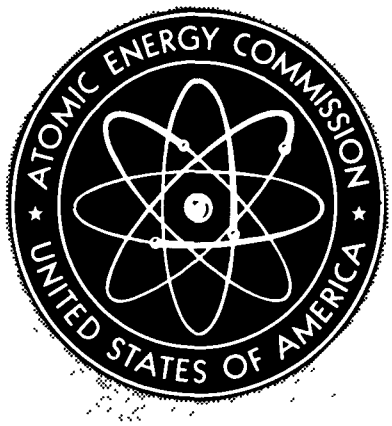

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HANDBOOK OF SHIELDING REQUIREMENTS AND RADIATION CHARACTERISTICS OF ISOTOPIC POWER SOURCES FOR TERRESTRIAL, MARINE, AND SPACE APPLICATIONS

By
E. D Arnold

April 1964

Oak Ridge National Laboratory
Oak Ridge, Tennessee

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OAK RIDGE NATIONAL LABORATORY
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This study was originally suggested by Mr. W. K. Eister of the Division of Isotopes Development, United States Atomic Energy Commission.

ABSTRACT

Isotopic power or radiation sources for use in terrestrial, marine, and space applications require shielding or a degree of isolation to prevent excessive radiation doses to personnel handling the source, to prevent radiation damage to instrument systems associated with the mission or application, and to prevent interference with experimental measurements that use radiation detection instruments.

This report is in essence a handbook of radiation properties and shielding requirements for isotopic power or radiation sources and is intended primarily for use in preparation of preliminary design estimates by design engineers in the field of isotopic source development and application. The calculated radiation intensities are probably slightly pessimistic in that they are overestimated no more than 50%. It therefore may be necessary to optimize the shield for an actual source from experimental data on the source itself. Actual measurements, proving the integrity of a shield, are necessary for licensing by the federal government.

Calculations have been made of the radiation intensities from shielded and unshielded sources fabricated from seventeen isotopes that show promise for use in isotopic power or radiation applications. Source sizes in the range of 100 to 20,000 thermal watts were evaluated. All shielded sources were assumed to be attenuated by iron, lead, and uranium; and in those cases where the source also emitted neutrons, neutron and gamma attenuation through water was determined.

The isotopes studied and their physical form are as follows: Co^{60} (metal), Kr^{85} (liquefied gas), Sr^{90} (oxide and titanate), Zr-Nb^{95} (oxide), Ru^{106} (metal), Cs^{137} (glass), Ce^{144} (oxide), Pm^{147} (oxide), Tm^{170} and Tm^{171} (both as oxides), Tl^{204} (metal), Po^{210} (metal matrix with void space for gas collection), U^{232} (oxide), Th^{228} (oxide matrix with void space for gas collection), Pu^{238} (oxide), Cm^{242} (oxide matrix with void space for gas collection), and Cm^{244} (oxide).

For the readers' convenience, several examples of how the graphical results may be used to calculate separation distance, shield thickness, and shield weight are included.

INTRODUCTION

Isotopes that are used for power production or radiation applications emit penetrating radiations to some degree. All radioisotopes that have been studied emit gamma, bremsstrahlung, or neutron radiation. Safe handling procedures, from both the radiation and contamination criteria, must be used during the shipment of radioisotopes, during the placing in operation (i.e., loading the power generator, radiation source unit, or thermal utilization system), and during operation of the systems. Hence, protective measures will be required on space missions in which astronauts or long-life electronic equipment are exposed to the radiation from isotopic power generators. Although most radioisotopic sources are intended primarily for power applications, several isotopes, including Cs^{137}, Zr-Nb^{95}, Co^{60}, Kr^{85}, and Ru^{106} also show promise as sources of high-energy gamma radiation in those applications where this radiation is most useful: sterilization, initiation of chemical reactions, etc.

This report is intended primarily as a handbook to be used by scientists and engineers evaluating types of radiation, hazards, shielding requirements, or design of isotopic power sources for terrestrial, marine, or space application. Since there cannot be one set of conditions applicable to all possible uses of a given source, the data covers a very wide set of conditions. It should be possible to obtain shield material or separation-distance requirements for any set of established conditions using the isotopes covered. For the reader's convenience, we include sample problems which illustrate how shield thickness, separation distance or shield weight can be determined.

Evaluations of the Use of the Seventeen Isotopes Analyzed in This Report

We present here the results of radiation-intensity calculations for shielded and unshielded isotopic power sources fabricated from 17 different isotopes. The results are presented in graphical form by plotting gamma radiation intensity (millirads/hour at 1 m) as a function of thickness of iron, lead, uranium, and water (only in those cases where the source also emits neutrons) shield materials or neutron radiation intensity (millireps

per hour at 1 m) as a function of water thickness for shielded neutron emitting sources, or by plotting gamma radiation intensity as a function of distance from the center of unshielded sources. Radiation intensities for sources of all 17 materials in the range of 100 to 20,000 thermal watts are presented.

In a previous study,[1] one which was a part of this overall study of shielding requirements and which determined radiation and shielding requirements for the more promising isotopes, Sr^{90}, Cs^{137}, Ce^{144}, Pm^{147}, Po^{210}, Pu^{238}, Cm^{242} and Cm^{244}, it was established that shield weights ranging from approximately 20 lb for Pu^{238} to over 500 lb for Cs^{137} would be required for 10,000-thermal-watt sources when used in an instrumented space mission. That study also indicated that weight penalties can be avoided if separation distance, rather than shielding, is relied on for protection. Separation distances ranging from 2 ft for Pu^{238} to 40 ft for Cs^{137} give protection equivalent to the shield weights quoted.

The fission products, Sr^{90}, Cs^{137}, Ce^{144} and Pm^{147}, have special appeal because they are produced in large quantities from nuclear reactor operation.

Among these isotopes producible by irradiation of special target materials (Co^{60}, Tl^{204}, U^{232}, Pu^{238}, Cm^{242}, and Cm^{244}) Co^{60}, though easy to produce, requires a special design for the heat-source generator because much of its emitted energy is penetrating gamma radiation, and therefore Co^{60} may be limited in its prospects for use as a power source but would therefore have special interest as a source of radiation.

Plutonium-238 is most generally favored as a heat source because of its long half-life and the fact that it can be used with almost no special shielding. However, its biological hazard encourages the search for a competitive material.

It is expected that whenever plutonium assumes a significant place as a recycled fuel in thermal reactors for power production, Cm^{244} can then become available at a reasonable cost.

Promethium-147, although its half-life is shorter than the other very promising isotopes, can be considered realistically for some uses as a substitute for Pu^{238}. Energy, cost, low shielding requirements, and minimum biological hazard are all factors favorable to Pm^{147} when nearly pure.[2,3]

The strong points for U^{232} (and its daughter Th^{228}) are high power densities and unusually long life at nearly constant heat output.

There are other characteristics of radioisotopes not essential to their use or production, but which may be factors in the choice of an isotope for a particular use. For example, although all radioisotopes are hazardous, they differ widely in their biological effects. Strontium and plutonium are especially hazardous because of their tendency to lodge in bone tissue. Promethium and cesium, on the other hand, do not exhibit the same biological retention. The type of radiation emitted by the isotope is another factor that may determine the preferred manner for use of an isotope. Those which are almost pure alpha emitters and also have a low neutron production rate, such as Pu^{238} and Po^{210}, or low energy, pure beta emitters, such as Pm^{147}, require the least shielding - a significant advantage where direct-contact fabrication and minimum shield weight are of interest.

Reliability of the Data

The results as plotted should be within ±50 % of the expected radiation value, based on the assumptions used and upon the accuracy of the input source strengths and the permanent data tables of the computer code. It is probably most likely that the dose rates are overestimated rather than underestimated. An error of ±50 % in radiation level would correspond to an error of ±0.175 "tenth thickness" (thickness of shielding necessary to attenuate radiation dose rate by a factor of 10).

The results are also only as accurate as the knowledge of the neutron, gamma, beta, and/or bremsstrahlung spectra. This is especially true for those sources whose spectra lie entirely in the lower energy (less than 0.5 Mev) range. Conversely, the calculated radiation intensities from those sources that have high-energy gamma spectra of high or well-known yield are in all probability reliable and accurate. This is especially true for Co^{60}, Zr-Nb^{95}, Cs^{137}, Ce^{144}, Po^{210}, Th^{228}, and U^{232}. Those sources that have only low-energy bremsstrahlung or gamma radiation are only as accurate as the estimates of the spectra and its self-absorption within the source itself. For the bremsstrahlung sources, the calculated dose rates and/or shield thicknesses are more accurate for the high-energy

spectra (within the accuracy of the actual spectra calculation) than for the low-energy spectra.

It should be emphasized that in many cases the total energy production rate of a given isotope is much greater than the recoverable heat rate in the source itself. This is especially true for those isotopes that produce considerable gamma radiation. For these isotopes, an external heat recovery unit (partial shield) must be provided to convert this gamma energy to heat. This may even be so for the high-energy beta emitters that produce bremsstrahlung energy equivalent to a few percent of the average beta energy. Here a decision must be made concerning the economic justification for provision of an external heat recovery unit for a few percent of the total energy.

It therefore should be stated that where minimizing shield weight is of utmost importance a final experimental determination of the shield thickness, and even a source mockup (especially for the lower energy gamma or bremsstrahlung sources), may be necessary before an accurate final source and shield design can be decided upon.

It is always necessary to provide experimental verification of the shielding performance and mechanical integrity of a shield for an isotopic source before a license can be obtained.

The results were plotted in terms of dose rate vs shield thickness in order to make this report as general as possible for all workers in the field of terrestrial, marine or space source design. It was decided not to plot shield weights as a function of dose rate since a particular type of shield (cylinder, contoured cylinder, shadow, spherical, etc.) and the radiation limitations based on application would have to be decided on before the actual weights could be calculated. It is very easy to calculate the shield weight once the thickness is known and the type of shield has been decided upon. These techniques are demonstrated in this report, beginning on page 6.

Pattern of the Presentation

This handbook discusses the subjects of allowable radiation levels and biological hazards, the calculation of the radiation levels, radiation production during the decay processes, and the radiation and shielding

requirements for 17 radioisotopes. In addition, Appendix A lists the computer code (for the CDC-1604 computer) used to calculate bremsstrahlung sources and the tabulation of the actual bremsstrahlung source strengths.

The section on allowable radiation levels and biological hazards defines radiation exposure and the limits of radiation exposure for various applications. This section also defines and tabulates the relative biological hazards for the various isotopes.

The section on calculation of the radiation levels describes the spatial relationships between source, shield and dose point assumed for the sources and describes the approach used for calculating the radiation intensities.

The section on radiation production during the decay process describes the various radiations produced during decay and the methods used to calculate the production rates for these radiations.

The section on radiation and shielding requirements contains the bulk of the calculated results of this study in which the physical characteristics of the sources are tabulated and the radiation intensities are plotted as a function of distance and/or shield thickness.

In order to make the handbook more meaningful to users who are new to the field, seven sample problems are given and solved, also showing where pertinent data can be found in the curves and tables presented here.

EXAMPLES OF USE OF DATA

The following seven example problems show, in increasing complexity, the types of problems that may be met when calculating shielding requirements from the tabular and graphical data of this report.

Example 1: Shield-thickness determination. This example is straightforward and is based on determination of shield thickness for a given attenuation. Sky shine or gamma scattering around a slab or shadow shield which may be approximately 10% of the direct dose rate is not determined.

Problem: A 10,000-w Sr^{90} oxide source is to be used as the power source for an instrumented satellite. The mission length is 5.7 yr (50,000 hr), and the allowable radiation exposure to the instrument package is 10^7 rads. Calculate the thickness of a uranium shield necessary to attenuate the dose rate at 1 m to the desired level.

Solution: The desired dose rate = 10^7 rads/50,000 hr = 200 rads/hr =

2×10^5 millirads/hr. Reading directly from Fig. 17 we find that 1.5 cm of uranium would be necessary.

Example 2: Separation-distance determination. This problem is also straightforward and is based on determination of separation distance for a given attenuation.

Problem: Using the same source as in example 1, calculate the distance required to achieve the same attenuation for the same source size:

Solution: Read directly (from Fig. 14) 860 cm as the distance from center of source for 2×10^5 millirads/hr dose rate.

Example 3: Separation-distance determination with shield thickness given. This problem is a combination of examples 1 and 2 and illustrates how separation distance may be determined, given the total attenuation and the thickness of shield.

Problem: Using the same source as in example 1, calculate the distance required to achieve the same attenuation for the same source if the source is also shielded by 1 cm of lead

Solution: The radiation dose rate through 1 cm of lead is 1.1×10^6 millirads/hr at 1 m (from Fig. 16) and the desired attenuation factor is then $2 \times 10^5 / 1.1 \times 10^6 = 0.183$. Using Fig. 14 and a 10,000-w source, we find for an unshielded source a dose rate of 1.45×10^7 millirads/hr at 1 m. An attenuation factor of 0.183 indicates that the effective unshielded dose rate is $0.183 \times 1.45 \times 10^7 = 2.65 \times 10^6$ millirads/hr. The desired separation distance would then be 235 cm, as read from Fig. 14.

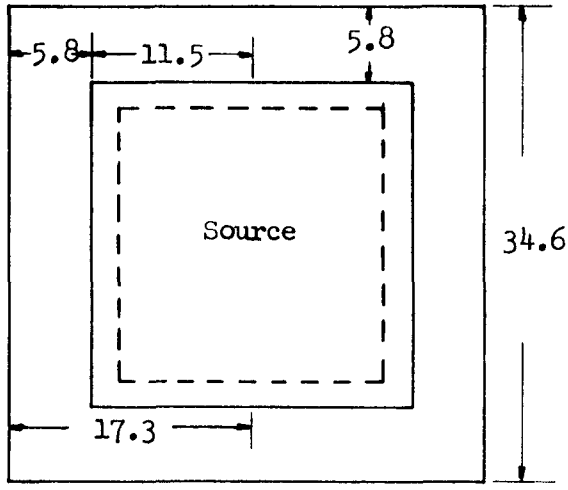
Example 4: Shield weight for a cylindrical shield. This example combines the determination of shield thickness, followed by the calculation of the shield weight for a source of given size.

Problem: Calculate the weight of a square cylindrical uranium shield necessary to attenuate a 10,000-w Sr^{90} oxide source to 1 rad/hr at 1 m.

Solution: From Fig. 17 we find that a thickness of 5.8 cm of uranium would be necessary. From Table 11 we find that the power density is 1.4 w/cc. The volume of a 10,000-w source is then equal to 10,000/1.4 = 7150 cc, and the radius of a cylinder whose height is equal to its diameter

would be $r = \sqrt[3]{V/2\pi} = \sqrt[3]{7150/2\pi} = 10.5$ cm.

Assume that the inside of the shield is 1 cm from the source as indicated in the sketch.



All dimensions in centimeters.

Then the volume of the shield would be:

$V = \pi HR^2 - \pi hr^2 = \pi(HR^2 - hr^2)$

where

r = radius of cavity = 11.5 cm,

h = height of cavity 23 cm,

R = radius of shield = 11.5 + 5.8 = 17.3 cm,

H = height of shield = 23 + 2(5.8) = 34.6 cm.

Then,

$V = \pi(34.6 \times 17.3^2 - 23 \times 11.5^2)$

$= \pi(34.6 \times 299.29 - 23 \times 132.25)$

$= \pi(10{,}355.4 - 3041.8) = \pi \times 7313.6 = 22{,}976$ cc,

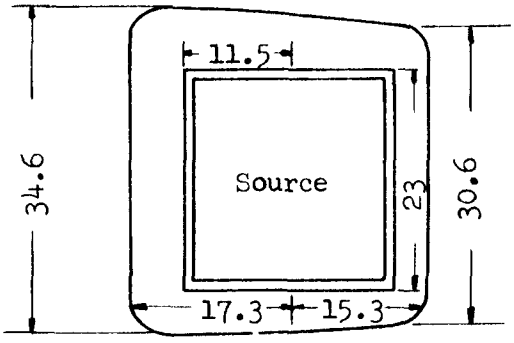
and

mass = 22,976 x 18.6 = 427,354 g = 427.4 kg.

Example 5: Shield weight for a contoured shield. This example is somewhat more complex than example 4 in that the shield is contoured to save weight. In order to save weight the attenuation in the direction away from the instruments is specified to be a tenth of that in the direction toward the instruments. The results for this example, compared with that from example 4, shows how much weight may be saved by this design.

Problem: Estimate the weight of a contoured cylindrical uranium shield necessary to attenuate a 10,000-w Sr^{90} oxide source to 1 rad/hr at 1 m over 1/3 of the total solid angle and to 10 rads/hr at 1 m over 2/3 of the total solid angle.

Solution: From Fig. 17 we find that a thickness of 5.8 cm of uranium would be necessary to attenuate to 1 rad/hr and that 3.8 cm of uranium would be necessary to attenuate to 10 rads/hr. From Table 11 we find that the power density is 1.4 w/cc. The volume and radius can then be calculated to be 7150 cc and 10.5 cm, respectively. Assume that the inside of the shield is 1 cm from the source, as indicated in the sketch.



All dimensions in centimeters.

Then the volume of the shield may be estimated by:

$V = 1/3\ V_L + 2/3\ V_S$,

where

V_L - volume of shield if entirely shielded by the thicker material,

V_S = volume of shield if entirely shielded by the thinner material.

Then,

$V = 1/3(\pi H_L R_L^2 - \pi hr^2) + 2/3(\pi H_S R_S^2 - \pi hr^2)$,

where

H_L = height of cylindrical shield if entirely shielded by thicker material = 34.6 cm,

R_L = outside radius of cylindrical shield if entirely shielded by thicker material = 17.3 cm,

H_S = height of cylindrical shield if entirely shielded by thinner material = 30.6 cm,

R_S = outside radius of shield if entirely shielded by thinner material = 15.3 cm,

h = height of cavity = 23 cm,

r = radius of cavity = 11.5 cm.

Then,

$$V = \pi(1/3H_L R_L^2 + 2/3H_S R_S^2 - hr^2)$$

$$= \pi(1/3 \times 34.6 \times 17.3^2 + 2/3 \times 30.6 \times 15.3^2 - 23 \times 11.5^2)$$

$$= \pi(1/3 \times 34.6 \times 299.29 + 2/3 \times 30.6 \times 234.09 - 23 \times 132.25)$$

$$= \pi(3451.8 + 4775.4 - 3041.8)$$

$$= \pi \times 5185.4 = 16{,}290 \text{ cc},$$

and

mass = 16,290 x 18.6 = 302,994 g = 303 kg.

This result is approximately 0.7 of that for the completely shielded source of example 4.

Example 6. Shield weight for a cylindrical shield. Sometimes it is forgotten that even though the penetration depth (shield thickness in units of g/cm^2) for low-density shields may be less than that for high-density shields, the weight of a cylindrical shield of the low-density material may still be greater than that of a high-density shield. This example is based on a total gamma dose of 10^7 rads during a 50,000-hr mission. (A gamma dose of 10^7 rads is the accepted design limit for transistorized instruments.)

Problem: Calculate the weight of a square cylindrical uranium shield necessary to attenuate a 10,000-w Sr^{90} oxide source to 200 rads/hr (10^7 rads during a 50,000-hr mission) at 1 m.

Solution: From Fig. 17 we find that a thickness of 1.5 cm of uranium would be necessary. From Table 11 we find that the power density is 1.4 w/cc. The volume of a 10,000-w source is then equal to 10,000/1.4 = 7150 cc, and the radius of a cylinder whose height is equal to its diameter would be

$$r = \sqrt[3]{V/2\pi} = \sqrt[3]{7150/2\pi} = 10.5 \text{ cm}.$$

Assume that the inside of the shield is 1 cm from the source. Then the volume of the shield would be:

$$V = \pi HR^2 - \pi hr^2 = \pi(HR^2 - hr^2),$$

where

r = radius of cavity = 11.5 cm,

h = height of cavity = 23 cm,

R = radius of shield = 11.5 + 1.5 = 13 cm,

H = height of shield = 23 + 2(1.5) = 26 cm.

Then,

$V = \pi(26 \times 13^2 - 23 \times 11.5^2) = \pi(26 \times 169 - 23 \times 132.25)$

$= \pi(4394.0 - 3041.8) = \pi \times 1352.2 = 4{,}248$ cc,

and

mass = 4,248 x 18.6 = 79,000 g = 79 kg.

Example 7. Shield weight for a neutron shield. This example, as well as comparing shield weight with that calculated in example 6, also illustrates the determination of the thickness required for a neutron shield. This example is based on a total neutron dose of 10^4 reps during a 50,000-hr mission. (A neutron dose of 10^4 reps, or rads, is the accepted design limit for transistorized instruments and is equivalent in radiation damage to 10^7 rads of gamma radiation.)

Problem: Estimate the weight of a square cylindrical lithium hydride shield necessary to attenuate a 10,000-w Cm^{244} source to 0.2 rep/hr (10^4 rep during a 50,000-hr mission) neutron dose rate at 1 m.

Solution: From Fig. 112 we find that a thickness of 23.5 g/cm^2 of water would be necessary. Using Fig. 2 we find that 18.0 g/cm^2 of lithium hydride is equivalent to 23.5 g/cm^2 of water as shielding for neutrons. The density of lithium hydride is 0.78 g/cc, thus the shield thickness would be 23.1 cm. From Table 23 we find that the power density of Cm^{244} sources is 26.4 w/cc. Thus the volume of a 10,000-w source is then equal to 10,000/26.4 = 379 cc, and the radius of a cylinder whose height is equal to its diameter would be $r = \sqrt[3]{V/2\pi} = \sqrt[3]{379/2\pi} = 3.9$ cm.

Assume that the inside of the shield is 1 cm from the source. Then the volume of the shield would be:

$V = \pi HR^2 - \pi hr^2 = \pi(HR^2 - hr^2)$,

where

r = radius of cavity = 4.9 cm,

h = height of cavity = 9.8 cm,

R = radius of shield = 4.9 + 23.1 = 28.0 cm,

H = height of shield = 9.8 + 2(23.1) = 56.0 cm.

Then,

$V = \pi(56 \times 28^2 - 9.8 \times 4.9^2)$

$= \pi(56 \times 784 - 9.8 \times 24.01)$

$= \pi(43{,}904 - 235.3) = \pi(43{,}668.7) = 137{,}190$ cc,

and

mass = 0.78 x 137,190 = 107,000 g = 107.0 kg.

The table below shows the comparison of shield thicknesses, penetration depths, and shield weights for the uranium-shielded Sr^{90} source (of example 6) and for the lithium hydride-shielded Cm^{244} source calculated in this example. Both sources generate 10,000 thermal watts.

	Uranium-Shielded Sr^{90} Source	LiH-Shielded Cm^{244} Source
Thickness, cm	1.5	23.1
Penetration depth, g/cm^2	27.9	18.0
Weight of cylindrical shield, kg	79	107

Caution: One must calculate the volume and weight when comparing cylindrical shields. A comparison of shields based on penetration depths (g/cm^2) is reliable for slab shields but not for cylindrical shields.

ALLOWABLE RADIATION LEVELS AND DISCUSSION OF BIOLOGICAL HAZARDS

Before allowable radiation levels are discussed, a few definitions of what are radiation levels and what they mean should be set. In this report gamma dose rates have been calculated in rads (or millirads) per hour, with human tissue assumed to be the absorption media. Neutron dose rates were plotted in terms of millireps per hour since the data were expressed this way in most of the literature. In addition, when talking about manned satellites and allowable radiation exposure to man, we must talk in terms of doses in rems.

Definitions of Dosage Terms

The following units of radiation dosage are used to measure the effect of radiation:

(1) The roentgen (r): The roentgen is the basic unit of radiation-dose measurement. It is a physical unit and is primarily used to express dosages of gamma or x radiation. One roentgen is defined as the quantity of gamma radiation which will produce, by ionization, one electrostatic unit of electricity of either sign (i.e., 2×10^9 ion pairs) in 1 cm^3 of dry air, measured at standard conditions of temperature and pressure.

(2) The roentgen equivalent physical (rep): The rep measures the energy absorbed by the human body rather than the energy absorbed by air. It is therefore applicable to any form of radiation, unlike the roentgen which applies only to gammas and x rays. One rep is defined as that radiation dosage that produces energy absorption in human tissue equal to 96.5 ergs per gram of soft tissue. This is roughly equal to the energy absorption in tissue caused by 1 r of gamma radiation.

(3) The rad is a unit of dose useful for correlating radiation damage in various materials. A rad is defined as an absorbed dose of 100 ergs/g. One rad is received by a material exposed to about 1 r of gamma radiation. Different materials exposed to the same radiation flux for the same period of time will absorb different amounts of energy per gram and thus will have a different exposure expressed in rads but the same exposure expressed in reps. In our usage of millireps per hour for neutron dose rates an error of 3.5% is made by assuming that millireps and millirads are equivalent. Gamma dose rates were calculated in terms of millirads per hour for human tissue.

(4) The roentgen equivalent man (rem): Specifications for maximum permissible exposure to radiation are usually expressed in rems. The rem is not a directly measurable physical quantity as are the roentgen and the rep, but, rather, it is an index of the damage effects produced in the human body by the various types of radiation. The varying biological effects of the various types of radiation are frequently correlated by a coefficient called the relative biological effectiveness (RBE). The RBE is defined as the ratio of the body damage from a given type of radiation to that caused by the same dose of gammas. The rem is defined as the doage in rads multiplied by the RBE for the type of radiation involved.

Roentgens, reps, rads and rems refer to doses; the dose rates are given, respectively, as roentgens per hour, reps per hour, rads per hour, or rems per hour.

Relative Biological Effectiveness of Radiation

Type of Radiation	RBE
x and gamma	1
Beta	1
Fast-neutron	10
Alpha particle (internal)	10

Allowable Radiation Levels

The shielding or separation-distance requirements for isotopic power sources are determined by maximum allowable radiation levels derived from one or more of the following three considerations:

Man

The maximum permissible radiation dose rates established by the National Committee on Radiation Protection for occupational personnel exposure are assumed to apply to those working around isotopic sources and preparing them for a launch. The recommended maximum allowable doses, integrated over a 13-week period, without reference to instantaneous dose rate, are 3 rads of gamma and 0.3 rad of fast neutrons for whole body exposure, and 25 rads of gamma and 2.5 rads of neutrons for exposure of hands and arms.

Transistorized Instruments

Data on the effects of radiation on instrument systems are available in reports of the REIC series from the Radiation Effects Information Center at Battelle Memorial Institute, Columbus, Ohio, and the proceedings of a recent symposium on protection against radiation hazards in space.[4] Allowable radiation doses vary widely with the type of radiation and type of instrument system. Maximum allowable doses $\lceil 10^7$ rads of gamma, 10^4 rads of fast neutrons (approximately 10^{12} neutrons/cm^2) and 10^4 rads of solar protons (approximately 10^{12} protons /cm^2)$\rfloor$ were chosen on the basis that they probably would not cause significant radiation damage in typical transistorized instrument systems. Some damage has been observed for doses 10 times greater, and there is significant damage at doses 100 times greater.

Radiation Detection Instruments

Radiation detection instruments (including scintillation counters, GM counters and ionization chambers) are often used to measure the radiation background in space, on or near the moon or planets, or physical properties such as density and composition. Thus, power sources emitting radiation of a type and energy similar to that being measured must be shielded or separated from the radiation detection instruments to prevent interference with the measurements. For some measurements it may be necessary to keep the radiation level due to the power source as low as 0.01 mrad/hr in the vicinity of the instrument.

In providing shielding or separation distance to satisfy the above maximum levels, one must remember to take into account the natural radiation levels found at various points in space (see Table 1). These are often high enough, especially for humans, to require shielding of the payload itself. The shield thickness or separation distance from the isotopic supply can then be correspondingly smaller.

Table 1. Radiation Levels in Space

Radiation Source	Radiation Levels (rads/hr) Through Shield Thicknesses of: 0 g/cm^2	10 g/cm^2
Van Allen belts:		
electrons	10^3-10^5	~0
bremsstrahlung	1-100	1-30
protons	10-100	1-10
Solar-flare protons (typical)	10-100	1-10
Galactic cosmic rays	~10^{-3}	~10^{-3}

Biological Hazards

This section is not intended as a complete discussion of the biological hazards of radiation or power sources. Since we are primarily interested in shielding, the relative hazards are based on working-level limits as used by operators of fuel-handling facilities. The Code of Federal Regulations, Title 10, Part 20[5] should be consulted for a more complete discussion of hazards of radiation sources.

Although all radioisotopes must be regarded as biologically hazardous materials, the effects of isotopes in biological systems may be a significant factor in the choice of what isotope to use for a power source. There are two comparisons that may be made in order to determine the relative biological hazard of several isotopes. These are: (1) a comparison of the maximum permissible concentrations[6] in air $(MPC)_a$ or in water $(MPC)_w$, and (2) a comparison of the hazard equivalent to some standard (1 g of Pu^{239} is used as the standard at ORNL[7]). The value of the HEP^X [Inhalation Hazard Amounts (curies) Equivalent to 1 g (6.1×10^{-2} curie) of Pu^{239}] of any given isotope, X, can be found from the following equation:

$$HEP^X = 2.16 \times 10^9 (MPC)^X_{a(40\ hr/wk)} [Max(g/curie,\ 0.1)]^X,$$

where

$(MPC)^X_{a(40\ hr/wk)}$ = maximum permissible concentration in air for isotope X for a 40 hour/week working time,

g/curies = grams of isotope X equivalent to one curie. (This term is used in order to allow for the relative activity dispersibility of the low specific power materials.)

Table 2 lists the HEP and $(MPC)_{a(40\ hr/wk)}$ values for the isotopes studied.

METHOD FOR CALCULATING RADIATION LEVELS

Calculation of the gamma and/or bremsstrahlung radiation level and shielding requirements were made by using the SDC code.[8] Sources were assumed to be right circular cylinders with height equal to diameter and clad with 0.2 cm (this thickness would correspond to 0.54 g/cm^2) of aluminum. The outside surface of the source was assumed to be 1 cm from the inside face of the shield for all shielded source cases. The dose point was assumed to be 100 cm from the center of the source in the case of shielded sources and was varied from 50 to 1000 cm from the source center in the case of the unshielded sources. Configurations for these two cases are shown in Fig. 1.

Self-attenuation within the sources is determined in the SDC code by calculating self-absorption factors or self-absorption exponents (self-absorption distances), the common practice in hand calculations as described in the shielding literature.[9,10,11,12,13]

Table 2. Biological Hazards of Source Isotopes

Isotope	HEP[a]	Minimum Value of $(MPC)_{a(40\ hr/wk)}$
Co^{60}	1.9	9×10^{-9}
Kr^{85}	2.2×10^{3}	10^{-5}
Sr^{90}	6.5×10^{-2}	3×10^{-10}
Zr^{95}	6.5	3×10^{-8}
Nb^{95}	22	10^{-7}
Ru^{106}	1.3	6×10^{-9}
Cs^{137}	2.2	10^{-8}
Ce^{144}	1.3	6×10^{-9}
Pm^{147}	13	6×10^{-8}
Tm^{170}	6.5	3×10^{-8}
Tm^{171}	22	10^{-7}
Tl^{204}	6.5	3×10^{-8}
Po^{210}	4.3×10^{-2}	2×10^{-10}
Th^{228}	1.3×10^{-3}	6×10^{-12}
U^{232}	6.5×10^{-3}	3×10^{-11}
Pu^{238}	4.3×10^{-4}	2×10^{-12}
Cm^{242}	2.2×10^{-2}	10^{-10}
Cm^{244}	1.9×10^{-3}	9×10^{-12}

[a]Inhalation hazard amounts (curies) equivalent to 1 g (6.1×10^{-2} curies) of Pu^{239}. Formula used: $HEP^{X} = 2.16 \times 10^{9}\ (MPC)^{X}_{a(40\ hr/wk)} \times [\text{Max (g/curie, 0.1)}]^{X}$.

The basic equation for the uncollided gamma flux of energy E_o from a cylindrical source with shield at the side and the dose point projection between the ends of the cylinder is:

$$\phi(E_o)=\frac{S_v R_o^2}{4(R + t_c)} \left[F(\theta_2, \mu t + \mu_c t_c) + F(\theta_1, \mu t + \mu_c t_c) \right],$$

where

E_o = gamma energy, Mev,

S_v = volumetric source strength, photons/cm^3,

R_o = radius of cylinder, cm,

R = separation distance from edge of cylinder to dose point, cm,

t_c = self-absorption distance, cm (see Fig. 1),

μ_c = linear attenuation coefficient for a gamma ray of energy E_o for the material from which the source is fabricated, cm^{-1},

$\mu_c t_c$ = self-absorption exponent,

μ = linear attenuation coefficient for a gamma ray of energy E_o for the shield material, cm^{-1},

t = thickness of shield, cm,

θ_1 = angle subtended at the dose point by a horizontal line and a line connecting the dose point with the intersection of the t_c line with the top of the cylinder (see Fig. 1),

θ_2 = angle subtended at the dose point by a horizontal line and a line connecting the dose point with the intersection of the t_c line with the base of the cylinder (see Fig. 1).

$F(\theta,X)$ = Sievert's integral = $\int_0^{\theta} e^{-X \sec \theta'} \, d\theta'$.

(Values of μ_c, μ, $\mu_c t_c$ and t_c may be found in ref. 11, Chapters 10 and 11. The values of $\mu_c t_c$ and t_c are obtained from Figs. 11.14 and 11.15 in Chapter 11 of ref. 11.)

Then the dose rate (in millirads/hr) is determined from the flux by the following equation:

$$D(R,E_o) = 5.767 \times 10^{-2} \, (\mu_a/\rho) \, E_o B(E_o, \mu t) \, \phi \, (E_o)$$

where

$D(R,E_o)$ = dose rate at distance R from uncollided gamma of initial energy E_o, including buildup contribution, millirads/hr,

(μ_a/ρ) = energy absorption mass attenuation coefficient for tissue, cm^2/g,

E_o = photon energy, Mev,

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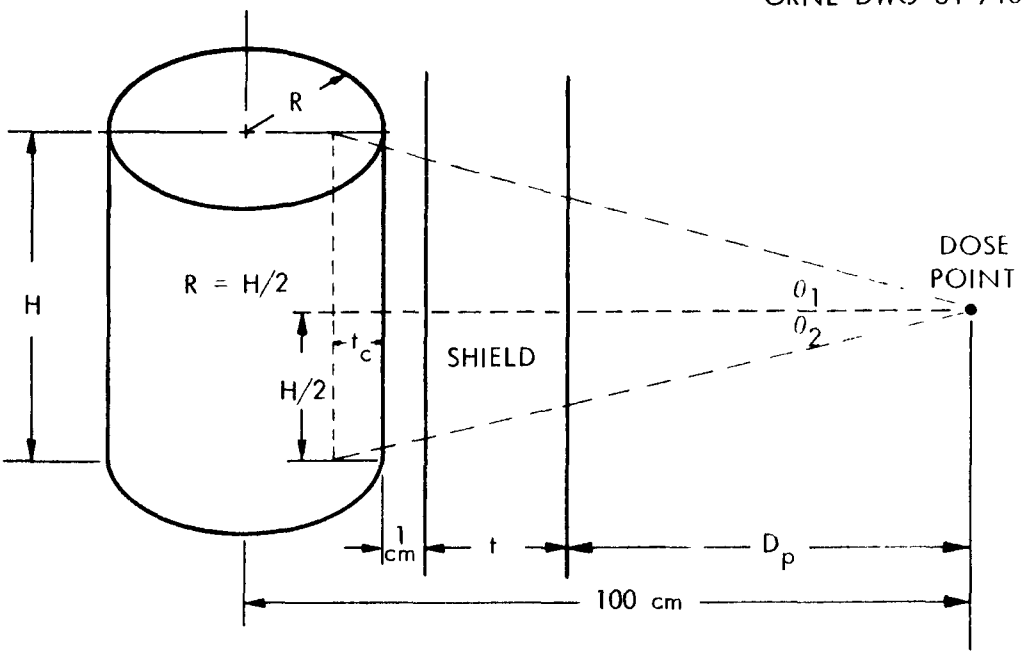
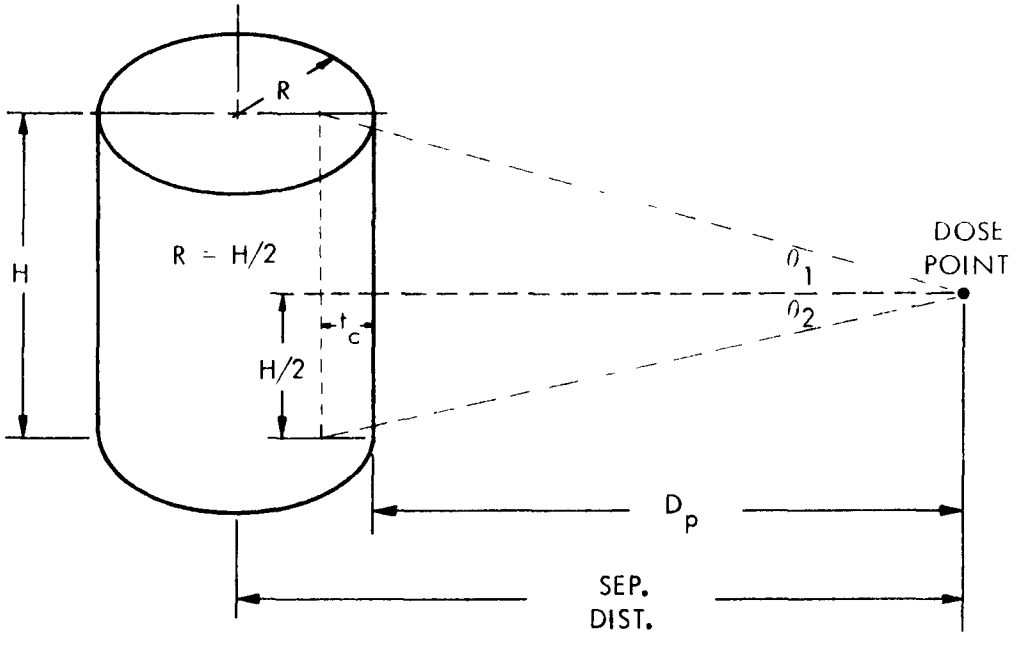


SPATIAL RELATIONSHIP FOR SHIELDED SOURCES



SPATIAL RELATIONSHIP FOR UNSHIELDED SOURCES

Fig. 1. Spatial Relationships Between Sources, Shield, and Dose Point Used for Calculations in This Report.

$B(E_o,\mu t)$ = buildup factor for gamma radiation of initial energy E_o and a shield which has a thickness of μt relaxation lengths.[10] (Buildup factors may be obtained from Tables 10.10 and 10.11 of Chapter 10, ref. 11.)

Fluxes and dose rates are calculated for each gamma group and then summed over the total number of gamma groups specified for the spectrum to obtain the total gamma flux and gamma dose rate.

The self-absorption factor for gamma or bremsstrahlung radiation within a source may be obtained from the following expression:[9]

$$F_\rho(\mu_c R_o) = \frac{\Gamma(\rho + 1)}{\Gamma(3/2)\Gamma(\rho + 1/2)} \int_0^{\pi/2} e^{-2\mu_c R_o \cos\theta} \sin^{2\rho}\theta d\theta,$$

where

$F_\rho(\mu_c R_o)$ = self-absorption factor = ϕ/ϕ_o,

ϕ = gamma flux with self-absorption, photons $cm^{-2}sec^{-1}$,

ϕ_o = gamma flux without self-absorption, photons $cm^{-2}sec^{-1}$,

ρ = 1/2, linear of slab source,

ρ = 1, cylindrical source,

ρ = 3/2, spherical source,

$\Gamma(N)$ = gamma function of N.

The SDC code allows as many as 12 gamma groups per calculation; however, only a few sources required this many to describe the spectrum. The actual gamma groups and source strengths for each source are listed in the tables of heat source data for each isotope.

The lowest gamma energy provided in the permanent data of the SDC code is 0.1 Mev.[8] Where an emission energy E was less than 0.1 Mev, an effective gamma energy of 0.1 Mev was used, and the true source strength was multiplied by $(E/0.1)^2$ to obtain an effective source strength for the calculations. This approximation holds for self-absorption and attenuation through a few relaxation lengths of the shield. The dose rate will be overestimated for thicker shields. However, this is of little importance since in most cases the harder components control the shield thickness. Even for monoenergetic sources this approximation is adequate since only thin shields are required, even for the 0.1-Mev effective gamma.

RADIATION PRODUCTION DURING THE DECAY PROCESSES

Calculations relating to beta, gamma, and neutron emissions are discussed, with special emphasis on processes often ignored but which are important when more accurate determinations of shield requirements are necessary. These processes include beta spectra analyses to permit accurate evaluation of average beta energies, bremsstrahlung production in beta decay, decay gamma production, internal conversion in gamma decay, other sources of gamma rays, and neutron production.

Beta Radiation

In beta decay, an electron and a neutrino are simultaneously emitted from the nucleus of an atom, with the sum of the energies of the two emissions being equal to the total beta decay energy for that particular transition. The amount of energy being carried off by either particle may vary from zero to almost the total beta decay energy, thus yielding particles with a continuous energy spectrum and making the maximum particle energy virtually equal to the beta decay energy. The neutrino can be neglected since it is of no practical concern in biological or heat generation calculations. However, the beta particle is important because of its biological and heating effects.

Two different types of beta transitions based on spin and parity changes are of interest in the study of beta spectra.[14] The "allowed" transitions involve a spin change of 0 or 1, and no parity change (0 or 1, no) while the "unique first forbidden" transitions involve a spin change of 2 and a change of parity (2, yes). Spectra for these two types of transitions can be calculated, but spectra for most other transitions cannot be simply or accurately predicted as yet.

The shape of the energy spectrum of beta particles depends on several factors, in particular the decay energy, the atomic number of the emitting nucleus, and the transition type. The shape of the energy spectrum of beta particles is important to radiation and shielding calculations for determining the average beta particle energy for heat generation calculations, and in determining the bremsstrahlung energy spectrum.

The relative number of betas, P(E), of kinetic energy E from an emitter of atomic number Z_e and maximum kinetic energy E_o is given for an allowed transition by:[14]

$$P(E) = K\eta WF(Z_e, W)(E_o - E)^2, \tag{1}$$

where

K is an arbitrary constant,

W = total beta energy in rest mass units,

$= \frac{E(\text{in Mev})}{0.511} + 1,$

η = beta momentum $= \sqrt{W^2 - 1},$

F(Z,W) = Fermi Differential Function,

Z_e = atomic number of the emitting nucleus.

This Fermi differential function, whose values have been tabulated by the National Bureau of Standards,[15] can be further defined by:

$$F(Z_e,W) = f(Z_p,\eta)/\eta^2. \tag{2}$$

The numerator of the right side of the above equations is the actual function tabulated by the National Bureau of Standards,

where

$$f(Z_p,\eta) = \eta^{2+2S}\, e^{\pm\pi\delta}\, \left|\Gamma(1 + S + i)\right|^2, \tag{3}$$

Z_p = atomic number of product nucleus,

$S = \sqrt{1 - \gamma^2} - 1,$

$\gamma = Z/137.0,$

$\delta = \gamma/(\eta/\sqrt{1 + \eta^2}).$

The $\pm$ sign in the exponential term applies to the spectra of negative (beta) and positive (positron) electrons, respectively.

Distributions for unique first-forbidden transitions are given by:[14]

$$P(E) = K\eta WF(Z_e,W)(E_o - E)^2 \left[\eta^2 + (W_o - W)^2\right]. \tag{4}$$

The average beta energy has in the past been expressed as approximately one-third of the maximum energy, but it varies from about 25 to 50% of the maximum energy, depending on the shape of the spectra. The above expressions for P(E) can be used to determine the average beta energy from:

$$\bar{E} = \left(\sum_{E=0}^{E_o} EP(E)\right)/\left(\sum_{E=0}^{E_o} P(E)\right) \tag{5}$$

The calculated average beta energies and other important nuclear data for the proposed isotopes are listed in Table 3.

Bremsstrahlung Radiation

When an electron is decelerated (or accelerated) it emits a fraction of its energy as electromagnetic radiation. This occurs both when an electron leaves a nucleus (called inner bremsstrahlung) and when it is absorbed (called external bremsstrahlung). In either case the spectral distribution is from zero to the maximum beta energy, but most of the energy is released as less energetic radiations. The energy emitted as inner bremsstrahlung depends only on beta energy, and is small even for the very energetic electrons.

External bremsstrahlung increases with increasing beta energy and increasing atomic number (Z_a) of the absorber. The ratio of energy loss by radiation (E_{rad}) to that by ionization (E_{ion}) for a monoenergetic electron is approximately:

$$E_{rad}/E_{ion} = Z_a E/800, \qquad (6)$$

where E is the kinetic energy of the electron in Mev. This approximation is rather inaccurate for low-energy electrons but is usually considered adequate because the high self- or external absorption of low-energy bremsstrahlung produced from low-energy electrons does not make high accuracy necessary. Application of this formula to beta emission requires division of the beta spectrum into several energy groups with subsequent calculation of the total bremsstrahlung energy resulting from each group. This gives only the total amount of bremsstrahlung energy but not the energy spectrum. For each energy group of betas, the maximum bremsstrahlung energy is equal to the maximum beta energy.

Equation (6) would also indicate that a few percent of the beta energy would escape the source as bremsstrahlung and would not be recoverable as heat within the source itself. If maximum heat recovery is required, it would then be necessary to encapsulate the source in an absorption medium of approximately one-tenth thickness in order to recapture the energy that escapes the source. The actual amount of bremsstrahlung escaping the source would be less than the total bremsstrahlung produced because of the self-absorption within the source. The self-absorption factors depend upon

Table 3. Nuclear Data for Isotopes Considered for Power Sources

Nuclide	Half-Life	Particles				Photons		Power (w/Kilocurie)			
		Type	Max E (Mev)	Avg E (Mev)	Abund. (%)	E (Mev)	Abund. (%)	Alpha	Beta	Gamma	Total
Co^{60}	5.27 y	β	0.312	0.095	100	1.172	100	--	0.56	14.80	15.36
						1.333	100				
Kr^{85}	10.3 y	β	0.672	0.253	100	0.517	0.7	--	1.498	0.024	1.522
Sr^{90}	28 y	β^f	0.545	0.20	100	--	--	--	1.184	--	1.184
Y^{90}	64.2 h	β^f	2.27	0.944	100	--	--	--	5.588	--	5.588
Zr^{95}	65 d	β_f^f	0.89	0.34	2	0.76	55	--	0.73	4.33	5.06
		β_f^f	0.40	0.122	43	0.72	43				
		β^f	0.36	0.109	55						
Nb^{95m}	90 h	IT	--	--	--	0.21	2		Included in above		
Nb^{95}	35 d	β	0.16	0.044	100	0.76	100	--	0.26	4.5	4.76
Ru^{106}	1.0 y	β	0.04	0.009	100	--	--	--	0.05		0.05
Rh^{106}	30 s	β	3.54	1.52	79	2.42	0.3	--	8.4	1.2	9.6
			3.03	1.30	7	1.55	0.5				
			2.40	0.98	11	1.04	2				
			1.99	0.79	2	0.62	11				
			1.42	0.40	1	0.51	20				

[f]Unique first forbidden transition.

Table 3 (cont'd)

Nuclide	Half-Life	Particles Type	Particles Max E (Mev)	Particles Avg E (Mev)	Particles Abund. (%)	Photons E (Mev)	Photons Abund. (%)	Power (w/ Kilocurie) Alpha	Beta	Gamma	Total
Cs^{134}	2.2 y	β	0.68	0.22	9	0.473	2	--	0.87	10.13	11.0
			0.65	0.20	58	0.567	22				
			0.41	0.122	4	0.605	100				
			0.08	0.02	27	0.796	91				
						0.801	18				
						1.038	0.9				
						1.168	3				
						1.367	5				
Cs^{137}	26.6 y	β^f	1.18	0.42	8	--	--	--	1.23	--	1.23
		β^f	0.52	0.19	92	--	--	--			
Ba^{137m}	2.6 m	IT	--	--	92	0.662	83.5	--	--	3.61	3.61
Ce^{144}	285 d	β	0.31	0.090	65	0.134	17	--	0.44	0.20	0.64
			0.22	0.060	5	0.10	2				
			0.17	0.045	30	0.08	2				
						0.04	17				
Pr^{144}	17.5 m	β_f	2.98	1.21	97.7	2.2	0.8	--	7.08	0.19	7.27
		β_f	2.3	0.89	1.3	1.5	0.25				
		β^f	0.8	0.28	1.0	0.7	1.6				
Pm^{147}	2.67 y	β	0.23	0.067	100	0.121	<0.1	--	0.397	--	0.397
Pm^{146}	1.94 y	β	0.78	0.27	35			--	0.50		
		EC	--	--	65	0.45	65		--	4.62	5.12
						0.75	65				

[f] The first forbidden transition.

Table 3 (cont'd)

Nuclide	Half-Life	Particles				Photons		Power (w/Kilocurie)			
		Type	Max E (Mev)	Avg E (Mev)	Abund. (%)	E (Mev)	Abund. (%)	Alpha	Beta	Gamma	Total
Eu^{152}	12.7 y	EC	--	0.04	73	0.122	10.9	--	0.17 EC		
		β	1.46	0.53	5.7	0.245	5.0		0.42	4.05	4.64
			1.05	0.36	1.6	0.344	18.5				
			0.68	0.22	13.8	0.780	8.4				
			0.36	0.10	3.5	0.965	10.1				
			0.22	0.06	2.4	1.087	8.4				
						1.113	9.3				
						1.409	16.8				
Eu^{154}	16 y	β	1.84	0.70	7	0.123	35	--	1.17	6.99	8.16
			1.60	0.60	3	0.593	4				
			0.83	0.28	20	0.725	21				
			0.55	0.17	30	0.875	13				
			0.25	0.07	28	0.998	14				
			0.15	0.04	12	1.007	17				
						1.277	42				
Pm^{148}	42 d	β	2.5	0.99	6	0.10	8	--	1.49	11.28	12.77
			2.0	0.72	1	0.19	5				
			1.2	0.42	3	0.29	12				
			1.07	0.37	5	0.41	16				
			0.76	0.25	19	0.43	11				
			0.64	0.21	11	0.55	92				
			0.57	0.18	15	0.63	90				
			0.47	0.14	40	0.73	36				
						0.91	19				
						1.01	20				
						1.46	2				

Table 3 (cont'd)

Nuclide	Half-Life	Particles Type	Max E (Mev)	Avg E (Mev)	Abund. (%)	Photons E (Mev)	Abund. (%)	Power (w/Kilocurie) Alpha	Beta	Gamma	Total
Tm^{170}	127 d	β	0.968	0.33	76	0.084	3.4	--	1.91	0.02	1.93
			0.884	0.30	24						
Tm^{171}	1.94 y	β	0.097	0.029	98	0.0667	2	--	0.17	0.008	0.178
			0.03	0.01	2						
Tl^{204}	3.9 y	β^f	0.764	0.27	98			--	1.57		
		EC		0.38	2	0.071(x)	2	--	0.045	0.008	1.623
Po^{210}	138.4 d	α	5.305	5.305	100	0.803	1.2×10^{-3}	31.41	--	--	31.41
U^{232}	74 y	α	5.318	5.318	68	0.0579	0.21	31.37	--	--	31.37
			5.261	5.261	31.68	0.131	0.075				
			5.134	5.134	0.32	0.268	4×10^{-3}				
						0.326	4×10^{-3}				
Th^{228}	1.91 y	α	5.421	5.421	71	0.0845	1.6	31.95	--	0.008	31.96
			5.338	5.338	28						
			5.208	5.208	0.4						
			5.173	5.173	0.2						
			5.137	5.137	0.03						
Ra^{224}	3.64 d	α	5.681	5.681	95	0.2411	3.7	33.56	--	0.053	33.61
			5.445	5.445	5						
Rn^{220}	51.5 s	α	6.282	6.282	100	0.542	0.03	37.19	--	0.001	37.19
Po^{216}	0.158 s	α	6.775	6.775	100	--	--	40.11	--	--	40.11

[f] Unique first forbidden transition.

Table 3 (cont'd)

Nuclide	Half-life	Particles Type	Max E (Mev)	Avg E (Mev)	Abund. (%)	Photons E (Mev)	Abund. (%)	Power (w/Kilocurie) Alpha	Beta	Gamma	Total
Pb^{212}	10.64 h	β	0.580	0.19	16	0.115	3	--	0.71	1.07	1.78
			0.340	0.11	80	0.239	74				
			0.160	0.05	4	0.415	0.01				
Bi^{212}	60.5 m	β	2.25	0.95	63.8	0.72	19	--	3.59	4.38	7.97
						0.83	19				
		α	6.09	6.09	9.85	1.03	6	12.97	--	0.06	13.03
			6.05	6.05	25.3	1.34	5				21.00
			5.77	5.77	0.6	1.61	7				
			5.622	5.622	0.05	1.81	7				
			5.603	5.603	0.4	2.20	3.5				
						0.04	25				
						Others	Weak				
Tl^{208}	3.1 m	β	1.795	0.67	50	0.277	10	--	3.40	19.95	23.35
			1.518	0.55	21	0.510	25				
			1.284	0.45	25	0.582	80		1.23[a]	7.22[a]	8.45[a]
			1.032	0.30	4	0.859	15				
						2.620	100				
Po^{212}	0.30 μs	α	8.78	8.78	100	--	--	51.98			51.98
								33.16[a]			33.16[a]
Total all daughters in equilibrium with U - 232											238.63

[a]When existing as a U^{232} or Th^{228} daughter.

Table 3 (cont'd)

Nuclide	Half-life	Particles				Photons		Power (w/kilocurie)			
		Type	Max E (Mev)	Avg E (Mev)	Abund. (%)	E (Mev)	Abund. (%)	Alpha	Beta	Gamma	Total
Pu^{238}	89.8 y	α	5.495 5.452	5.495 5.452	72 28	0.0435 0.099 0.150 0.203 0.76 0.875	0.038 8×10^{-3} 1×10^{-3} 4×10^{-6} 5×10^{-5} 2×10^{-5}	32.46	--	--	32.46
Cm^{242}	163 d	α	6.11 6.066	6.11 6.066	73.7 26.3	0.044 0.1018 0.1576	0.039 3.5×10^{-3} 2.3×10^{-3}	36.10	--	--	36.10
Cm^{244}	18.4 y	α	5.801 5.759	5.801 5.759	76.7 23.3	0.043 0.1 0.15	2.1×10^{-2} 1.5×10^{-3} 1.3×10^{-3}	34.28	--	--	34.28

source size and attenuation coefficients of the material from which the source is fabricated. Fifty percent of the bremsstrahlung energy would be absorbed within the source if the product of $\mu_c R_o$ (linear attenuation coefficient x radius of cylindrical source) = 1.0. The method for calculating self-absorption was given on page 19.

The total bremsstrahlung energy release per beta emission for the important bremsstrahlung emitters is given in the Tables of Appendix A.

According to Wyard,[6] the relative amount of energy I radiated by a monoenergetic electron of kinetic energy E to produce bremsstrahlung photons of energy k is given approximately by:

$$I = C\left[4\left(1 - \frac{k}{E}\right) + 3\,\frac{k}{E}\,\ln\left(\frac{k}{E}\right)\right]\Delta k, \qquad (7)$$

where C is an arbitrary constant. The total energy radiated by the electron is given by summing over all values of k (which gives 1.25 CE), and this energy is equal to the energy loss by radiation as calculated above. For the determination of bremsstrahlung spectra of beta emitters it is necessary to divide both the beta and bremsstrahlung spectra into small energy intervals, to determine the photon intensity for each appropriate combination of intervals, and to combine the results to obtain a photon spectrum.

The bremsstrahlung spectrum, S(k), may be evaluated as the product of three factors, summed over all beta energies from k to E_o. The factors are: (1) the fraction of the total beta energy at energy E, (2) the fraction of this energy which is lost by radiation, and (3) the fraction of this radiated energy which has energy k. The total energy of bremsstrahlung of energy k per Δk photon energy interval per unit beta energy is thus given by:[14]

$$S(k) = \sum_{E=k}^{E_o} \left[\frac{EP(E)}{\Sigma EP(E)}\right]\left[\frac{Z_a E}{Z_a E + 800}\right]\left[\frac{1}{1.25E}\right]\left[4\left(1 - \frac{k}{E}\right) + 3\,\frac{k}{E}\,\ln\frac{k}{E}\right]\Delta k, \qquad (8)$$

where

$\Sigma EP(E)$ is from $E = 0$ to $E = E_o$.

Then the total number of bremsstrahlung photons, N(k), of energy k per Δk photon energy interval per beta emission is given by:

$$N(k) = S(k)\,\bar{E}/k, \qquad (9)$$

where $\bar{E}$ is the average beta energy as defined previously.

Equations (1) through (9) were programmed for the CDC-1604 as Program Spectra (see Appendix A). The calculated results which are tabulated in Tables A.1 through A.14 of Appendix A were used to obtain the average beta energies of Table 3 and the bremsstrahlung groups used in the shielding calculations and listed in Tables 5 through 23. The actual calculations used much smaller photon energy intervals than those listed in Tables 5 through 23; combinations and averaging were necessary in some cases in order to get all photons included in 12 groups or less for the SDC program.

Gamma Radiation

Characteristic gamma rays are emitted in the decay of most radioisotopes: alpha, beta and positron decay, and electron capture generally leave the product nucleus in an excited state, which subsequently decays to the ground state with the emission of one or more photons. These gamma rays vary widely in energy and abundance from one isotope to another. For example, abundant 1- to 3-Mev gamma rays accompany the decay of Co^{60}, Ce^{144}, Rh^{106} (daughter of Ru^{106}) and Tl^{208} (daughter of U^{232}), while there is little gamma accompanying the decay of Pm^{147} or Pu^{238}. Decay schemes and photon spectra are given by Strominger, Hollander, and Seaborg,[17] by Dzhelepov and Peker,[18] and in the Isotope Datadex.[19] These references were used to obtain the gamma spectra given in Table 3.

Internal Conversion

All gamma photons are emitted with the same energy for any given gamma transition, instead of a spectrum of energies as with beta particles. However, calculations involving gamma-ray emission are complicated somewhat by internal conversion, in which part or all of the gamma photons emitted from a nucleus are absorbed by an extranuclear electron of the emitting atom. The kinetic energy of the electron is equal to the initial photon energy less the binding energy of the electron. This binding energy is equal to the energy of the x ray for the shell from which the electron was ejected. An x ray is also emitted from the atom when an electron drops into the site vacated in the conversion process. Thus the radiation observed from gamma transitions in a given nuclide may consist partly of monoenergetic electrons and x rays, and fewer gamma photons than expected.

Internal conversion is very common for gamma photons of energy less than about 0.2 Mev, and usually negligible for photons above about 0.7 Mev. The extent of internal conversion has been determined experimentally for most of the nuclides of interest and often need to be included in calculations involving gamma rays.[14]

The radiation from gamma emitters where internal conversion is appreciable should therefore be treated as at least two separate types of radiation for most purposes. Gamma shielding and dose rate calculations should consider only the photons that escape the atom rather than all of those emitted from the nucleus. Heating calculations should include the energy imparted to the conversion electrons along with the heat generated by beta radiation, while the gamma photons should be treated separately since their energy is ordinarily lost at some distance from the source. The x rays may either be treated as a third class, or lumped with the betas and conversion electrons when their penetrating power is small compared with the gamma radiation. If the ratio of conversion electrons to gamma photons is large, and bremsstrahlung is small, x ray emission may become important. In such cases it might be necessary to include the internal conversion of the x rays, which is then followed by the emission of lower-energy x rays.[20]

Other Sources of Gamma Radiation

Gamma rays are also emitted by spontaneous fission, by fission products produced during spontaneous fission, by inelastic scattering of neutrons, and by neutron capture.

Prompt Gammas from Spontaneous Fission

A continuous spectrum of gamma rays, varying in energy from approximately 0.1 to 8 Mev, is emitted in the process of spontaneous fission. About 7.5 Mev is emitted in this fashion per fission.

Fission-Product Decay Gammas

Fission products formed in spontaneous fission ultimately emit a total of approximately 5.5 Mev in gamma rays per fission. These gamma rays generally have lower energy than the prompt gammas from spontaneous fission. The fission-product gammas approach a steady-state decay rate within a few hours after a spontaneously fissioning nuclide is isolated.

Data on the gamma spectrum from U^{235} fission products as a function of irradiation and cooling time are given by Goldstein[12] and by Blizard and Abbott.[11] The yield of fission products from many of the actinide elements is given by Hyde.[21]

Inelastic Gammas

Inelastic scattering of neutrons, the predominant process of slowing down in the heavier elements, is accompanied by the emission of photons. The total photon energy is less than the incident neutron energy. Ordinarily the energy absorbed from the neutron is emitted as several photons.

Neutron-Capture Gammas

Nearly all isotopes, with the exception of Li^6 and B^{10}, emit energetic gamma rays upon the capture of a neutron. This gamma source is especially important for those elements with high neutron capture cross sections since these elements compete most strongly for the available neutrons.

One must know the neutron flux and spectrum in a shield to calculate inelastic and capture gammas. Approximate answers for the ordinarily small sources and shields of interest in space applications may be obtained with the DSN reactor transport code or other reactor codes. Renupak and NIOBE are available for more rigorous calculation. Goldstein,[12] and Price, Horton and Spinney[13] present data for calculating sources of capture and inelastic gammas; but, in general, it is not practical to perform the calculations by hand, especially when fast and intermediate neutrons are important.

The calculation of gamma radiation produced by neutron capture and inelastic scattering was not performed for this study. The neglection of this source of gamma radiation should not be serious. The sources of gamma from spontaneous fission and from spontaneous fission produced fission products were included.

Neutron Radiation

Neutron radiation may be produced by any or all of the following processes: spontaneous fission, α,n reactions, γ,n reactions, and n,2n reactions.

Neutrons from Spontaneous Fission

A continuous spectrum of neutrons varying in energy from about 0.1 to 18 Mev is emitted in the spontaneous fission process. An average of two to four neutrons are emitted per fission, depending on the fissioning isotopes. Yields and spectra of prompt neutrons and gammas from spontaneous fission are given by Hyde.[21]

α,n Neutrons

Neutrons are produced from mixtures of alpha emitters and certain elements, including most of the light elements (see Table 4). The energy spectrum of the neutrons is a continuous distribution extending from very low energies up to a maximum energy that is slightly less than the sum of the alpha energy and the energy liberated in the reaction. The yield of neutrons varies with the target elements and the composition of the mixture. Beryllium produces the most neutrons per alpha. Aluminum, carbon, and oxygen produce progressively fewer neutrons.

Experimental data on α,n sources are given by Price, Horton and Spinney.[13] The α,n emission rate and maximum neutron energy of mixtures of Po^{210}, Pu^{238}, Cm^{242} and Cm^{244} with various light elements are shown in Table 4. It is conservative to use the maximum neutron energy to calculate shielding and radiation levels if the spectrum is not known. Approximately 5×10^6 neutrons per sec per α curie with neutron energy up to 11 Mev are emitted from mixtures of beryllium and alpha emitters. The yields and energy increase with alpha energy. Yields and maximum neutron energies are lower for aluminum, oxygen, and carbon.

Photoneutrons

A photon can eject a neutron when its energy exceeds the neutron binding energy. A photon energy greater than 7 Mev is required for all isotopes except deuterium, Be^9, C^{14} and Li^6. Cross sections for the reaction are of the order of 1 mb, and the neutrons generally have energy of 0.1 to 1 Mev. Data for photoneutron calculations are given by Goldstein[12] and by Price, Horton, and Spinney.[13]

n,2n Neutrons

If the bombarding neutron has sufficient energy, it is possible for the compound nucleus to emit two neutrons. The threshold energy is relatively high for all isotopes except deuterium and Be^9. Thus, beryllium,

which may be used as a heat shield in space, may serve to multiply the neutron flux, though usually to only a slight degree. Price, Horton, and Spinney[13] give data for n,2n calculations, and the DSN code calculates n,2n reactions.

Neutron Scattering and Attenuation

Most materials elastically and inelastically scatter neutrons through relatively large angles. In general, neutrons scatter more readily than gamma rays. Scattering at large angles is most predominant with heavier elements.

Computer Codes for Neutron Dose Calculations

The calculation of dose and spectra from fast neutrons in thick shields has become precise only in the last few years with the development of special transport codes for handling very fast neutrons. While the reactor transport and diffusion codes, which have been in use for many years, may adequately calculate the behavior of intermediate- to low-energy neutrons, they are not set up to handle the high-energy penetrating neutrons that are important in thick neutron shields.

In addition to Monte Carlo codes, there are available two new transport codes called Renupak and NIOBE. They are both codes for an IBM-7090 computer, and they calculate neutron current, flux, spectrum, and dose rate as a function of shield thickness. Correlated data from calculations with Renupak (moments method), for some neutron shields, are presented by Goldstein.[12] The Radiation Shielding Information Center at Oak Ridge[21] can furnish references to other calculations including those for lithium hydride and concrete.

The neutron dose rate attenuation factors for a fission source are given in Fig. 2 for beryllium, water, lithium hydride, and two hydrocarbons whose structure may be represented by CH or CH_2. This figure along with the actual dose rates through water shields which are presented graphically for the neutron emitting isotopes may be used to determine neutron dose rates through any of the five materials.

Table 4. α,n Emission from Mixtures of Alpha Emitters and Light Elements

Target Element	Property	Radioisotope Po^{210}	Pu^{238}	Cm^{242}	Cm^{244}
Be	Neutron emission rate (n/sec/α curie)	3×10^6	4×10^6	7×10^6	6×10^6
	Max neutron energy (Mev)	10.8	10.9	11.5	11.2
C	Neutron emission rate (n/sec/α curie)	4×10^3	5×10^3	1×10^4	8×10^3
	Max neutron energy (Mev)	7.2	7.4	8.0	7.6
O	Neutron emission rate (n/sec/α curie)	2.4×10^3	3×10^3	6×10^3	5×10^3
	Max neutron energy (Mev)	5.6	5.8	6.3	6.1
Al	Neutron emission rate (n/sec/α curie)	3×10^4	4×10^4	7×10^4	5×10^4
	Max neutron energy (Mev)	2.3	2.5	3.1	2.7

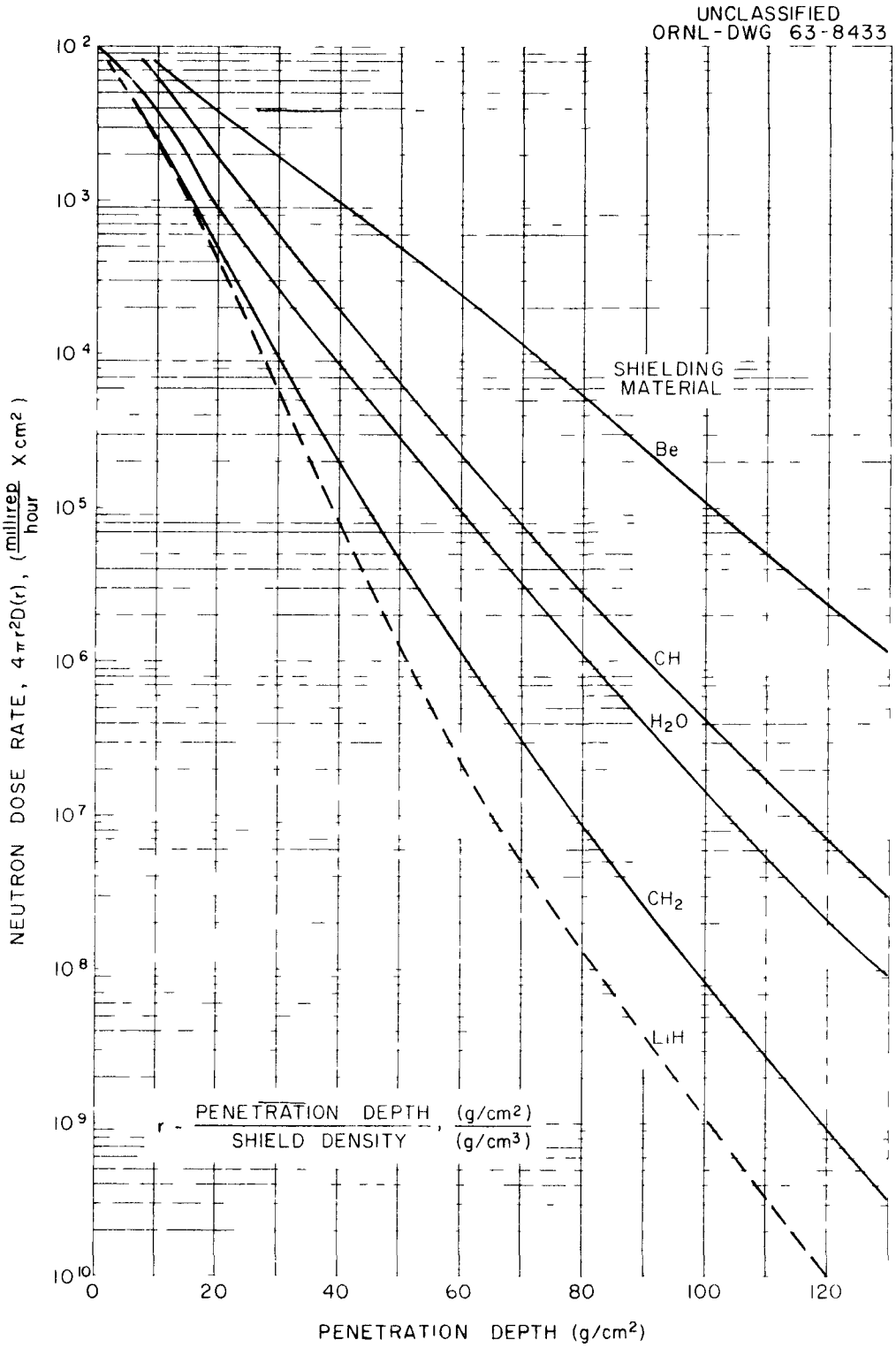


Fig. 2. Fast Neutron Dose Rate (Multiplied by $4\pi r^2$) in Various Materials as a Function of Penetration Depth from a Unit Point Isotropic Fission Source.

RADIATION AND SHIELDING REQUIREMENTS FOR ISOTOPIC SOURCES

This section presents an evaluation of the studied isotopes in terms of their radiation characteristics and shielding requirements. Each isotope is handled as an individual entity. Nuclear data is given in tabular form for each type of heat source calculated. This table lists such characteristics as specific activity, specific power, neutron emission rates (if any), type of compound and density, and gamma emission rates for each of the energy groups selected for the SDC shielding calculations. The table also notes the type of emission (decay gamma, bremsstrahlung, x ray, spontaneous fission gamma) and the most predominant radiation from an unshielded source. Of course the predominant radiation would shift to the higher and higher components of the spectra as the shield thickness increases. This shift can cause a very marked initial attenuation from a small amount of shielding. This is especially true when the predominant energy for the unshielded source is low. The actual gamma spectra for each isotope, listed in Table 3, was used to group the energies for input data to the SDC code. These gamma spectra were obtained from several sources.[17,18,19]

The results of the shielding calculations are presented for the following conditions: (1) gamma dose rates from unshielded sources as a function of distance from center of source; (2) gamma dose rates at 1 m from the center of the source as a function of shield thickness for iron, lead, and uranium shield (in some cases, water) materials; and (3) neutron dose rates at 1 m from the center of the source as a function of water shield thickness. The neutron dose rates through other materials may be estimated by using Fig. 2, which gives the attenuation factors for beryllium, water, lithium hydride, and two hydrocarbons whose structure may be represented by CH or CH_2.

Gamma dose rates from shielded sources at distances other than 1 m may be estimated by finding the relative distance attenuation factor from the graph for the unshielded case and applying the factor to the corresponding value for the shielded source. Neutron dose rates at distances other than 1 m may be estimated by using the inverse-square law since the neutron dose rate calculations assumed that the entire neutron source was located as a point at the center of the source.

Cobalt-60

Cobalt-60 is produced in substantial quantities today for use as a gamma source. In portable radiographic devices, it has largely replaced radium. Its half-life is fairly long, 5.3 yrs; it is of lower biological toxicity than most other possible isotopic source materials (Sr^{90} and all the alpha emitters). The fact that the inactive normal cobalt can be encapsulated and then irradiated to the desired activity without further processing is a distinct advantage specific to this isotope. The characteristics of its radiation are unfortunate for the present applications because of the major component of high-energy gammas that must be absorbed to yield appreciable heat. Of all of the isotopes considered in this report, only Co^{60} and $Zr\text{-}Nb^{95}$ require a special design for the heat generator, apart from also requiring heavy shielding. For those applications where weight is an important consideration, Co^{60} is, therefore, at a disadvantage. Furthermore, its intense and energetic gamma activity inhibits its use as a typical or representative heat source for use in demonstration devices. Like all isotopes that have a substantial portion of their heat in a gamma component or in hard bremsstrahlung, an appreciable part of this energy is not recoverable because the penetrating radiation cannot be completely absorbed and converted to heat only in that section of the device in contact with the thermoelectric elements. The specific power of Co^{60} is high, 17.4 w/g, but the concentrations which can be practically attained would limit the heat output to about 20% of that figure. Even so, for metal, a power density of 27.3 w/cc (200-curies-per-gram sources) appears achievable.

Isotopic heat source data for Co^{60} sources appear in Table 5, and the nuclear data appear in Table 3. The results of the radiation intensity calculations for shielded and unshielded sources are given in Figs. 4 through 6 for 75-curies-per-gram sources, and in Fig. 3 and Fig. 7 through 9 for 200-curies-per-gram sources. The radiation intensity from unshielded sources having a specific activity of 75 curies/g were not calculated or plotted.

Table 5. Physical Properties of Cobalt-60 as an Isotopic Power Source

General Properties	
Half-life	5.27 y
Specific activity of pure isotope, curies/g	1124
Specific power of pure isotope, w/g	17.26
Compound	metal
Weight of active isotope in metal, g/cc	0.59[a], 1.58[c]
Effective density of metal, g/cc	8.9
Power, w/cc of metal	10.3[a], 27.3[c]
Activity, curies/cc of metal	667.5[a], 1780[c]

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
2.385×10^{12}	1.172
2.385×10^{12}	1.333[e]

[a]Source: 75 curies per gram of cobalt metal.

[c]Source: 200 curies per gram of cobalt metal.

[e]Predominant radiation from unshielded 5000 watt source.

