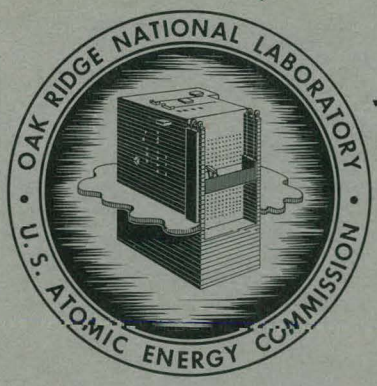
325
12-21-61

MASTER

ORNL-3126
UC-41 – Health and Safety
TID-4500 (16th ed., Rev.)

THE ORNL BADGE DOSIMETER AND ITS PERSONNEL MONITORING APPLICATIONS

W. T. Thornton
D. M. Davis
E. D. Gupton



OAK RIDGE NATIONAL LABORATORY
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

DISCLAIMER

Printed in USA. Price $1.50. Available from the
Office of Technical Services
Department of Commerce
Washington 25, D.C.

ORNL-3126

Contract No. W-7405-eng-26

HEALTH PHYSICS DIVISION

THE ORNL BADGE DOSIMETER AND ITS PERSONNEL MONITORING APPLICATIONS

W. T. Thornton
D. M. Davis
E. D. Gupton

DATE ISSUED

DEC 21 1967

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
for the
U. S. ATOMIC ENERGY COMMISSION

THIS PAGE

WAS INTENTIONALLY

LEFT BLANK

TABLE OF CONTENTS

I. INTRODUCTION

The film dosimeter in a variety of forms has been used since the beginning of the Atomic Energy Program by nearly all installations handling significant quantities of radioactive materials. Methods for overcoming or circumventing most of the disadvantages of film dosimetry, such as the photon energy dependence, have been achieved, mainly by placing the film in a properly designed package or "badge".

Pardue, Goldstein and Wollan[1] stated as early as 1944 that a good pocket dosimeter should have the following properties:

1. Response for equal exposures in roentgens should be, as far as possible, independent of the quantum energy of radiation.
2. The meter should provide a means for accurately measuring doses within the range 0.1 r to 20 r.
3. The response should be producible by ionizing radiation only.
4. The dosimeter should be small and light and should be adapted to routine processing in large numbers.

These fundamental requirements are still included in a properly designed and engineered film dosimeter, but the requirements have been greatly extended to provide a method to differentiate between beta radiation and soft X or gamma radiation and to provide for emergency monitoring in case of high gamma and/or neutron exposures, particularly criticality accidents.

---

(1) L. A. Pardue, N. Goldstein and E. O. Wollan. CH-1553 (1944).

II. HISTORICAL DEVELOPMENTS OF ORNL DOSIMETER

A film badge using a 1 mm Cd filter was fabricated from tinplate on the basis of the pioneer work by Pardue, Goldstein and Wollan. This badge was first used on the Manhattan Project at the University of Chicago. Later, it was adopted at some of the national laboratories, including the Oak Ridge National Laboratory (ORNL) and for a few years was widely used. Since the introduction of this "tin" badge at ORNL, there have been five basic changes in the badge design. Figure 1 is a photograph of these badges including the "tin" model, No. 1. The badges numbered 4 are identical, except for color, as are those numbered 5 and those numbered 6.

Basic design criteria for badge meters at ORNL have always included attempts to develop badges that would permit dose interpretation according to the current recommendations of the National and International Committee on Radiation Protection. However, changes in badge design have not always been a direct result of the evolution of rules governing the maximum permissible dose. Table 1 lists the types of badges and dose measurement methods used at ORNL during the past 17 years, including the present badge, ORNL Meter, Model II.

III. DESCRIPTION OF THE ORNL BADGE METER, MODEL II

In July 1960, the ORNL Badge Meter, Model II, was put into service at five AEC sites. Figure 2 shows the components of the Model II Badge in exploded view.

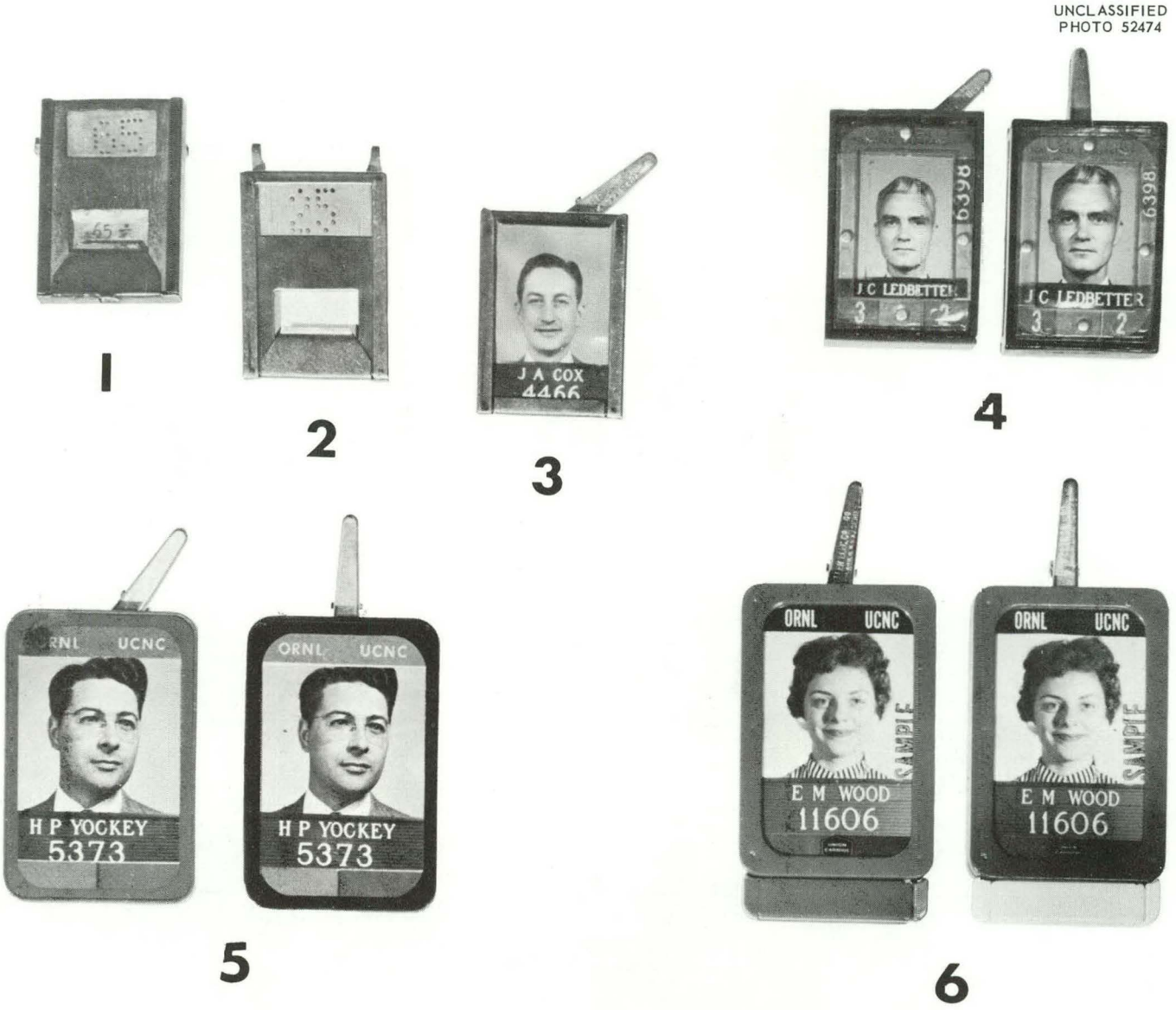


Fig. 1. Film badges that have been used at ORNL.

Table 1. Types of Badges and Dose Measurements at ORNL from 1944 to 1961

Year[a]	Type Badge (Fig. 1)	Material Used in Construction of Badge	Filters	Maximum Permissible Dose	Doses Reported	Method of Determining Dose[b]	Films Used
1944	1	Tin	Cd W	0.1 rem/day	mrep mr	W Cd	552
1949	2	Stainless Steel	Cd W	0.3 rem/wk	mrep mr	W Cd	552 and fine grained particle
1951	3	Stainless Steel	Cd W	0.3 rem/wk penetrating + 0.2 rem/wk "soft" 1.5 rem/wk, hands, feet	mrep (PTR) mr	W - Cd Cd	552 NTA
1953	4	Plastic	Pb Cd Cu Pl W	0.3 rem/wk penetrating + 0.2 rem/wk "soft" 1.5 rem/wk, hands, feet	mrep (PTR) mr (special analysis of films indicating X-ray exposures)	W - Cd Cd	552 NTA
1954	4	Plastic	Pb Cd Cu Pl W	Blood forming organs, gonads-300 mrem/wk Lens of eyes-300 mrem/wk Skin-600 mrem/wk Soft radiation-1500 mrem/wk (See Rule III HB 59)	mrep (PTR) mr (special analysis of films indicating X-ray exposures)	W - Cd Cd	553 NTA
1956	4	Plastic	Pb Cd Cu Pl W	Blood forming organs gonads-300 mrem/wk Lens of eyes-300 mrem/wk Skin-600 mrem/wk Soft Radiation-1500 mrem/wk	Symbol - Depth of Dose D_p > 1000 mg/cm^2 D_m - 130 mg/cm^2 D_l - 300 mg/cm^2 D_s - 7 mg/cm^2	Cd (W-Pl) + Cd Cd or 1/2 of (W-Pl) + Cd, whichever was greater. Not determined from film.	552 Type B

[a] Changes in badges, maximum permissible dose, and methods of reporting dose did not normally occur at the same time. Hence, the year does not necessarily indicate that this was the date of a given change.

[b] Density behind the particular shield referred to a standard calibration curve.

Table 1 (continued)

Year[a]	Type Badge (Fig. 1)	Material Used in Construction of Badge	Filters	Maximum Permissible Dose	Doses Reported	Method of Determining Dose[b]	Films Used
1958	5	Plastic	Cd Al Pl W	Blood forming organs Gonads-300 mrem/wk > 5(N-18) Skin-600 mrem/wk > 10(N-18) Hands and feet-1500 mrem/wk Soft radiation-1500 mrem/wk	D_p D_m D_l D_s	Cd (W-P + Cd Cd or 1/2 [(W-Pl) + Cd] whichever was greater Not determined from film.	552 Type B-2
1959	6	Plastic	Cd Al Pl W	Critical organs 3 rem/qtr > 5(N-18) Other organs: Skin-6 rem/qtr > 10(N-18) Hands, etc.-25 rem/qtr, 75 rem/yr	D_p D_m D_l D_s	Cd (W-P) + Cd Cd or 1/2 [(W-Pl) + Cd] whichever was greater. Not determined from film.	552 Type A
1961	6	Plastic	Cd Al Pl W	Critical organs: 3 rem/qtr, > 5(N-18) Skin: 10 rem/qtr 30 rem/yr	D_c (critical organs) D_s (skin) (This dose based on beta energies equivalent to those from U metal.)	Cd 2.5 (W-Pl) + Cd	544 Type A

Symbols:
Cd - Cadmium (after 1958, Cd-Au-Cd)
W - Window
Pb - Lead
Cu - Copper
Pl - Plastic
PTR - Probable Total Reading

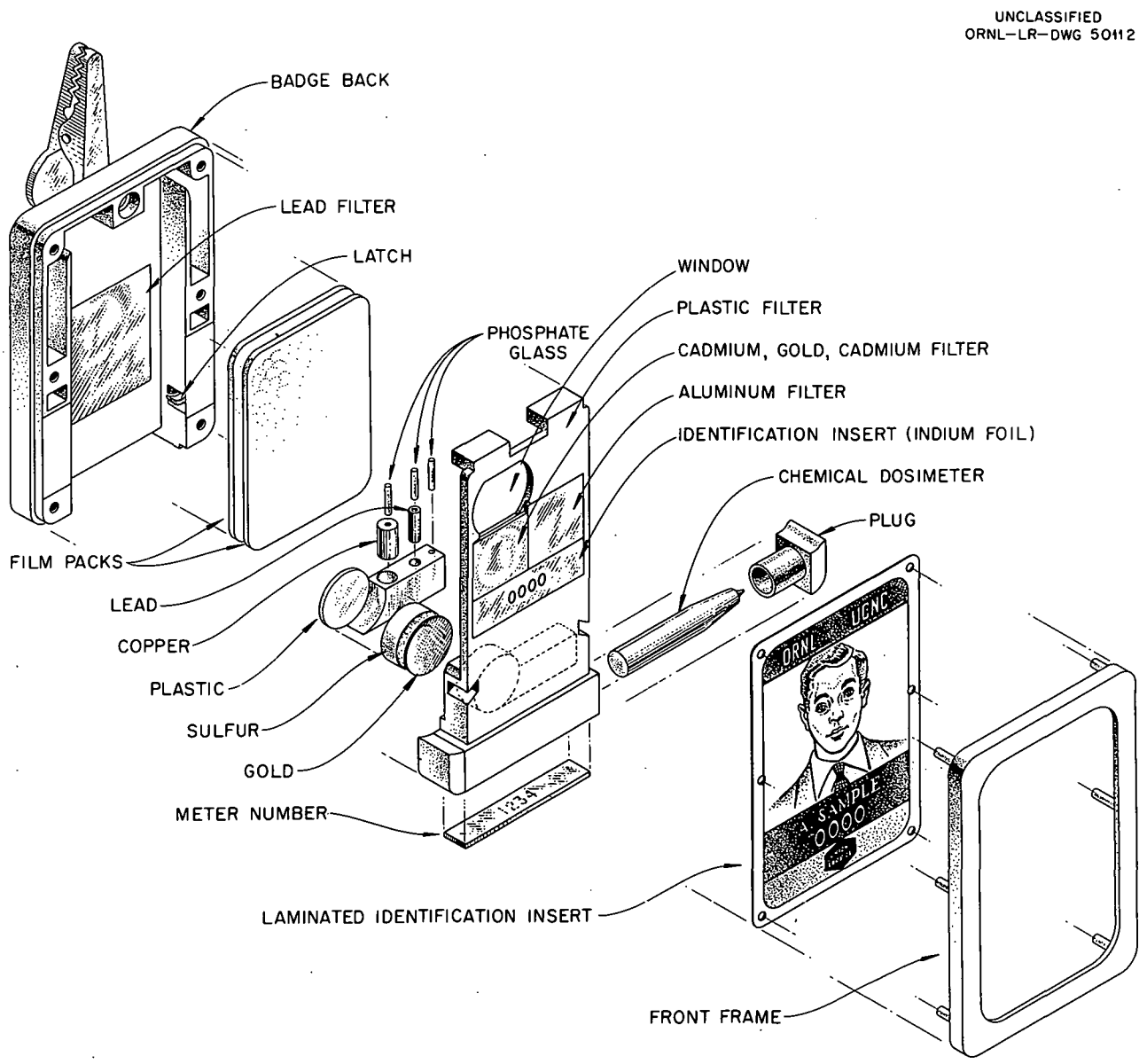


Fig. 2. ORNL badge-meter model II.

Model II is composed of four basic parts as follows: (1) a laminated identification insert, (2) a front frame, (3) a filter assembly (slide) and (4) the badge back. All of these parts, except the laminated photo, are molded from high impact styrene.

The laminated photo is fabricated from the following components: (1) two sheets of polyester photo ID, each of which is 0.005 inches thick, (2) a paper insert from 12-16 pound paper, 0.003 inches thick, and (3) a photo on coated printing paper. When laminated, the badge front is approximately 0.017 inches thick and represents an absorber of approximately 52 mg/cm^2. It is held in place by the front frame.

The slide provides for different absorbers which are essential to the system in the determination of doses from various energies of mixed radiations. The system requires that the radiation incident on the face of the badge be filtered in at least four areas over the useable portion of the photographic emulsions used in the badge. The four filtered areas include: (1) a gold foil 0.005 mils thick placed between two pieces of Cd 0.017 inches thick each and this combination represents an absorber of approximately 1000 mg/cm^2, (2) an aluminum filter 0.040 inches thick which is an absorber of approximately 275 mg/cm^2, (3) a "thick" plastic area 0.070 inches thick which is an absorber of approximately 215 mg/cm^2, and (4) a "window" position where the only absorbers between the photographic emulsion and the front of the badge are the paper wrapper around the film (approximately 28 mg/cm^2) and the badge front (approximately 52 mg/cm^2) which taken together represents a total absorber of approximately 80 mg/cm^2. Other components used in the slide include: (5) an indium foil, (6) a sulfur

pellet, (7) a "bare" gold foil and (8) three meta-phosphate glass rods with associate filters and plastic holder.

Applications of the Model II badge-meter include (1) routine beta-gamma and neutron dosimetry, (2) criticality applications, (3) high doses resulting from accidents involving gamma radiation only, and (4) security identification.

IV. BETA-GAMMA DOSIMETRY TECHNIQUES

A. Film Dosimetry

Beta-gamma film has the characteristics of an absorbed energy dosimeter, and within the useful range, the density produced in the film is proportional to the rad dose of ionizing radiation absorbed in the film emulsion. Because of the K-absorption and emission of silver and bromine, which are constituents of the film emulsion, photon radiation with energy slightly above these K-absorption frequencies will be absorbed to a degree much greater than if only Compton scattering were involved. This photoelectric absorption with attendant emission of accelerated orbital electrons and K and L X-rays produce secondary radiations within the emulsion far out of proportion to that which exist in air or soft tissue; thus, relative to air or tissue the film is "energy dependent" at these energies.

In order to determine the energy absorbed from an unknown dose of radiation in a given film emulsion, it is necessary to determine the quantity of free silver (converted from AgBr) per unit volume of the emulsion, and to compare this quantity with that produced by known doses of radiation.

This is generally done by developing the film and comparing transmission densities of the unknowns with those of calibrated films. It should be emphasized, however, that direct comparison of densities in order to determine exposure should be performed only with films of the same production "lot", stored and handled under similar conditions, developed concurrently in the same solutions, and read with a single densitometer. Under these conditions 95% of the films will yield readings within $\pm$ 10% of the actual value.

