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MONTE CARLO AIR-SCATTERING DATA FOR MONOENERGETIC FAST NEUTRONS FROM POINT ISOTROPIC SOURCES

U. S. AIR FORCE

Nuclear Aerospace Research Facility
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GENERAL DYNAMICS/FORT WORTH

A Division of General Dynamics Corporation

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ABSTRACT

The Monte Carlo fast-neutron air-scattering data presented in FZK-9-147, Volumes I and II, have been integrated to obtain the angular distributions and energy spectra for a point isotropic source emitting one neutron per second at a given energy E_0. These calculations were performed for source-detector separations of 10, 35, 64, and 100 feet and for initial neutron energies of 0.33, 1.1, 2.7, 4.0, 6.0, 8.0, 10.9, and 14.0 Mev.

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I. INTRODUCTION

Air scattering is the principal process by which neutrons leaving a source in air are transported to other positions some distance away. In order to make comprehensive shield-design studies, one must know the energy and angular distributions of the scattered neutron flux at the positions of interest. It is possible to design and apply Monte Carlo procedures which solve this problem directly for specific cases; however, it is often more practical to use the Monte Carlo procedures to generate data in a suitable parametric form so as to be able to apply the results without having to run a new problem for each new source term.

This report presents parametric air-scattering data for isotropic sources of monoenergetic fast neutrons. For source energies of 0.33, 1.1, 2.7, 4.0, 6.0, 8.0, 10.9, and 14.0 Mev, the angular distributions of the scattered flux and dose rate and the energy spectra of the scattered flux are given for source-receiver separation distances of 10, 35, 64, and 100 feet.

The data presented here were computed from the results of Wells' Monte Carlo parameter study of neutron scattering for directional point sources in an infinite, homogeneous medium of air (Refs. 1 and 2). A straight-forward integration procedure was formulated and programmed in Fortran for the IBM-704 in order to carry out the calculations. The procedure was designed to utilize directly the punched card output of Wells' Monte Carlo calculations.

Section II of this report deals with the geometry of the neutron-scattering problem while Section III describes the Monte Carlo data used in these calculations. The integration scheme and Fortran procedure are discussed in Section IV and the results are presented in Section V in the form of tables and graphs giving the angular distributions of the scattered flux and dose rate and the energy spectra of the scattered flux.

II. NEUTRON SCATTERING GEOMETRY

The geometry (Fig. 1) consists of a neutron source S located in an infinite, homogeneous medium of air. A detector D of unit cross section is located a distance $\underset{\sim}{a}$ from the source S. The neutron current leaving the source is described by a polar angle K and an azimuthal angle ϕ. The polar angle is measured with respect to the source-detector axis while the azimuthal angle is the angle between the positive y axis and the projection of the neutron direction on the y,z plane. The azimuthal angle is measured in a clockwise direction.

The detector angle β is the polar angle between the direction of the incoming neutron flux and the source-detector axis. The azimuthal angle ϕ' at the detector is defined in the same manner as ϕ. The neutron current of energy E_o moving in the direction (K,ϕ) at a point Q on the surface of a unit sphere about S is defined as $S(K,\phi,E_o)$. The number of neutrons passing through a surface element dA at Q per unit time is given by $S(K,\phi,E_o)dA$.

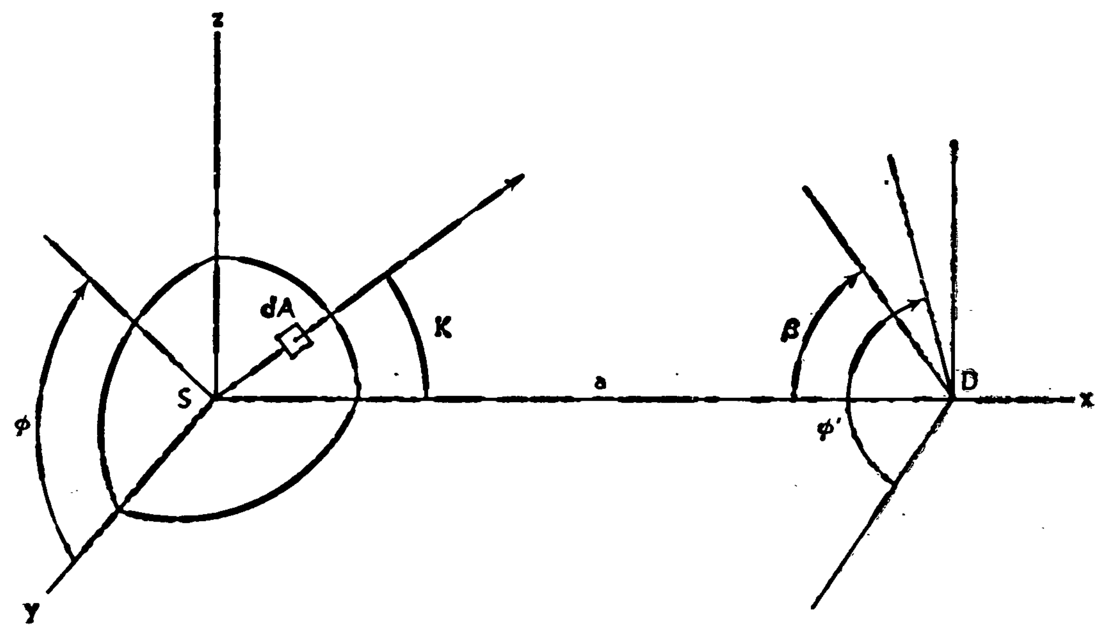

FIGURE 1. SCATTERING GEOMETRY

III. THE MONTE CARLO DATA

Since the generation of the Monte Carlo data used in these calculations has been thoroughly described by Wells in Reference 1, only a brief review is needed here.

The effects of elastic scattering, inelastic scattering and absorption were taken into account in the Monte Carlo air-scattering calculations. The neutron cross sections for nitrogen and oxygen used in these calculations were taken from data giving by Lustig, Goldstein, and Kalos (Refs. 3 and 4). The neutron cross sections for air were computed on the basis of a mixture of 78% nitrogen and 22% oxygen and a density of 5.37×10^{19} atoms/cc or 0.1293×10^{-2} gm/cc. The flux-to-dose conversion factors F(E) used in computing the tissue dose rates were those calculated by Hurst and Ritchie (Ref. 5).

The calculated results presented in Reference 2 represent the angular distributions of the neutron flux, the angular distributions of the tissue dose rate, and the energy distribution of the neutron flux at the detector D for a point monodirectional source S emitting one neutron per second of energy E_0 in the given direction (K,ϕ). The calculations were performed for four separation distances of 10, 35, 64, and 100 feet, eight initial neutron energies of 0.33, 1.1, 2.7, 4.0, 6.0, 8.0, 10.9 and 14.0 Mev, and eight source angles, K, of 5, 15, 30, 60, 90, 120, 150, and 180 degrees. This represents a total of 64 different source descriptions and four source-detector separations which may be combined to describe any arbitrary source distribution.

The scattered flux reaching the detector is sorted into eighteen equal angle intervals, β_i, and ten arbitrary energy intervals, E_j. $N(K,\phi,E_o,a,\beta_i,E_j)$ is defined as the fraction of neutrons with initial energy E_o and direction (K,ϕ) passing through the incremental area dA per unit time which reach the detector at a distance a in the detector angle increment β_i with energy in the energy group E_j. $D(K,\phi,E_o,a,\beta_i)$ is defined as the neutron dose rate due to neutrons with initial energy E_o and direction (K,ϕ) which arrive at the detector D a distance a from the source in the detector angle increment β_i. Since there is symmetry about the source-detector axis and both source and detector are located in the infinite homogeneous medium, angular distributions at the detector are independent of both source and detector azimuthal angle. Therefore, the data are not sorted with respect to this angle, and the variable ϕ is dropped from the expressions.

The following quantities are tabulated in Reference 2: $N(K,E_o,a,\beta_i,E_j)$, $N(K,E_o,a,E_j)$, $N(K,E_o,a,\beta_i)$, $D(,E_o,a,\beta_i)$, $N(K,E_o,a)$, and $D(K,E_o,a)$. $N(K,E_o,a,\beta_i,E_j)$ is tabulated in the form of an 18 x 10 matrix for each of the four values of a, eight values of E_o, and eight values of K. In addition to the 18 x 10 matrices, the energy distribution of the air-scattered flux is given by

$$N(K,E_o,a,E_j) = \sum_{i=1}^{18} N(K,E_o,a,\beta_i,E_j) ; \qquad (1a)$$

the angular distributions of the air-scattered flux is given by

$$N(K,E_o,a,\beta_i) = \sum_{j=1}^{10} N(K,E_o,a,\beta_i,E_j) . \qquad (2a)$$

The dose rate $D(K,E_o,a,\beta_i)$ is calculated and stored as a function of K,E_o,a and β_i. It is printed as a function of angle β for each combination of K,E_o, and a.

The total scattered flux and dose rate are given, respectively, by

$$N(K,E_o,a) = \sum_{i=1}^{18} \sum_{j=1}^{10} N(K,E_o,a,\beta_i,Ej) , \qquad (4a)$$

and

$$D(K,E_o,a) = \sum_{i=1}^{18} D(K,E_o,a,\beta_i) . \qquad (5a)$$

The units of the scattered fluxes are neutrons/cm^2-sec per source neutron/sec and those of the scattered dose rates are rem/hr per source neutron/sec.

The total scattered flux from a source $S(K,E_o)$ reaching the detector D at a distance a is

$$N(E_o,a) = \int_0^{\pi}\int_0^{2\pi} S(K,E_o)N(K,E_o,a)\ \sin K\ dK\ d\phi \tag{6a}$$

$$= 2\pi \int_0^{\pi} S(K,E_o)N(K,E_o,a)\ \sin K\ dK. \tag{6b}$$

Similarly, the total scattered dose rate is

$$D(E_o,a) = \int_0^{\pi}\int_0^{2\pi} S(K,E_o)D(K,E_o,a)\ \sin K\ dK\ d\phi \tag{7a}$$

$$= 2\pi \int_0^{\pi} S(K,E_o)D(K,E_o,a)\ \sin K\ dK. \tag{7b}$$

Similarly, the angular distributions of flux and dose rate are given, respectively, by

$$N(E_o,a,\beta) = \int_0^{\pi}\int_0^{2\pi} S(K,E_o)N(K,E_o,a,\beta)\ \sin K\ dK\ d\phi \tag{8a}$$

$$= 2\pi \int_0^{\pi} S(K,E_o)N(K,E_o,a,\beta)\ \sin K\ dK, \tag{8b}$$

and

$$D(E_o,a,\beta) = \int_0^{\pi}\int_0^{2\pi} S(K,E_o)D(K,E_o,a,\beta)\ \sin K\ dK\ d\phi \tag{9a}$$

$$= 2\pi \int_0^{\pi} S(K,E_o)D(K,E_o,a,\beta)\ \sin K\ dK. \tag{9b}$$

The energy distribution of the scattered flux is given by

$$N(a,E_oE) = \int_0^{\pi}\int_0^{2\pi} S(K,E_o)N(K,E_o,a,E) \sin K \, dK \, d\phi \qquad (10a)$$

$$= 2\pi \int_0^{\pi} S(K,E_o)N(K,E_o,a,E) \sin K \, dK. \qquad (10b)$$

An analysis of the Monte Carlo data showed that the standard deviation of the total scattered flux $N(K,E_o,a)$ was less than 0.1 $N(K,E_o,a)$ in 60.2% of the problems, less than 0.15 $N(K,E_o,a)$ in 91.4% of the problems, and in only 2.73% of the problems did the standard deviation exceed 0.2 $N(K,E_o,a)$.

IV. INTEGRATION OF THE MONTE CARLO DATA

Since the quantities $N(K,E_o,a,\beta)$, $D(K,E_o,a,\beta)$, and $N(K,E_o,a,E)$ are not given in terms of analytic functions, it is necessary to devise some integration scheme that will produce accurate results. For the purpose of numerical integrations, Equations 6b, 7b, 8b, 9b, and 10b become, in turn

$$N(E_o,a) = 2\pi \sum_{n=1}^{8} S(K_n,E_o)N(K_n,E_o,a) \sin K_n \,(\Delta K_n), \qquad (6c)$$

$$D(E_o,a) = 2\pi \sum_{n=1}^{8} S(K_n,E_o)D(K_n,E_o,a) \sin K_n \,(\Delta K_n), \qquad (7c)$$

$$N(E_o,a,\beta_i) = 2\pi \sum_{n=1}^{8} S(K_n,E_o)N(K_n,E_o,a,\beta_i) \sin K_n \,(\Delta K_n), \qquad (8c)$$

$$D(E_o,a,\beta_i) = 2\pi \sum_{n=1}^{8} S(K_n,E_o)D(K_n,E_o,a,\beta_i) \sin K_n \,(\Delta K_n), \qquad (9c)$$

$$N(E_o,a,E_j) = 2\pi \sum_{n=1}^{8} S(K_n,E_o)N(K_n,E_o,a,E_j) \sin K_n \,(\Delta K_n), \qquad (10c)$$

where $i = 1, 2, \ldots, 18$ and $j = 1, 2, \ldots, 10$.

The calculations presented here were performed for a point isotropic source emitting 1 neutron/sec with energy E_0. Therefore,

$$S(K,E_0) = \text{constant} = \frac{1}{4\pi} \qquad (11)$$

The values of N and D were taken from the tables in Reference 2.

The summations in Equations 6c and 7c were performed using a machine code designed primarily for matrix calculations but adaptable for these calculations.

An IBM Fortran procedure was written to provide for the integration of Equations 8c, 9c, and 10c with a nonisotropic source term depending only on the polar angle K. The isotropic case presented here is a special case of this more general source term. A copy of the Fortran program is shown in the Appendix. Certain subroutines, such as SETUP, END 9, LIB 1, LIB, and END(2,1), are special General Dynamics/Fort Worth subroutines. However, the basic program should remain the same anywhere.

The values of (ΔK_n) used in the numerical integration were chosen after considering which histogram would best represent the smooth curve for integration purposes. The histogram-fit tends to overestimate in the first 20 degrees and underestimate in the last 20 degrees.

The validity of the integration scheme was checked by comparing the results of numerical integration with those obtained using a planimeter on smooth curves drawn through the eight points. In the seven cases chosen at random for comparison, the numerical integration overestimated the result by less than 2.5%. In all cases, the numerical integration overestimated the result.

V. RESULTS

Equations 6c, 7c, 8c, 9c, and 10c have been numerically integrated for a point isotropic source. The results are presented in this section in both tabulated and graphical form.

The scattered neutron flux, $N(E_o,a,E_j)$, Tables II through V, and the neutron flux per Mev, $N(E_o,a,E_j)/\Delta E_j$, Tables VI through IX, have been tabulated as functions of E_o, the initial energy, and $(E_m)_j$. The subscript m indicates that $(E_m)_j$ is the lower limit of the j^{th} energy interval. $(E_m)_j$ for each E_o is shown in Table I. The minimum energy cutoff for each source energy is given in Table I by $(E_m)_j$ for each source energy. ΔE_j is the width of the j^{th} energy group. The scattered neutron flux, $N(E_o,a,E_j)$, is reported in units of

$$\frac{\text{neutrons/cm}^2\text{-sec}}{\text{source neutron/sec}}$$

and the neutron flux per Mev, $N(E_o,a,E_j)/\Delta E_j$ in units of

$$\frac{\text{neutrons/cm}^2\text{-sec/Mev}}{\text{source neutron/sec}}.$$

The angular distributions of the scattered neutron flux and dose rate in a 10^o interval of β, $N(E_o,a,\beta_i)$ and $D(E_o,a,\beta_i)$, are tabulated as functions of E_o and β_i in Tables X through XIII and XIV through XVII, respectively, where β_i is the upper limit of the i^{th} angular interval. The units here are

$$\frac{\text{neutrons/cm}^2\text{-sec in } 10^\circ \text{ interval}}{\text{source neutron/sec}}$$

and

$$\frac{\text{rem/hr in } 10^\circ \text{ interval}}{\text{source neutron/sec}} .$$

The quantities $N(E_o,a,E_j)/\Delta E_j$, $N(E_o,a,\beta_i)$, $D(E_o,a,\beta_i)$, $N(E_o,a)$, and $D(E_o,a)$ are shown in Figures 2-9, 10-17, 18-25, 26 and 27, respectively.

The angular distributions are plotted as function of $\bar{\beta}_i$, the midpoint of the angular interval β_{i-1} to β_i.

In Figures 2 through 9, the energy spectrum is plotted against the midpoint of the energy group E_j.

The total scattered flux and dose rate, $N(E_o,a)$ and $D(E_o,a)$, respectively, are shown in Figures 26 and 27 as functions of E_o for each of the four separation distances considered.

The scattered dose rates for source energies of 1.1 Mev or greater result from neutrons with energies greater than the minimum energy used for each source energy, but the scattered dose rates for the 0.33 Mev source are those resulting from neutrons with energies greater than 0.22 Mev.

TABLE I. LOWER BOUNDS OF THE ENERGY GROUPS USED TO DEFINE THE SCATTERED NEUTRON FLUX FOR EACH SOURCE ENERGY

$(E_m)_j$ (Mev)

Energy Group Index j	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.070	0.20	0.60	0.33	0.33	0.33	0.33	0.33
2	0.096	0.29	0.90	0.50	0.75	1.75	1.75	1.75
3	0.122	0.38	1.10	0.80	1.50	2.50	2.50	2.50
4	0.148	0.47	1.30	1.20	2.25	3.00	3.75	3.50
5	0.174	0.56	1.50	1.60	2.75	3.50	4.75	4.50
6	0.200	0.65	1.70	2.00	3.50	4.50	5.50	5.50
7	0.226	0.74	1.90	2.40	4.00	5.50	6.75	6.75
8	0.252	0.83	2.10	2.80	4.50	6.00	8.00	8.00
9	0.278	0.92	2.30	3.20	5.00	6.50	9.00	10.0
10	0.304	1.01	2.50	3.60	5.50	7.25	10.0	12.0

TABLE II. TOTAL SCATTERED NEUTRON FLUX
Separation Distance - 10 Feet

[(neutrons/cm^2-sec)/(source neutron/sec)]

Energy Group Index j	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.1122-12	0.	0.5394-12	0.1167-13	0.1128-08	0.2512-08	0.4350-08	0.8118-09
2	0.2790-10	0.4869+10	0.2064-10	0.5279-12	0.1870-08	0.2370-08	0.2482-09	0.3264-08
3	0.2977-09	0.6109-10	0.1182-09	0.7102-10	0.1409-08	0.1955-08	0.2358-08	0.1626-08
4	0.7100-09	0.2407-09	0.1914-09	0.3504-10	0.1764-09	0.1092-08	0.4100-08	0.1231-09
5	0.3901-08	0.7973-09	0.6295-09	0.8408-10	0.2141-09	0.6667-09	0.2416-08	0.5218-09
6	0.2775-08	0.2744-09	0.2014-08	0.6066-09	0.4960-09	0.4670-09	0.3055-08	0.3969-08
7	0.1578-07	0.5649-08	0.8703-08	0.2911-08	0.3923-09	0.3684-09	0.5132-10	0.3698-08
8	0.2731-07	0.5370-07	0.1313-07	0.2503-07	0.1420-07	0.6227-08	0.4220-08	0.2806-08
9	0.2144-07	0.1407-07	0.8017-08	0.1431-07	0.6063-08	0.4763-08	0.2298-08	0.5662-08
10	0.2060-07	0.1475-07	0.8423-08	0.1322-07	0.1249-07	0.1104-07	0.8306-08	0.1423-07

0.1122-12 = 1.122×10^{-13}

TABLE III. TOTAL SCATTERED NEUTRON FLUX
Separation Distance - 35 Feet

[(neutrons/cm^2-sec)/(source neutron/sec)]

Energy Group Index J	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.7994-13	0.	0.1401-11	0.7245-14	0.5-10-09	0.1047-08	0.1638-08	0.4155-09
2	0.2408-10	0.7550-12	0.2016-10	0.496[illegible]-1[illegible]	0.6236-09	0.9556-09	0.9950-10	0.8967-09
3	0.1916-09	0.3998-10	0.12-7-09	0.5142-10	0.1875-09	0.6682-09	0.7220-09	0.3803-09
4	0.4971-09	0.2309-09	0.1843-09	0.1496-10	0.9951-10	0.2960-09	0.8941-09	0.1023-09
5	0.1247-08	0.7369-09	0.3327-09	0.8966-10	0.1712-09	0.3182-09	0.6678-09	0.2930-09
6	0.2157-08	0.1620-08	0.5705-09	0.5021-09	0.2181-09	0.1877-09	0.4181-09	0.1457-08
7	0.3934-08	0.2279-08	0.2448-08	0.1370-08	0.4158-09	0.1005-09	0.1688-09	0.9020-09
8	0.6643-08	0.1272-07	0.3433-08	0.6336-08	0.3368-08	0.1595-08	0.9789-09	0.6838-09
9	0.4941-08	0.3213-08	0.2513-08	0.4132-08	0.1582-08	0.1165-08	0.8691-09	0.1459-08
10	0.5449-08	0.3962-08	0.2078-08	0.3972-08	0.3485-08	0.3118-08	0.2571-08	0.4062-08

