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INDUSTRY PROGRAM OF THE COLLEGE OF ENGINEERING

STUDY OF GAMMA SHIELD REQUIREMENTS FOR AN IDEALIZED POINT U-235 NUCLEAR REACTOR

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PREFACE

In any reactor, the gamma shield is by far the heaviest single component. This report presents the results of a study of the minimum weights of gamma radiation shields which might be achieved for nuclear reactors in order to assure freedom from excessive exposure of personnel to radiation from the reactors. A nuclear reactor occupying only a "point volume" was assumed in this study, and the neutron radiation from the reactor was disregarded.

The point configuration of a reactor yields the lightest possible weight of shield for any desired power of the nuclear reactor. Any practical reactor would require greater shielding weights than those shown in this report, since the reactor would be larger than a point and since some additional weight of shield would be required to stop neutrons.

Consequently, the weights of shielding estimated in this report are believed to provide minimum weights of nuclear reactors for given ranges of power. This information will assist in determining ranges of shaft horsepower output required to meet allowable ratios of "pounds weight per shaft horsepower" in nuclear reactor systems. Such information may assist in defining the ranges of feasibility of nuclear power for mobile applications.

For those who do not wish to follow the details of the methods of calculation, a graphical summary of the results will be found in Figures 10 through 15. These results are easily extended to other reactor powers, and other point source spectra.

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ABSTRACT

The method and results of calculations are presented for the thickness, weight, and weight per shaft horsepower of a gamma shield for a point nuclear reactor having a U-235 fission spectrum. The shield materials considered are water, aluminum, iron, tin, lead, tungsten, and uranium. Also included in this report are methods of extending the results to other shielding problems.

The results are summarized in graphical form in addition to the detailed calculations presented in the text.

I. INTRODUCTION

With the increasing demand for the application of nuclear power in industry, one question which must always be considered as an initial step is whether a reactor can be constructed which will meet required specifications. In particular, for those applications in which the weight of the nuclear reactor is a limiting factor in the feasibility of the power supply, the weight of the gamma shield necessary for safety is often a crucial part of the problem.

The reason for this importance of gamma shielding where weight must be kept to a minimum will be evident from the following considerations. A reactor emits gamma radiation, neutrons, and other particles. The "other particles" are easily stopped by the shielding for neutrons and gamma radiation and thus are not of primary importance to shield weight. Neutron shielding requires a material of low atomic mass for thermalization and an absorbing material to catch the neutrons once they are thermalized. The difficulty encountered is that when neutrons are slowed down (thermalized) and captured they often produce gamma radiation which must be stopped. The gamma radiation arising from the fission process in the reactor can be attenuated at any location outside the reactor core, but the neutron capture gammas produced by the capture of neutrons in the shield can be stopped only outside of the neutron shield. Conversely, it is also possible for the original fission gammas to produce neutrons, which must in turn be absorbed. However, the absorption of these neutrons requires only a thin layer of the light weight neutron shield material and therefore is not of primary importance in reactor weight considerations.

Gamma radiation is most efficiently absorbed by a dense material of high atomic number. With the use of such a material, the thickness and weight of the gamma shield are reduced to the lowest possible values. The thickness of such a shield for gamma radiation will ordinarily be less than the thickness of the neutron shield. However, because of the high density of the gamma shield material, and because of the necessity of shielding neutron capture gammas, the gamma shield will usually be of much greater weight than the neutron shield of a nuclear reactor.

Therefore, the gamma shield should be one of the first components considered where the weight of the nuclear reactor is a required specification.

In recent years, extensive calculations have been performed for the attenuation of gamma radiation by shielding materials. It is the purpose of this report to give numerical values for shielding weights and shielding thickness in the form of easily read graphs using the results of these calculations. Reactors are, of course, being designed and built in many different forms and for many different purposes. It would be much too ambitious to attempt to devise one simple formula for shielding any reactor. However, by considering only a point reactor surrounded directly by its gamma shield, it is hoped that the relative merits of different shielding material may be more easily visualized, and particularly, that some insight may be gained into the requirements of shielding relative to the power level desired.

To accomplish these objectives, a point gamma source emitting the spectrum of a typical U-235 reactor has been assumed. A means has been given, however, for applying the results of this report to the shielding of any strong gamma source without re-doing the longer calculations performed in arriving at the initial set of dose-rate versus shielding-thickness curves.

The initial calculations have been carried out for a one megawatt reactor, but could equally well have been performed for any other reasonable power since the dose rate solved for is linearly related to the power level.

The fact that the reactor and the neutron shield are considered reduced in size to a point instead of being one of the usual shapes and of finite size cannot be dismissed easily. With this assumption the required gamma shield thickness determined will be larger at a given power level than for any finite size of reactor. However, due to the negligible size of the reactor a very useful bit of information is obtained. The weight of this gamma shield is the absolute minimum that can be allowed for a reactor of the given power if the reactor is to be safe. In other words, when using fissioning uranium for a power source, no amount of careful designing can result in a power plant which weighs less than the values obtained for this case if the reactor is to be safely approached by personnel outside the shield.

II. THEORETICAL CONSIDERATIONS

In this paper, the shielding of an idealized point U-235 reactor is considered. The purpose of the paper is to determine the minimum weight of reactor gamma shield for safe operation of a nuclear power plant. The gamma shield has a spherical geometry and is placed immediately adjacent to the core. This is shown in Figure 1. The source reactor core is at the origin "O" and the dosage is taken adjacent to the outside of the gamma shield at distance "r".

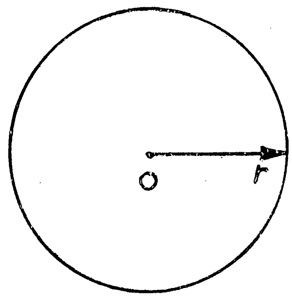

Figure 1. Point Reactor Shield Geometry

The gamma spectrum employed as the basis for calculation in this report is that of an operating U-235 reactor. For this reason, gammas in the reactor core resulting from processes other than fissioning are tacitly included in the U-235 fission spectrum.

Neutron capture gammas produced in the nuclear reactor shield are neglected in this paper. A calculation to determine the magnitude of the error introduced by this omission was performed using cadmium as the thermal neutron shield material. Cadmium, of all the thermal neutron absorbers would most alter the gamma spectrum shielded, and therefore this calculation would yield the greatest possible error. Even in this case, the error introduced is within the accuracy of the calculation for gamma shield thickness and weight, and thus the effect of neutron capture gammas on gamma shield thickness is negligible.

Since the reactor is a point, the shield is a solid sphere. This configuration gives the greatest shield thickness necessary to attentuate gamma radiation to any set dose rate. If the shield had a spherical shell configuration around an actual reactor, the shield thickness would be reduced due to the inverse square law; but for reduction of the radiation to tolerance the weight of the shield would still be greater than that around an equivalent point source. Thus the case considered gives the minimum weight of shielding required.

III. METHODS

The gamma spectrum that is used in this report was obtained from experimental data taken at the Los Alamos Water Boiler and Fast Reactors, by J. W. Motz[9]. The original data gives the gamma ray spectra of the Los Alamos Reactors in units of photons per Kev interval per square centimeter per second.

A graphical integration of this data was performed by R. Stephenson,[12] and the gamma spectrum per fission for the energy ranges of 0-2, 2-4, 4-6, 6-8, and 8-10 Mev obtained. Stephenson assumed 12.2 Mev per fission released in the form of gamma rays. This gamma ray spectrum includes prompt, fission product, and U-235 capture gammas.

Another experimental spectrum has been obtained by Maienschein and Love[6] from the Bulk Shielding Reactor at the Oak Ridge National Laboratory. The BSR is a swimming pool type with aluminum clad enriched fuel elements. A rough graphical integration of the data was performed and the spectrum normalized to 12.2 Mev per fission as before.

A comparison of the spectra from the two laboratories was made which showed the same general profile with minor variations. The variations are probably introduced by the difference in structural material in the reactors and the nuclear interactions with these materials. It should be noted however, that there is a marked difference in the extension of the data curve from the Los Alamos Reactors to cover the 8 to 10 Mev range and the curve of recorded data from the BSR at ORNL. The former yields a continuous curve, the latter a sudden fall off of several decades in the 8 to 10 Mev range. However, this is a minor detail where this report is concerned as it will be seen that the dose rate contribution of this range is small compared to that of the intermediate ranges.

The gamma spectrum per fission used as a basis for calculation is shown in Table 1.

TABLE 1. U-235 GAMMA SPECTRUM PER FISSION[12]

Energy Interval Mev	Average Photon Energy Mev	Number of Gammas/Fission	Total Energy Mev
0-2	1	9.31	9.31
2-4	3	0.75	2.25
4-6	5	0.099	0.495
6-8	7	0.154	0.1078
8-10	9	0.0029	0.0261

A verification of the spectrum was made by performing a graphical integration for the area under the spectrum curve obtained by Motz[9]; and 12.2 Mev per fission produced in the form of prompt, fission product and fuel capture gammas is in reasonable agreement with the values used by other authorities. These authors are listed in the references (2, 3, 5, 7, 10, 12).

An extensive series of calculations on the penetration of gamma rays in infinite homogeneous media has been performed by H. Goldstein and J. E. Wilkins, Jr.[4] with the aid of a high speed automatic computer (the SEAC). In the report containing these calculations are the results obtained using a monoenergetic point isotropic source, the differential energy spectra in several materials after various numbers of relaxations are tabulated.

Plots of the tabulated values of the differential energy spectra versus the energy of the emergent photons were made from these machine calculations for each incident energy and several thicknesses of shield (numbers of relaxations). Using the gamma spectrum given by Stephenson[12] the assumption was made that all of the photons emitted in any one 2 Mev energy range are monoenergetic at the median energy, i.e., in the 0-2 Mev range 9.31 photons are emitted; all of these are considered to have an energy of 1 Mev. (Table 1).

Calculations are all based on the point reactor source having a power of one megawatt. The results of the calculations are easily modified for any desired source power.

