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A PERSONNEL NEUTRON DOSIMETER USING LITHIUM FLUORIDE THERMOLUMINESCENT DOSIMETERS

Roger B. Falk

Abstract. A neutron dosimetry system using commercial lithium-fluoride ribbon thermoluminescent dosimeters has been developed. The system is designed to cover the range of neutron spectra found in plutonium glove-box operations. The physical size is approximately the same as a conventional neutron film packet ($1^5/_8$ by $1^5/_{32}$ by $^1/_8$ inches). The minimum dose measurable is 10 to 20 millirem in the presence of 100 millirem photon dose. The dose response is essentially independent up to 10 centimeters of moderator thickness.

INTRODUCTION

At Rocky Flats Plant, neutron dosimeters are required for personnel engaged in glove-box operations. Various thicknesses and configurations of neutron shielding used on glove boxes result in a variety of neutron spectra. Such spectra range from unmoderated to heavily moderated fluences from plutonium spontaneous fission and plutonium (α, n) reactions on various light elements. An adequate neutron dosimetry system should therefore yield a dose indication which is essentially independent of moderation over a range of moderator thicknesses for the various neutron sources.

A neutron dosimetry system, based on the use of lithium fluoride (LiF) thermoluminescent dosimeters (TLD), has been developed which covers the range of situations routinely encountered by personnel in glove-box operations with plutonium. Described are the theoretical considerations, calibrations, and results of field tests. Applications and limitations of the system are noted also.

DISCUSSION

Description: Since the thermoluminescence of LiF is based on the (α, n) reaction in the lithium-6 isotope, the LiF is more sensitive to intermediate and thermal neutrons than to fast neutrons. However, the sensitivity of the LiF to thermal neutrons can be decreased by shielding the LiF with cadmium, and the response to fast neutrons can be enhanced by moderating incident fast neutrons to lower energies. Appropriate cadmium shielding and use of a person's body as the fast-neutron moderator are incorporated in the design of the system.

The neutron dosimetry system consists of three pairs of LiF dosimeters shielded with cadmium (0.016 inches thick). One of the pairs is shielded from the front, one pair from the back, and the third pair is shielded both front and back. The system can be modified to two pairs of dosimeters by the elimination of the pair shielded front and back.

Each pair of dosimeters includes a TLD-600 and TLD-700 high sensitivity ribbon manufactured by the Harshaw Chemical Company, Cleveland, Ohio. Since the TLD-600 dosimeter (according to Harshaw) is about 95 percent lithium-6 and responds to both neutron and photon radiations, the TLD-700 dosimeter (99+ percent lithium-7) is used to subtract the photon response. The system is contained in a plastic holder the approximate size of a piece of neutron dosimetry film ($1^5/_8$ by $1^5/_{32}$ by $^1/_8$ inches).

Theory:

The general concepts of the system are:

1. The net response of the dosimeter shielded from the front with cadmium (designated *FCd*) results from *(a)* incident fast and intermediate neutrons; *(b)* thermal, intermediate, and fast neutrons moderated and reflected by the person's body; and *(c)* intermediate and fast neutrons thermalized and reflected by the person's body.

2. The net response of the dosimeter shielded from the back with cadmium *(BCd)* results from *(a)* incident fast, intermediate, and thermal neutrons; and *(b)* fast and intermediate neutrons moderated and reflected by the person's body.

3. The net response of the dosimeter shielded front and back with cadmium *(2Cd)* results from *(a)* incident fast and intermediate neutrons; and *(b)* moderated and reflected fast and intermediate neutrons.

The quantity $(BCd - 2Cd)$ is then the response to incident thermals. If this quantity is multiplied by some factor f, which would represent the fraction of incident thermal neutrons which are reflected by the body, the quantity $f(BCd - 2Cd)$ is a measure of the reflected thermals.

The quantity $[(FCd) - f(BCd - 2Cd)]$, then gives the response to incident fast and intermediate neutrons and to those fast and intermediate neutrons which are moderated and reflected by the person's body. When the system is calibrated, with C as the calibration factor in units of millirem (mrem) per net count, the neutron dose equivalent[1] is given by:

$$\text{Neutron Dose Equivalent} = C\,[(FCd) - f(Cd - 2Cd)] \quad (1)$$

The system can be reduced to a two-pair system by the elimination of the pair shielded front and back with cadmium $(2Cd)$. The equation to calculate the neutron dose equivalent for the two-pair system is:

$$\text{Neutron Dose Equivalent} = C\,[(FCd) - f(BCd)] \quad (2)$$

In Equation 2, the quantity $[(FCd) - f(BCd)]$ is the response to fast and intermediate neutrons thermalized and reflected by the body plus $(1 - f)$ times the incident and reflected fast and intermediate neutrons.

Calibration:

The approach in calibrating the system is to determine the response to the total spectrum for a range of spectra from unmoderated to heavily moderated; and to select the parameters so that the system response is independent, within stated limits, of the degree of moderation of the spectra. For practical applications, the system should be calibrated to the source and range of moderations which the system will encounter in routine operations.

At Rocky Flats, a plutonium fluoride (PuF_4) source was used for calibration of a range of 0 to 10 centimeters (cm) polyethylene moderation. The source consisted of 210 grams of PuF_4 enclosed in a steel sphere. A set of polyethylene shells of thicknesses 2.5, 4.0, 4.9, and 7.0 cm was used to moderate the spectrum. Data curves were extrapolated to a total moderator thickness of 10 cm. For all calibrations, the system holder was placed inside a plastic dosimetry badge in the cavity reserved for the neutron-film packet. The badge was taped to the chest of a water-filled, plastic phantom having lung equivalent material in the lung cavity. Dosimeter preparation consisted of annealing at 400 °C for one hour and then cooling rapidly to ambient temperature by spreading in a large aluminum tray. No low temperature annealing or pre-readout annealing was performed. Dosimeter readout occurred 16 to 24 hours after the end of the irradiation period. A gamma

TABLE I. Net counts per millirem versus moderator thickness for the three pairs of dosimeters.

Moderator Thickness (centimeters)	Shielded from Front with Cadmium (FCd)	Shielded Front and Back with Cadmium (2Cd) (counts per millirem)	Shielded from Back with Cadmium (BCd)
0	0.37	0.11	0.17
1	0.60	0.20	0.37
2	0.83	0.29	0.63
3	1.06	0.37	0.98
4	1.29	0.45	1.40
5	1.52	0.54	1.88
6	1.75	0.63	2.48
7	1.98	0.72	3.17
8	2.21	0.80	3.93
9	2.45	0.89	4.80
10	2.68	0.98	5.75

calibration indicated that the gamma sensitivity of the TLD-600 and TLD-700 dosimeters agreed within 5 percent. Thus the response of the TLD-700 dosimeter could be subtracted directly from the response of the TLD-600 dosimeter to yield the neutron response of the TLD-600 dosimeter.

The first step in the calibration was to expose the TLD system frontally to the source with the set of moderators to obtain the neutron response in terms of net counts per mrem versus moderator thickness for each paired dosimeter set. The results are shown in Figure 1. From this plot, counts per mrem information can be obtained for increments of one-centimeter moderation, extrapolated to a total of 10-cm moderator thickness. The information is given in Table I.

The information in Table I is used to determine the values of C and f, which yield a system response most nearly independent of moderation over the optimum range of moderation to obtain the system counts per mrem versus moderation for various values of f. A family of curves were computed from the expression:

$$\frac{\text{System Counts}}{\text{mrem}} = [(FCd) - f(BCd - 2Cd)]$$

The family of curves noted in Figure 2 is used to determine the range for which the curves are independent of moderation, within specified limits. The ranges for which selected curves are independent of moderation within ±10 and ±20 percent, with the corresponding value of C, are given in Table II.

The choice of the curve and the range depends upon the circumstances in which the system would be used as a neutron dosimeter.

[1] The product of absorbed dose and modifying factors is termed the *dose equivalent* (DE). The unit of dose equivalent, the rem, is numerically equal to the dose in rads multiplied by the appropriate modifying factors.

