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Radiation & Shielding Data for Pu-Al Components

Radiation intensities associated with the various Pu-Al fuel and target components considered to date for a Cm^{244} production program vary according to quantity and isotopic content of the Pu used, age of the Pu since separation, and component design. This report summarizes the results of calculations made to determine expected radiation intensities, and includes associated shielding data for use in evaluating future shielding requirements for billet preparation and tube extrusion facilities. Results are shown in tabular form in Table I. In addition, shielding information is presented graphically in attachments.

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TABLE I

Pu-Al Component	Calculated Radiation Intensity, mr(cm)/hr						
	Gamma				Neutrons		
	3"	1'	18"	1 meter	3"	1'	18"
Case I							
18% Pu^{240} Fuel Core (single)*	3200	200	90	—	30	2	1
" " " Billet**	1340	85	40	—	30	2	1
" " " Tube	1800	400	—	90	2	<1	<1
Case II							
8% Pu^{240} Fuel Core (single)	300	20	10	—	8	<1	<1
" " " Billet	140	9	5	—	8	<1	<1
" " " Tube	100	25	—	5	<1	<1	<1
Case III							
53% Pu^{240} Target Core (single)	2100	135	60	—	240	15	7
" " " Billet	1000	65	30	—	240	15	7
" " " Tube	1500	350	—	85	15	4	2
Case IV							
13% Pu^{240} Fuel Core (single)	500	30	15	—	20	1.5	<1
" " " Billet	225	15	7	—	20	1.5	<1

* Radiation intensities at 1' and 18" from double-core castings will be ~ twice that of the single core; from risers ~ 2/3 that of the single core; from scrap cans ~ the same as the single core.

** The billet radiation intensities are those from the cores multiplied by the shielding transmission factor for the 0.5" Al cladding.

The basic guides and assumptions used in the calculations are as follows:

1. The Am^{241} and U^{237} daughters of Pu^{241} contribute the majority of the gamma radiation.

2. The number of curies of Am^{241}/gram Pu^{241} is given by $4.28 \times 10^{-4}\ t^{(1)}$, where t = time in days since separation of the Pu.

3. The number of curies of U^{237}/gram Pu^{241} is given by $4.7 \times 10^{-3} (1-e^{-.102t})$[1]. When time since separation is $\leqslant 14$ years but $>$ a few weeks, the value of 4.7×10^{-3} may be used.

4. The yields or relative intensities used for the significant gamma energies associated with the decay of Pu^{241} were:

Am^{241}		U^{237}		Pu^{241}	
17 Kev	.37	60 Kev	.37	145 Kev (with 4.4×10^{-3}% of decays)	.07
26 Kev	.027	207 Kev	.21		
60 Kev	.37	334 Kev	.025		

5. 45 Kev gammas were assumed to be coincident with 10^{-4} of the Pu^{240} and Pu^{242} decays.

6. Neutron intensities were calculated from the Pu^{240} and Pu^{242} content.

7. The neutron dose rate per unit flux density used was 0.115 mrem/hr per neutron/sec/cm^2[1].

8. The basic formula R = 6 CE was used in calculating radiation intensities from cores and billets, as results compared favorably with those obtained by use of more complex formulae.

9. The basic formula $I = \frac{I_o}{a} (\phi_1 + \phi_2)$ was used in calculating intensities from tubes and double core castings.

10. The basic formula $\frac{n/s}{4\pi r^2}$ was used for determining N_f flux density.

11. Gamma shielding was calculated by determining an effective transmission factor, Be^{-ux}, for all gamma energies emitted by the source.

12. Neutron shielding was determined by adapting the formula $D(R, t) = \frac{N_f \, 4\pi r^2 D_H(r)}{4\pi R^2} e^{-\Sigma t}$[2] for hydrogenous materials and $I = I_o e^{-\Sigma_r t}$ for non-hydrogenous materials.

13. Self absorption of gammas within the source was considered in core and billet calculations but neglected in tube calculations (for energies $\geqslant 26$ Kev).

14. Values of u for determining self absorption in the Pu-Al cores were obtained by multiplying the linear absorption coefficient, u, for the various energies in Pu by the % by volume of Pu in the core[3].