The differential energy spectra obtained by Goldstein and Wilkins[4] is in units of photons/sec-Mev. The detailed method of calculating dose rate as a function of shield thickness for any one shield material using this information is as follows. After plotting curves of emitted differential energy spectra versus energy from the calculations of Goldstein and Wilkins for the shield material considered the calculations proceed in tabular form (Table 2 contains a sample calculation for iron). The incident gamma spectrum is divided into monoenergetic beams and the gamma linear absorption coefficient is obtained for each incident energy beam from Moteff[8] and/or Goldstein and Wilkins.[4] A separate tabulation of results is made for each value of $\mu_o r$ (number of relaxations) used.

Select one value of $\mu_o r$. From the known value of μ_o, r is found for each incident beam, i.e., 1, 3, 5, 7, and 9 Mev. The differential energy spectrum at this value of $\mu_o r$ is given by Goldstein and Wilkins as a plot of $4\pi r^2 e^{\mu_o r} I_o$ (I_o is the emitted gamma ray intensity) versus the energy in Mev of the emitted photons for each incident monoenergetic beam. By dividing the quantity $4\pi r^2 e^{\mu_o r} I_o$ by $4\pi r^2 e^{\mu_o r}$ for each incident energy, we have the intensity of radiation spectrum emitted from the shield for each incident energy. The emitted spectra resulting from each of the monoenergetic incident beams are now divided into 0-2, 2-4, 4-6, 6-8, and 8-10 Mev groups, and the average number of photons in each range times the number of Mev intervals spanned in the range yields the number of photons in any one group. This average is actually a weighted one as it is the purpose of this step in the calculation

TABLE 2. SAMPLE CALCULATION - IRON

$\rho = 7.60$ gm/cm^3 $\quad Z = 26 \quad A = 56$

$\mu_o r = 20 \quad e^{\mu_o r} = 4.85 \times 10^8$

E_o Incident Gamma Energy (Mev)	Linear γ Abs. Coef. μ_o (cm^{-1})	Shield Thickness r(cm)	$4\pi r^2$	No. of Photon Per Energy Interval ($n_o n_i$)	$\frac{n_o n_i}{4\pi r^2 e^{\mu_o r}}$	Emergent γ Spectrum (Mev)	Phot/sec-Mev in Emergent γ Spectrum [$4\pi r^2 e^{\mu_o r} I_o$]	Phot/cm^2-sec in Emergent γ Spectrum	Dose Rate Contribution (mr/hr)	Total Dose Rate (mr/hr)
1	0.464	43.1	2.09×10^4	2.89×10^{17}	2.85×10^4	0-2	50	1.43×10^6	2.9×10^3	2.9×10^3
3	0.270	74.1	6.90×10^4	2.325×10^{16}	6.95×10^2	0-2	32	2.22×10^4	4.5×10	93
						2-4	15	1.04×10^4	4.8×10	
5	0.234	85.5	9.19×10^4	3.07×10^{15}	6.89×10	0-2	25	1.72×10^3	3.5	
						2-4	25.4	1.75×10^3	8.0	15.9
						4-6	10	6.89×10^2	4.4	
7	0.223	89.7	1.14×10^5	4.77×10^{14}	8.63	0-2	22	1.90×10^2	3.8×10^{-1}	
						2-4	23	1.98×10^2	9.0×10^{-1}	2.87
						4-6	19.4	1.67×10^2	1.08	
						6-8	7.3	6.30×10	.51	
9	0.223	89.7	1.14×10^5	9.0×10^{13}	1.63	0-2	20	32.6	$.67 \times 10^{-1}$	
						2-4	22	35.9	1.65×10^{-1}	.717
						4-6	19.8	32.3	2.1×10^{-1}	
						6-8	14.4	23.5	1.9×10^{-1}	
						8-10	5.3	8.64	$.85 \times 10^{-1}$	

to place a number of photons at the median energy of the group which will yield an equivalent effect to that of the gamma radiation throughout the range. This number of photons is now assumed to be monoenergetic at the median energy of the range, and the flux is converted to dose rate by means of plots of the equation (see Figure 2):

$$\text{Dose Rate in mr/hr} = \frac{3600\ (\Sigma_r\text{-}\Sigma_s)_{air}\ E\Phi}{6.77 \times 10}$$

where

E = gamma energy in Mev

Φ = gamma flux in photon/cm^2/sec

$(\Sigma_r\text{-}\Sigma_s)_{air}$ = differential gamma cross section of air in cm^{-1}

The tabulation thus far yields a separate table for each $\mu_0 r$. Each table contains five rows (corresponding to five monoenergetic beams); and each row contains a value of r and the dose-rate at that r. (See Table 2). For each material studied four values of $\mu_0 r$ were used. Thus by plotting the dose rate versus r for one incident energy for all values of $\mu_0 r$, the contribution to the total dose rate by that single incident energy is obtained in the form of a curve. For a point source reactor, this curve was found to be a straight line on semi-log paper after a few relaxations.* This operation is performed for each incident energy giving five dose rate contribution curves for each material. Figures 3 through 9 contain these straight line curves for the shield materials uranium, tungsten, lead, tin, iron, aluminum, and water. The coordinates of two points on each curve are given in order that the curves may be accurately replotted for further calculations.

By a simple addition of the dose rates due to each incident gamma radiation beam (ordinates of the curves in Figures 3-9) at different values of r (shield thickness), a curve of total dose rate versus shield thickness is obtained for each shield material for a 1 megawatt U-235 point reactor. (See Figures 10 and 11). For any desired dose rate, a plot of shield weight versus power or weight of shield per unit of power versus power is easily found. The method for doing this for the general case of a source power other than one megawatt or for a source spectrum different from the one used in this report is included in the section entitled "Extension of Results". Also included in this section is the explanation of how to obtain a plot of source power versus shield thickness for any source spectrum.

For the case of a reactor with a U-235 spectrum and an assumed safe dose rate of 6 mr/hr, curves of reactor power versus shield thickness are

* Since this report is concerned with the reduction to tolerance of gamma radiation from a strong source, no study has been made of the attenuation of gamma rays in the first few relaxations where a simple exponential relation would probably not be found.

given. For 20 percent thermal efficiency, curves of shield weight versus shaft horsepower and curves of shield weight per shaft horsepower versus horsepower are given. These curves are shown in Figures 12 through 15.

In order to obtain the curves for the weight of shield material versus shaft horsepower for the various materials discussed (Figure 12) and the curve for weight of shield per shaft horsepower versus shaft horsepower (Figure 13), it is necessary to discuss the basis of Figures 14 and 15 (power as a function of shield thickness).

It will be noticed that the curves drawn through the calculated points in Figure 10 can be closely approximated by straight lines. The curves from which Figure 10 is plotted (Figures 3 through 9) are all straight lines. The positions of the ordinates on these curves (Figures 3 through 9) are linearly dependent on the number of photons per second emitted in the energy range corresponding to the curve; thus for the U-235 source, changing the power level merely moves the curves up or down. For example, if the source power were increased by a factor of 10, each of the dose contribution curves would be raised by one decade; and as Figure 10 is merely a summation of the curves in Figures 3 through 9, each point on the ordinate of the curves of Figure 10 would be raised by one decade. This is equivalent to replotting Figure 10 exactly and relabeling the ordinate in decreasing reactor power units of megawatts. Such a set of curves must be made for each dose rate desired because the unit of one megawatt power would fall on the ordinate point in Figure 10 where the desired dose rate is located. As an example, suppose the thickness of shield to reduce the gamma radiation to 3 mr/hr is desired; then by labeling the ordinate on the new plot of power versus shield thickness such that one megawatt corresponds to the point of 3 mr/hr on the dose rate versus shield thickness curve (Figure 10), the shield thickness of a 0.1 megawatt source would correspond to the point for 30 mr/hr on the dose rate curve and the 10 megawatt point would be that of the 0.3 mr/hr. This process effectively does nothing but shift the ordinate of Figure 10. Sample curves of reactor power versus shield thickness, are plotted for the shield materials discussed assuming a desired dose rate of 6 mr/hr.

Using the data available in a set of curves such as those shown, Figures 14 and 15, plots of weight of shield versus shaft horsepower and weight of shield per shaft horsepower versus shaft horsepower are readily calculated. Examples of such plots for a dose rate of 6 mr/hr are shown in Figures 12 and 13.

IV. RESULTS

The results of this study are briefly as follows:

Calculations are carried through to the graphical representation of horsepower versus minimum weight of shielding and weight of shielding per shaft horsepower versus shaft horsepower for each of the several materials considered. The calculations are all based on the gamma spectrum of an operating reactor and on an assumed plant efficiency of 20 percent.

V. EXTENSION OF RESULTS AND POTENTIAL FOR FURTHER STUDY

In the section entitled "Methods", the means of altering the presented calculation for a source power other than a one megawatt U-235 reactor has already been outlined. The final curves of weight versus shaft horsepower, and weight per shaft horsepower versus shaft horsepower are calculated by the use of the following equation:

$$\text{Shaft horsepower} = 1.341 \times 10^3 \times \text{percent plant thermal efficiency} \times \text{reactor power (in megawatts)}$$

and

$$\text{weight in tons} = 4\pi r^3 \rho \times 1.1023 \times 10^{-6}$$
(2000 lbs/ton)

where

ρ is the density of the shield material in grams/cm^3, and

r is the shield thickness in centimeters.

To use the results of this report for the more general problem of any source spectrum which lies in the energy range 0-10 Mev for any of the shielding materials discussed, the following technique is employed. Divide the new incident gamma spectrum into the energy ranges 0-2, 2-4, 4-6, 6-8, and 8-10 Mev as was done for the U-235 fission gamma spectrum. Determine the number of photons per second in each energy group and treat them as monoenergetic at the median energy in the method outlined. Then divide the number of photons per second in each energy group of the new gamma spectrum by the number of photons per second in the corresponding energy group of the U-235 reactor spectrum. (To find the number of photons in the corresponding energy group of the U-235 reactor spectrum, multiply the values in Table 1 by 3.1×10^{16} fissions per second for a one megawatt reactor.) Multiply the ordinate of the corresponding curve found on one of the Figures 3-9 by this quotient and the result is a new dose rate contribution curve for the new incident spectrum. This procedure must be followed for each of the five dose rate contribution curves of any one shield material; each time dividing corresponding numbers of photons of the new spectrum by the U-235 fission spectrum times 3.1×10^{16}. This gives five new dose rate contribution curves for each material for the new spectrum. From this point, the calculations for the new spectrum are identical to the ones employed in this paper.