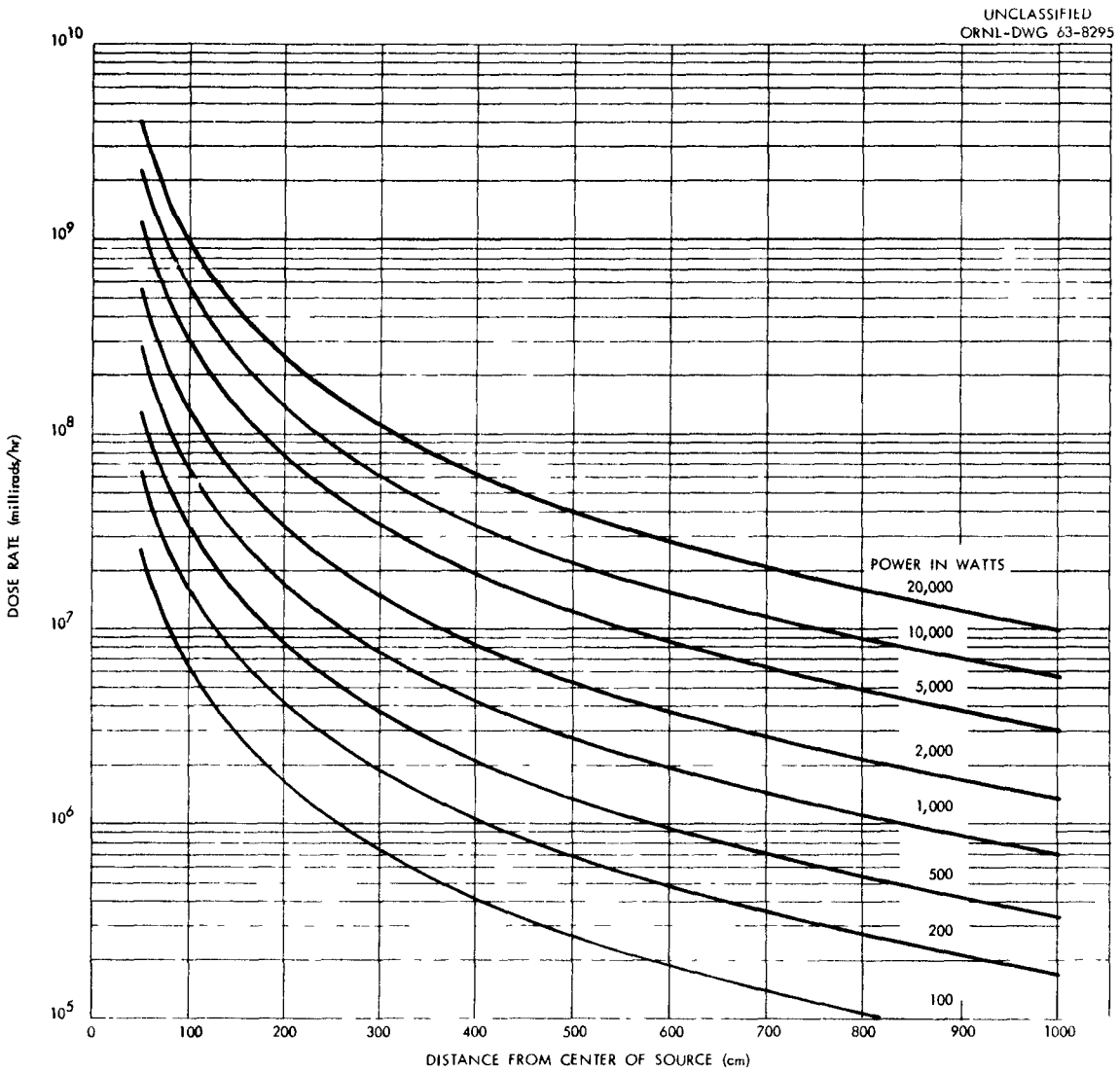


Fig. 3. Gamma Dose Rates from Unshielded Isotopic Power Sources of Cobalt-60 as a Function of Distance from Center of Source. Specific activity of source = 200 curies per gram of cobalt.

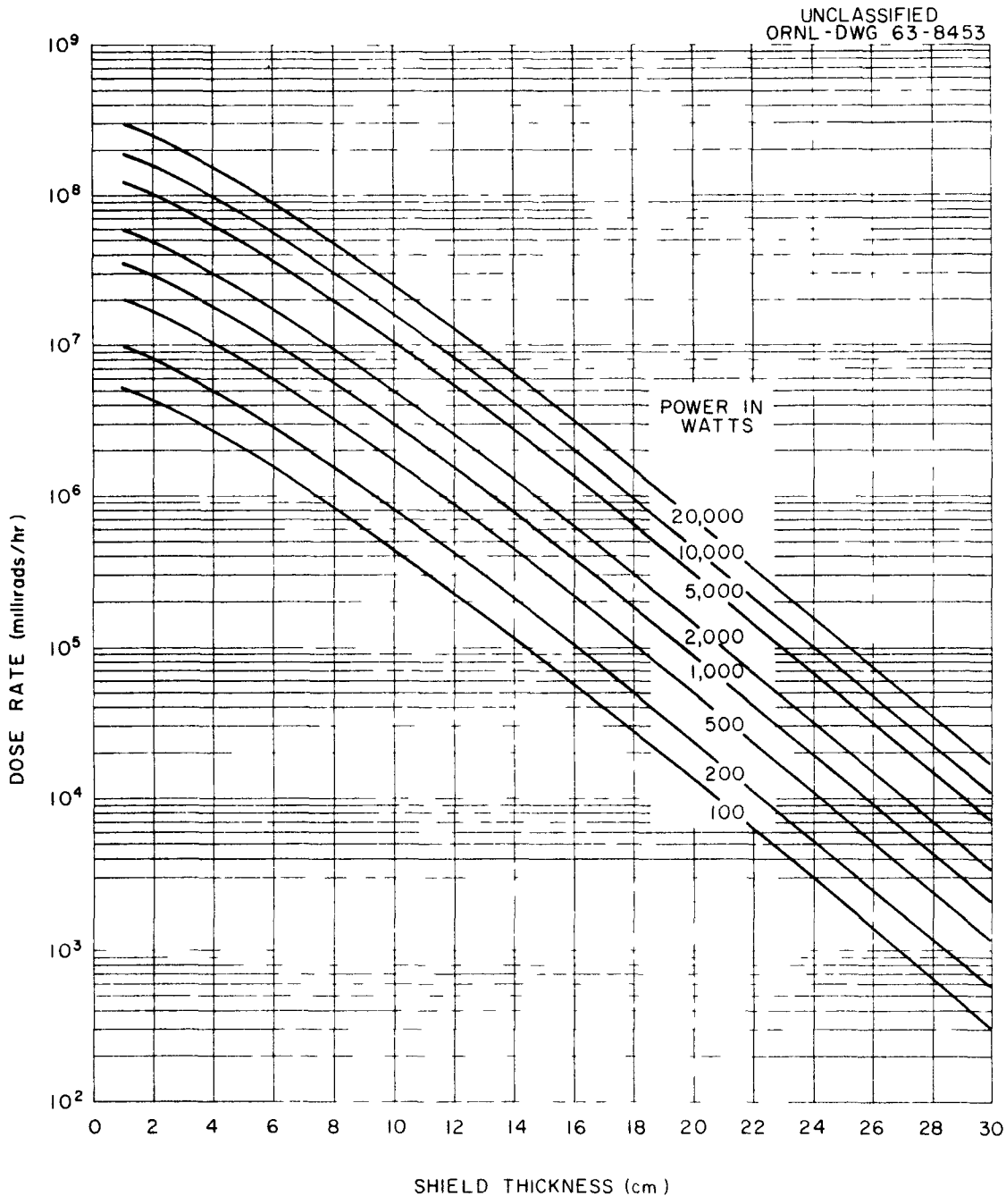


Fig. 4. Gamma Dose Rates from Iron-Shielded Isotopic Power Sources of Cobalt-60. Center of source to dose point separation distance = 100 cm. Specific activity of source = 75 curies per gram of cobalt.

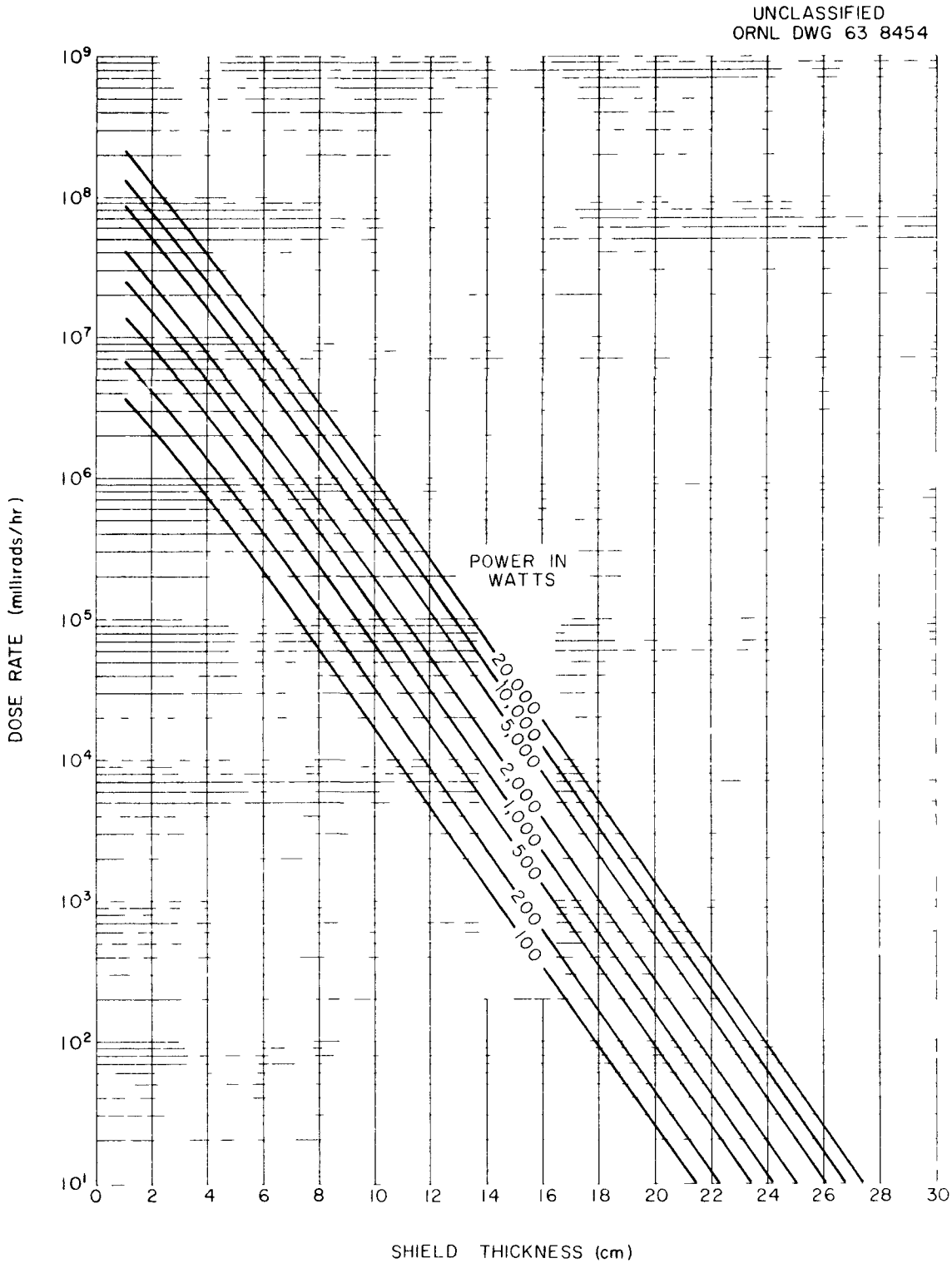


Fig. 5. Gamma Dose Rates from Lead-Shielded Isotopic Power Sources of Cobalt-60. Center of source to dose point separation distance = 100 cm. Specific activity of source = 75 curies per gram of cobalt.

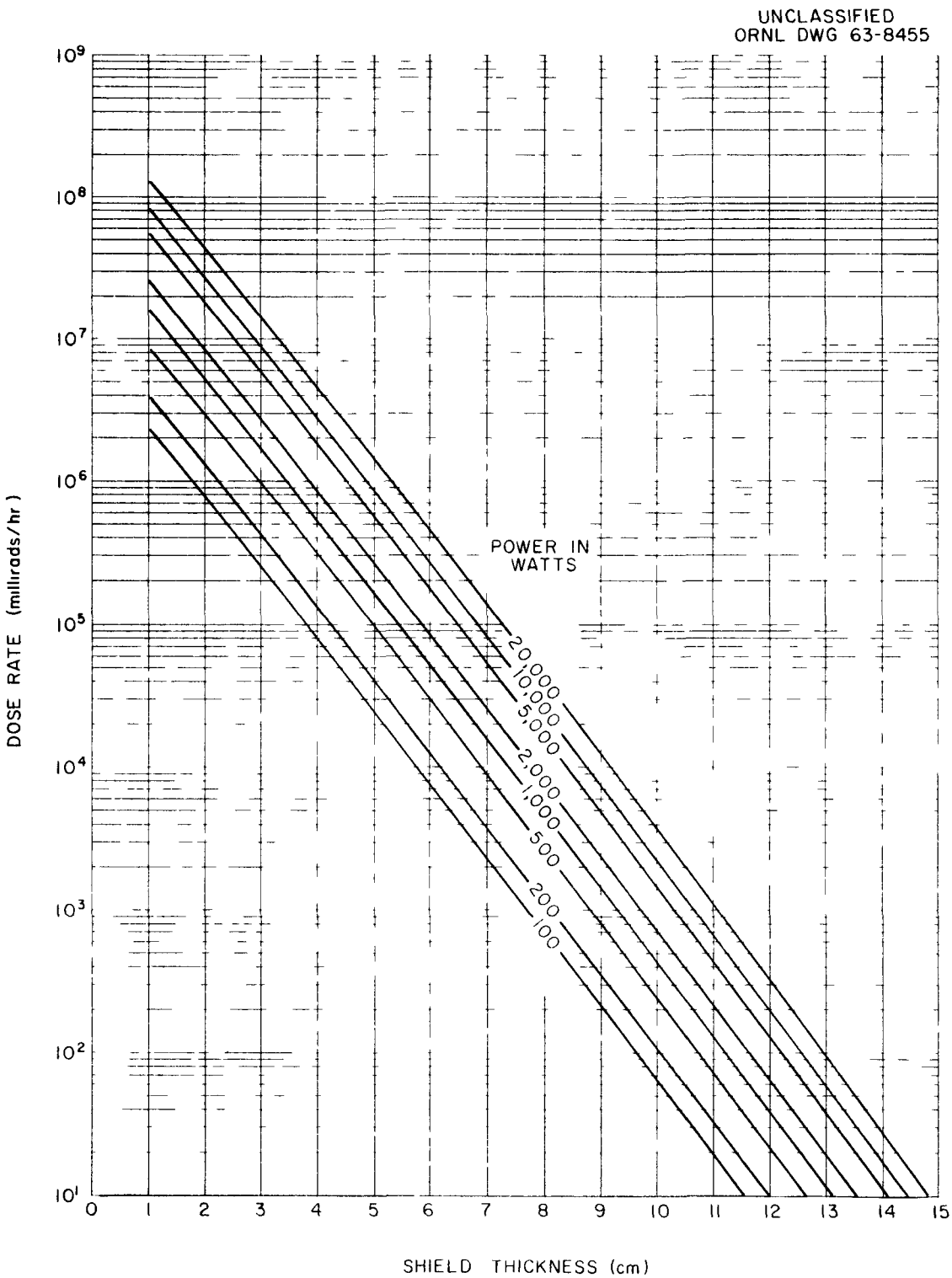


Fig. 6. Gamma Dose Rates from Uranium-Shielded Isotopic Power Sources of Cobalt-60. Center of source to dose point separation distance = 100 cm. Specific activity of source = 75 curies per gram of cobalt

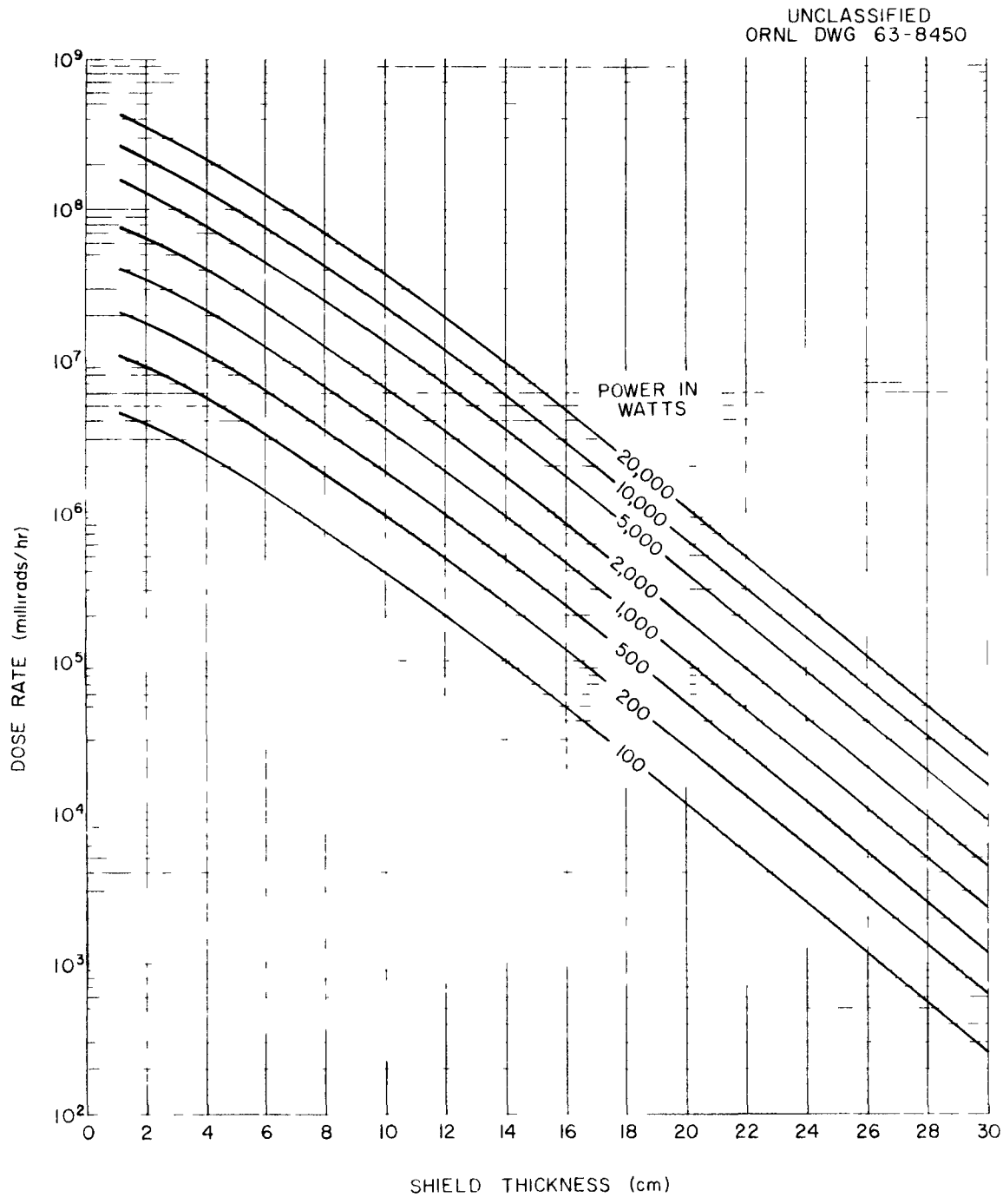


Fig. 7. Gamma Dose Rates from Iron-Shielded Isotopic Power Sources of Cobalt-60. Center of source to dose point separation distance = 100 cm. Specific activity of source = 200 curies per gram of cobalt.

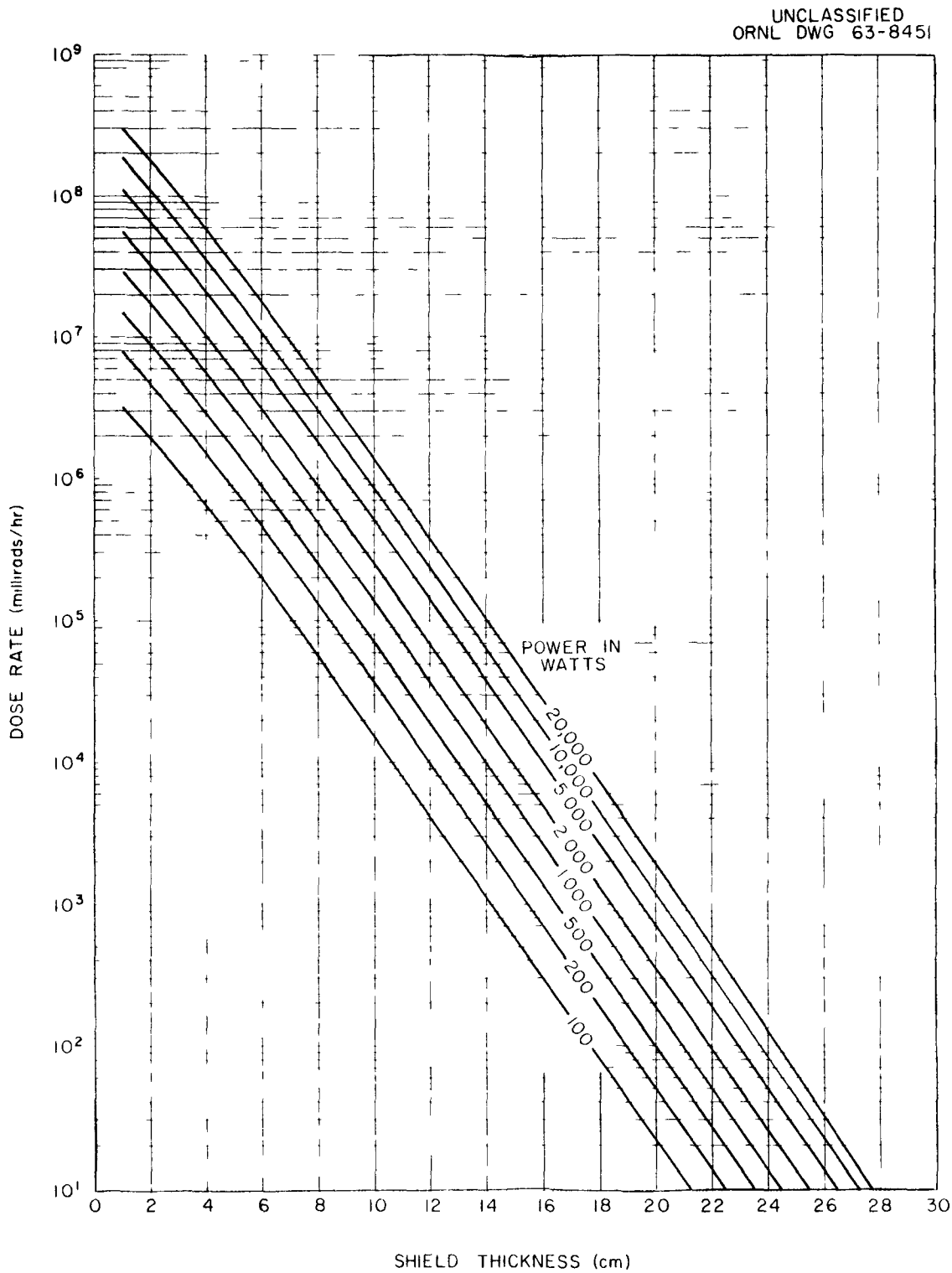


Fig. 8. Gamma Dose Rates from Lead-Shielded Isotopic Power Sources of Cobalt-60. Center of source to dose point separation distance = 100 cm. Specific activity of source = 200 curies per gram of cobalt.

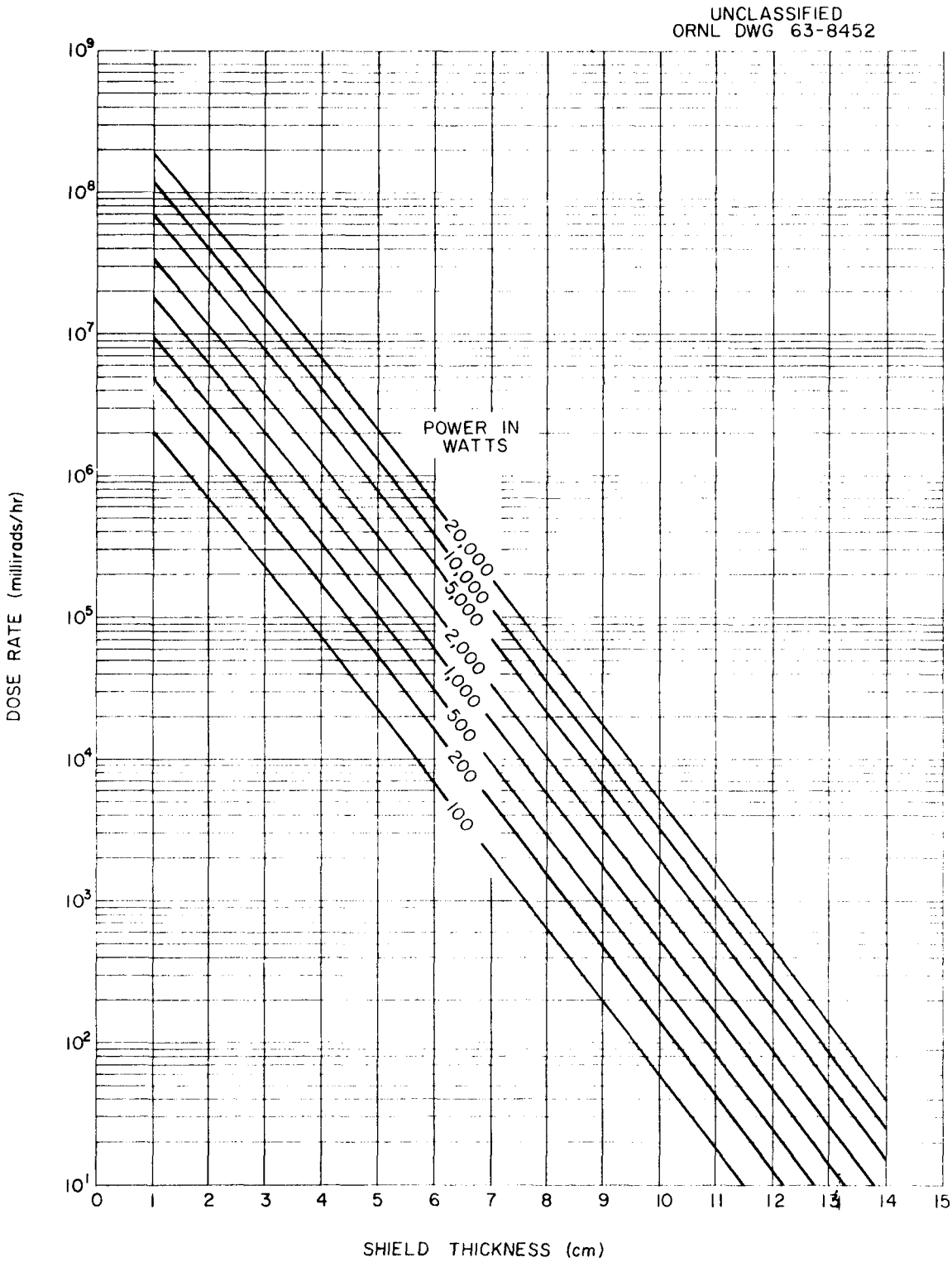


Fig. 9. Gamma Dose Rates from Uranium-Shielded Isotopic Power Sources of Cobalt-60. Center of source to dose point separation distance = 100 cm. Specific activity of source = 200 curies per gram of cobalt.

Krypton-85

Krypton-85 has a suitable half-life (10.3 yr) to consider it as a possible power source. Its specific power is about 0.62 w/g, mostly from beta (which produces a small amount of bremsstrahlung x rays) but with some fairly energetic gamma (0.52 Mev) derived from less than 1% of the beta decays. Its fission yield is only about 0.3% from U^{235} fission.[21,23,24] It should be recoverable along with stable isotopes of krypton such that without isotope separation, it would comprise only about 7.5% of the krypton mixture. In this study, isotopic separation was assumed, and a krypton mixture containing 50 wt % Kr^{85} was used in the computations. Computations were based on liquid krypton at a density of 2.16 g/cc and a power density of 0.67 w/cc.

Isotopic heat source data for Kr^{85} sources appear in Table 6, and the nuclear data appear in Table 3. The results of the radiation intensity calculations for shielded and unshielded sources are given in Figs. 10 through 13.

Table 6. Physical Properties of Krypton-85 as an Isotopic Power Source

General Properties	
Half-life	10.3 y
Specific activity of pure isotope, curies/g	410
Specific power of pure isotope, w/g	0.623
Compound	liquid krypton
Weight of active isotope in liquid, g/cc	1.08
Effective density of liquid, g/cc	2.16
Power, w/cc of liquid	0.673
Activity, curies/cc of liquid	443

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
7.805×10^{11}	0.1[b]
6.945×10^{10}	0.25[b]
8.41×10^{9}	0.40[b]
4.185×10^{8}	0.55[b]
1.702×10^{11}	0.517[e]

[b]Bremsstrahlung x ray group.

[e]Predominant radiation from unshielded 5000 watt source.

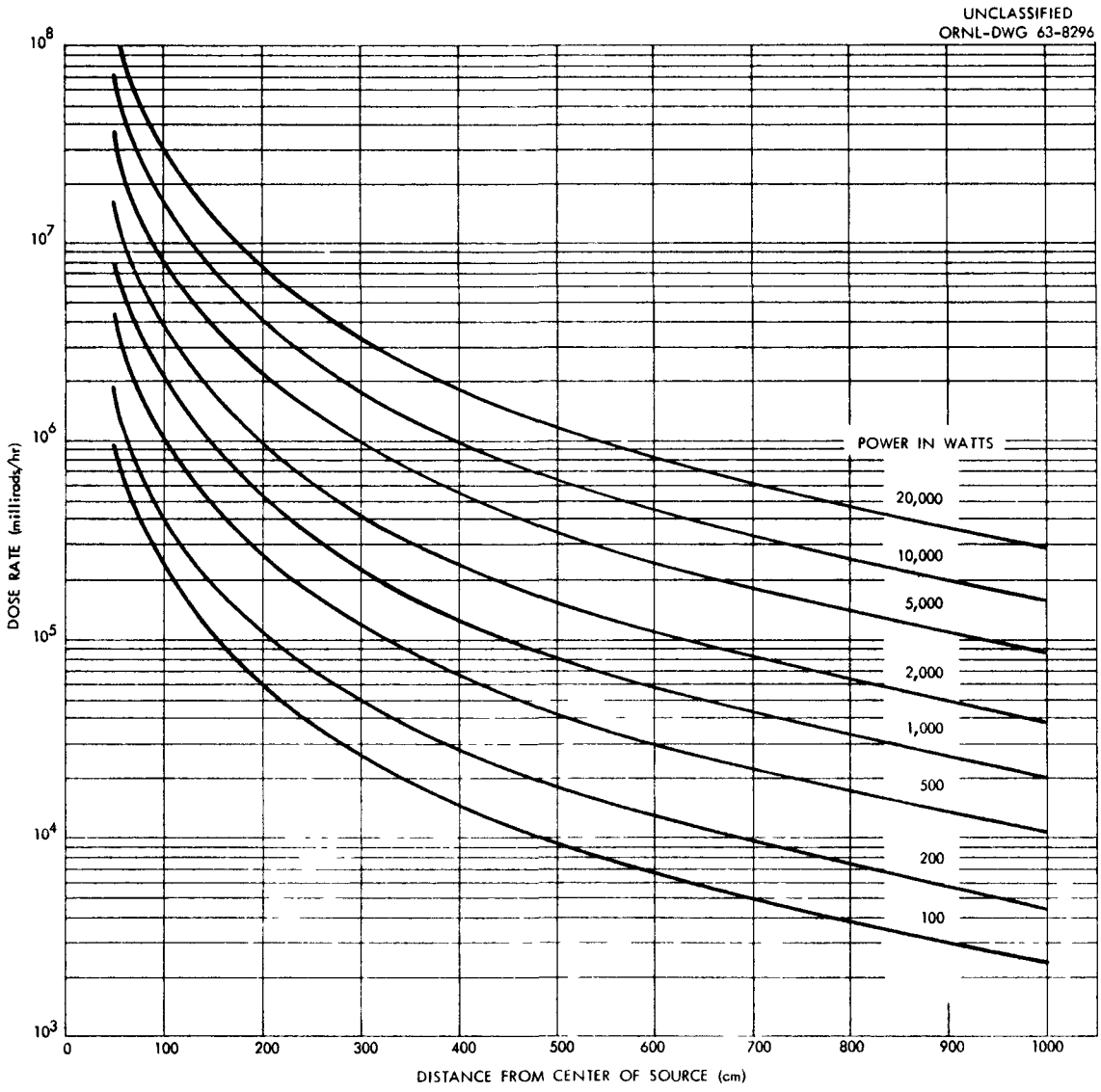


Fig. 10. Gamma Plus Bremsstrahlung Dose Rates from Unshielded Isotopic Power Sources of Krypton-85 as a Function of Distance from Center of Source. Source is assumed to be liquid krypton containing 50% Kr^{85}.

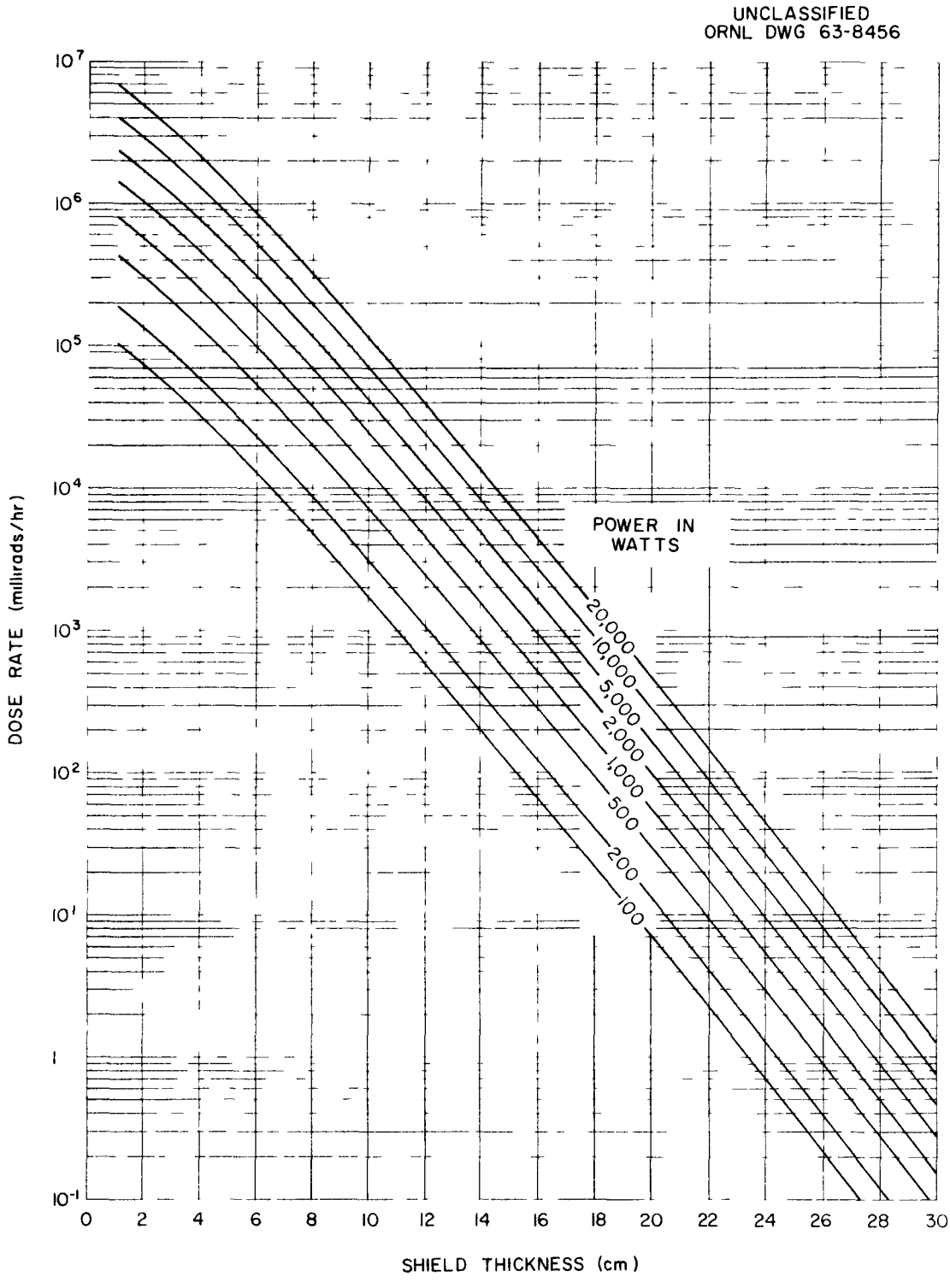


Fig. 11. Gamma Plus Bremsstrahlung Dose Rates from Iron-Shielded Isotopic Power Sources of Krypton-85. Center of source to dose point separation distance = 100 cm. Source is assumed to be liquid krypton containing 50% Kr^{85}.

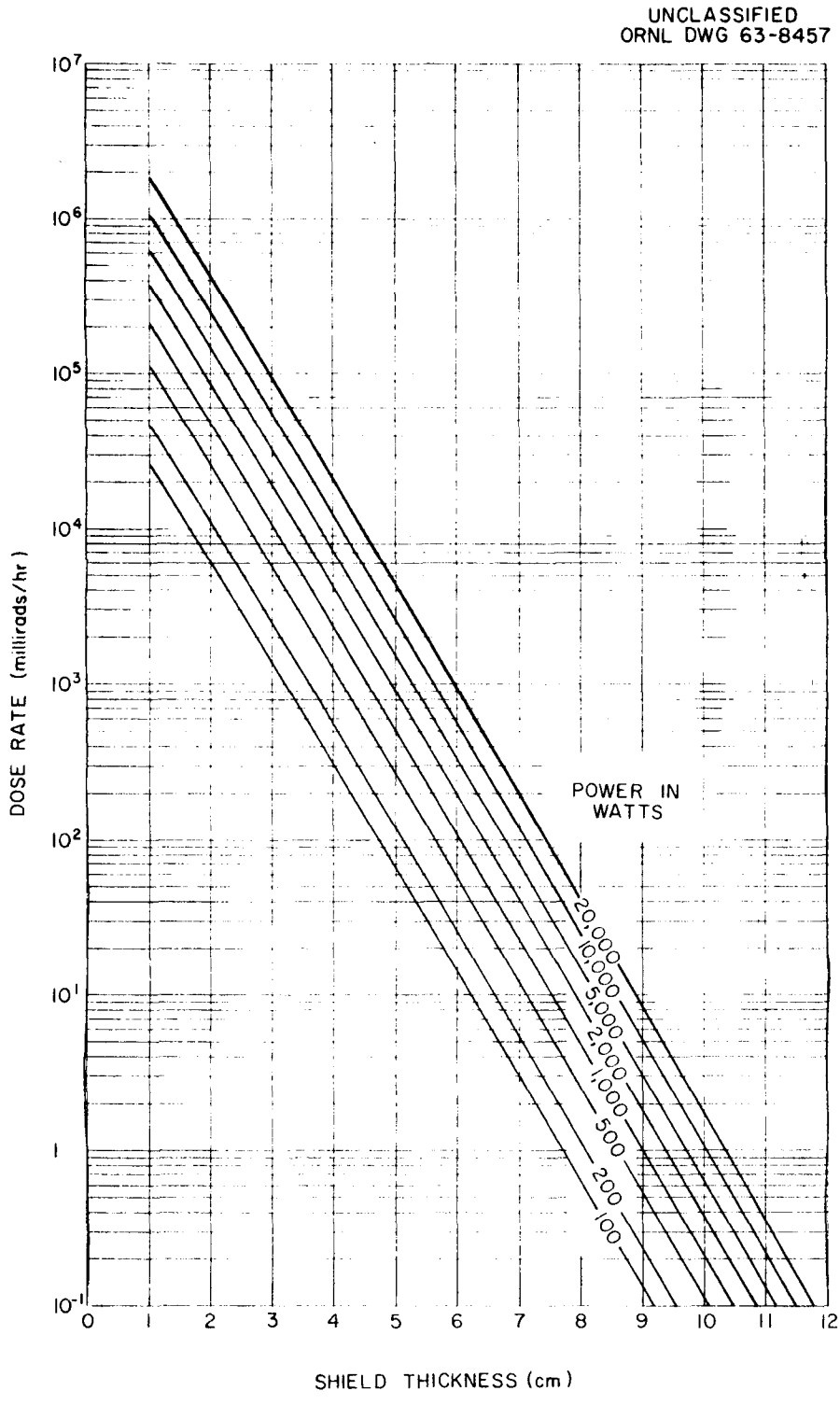


Fig. 12. Gamma Plus Bremsstrahlung Dose Rates from Lead-Shielded Isotopic Power Sources of Krypton-85. Center of source to dose point separation distance = 100 cm. Source is assumed to be liquid krypton containing 50% Kr^{85}.

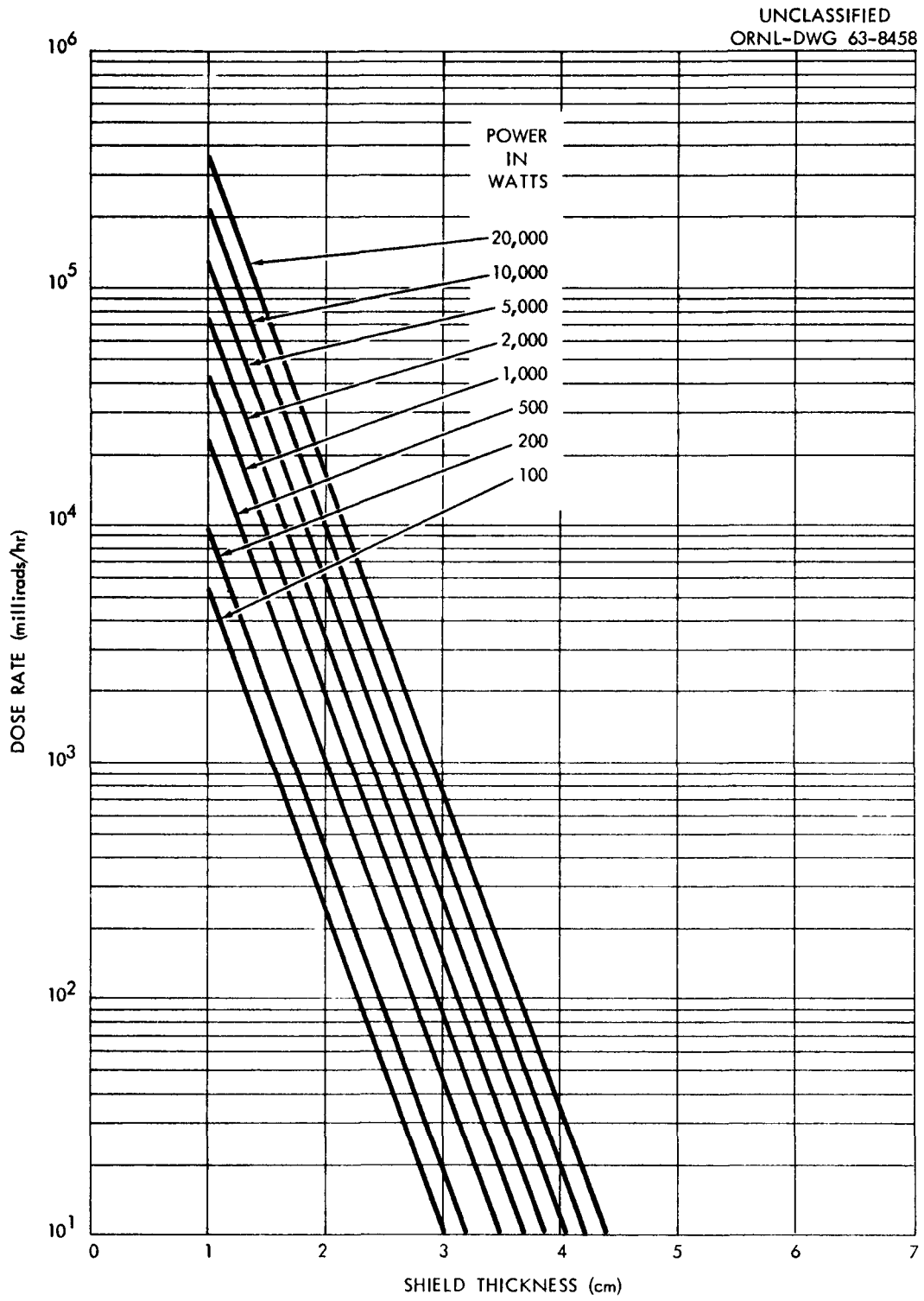


Fig. 13. Gamma Plus Bremsstrahlung Dose Rates from Uranium-Shielded Isotopic Power Sources of Krypton-85. Center of source to dose point separation distance = 100 cm. Source is assumed to be liquid krypton containing 50% Kr^{85}.

Strontium-90

Strontium-90 ($T_{1/2}$ = 28 yr) is produced in reactor operation along with other strontium isotopes (including the stable isotopes Sr^{86}, Sr^{88}) and Sr^{89}, which has a half-life of 51 days and emits a very high energy beta particle. As recovered, the strontium isotopic mixture should consist of approximately 50% Sr^{90}. Strontium-90 has a decay daughter, Y^{90}, in secular equilibrium, which emits a very-high-energy beta particle. The intense, high-energy beta activity of Sr^{89} and Y^{90}, as well the softer beta of Sr^{90}, results in high-energy bremsstrahlung for which heavy shielding is required. Unless the strontium has aged sufficiently, the 51-day Sr^{89} contributes a substantial fraction of heat from the isotopic mixture. Some degree of aging is advisable in order to provide a heat source with minor contribution from Sr^{89}; otherwise, the source will show a high initial decrease in heat output characteristic of the short-lived Sr^{89}. A curie of Sr^{89} will contribute nearly 50% of the heat and 40% of the radiation produced by a curie of Sr^{90} in equilibrium with Y^{90}. The specific power of Sr^{90} is 0.93 w/g. The power density of strontium oxide is approximately 1.4 w/cc, while the power density of strontium titanate is 0.85 w/cc. The bremsstrahlung production in the titanate is about 88% of that in the oxide.

Although all radioisotopes are hazardous and have a very low maximum permissible concentration in air or water (MPC),[8] Sr^{90} has a far larger MPC than any of the alpha emitters; however, among the fission products and the beta-active transmutation products, Sr^{90} is more hazardous.

Isotopic heat source data for Sr^{90} sources appear in Table 7 for strontium oxide sources and in Table 8 for strontium titanate sources, and the nuclear data appear in Table 3. The results of the radiation intensity calculations for shielded and unshielded sources of strontium oxide are given in Figs. 14 through 17, while the results of these same calculations for strontium titanate are given in Figs. 18 through 21.

Table 7. Physical Properties of Strontium-90 as an Isotopic Power Source

General Properties

Half-life	28 y
Specific activity of pure isotope, curies/g	142
Specific power of pure isotope, w/g	0.962
Compound	SrO
Weight of active isotope in compound, g/cc	1.46
Effective density of compound, g/cc	3.7
Power, w/cc of compound	1.4
Activity, curies/cc of compound	208

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
1.923×10^{11}	0.25[b,e]
8.695×10^{10}	0.50[b]
2.649×10^{10}	0.80[b]
8.255×10^{9}	1.10[b]
2.229×10^{9}	1.40[b]
4.086×10^{8}	1.70[b]
2.806×10^{7}	2.00[b]

[b] Bremsstrahlung x ray group.

[e] Predominant radiation from unshielded 5000 watt source.

Table 8. Physical Properties of Strontium-90 as an Isotopic Power Source

General Properties	
Half-life	28 y
Specific active of pure isotope, curies/g	142
Specific power of pure isotope, w/g	0.962
Compound	$SrTiO_3$
Weight of active isotope in compound, g/cc	0.884
Effective density of compound, g/cc	3.93
Power, w/cc of compound	0.85
Activity, curies/cc of compound	126

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
1.705×10^{11}	0.25[b,e]
6.612×10^{10}	0.50[b]
1.958×10^{10}	0.80[b]
6.103×10^{9}	1.10[b]
1.650×10^{9}	1.40[b]
3.027×10^{8}	1.70[b]
2.082×10^{7}	2.00[b]

[b] Bremsstrahlung x ray group.

[e] Predominant radiation from unshielded 5000 watt source.

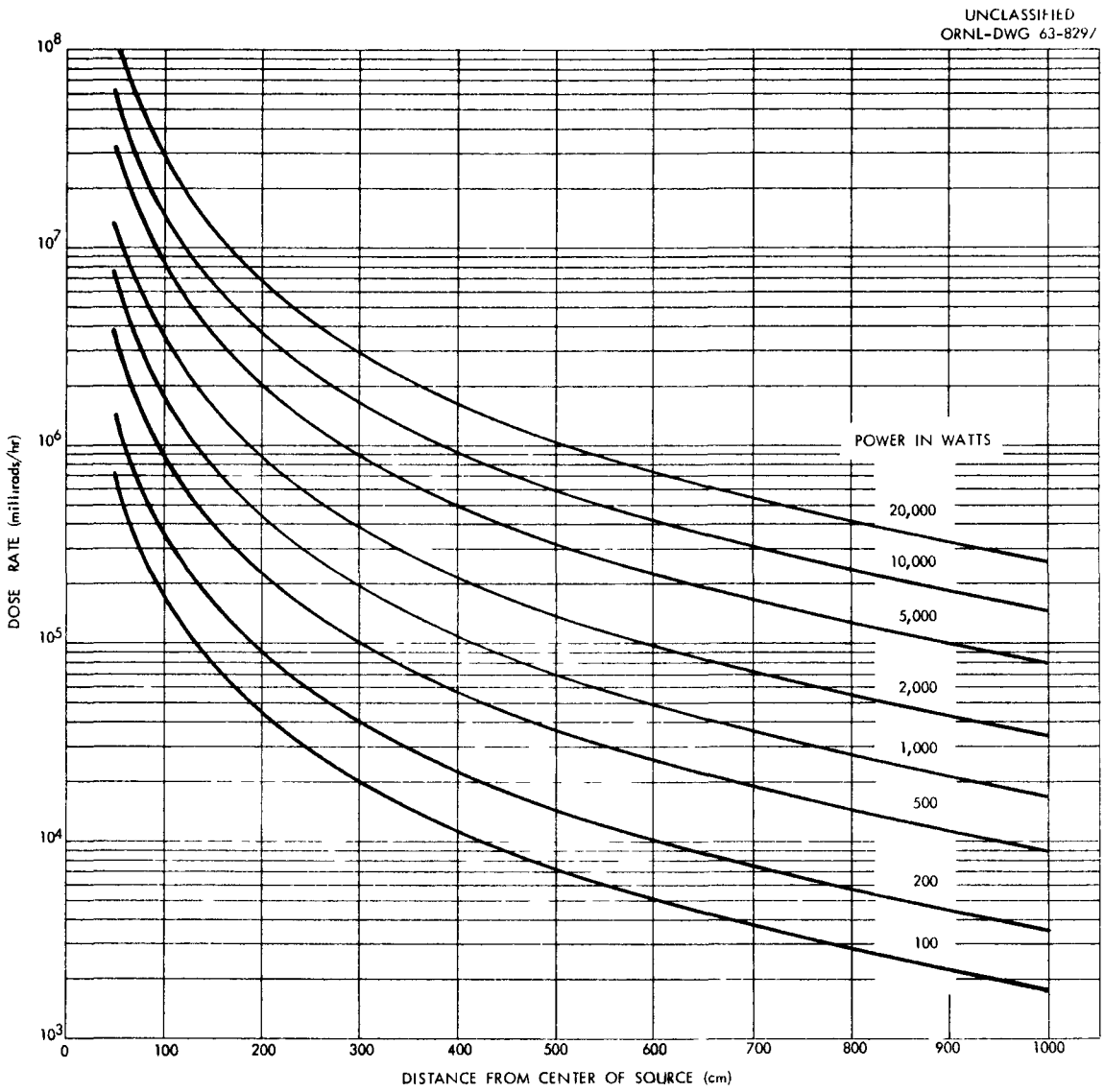


Fig. 14. Bremsstrahlung Dose Rates from Unshielded Isotopic Power Sources of Strontium-90 (Strontium Oxide) as a Function of Distance from Center of Source.

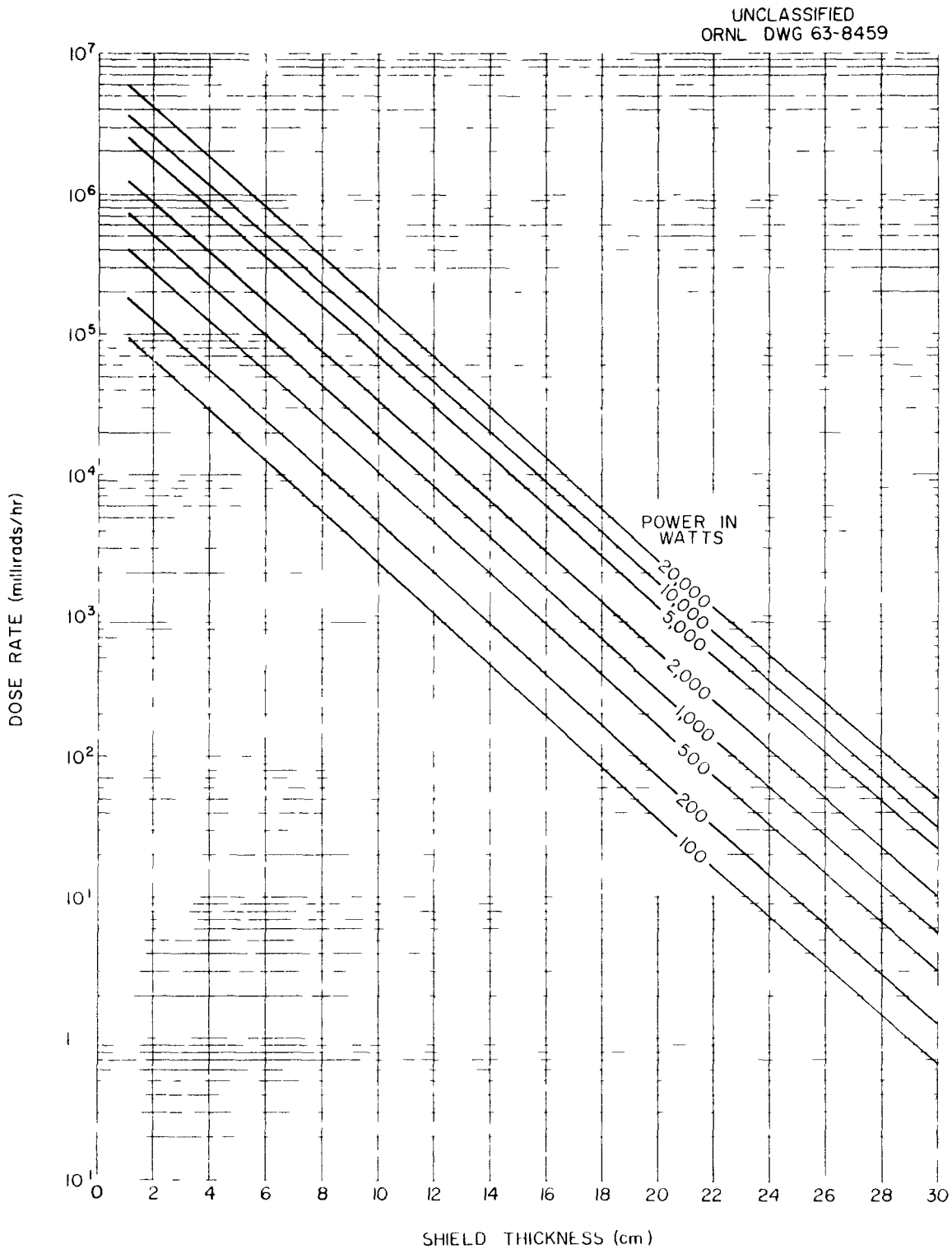


Fig. 15. Bremsstrahlung Dose Rates from Iron-Shielded Isotopic Power Sources of Strontium-90 (Strontium Oxide). Center of source to dose point separation distance = 100 cm.

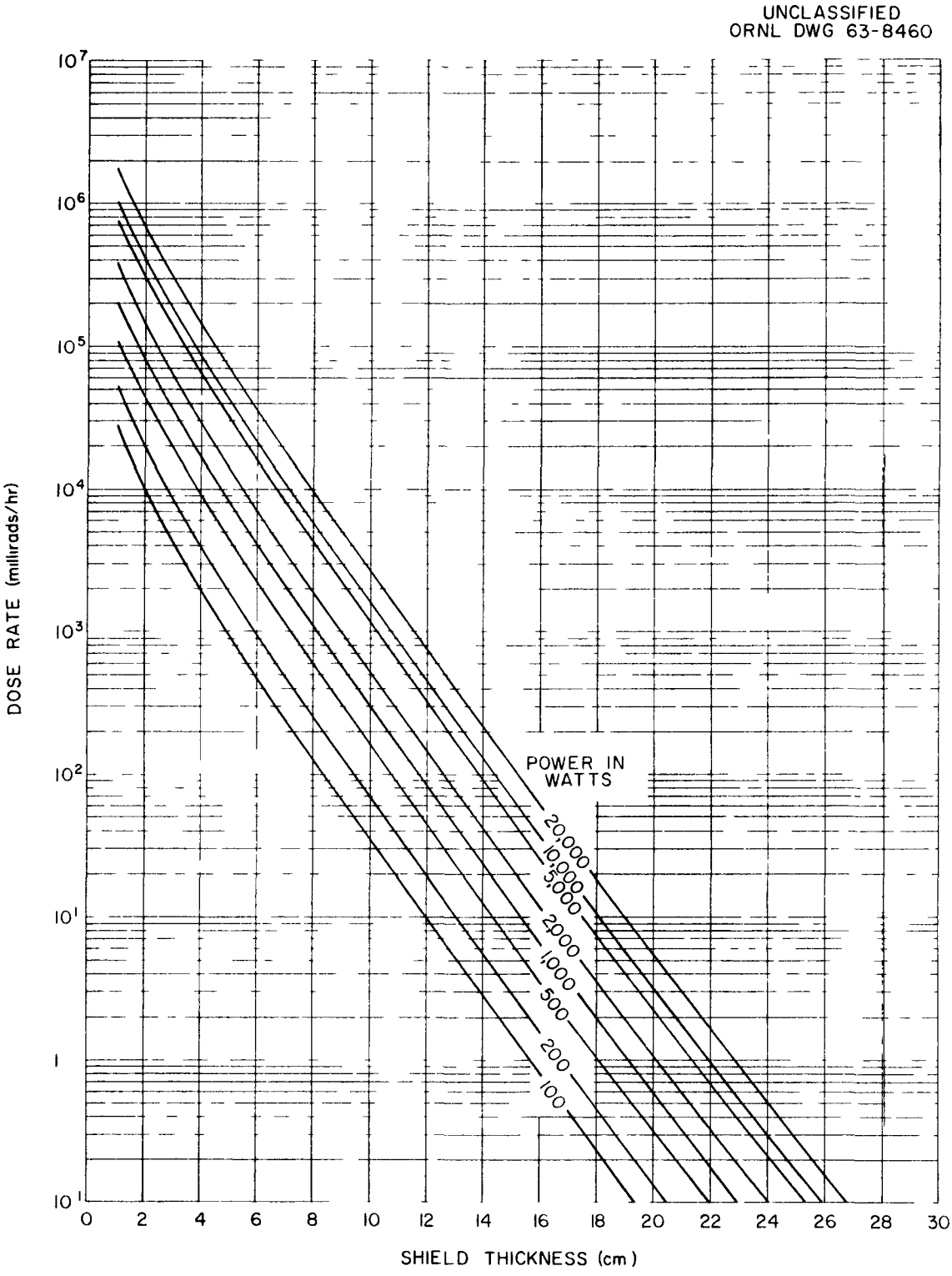


Fig. 16. Bremsstrahlung Dose Rates from Lead-Shielded Isotopic Power Sources of Strontium-90 (Strontium Oxide). Center of source to dose point separation distance = 100 cm.

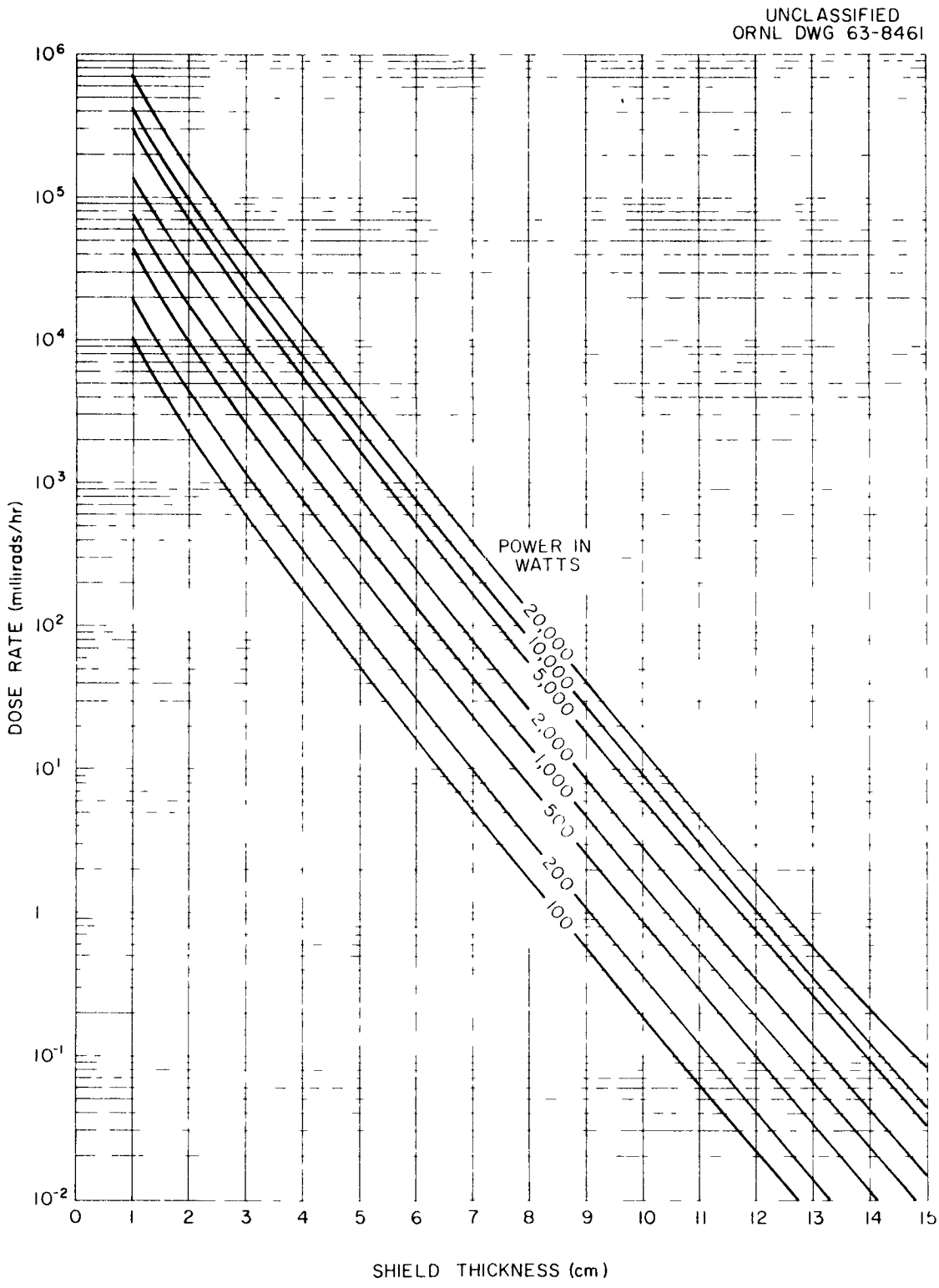


Fig. 17. Bremsstrahlung Dose Rates from Uranium-Shielded Isotopic Power Source of Strontium-90 (Strontium Oxide). Center of source to dose point separation distance = 100 cm.

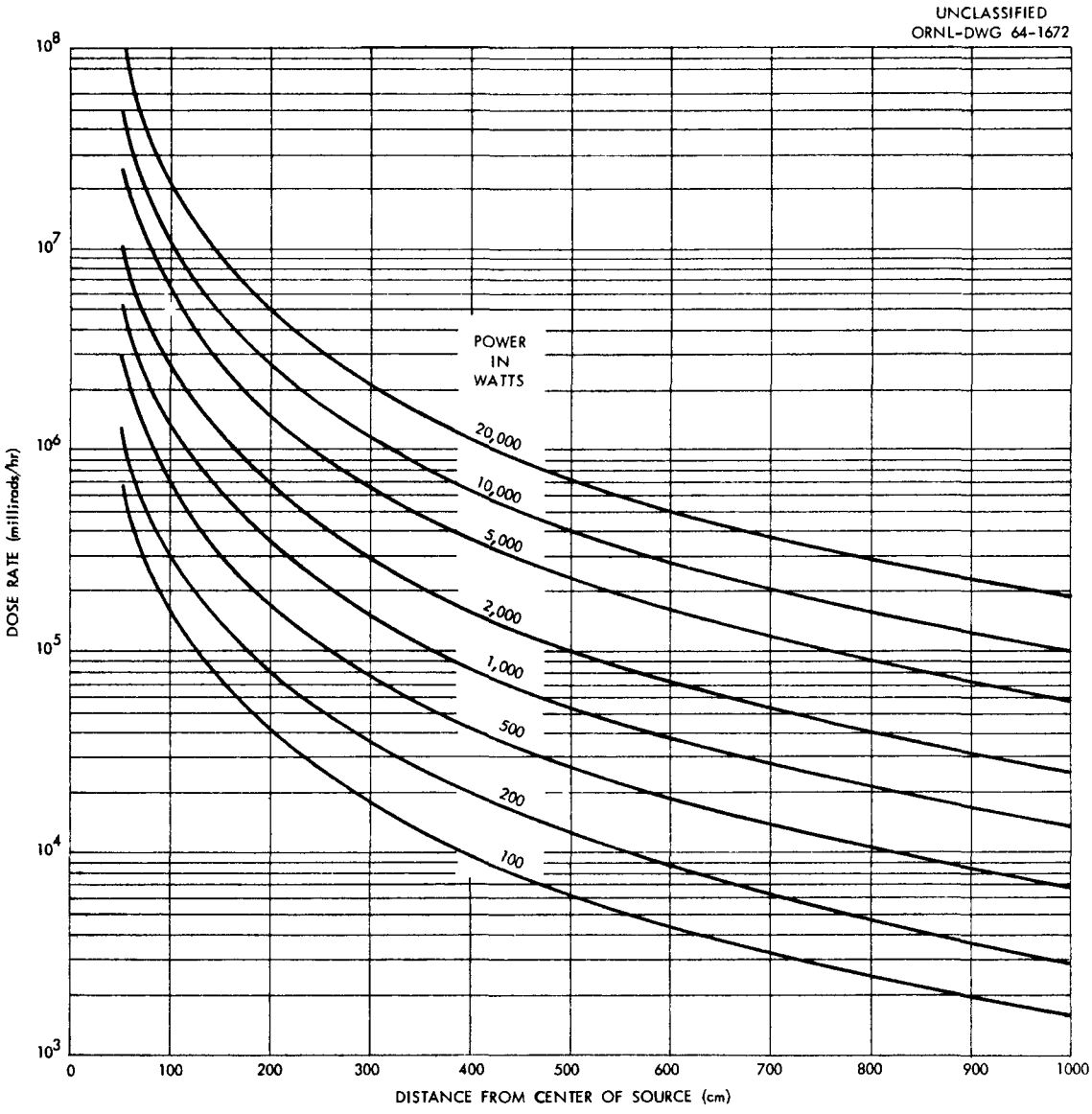


Fig. 18. Bremsstrahlung Dose Rates from Unshielded Isotopic Power Sources of Strontium-90 (Strontium Titanate) as a Function of Distance from Center of Source.

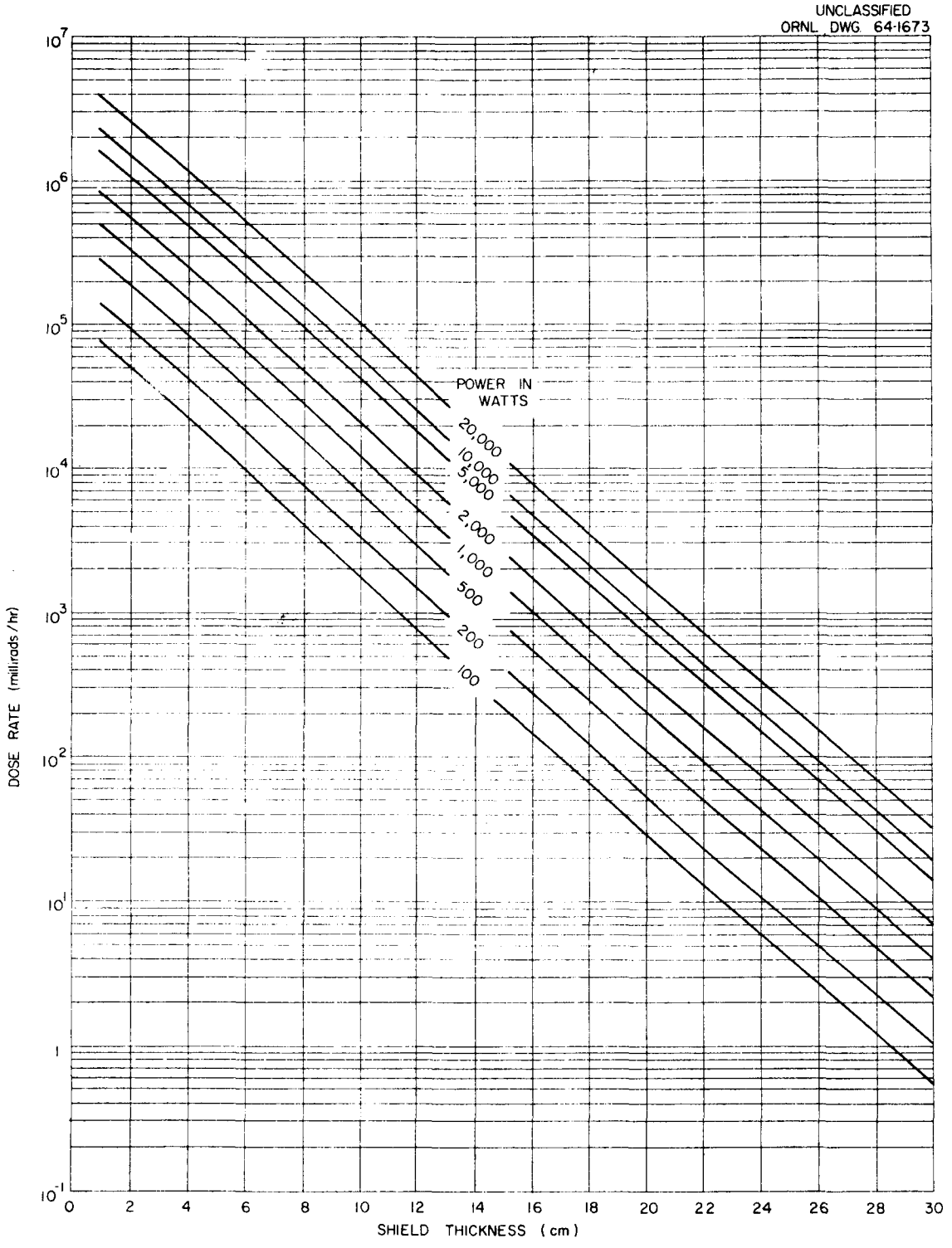


Fig. 19. Bremsstrahlung Dose Rates from Iron-Shielded Isotopic Power Sources of Strontium-90 (Strontium Titanate). Center of source to dose point separation distance = 100 cm.

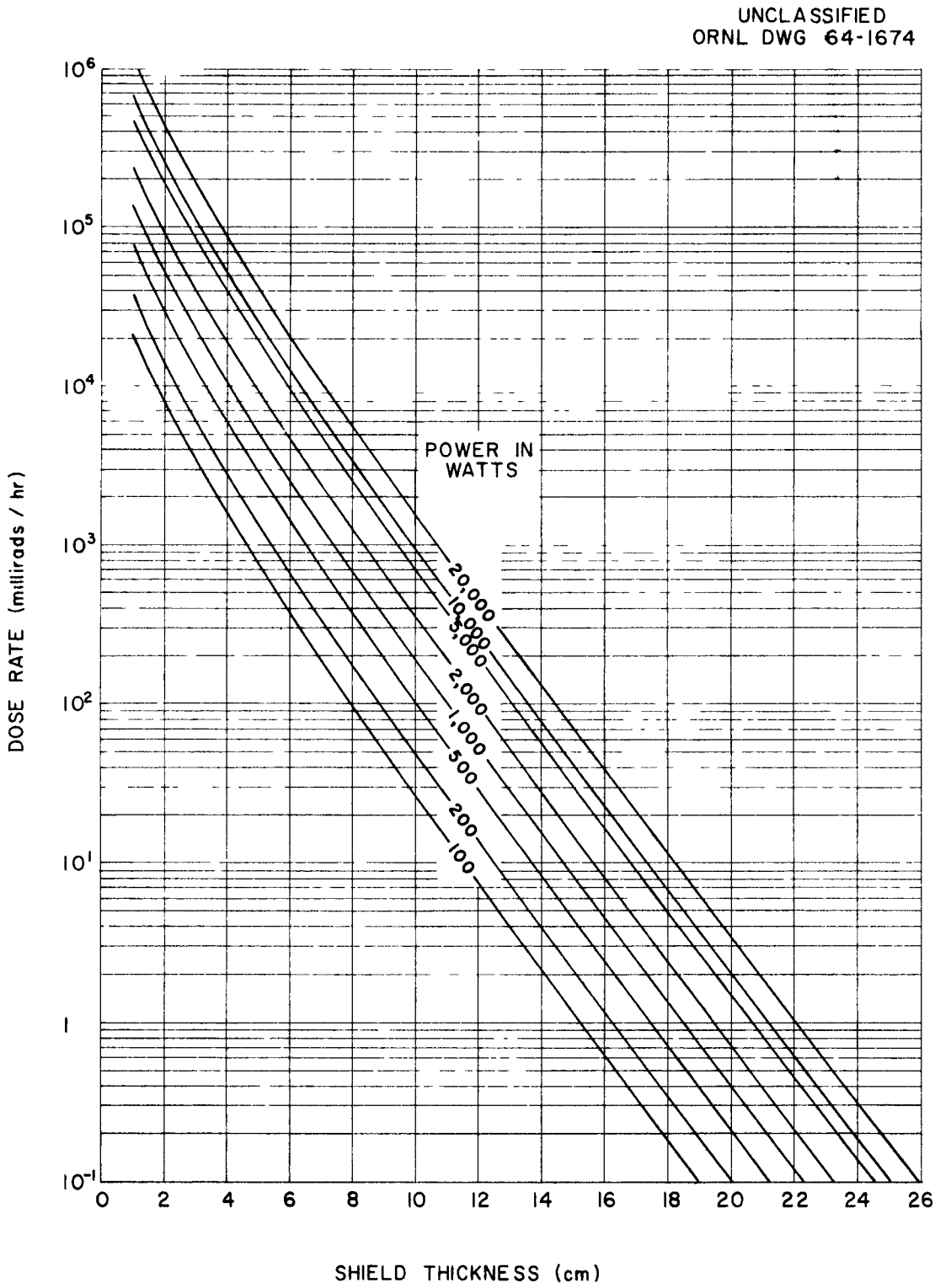


Fig. 20. Bremsstrahlung Dose Rates from Lead-Shielded Isotopic Power Sources of Strontium-90 (Strontium Titanate). Center of source to dose point separation distance = 100 cm.