The density produced in the film emulsion by a calibrated roentgen dose is a function of (1) the manufacturer's emulsion type, (2) the production lot, (3) the base fog in the emulsion prior to development, (4) the energy of the radiation to which the film was exposed, (5) the type of developing solution, (6) the concentration and age of the developing solution, (7) the development time, (8) the amount of agitation during development, and (9) the rinse, stop bath, and fixing processes. The density determination is also subject to the accuracy of (10) the densitometer and (11) the capability of the operator. Of these eleven considerations, none can be considered constant or controllable over long periods of time. For example, manufacturers modify or discontinue film and developer types; there may be differences of $\pm$ 10% in the dose vs. density response within a production lot; the base fog varies with the age and environmental history of the film, etc. Therefore, systems of dosimetry which are based upon calculations involving comparative densities and which are evolved from studies of density vs. dose under a given set of conditions at a given time are subject to considerable error in interpretation at later times.

This is due to the fact that the emulsions most used for dosimetry do not have produced in them a density which is a linear function of the dose which produced the density, and the degree of non-linearity for a given emulsion type varies with the production lot, the base fog, and the total density. All the foregoing considerations (1) through (11) except numbers (3) and (4) may be obviated by using calibrated films of the same type and lot as the monitoring films. The monitoring and calibrated films should be developed concurrently and read by a single operator on a single densitometer.

The base fog, consideration (3), is difficult to control or predict for films which are subjected to elevated temperatures, certain chemical vapors and excessive aging. Every reasonable effort should be made to use films which are well within the manufacturer's expiration date and to avoid unnecessary exposure to heat.

With reference to consideration (4), one of the more difficult problems with film as a dosimeter for determining air or tissue dose involves the difference in photon energy dependence between film emulsion and air. If the film were to be exposed only to photon radiation of energies 200 to 5000 kev, the bare film packet would suffice as a dosimeter, or if the film were exposed only to photon radiation of energies 20 to 5000 kev, a simple system of metallic filters would permit conversion to tissue dose. If beta (or electron) radiation were the only source of exposure or if it accompanied photon radiation of energies 200 to 5000 kev, the film dose may be compared directly with the absorbed dose in tissue, because the film is not energy dependent relative to tissue for these radiations. (However,

for beta radiation one must consider the quantity of absorber and the consequent attenuation of the beta radiation prior to penetration to the film emulsion, and interpret the tissue depth dose accordingly.) If the exposure is to an unknown mixture of beta and photon radiation, the dose may be difficult to interpret. Much of the work in film dosimetry at ORNL during the past few years has been directed toward this problem. Based upon this work techniques have been developed which permit us to determine, within limits, the dose in rads from mixed beta and photon radiation at various assumed depths in tissue.

NBS Handbook 59 and supplements thereto contain recommendations for permissible doses to persons from external sources of radiation. These permissible doses are functions of the type of radiation, the portion of the body exposed to the radiation, the age and previous exposure history of the person, and whether or not the person is occupationally exposed. Due to the differences in biological importance and radiosensitivity among the organs and tissues of the body, they are divided into groups for dosimetry. Group I includes the whole body, the blood forming organs at an average tissue depth of 5 cm, the lens of the eye at a depth of 3 mm, and the gonads at an average tissue depth of 1 cm (for the male); and Group II is all other organs including the skin.

Since the permissible dose to a large volume of the body (organs of Group I) is no less than that for the male gonads, which are at an assumed depth of one centimeter in wet tissue, the dose for Group I may be determined at an equivalent tissue depth of one centimeter. The lens of the eye, at a tissue depth of three millimeters, may become the

limiting organ in Group I under certain circumstances which in practice occur rarely. Where the exposure is primarily to beta rays, the limiting organ will normally be the skin. To quote from the recommendations of the ICRP, Recommendations of the International Commission on Radiological Protection, paragraph 51 f, "Where work involves exposure to β-rays of E_{max} >2.5 Mev, eye shields or other suitable shielding may be necessary to keep the dose to the lens within permissible limits. In the case of exposure to β-rays of lower energy, if the provision of such shielding is impracticable, the small additional β-ray dose in the lens over the dose already permitted for more penetrating radiation, such as γ-rays and neutrons, is permissible, provided the dose in the skin is limited to the level recommended in paragraph 52 (a)." (Paragraph 52 (a) specifies the recommended maximum permissible skin dose.)

The methods for determining the doses at ORNL are as follows:

Step 1. Calibration curves for the film types to be analyzed are based upon the doses for which the films are calibrated using a radium source, and the net densities obtained at the Cd(Cd-Au-Cd) filter area (Figure 2).

Step 2. The densities of the dosimetry films are determined for the Window (W), Plastic (Pl), Aluminum (Al)* and Cd filter areas. These densities are denoted by the symbols W_D, Pl_D, Al_D, and Cd_D respectively.

---

* At the present time no use is routinely made of the Aluminum reading. In Section VII proposals are made for its use in determining the effective energy of photon exposures.

Step 3. The densities of Step 2 for each filter are converted to readings by reference to the calibration curve (the term "reading" is preferable, at this stage, to the term "dose", because the dose is yet to be calculated). These readings are denoted by W_R, Pl_R, Al_R, and Cd_R respectively.

Step 4. The readings obtained in Step 3 are used to calculate the doses as follow:

A. Group 1 dose, D_C

$$D_C = Cd_R \qquad (1)$$

B. Group II dose, D_S

$$D_S = 2.5\ (W_R - Pl_R) + D_C \qquad (2)$$

The skin dose, D_S, is the dose at 7 mg/cm^2 of tissue. The first factor in the D_S formula represents the beta radiation contribution to the total dose and presents the main problem in determining D_S. The 80 mg/cm^2 minimum absorber preceding the film emulsion in the radiation path does not allow an interpretation of film density or dose for all energies of beta rays. The constant, 2.5, represents the compensation required by the difference in beta transmission through the 80 mg/cm^2 window and the 300 mg/cm^2 plastic filter when the beta radiation is from a slab of normal uranium. D_S is routinely determined making this assumption; however, if there is evidence that the exposure was to beta radiation of another energy, adjustment of the constant may be made by referring to beta absorption curves, Figure 3. Permissible doses for employees at ORNL are summarized in Table 2.

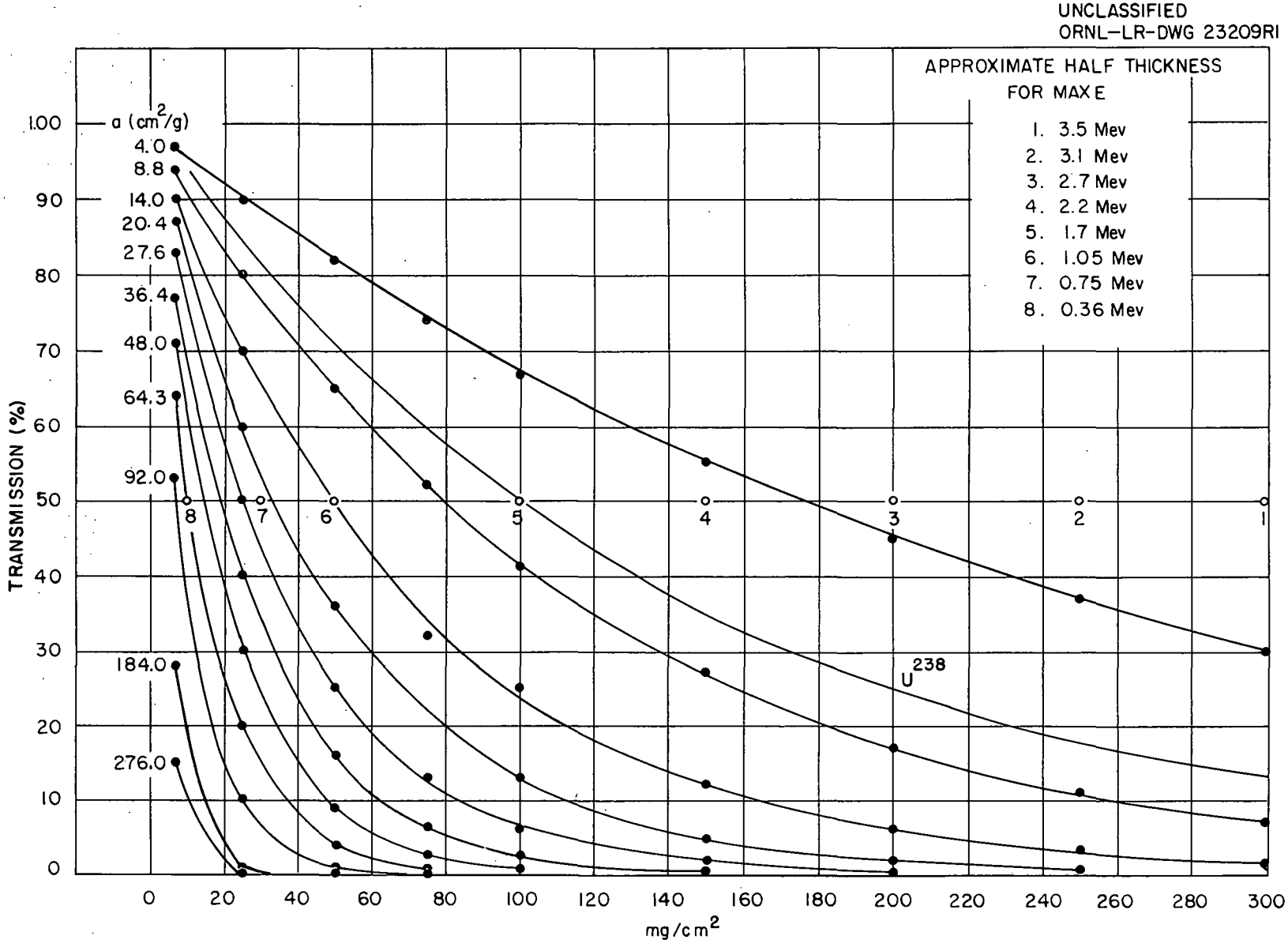


Fig. 3. Beta absorption data—typical curves.

Table 2

Group	Limiting Tissue or Organ	Age Proration Formula Rem	Maximum Yearly Dose Rem	Maximum Quarterly Dose Rem	Tissue Depth (Absorber)	Dose Symbol
I	Male Gonad	5(N-18)*	12	3	1 cm (1000 mg/cm^2)	D_C
II	Skin	30(N-18)	30	10	0.07 mm (7 mg/cm^2)	D_S

* N is the age in years of the employee.

B. Glass Dosimetry

Silver meta-phosphate glass rods are used in the ORNL badge dosimeter to measure gamma radiation doses above 100 rads. The components of the glass system and their orientation in the badge have been described earlier in this report. There is a dual reason for having a three-rod system. First, the approximate effective energy of the radiation may be determined by comparing the relative responses of the lead, copper, and plastic shielded rods; and second, the possibility of losing dose information as a result of damaging a rod is greatly decreased.

The glass has been studied by Schulman[2,3], and others[4,5,6] with regard to its dosimetry characteristics and its reponse is found to depend on the energy of the incident photons. Figure 4 shows the calculated relative energy response of the shielded and unshielded glass based on energy absorbed.[7] Figure 5 is a plot of the ratios of unshielded to Cu-shielded and Pb-shielded glass respectively. With these ratios a good estimate of the effective photon energy may be made and the dose determined by correcting for energy dependence. To evaluate the dose

---

(2) J. H. Schulman, R. J. Ginther, and C. C. Klick. J. Appl. Phys. 22, 1479 (1951).

(3) J. H. Schulman and H. W. Etzel. Science 118, 184 (1953).

(4) N. J. Kriedl and G. E. Blair. Nucleonics 14 (3), 82 (1955).

(5) H. W. Etzel, R. D. Kirk and J. H. Schulman. Ra-Det 8(2), 49 (1955).

(6) A. L. Riegart, H. E. Johns and J. W. T. Spinks. Nucleonics 14(11), 134 (1956).

(7) W. T. Thornton and J. A. Auxier. ORNL-2912 (1960).

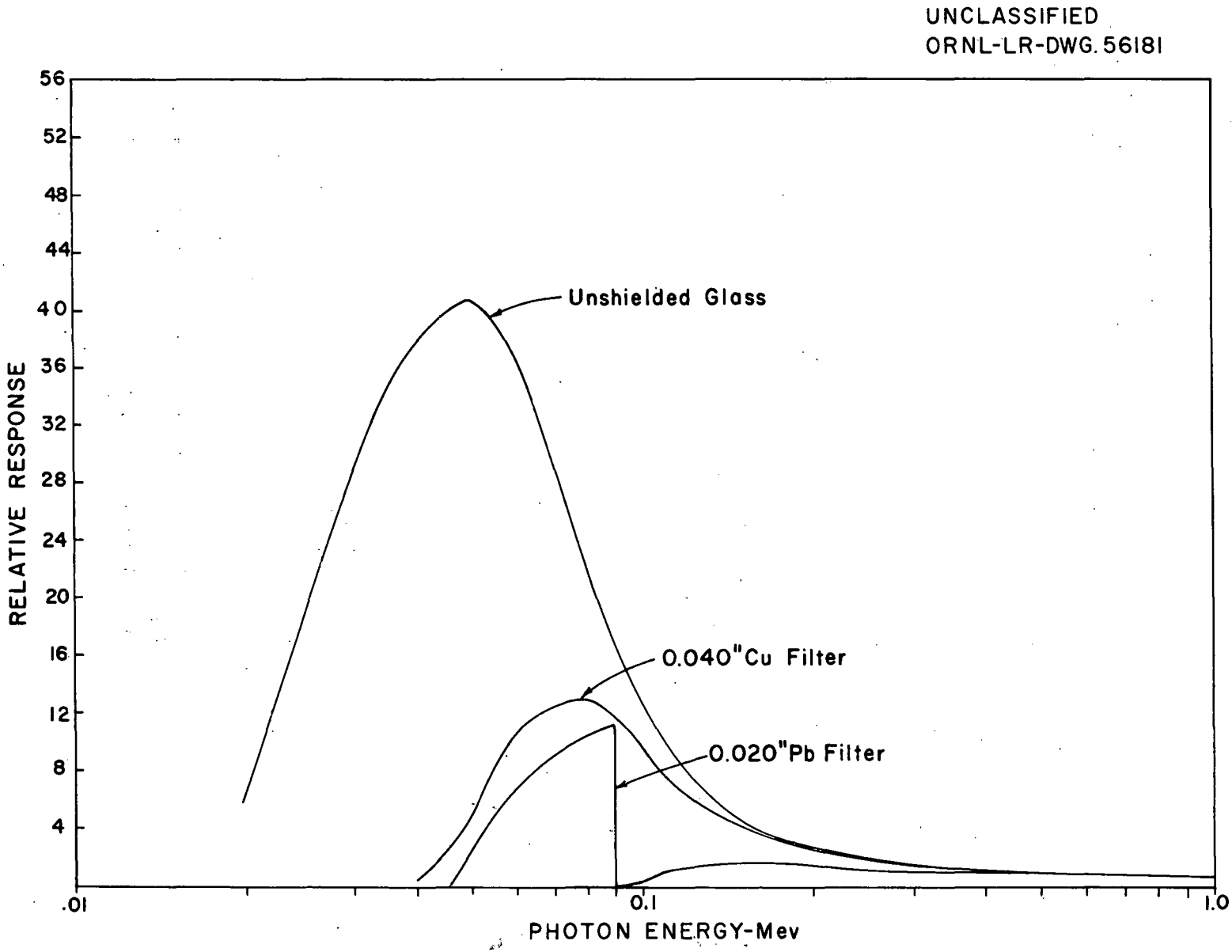


Fig. 4. Calculated relative response of silver metaphosphate glass.

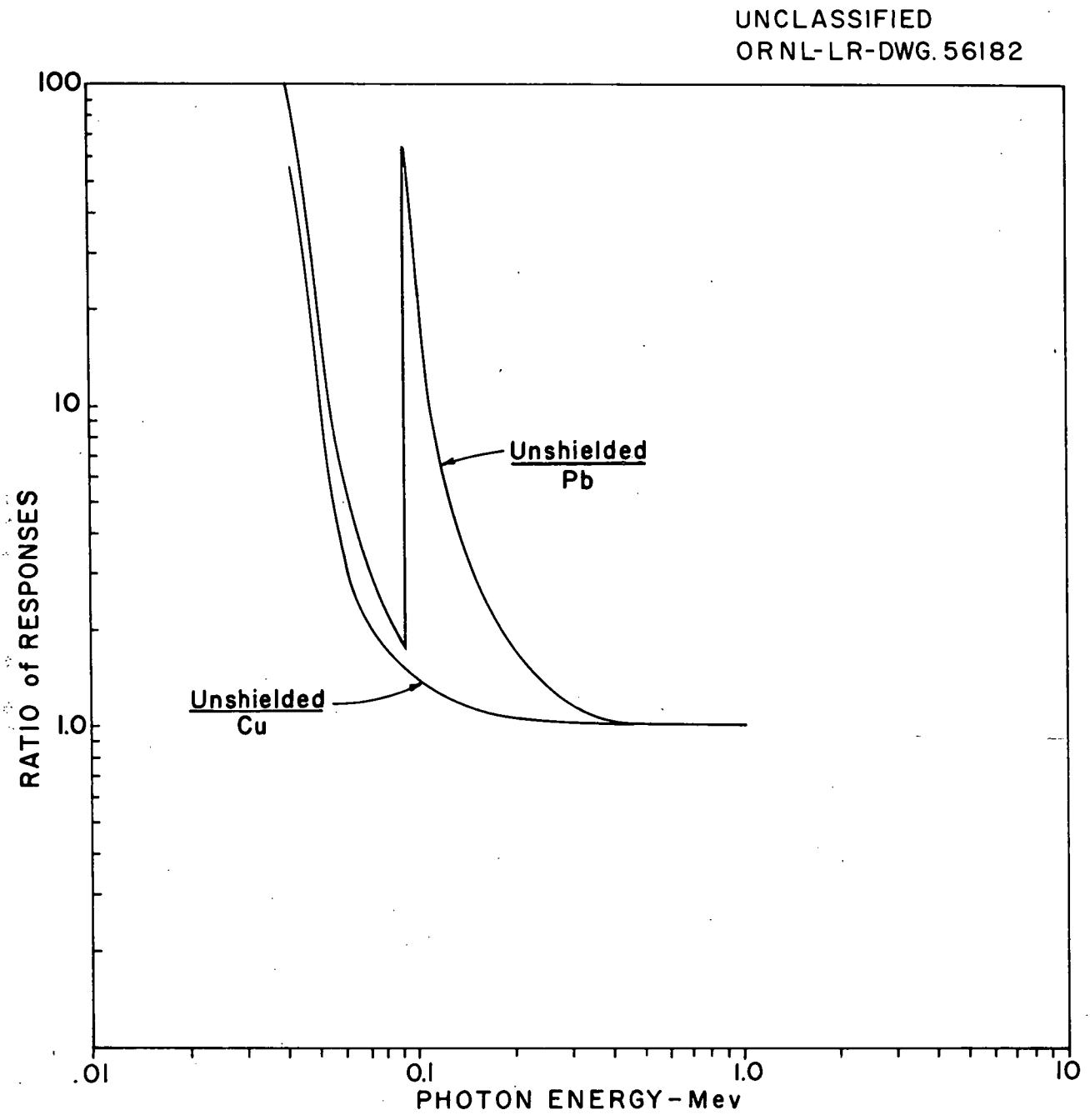


Fig. 5. Ratios of the response of unshielded glass to lead and copper filters in badge dosimeter.

received by a glass rod a comparison of its fluorescence reading with those of a set of rods whose responses have been calibrated is made. Both the standards and the samples are read using the same instrument.

