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MONTE CARLO CALCULATIONS OF FLUXES AND DOSE RATES RESULTING FROM NEUTRONS MULTIPLY SCATTERED IN AIR

F. L. Keller
C. D. Zerby
J. Hilgeman

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F. L. Keller
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ABSTRACT

A Monte Carlo code previously developed for the Oracle was used for a parameter study to calculate the fluxes and dose rates resulting from neutrons multiply scattered in air. Both the source and the detector were assumed to be suspended in an infinite body of air ($\rho = 0.00125$ g/cm^3), and the source was assumed to be a point source which emitted a monoenergetic line-beam of neutrons at various angles with respect to the source-detector axis. Calculations were performed for separation distances, g, of 5, 10, 20, 40, 64, and 100 ft; for source energies, E_o, of 5, 3, 2, 1.2, and 0.55 Mev; and for beam angles, θ_o, of 2, 15, 30, 60, 90, 135, and 180 deg. The scattering was assumed to be isotropic in the center-of-mass system. The results of the flux calculations were found to fit the expression $\frac{\Sigma_s(E_o)}{4\pi g} \cdot \frac{\pi - \theta_o}{\sin\theta_o}$, where $\Sigma_s(E_o)$ is the total macroscopic scattering cross section of air at the initial energy of the neutron. The results of the dose-rate calculations were found to fit the expression $\frac{1}{1.2} \cdot \frac{C(E_o)\Sigma_s(E_o)}{4\pi g} \cdot \frac{\pi - \theta_o}{\sin\theta_o}$, where $C(E_o)$ is the flux-to-dose rate conversion factor.

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INTRODUCTION

A complete description has been given previously of a Monte Carlo code which was devised for the ORACLE to determine the angular distribution and energy spectrum of air-scattered neutrons.[1] The particular form of Monte Carlo calculation used involved the integration of the Boltzmann equation. The problem was idealized to have the source in an infinite body of air and used isotropic scattering in the center-of-mass system. The output includes the total flux, total tissue dose rate, angular distributions of both flux and dose rate, and the energy spectrum which results from air-scattered neutrons which reach a detector located an arbitrary distance from the source.

The results of a parameter study for point sources which emit line beams of monoenergetic neutrons is reported here.

GENERAL DESCRIPTION OF CALCULATIONS

The geometry of the source and detector is shown in Fig. 1. The source was taken to be a point source which emitted a monoenergetic line beam of neutrons of energy E_o at an angle θ_o with respect to the source-detector axis. The detector was divided into a number of solid angles with apexes at the detector point. The polar and azimuthal angles were each divided into a number of equal angle intervals. The meridians and parallels of latitude scribed on a sphere centered at the detector point established the intersection of the set of solid angles with the arbitrary sphere. Because of symmetry, the lower 2π space is just a reflection of the upper 2π space. The number of neutrons which entered each of these solid angle intervals was recorded. The energy spectrum of the radiation which entered each solid angle interval was determined by dividing the energy range from a specified cut-off energy to the source energy into an arbitrary number of equal intervals and storing the contribution in the appropriate

1. C. D. Zerby, "A Monte Carlo Calculation of Air-Scattered Neutrons," ORNL-2277 (1957).

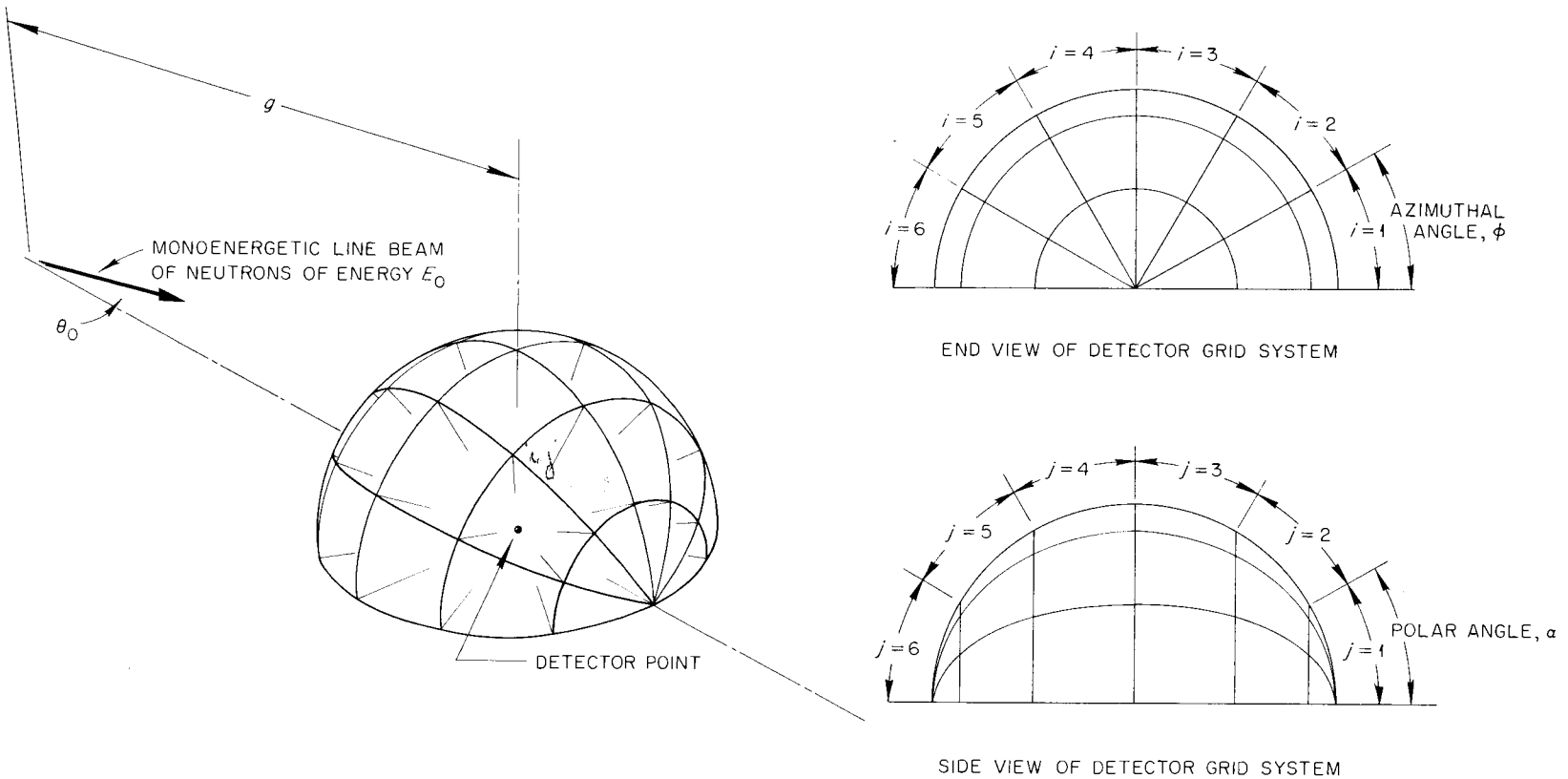

Fig. 1. Geometry Used in the Calculations of Multiply Scattered Neutron Dose Rates.

interval. This energy spectrum was then used by the machine to determine the contribution to the tissue dose rate at the detector from neutrons which entered in each energy interval of each solid angle interval. The reader is referred to the previous reference for a complete description of the method of computation.

Two separate tissue dose-rate calculations were incorporated into the code. The first, which was designated as Dose 1, used the tissue dose-rate conversion factors calculated in the manner described by Hurst and Ritchie for calibration of the Hurst-type dosimeter.[2] The second calculation, which was designated as Dose 2, used the tissue dose-rate conversion factors calculated by Snyder[3] who used a Monte Carlo technique. Thus Dose 1 corresponds to the response of the dosimeter in air and Dose 2 corresponds to the response of the dosimeter when it is in a tissue phantom. The dose-rate results which are given in this report are all of the Dose 1 type. A plot of the Hurst and Ritchie type conversion factor versus neutron energy is reproduced in Fig. 2.

The total cross sections for oxygen and nitrogen were taken from the graphs presented in the report BNL-325 (Ref. 4) and were assumed to be isotropic in the center-of-mass system.

Variables which were required as input data to the code were: the angle, θ_o; the initial energy, E_o; the cutoff energy, E_c; the number of energy intervals; the number of polar and azimuthal angle intervals; the separation distance, g; the density of dry air and the density of moisture in the air; and the number of case histories desired.

RESULTS

The neutron air-scattering problem is of sufficient interest that a complete parameter study was undertaken and completed. For general application it was decided to use a monodirectional, monoenergetic point source at several source-detector separation distances. The energies of

2. G. S. Hurst and R. H. Ritchie, "A Count-Rate Method of Measuring Fast-Neutron Tissue Dose," ORNL-930 (1951).
3. W. S. Snyder, Health Phys. Div. Prog. Rep. July 31, 1956, ORNL-2151 (1956), p. 64.
4. D. J. Hughes and J. A. Harvey, "Neutron Cross Sections," BNL-325 (1955).