The problem of extending the results of this report to other geometries is a difficult one, but some simplifying approximations can be made. In the case where reactor core geometries can be equated to an equivalent spherical core and the gamma spectrum at the surface of the core can be determined, at distances from the center of the core large compared to the radius, the core can be assumed to be a point reactor with this spectrum. The total number of photons per second emitted from the core will yield an equivalent power level for the point reactor. The shield thickness needed can now be found by following the methods of calculation presented in this paper or by using one of the plots in Figure 10 or Figure 14 if they apply. It must be noted however that finding a 1/10 value thickness and multiplying by a suitable factor would probably yield better results than calculating for tolerance in one step. This method is very rough, but is useful as a first approximation for the amount of gamma shielding required for an actual reactor. Further development of this problem of finite geometries may well yield accurate methods for the calculation of gamma shield thickness from the results of this paper.

An attempt was made to find a general correlation formula for finding the amount of shielding needed for a reactor as a function of the gamma cross section, density, atomic number, and atomic mass of the shielding material. It is thought that this may be possible because the curves of dose rate versus shield thickness are very nearly straight lines on semi-log paper. (See Figure 10.)

These straight lines are of the general form:

$$\text{Dose rate} = \alpha\, e^{-\Sigma_{ave} r}$$

where

> α is the intercept at $r = 0$, and Σ_{ave} is an average gamma linear absorption cross-section. Both are probably functions of density, cross-section, atomic number, and atomic mass.

As yet, no general correlation has been arrived at. It is to be noted however that the lowest point on the total linear gamma cross-section curve[8] in the 0-10 Mev range for the materials studied is approximately 80 to 85 percent of the graphically determined values of Σ_{ave}. The possibility of a general correlation definitely warrants further investigation. If such a general correlation cannot be found for all materials, it is still possible by a very involved technique to calculate shield thickness for any element or material; for some materials, however, this might necessitate a calculation of the type performed by Goldstein and Wilkins.[4]

The calculations made by Goldstein and Wilkins on which this report is based are for infinite homogeneous media, but the results are presently being employed for finite layer shields with results that are in excellent agreement with experiment data. There are however certain rules of thumb which must be followed for good results in calculations in the cases where laminar shielding is being used, but for a shield of a single material the the calculations fit experimental data with no alteration or special technique of application.

For thick layers in a laminar shield the present calculations can be used by assuming the exterior of each successive shield as a new reactor with a different gamma spectrum and utilizing the technique mentioned with reference to the extension of point source calculations to other geometries. One precaution must be observed in this procedure, however, and this is: the outside radius of the "new reactor" must be small compared to the distance from the center of the core to the point at which the radiation would first come in contact with operating personnel.

The chief merit of this paper in its present state is that it will yield formally accurate results for any geometry which can be approximated by a point source. Examples of this are small spherical reactor cores requiring a shield thickness of a magnitude much greater than the core radius, and containers for holding or transporting materials which emit a high gamma flux in the 0-10 Mev energy region. Care must be taken not to extend these results to the shielding of sources so weak that less than about four relaxations are required for tolerance dosage, but there seems to be no upper limit to the number of relaxations for which a linear extrapolation of the results is not accurate.

VII. NOMENCLATURE AND CONVERSION FACTORS

mr/hr = milliroentgen per hour

Mev = million electron volts

r = shield thickness (cms)

ρ = density (gms/cm^3)

Σ = total macroscopic cross-section (cm^{-1})

Σ_s = macroscopic scattering cross-section

μ_o = gamma linear absorption coefficient (cms^{-1}) at energy E_o

n_o = number of fissions per second for a particular reactor power level

n_i = number of photons per fission emitted from a reactor core in the ith energy group

I_o = number of photons per second per Mev emitted from a shield material of a particular thickness at any single energy

E_o = energy of the incident gamma beam

1 watt = 3.1×10^{10} fissions of U-235 per second (at 200 Mev per fission)