The thermal neutron sensitivity of the glass has been reported.(8) One roentgen equivalent gamma response is produced by 3×10^9 n_t/cm^2. Therefore, for the determination of a gamma dose in a mixed neutron-gamma field, the thermal component as measured by the Cd-shielded and unshielded Au-foils must be subtracted. There is no significant reading induced in the glass by fast neutrons of the energies most prevalent in a criticality event. The response of the glass per rad of fast neutrons relative to a rad of Co^{60} gamma rays has been found to be less than 0.7%(7)

Presently, at ORNL, each employee has three rods in each of two badges which are worn alternately for 13-week periods. A survey of about 600 rods worn for a quarter shows no detectable damage due to installation, wearing, or removal. Figures 6 and 7 show the Bausch and Lomb Microdosimeter Reader and handling equipment.

C. Chemical Dosimeter

Chemical dosimetry continues to be considered as a potentially important adjunct to personnel monitoring. An acceptable chemical dosimeter would perform the very useful function of allowing a quick visual indication of the exposed persons following an incident involving gamma radiation and/or thermal neutrons.

---

(8) S. Kondo. Health Phys. 4(1), 21 (1960).

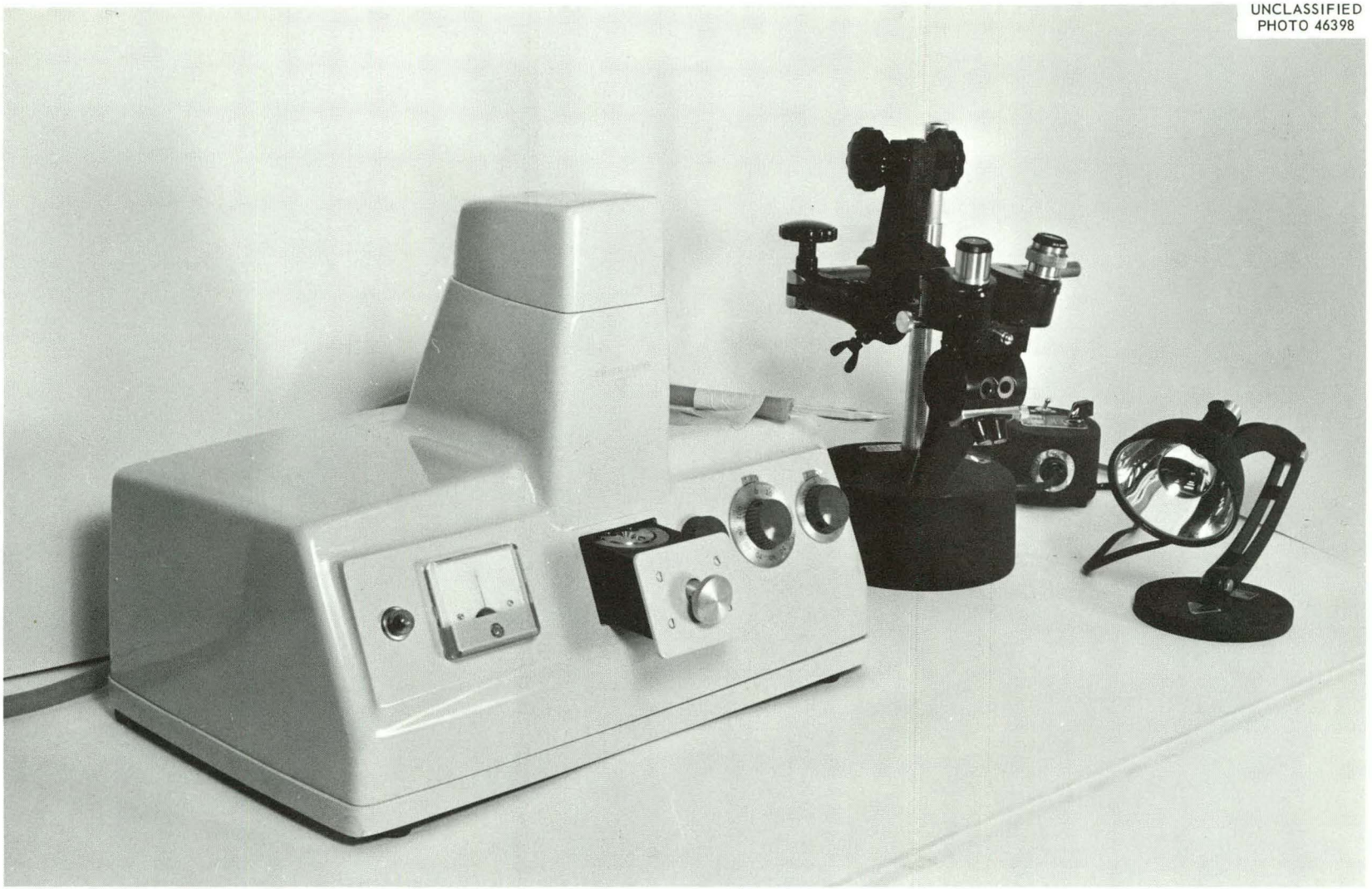


Fig. 6. Bausch and Lomb microdosimeter reader and glass rod inspection equipment.

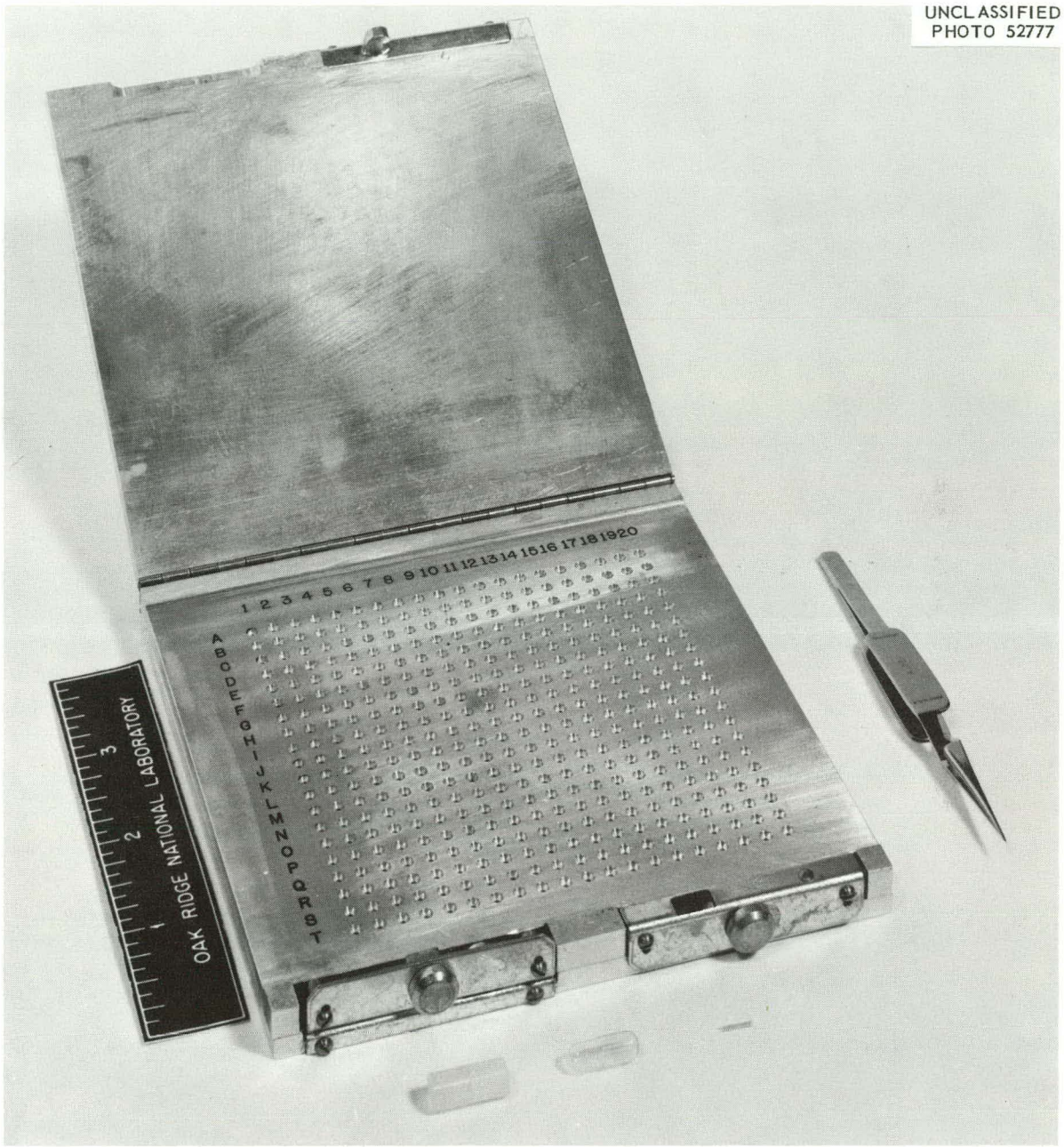


Fig. 7. Silver metaphosphate glass handling equipment.

V. NEUTRON DOSIMETRY TECHNIQUES

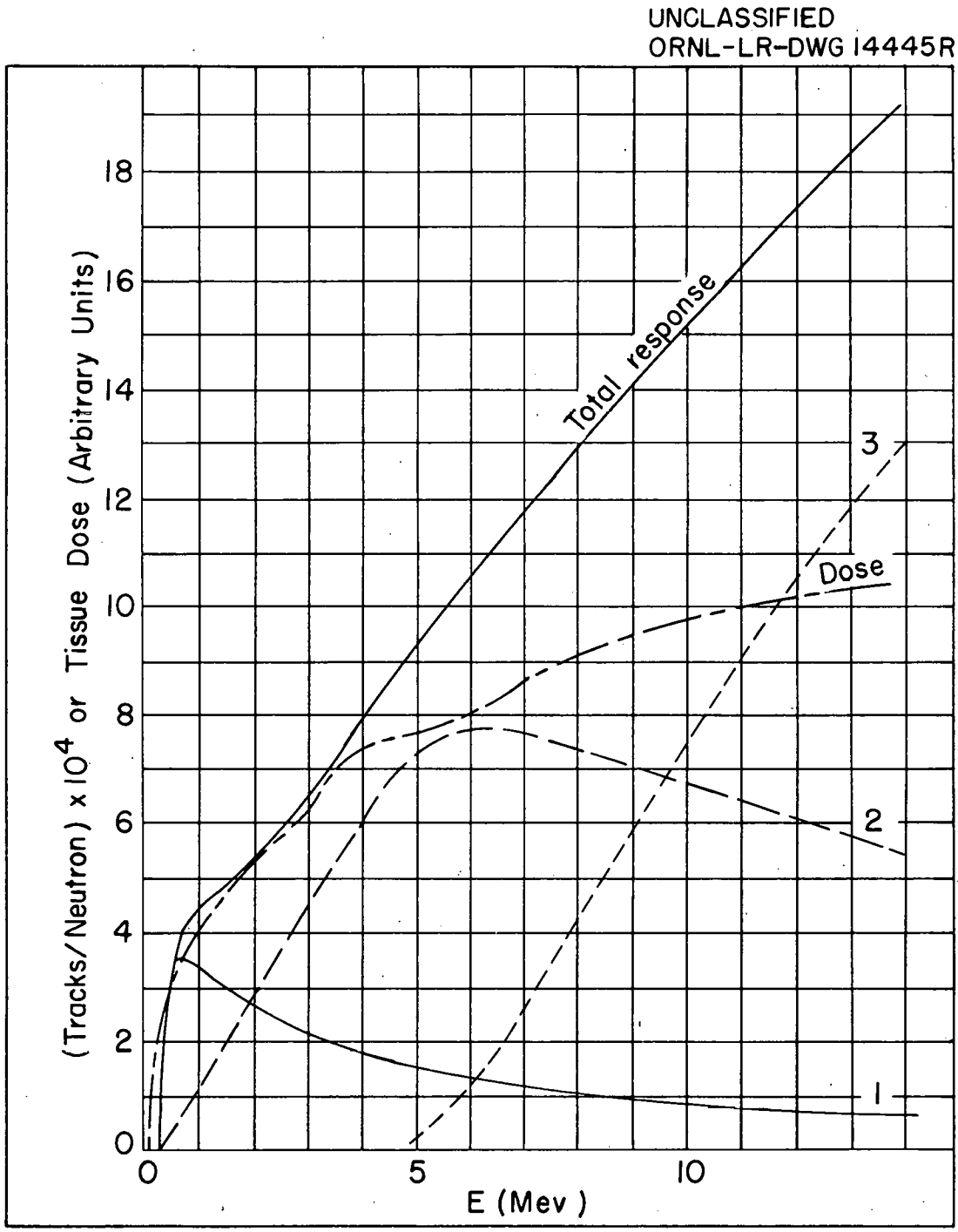
A. Routine

Routine neutron dosimetry is accomplished through the use of the NTA neutron sensitive emulsions manufactured by the Eastman Kodak Company. When exposed to fast neutrons, proton recoil tracks are produced in this film because of the elastic collision of neutron and hydrogen nuclei in the film base, the emulsion, and the film wrapper as shown in Figure 8. The emulsion at the "window" position of Model II is sensitive to thermal neutrons which are captured by elemental nitrogen in the film resulting in the production of C^{14} and a recoil proton according to the reaction ${}_7N^{14}$ (n,p) ${}_6C^{14}$. Thus the fast neutron flux is proportional to the number of proton tracks in the film behind the cadmium-gold-cadmium shield, and the thermal neutron flux is proportional to the difference in the number of proton tracks in the portion of the film at the open window and the portion behind the cadmium-gold-cadmium shield.

The measurement of neutron flux is not a measure of dose since dose per neutron is a function of energy. Based on tissue composition and the probable histories of neutrons of several specific energies, Snyder and Neufeld[9] of ORNL have calculated total or multiple collision dose per neutron incident normally on a body. The values obtained by Snyder and Neufeld are used to determine Maximum Permissible Exposure (MPE). However, since a film badge meter is worn next to the body so that reflected neutrons

---

(9) National Bureau of Standards, Handbook 63.



1. Emulsion
2. Film Base
3. Film Wrapper

Fig. 8. Response of Eastman NTA film to Neutrons.

again impinge on the film, a first-collision dose curve will give a better measure of the body dose. Such a curve was calculated on the basis of published cross sections, and, in general, its magnitude is about 2/3 of the total dose.

As observed in the Eastman Type A packet (Figure 8) the total response curve corresponds to the dose curve up to about 4 Mev, beyond which it becomes increasingly too high until at 14 Mev it is approximately twice as high as the dose curve. Cheka[10] has demonstrated that by replacing the black paper between the film and wrapper with a 85 mg/cm^2 aluminum foil (Table 3) the film response will be proportional to the dose up to approximately 14 Mev (Figure 9).

While the fading of the latent image is very slight for beta-gamma monitoring films, the NTA emulsions are fairly unstable unless they are packaged in a moisture-proof wrapper. The rate of deterioration is a function of temperature and humidity. The effect on track population of three different sets of storage conditions between exposure and development are shown in Figure 10. As indicated by line II of Figure 11, film sealed in "pouch paper" (a three-layer lamination of paper, aluminum foil, and plastic) shows a loss of only about 5% of the tracks in 60 days. Thus, film packaged in this manner may be processed on a monthly, or longer, cycle with no appreciable track loss.

The average permissible exposure rate to fast neutrons over a period of one week, 100 mrem/wk, produces approximately 1000 recognizable tracks/cm^2.

---

(10) J. S. Cheka. ORNL-547 (1950).

Table 3. Laminations of Hydrogeneous Material and Aluminum Foil Around NTA Film to Adjust Track Response to Fast Neutron Tissue Dose

Material	Thickness (mg/cm^2)	Energy of Proton Whose Range Equals Cumulative Thickness of Material to Emulsion (Mev)
Cellulose (or front film)	$\geq$ 76	$\geq$ 14
Aluminum	85	11.05
Cellulose	24.2	8.0
Cellulose (Film Wrapper)*	10.3	6.6
Aluminum*	27	5.9
Film Base*	28.5	4.37
Emulsion*		
Blank Film*	28.5	4.37
Aluminum*	27	5.9
Cellulose (Film Wrapper)*	10.3	6.6
Cellulose	24.2	8.0
Aluminum	85	11.05
Cellulose	$\geq$ 76	$\geq$ 14

* These constituents are present in the packet as supplied by the manufacturer.