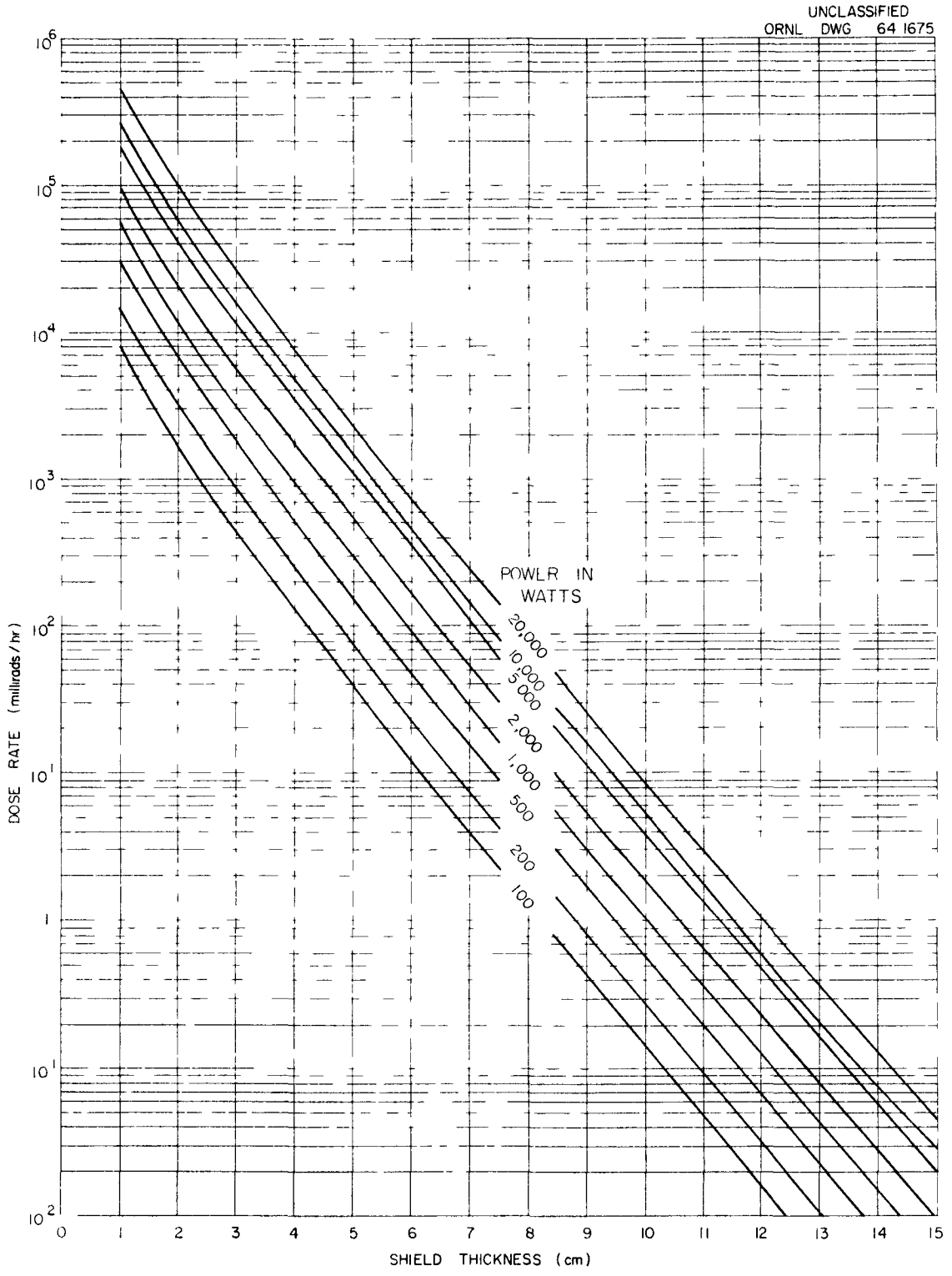


Fig. 21. Bremsstrahlung Dose Rates from Uranium-Shielded Isotopic Power Sources of Strontium-90 (Strontium Titanate). Center of source to dose point separation distance = 100 cm.

Zirconium-Niobium^{95}

The Zr-Nb^{95} couple can only be considered for those applications involving radiation devices or where a high specific power short-half-life isotope is best suited. The half-lives are very short (Zr^{95}, 65 days; Nb^{95}, 35 days) when compared with the other isotopes suggested as power sources. Zirconium-95 has a very high specific activity and power when pure. These values are 20,000 curies/g and 100 w/g, respectively. In addition, Nb^{95} in transient equilibrium with Zr^{95} following an irradiation time of 120 days at 3×10^{14} neutrons cm^{-2} sec^{-1} thermal flux and a decay period of 30 days would contribute an additional 30,000 curies and 143 w per gram of the associated Zr^{95}.

However, these figures do not present the entire story; pure Zr^{95} can not be obtained without isotope separation. Like ruthenium, the fission process produces zirconium isotopes with a very high yield.[21,23,24] With the exception of Zr^{95}, these isotopes are either stable or have a very long half-life. With the irradiation and decay conditions given above, Zr^{95} would consitute about 8% of the total zirconium isotopes in the fission produced mixture. Thus the specific activity and power of a fission produced mixture of zirconium isotopes (plus Nb^{95}) would become 4000 curies/g and 19.5 w/g, respectively. A power density of 75 w/cm^3 could be achieved with a zirconium oxide power source.

Zirconium-95 would also be similar to Co^{60} in power recovery requirements since 90% of the total heat produced is produced as gamma radiation.

The biological hazard for Zr^{95} in relatively low, about the same as that for Co^{60} or Ru^{106}. Both Zr^{95} and Nb^{95} emit a large number of relatively high-energy gamma rays. Shielding requirements are therefore substantial.

Isotopic heat source data for Zr-Nb^{95} sources appear in Table 9, and the nuclear data appear in Table 3. The results of the radiation intensity calculations for shielded and unshielded sources are given in Figs. 22 through 25.

Table 9. Physical Properties of Zirconium-Niobium^{95} as an Isotopic Power Source

General Properties

Half-life	65-35 d
Specific activity of pure isotope, curies/g	(20,000(Zr^{95}) + 30,000(Nb^{95}))/g Zr^{95}
Specific power of pure isotope, w/g	243.6/g Zr^{95}
Compound	ZrO_2 - Nb_2O_5
Weight of active isotope in compound, g/cc	0.31
Effective density of compound, g/cc	5.2
Power, w/cc of compound	75
Activity, curies/cc of compound	15,500

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
1.671×10^{12}	0.76[a]
1.306×10^{12}	0.72[a]
4.557×10^{12}	0.76[c,e]

[a] From Zr^{95} activity.

[c] From Nb^{95} activity.

[e] Predominant radiation from unshielded 5000 watt source.

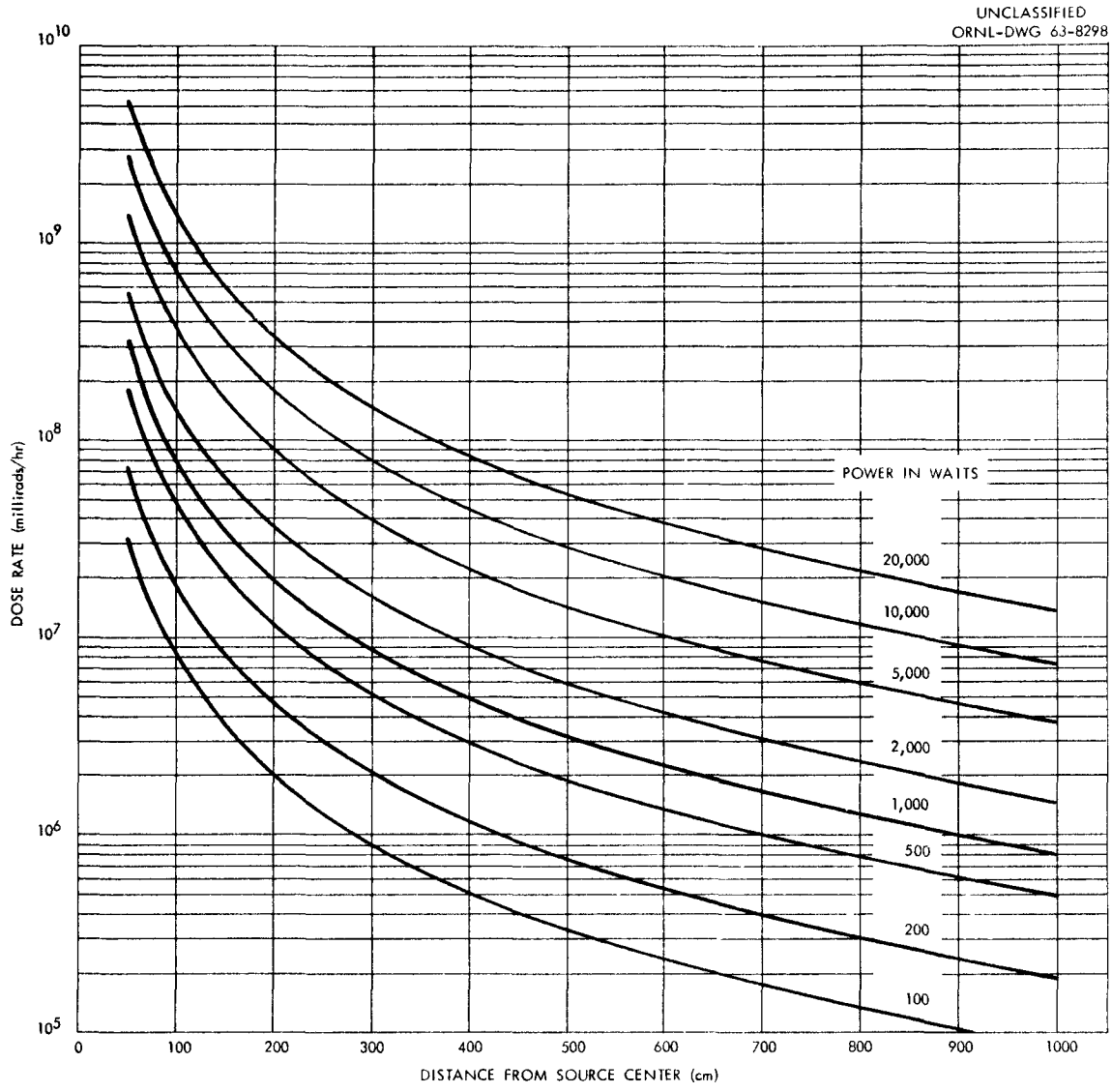


Fig. 22. Gamma Dose Rates from Unshielded Isotopic Power Sources of Zirconium-Niobium-95 as a Function of Distance from Center of Source.

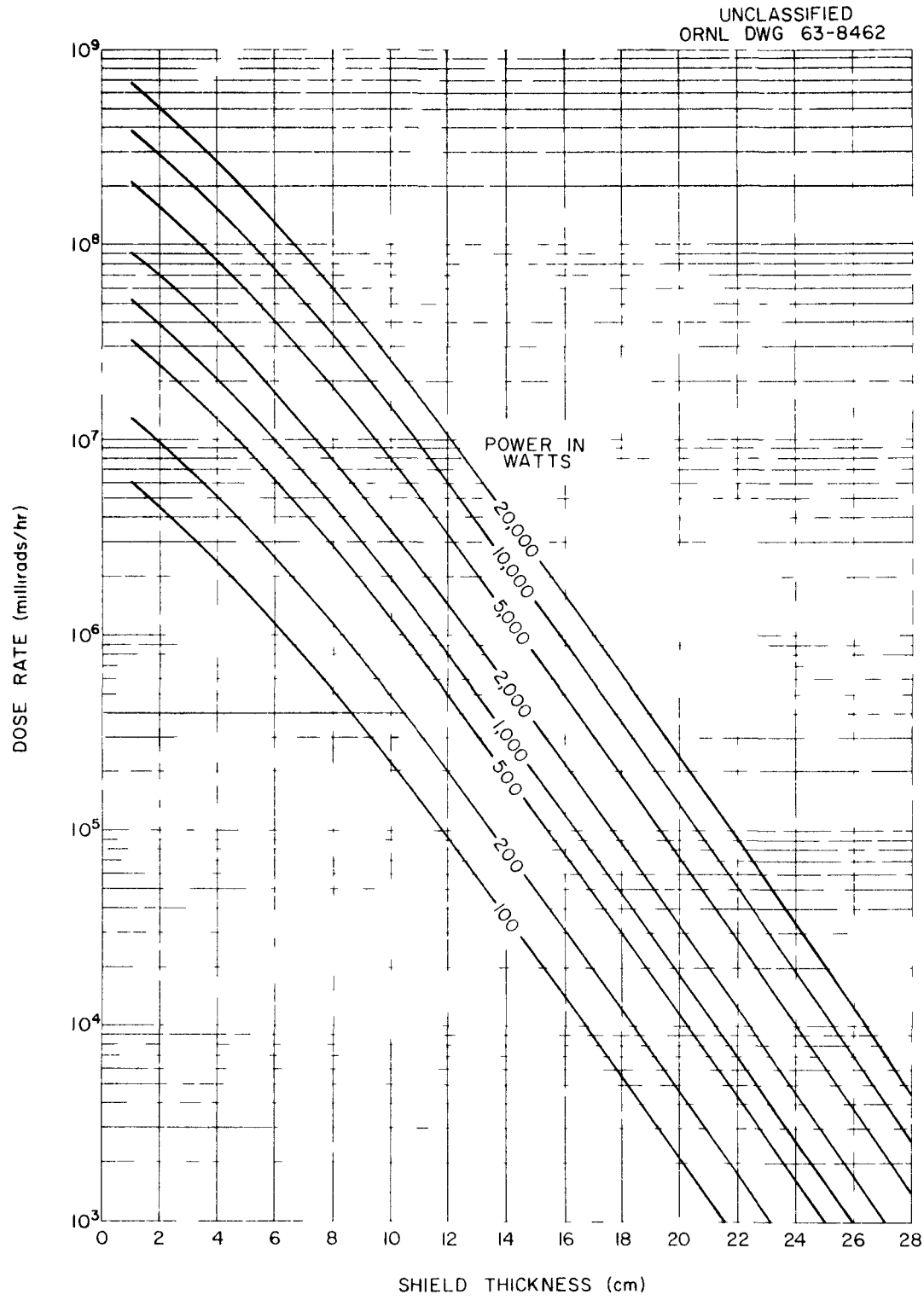


Fig. 23. Gamma Dose Rates from Iron-Shielded Isotopic Power Sources of Zirconium-Niobium-95. Center of source to dose point separation distance = 100 cm.

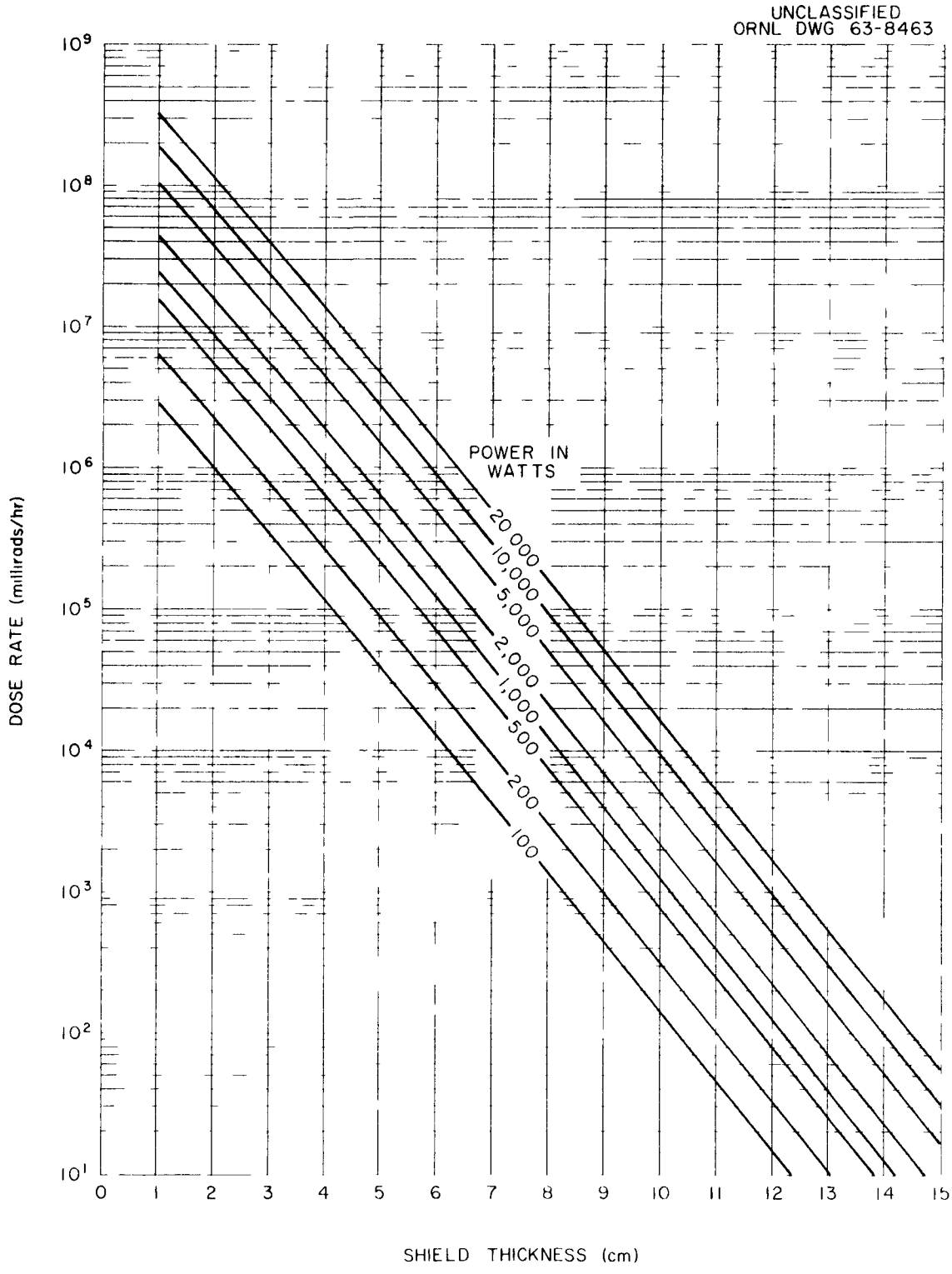


Fig. 24. Gamma Dose Rates from Lead-Shielded Isotopic Power Sources of Zirconium-Niobium-95. Center of source to dose point separation distance = 100 cm.

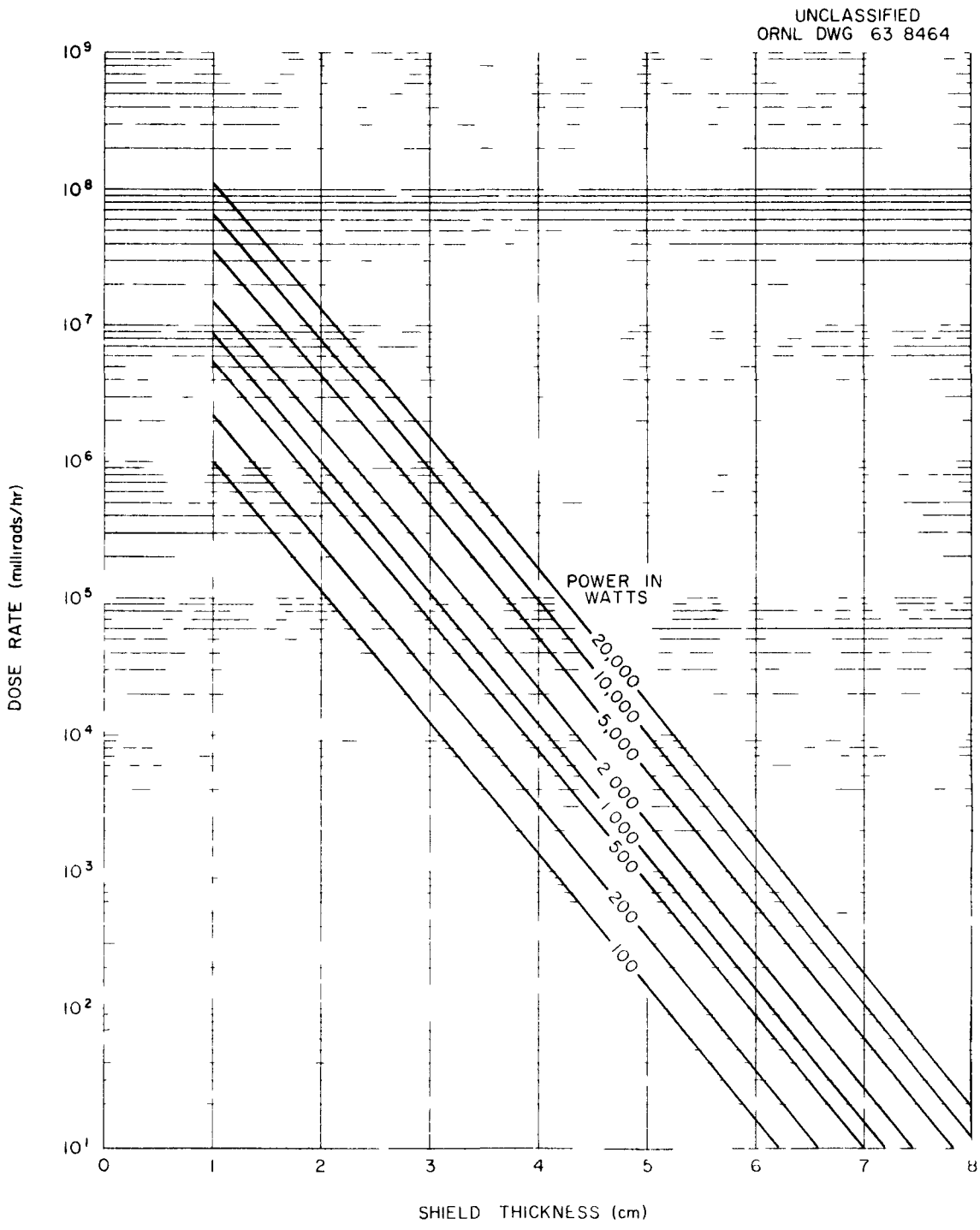


Fig. 25. Gamma Dose Rates from Uranium-Shielded Isotopic Power Sources of Zirconium-Niobium-95. Center of source to dose point separation distance = 100 cm.

Ruthenium-106

Next to xenon,[a] ruthenium is the fission product produced in highest yield.[21,23,24] It comprises a mixture of at least seven isotopes of which Ru^{106} has a half-life of 1 yr. Three of the other isotopes are stable. The specific power from Ru^{106} is fairly high, 31.8 w/g (Table 10). However, yield of Ru^{106} from U^{235} fission is the smallest of the ruthenium isotopes (0.38%), such that the isotopic purity of Ru^{106} in the mixture with its stable relatives in only about 3.3%. However, Ru^{103} has a 40-day half-life, and a decay period of nearly 190 days must be used to decay this isotope down to a power level of 20% of the Ru^{106} isotope (when in equilibrium with its rhodium daughter). This decay period would in turn reduce the Ru^{106} activity to about 70% of its reactor discharged value.

The yield of Ru^{106} in plutonium fission[21,23,24] is, however, substantially greater (about 4.5%). Thus, the purity would also appear to be much greater under conditions of plutonium fission - possibly as high as 20%. In power reactors operating on low enrichment uranium, Ru^{106} would result from both plutonium and U^{235} fission; the isotopic purity of the Ru^{106} would then lie between 3.3% and 20%, possibly near 12%. The specific power from ruthenium containing such concentrations of the 106 isotope would still be appreciable, possibly 3.7 w/g. The power density for ruthenium metal would then be nearly 47 w/cc. Calculations were based on a ruthenium mixture containing 2.4 wt % Ru^{106}.

The biological hazard for Ru^{106} is also relatively low, about the same as that for Co^{60}. Although Ru^{106} is a beta emitter, its daughter, Rh^{106} emits a number of high-energy gammas and bremsstrahlung x rays. Shielding requirements are therefore substantial. The chemistry of ruthenium is complex because of the various valence states. Since it is a noble metal, once obtained, it should be easily maintained in a stable form as the metal. Under today's conditions of availability, it perhaps should be receiving as much attention as Ce^{144}.

Isotopic heat source data for Ru^{106} sources appear in Table 10, and the nuclear data appear in Table 3. The results of the radiation intensity calculations for shielded and unshielded sources are given in Figs. 26 through 29.

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[a]Xenon isotopes can not be considered as radiation or power sources since all xenon isotopes have very short half-lives or are stable.

Table 10. Physical Properties of Ruthenium-106 as an Isotopic Power Source

General Properties	
Half-life	365 d
Specific activity of pure isotope, curies/g	3300
Specific power of pure isotope, w/g	31.85
Compound	metal
Weight of active isotope in metal, g/cc	0.296
Effective density of metal, g/cc	12.3
Power, w/cc of metal	9.43
Activity, curies/cc of metal	977

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
1.378×10^{12}	0.35[b]
1.263×10^{11}	0.95[b]
2.539×10^{10}	1.55[b]
4.518×10^{9}	2.15[b]
5.225×10^{8}	2.75[b]
8.255×10^{6}	3.30[b]
1.127×10^{10}	2.42
1.878×10^{10}	1.55
7.51×10^{10}	1.04
1.239×10^{12}	0.55[e]

[b] Bremsstrahlung x ray group.

[e] Predominant radiation from unshielded 5000 watt source.

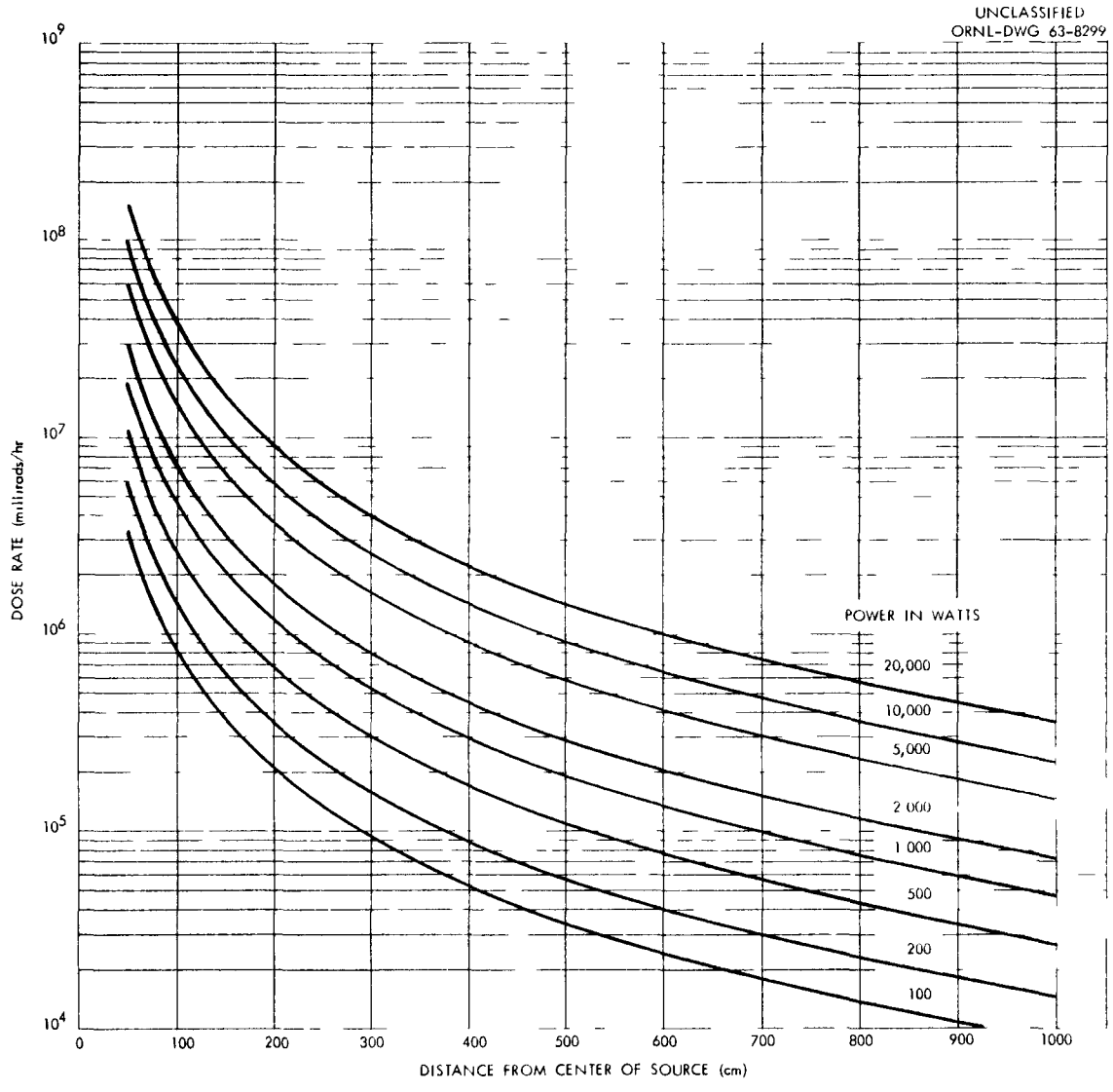


Fig. 26. Gamma Plus Bremsstrahlung Dose Rates from Unshielded Isotopic Power Sources of Ruthenium-106 as a Function of Distance from Center of Source.

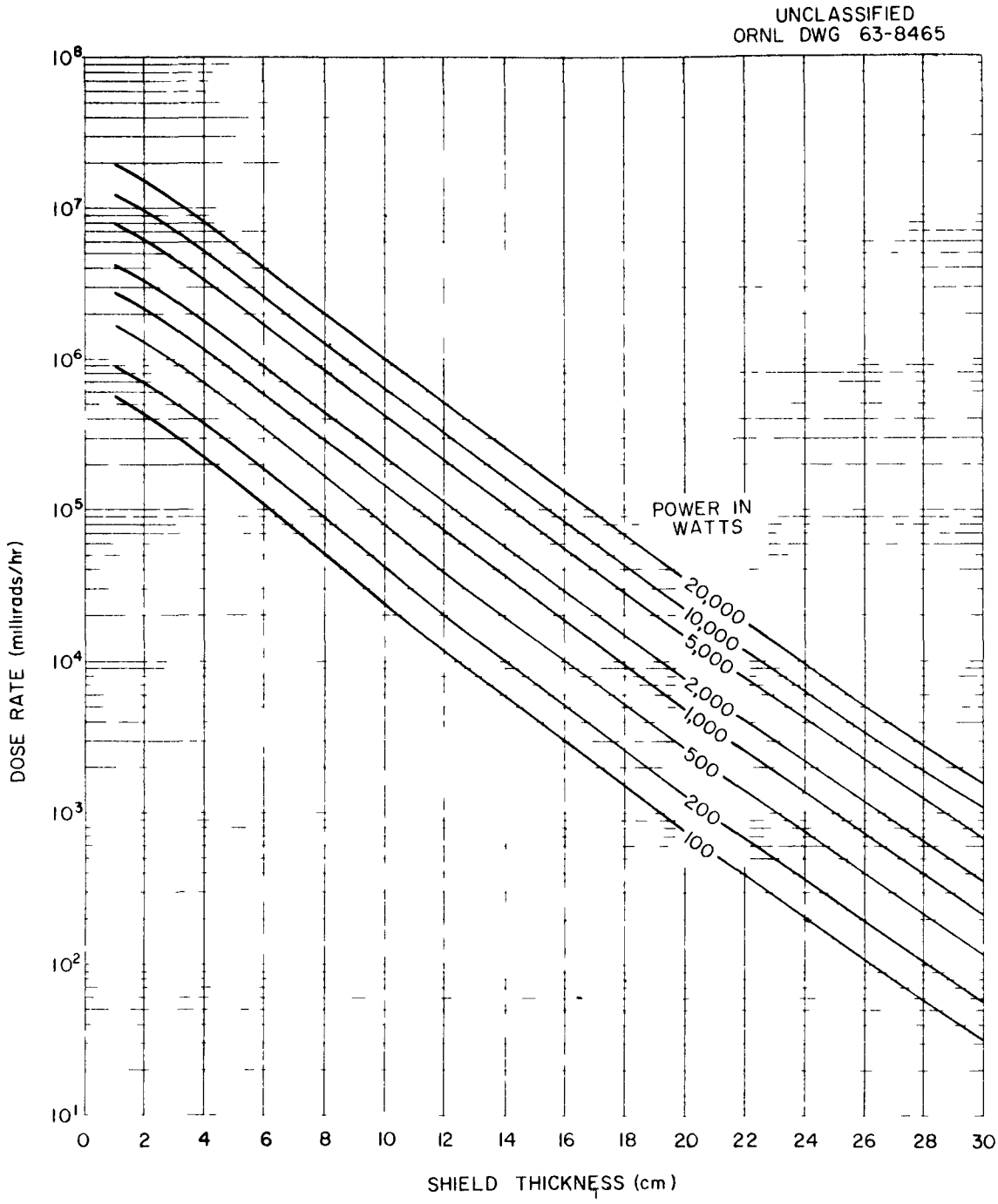


Fig. 27. Gamma Plus Bremsstrahlung Dose Rates from Iron-Shielded Isotopic Power Sources of Ruthenium-106. Center of source to dose point separation distance = 100 cm.

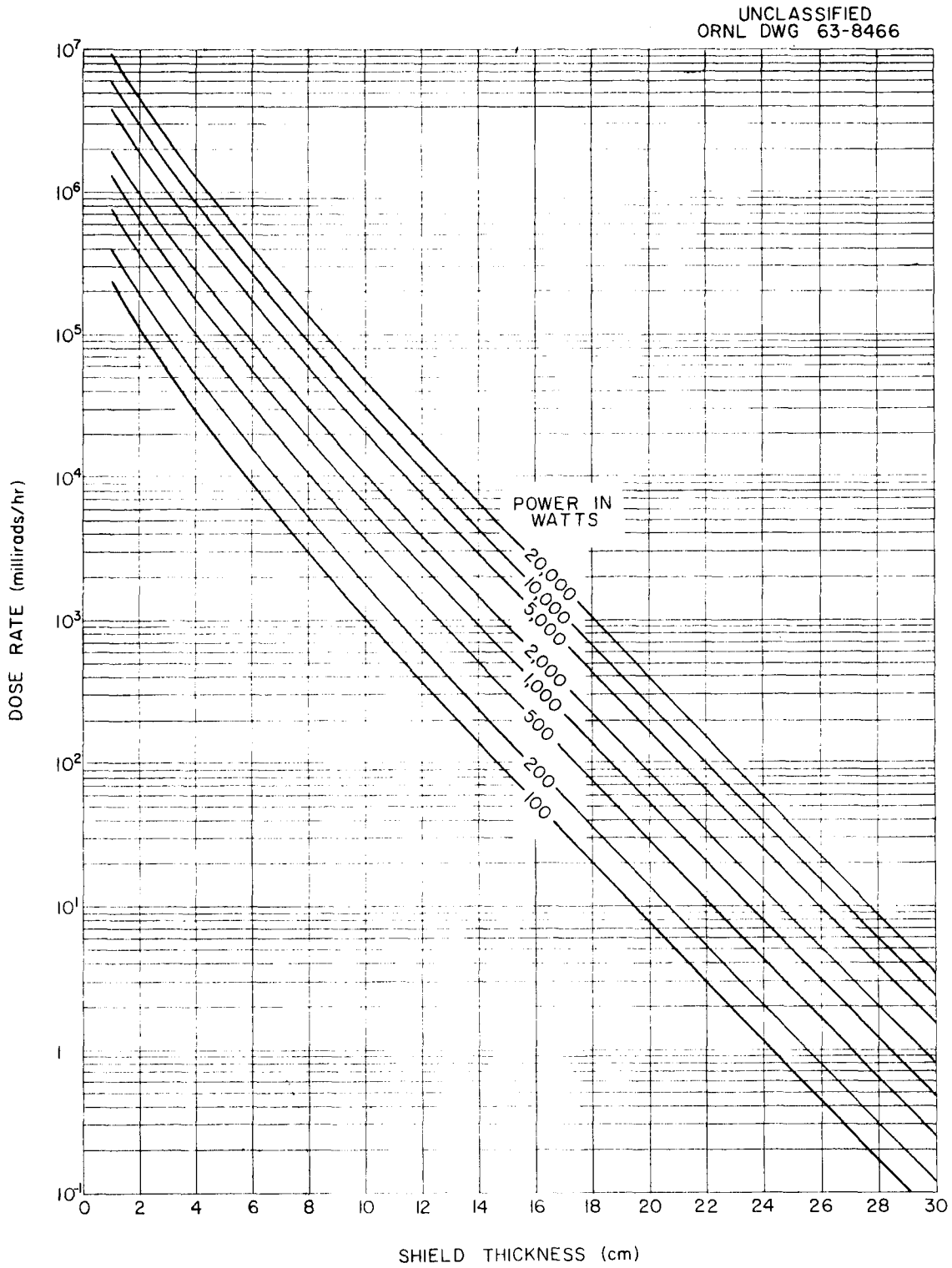


Fig. 28. Gamma Plus Bremsstrahlung Dose Rates from Lead-Shielded Isotopic Power Sources of Ruthenium-106. Center of source to dose point separation distance = 100 cm.

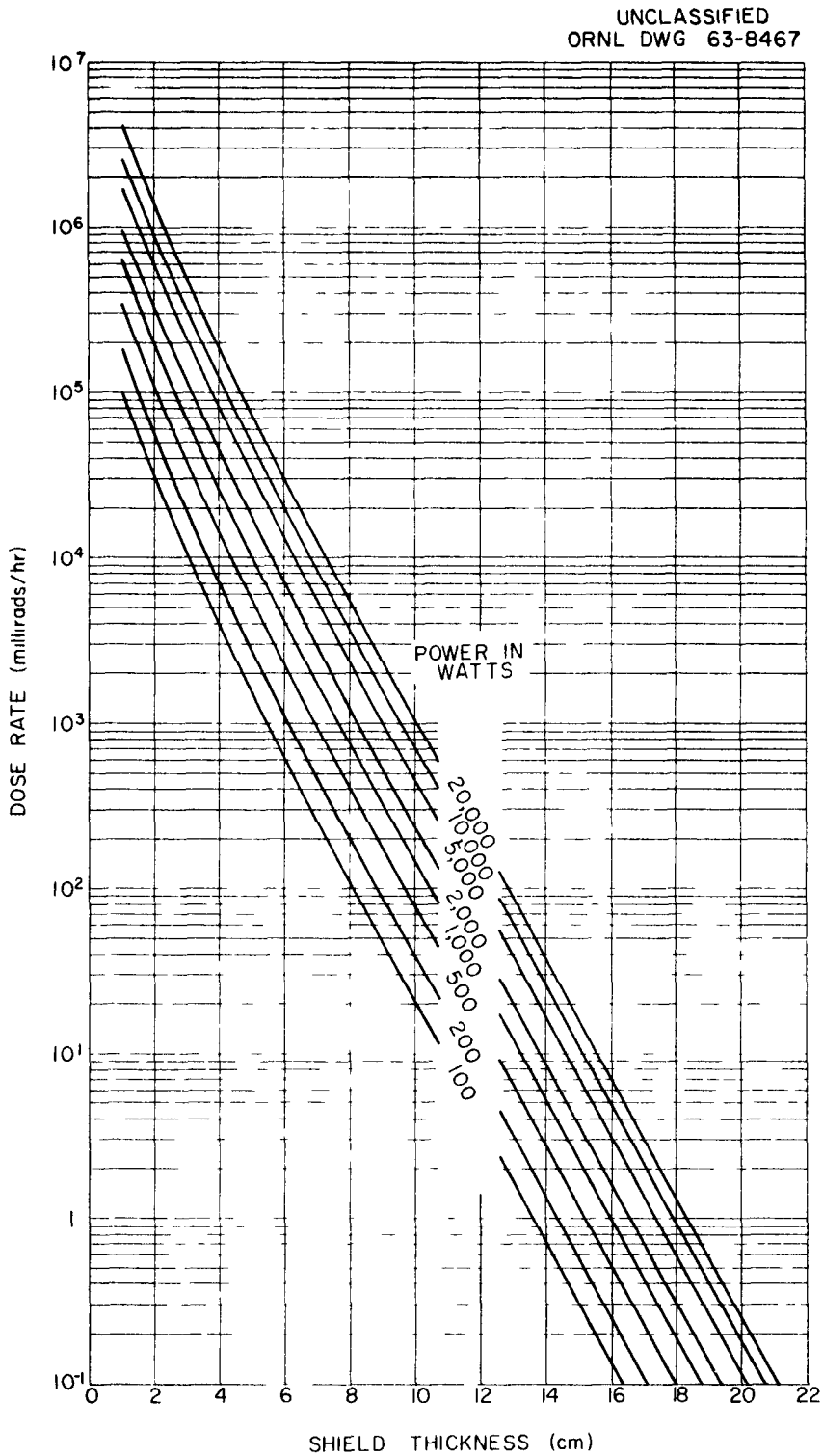


Fig. 29. Gamma Plus Bremsstrahlung Dose Rates from Uranium-Shielded Isotopic Power Sources of Ruthenium-106. Center of source to dose point separation distance = 100 cm.

Cesium-137

Cesium-137 ($T_{1/2}$ = 26.6 yr) comprises about a third of the cesium isotopes discharged from nuclear reactors. The other isotopes are stable Cs^{133}, 2.3-year half-life Cs^{134}, and slightly active 2-million-year half-life Cs^{135}. Thus, for practical purposes, Cs^{135} can also be considered to be stable. The specific power for Cs^{137} is 0.475 w/g. The power density of cesium "glass" is about 0.21 w/cc. Although Cs^{137} is a beta emitter, its 2.6-min daughter, Ba^{137}, is the source of fairly energetic gamma activity. For this reason Cs^{137} can also be considered for radiation sources. Cesium-137 requires slightly less shielding than Sr^{90} because of the higher energy of some of the bremsstrahlung from the Sr^{90} decay chain. In the value for specific power about one-fourth is from beta energy and three-fourths is from the gamma energy from Ba^{137} decay. It is concluded that the energy of the gamma activity is low enough (in contrast with Co^{60}) to ensure that most of it will be absorbed and will produce heat in the mass of the compound and encapsulating material. Biologically, Cs^{137} is far less hazardous than Sr^{90}. The fission yield is high, but the isotopic purity, low density of the compound form (borosilicate glass),[25] and principally, the low content of Cs^{137} in this compound form result in a heat source of rather low power density - about one-fifth that of strontium oxide.

Isotopic heat source data for Cs^{137} sources appear in Table 11, and the nuclear data appear in Table 3. The results of the radiation intensity calculations for shielded and unshielded sources are given in Figs. 30 through 33. The radiation intensities from a 1% impurity (beta-activity basis) of Cs^{134} in Cs^{137} sources are given in Figs. 34 through 37. The contribution from an actual known Cs^{134} contamination can then be found from "ratioing" with the plotted results.

Table 11. Physical Properties of Cesium-137 as an Isotopic Power Source

General Properties	
Half-life	26.6 y
Specific activity of pure isotope, curies/g	98.25
Specific power of pure isotope, w/g	0.475
Compound	glass
Weight of active isotope in glass, g/cc	0.442
Effective density of glass, g/cc	3.2
Power, w/cc of glass	0.21
Activity, curies/cc of glass	43.4

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
6.39×10^{12}	0.662

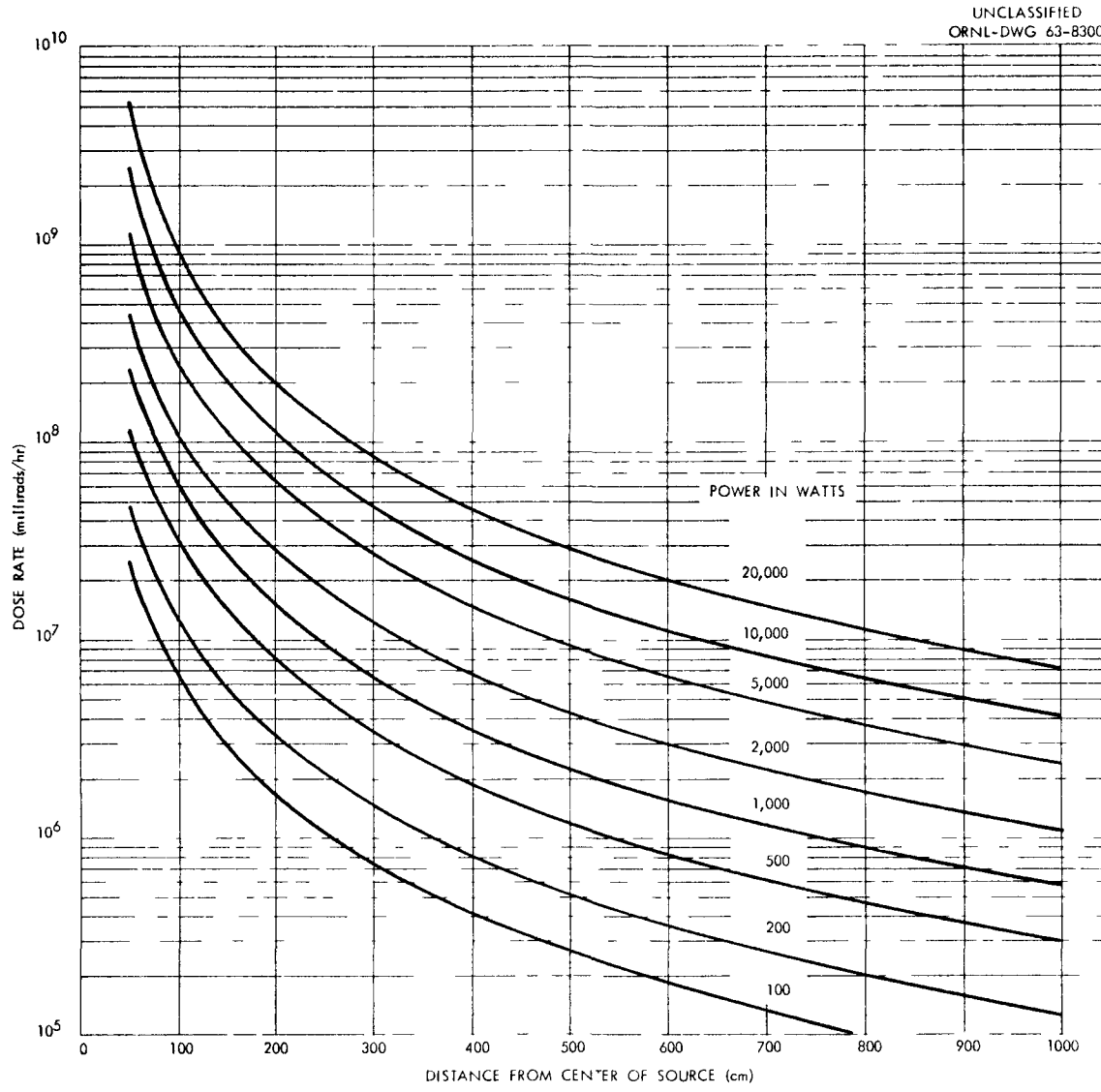


Fig. 30. Gamma Dose Rates from Unshielded Isotopic Power Sources of Cesium-137 as a Function of Distance from Center of Source.

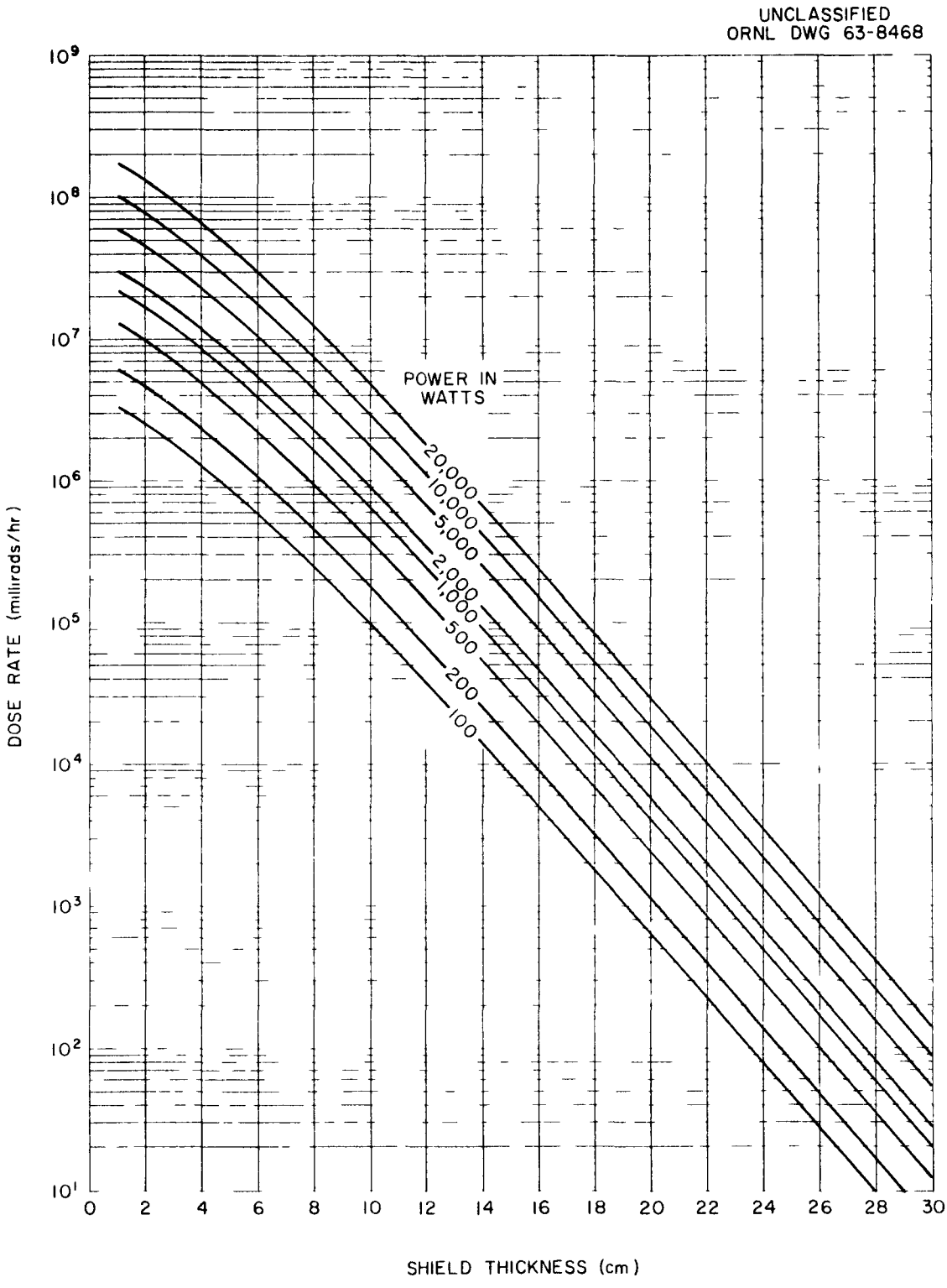


Fig. 31. Gamma Dose Rates from Iron-Shielded Isotopic Power Sources of Cesium-137. Center of source to dose point separation distance = 100 cm.

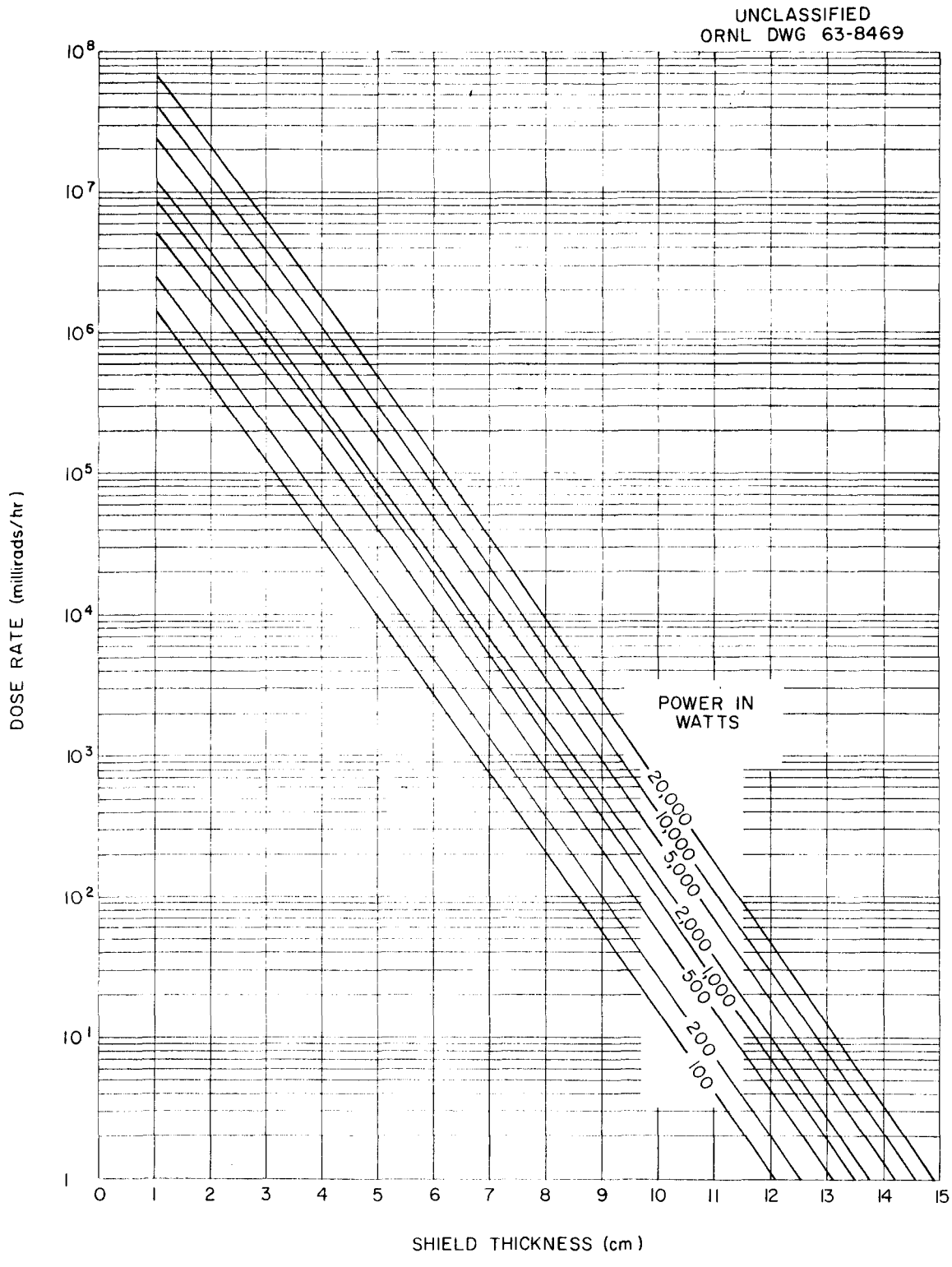


Fig. 32. Gamma Dose Rates from Lead-Shielded Isotopic Power Sources of Cesium-137. Center of source to dose point separation distance = 100 cm.

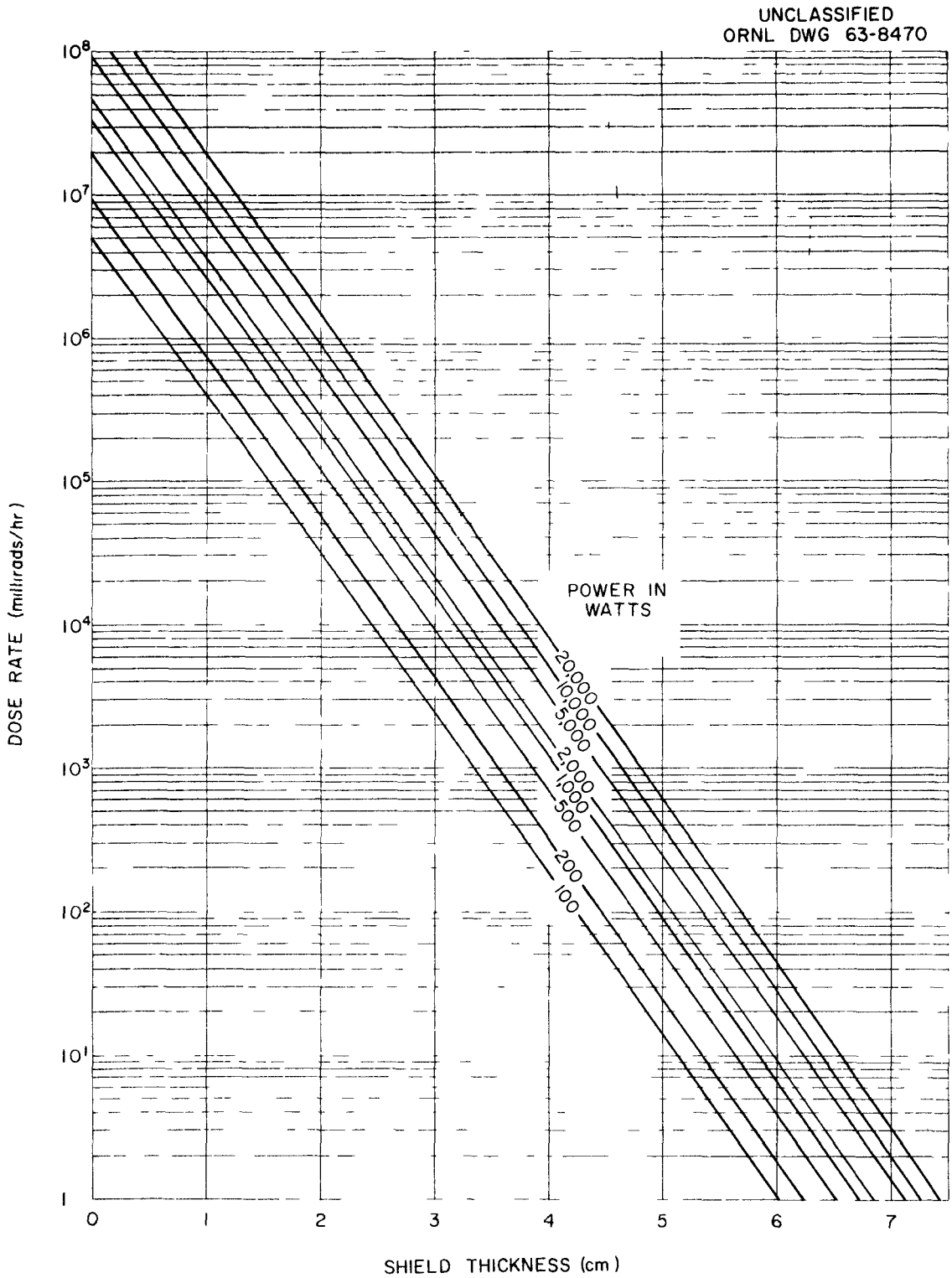


Fig. 33. Gamma Dose Rates from Uranium-Shielded Isotopic Power Sources of Cesium-137. Center of source to dose point separation distance = 100 cm.

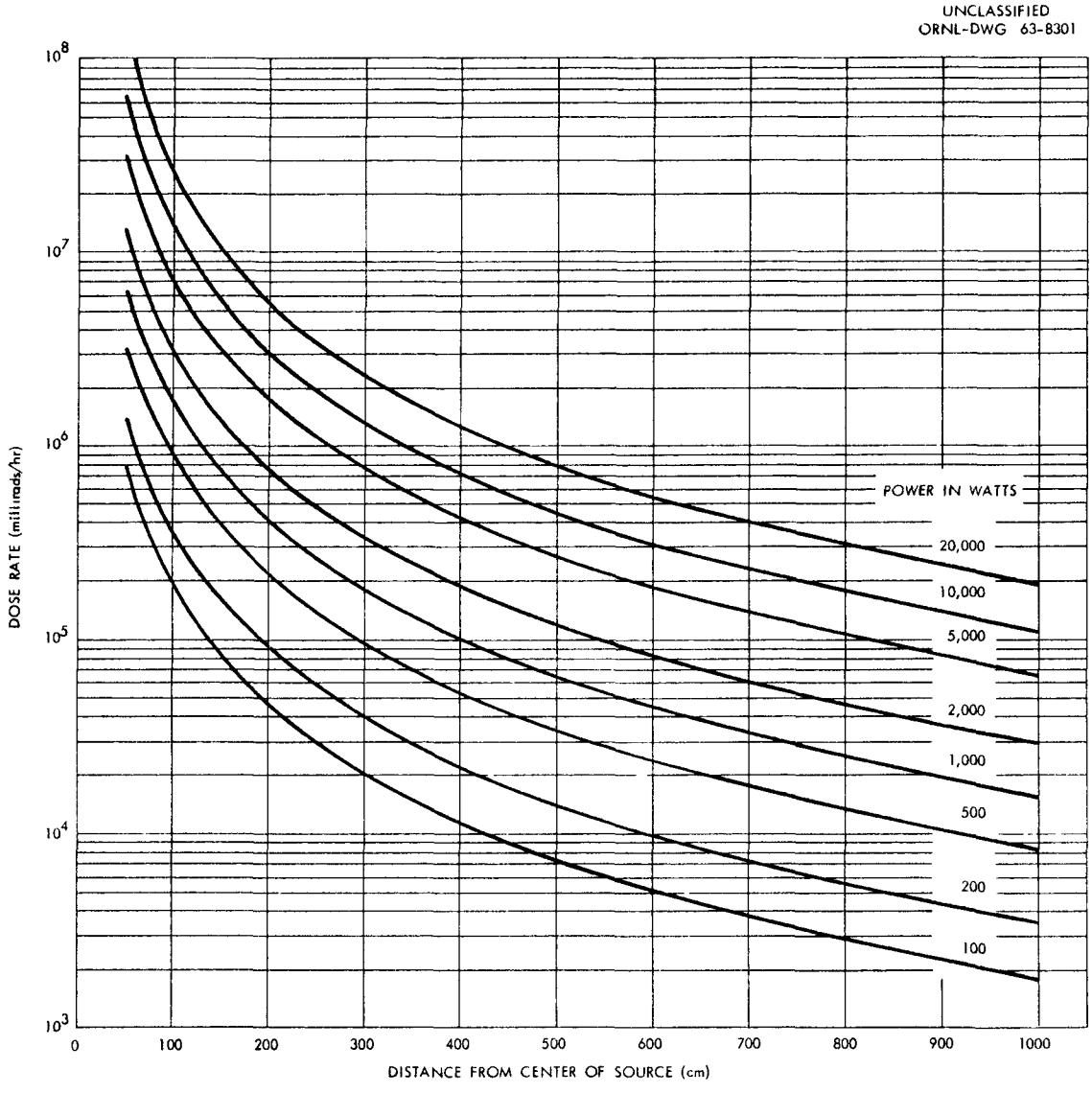


Fig. 34. Gamma Dose Rates Contributed by a 1% Impurity of Cesium-134 in Cesium-137 Isotopic Power Sources as a Function of Distance from Center of Source. Cesium-134 is assumed to contribute 1% of the beta activity due to Cs^{137}; refer to Fig. 30.

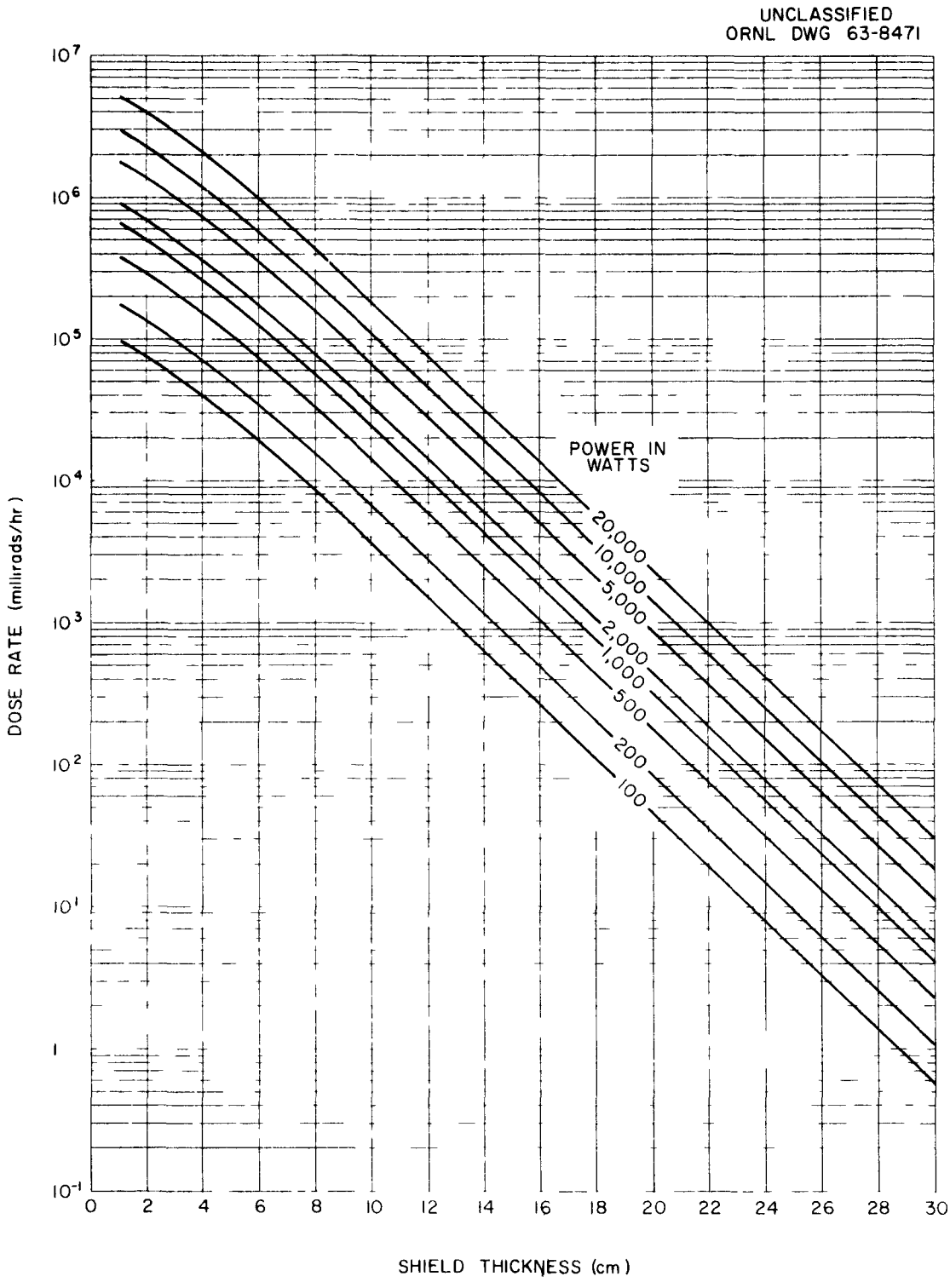


Fig. 35. Gamma Dose Rates Contributed by a 1% Impurity of Cesium-134 in Iron-Shielded Isotopic Power Sources of Cesium-137. Center of source to dose point separation distance = 100 cm. Cesium-134 is assumed to contribute 1% of the beta activity due to Cs^{137}; refer to Fig. 31.

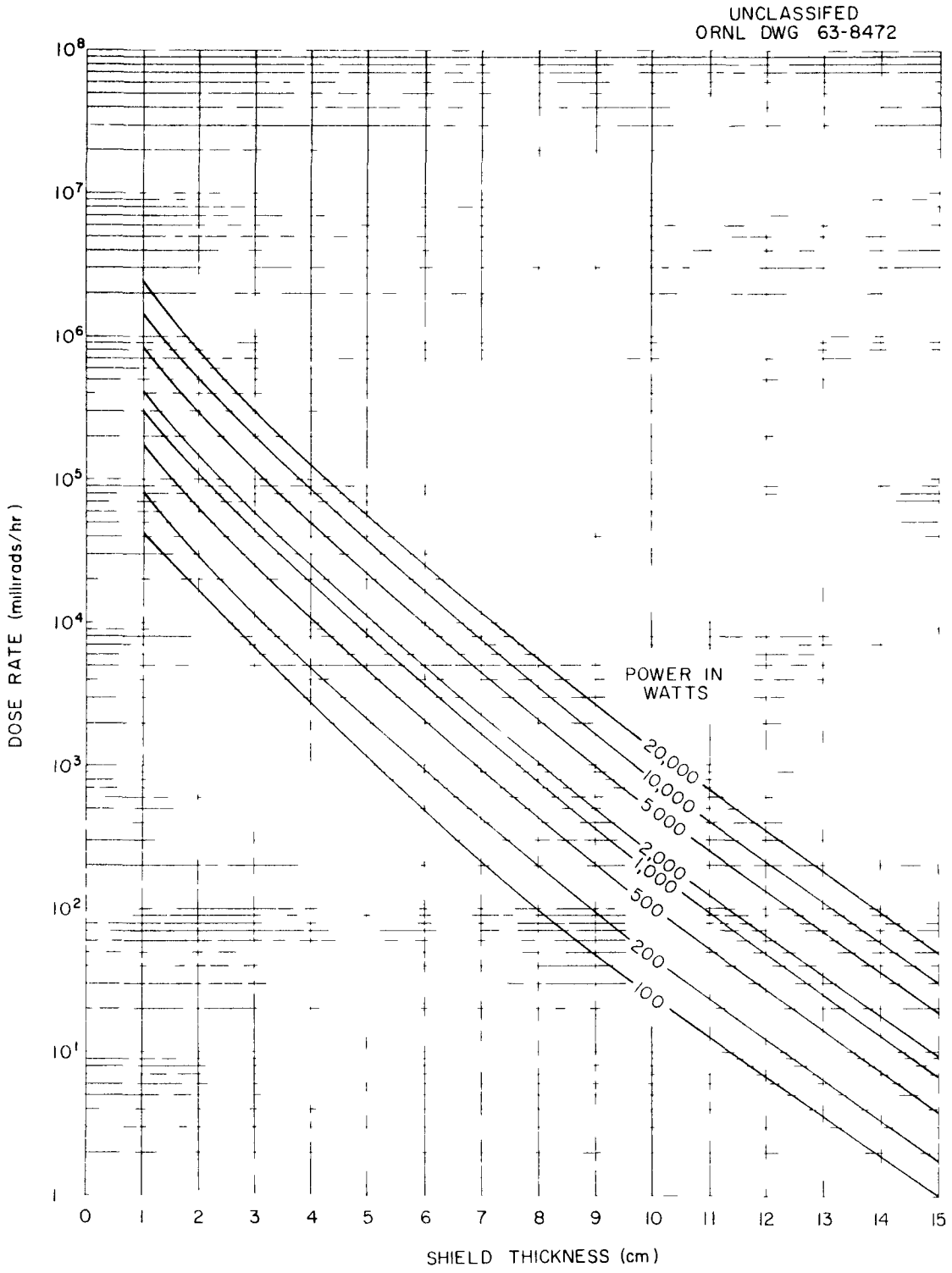


Fig. 36. Gamma Dose Rates Contributed by a 1% Impurity of Cesium-134 in Lead-Shielded Isotopic Power Sources of Cesium-137. Center of source to dose point separation distance = 100 cm. Cesium-134 is assumed to contribute 1% of the beta activity due to Cs^{137}; refer to Fig. 32.

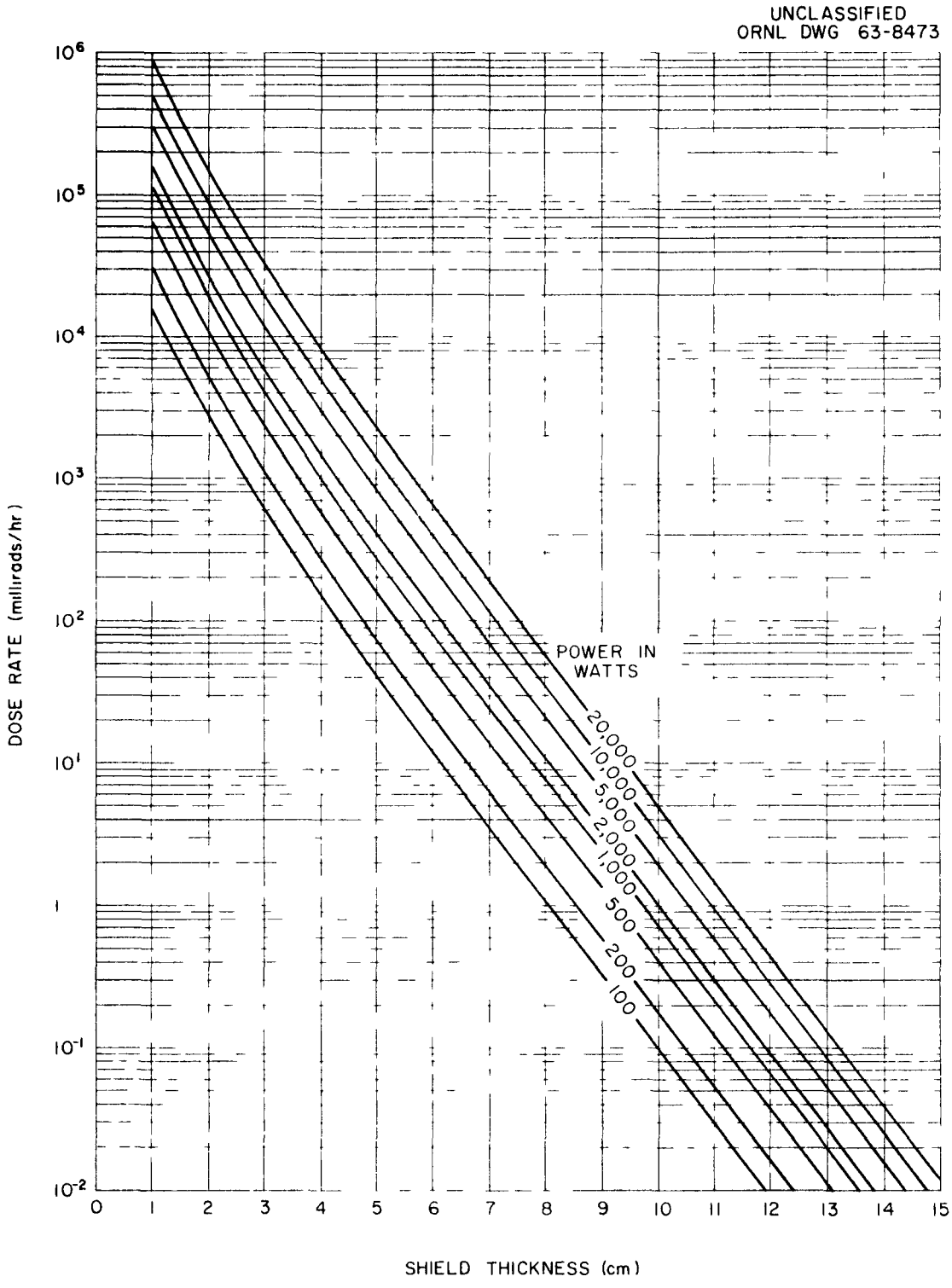


Fig. 37. Gamma Dose Rates Contributed by a 1% Impurity of Cesium-134 in Uranium-Shielded Isotopic Power Sources of Cesium-137. Center of source to dose point separation distance = 100 cm. Cesium-134 is assumed to contribute 1% of the beta activity due to Cs^{137}; refer to Fig. 33.