0.7994-13 = 7.994×10^{-14}

TABLE IV. TOTAL SCATTERED NEUTRON FLUX
Separation Distance - 64 Feet

[(neutrons/cm^2-sec)/(source-neutron/sec)]

Energy Group Index J	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.6086-13	0.	0.1209-11	0.1068-14	0.3648-09	0.6106-09	0.8175-09	0.2784-09
2	0.2306-10	0.7801-12	0.2046-10	0.5026-12	0.5169-09	0.4549-09	0.6783-10	0.4817-09
3	0.1588-09	0.3263-10	0.3247-09	0.3786-10	0.1105-09	0.3527-09	0.4818-09	0.2851-09
4	0.4821-09	0.2229-09	0.1417-09	0.1988-10	0.9799-10	0.2062-09	0.4905-09	0.7119-10
5	0.7371-09	0.5434-09	0.2486-09	0.1052-09	0.1588-09	0.2452-09	0.3351-05	0.1776-09
6	0.1148-08	0.1176-08	0.4671-09	0.3451-09	0.1897-09	0.2141-09	0.2462-09	0.6094-09
7	0.2176-08	0.1685-08	0.1179-08	0.1051-08	0.3678-09	0.1174-09	0.9096-10	0.6088-09
8	0.3300-08	0.5393-08	0.1683-08	0.3358-08	0.1700-08	0.8854-09	0.4840-09	0.3930-09
9	0.2528-08	0.1696-08	0.1323-08	0.2046-08	0.1054-08	0.7237-09	0.4625-09	0.7167-09
10	0.2497-08	0.1919-08	0.1162-08	0.1929-08	0.1599-08	0.1570-08	0.1312-08	0.1845-08

0.6086-13 = 6.086×10^{-14}

TABLE V . TOTAL SCATTERED NEUTRON FLUX
Separation Distance - 100 Feet

[(neutrons/cm^2-sec)/(source neutron/sec)]

Energy Group Index J	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.4276-13	0.	0.1111-11	0.1103-14	0.1795-09	0.4964-09	0.3456-09	0.1845-09
2	0.2850-10	0.4220-12	0.2009-10	0.2177-11	0.2871-09	0.3141-09	0.5919-10	0.3015-09
3	0.1530-09	0.4332-10	0.8127-10	0.1287-10	0.6905-10	0.2214-09	0.4010-09	0.1335-09
4	0.3945-09	0.2484-09	0.1386-09	0.2256-10	0.1065-09	0.1082-09	0.2241-09	0.6058-10
5	0.7213-09	0.4144-09	0.2417-09	0.8228-10	0.1350-09	0.0957-10	0.2154-09	0.1242-09
6	0.6783-09	0.8503-09	0.3731-09	0.3470-09	0.1602-09	0.1100-09	0.1281-09	0.4271-09
7	0.1348-08	0.9637-09	0.7032-09	0.8224-09	0.2867-09	0.7662-10	0.5522-10	0.4062-09
8	0.1754-08	0.2784-08	0.9533-09	0.1871-08	0.5475-09	0.4564-09	0.2474-09	0.2194-09
9	0.1337-08	0.1175-08	0.7217-09	0.1149-08	0.6040-09	0.4504-09	0.3146-09	0.4098-09
10	0.1343-08	0.1165-08	0.6402-09	0.1170-08	0.9556-09	0.9477-09	0.7774-09	0.1217-08

0.4276-13 = 4.276×10^{-14}

TABLE VI. TOTAL SCATTERED NEUTRON FLUX PER MEV
Separation Distance - 10 Feet

[(neutron/cm^2-sec-Mev)/(source neutron/sec)]

Energy Group Index j	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.4315-11	0.0000-00	0.1798-11	0.6865-13	0.2686-08	0.1769-08	0.3063-08	0.5717-09
2	0.1073-08	0.8372-11	0.1032-09	0.1760-11	0.2493-08	0.3160-08	0.3309-09	0.4352-08
3	0.1145-07	0.6788-09	0.5910-09	0.1776-09	0.1879-08	0.3910-08	0.1886-08	0.1626-08
4	0.2731-07	0.2674-08	0.9570-09	0.8760-10	0.3526-09	0.2184-08	0.4099-08	0.1231-09
5	0.1500-06	0.8859-08	0.3148-08	0.2102-09	0.2855-09	0.6667-09	0.3221-08	0.5218-09
6	0.1066-06	0.3048-07	0.1007-07	0.1517-08	0.9920-09	0.4669-09	0.2444-08	0.3175-08
7	0.6069-06	0.6277-07	0.4352-07	0.7278-08	0.7846-09	0.7368-09	0.4106-10	0.2958-08
8	0.1050-05	0.5967-06	0.6565-07	0.6258-07	0.2840-07	0.1245-07	0.4220-08	0.1403-08
9	0.8246-06	0.1563-06	0.4009-07	0.3578-07	0.1213-07	0.6351-08	0.2298-08	0.2831-08
10	0.7923-06	0.1639-06	0.4212-07	0.3305-07	0.2498-07	0.1471-07	0.9229-08	0.7115-08

0.4315-11 = 4.315x10^{-12}

TABLE VII. TOTAL SCATTERED NEUTRON FLUX PER MEV

Separation Distance - 35 Feet

[(neutron/cm^2-sec-Mev)/(source neutron/sec)]

Energy Group Index j	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.3075-12	0.0000-00	0.4670-11	0.4261-14	0.1288-08	0.7373-09	0.1153-08	0.2925-09
2	0.9262-09	0.8388-11	0.1008-09	0.1656-11	0.8314-09	0.1274-08	0.1327-09	0.1196-08
3	0.7369-08	0.4442-09	0.6235-09	0.1286-09	0.2500-09	0.1336-08	0.5776-09	0.3803-09
4	0.1912-07	0.2566-08	0.9215-09	0.3740-10	0.1990-09	0.5920-09	0.8941-09	0.1023-09
5	0.4796-07	0.8209-08	0.1914-08	0.2241-09	0.2283-09	0.3182-09	0.8904-09	0.2930-09
6	0.8296-07	0.1800-07	0.2853-08	0.1255-08	0.4362-09	0.1877-09	0.3345-09	0.1166-08
7	0.1513-06	0.2532-07	0.1224-07	0.3425-08	0.8316-09	0.2010-09	0.1350-09	0.7216-09
8	0.2555-06	0.1413-06	0.1717-07	0.1584-07	0.6776-08	0.3190-08	0.9789-09	0.3419-09
9	0.1900-06	0.3570-07	0.1257-07	0.1033-07	0.3164-08	0.1553-08	0.8690-09	0.7295-09
10	0.2096-06	0.4402-07	0.1039-07	0.9930-08	0.6970-08	0.4157-08	0.2856-08	0.2031-08

0.3075-12 = 3.075×10^{-13}

TABLE VIII. TOTAL SCATTERED NEUTRON FLUX PER MEV
Separation Distance = 64 Feet

[(neutron/cm^2-sec-Mev)/(source neutron/sec)]

Energy Group Index j	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.2341-11	0.0000-00	0.4030-11	0.6282-14	0.8686-09	0.4300-09	0.5757-09	0.1960-09
2	0.8869-09	0.8667-11	0.1023-09	0.1675-11	0.6892-09	0.6065-09	0.9042-10	0.6423-09
3	0.6104-08	0.3626-09	0.1624-08	0.9465-10	0.1473-09	0.7052-09	0.3854-09	0.2851-09
4	0.1854-07	0.2477-08	0.7085-09	0.4970-10	0.1960-09	0.4124-09	0.4905-09	0.7119-10
5	0.2835-07	0.6037-08	0.1243-08	0.2628-09	0.2117-09	0.2452-09	0.4468-09	0.1776-09
6	0.4415-07	0.1307-07	0.2336-08	0.8628-08	0.3790-09	0.2141-09	0.1970-09	0.4875-09
7	0.8369-07	0.1883-07	0.5895-08	0.2628-08	0.7354-09	0.2348-09	0.7277-10	0.4870-09
8	0.1269-06	0.5992-07	0.8415-08	0.8395-08	0.3400-08	0.1771-08	0.4840-09	0.1965-09
9	0.9723-07	0.1884-07	0.6610-08	0.5115-08	0.1108-08	0.9649-09	0.4625-09	0.3583-09
10	0.9604-07	0.2132-07	0.5810-08	0.4823-08	0.3198-08	0.2093-08	0.1458-08	0.9225-09

0.2341-11 = 2.341x10^{-12}

TABLE IX. TOTAL SCATTERED NEUTRON FLUX PER MEV
Separation Distance - 100 Feet

[(neutron/cm^2-sec-Mev)/(source neutron/sec)]

Energy Group Index j	Source Energy (Mev) 0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.1644-11	0.0000-00	0.3703-11	0.6488-14	0.4244-09	0.3495-09	0.3842-09	0.1299-09
2	0.1096-08	0.4689-11	0.1005-09	0.7257-11	0.3828-09	0.4188-09	0.7892-10	0.4020-09
3	0.5885-08	0.4813-09	0.4063-09	0.3218-10	0.9205-10	0.4428-09	0.3208-09	0.1335-09
4	0.1517-07	0.2760-08	0.6930-09	0.5640-10	0.2130-09	0.2164-09	0.2241-09	0.6058-10
5	0.2774-07	0.4604-08	0.1209-08	0.2057-09	0.1800-09	0.8987-10	0.2872-09	0.1242-09
6	0.2609-07	0.9447-08	0.1866-08	0.8675-09	0.3204-09	0.1180-09	0.1025-09	0.3417-09
7	0.5181-07	0.1071-07	0.3516-08	0.2056-08	0.5774-09	0.1576-09	0.4418-10	0.3202-09
8	0.6746-07	0.3093-07	0.4767-08	0.4678-08	0.1895-08	0.9728-09	0.2474-09	0.1097-09
9	0.5142-07	0.1306-07	0.3609-08	0.2873-08	0.1208-08	0.6004-09	0.3146-09	0.2049-09
10	0.5165-07	0.1294-07	0.3201-08	0.2923-08	0.1799-08	0.1264-08	0.8638-09	0.6085-09

0.1644-11 = 1.644x10^{-12}

TABLE X . ANGULAR DISTRIBUTION OF TOTAL SCATTERED NEUTRON FLUX
Separation Distance - 10 Feet

[(neutrons/cm^2-sec)/(source neutron/sec)]

Detector Angular Interval (degrees)	Source Energy (Mev) 0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
10	0.1122-07	0.9942-08	0.4357-08	0.6558-08	0.6104-08	0.6060-08	0.4696-08	[illegible]
20	0.1054-07	0.1060-07	0.4611-08	0.6806-08	0.5373-08	0.5074-08	0.5240-08	[illegible]
30	0.9124-08	0.6955-08	0.5428-08	0.5212-08	0.4776-08	0.2505-08	0.2489-08	0.372[illegible]-08
40	0.1019-07	0.6924-08	0.4250-08	0.4590-08	0.3577-08	0.2746-08	0.2035-08	0.3344-08
50	0.5104-08	0.7208-08	0.3178-08	0.4972-08	0.3241-08	0.2651-08	0.1965-08	0.7542-09
60	0.6367-08	0.6285-08	0.3880-08	0.4228-08	0.2061-09	0.1574-08	0.2246-08	0.1411-08
70	0.6506-08	0.5420-08	0.2317-08	0.3174-08	0.2327-08	0.1425-08	0.1516-08	0.2283-08
80	0.5911-08	0.6200-08	0.1905-08	0.3500-08	0.2701-08	0.1467-08	0.1101-08	0.1501-08
90	0.5794-08	0.4625-08	0.2566-08	0.2769-08	0.2273-08	0.1186-08	0.2144-08	0.1035-08
100	0.4296-08	0.4814-08	0.1853-08	0.7932-08	0.1362-08	0.1015-08	0.1695-08	0.1828-08
110	0.4096-08	0.5075-08	0.1605-08	0.2246-08	0.1045-08	0.1655-08	0.1340-08	0.7659-09
120	0.3297-08	0.4353-08	0.1184-08	0.2780-08	0.1701-08	0.7857-09	0.1032-08	0.8606-09
130	0.1838-08	0.2454-08	0.1081-08	0.1211-08	0.4333-09	0.6927-09	0.5874-09	0.5031-09
140	0.1719-08	0.3227-08	0.6343-09	0.1643-09	0.7180-09	0.5468-09	0.4502-09	0.5939-09
150	0.1596-08	0.2491-08	0.8437-09	0.1336-08	0.5656-09	0.5291-09	0.5254-09	0.5987-09
160	0.1375-08	0.2004-08	0.4178-09	0.1581-08	0.3716-09	0.3282-09	0.2771-09	0.3334-09
170	0.6936-09	0.1046-08	0.2567-09	0.5276-09	0.2190-09	0.1630-09	0.1938-09	0.2383-09
180	0.1763-09	0.4016-09	0.8089-10	0.1517-09	0.6916-10	0.5092-10	0.6550-10	0.6101-10

0.1122-07 = 1.122×10^{-8}

TABLE XI . ANGULAR DISTRIBUTION OF TOTAL SCATTERED NEUTRON FLUX
Separation Distance - 35 Feet

[(neutrons/cm^2-sec)/(source neutron/sec)]

Detector Angular Interval (degrees)	Source Energy (Mev) 0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
10	0.2985-08	0.2676-08	0.1389-08	0.2127-08	0.1773-08	0.1717-08	0.1635-08	0.2094-08
20	0.2951-08	0.2360-08	0.1358-08	0.1799-08	0.1516-08	0.1244-08	0.1454-08	0.1625-08
30	0.2792-08	0.2277-08	0.1235-08	0.1575-08	0.10[illegible]-08	0.1181-08	0.1012-08	0.1104-08
40	0.2375-08	0.[illegible]366-08	0.1069-08	0.1366-08	0.1102-08	0.7425-09	0.7318-09	0.7057-09
50	0.2305-08	0.1944-08	0.1059-08	0.1330-08	0.8109-09	0.7796-09	0.7732-09	0.7480-09
60	0.1927-08	0.1569-08	0.9593-09	0.1119-08	0.8159-09	0.4075-09	0.5154-09	0.6119-09
70	0.1673-08	0.1590-08	0.6521-09	0.1206-08	0.7062-09	0.4730-09	0.4654-09	0.9164-09
80	0.1327-08	0.1447-08	0.9532-09	0.9720-09	0.4575-09	0.4717-09	0.4767-09	0.5055-09
90	0.1611-08	0.1505-08	0.7025-09	0.[illegible]634-09	0.5651-09	0.3683-09	0.2716-09	0.4188-09
100	0.1102-08	0.1408-08	0.5741-09	0.1062-08	0.4786-09	0.3484-09	0.4780-09	0.4521-09
110	0.1096-08	0.1265-08	0.4459-09	0.7177-09	0.4450-09	0.3365-09	0.4454-09	0.3209-09
120	0.6797-09	0.[illegible]925-09	0.3566-09	0.5843-09	0.2619-09	0.1616-09	0.[illegible]314-09	0.1765-09
130	0.4514-09	0.5038-09	0.2533-09	0.3173-09	0.2774-09	0.2194-09	0.2158-09	0.2644-09
140	0.7216-09	0.9110-09	0.2198-09	0.4422-09	0.2467-09	0.2037-09	0.1749-09	0.2244-09
150	0.4854-09	0.6913-09	0.2007-09	0.3452-09	0.1765-09	0.2471-09	0.1463-09	0.1216-09
160	0.2729-09	0.4520-09	0.1413-09	0.1742-09	0.1643-09	0.1101-09	0.[illegible]57-10	0.9820-10
170	0.1773-09	0.3616-09	0.8774-10	0.1512-10	0.1015-09	0.7342-10	0.4004-10	0.5673-10
180	0.9237-10	0.3167-10	0.3210-10	0.3523-10	0.2411-10	0.5485-10	0.1967-10	0.1351-10

0.2985-08 = 2.985x10^{-9}

TABLE XII. ANGULAR DISTRIBUTION OF TOTAL SCATTERED NEUTRON FLUX
Separation Distance - 64 Feet

[(neutrons/cm^2-sec)/(source neutron/sec)]

Detector Angular Interval (degrees)	Source Energy (Mev) 0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
10	0.1445-08	0.1291-08	0.7224-09	0.9822-09	0.9024-09	0.8763-09	0.9245-09	0.1038-08
20	0.1471-08	0.1258-08	0.7040-09	0.9266-09	0.8263-09	0.7580-09	0.7338-09	0.8059-09
30	0.1354-08	0.1205-08	0.6535-09	0.8988-09	0.6084-09	0.5679-09	0.5170-09	0.6273-09
40	0.1227-08	0.1053-08	0.7814-09	0.7660-09	0.4580-09	0.4663-09	0.4173-09	0.4947-09
50	0.1197-08	0.1483-08	0.5389-09	0.6667-09	0.4799-09	0.3683-09	0.2924-09	0.3868-09
60	0.9442-09	0.8502-09	0.4657-09	0.6445-09	0.4905-09	0.3996-09	0.2795-09	0.3307-09
70	0.8803-09	0.7493-09	0.4568-09	0.6775-09	0.3432-09	0.4103-09	0.2697-09	0.3153-09
80	0.7880-09	0.7260-09	0.3556-09	0.5836-09	0.3365-09	0.2565-09	0.2316-09	0.2662-09
90	0.7578-09	0.7211-09	0.3576-09	0.4172-09	0.3130-09	0.2395-09	0.2201-09	0.2362-09
100	0.8749-09	0.7095-09	0.3246-09	0.4547-09	0.2456-09	0.1967-09	0.1941-09	0.2209-09
110	0.4348-09	0.5938-09	0.2234-09	0.3945-09	0.3394-09	0.2041-09	0.2091-09	0.2036-09
120	0.3927-09	0.5000-09	0.1668-09	0.3110-09	0.2117-09	0.1276-09	0.1103-09	0.1441-09
130	0.4509-09	0.4402-09	0.1792-09	0.4457-09	0.1355-09	0.1134-09	0.1249-09	0.1029-09
140	0.2934-09	0.3820-09	0.1375-09	0.2467-09	0.2125-09	0.1329-09	0.8959-10	0.1328-09
150	0.2774-09	0.3095-09	0.1218-09	0.2373-09	0.1108-09	0.8370-10	0.7040-10	0.7653-10
160	0.1457-09	0.2258-09	0.8080-10	0.1199-09	0.7332-10	0.1272-09	0.6981-10	0.4807-10
170	0.8827-10	0.1403-09	0.2718-09	0.9904-10	0.6227-10	0.3386-10	0.2708-10	0.2823-10
180	0.2796-10	0.2962-10	0.9061-11	0.2091-10	0.9400-11	0.1831-10	0.7067-11	0.9389-11

0.1445-08 = 1.445x10^{-9}

TABLE XIII. ANGULAR DISTRIBUTION OF TOTAL SCATTERED NEUTRON FLUX
Separation Distance - 100 Feet

[(neutrons/cm^2-sec)/(source neutron/sec)]

Detector Angular Interval (degrees)	Source Energy (Mev) 0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
10	0.8001-09	0.7332-09	0.4817-09	0.5678-09	0.5540-09	[illegible]	0.1163-09	0.6296-09
20	0.7540-09	0.7368-09	0.4401-09	0.5687-09	0.4634-09	[illegible]	0.1165-09	0.5366-09
30	0.8856-09	0.6932-09	0.3929-09	0.5655-09	0.4337-09	[illegible]	[illegible]376-09	0.3771-09
40	0.8172-09	0.7226-09	0.3519-09	0.5679-09	0.3525-09	0.7742-09	[illegible]468-09	0.3231-09
50	0.6620-09	0.5291-09	0.3279-09	0.4656-09	0.2712-09	0.2145-09	[illegible]392-09	0.2969-09
60	[illegible]	0.5719-09	0.2800-09	0.4267-09	0.2382-09	[illegible]	[illegible]688-09	0.1892-09
70	0.4812-09	0.5053-09	0.2303-09	0.3089-09	0.2321-09	0.2135-09	0.1592-09	0.2140-09
80	0.4513-09	0.6461-09	0.2714-09	0.3104-09	0.1672-09	0.1745-09	0.1783-09	0.1644-09
90	0.2168-09	0.6617-09	0.1821-09	0.3280-09	0.1889-09	0.1419-09	0.1279-09	0.1456-09
100	[illegible]158-09	0.3602-09	0.1781-09	0.2713-09	0.2723-09	0.1404-09	0.1152-09	0.1556-09
110	0.3130-09	0.3872-09	0.1604-09	0.2530-09	0.1558-09	0.9845-10	0.1116-09	0.1325-09
120	0.2016-09	0.2593-09	0.1174-09	0.2039-09	0.1255-09	0.1100-09	0.8729-10	0.7965-10
130	0.1427-09	0.7487-09	0.1680-09	0.1733-09	0.1267-09	0.7285-10	0.7919-10	0.6276-10
140	0.1370-09	0.2123-09	0.7924-10	0.1819-09	0.8017-10	0.7069-10	0.6165-10	0.6496-10
150	0.1366-09	0.1586-09	0.1072-09	0.1391-09	0.6165-10	0.1030-09	0.3765-10	0.4268-10
160	0.1936-09	0.1358-09	0.7867-10	0.8375-10	0.4968-10	0.4762 10	0.4451-10	0.3769-10
170	0.2688-10	0.7035-10	0.2268-10	0.5032-10	0.2927-10	0.3042-10	0.1647-10	0.2034-10
180	0.8696-11	0.1351-10	0.4655-11	0.8472-11	0.4668-11	0.5584-11	0.5059-11	0.4775-11