ORNL-LR-DWG. 14447
UNCLASSIFIED

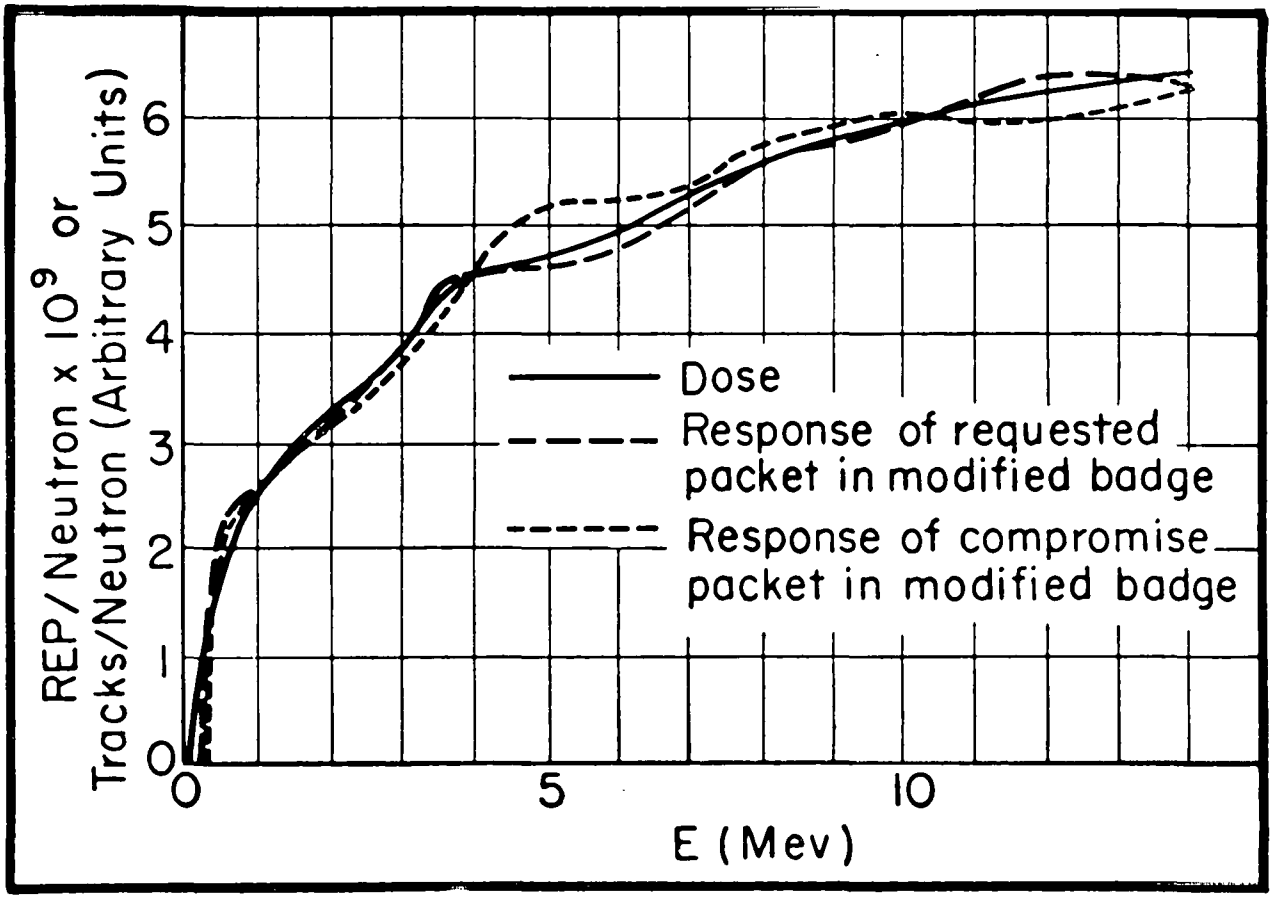


Fig. 9. Film response to calculated first-collision dose.

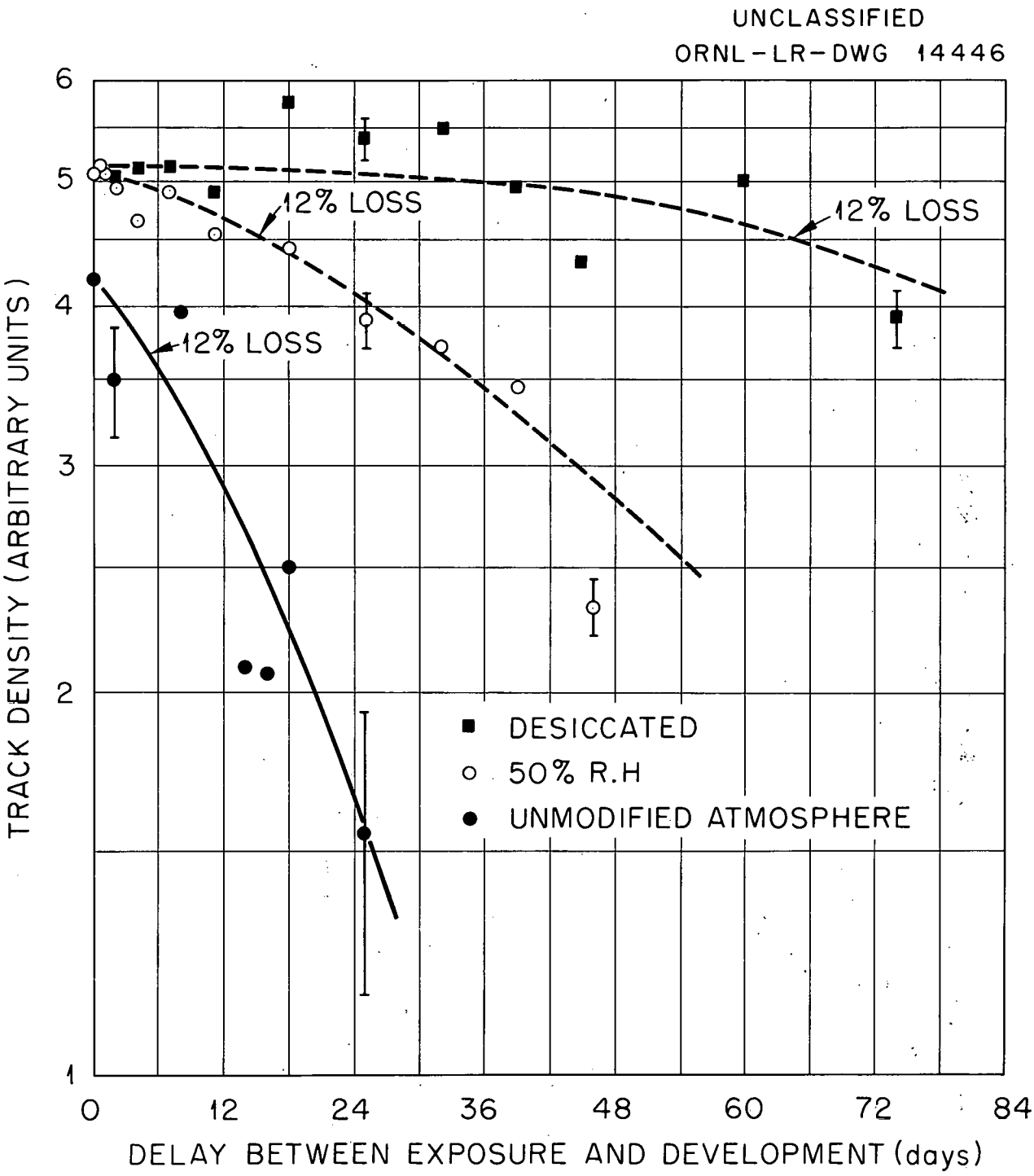


Fig. 10. Effect of relative humidity on latent image stability.

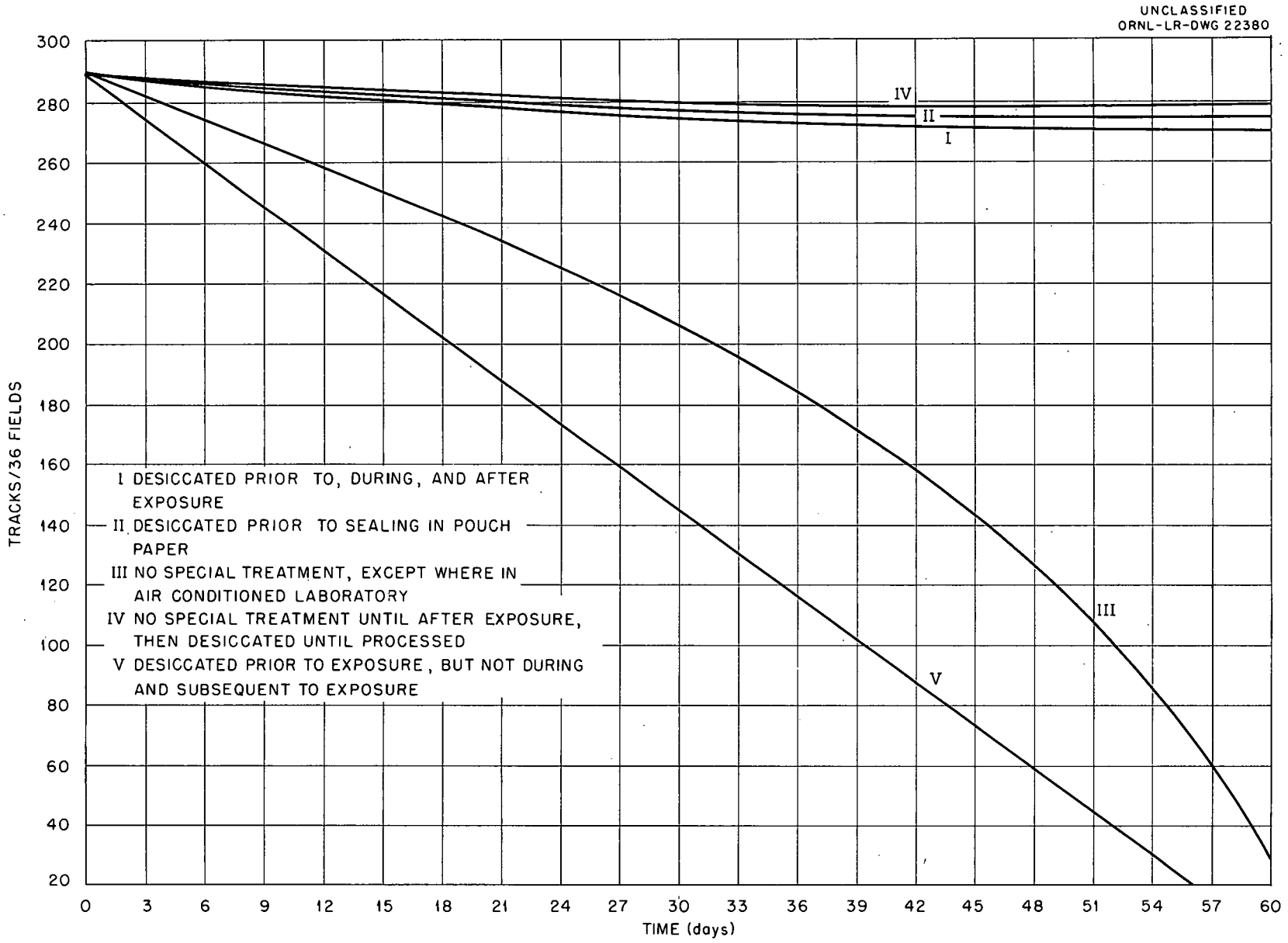


Fig. 11. Effect of sealing neutron film in moisture proof paper.

The figure for thermal neutrons is almost the same. Considering the small size of the developed silver grains, i.e., a few tenths of a micron, and the fact that tracks down to 3 grains are counted, it is necessary to use high magnification and dark field illumination. At 950X the area of one field is 2×10^{-4} cm^2, and consequently a maximum permissible exposure rate for one week produces approximately 0.2 track/field.

B. Criticality

The use of elemental "foils" in Model II of the ORNL badge-meter depends for accurate dosimetry upon concurrent use with the Hurst threshold detector system.(11) As shown in Figure 2, these "foils" include a 1/2 gram pellet of sulfur and two 0.005 inch thick by 7/16 inch diameter gold foils. The so-called bare gold foil is inserted in the cavity behind the sulfur. The other gold foil is the gold component of the cadmium-gold-cadmium sandwich filter described previously. When bombarded by neutrons the foils show a response which can be related to dose through the use of relatively simple counting techniques.(11) Thermal neutron response is determined from the gold foils; high energy (>2.5 Mev) neutron response is determined from the sulfur. Exposure to intermediate energy neutrons is determined by referring the data from the badge to that from dosimeter stations in the vicinity of the badge wearer, or by fitting the data from the badge to the spectrum at the point of exposure, if such otherwise is known. In effect, the badge allows for a determination of a "point in space" dose where the point in space represents the "effective" position of the person

(11) G. S. Hurst and R. H. Ritchie, "Radiation Accidents: Dosimetric Aspects of Neutron and Gamma-Ray Exposures", ORNL-2748.

on whom the badge is located. The dose to the person can then be calculated when the orientation of the badge and the person is known with respect to the source of radiation. Thus, if orientation factors are resolved, and the badge accompanies the person throughout the exposure interval, the dose received by the badge is indicative of that received by the person. One solution to the orientation problem includes the use of a dosimeter belt which contains a series of β, γ, and N detectors spaced at intervals about the mid-section of the wearer. Other methods include techniques involving activation analysis of samples of hair removed from portions of the body, pieces of jewelry, buttons, pencil clips and other substances which might be carried on the person of an exposed individual. Even though these latter techniques are useful and in many cases adequate, the resolution of the orientation problem where the gamma dose is a prominent factor lies with the use of multiple detectors spaced over the body as is provided in the dosimeter belt concept.

Immediately following a nuclear accident those persons who may have received significant exposure to neutrons may be determined easily by a simple measurement of the radiation which has been induced in the indium foil located in the badge. (Indium 115, which constitutes 96% of that occurring naturally in nature, has a thermal neutron activation cross section of 155 barns with a half life of 54 minutes for the resultant In^{166}).[12] Table 4 is a listing of resultant dose rate readings obtained subsequent to exposing badges to fast and thermal neutrons in the west animal tunnel of the X-10 graphite reactor and to thermal neutrons in the

---

(12) Neutron Cross Section, AECU-2040(1952).

Table 4. Neutron Activation of the ORNL Film Badge

Neutron Exposure	Instrument Reading in Radium Equivalent mr/hr when in "Contact" with Badge (Hours after exposure as indicated)			
	0.5	2	3	6
10^{11} thermal neutrons/cm^2 (thermal column of X-10 graphite reactor)	150(a)	40(a)	20(b)	3(b)
100 rads fast neutron dose + 10^{11} thermal neutrons/cm^2 (west animal tunnel of X-10 graphite reactor)	220(a)	70(a)	20(b)	5(b)
10^9 thermal neutron/cm^2 (thermal column of X-10 graphite reactor)	12(b)	3(b)		

(a) Readings taken with ORNL Cutie Pie in "contact" with badge.

(b) Readings taken with Victoreen model 389C probe in "contact" with badge, shield closed. The shield open to shield closed reading ratio was 2 to 1.

thermal column of the same reactor. The activated gold foil also serves to identify the persons exposed to neutrons. Although gold is less sensitive to neutron activation than is indium, gold serves as a valuable complement since it has a much longer half life than the indium.

VI. SUMMARY OF ORNL FILM DOSIMETRY EXPERIMENTS

A. Dose Range of Film

The sensitivity of several commercially available emulsions has been measured using the film calibration facility (Figure 12) which contains two radium-gamma sources, one 98 mg and the other 500 mg. These sources are rapidly positioned by means of a vacuum pump. Each source is contained in an aluminum can and is enclosed in a plastic tube at the time of exposure. One-half centimeter of plastic in front of the film ensures electronic equilibrium for the radium exposure. Behind the film 5 cm of plastic simulates backscatter conditions which would prevail in a personal exposure being monitored by the film badge.

A comparison of sensitivity of the low range emulsions considered is offered in Figure 13, and the high range emulsions in Figure 14. The upper limit of an emulsion's range is determined by the dose at which a density of 2.0 is produced. At this density the film transmits only one per cent of the incident visible light. The emulsions for the ORNL film dosimeter, DuPont 555 and 834, were chosen to cover the desired range of doses without exceeding a density of 2.0. The 555 emulsion range covers 50 mrad to 5 rad between density limits of 0.04 and 2.0. The range of the 834 emulsion is 5 rad to 150 rad for respective densities of 0.10 and 2.0.

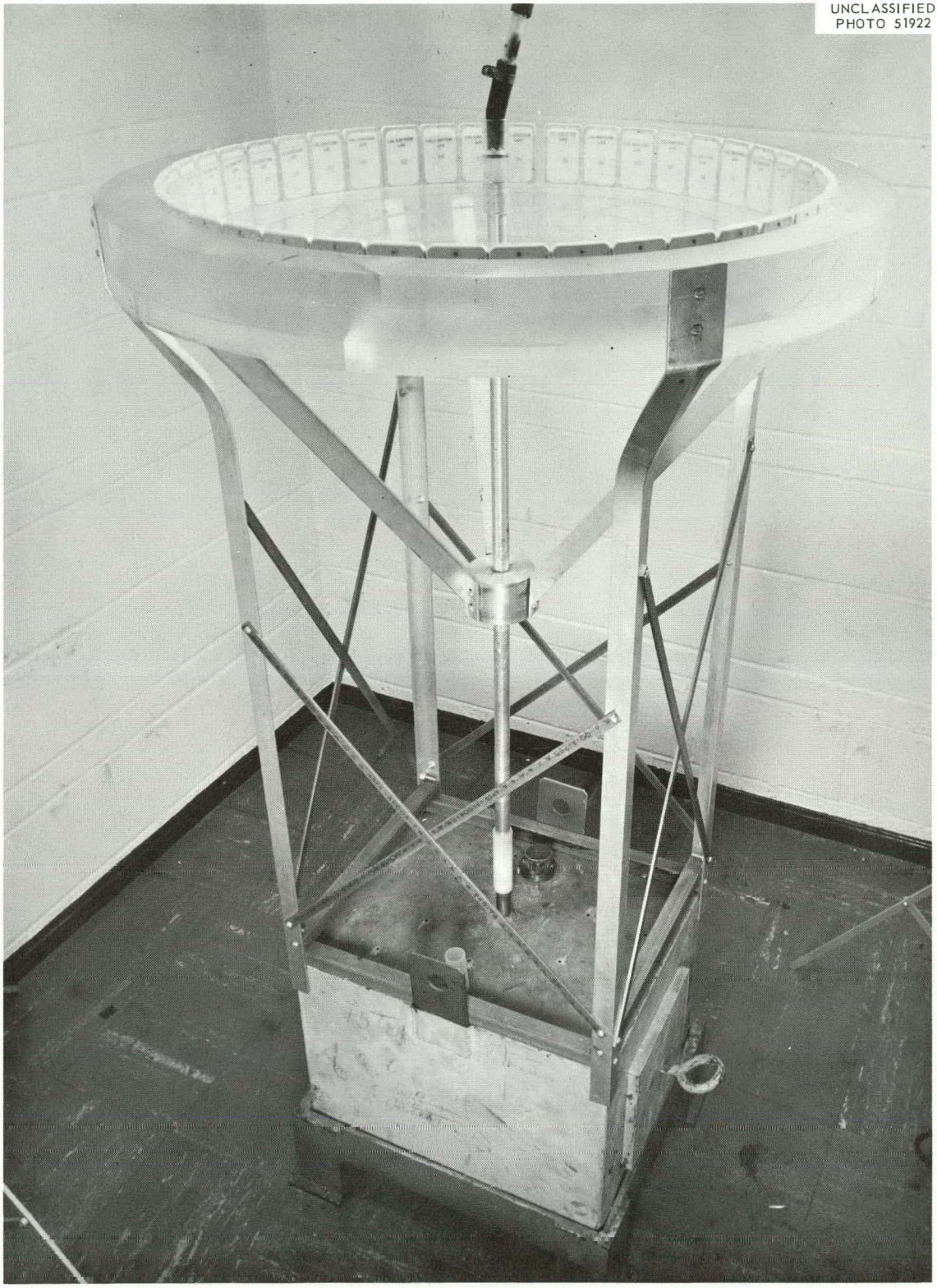


Fig. 12. Radium-gamma calibration ring.