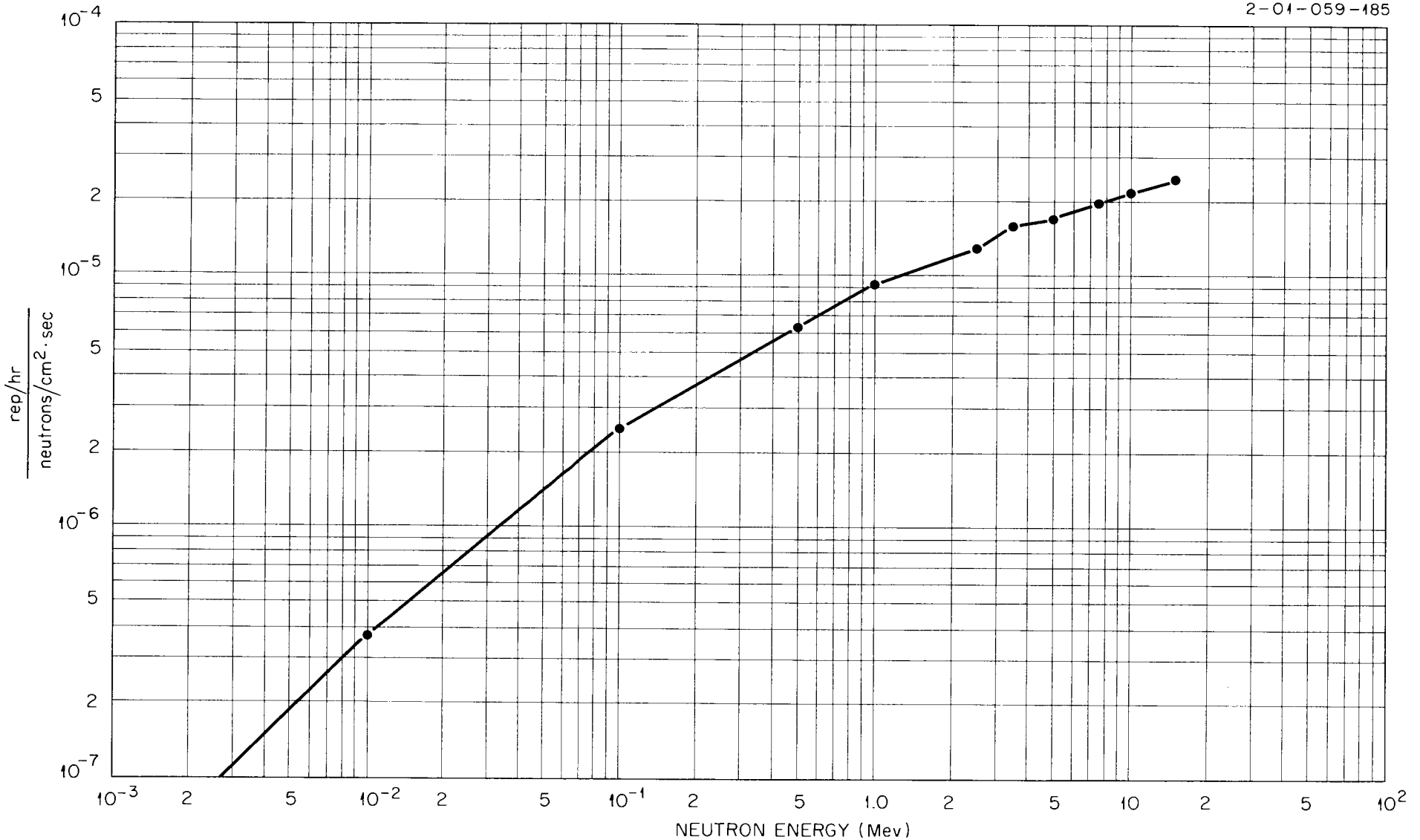

Fig. 2. Neutron Flux to Tissue Dose Rate Conversion Factors Calculated by the Hurst–Ritchie Method.

the source which appeared to be of interest and were within the limits of application of the code were 5, 3, 2, 1.2, and 0.55 Mev. These particular values were selected because the cross section at those energies appeared to represent an average within a reasonable energy interval. Beam angles, θ_o, of 2, 15, 30, 60, 90, 135, and 180 deg were also selected. The source-detector separation distances, g, selected were 5, 10, 20, 40, 64, and 100 ft. The density of dry air used in every problem was 0.00125 g/cm^3. No moisture was included. The energy cutoff was taken as 0.1 Mev. This value was taken to correspond with the approximate cutoff energy of the Hurst-type fast-neutron dosimeter. All of the results were normalized to one neutron per second emitted from the source.

Figures 3 through 7 show the total flux at the detector plotted as a function of the source-detector separation distance for the various angles of emission. In each of these figures the individual points which are plotted are the results of the Monte Carlo calculations, whereas the solid curves are plots of the function

$$\frac{\Sigma_s(E_o)}{4\pi g} \cdot \frac{\pi - \theta_o}{\sin\theta_o} \tag{1}$$

where

$\Sigma_s(E_o)$ = total macroscopic scattering cross section of air at the initial energy of the neutron,

g = source-detector separation distance,

θ_o = angle of emission of line beam with respect to the source-detector axis.

The expression given in Eq. 1 is the result obtained for the flux from a line-beam source if a single scattering calculation is performed in which the scattering cross section is assumed to be isotropic in the laboratory system and the material attenuation is neglected on both legs of the scattering event. It was pointed out some time ago[5] that an approximate method of correcting for multiple scattering is to omit the material attenuation in single scattering; however, it is rather surprising that this result agrees with the Monte Carlo results as accurately as is shown in the

5. A. Simon and R. H. Ritchie, "Background Calculations for the Proposed Tower Shielding Facility, Oak Ridge National Laboratory," ORNL-1273, App. B (Nov. 3, 1952).

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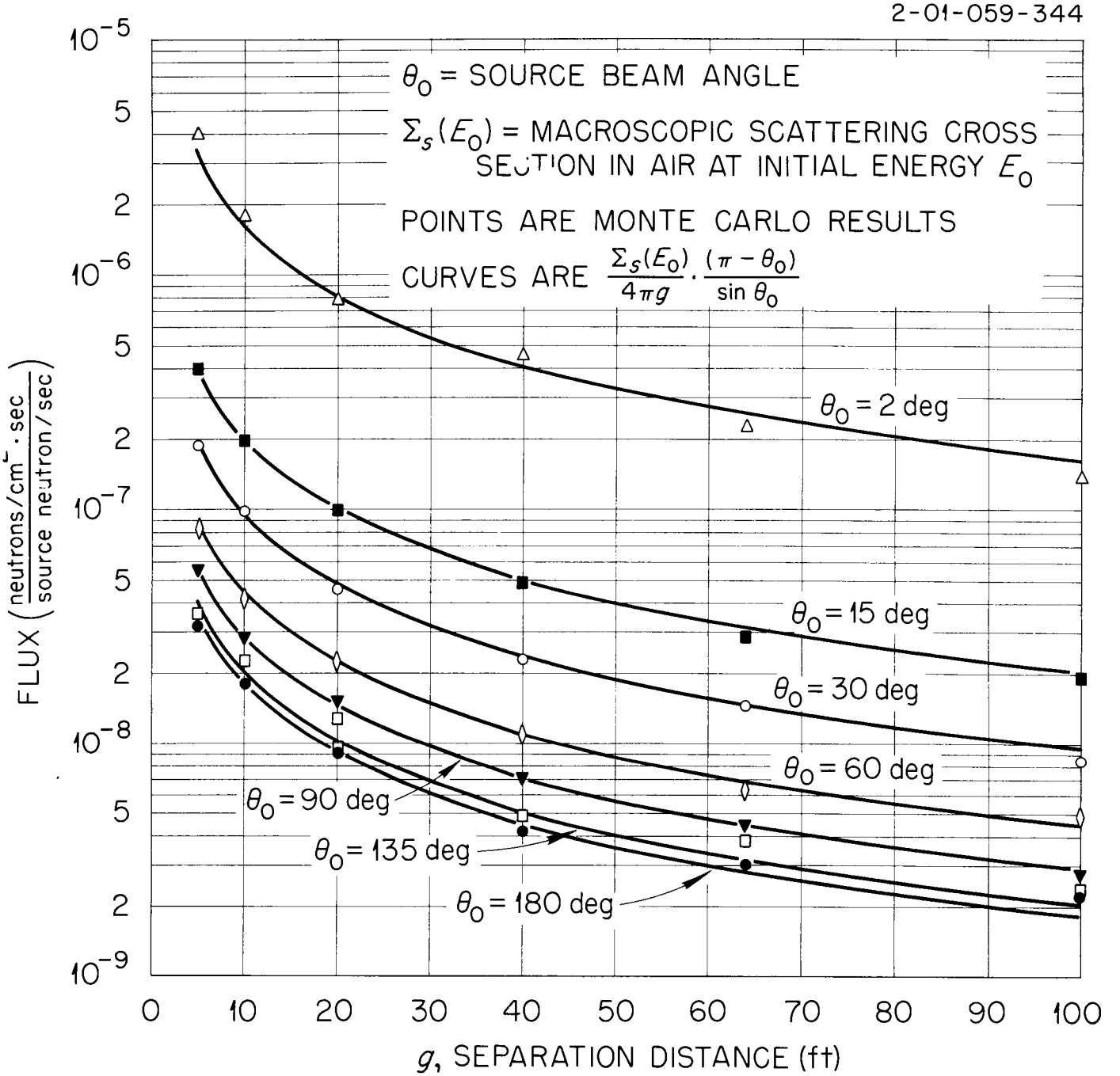

Fig. 3. Neutron Flux in Air at a Source–Detector Separation Distance g Resulting from Multiply Scattered Neutrons. 5-Mev Line-Beam Source.