15. Values of u, cm^{-1}, used for the various gammas in Pu were:

26 Kev	800	145 Kev	40
45 Kev	160	207 Kev	22
60 Kev	82	334 Kev	7

16. Isotopic content and quantity of the Pu, age since separation, and dimensions of the various components assumed in calculating radiation intensities are as follows:

	Component	Isotopic Content, Quantity, Age Since Separation	Dimensions
Case I	18% Fuel Cores, Billets, Tubes	77% Pu^{239}, 18% Pu^{240}, 5% Pu^{241} - 700g Pu/core, billet or tube - 2 years since separation	Core - hollow cylinder, 5.7" O.D., 4.4" I.D., 7.4" length Billet - same as core, with 0.5" Al cladding Tube - 10' active length
Case II	8% Fuel Cores, Billets, Tubes	91% Pu^{239}, 8% Pu^{240}, 1% Pu^{241} - 435g Pu/core, billet or tube - 1 year since separation	Core & Billet - same as above Tube - 13' active length
Case III	53% Target Cores, Billets, Tubes	10% Pu^{239}, 53% Pu^{240}, 15% Pu^{241}, 22% Pu^{242} - 1200g Pu/core, billet or tube - 100 days since separation	Core - hollow cylinder, 5.7" O.D., 4.4" I.D., 6.7" length Billet - same as core, with 0.5" Al cladding Tube - 13' active length
Case IV	13% Fuel Cores, Billets	84% Pu^{239}, 13% Pu^{240}, 3% Pu^{241} - 700g Pu/core or billet - 100 days since separation	Core & Billet - same as for 18% fuel cores and billets

All radiation and shielding calculations made in obtaining the information presented in the attachments are not included for the sake of brevity. However, examples of each type calculation are shown below and represent the methods used for all the various Pu-Al components.

Radiation Intensities from 8% Pu^{240} Fuel Cores - Case II

1. Isotopic content - 91% Pu^{239} Pu Density = 19 g/cm^3
   8% Pu^{240}
   1% Pu^{241}

2. Time since separation - 1 year

3. Amount of Pu/core - 435g Pu ($\backsim$ 400g Pu^{239})

4. Core dimensions - 5.7" O.D.
   4.4" I.D.
   7.4" length

5. Core volume $= \pi r^2_{O.D.} h - \pi r^2_{I.D.} h = 3.14\ (2.85)^2\ 7.4 - 3.14\ (2.2)^2\ 7.4$

$= 77\ in^3 = 1260\ cm^3$

6. Projected volume $=$ Projected Area (cm^2) X mean free path (MFP)

$= 7.4\ (2.54) \times 5.7\ (2.54) \times$ MFP

$= 272\ cm^2 \times$ MFP

7. $\text{MFP} = \dfrac{1}{u} = \dfrac{1\ (100)}{u \times \%\ \text{Pu by volume in core}} = \dfrac{1\ (100)}{u \times \frac{435g}{19g/cm^3} \div 1260\ cm^3}$

$= \dfrac{1}{u \times 0.0182}$

8. Therefore, projected volume $= 272 \times \dfrac{1}{u \times 0.0182}$

For 60 Kev Am^{241} gamma:

Projected volume $= 272 \times \dfrac{1}{82 \times 0.0182} = 182\ cm^3$

Pu in projected volume $= 182\ cm^3 \times$ g Pu/cm^3 in core $= 182 \times \dfrac{435}{1260} = 63g$

Pu^{241} in projected volume $= 63g \times 1\% = 0.63g$

Curies 60 Kev Am^{241} in projected volume $= 4.28 \times 10^{-4}$ t X 60 Kev yield X g Pu^{241}

$= 4.28 \times 10^{-4}\ (365) \times 0.37 \times 0.63g$

$= .036c$

R $= 6$ CE $= 6 \times .036 \times .06 = 0.013$ = <u>13 mr/hr @ 1'</u>

= <u>6 mr/hr @ 18"</u>

= <u>210 mr/hr @ 3"</u> (using inverse square)

(Contributions from other significant gammas, 207 Kev, 330 Kev, etc. were determined in the same manner as above.)