Cerium-144

Cerium-144 has several outstanding advantages for the majority of the special short-term applications, namely, its high fission yield (6.0 % from U^{235}) and high specific power (25.2 w/g). Since its half-life is less than a year (285 days), Ce^{144} is not generally considered as a major source material for long-term power applications. The extremely energetic gamma and bremsstrahlung activity of its Pr^{144} daughter and its dilution with stable cerium isotopes (about 16 wt % Ce^{144}) are distinct disadvantages. On the basis of the ratio of gamma energy per watt of total energy, Ce^{144} and U^{232} and Th^{228} are almost identical. Because of the fairly short half-life of Ce^{144}, its concentration with relation to the stable Ce^{140} and Ce^{142} will vary significantly, depending on the age of the material since formation in the fission process. However, there does not appear to be the aging problem for 33-day Ce^{141} in Ce^{144} as there is for Sr^{89} in Sr^{90}. Even at dilution to 16% Ce^{144}, the power density of the oxide at 90% of theoretical density is about 20 w/cc.

The biological hazard of Ce^{144} is relatively low, being about the same as that for Co^{60} and Cs^{137}. Because of both its very high specific power and high fission yield, Ce^{144} is potentially the cheapest heat source (on a watt basis). However, the short half-life and energetic gamma activity detract very severely from these apparent advantages.

Isotopic heat source data for Ce^{144} sources appear in Table 12, and the nuclear data appear in Table 3. The results of the radiation intensity calculations for shielded and unshielded sources are given in Figs. 38 through 41.

Table 12. Physical Properties of Cerium-144 as an Isotopic Power Source

General Properties	
Half-life	285 d
Specific activity of pure isotope, curies/g	3180
Specific power of pure isotope, w/g	25.2
Compound	Ce_2O_3
Weight of active isotope in compound, g/cc	0.80
Effective density of compound, g/cc	6.06
Power, w/cc of compound	20
Activity, curies/cc of compound	2544

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
7.484×10^{10}	0.7
1.17×10^{10}	1.5
3.74×10^{10}	2.2
8.204×10^{11}	0.35[b,e]
6.15×10^{10}	0.95[b]
9.644×10^{9}	1.55[b]
9.644×10^{8}	2.15[b]
2.00×10^{7}	2.60[b]

[b] Bremsstrahlung x ray group.

[e] Predominant radiation from unshielded 5000 watt source.

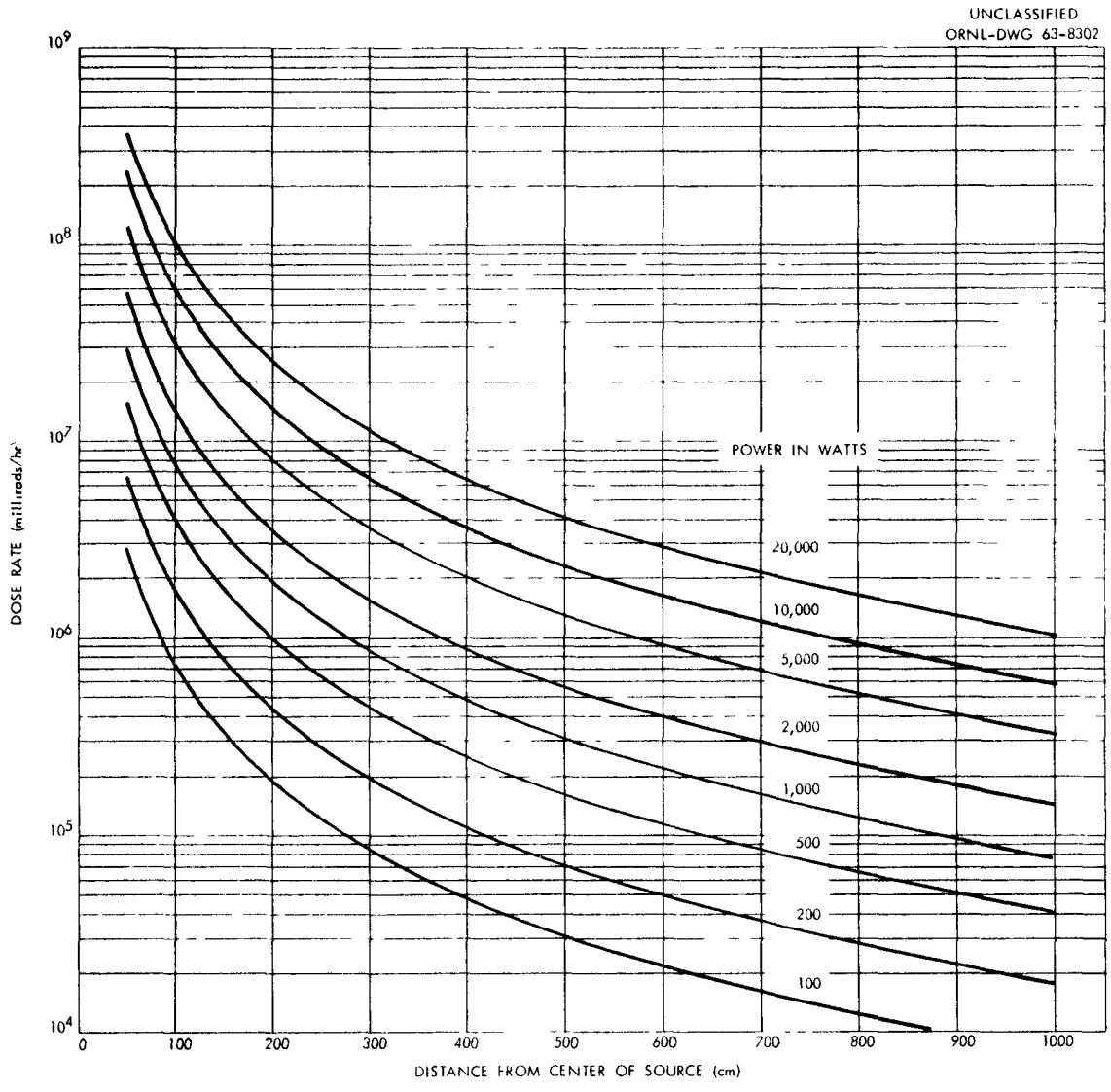


Fig. 38. Gamma Plus Bremsstrahlung Dose Rates from Unshielded Isotopic Power Sources of Cerium-144 as a Function of Distance from Center of Source.

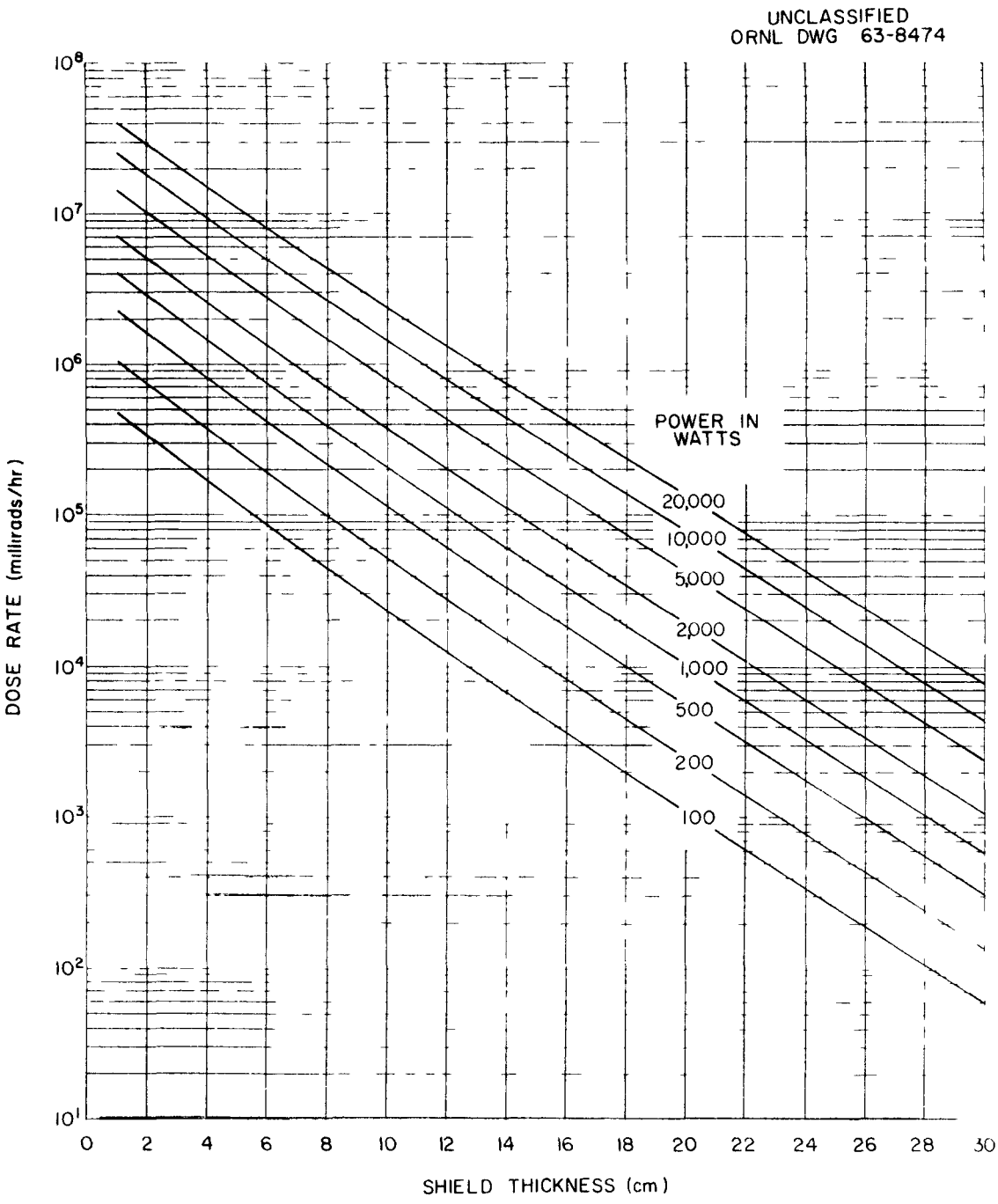


Fig. 39. Gamma Plus Bremsstrahlung Dose Rates from Iron-Shielded Isotopic Power Sources of Cerium-144. Center of source to dose point separation distance = 100 cm.

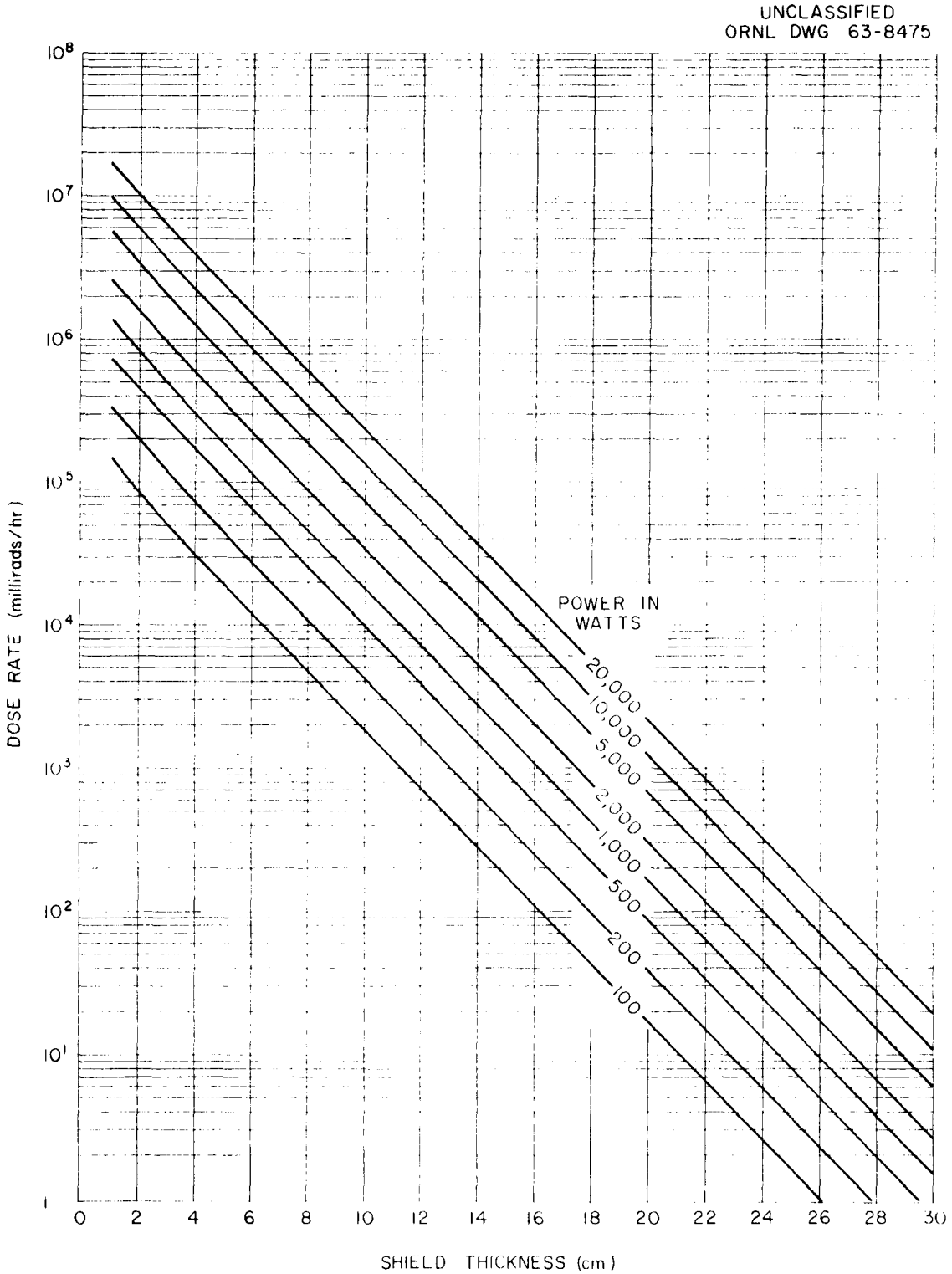


Fig. 40. Gamma Plus Bremsstrahlung Dose Rates from Lead-Shielded Isotopic Power Sources of Cerium-144. Center of source to dose point separation distance = 100 cm.

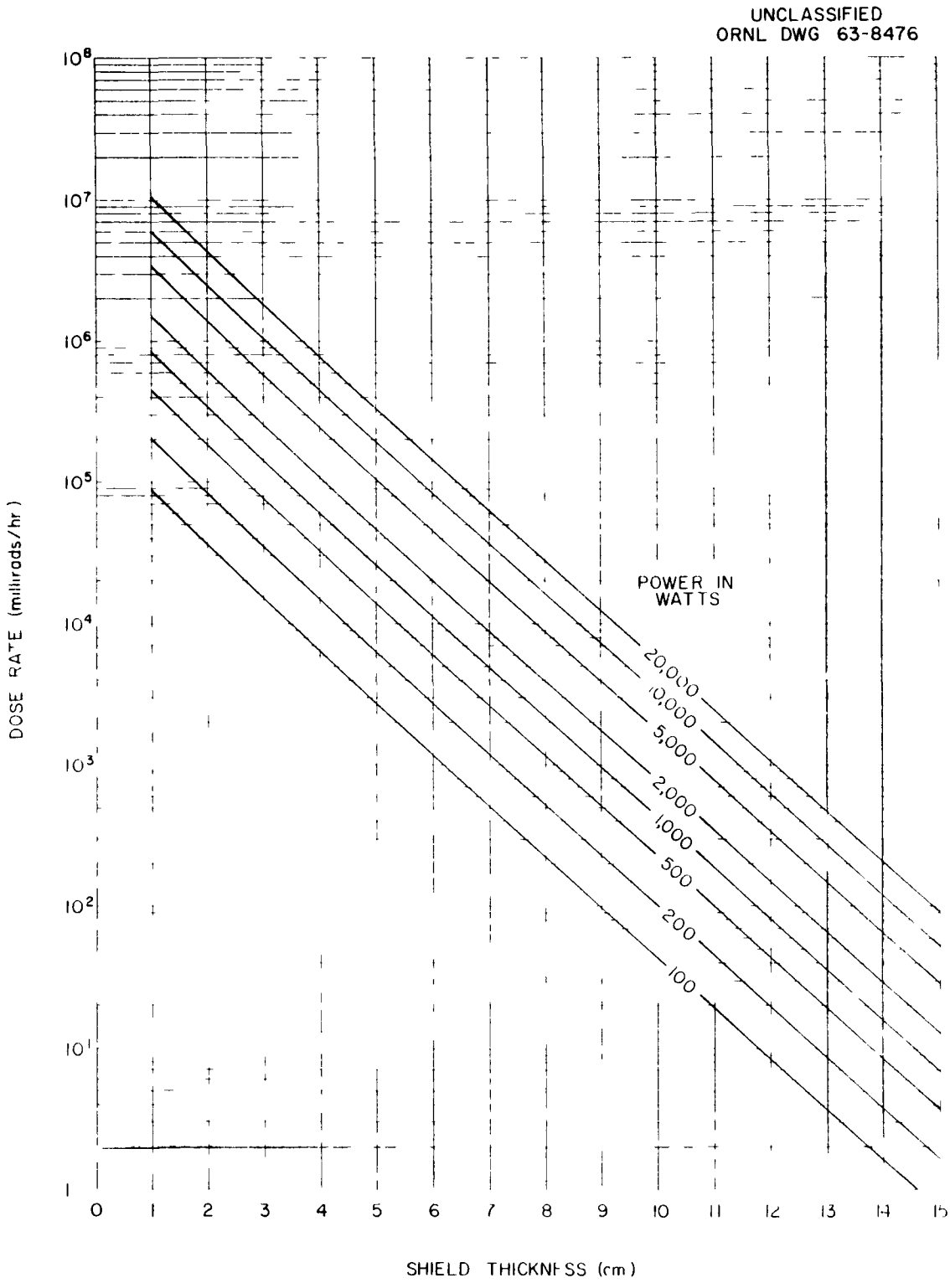


Fig. 41. Gamma Plus Bremsstrahlung Dose Rates from Uranium-Shielded Isotopic Power Sources of Cerium-144. Center of source to dose point separation distance = 100 cm.

Promethium-147

The properties of Pm^{147} ($T_{1/2}$ = 2.67 yr) are extremely desirable, and therefore a more critical and favorable review of this isotope is justified.[2] In the fission process, it is the only isotope of promethium realized in appreciable yield. Although its yield is only about half that of either Sr^{90} or Cs^{137}, it must still be regarded as having a reasonably high yield and should, therefore, be regarded as a fairly available material. Its higher atomic weight reduces the apparent disadvantage of fission yield in comparison with Sr^{90}. Since it has only one step in its decay scheme to "stable" (10^{11}-yr) Sm^{147}, it may be regarded as a pure beta emitter. The energy of the beta emission is relatively low; therefore, the bremsstrahlung will have low energy, which should be easily shielded with a minimum of mass. However, in the irradiation process, energetic, gamma emitting, 42-day Pm^{148} is formed by neutron capture in Pm^{147}; this contaminating isotope though present in very small concentrations is sufficient to mask the pure beta activity advantage of Pm^{147} (ref. 3). When recovered from fuel of only moderate age (less than a year), promethium has enough gamma energy from Pm^{148} such that shielding can be appreciable. Recently it has been noted that some gamma radiation can also be expected from the trace amounts of Pm^{146} which is also present. Promethium-146 has a half-life of about 1.9 yrs, which approximates that of Pm^{147}. For Pm^{148} with its half-life of 42 days, aging for about one Pm^{147} half-life (see Fig. 42) is sufficient to reduce the total shielding requirements to about that which would be expected for shielding bremsstrahlung and gammas from Pm^{147} and Pm^{146}. This is regarded as a minor shielding requirement and can be met by the mass of the usual encapsulating material. This aging step would, however, further reduce the apparent yield of this type of material. To achieve the highest power densities, the daughter products accumulated during aging would have to be separated.

Promethium has a shorter half-life than many of the other attractive heat source isotopes, but this 2.67-yr figure is considered long enough for a number of useful missions. Its specific power is 0.36 w/g; the power density of the oxide is expected to be about 2.0 w/cc.

Biologically, Pm^{147} is far less hazardous than any of the heat source isotopes which have been considered.

As a heat source, if power density is an important consideration, Pm^{147} in the oxide form is a superior choice since it is about twice as good as Sr^{90} in either the titanate or oxide form, and nine times as good as Cs^{137} in the form of the borosilicate glass.[2] In addition, it should be noted that Pm^{147} as the oxide still has over half the power density of Pu^{238} oxide. If its low biological hazard could be taken into account, such that a less chemically stable form like the metal could also be used, some advantages in thermal conductivity and heat source fabrication may also be realized.

Promethium-147 sources can be contaminated with europium isotopes[26] to such an extent that Eu^{152} plus Eu^{154} activities can easily control the shielding. Some ORNL experience has indicated that as much as 0.5% (beta-activity basis) Eu^{152} plus Eu^{154} could be present in the recovered product. If this is the case, the shield requirements would be little different from those for strontium or cesium. It thus becomes necessary to recover as pure a promethium product as possible with europium contamination being reduced to less than 0.01%. Van Tuyl, Roberts, and Wheelwright[3] have quoted a Pm^{146}/Pm^{147} ratio of 6.5 to 5×10^{-7}, while some ORNL data[26] has been found to indicate a Pm^{146}-to-Pm^{147} ratio of 8.3×10^{-4}. The data of all three types of sources were plotted.

Isotopic heat source data for Pm^{147} sources appear in Table 13, and the nuclear data appear in Table 3. The result of the radiation intensity calculations for shielded and unshielded sources are given in Figs. 43 through 46 for a promethium source containing 0.5% Eu^{152} plus Eu^{154} plus 8.37×10^{-2} % Pm^{146}; in Figs. 47 through 50 for a promethium source containing 8.37×10^{-2} % Pm^{146}; and in Figs. 51 through 54 for a promethium source containing 5×10^{-5} % Pm^{146}.

In addition, radiation intensities from a 0.01% impurity (beta-activity basis) of Pm^{148} were also plotted (Figs. 55 through 58). Promethium-148 was calculated and plotted separately as an impurity in the actual Pm^{147} sources since the activity due to Pm^{148} decreases rapidly with time. A beta activity of 0.01% of that due to Pm^{147} can be achieved with an aging period of 560 days. The effect of decay time on Pm^{146} and Pm^{148} activities is shown in Fig. 42.

Table 13. Physical Properties of Promethium-147 as an Isotopic Power Source

General Properties	
Half-life	2.67 y
Specific activity of pure isotope, curies/g	912
Specific power of pure isotope, w/g	0.362
Compound	Pm_2O_3
Weight of active isotope in compound, g/cc	5.53
Effective density of compound, g/cc	6.72
Power, w/cc of compound	2.0
Activity, curies/cc of compound	5043

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
9.765×10^{10}	0.15[e]
9.015×10^{9}	0.156[b]
2.909×10^{8}	0.25[b]
6.350×10^{9}	0.45[c]
6.350×10^{9}	0.75[c,f]
3.276×10^{11}	0.316[d]
3.425×10^{11}	0.919[d]
3.743×10^{11}	1.275[d,g]

[b] Pm^{147} bremsstrahlung x ray group.

[c] Pm^{146} gamma at a Pm^{146}/Pm^{147} activity ratio = 1×10^{-4} (actual calculations based on activity ratios of 8.37×10^{-4} and 5×10^{-7}.

[d] Gamma rays from 0.25% Eu^{152} + 0.25% Eu^{154} (beta activity basis) impurity.

[e] Predominant radiation for Pm^{147} + 5×10^{-5}% Pm^{146}.

[f] Predominant radiation for Pm^{147} + 8.37×10^{-2}% Pm^{146}.

[g] Predominant radiation for Pm^{147} + 8.37×10^{-2}% Pm^{146} + 0.25% Eu^{152} + 0.25% Eu^{154}.

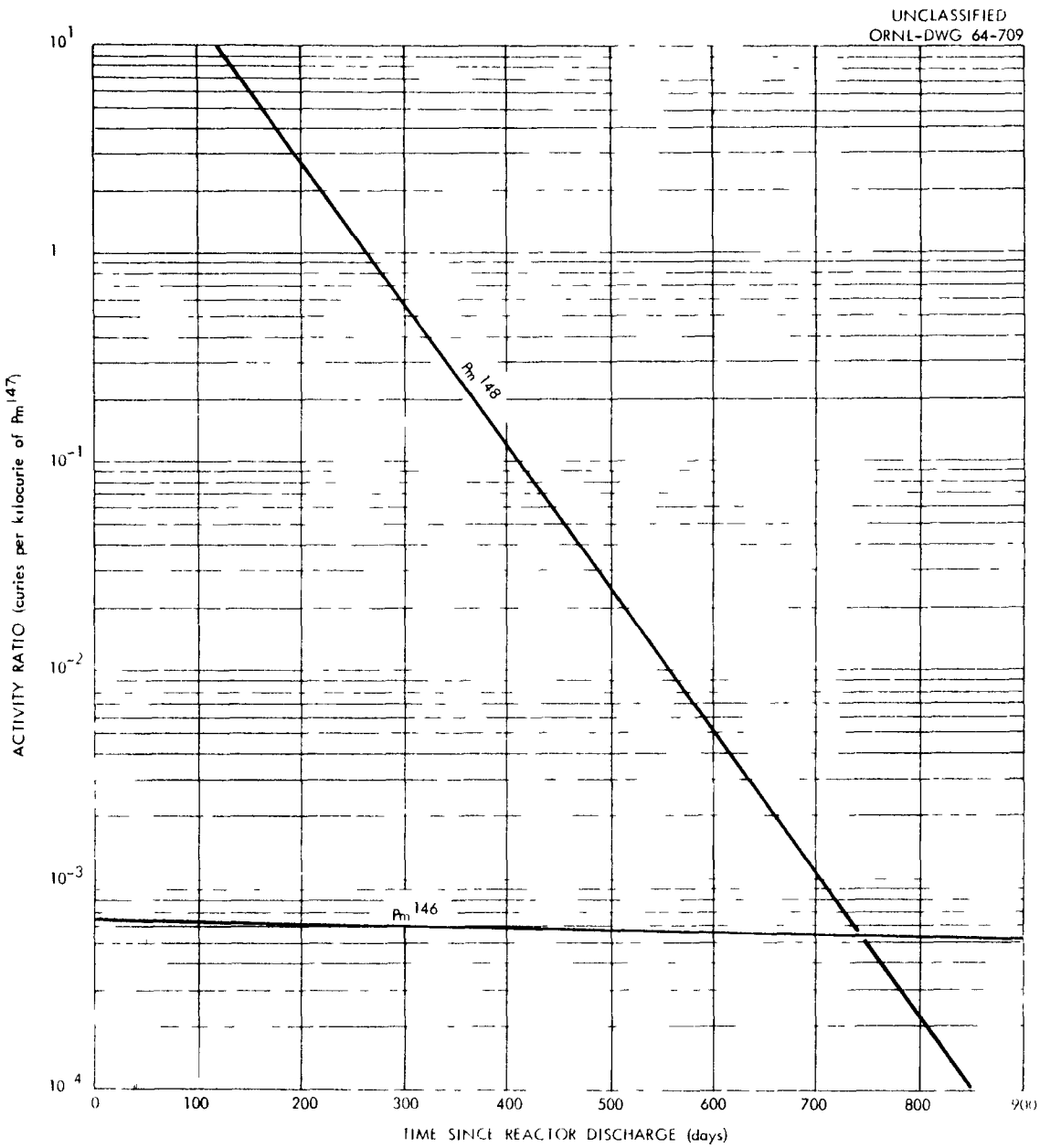


Fig. 42. Decay of Promethium-146 and Promethium-148 Impurities Relative to Promethium-147.

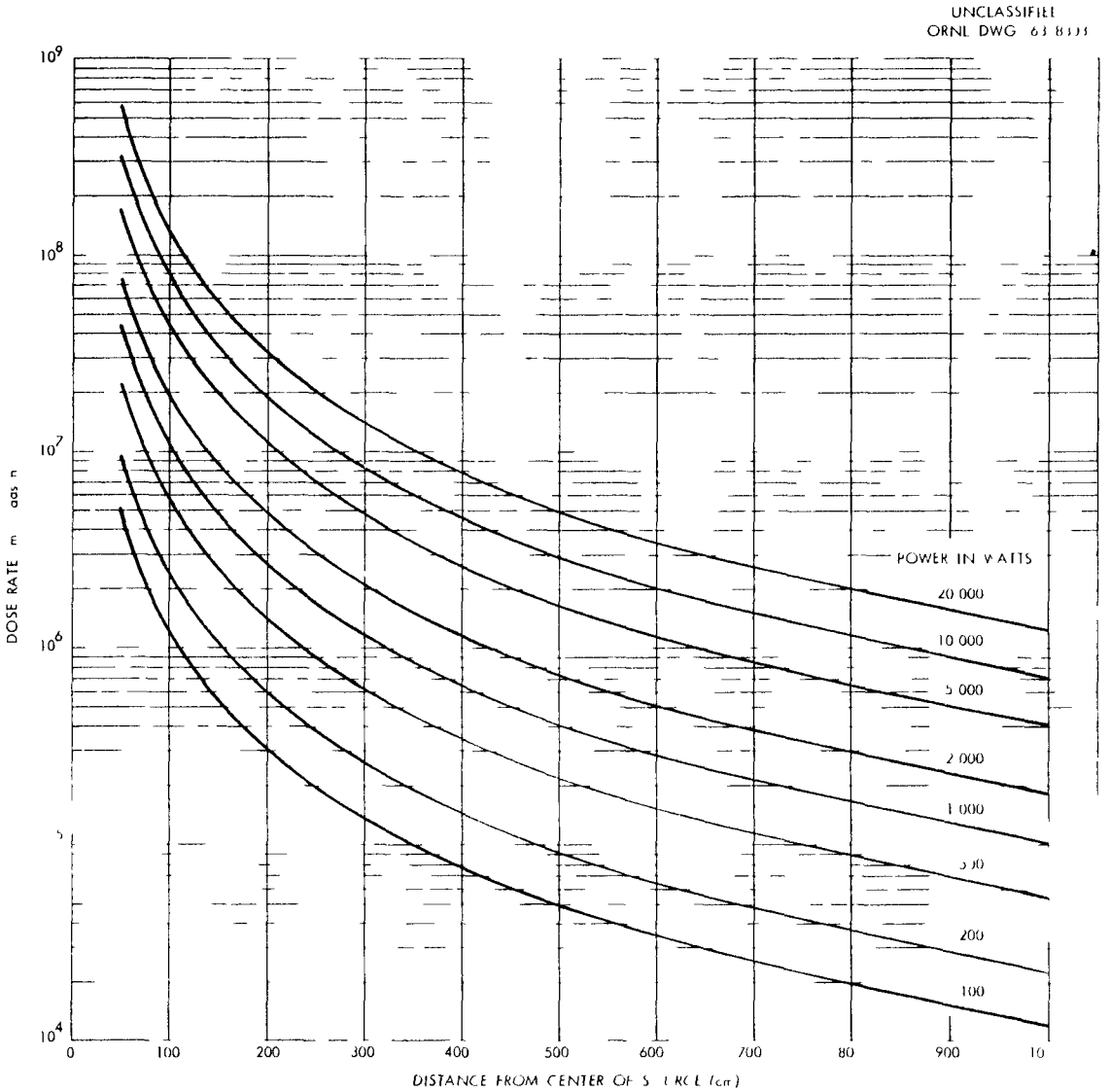


Fig. 43. Gamma Plus Bremsstrahlung Dose Rates from Unshielded Isotopic Power Sources of Promethium-147 as a Function of Distance from Center of Source. Source is assumed to contain 0.25% Eu^{152} + 0.25% Eu^{154} + 8.37 x 10^{-2}% Pm^{146} (beta activity basis) as impurities.

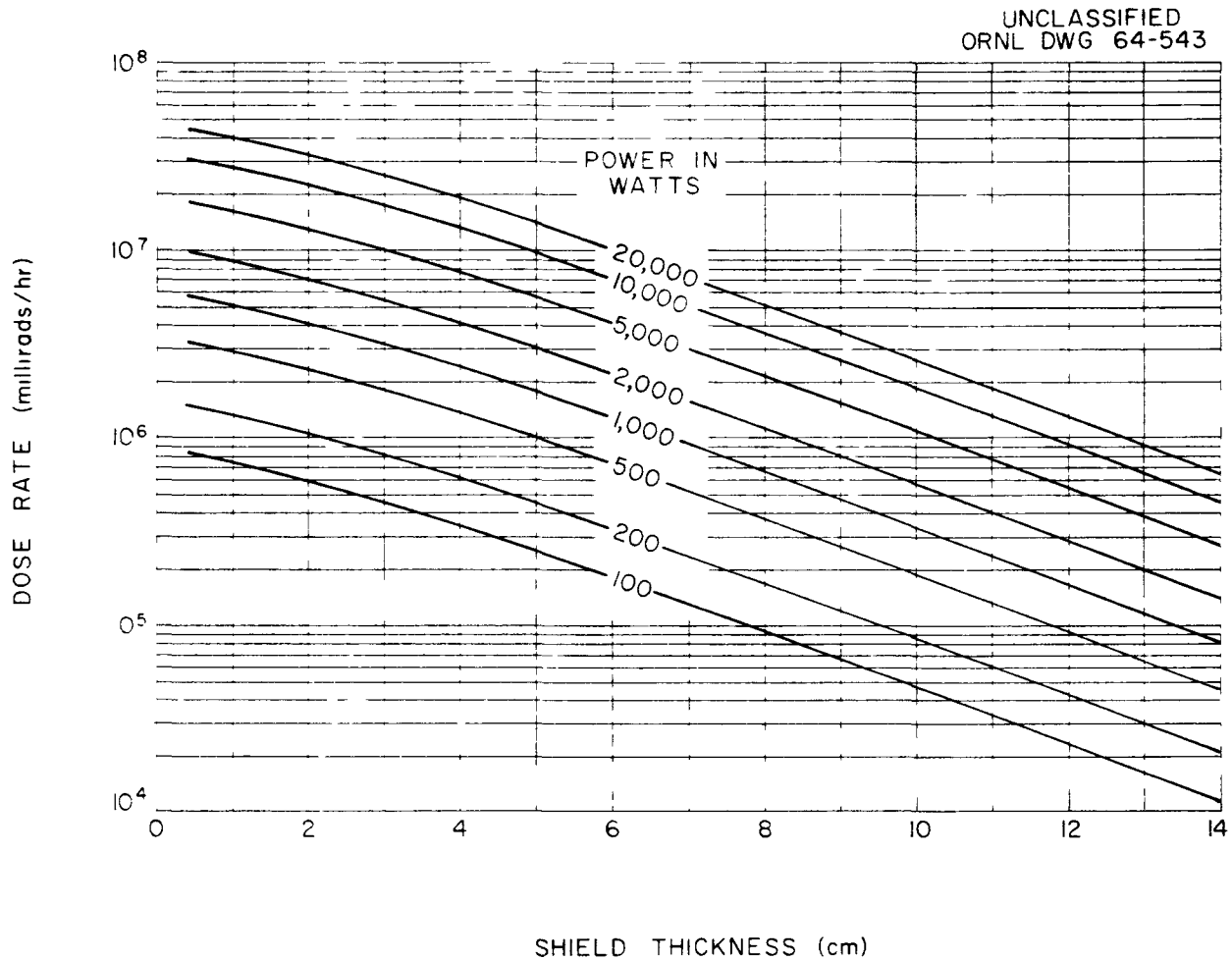


Fig. 44. Gamma Plus Bremsstrahlung Dose Rates from Iron-Shielded Isotopic Power Sources of Promethium-147. Center of source to dose point separation distance = 100 cm. Source is assumed to contain 0.25% Eu^{152} + 0.25% Eu^{154} + 8.37 x 10^{-2}% Pm^{146} (beta activity basis) as impurities.

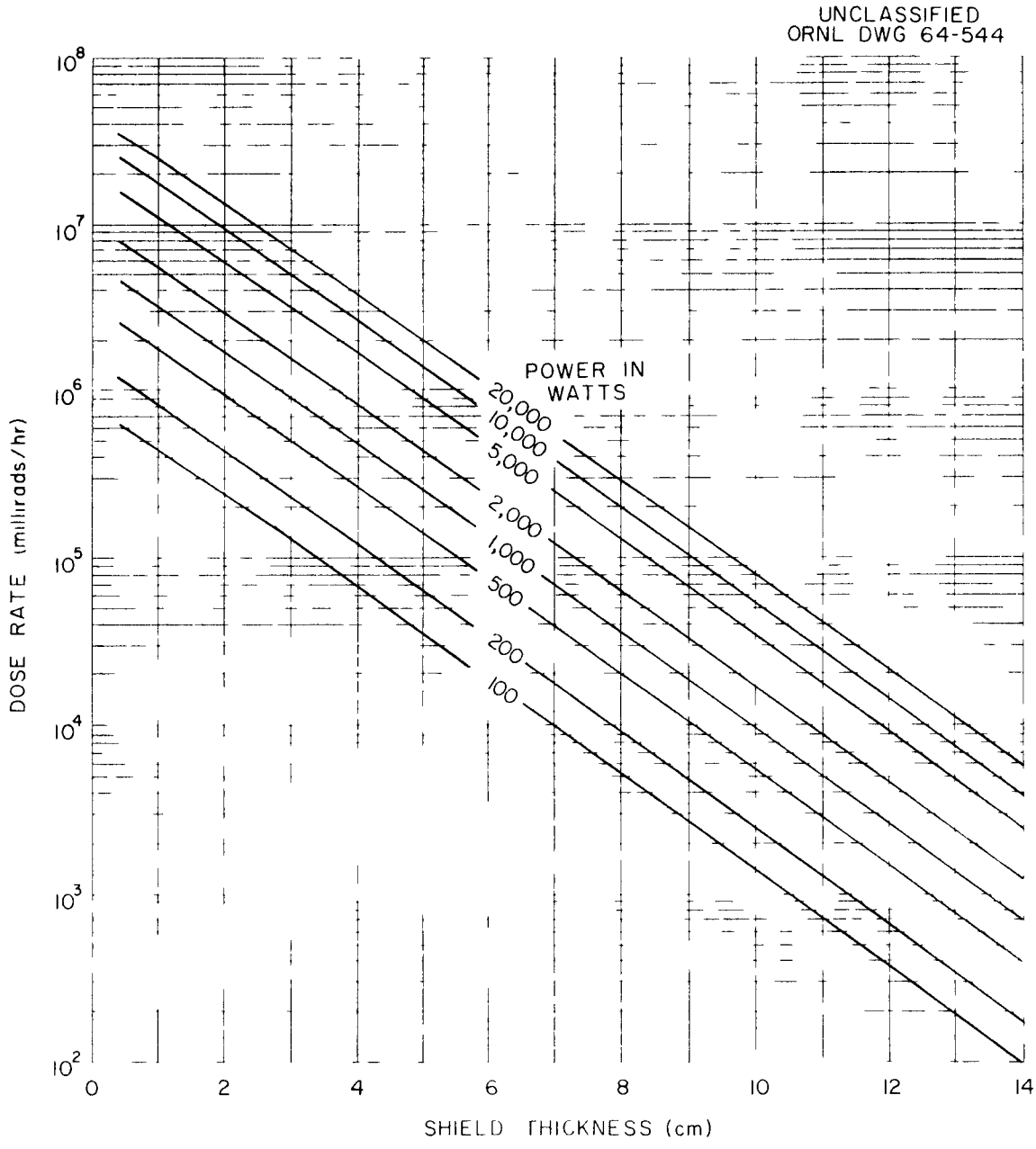


Fig. 45. Gamma Plus Bremsstrahlung Dose Rates from Lead-Shielded Isotopic Power Sources of Promethium-147. Center of source to dose point separation distance = 100 cm. Source is assumed to contain 0.25% Eu^{152} + 0.25% Eu^{154} + 8.37 x 10^{-2}% Pm^{146} (beta activity basis) as impurities.

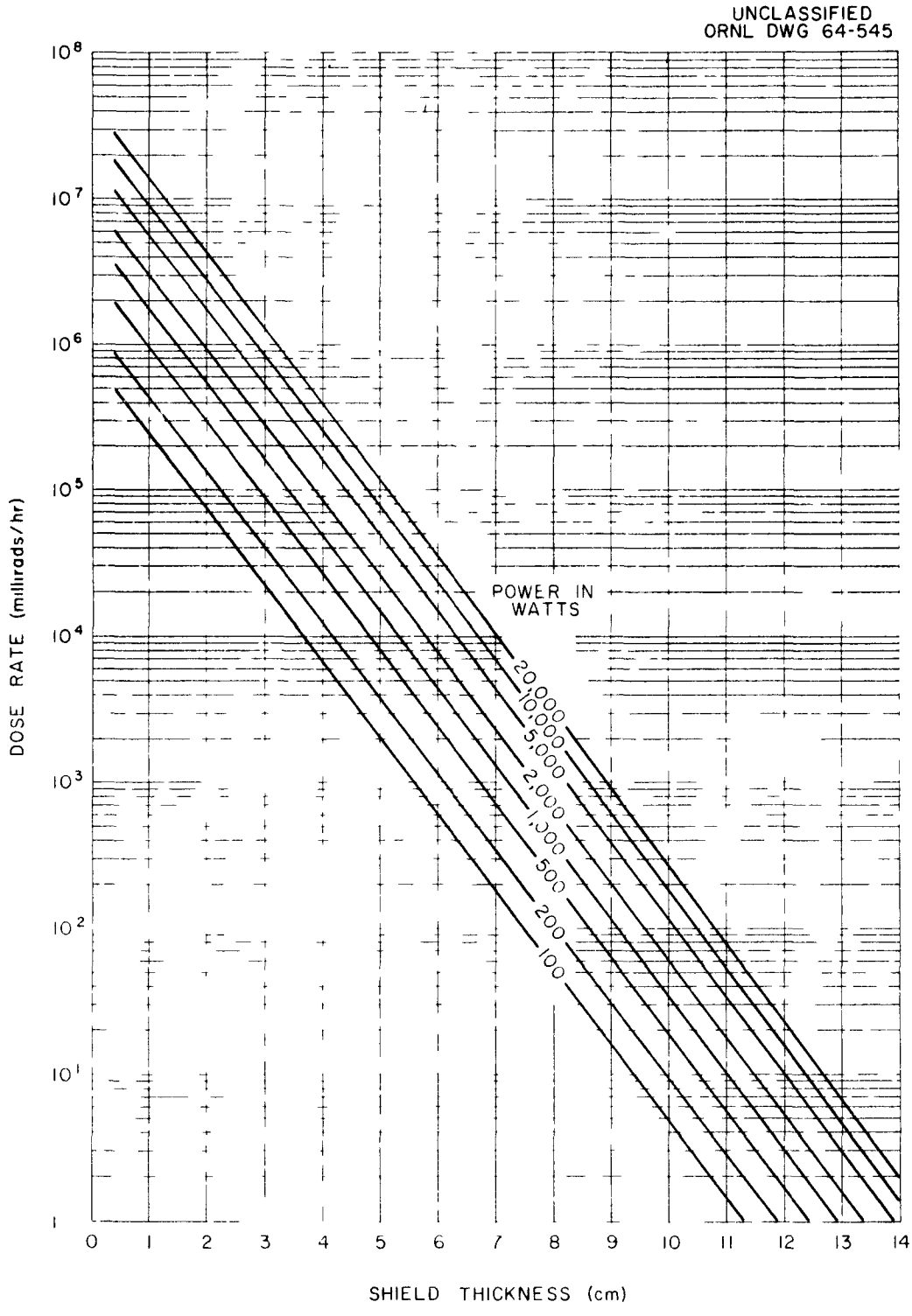


Fig. 46. Gamma Plus Bremsstrahlung Dose Rates from Uranium-Shielded Isotopic Power Sources of Promethium-147. Center of source to dose point separation distance = 100 cm. Source is assumed to contain 0.25% Eu^{152} + 0.25% Eu^{154} + 8.37 x 10^{-2}% Pm^{146} (beta activity basis) as impurities.

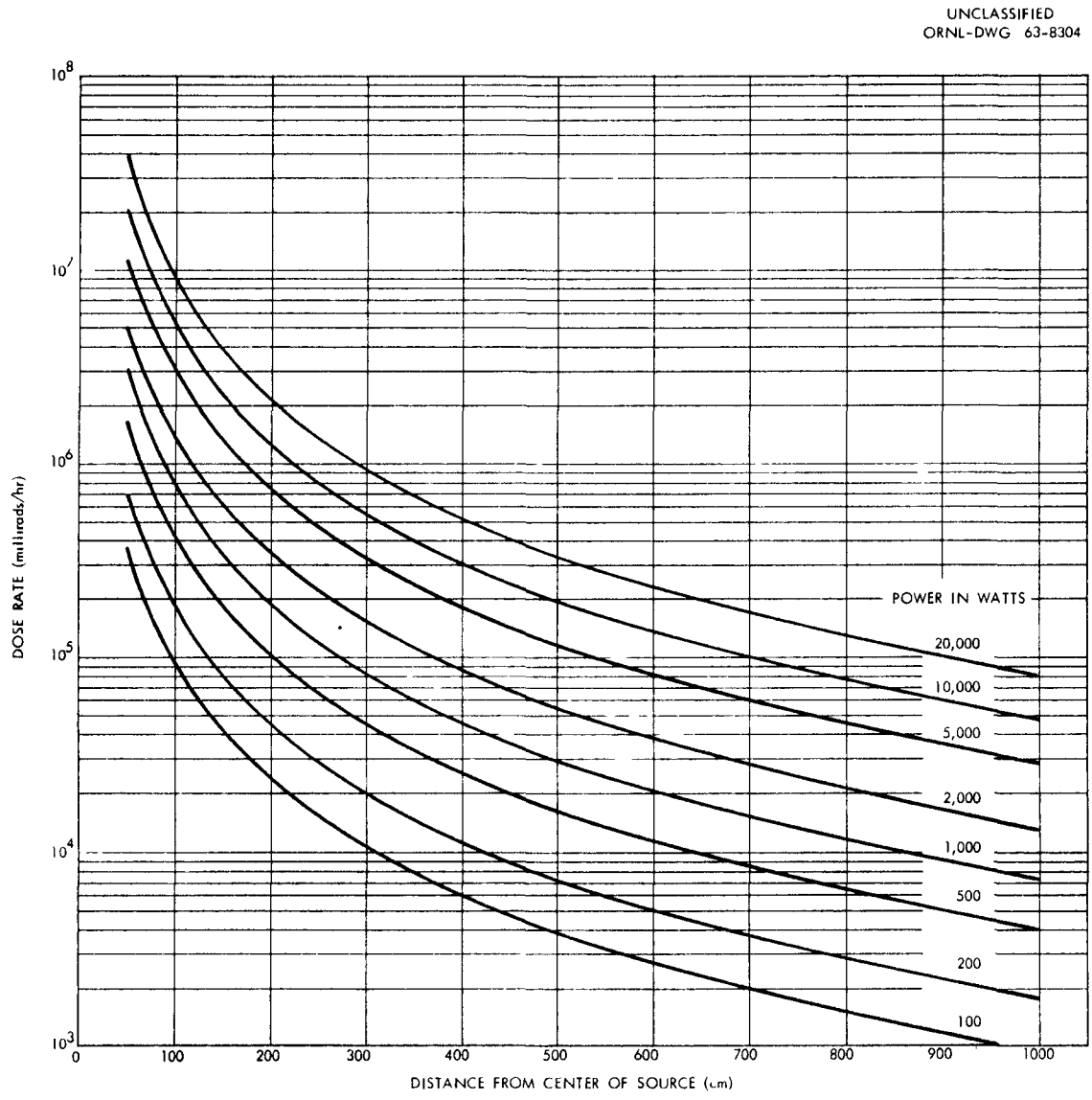


Fig. 47. Gamma Plus Bremsstrahlung Dose Rates from Unshielded Isotopic Power Sources of Promethium-147 as a Function of Distance from Center of Source. Source is assumed to contain 8.37×10^{-2}% Pm^{146} (beta activity basis) as an impurity.

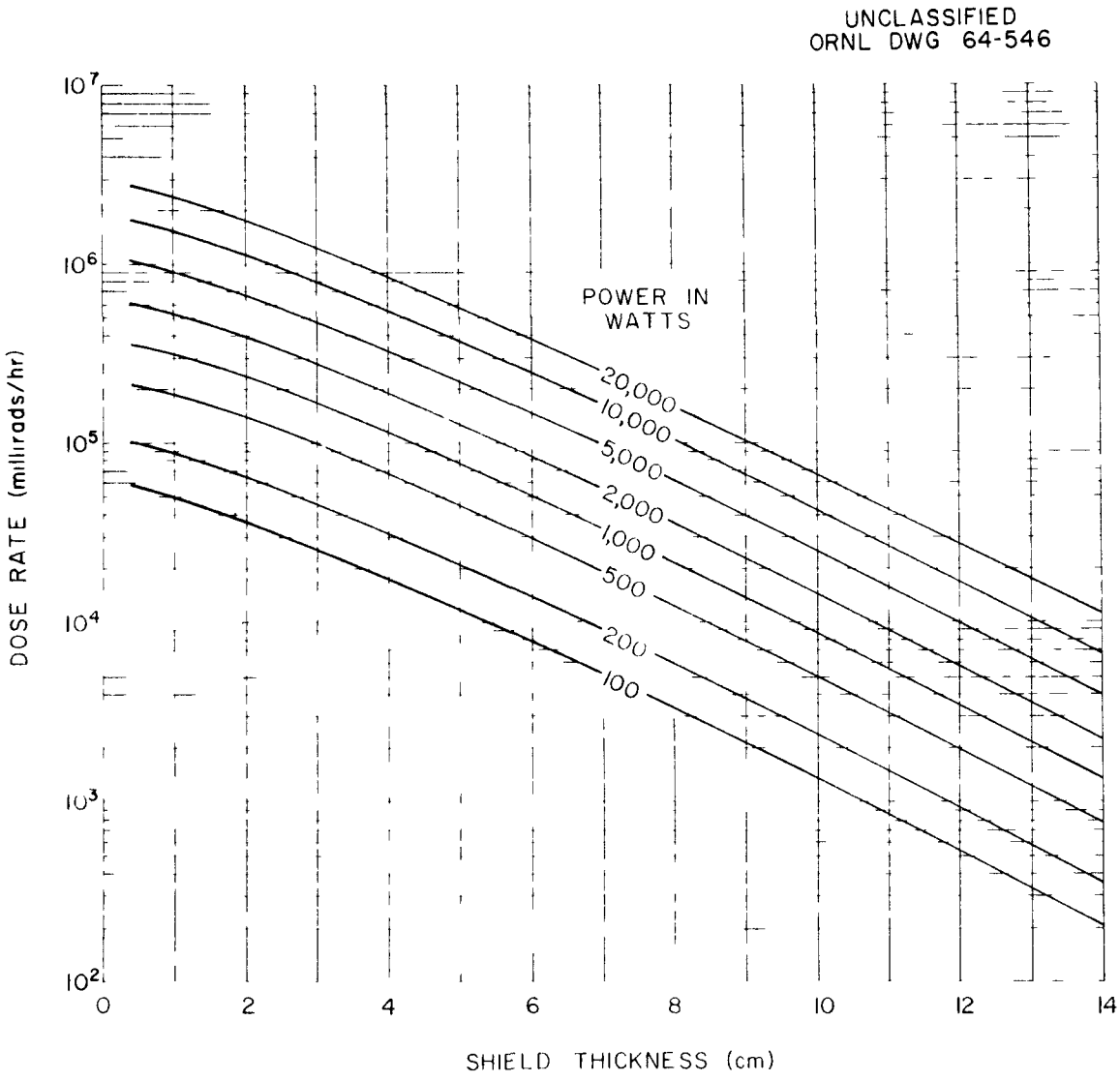


Fig. 48. Gamma Plus Bremsstrahlung Dose Rates from Iron-Shielded Isotopic Power Sources of Promethium-147. Center of source to dose point separation distance = 100 cm. Source is assumed to contain 8.37×10^{-2}% Pm^{146} (beta activity basis) as an impurity.

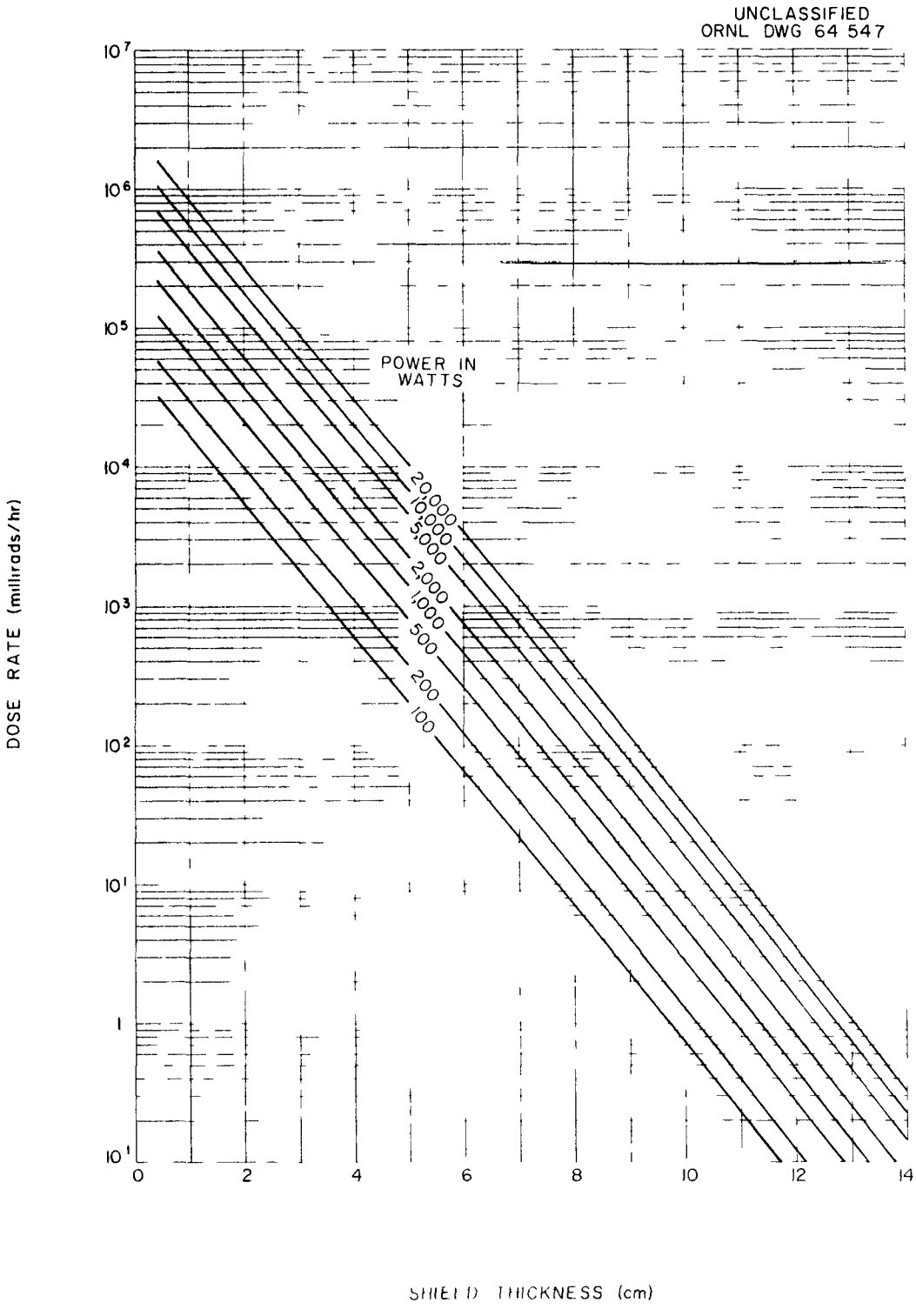


Fig. 49. Gamma Plus Bremsstrahlung Dose Rates from Lead-Shielded Isotopic Power Sources of Promethium-147. Center of source to dose point separation distance = 100 cm. Source is assumed to contain $8.37 \times 10^{-2}\%$ Pm^{146} (beta activity basis) as an impurity.

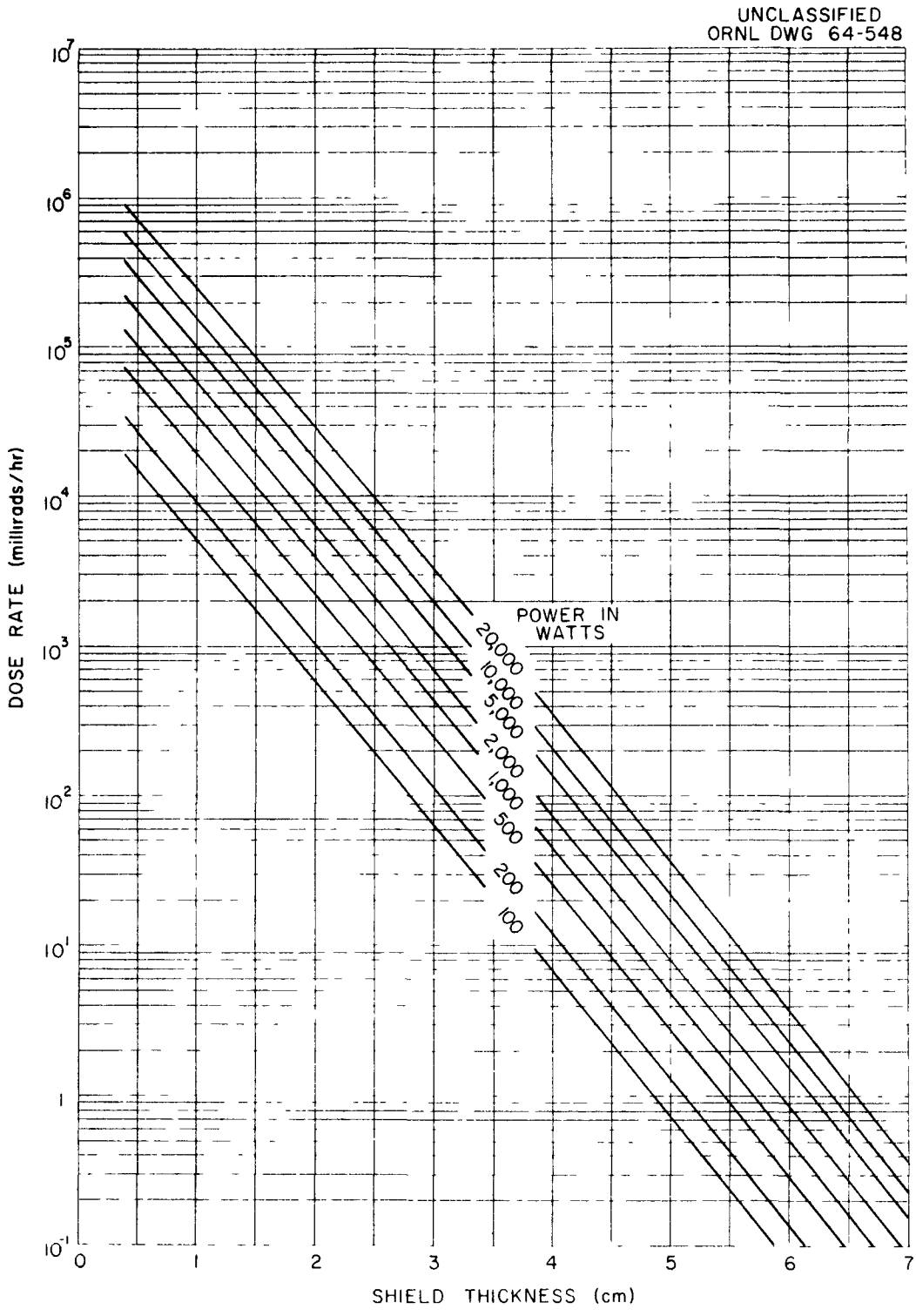


Fig. 50. Gamma Plus Bremsstrahlung Dose Rates from Uranium-Shielded Isotopic Power Sources of Promethium-147. Center of source to dose point separation distance = 100 cm. Source is assumed to contain $8.37 \times 10^{-2}\%$ Pm^{146} (beta activity basis) as an impurity.

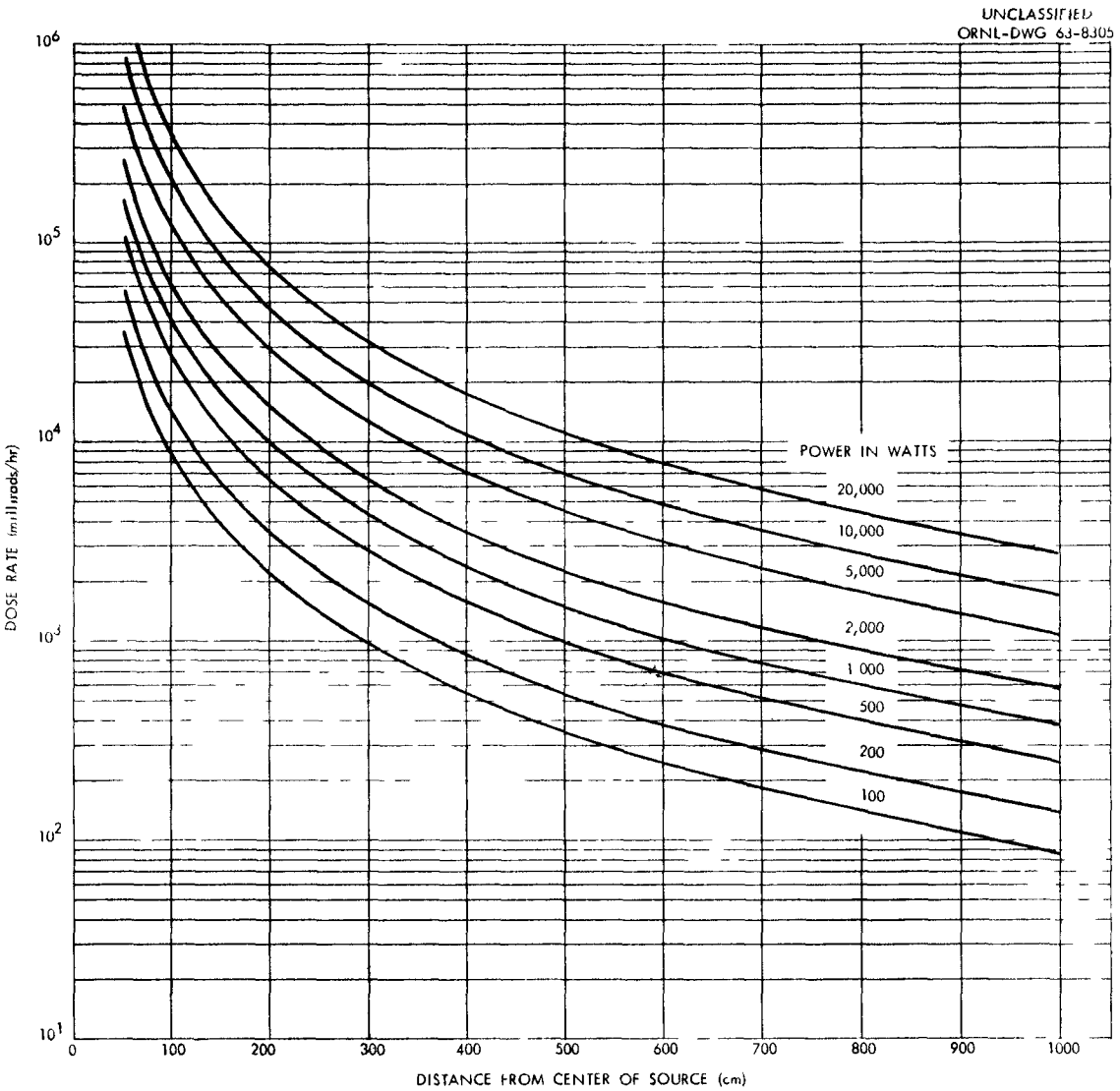


Fig. 51. Gamma Plus Bremsstrahlung Dose Rates from Unshielded Isotopic Power Sources of Promethium-147 as a Function of Distance from Center of Source. Source is assumed to contain 5 x 10^{-5}% Pm^{146} (beta activity basis) as an impurity.

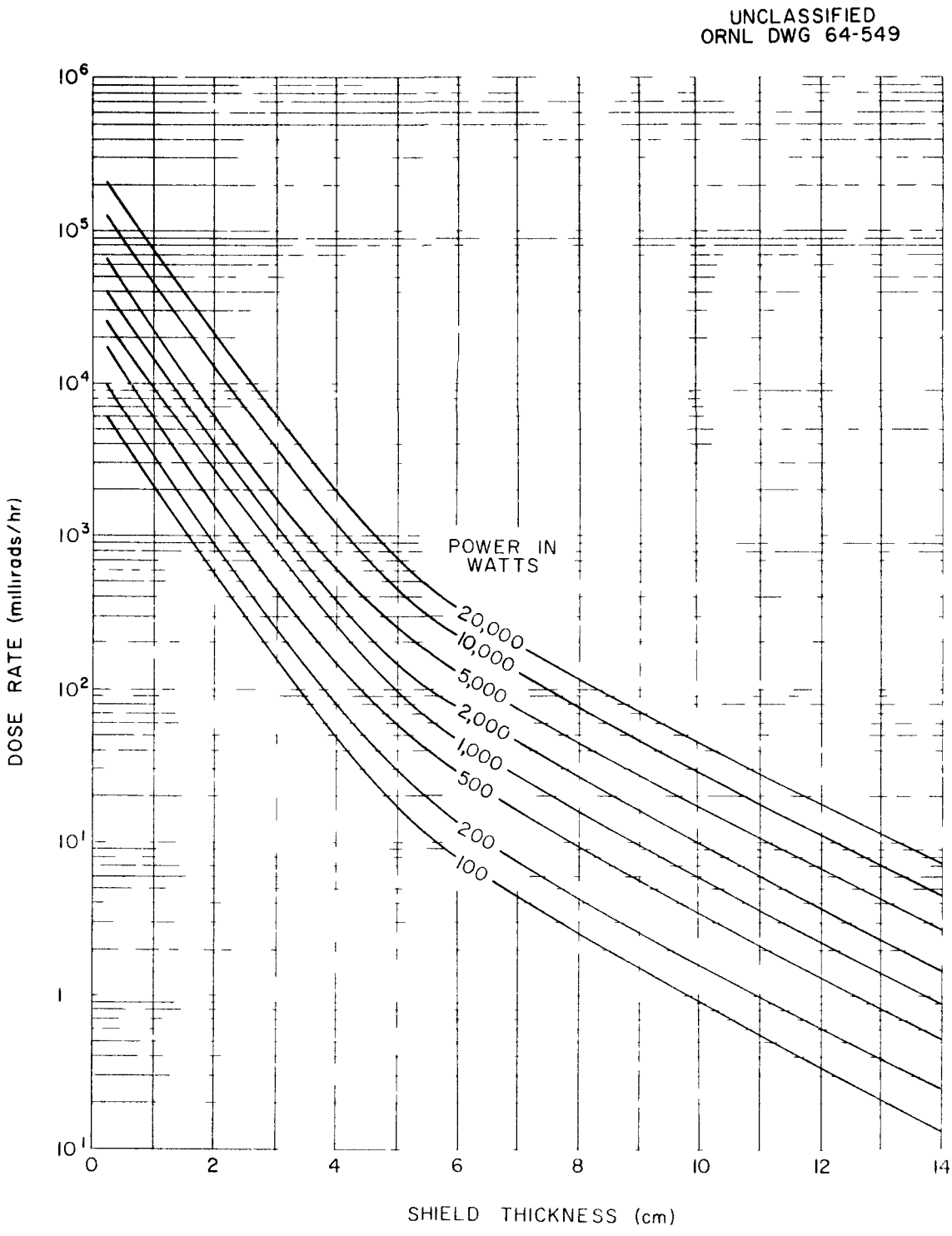


Fig. 52. Gamma Plus Bremsstrahlung Dose Rates from Iron-Shielded Isotopic Power Sources of Promethium-147. Center of source to dose point separation distance = 100 cm. Source is assumed to contain 5 x 10^{-5}% Pm^{146} (beta activity basis) as an impurity.

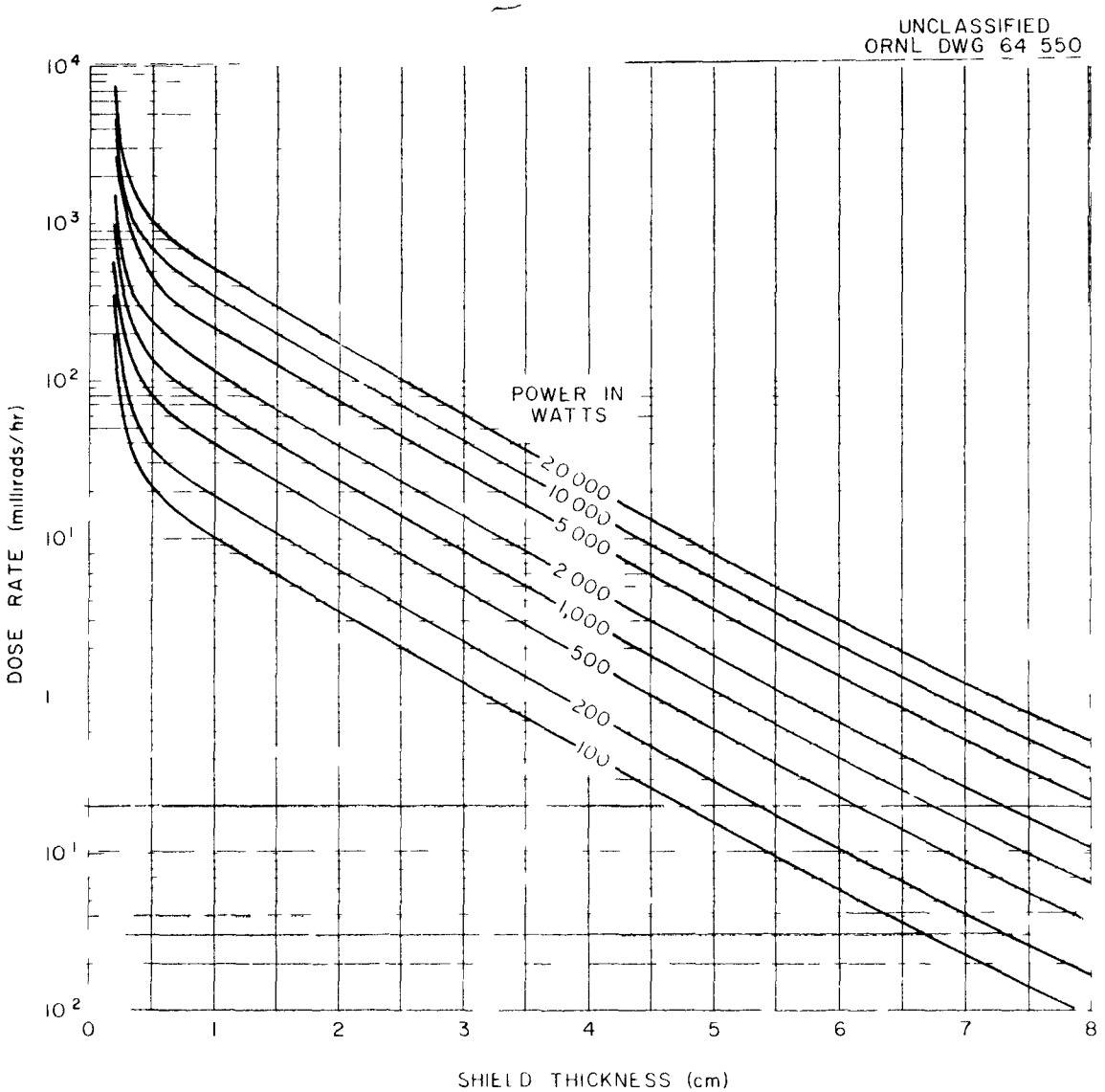


Fig. 53. Gamma Plus Bremsstrahlung Dose Rates from Lead-Shielded Isotopic Power Sources of Promethium-147. Center of source to dose point separation distance = 100 cm. Source is assumed to contain $5 \times 10^{-5}\%$ Pm^{146} (beta activity basis) as an impurity.