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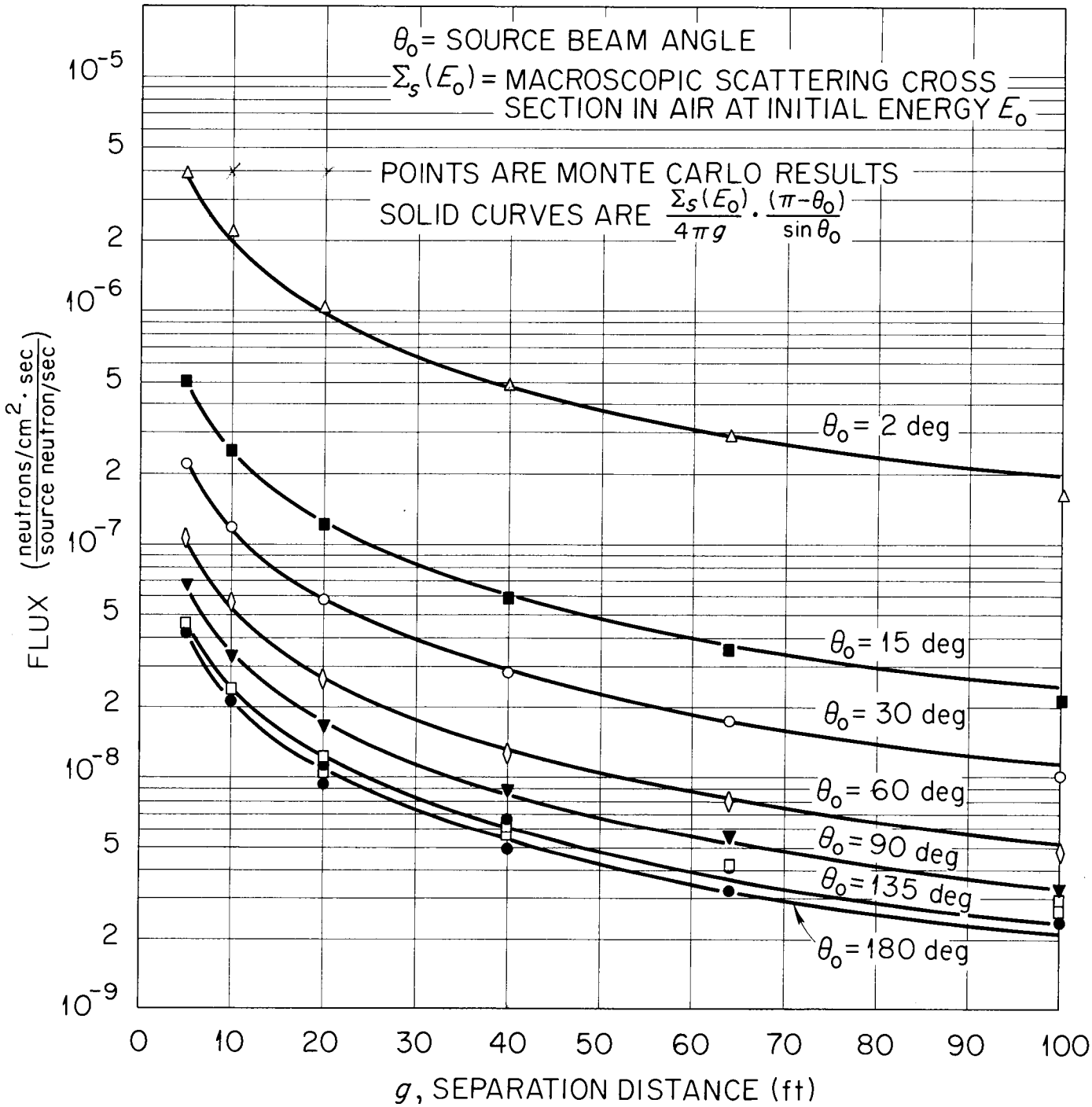

Fig. 4. Neutron Flux in Air at a Source-Detector Separation Distance g Resulting from Multiply Scattered Neutrons. 3-Mev Line-Beam Source.

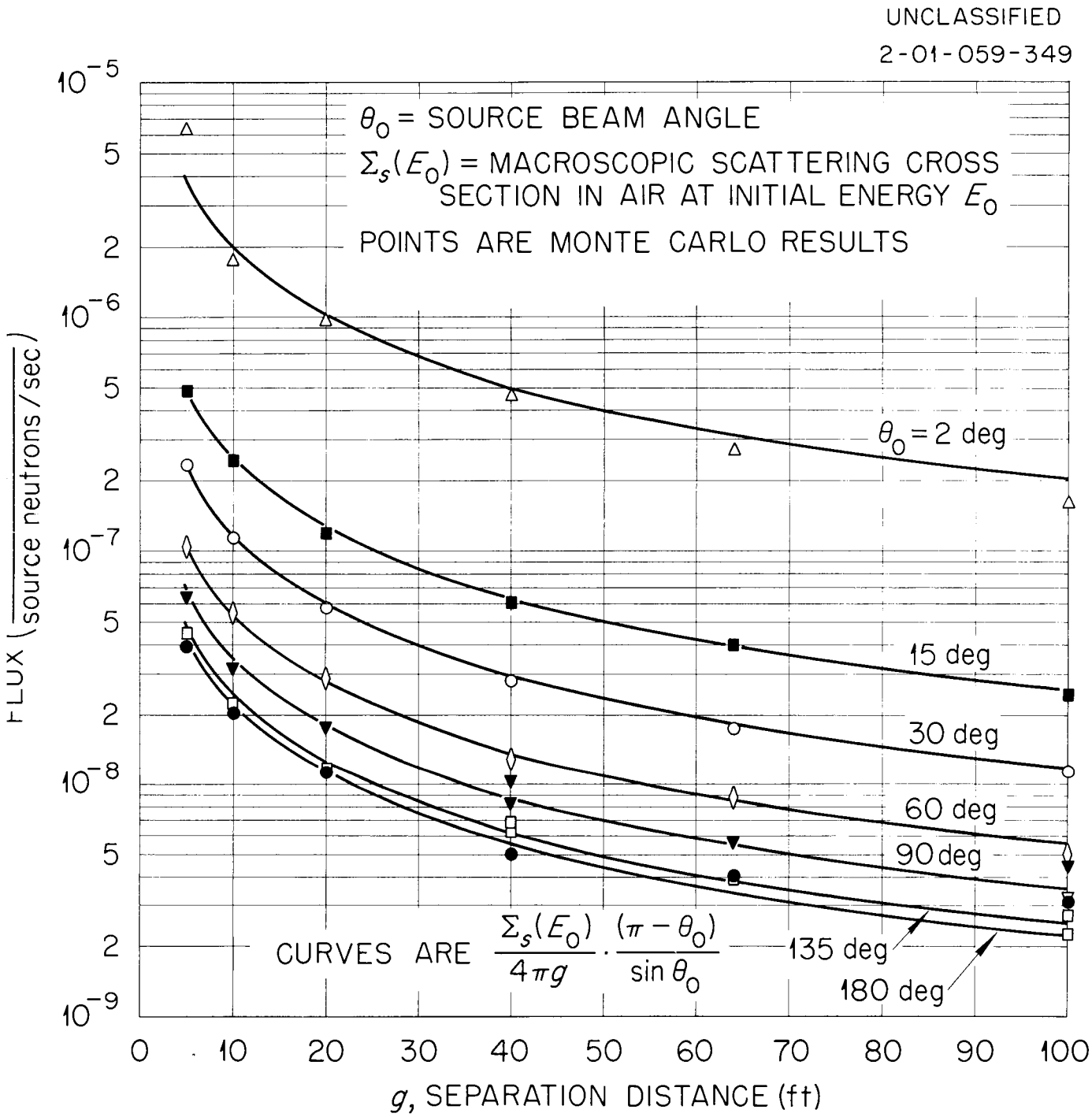

Fig. 5. Neutron Flux in Air at a Source–Detector Separation Distance g Resulting from Multiply Scattered Neutrons. 2-Mev Line-Beam Source.