0.8001-09 = 8.001x10^{-10}

TABLE XIV. ANGULAR DISTRIBUTION OF TOTAL SCATTERED NEUTRON DOSE RATE
Separation Distance - 10 Feet

[(rem/hr)/(source-neutron/sec)]

Detector Angular Interval (degrees)	Source Energy (Mev) 0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
10	0.5081-12	0.8819-12	0.6244-12	0.1066-11	0.1107-11	0.1208-11	0.1037-11	0.1895-11
20	0.4753-12	0.9263-12	0.5761-12	0.1107-11	0.9700-12	0.1009-11	0.1158-11	0.1397-11
30	0.4011-12	0.7667-12	0.6691-12	0.8460-12	0.8641-12	0.6956-12	0.7675-12	0.9061-12
40	0.4031-12	0.5843-12	0.5221-12	0.7422-12	0.6423-12	0.5414-12	0.6126-12	0.7987-12
50	0.3057-12	0.5947-12	0.3803-12	0.7978-12	0.5852-12	0.5229-12	0.4304-12	0.6745-12
60	0.2623-12	0.5226-12	0.4769-12	0.6847-12	0.3725-12	0.3071-12	0.4812-12	0.5572-12
70	0.2681-12	0.4548-12	0.2768-12	0.5105-12	0.4167-12	0.2772-12	0.3119-12	0.5404-12
80	0.2301-12	0.5128-12	0.2261-12	0.5662-12	0.3934-12	0.2882-12	0.2323-12	0.3458-12
90	0.2433-12	0.3680-12	0.3172-12	0.4471-12	0.4022-12	0.2329-12	0.4491-12	0.2407-12
100	0.1693-12	0.3945-12	0.2182-12	0.4727-12	0.2447-12	0.1987-12	0.3573-12	0.4236-12
110	0.1632-12	0.4076-12	0.2212-12	0.3632-12	0.1866-12	0.3241-12	0.2883-12	0.1763-12
120	0.1296-12	0.3516-12	0.1386-12	0.4480-12	0.2994-12	0.1513-12	0.2211-12	0.2022-12
130	0.7431-13	0.1983-12	0.1317-12	0.2082-12	0.7225-13	0.1312-12	0.1202-12	0.1130-12
140	0.6860-13	0.2615-12	0.7394-13	0.2638-12	0.1281-12	0.1009-12	0.9248-13	0.1325-12
150	0.6438-13	0.1999-12	0.1038-12	0.2137-12	0.1012-12	0.1022-12	0.1101-12	0.1396-12
160	0.4147-13	0.1734-12	0.5185-13	0.2494-12	0.6592-13	0.6272-13	0.5757-13	0.7834-13
170	0.3001-13	0.9408-13	0.3304-13	0.8504-13	0.3755-13	0.3052-13	0.4160-13	0.5405-13
180	0.8234-14	0.3679-13	0.1064-13	0.2444-13	0.1219-13	0.9518-14	0.1426-13	0.1435-13

$0.5081\text{-}12 = 5.081 \times 10^{-13}$

TABLE XV ANGULAR DISTRIBUTION OF TOTAL SCATTERED NEUTRON DOSE RATE
Separation Distance - 35 Feet

[(rem/hr)/(source-neutron/sec)]

Detector Angular Interval (degrees)	Source Energy (Mev) 0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
10	0.1321-12	0.2368-12	0.1742-12	0.3433-12	0.3207-12	0.3416-12	0.3611-12	0.5010-12
20	0.1267-12	0.2040-12	0.1690-12	0.2909-12	0.2732-12	0.2448-12	0.3111-12	0.3865-12
30	0.1183-12	0.1958-12	0.1528-12	0.2536-12	0.1960-12	0.2318-12	0.2171-12	0.2613-1[illegible]
40	0.9623-13	0.1978-12	0.1306-12	0.2200-12	0.1985-12	0.1434-12	0.1547-12	[illegible].1771-12
50	0.8775-13	0.1603-12	0.1283-12	0.2134-12	0.1411-12	0.1[illegible]1-1[illegible]	0.[illegible]623-12	0.1727-12
60	0.7571-13	0.1295-12	0.1162-12	0.1791-12	0.[illegible]09[illegible]-1[illegible]	0.1131-12	0.1085-12	0.1234-12
70	0.6514-13	0.1266-12	0.8287-13	0.1[illegible]9-1[illegible]	0.[illegible]073-12	0.8911-13	[illegible]689-13	0.2043-12
80	0.4986-13	0.1166-12	0.115[illegible]-1[illegible]	0.1540-12	0.8634-13	[illegible].8579-13	0.8394-13	0.1135-12
90	0.5749-13	0.11[illegible]-12	0.8685-13	0.1378-1[illegible]	0.[illegible]464-13	0.7027-13	0.7580-13	0.9541-13
100	0.41[illegible]5-13	0.1113-12	0.[illegible]27[illegible]-1[illegible]	0.1706-12	0.7381-13	0.7131-13	0.7308-13	0.1039-12
110	0.3828-13	0.17[illegible]8-12	0.5286-13	0.1119-12	0.7852-13	0.6469-13	0.6822-13	0.6996-13
120	0.[illegible]427-1[illegible]	0.7837-13	0.4260-13	0.8975-13	0.4535-13	0.4682-13	0.4428-13	0.6531-13
130	0.1657-13	0.7101-13	0.3418-13	0.7879-13	0.4846-13	0.4258-13	0.4415-13	0.5542-13
140	0.2390-13	0.7415-13	0.2626-13	0.6989-13	0.4268-13	0.3876-13	0.3611-13	0.5344-13
150	0.1720-13	0.5562-13	0.2425-13	0.5461-13	0.2757-13	0.3791-13	0.2729-13	0.2734-13
160	0.1028-13	0.3848-13	0.1726-13	0.4381-13	0.2728-13	0.2055-13	0.1460-13	0.2129-13
170	0.7314-14	0.3156-13	0.1096-13	0.2364-13	0.1722-13	0.1180-13	0.9576-14	0.1283-13
180	0.1963-14	0.5607-14	0.3570-14	0.6086-14	0.4169-14	0.4597-14	0.3847-14	0.3130-14

0.1321-12 = 1.321×10^{-13}

TABLE XVI. ANGULAR DISTRIBUTION OF TOTAL SCATTERED NEUTRON DOSE RATE
Separation Distance - 64 Feet

[(rem/hr)/(source-neutron/sec)]

Detector Angular Interval (degrees)	Source Energy (Mev) 0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
10	0.6339-13	0.1139-12	0.9070-13	0.1590-12	0.1629-12	0.1740-12	0.2025-12	0.2480-12
20	0.6190-13	0.1083-12	0.8733-13	0.1492-12	0.1475-12	0.1477-12	0.1575-12	0.1900-12
30	0.5506-13	0.1018-12	0.8001-13	0.1431-12	0.1074-12	0.1052-12	0.1100-12	0.1408-12
40	0.4814-13	0.9657-13	0.9623-13	0.1220-12	0.8015-13	0.8824-13	0.8603-13	0.1133-12
50	0.4551-13	0.1192-12	0.6508-13	0.1056-12	0.8080-13	0.7030-13	0.5934-13	0.8586-13
60	0.3612-13	0.6865-13	0.5571-13	0.1023-12	0.9529-13	0.7431-13	0.5586-13	0.7199-13
70	0.3144-13	0.5943-13	0.5426-13	0.1056-12	0.5649-13	0.7239-13	0.5247-13	0.7049-13
80	0.2896-13	0.5760-13	0.4274-13	0.9138-13	0.5095-13	0.4725-13	0.4500-13	0.5764-13
90	0.2435-13	0.5700-13	0.4200-13	0.6572-13	0.4857-13	0.4285-13	0.4284-13	0.5190-13
100	0.3028-13	0.5573-13	0.3882-13	0.7146-13	0.4162-13	0.3656-13	0.3792-13	0.4689-13
110	0.1333-13	0.4630-13	0.2631-13	0.6196-13	0.5016-13	0.3711-13	0.3853-13	0.4379-13
120	0.1089-13	0.3853-13	0.1987-13	0.4844-13	0.3566-13	0.2194-13	0.2122-13	0.3022-13
130	0.1534-13	0.3461-13	0.2119-13	0.7027-13	0.2329-13	0.2027-13	0.2317-13	0.2179-13
140	0.1062-13	0.3010-13	0.1629-13	0.3889-13	0.3678-13	0.2371-13	0.1672-13	0.2852-13
150	0.7533-14	0.2470-13	0.1468-13	0.3695-13	0.1809-13	0.1551-13	0.1374-13	0.1625-13
160	0.5139-14	0.1901-13	0.9737-14	0.1896-13	0.1252-13	0.2369-13	0.1348-13	0.1045-13
170	0.3144-14	0.1184-13	0.2856-13	0.1496-13	0.1046-13	0.6111-14	0.4907-14	0.6132-14
180	0.8539-15	0.1549-14	0.1140-14	0.3317-14	0.1428-14	0.7362-14	0.1374-14	0.2003-14

$0.6339\text{-}13 = 6.339 \times 10^{-14}$

TABLE XVII. ANGULAR DISTRIBUTION OF TOTAL SCATTERED NEUTRON DOSE RATE
Separation Distance - 100 Feet

[(rem/hr)/(source-neutron/sec)]

Detector Angular Interval (degrees)	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
10	0.3489-13	0.6425-13	0.5977-13	0.9183-13	0.3936-13	0.1065-12	0.1133-12	0.1492-12
20	0.3275-13	0.6277-13	0.5441-13	0.9085-13	0.8561-13	0.8671-13	0.8934-13	0.1236-12
30	0.3386-13	0.5703-13	0.4814-13	0.9007-13	0.7470-13	0.7770-13	0.7000-13	0.8754-13
40	0.3148-13	0.6118-13	0.4274-13	0.8975-13	0.6046-13	0.5098-13	0.4867-13	0.7169-13
50	0.2214-13	0.4264-13	0.3923-13	0.7317-13	0.4703-13	0.3938-13	0.5355-13	0.6380-13
60	0.2180-13	0.4537-13	0.3343-13	0.6625-13	0.4069-13	0.3298-13	0.3274-13	0.4111-13
70	0.1711-13	0.3995-13	0.2749-13	0.4823-13	0.3608-13	0.3490-13	0.3160-13	0.4635-13
80	0.1314-13	0.4922-13	0.3191-13	0.4825-13	0.2847-13	0.3264-13	0.3416-13	0.3465-13
90	0.1521-13	0.5474-13	0.2129-13	0.5062-13	0.2868-13	0.2321-13	0.3355-13	0.3148-13
100	0.1394-13	0.2814-13	0.2105-13	0.4158-13	0.3608-13	0.2336-13	0.2131-13	0.3318-13
110	0.9133-14	0.2995-13	0.1897-13	0.3962-13	0.2666-13	0.1734-13	0.2100-13	0.2750-13
120	0.6591-14	0.2006-13	0.1376-13	0.3166-13	0.2162-13	0.1713-13	0.1843-13	0.1652-13
130	0.6510-14	0.1982-13	0.1967-13	0.2698-13	0.1937-13	0.1546-13	0.1365-13	0.1311-13
140	0.4228-14	0.1607-13	0.9407-14	0.2727-13	0.1302-13	0.1199-13	0.1137-13	0.1367-13
150	0.3163-14	0.1279-13	0.1245-13	0.2153-13	0.1038-13	0.1829-13	0.7176-14	0.9263-14
160	0.3668-14	0.1072-13	0.9068-14	0.1269-13	0.8464-14	0.5490-14	0.6895-14	0.8015-14
170	0.1990-14	0.5938-14	0.2794-14	0.7959-14	0.4975-14	0.3381-14	0.3108-14	0.4317-14
180	0.3009-15	0.1102-14	0.5746-15	0.1304-14	0.7974-15	0.8988-15	0.9027-15	0.1023-14

