36+00 -- 4.96+00	1.409-04	4.267-05	1.266-05	3.695-06	1.064-06	3.030-07	1.305-07	5.594-08
4.96+00 -- 4.06+00	2.616-04	8.324-05	2.598-05	7.980-06	2.419-06	7.255-07	3.233-07	1.435-07
4.06+00 -- 3.01+00	1.792-04	5.478-05	1.654-05	4.943-06	1.464-06	4.305-07	1.896-07	8.327-08
3.01+00 -- 2.46+00	4.087-04	1.244-04	3.732-05	1.108-05	3.263-06	9.543-07	4.192-07	1.837-07
2.46+00 -- 2.35+00	8.450-04	2.659-04	8.203-05	2.493-05	7.488-06	2.228-06	9.888-07	4.375-07
2.35+00 -- 1.83+00	7.523-04	2.343-04	7.166-05	2.161-05	6.449-06	1.908-06	8.438-07	3.721-07
1.83+00 -- 1.11+00	9.822-04	3.047-04	9.281-05	2.788-05	8.287-06	2.443-06	1.078-06	4.743-07
1.11+00 -- 5.50-01	2.259-03	7.030-04	2.143-04	6.433-05	1.910-05	5.621-06	2.477-06	1.089-06
5.50-01 -- 1.11-01	5.537-03	1.734-03	5.293-04	1.588-04	4.702-05	1.381-05	6.084-06	2.676-06
1.11-01 -- 3.35-03	3.606-02	1.137-02	3.483-03	1.046-03	3.100-04	9.105-05	4.011-05	1.764-05
3.35-03 -- 5.83-04	5.958-01	1.898-01	5.880-02	1.782-02	5.304-03	1.555-03	6.811-04	2.968-04
5.83-04 -- 1.01-04	3.377+00	1.071+00	3.301-01	9.984-02	2.976-02	8.771-03	3.863-03	1.695-03
1.01-04 -- 2.90-05	1.570+01	4.978+00	1.533+00	4.629-01	1.379-01	4.069-02	1.795-02	7.895-03
2.90-05 -- 1.07-05	4.849+01	1.540+01	4.745+00	1.433+00	4.269-01	1.258-01	5.550-02	2.440-02
1.07-05 -- 3.06-06	1.372+02	4.362+01	1.346+01	4.066+00	1.211+00	3.570-01	1.574-01	6.921-02
3.06-06 -- 1.12-06	3.872+02	1.233+02	3.807+01	1.151+01	3.429+00	1.010+00	4.453-01	1.957-01
1.12-06 -- 4.14-07	8.861+02	2.823+02	8.719+01	2.636+01	7.854+00	2.315+00	1.021+00	4.489-01
- T H E R M A L -	2.167+03	6.920+02	2.140+02	6.474+01	1.928+01	5.679+00	2.502+00	1.100+00