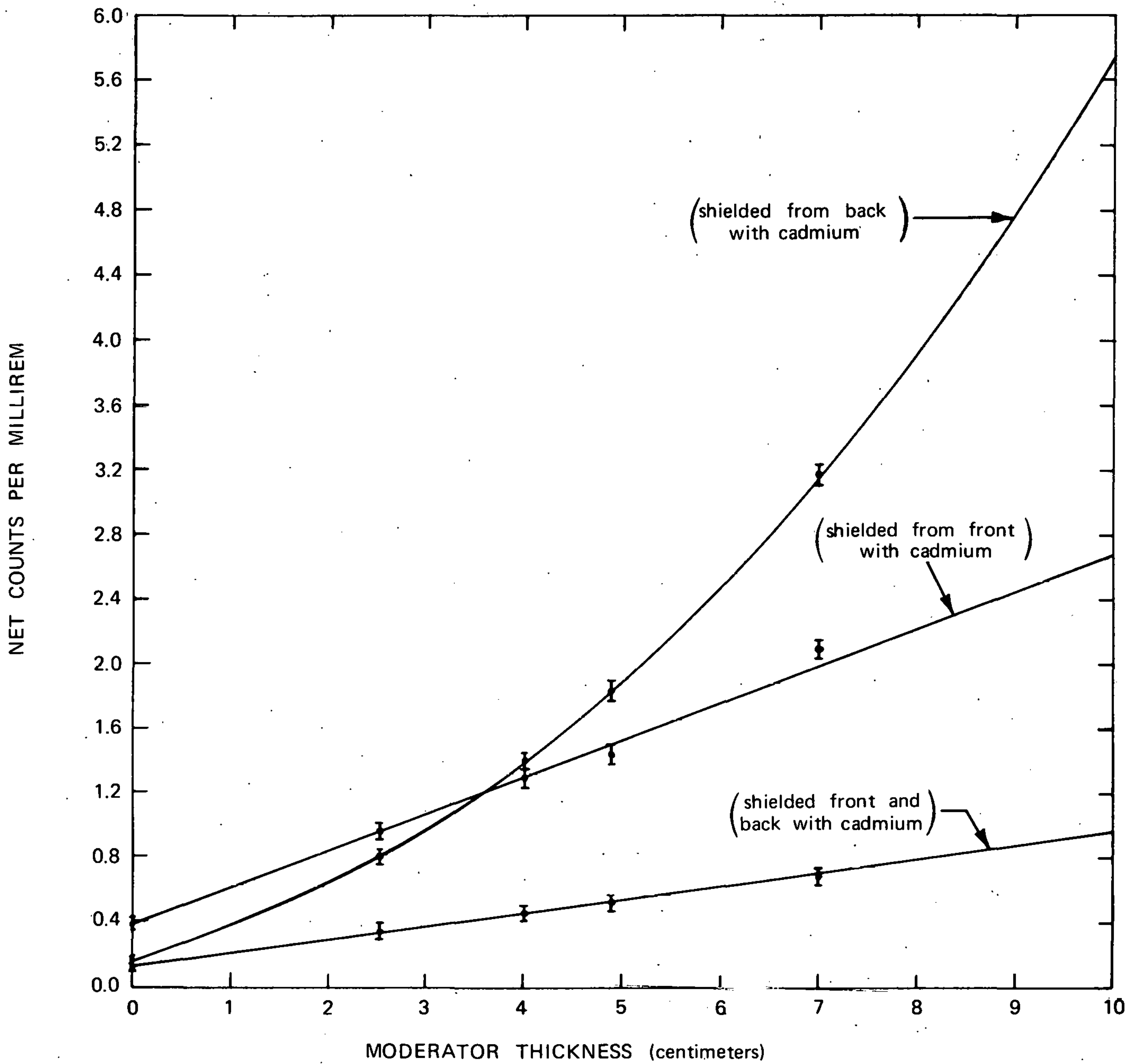


FIGURE 1. Net neutron response of the three sets of dosimeters as a function of moderation for a plutonium fluoride source.

The data in Tables I and II were obtained from the phantom exposed frontally to the neutron source. Since most practical situations include neutrons incident at other angles, the response of the system was determined for neutron incident at angles other than normal. Calibration curves (counts per mrem versus cm moderation) could then be obtained for neutrons incident on the system for various distributions of angles of incidence. Two cases considered were: (1) for neutrons incident equally at angles 0 to 90° from normal, and (2) for neutrons incident as if from an extended line source. These cases are illustrated in Figure 3.

The response (counts per mrem) of each of the three sets of dosimeters for angles of incidence of 0°, 45°, 60°, 75°, and 90° are given in Figures 4, 5, and 6. Values from these plots were used to obtain a family of curves of counts per mrem versus angle for increments of moderation for each of the sets of dosimeters (Figures 7, 8, 9). Values of

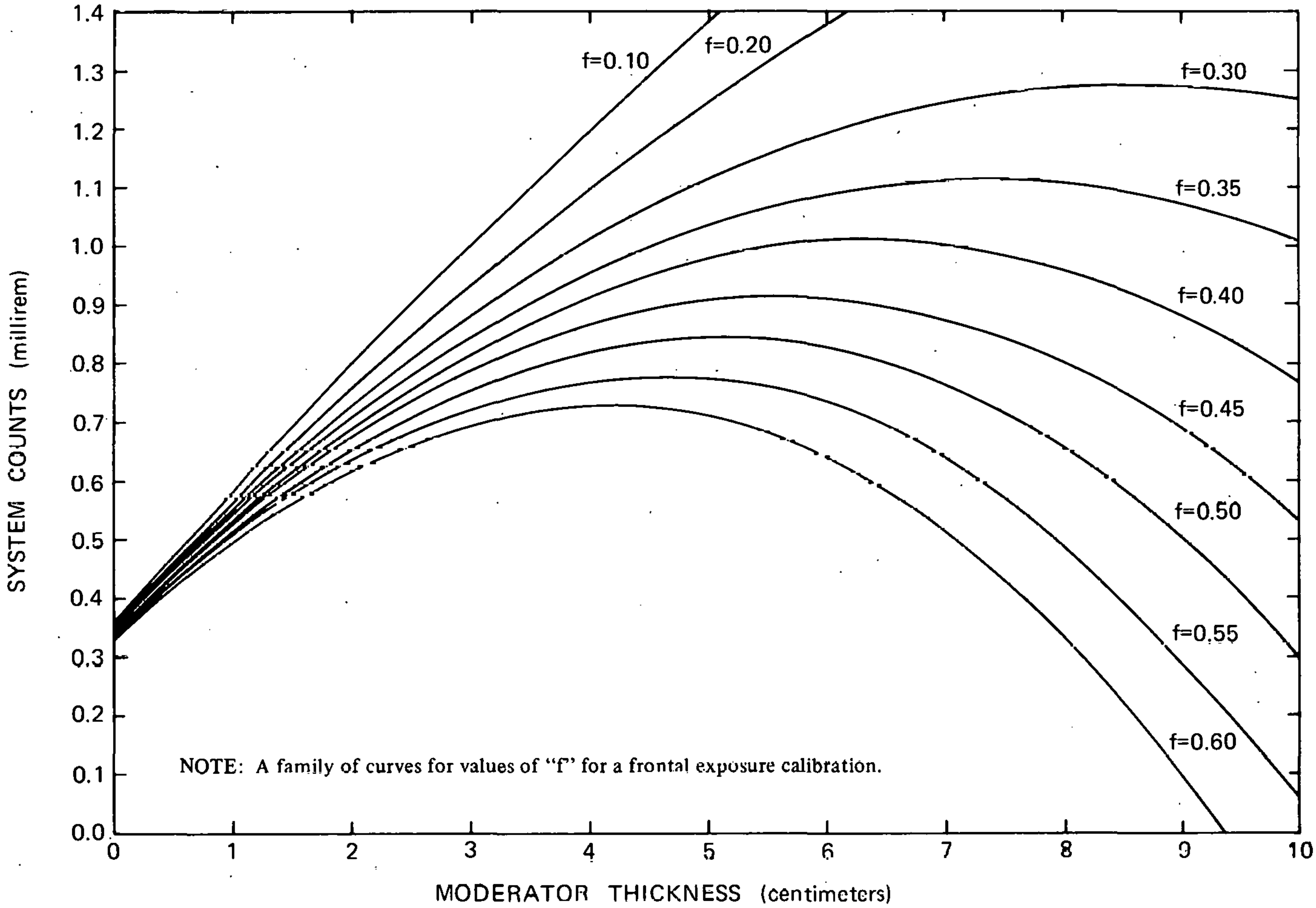


FIGURE 2. The neutron response for a three-pair system.

counts per mrem for increments of 5 degrees of angle, obtained from these curves, were then weighted according to the assumption of the exposure distribution as a function of the angle of incidence. When the weighting factor (the fraction of the total dose per angle increment) is multiplied by corresponding increments of each curve in the family of curves in Figures 7, 8, and 9, the sum of the incremental results for each of the curves is the adjusted value of the counts per mrem for that thickness of moderation for that distribution. The adjusted values for the equal angle and extended line-neutron incidence distributions are given in Table III.