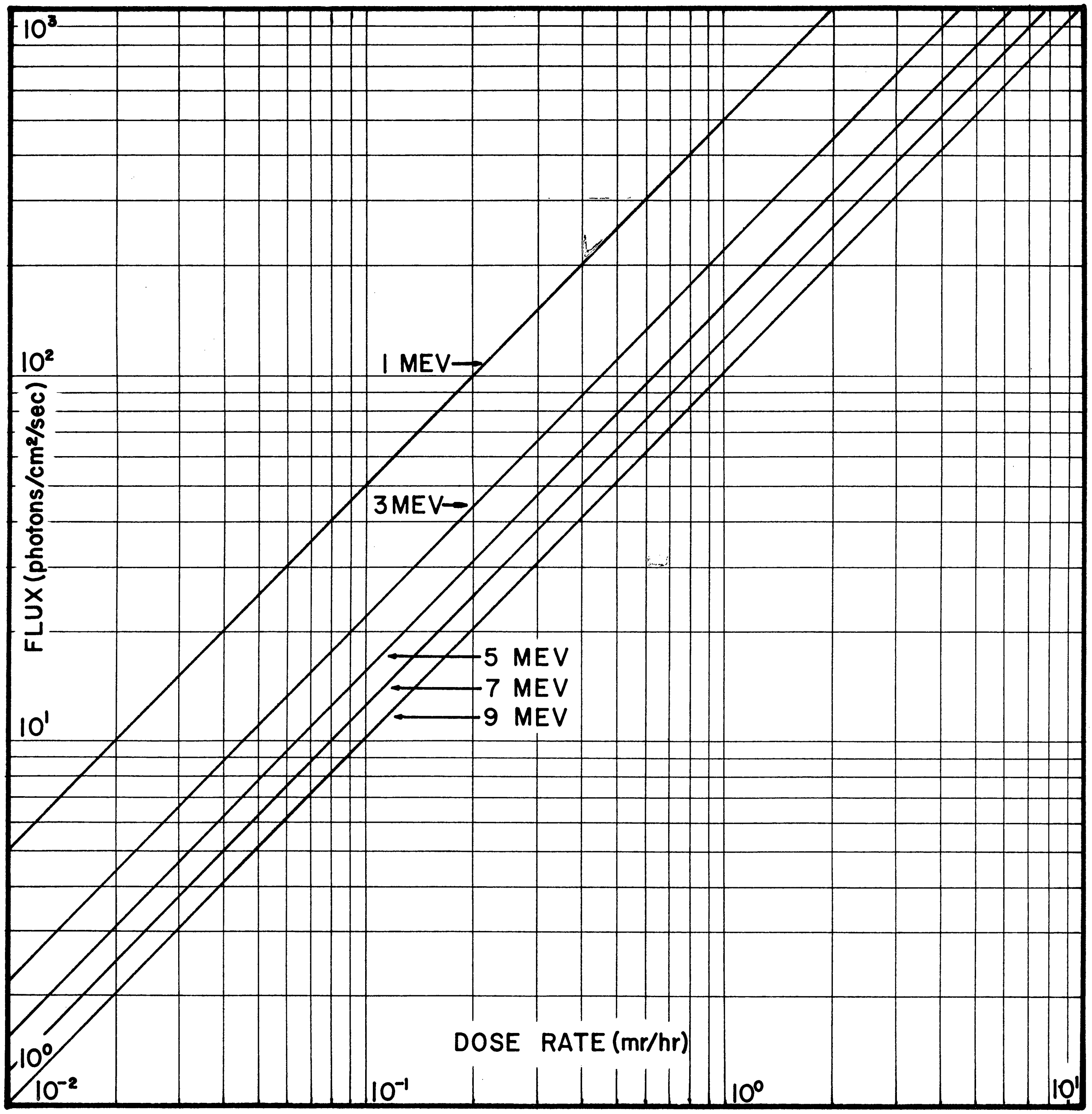

FIG. 2
FLUX-DOSE RATE CONVERSION CURVES

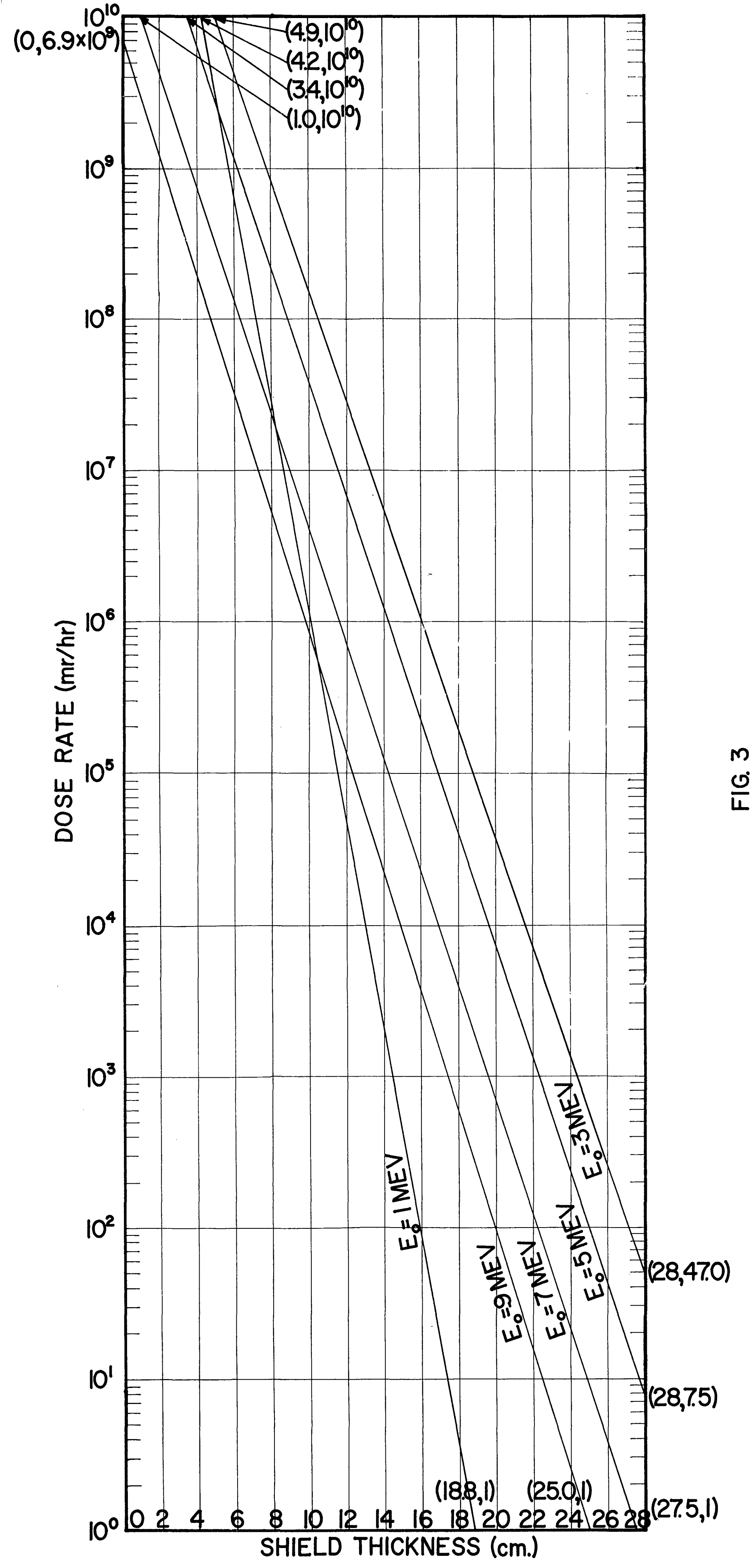

FIG. 3

DOSE RATE CONTRIBUTION CURVES

SHIELD MATERIAL - URANIUM

SOURCE - 1 MEGAWATT U^{235} POINT REACTOR

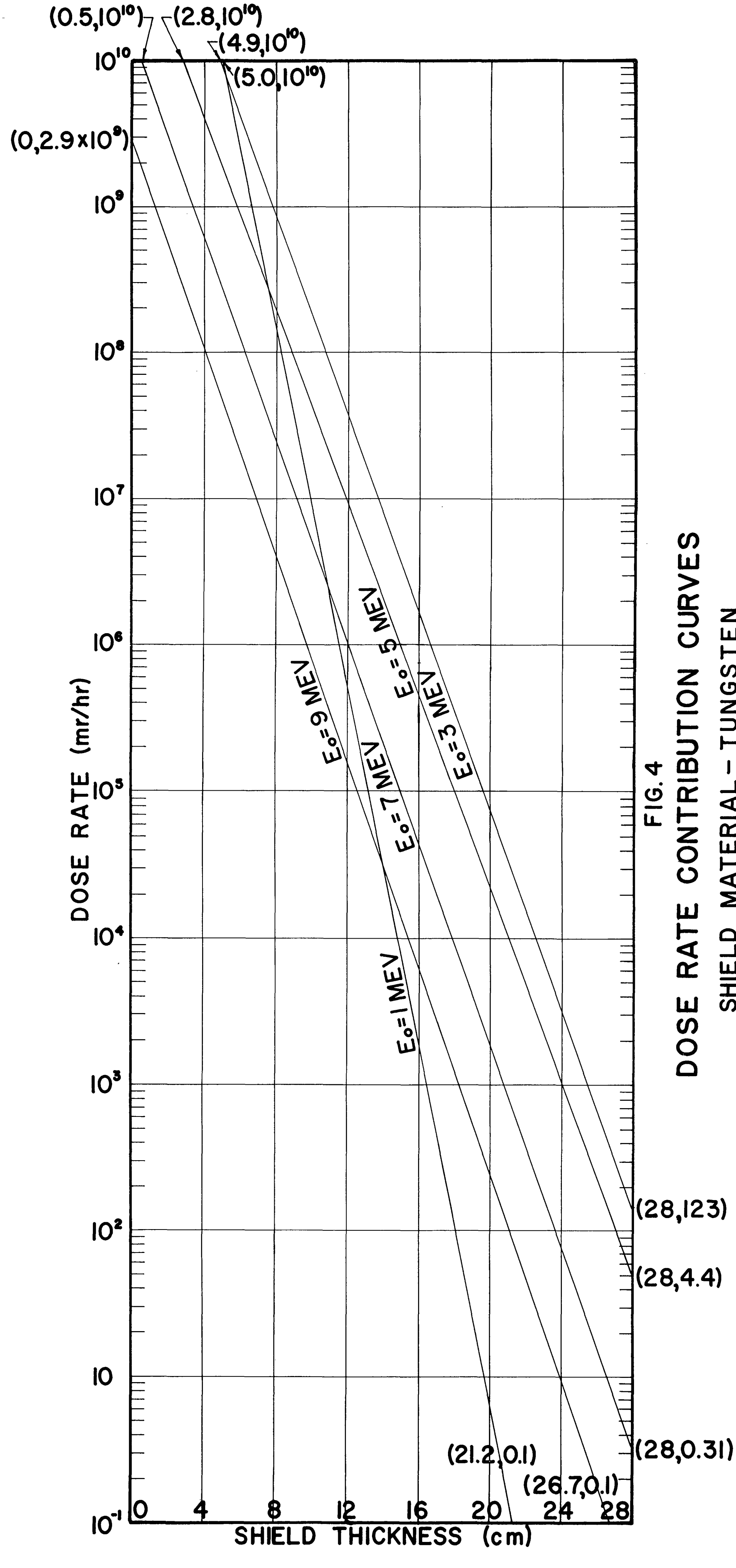

FIG. 4

DOSE RATE CONTRIBUTION CURVES

SHIELD MATERIAL – TUNGSTEN
SOURCE – 1 MEGAWATT U^{235} POINT SOURCE

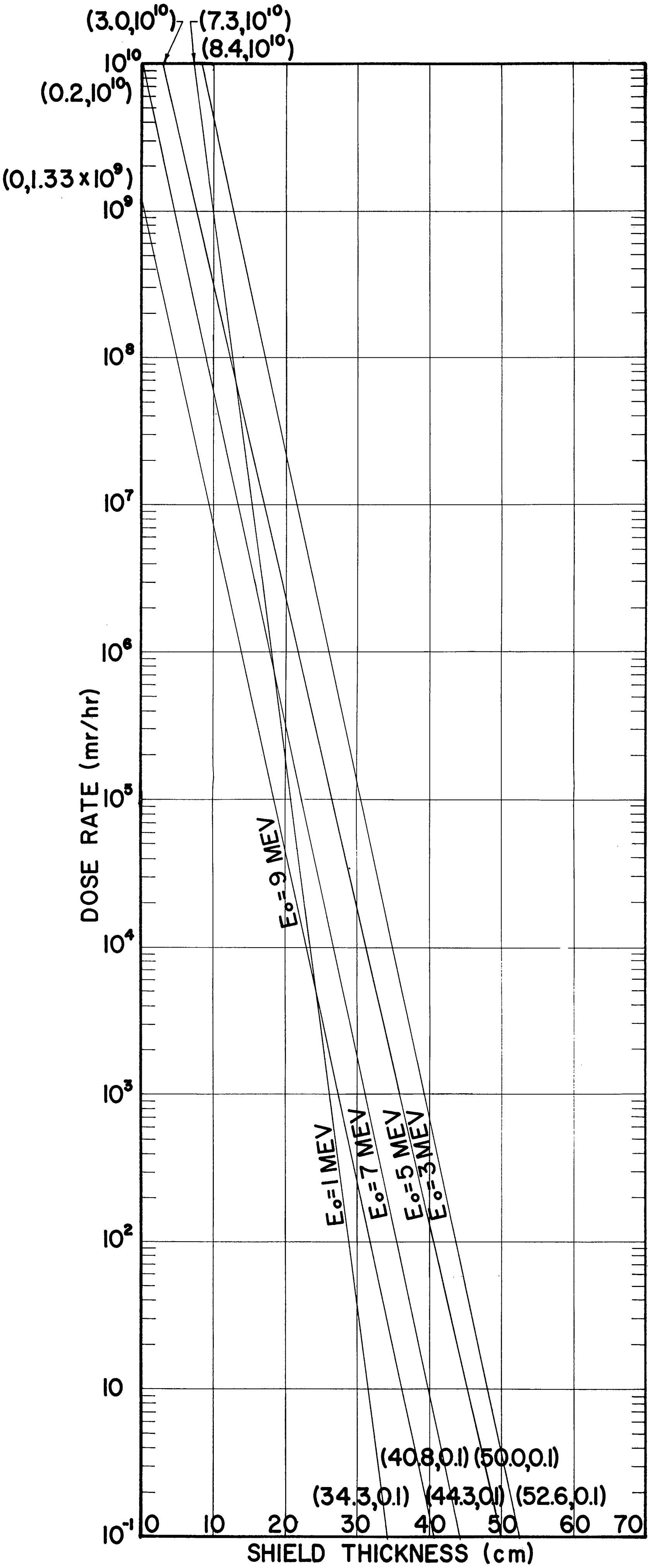

FIG.5

DOSE RATE CONTRIBUTION CURVES

SHIELD MATERIAL - LEAD