For Pu^{240} neutron intensity:

N_f emission from Pu^{240} = 1380 n/sec/g [1]

$1380 \times 435g \times 8\%\ (Pu^{240}) = 4.8 \times 10^4$ n/sec

$$\frac{n/s}{4\pi r^2} = \frac{4.8 \times 10^4}{4(3.14)(30.5)^2} = 4.1 \ n/sec/cm^2 \ @ \ 30.5 \ cm \ (1')$$

$$= 4.1 \times .115 = .47 = \underline{0.5 \ mrem/hr \ @ \ 1'}$$

$$= \underline{0.2 \ mrem/hr \ @ \ 18''}$$

$$= \underline{8 \ mrem/hr \ @ \ 3''}$$

<u>Radiation Intensities from 8% Pu^{240} Fuel Tubes - Case II</u>

1. Isotopic content and time since separation same as for 8% fuel cores.
2. Active tube length - 13'.
3. Self-absorption of energies $\geqslant$ 26 Kev and absorption by the 45 mil Al cladding was neglected.

For 60 Kev Am^{241} gamma:

$$\text{Curies 60 Kev } Am^{241}/g \ Pu^{241} = 4.28 \times 10^{-4} \ t \times 60 \text{ Kev yield}$$

$$= 4.28 \times 10^{-4} (365) \times 0.37$$

$$= 0.058c$$

$$R = 6 \ CE = 6 \times 0.058 \times .06 = .021 = 21 \ mr/hr \ @ \ 1'/g \ Pu^{241}$$

$$1' \text{ of tube} = \frac{1}{13} \times 435g = 33.5g \ Pu = 33.5 \times 1\% = 0.34g \ Pu^{241}$$

intensity @ 1' from 1' length of tube = 21 mr/hr X 0.34g = 7.1 mr/hr

$$\text{intensity @ 1' from 13' tube} = I = \frac{Io}{a}(\phi_1 + \phi_2)$$

$$= \frac{7.1}{1}(\text{arctan } 6.5 + \text{arctan } 6.5)$$

$$= 7.1 \ (1.42 + 1.42 \text{ radians})$$

$$= \underline{20 \ mr/hr \ @ \ 1'}$$

$$@ \ 1 \text{ meter} = \frac{7.1}{3.2}(1.11 + 1.11)$$

$$= \underline{5 \ mr/hr}$$

$$@ \ 3'' = \frac{7.1}{.25}(1.53 + 1.53)$$

$$= \underline{87 \ mr/hr}$$

(Contributions from other gammas were determined in same manner as above.)

Gamma Transmission Factors through Various Shielding Media - 16% Pu^{240} Fuel Cores
Case I

1. Fraction of total gamma due to Am^{241} 60 Kev - 0.875
   " " " " " " U^{237} 207 Kev - 0.075
   " " " " " " U^{237} 330 Kev - 0.045

2. Linear absorption coefficients, u, in various materials for the significant gammas are:

u (cm^{-1})

	Al	Pb	Pb Glass*	Lucite	Safety Glass(4)
60 Kev	0.75	38	19.7	0.2	1.97
207 Kev	0.33	9	1.2	0.14	0.28
330 Kev	0.28	3.3	0.8	0.12	0.16

* density - 3.3 g/cm^3

Al - 0.5" (1.27 cm)

	u	x	ux	B	e^{-ux}	Be^{-ux}	**Effective Be^{-ux}
60 Kev	0.75	1.27	.95	1	0.387	0.387	0.338
207 Kev	0.33	1.27	.42	1	0.658	0.658	0.049
330 Kev	0.28	1.27	.36	1	0.701	0.701	0.032

Transmission Factor = 0.419

** obtained by multiplying Be^{-ux} by the fraction of total gamma due to each energy

Safety Glass (2.3 g/cm^3) - 0.25" (0.63 cm)

	u	x	ux	B(4)	e^{-ux}	Be^{-ux}	Effective Be^{-ux}
60 Kev	1.97	0.63	1.25	3	0.288	0.865	0.756
207 Kev	0.28	0.63	0.175	1	0.839	0.839	6.3×10^{-2}
330 Kev	0.16	0.63	0.10	1	0.905	0.905	4.07×10^{-2}

Transmission Factor = 0.860

(Transmission factors for other shielding media and varying thicknesses were determined in same manner as above.)