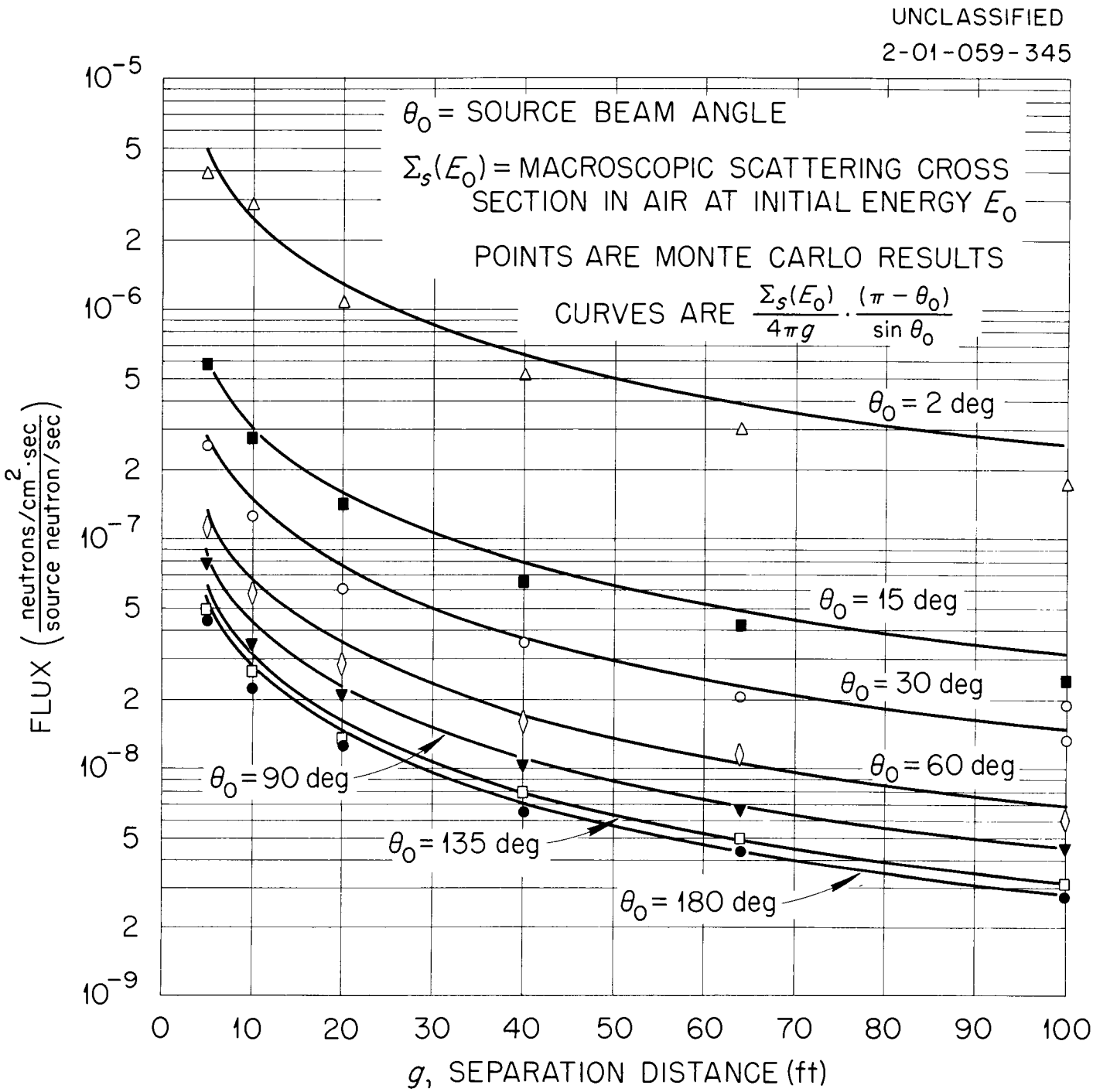

Fig. 6. Neutron Flux in Air at a Source–Detector Separation Distance g Resulting from Multiply Scattered Neutrons. 1.2-Mev Line-Beam Source.

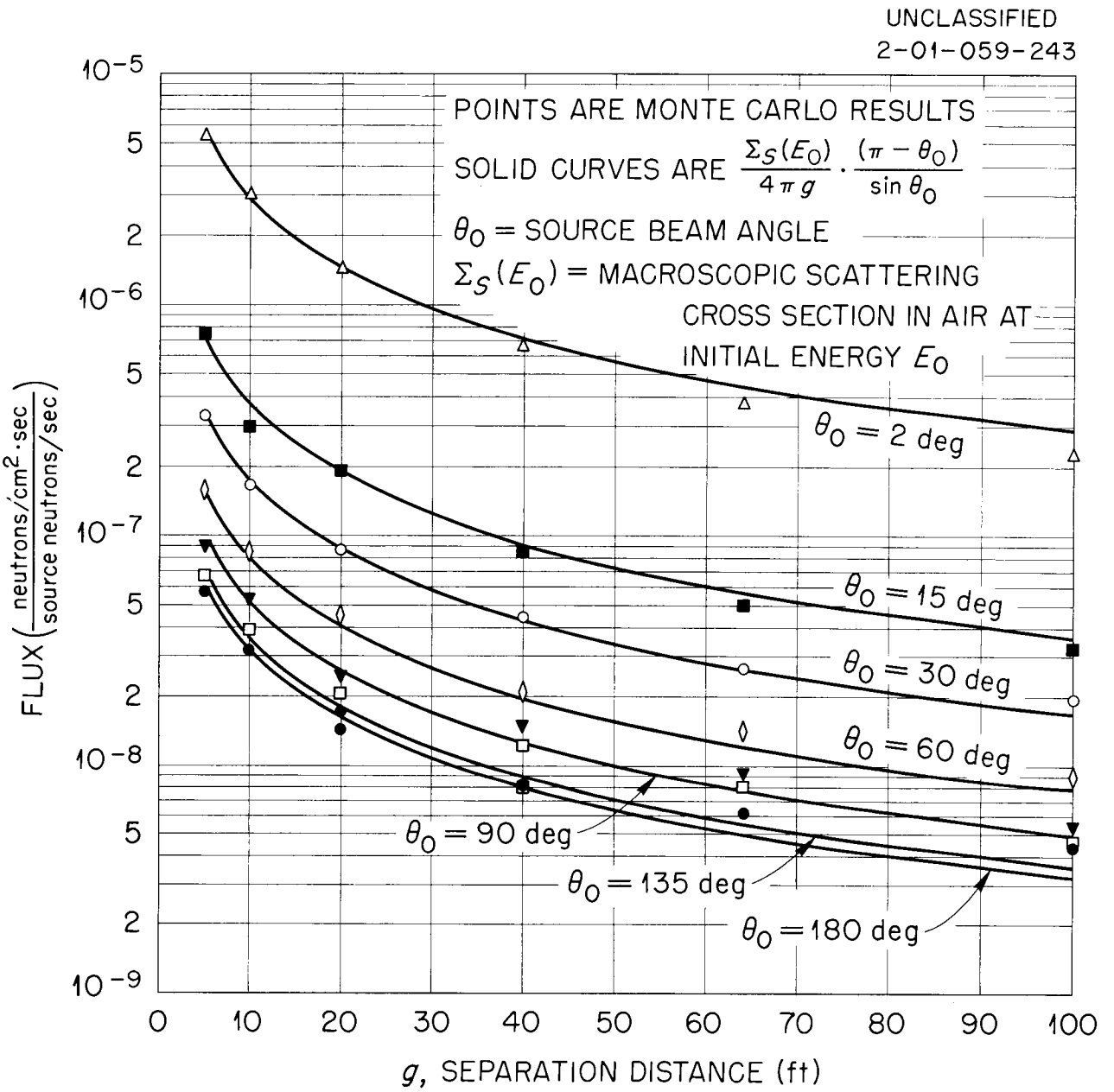

Fig. 7. Neutron Flux in Air at a Source-Detector Separation Distance g Resulting from Multiply Scattered Neutrons. 0.55-Mev Line-Beam Source.

figures for such large variations in energy, angle, and separation distance. The largest variations in the results appear to occur for the largest emission angles and separation distances where the Monte Carlo results are, of course, subject to the greatest statistical variations.

Figures 8 through 12 show plots of the total tissue dose rate at the detector versus the source-detector separation distance for the various angles of emission. Again, the individual points which are plotted are the results of the Monte Carlo calculation, whereas, in this case, the solid curves are plots of the function

$$\frac{1}{1.2} \cdot \frac{C(E_o)\Sigma_s(E_o)}{4\pi g} \cdot \frac{\pi - \theta_o}{\sin\theta_o} \tag{2}$$

where $C(E_o)$ is the flux-to-dose-rate conversion factor ($\text{reps}\cdot\text{hr}^{-1}/\text{neutron}\cdot\text{cm}^{-2}\cdot\text{sec}^{-1}$) at the initial energy of the neutron (see Fig. 2).

Except for the factor 1/1.2, Eq. 2 is the dose rate which results from a single-scattering calculation for a line-beam source in which the scattering cross section is taken to be isotropic in the laboratory system, the material attenuation is omitted on each leg, and energy degradation upon scattering is neglected. From the plots it is seen that Eq. 2 fits the Monte Carlo results extremely well over the entire range of variables.

Since Eq. 1 fits the flux data almost exactly and Eq. 2 gives an almost exact fit to the tissue dose-rate data, it is obvious that the factor $C(E_o)/1.2$ in Eq. 2 must represent the average flux-to-dose-rate conversion factor for the scattered neutrons which reach the detector from a line beam of neutrons of initial energy E_o. That is, if $\Phi_{E_o,\theta_o,g}(E)dE$ represents the total flux of air-scattered neutrons with energies between E and E + dE at a detector located a distance g from a point source which emits a line beam of neutrons with initial energy E_o at an angle θ_o, then Eqs. 1 and 2 require that