SOURCE - 1 MEGAWATT U^{235} POINT REACTOR

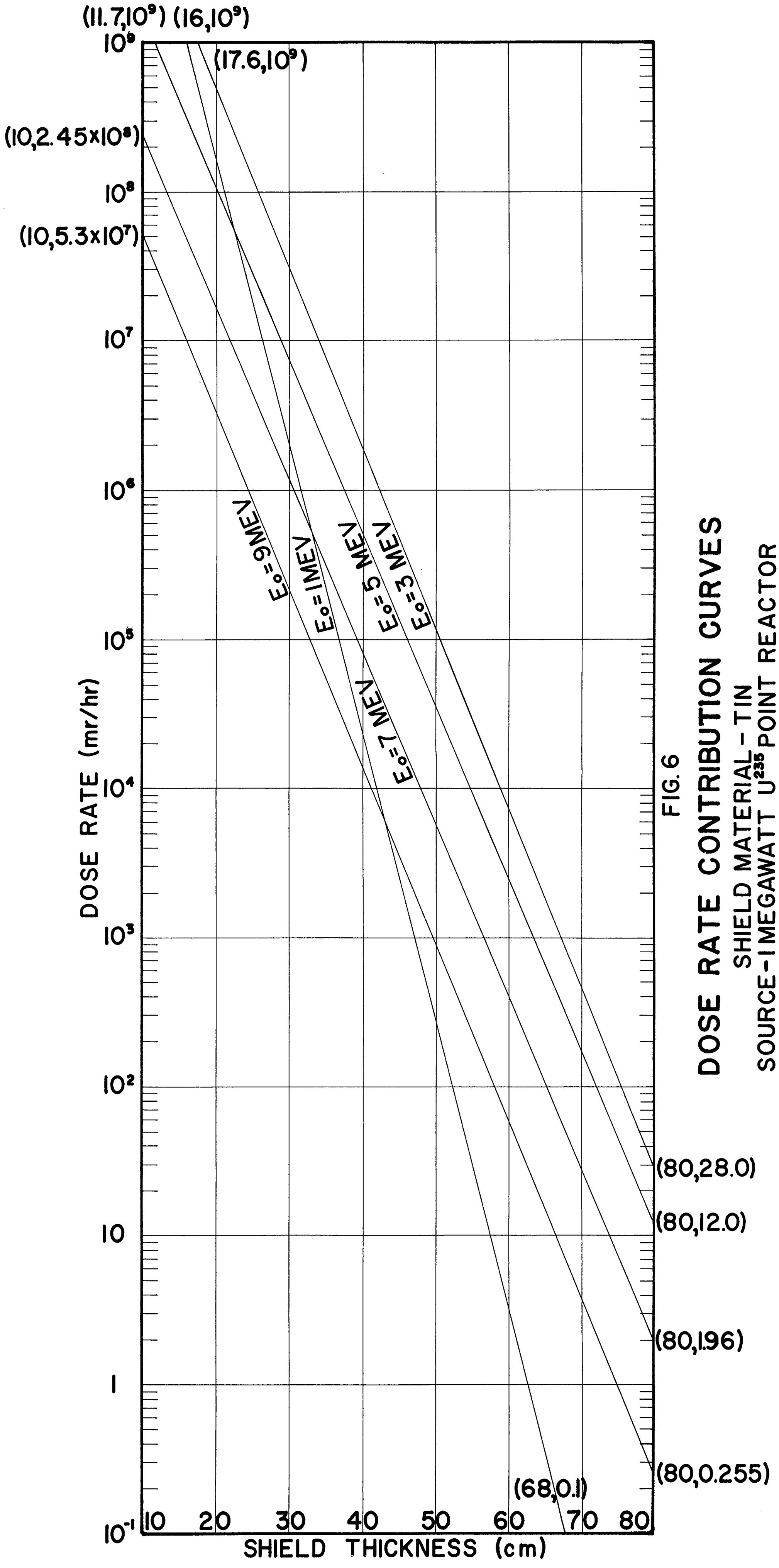

FIG. 6

DOSE RATE CONTRIBUTION CURVES

SHIELD MATERIAL - TIN

SOURCE - 1 MEGAWATT U^{235} POINT REACTOR

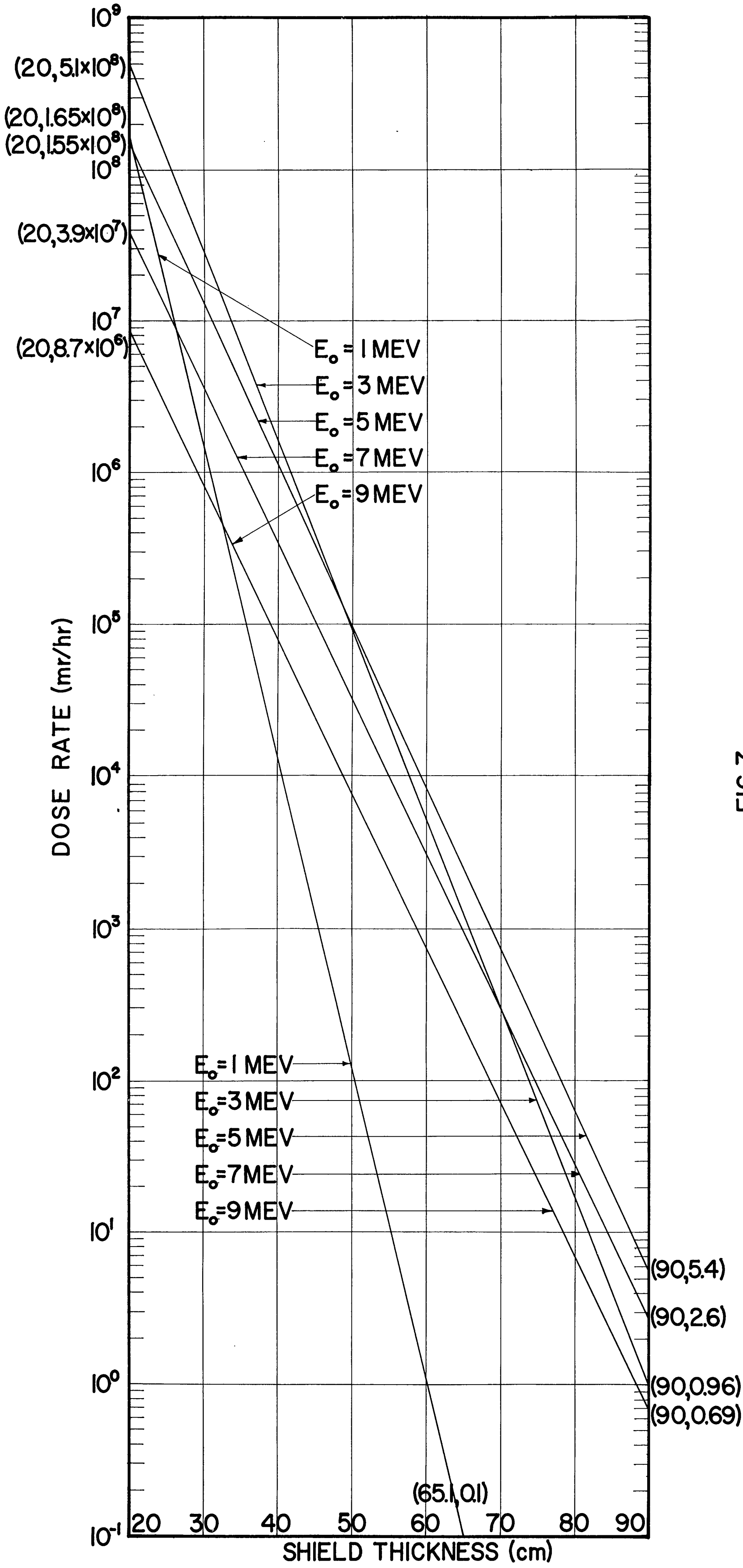

FIG. 7

DOSE RATE CONTRIBUTION CURVES

SHIELD MATERIAL - IRON

SOURCE - 1 MEGAWATT U^{235} POINT REACTOR

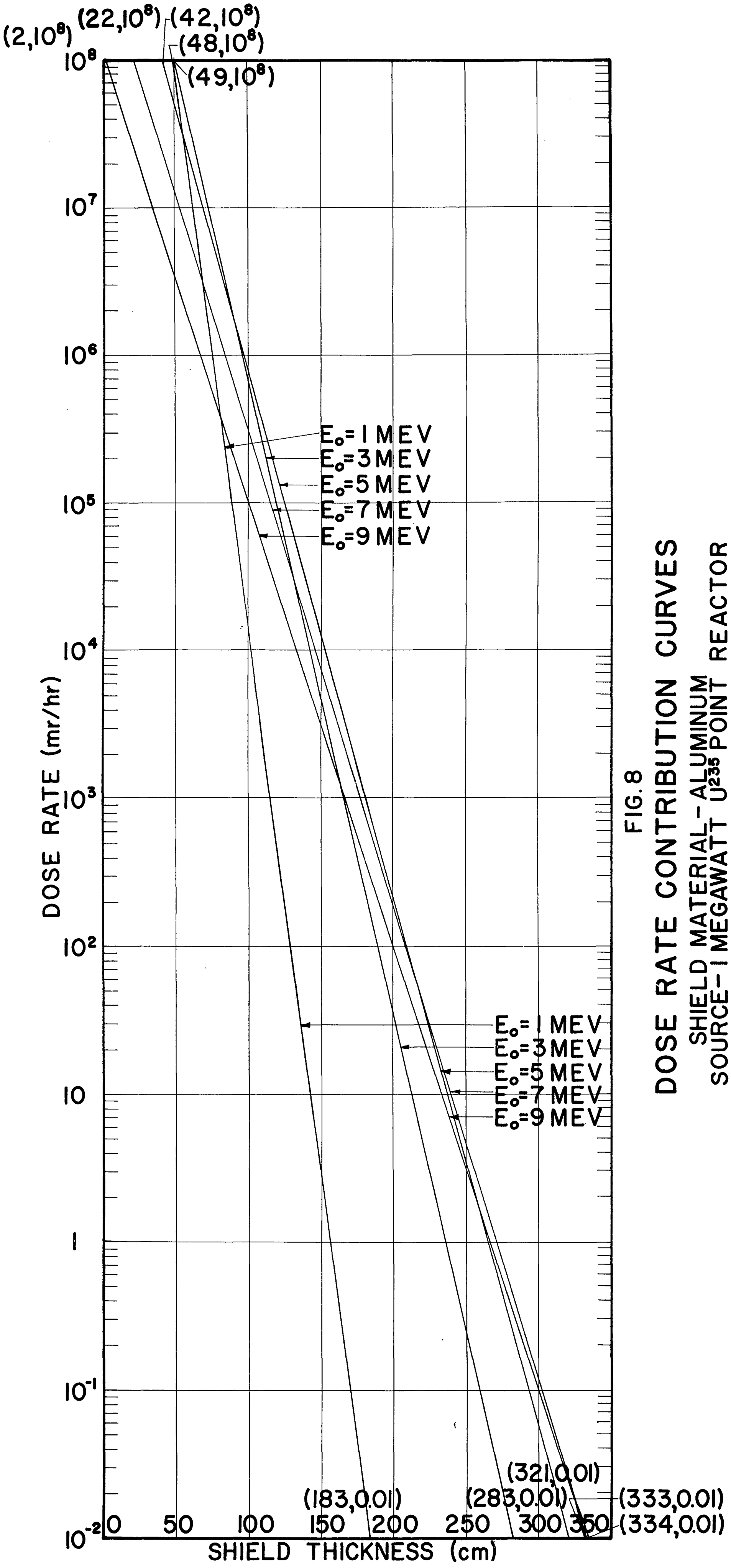

FIG. 8

DOSE RATE CONTRIBUTION CURVES

SHIELD MATERIAL - ALUMINUM

SOURCE - 1 MEGAWATT U^{235} POINT REACTOR