Values in Tables I and III can be used to obtain families of curves of counts per mrem versus moderation for values of *f* in the same manner as for the frontal incidence calibration, for the two-pair and the three-pair systems. A comparison of curves selected from each of the families of curves is given in Figure 10. The response of these curves relative to 0-cm moderation, which can be used as an indicator of the relative shape of the curves, is given in Table IV.

The significance of the similarities noted in Table IV is that the value of the dose calculated from Equation 1 or 2 is independent of the distribution of the neutron incidence, within the range of the three cases for each pair system.

The ratio of the net counts of the dosimeter shielded from the back with cadmium divided by the net counts of the dosimeter shielded from the front with cadmium *(BCd/FCd)* can be used as an index of the degree of moderation. Figure 11 shows the *BCd/FCd* ratio versus moderation for frontal, equal angle, and line neutron-incidence distributions. A *BCd/FCd* ratio of less than 0.75 indicates a moderation of from 0 to 2 cm or for mixtures of moderations not greater than 2 cm. This feature can be utilized to identify unmoderated neutron fluences and then to use an appropriate value of *C* to calculate the neutron dose for those situations.

Performance information for the two-pair and three-pair systems, based on calibration data adjusted for the extended line-source distribution, is given in Table V. Performance information for the frontal and equal angle incidence data is similar

The minimum neutron dose which the system can measure is limited mainly by the magnitude of the error in the readout of each dosimeter, which depends in part on the magnitude of the photon response of each dosimeter. The magnitude of the error in the neutron dose indication caused by readout error was calculated for various combinations of assumed neutron and gamma doses.

Data are given in Figure 12 for the two-pair system. The readout error was assumed to be ±5 percent for dosimeter readings ≥ 100 units. For dosimeter readings less than 100 units, where R is the dosimeter reading, the function representing the readout error is:

$$\delta R = 0.803\ R^{0.397}$$

The instrument is assumed to be calibrated so that one unit reading corresponds to an accumulated gamma dose of one mrem. The results indicate that a neutron dose of 10 mrem can be measured with an error of ±7 mrem in the presence of 100 mrem gamma dose and 20 ±14 mrem in the presence of 200 mrem gamma dose. These errors are in addition to the ±ranges of dose indication given in Table V.

Routine exposures are likely to consist of various mixtures of unmoderated and moderated fluences, especially in a glove-box situation where thick shielding can be voided by glove ports. The effect of mixtures of moderation on the response of the system is shown in Figure 13.

TABLE II. Ranges in which the system response is independent of moderation.

*f	Range (±10 percent) (centimeters)	Calibration Factor (C) **(mrem/per net count)	Range (±20 percent) (centimeters)	Calibration Factor (C) (mrem/net count)
0.30	4.35 to > 10	0.86	2.80 to > 10	0.94
0.35	3.60 to > 10	0.98	2.25 to > 10	1.07
0.40	3.10 to 9.50	1.09	1.90 to > 10	1.19
0.45	2.60 to 8.50	1.20	1.55 to 9.50	1.30
0.50	2.35 to 7.65	1.29	1.30 to 8.60	1.41
0.55	1.95 to 7.00	1.41	1.05 to 7.80	1.53
0.60	1.70 to 6.45	1.53	0.90 to 7.20	1.67

*f = Arbitrary calibration factor selected to give optimum system response.
**C = Calibration factor in units of millirem per net count.

TABLE III. Adjusted values of counts per millirem versus moderation for neutrons.

Moderator Thickness (centimeters)	*Equal Angle Distribution (adjusted counts per mrem) FCd	2Cd	BCd	**Extended Line Distribution (adjusted counts per mrem) FCd	2Cd	BCd
0	0.32	0.08	0.15	0.36	0.10	0.16
1	0.51	0.15	0.32	0.58	0.18	0.36
2	0.71	0.22	0.51	0.80	0.26	0.59
3	0.90	0.28	0.74	1.02	0.33	0.87
4	1.10	0.34	0.99	1.24	0.40	1.20
5	1.29	0.41	1.32	1.46	0.48	1.62
6	1.49	0.47	1.74	1.68	0.55	2.13
7	1.68	0.57	2.22	1.90	0.63	2.73
8	1.88	0.64	2.76	2.12	0.70	3.39
9	2.07	0.71	3.36	2.34	0.78	4.13
10	2.27	0.77	4.03	2.56	0.86	4.95

* Incident equally from all angles from 0° to 90° of normal.
** Incident as if from an extended line source.

TABLE IV. System response relative to zero moderation for selected curves.

Moderator Thickness (centimeters)	Three-Pair System: Frontal (f = 0.43)	Equal Angle (f = 0.55)	Line (f = 0.48)	Two-Pair System: Frontal (f = 0.35)	Equal Angle (f = 0.45)	Line (f = 0.39)
0	1.00	1.00	1.00	1.00	1.00	1.00
1	1.56	1.50	1.48	1.52	1.52	1.50
2	2.00	1.96	1.94	1.97	1.96	1.93
3	2.35	2.32	2.30	2.32	2.32	2.30
4	2.59	2.64	2.61	2.58	2.60	2.57
5	2.76	2.82	2.76	2.77	2.80	2.77
6	2.79	2.82	2.79	2.84	2.84	2.83
7	2.74	2.75	2.70	2.81	2.76	2.80
8	2.53	2.54	2.52	2.68	2.56	2.67
9	2.26	2.18	2.21	2.48	2.24	2.43
10	1.85	1.71	1.82	2.16	1.84	2.10