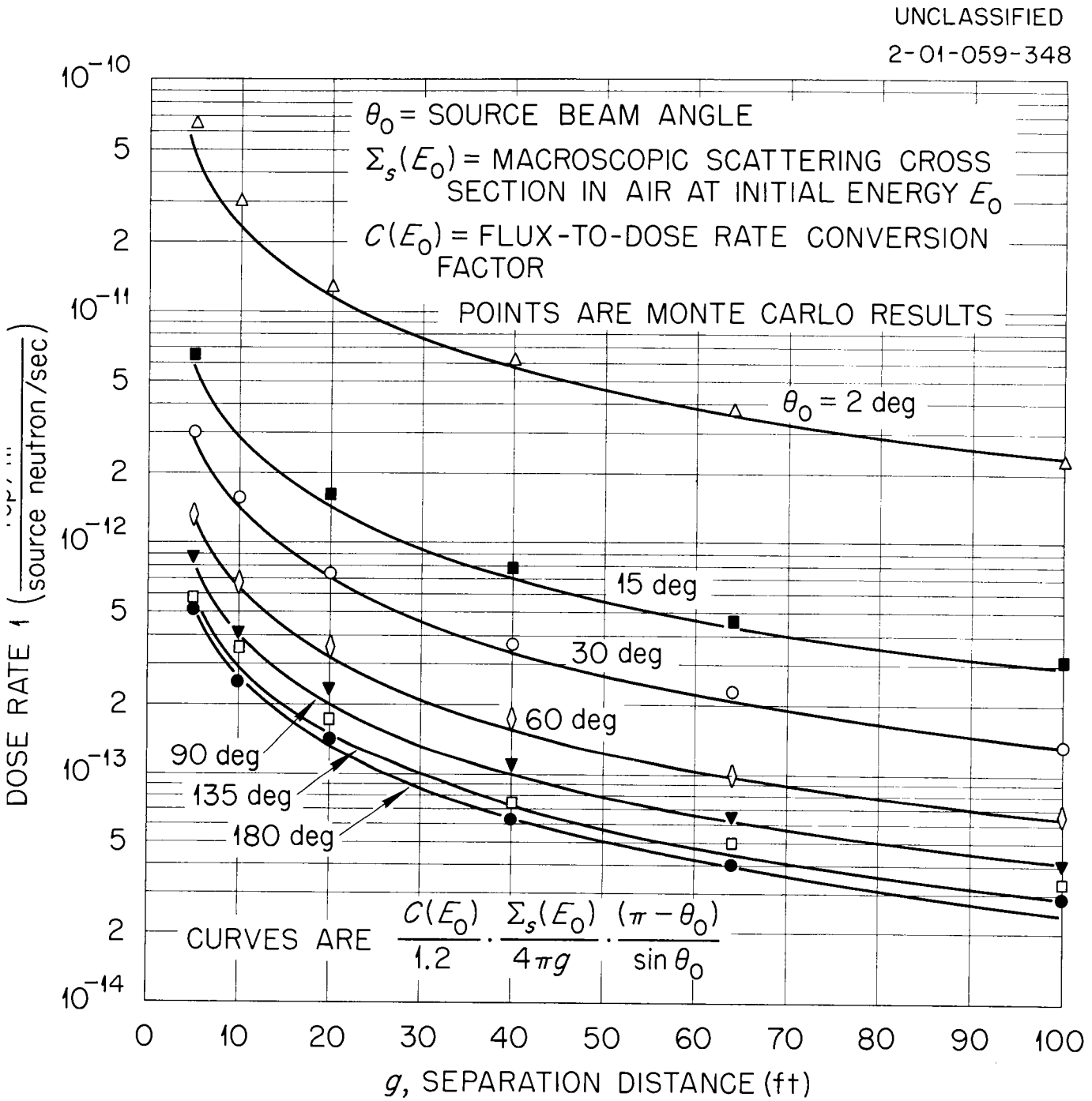

Fig. 8. Dose Rate in Air at a Source–Detector Separation Distance g Resulting from Multiply Scattered Neutrons. 5-Mev Line-Beam Source.

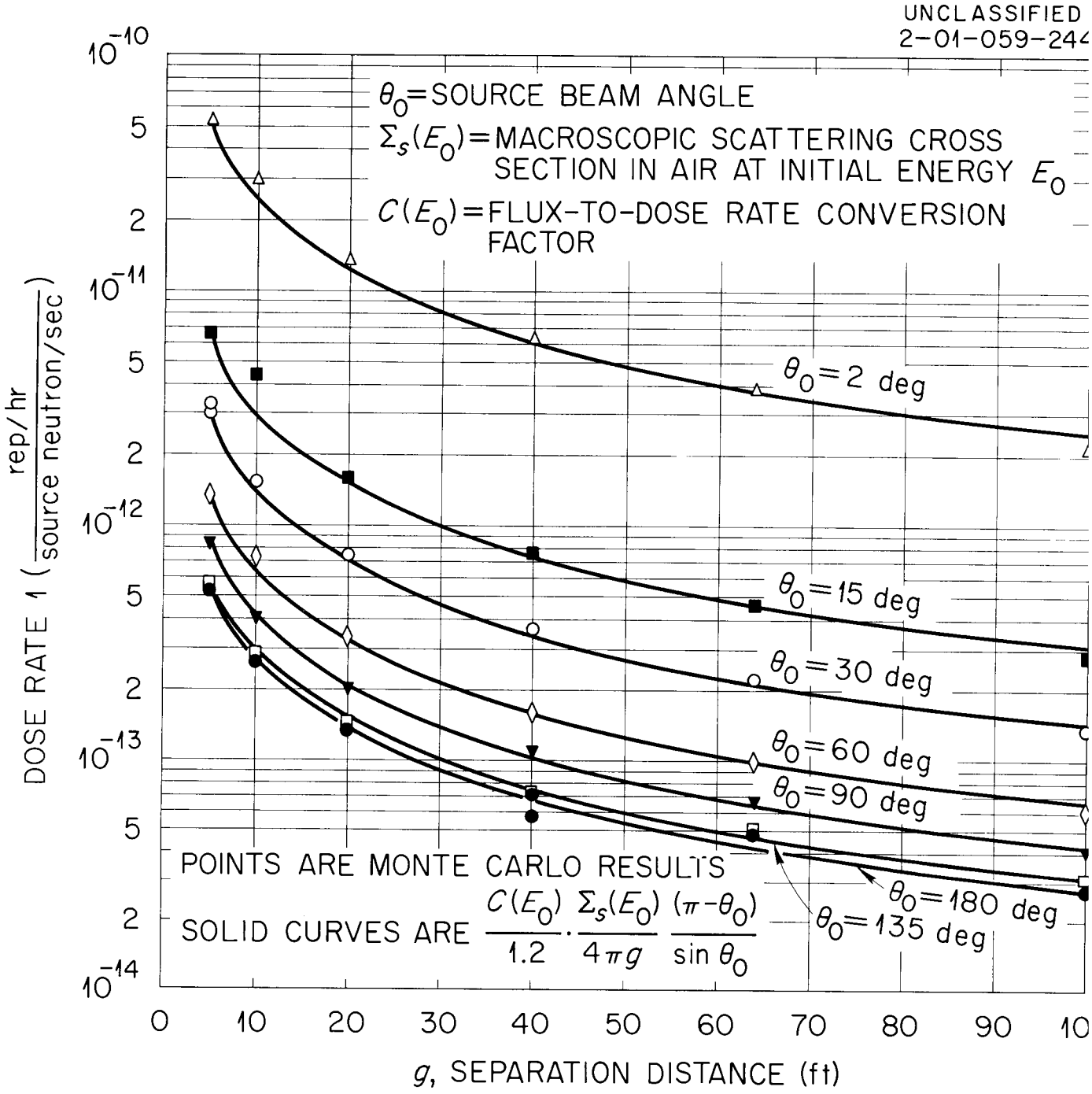

Fig. 9. Dose Rate in Air at a Source-Detector Separation Distance g Resulting from Multiply Scattered Neutrons. 3-Mev Line-Beam Source.

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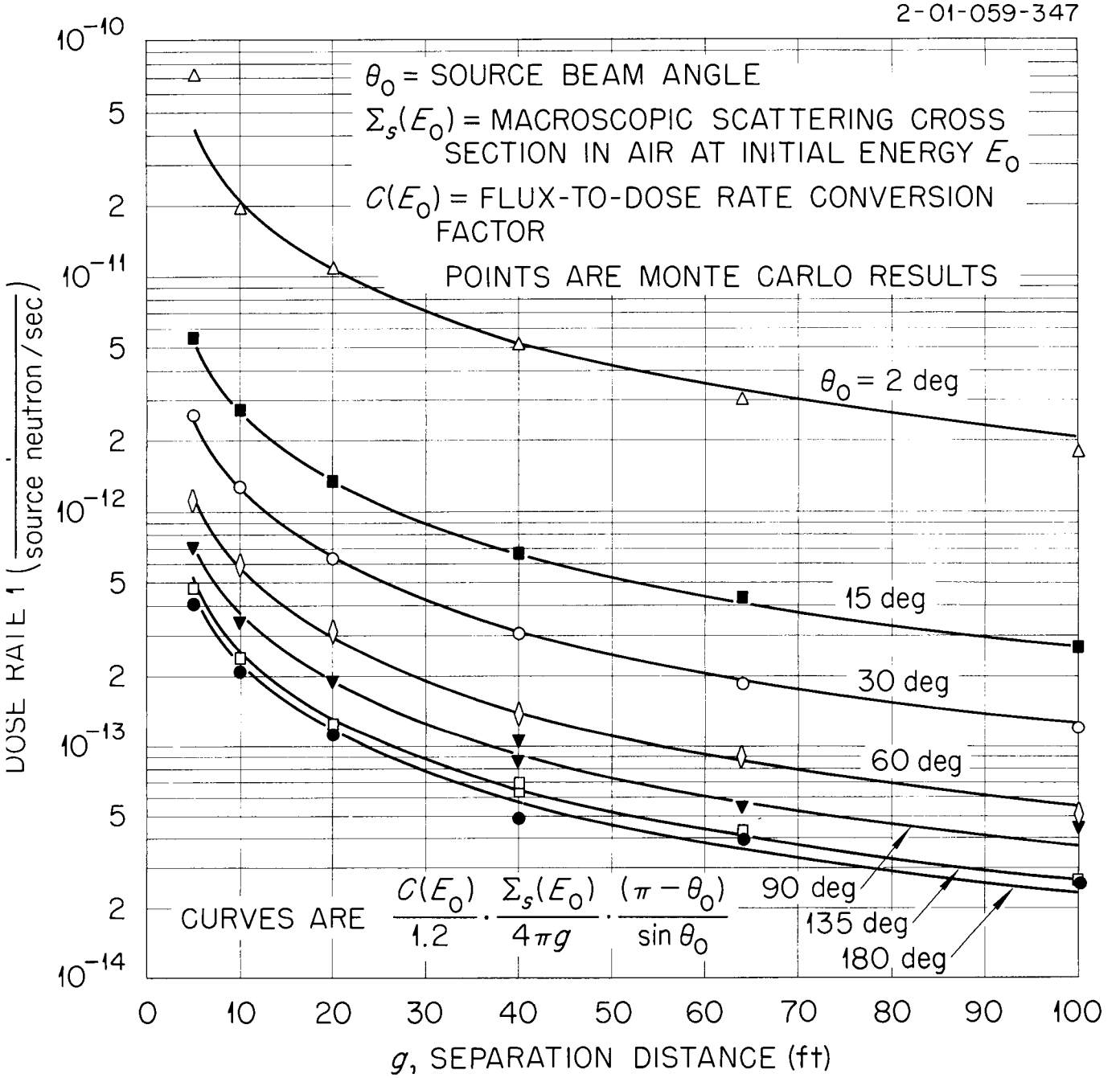

Fig. 10. Dose Rate in Air at a Source–Detector Separation Distance g Resulting from Multiply Scattered Neutrons. 2-Mev Line Beam Source.