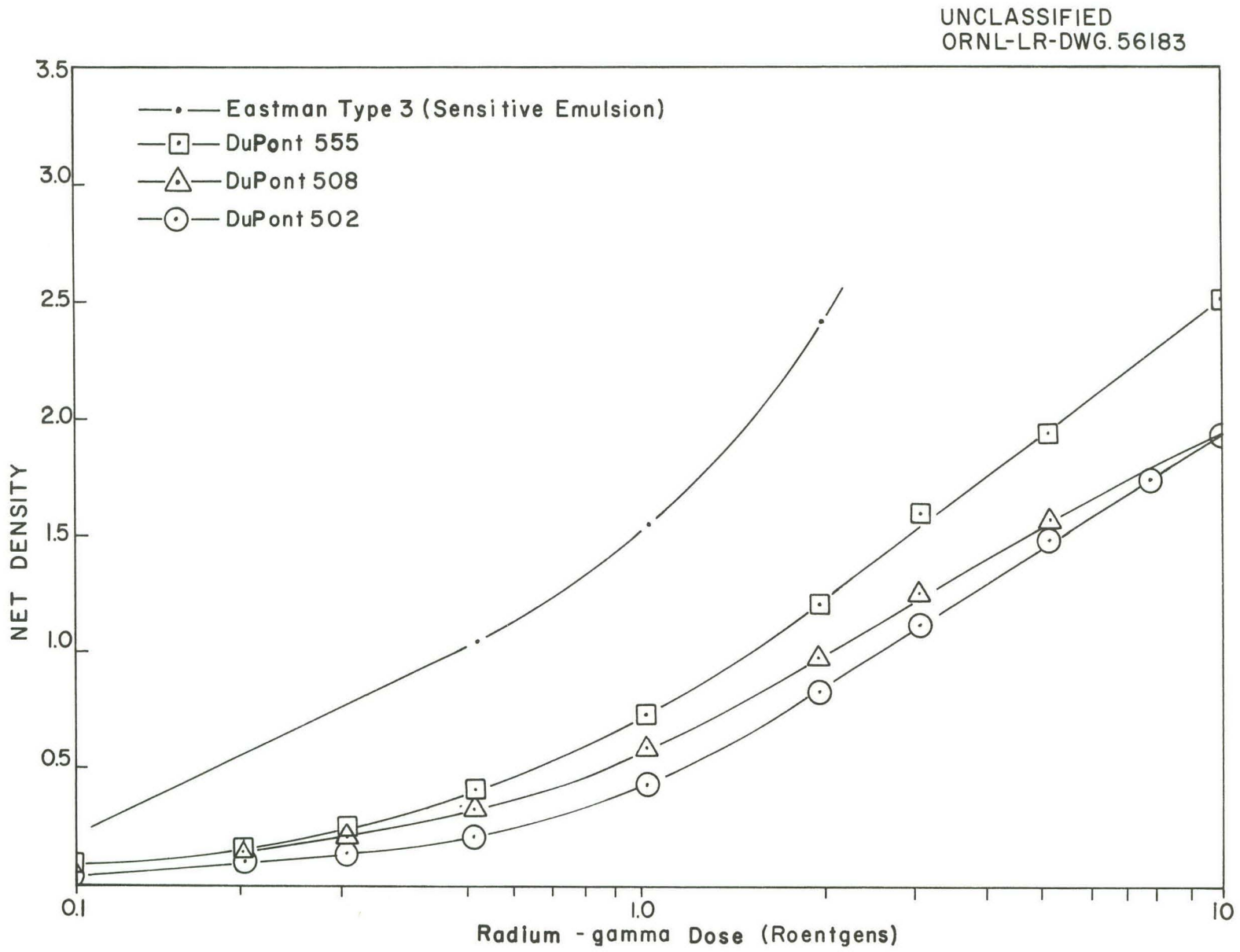


Fig. 13. Radium calibration of low-range emulsions.

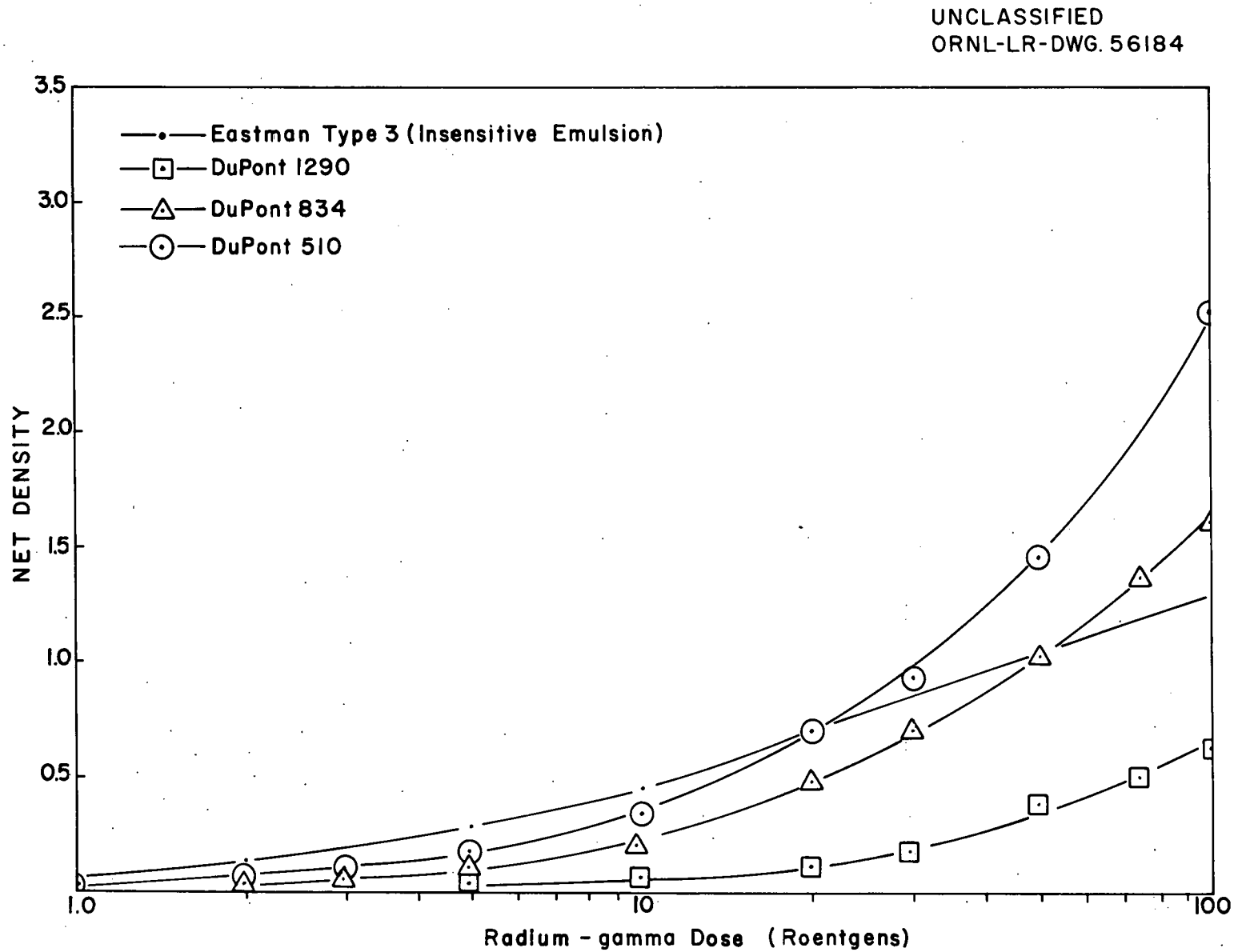


Fig. 14. Radium calibration of high-range emulsions.

Customarily the film is not used as the primary dosimeter above 100 rad; the Ag-metaphosphate glass is considered primary above 100 rad.

B. Photon Energy Response of Film

The response of the aforementioned emulsions to x-rays in the 20 to 200 Kev range was determined using a modified Westinghouse Quadrocondex X-ray machine. By using standard NBS filters, the energy spectrum of the x-rays was narrowed and the exposure energies reported are effective energies* as determined by Villforth, et al[13] who described the filtered spectrum for this machine. It was impractical to narrow the x-ray spectrum sufficiently to produce ideal mono-energetic conditions because of the lengthy exposure time that would be required to expose the large number of films involved. Therefore, this information should not be construed as the real energy response of an emulsion, but it gives the comparative relative energy responses of the emulsion tested. Exposure field mapping showed a uniformity within $\pm$ 2% over an area sufficient to expose four badges simultaneously. Two similar emulsions were used per exposure. The exposure dose rates were measured with a 25r Victoreen condenser r-meter. Figures 15 through 22 show the results of these exposures for the various filters in the badge when normalized to the response for Co^{60} gamma rays. The nonlinearity of the dose vs density curve for film was taken into account by comparing each density to a Co^{60}

---

* The effective energy of a heterochromatic x-ray beam is the energy of a monochromatic beam which has the same absorption coefficient as a given beam in an incremental thickness of standard filter material.

(13) J. C. Villforth, R. D. Birkhoff, H. H. Hubbell, Jr. ORNL-2529 (1958).

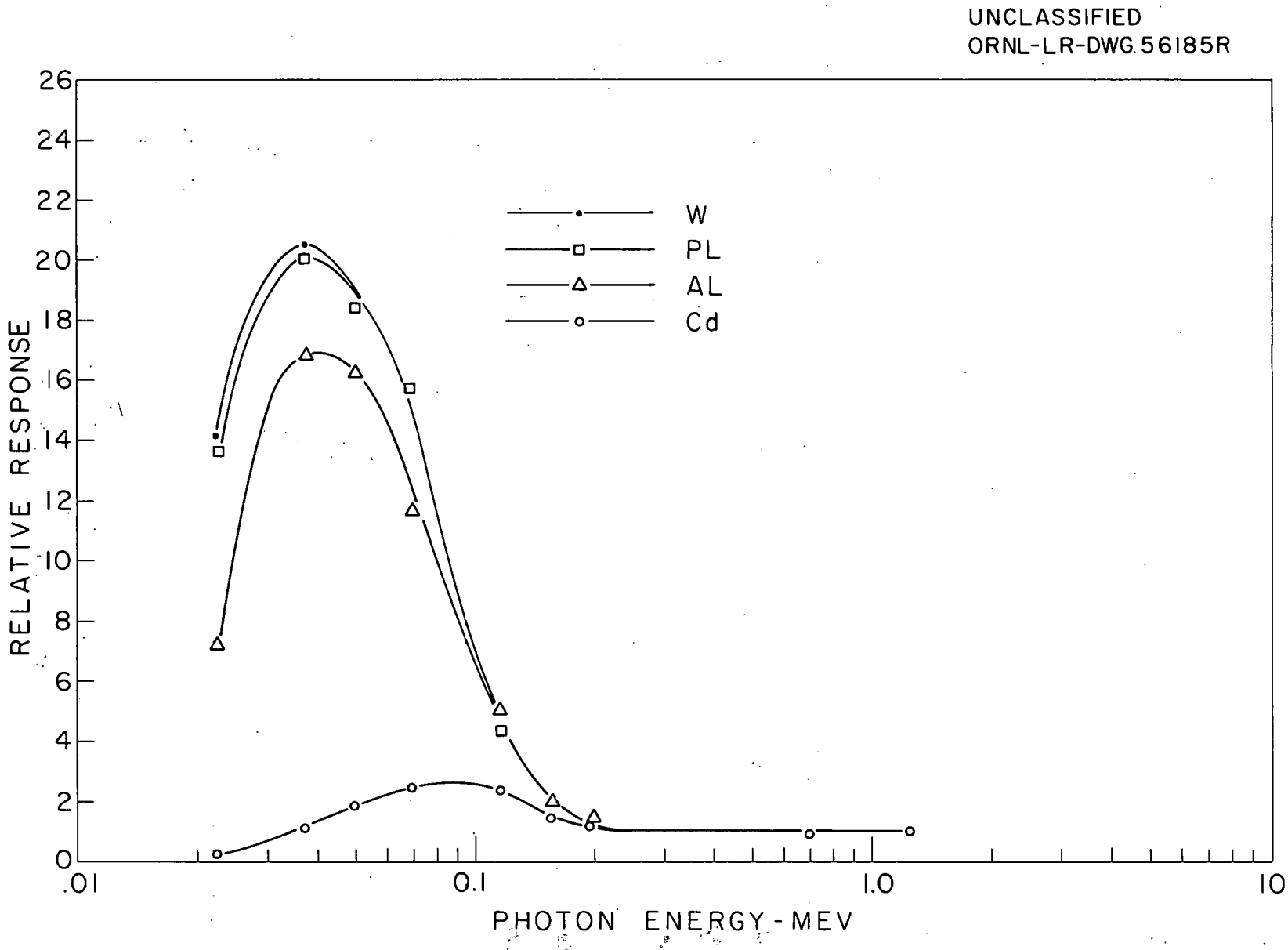


Fig. 15. Relative response of the Du Pont 502 emulsion irradiated in the badge dosimeter.

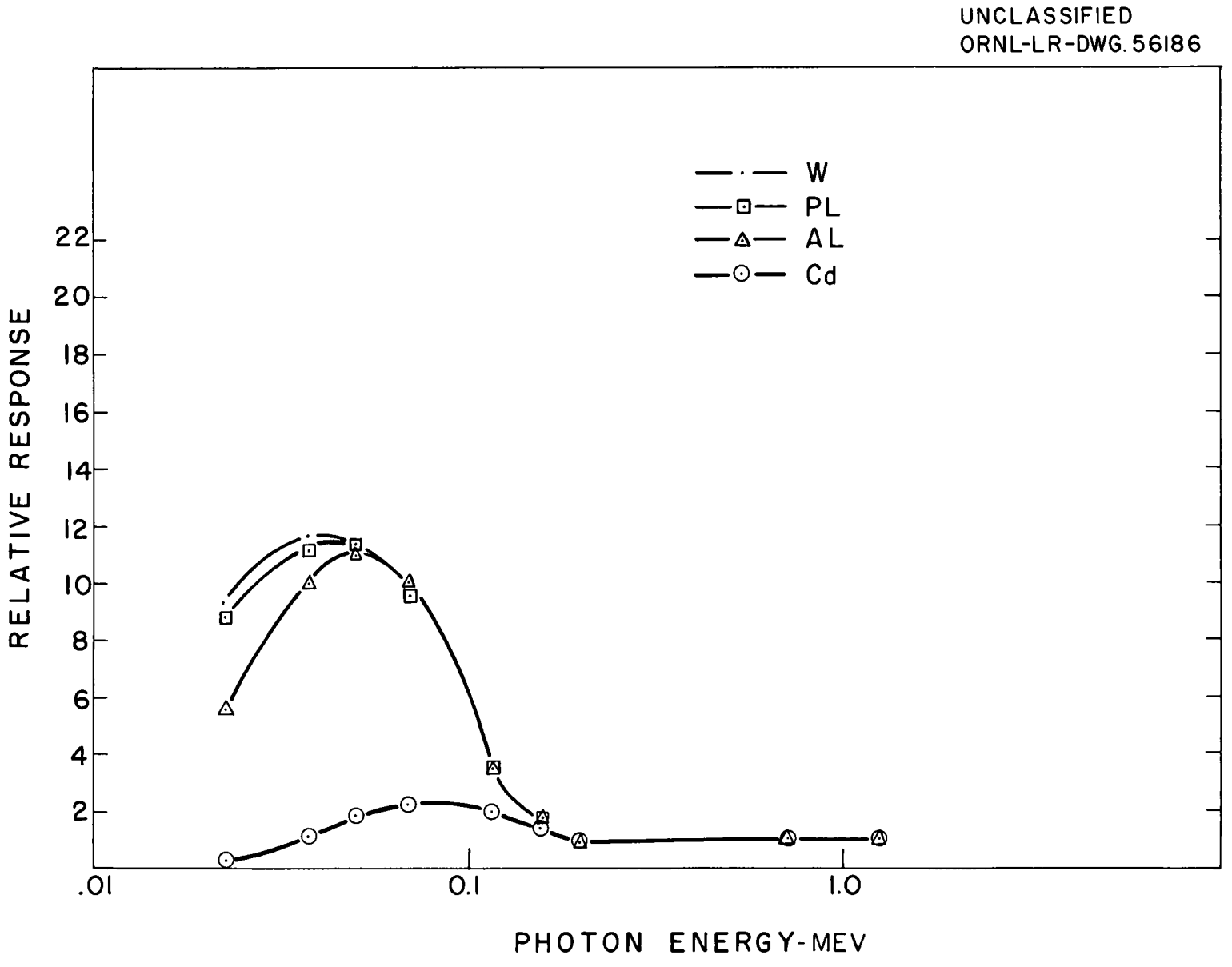


Fig. 16. Relative response of the Du Pont 508 emulsion irradiated in the badge dosimeter.