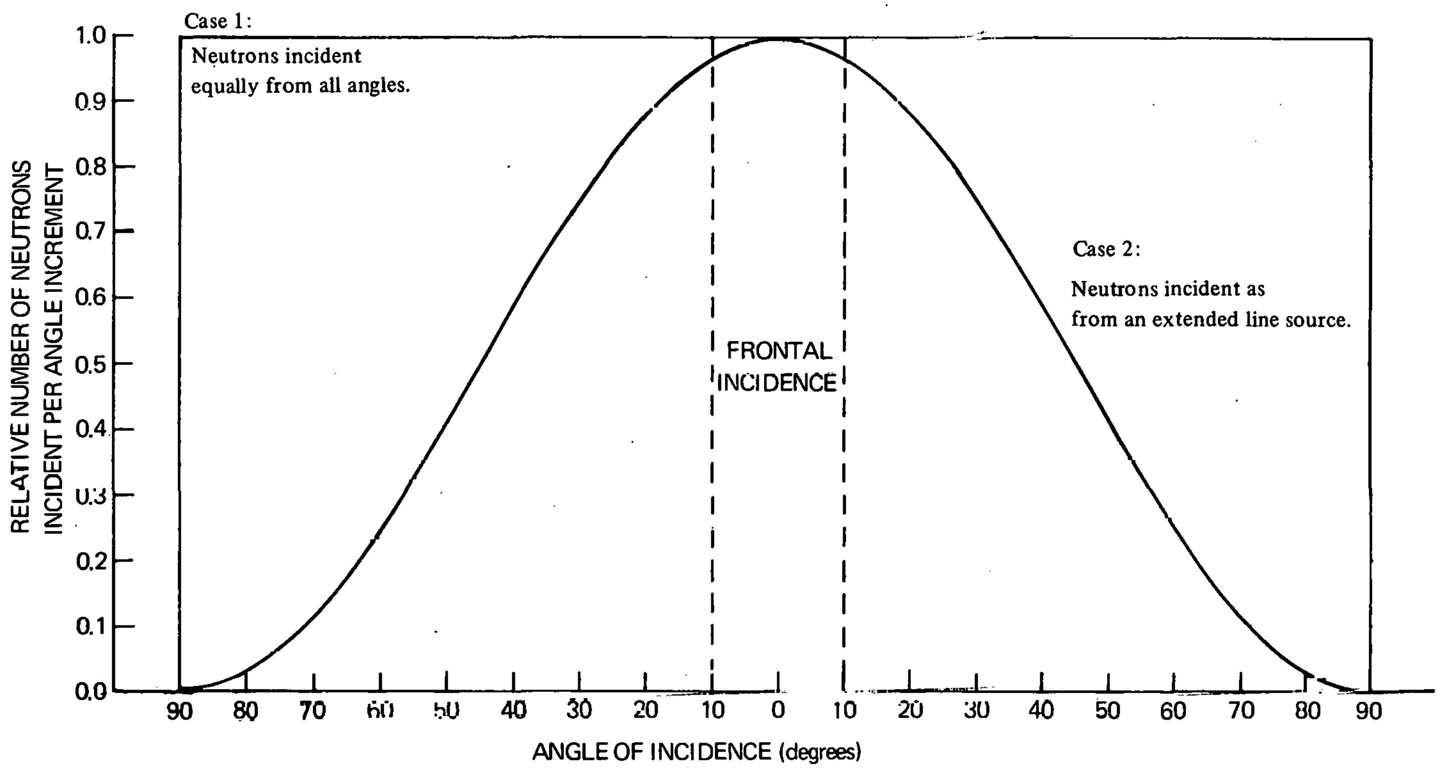
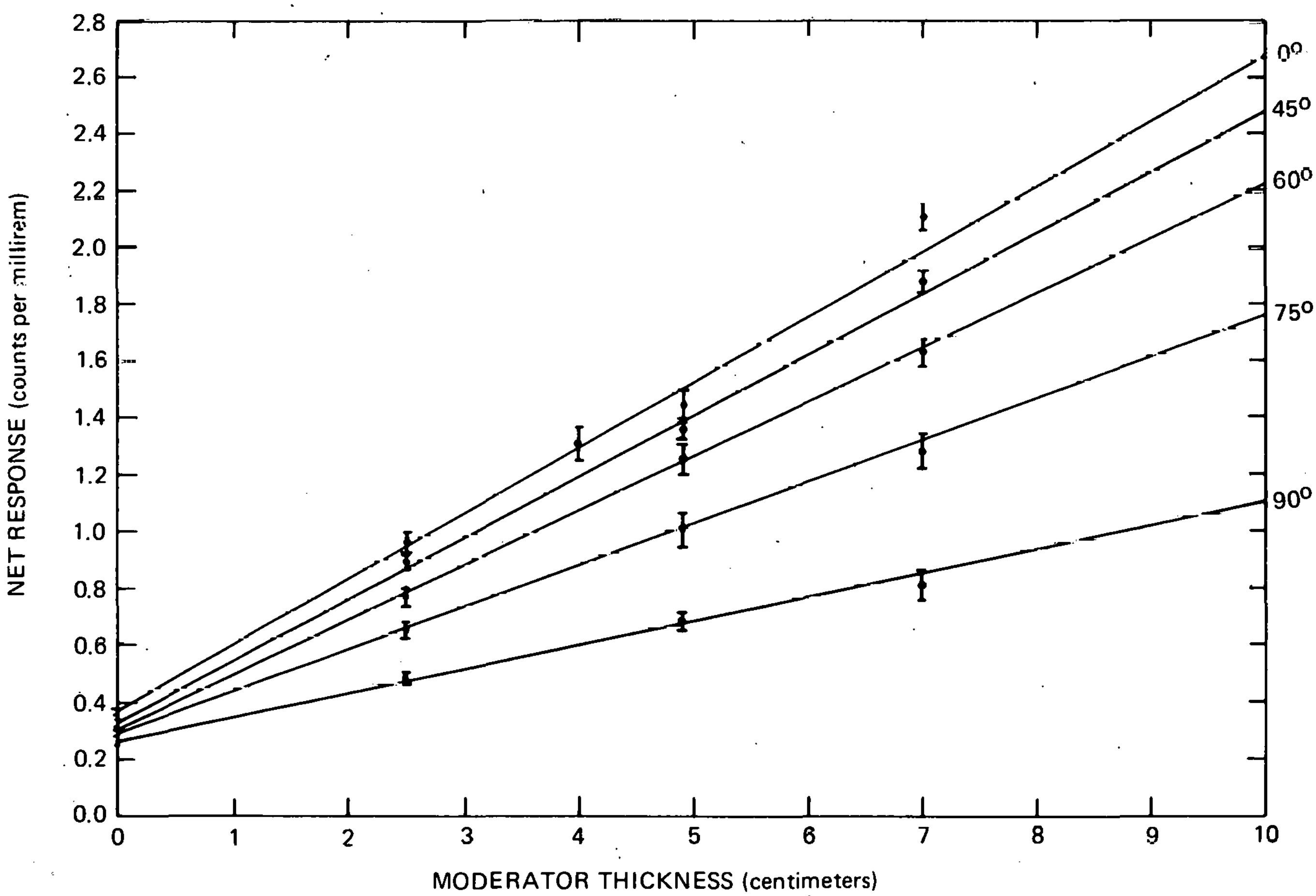


FIGURE 3. Distributions of Angle of Incidence.

FIGURE 4. Net response of dosimeter pair shielded from the front with cadmium for angles of neutron incidence.



Dose values were calculated, using the data adjusted for the extended line-source distribution, for 1:1 mixtures of moderations for values of f = 0.39 and C = 1.29 for the two-pair system. A given dose of 100 mrem was assumed. The calculated TLD system dose is plotted in modules of the mixtures of moderations. First mixtures of 0-cm moderation with all others are plotted, then mixtures of 1-cm moderation with all others, etc. If other values of *C* are used, results are obtained which are the multiple *C/1.29* of the values given in Figure 13. Plotted also are the results if a value of C = 2.58 is used for all *BCd/FCd* indexes ≤0.75. The system TLD dose can be graphically obtained from this plot for any percentage mixture of any two moderations and is within the range of the curves presented here.

Field Tests:

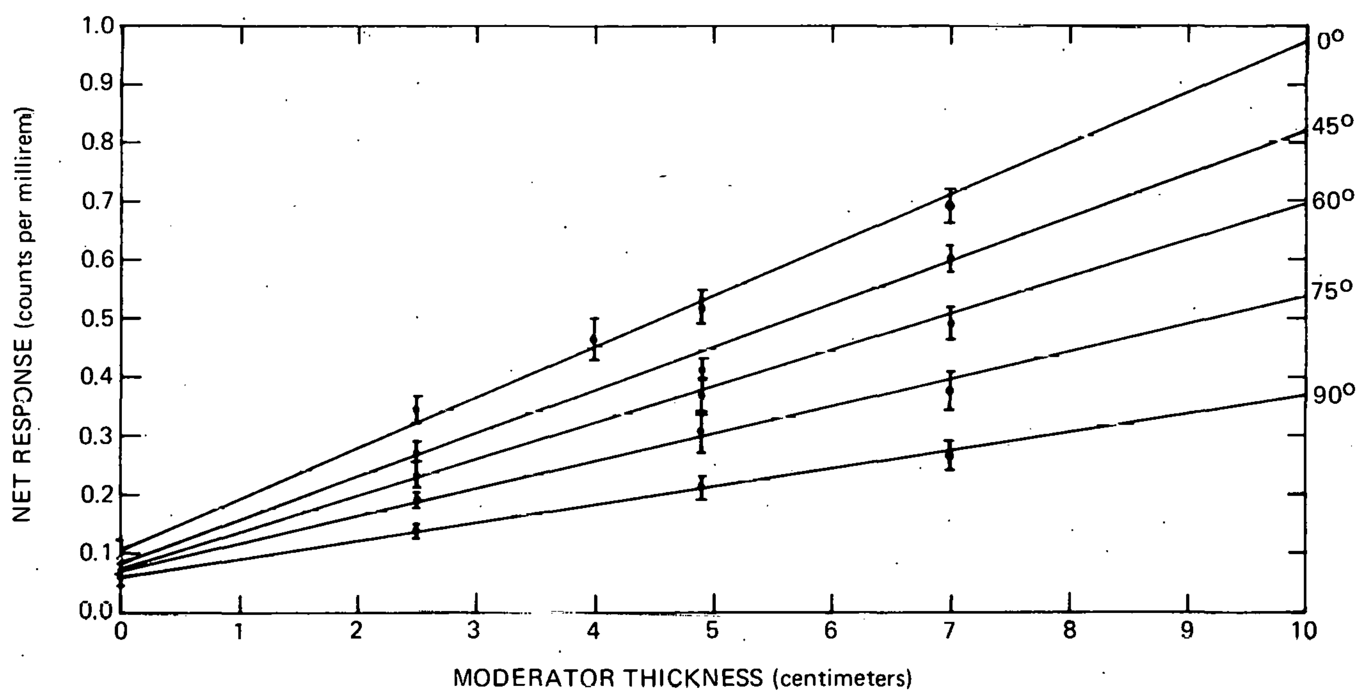
Field tests were conducted to determine the performance of the system for the range of neutron fluences and spectra routinely encountered in the various glove-box and material-handling situations. For each test, two dosimetry badges were placed on the chest of the phantom, stationed one to two feet from the surface of the glove box or in a

TABLE V. Performance information for the two-pair and three-pair systems for a plutonium fluoride calibration in a line-source distribution.