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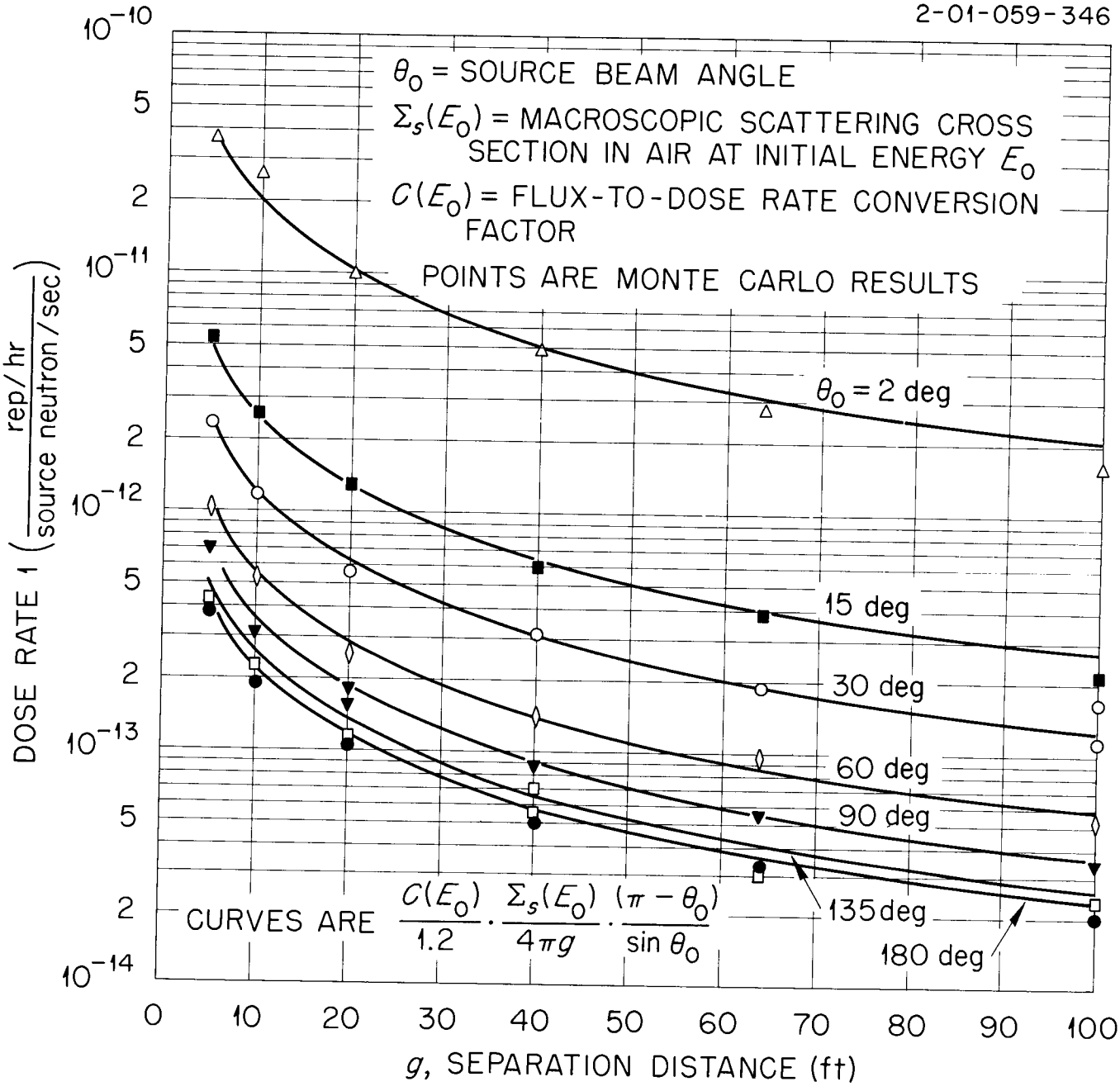

Fig. 11. Dose Rate in Air at a Source-Detector Separation Distance g Resulting from Multiply Scattered Neutrons. 1.2-Mev Line-Beam Source.

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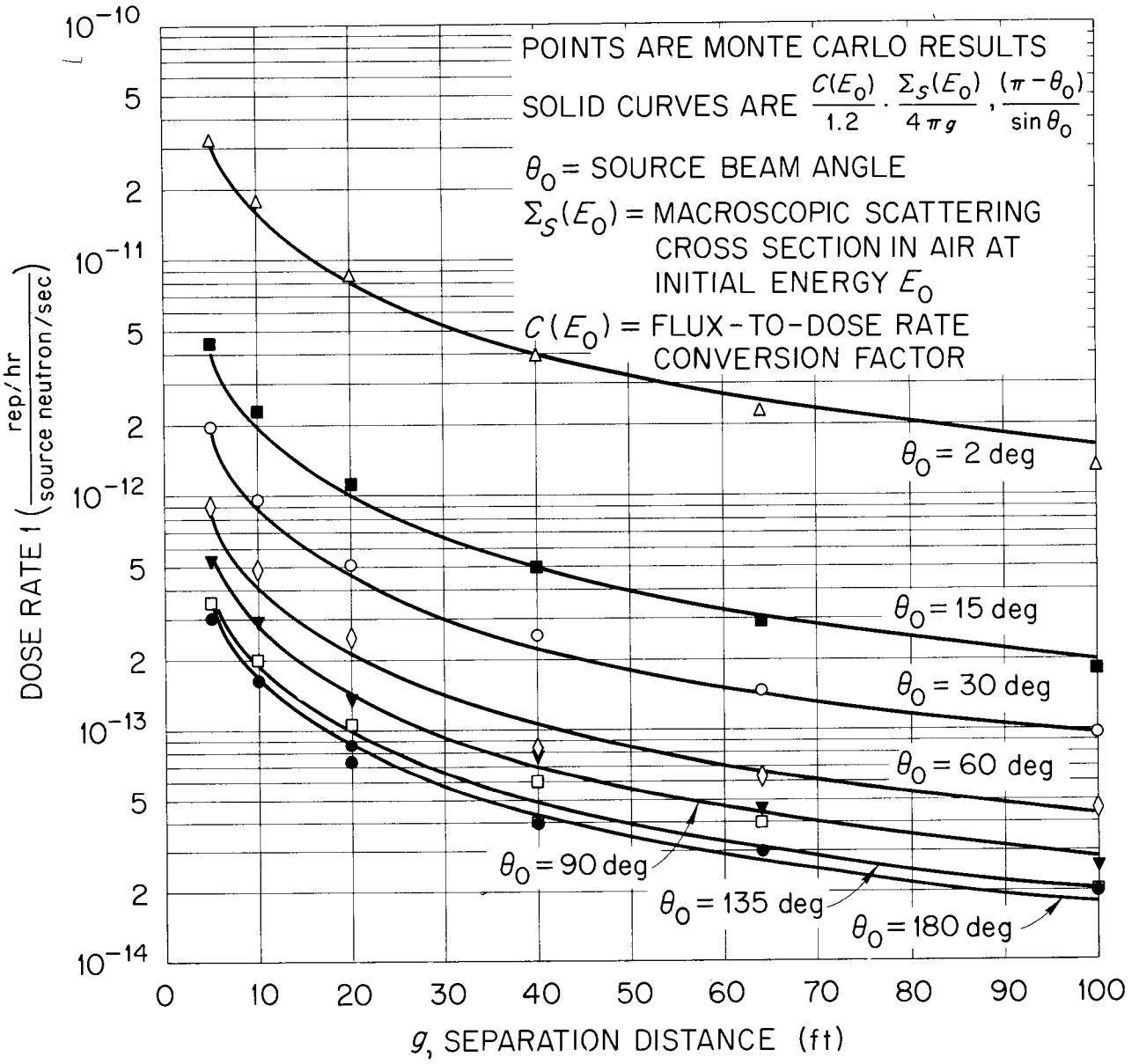

Fig. 12. Dose Rate in Air at a Source – Detector Separation Distance g Resulting from Multiply Scattered Neutrons. 0.55-Mev Line-Beam Source.

$$\frac{C(E_o)}{1.2} = \frac{\int_{E_c}^{E_o} C(E)\Phi_{E_o,\theta_o,g}(E)dE}{\int_{E_c}^{E_o} \Phi_{E_o,\theta_o,g}(E)dE} \tag{3}$$

If there were no energy degradation this average would just be $C(E_o)$. Hence, the factor 1/1.2 must be entirely due to the fact that there is energy degradation and the conversion factor varies with neutron energy. The fact that the same value, 1/1.2, applies for all initial neutron energies is not surprising in view of the results obtained for the flux and the fact that, from Fig. 2, over the range of energy of interest the conversion factor, C(E), changes by approximately the same fractional amount, at any energy, for a given fractional change in E. (A neutron, of course, loses the same fraction of its energy for a given angle of elastic scattering regardless of its original energy.) It is somewhat surprising, however, that this same factor applies as well as it does over such large variations in angle and source-detector separation distance.

As stated previously, all of the data presented in Figs. 3 through 12 were obtained for conditions of sea level air density. However, since the results are plotted as a function of source-detector separation distance they may easily be transformed to apply to other air densities by means of the transformations described in another report.[6] In fact, since Eqs. 1 and 2 give almost exact fits to the Monte Carlo results the altitude transformation may be made by merely substituting the correct total macroscopic scattering cross section appropriate to the altitude in question into these expressions. This, of course, will yield the same results as would be obtained from the transformation cited above.