0.3489-13 = 3.489×10^{-14}

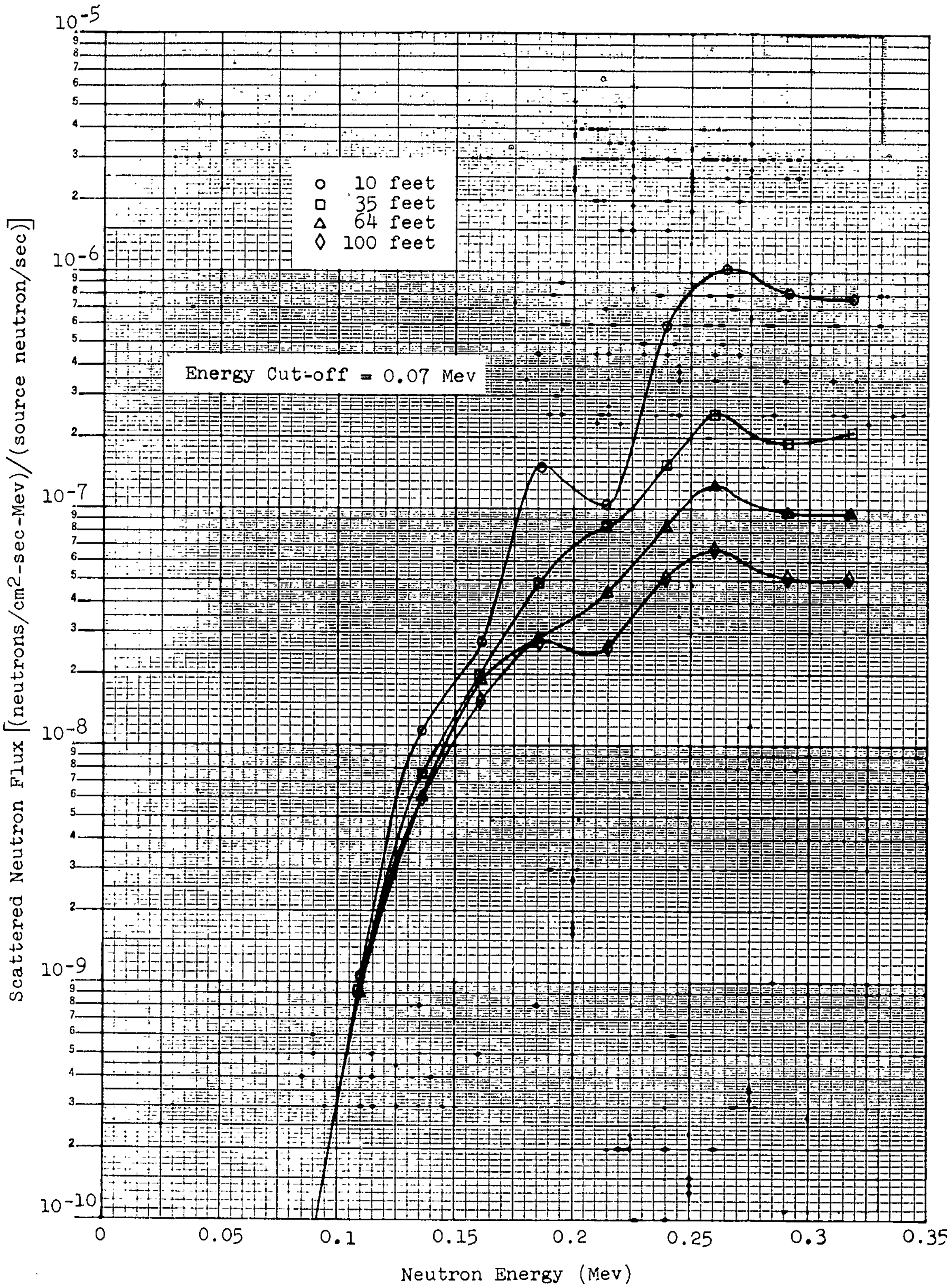

FIGURE 2. ENERGY SPECTRA OF TOTAL SCATTERED NEUTRON FLUX
Initial Energy 0.33 Mev

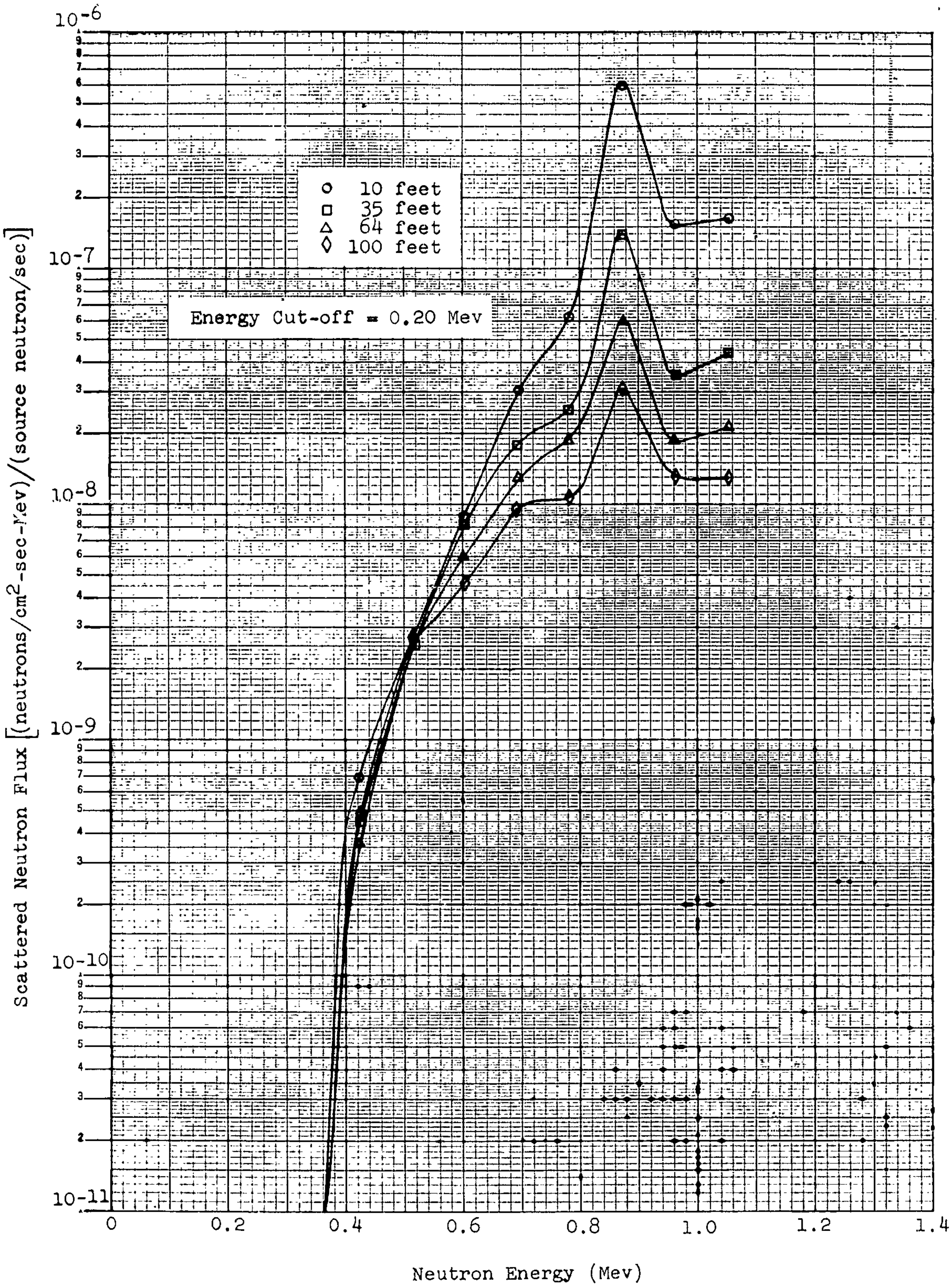

FIGURE 3. ENERGY SPECTRA OF TOTAL SCATTERED NEUTRON FLUX
Initial Energy 1.1 Mev

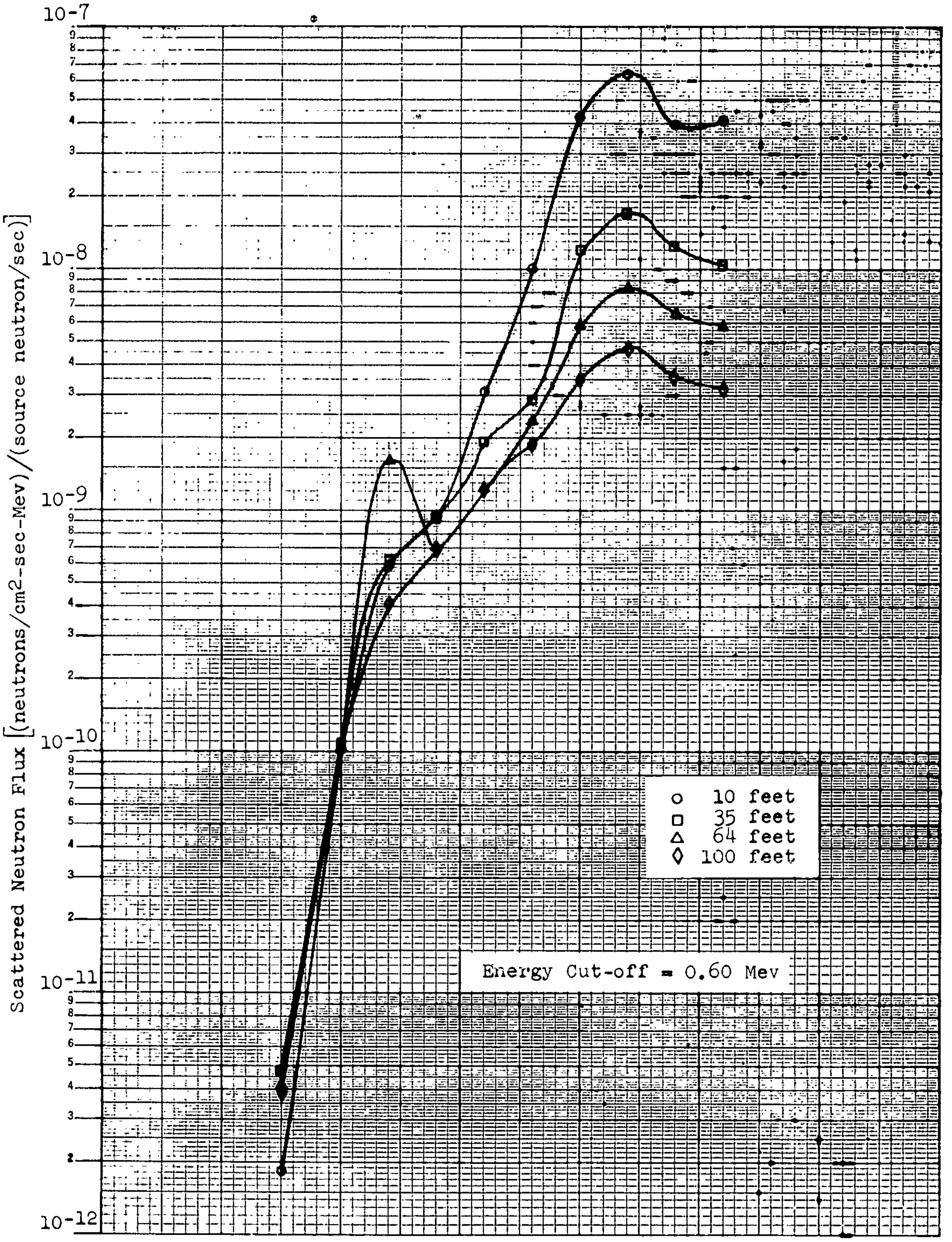

FIGURE 4. ENERGY SPECTRA OF TOTAL SCATTERED NEUTRON FLUX
Initial Energy 2.7 Mev

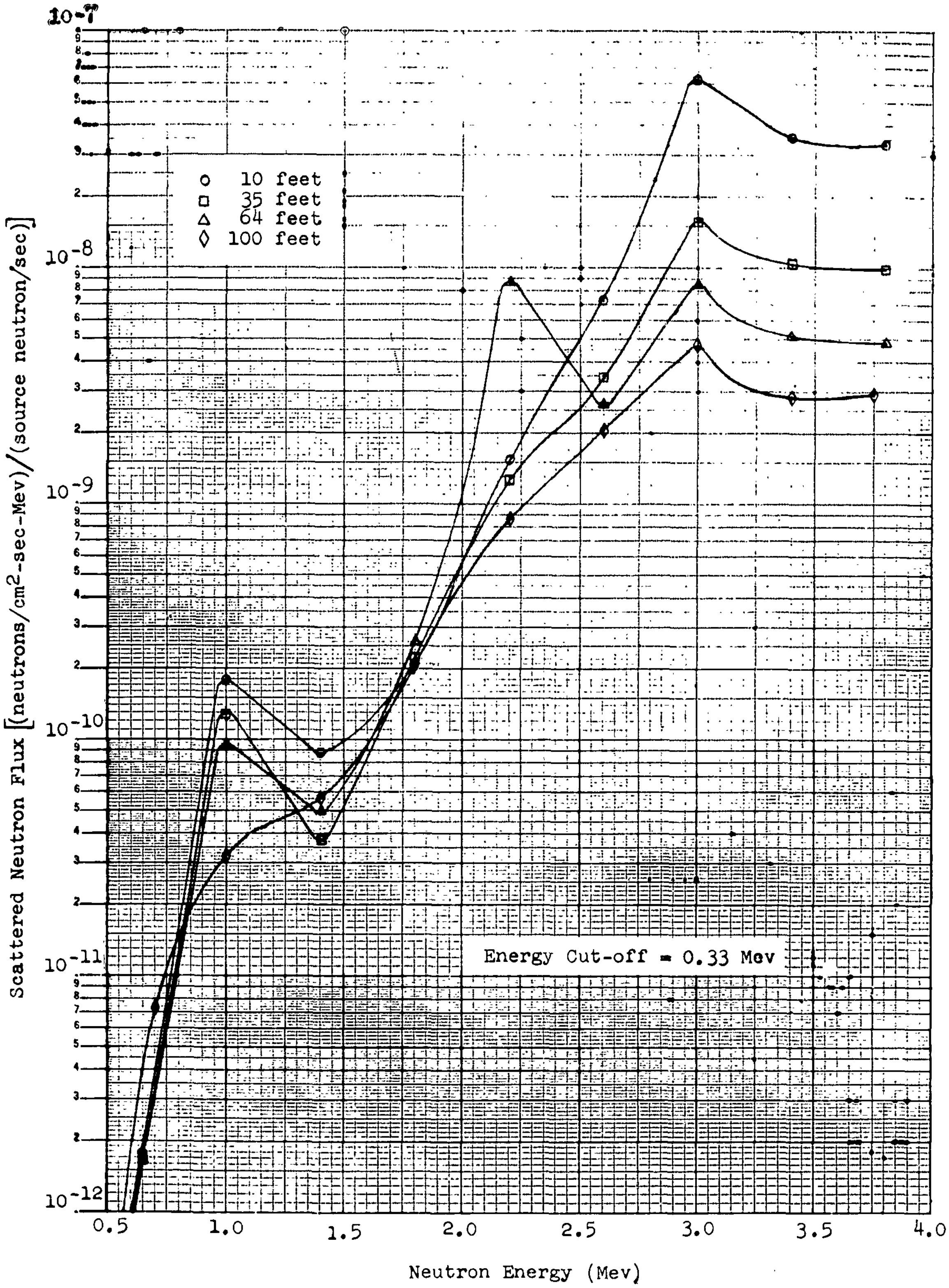

FIGURE 5. ENERGY SPECTRA OF TOTAL SCATTERED NEUTRON FLUX
Initial Energy 4.0 Mev

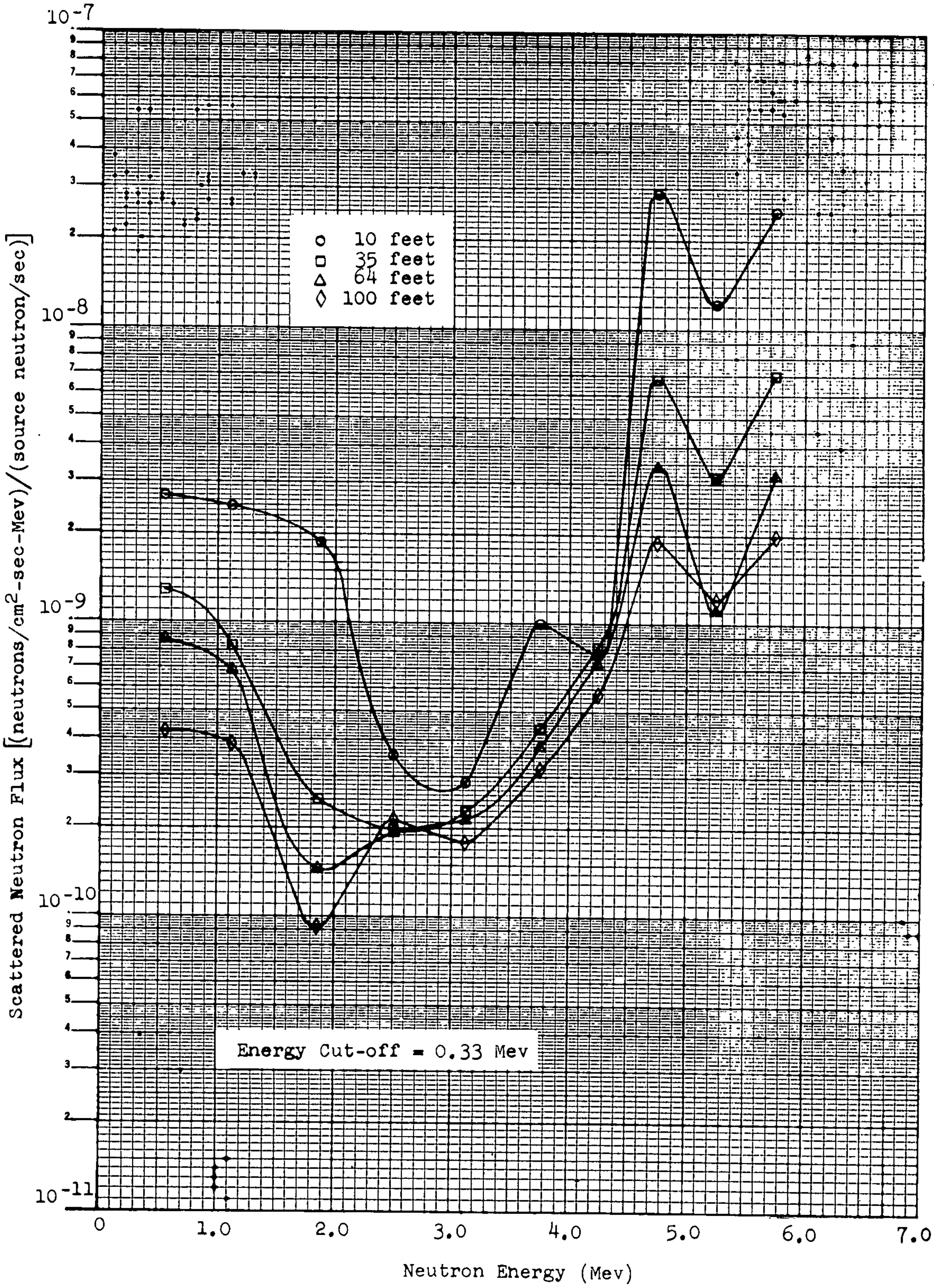

FIGURE 6. ENERGY SPECTRA OF TOTAL SCATTERED NEUTRON FLUX
Initial Energy 6.0 Mev

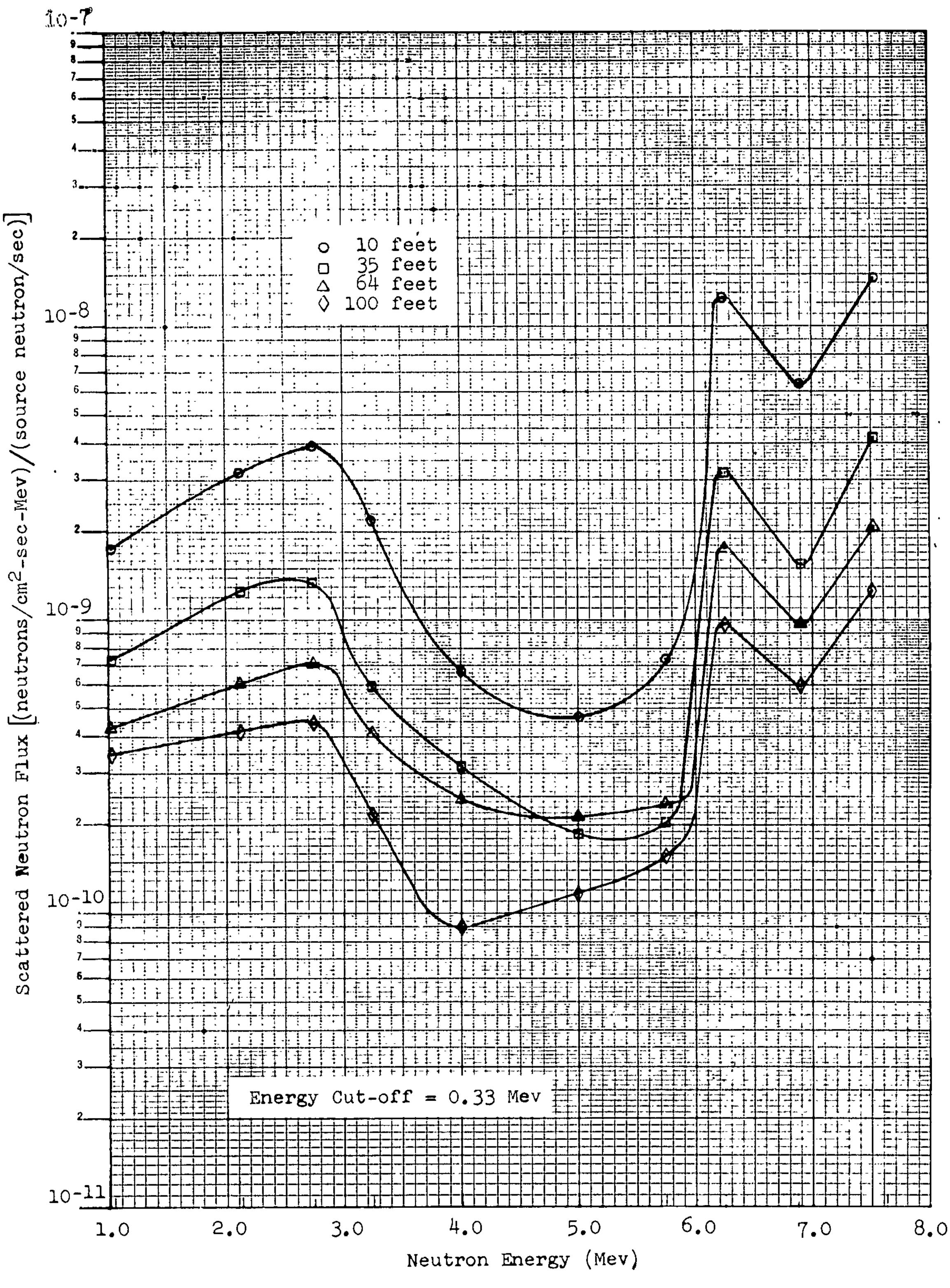

FIGURE 7. ENERGY SPECTRA OF TOTAL SCATTERED NEUTRON FLUX
Initial Energy 8.0 Mev

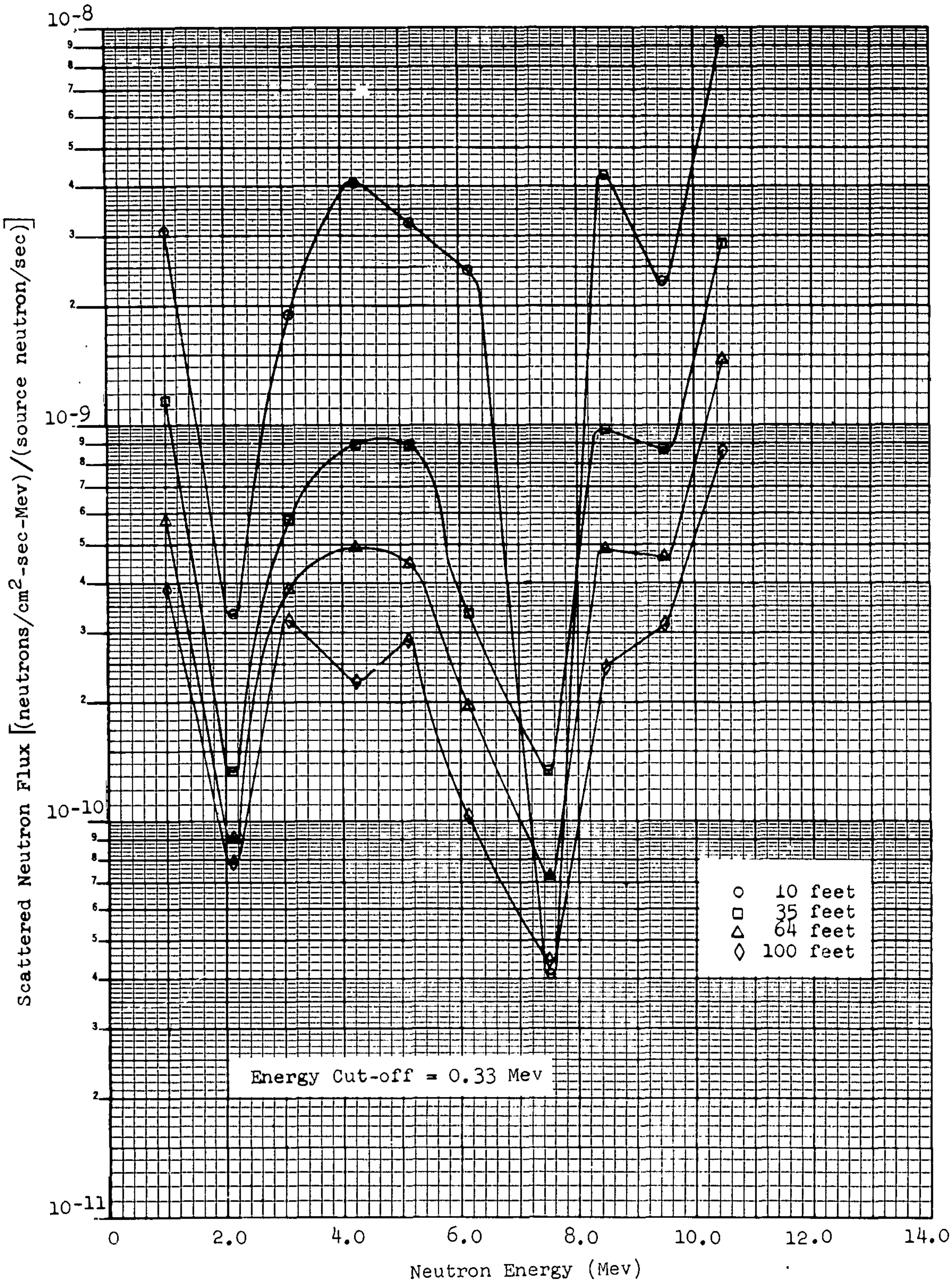