Energy Bounds	R(meters): 4200	4400	4600	4800	5000	5200	5400
1.50+01 -- 1.22+01	2.062-10	6.978-11	2.355-11	7.926-12	2.662-12	8.918-13	2.982-13
1.22+01 -- 1.00+01	5.291-10	1.883-10	6.678-11	2.361-11	8.321-12	2.925-12	1.025-12
1.00+01 -- 8.18+00	1.850-09	7.176-10	2.776-10	1.071-10	4.123-11	1.584-11	6.071-12
8.18+00 -- 6.36+00	8.276-09	3.425-09	1.413-09	5.813-10	2.386-10	9.767-11	3.990-11
6.36+00 -- 4.96+00	2.390-08	1.018-08	4.324-09	1.831-09	7.736-10	3.260-10	1.371-10
4.96+00 -- 4.06+00	6.347-08	2.798-08	1.230-08	5.393-09	2.358-09	1.029-09	4.478-10
4.06+00 -- 3.01+00	3.647-08	1.594-08	6.947-09	3.022-09	1.312-09	5.681-10	2.457-10
3.01+00 -- 2.46+00	8.032-08	3.505-08	1.527-08	6.637-09	2.881-09	1.248-09	5.400-10
2.46+00 -- 2.35+00	1.931-07	8.498-08	3.732-08	1.636-08	7.158-09	3.126-09	1.363-09
2.35+00 -- 1.83+00	1.637-07	7.186-08	3.148-08	1.376-08	6.005-09	2.616-09	1.138-09
1.83+00 -- 1.11+00	2.083-07	9.126-08	3.992-08	1.743-08	7.601-09	3.310-09	1.440-09
1.11+00 -- 5.50-01	4.777-07	2.092-07	9.143-08	3.991-08	1.740-08	7.582-09	3.300-09
5.50-01 -- 1.11-01	1.177-06	5.174-07	2.277-07	1.003-07	4.433-08	1.964-08	8.729-09
1.11-01 -- 3.35-03	7.759-06	3.413-06	1.503-06	6.631-07	2.931-07	1.299-07	5.772-08
3.35-03 -- 5.83-04	1.287-04	5.559-05	2.391-05	1.025-05	4.376-06	1.863-06	7.911-07
5.83-04 -- 1.01-04	7.410-04	3.228-04	1.402-04	6.073-05	2.622-05	1.129-05	4.852-06
1.01-04 -- 2.90-05	3.463-03	1.515-03	6.614-04	2.880-04	1.251-04	5.427-05	2.348-05
2.90-05 -- 1.07-05	1.070-02	4.684-03	2.046-03	8.918-04	3.881-04	1.686-04	7.311-05
1.07-05 -- 3.06-06	3.036-02	1.328-02	5.803-03	2.530-03	1.101-03	4.787-04	2.078-04
3.06-06 -- 1.12-06	8.580-02	3.754-02	1.640-02	7.149-03	3.113-03	1.354-03	5.884-04
1.12-06 -- 4.14-07	1.969-01	8.616-02	3.764-02	1.641-02	7.147-03	3.107-03	1.349-03
- T H E R M A L -	4.820-01	2.110-01	9.220-02	4.025-02	1.755-02	7.650-03	3.332-03

Table XVII. - continued.

THERMONUCLEAR SOURCE IN AIR

$4\pi R^2 \times$ Flux (Neutrons/MeV per Source Neutron)

Energy Bounds	R(meters): 0	25	50	75	100	150	200	250
1.50+01 -- 1.22+01	2.520-02	2.242-02	2.012-02	1.811-02	1.631-02	1.319-02	1.062-02	8.526-03
1.22+01 -- 1.00+01	1.160-02	1.151-02	1.124-02	1.083-02	1.034-02	9.221-03	8.054-03	6.927-03
1.00+01 -- 8.18+00	7.740-03	8.487-03	8.840-03	8.934-03	8.854-03	8.378-03	7.674-03	6.878-03
8.18+00 -- 6.36+00	8.080-03	8.966-03	9.552-03	9.898-03	1.005-02	9.951-03	9.478-03	8.794-03
6.36+00 -- 4.96+00	1.290-02	1.428-02	1.522-02	1.582-02	1.614-02	1.614-02	1.555-02	1.459-02
4.96+00 -- 4.06+00	1.890-02	2.145-02	2.314-02	2.418-02	2.471-02	2.469-02	2.373-02	2.223-02
4.06+00 -- 3.01+00	2.470-02	2.933-02	3.104-02	3.183-02	3.208-02	3.128-02	2.929-02	2.666-02
3.01+00 -- 2.46+00	3.450-02	4.544-02	5.124-02	5.471-02	5.679-02	5.807-02	5.655-02	5.323-02
2.46+00 -- 2.35+00	4.540-02	5.519-02	6.370-02	7.003-02	7.450-02	7.913-02	7.950-02	7.699-02
2.35+00 -- 1.83+00	5.710-02	6.691-02	7.434-02	7.973-02	8.337-02	8.646-02	8.539-02	8.156-02
1.83+00 -- 1.11+00	8.270-02	1.043-01	1.202-01	1.314-01	1.386-01	1.440-01	1.409-01	1.328-01
1.11+00 -- 5.50-01	1.520-01	2.096-01	2.562-01	2.924-01	3.191-01	3.481-01	3.523-01	3.397-01
5.50-01 -- 1.11-01	2.320-01	4.043-01	5.381-01	6.433-01	7.244-01	8.248-01	8.596-01	8.471-01
1.11-01 -- 3.35-03	3.300+00	5.262+00	8.812+00	1.150+01	1.287+01	1.266+01	1.081+01	8.847+00
3.35-03 -- 5.83-04	4.400+01	6.918+01	1.241+02	1.771+02	2.162+02	2.472+02	2.341+02	2.020+02
5.83-04 -- 1.01-04	4.980+01	2.031+02	5.142+02	8.309+02	1.086+03	1.354+03	1.361+03	1.225+03
1.01-04 -- 2.90-05	2.780+01	2.926+02	1.206+03	2.493+03	3.790+03	5.609+03	6.158+03	5.821+03
2.90-05 -- 1.07-05	0.000	-6.485+02	7.258+02	4.434+03	8.967+03	1.615+04	1.885+04	1.819+04
1.07-05 -- 3.06-06	0.000	5.804+02	4.493+03	1.174+04	2.064+04	3.705+04	4.636+04	4.813+04
3.06-06 -- 1.12-06	0.000	-7.231+04	-5.360+04	1.364+04	7.152+04	1.304+05	1.439+05	1.375+05
1.12-06 -- 4.14-07	0.000	-1.711+05	-1.392+05	9.419+03	1.413+05	2.794+05	3.152+05	3.058+05
- T H E R M A L -	0.000	-4.151+05	-3.721+05	-3.732+04	2.735+05	6.137+05	7.160+05	7.107+05