	Two-Pair System, f = 0.39		Three-Pair System, f = 0.48	
Range of Dose Indication (percent)	Range of Moderation (centimeters)	*C	Range of Moderation (centimeters)	*C
±10	3.05 to 9.40	1.29	2.90 to 8.85	1.20
+10 to -20	2.30 to > 10	1.29	2.20 to 9.50	1.20
+10 to -30	1.65 to > 10	1.29	1.60 to > 10	1.20
+10 to -40	1.05 to > 10	1.29	1.00 to > 10	1.20
+10 to -50	0.60 to > 10	1.29	0.55 to > 10	1.20
±20	1.90 to > 10	1.41	1.80 to 9.90	1.30
+20 to -30	1.35 to > 10	1.41	1.25 to > 10	1.30
+20 to -40	0.85 to > 10	1.41	0.80 to > 10	1.30
+20 to -50	0.40 to > 10	1.41	0.35 to > 10	1.30
±30	1.05 to > 10	1.53	1.00 to > 10	1.41
+30 to -40	0.60 to > 10	1.53	0.60 to > 10	1.41
+30 to -50	0.20 to > 10	1.53	0.20 to > 10	1.41

**C* = Calibration factor in units of millirem per net count.

FIGURE 5. Net response of dosimeter pair shielded front and back with cadmium for angles of neutron incidence.



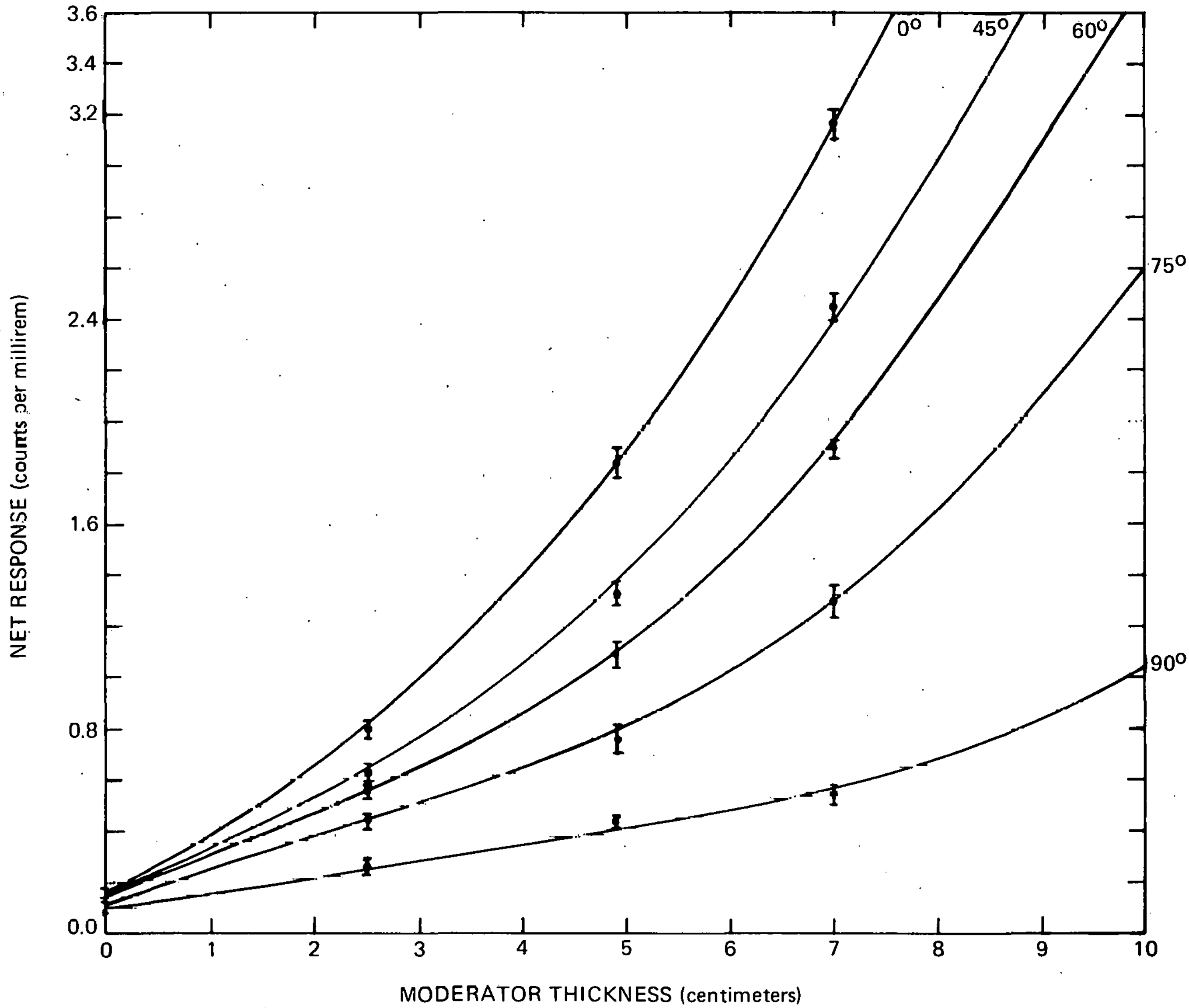


FIGURE 6. Net response of dosimeter pair shielded from the back with cadmium for angles of neutron incidence.

general work or material-storage area. The exposure time ranged from a couple of hours to several days depending upon the dose rate and the desired accumulated dose. Dose rate measurements, obtained with a Ludlum 10-inch sphere system for neutron dose rates and a Victoreen-440 survey meter for gamma dose rates, were made at the start and at the end of each exposure period. The measured dose rated multiplied by the exposure time yielded an apparent dose range which may or may not reflect fluctuations in the actual dose rates caused by passage of material through the processing lines. An integrating dose meter was not available for this study.

Results of the study are presented in Table VI. The TLD dose indications are presented for both the two-pair and the three-pair systems, using the parameters $f = 0.39$ and $C = 1.29$ for the two-pair system; and $f = 0.48$ and $C = 1.20$ for the three-pair system. These parameters were used to yield a ±10 percent dose indication range for moderation ranges of 3.05 to 9.40 cm and 2.90 to 8.85 cm, respectively, as indicated in Table V.

Gamma TLD dose results are also presented along with a brief description of the source and shielding conditions. Figure 4 gives a graphical comparison of the TLD dose indications with the apparent survey dose range for the data presented in Table VI for the two-pair TLD neutron system. This comparison indicates that the survey dose range is consistently within 20 percent of the TLD neutron dose indication. The significance of the results of the