FIGURE 8. ENERGY SPECTRA OF TOTAL SCATTERED NEUTRON FLUX
Initial Energy 10.9 Mev

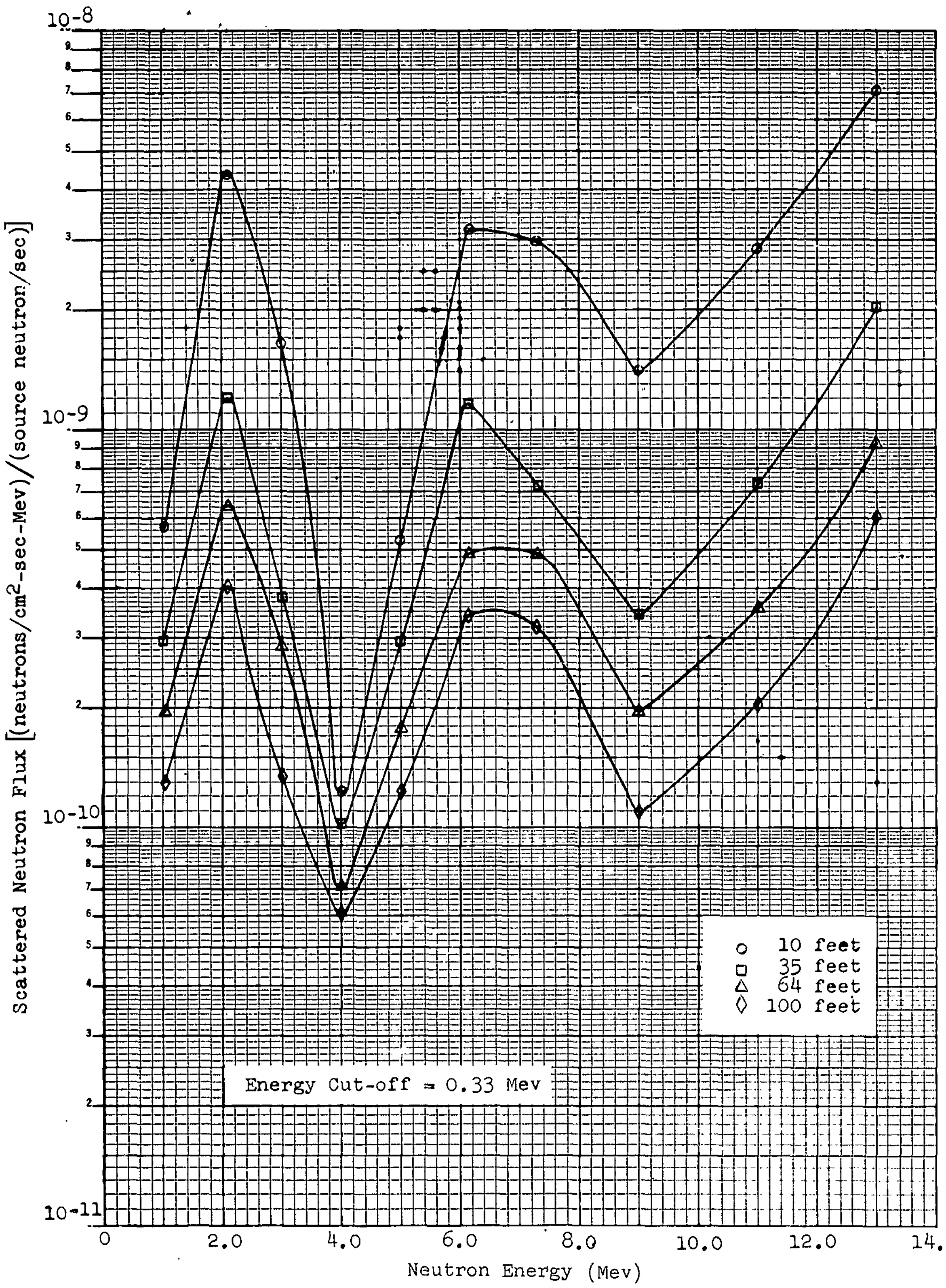

FIGURE 9. ENERGY SPECTRA OF TOTAL SCATTERED NEUTRON FLUX
Initial Energy 14.0 Mev

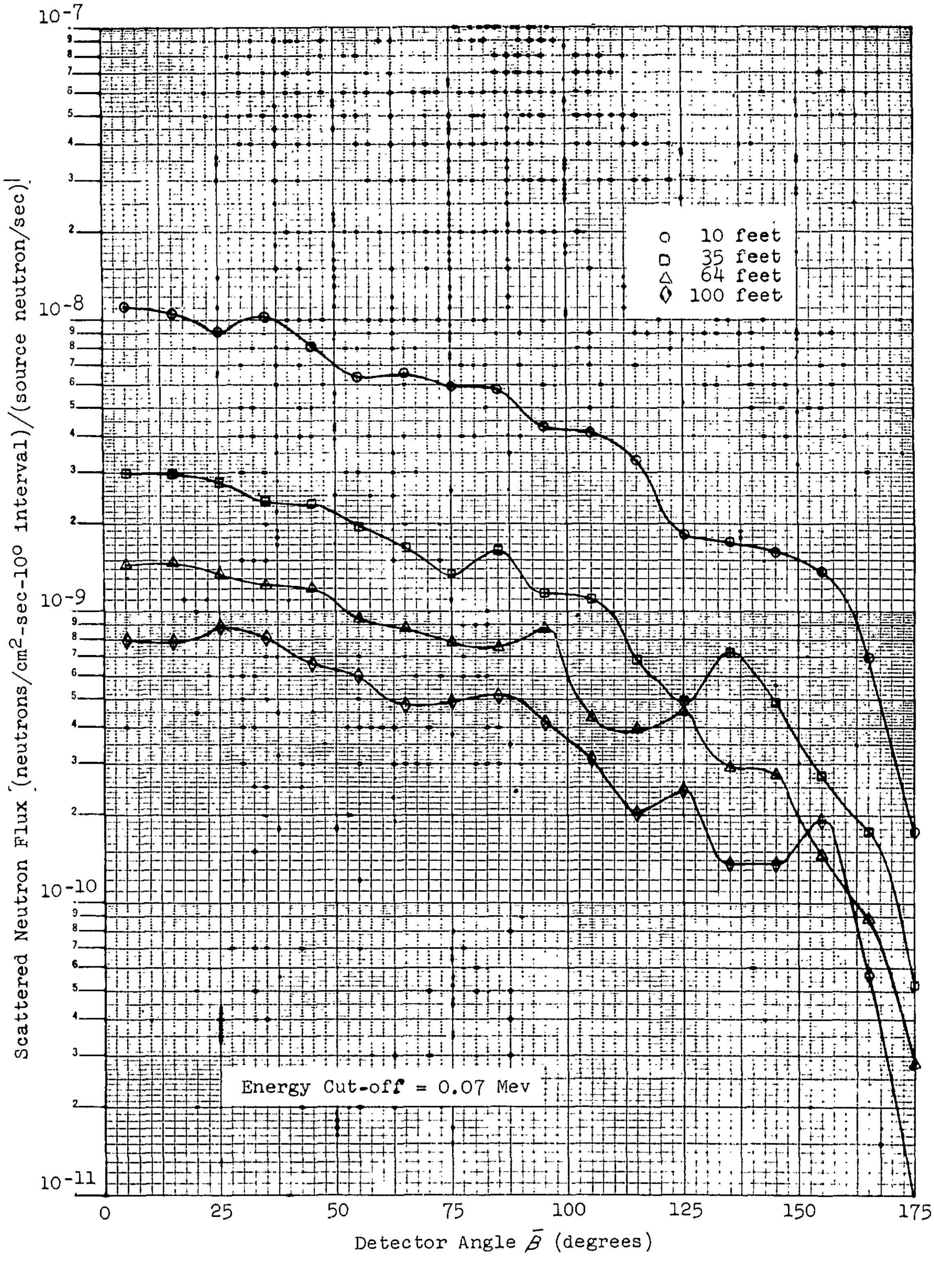

FIGURE 10. TOTAL SCATTERED NEUTRON FLUX VS. DETECTOR ANGLE
Initial Energy 0.33 Mev

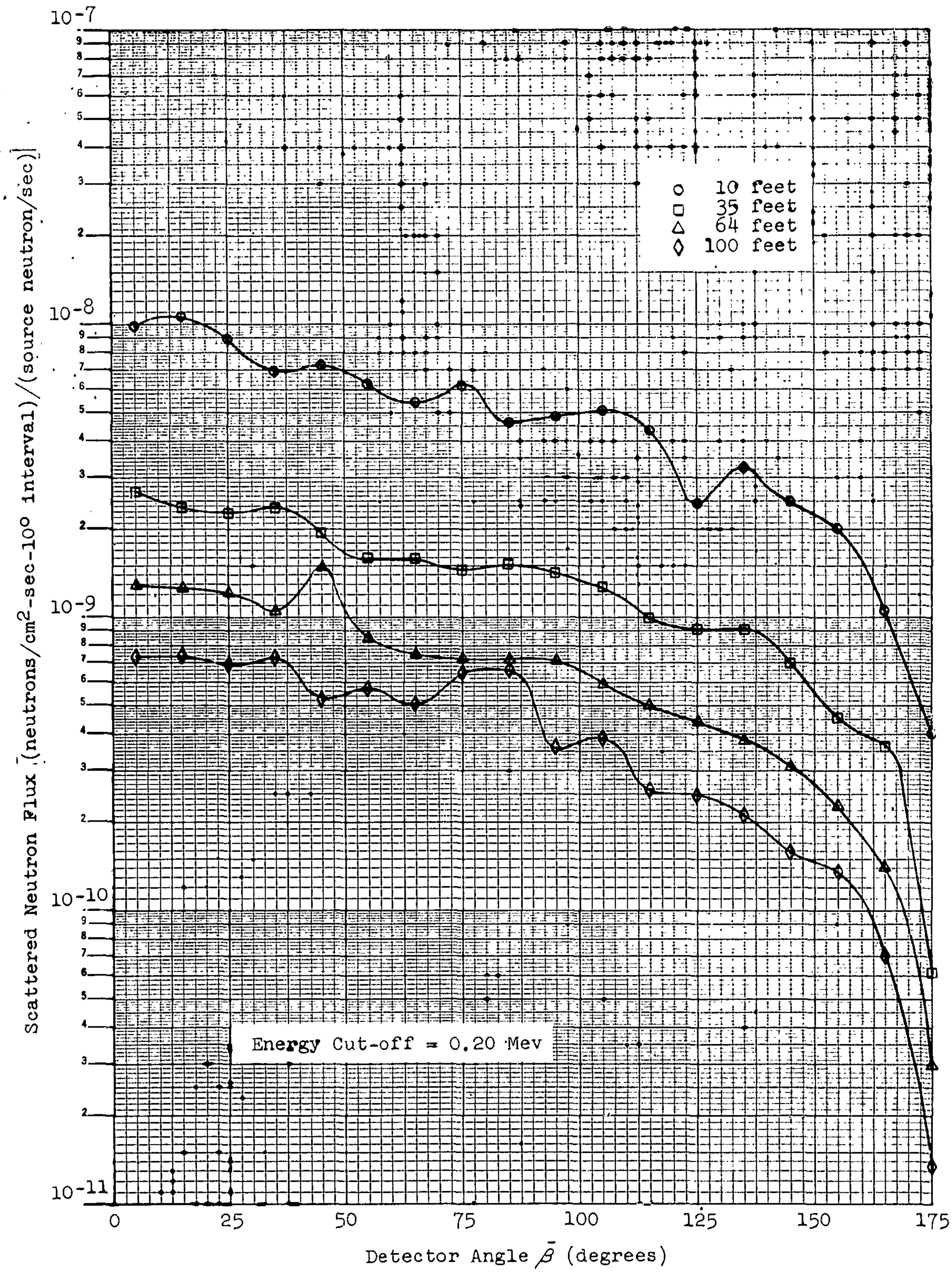

FIGURE 11. TOTAL SCATTERED NEUTRON FLUX VS. DETECTOR ANGLE
Initial Energy 1.1 Mev

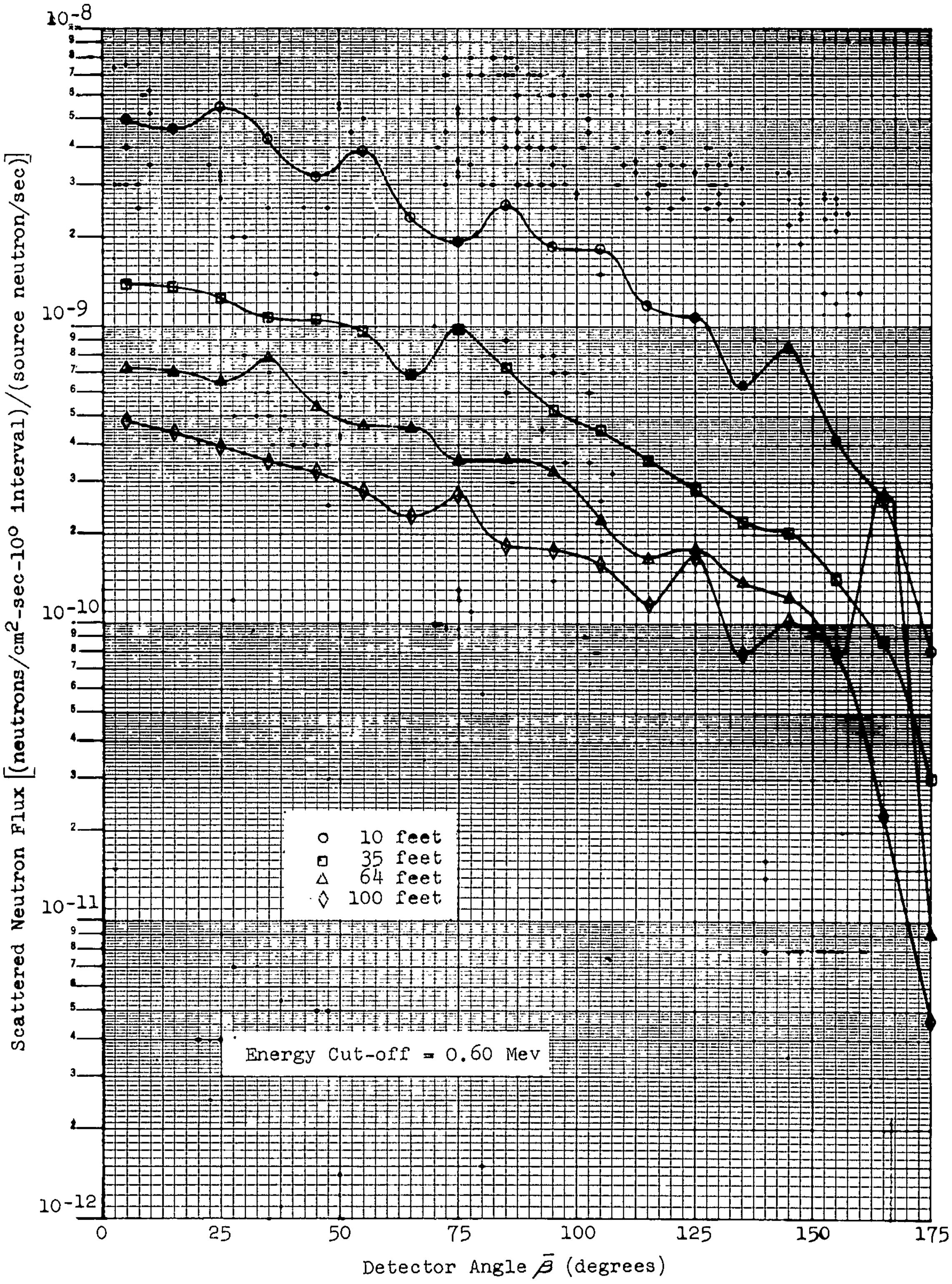

FIGURE 12. TOTAL SCATTERED NEUTRON FLUX VS. DETECTOR ANGLE
Initial Energy 2.7 Mev

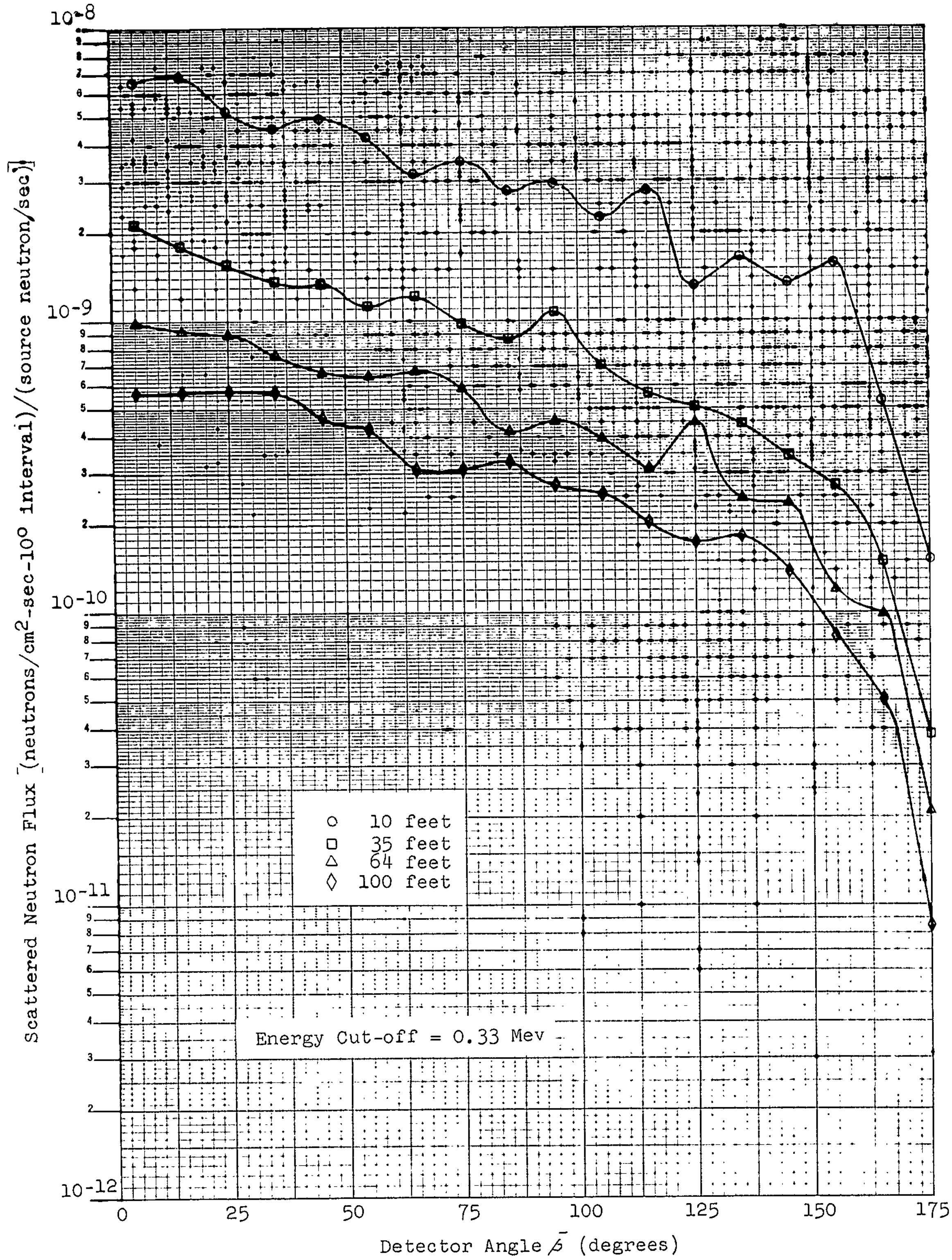

FIGURE 13. TOTAL SCATTERED NEUTRON FLUX VS. DETECTOR ANGLE
Initial Energy 4.0 Mev

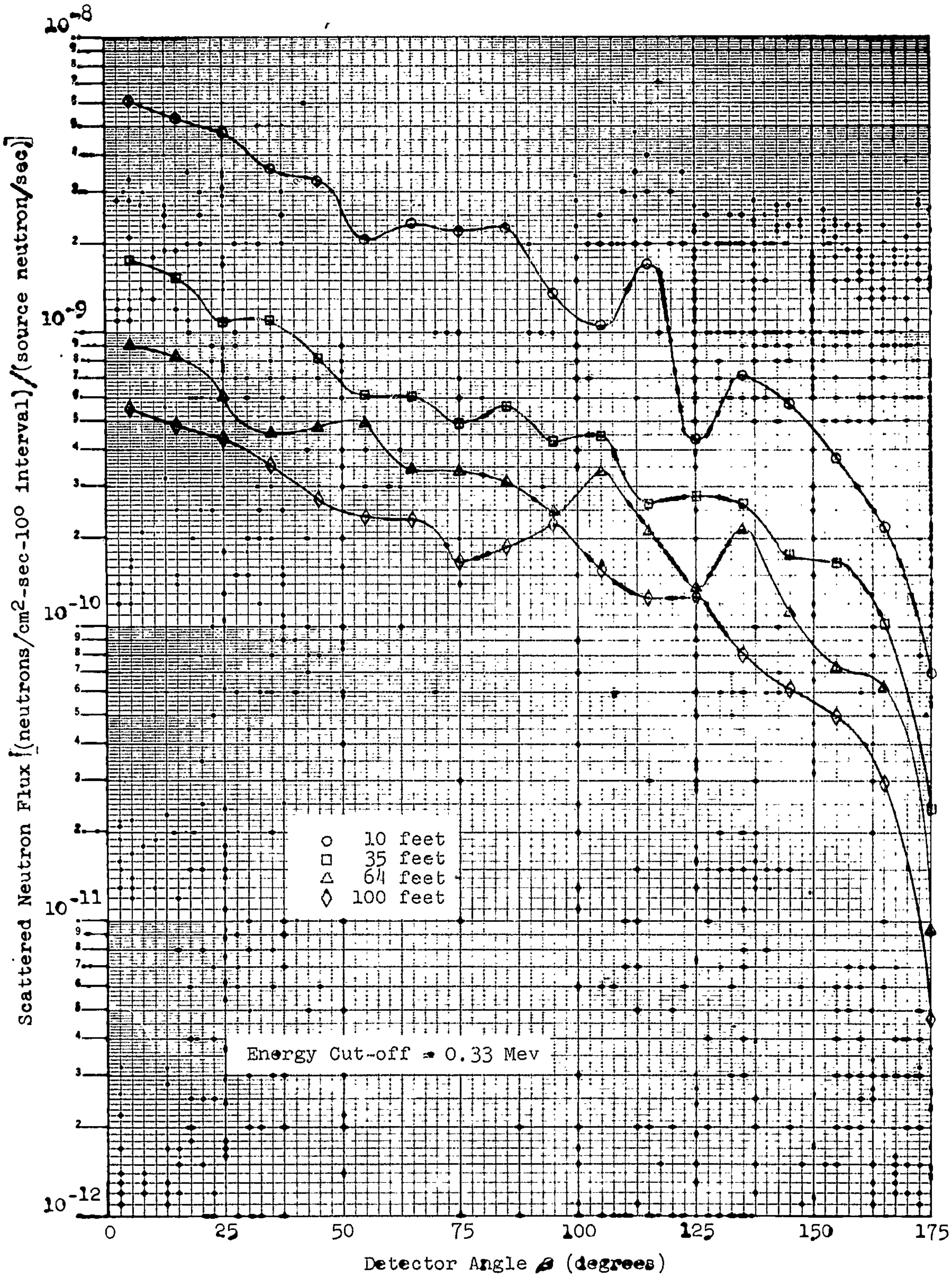

FIGURE 14. TOTAL SCATTERED NEUTRON FLUX VS. DETECTOR ANGLE
Initial Energy 6.0 Mev