Energy Bounds	R(meters): 300	400	500	600	900	1200	1500	1800
1.50+01 -- 1.22+01	6.824-03	4.344-03	2.745-03	1.723-03	4.131-04	9.532-05	2.135-05	4.671-06
1.22+01 -- 1.00+01	5.887-03	4.139-03	2.833-03	1.902-03	5.322-04	1.380-04	3.408-05	8.128-06
1.00+01 -- 8.18+00	6.066-03	4.554-03	3.305-03	2.340-03	7.572-04	2.245-04	6.312-05	1.710-05
8.18+00 -- 6.36+00	8.004-03	6.363-03	4.865-03	3.618-03	1.336-03	4.467-04	1.406-04	4.248-05
6.36+00 -- 4.96+00	1.342-02	1.089-02	8.482-03	6.418-03	2.488-03	8.711-04	2.870-04	9.074-05
4.96+00 -- 4.06+00	2.043-02	1.660-02	1.299-02	9.908-03	3.979-03	1.458-03	5.046-04	1.680-04
4.06+00 -- 3.01+00	2.377-02	1.815-02	1.338-02	9.650-03	3.371-03	1.115-03	3.588-04	1.132-04
3.01+00 -- 2.46+00	4.888-02	3.916-02	2.996-02	2.224-02	8.158-03	2.732-03	8.723-04	2.707-04
2.46+00 -- 2.35+00	7.261-02	6.108-02	4.881-02	3.766-02	1.517-02	5.444-03	1.831-03	5.917-04
2.35+00 -- 1.83+00	7.603-02	6.270-02	4.927-02	3.744-02	1.452-02	5.057-03	1.663-03	5.277-04
1.83+00 -- 1.11+00	1.219-01	9.757-02	7.461-02	5.540-02	2.045-02	6.923-03	2.234-03	7.000-04
1.11+00 -- 5.50-01	3.166-01	2.571-01	1.970-01	1.455-01	5.219-02	1.718-02	5.429-03	1.676-03
5.50-01 -- 1.11-01	8.034-01	6.690-01	5.210-01	3.886-01	1.400-01	4.549-02	1.415-02	4.308-03
1.11-01 -- 3.35-03	7.257+00	5.119+00	3.747+00	2.738+00	9.749-01	3.147-01	9.700-02	2.927-02
3.35-03 -- 5.83-04	1.665+02	1.087+02	7.222+01	4.951+01	1.682+01	5.416+00	1.663+00	4.995-01
5.83-04 -- 1.01-04	1.038+03	6.887+02	4.481+02	2.977+02	9.647+01	3.100+01	9.509+00	2.849+00
1.01-04 -- 2.90-05	5.071+03	3.437+03	2.229+03	1.460+03	4.559+02	1.453+02	4.457+01	1.335+01
2.90-05 -- 1.07-05	1.597+04	1.089+04	7.114+03	4.681+03	1.439+03	4.512+02	1.383+02	4.153+01
1.07-05 -- 3.06-06	4.480+04	3.258+04	2.144+04	1.386+04	4.108+03	1.292+03	3.954+02	1.182+02
3.06-06 -- 1.12-06	1.227+05	8.846+04	6.020+04	4.027+04	1.205+04	3.662+03	1.113+03	3.353+02
1.12-06 -- 4.14-07	2.766+05	2.035+05	1.403+05	9.448+04	2.814+04	8.452+03	2.555+03	7.698+02
- T H E R M A L -	6.554+05	4.979+05	3.508+05	2.391+05	7.114+04	2.100+04	6.285+03	1.891+03

Table XVIII. Multigroup spectra in air for a thermonuclear weapon point isotropic source. 1.11 g/l.