The fraction of the total dose rate at the detector contributed by singly scattered neutrons is plotted versus source-detector separation distance for various angles of emission of a beam of 5-Mev neutrons in

6. C. D. Zerby, "Radiation Flux Transformation as a Function of Density of An Infinite Medium with Anisotropic Point Source," ORNL-2100 (Oct. 9, 1956).

Fig. 13. It is seen that the fraction of the total dose contributed by singly scattered neutrons decreases with increasing source-detector separation distance and increasing angle of emission. This set of curves should be very nearly the same for all of the source energies if the abscissa is expressed in dimensionless mean free paths rather than in absolute distance.

Plots of the angular distribution of flux and dose rate at the detector position were generated for all of the problems which were run with a source-detector separation distance of 64 ft. These plots were obtained by drawing smooth curves through the histogram outputs from the machine calculations. In order to increase the accuracy of the method, the contributions from single scattering were plotted separately from the contributions from higher orders of scattering. The curves for the total angular distributions were then obtained by addition. A representative set of curves for the angular distribution of dose from 5-Mev source neutrons is shown in Figs. 14 through 15. In Fig. 14 the dose rate per unit $\cos\alpha$ is plotted versus $\cos\alpha$ (α = polar angle at the detector) for the contribution from neutrons which have scattered only once. These curves must obviously go to zero at values of $\cos\alpha$ for which $\alpha \leq \pi - \theta_o$. Similar curves for the contribution from neutrons which have undergone more than one scattering are shown in Fig. 15. It is possible, of course, for these neutrons to enter the detector at any angle α. Finally, curves of the total dose rate per unit $\cos\alpha$ versus $\cos\alpha$ are shown in Fig. 16. The sharp breaks in these curves, of course, occur at the points where the single-scattering contributions go to zero.

The shapes of the curves obtained for the angular distributions of both flux and dose for other source energies were essentially the same as the shapes of the representative curves shown in Figs. 14 through 16, to within the accuracy with which smooth curves could be drawn through the histograms. Hence, good approximations to the total angular distribution curves for the other cases at 64 ft may be obtained by merely renormalizing the representative curves by ratios based on total dose rate or flux values from Figs. 3 through 12. At smaller source-detector separation distances the single-scattering contribution becomes more important, as seen in Fig. 13.

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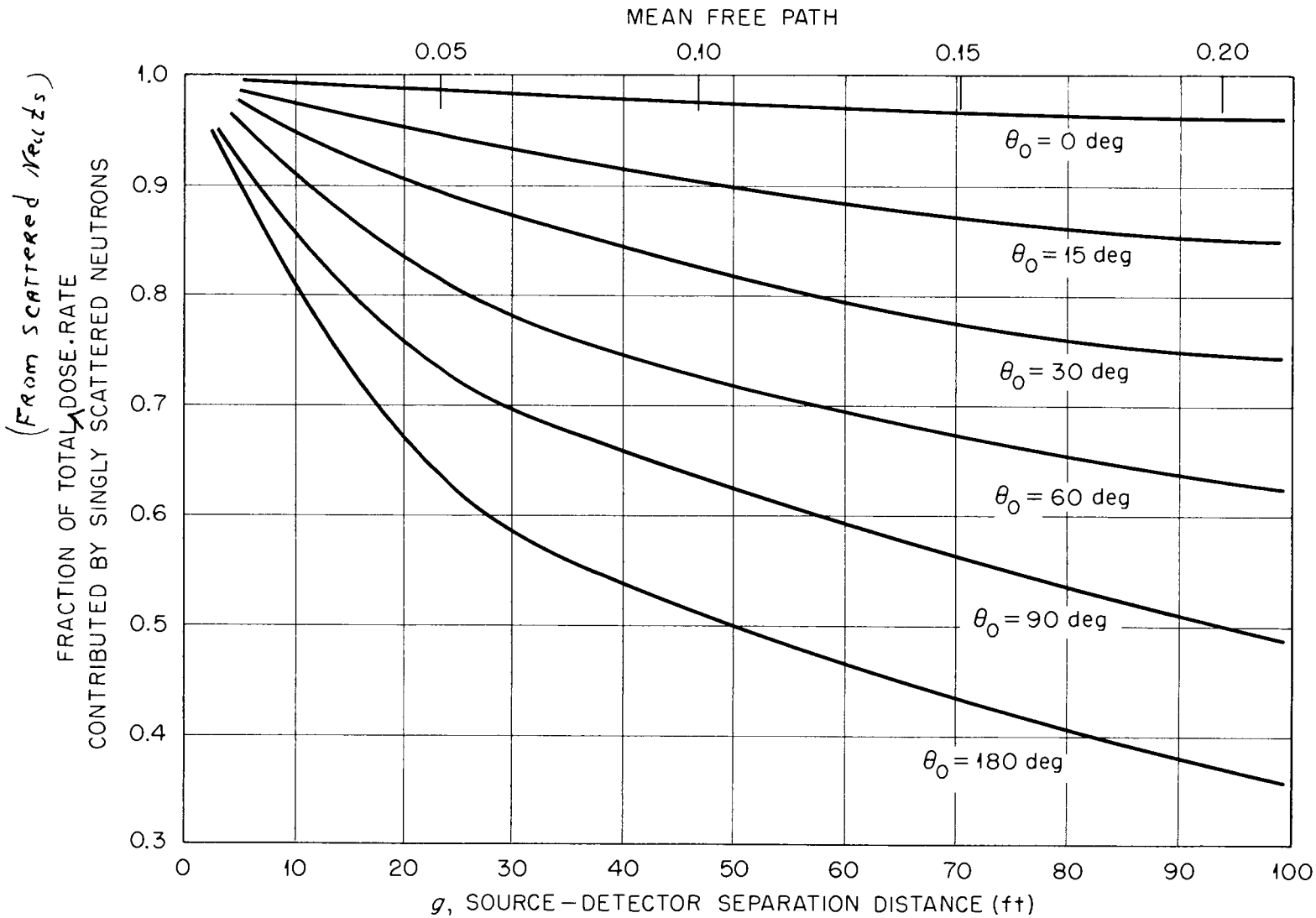

Fig. 13. Fraction of Total Dose Rate Contributed by Single Scattering as a Function of Source–Detector Separation Distance for Various Angles of Emission of a 5-Mev Neutron Beam.

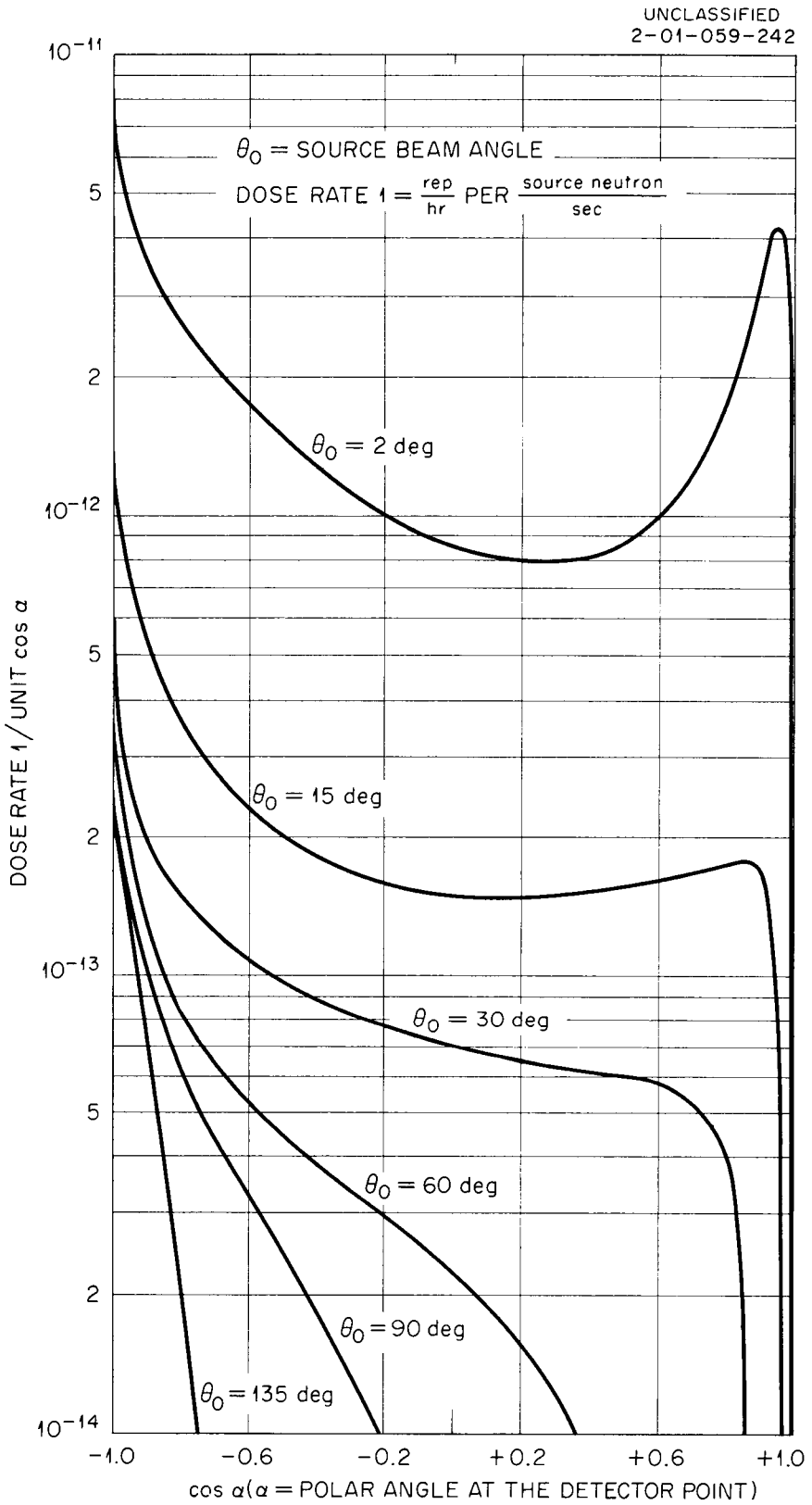

Fig. 14. Contribution to the Total Tissue Dose Rate in Air at a Source-Detector Separation Distance of 64 ft by Neutrons with Energy Greater than 0.1 Mev Which have Scattered Once in Air. The Source was a Point-Monodirectional Source of 5-Mev Neutrons. Air Density = 0.00125 g/cm^3.