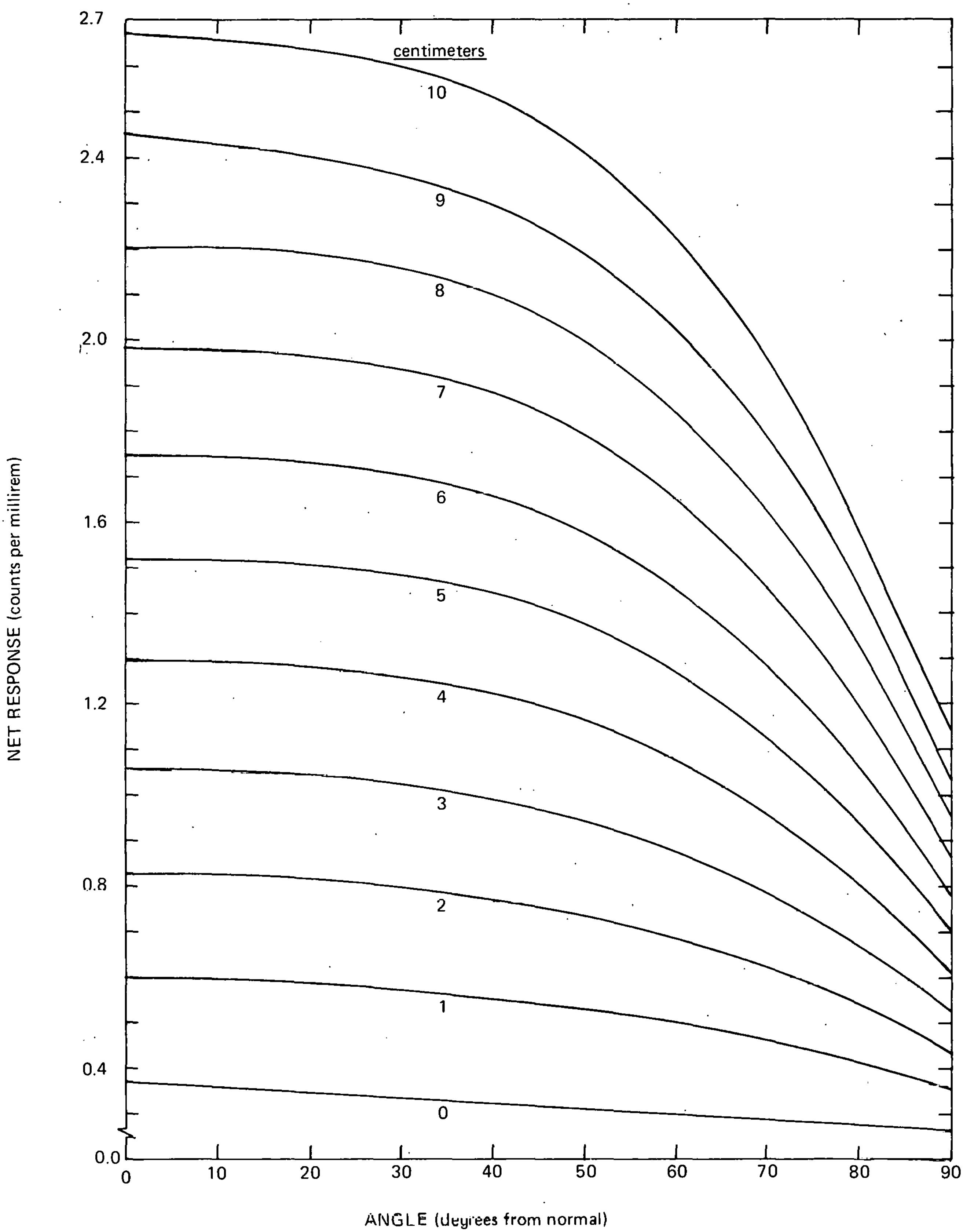


FIGURE 7. Net response as a function of angle of neutron incidence for dosimeter pair shielded from the front with cadmium.

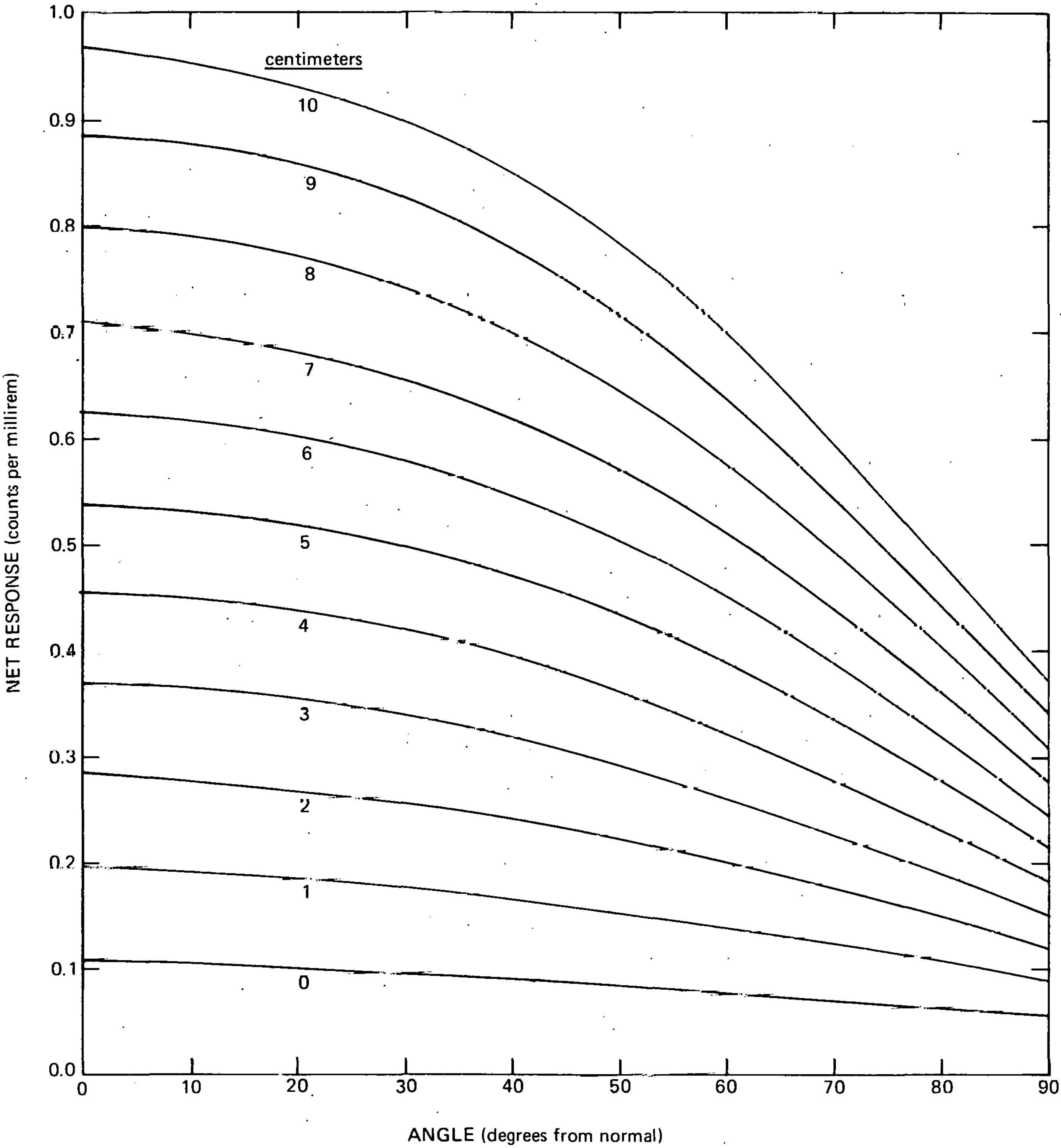


FIGURE 8. Net response as a function of angle of neutron incidence for dosimeter pair shielded front and back with cadmium.