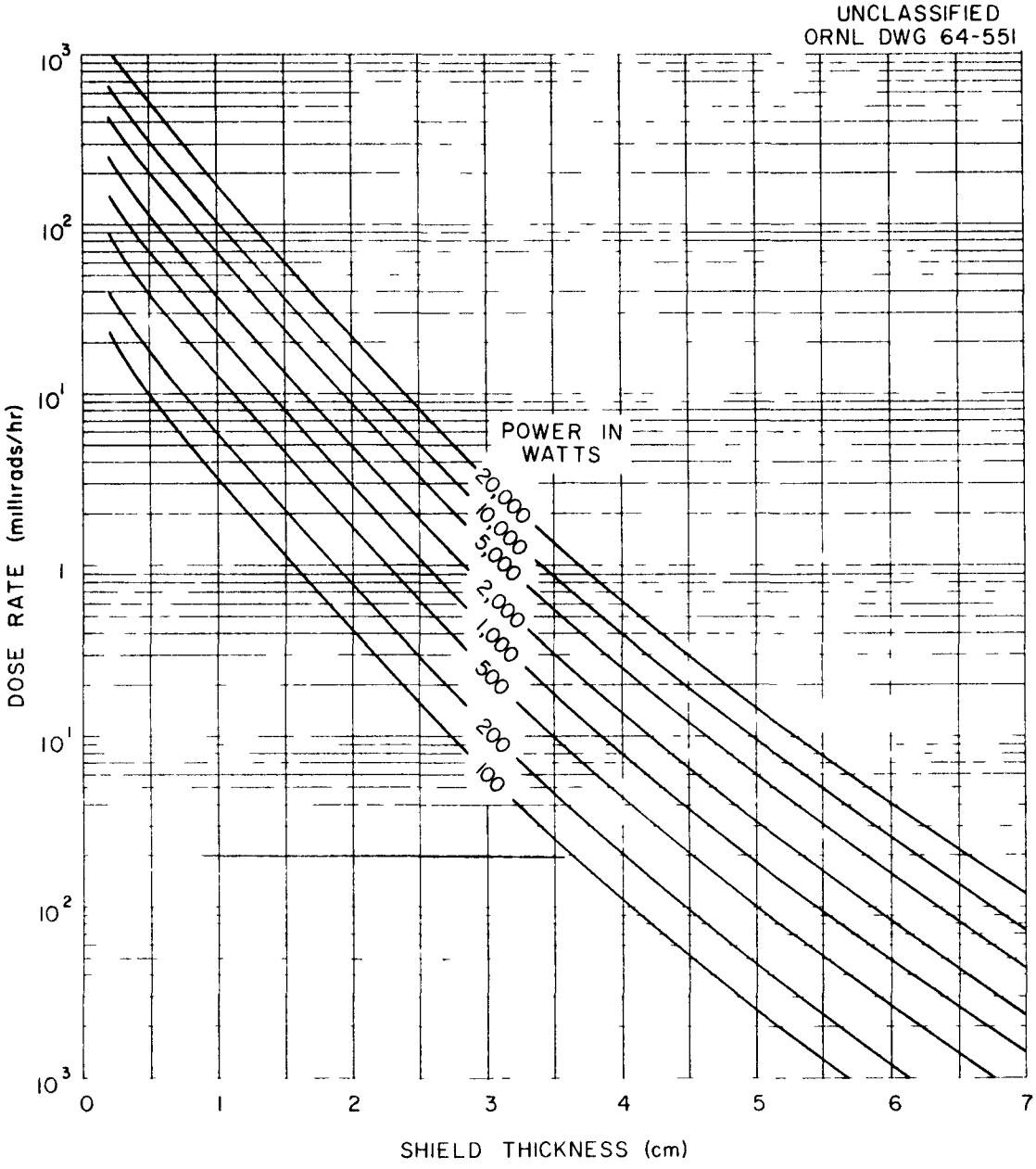


Fig. 54. Gamma Plus Bremsstrahlung Dose Rates from Uranium-Shielded Isotopic Power Sources of Promethium-147. Center of source to dose point separation distance = 100 cm. Source is assumed to contain 5 x 10^{-5}% Pm^{146} (beta activity basis) as an impurity.

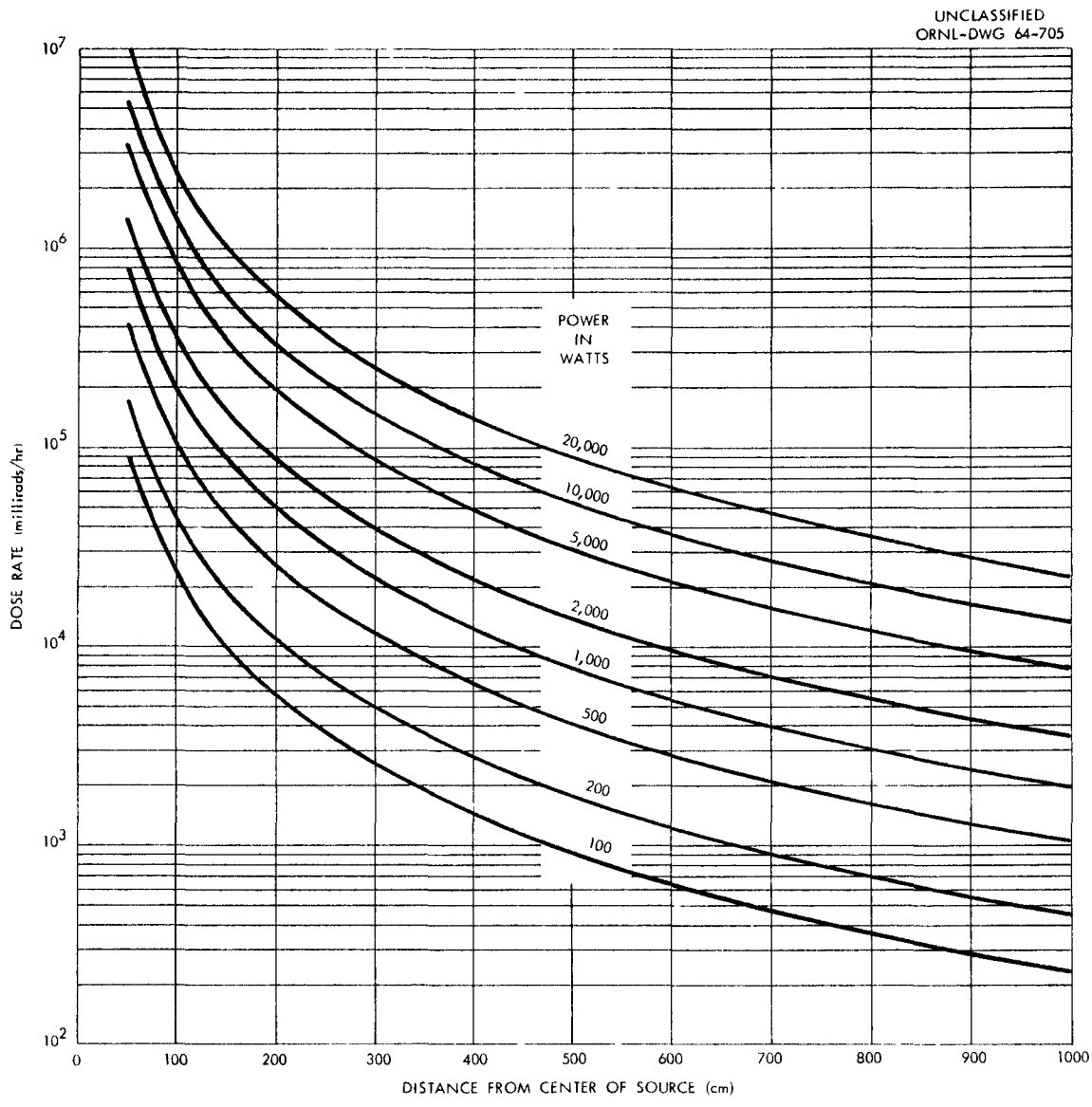


Fig. 55. Gamma Dose Rates Contributed by a 0.01% Impurity of Promethium-148 in Unshielded Isotopic Power Sources of Promethium-147 as a Function of Distance from Center of Source. Promethium-148 is assumed to contribute 0.01% of the beta activity due to Pm^{147}; refer to Figs. 43, 47, and 51.

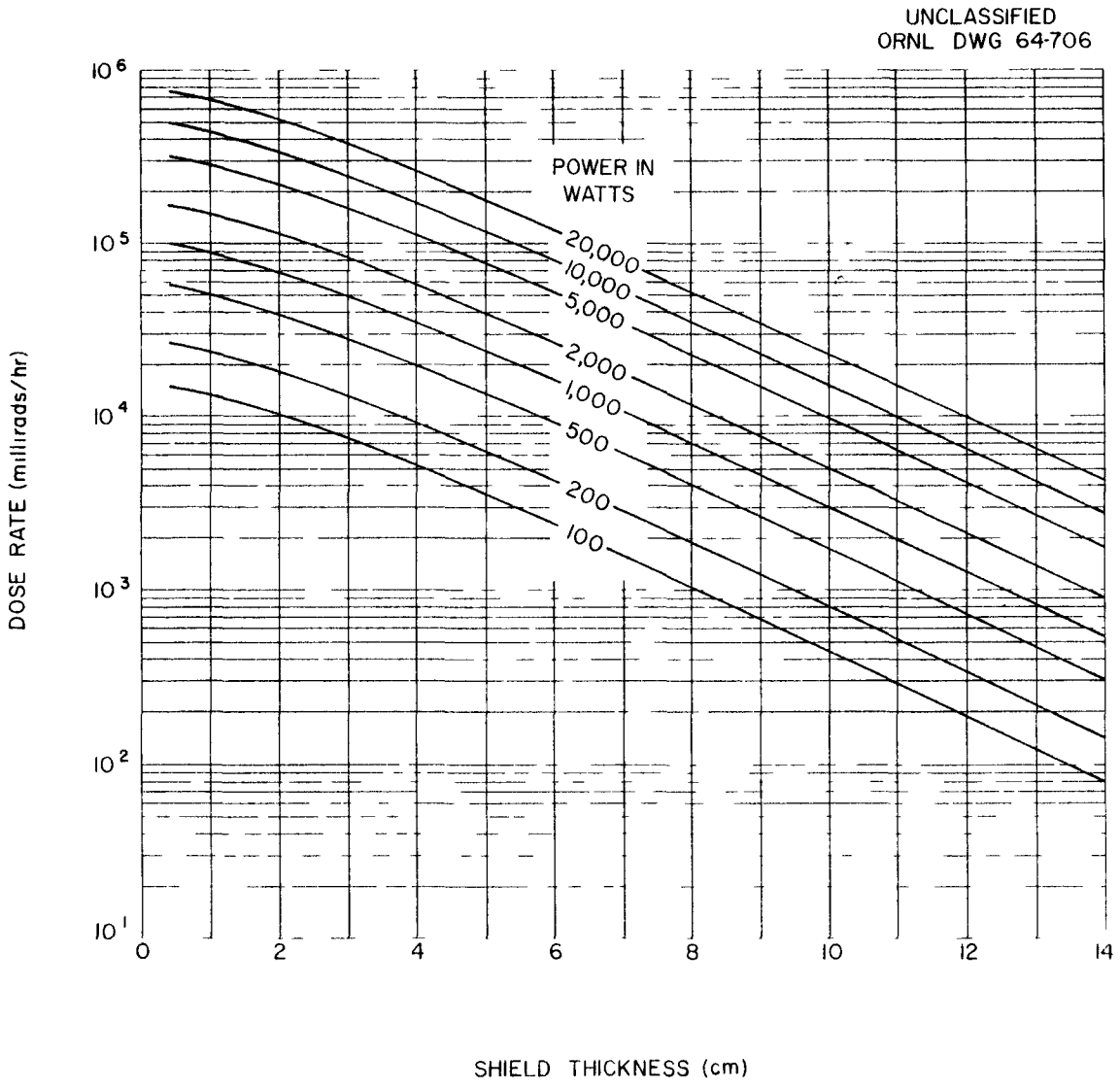


Fig. 56. Gamma Dose Rates Contributed by a 0.01% Impurity of Promethium-148 in Iron-Shielded Isotopic Power Sources of Promethium-147. Center of source to dose point separation distance = 100 cm. Promethium-148 is assumed to contribute 0.01% of the beta activie due to Pm^{147}; refer to Figs. 44, 48, and 52.

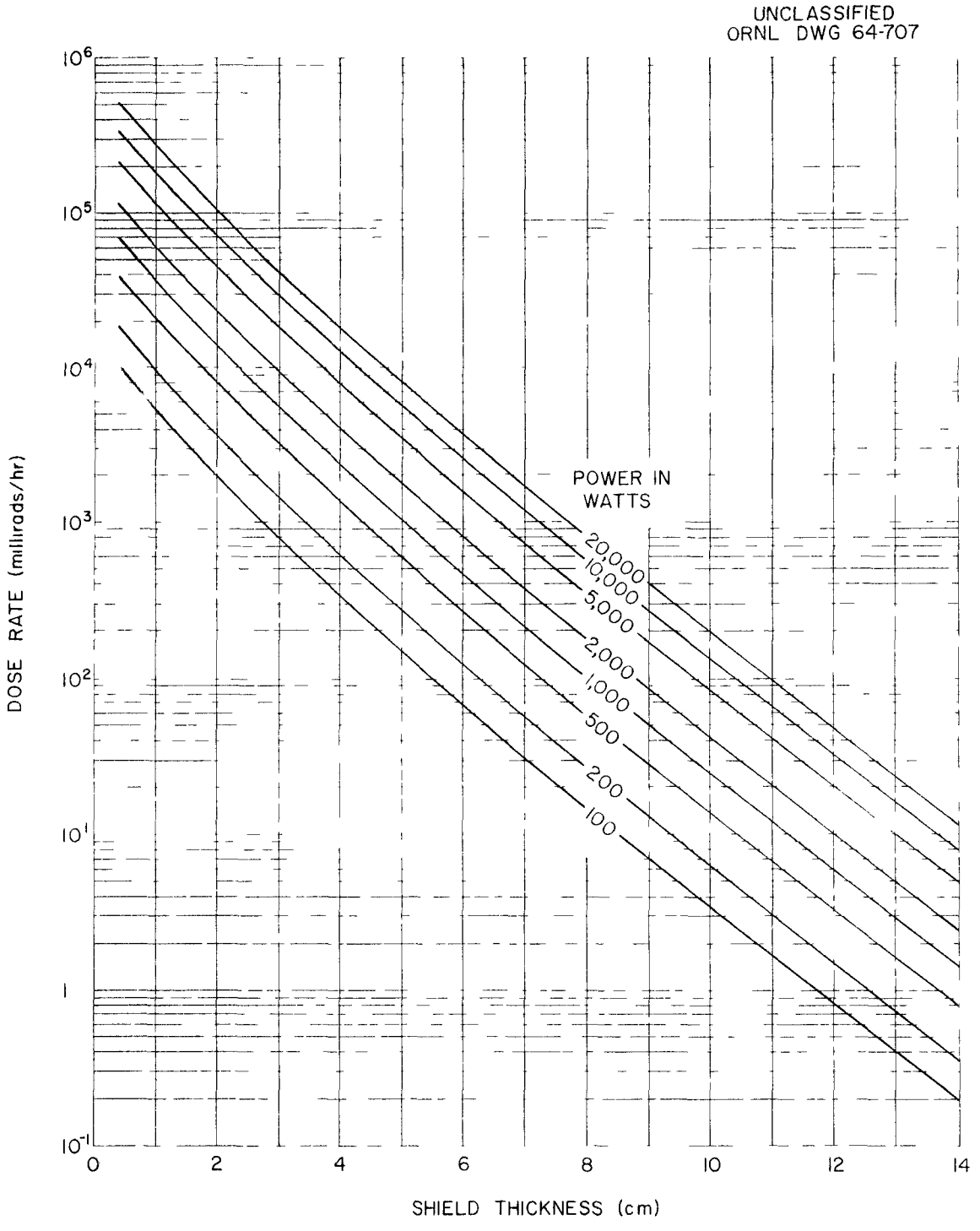


Fig. 57. Gamma Dose Rates Contributed by a 0.01% Impurity of Promethium-148 in Lead-Shielded Isotopic Power Sources of Promethium-147. Center of source to dose point separation distance = 100 cm. Promethium-148 is assumed to contribute 0.01% of the beta activity due to Pm^{147}; refer to Figs. 45, 49, and 53.

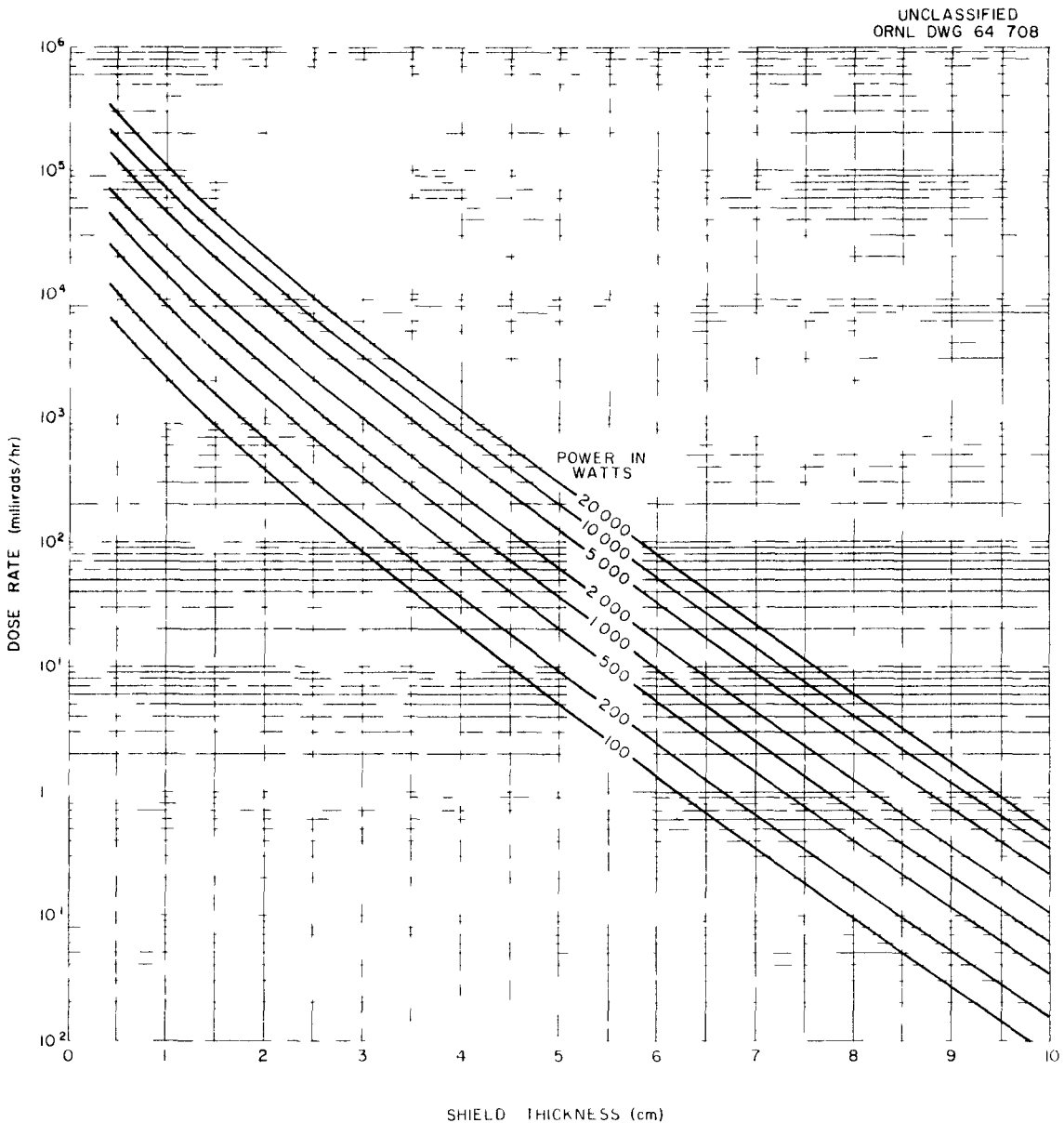


Fig. 58. Gamma Dose Rates Contributed by a 0.01% Impurity of Promethium-148 in Uranium-Shielded Isotopic Power Sources of Promethium-147. Center of source to dose point separation distance = 100 cm. Promethium-148 is assumed to contribute 0.01% of the beta activity due to Pm^{147}; refer to Figs. 46, 50, and 54.

Thulium-170

Thulium-170 appears to be a reasonable power source where a short half-life isotope is best suited but the availability of target material is limited at this time. It's half-life is only 127 days, which thus places it in the same category ($T_{1/2}$ less than 1 yr) with Zr^{95}, Ce^{144}, Po^{210} and Cm^{242}. Thulium-170 is produced by neutron irradiation of Tm^{169}, which constitutes 100% of the naturally occurring thulium. The specific activity and power of Tm^{170} are 6048 curies/g and 12.1 w/g, respectively. However, a practical irradiation would convert only 16.6% of the Tm^{169} to the 170 isotope. The resulting activity and power would then be 100 curies/g and 2 w/g of thulium metal, respectively. A power source of thulium oxide (Tm_2O_3) would have a specific power of 15 w/cc. Thulium-170 produced by irradiation at 3×10^{14} neutrons cm^{-2} sec^{-1} thermal flux to 1000 curies/g thulium would have a Tm^{171}/Tm^{170} atomic ratio of 0.27, or an activity ratio of 0.05. Thulium-171 produces much softer gamma and bremsstrahlung than Tm^{170} and thus this impurity would not be significant.

The biological hazard of Tm^{170} is low, being about the same as that of Co^{60} or Zr^{95}.

Isotopic heat source data for Tm^{170} sources appear in Table 14, and the nuclear data appear in Table 3. The results of the radiation intensity calculations for shielded and unshielded sources are given in Figs. 59 through 62.

Table 14. Physical Properties of Thulium-170 as an Isotopic Power Source

General Properties	
Half-life	127 d
Specific activity of pure isotope, curies/g	6048
Specific power of pure isotope, w/g	12.1
Compound	Tm_2O_3
Weight of active isotope in compound, g/cc	1.247
Effective density of compound, g/cc	8.6
Power, w/cc of compound	15.08
Activity, curies/cc of compound	7540

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
7.465×10^{11}	0.2[b,e]
5.01×10^{10}	0.475[b]
1.158×10^{9}	0.775[b]
3.208×10^{12}	0.1[c]

[b] Bremsstrahlung x ray group.

[c] Actual gamma emission rate = 6.4×10^{11} at 0.084 Mev + additional softer x rays.

[e] Predominant radiation from unshielded 5000 watt source. However, the 0.1 Mev and the 0.475 Mev γ each contribute ~80% of the dose delivered by the 0.2 Mev γ.

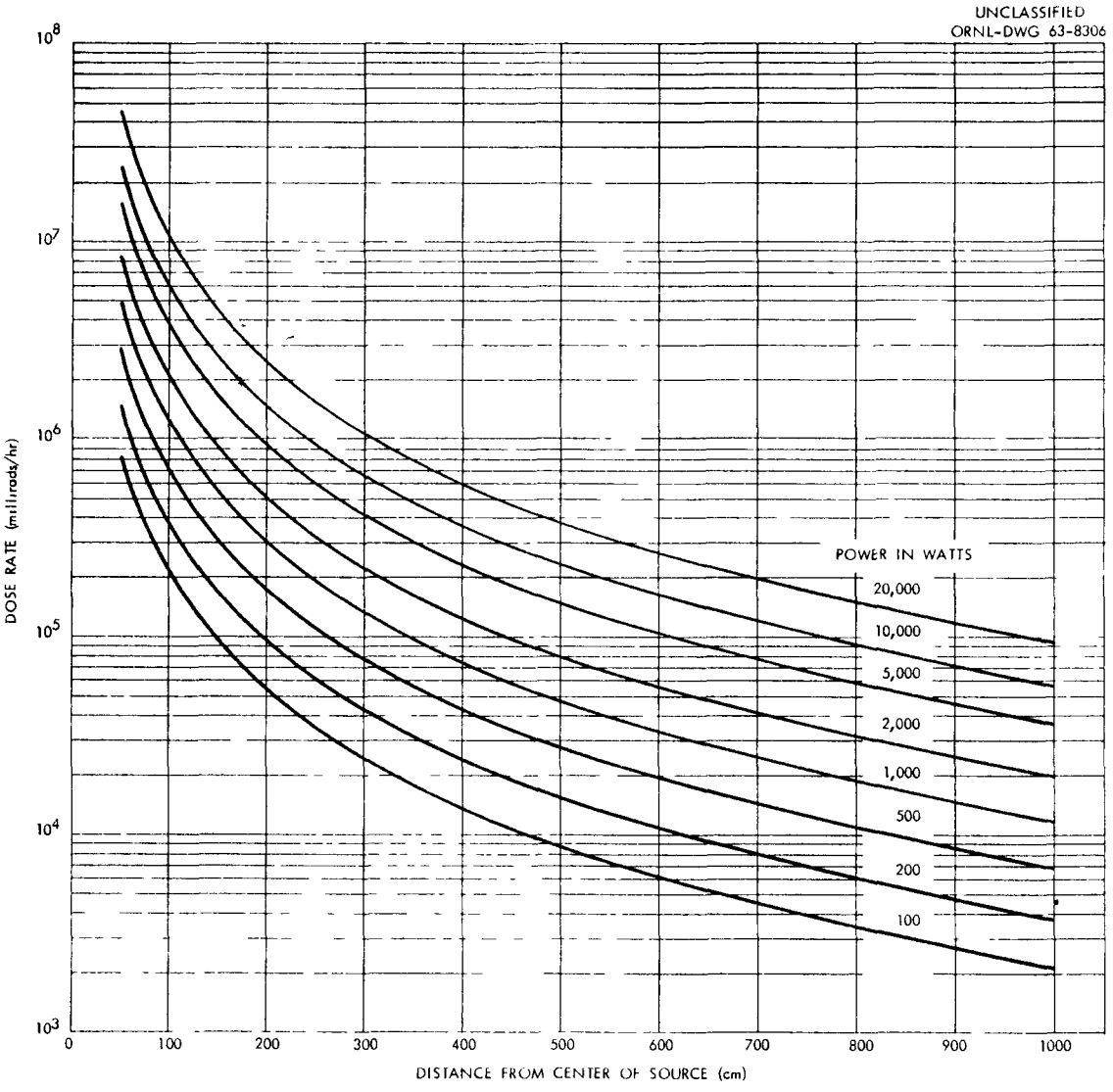


Fig. 59. Gamma Plus Bremsstrahlung Dose Rates from Unshielded Isotopic Power Sources of Thulium-170 as a Function of Distance from Center of Source.

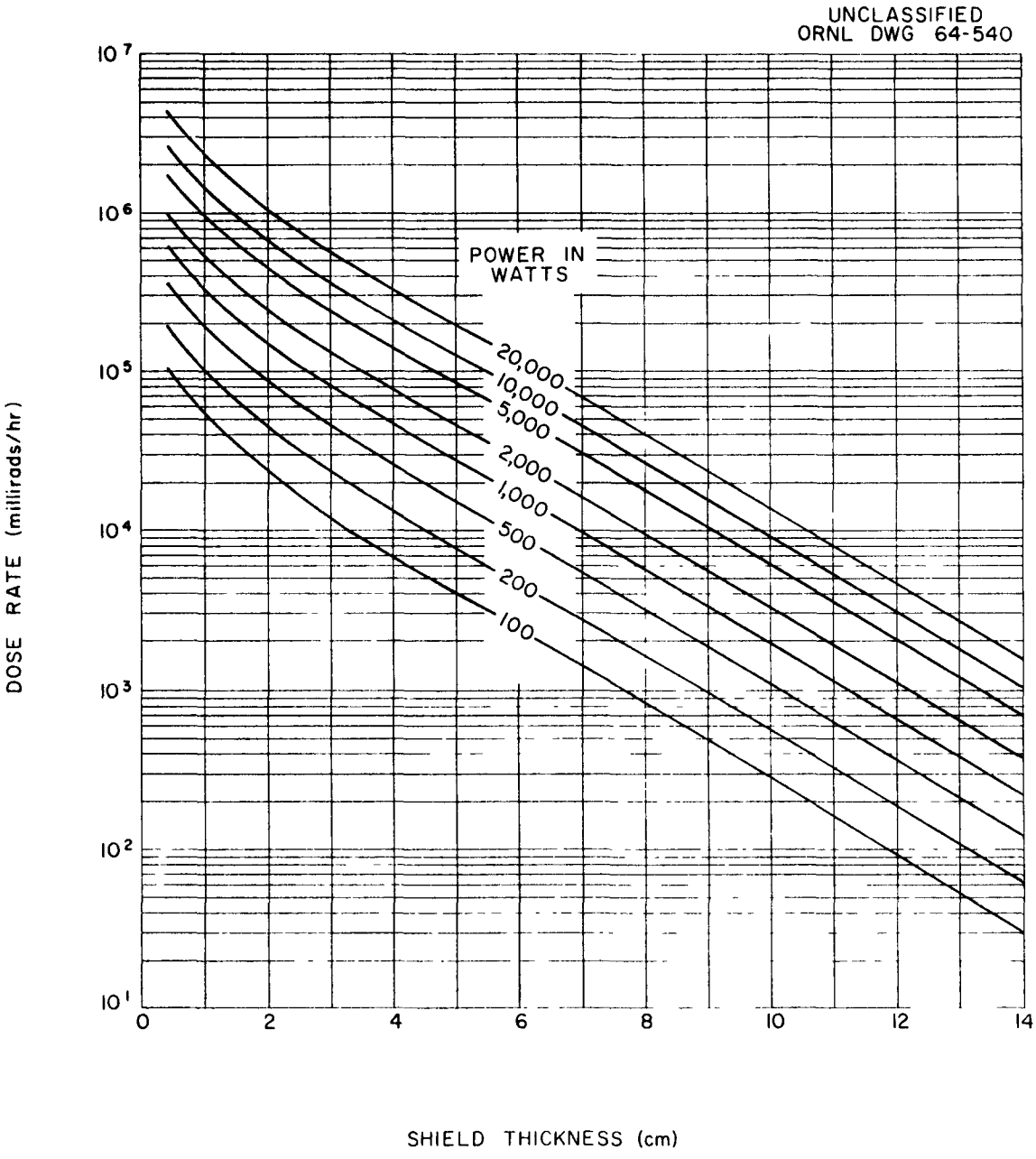


Fig. 60. Gamma Plus Bremsstrahlung Dose Rates from Iron-Shielded Isotopic Power Sources of Thulium-170. Center of source to dose point separation distance = 100 cm.

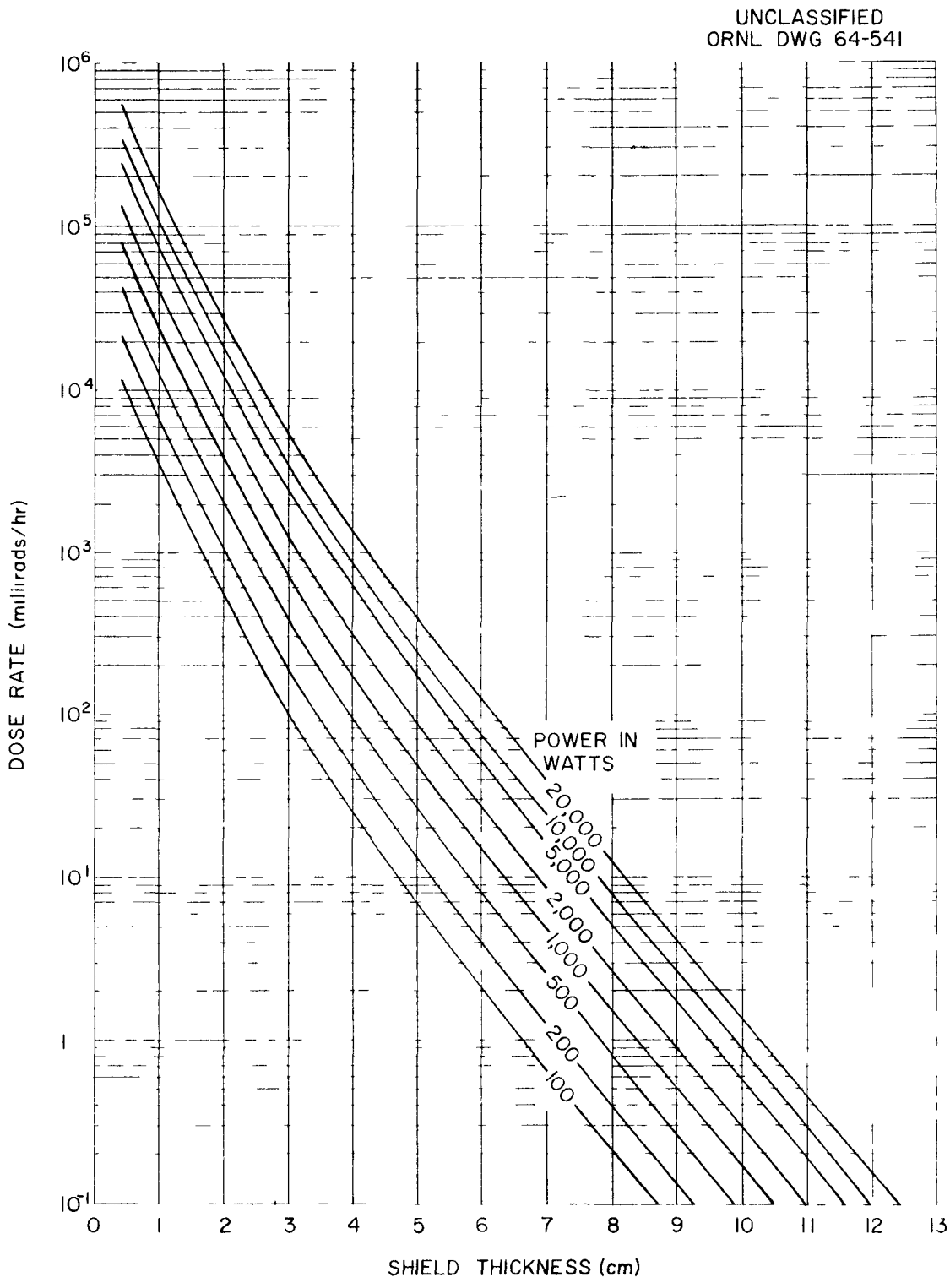


Fig. 61. Gamma Plus Bremsstrahlung Dose Rate from Lead-Shielded Isotopic Power Sources of Thulium-170. Center of source to dose point separation distance = 100 cm.

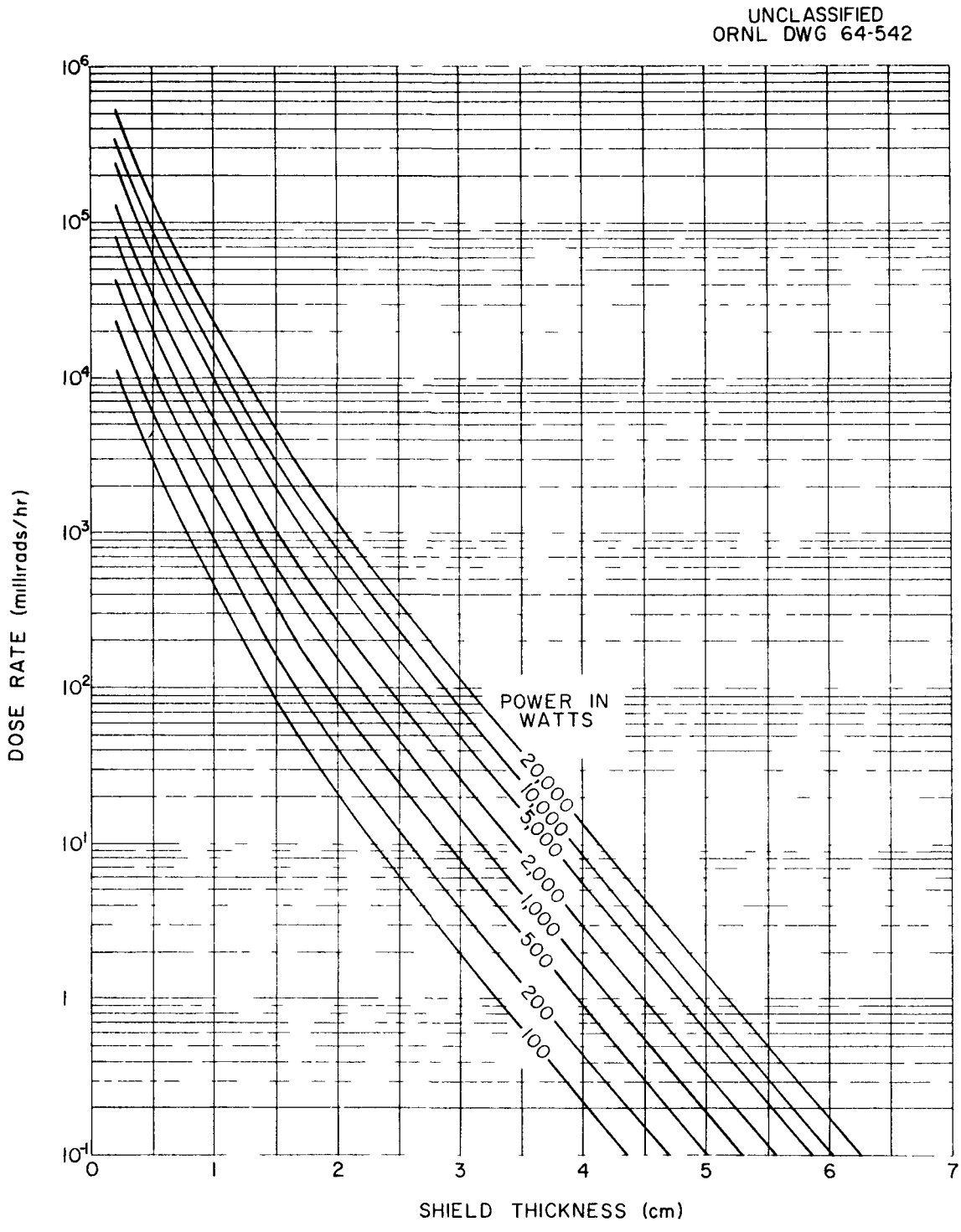


Fig. 62. Gamma Plus Bremsstrahlung Dose Rates from Uranium-Shielded Isotopic Power Sources of Thulium-170. Center of source to dose point separation distance = 100 cm.

Thulium-171

Thulium-171 has a favorable half-life (1.94 yrs) but a very low specific power - approximately 0.2 w/g. However, due to target availability and irradiation costs, the energy cost (dollars per watt) is greater for Tm^{171} than for any of the other isotopes considered. There may be some application for this isotope since very little shielding is required to attenuate its 0.067-Mev gamma. Thulium-171 requires less shielding per unit of power than any other power source considered here. It is produced by the neutron irradiation of Er^{170} according to the reaction:

$$Er^{170} \xrightarrow[9b]{n,\gamma} Er^{171} \xrightarrow[7.5h]{\beta^-} Tm^{171}.$$

Erbium-170 constitutes 14.88% of natural erbium. Erbium-168 (cross section, 2 barns) would also capture neutrons to form stable Tm^{169}, which would contaminate the final separated thulium product. Unless isotopically separated Er^{170} was irradiated, the thulium product would be less than 75% Tm^{171}. The calculations were based on pure Tm^{171} oxide with a specific power of 1.5 w/cc. The radiation intensities would be overestimated by about 25% by making this assumption.

Like Tm^{170}, the biological hazard is quite low.

Isotopic heat source data for Tm^{171} sources appear in Table 15, and the nuclear data appear in Table 3. The results of radiation-intensity calculations for unshielded sources are plotted in Fig. 63, and the results for iron-shielded sources are plotted in Fig. 64. The results for lead- and uranium-shielded sources could not be plotted because of the extremely high attenuation coefficient for these materials at 0.1 Mev. Shielding requirements for these two materials can best be estimated by using the relationship that 1 cm of iron is equivalent to 0.039 cm of lead and 0.137 cm of uranium for 0.1-Mev gamma rays. Lead is actually a better shield material than uranium for these very low energy gamma rays.

Table 15. Physical Properties of Thulium-171 as an Isotopic Power Source

General Properties	
Half-life	1.94 y
Specific activity of pure isotope, curies/g	1150
Specific power of pure isotope, w/g	0.205
Compound	Tm_2O_3
Weight of active isotope in compound, g/cc	7.52
Effective density of compound, g/cc	8.6
Power, w/cc of compound	1.54
Activity, curies/cc of compound	8648

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
2.325×10^{11}	0.1[b]
1.86×10^{12}	0.1[c,e]

[b] Bremsstrahlung x ray group.

[c] Actual gamma ray emission rate = 4.14×10^{12} at 0.067 Mev.

[e] Predominant radiation from unshielded 5000 watt source.

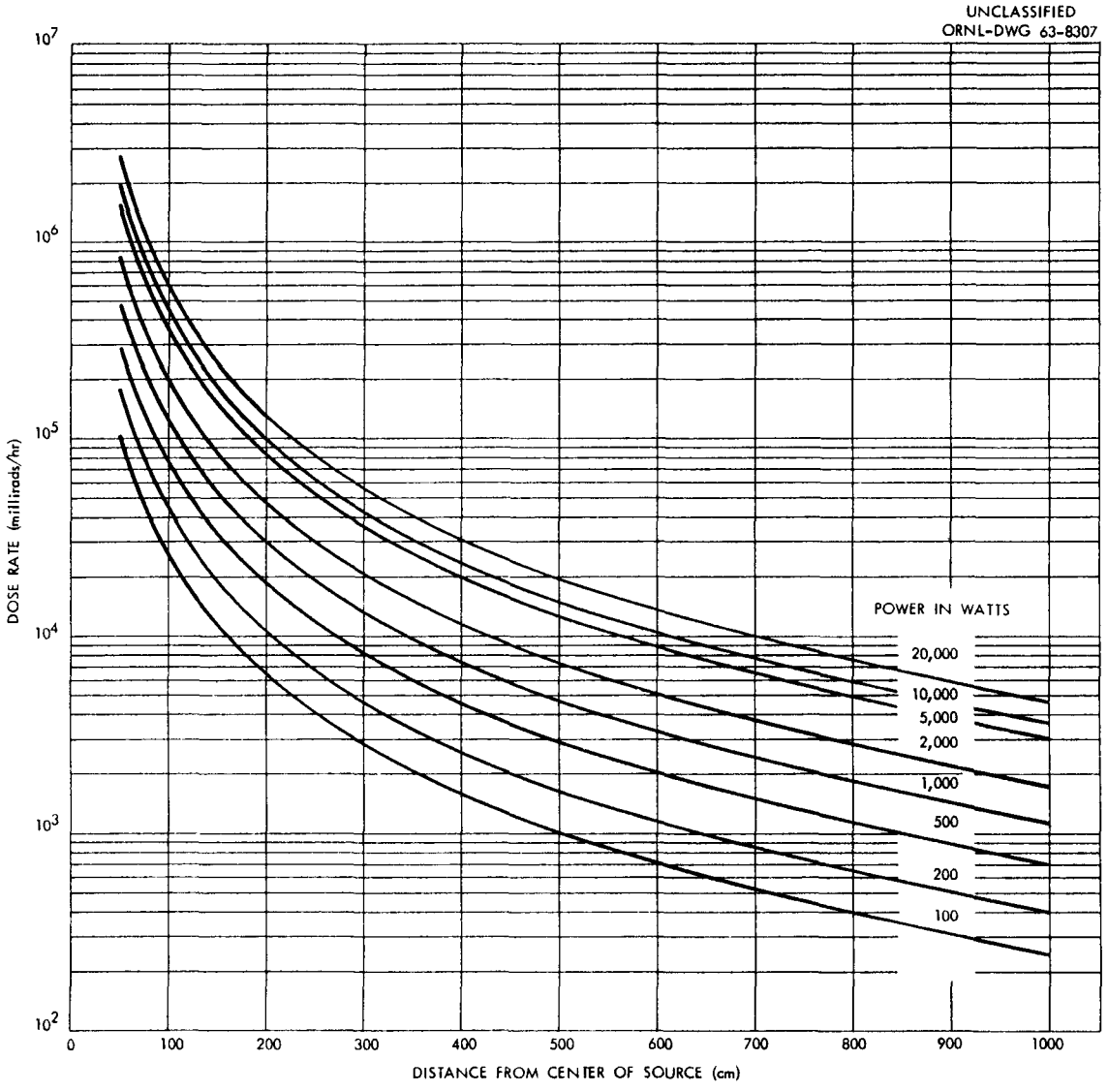


Fig. 63. Gamma Plus Bremsstrahlung Dose Rates from Unshielded Isotopic Power Sources of Thulium-171 as a Function of Distance from Center of Source.

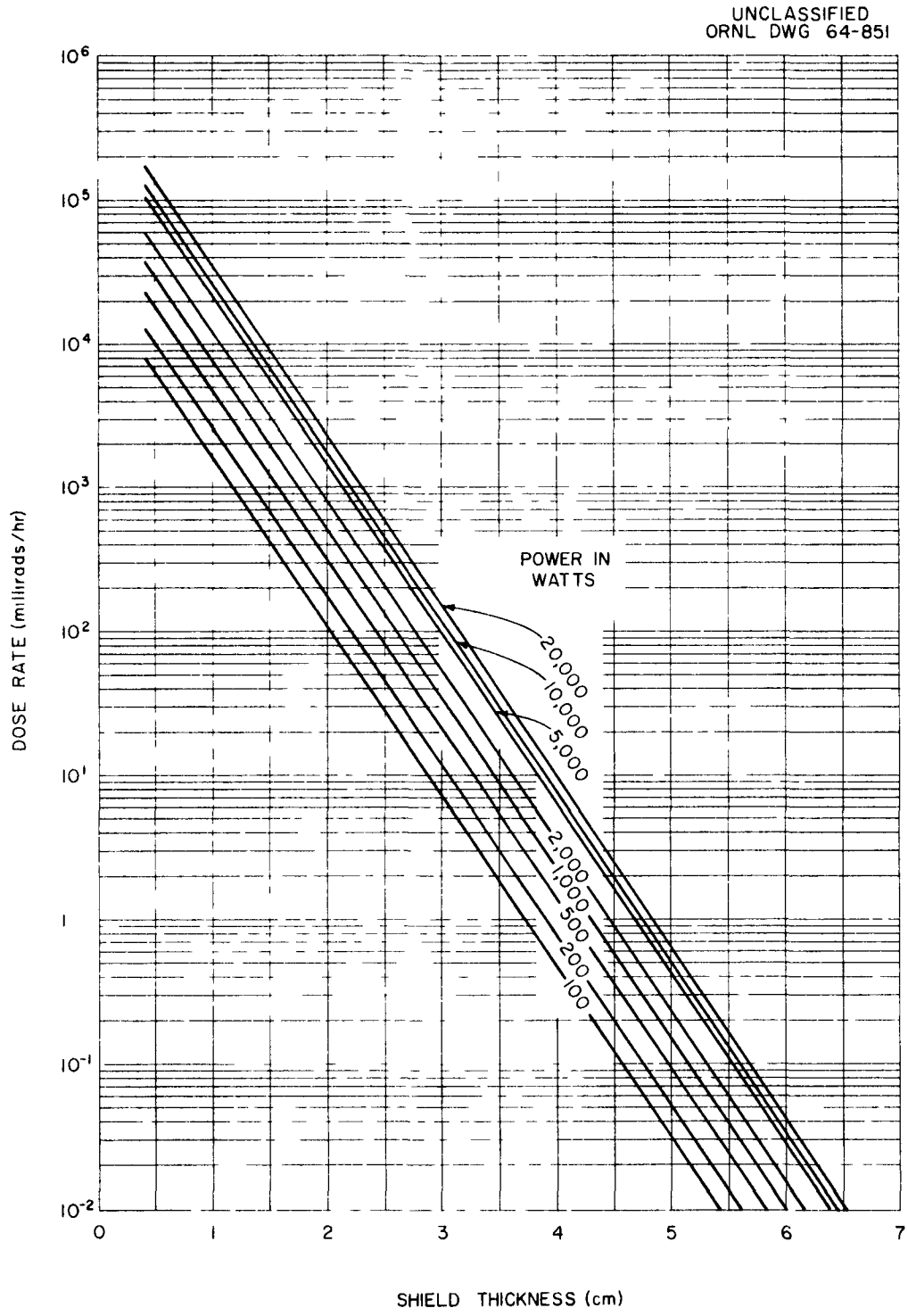


Fig. 64. Gamma Plus Bremsstrahlung Dose Rates from Iron-Shielded Isotopic Power Sources of Thulium-171. Center of source to dose point separation distance = 100 cm. Estimate required thickness of lead or uranium shield by using the following relationship: 1 cm of iron = 0.039 cm of lead and 0.137 cm of uranium for 0.1-Mev gamma rays.

Thallium-204

Thallium-204 ($T_{1/2}$ = 3.9 yrs) has very desirable properties as an isotopic power source. The specific power of the pure isotope is approximately 0.8 w/g, nearly all from beta decay. Shielding for bremsstrahlung is only moderate. The cross section and availability of the element favor the formation of Tl^{204}. However, to achieve a reasonable power density, isotopic separation of the radioactive Tl^{204} from Tl^{203} target material and inert Tl^{205} would be required. Isotopic separation does not appear to be feasible today. There is no known volatile compound of thallium, and therefore gaseous diffusion cannot be used. Electromigration might be possible, but the equilibrium time is very long.[27] The biological hazard is slightly less than that for Cs^{137}, Co^{60}, and Ce^{144}.

An irradiation at 2-3 x 10^{14} neutrons cm^{-2} sec^{-1} thermal neutron flux for 2 or 3 years would produce about 20 curies of Tl^{204} per gram of the normal mixture of thallium isotopes.

The calculations in this report were based on two different unit source strengths. These being 20 curies per gram of thallium metal (the as-irradiated condition) and 200 curies/g (an isotopically separated cut containing 45 wt % Tl^{204}). The specific powers would be 0.38 and 3.8 w/cc, respectively.

Isotopic heat source data for the two types of Tl^{204} sources appear in Table 16, and the nuclear data appear in Table 3. The results of the radiation intensity calculations for shielded and unshielded sources of 20 curies/g Tl^{204} are given in Figs. 65 through 68, while the radiation intensities for 200-curie-per-gram sources are given in Figs. 69 through 72.

Table 16. Physical Properties of Thallium-204 as an Isotopic Power Source

General Properties	
Half-life	3.9 y
Specific activity of pure isotope, curies/g	450
Specific power of pure isotope, w/g	0.729
Compound	metal
Weight of active isotope in metal, g/cc	0.524[a], 5.24[c]
Effective density of metal, g/cc	11.8
Power, w/cc of metal	0.382[a], 3.82[c]
Activity, curies/cc of metal	236[a], 2358[c]

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
4.57×10^{11}	0.071
7.87×10^{11}	0.2[b,e]
3.253×10^{10}	0.45[b,e]
3.394×10^{8}	0.625[b]

[a]Source produced from as irradiated natural thallium.

[b]Bremsstrahlung x ray group.

[c]Source produced from isotopically separated thallium.

[e]Predominant radiation from unshielded 5000 watt source, the two gammas are about equal in dose rate.

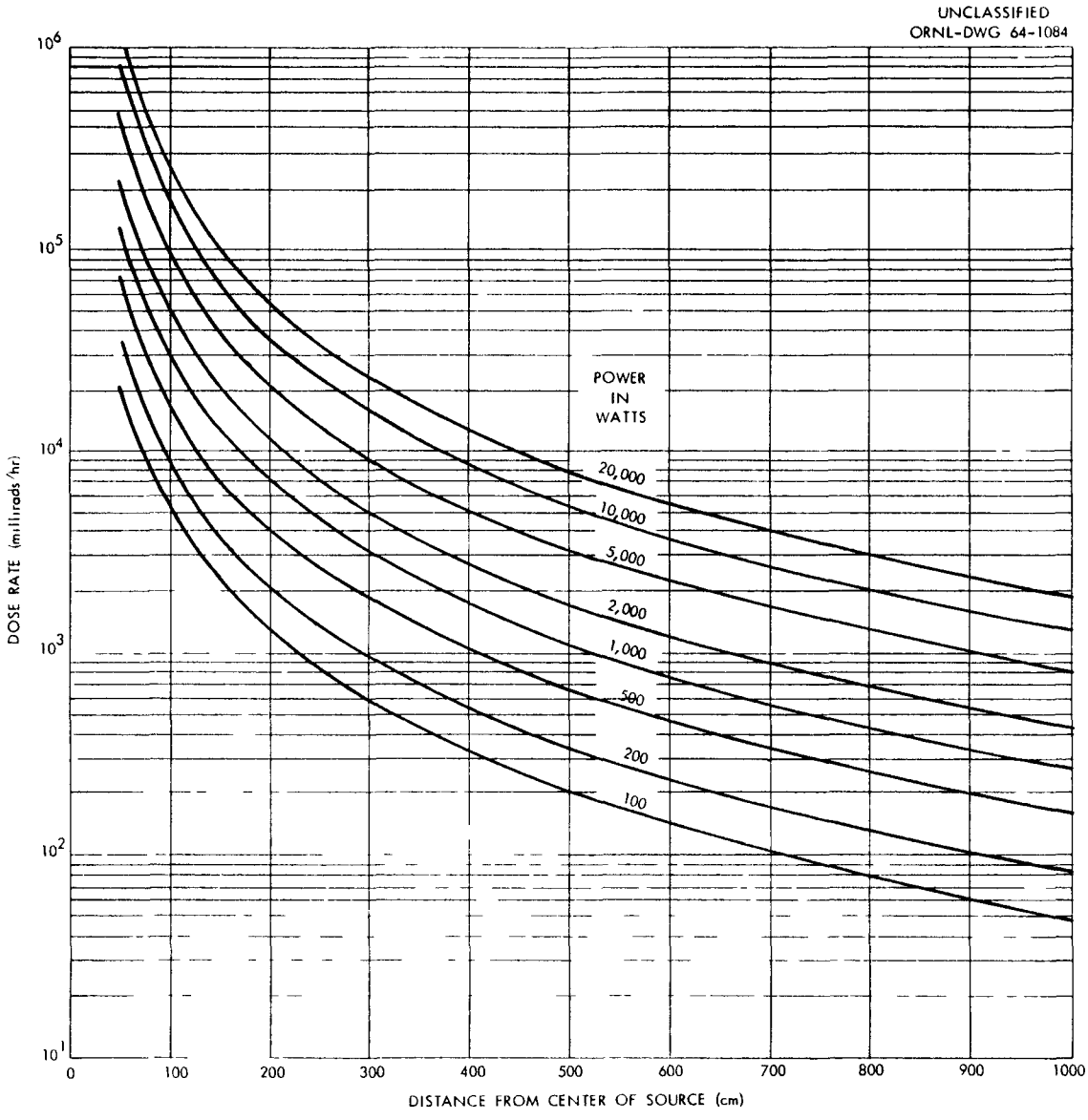


Fig. 65. Gamma Plus Bremsstrahlung Dose Rates from Unshielded Isotopic Power Sources of Irradiation Produced Thallium-204 as a Function of Distance from Center of Source. Source is assumed to contain 20 curies of Tl^{204} per gram of total thallium.

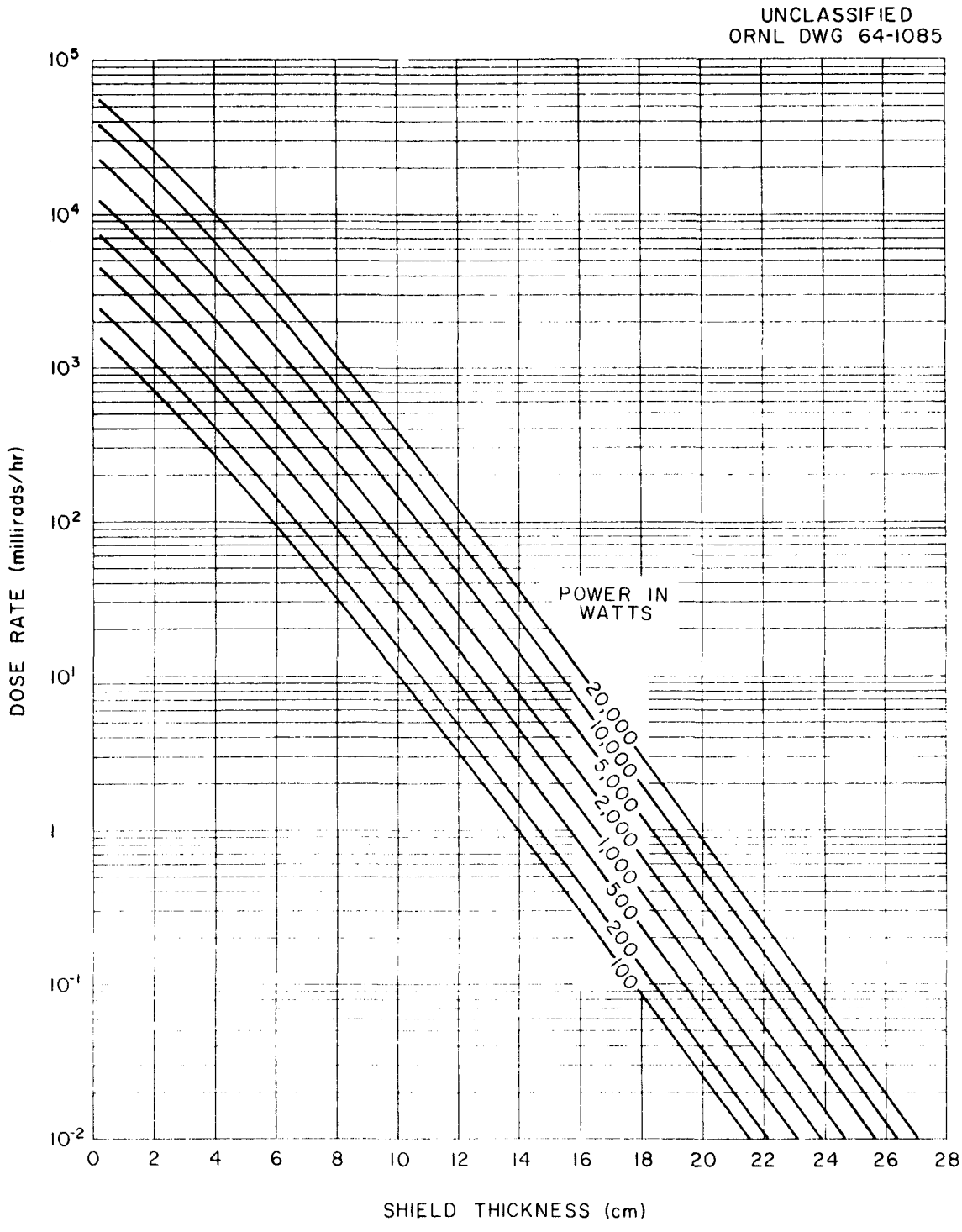


Fig. 66. Gamma Plus Bremsstrahlung Dose Rates from Iron-Shielded Isotopic Power Sources of Irradiation Produced Thallium-204. Center of source to dose point separation distance = 100 cm. Source is assumed to contain 20 curies of Tl^{204} per gram of total thallium.

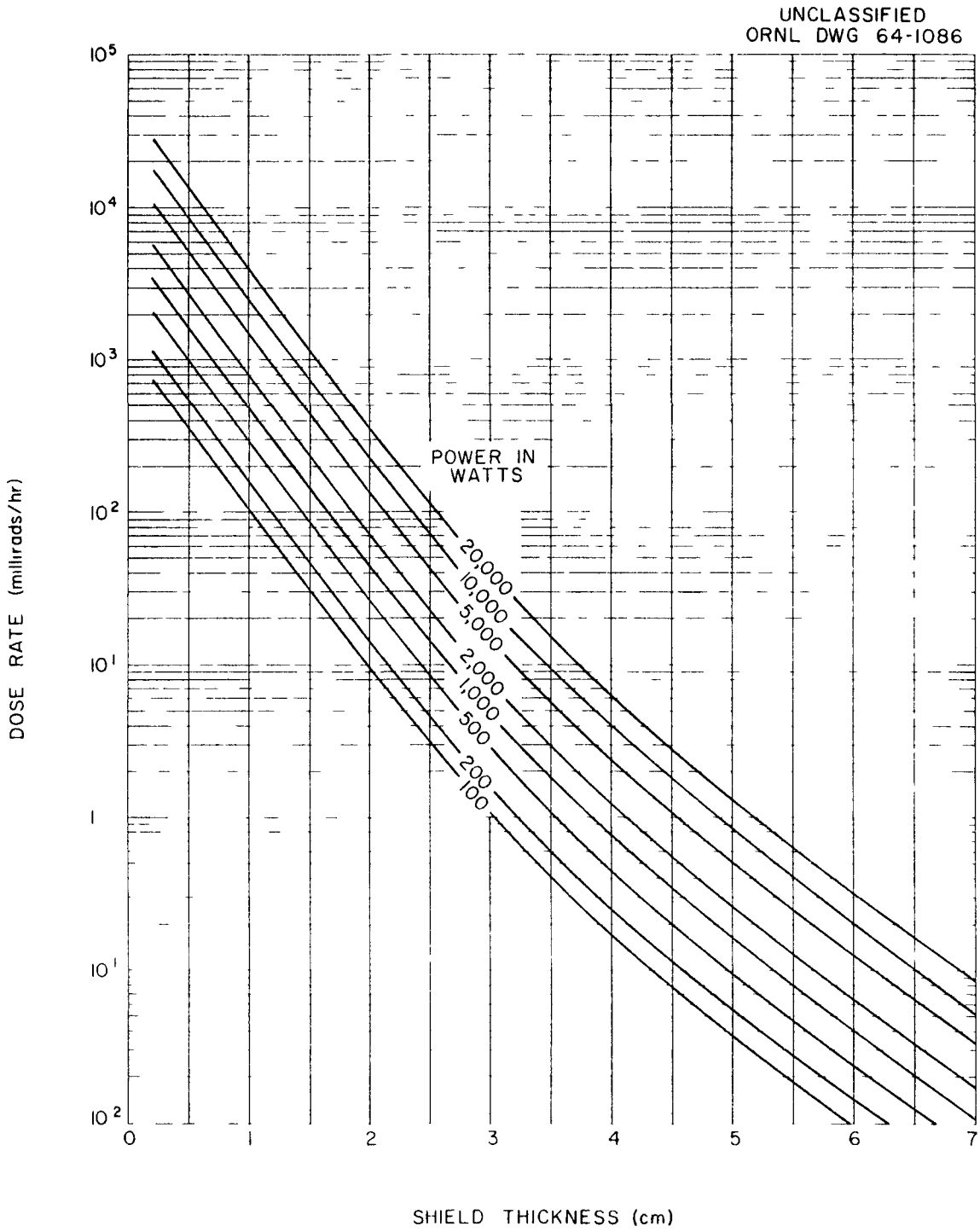


Fig. 67. Gamma Plus Bremsstrahlung Dose Rates from Lead-Shielded Isotopic Power Sources of Irradiation Produced Tl^{204}. Center of source to dose point separation distance = 100 cm. Source is assumed to contain 20 curies of Tl^{204} per gram of total thallium.

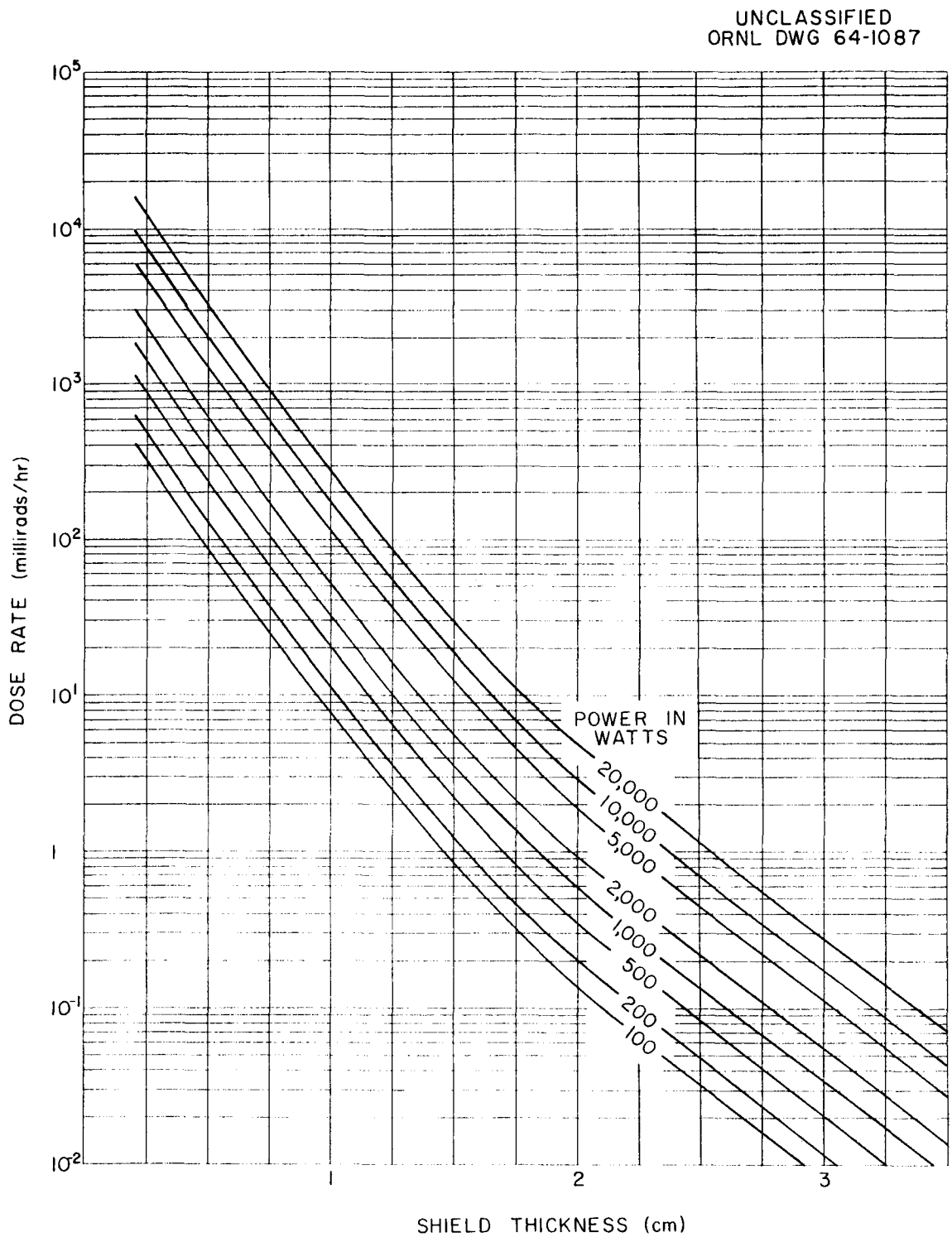


Fig. 68. Gamma Plus Bremsstrahlung Dose Rates from Uranium-Shielded Isotopic Power Sources of Irradiation Produced Thallium-204. Center of source to dose point separation distance = 100 cm. Source is assumed to contain 20 curies of Tl^{204} per gram of total thallium.

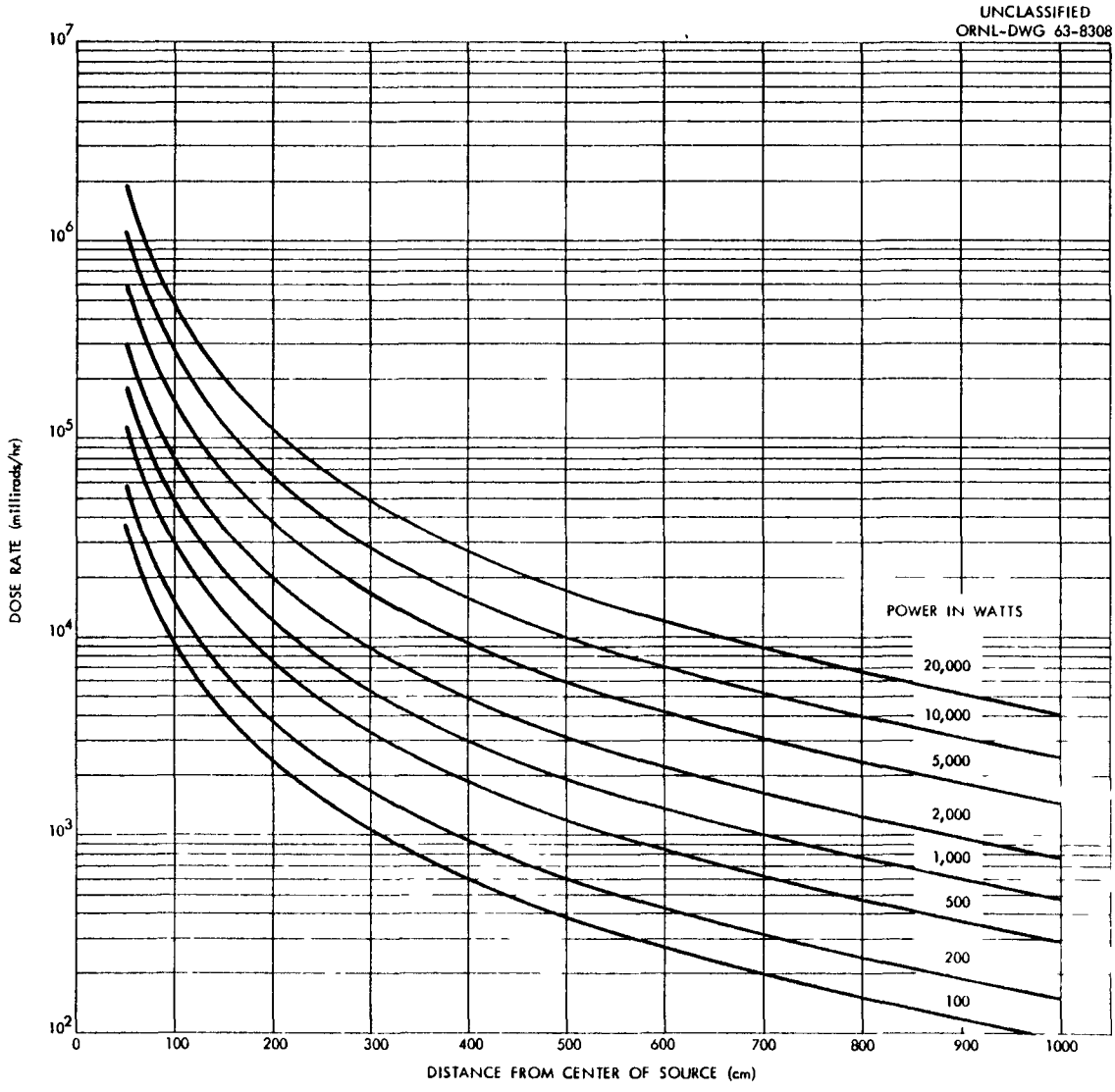


Fig. 69. Gamma Plus Bremsstrahlung Dose Rates from Unshielded Isotopic Power Sources of Isotopically Separated Thallium-204 as a Function of Distance from Center of Source. Source is assumed to be an isotopically separated cut from irradiated natural thallium. This cut is assumed to contain 200 curies of Tl^{204} per gram of total thallium (about 45 wt % Tl^{204}).

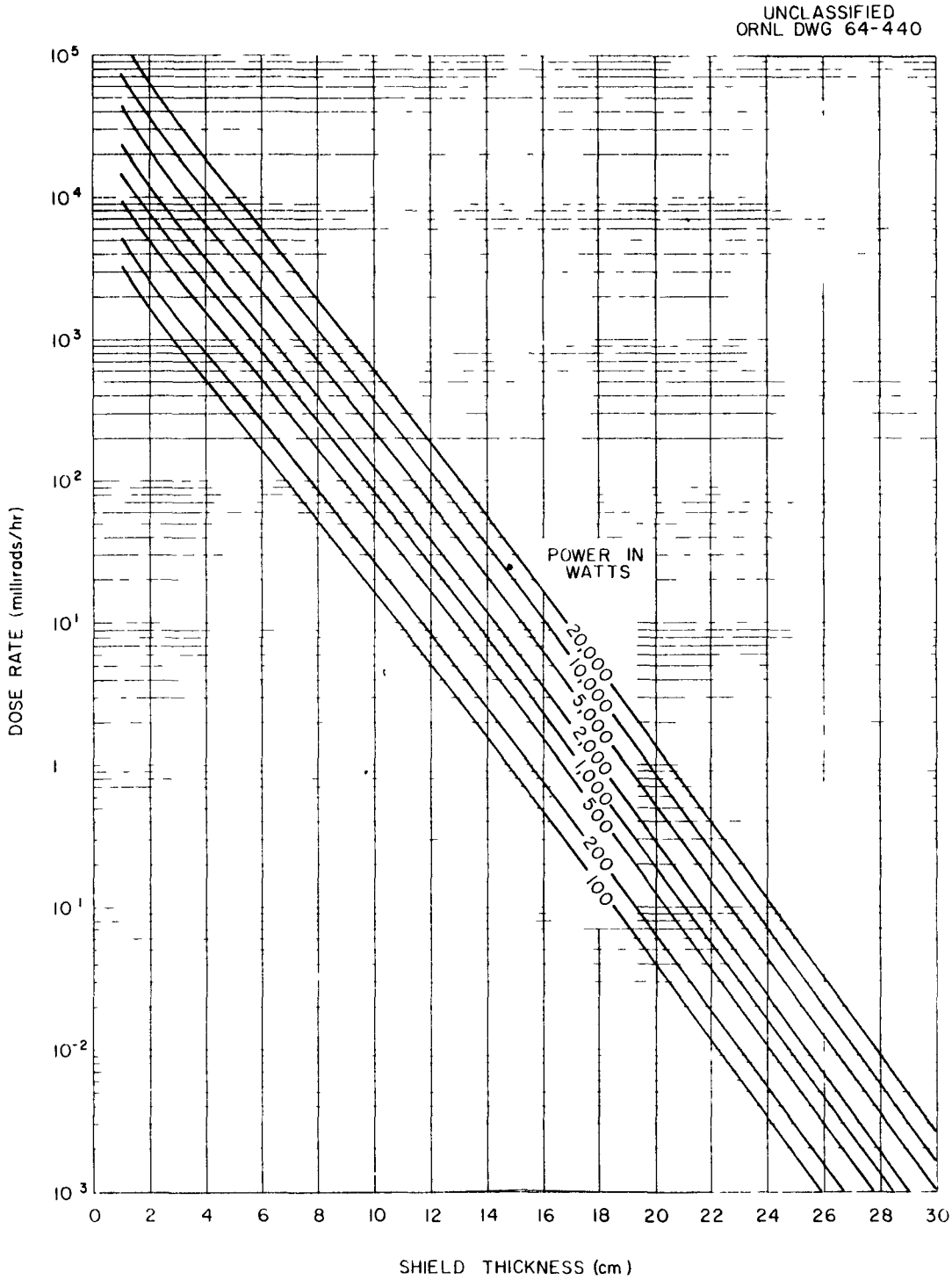


Fig. 70. Gamma Plus Bremsstrahlung Dose Rates from Iron-Shielded Isotopic Power Sources of Isotopically Separated Thallium-204. Center of source to dose point separation distance = 100 cm. Source is assumed to be an isotopically separated cut from irradiated natural thallium. This cut is assumed to contain 200 curies of Tl^{204} per gram of total thallium (about 45 wt % Tl^{204}).