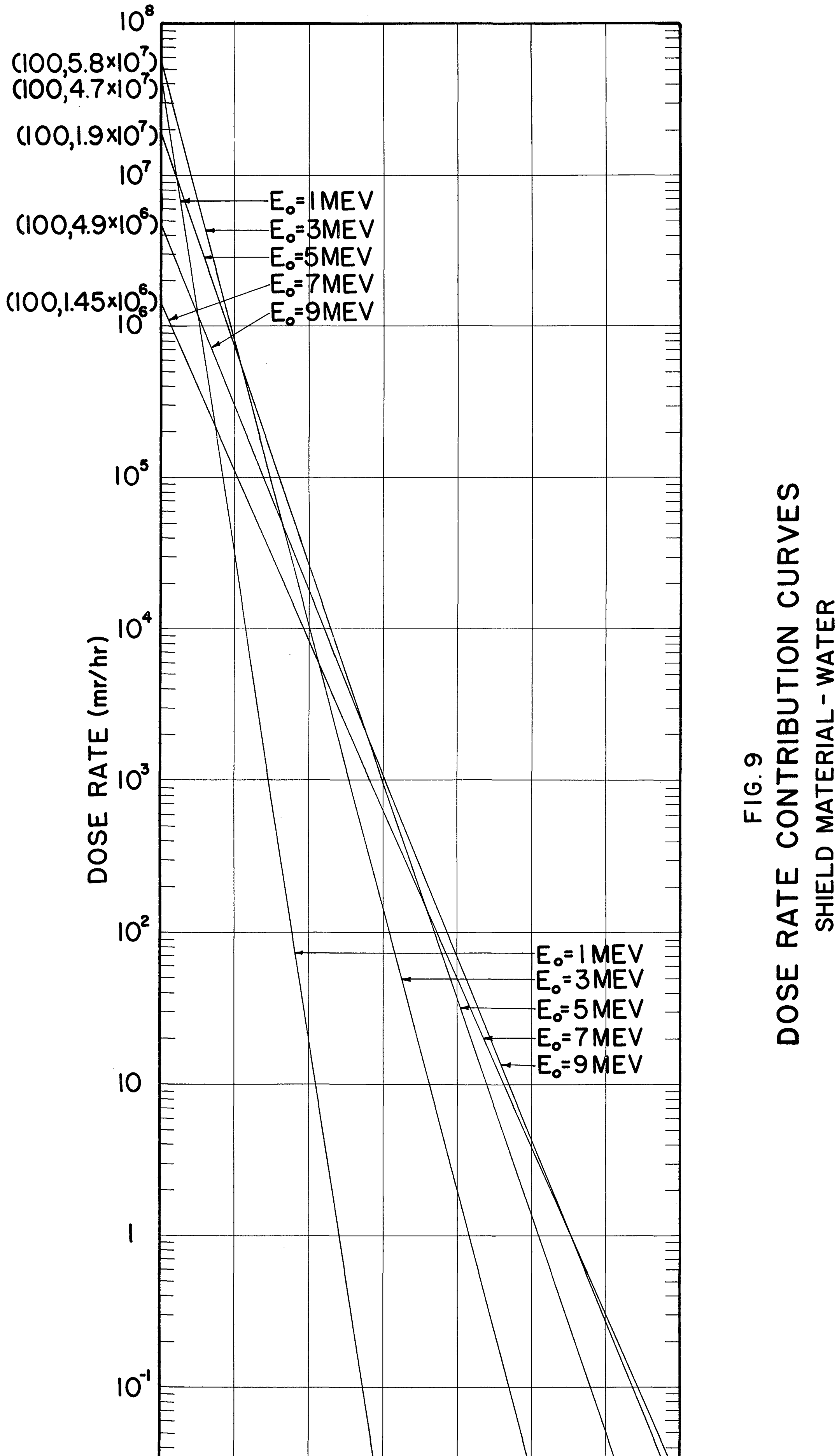

FIG. 9
DOSE RATE CONTRIBUTION CURVES
SHIELD MATERIAL - WATER
SOURCE-1 MEGAWATT U^{235} POINT REACTOR

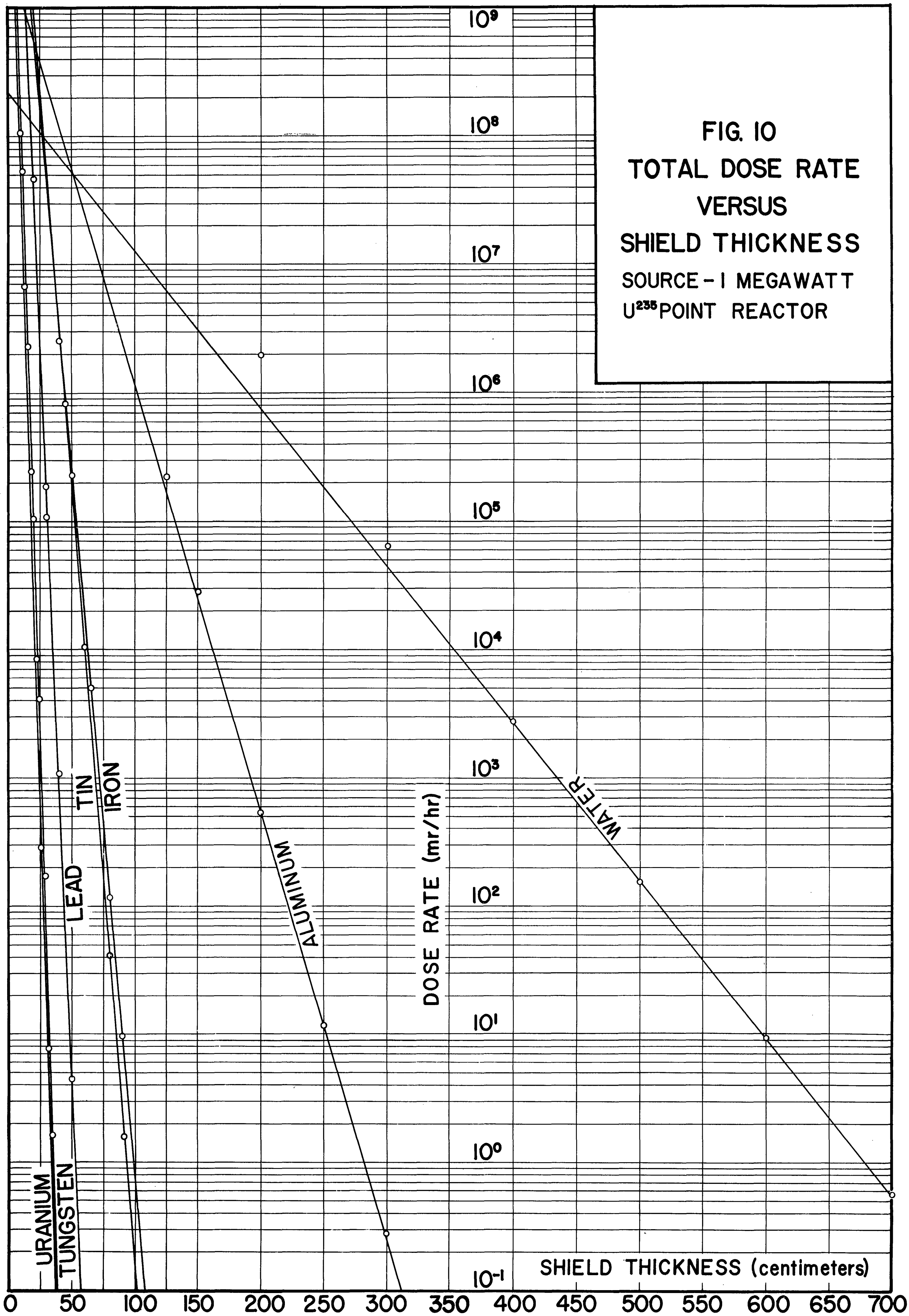
FIG. 10
TOTAL DOSE RATE
VERSUS
SHIELD THICKNESS
SOURCE - 1 MEGAWATT
U^{235} POINT REACTOR
DOSE RATE (mr/hr)
SHIELD THICKNESS (centimeters)
URANIUM
TUNGSTEN
LEAD
TIN
IRON
ALUMINUM
WATER
10^9
10^8
10^7
10^6
10^5
10^4
10^3
10^2
10^1
10^0
10^{-1}
0 50 100 150 200 250 300 350 400 450 500 550 600 650 700