THERMONUCLEAR SOURCE IN AIR

$4\pi R^2 \times$ Flux (Neutrons/MeV per Source Neutron)

Energy Bounds / R(meters):	2100	2400	2700	3000	3300	3600	3800	4000
1.50+01 -- 1.22+01	1.002-06	2.113-07	4.392-08	9.020-09	1.832-09	3.686-10	1.260-10	4.290-11
1.22+01 -- 1.00+01	1.889-06	4.297-07	9.613-08	2.120-08	4.616-09	9.947-10	3.556-10	1.267-10
1.00+01 -- 8.18+00	4.509-06	1.164-06	2.955-07	7.401-08	1.833-08	4.494-09	1.753-09	6.813-10
8.18+00 -- 6.36+00	1.246-05	3.572-06	1.006-06	2.789-07	7.640-08	2.071-08	8.630-09	3.584-09
6.36+00 -- 4.96+00	2.785-05	8.351-06	2.459-06	7.134-07	2.044-07	5.795-08	2.488-08	1.064-08
4.96+00 -- 4.06+00	5.431-05	1.717-05	5.329-06	1.630-06	4.923-07	1.471-07	6.543-08	2.899-08
4.06+00 -- 3.01+00	3.515-05	1.077-05	3.260-06	9.763-07	2.897-07	8.523-08	3.755-08	1.649-08
3.01+00 -- 2.46+00	8.252-05	2.484-05	7.407-06	2.193-06	6.450-07	1.887-07	8.290-08	3.634-08
2.46+00 -- 2.35+00	1.860-04	5.737-05	1.744-05	5.245-06	1.563-06	4.623-07	2.045-07	9.025-08
2.35+00 -- 1.83+00	1.637-04	4.994-05	1.505-05	4.494-06	1.331-06	3.917-07	1.728-07	7.600-08
1.83+00 -- 1.11+00	2.149-04	6.506-05	1.948-05	5.783-06	1.705-06	4.999-07	2.200-07	9.664-08
1.11+00 -- 5.50-01	5.093-04	1.529-04	4.552-05	1.345-05	3.951-06	1.155-06	5.073-07	2.225-07
5.50-01 -- 1.11-01	1.296-03	3.862-04	1.143-04	3.364-05	9.852-06	2.872-06	1.260-06	5.517-07
1.11-01 -- 3.35-03	8.733-03	2.588-03	7.630-04	2.239-04	6.540-05	1.903-05	8.339-06	3.649-06
3.35-03 -- 5.83-04	1.485-01	4.389-02	1.291-02	3.781-03	1.103-03	3.206-04	1.405-04	6.145-05
5.83-04 -- 1.01-04	8.449-01	2.494-01	7.329-02	2.145-02	6.256-03	1.818-03	7.960-04	3.481-04
1.01-04 -- 2.90-05	3.952+00	1.164+00	3.417-01	9.999-02	2.915-02	8.471-03	3.709-03	1.622-03
2.90-05 -- 1.07-05	1.231+01	3.617+00	1.059+00	3.092-01	9.015-02	2.623-02	1.150-02	5.036-03
1.07-05 -- 3.06-06	3.494+01	1.028+01	3.012+00	8.805-01	2.565-01	7.451-02	3.262-02	1.426-02
3.06-06 -- 1.12-06	9.977+01	2.935+01	8.566+00	2.488+00	7.212-01	2.089-01	9.150-02	4.008-02
1.12-06 -- 4.14-07	2.294+02	6.759+01	1.972+01	5.715+00	1.652+00	4.772-01	2.087-01	9.137-02
- T H E R M A L -	5.656+02	1.671+02	4.879+01	1.410+01	4.054+00	1.163+00	5.066-01	2.211-01