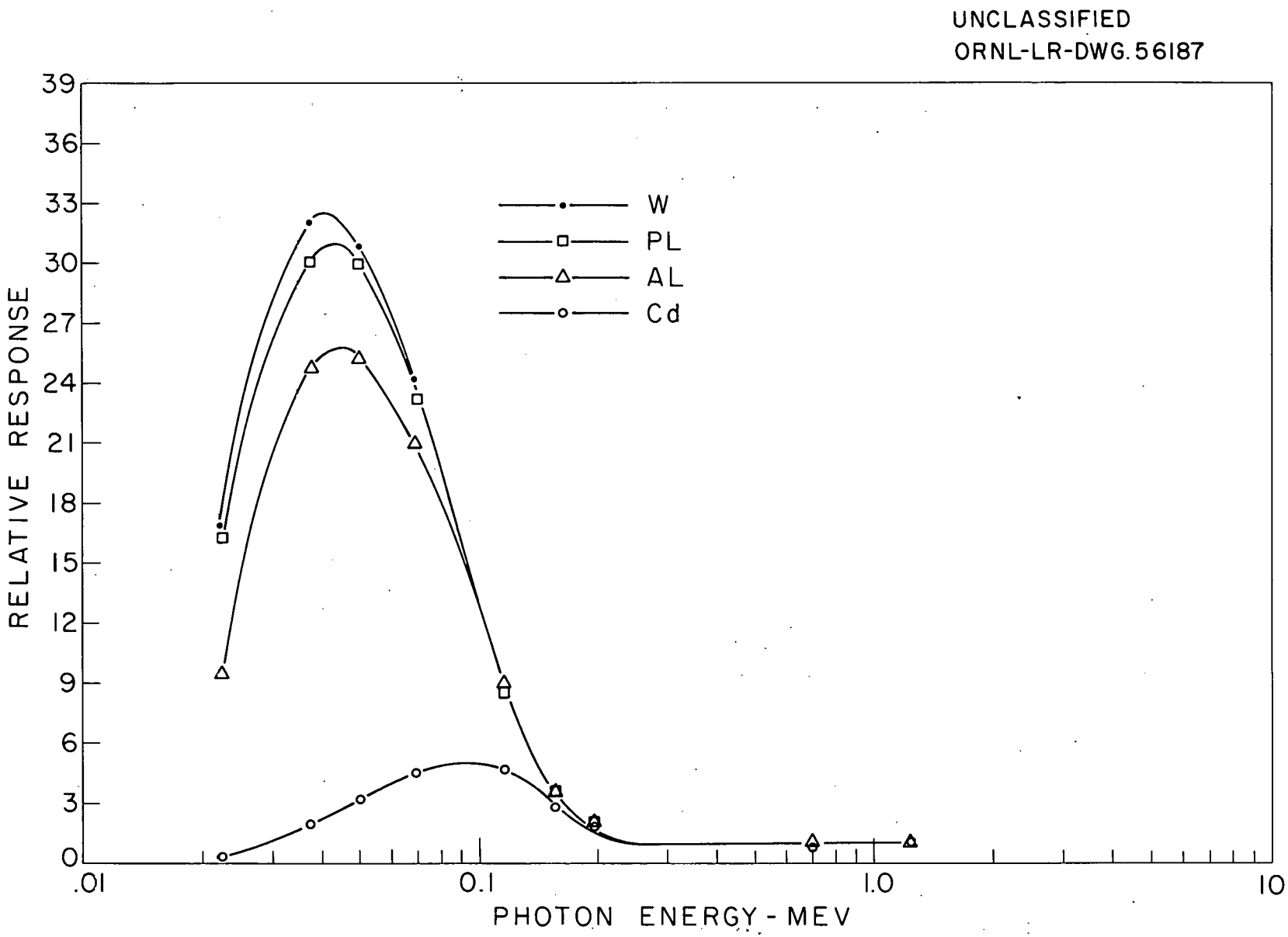


Fig. 17. Relative response of the Du Pont 510 emulsion irradiated in the badge dosimeter.

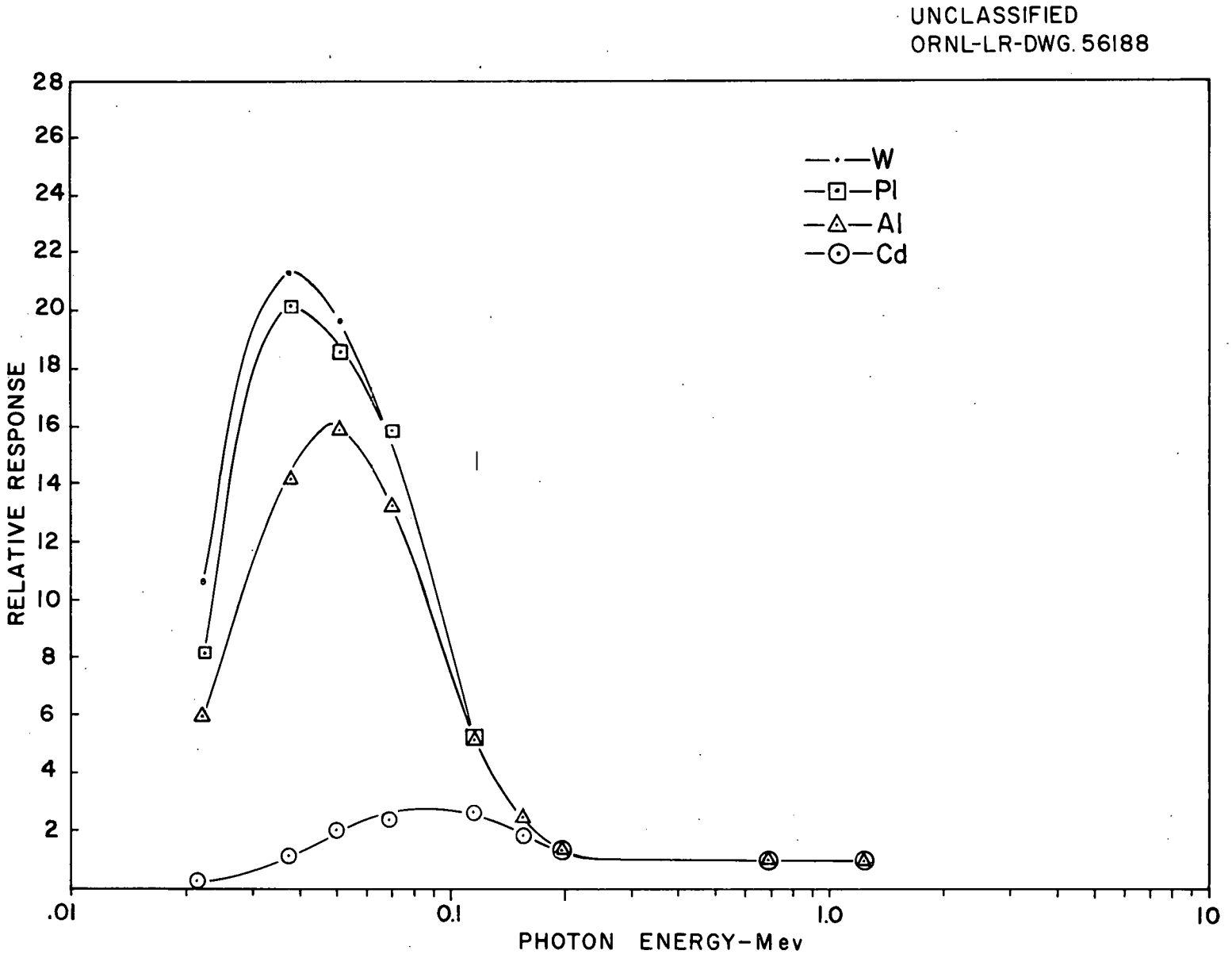


Fig. 18. Relative response of the Du Pont 555 emulsion irradiated in the badge dosimeter.

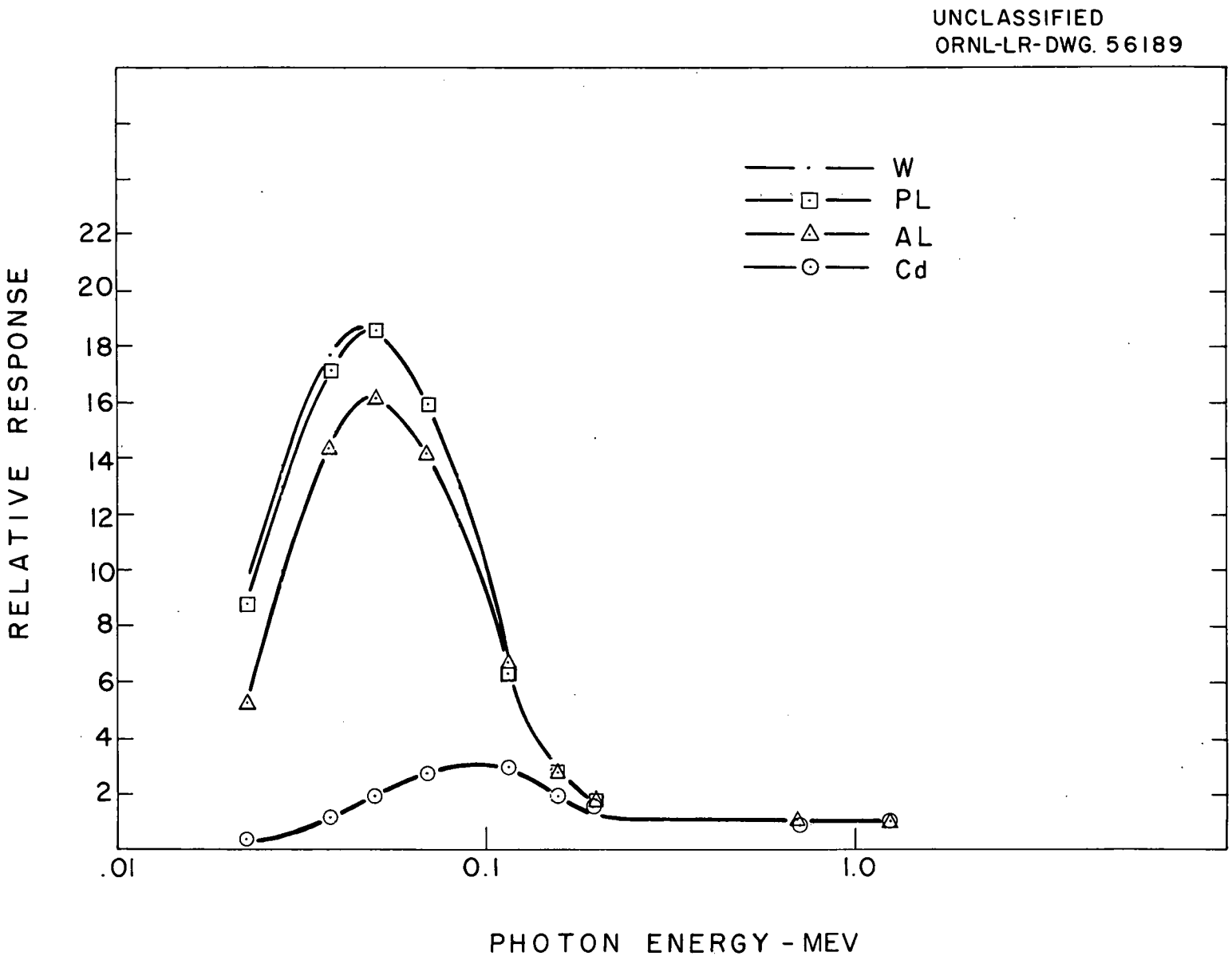


Fig. 19. Relative response of the Du Pont 834 emulsion irradiated in the badge dosimeter.

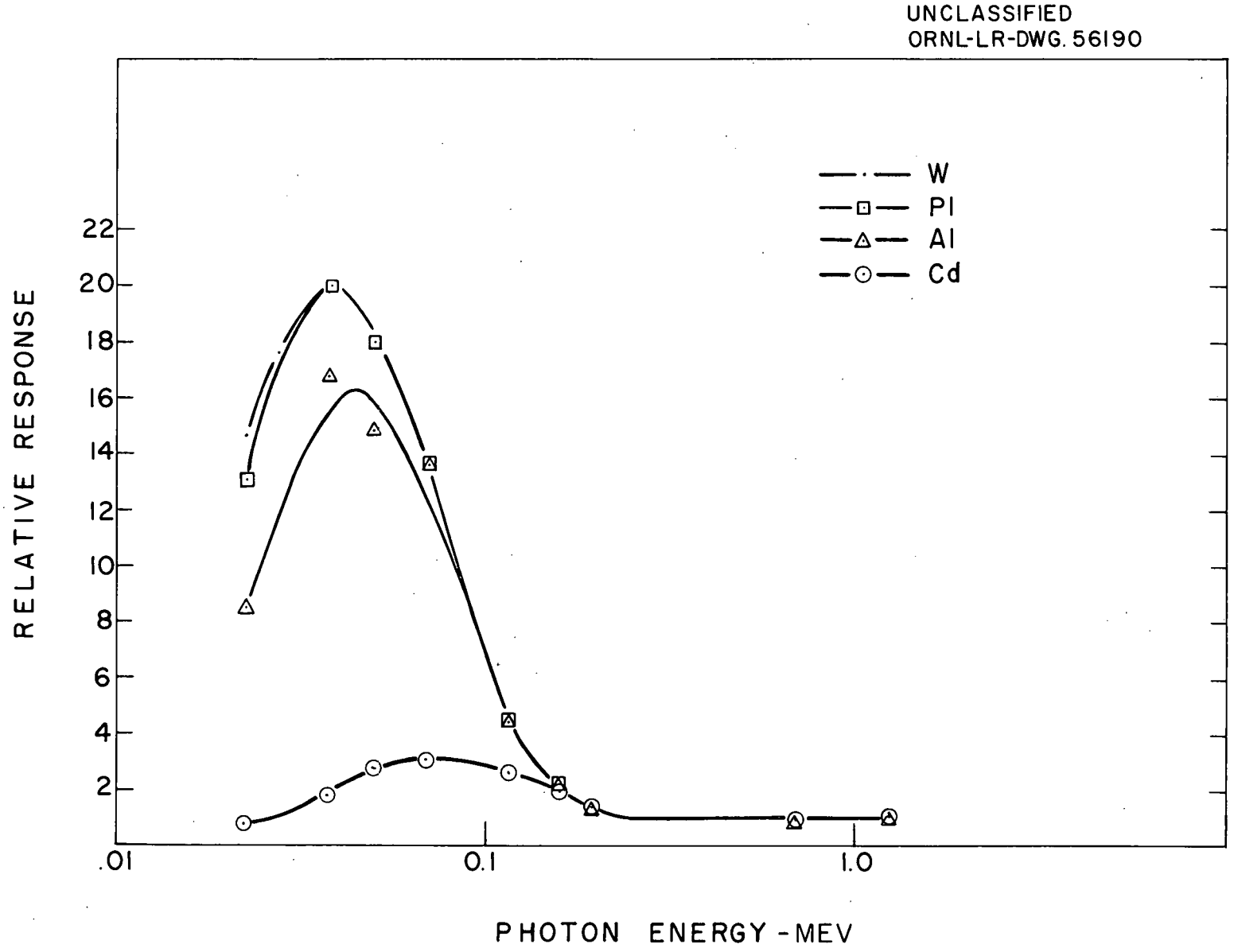


Fig. 20. Relative response of the Du Pont 1290 emulsion irradiated in the badge dosimeter.

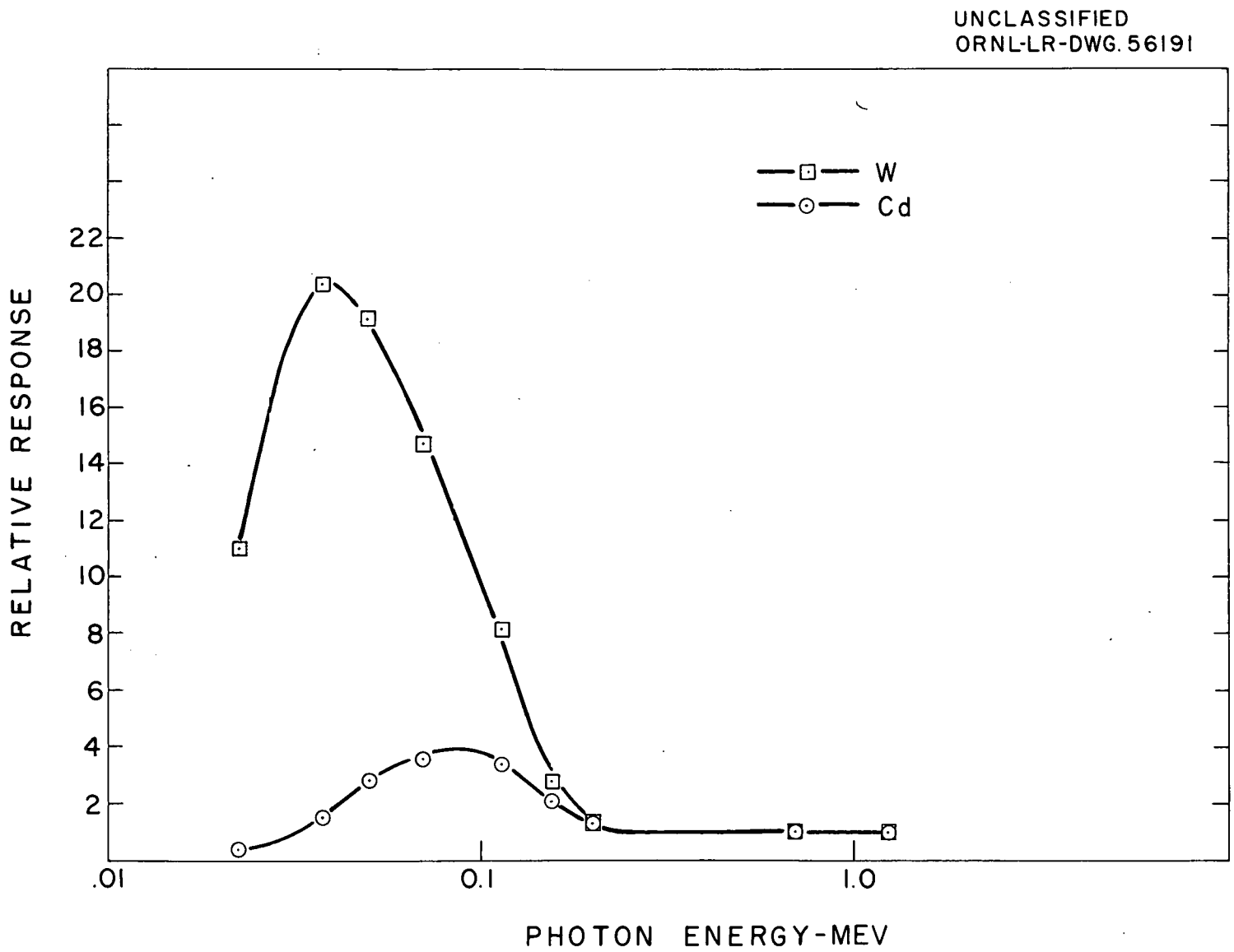


Fig. 21. Relative response of the Eastman type 3 (sensitive emulsion) irradiated in the badge dosimeter.