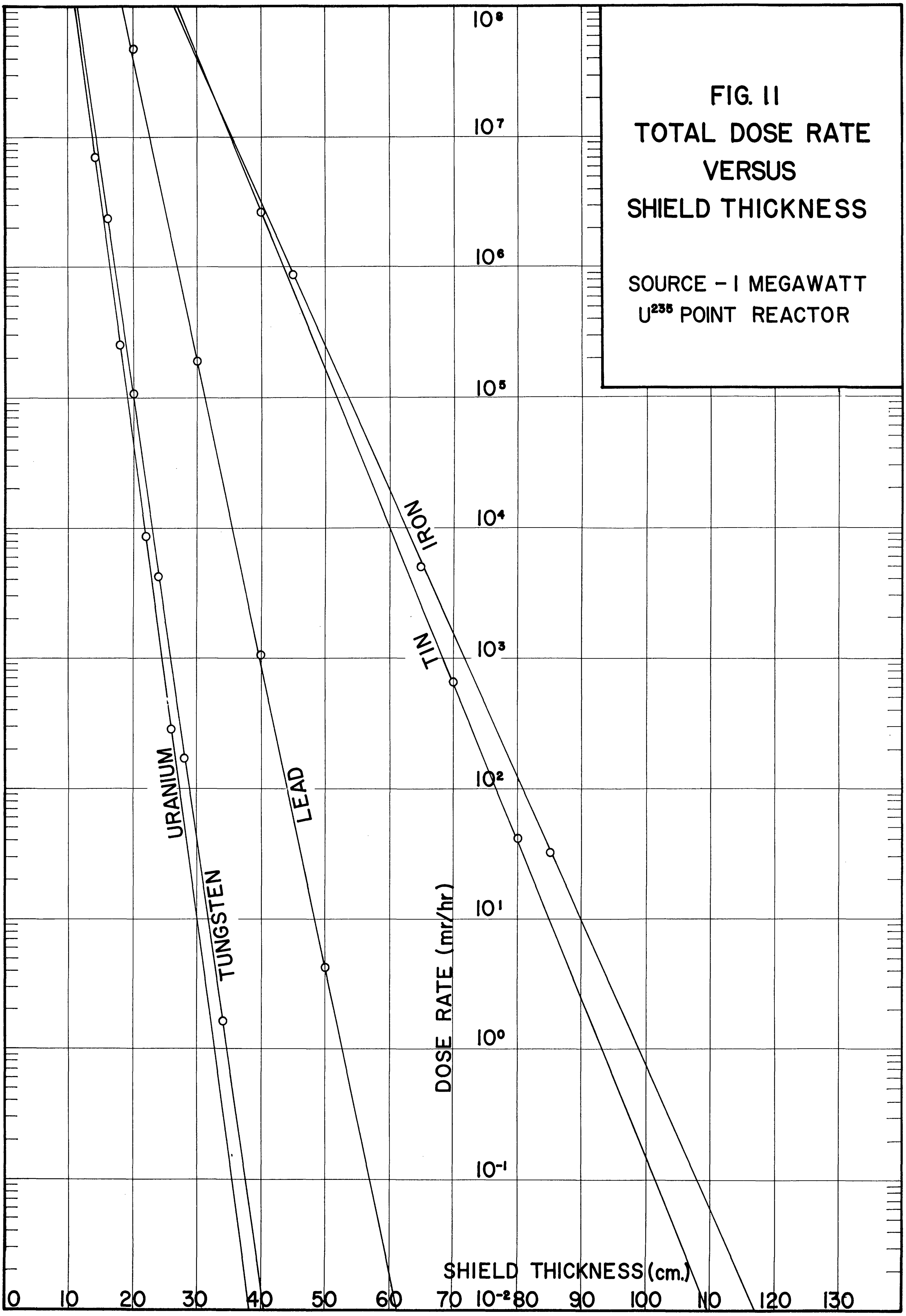
FIG. 11
TOTAL DOSE RATE
VERSUS
SHIELD THICKNESS
SOURCE - 1 MEGAWATT
U235 POINT REACTOR
IRON
TIN
LEAD
URANIUM
TUNGSTEN
DOSE RATE (mr/hr)
SHIELD THICKNESS (cm.)
10⁸
10⁷
10⁶
10⁵
10⁴
10³
10²
10¹
10⁰
10⁻¹
10⁻²
0
10
20
30
40
50
60
70
80
90
100
110
120
130

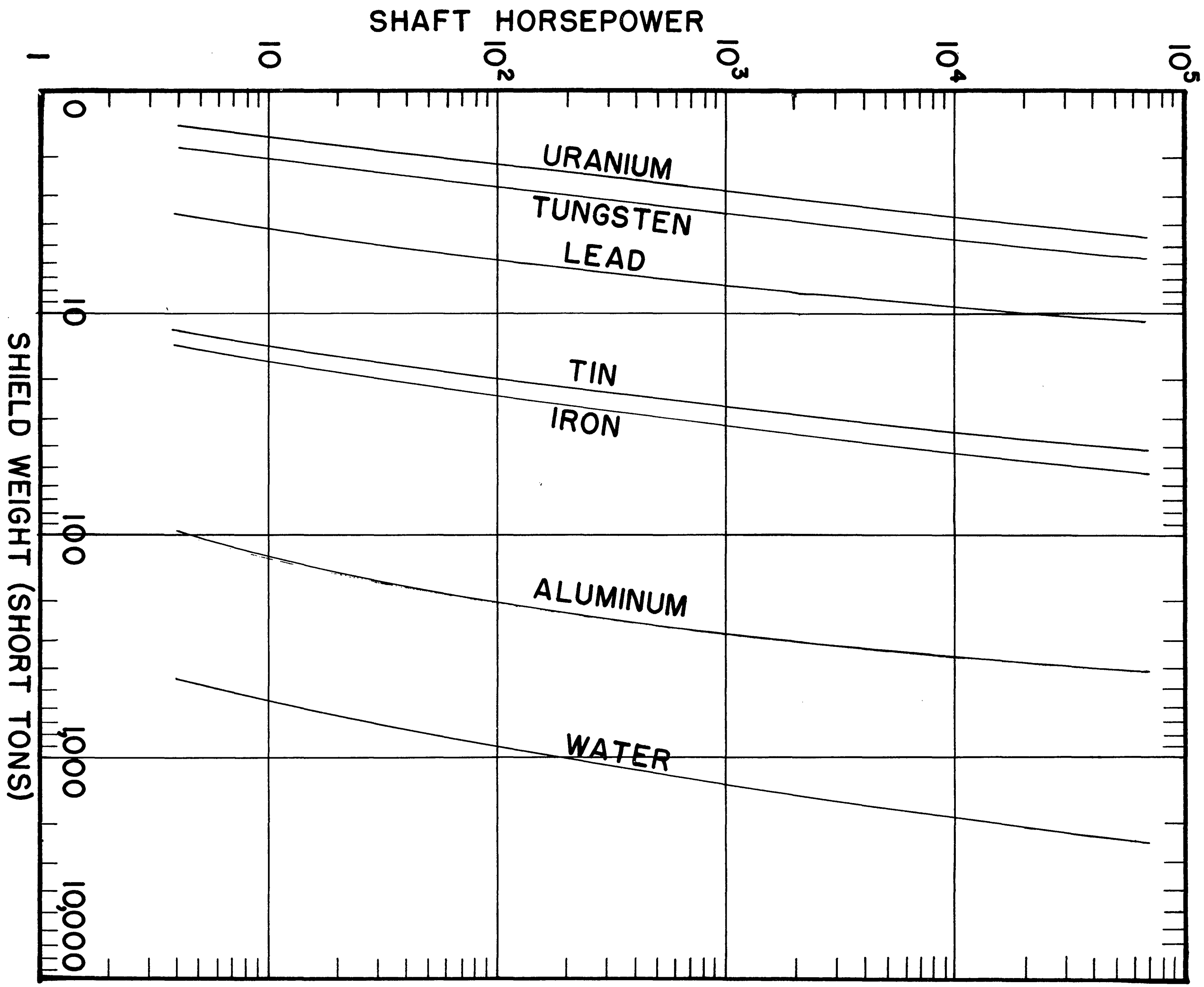

FIG. 12

SHAFT HORSEPOWER vs. SHIELD WEIGHT

(TO REDUCE DOSE RATE OF U^{235} POINT REACTOR TO 6 MR/HR)

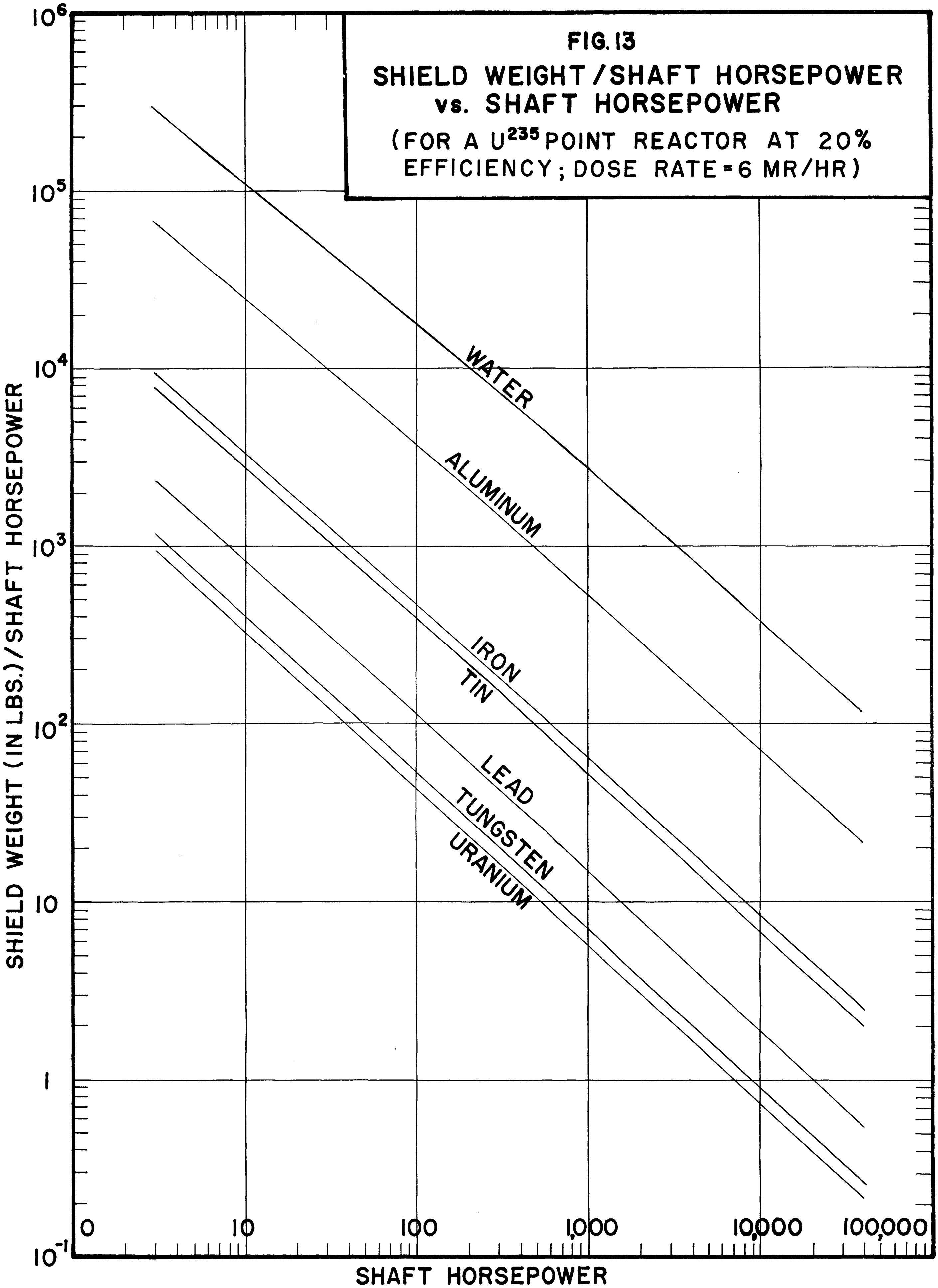
FIG. 13
SHIELD WEIGHT / SHAFT HORSEPOWER vs. SHAFT HORSEPOWER
(FOR A U235 POINT REACTOR AT 20% EFFICIENCY; DOSE RATE = 6 MR/HR)
WATER
ALUMINUM
IRON
TIN
LEAD
TUNGSTEN
URANIUM
SHIELD WEIGHT (IN LBS.) / SHAFT HORSEPOWER
SHAFT HORSEPOWER
0
10
100
1,000
10,000
100,000
1
10
10^2
10^3
10^4
10^5
10^6
10^{-1}