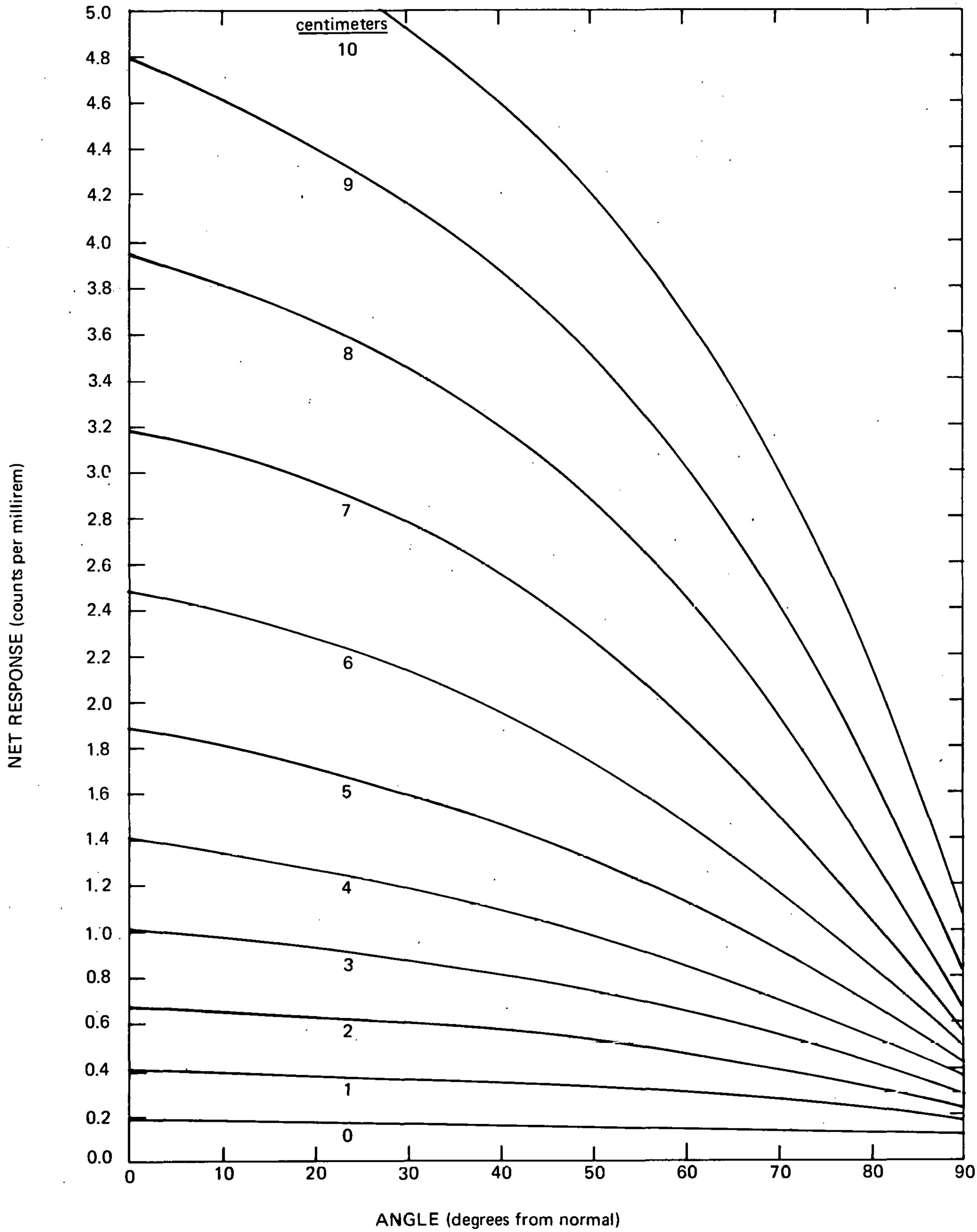


FIGURE 9. Net response as a function of angle of neutron incidence for the dosimeter pair shielded from the back with cadmium.

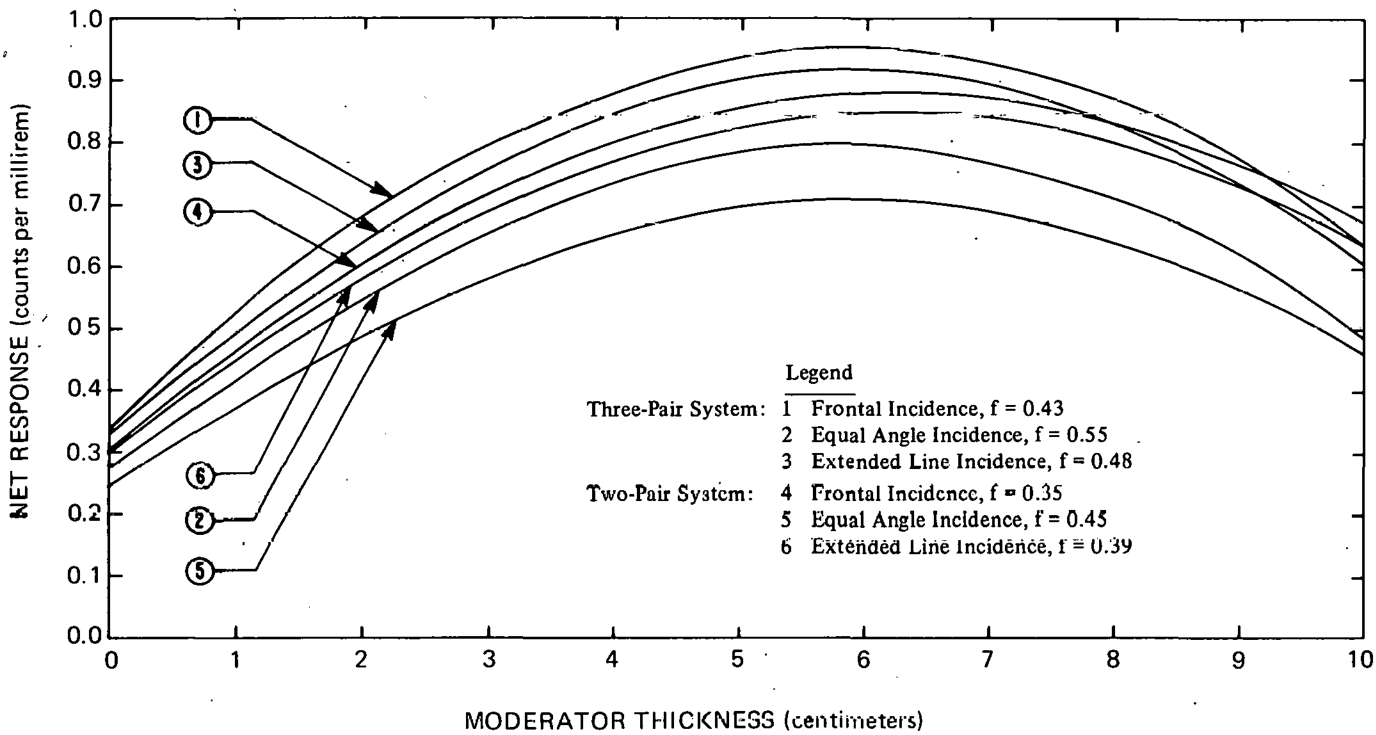
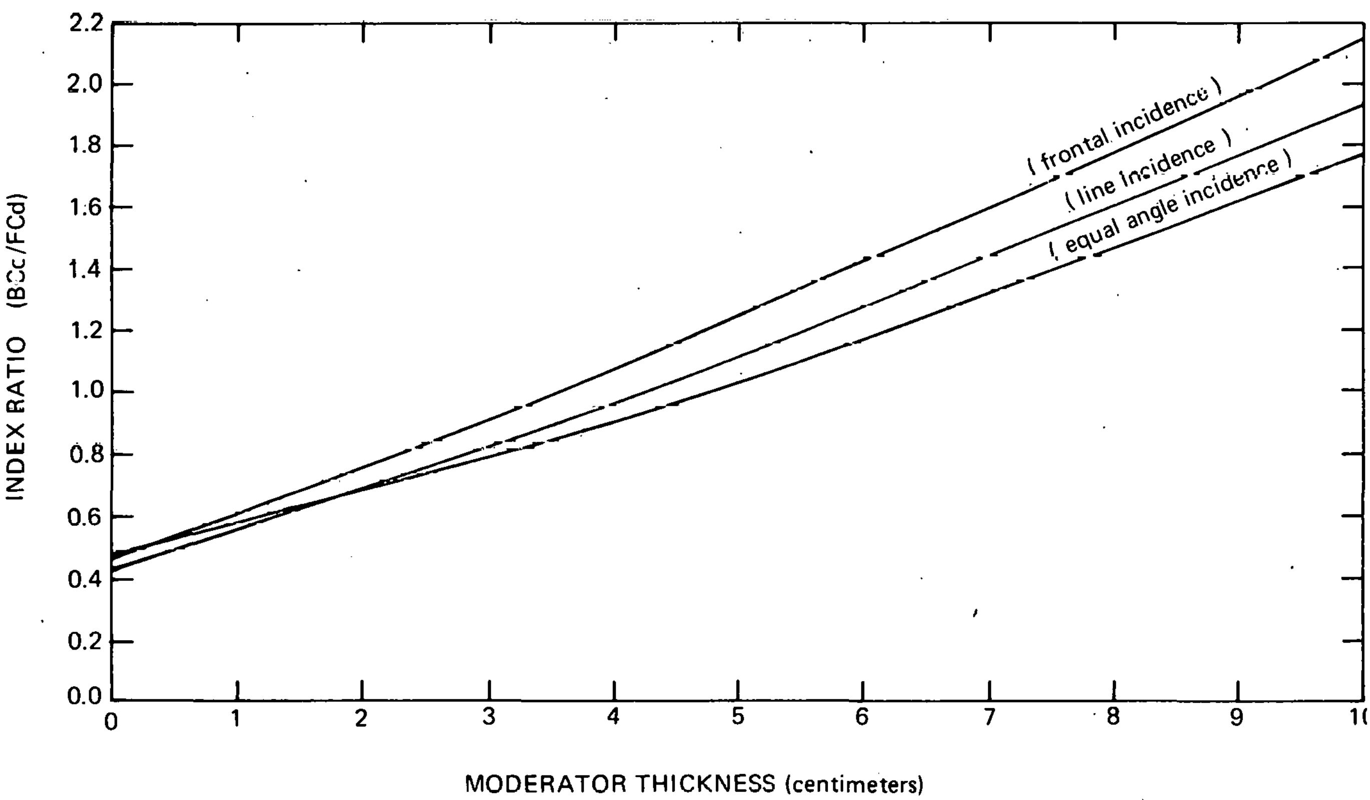


FIGURE 10. Selected curves for the two-pair and three-pair thermoluminescent dosimeter neutron systems.