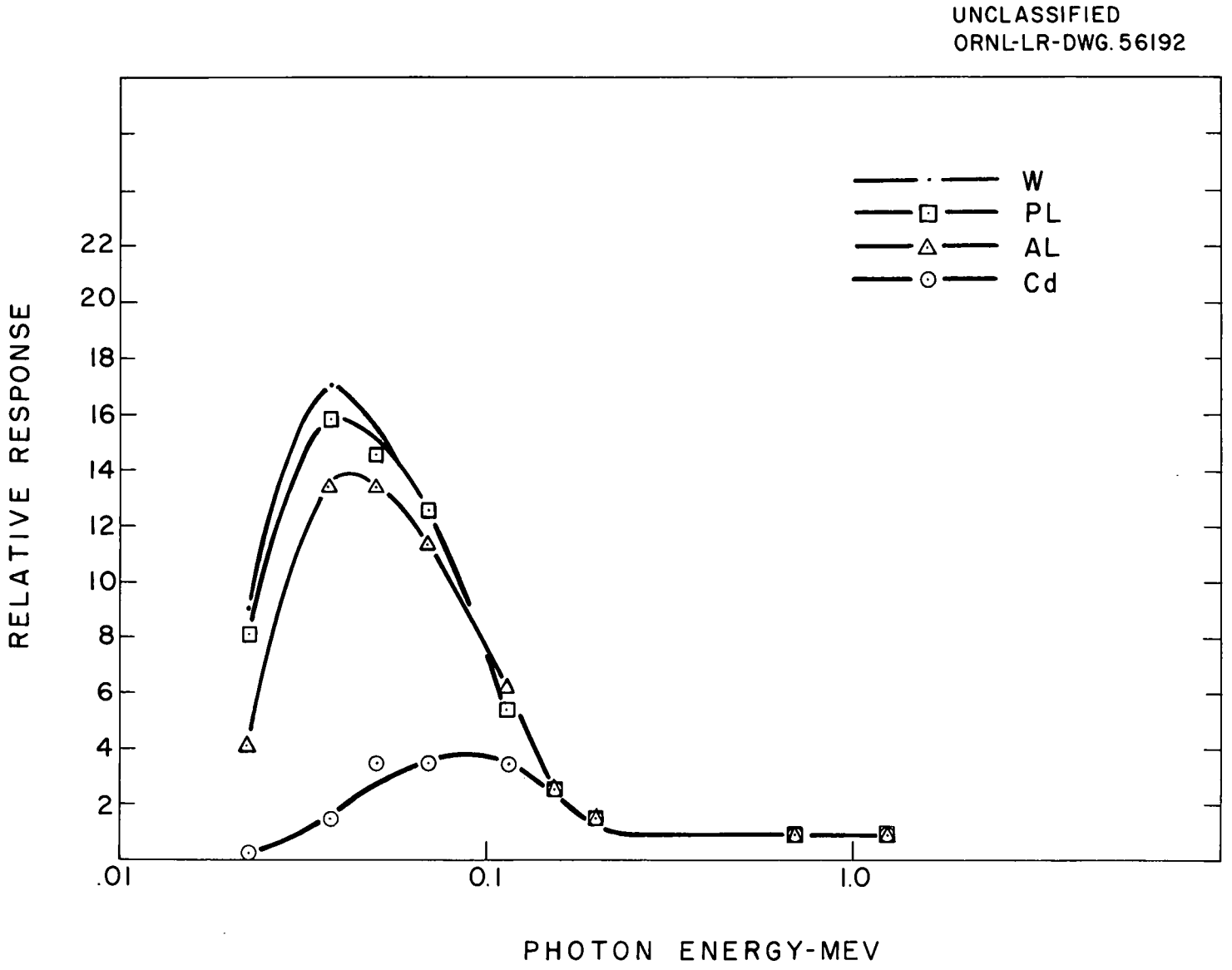


Fig. 22. Relative response of the Eastman type 3 (insensitive emulsion) irradiated in the badge dosimeter.

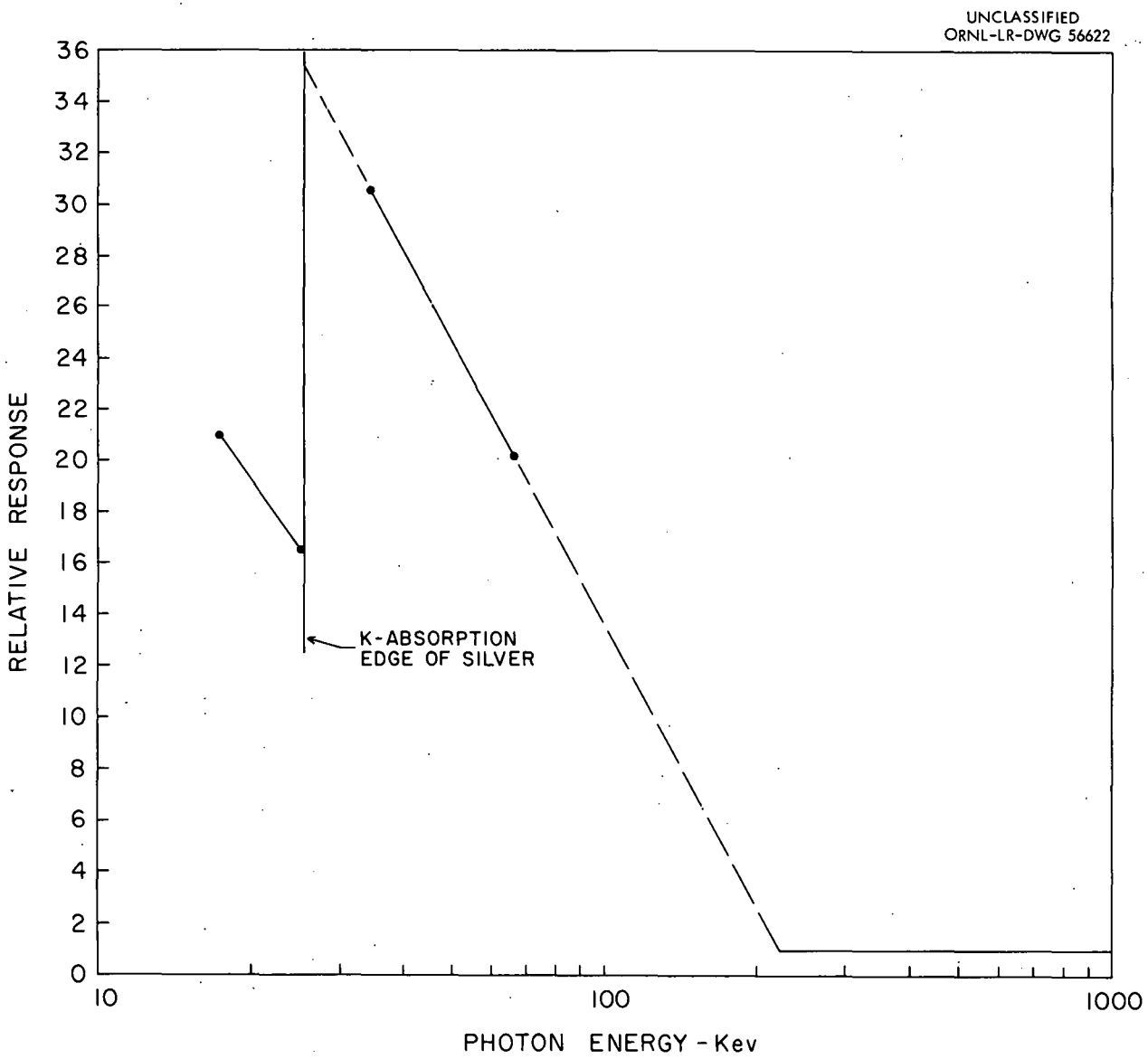


Fig. 23. Relative response of the Du Pont 555 emulsion when irradiated with fluorescence x-rays.

calibration curve to get the apparent dose, and then a ratio of the apparent dose to the exposure dose gave the relative response.

In the case of DuPont's 555 emulsion, measurements were made to verify the energy response to more nearly monoenergetic radiation. This was done using the 250 Kev x-ray machine and radiators to produce fluorescence radiation in a manner similar to that described by Seemann.[14] The choice of radiators was made so as to provide fluorescence of an energy which would bracket the energy where the film emulsion was most sensitive. This energy was assumed to be the K-absorption edge of silver, 25.5 Kev. Table 5 shows the radiators and filters used and the photon energies achieved. Villforth, et al[13] describe the fluorescence spectra of these radiators in greater detail. The filters were chosen to attenuate the K_β emission to 10 per cent of its original intensity, while attenuating the K_α component relatively little. The response of the 555 emulsion at the fluorescence x-ray energies was normalized to that for Co^{60} in the same way as was previously described. Exposure to Co^{60} was made using a NBS calibrated source and geometry similar to that for the x-ray exposures. All exposures were monitored by a 250 mr Victoreen condenser r-meter to which all film doses were related. The doses as measured by the Victoreen were corrected for the attenuation by the film packet. This varied from 48% at 17.4 Kv to 7% at 66.2 Kv. No correction was made for the energy dependence of the 250 mr Victoreen. The relative response of the 555 emulsion is illustrated in Figure 23. The choice of x-ray energies showed the marked effect of the K-edge of silver on the film sensitivity. Extrapolation of the response

---

(14) H. E. Seemann. Rev. of Sci. Inst. 21(4), 314 (1950).

Table 5

Applied KV	Radiator	Filter	Photon Energy (Kev)	Dose Rate mr/min
50	Mo	Zr 31 mc/cm^2	17.4	20.1
50	Sn	Cd 65 mg/cm^2	25.2	10.5
50	$Ce(NO_3)_3$	$BaCO_3$ 100 mg/cm^2	34.6	1.83
150	Pt	W 287 mg/cm^2	66.2	7.6
Co^{60}	--	--	1250	14.5

Table 6. Relative Density

Film No.	Filter			
	None	Lead .020"	Cadmium .030"	Copper .040"
I A	1.00	0.75	0.71	0.75
I B	1.00	1.12	1.04	1.00
II A	1.00	0.79	0.71	0.75
II B	1.00	0.91	0.86	0.91
III	1.00	1.33	1.19	1.09
IV	1.00	1.00	1.00	1.00

to the discontinuity at 25.5 Kev puts the peak response at 35 times the Co^{60} response. The extrapolated portions of the curve are shown as broken lines.

C. Beta to Gamma Response

For many years uranium beta as well as radium gamma calibration curves were used at ORNL to interpret densities. It was noticed that the ratio of the open-window uranium beta response to the Cd-shield radium gamma response was a constant within the linear range of the film. The ratio of beta to gamma response was tested by the following experiments: (1) film wrapped in a 7 mg/cm^2 absorber and placed in contact with a slab of normal uranium produced a density of 0.52 per rad compared with 0.50 per rad of radium gamma measured behind the Cd filter; and (2) a stack of film packets was given a dose of 5 rad from a normal uranium surface and the densities produced at various depths were used to extrapolate to a "surface" (7 mg/cm^2) density value of 1.68. A corresponding 5 rad dose from radium produced a density of 1.71 under Cd filter. It appears, therefore, that for betas from normal uranium, the density produced per rad in film is equal to the density produced under the Cd filter per rad in film by radium gamma.

Poddar[15] has reported that the density per unit dose produced by various beta emitters is a function of the specific ionization and the average energy of the betas, i.e.,

$$\text{Density} = \frac{Q(\text{electrons/cm})}{E(\text{Mev})}$$

---

(15) Poddar, R. K., Indian Journal of Physics 29(4) (1955).

and that the density per unit dose varies slowly with the average beta energy over a fairly wide range. Hence, in its application for routine personnel dosimetry, film is considered equally sensitive for beta and gamma radiations.

D. Intensification of Film Blackening with Metallic Filters

Since the meaning of blackening on a film is ambiguous without a knowledge of the type and approximate energy of the radiation producing it, one must have a system of filters in a film dosimeter. The necessary proximity of these filters gives rise to blackening on the film which does not always signify an exposure dose.

Using radiation sources, as described above, and metallic filters representing a wide range of atomic numbers, film was exposed to study this effect. The cases shown in Figure 24 illustrate the experimental arrangements used to demonstrate these effects. The radium-gamma source used was enclosed in 860 mg/cm^2 of aluminum and 300 mg/cm^2 of lucite. Table 6 (page 47) shows the densities found using various filters relative to the density of an unshielded film. The following information should be considered in applying heavy metal filters in film dosimetry:

Case I A - The metallic filters reduce the gamma radiation by functioning as absorbers. Most, if not all, of the electrons forescattered from the metals, are absorbed in the 110 mg/cm^2 of plastic and film wrapper between the metal and the film.

Case I B - The absorbing effect of the filters is the same as for Case I A, but the electrons backscattered from the metals are not all absorbed in the 30 mg/cm^2 film wrapper, with the result that the

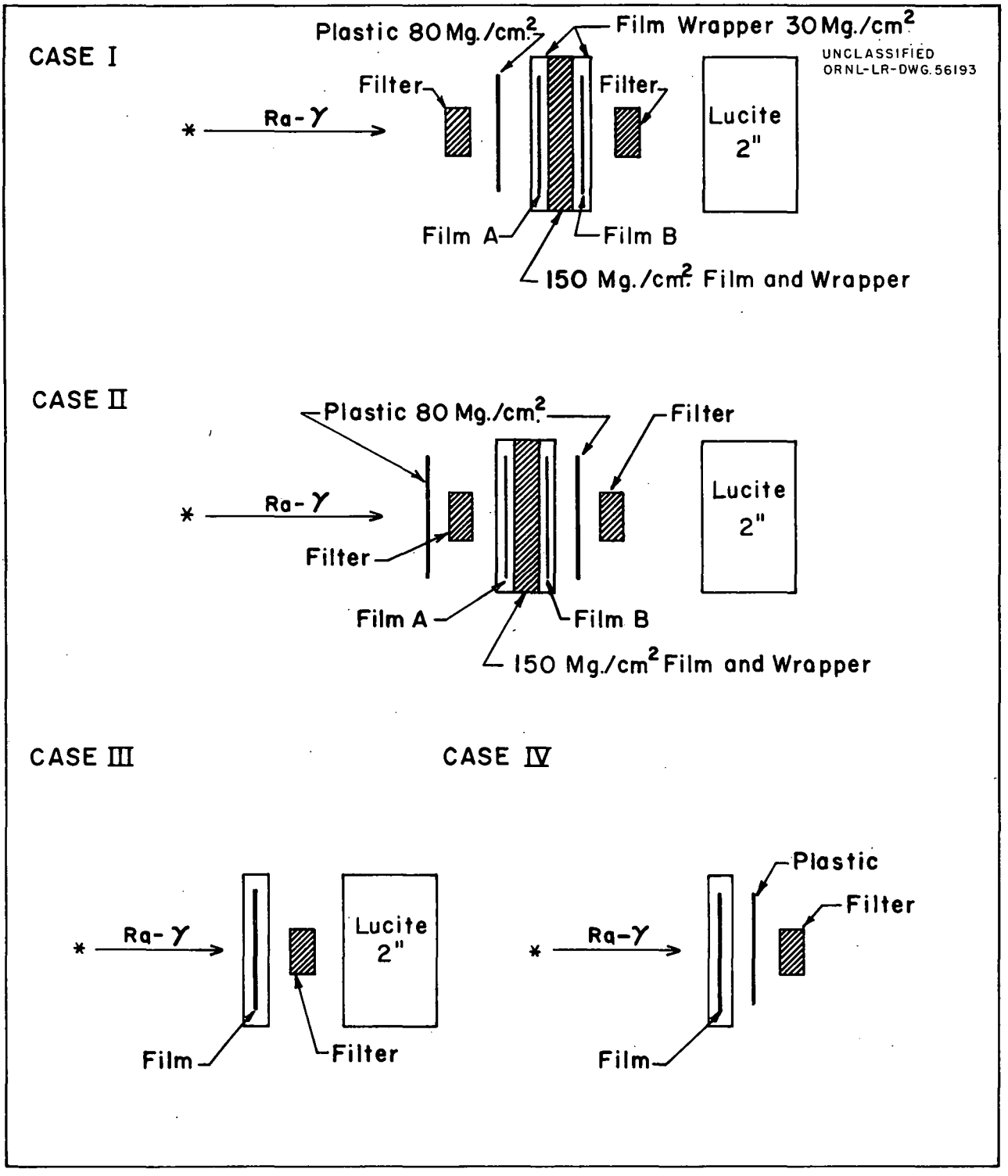


Fig. 24. Schematic representation of film and filter orientations.

effect of backscatter is equal to or greater than that of the incident filtration. Such exposures with only Cd filters could be misinterpreted to involve thermal neutrons.

Case II A - This case is similar to Case I A, except that the absorber between the metals and the film is 30 mg/cm^2 instead of 110 mg/cm^2. The 5 per cent increase in film density under the lead filter over that for Case I A is probably a real indication of the effect of forescattered electrons. Since the plastic between the Cd or the Cu and film makes no difference in the film density, it would appear that these lower atomic number metals do not contribute significant electron forescatter capable of penetrating the 30 mg/cm^2 wrapper.

Case II B - The differences in the film densities of the respective filters in Cases I B and II B indicate the increase in backscatter with the atomic number of the backscatterer.

Case III - With no incident filtration and only 30 mg/cm^2 absorber between the metal and the film, the backscatter increases the dose to the film up to 33% for a lead backscatterer.

Case IV - Essentially all backscatter is absorbed in the 110 mg/cm^2 layer of plastic and wrapper between the backscatterer and the film.

E. Latent Image Stability of Beta-Gamma Film

There have been three separate tests devised to determine the stability of the beta-gamma emulsions which are commercially available. The following is a description of these and a summary of the results.