Energy Bounds / R(meters):	4200	4400	4600	4800	5000	5200	5400
1.50+01 -- 1.22+01	1.456-11	4.927-12	1.662-12	5.593-13	1.877-13	6.281-14	2.097-14
1.22+01 -- 1.00+01	4.495-11	1.590-11	5.608-12	1.972-12	6.918-13	2.420-13	8.450-14
1.00+01 -- 8.18+00	2.640-10	1.021-10	3.934-11	1.513-11	5.808-12	2.225-12	8.507-13
8.18+00 -- 6.36+00	1.483-09	6.120-10	2.518-10	1.034-10	4.233-11	1.730-11	7.055-12
6.36+00 -- 4.96+00	4.539-09	1.930-09	8.181-10	3.459-10	1.460-10	6.145-11	2.582-11
4.96+00 -- 4.06+00	1.280-08	5.634-09	2.473-09	1.082-09	4.725-10	2.058-10	8.947-11
4.06+00 -- 3.01+00	7.222-09	3.154-09	1.374-09	5.975-10	2.592-10	1.122-10	4.850-11
3.01+00 -- 2.46+00	1.590-08	6.939-09	3.022-09	1.314-09	5.697-10	2.465-10	1.065-10
2.46+00 -- 2.35+00	3.973-08	1.745-08	7.648-09	3.346-09	1.461-09	6.369-10	2.772-10
2.35+00 -- 1.83+00	3.337-08	1.462-08	6.395-09	2.792-09	1.217-09	5.299-10	2.304-10
1.83+00 -- 1.11+00	4.237-08	1.855-08	8.105-09	3.537-09	1.541-09	6.708-10	2.916-10
1.11+00 -- 5.50-01	9.739-08	4.257-08	1.858-08	8.099-09	3.525-09	1.533-09	6.657-10
5.50-01 -- 1.11-01	2.412-07	1.053-07	4.592-08	2.000-08	8.697-09	3.778-09	1.639-09
1.11-01 -- 3.35-03	1.595-06	6.960-07	3.033-07	1.321-07	5.742-08	2.494-08	1.082-08
3.35-03 -- 5.83-04	2.685-05	1.171-05	5.105-06	2.222-06	9.661-07	4.195-07	1.819-07
5.83-04 -- 1.01-04	1.520-04	6.630-05	2.889-05	1.257-05	5.465-06	2.373-06	1.029-06
1.01-04 -- 2.90-05	7.081-04	3.088-04	1.345-04	5.850-05	2.542-05	1.104-05	4.786-06
2.90-05 -- 1.07-05	2.202-03	9.609-04	4.186-04	1.820-04	7.898-05	3.421-05	1.479-05
1.07-05 -- 3.06-06	6.225-03	2.714-03	1.182-03	5.142-04	2.234-04	9.697-05	4.205-05
3.06-06 -- 1.12-06	1.756-02	7.696-03	3.373-03	1.478-03	6.474-04	2.835-04	1.241-04
1.12-06 -- 4.14-07	4.004-02	1.757-02	7.720-03	3.395-03	1.495-03	6.585-04	2.903-04
- T H E R M A L -	9.676-02	4.251-02	1.875-02	8.309-03	3.699-03	1.655-03	7.437-04

Table XVIII. - continued.

ing a Legendre sum of the harmonic coefficients for the appropriate values of E and z. In practice, the number of coefficients available is insufficient - particularly for high energy neutrons - and spurious oscillations or even negative values occur. These oscillations can be reduced considerably by using an extrapolation technique devised by Spencer [6] for estimating harmonic coefficients for higher ℓ values. Although this technique does not eliminate oscillations, the remaining peculiarities are very often limited to negative values of $\cos\theta$, where the angular distribution has relatively low values and is therefore of little practical interest. The angular distribution near the source, particularly for point sources, tend to exhibit trends associated with the single scattered angular distribution. Thus near the source, one expects that the angular distribution would exhibit oscillations similar to those of the scattering kernel.

In Figure 4 we show a comparison between moments and S_n calculations of the Snyder-Neufeld dose angular distribution for a fission source. These comparisons indicate that the moment results tend to be more peaked forward than the S_n results. Consequently more neutrons would transport further from the source. This phenomena also contributes to the higher dose values at large distances that is predicted by the moments calculations.

The authors wish to express their thanks to Dr. Lewis V. Spencer, whose encouragement and advice helped make this paper possible.