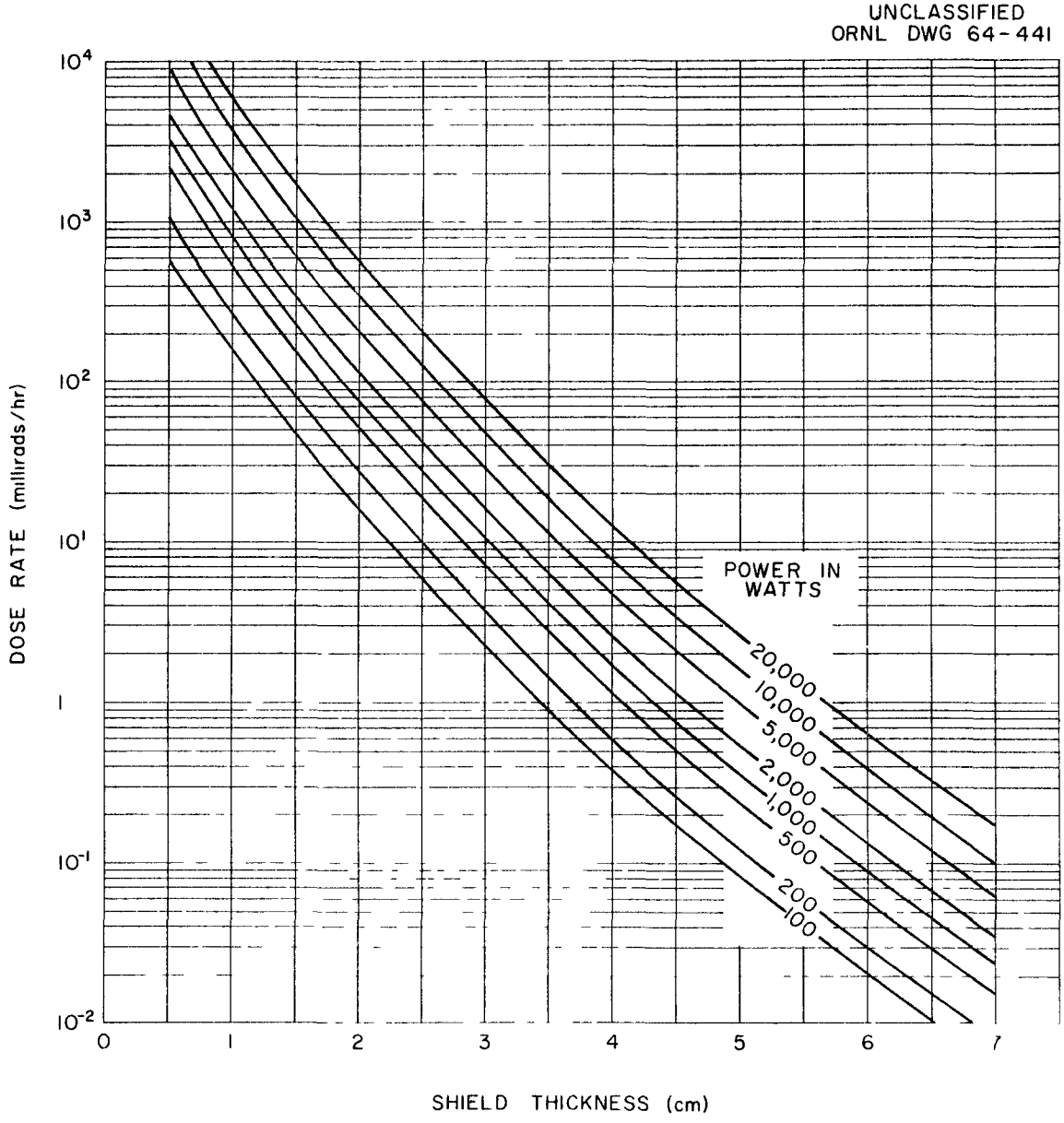


Fig. 71. Gamma Plus Bremsstrahlung Dose Rates from Lead-Shielded Isotopic Power Sources of Isotopically Separated Thallium-204. Center of source to dose point separation distance = 100 cm. Source is assumed to be an isotopically separated cut from irradiated natural thallium. This cut is assumed to contain 200 curies of Tl^{204} per gram of total thallium (about 45 wt % Tl^{204}).

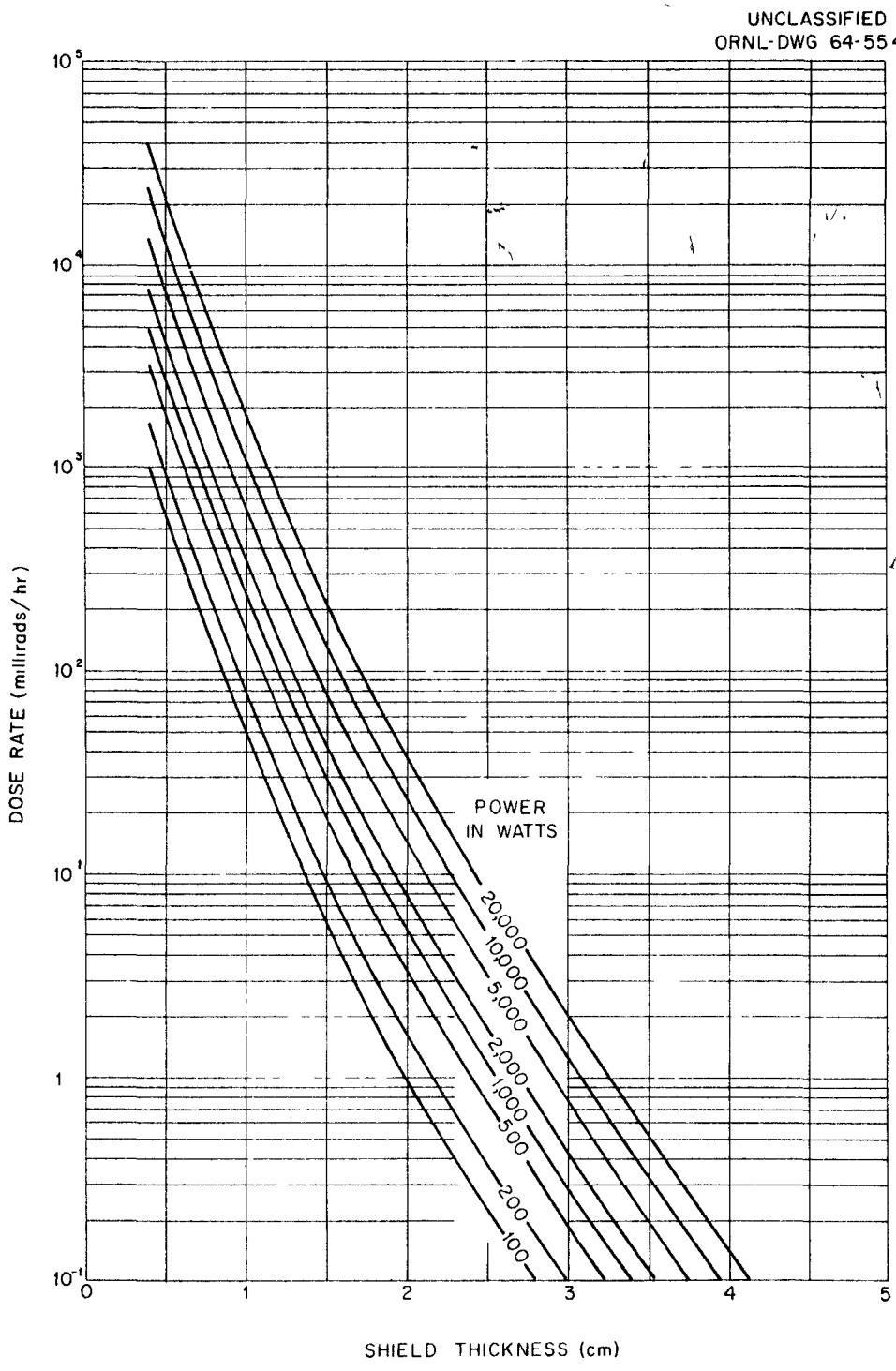


Fig. 72. Gamma Plus Bremsstrahlung Dose Rates from Uranium-Shielded Isotopic Power Sources of Isotopically Separated Thallium-204. Center of source to dose point separation distance = 100 cm. Source is assumed to be an isotopically separated cut from irradiated natural thallium. This cut is assumed to contain 200 curies of Tl^{204} per gram of total thallium (about 45 wt % Tl^{204}).

Polonium-210

Polonium-210, a very rare element occurring in the decay chain of U^{238}, and also producible by the irradiation of bismuth was the first radioisotope used to demonstrate the "SNAP III," a prototype power device, early in 1959.[2] Its specific power is a very high 140 w/g. However, its half-life is only 138 days. It is an alpha emitter with negligible (accompanying 1.2×10^{-3}% of the alpha disintegrations) gamma activity such that very little shielding is needed. Compared with other alpha emitters its biological hazard is low. It can be prepared with high purity. Production in significant quantities appears feasible, and therefore Po^{210} is likely to have its most important uses in manned space applications where low radiation levels and light-weight shields are necessary.

The calculations in this study were based on dilution of the polonium metal with an inert metal matrix to achieve a reasonable specific power of 75 w/cc for the actual power source. The actual source is assumed to be in the form of a pellet at a power density of 150 w/cc with an equal volume of void space for gas collection.

Isotopic heat source data for Po^{210} sources appear in Table 17, and the nuclear data appear in Table 3. The results of the gamma radiation intensity calculations for shielded and unshielded sources are given in Figs. 73 through 77. In addition, Fig. 78 shows the neutron intensities from water-shielded sources. Figure 2 should be used in conjunction with this figure in order to get neutron radiation intensities when polonium is shielded by other materials.

Table 17. Physical Properties of Polonium-210 as an Isotopic Power Source

General Properties	
Half-life	138.4 d
Specific activity of pure isotope, curies/g	4500
Specific power of pure isotope, w/g	141.3
α,n rate, neutrons/g-sec	4.51×10^5
Compound	metal matrix
Weight of active isotope in matrix, g/cc	0.53
Effective density of matrix, g/cc	4.25
Power, w/cc of matrix	75
Activity, curies/cc of matrix	2388

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
1.415×10^7	0.8

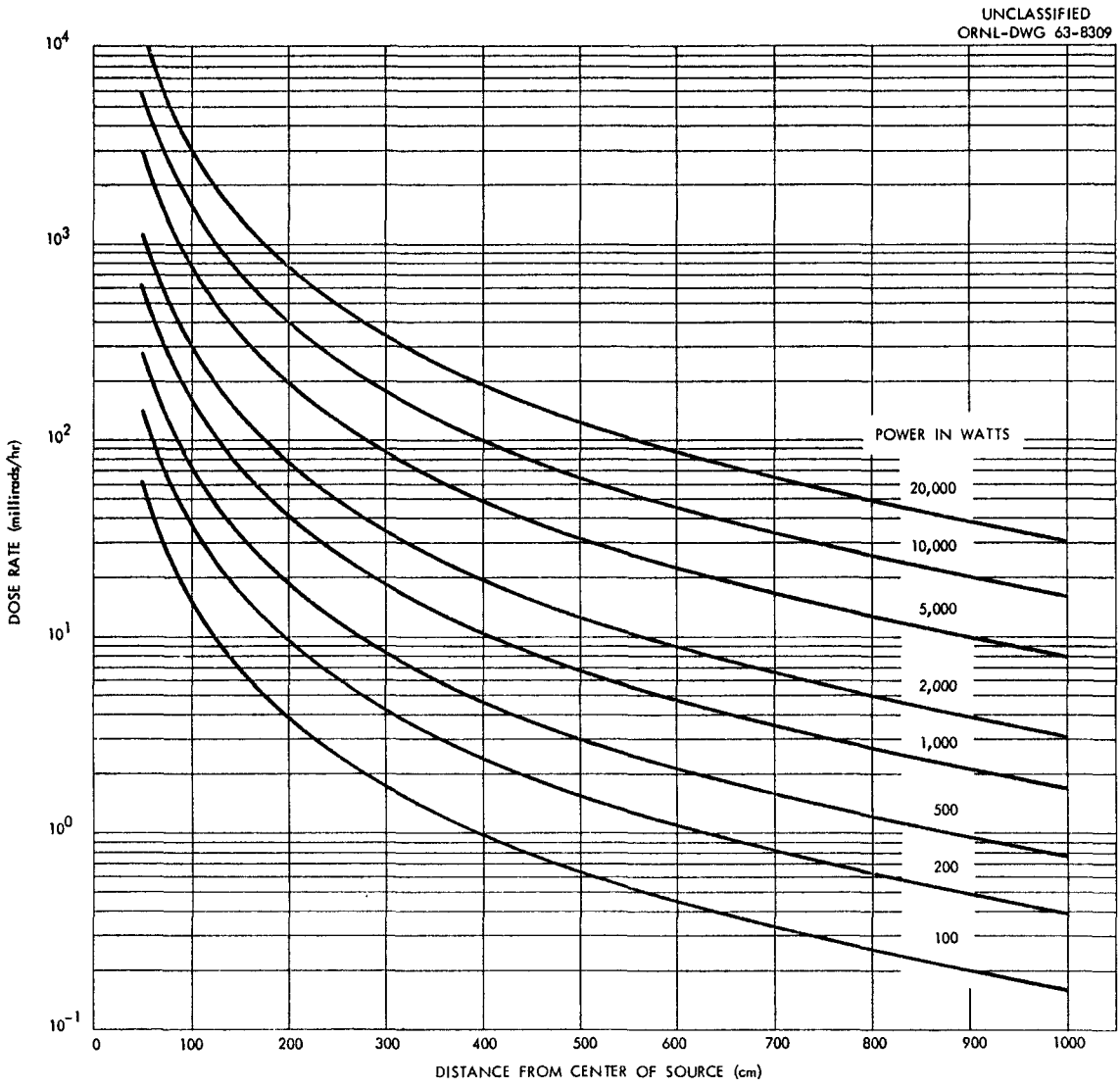


Fig. 73. Gamma Dose Rates from Unshielded Isotopic Power Sources of Polonium-210 as a Function of Distance from Center of Source.

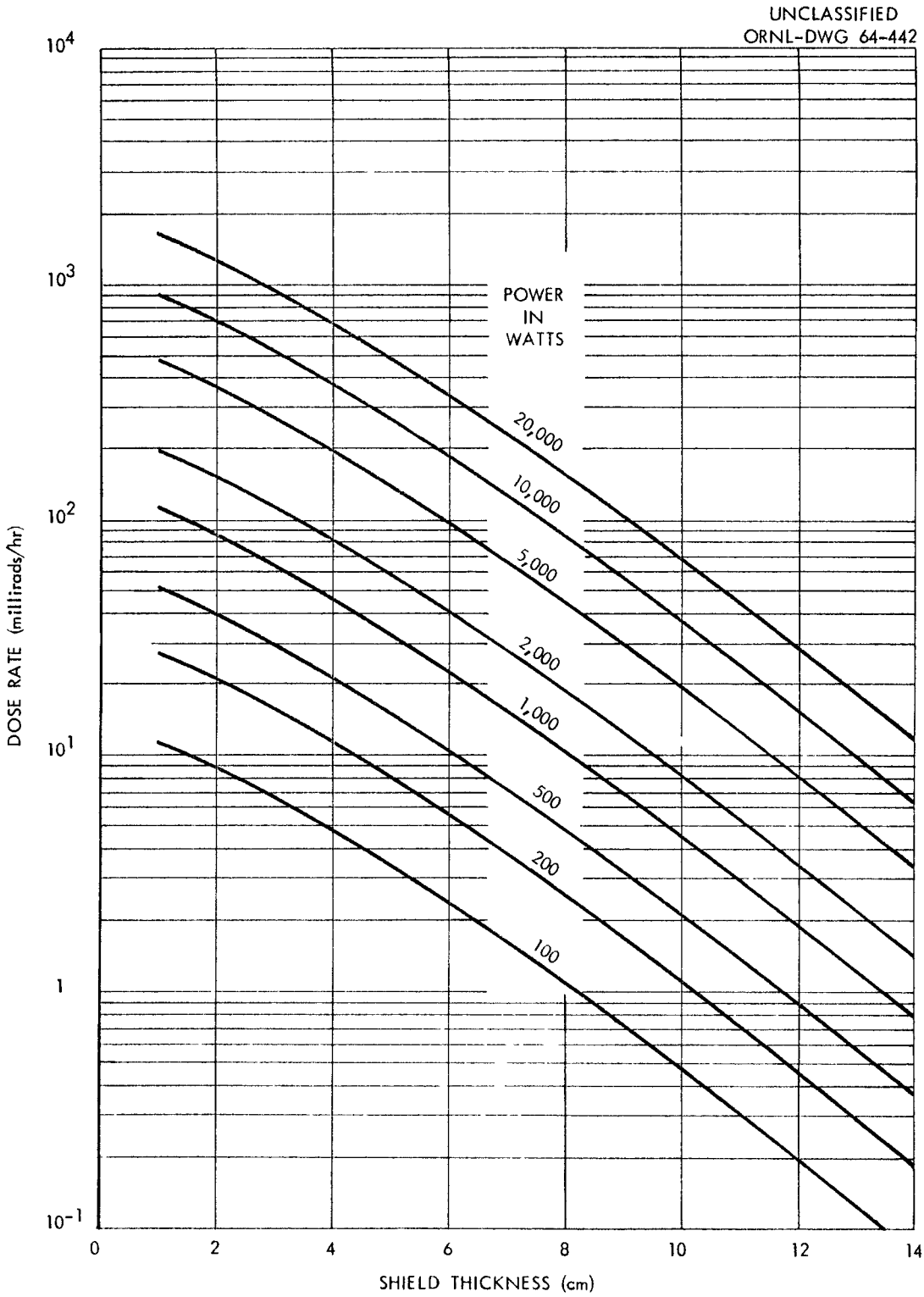


Fig. 74. Gamma Dose Rates from Iron-Shielded Isotopic Power Sources of Polonium-210. Center of source to dose point separation distance = 100 cm.

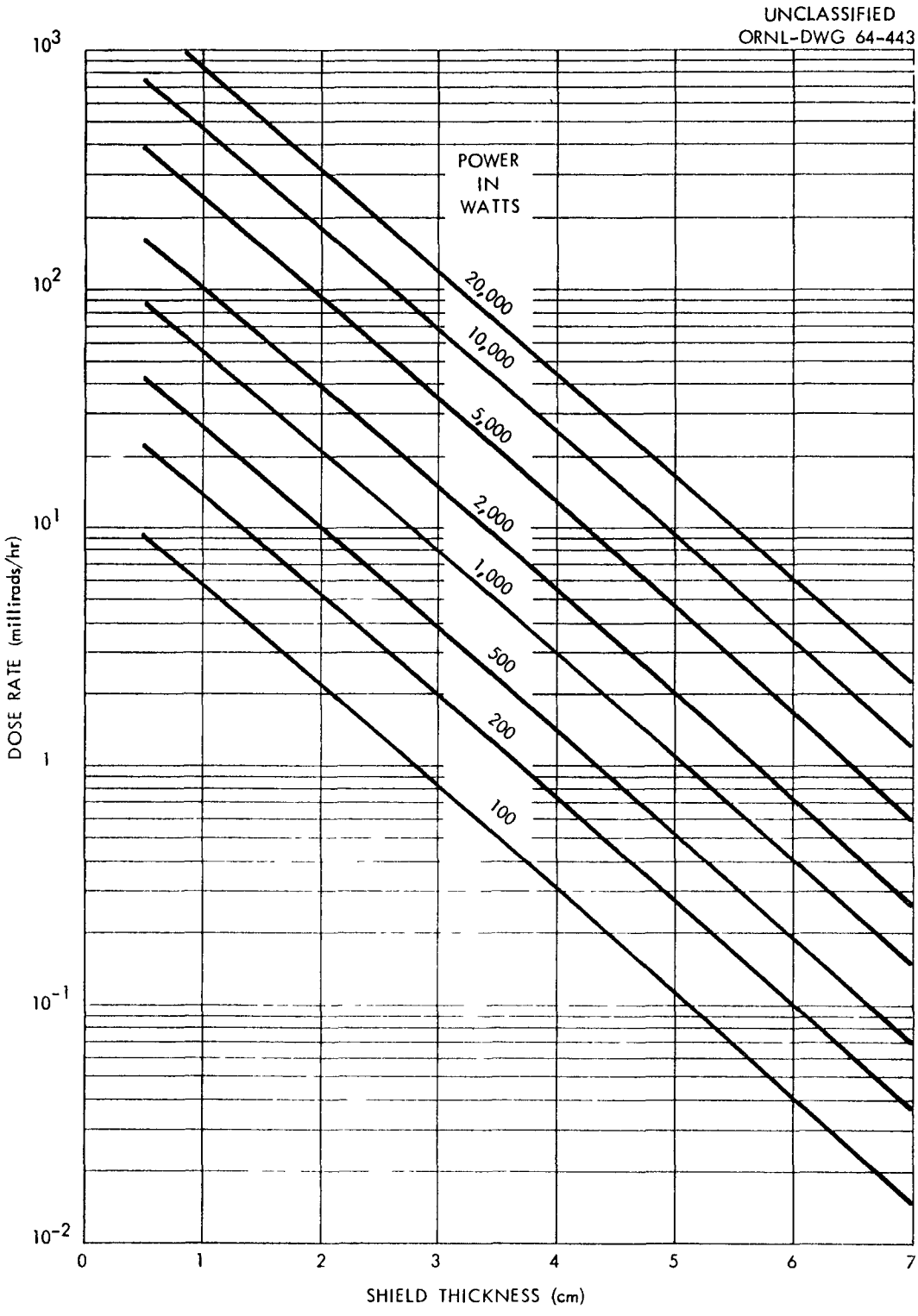


Fig. 75. Gamma Dose Rates from Lead-Shielded Isotopic Power Sources of Polonium-210. Center of source to dose point separation distance = 100 cm.

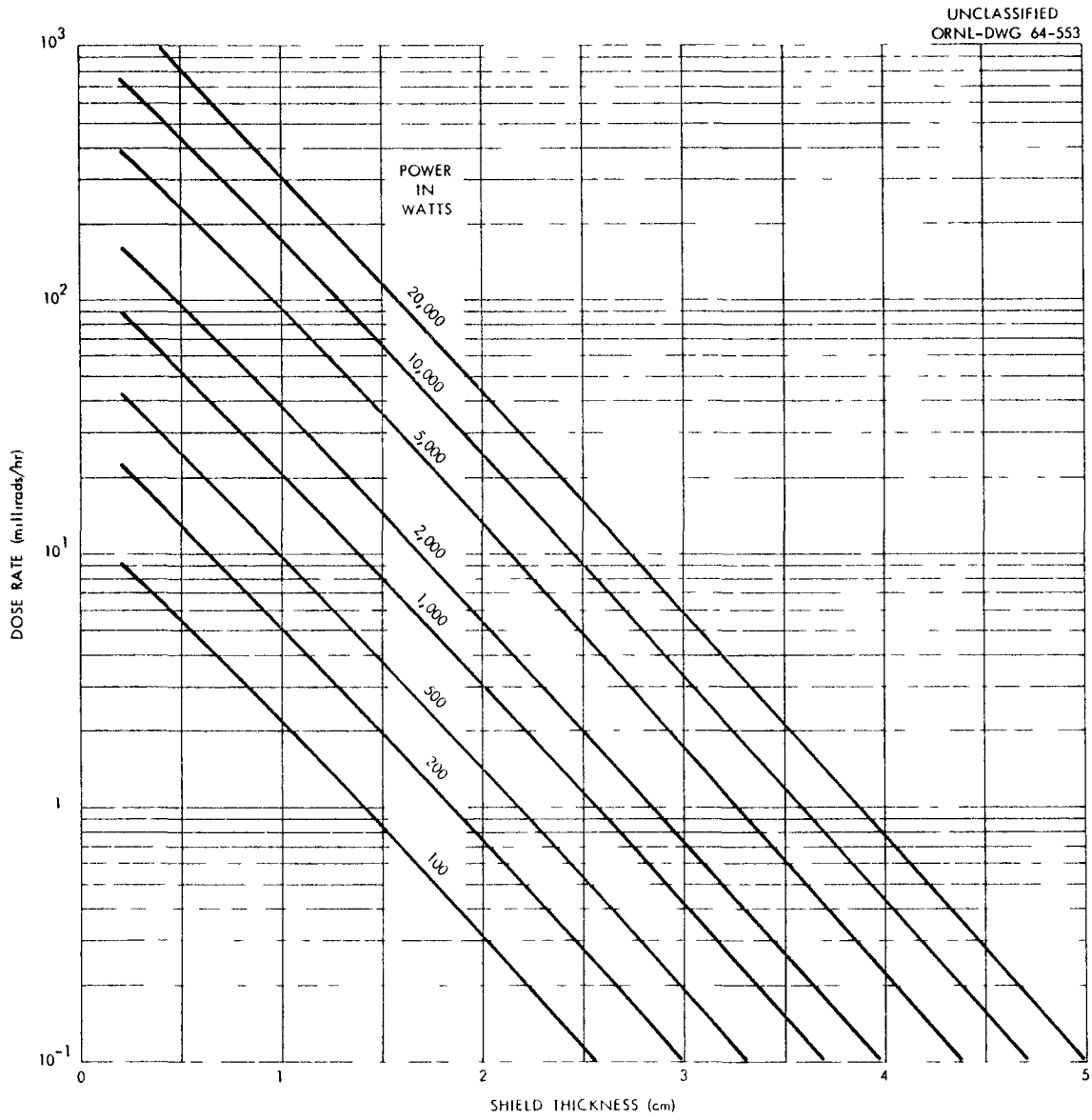


Fig. 76. Gamma Dose Rates from Uranium-Shielded Isotopic Power Sources of Polonium-210. Center of source to dose point separation distance = 100 cm.

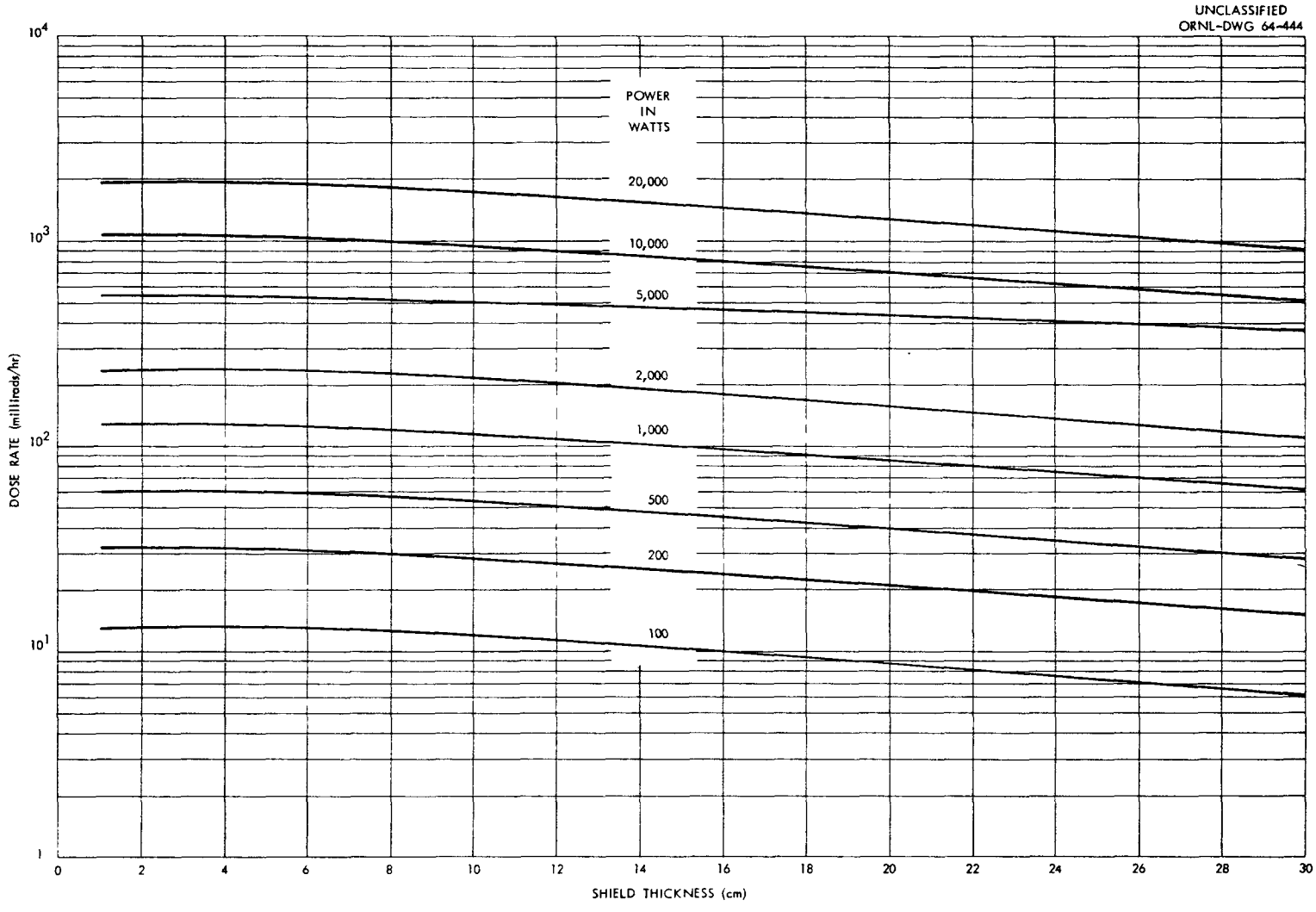


Fig. 77. Gamma Dose Rates from Water-Shielded Isotopic Power Sources of Polonium-210. Center of source to dose point separation distance = 100 cm.

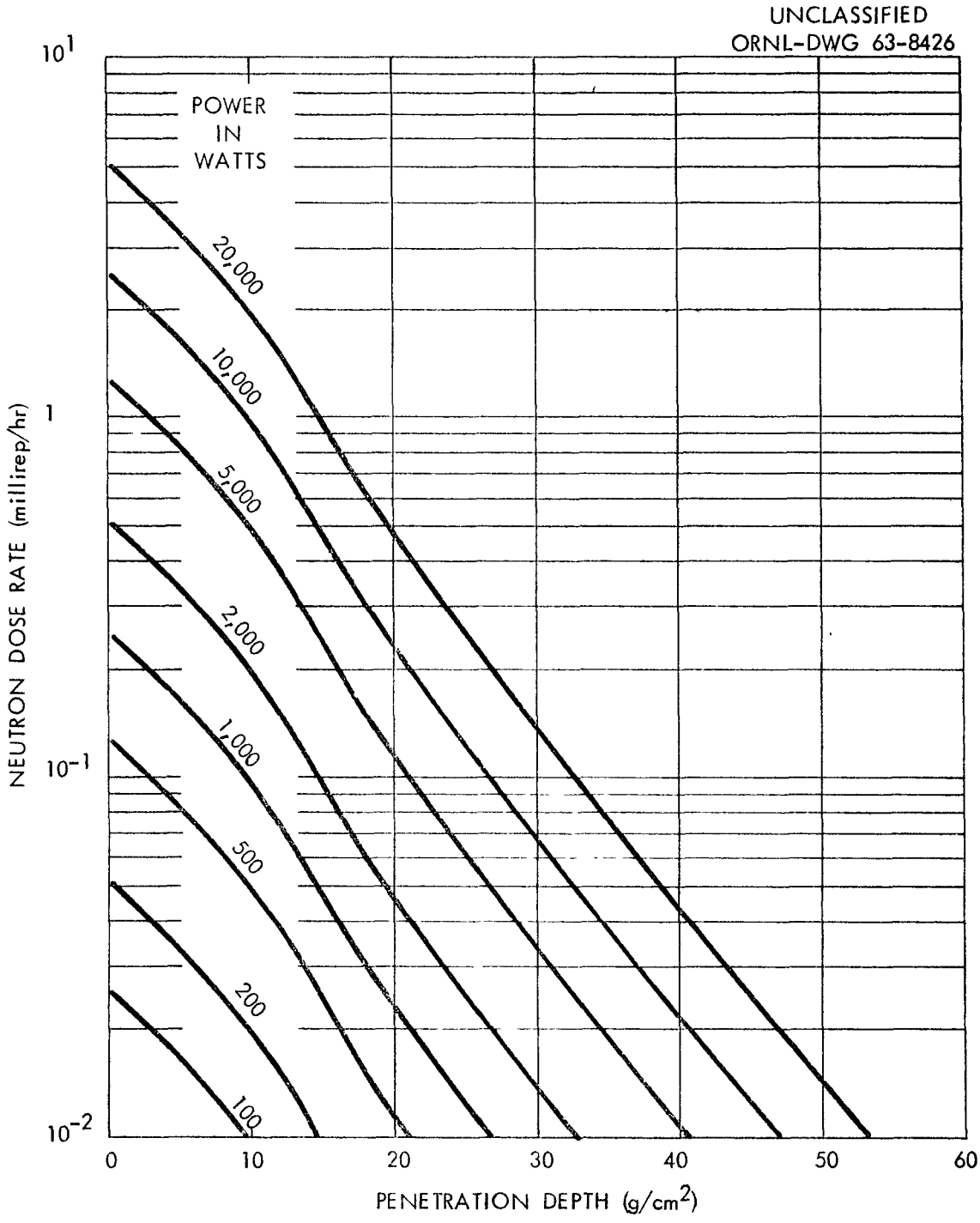


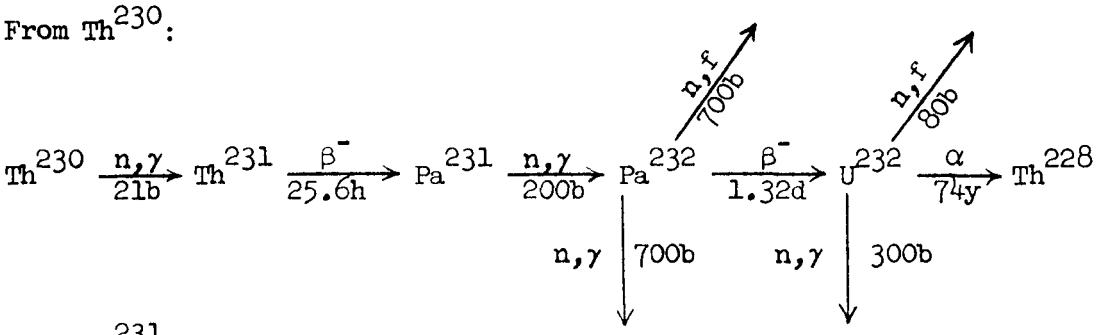
Fig. 78. Neutron Dose Rates from Water-Shielded Isotopic Power Sources of Polonium-210 as a Function of Penetration Depth of Shielding Material. Center of source to dose point separation distance = 100 cm. Refer to Fig. 2 to obtain dose rate through other materials.

Uranium-232

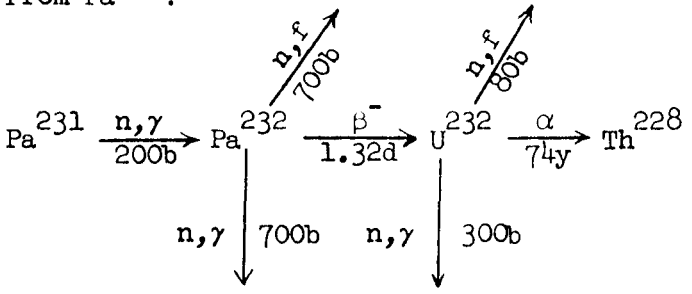
The technological and economic feasibility of the production of U^{232} and its daughter Th^{228} have been covered in the literature.[28,29,30] However, we believe that the feasibility of production on a large scale has not been fully established. Production of a gram of U^{232} via the irradiation of Pa^{231} was accomplished during 1962 at the Oak Ridge National Laboratory.[29] Source of thorium sufficiently high in the Th^{230} isotope to provide economical U^{232} manufacture without need for isotopic separation have been located.[28] Uranium-232 can be produced by: (1) the irradiation of Th^{230}, a long-lived isotope of thorium formed in uranium ores by decay of U^{238}; or (2) by irradiation of Pa^{231}, which is also found in uranium ores and formed by decay of U^{235}.

The two routes of production would be as follows:

1. From Th^{230}:

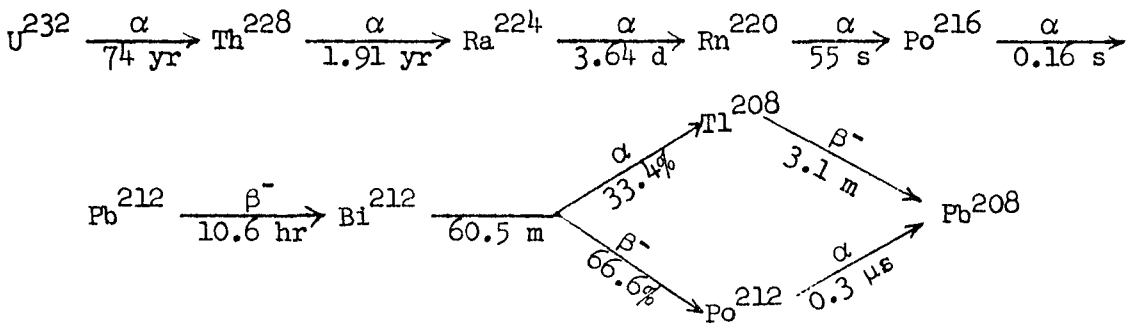


2. From Pa^{231}:



Uranium-232 ($T_{1/2}$ = 74 g) is characterized by its high specific power (4.9 w/g when daughters are in transient equilibrium) and very high power density (42.6 w/cc in the form of the dioxide). These high power values result from the unique decay of U^{232} which, in contrast with other radioisotopes, consists of a long chain of energetic alpha emitters, including a fairly high yield of its reasonably long-lived daughter, Th^{228}.

This decay chain is as follows:



In this chain the Tl^{208} is the source of intense and energetic gamma activity. This makes the shielding requirements per watt of thermal energy almost identical with that for Ce^{144}. (Compare Figs. 41 and 87 for example.) Another peculiarity of U^{232} relates to the effect of its first decay product, its daughter Th^{228}. This daughter is the longest-lived member in the decay chain. Thus, U^{232} does not reach its maximum specific power until saturation with the growing in of the daughter takes place. For pure U^{232}, this is not reached fully until about 8 yrs after separation. If, however, a stock of U^{232} is at hand from which Th^{228} can be "milked," and if this Th^{228} is then used to "activate" the U^{232}, heat output near the maximum for U^{232} can be achieved at once and retained near maximum for over 30 yr, with very small amounts of Th^{228} being required.[2] This is indeed a unique characteristic and one which may be extremely useful in applications which require a very long period of nearly uniform heat evolution. This characteristic is specific to U^{232} "activated" in this way and is probably not duplicated by any other isotope or combination. The spontaneous fission half-life for U^{232} is long, 8×10^{13} yr. However, α,n reactions on light elements which exist as impurities or in uranium compounds will produce many more neutrons than spontaneous fission.

The biological hazard is somewhat less than that for Pu^{238} but still substantial compared with that for some of the other promising power sources.

Dose rates from two types of U^{232} sources were calculated, these being: (1) 2-yr-aged material and (2) equilibrium or 9-yr-aged material. The 2-yr material was considered mainly for comparison to show the effect of little self absorption in the source itself since there is little difference in dose rates for any given power. However, the power per unit volume will increase with time over the period of 2 to 8 yr and thus this

source could be used only for a very special application where an increasing power generation rate is needed. The buildup and decay of U^{232} and Th^{228} activity as a function of time since final separation is plotted as Fig. 79.

Isotopic heat source data for 2-yr-aged U^{232} sources appear in Table 18, and the nuclear dat appear in Table 3. The results of the radiation intensity calculations for shielded and unshielded 2-yr-aged sources are given in Figs. 80 through 83. Isotopic heat source data for equilibrium U^{232} sources appear in Table 19. The results of the radiation intensity calculations for these shielded and unshielded sources are given in Figs. 84 through 87. In addition, Figs. 88 and 89 show the neutron intensities from water-shielded sources. Figure 2 should be used in conjunction with these figures in order to determine neutron radiation intensities when U^{232} is shielded by other materials.

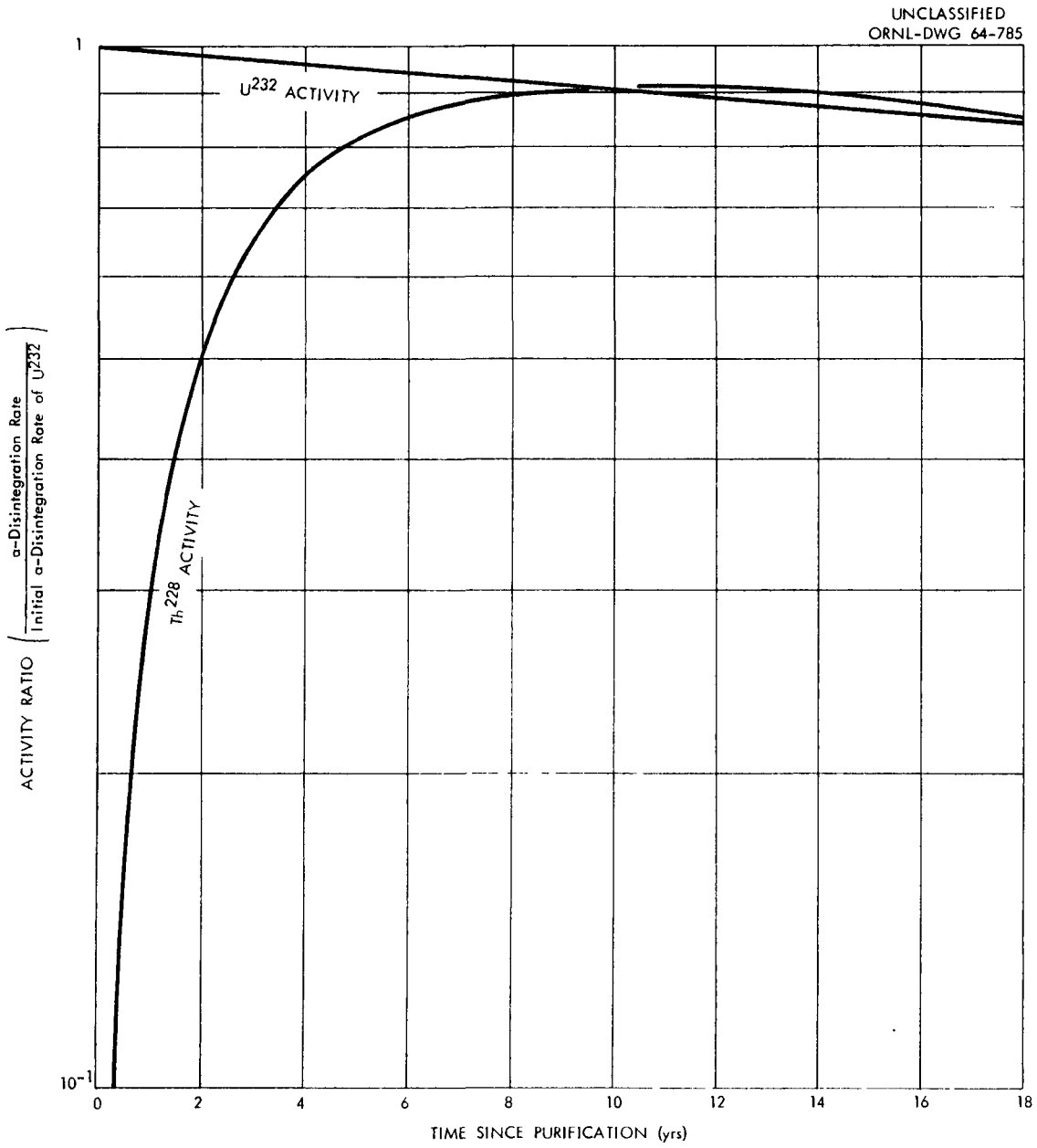


Fig. 79. Buildup and Decay of Uranium-232 and Thorium-228 Daughter Activities Following Purification.

Table 18. Physical Properties of Uranium-232[a] as an Isotopic Power Source

General Properties

Half-life	74 y
Specific activity of pure isotope, curies/g	20.9[c]
Specific power of pure isotope, w/g	2.82[d]
α,n rate, neutrons/g-sec	4×10^5
Spontaneous fission rate, neutrons/g-sec	1.79
Compound	UO_2
Weight of active isotope in compound, g/cc	8.53
Effective density of compound, g/cc	9.7
Power, w/cc of compound	24.09
Activity, curies/cc of compound	178[c]

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
1.296×10^{11}	0.25
3.719×10^{10}	0.83
1.075×10^{10}	1.34
1.367×10^{10}	1.81
6.835×10^{9}	2.20
4.949×10^{10}	2.62[e]
5.085×10^{10}	0.634

[a]Two-year-aged U^{232} source.

[c]Uranium-232 curies--does not include activity due to daughters.

[d]Includes heat generation from daughters which build in during two year storage time.

[e]Predominant radiation from unshielded 5000 watt source.

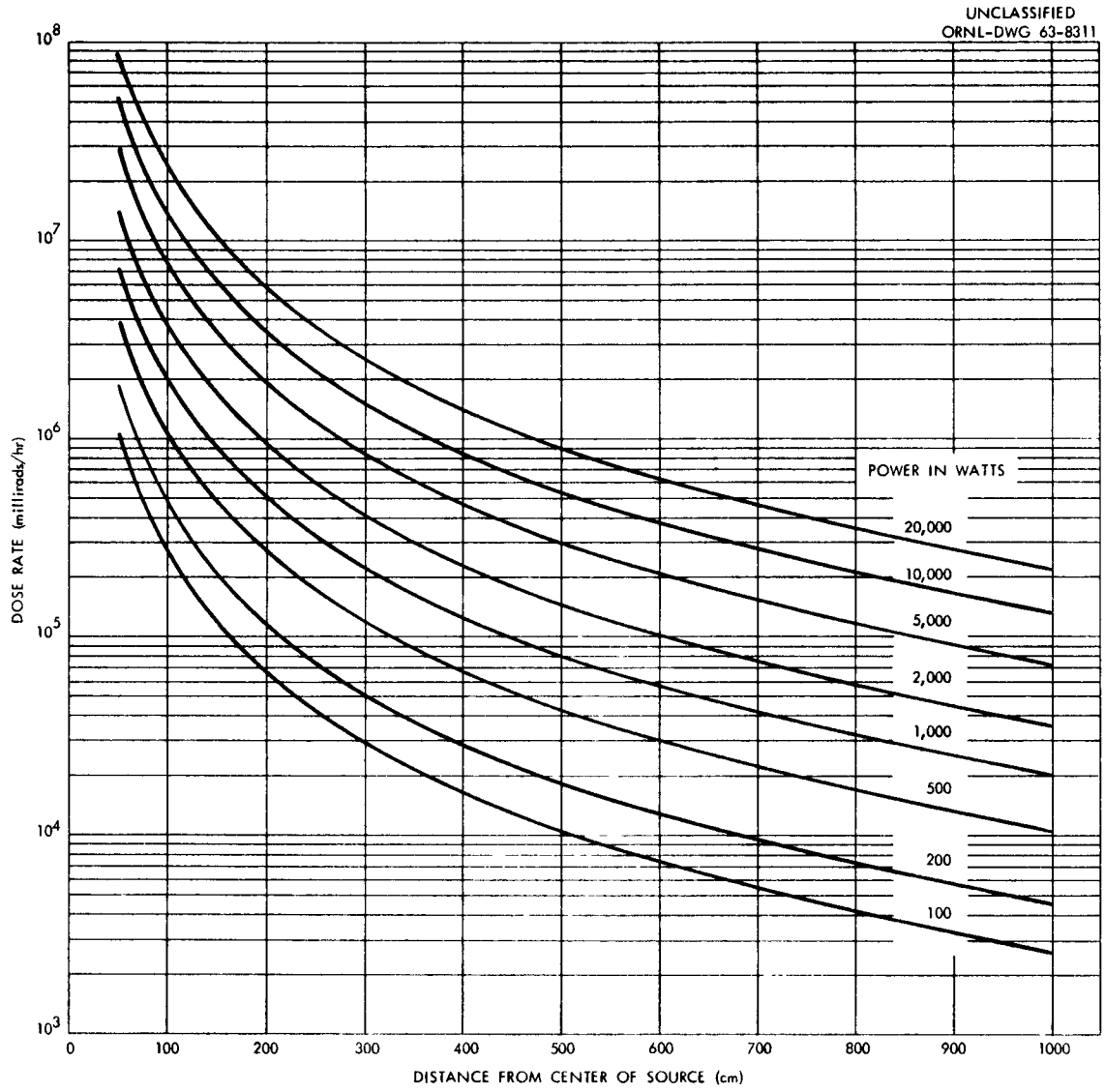


Fig. 80. Gamma Dose Rates from Unshielded Isotopic Power Sources of Uranium-232 as a Function of Distance from Center of Source. Source is assumed to be 2-yr-aged U^{232} and contains decay daughter activity equivalent to that which builds up during a period of 2 yr since final separation.

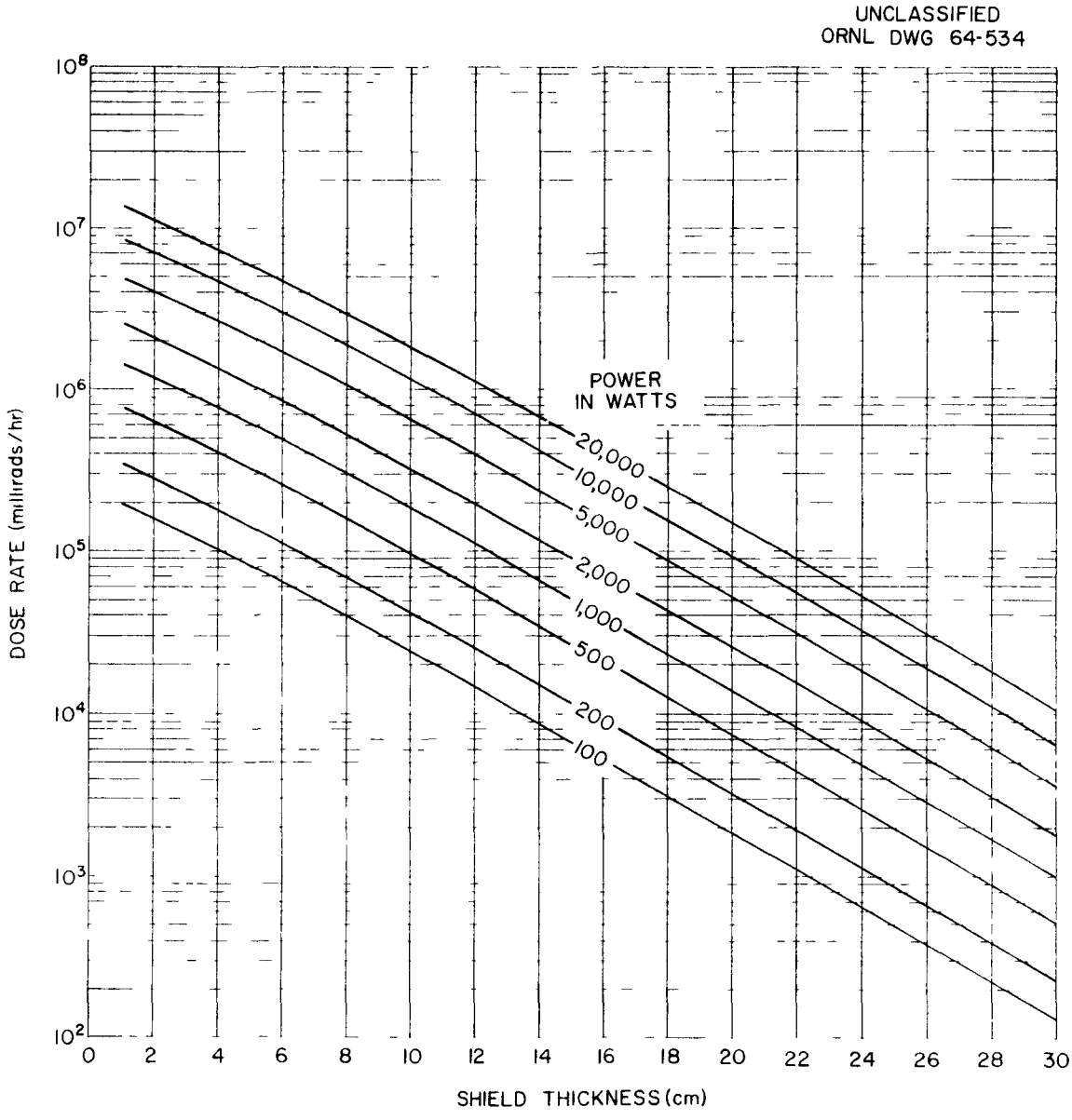


Fig. 81. Gamma Dose Rates from Iron-Shielded Isotopic Power Sources of Uranium-232. Center of source to dose point separation distance = 100 cm. Source is assumed to be 2-yr-aged U^{232} and contains decay daughter activity equivalent to that which builds up during a period of 2 yr since final separation.

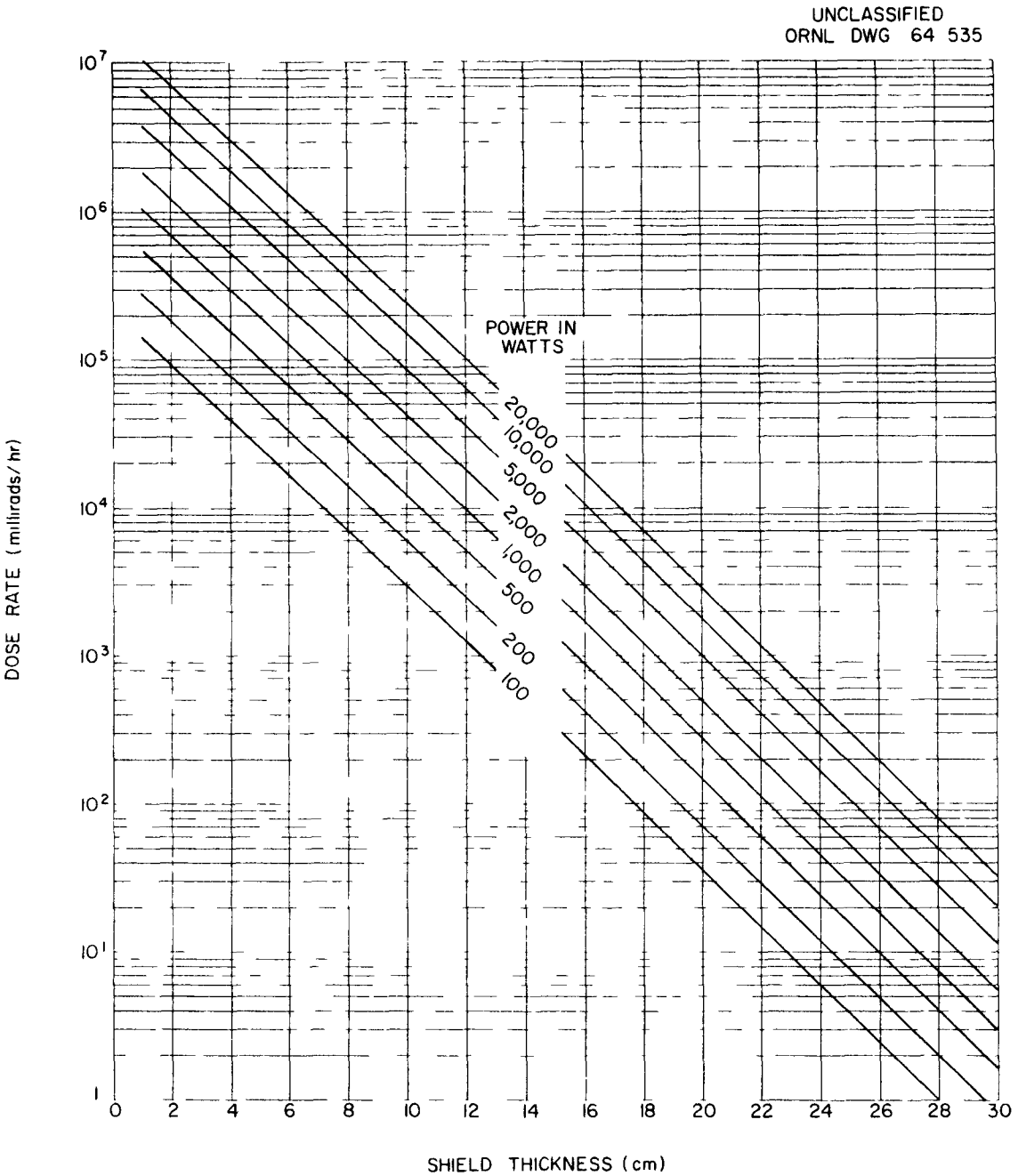


Fig. 82. Gamma Dose Rates from Lead-Shielded Isotopic Power Sources of Uranium-232. Center of source to dose point separation distance = 100 cm. Source is assumed to be 2-yr-aged U^{232} and contains decay daughter activity equivalent to that which builds up during a period of 2 yr since final separation.

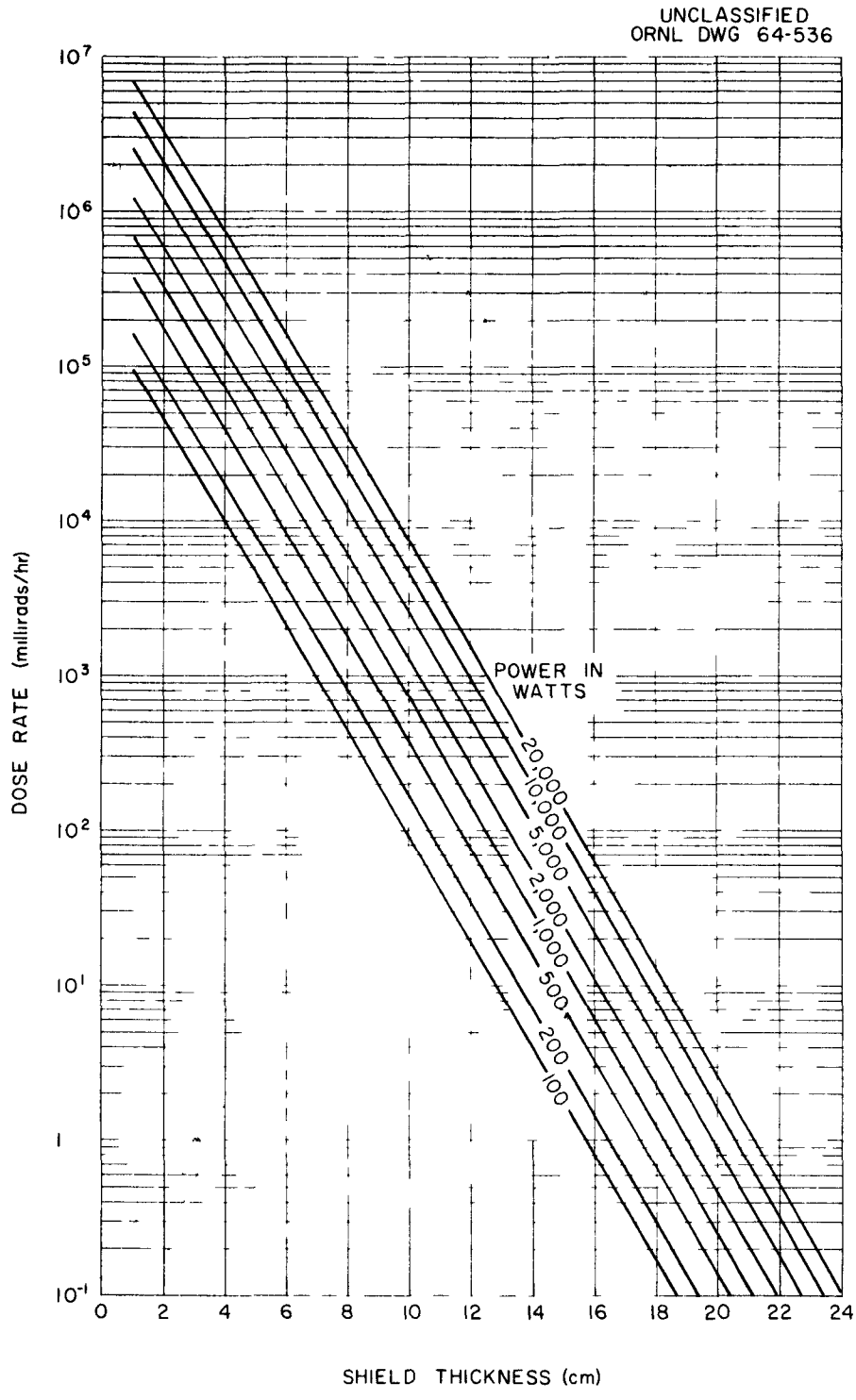


Fig. 83. Gamma Dose Rates from Uranium-Shielded Isotopic Power Sources of Uranium-232. Center of source to dose point separation distance = 100 cm. Source is assumed to be 2-yr-aged U^{232} and contains decay daughter activity equivalent to that which builds up during a period of 2 yr since final separation.

Table 19. Physical Properties of Uranium-232[a] as an Isotopic Power Source

General Properties	
Half-life	74 y
Specific activity of pure isotope, curies/g	20.9[c]
Specific power of pure isotope, w/g	4.99[d]
α,n rate, neutrons/g-sec	7.95×10^5
Spontaneous fission rate, neutrons/g-sec	1.79
Compound	UO_2
Weight of active isotope in compound, g/cc	8.53
Effective density of compound, g/cc	9.7
Power, w/cc of compound	42.58
Activity, curies/cc of compound	178[c]

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
1.466×10^{11}	0.25
4.207×10^{10}	0.83
1.214×10^{10}	1.34
1.547×10^{10}	1.81
7.735×10^{9}	2.20
5.60×10^{10}	2.62[e]
5.755×10^{10}	0.634

[a] Nine-year-aged U^{232} source.

[c] Uranium-232 curies--does not include activity due to daughters.

[d] Includes heat generation from daughters which build in during nine year storage time.

[e] Predominant radiation from unshielded 5000 watt source.

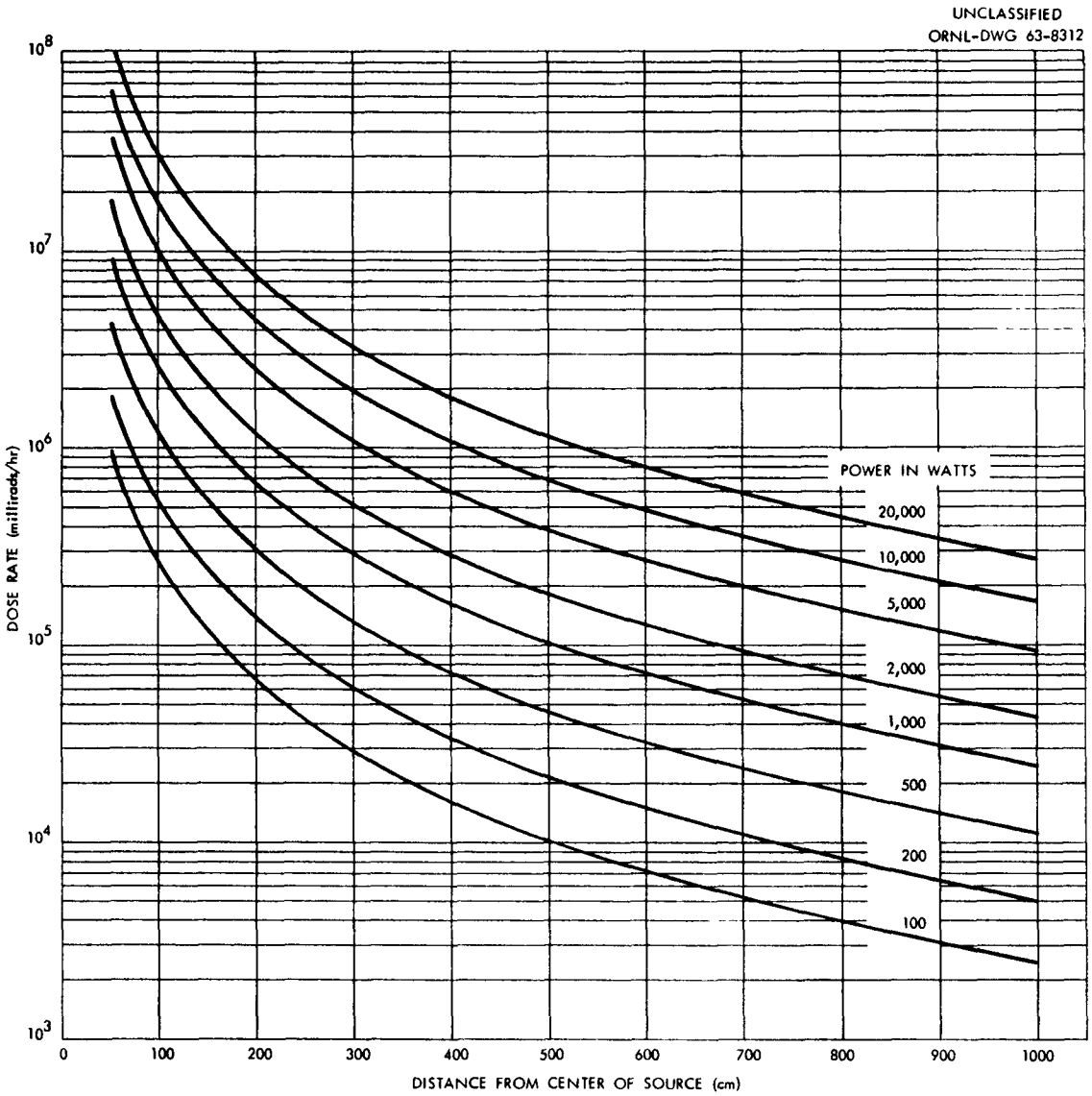


Fig. 84. Gamma Dose Rates from Unshielded Isotopic Power Sources of Uranium-232 as a Function of Distance from Center of Source. Source is assumed to be 9-yr-aged U^{232} and contains decay daughter activity equivalent to that which builds up during a period of 9 yr since final separation. Total activity remains almost constant during a time period of 9 to 14 yr since separation.

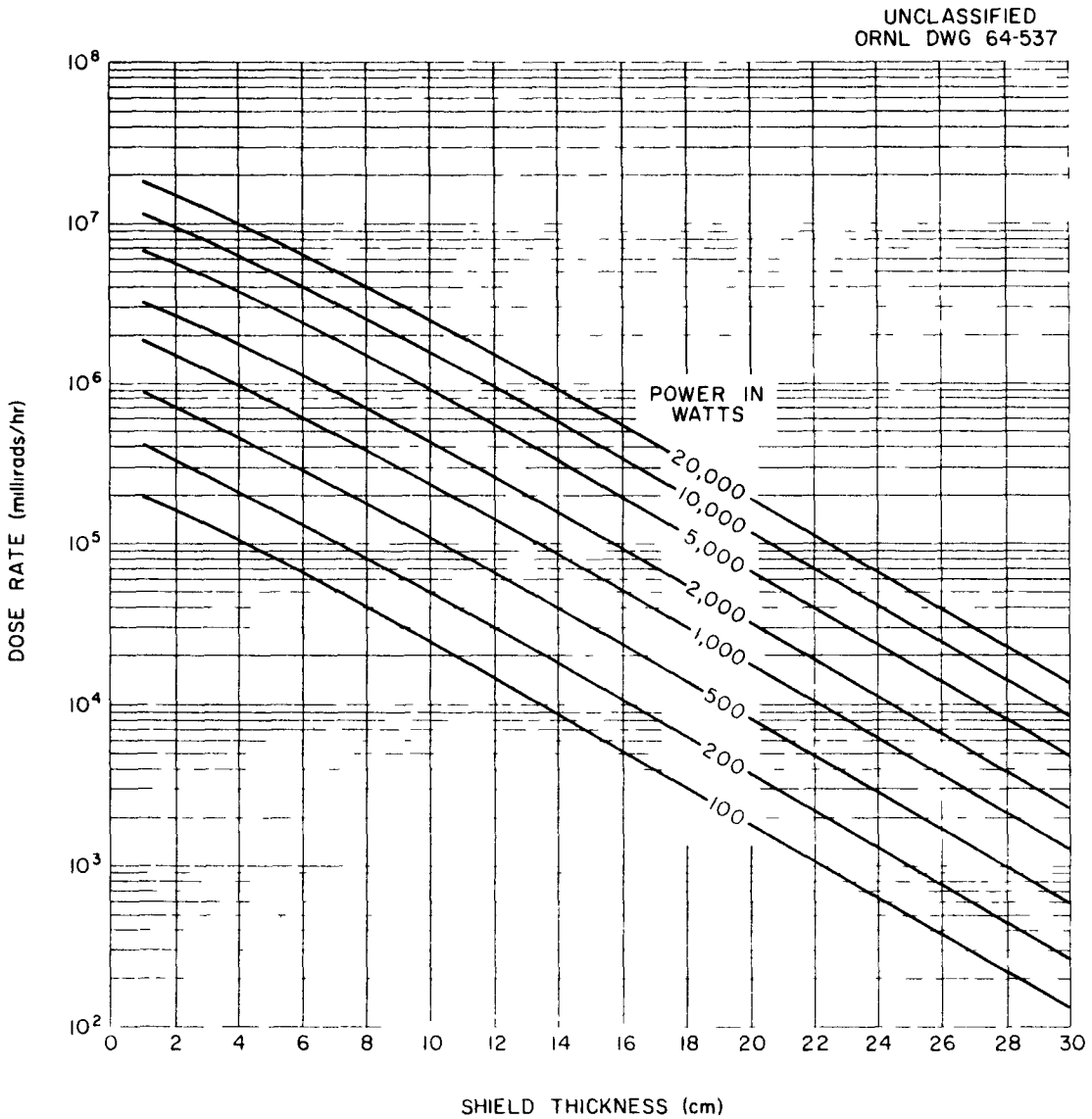


Fig. 85. Gamma Dose Rates from Iron-Shielded Isotopic Power Sources of Uranium-232. Center of source to dose point separation distance = 100 cm. Source is assumed to be 9-yr-aged U^{232} and contains decay daughter activity equivalent to that which builds up during a period of 9 yr since final separation. Total activity remains almost constant during a time period of 9 to 14 yr since separation.

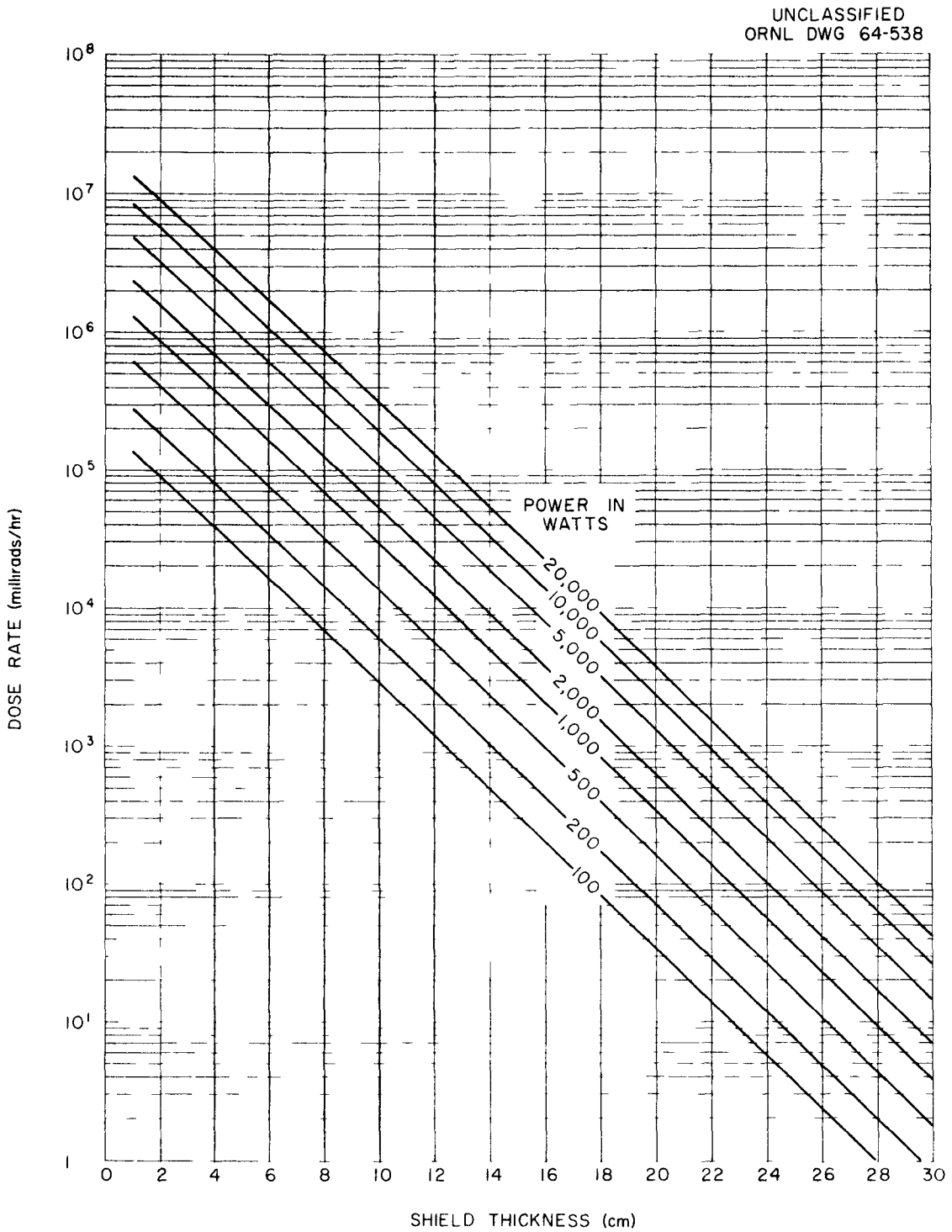


Fig. 86. Gamma Dose Rates from Lead-Shielded Isotopic Power Sources of Uranium-232. Center of source to dose point separation distance = 100 cm. Source is assumed to be 9-yr-aged U^{232} and contains decay daughter activity equivalent to that which builds up during a period of 9 yr since final separation. Total activity remains almost constant during a time period of 9 to 14 yr since separation.