[redacted]

<u>Neutron Transmission Factors through Various Shielding Media</u>

1. $$D(R, t) = \frac{N_{\xi}\, 4\pi r^2\, D_H(r)\, e^{-\xi_r t}\ \text{mrem/hr}}{4\pi R^2} \quad (2)$$

Where, $D(R, t)$ = radiation intensity through a hydrogenous non-aqueous shield of t thickness @ R distance, in cm, from a source

N_{ξ} = source strength, n/sec

$r = P_H t_H/0.111$ cm

$P_H t_H$ = density of hydrogen, g/cm^3, in shield multiplied by the shield thickness, t, in cm

$4\pi r^2 D_H(r)$ = radiation intensity, mrem/hr, through a pure hydrogen shield of thickness r cm, multiplied by $4\pi r^2$

ξ_r = macroscopic removal cross section, cm^{-1}, for all elements in the shield other than hydrogen

t = shield thickness, cm

R = distance from receptor to source, cm

<u>Lucite - 2" (5.08 cm)</u>

$$\xi_r = \left[\frac{\xi_c}{P_c}\, a_1 + \frac{\xi_o}{P_o}\, a_2\right] P$$

Where ξ_c = macroscopic removal cross section, cm^{-1} for carbon

ξ_o = " " " " " " oxygen

P_c = Density of carbon, g/cm^3

P_o = " " oxygen, g/cm^3

a_1, a_2 = fractions by weight of carbon and oxygen in the material

P = density of shield material, g/cm^3

$$= \left[(5.5 \times 10^{-2})\, 0.6 + (4.3 \times 10^{-2})\, 0.32\right] 1.2$$

$$= 5.62 \times 10^{-2}$$

$$= .0562\ \text{cm}^{-1}$$

[redacted]

$P_H = 0.096$

$t_H = 5.08$

$P_H t_H = 0.486$

$r = 4.38$

$4 \pi r^2 D_H (r) = 7 \text{ X } 10^{-2}$

$\Sigma_r = 0.0562$

$\Sigma_r t = 0.285$

$e^{-\Sigma_r t} = 0.753$

$4 \pi r^2 D_H (r) e^{-\Sigma_r t} = 5.27 \text{ X } 10^{-2}$

$\dfrac{4 \pi r^2 D_H (r) e^{-\Sigma_r t}}{0.115^{*}} = 4.58 \text{ X } 10^{-1}$ = transmission factor through 2" lucite

* 0.115 mrem/hr/n/sec/cm^2 was combined in the value D_H in the formula: therefore, a correction is necessary, dividing by 0.115, to obtain the correct transmission factor when applied to radiation intensities previously calculated by using the 0.115 value.

2. $I = Io\, e^{-\Sigma_T t}$

Where I = radiation intensity through a non-hydrogenous shield of t thickness

Io = unshielded radiation intensity

Σ_T = total removal cross section for the shield material, cm^{-1}

t = shield thickness, cm

<u>Safety Glass - 1"</u>

$$\Sigma_T = \left[\frac{\Sigma_{Si}}{P_{Si}} a_1 + \frac{\Sigma_o}{P_o} a_2 \right] P$$

Where $\dfrac{\Sigma}{P}$ = removal cross section divided by density for silicon and oxygen

a_1, a_2 = fractions by weight of silicon and oxygen in material

P = density of material, g/cm^3

$$\xi_T = \left[(3 \times 10^{-2})\ 0.47 + (4.5 \times 10^{-2})\ 0.53 \right] 2.3$$

$$= 3.8 \times 10^{-2}\ (2.3)$$

$$= 0.0874$$

$$e^{-\xi_T t} = e^{-(0.0874 \times 2.54)} = e^{-0.222} = 0.801 = \text{transmission factor through 1" safety glass}$$

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(1) Surface Dose from Plutonium, W. C. Roesch, reprint from "Progress in Nuclear Energy, Series XII - Volume I - Health Physics", Pergamon Press, 1959, and HW-61-358.