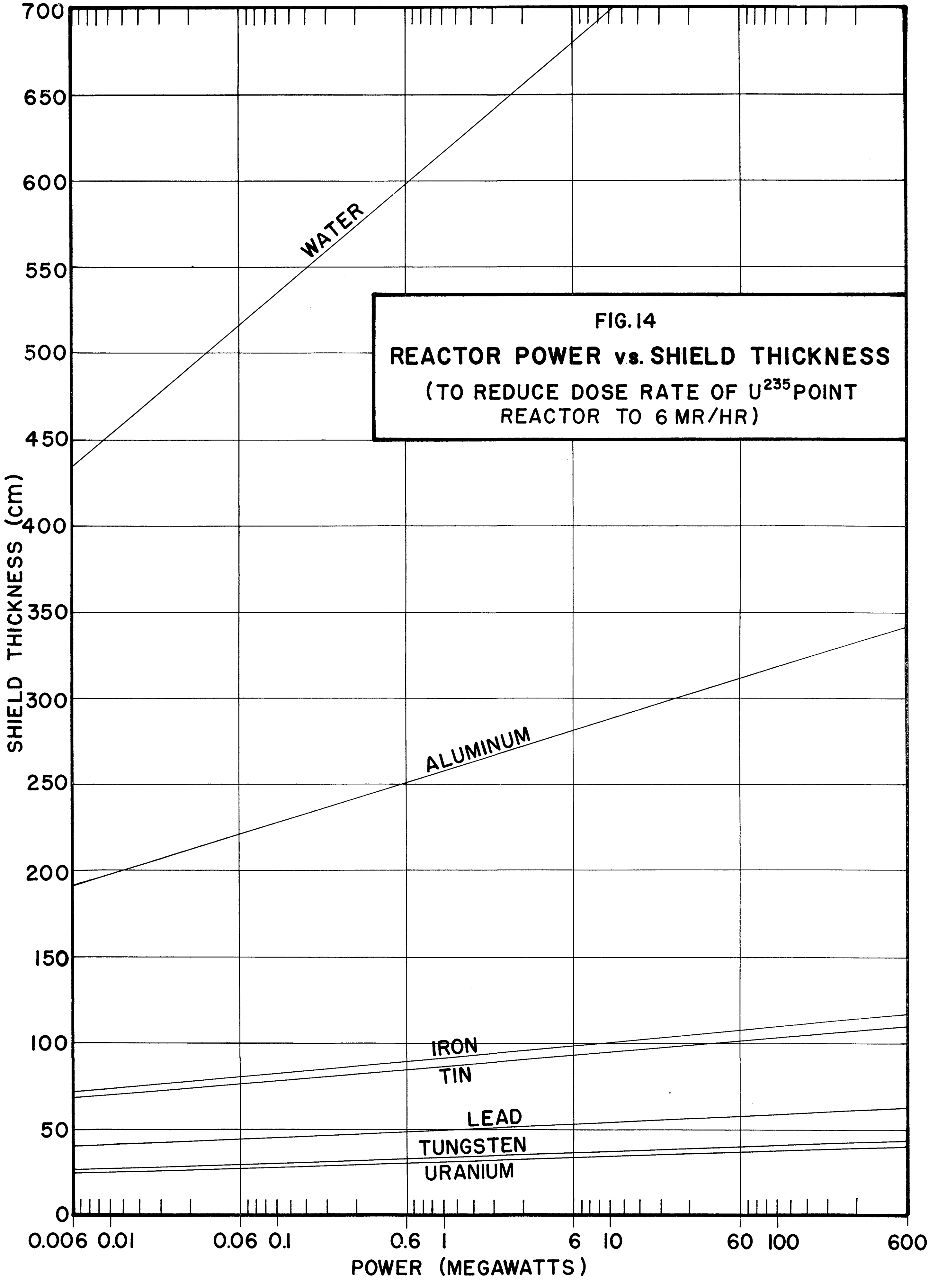
FIG. 14
REACTOR POWER vs. SHIELD THICKNESS
(TO REDUCE DOSE RATE OF U235 POINT REACTOR TO 6 MR/HR)
WATER
ALUMINUM
IRON
TIN
LEAD
TUNGSTEN
URANIUM
SHIELD THICKNESS (cm)
700
650
600
550
500
450
400
350
300
250
200
150
100
50
0
0.006
0.01
0.06
0.1
0.6
1
6
10
60
100
600
POWER (MEGAWATTS)

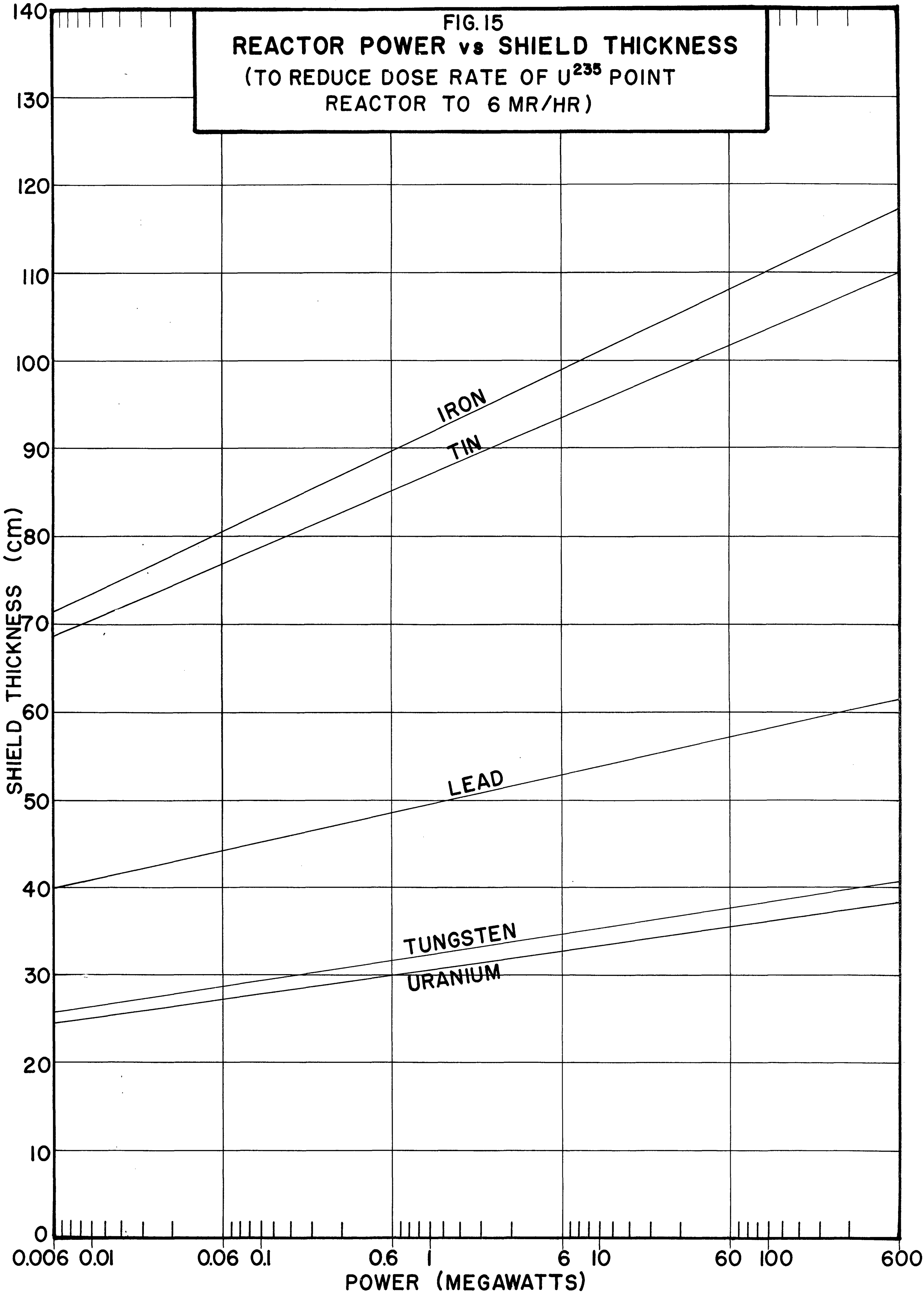
FIG. 15
REACTOR POWER vs SHIELD THICKNESS
(TO REDUCE DOSE RATE OF U235 POINT
REACTOR TO 6 MR/HR)
IRON
TIN
LEAD
TUNGSTEN
URANIUM
SHIELD THICKNESS (cm)
0
10
20
30
40
50
60
70
80
90
100
110
120
130
140
0.006
0.01
0.06
0.1
0.6
1
6
10
60
100
600
POWER (MEGAWATTS)

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