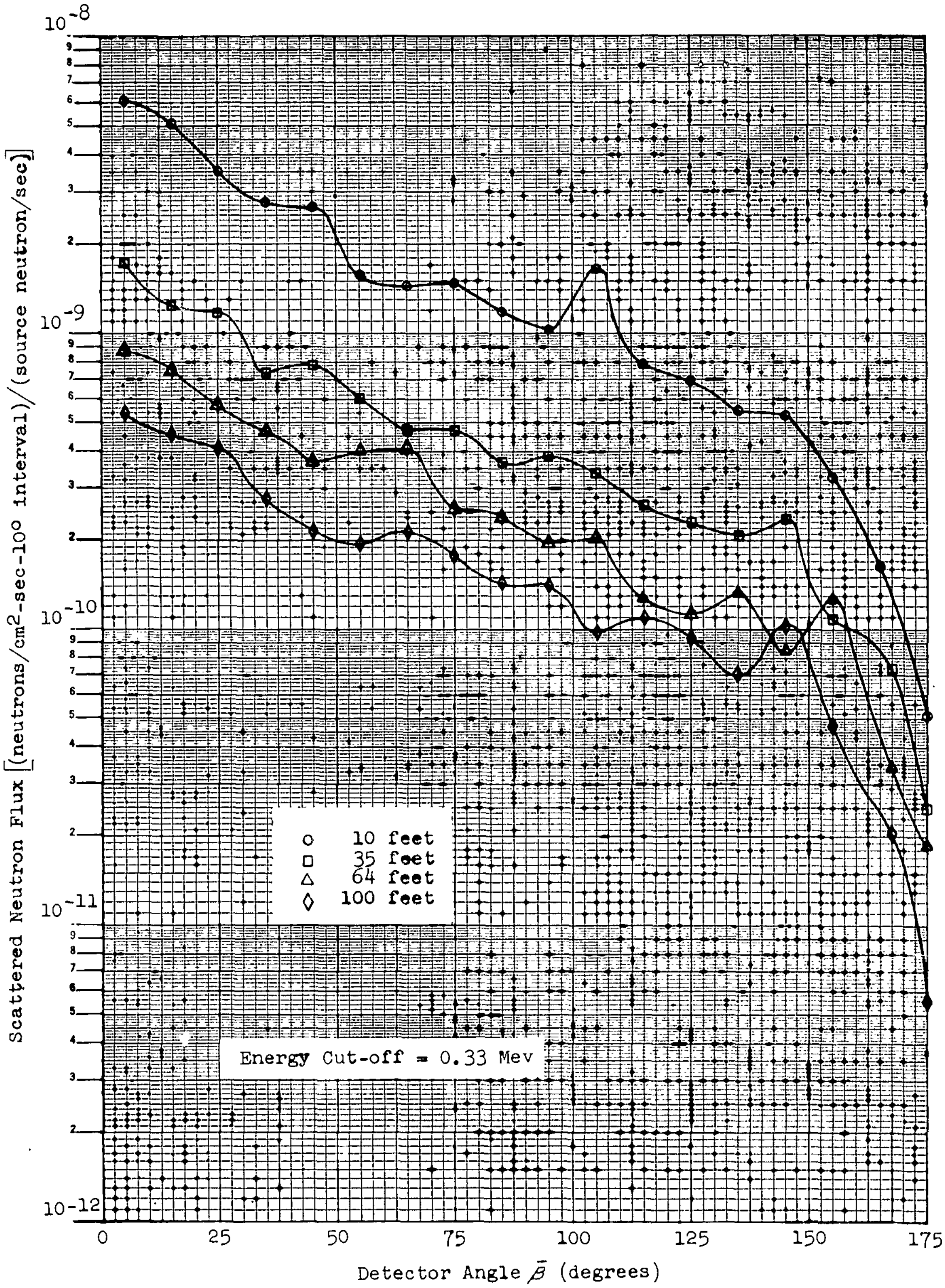

FIGURE 15. TOTAL SCATTERED NEUTRON FLUX VS. DETECTOR ANGLE
Initial Energy 8.0 Mev

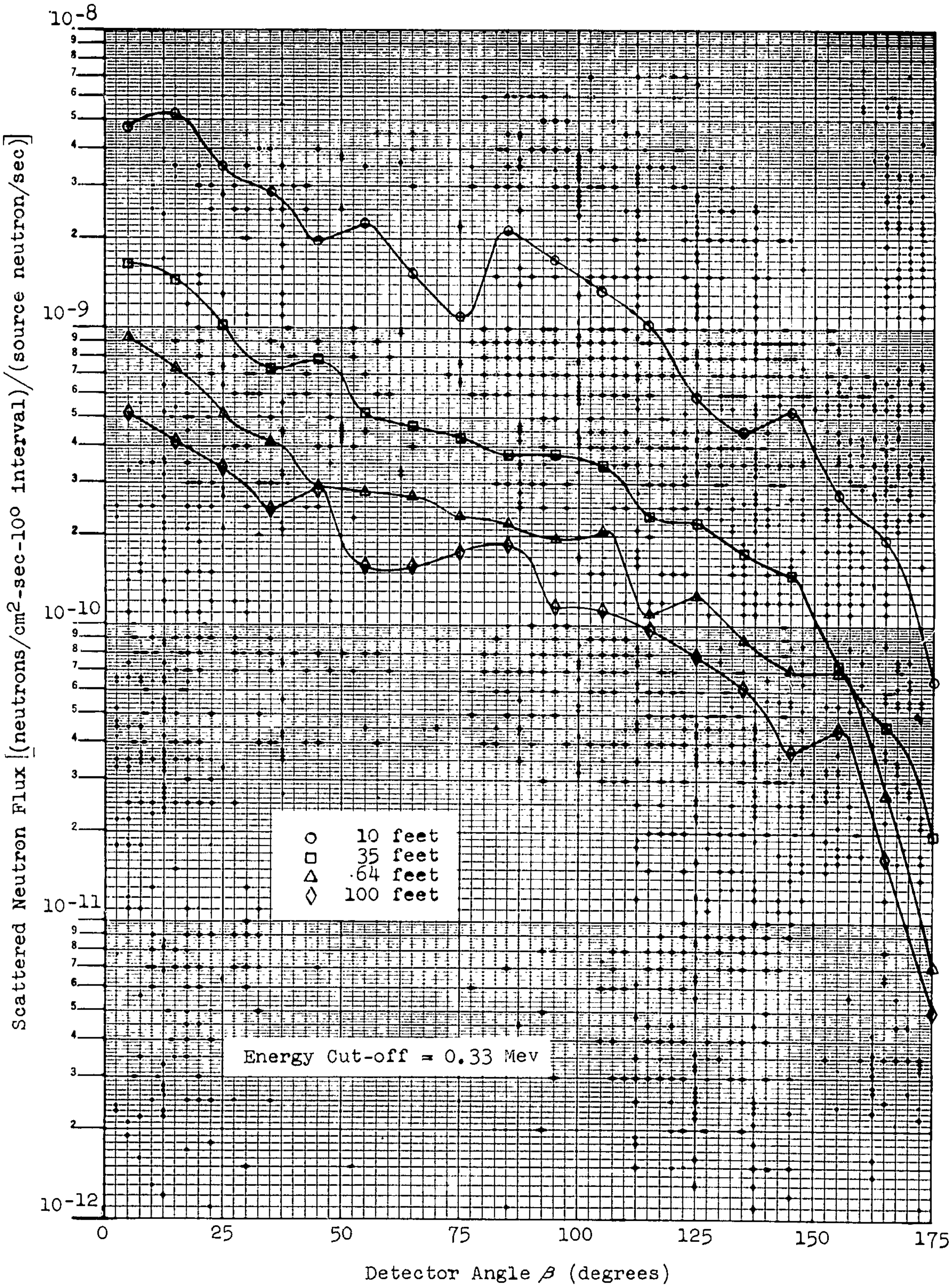

FIGURE 16. TOTAL SCATTERED NEUTRON FLUX VS. DETECTOR ANGLE
Initial Energy 10.9 Mev

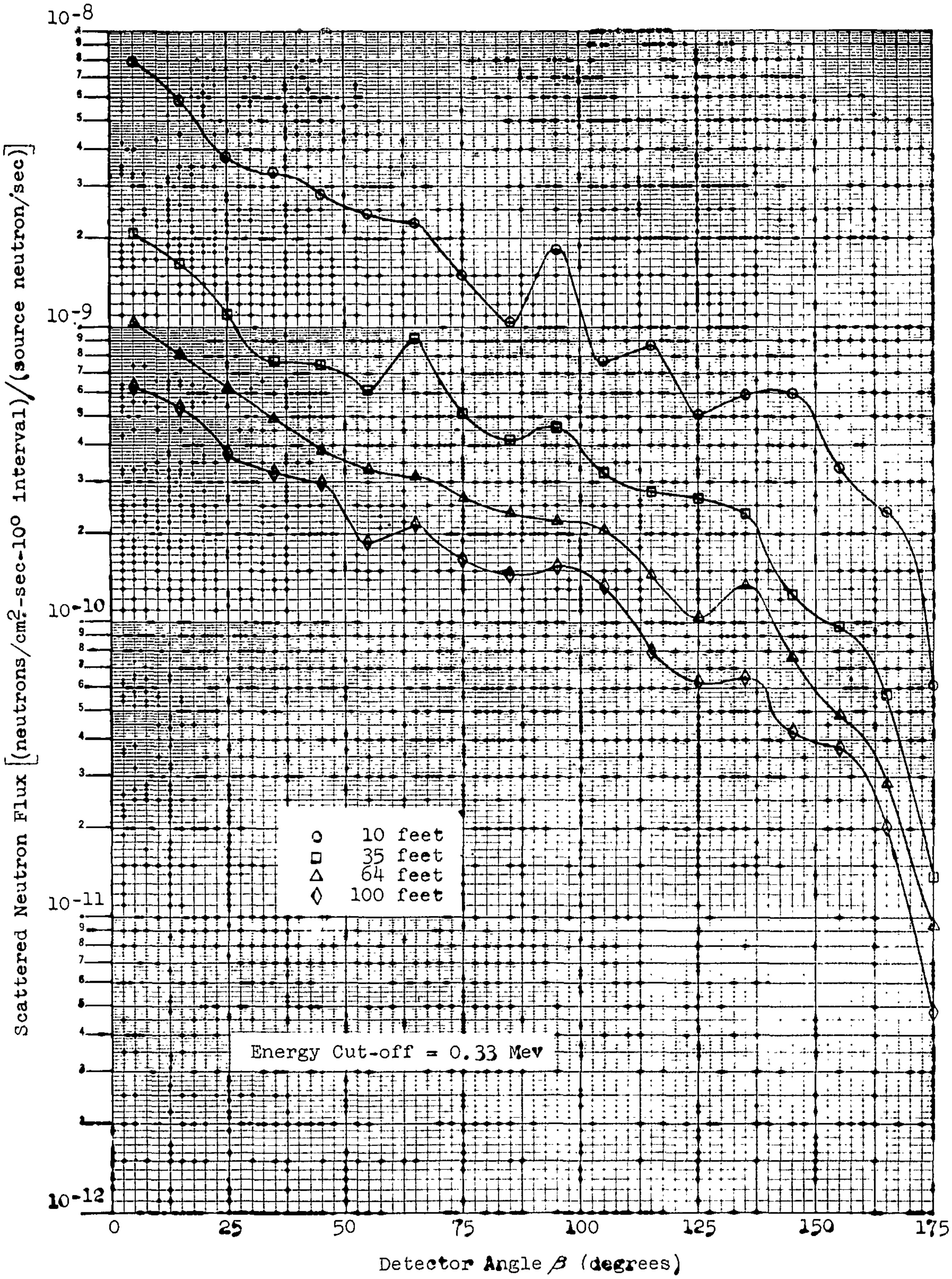

FIGURE 17. TOTAL SCATTERED NEUTRON FLUX VS. DETECTOR A[illegible]
Initial Energy 14.0 Mev