(2) Nuclear Engineering Handbook, H. Etherington, Editor: McGraw Hill, 1958.

(3) Radiation Shielding, B. T. Price, C. C. Horton and K. T. Spinney; Pergamon Press, 1957.

(4) Nucleonics, Volume 16, No. 10 - October, 1958.

PBV:RLH:lra

CASE I - Gamma Shielding Transmission Factors

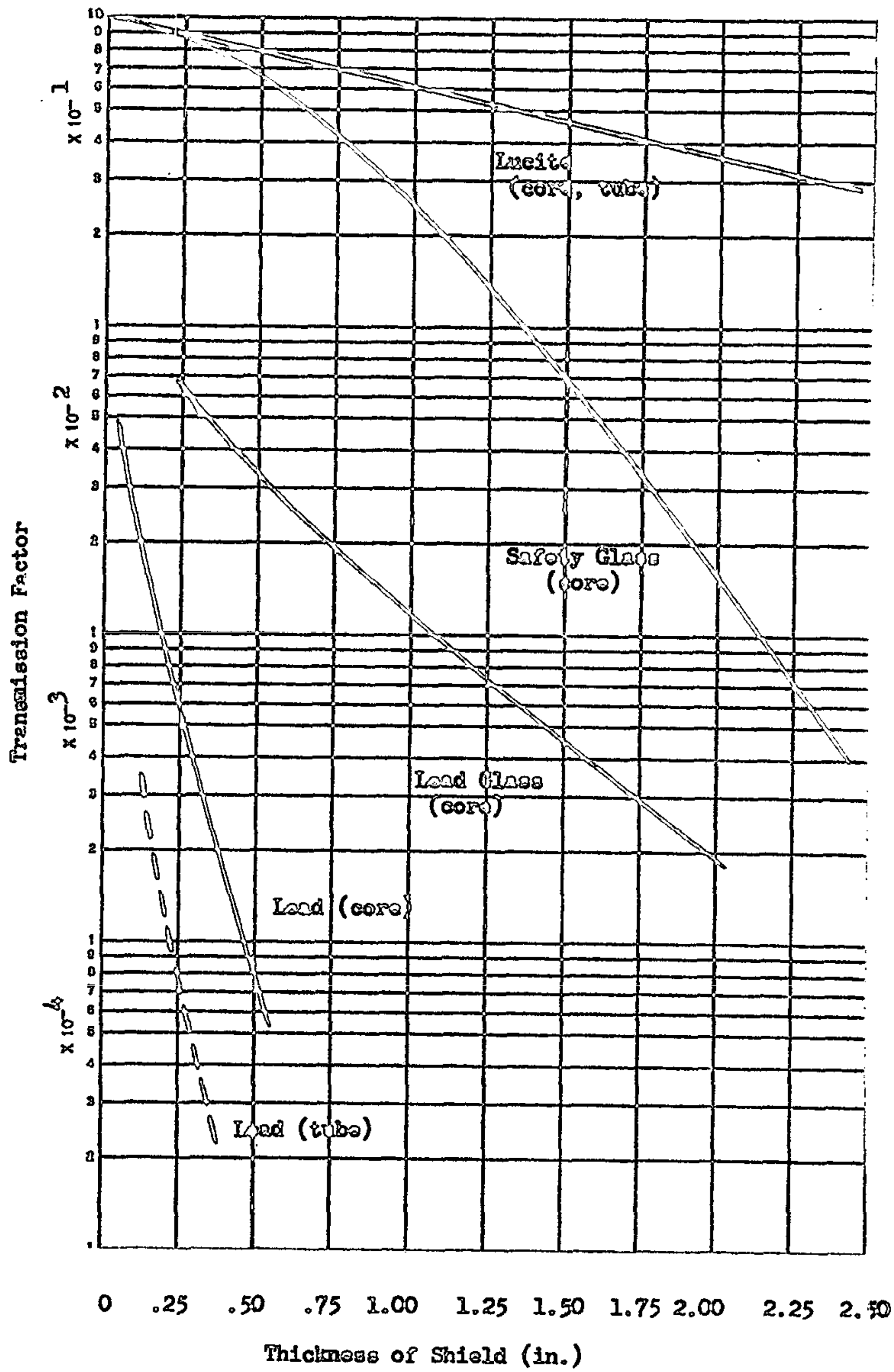