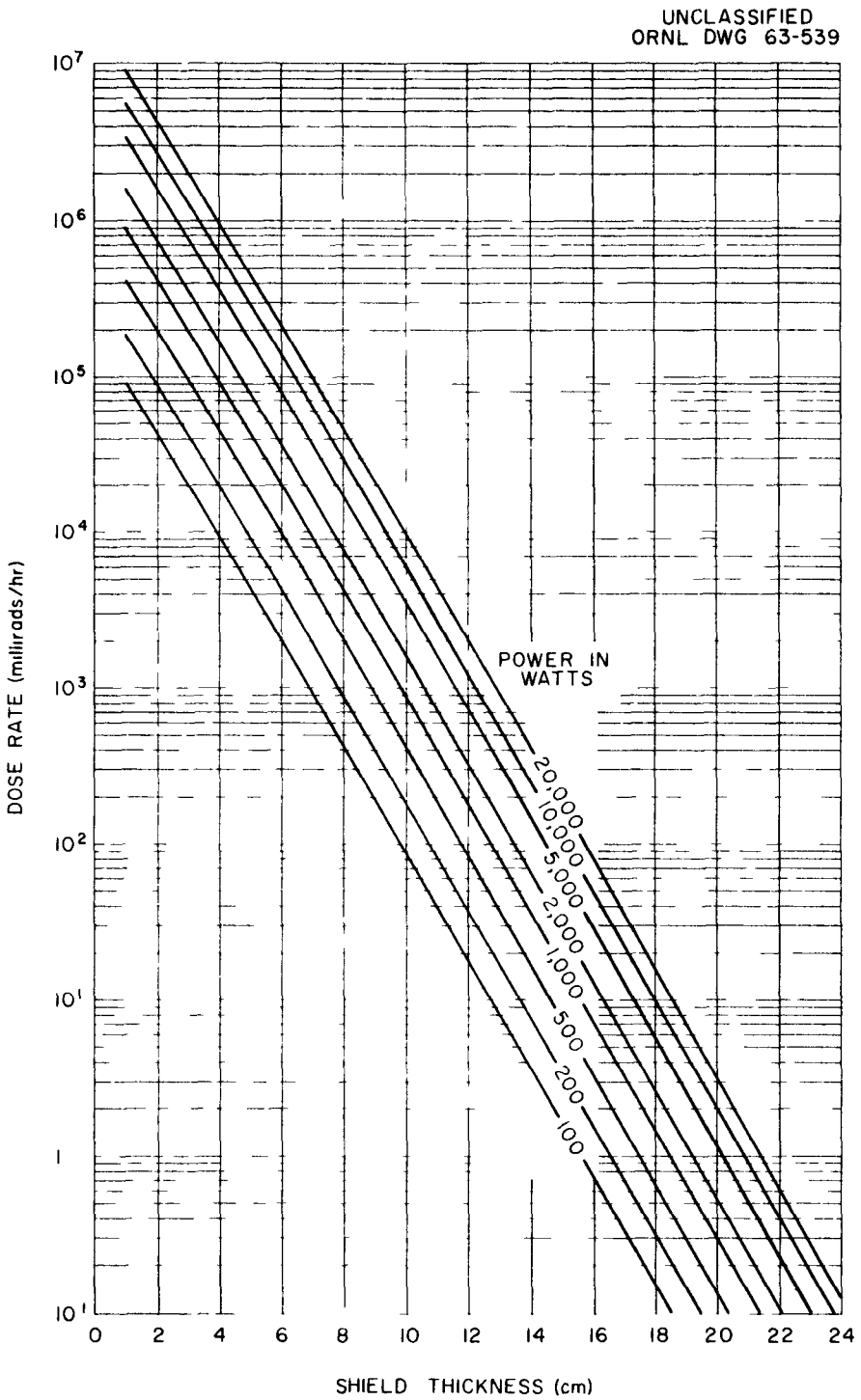


Fig. 87. Gamma Dose Rates from Uranium-Shielded Isotopic Power Sources of Uranium-232. Center of source to dose point separation distance = 100 cm. Source is assumed to be 9-yr-aged U^{232} and contains decay daughter activity equivalent to that which builds up during a period of 9 yr since final separation. Total activity remains almost constant during a time period of 9 to 14 yr since separation.

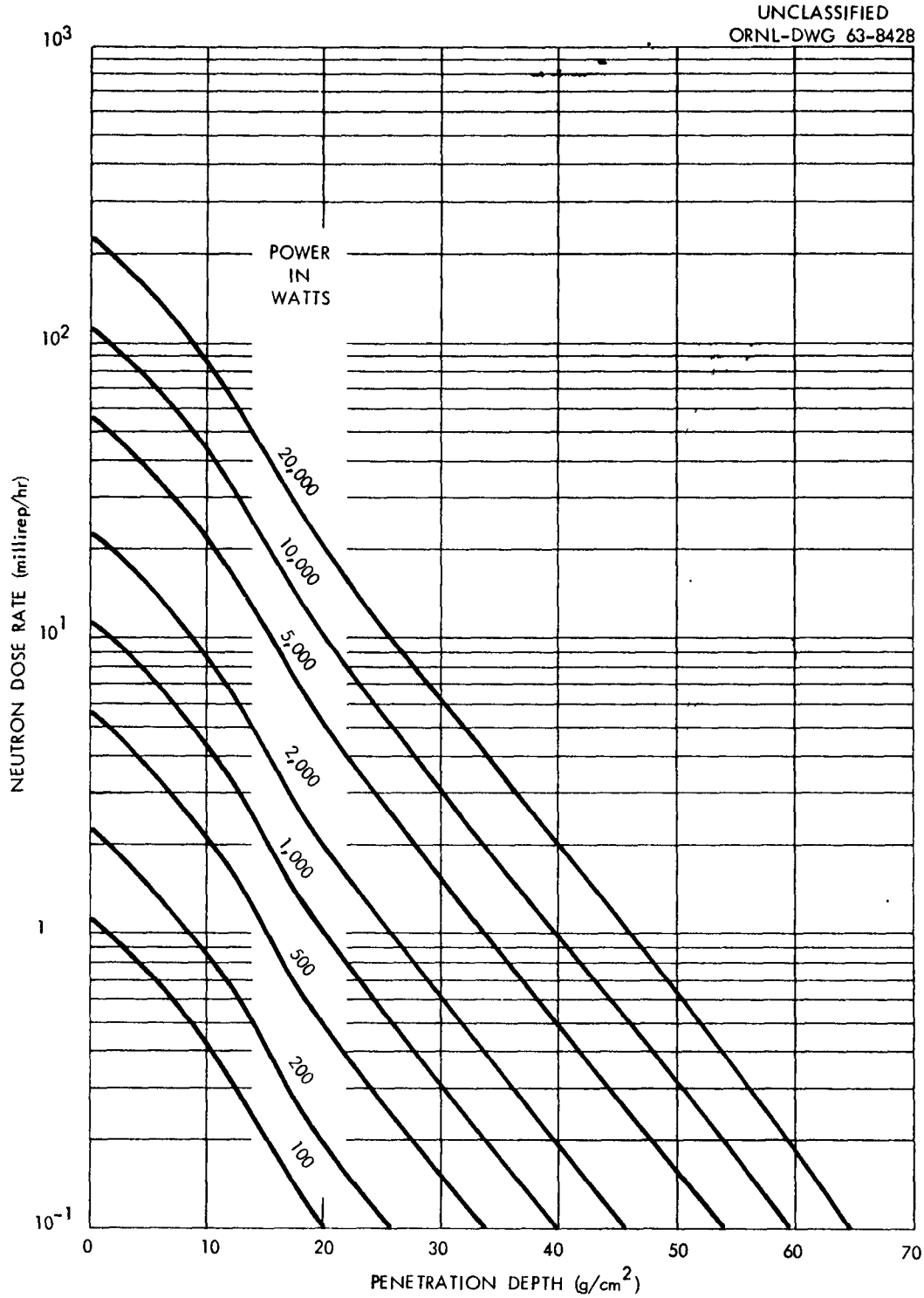


Fig. 88. Neutron Dose Rates from Water-Shielded Isotopic Power Sources of Two-Year-Aged Uranium-232 as a Function of Penetration Depth of Shielding Material. Center of source to dose point separation distance = 100 cm. Refer to Fig. 2 to obtain dose rates through other materials.

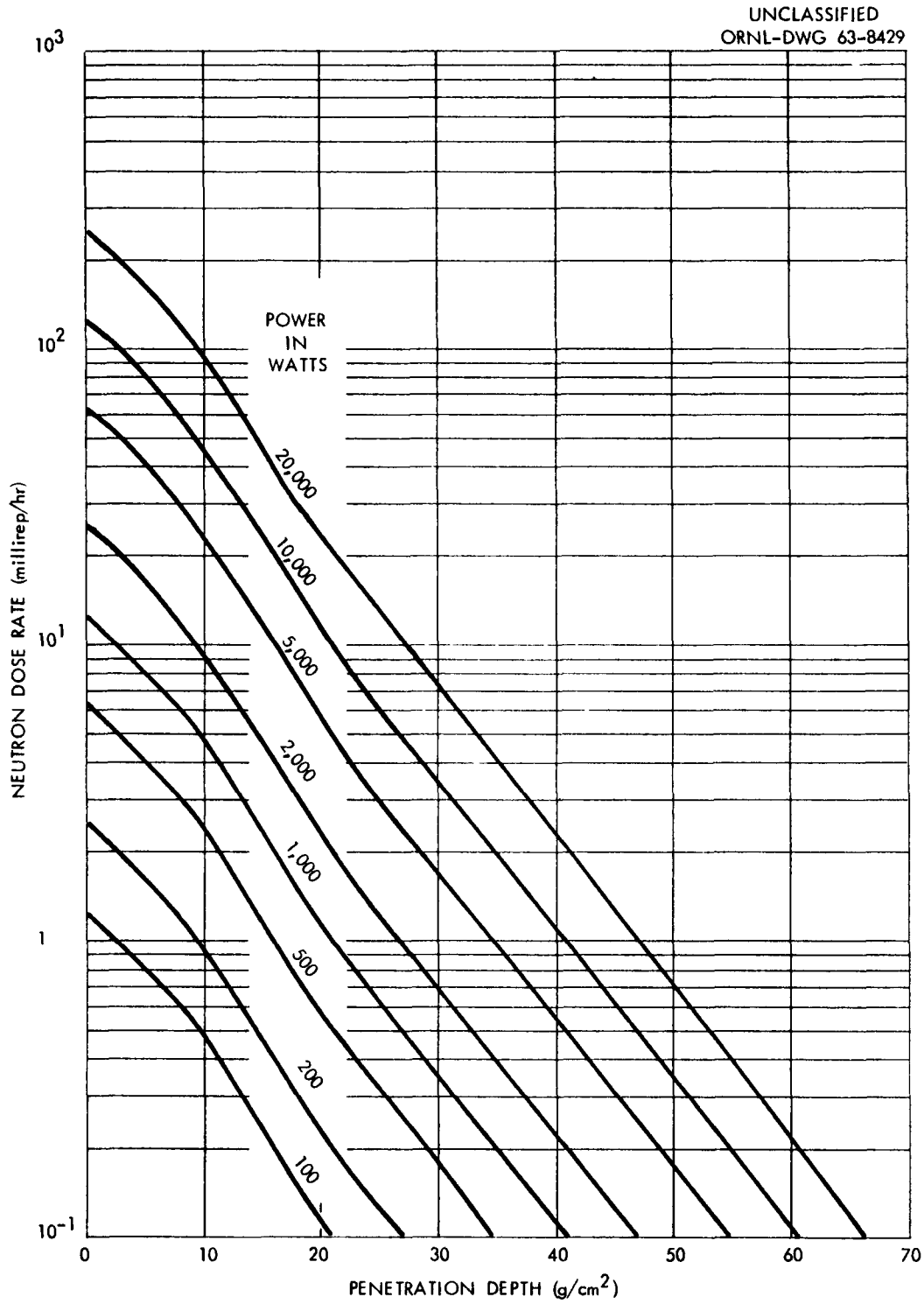


Fig. 89. Neutron Dose Rates from Water-Shielded Isotopic Power Sources of Nine-Year-Aged Uranium-232 as a Function of Penetration Depth of Shielding Material. Center of source to dose point separation distance = 100 cm. Refer to Fig. 2 to obtain dose rates through other materials.

Thorium-228

The same production techniques and decay-daughter characteristics described for U^{232} would apply to Th^{228}, and for this reason the description of Th^{228} follows that for U^{232}, rather than being in the normal periodic order.

With a production effort on U^{232} and a stock of such material at hand, Th^{228} could also be routinely recovered. However, this stockpile would have to contain 64 kg of U^{232} in order to produce enough Th^{228} equivalent to 100 thermal kilowatts of power per year. The production and utilization of Th^{228} depends on the feasibility of large-scale production of U^{232}. From heat output and half-life standpoints, Th^{228} has no equal (170 w/g, 1.9-yr half-life). This is particularly impressive when the power density is noted, 1270 w/cc. Even at the high estimated cost per gram, the large thermal yield is sufficient to make this energy source competitive with the cheapest power sources. No spontaneous fission half-life is shown for Th^{228} (ref. 31). However, the decay daughters of Th^{228} release high-energy alpha particles which would produce α,n reactions on light elements present either as impurities or as a compound of thorium, such as the oxygen of thorium oxide.

The heat source calculations were based on thorium oxide (ThO_2) diluted with normal thorium (Th^{232}) oxide to a power density of 150 w/cc. An equal volume of void space was provided for gas collection thus reducing the effective power density to 75 w/cc. Since there is little self-absorption in thorium oxide for the 2.62-Mev gamma of Tl^{208}, there would be very little increase in radiation dose rates from higher specific power sources of the same total power.

Isotopic heat source data for Th^{228} sources appear in Table 20, and the nuclear data appear in Table 3. The results of the radiation-intensity calculations for shielded and unshielded sources are given in Figs. 90 through 93. In addition, Fig. 94 shows the neutron intensities from water shielded sources. Figure 2 should be used in conjunction with this figure in order to determine neutron radiation intensities when Th^{228} is shielded by other materials.

Table 20. Physical Properties of Thorium-228 as an Isotopic Power Source

General Properties

Half-life	1.91 y
Specific activity of pure isotope, curies/g	822.4[a]
Specific power of pure isotope, w/g	170.5[c]
α,n rate, neutrons/g-sec	3×10^7
Compound	ThO_2 in matrix
Weight of active isotope in matrix, g/cc	0.439
Effective density of matrix, g/cc	9.0
Power, w/cc of matrix	75
Activity, curies/cc of matrix	361[a]

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
1.689×10^{11}	0.25
4.846×10^{10}	0.83
1.399×10^{10}	1.34
1.782×10^{10}	1.81
8.91×10^{9}	2.20
6.45×10^{10}	2.62[e]
6.625×10^{10}	0.634

[a]Thorium-228 curies--does not include activity due to daughters.

[c]Includes heat generation from daughters in transient equilibrium with Th^{228}.

[e]Predominant radiation from unshielded 5000 watt source.

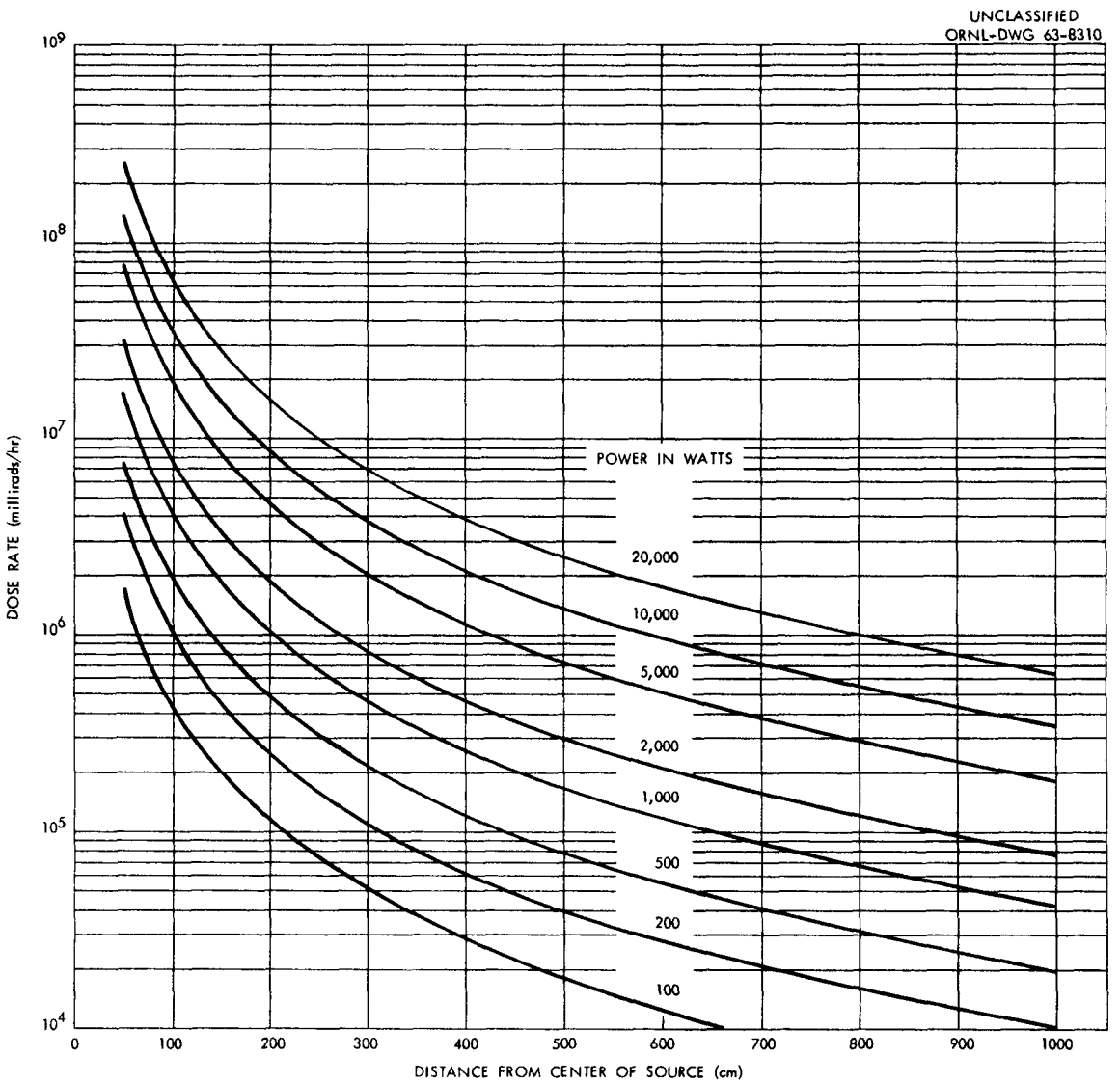


Fig. 90. Gamma Dose Rates from Unshielded Isotopic Power Sources of Thorium-228 as a Function of Distance from Center of Source. Decay daughters are assumed to be in transient equilibrium with Th^{228}.

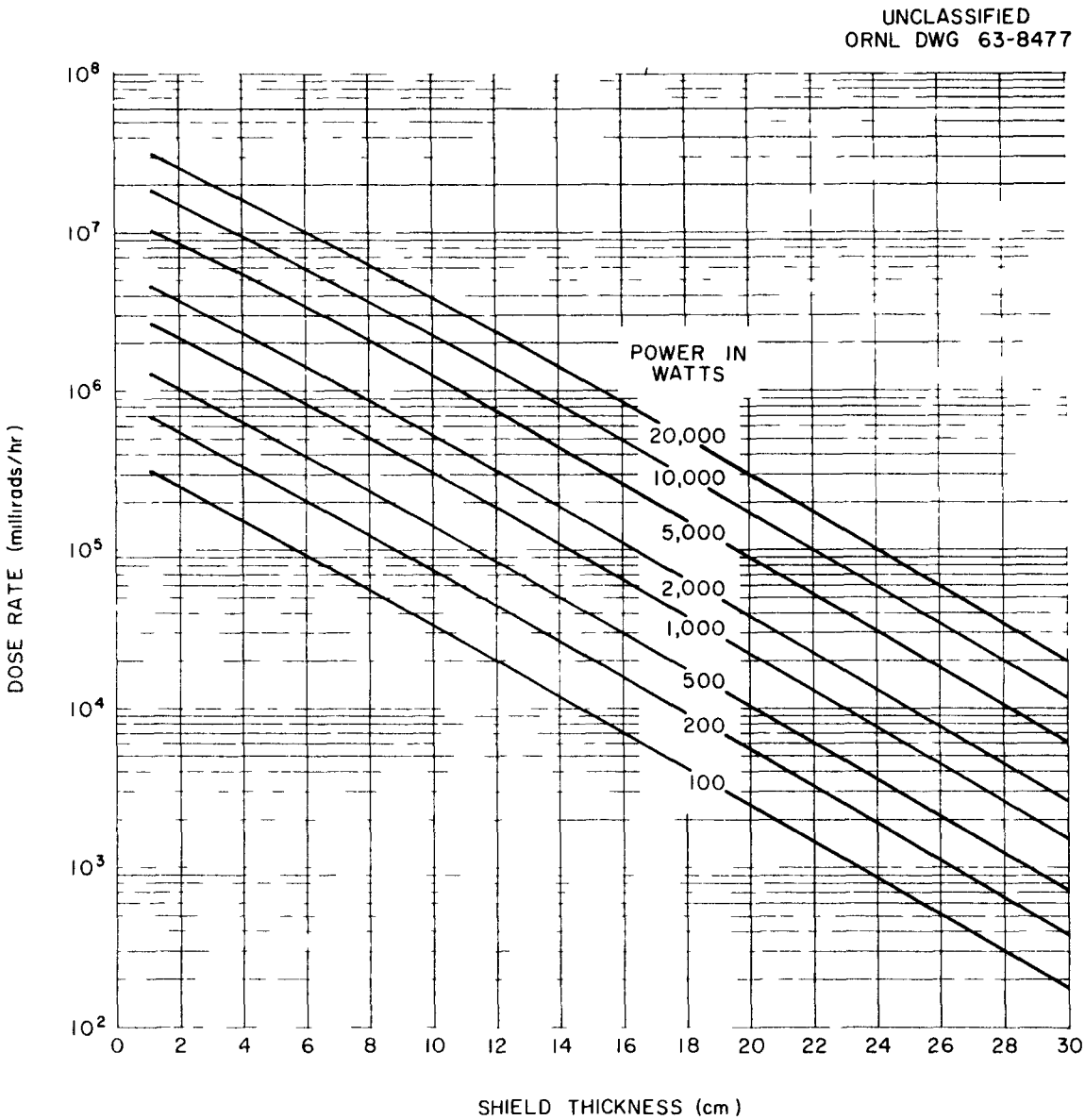


Fig. 91. Gamma Dose Rates from Iron-Shielded Isotopic Power Sources of Thorium-228. Center of source to dose point separation distance = 100 cm. Decay daughters are assumed to be in transient equilibrium with Th^{228}.

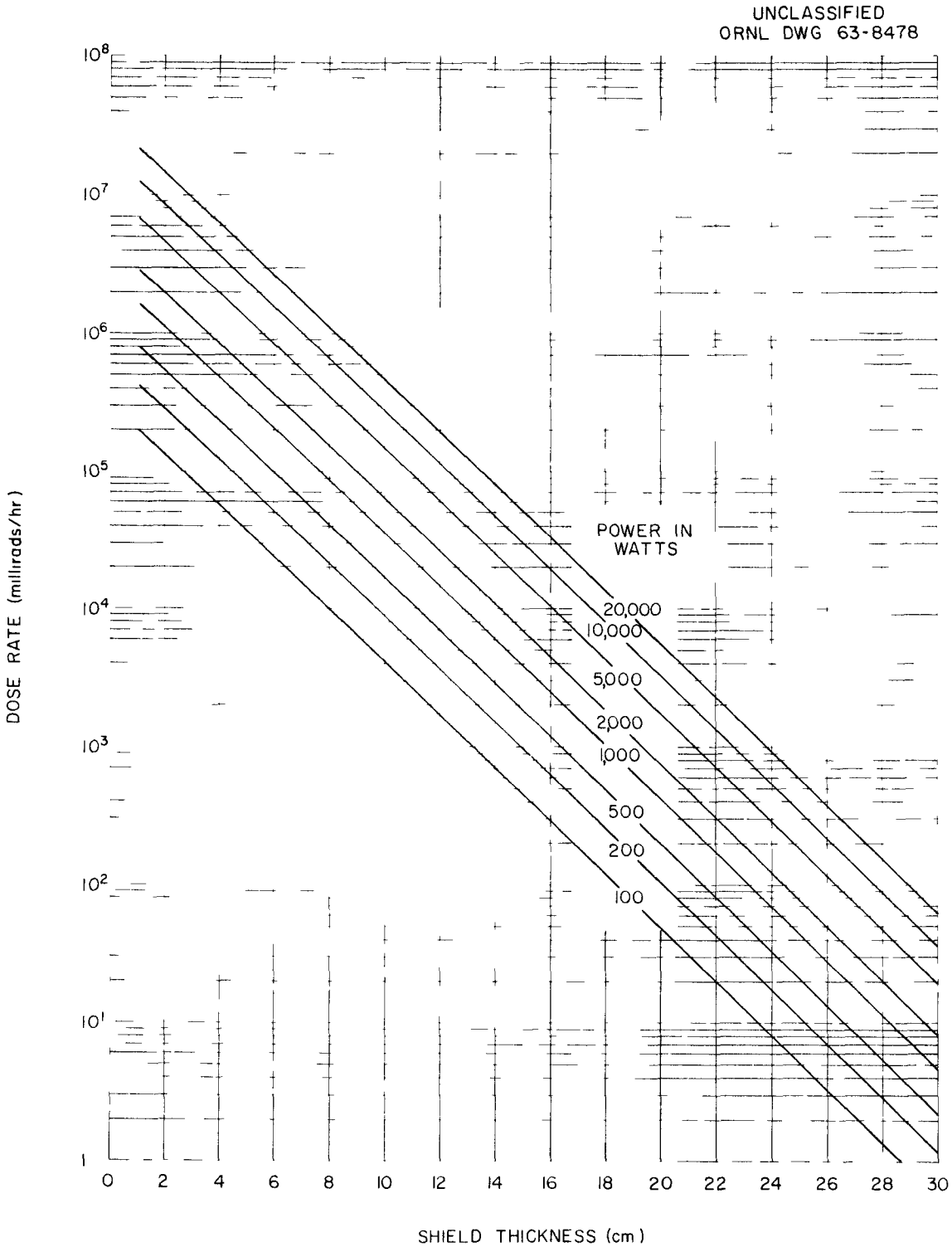


Fig. 92. Gamma Dose Rates from Lead-Shielded Isotopic Power Sources of Thorium-228. Center of source to dose point separation distance = 100 cm. Decay daughters are assumed to be in transient equilibrium with Th^{228}.

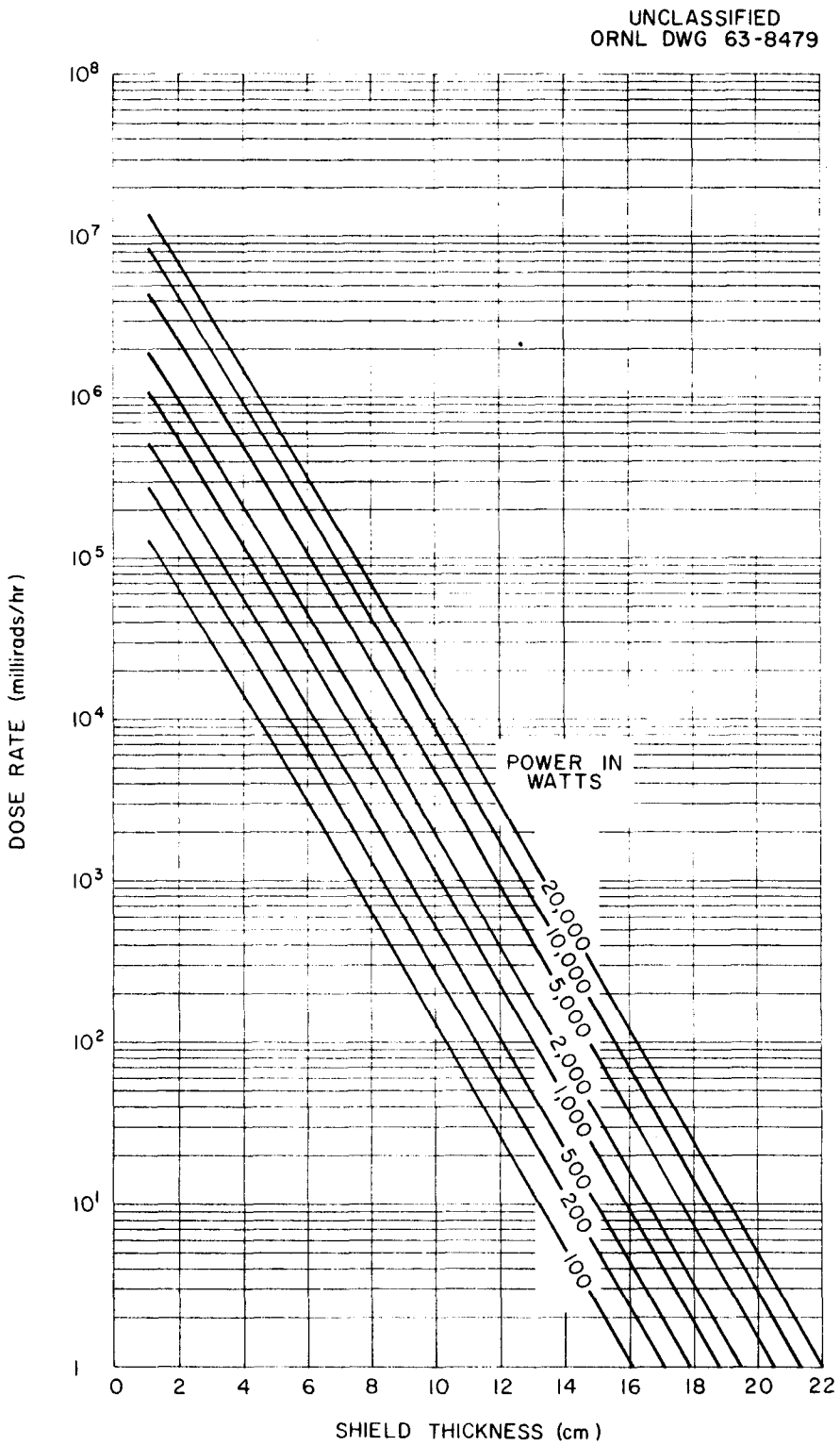


Fig. 93. Gamma Dose Rates from Uranium-Shielded Isotopic Power Sources of Thorium-228. Center of source to dose point separation distance = 100 cm. Decay daughters are assumed to be in transient equilibrium with Th^{228}.

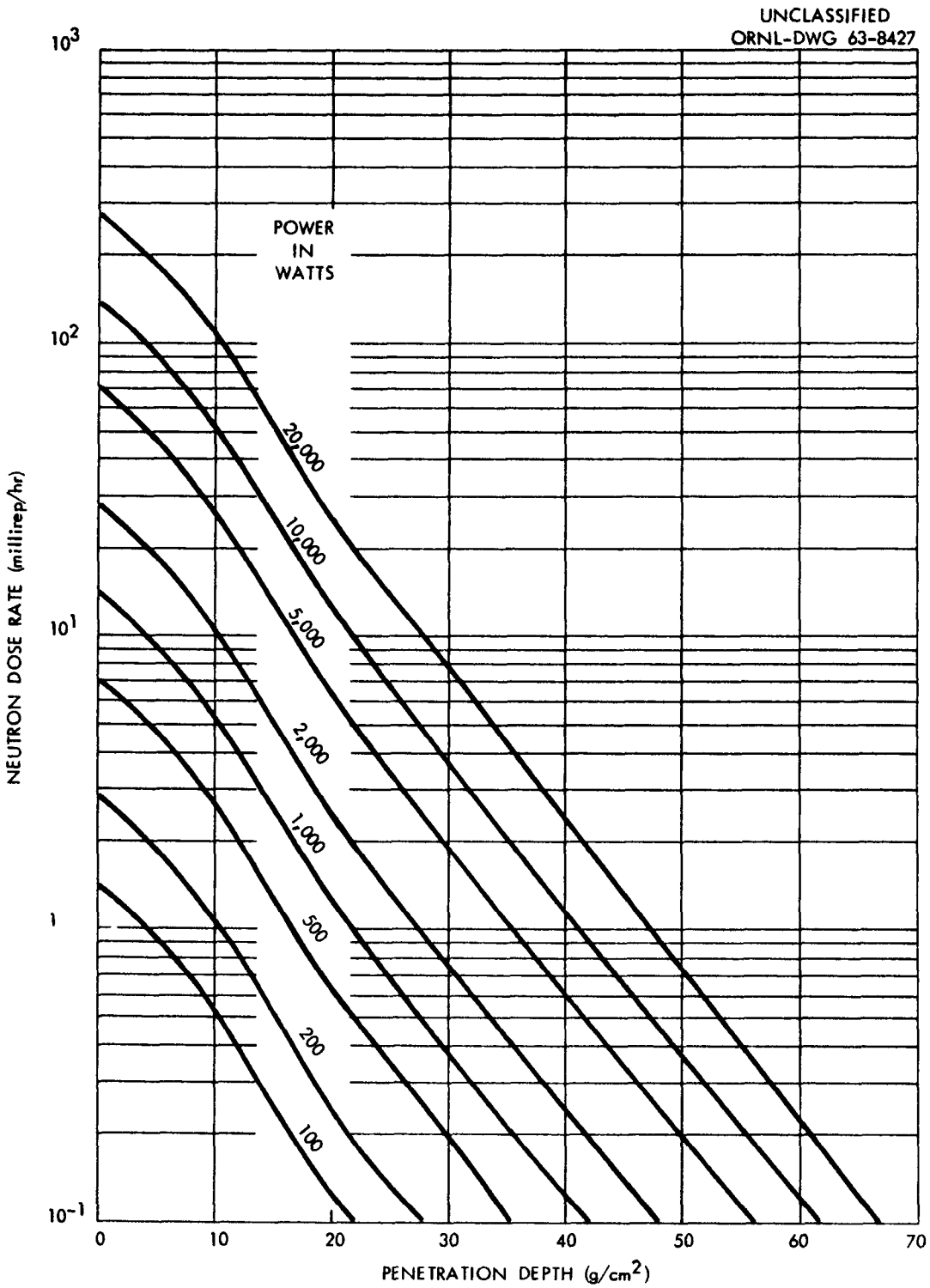


Fig. 94. Neutron Dose Rates from Water-Shielded Isotopic Power Sources of Thorium-228 as a Function of Penetration Depth of Shielding Material. Center of source to dose point separation distance = 100 cm. Refer to Fig. 2 to obtain dose rates through other materials.

Plutonium-238

Plutonium-238 is produced by the irradiation of Np^{237}. The latter isotope is recovered from irradiated uranium processing streams. Production of Pu^{238} during reactor irradiation is similar to that of fission products in that it is a function of the irradiation exposure. However, the production rate can be increased by recycling the U^{236} with the uranium streams, with U^{235} burnup being compensated by blending rather than re-enrichment in the diffusion cascade, where U^{236} is partially removed. A Pu^{238} production rate of about 3 kg/yr per 1000 thermal megawatts of reactor power (0.8 load factor) can be achieved.[32] Neptunium-237 is formed in very small quantities by two processes: in one, an n,2n reaction on U^{238} produces U^{237}, which rapidly decays to Np^{237}; in the other, neutron capture in U^{235} produces U^{236} which, in turn, captures a neutron to produce U^{237} (6.7-day half-life), which decays to Np^{237} (very long half-life). The total cycle for Pu^{238} production involves recovery and purification of Np^{237}, fabrication of Np^{237} into target elements, irradiation of the target elements to convert some of the Np^{237} to Pu^{238}, and chemical processing for the recovery and purification of the Pu^{238} and unconverted Np^{237} (ref. 33).

Plutonium-238 has the longest half-life (89.6 yr) of any of the radioisotopes which have promise as isotopic power sources. Its specific power is only moderate, 0.55 w/g, but its high density, particularly as the metal, puts it near the top of the attractive heat-source isotopes on hand today. Its extremely weak or low-yield (spontaneous-fission) gamma components allow its use without special shielding. Among those on hand today, it is the only radioisotope with which shielding is not a major problem. However, its limited availability,[34] extremely high biological hazard, and high cost per watt are factors unfavorable to plans for widespread use. Although the purity obtainable is quite high, it still can be expected to be contaminated with other plutonium isotopes, which could be regarded as diluents. For Pu^{238}, as with all alpha emitters, allowance must be made in the design of the capsule for the accumulation of gaseous helium. (Helium is the end-product of alpha particles.) Unless adequate space is provided, such accumulations will result in high pressures. A lesser problem also common among some of the high-atomic-weight alpha emitters is that of the neutron and resulting gamma activity derived from spontaneous fission. Fortunately,

the spontaneous fission half-lives are long and thus afford relatively minor fluxes which can be accommodated in the design of the device. The spontaneous fission half-life of Pu^{238} is 4.9×10^{10} yr.[17]

In addition to spontaneous fission, neutrons can be generated by α,n reactions on light-element atoms present in plutonium compounds or as impurities. For plutonium oxide, the production of neutrons by α,n reactions on oxygen is about 15 times that from spontaneous fission.

Isotopic heat source data for Pu^{238} sources appear in Table 21, and the nuclear data appear in Table 3. The result of the gamma-radiation-intensity calculations for shielded and unshielded sources are given in Figs. 95 through 99. Figure 100 shows the neutron intensities from water-shielded sources. Figure 2 should be used in conjunction with this figure in order to get neutron radiation intensities when plutonium is shielded by other materials.

The results are presented for plutonium oxide sources. If plutonium metal sources are used, the neutron dose would be decreased somewhat (maybe as much as a factor of two) depending upon impurities or alloying agents, while there would be only a small decrease in the gamma intensities due to an increase in density, with its increase in self-absorption being partially compensated by an increase in specific power and decrease in the radius of the source.

Table 21. Physical Properties of Plutonium-238 as an Isotopic Power Source

General Properties

Half-life	89.8 y
Specific activity of pure isotope, curies/g	16.9
Specific power of pure isotope, w/g	0.55
α,n rate, neutrons/g-sec	5.05×10^4
Spontaneous fission rate, neutrons/g-sec	3.41×10^3
Compound	PuO_2
Weight of active isotope in compound, g/cc	6.36
Effective density of compound, g/cc	10.0
Power, w/cc of compound	3.5
Activity, curies/cc of compound	107.5

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
8.77×10^7	0.15
5.68×10^5	0.76[e]
6.60×10^3	1.0[a]
1.654×10^3	1.5[a]
1.75×10^3	2.3[a]
3.09×10^2	3.0[a]
4.12×10^2	5.0[a]
5.84×10^3	0.63[c]
2.16×10^3	1.1[c]
2.53×10^3	1.55[c]
4.89×10^2	2.38[c]
7.24×10^2	2.75[c]

[a]Spontaneous fission gamma source.

[c]Fission product (from spontaneous fission) gamma source.

[e]Predominant radiation from unshielded 5000 watt source.

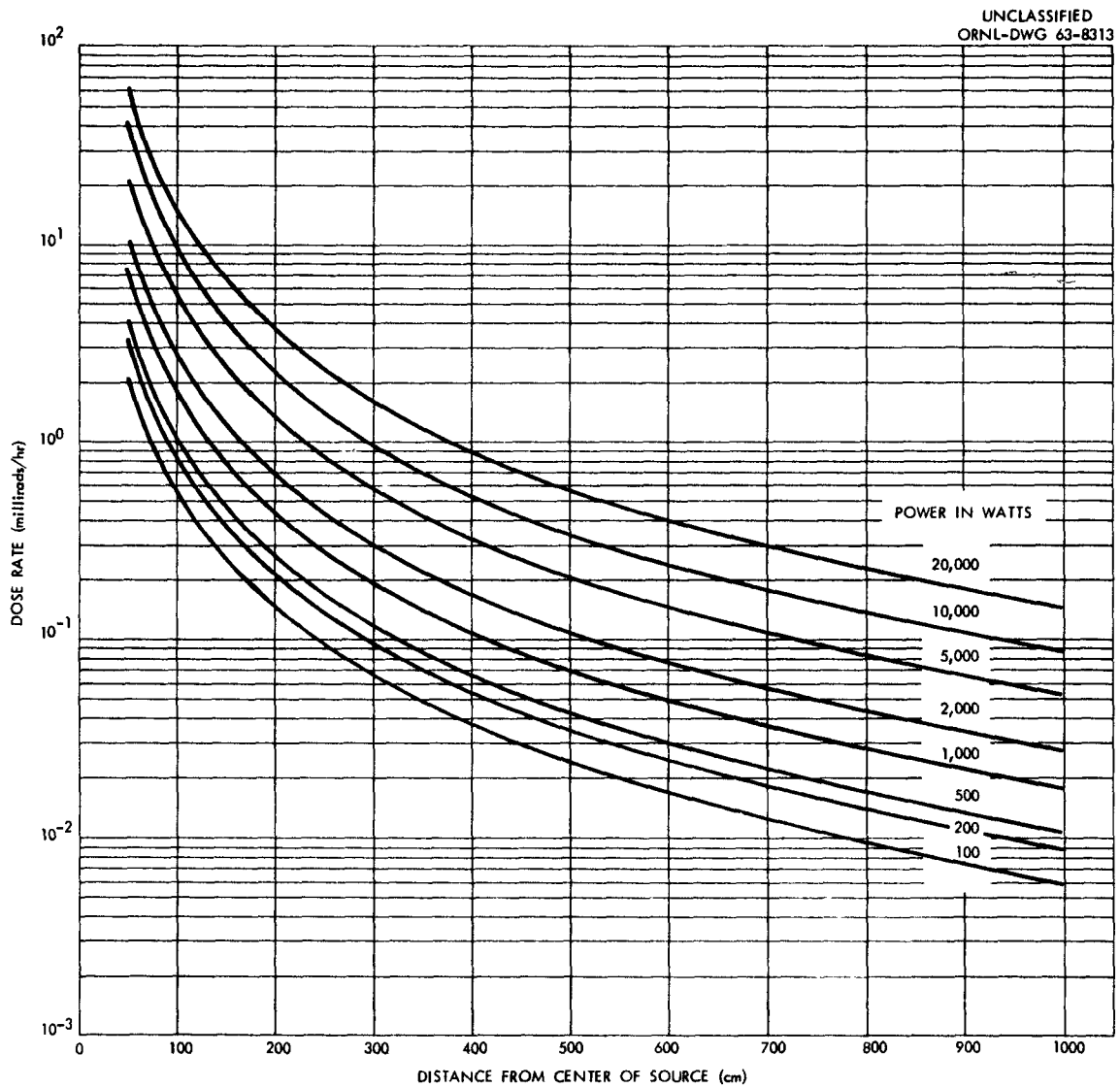


Fig. 95. Gamma Dose Rates from Unshielded Isotopic Power Sources of Plutonium-238 as a Function of Distance from Center of Source.

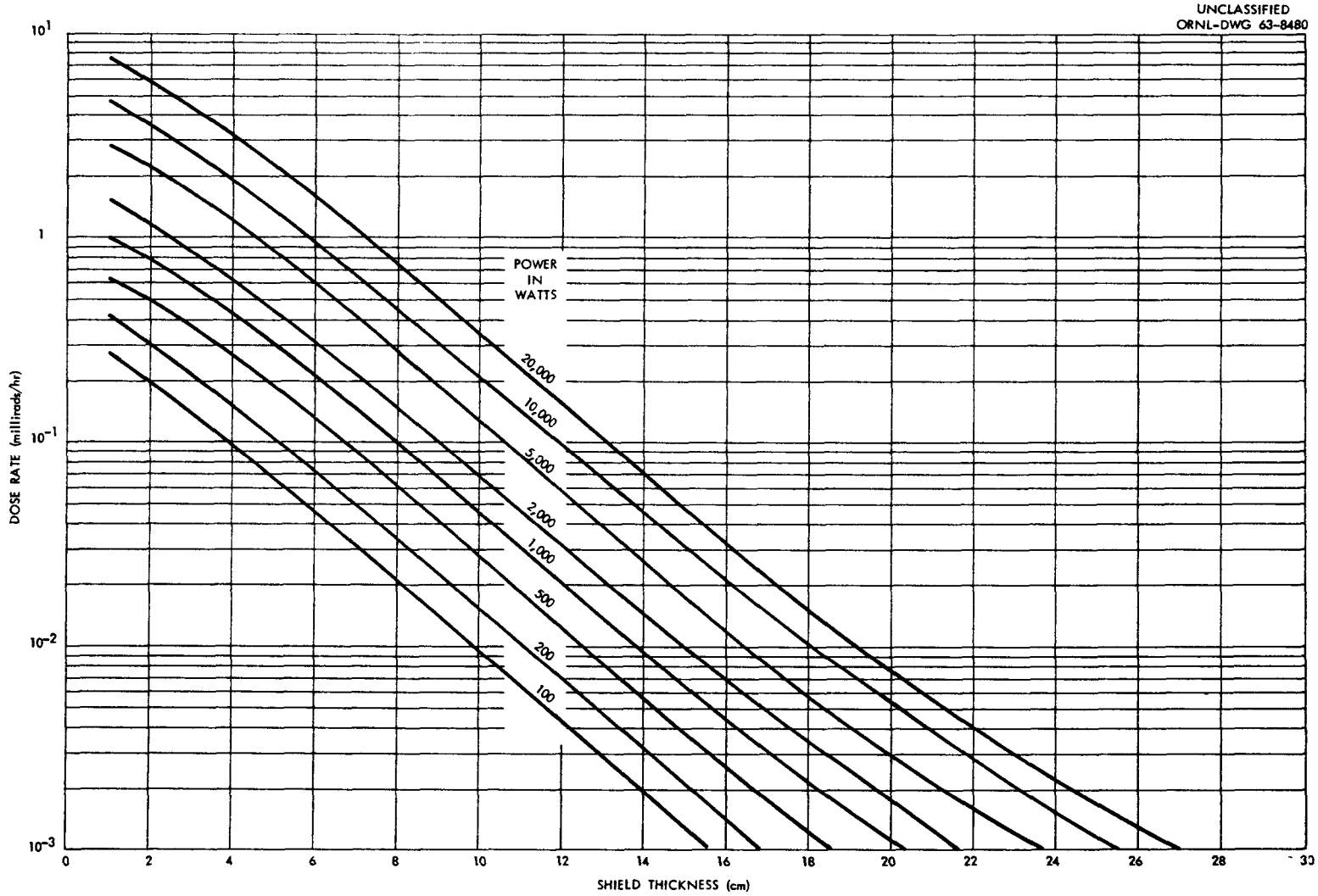


Fig. 96. Gamma Dose Rates from Iron-Shielded Isotopic Power Sources of Plutonium-238. Center of source to dose point separation distance = 100 cm.

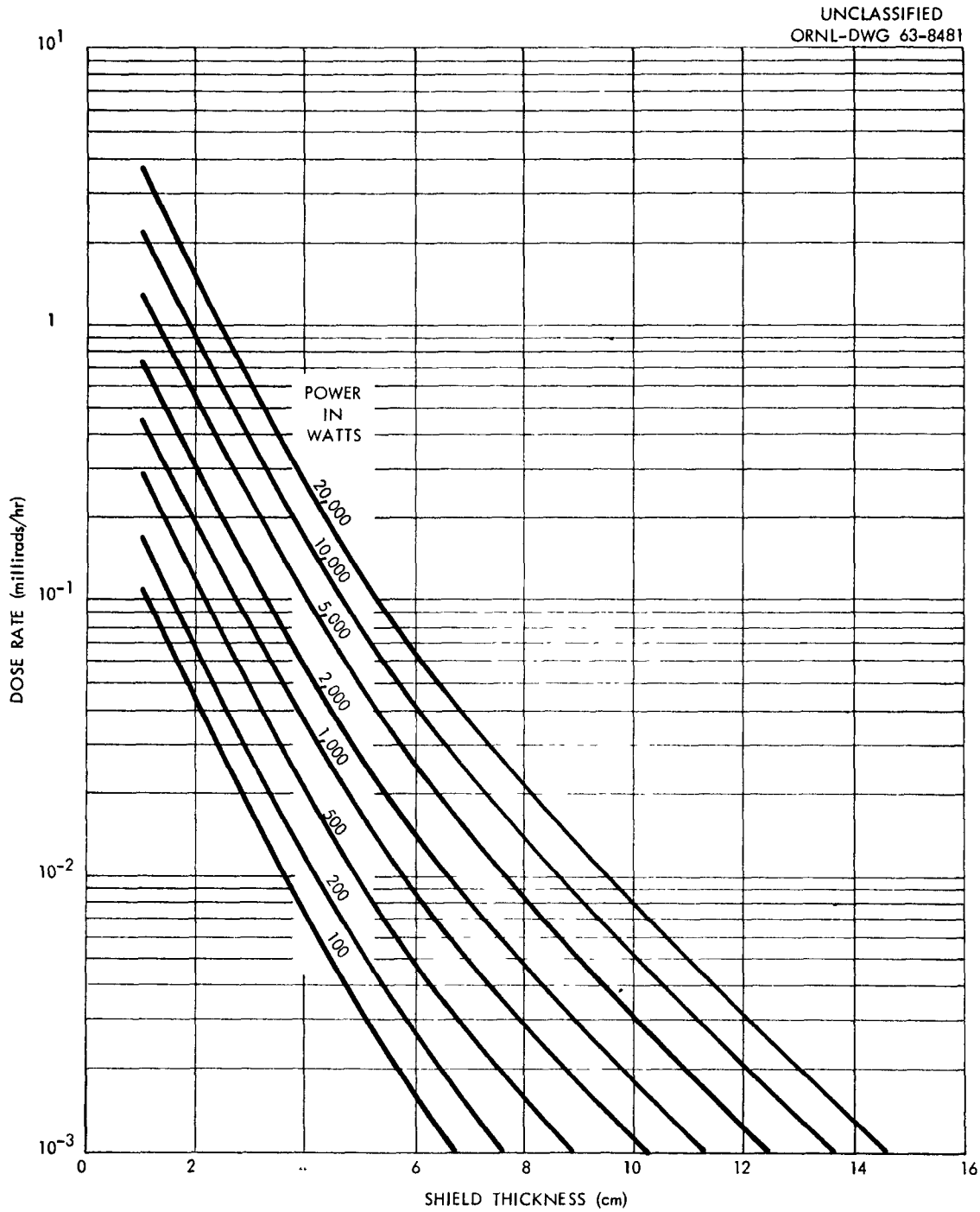


Fig. 97. Gamma Dose Rates from Lead-Shielded Isotopic Power Sources of Plutonium-238. Center of source to dose point separation distance = 100 cm.

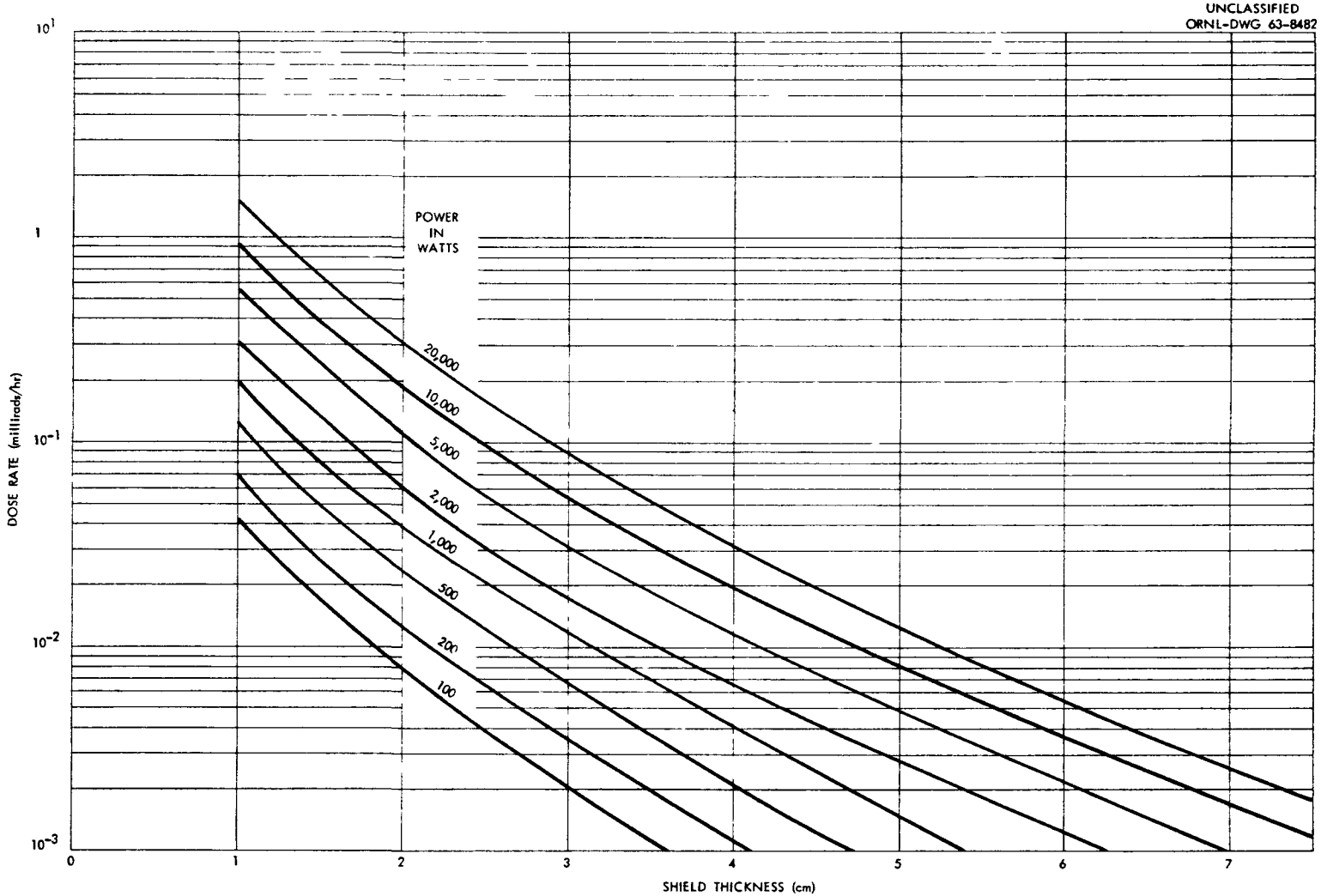


Fig. 98. Gamma Dose Rates from Uranium-Shielded Isotopic Power Source of Plutonium-238. Center of source to dose point separation distance = 100 cm.

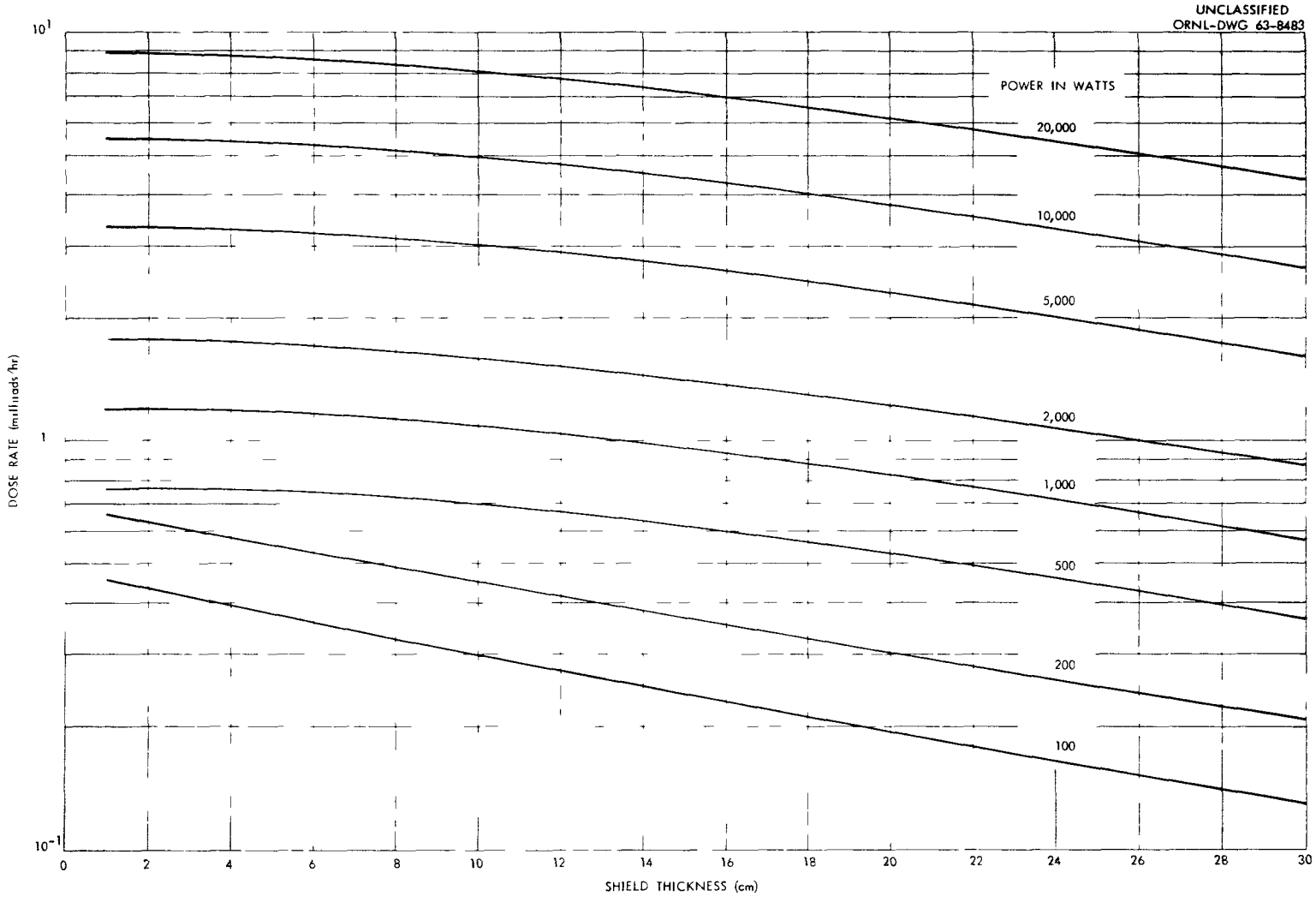


Fig. 99. Gamma Dose Rates from Water-Shielded Isotopic Power Sources of Plutonium-238. Center of source to dose point separation distance = 100 cm.

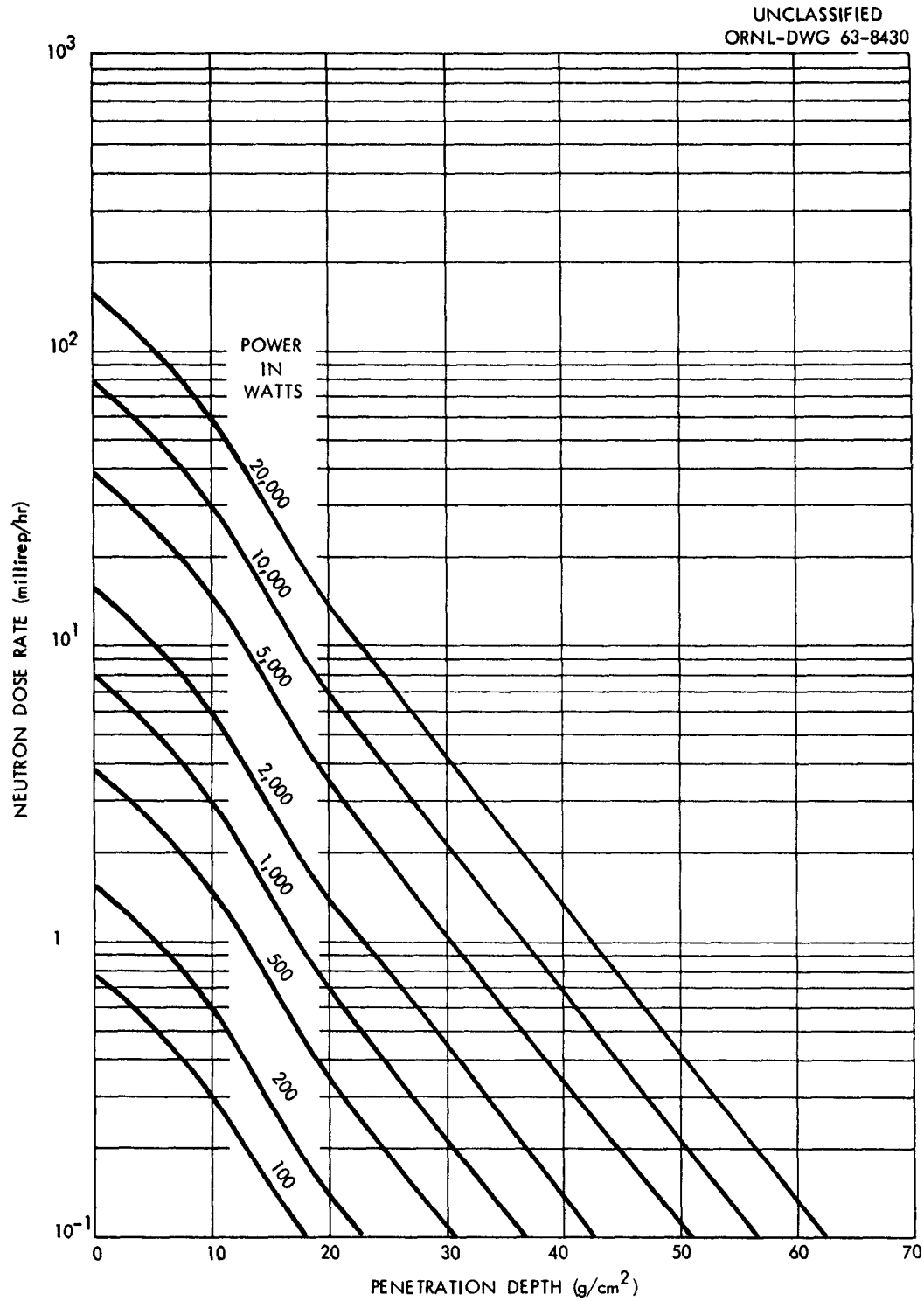


Fig. 100. Neutron Dose Rates from Water-Shielded Isotopic Power Sources of Plutonium-238 as a Function of Penetration Depth of Shielding Material. Center of source to dose point separation distance = 100 cm. Refer to Fig. 2 to obtain dose rates through other materials.

Curium-242

For some of the investigations on the moon, Cm^{242} is being developed as the radioisotopic power source for the necessary, remote, unmanned monitoring and instrumentation.[35] Production capability for this isotope is being developed. A reason for the use of this material is that since Cm^{242} and its daughter, Pu^{238}, are man-made elements, their existence as contaminants in the lunar chemicals would not interfere with precise analytical efforts to determine lunar compositions. Curium-242 has a half-life of 163 days. Its specific power is about 120 w/g, and the power density for pure Cm_2O_3 should be about 1170 w/cc. It is almost a pure alpha emitter and thus requires minimum gamma shielding. Its spontaneous fission half-life is 7.2×10^6 yr, about half that of Cm^{244}. Spontaneous fission would produce about 2.3×10^7 neutrons per second per gram of Cm^{242}. In addition α,n reactions on oxygen in Cm_2O_3 would produce approximately 2×10^7 neutrons per second per gram of Cm^{242}. This neutron production rate is sufficient to require a moderate neutron shield (water, lithium hydride, etc.) to attenuate neutron dose rates to a tolerable level. The neutron production rate from Cm^{242} would be approximately 1/12 of that from Cm^{244} per watt of source power. As with other alpha emitters, its biological hazard is much higher than that of fission products or Co^{60}. It is made by neutron irradiation of Am^{241}, which is available only as a decay product of Pu^{241}. In spite of the high cost of the raw material and other costs of irradiation and chemical processing the high specific power may lead to fairly reasonable costs per initial watt. The short half-life limits its utility and thus establishes Cm^{242} in the special-applications category where a short half-life product can be used. The calculations in this study were based on dilution of the curium oxide with an inert oxide matrix to achieve a reasonable specific power of 150 w/cc for the source pellet, with an equal volume of void space provided for gas collection, thus reducing the effective power density to 75 w/cc.

Isotopic heat source data for Cm^{242} sources appear in Table 22, and the nuclear data appear in Table 3. The results of the radiation-intensity calculations for shielded and unshielded sources are given in Figs. 101

through 105. In addition, Fig. 106 shows the neutron intensities from water-shielded sources. Figure 2 should be used in conjunction with this figure in order to determine neutron radiation intensities when Cm^{242} is shielded by other materials.

Table 22. Physical Properties of Curium-242 as an Isotopic Power Source

General Properties

Half-life	163 d
Specific activity of pure isotope, curies/g	3320
Specific power of pure isotope, w/g	120
α,n rate, neutrons/g-sec	1.99×10^7
Spontaneous fission rate, neutrons/g-sec	2.30×10^7
Compound	Cm_2O_3 in matrix
Weight of active isotope in matrix, g/cc	0.615
Effective density of matrix, g/cc	4.85
Power, w/cc of matrix	75
Activity, curies/cc of matrix	2050

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
1.19×10^8	0.10[e]
2.36×10^7	0.158
2.03×10^5	1.0[a]
5.08×10^4	1.5[a]
5.38×10^4	2.3[a]
9.55×10^3	3.0[a]
1.27×10^4	5.0[a]
1.80×10^5	0.63[c]
6.66×10^4	1.1[c]
7.80×10^4	1.55[c]
1.50×10^4	2.38[c]
2.22×10^4	2.75[c]

[a]Spontaneous fission gamma source.

[c]Fission product (from spontaneous fission) gamma source.

[e]Predominant radiation from unshielded 5000 watt source.

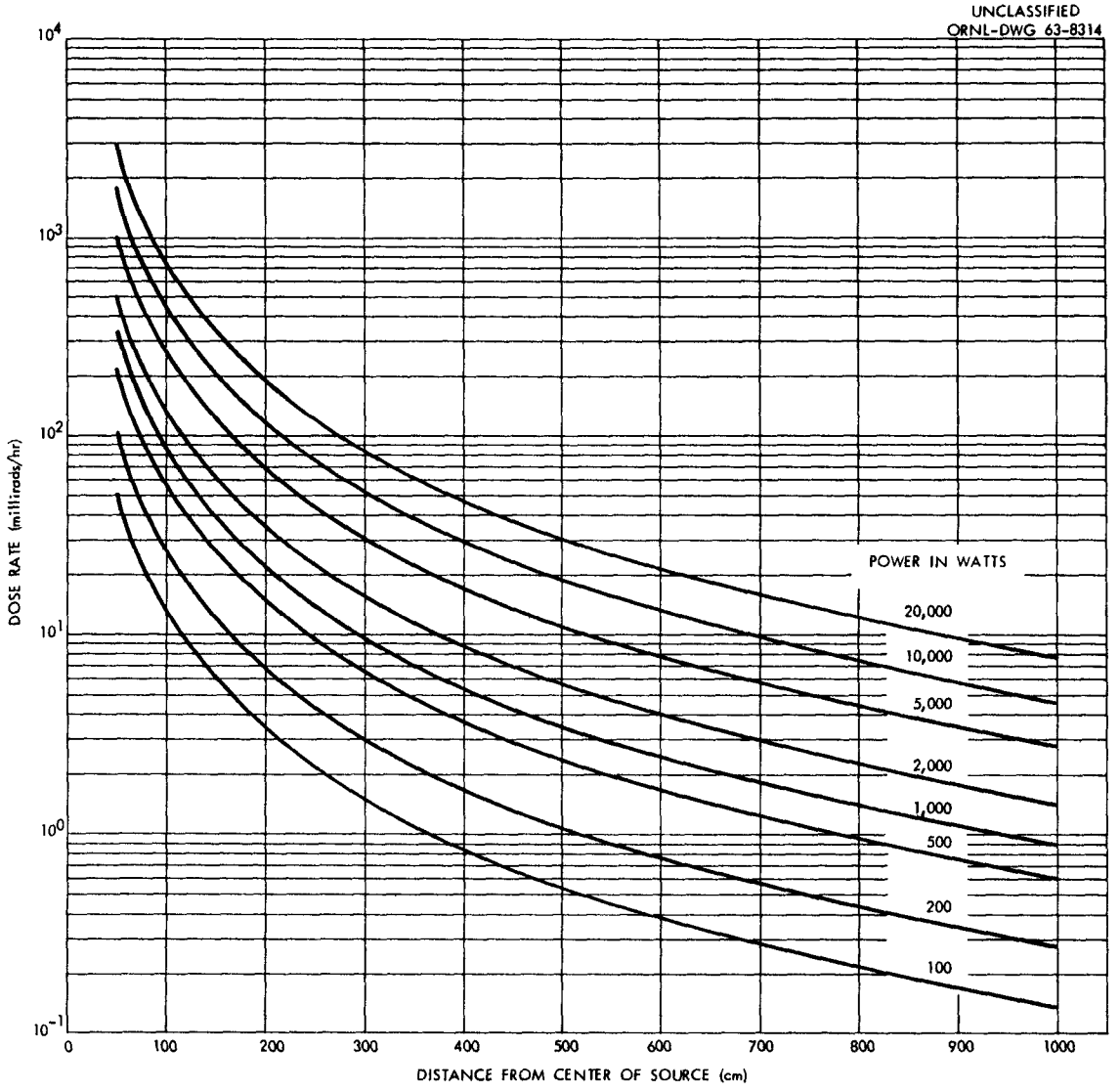


Fig. 101. Gamma Dose Rates from Unshielded Isotopic Power Sources of Curium-242 as a Function of Distance from Center of Source.

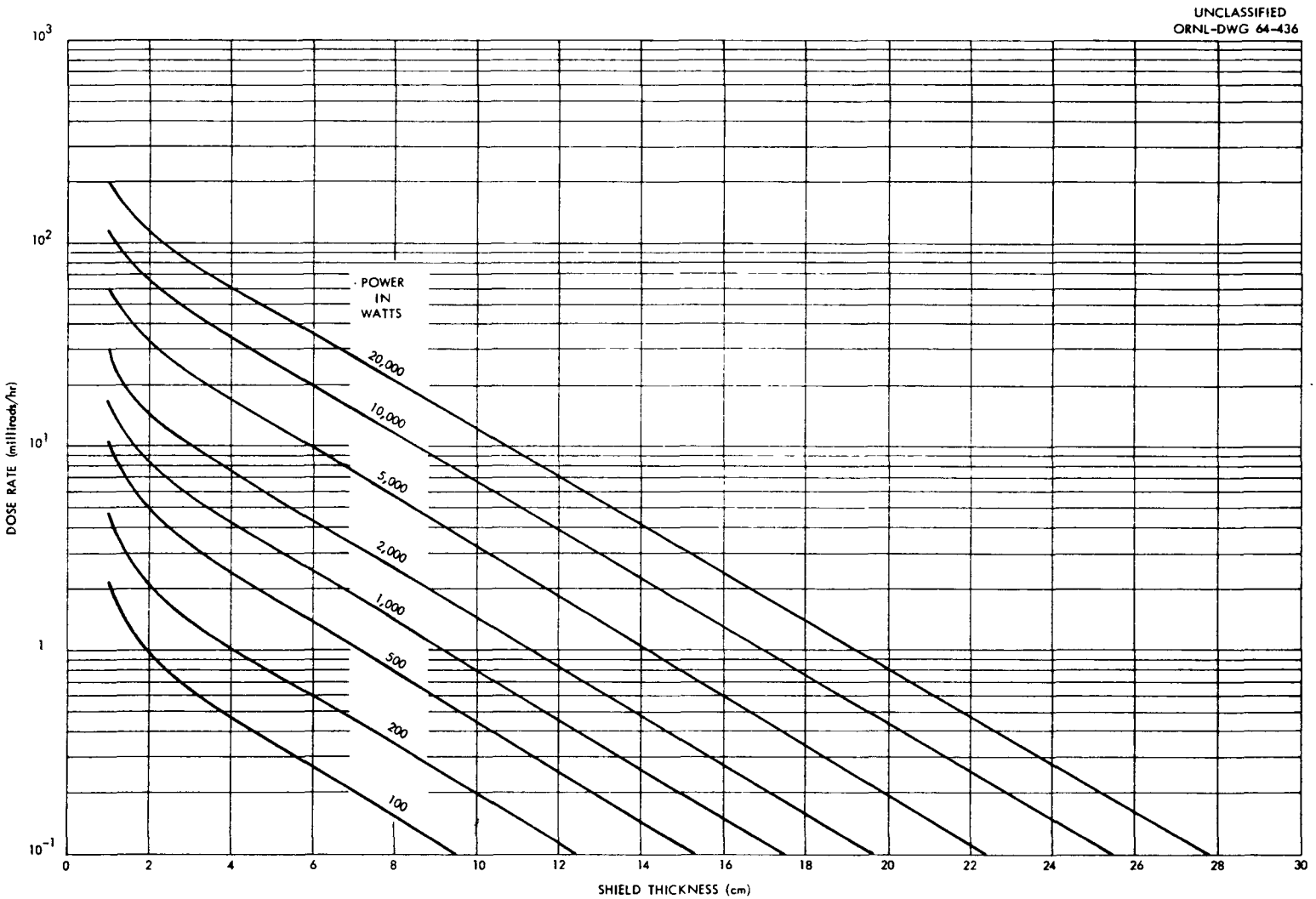


Fig. 102. Gamma Dose Rates from Iron-Shielded Isotopic Power Sources of Curium-242. Center of source to dose point separation distance = 100 cm.