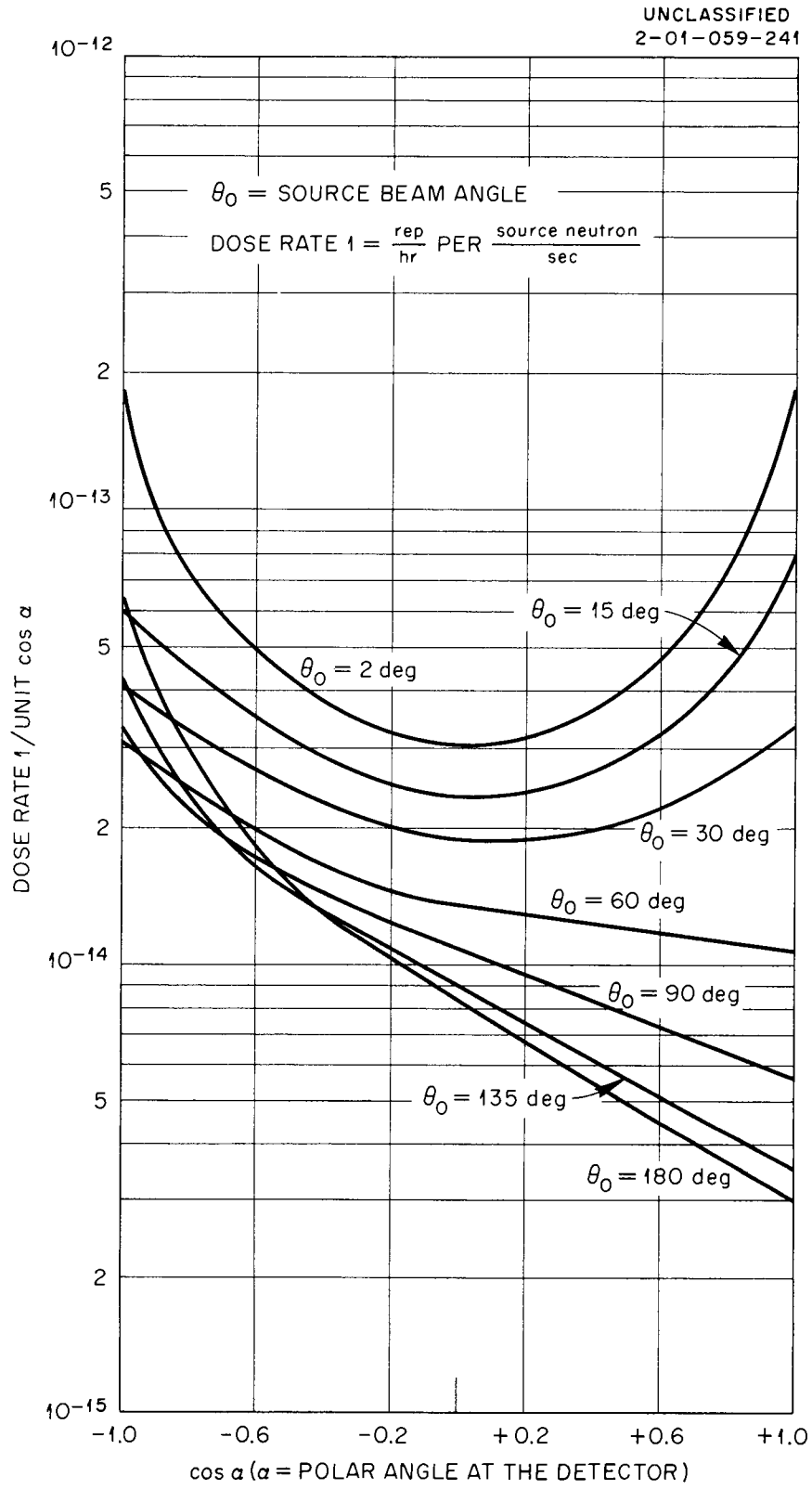

Fig. 15. Contribution to the Total Tissue Dose Rate in Air at a Source-Detector Separation Distance of 64 ft by Neutrons with Energy Greater than 0.1 Mev Which have Scattered in Air Two or More Times. Point-Monodirectional Source of 5-Mev Neutrons. Air Density = 0.00125 g/cm^3.

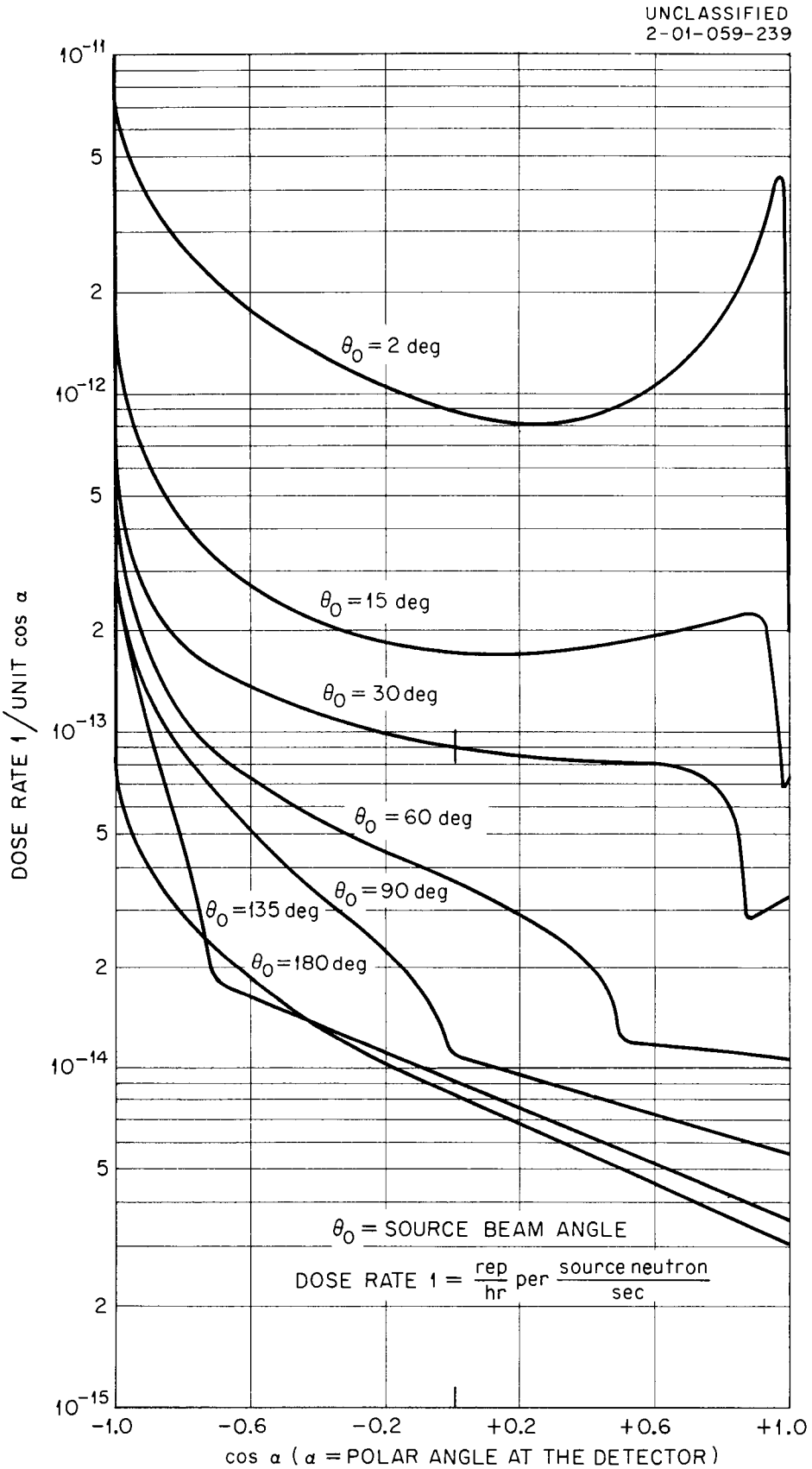

Fig. 16. Angular Distribution of Total Tissue Dose Rate in Air at a Source-Detector Separation Distance of 64-ft Resulting from Neutrons with Energy Greater than 0.1 Mev Point-Monodirectional Source of 5 Mev Neutrons. Air Density = 0.00125 g/cm^3.

The shapes of the angular distribution curves for single scattering, however, should not vary appreciably from those in Fig. 15 over the ranges of distances considered here. Hence, the curves of Fig. 13 can be used together with the values from Figs. 3 through 12 to renormalize the angular distribution curves of single-scattering contributions and multiple-scattering contributions separately (Figs. 14 and 15) to give approximate angular distributions at the other separation distances.

In all of the calculations reported here the scattering probability was taken as isotropic in the center-of-mass system. A discussion of the effects of anisotropic scattering on these results may be found elsewhere.[7]

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7. C. D. Zerby, "The Effect of Anisotropic Scattering on the Neutron Flux in Air," ORNL-CF-57-12-10 (1957).

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