Figure 2 CASE II - Gamma Shielding Transmission Factors

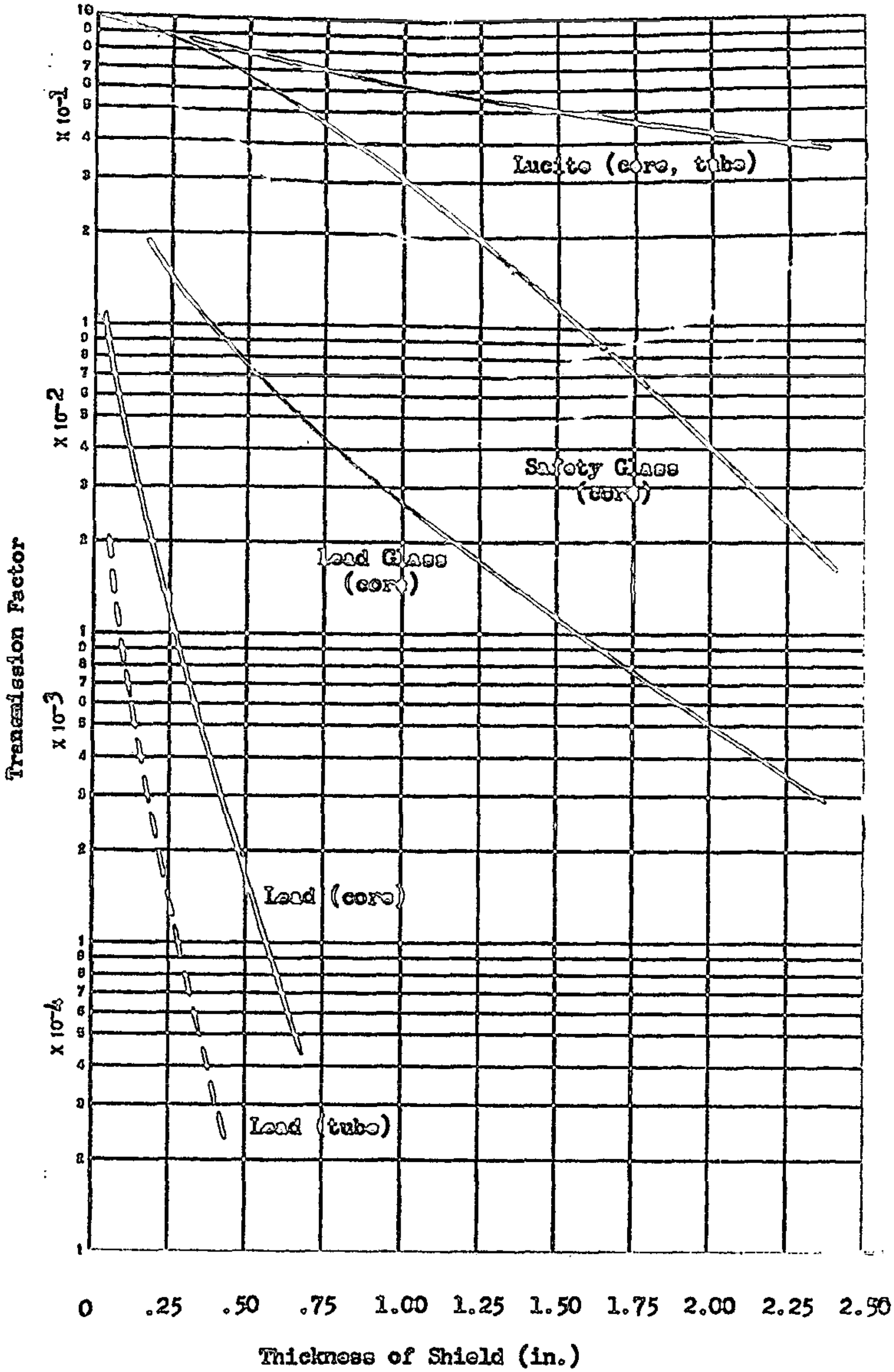

Figure 3

CASE III - Gamma Shielding Transmission Factors

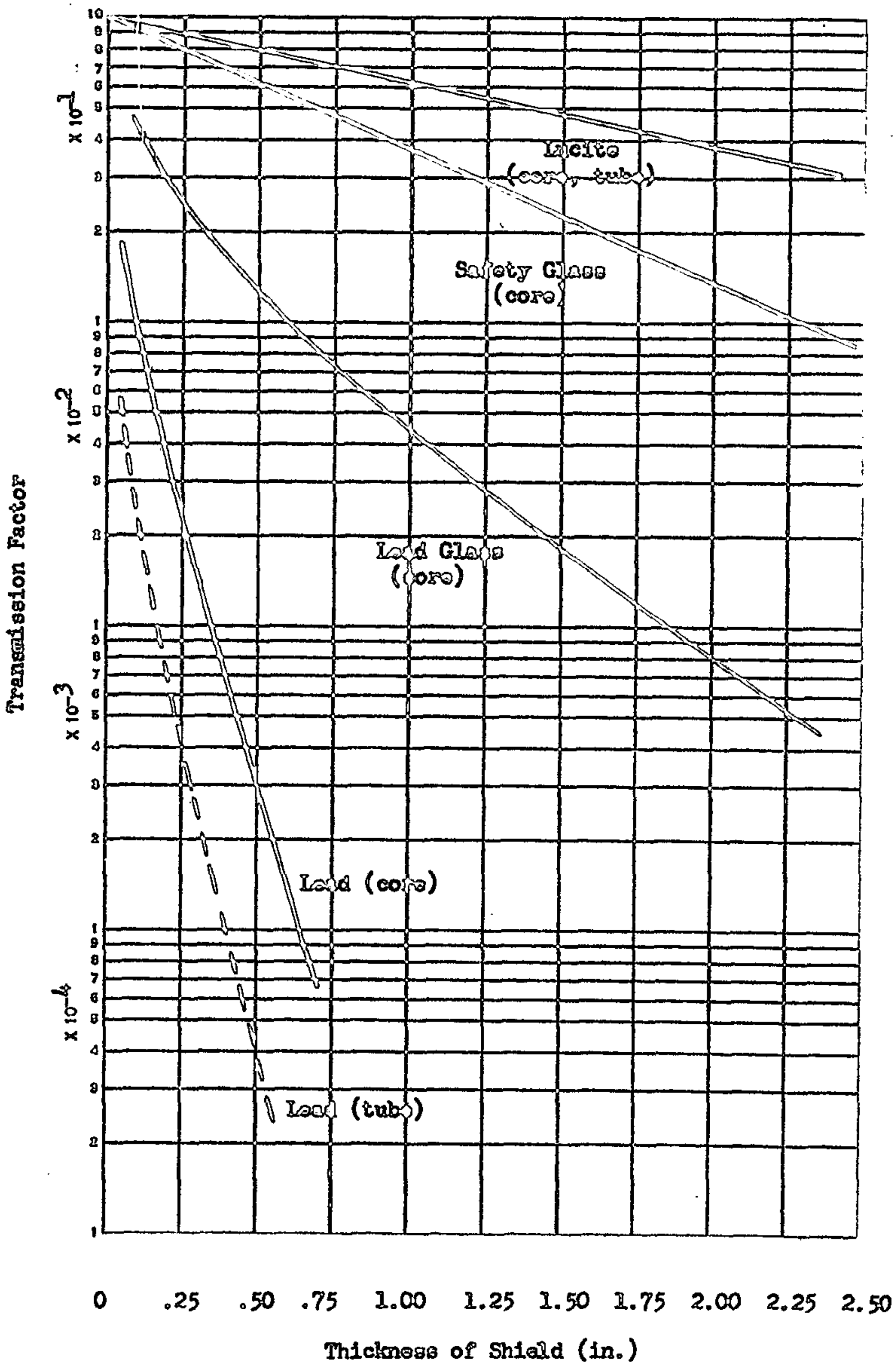