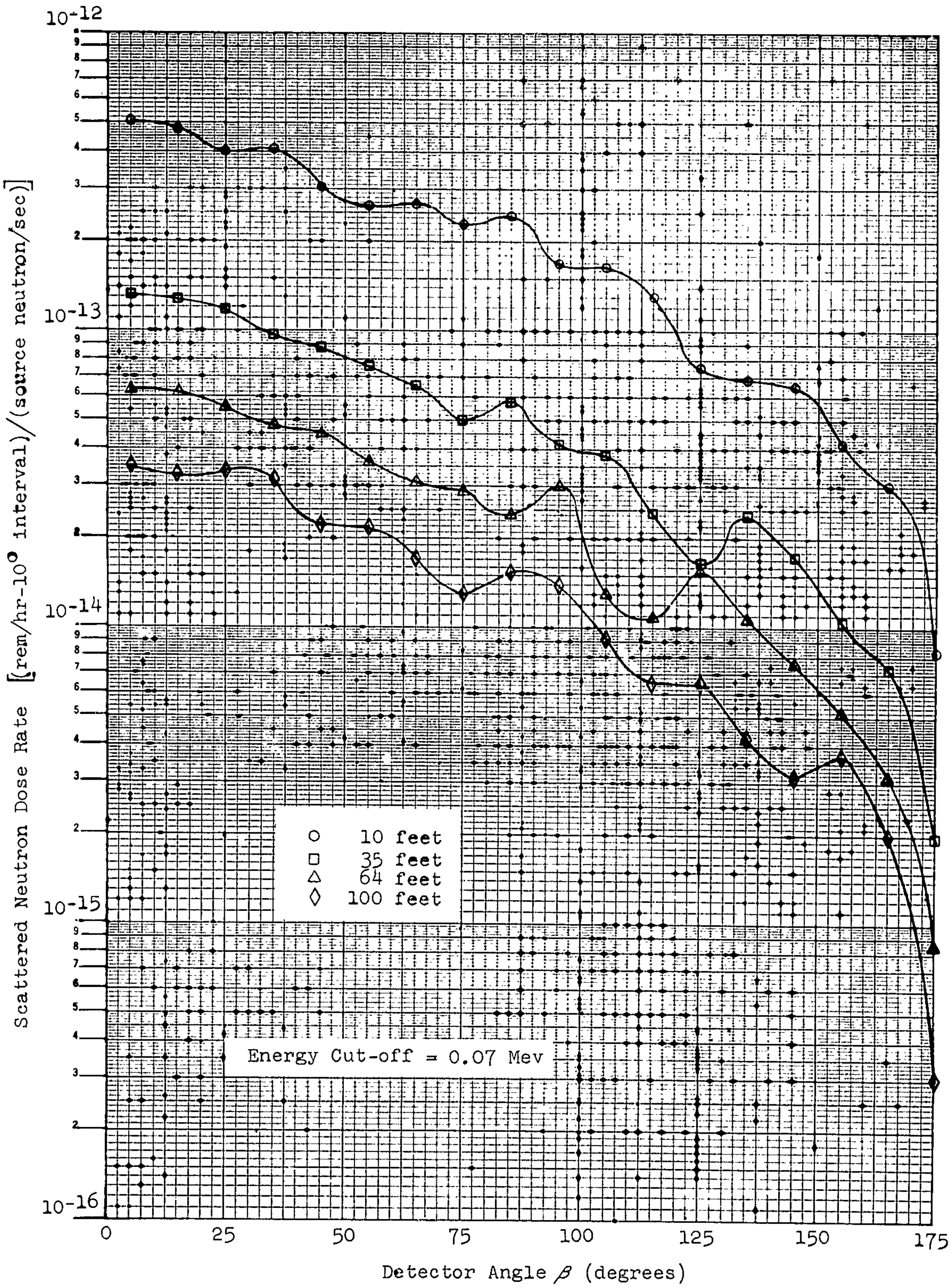

FIGURE 18. TOTAL SCATTERED NEUTRON DOSE RATE VS. DETECTOR ANGLE
Initial Energy 0.33 Mev

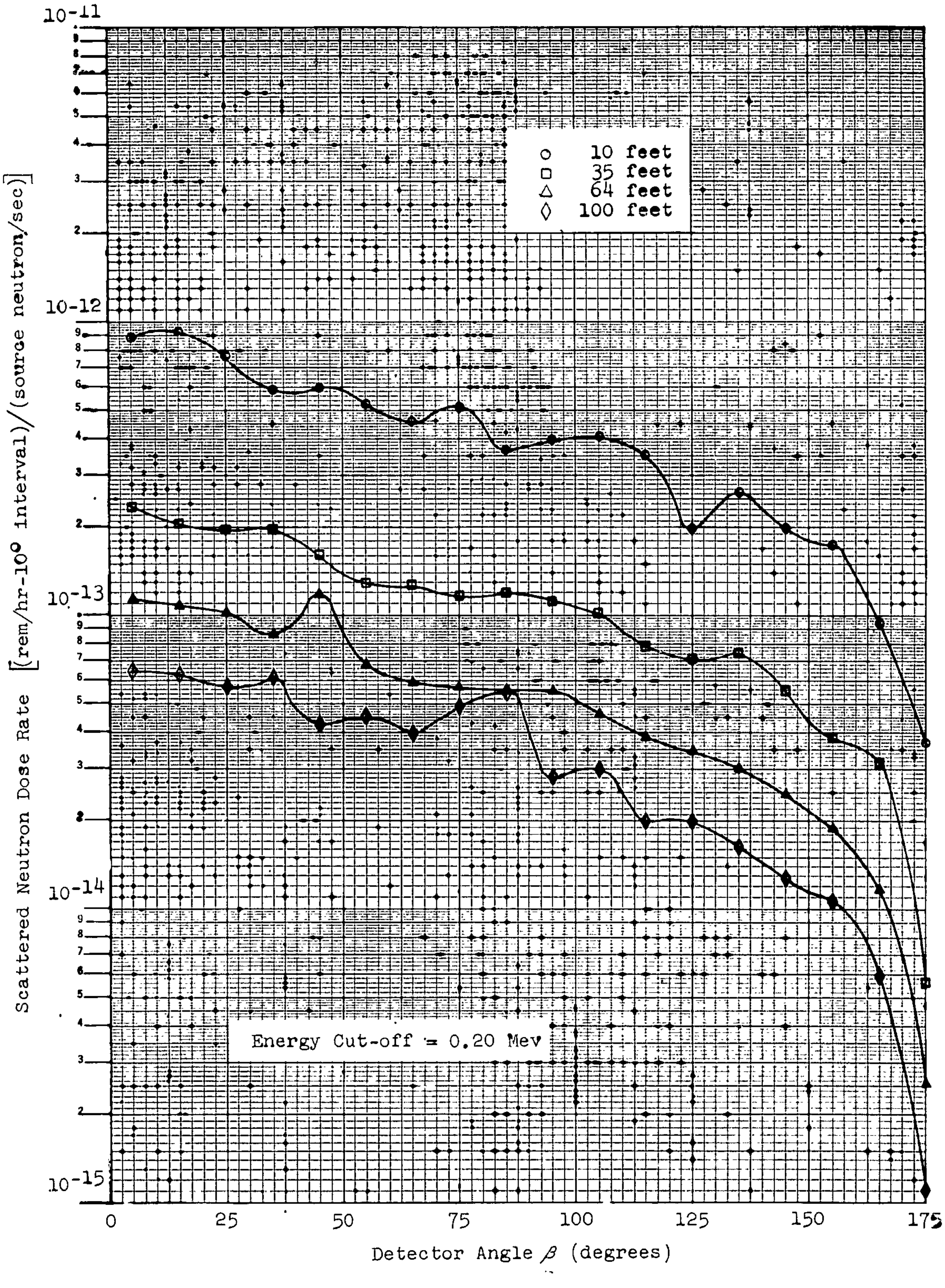

FIGURE 19. TOTAL SCATTERED NEUTRON DOSE RATE VS. DETECTOR ANGLE
Initial Energy 1.1 Mev

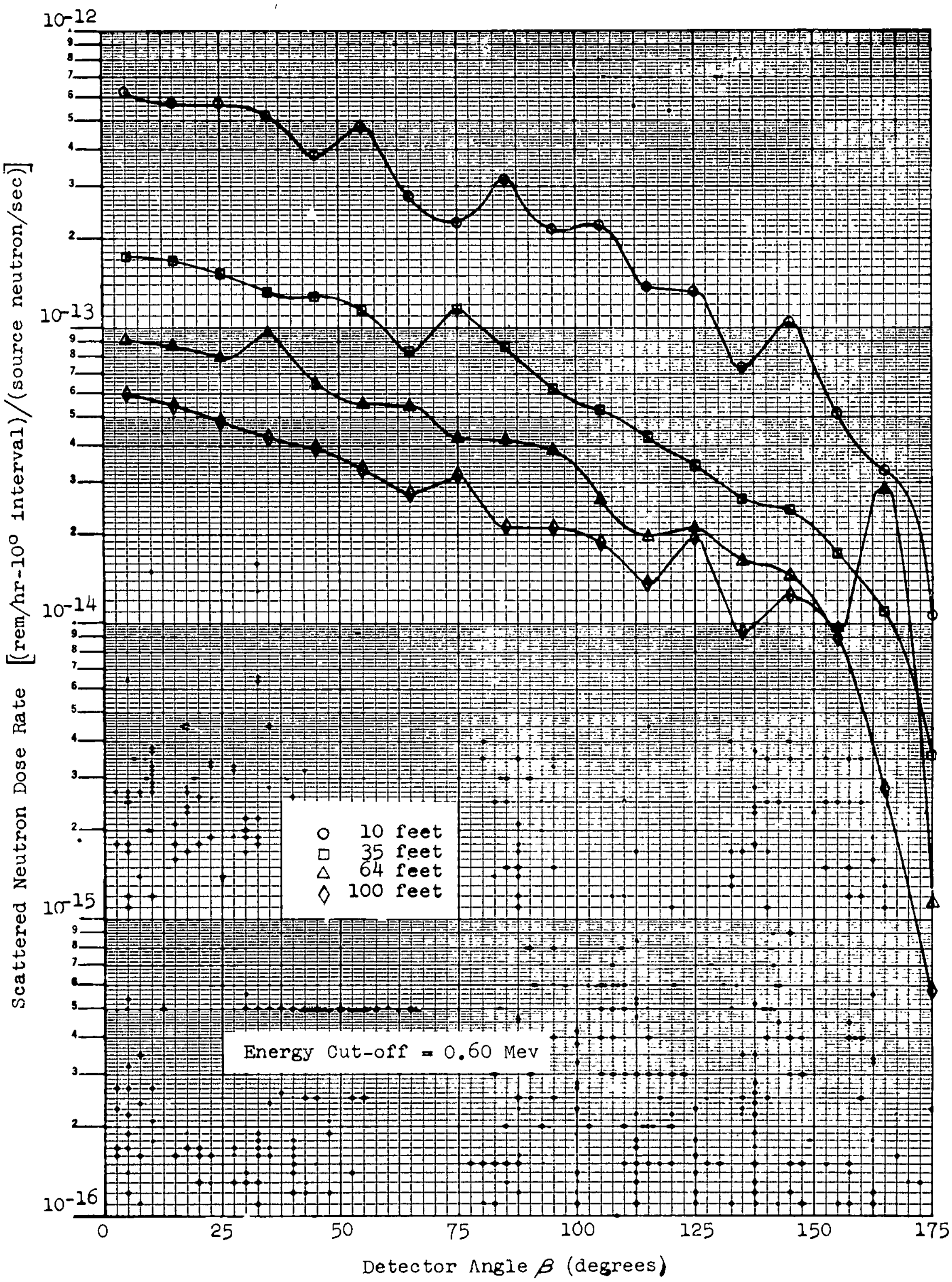

FIGURE 20. TOTAL SCATTERED NEUTRON DOSE RATE VS. D[illegible] ANGLE
Initial Energy 2.7 Mev