1. Sensitive Emulsions - Exposures of the DuPont 502, 508, and 555, and the Eastman Type 3 (sensitive) were made for a period of four months. The magnitude of the exposures was one roentgen of radium gamma radiation and the frequency of exposure increased as the time for development approached. All films were developed simultaneously and read on the same Ansco Densitometer. Figure 25 shows the results obtained. The dashed line associated with each emulsion represents the average density of all its exposures. The total variation in the density per roentgen over this period will be assumed due to emulsion instability and reproduction of the exposure on the film ring (Figure 12). The error limits represented show the spread in the readings of the four films used at each exposure.

For the DuPont 502, 508, and 555 emulsions, the maximum variations from the average densities are 5%, 15%, and 9%, respectively. For the Eastman Type 3 (sensitive) the maximum variation is 15%. Of this about 2% is due to the non-reproducibility in the exposures. It will be noted from Figure 25 that the density per exposure for the Eastman Type 3 increased instead of faded with time. In the other emulsions, the fogging and fading effects seem to offset each other so that no obvious trend is seen.

2. Insensitive Emulsions - This test is similar to the one mentioned above except that a 100 roentgen exposure was given and the duration of the test was one month instead of four months. Figure 26 shows

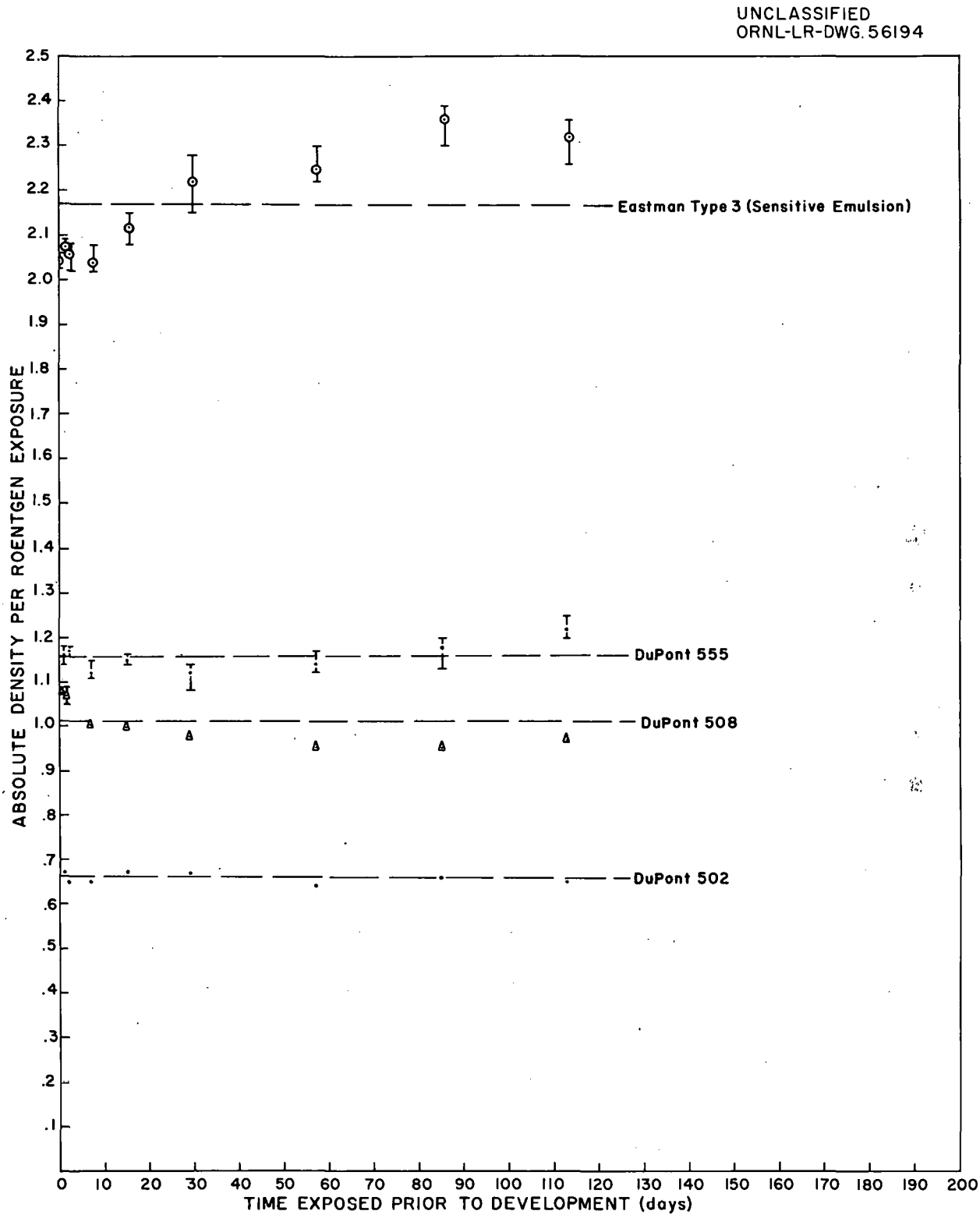


Fig. 25. Latent image stability of various sensitive emulsions.

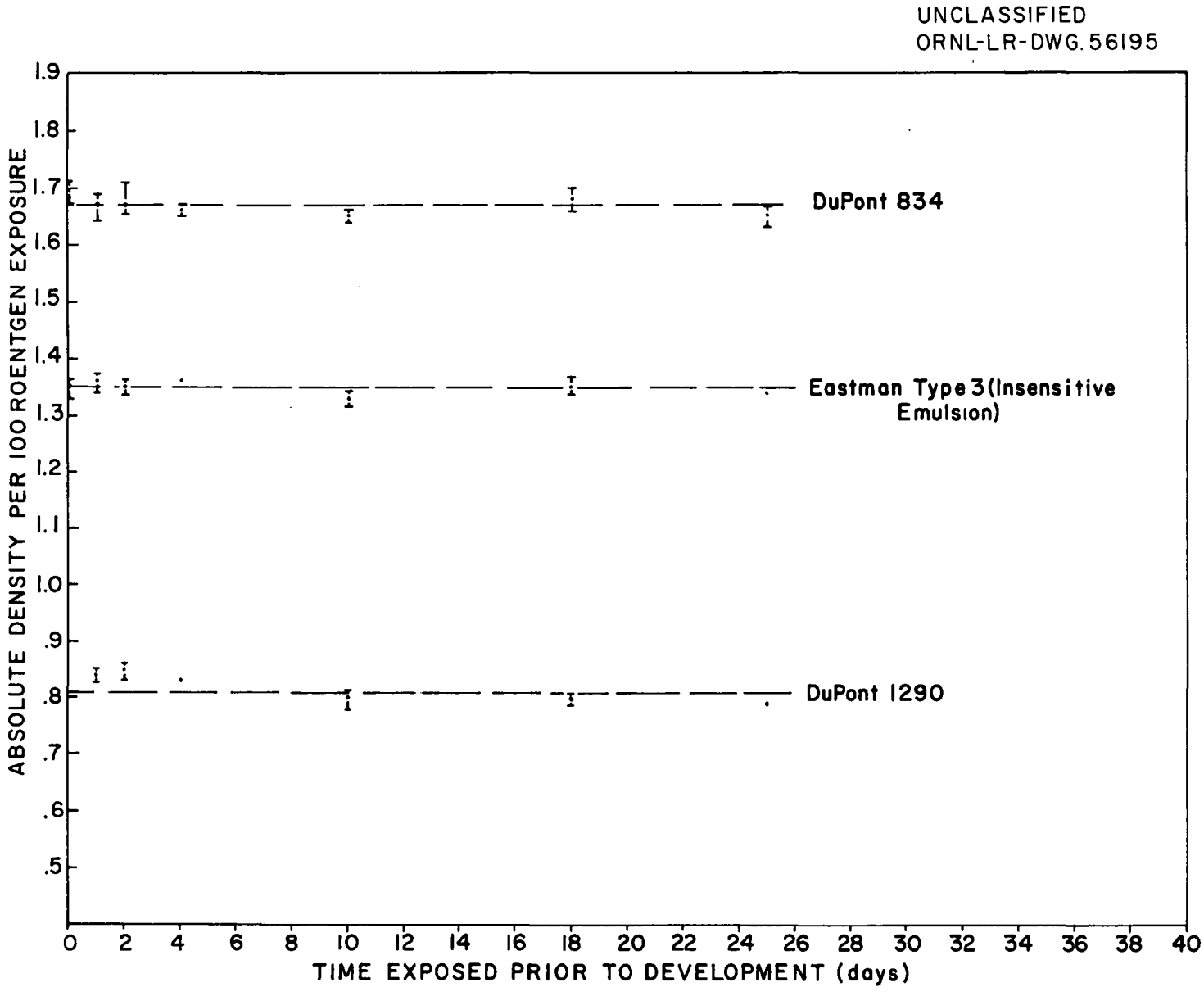


Fig. 26. Latent image stability of various insensitive emulsions.

the Eastman Type 3 (insensitive) and DuPont 834 and 1290 to be very stable over this period of time. No emulsion showed greater than 2% variation.

3. Integration Consistency - The DuPont 502 and 555 have each been exposed to 10 mr/day for three months, and it was found in each case that the accumulated dose read from the film density agreed with the exposure dose within 5%. This experiment is continuing and included in the study are 10 mr/day exposures of the glass rods to evaluate their use as a long term integrator.

VII. ANALYSIS OF BETA-GAMMA DOSIMETRY TECHNIQUES

In view of the maximum permissible limits and the dose determinations required by the NCRP as presented in Section III A, it is necessary to consider how well the present system fulfills these requirements. The practical measurement of the dose to the critical organ, D_C, presents the problem of measuring the gamma and x-ray exposures over a range of 0.02 to 5 Mev. That is, does using the Cd filter reading to simulate the dose which would penetrate to a depth of one centimeter in tissue actually represent this dose over the entire energy range? A calculation of the relative transmission of one cm of tissue and a similar calculation for the Cd filter are shown as a function of energy in Figure 27. They are curves A and B respectively. When the increase in apparent dose to the film due to the increased sensitivity of the film below 200 Kev is added to the transmission through the Cd-Au filter, Curve C in Figure 27 results. The energy response used was that determined for the DuPont 555 emulsion

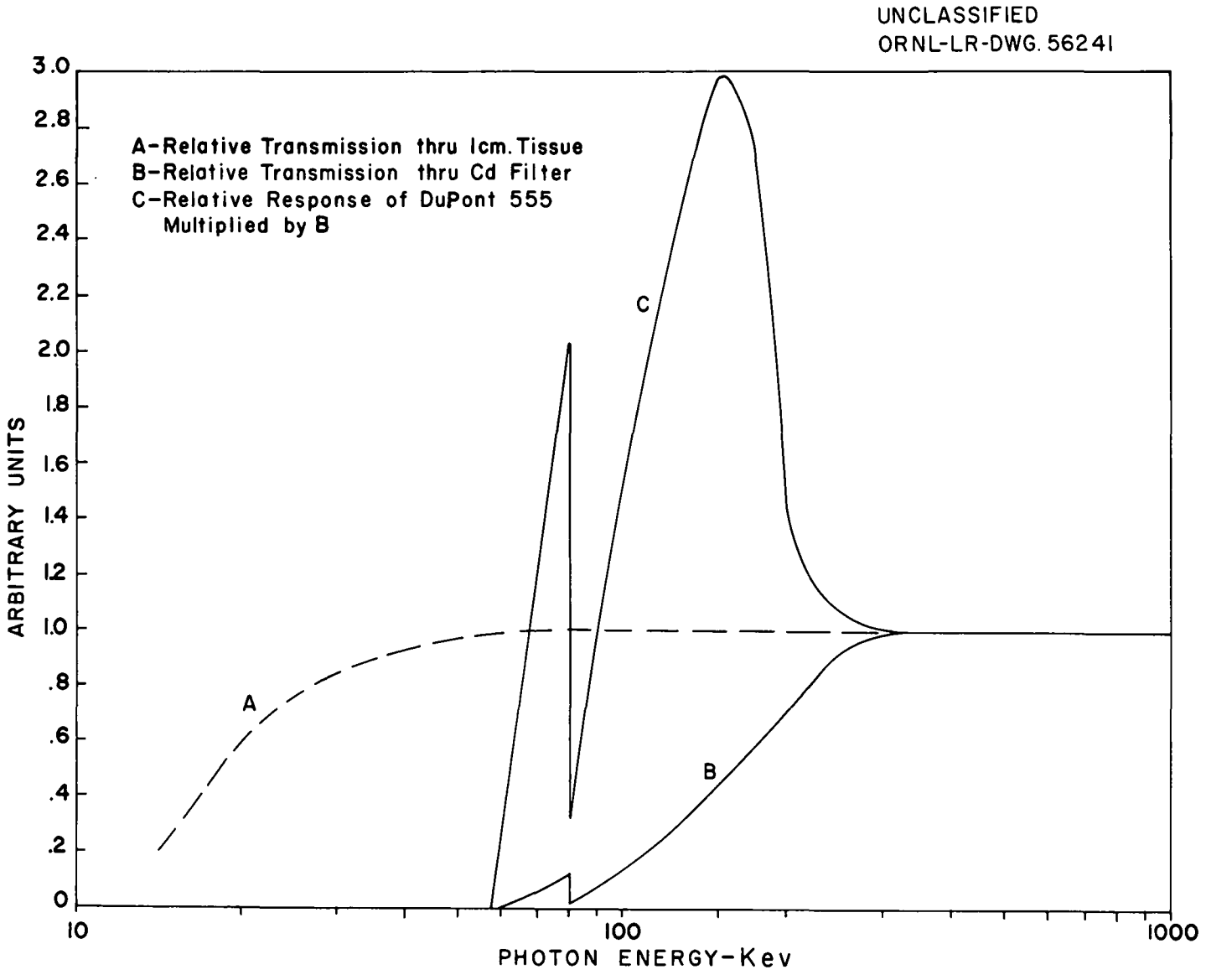


Fig. 27. Comparison of the relative dose reading under Cd filter with the relative air dose which will penetrate 1 cm in tissue.

by the monochromatic x-ray source reported in Section V B. From Figure 27 two things become apparent. First, an overestimation of D_C occurs between 70 and 80 Kev and between 90 and about 200 Kev; and second, the dose is underestimated below 70 Kev and not recorded at all below 60 Kev. However, low energy photons could be scattered under the Cd filter.

To determine the D_S dose the beta contribution must be measured in addition to the X and gamma radiation. Since there is an 80 mg/cm^2 minimum absorber interposed between the film and the source, it is impossible to interpret the film density as the dose at 7 mg/cm^2 without knowing the beta energy. Routinely, it is assumed that the radiation is from natural uranium which emits a 1.7 Mev beta. Beta rays of this energy are approximately 40% absorbed in 80 mg/cm^2. Recognizing the limitation of the badge dosimeter with regard to beta dosimetry is essential if meaningful results are to be obtained from it.

VIII. PROPOSALS AND RECOMMENDATIONS

The following are proposed as additions to the film dosimetry system which would make it more useful in cases of unusual exposure:

1. Use the ratio of the aluminum and cadmium filter readings to make an estimation of the effective energy of the photon radiation. Once this is done, the dose could be determined from the aluminum reading by correcting for energy dependence.
2. Verify the response of film, when shielded by 0.040 in. of aluminum and by 0.034 in. of Cd and 0.005 in. of Au, to

mono-energetic x-rays similar to those described in Section V B. Those energies should be expanded to cover the range from 70 to 100 Kev.

3. Use IBM to process all film densities so that every exposure may be examined to see if the effective energy of the gamma exposure is such that a correction need be made in D_C.

4. Maintenance of close liaison between the area surveyor and the personnel monitoring section to intelligently interpret beta exposures.

ORNL-3126
UC-41 — Health and Safety
TID-4500 (16th ed., Rev.)

INTERNAL DISTRIBUTION

1. C. E. Larson
2. Biology Library
3-4. Central Research Library
5. Reactor Division Library
6. ORNL — Y-12 Technical Library Document Reference Section
7-26. Laboratory Records Department
27. Laboratory Records, ORNL R.C.
28. R. G. Jordan (Y-12)
29. J. P. Murray (K-25)
30. A. M. Weinberg
31. J. A. Swartout
32. F. R. Bruce
33. W. H. Jordan
34. A. H. Snell
35. C. E. Winters
36. J. D. McLendon
37. H. G. MacPherson
38. W. Y. Gissel
39. H. E. Seagren
40. J. H. Gillette
41. J. A. Cox
42. G. C. Cain
43. E. P. Blizard
44. M. L. Nelson
45. A. S. Householder
46. C. J. Borkowski
47. R. S. Livingston
48. J. L. Fowler
49. J. H. Frye, Jr.
50. M. T. Kelley
51. E. H. Taylor
52. A. Hollaender
53. F. L. Culler
54. K. Z. Morgan
55. E. G. Struxness
56. G. S. Hurst
57. W. S. Snyder
58. S. I. Auerbach
59. B. R. Fish
60. J. C. Hart
61-86. D. M. Davis
87. A. D. Warden
88-113. E. D. Gupton
114-139. W. T. Thornton
140. T. J. Burnett
141. E. J. Witkowski
142. J. C. Frye (consultant)
143. W. H. Langham (consultant)
144. G. M. Fair (consultant)
145. L. S. Taylor (consultant)
146. R. L. Platzman (consultant)
147. E. T. Odum (consultant)

EXTERNAL DISTRIBUTION

148. C. P. Straub, Public Health Service, Robert A. Taft Sanitary Engineering Center.
149. W. T. Ham, Medical College of Virginia, Richmond, Va.
150. O. W. Kochtitzky, Tennessee Valley Authority, 717 Edney Building, Chattanooga, Tenn.
151. C. S. Shoup, Biology Branch, USAEC, Oak Ridge, Tenn.
152. T. S. Chapman, Health Physics and Medical Division, Dow Chemical Company, P. O. Box 2131, Denver, Colo.
153. Carnick A. Markarian, Captain, USAF, MSC, Director
154. S. Allen Lough, Division of Biology and Medicine, USAEC, Washington
155. James E. Turner, Division of Biology and Medicine, USAEC, Washington
156. Division of Research and Development, AEC, ORO
157-819. Given distribution as shown in TID-4500 (16th ed., Rev.) under Health and Safety category (100 copies — OTS)