Figure 4 CASE IV - Gamma Shielding Transmission Factors

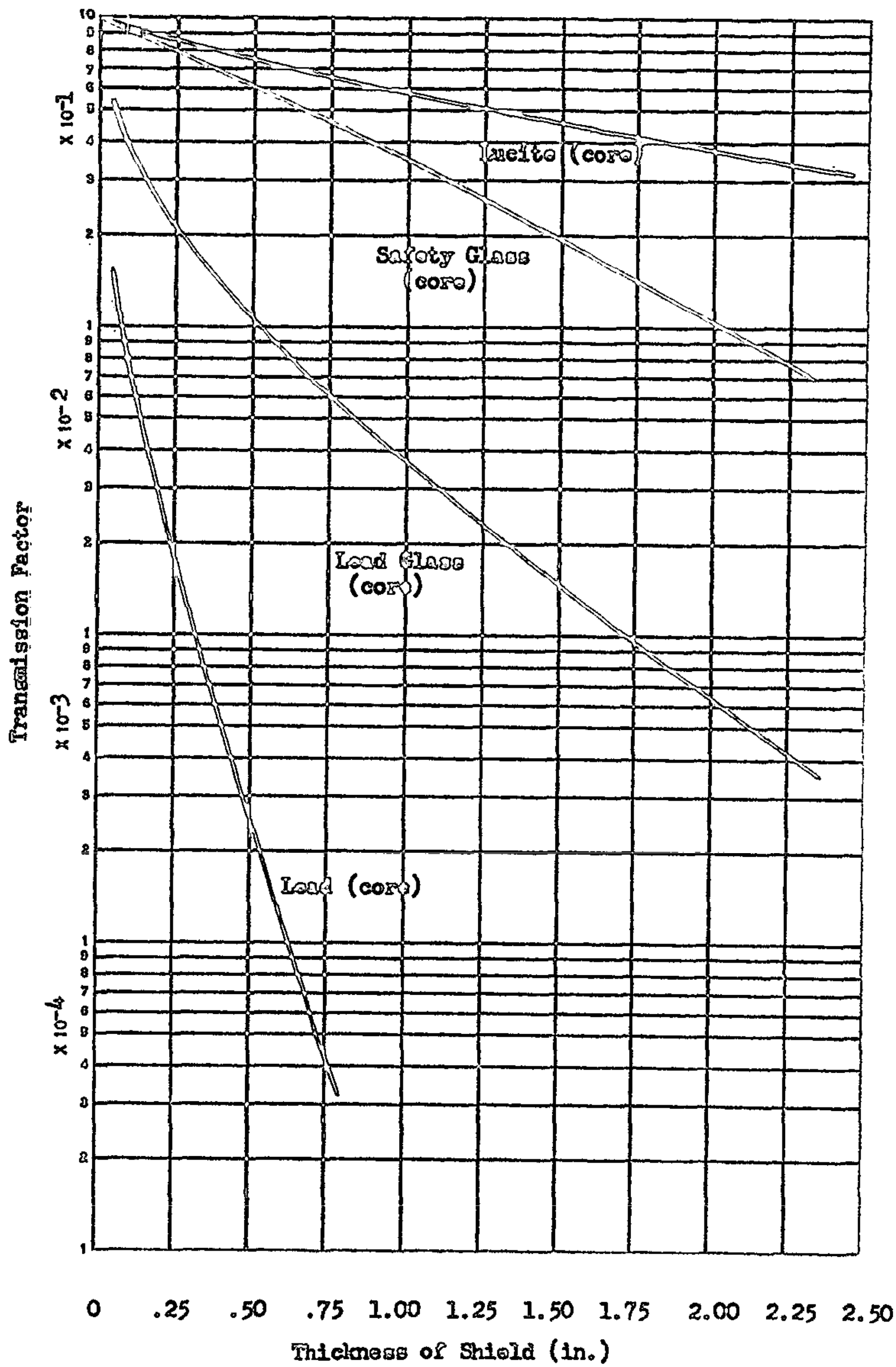

Figure 5

Neutron Shielding Transmission
Factors For Cases I, II, III & IV