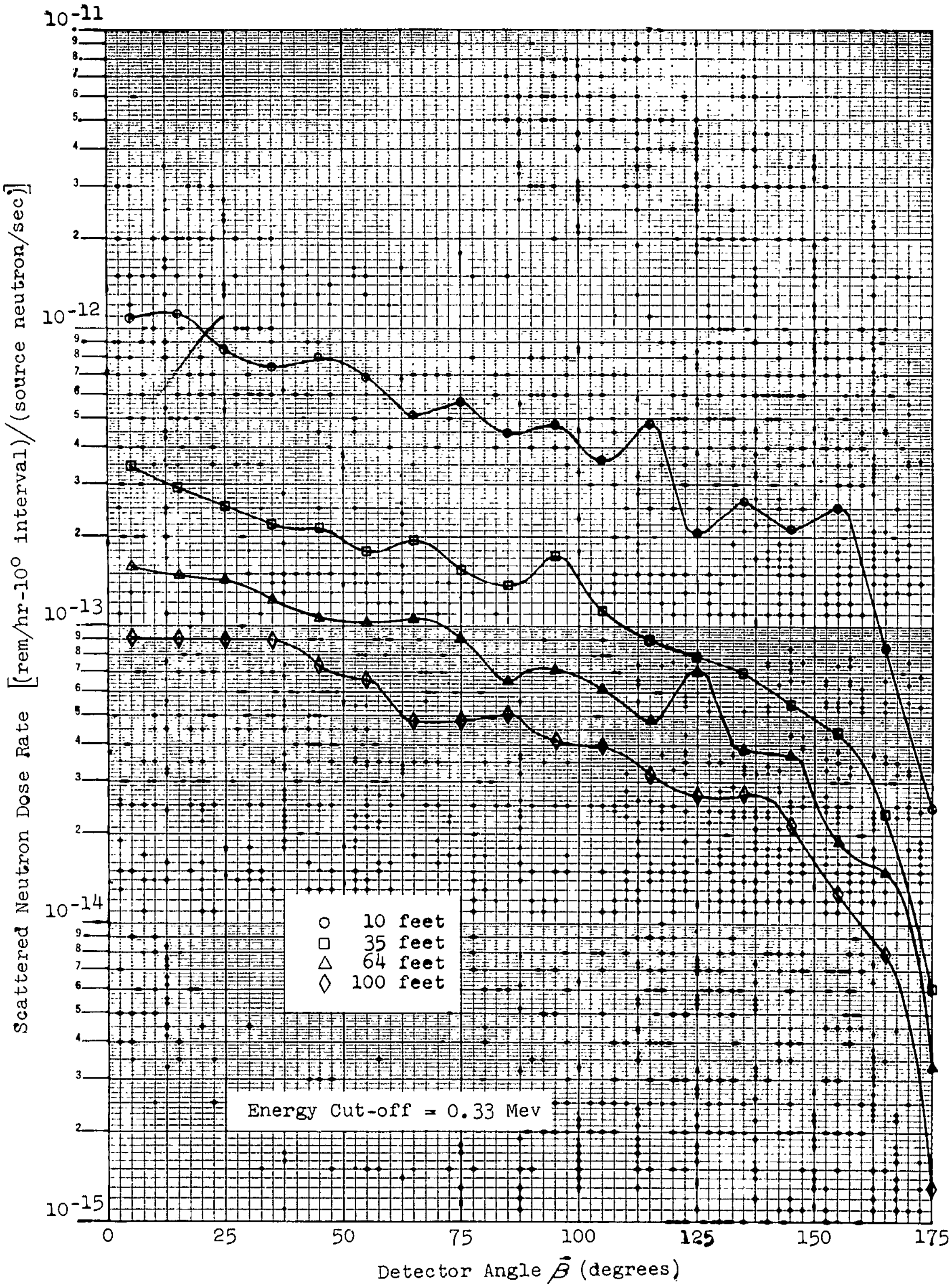

FIGURE 21. TOTAL SCATTERED NEUTRON DOSE RATE VS. DETECTOR ANGLE
Initial Energy 4.0 Mev

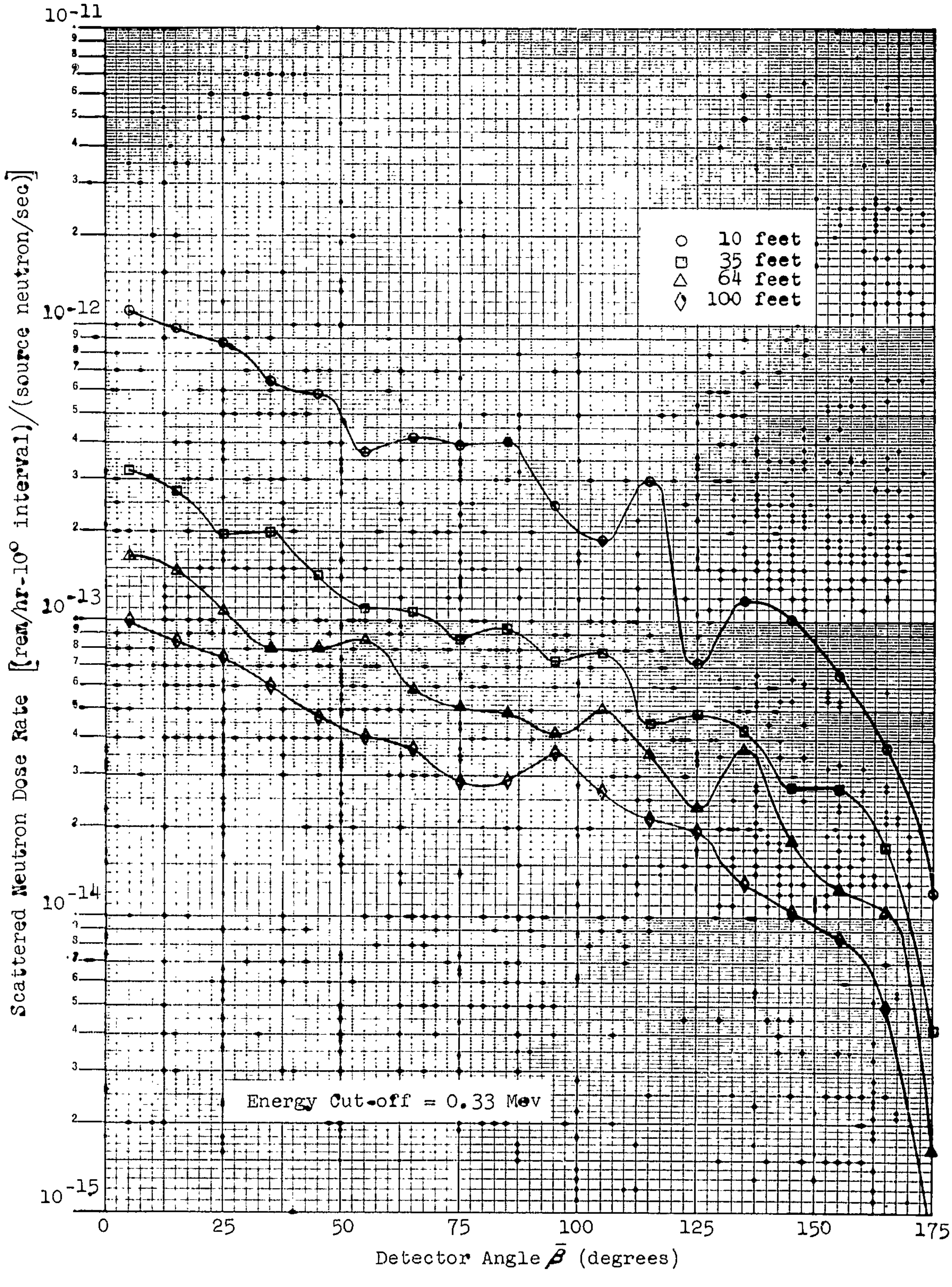

FIGURE 22. TOTAL SCATTERED NEUTRON DOSE RATE VS. DETECTOR ANGLE
Initial Energy 6.0 Mev

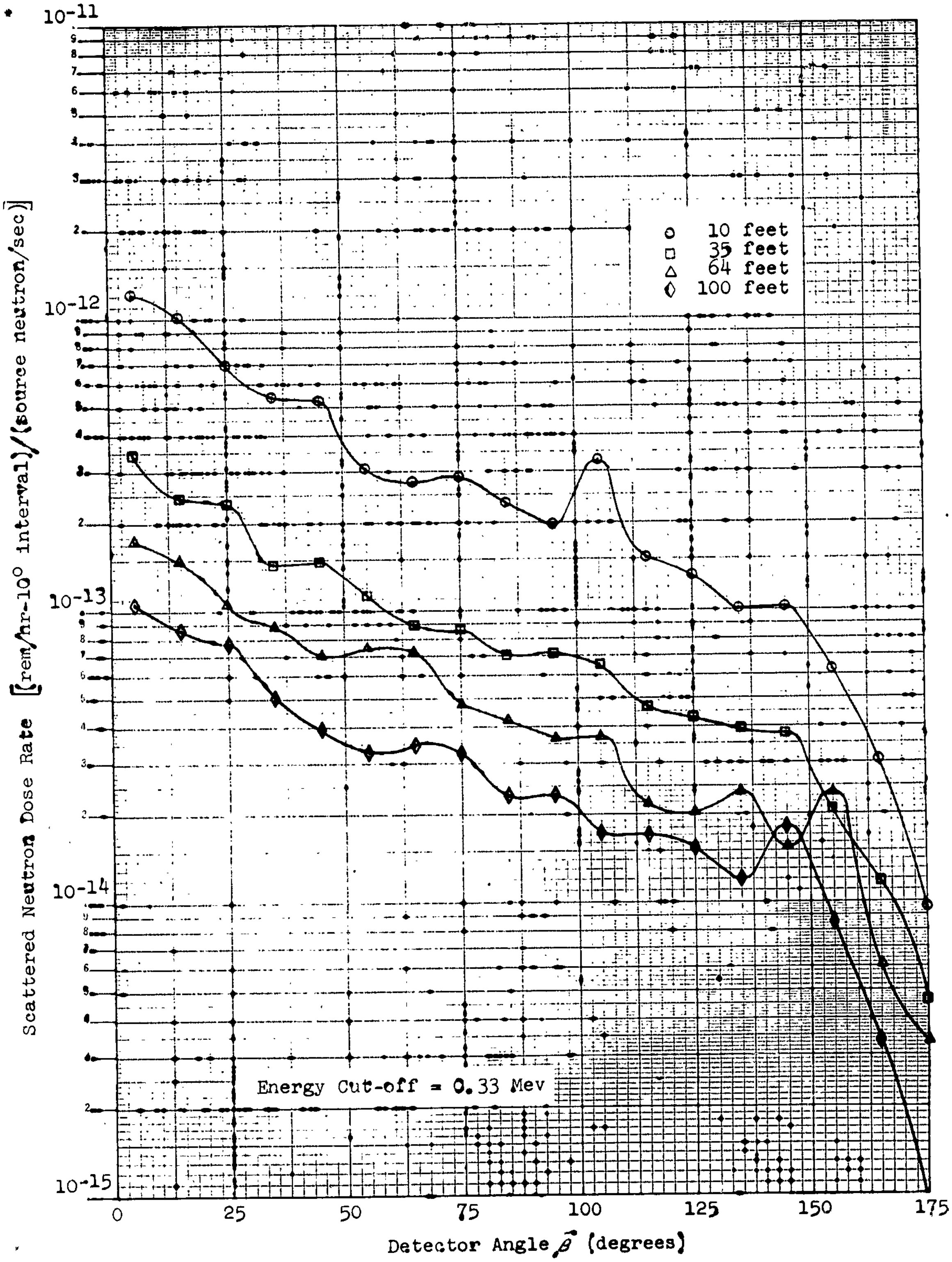

FIGURE 23. TOTAL SCATTERED NEUTRON DOSE RATE VS. DETECTOR ANGLE
Initial Energy 8.0 Mev

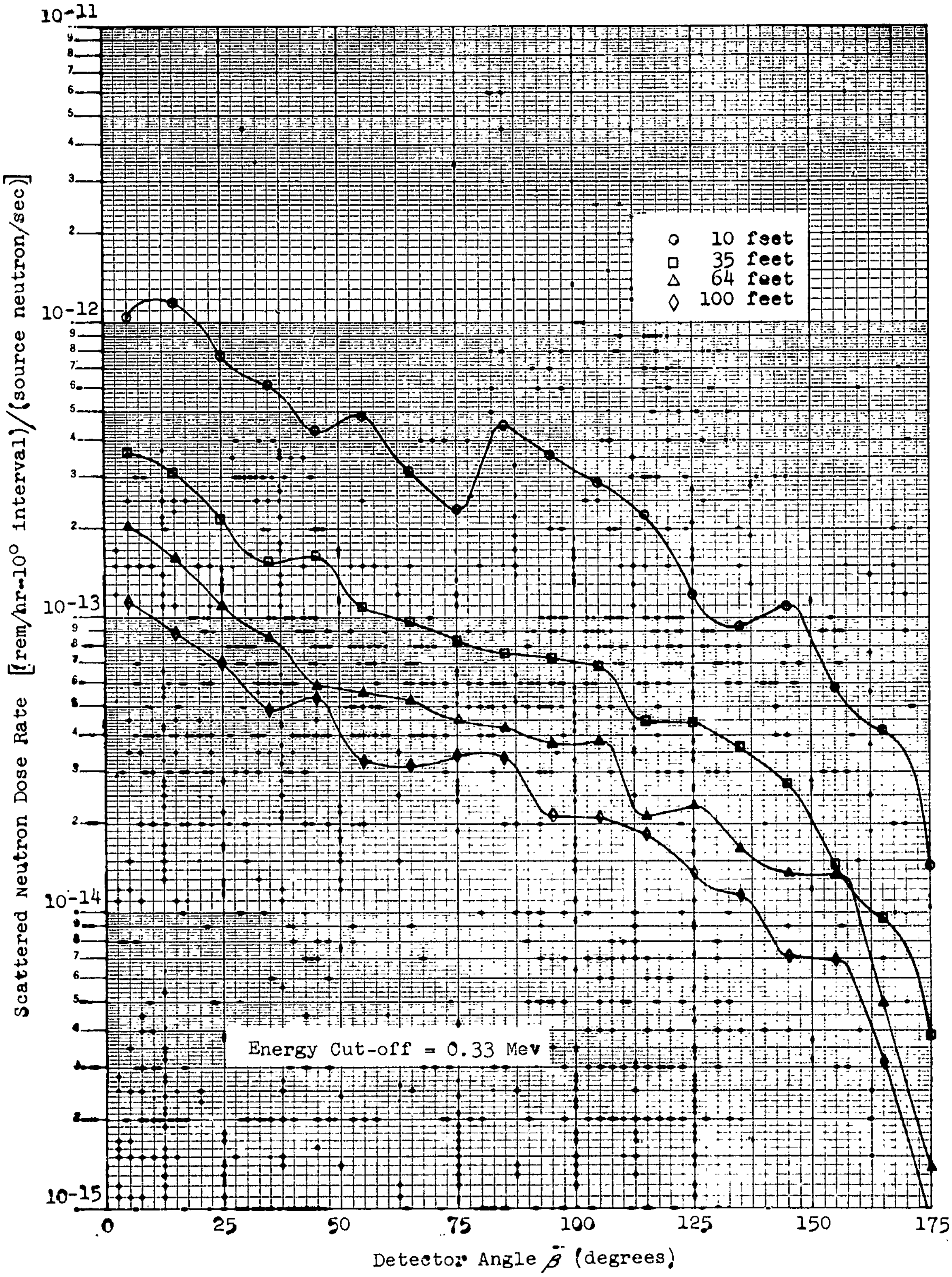

FIGURE 24. TOTAL SCATTERED NEUTRON DOSE RATE VS. DETECTOR ANGLE
Initial Energy 10.9 Mev

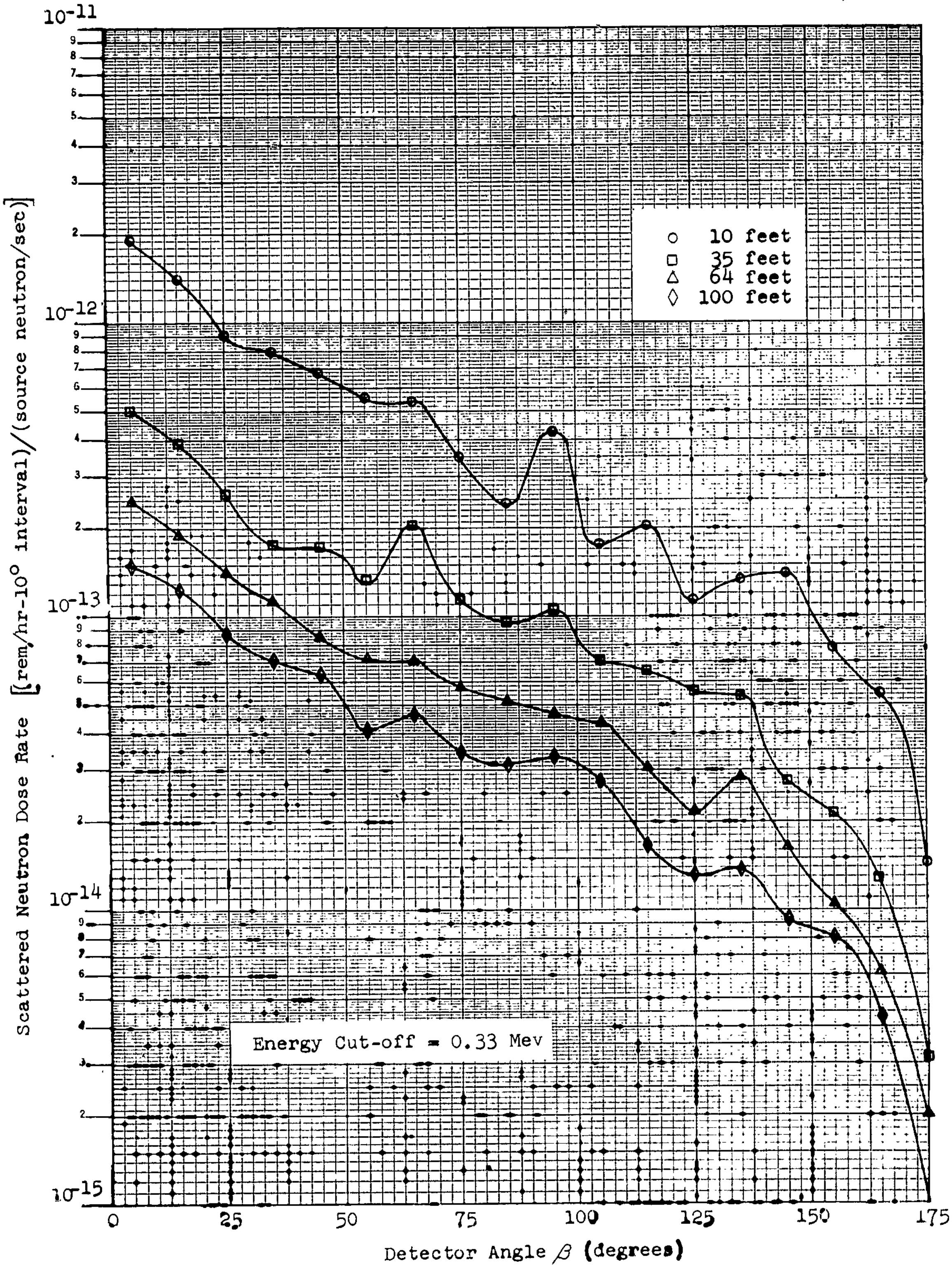

FIGURE 25. TOTAL SCATTERED NEUTRON DOSE RATE VS. DETECTOR ANGLE
Initial Energy 14.0 Mev

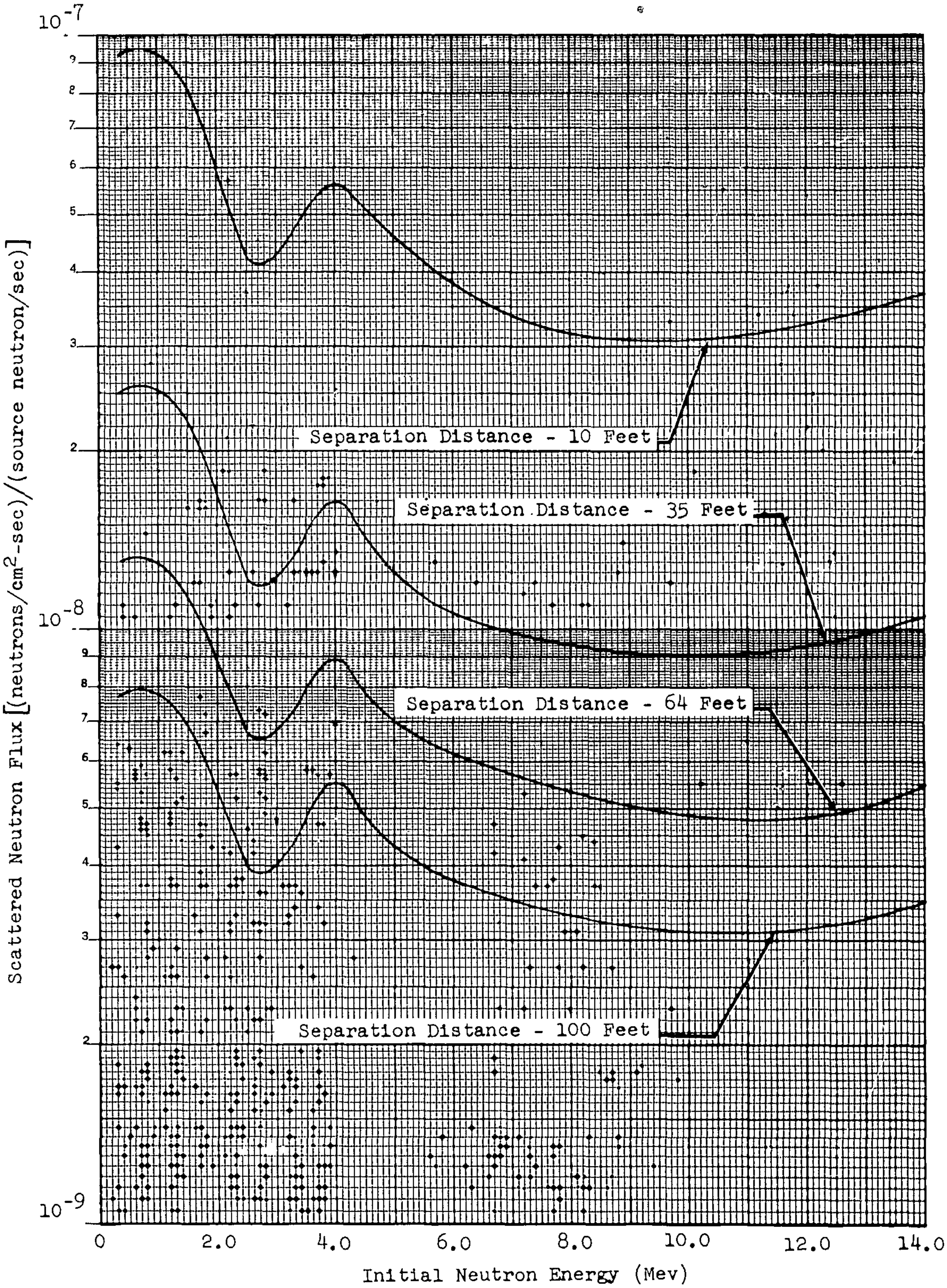

FIGURE 26. TOTAL SCATTERED NEUTRON FLUX VS. INITIAL ENERGY

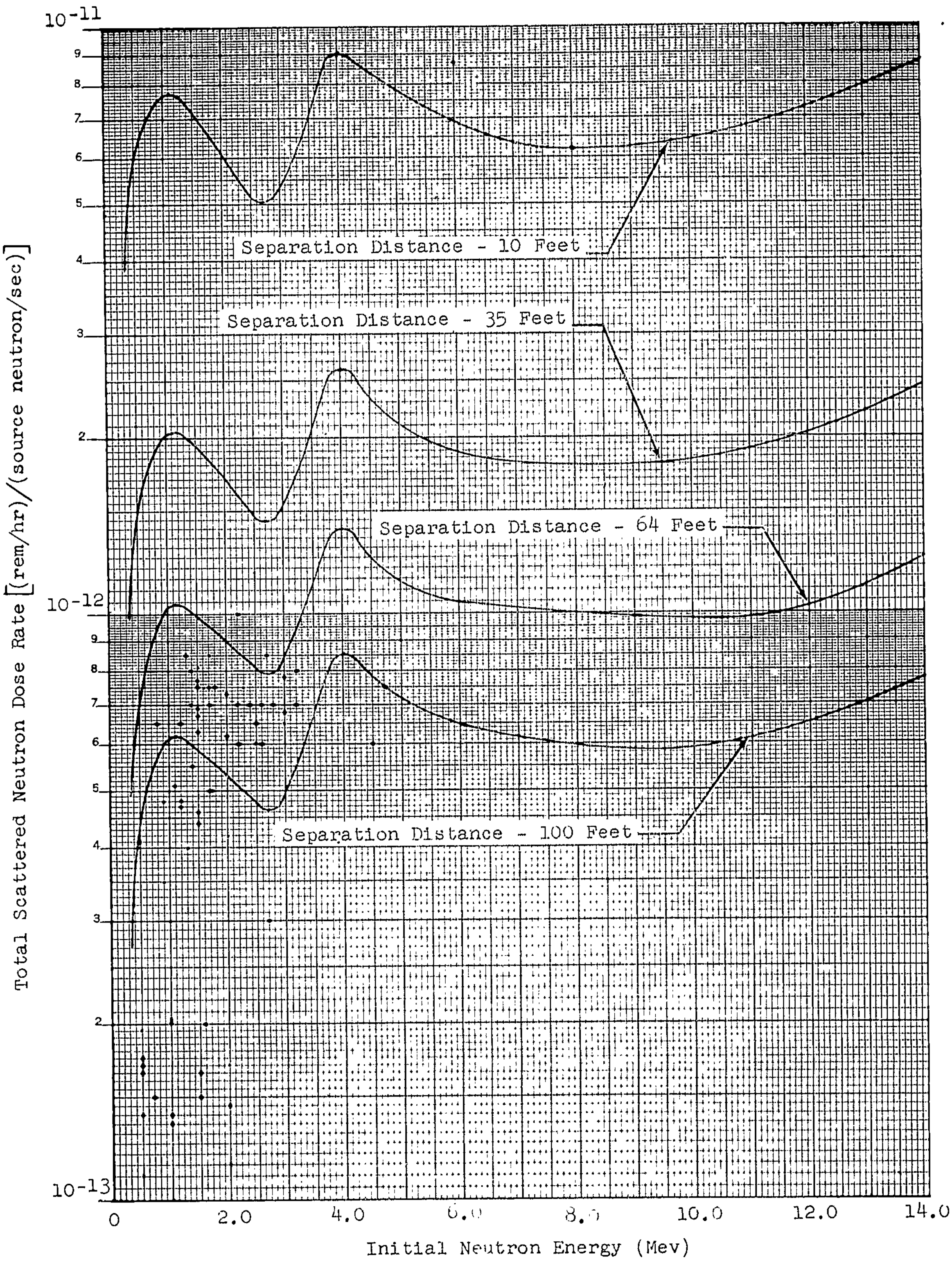

FIGURE 27. TOTAL SCATTERED NEUTRON DOSE RATE VS. INITIAL ENERGY

APPENDIX

R-55 FORTRAN STATEMENTS

APPENDIX

'55 FORTRAN STATEMENTS

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 7171  BEG XTRA R55 JACK.GRIGSBY (CAROL DIFFEY--3987)                        0001 R55
EZE    EZEC1 EZED2 EZED3 EOF   EZED5 EZEB1AEZEB3 EZEB4 EZEB1AEZEB5
C      IBM 704 PROCEDURE R55                                                 0002 R55
C      INTEGRATION OF MONTE CARLO CALCULATIONS OF FAST NEUTRON SCATTERING0003 R55
C      IN AIR FOR NON-ISOTROPIC SOURCES                                      0004 R55
C                                                                            0005 R55
       DIMENSION  Z(8),  DZ(8),  FLUXA (576,8),  DOSEA (576,8),              0006 R55
      1          FLUXE (320,8),  S(8,8), SRZ(8),  A(8),                      0007 R55
      2          SS(576,8)    , FD(576),  SOS (18,4,8),                      0008 R55
      3          FAD (576,8)  , FLIB(5),  RZ(8)                              0009 R55
       COMMON  Z ,  DZ ,  FLUXA ,  DOSEA ,  FLUXE ,  S , SRZ ,               0010 R55
      1         A ,  SS ,  FD ,  SOS ,  FAD , FLIB ,  RZ                     0011 R55
    10 CALL LIB1 (M)                                                         0012 R55
       GO TO (11,50) , M                                                     0013 R55
    11 CALL LIB (3HR55,ED)                                                   0014 R55
       READ 500, FLIP                                                        00151R55
        IF ( FLIP ) 14, 14, 13                                               00152R55
    13 BACKSPACE 9                                                           00153R55
       GO TO 50                                                              0016 R55
    14 FLIP  =  MODF ( ED , 10.0)                                            0017 R55
       LIBT  =  FLIP                                                         0018 R55
       GO TO (15,20,25,30,35) , LIBT                                         0019 R55
    40 FORMAT (6E10.4)                                                       0020 R55
    15 READ 40, (  Z(J), J=1,8 )                                             0021 R55
       XID1 = ED                                                             0022 R55
       GO TO 11                                                              0023 R55
    20 READ 40,    (DZ(J), J=1,8 )                                           0024 R55
       XID2 =  ED                                                            0025 R55
       GO TO 11                                                              0026 R55
    25 READ 40,  ((FLUXA (I,J), J=1,8) I=1,576 )                             0027 R55
       XID3 =  ED                                                            0028 R55
       GO TO 11                                                              0029 R55
    30 READ 40,  ((DOSEA (I,J), J=1,8) I=1,576 )                             0030 R55
       XID4 =  ED                                                            0031 R55
       GO TO 11                                                              0032 R55
    35 READ 40,  ((FLUXE (I,J), J=1,8) I=1,320 )                             0033 R55
       XID5 =  ED                                                            0034 R55
       GO TO 11                                                              0035 R55
    50 CALL SETUP (3HR55,XID)                                                0036 R55
       READ 500,  (FLIB(N), N=1,5)                                           0037 R55
   500 FORMAT (5F7.0)                                                        0038 R55
       IF (XID1 - FLIB(1))   55, 51, 55                                      0039 R55
    51 IF (XID2 - FLIB(2))   55, 52, 55                                      0040 R55
    52 IF (XID3 - FLIB(3))   55, 53, 55                                      0041 R55
    53 IF (XID4 - FLIB(4))   55, 54, 55                                      0042 R55
    54 IF (XID5 - FLIB(5))   55, 58, 55                                      0043 R55
    55 PRINT 56                                                              0044 R55
    56 FORMAT (54H LIBRARY DECKS CALLED FOR IN PROBLEM ARE NOT AVAILABLE)0045 R55
       CALL END9                                                             0046 R55
    58 READ 59, (NOP1  , NOP2 , NOP3 )                                       0047 R55
    59 FORMAT (3I3)                                                          0048 R55
       IF (NOP1 + NOP2 + NOP3 )  1000, 1000, 60                              0049 R55
    60 READ 40,  ((S(I,J), J=1,8),I=1,8 )                                    0050 R55
       DO 70  J=1,8                                                          0051 R55
       RZ(J) =  0.01745329  *  Z(J)                                          0052 R55
       SRZ(J)=  SINF (RZ(J))                                                 0053 R55
       A(J)  =  SRZ(J)   *  DZ(J)                                            0054 R55
    70 CONTINUE                                                              0055 R55
```

FORTRAN STATEMENTS (cont'd.)

```
  900 IF ( NOP1 + NOP2 ) 1000, 1100, 1200
 1000 PRINT 1001
 1001 FORMAT (37H ERROR ON SECOND CARD OF PROBLEM DECK)
      CALL END9
 1100 NM = 320
      GO TO 2000
 1200 NM = 576
C
C     CONSTRUCT A LARGER MATRIX FROM S(I,J)
 2000 DO 2022  I=1,NM
      DO 2022  J=1,8
      IF (I-8)  2010, 2010, 2012
 2010 SS(I,J)  =  S(I,J)
      GO TO 2022
 2012 SS(I,J)  =  SS (I-8, J)
 2022 CONTINUE
   65 FORMAT (8E10.4)
      IF ( SENSE SWITCH   2  )   2023, 2024
 2023 PRINT 65, (( SS(I,J), J=1,8 ),I=1,NM )
 2024 DO   2027   I=1,NM
      DO   2027   J=1,8
      SS(I,J)  =  6.2831853  * SS(I,J)
 2027 CONTINUE
      IF (SENSE SWITCH 2 )   2028, 2029
 2028   PRINT 65, ((SS(I,J), J=1,8 ), I=1, NM )
 2029 IF  (NOP1  )  1000, 2150, 2030
C
C     COMPUTE ANGULAR DISTRIBUTION OF FLUX.
 2030 DO 2035    I= 1,576
      DO 2035    J= 1,8
      FAD (I,J)  =   SS(I,J)  *  FLUXA (I,J)
      FAD (I,J)  =  FAD(I,J)  *   A(J)
 2035 CONTINUE
C     SUMMATION
      DO 2040  I=1,576
      FD(I) = 0.0
      DO 2040  J=1,8
      FD(I) = FD(I)   +   FAD (I,J)
 2040 CONTINUE
      DO 2050  I=1,18
      DO 2050  J=1,4
      DO  2050  K=1,8
      L = K  +  8* (J-1)  +  32 * (I-1)
      SOS (I,J,K)  =  FD(L)
 2050 CONTINUE
C     PRINT OUT RESULTS AS 4  18X8 MATRICES.
      DO 2100  J=1,4
      PRINT 2060
 2060 FORMAT (29H1ANGULAR DISTRIBUTION OF FLUX)
      PRINT 2070 , J
 2070 FORMAT (22H0SEPARATION DISTANCE A,I1)
      PRINT 2080
 2080 FORMAT (100H0 K      E01              E02              E03          E04
     1  E05              E06              E07              E08      )
      DO 2100 I=1,18
      PRINT 2090 ,  (I, (SOS (I,J,K), K=1,8 ))
 2090 FORMAT  (1H ,I2,8E12.4)
 2100 CONTINUE
```

```
 2150 IF (NOP 2 )  1000, 3000, 2200
C
C     COMPUTE ANGULAR DISTRIBUTION OF DOSE RATE.
 2200 DO 2210   I = 1,576
      DO 2210   J = 1,8
      FAD (I,J) = SS(I,J)  *  DOSEA  (I,J)
      FAD (I,J) = FAD(I,J) *  A(J)
 2210 CONTINUE
C     SUMMATION
      DO 2220   I=1,576
      FD(I)  = 0.0
      DO 2220   J=1,8
      FD(I) = FD(I)  +  FAD (I,J)
 2220 CONTINUE
      DO 2230   I=1,18
      DO 2230   J=1,4
      DO 2230   K=1,8
      L  =  K   +  8 * (J-1)  +  32 * (I-1)
      SOS(I,J,K)  =  FD(L)
 2230 CONTINUE
C     PRINT OUT ANGULAR DISTRIBUTION OF DOSE RATE
      DO 2250  J=1,4
      PRINT 2240
 2240 FORMAT (38H1ANGULAR DISTRIBUTION OF THE DOSE RATE)
      PRINT 2070,  J
      PRINT 2080
      DO 2250   I=1,18
      PRINT 2090 ,  (I , (SOS (I,J,K), K=1,8) )
 2250 CONTINUE
      IF (NOP 3 )   1000,  50,  3000
C
C     COMPUTE ENERGY SPECTRUM OF FLUX
 3000 DO 3035   I=1,320
      DO 3035   J=1,8
      SS(I,J) = SS(I,J)  * FLUXE (I,J)
      SS(I,J) = SS(I,J)  * A(J)
 3035 CONTINUE
C     SUMMATION
      DO 3040   I=1,320
      FD(I)  = 0.0
      DO 3040   J=1,8
      FD(I)  = FD(I) +  SS (I,J)
 3040 CONTINUE
      DO 3050   I= 1,10
      DO 3050   J= 1,4
      DO 3050   K= 1,8
      L  =  K  +   8 * (J-1)  +  32 * (I-1)
      SOS (I,J,K )  = FD(L)
 3050 CONTINUE
C     PRINT OUT ENERGY SPECTRUM OF FLUX
      DO 3070  J=1,4
      PRINT 3060
 3060 FORMAT (29H1TOTAL FLUX IN ENERGY GROUP K)
C
      PRINT 2070 , J
      PRINT 2080
      DO 3070  I=1,10
      PRINT 2090 , ( I,  (SOS(I,J,K),K=1,8) )
```

FORTRAN STATEMENTS (cont'd.)

```
3070 CONTINUE                                                    0164 R55
4000 GO TO 50                                                    0165 R55
     END(2,1)                                                    0166 R55
```

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