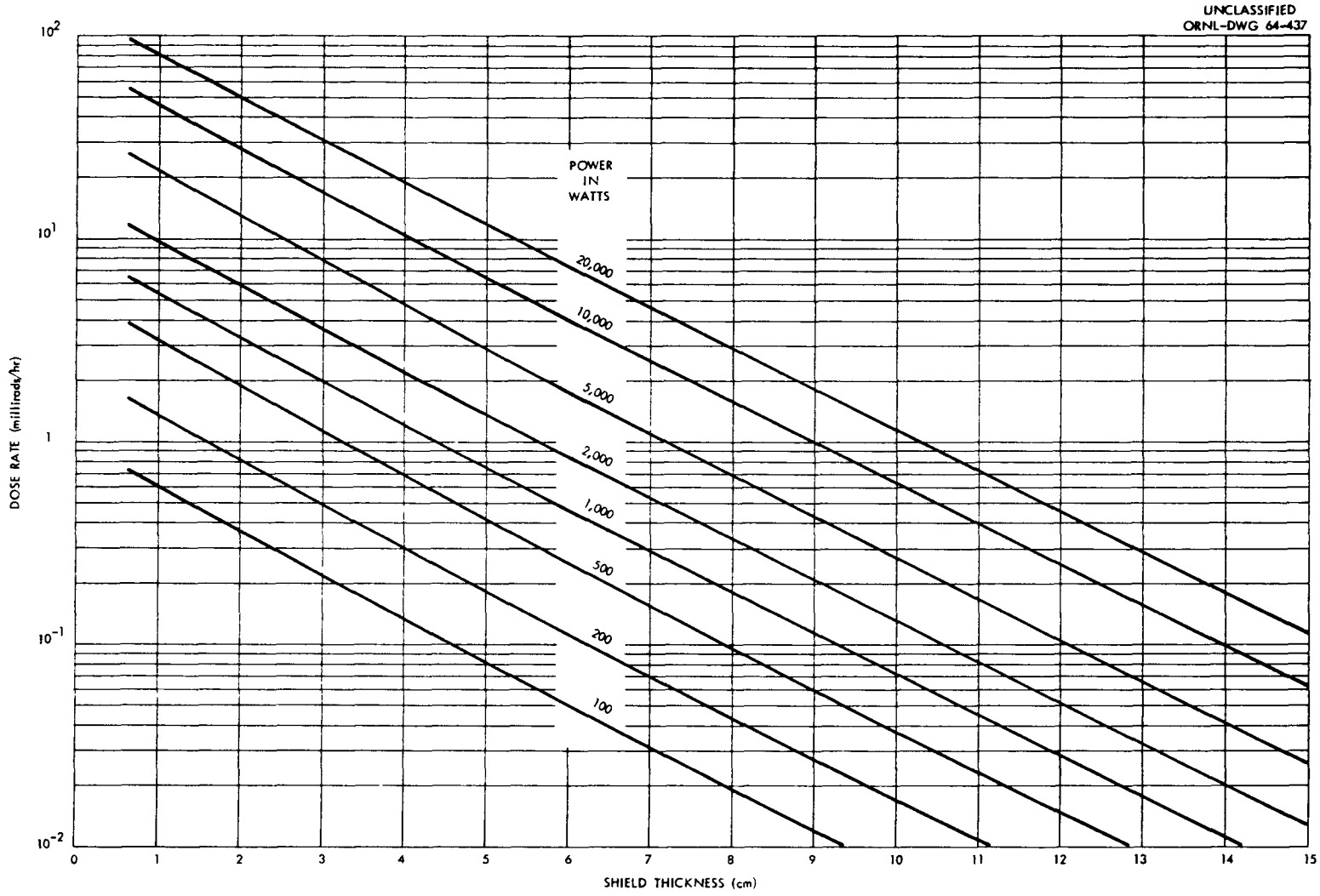


Fig. 103. Gamma Dose Rates from Lead-Shielded Isotopic Power Sources of Curium-242. Center of source to dose point separation distance = 100 cm.

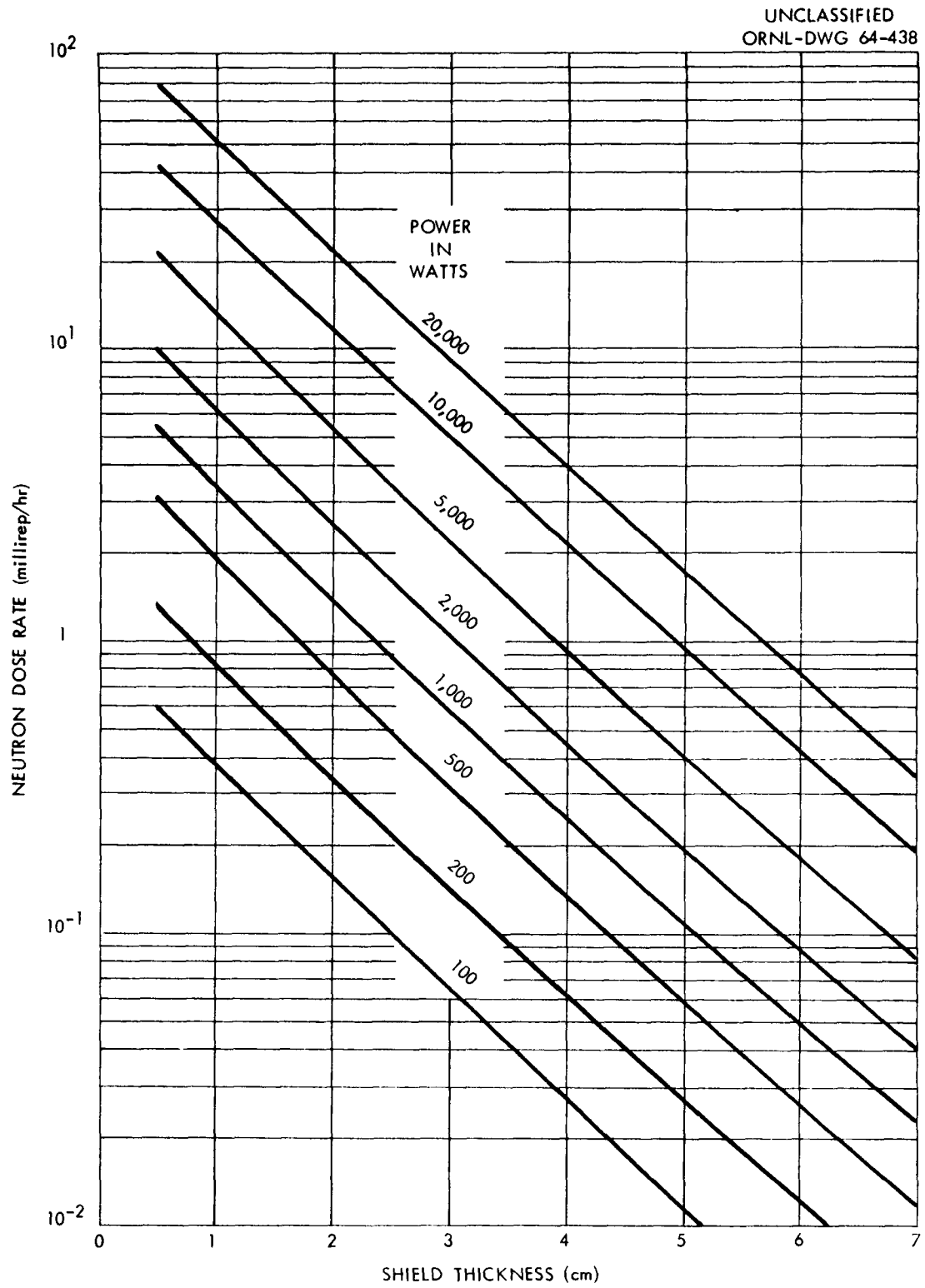


Fig. 104. Gamma Dose Rates from Uranium-Shielded Isotopic Power Sources of Curium-242. Center of source to dose point separation distance = 100 cm.

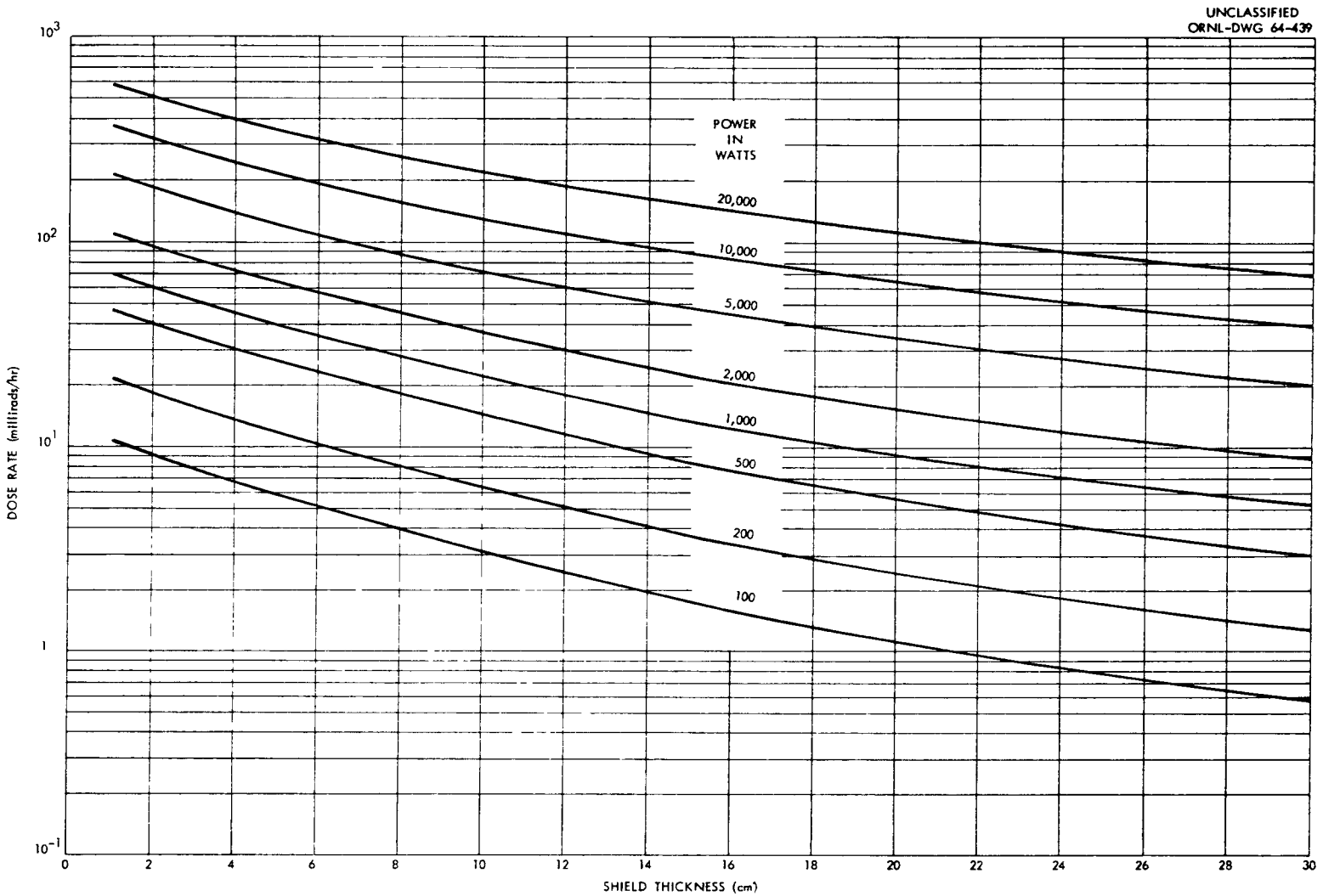


Fig. 105. Gamma Dose Rates from Water-Shielded Isotopic Power Sources of Curium-242. Center of source to dose point separation distance = 100 cm.

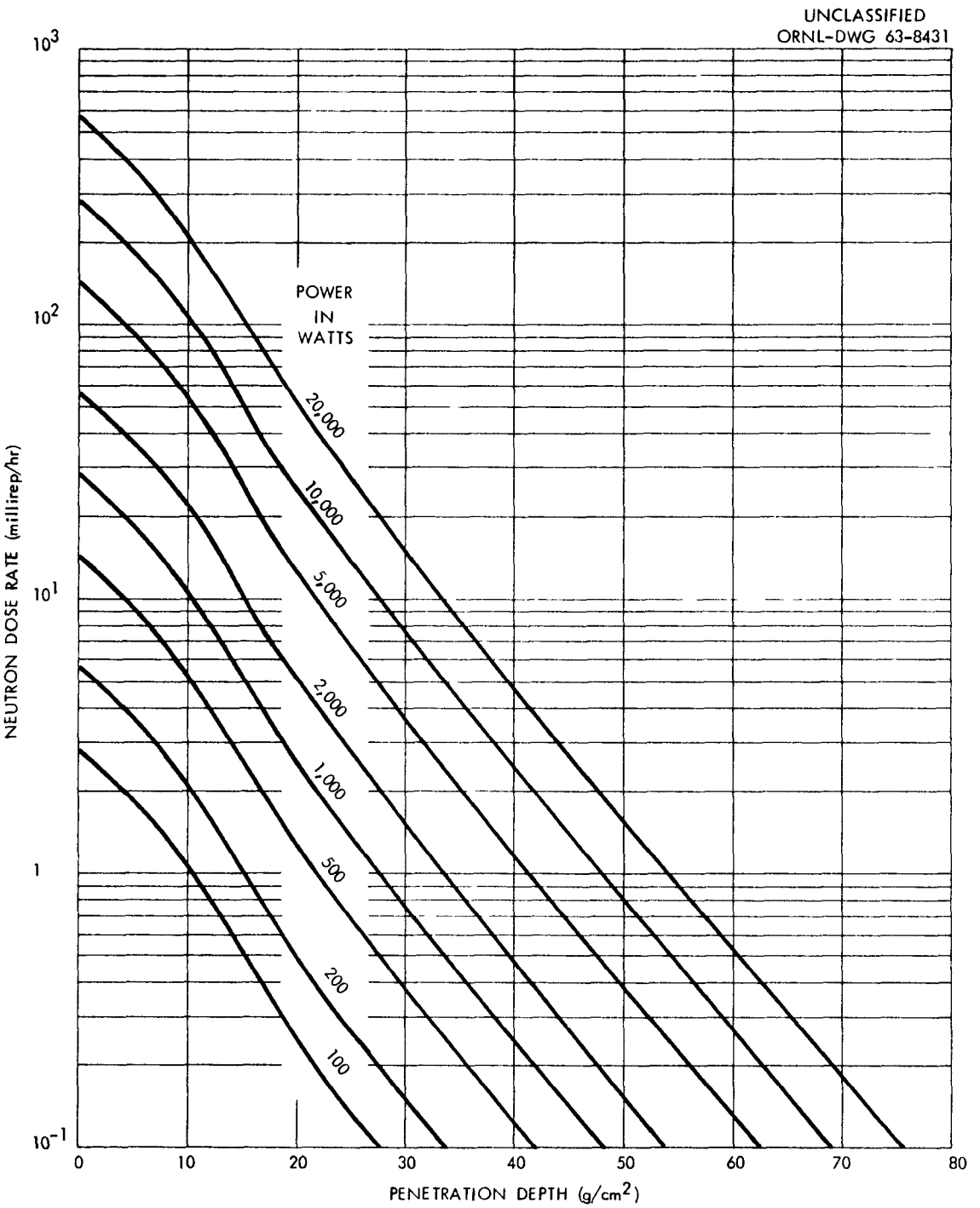


Fig. 106. Neutron Dose Rates from Water-Shielded Isotopic Power Sources of Curium-242 as a Function of Penetration Depth of Shielding Material. Center of source to dose point separation distance = 100 cm. Refer to Fig. 2 to obtain dose rates through other materials.

Curium-244

Curium-244 has a very desirable half-life of 18.4 yr. It is an alpha emitter with radiation characteristics very similar in composition and energy to those of Pu^{238}. Gamma shielding should therefore also be minor. However, the spontaneous fission half-life (1.4×10^7 yr) is much shorter than that for Pu^{238}. Spontaneous fission would produce approximately 1.2×10^7 neutrons per second per gram of Cm^{244}. In addition, α,n reactions on oxygen in Cm_2O_3 would produce approximately 4.2×10^5 neutrons per second per gram of Cm^{244}. This neutron production rate is sufficient to require a thick neutron shield (water, lithium hydride, etc.) to attenuate neutron dose rates to a tolerable level. The resulting weights of cylindrical shields for the sources used for space applications would be greater for Cm^{244} than for many of the other possible source materials. Although the neutron production rate from Cm^{244} would be about half that from Cm^{242} on the weight basis, Cm^{244} would produce about 12 times as many neutrons as Cm^{242} per watt of source power.

Its specific power is calculated to be 2.86 w/g. The power density for the oxide should then be about 27 w/cc. Like other alpha emitters, Cm^{244} is a great biological hazard. The material is only slightly less hazardous than Pu^{238}. It is produced by successive neutron captures in plutonium and finally in Am^{243}. It is unique in that in this whole irradiation chain it has the smallest neutron capture cross section. Thus, when plutonium is exposed to prolonged irradiation, one of the transmutation products reaching highest concentration in the residues is Cm^{244}. In addition, this minimum cross section should ensure material of good purity since the adjacent isotopes have very high cross sections. The current Transuranium Program is providing data on this buildup of Cm^{244} (refs. 36 and 37). In this program, several kilograms of plutonium are being irradiated to produce ultimately a few hundred milligrams of Cf^{252}. In view of the conveniently low cross section of Cm^{244}, it provides a sort of stopping place or plateau on the route to Cf^{252}. At this point where a large fraction of the plutonium has disappeared, the residual plutonium is nearly pure Pu^{242}, with relatively high concentrations of Am^{243} and Cm^{244}. These are to be separated and are then to be converted to the Cf^{252} in the HFIR via a few more years of irradiation.

It is concluded that on a watt-of-product (Cm^{244}) basis today, production by irradiation of Pu^{239} is probably an economical method of producing Cm^{244} for use as a heat source, in comparison with present or projected costs for Pu^{238}. This is fairly certain to be the case, if a low value, such as $8.00 per gram, is used for the cost of Pu^{239}. However, when plutonium is used routinely in power reactors to provide a significant fraction of the fuel, substantial Cm^{244} production may be realized by relatively simple treatments or alterations in the normal fuel recovery process. The cost of Cm^{244} should then be much less than projected Pu^{238} costs. This possibility, it should be emphasized, represents a situation that will exist only in the somewhat indefinite future.

Isotopic heat source data for Cm^{244} sources appear in Table 23, and the nuclear data appear in Table 3. The results of the radiation intensity calculations for shielded and unshielded sources are given in Figs. 107 through 111. In addition, Fig. 112 shows the neutron intensities from water-shielded sources. Figure 2 should be used in conjunction with this figure in order to determine neutron radiation intensities when Cm^{244} is shielded by other materials.

Table 23. Physical Properties of Curium-244 as an Isotopic Power Source

General Properties

Half-life	18.4 y
Specific activity of pure isotope, curies/g	79.8
Specific power of pure isotope, w/g	2.74
α,n rate, neutrons/g-sec	4.2×10^5
Spontaneous fission rate, neutrons/g-sec	1.164×10^7
Compound	Cm_2O_3
Weight of active isotope in compound, g/cc	9.63
Effective density of compound, g/cc	10.6
Power, w/cc of compound	26.4
Activity, curies/cc of compound	770

Gamma Radiation Properties

Gamma Emission Rate For Shielding Calculations (photons/w-sec)	Photon Energy (Mev)
5.80×10^7	0.1
1.399×10^7	0.15
4.34×10^6	1.0[a,e]
1.084×10^6	1.5[a]
1.15×10^6	2.3[a]
2.033×10^5	3.0[a]
2.71×10^5	5.0[a]
3.84×10^6	0.63[c]
1.42×10^6	1.1[c]
1.66×10^6	1.55[c]
3.198×10^5	2.38[c]
4.732×10^5	2.75[c]

[a]Spontaneous fission gamma source.

[c]Fission product (from spontaneous fission) gamma source.

[e]Predominant radiation from unshielded 5000 watt source.

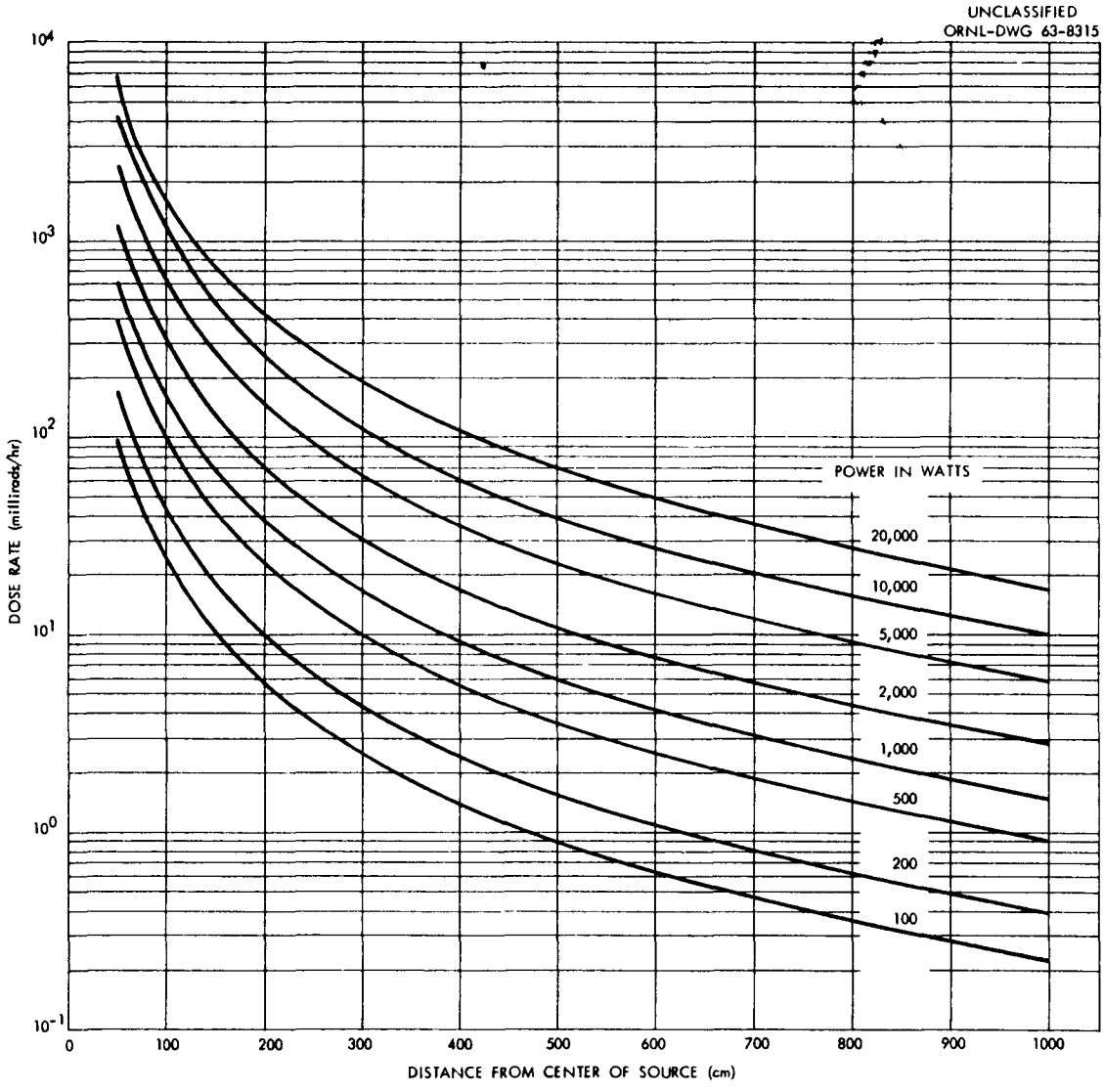


Fig. 107. Gamma Dose Rates from Unshielded Isotopic Power Sources of Curium-244 as a Function of Distance from Center of Source.

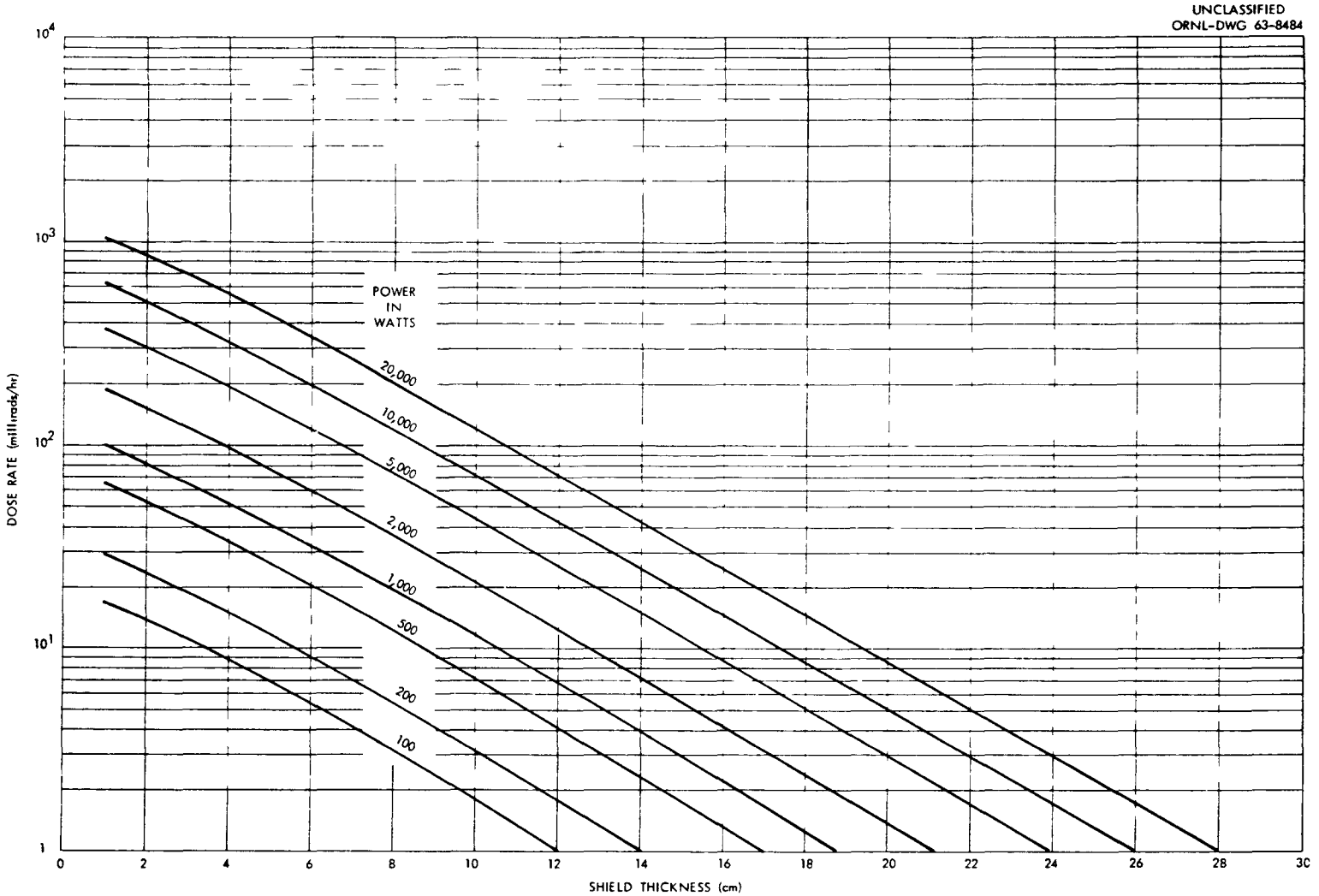


Fig. 108. Gamma Dose Rates from Iron-Shielded Isotopic Power Sources of Curium-244. Center of source to dose point separation distance = 100 cm.

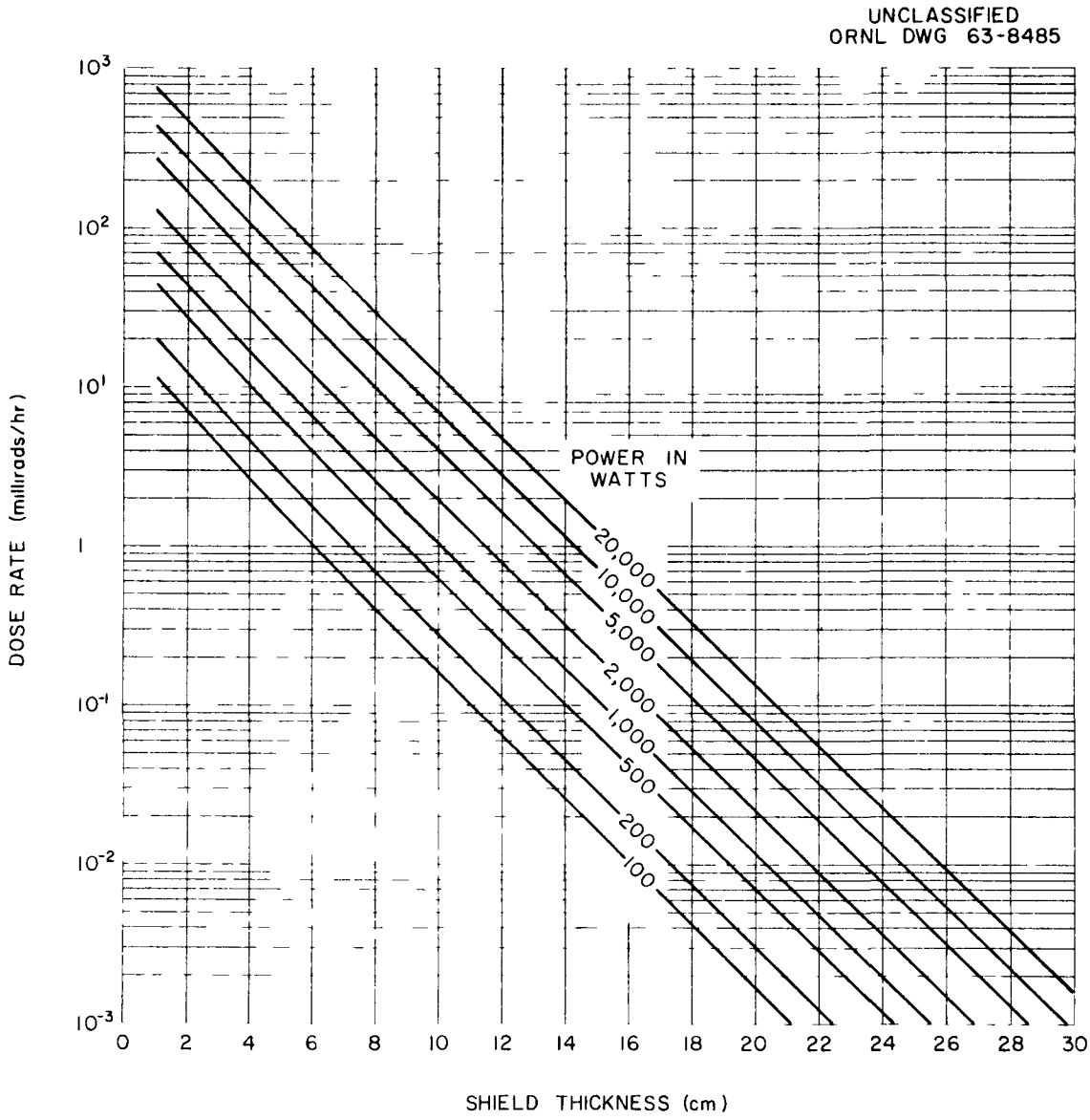


Fig. 109. Gamma Dose Rates from Lead-Shielded Isotopic Power Sources of Curium-244. Center of source to dose point separation distance = 100 cm.

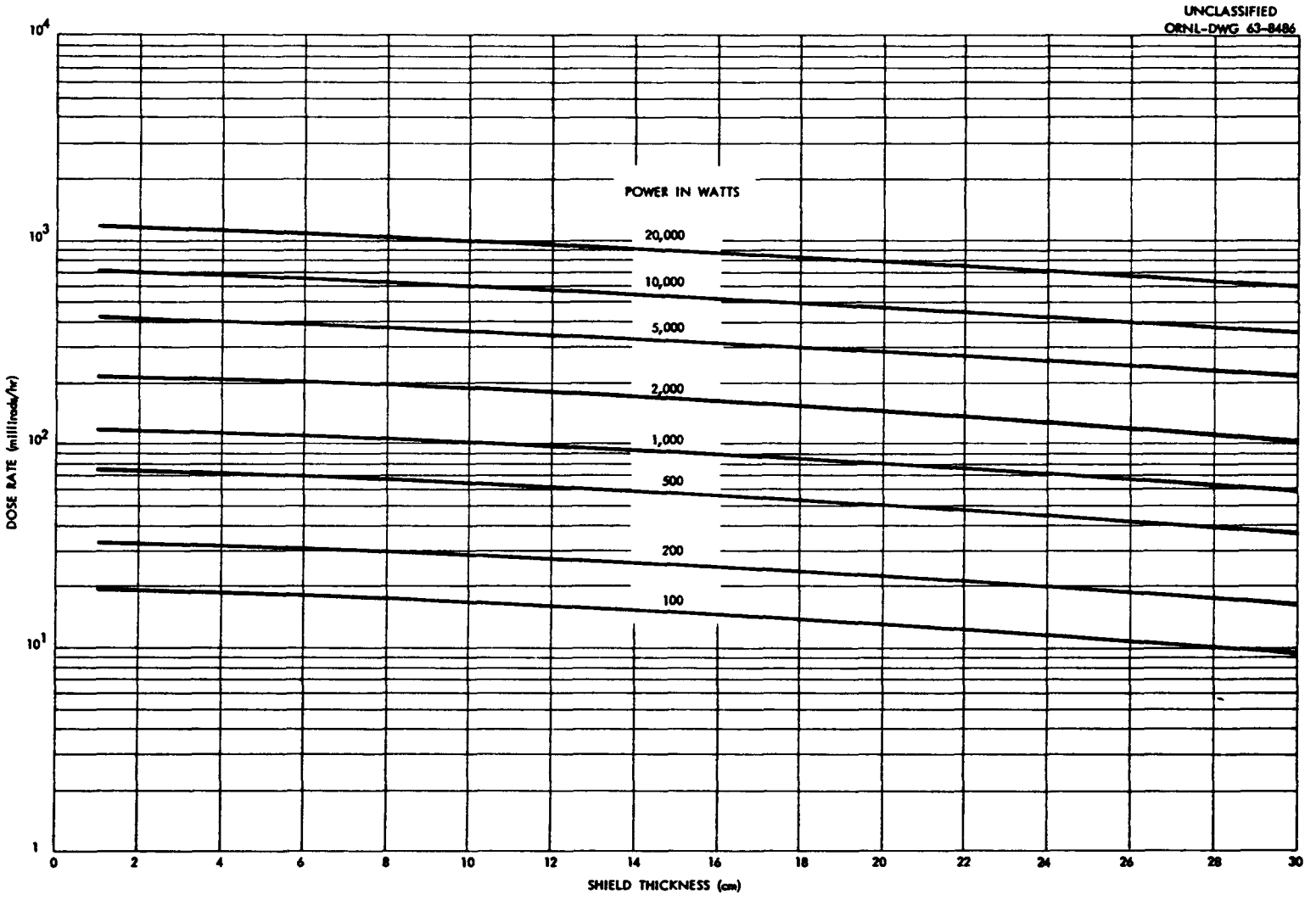


Fig. 110. Gamma Dose Rates from Water-Shielded Isotopic Power Sources of Curium-244. Center of source to dose point separation distance = 100 cm.

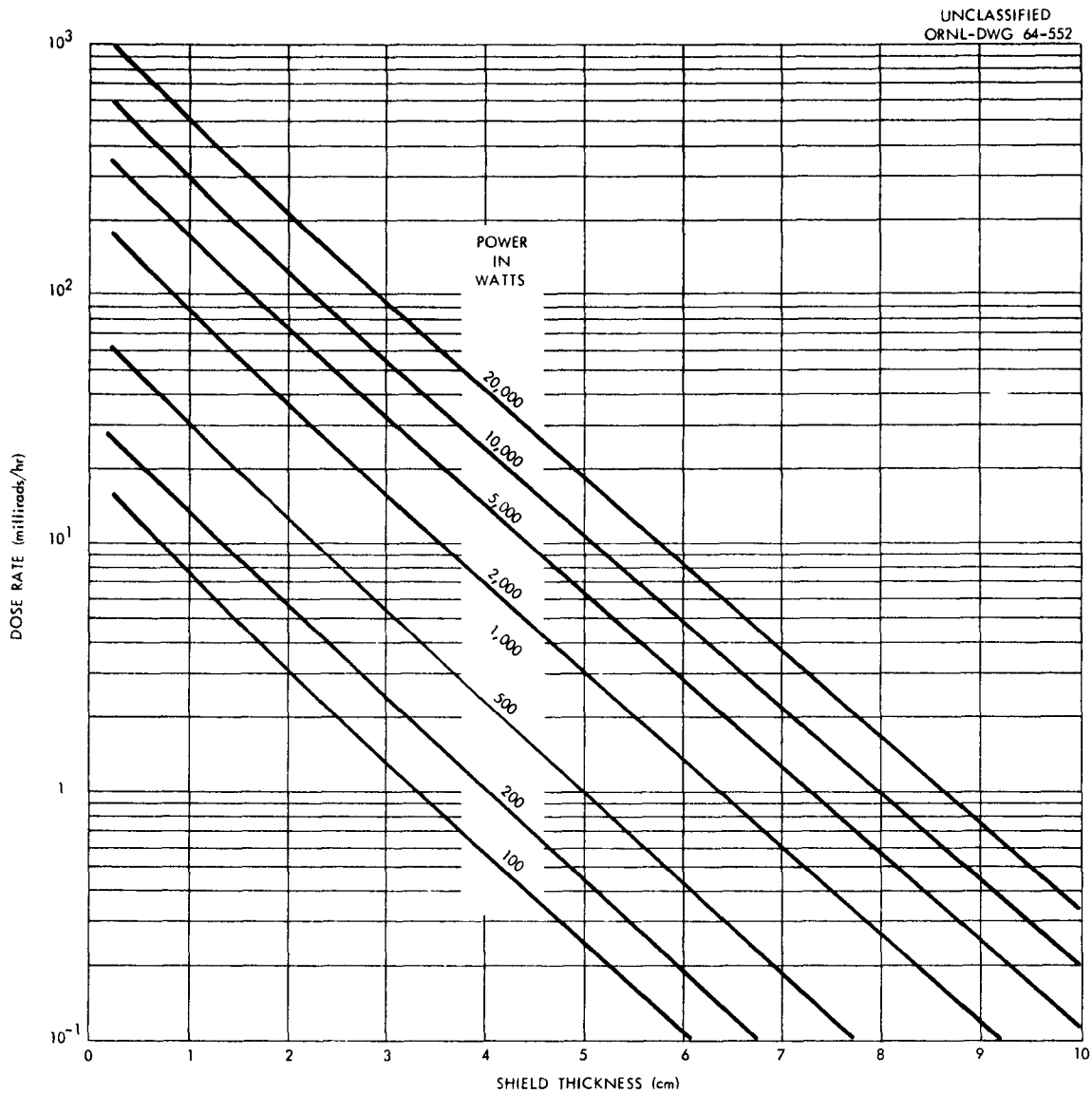


Fig. 111. Gamma Dose Rates from Uranium-Shielded Isotopic Power Sources of Curium-244. Center of source to dose point separation distance = 100 cm

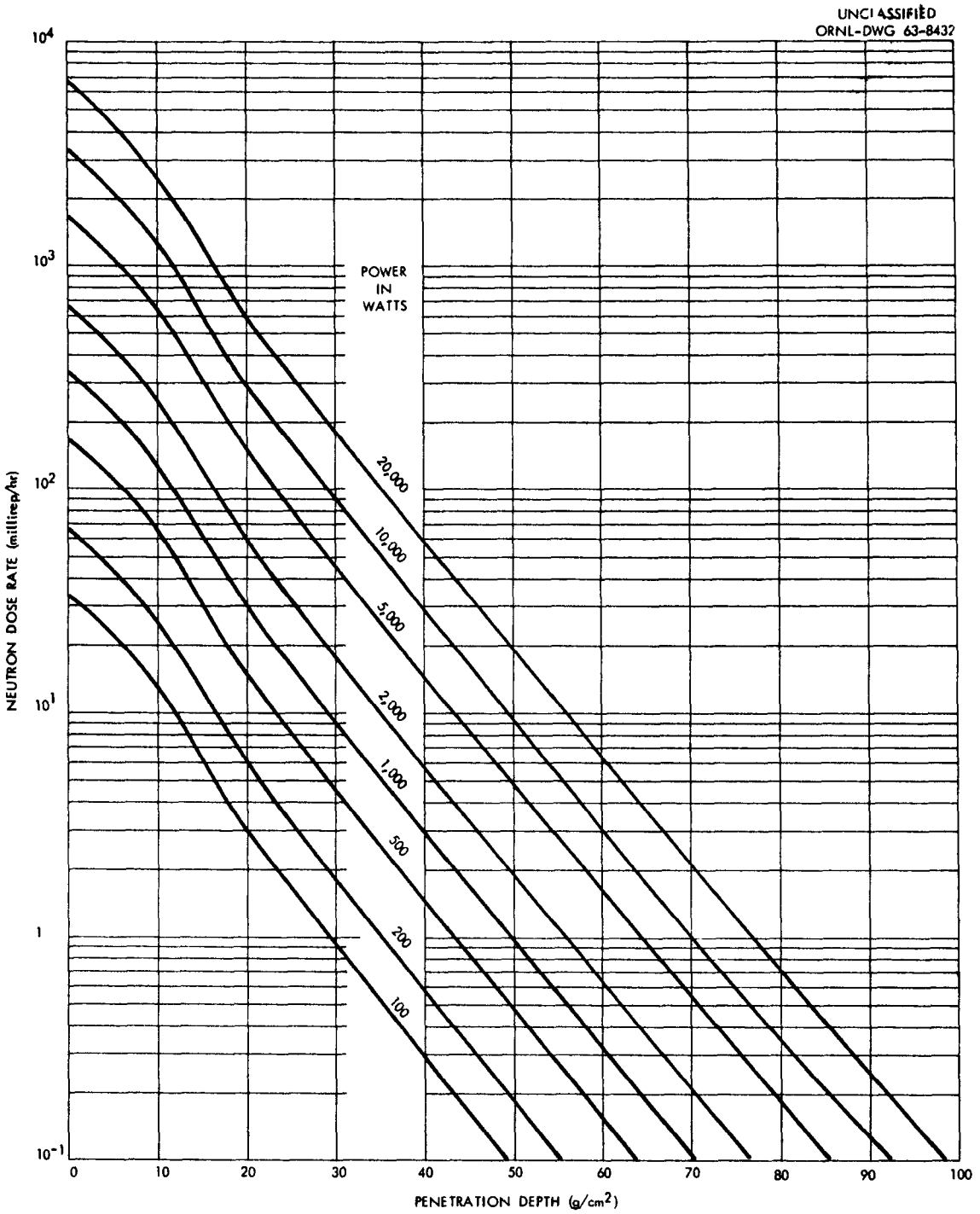


Fig. 112. Neutron Dose Rates from Water-Shielded Isotopic Power Sources of Curium-244 as a Function of Penetration Depth of Shielding Material. Center of source to dose point separation distance = 100 cm. Refer to Fig. 2 to obtain dose rates through other materials.

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APPENDIX A

This appendix contains copies of FORTRAN lists of Program Spectra, the necessary mathematical subroutines for this program, and the results from the output of Program Spectra, which calculated the average beta particle energy and the bremsstrahlung spectra. The program was originally written by Miss Robin Smith and then revised by the author to allow a more narrow energy spacing. The associated subroutines which calculate the gamma function of complex argument in the evaluation of the Fermi Differential Function were supplied by the CDC-1604 Subroutine Library of the Oak Ridge National Laboratory Mathematics Division. The tables listing the results from Program Spectra also list the hand calculated bremsstrahlung energy production in units of Mev/beta particle.

NOTE: All letter O's are marked as ∅ to distinguish them from zeros.

FORTRAN Listing of Program Spectra and Subroutines

```
      PRØGRAM SPECTRA
C  PRØGRAM TØ FIND N(L), THE TØTAL NUMBER ØF
C  BREMSSTRAHLUNG X-RAYS ØF ENERGY K PER DELTA L
C  PHØTØN ENERGY INTERVAL PER BETA EMISSIØN
C  DELTA L = SPACE*0.01 MEV = SPACE*DELTA K
C  DELTA K = 0.01 MEV,
      DIMENSIØN P(500), S1(500),XN(500),XM(500),TITLE(9)
      EQUIVALENCE (P,XN)
   10 READ (50,100)TITLE,EO,Z,ZSTAR,TQ,SIGN,SPACE
  100 FØRMAT (9A8/6E10.0)
      WO=EO/0.511+1.0
      GAMMA=Z/137.04
      S=SQRTF(-GAMMA*GAMMA+1.0)-1.0
      IEO=100.0*EO
      DELTAK=0.01
      DØ 6 I=1,IEO
      W=DELTAK/0.511+1.0
      ETA=SQRTF(W*W-1.0)
      IF(TQ)2,2,1
    1 Q=1.0
      GØ TØ 3
    2 Q=ETA*ETA+(WO-W)*(WO-W)
    3 DELT=GAMMA*W/ETA
      EPI=EXPF(3.14159265*DELT)
      IF(SIGN)4,4,5
    4 EPI=1.0/EPI
    5 REAL=S+1.0
      CALL GAMEXT(REAL,DELT,U,V,PHI)
      P(I)=ETA**(S+REAL)*W*EPI*PHI
     1(EO-DELTAK)*(EO-DELTAK)*Q
    6 DELTAK=DELTAK+0.01
      ABØVE=0.0
      BELØW=0.0
      FJ=0.01
      DØ 8 J=1,IEO
      S1(J)=0.0
      DELTAK=FJ
      DØ 7 K=J,IEO
      FJK=FJ/DELTAK
      T=ZSTAR*DELTAK
      S1(J)=((T*P(K))/(T+800.0))*(4.0*(1.0-FJK)+3.0*FJK*
     1LØGF(FJK))+S1(J)
    7 DELTAK=DELTAK+0.01
      S1(J)=0.008*S1(J)
      DELTAK=FJ
      ABØVE=DELTAK*P(J)+ABØVE
      BELØW=P(J)+BELØW
      FJ=FJ+0.01
    8 XN(J)=S1(J)/DELTAK
      EBAR=ABØVE/BELØW
      WRITE(51,101)TITLE,EO,EBAR
```

```
101 FØRMAT(1H1,47X,9A8/86X,21HMAXIMUM BETA ENERGY =
   1E13.5/86X,21HAVERAGE BETA ENERGY =E13.5///
   233X,14HBREMSSTRAHLUNG,20X,17HNUMBER ØF PHØTØNS/
   336X,11HENERGY(MEV),22X,15HPER BETA  N(K)///)
    DELTAK=0.01
    DELTAL=SPACE*0.01
    LEO=IEO/(XFIXF(SPACE))
    DØ 9 L=1,LEO
    LSP=L*(XFIXF(SPACE))
    DØ 19 I=LSP,IEO
    IF(I-LSP)19,99,9
 99 XM(L)=XN(I)*SPACE/BELØW
    WRITE(51,102)DELTAL,XM(L)
102 FØRMAT(35X,F10.5,24X,E13.5)
    DELTAL=DELTAL+(0.01*SPACE)
    GØ TØ 9
 19 CØNTINUE
  9 CØNTINUE
    GØ TØ 10
    END SPECTRA
```

```
    SUBRØUTINE GAMEXT(XX.YY,GRE,GIM,GMØD)
    X=XX
    Y=YY
    IF (Y.EQ.0.)200,201
200 CALL RCPGAMR(X,U)$V=0.$GØ TØ 202
201 NN=X
    X1=NN
    FRACT=X-X1
    IF(FRACT.EQ.0..AND.X.LE.0.)203,204
203 WRITE(51,1)X$GØ TØ 999
  1 FØRMAT(16H1ARG ØF GAMEXT =E12.2)
204 YABS=ABSF(Y)
    IF(YABS-1.)100,100,101
100 CALL RCPGAMI(FRACT,Y,U,V)
202 RAD= 1./(U*U+V*V)
    FO=U*RAD
    GO=-V*RAD
    IF(Y.EQ.0.)205,111
205 F=FO$G=GO$GØ TØ 206
101 IF(YABS-2.)102,102,104
102 TRE=FRACT*0.5
    TIM=Y*0.5
    TRE1=TRE+0.5
    CALL RCPGAMI(TRE,TIM,U,V)
    RAD=1./(U*U+V*V)
    SO=U*RAD
    TO=-V*RAD
    CALL RCPGAMI(TRET,TIM,U,V)
    RAD=1./(U*U+V*V)
    S1=U*RAD
    T1=-V*RAD
103 EX=EXPF(FRACT*0.6931471806)
    ARG=Y*0.6931471806
    CØSINE=CØSF(ARG)
    SINE=SINF(ARG)
    U1=SO*S1-TO*T1
    V1=SO*T1*S1*TO
    ARG=0.282094792*EX
    FO=ARG*(CØSINE*UT-SINE*V1)
    GO=ARG*(CØSINE*VT+SINE*U1)
    GØ TØ 111
104 IF(YABS-4.)105,105,999
105 TRE=0.25*FRACT
    TIM=0.25*Y
    DØ 110 N=1,4
    N=N
    CALL RCPGAMI(TRE,TIM,U,V)
    RAD=1./(U*U+V*V)
    S1=U*RAD
    T1=-V*RAD
    GØ TØ (106,107,108,109),N
```

```
106 U0=S1
    V0=T1
    GØ TØ 110
107 U1=S1
    V1=T1
    GØ TØ 110
108 U2=S1
    V2=T1
    GØ TØ 110
109 U3=S1
    V3=T1
110 TRE=TRE+0.25
    EX=EXPF(FRACT*0.3465735903)
    ARG=Y*0.3465735903
    CØSINE=CØSF(ARG)
    SINE=SINF(ARG)
    TRE=U0*U2-V0*V2
    TIM=U0*V2+V0*U2
    ARG=EX*0.282094792
    S0=ARG*(CØSINE*TRE-SINE*TIM)
    T0=ARG*(CØSINE*TIM+SINE*TRE)
    ARG=ARG*1.41421356
    TRE=U1*U3-V1*V3
    TIM=U1*V3+V1*U3
    S1=ARG*(CØSINE*TRE-SINE*TIM)
    T1=ARG*(SINE*TRE+CØSINE*TIM)
    GØ TØ 103
111 F=F0
    G=G0
    FPN=FRACT
    DØ 112 N=1,NN
    TRE=FPN*F-Y*G
    G=FPN*G+Y*F
    F=TRE
112 FPN=FPN+1.
206 GRE=F
    GIM=G
    GMØD=SQRTF(F*F+G*G)
999 RETURN
    END
```

```
      SUBRØUTINE RCPGAMI(S,P,Q,R)
      DIMENSIØN C(21)
      X=S
      Y=P
      IF(C(2)-0.25)100,101,100
  100 C(21)=2.E-12
      C(20)=2.6E-11
      C(19)=-2.95E-10
      C(18)=1.251E-9
      C(17)=1.529E-9
      C(16)=-5.1408E-8
      C(15)=2.83257E-7
      C(14)=-3.12624E-7
      C(13)=-5.033714E-6
      C(12)=3.2012571E-5
      C(11)=-5.3810419E-5
      C(10)=-2.91291898E-4
      C(9)=.001804735812
      C(8)=-.002405492882
      C(7)=-.010549433639
      C(6)=.041634652846
      C(5)=-.010500658759
      C(4)=-.163969517880
      C(3)=.144303916225
      C(2)=0.25
      C(1)=0.
  101 U=X*C(21)+C(20)
      V=Y*C(21)
      DØ 102 I=1,19
      K=20-I
      G=U*X-V*Y
      V=U*Y+V*X
  102 U=G+C(K)
      Q=U*4.
      R=V*4.
      RETURN
      END
```

```
      SUBRØUTINE RCPGAMR (V,W)
C1 UCSD RCPGAMR
C     CALCULATIØN ØF THE RECIPRØCAL ØF THE GAMMA FUNCTIØN
C     THE RANGE ØF THE ARGUMENT V IS UNRESTRICTED, SINCE THE
C     RELATION
C           GAMMA(1+X)=X*GAMMA(X)
C     IS USED TØ BRING THE ARGUMENT INTØ THE RANGE
C           -1 LEQ X LEQ 1
C     FØR USE IN THE SUBSIDIARY RØUTINE RGAM,WHICH CØMPUTES THE
C     RECIPRØCAL ØF THE GAMMA FUNCTIØN USING A 14-TH DEGREE PØLYNØMIAL.
C
      R=V
      IF(R)2,1,2
    1 S=0.0
      GØ TØ 100
    2 IF(R-1.)8,6,3
    3 T=R
      Y=1.
    4 R=R-1.
      Y=Y*R
      IF(R-1.)5,7,4
    5 CALL RGAM(R,S)
      S=S/Y
      GØ TØ 100
    6 S=1.
      GØ TØ 100
    7 S=1./Y
      GØ TØ 100
    8 IF(R+1.)10,1,9
    9 CALL RGAM(R,S)
      GØ TØ 100
   10 IR=R
      PQ=IR
   18 IF(PQ-R)12,11,12
   11 S=0.
      GØ TØ 100
   12 Y=R
   13 R=R+1.
      IF(R+1.)14,15,16
   14 Y=Y*R
      GØ TØ 13
   15 STØP 0001
   16 CALL RGAM(R,S)
      S=S*Y
  100 W = S
      RETURN
      END
```

```
      SUBRØUTINE RGAM(U,V)
      CØMMØN/BCØE/B(14)
      P=U
      Q=B(1)
      DØ 2 N=2,14
    2 Q=Q*P+B(N)
      Q=Q*P*(P+B(14))
      V=Q
      RETURN
      END
         IDENT     BTBLE                TABLE ØF B FØR RGAM. THEY ARE
BCØE     BLØCK                          STØRED IN REVERSE ØRDER TØ SIMPLIFY
         CØMMØN    B(14)                THE CALCULATIØN ØF THE PØLYNØMIAL.
         ØRGR      B
 B       DEC       -1.8122E-7           B(14)
         DEC       +1.328554E-6         B(13)
         DEC       -2.625721E-6         B(12)
         DEC       -1.7527917E-5        B(11)
         DEC       +1.45624324E-4       B(10)
         DEC       -3.60851496E-4       B(9)
         DEC       -8.04341335E-4       B(8)
         DEC       +.008023278113       B(7)
         DEC       -.017645242118       B(6)
         DEC       -.024552490887       B(5)
         DEC       +.191091101162       B(4)
         DEC       -.233093736365       B(3)
         DEC       -.422784335092       B(2)
         DEC       +1.                  B(1)
         END
```

Table A.1. Production of Bremsstrahlung Photons From Krypton-85 Beta in Liquid Krypton

Maximum beta-particle energy, Mev	0.672
Average beta-particle energy, Mev	0.253

Bremsstrahlung Energy Group (Mev)	Number of Photons per Beta Particle Within ΔE Energy Group
0.050 ± 0.025	2.165×10^{-2}
0.100 ± 0.025	7.228×10^{-3}
0.150 ± 0.025	3.238×10^{-3}
0.200 ± 0.025	1.607×10^{-3}
0.250 ± 0.025	8.264×10^{-4}
0.300 ± 0.025	4.233×10^{-4}
0.350 ± 0.025	2.094×10^{-4}
0.400 ± 0.025	9.668×10^{-5}
0.450 ± 0.025	3.981×10^{-5}
0.500 ± 0.025	1.355×10^{-5}
0.550 ± 0.025	3.285×10^{-6}
0.600 ± 0.025	3.831×10^{-7}
0.650 ± 0.025	2.628×10^{-9}
Total bremsstrahlung energy, Mev/beta particle	3.085×10^{-3}

Table A.2. Production of Bremsstrahlung Photons From Strontium-90 Beta in Strontium Oxide Matrix

Maximum beta-particle energy, Mev	0.545
Average beta-particle energy, Mev	0.201

Bremsstrahlung Energy Group (Mev)	Number of Photons per Beta Particle Within ΔE Energy Group
0.020 ± 0.01	1.378×10^{-2}
0.040 ± 0.01	5.523×10^{-3}
0.060 ± 0.01	2.998×10^{-3}
0.080 ± 0.01	1.841×10^{-3}
0.100 ± 0.01	1.208×10^{-3}
0.120 ± 0.01	8.239×10^{-4}
0.140 ± 0.01	5.759×10^{-4}
0.160 ± 0.01	4.086×10^{-4}
0.180 ± 0.01	2.923×10^{-4}
0.200 ± 0.01	2.099×10^{-4}
0.220 ± 0.01	1.505×10^{-4}
0.240 ± 0.01	1.074×10^{-4}
0.260 ± 0.01	7.604×10^{-5}
0.280 ± 0.01	5.315×10^{-5}
0.300 ± 0.01	3.654×10^{-5}
0.320 ± 0.01	2.458×10^{-5}
0.340 ± 0.01	1.609×10^{-5}
0.360 ± 0.01	1.016×10^{-5}
0.380 ± 0.01	6.135×10^{-6}
0.400 ± 0.01	3.491×10^{-6}
0.420 ± 0.01	1.835×10^{-6}
0.440 ± 0.01	8.638×10^{-7}
0.460 ± 0.01	3.456×10^{-7}
0.480 ± 0.01	1.065×10^{-7}
0.500 ± 0.01	2.007×10^{-8}
0.520 ± 0.01	1.000×10^{-9}
0.540 ± 0.01	0.000
Total bremsstrahlung energy, Mev/beta particle	1.411×10^{-3}

Table A.3. Production of Bremsstrahlung Photons From Yttrium-90 Beta in Strontium Oxide Matrix

Maximum beta-particle energy, Mev	2.27
Average beta-particle energy, Mev	0.944

Bremsstrahlung Energy Group (Mev)	Number of Photons per Beta Particle Within ΔE Energy Group
0.100 ± 0.05	6.152×10^{-2}
0.200 ± 0.05	2.415×10^{-2}
0.300 ± 0.05	1.281×10^{-2}
0.400 ± 0.05	7.674×10^{-3}
0.500 ± 0.05	4.894×10^{-3}
0.600 ± 0.05	3.234×10^{-3}
0.700 ± 0.05	2.181×10^{-3}
0.800 ± 0.05	1.485×10^{-3}
0.900 ± 0.05	1.014×10^{-3}
1.000 ± 0.05	6.887×10^{-4}
1.100 ± 0.05	4.630×10^{-4}
1.200 ± 0.05	$3.0\ 0 \times 10^{-4}$
1.300 ± 0.05	1.973×10^{-4}
1.400 ± 0.05	1.231×10^{-4}
1.500 ± 0.05	7.337×10^{-5}
1.600 ± 0.05	4.117×10^{-5}
1.700 ± 0.05	2.125×10^{-5}
1.800 ± 0.05	9.740×10^{-6}
1.900 ± 0.05	3.728×10^{-6}
2.000 ± 0.05	1.058×10^{-6}
2.100 ± 0.05	1.670×10^{-7}
2.200 ± 0.05	4.768×10^{-9}
Total bremsstrahlung energy, Mev/beta particle	2.814×10^{-2}

Table A.4. Production of Bremsstrahlung Photons From Strontium-90 Beta in Strontium Titanate Matrix

Maximum beta-particle energy, Mev	0.545
Average beta-particle energy, Mev	0.201

Bremsstrahlung Energy Group (Mev)	Number of Photons per Beta Particle Within ΔE Energy Group
0.020 ± 0.01	1.009×10^{-2}
0.040 ± 0.01	4.044×10^{-3}
0.060 ± 0.01	2.195×10^{-3}
0.080 ± 0.01	1.348×10^{-3}
0.100 ± 0.01	8.845×10^{-4}
0.120 ± 0.01	6.034×10^{-4}
0.140 ± 0.01	4.217×10^{-4}
0.160 ± 0.01	2.993×10^{-4}
0.180 ± 0.01	2.141×10^{-4}
0.200 ± 0.01	1.537×10^{-4}
0.220 ± 0.01	1.103×10^{-4}
0.240 ± 0.01	7.871×10^{-5}
0.260 ± 0.01	5.571×10^{-5}
0.280 ± 0.01	3.895×10^{-5}
0.300 ± 0.01	2.678×10^{-5}
0.320 ± 0.01	1.801×10^{-5}
0.340 ± 0.01	1.179×10^{-5}
0.360 ± 0.01	7.448×10^{-6}
0.380 ± 0.01	4.497×10^{-6}
0.400 ± 0.01	2.559×10^{-6}
0.420 ± 0.01	1.345×10^{-6}
0.440 ± 0.01	6.333×10^{-7}
0.460 ± 0.01	2.534×10^{-7}
0.480 ± 0.01	7.807×10^{-8}
0.500 ± 0.01	1.472×10^{-8}
0.520 ± 0.01	7.336×10^{-10}
0.540 ± 0.01	0.000

Total bremsstrahlung energy, Mev/beta particle 9.924×10^{-4}

Table A.5. Production of Bremsstrahlung Photons From Yttrium-90 Beta in Strontium Titanate Matrix

Maximum beta-particle energy, Mev	2.27
Average beta-particle energy, Mev	0.944

Bremsstrahlung Energy Group (Mev)	Number of Photons per Beta Particle Within ΔE Energy Group
0.100 ± 0.05	4.537×10^{-2}
0.200 ± 0.05	1.782×10^{-2}
0.300 ± 0.05	9.456×10^{-3}
0.400 ± 0.05	5.665×10^{-3}
0.500 ± 0.05	3.613×10^{-3}
0.600 ± 0.05	2.389×10^{-3}
0.700 ± 0.05	1.611×10^{-3}
0.800 ± 0.05	1.098×10^{-3}
0.900 ± 0.05	7.493×10^{-4}
1.000 ± 0.05	5.092×10^{-4}
1.100 ± 0.05	3.425×10^{-4}
1.200 ± 0.05	2.264×10^{-4}
1.300 ± 0.05	1.460×10^{-4}
1.400 ± 0.05	9.110×10^{-5}
1.500 ± 0.05	5.434×10^{-5}
1.600 ± 0.05	3.050×10^{-5}
1.700 ± 0.05	1.576×10^{-5}
1.800 ± 0.05	7.220×10^{-6}
1.900 ± 0.05	2.764×10^{-6}
2.000 ± 0.05	7.850×10^{-7}
2.100 ± 0.05	1.239×10^{-7}
2.200 ± 0.05	3.540×10^{-9}
Total bremsstrahlung energy, Mev/beta particle	2.078×10^{-2}

Table A.6. Production of Bremsstrahlung Photons From Rhodium-106 Beta 1 in Metal Matrix

Maximum beta-particle energy, Mev 3.54
Average beta-particle energy, Mev 1.515

Bremsstrahlung Energy Group (Mev)	Number of Photons per Beta Particle Within ΔE Energy Group
0.100 ± 0.05	1.950×10^{-1}
0.200 ± 0.05	8.246×10^{-2}
0.300 ± 0.05	4.703×10^{-2}
0.400 ± 0.05	3.033×10^{-2}
0.500 ± 0.05	2.090×10^{-2}
0.600 ± 0.05	1.500×10^{-2}
0.700 ± 0.05	1.106×10^{-2}
0.800 ± 0.05	8.306×10^{-3}
0.900 ± 0.05	6.317×10^{-3}
1.000 ± 0.05	4.846×10^{-3}
1.100 ± 0.05	3.737×10^{-3}
1.200 ± 0.05	2.891×10^{-3}
1.300 ± 0.05	2.238×10^{-3}
1.400 ± 0.05	1.731×10^{-3}
1.500 ± 0.05	1.335×10^{-3}
1.600 ± 0.05	1.025×10^{-3}
1.700 ± 0.05	7.826×10^{-4}
1.800 ± 0.05	5.928×10^{-4}
1.900 ± 0.05	4.448×10^{-4}
2.000 ± 0.05	3.299×10^{-4}
2.100 ± 0.05	2.413×10^{-4}
2.200 ± 0.05	1.736×10^{-4}
2.300 ± 0.05	1.224×10^{-4}
2.400 ± 0.05	8.428×10^{-5}
2.500 ± 0.05	5.639×10^{-5}
2.600 ± 0.05	3.642×10^{-5}
2.700 ± 0.05	2.251×10^{-5}
2.800 ± 0.05	1.316×10^{-5}
2.900 ± 0.05	7.162×10^{-6}
3.000 ± 0.05	3.534×10^{-6}
3.100 ± 0.05	1.519×10^{-6}
3.200 ± 0.05	5.285×10^{-7}
3.300 ± 0.05	1.281×10^{-7}
3.400 ± 0.05	1.447×10^{-8}
3.500 ± 0.05	8.903×10^{-11}

Total bremsstrahlung energy, Mev/beta particle 1.288×10^{-1}

Table A.7. Production of Bremsstrahlung Photons From Rhodium-106 Beta 2 in Metal Matrix

Maximum beta-particle energy, Mev	2.40
Average beta-particle energy, Mev	0.976

Bremsstrahlung Energy Group (Mev)	Number of Photons per Beta Particle Within ΔE Energy Group
0.100 ± 0.05	1.185×10^{-1}
0.200 ± 0.05	4.663×10^{-2}
0.300 ± 0.05	2.477×10^{-2}
0.400 ± 0.05	1.485×10^{-2}
0.500 ± 0.05	9.475×10^{-3}
0.600 ± 0.05	6.263×10^{-3}
0.700 ± 0.05	4.225×10^{-3}
0.800 ± 0.05	2.880×10^{-3}
0.900 ± 0.05	1.968×10^{-3}
1.000 ± 0.05	1.341×10^{-3}
1.100 ± 0.05	9.061×10^{-4}
1.200 ± 0.05	6.039×10^{-4}
1.300 ± 0.05	3.947×10^{-4}
1.400 ± 0.05	2.514×10^{-4}
1.500 ± 0.05	1.548×10^{-4}
1.600 ± 0.05	9.114×10^{-5}
1.700 ± 0.05	5.064×10^{-5}
1.800 ± 0.05	2.602×10^{-5}
1.900 ± 0.05	1.199×10^{-5}
2.000 ± 0.05	4.707×10^{-6}
2.100 ± 0.05	1.431×10^{-6}
2.200 ± 0.05	2.723×10^{-7}
2.300 ± 0.05	1.633×10^{-8}
2.400 ± 0.05	0.000
Total bremsstrahlung energy, Mev/beta particle	5.455×10^{-2}

Table A.8. Production of Bremsstrahlung Photons From Praseodymium-144 Beta in Cerium Oxide Matrix

Maximum beta-particle energy, Mev	3.00
Average beta-particle energy, Mev	1.23

Bremsstrahlung Energy Group (Mev)	Number of Photons per Beta Particle Within ΔE Energy Group
0.100 ± 0.05	9.016×10^{-2}
0.200 ± 0.05	3.703×10^{-2}
0.300 ± 0.05	2.054×10^{-2}
0.400 ± 0.05	1.289×10^{-2}
0.500 ± 0.05	8.620×10^{-3}
0.600 ± 0.05	6.001×10^{-3}
0.700 ± 0.05	4.284×10^{-3}
0.800 ± 0.05	3.107×10^{-3}
0.900 ± 0.05	2.277×10^{-3}
1.000 ± 0.05	1.677×10^{-3}
1.100 ± 0.05	1.238×10^{-3}
1.200 ± 0.05	9.123×10^{-4}
1.300 ± 0.05	6.698×10^{-4}
1.400 ± 0.05	4.883×10^{-4}
1.500 ± 0.05	3.527×10^{-4}
1.600 ± 0.05	2.516×10^{-4}
1.700 ± 0.05	1.766×10^{-4}
1.800 ± 0.05	1.216×10^{-4}
1.900 ± 0.05	8.176×10^{-5}
2.000 ± 0.05	5.334×10^{-5}
2.100 ± 0.05	3.353×10^{-5}
2.200 ± 0.05	2.011×10^{-5}
2.300 ± 0.05	1.135×10^{-5}
2.400 ± 0.05	5.915×10^{-6}
2.500 ± 0.05	2.760×10^{-6}
2.600 ± 0.05	1.095×10^{-6}
2.700 ± 0.05	3.363×10^{-7}
2.800 ± 0.05	6.451×10^{-8}
2.900 ± 0.05	3.898×10^{-9}
3.000 ± 0.05	0.000
Total bremsstrahlung energy, Mev/beta particle	5.075×10^{-2}

Table A.9. Production of Bremsstrahlung Photons From Promethium-146 Beta in Promethium Oxide Matrix

Maximum beta-particle energy, Mev	0.735
Average beta-particle energy, Mev	0.24

Bremsstrahlung Energy Group (Mev)	Number of Photons per Beta Particle Within ΔE Energy Group
0.050 ± 0.025	1.638×10^{-2}
0.100 ± 0.025	5.420×10^{-3}
0.150 ± 0.025	2.417×10^{-3}
0.200 ± 0.025	1.200×10^{-3}
0.250 ± 0.025	6.221×10^{-4}
0.300 ± 0.025	3.247×10^{-4}
0.350 ± 0.025	1.665×10^{-4}
0.400 ± 0.025	8.187×10^{-5}
0.450 ± 0.025	3.754×10^{-5}
0.500 ± 0.025	1.543×10^{-5}
0.550 ± 0.025	5.340×10^{-6}
0.600 ± 0.025	1.377×10^{-6}
0.650 ± 0.025	1.977×10^{-7}
0.700 ± 0.025	4.928×10^{-9}
Total bremsstrahlung energy, Mev/beta particle	2.336×10^{-3}

Table A.10. Production of Bremsstrahlung Photons From Promethium-147 Beta in Promethium Oxide Matrix

Maximum beta-particle energy, Mev 0.23
Average beta-particle energy, Mev 0.067

Bremsstrahlung Energy Group (Mev)	Number of Photons per Beta Particle Within ΔE Energy Group
0.010 ± 0.005	5.230×10^{-3}
0.020 ± 0.005	1.929×10^{-3}
0.030 ± 0.005	9.687×10^{-4}
0.040 ± 0.005	5.507×10^{-4}
0.050 ± 0.005	3.339×10^{-4}
0.060 ± 0.005	2.100×10^{-4}
0.070 ± 0.005	1.348×10^{-4}
0.080 ± 0.005	8.737×10^{-5}
0.090 ± 0.005	5.674×10^{-5}
0.100 ± 0.005	3.669×10^{-5}
0.110 ± 0.005	2.345×10^{-5}
0.120 ± 0.005	1.474×10^{-5}
0.130 ± 0.005	9.042×10^{-6}
0.140 ± 0.005	5.367×10^{-6}
0.150 ± 0.005	3.050×10^{-6}
0.160 ± 0.005	1.635×10^{-6}
0.170 ± 0.005	8.095×10^{-7}
0.180 ± 0.005	3.581×10^{-7}
0.190 ± 0.005	1.337×10^{-7}
0.200 ± 0.005	3.764×10^{-8}
0.210 ± 0.005	5.924×10^{-9}
0.220 ± 0.005	1.223×10^{-26}
0.230 ± 0.005	0.000

Total bremsstrahlung energy, Mev/beta particle 2.020×10^{-4}

Table A.11. Production of Bremsstrahlung Photons from Thulium-170 Beta 1 in Thulium Oxide Matrix

Maximum beta-particle energy, Mev	0.968
Average beta-particle energy, Mev	0.326

Bremsstrahlung Energy Group (Mev)	Number of Photons per Beta Particle Within ΔE Energy Group
0.050 ± 0.025	5.736×10^{-2}
0.100 ± 0.025	2.085×10^{-2}
0.150 ± 0.025	1.026×10^{-2}
0.200 ± 0.025	5.686×10^{-3}
0.250 ± 0.025	3.340×10^{-3}
0.300 ± 0.025	2.020×10^{-3}
0.350 ± 0.025	1.236×10^{-3}
0.400 ± 0.025	7.551×10^{-4}
0.450 ± 0.025	4.562×10^{-4}
0.500 ± 0.025	2.696×10^{-4}
0.550 ± 0.025	1.542×10^{-4}
0.600 ± 0.025	8.410×10^{-5}
0.650 ± 0.025	4.296×10^{-5}
0.700 ± 0.025	2.000×10^{-5}
0.750 ± 0.025	8.138×10^{-6}
0.800 ± 0.025	2.680×10^{-6}
0.850 ± 0.025	6.111×10^{-7}
0.900 ± 0.025	6.278×10^{-8}
0.950 ± 0.025	2.196×10^{-10}
Total bremsstrahlung energy, Mev/beta particle	1.033×10^{-2}

Table A.12. Production of Bremsstrahlung Photons from Thulium-170 Beta 2 in Thulium Oxide Matrix

Maximum beta-particle energy, Mev	0.884
Average beta-particle energy, Mev	0.293

Bremsstrahlung Energy Group (Mev)	Number of Photons per Beta Particle Within ΔE Energy Group
0.050 ± 0.025	4.999×10^{-2}
0.100 ± 0.025	1.765×10^{-2}
0.150 ± 0.025	8.429×10^{-3}
0.200 ± 0.025	4.522×10^{-3}
0.250 ± 0.025	2.562×10^{-3}
0.300 ± 0.025	1.487×10^{-3}
0.350 ± 0.025	8.674×10^{-4}
0.400 ± 0.025	5.011×10^{-4}
0.450 ± 0.025	2.828×10^{-4}
0.500 ± 0.025	1.538×10^{-4}
0.550 ± 0.025	7.907×10^{-5}
0.600 ± 0.025	3.753×10^{-5}
0.650 ± 0.025	1.584×10^{-5}
0.700 ± 0.025	5.592×10^{-6}
0.750 ± 0.025	1.459×10^{-6}
0.800 ± 0.025	2.091×10^{-7}
0.850 ± 0.025	4.896×10^{-9}

Total bremsstrahlung energy, Mev/beta particle 8.310×10^{-3}

Table A.13. Production of Bremsstrahlung Photons From Thulium-171 Beta in Thulium Oxide Matrix

Maximum beta-particle energy, Mev	0.097
Average beta-particle energy, Mev	0.029

Bremsstrahlung Energy Group (Mev)	Number of Photons per Beta Particle Within ΔE Energy Group
0.010 ± 0.005	1.632×10^{-3}
0.020 ± 0.005	4.124×10^{-4}
0.030 ± 0.005	1.367×10^{-4}
0.040 ± 0.005	4.785×10^{-5}
0.050 ± 0.005	1.599×10^{-5}
0.060 ± 0.005	4.632×10^{-6}
0.070 ± 0.005	9.973×10^{-7}
0.080 ± 0.005	1.068×10^{-7}
0.090 ± 0.005	0.000

Total bremsstrahlung energy, Mev/beta particle 3.126×10^{-5}

Table A.14. Production of Bremsstrahlung Photons From Thallium-204 Beta in Metal Matrix

Maximum beta-particle energy, Mev 0.764
Average beta-particle energy, Mev 0.27

Bremsstrahlung Energy Group (Mev)	Number of Photons per Beta Particle Within ΔE Energy Group
0.050 ± 0.025	5.228×10^{-2}
0.100 ± 0.025	1.801×10^{-2}
0.150 ± 0.025	8.375×10^{-3}
0.200 ± 0.025	4.357×10^{-3}
0.250 ± 0.025	2.377×10^{-3}
0.300 ± 0.025	1.315×10^{-3}
0.350 ± 0.025	7.211×10^{-4}
0.400 ± 0.025	3.834×10^{-4}
0.450 ± 0.025	1.932×10^{-4}
0.500 ± 0.025	8.945×10^{-5}
0.550 ± 0.025	3.632×10^{-5}
0.600 ± 0.025	1.193×10^{-5}
0.650 ± 0.025	2.674×10^{-6}
0.700 ± 0.025	2.543×10^{-7}
0.750 ± 0.025	2.981×10^{-10}

Total bremsstrahlung energy, Mev/beta particle 8.098×10^{-3}