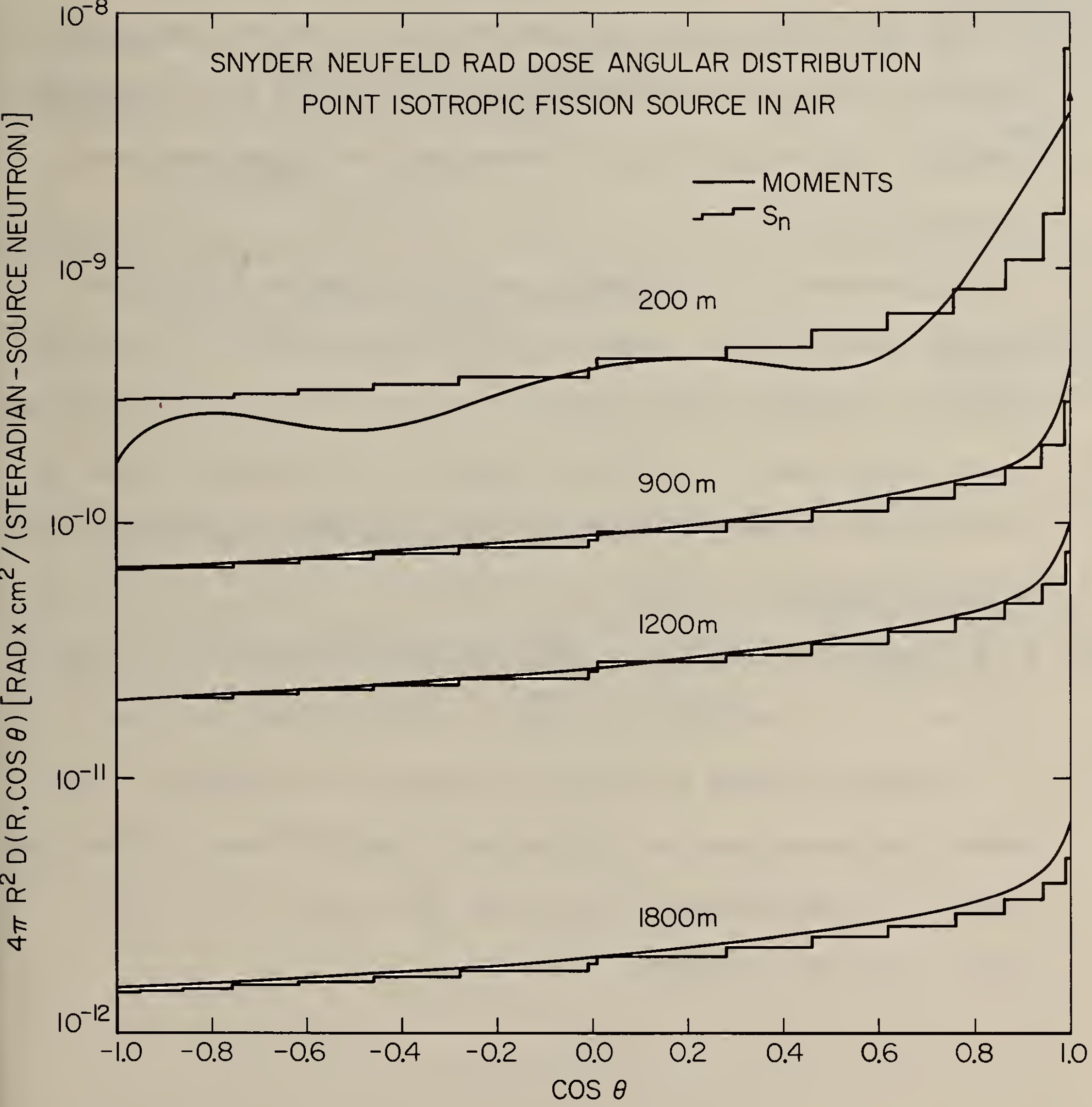

Figure 4. Comparison of moments and S_n (Ref. 5) calculations of the angular distribution of Snyder-Neufeld dose in rads at four depths in air.

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16. ABSTRACT (A 200-word or less factual summary of most significant information. If document includes a significant bibliography or literature survey, mention it here.)

The moments method is applied to the problem of calculating the neutron dose and fluence distributions in an infinite medium of air. These calculations are compared with Monte Carlo and Discrete Ordinates (S_n) results. Simple parametric representations for the distributions are given which facilitate the calculation of dose and flux distributions in air with a different density.

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