FIGURE 11. Back to front index ratio (BCd/FCd) as a function of moderator thickness.



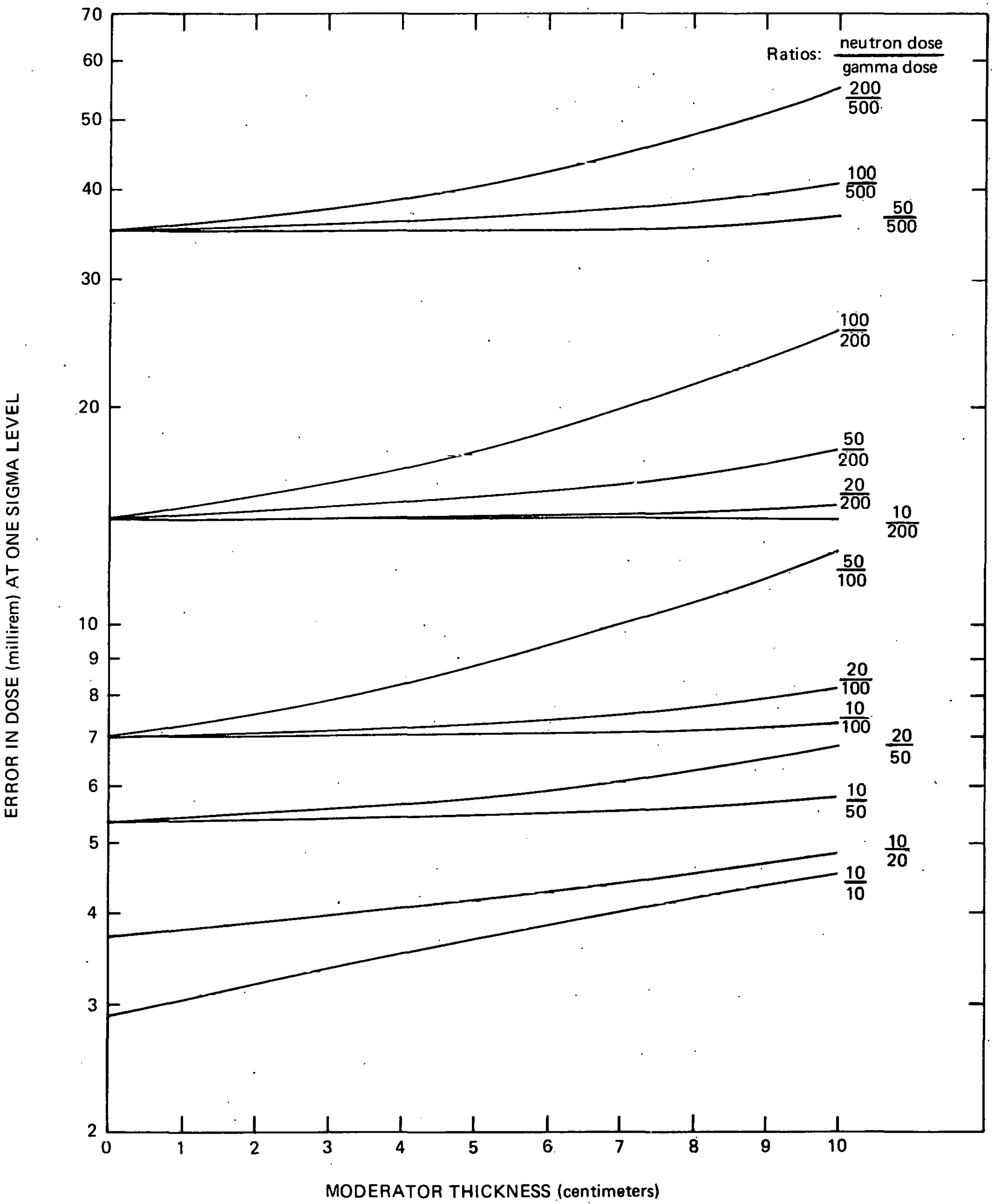


FIGURE 12. Calculated error in thermoluminescent neutron dose indication for combinations of neutron and gamma doses.

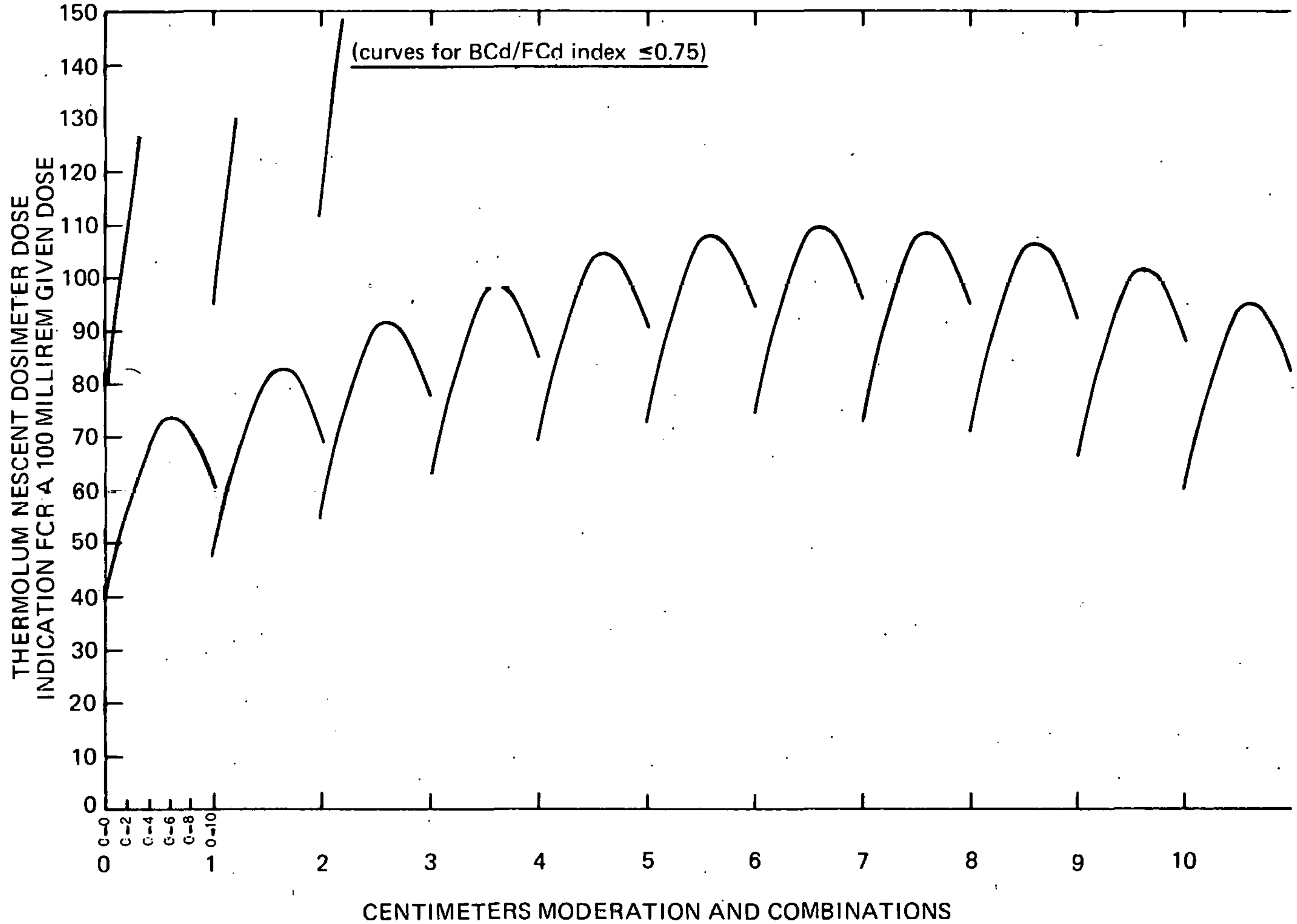


FIGURE 13. Calculated thermoluminescent dosimeter system for a given dose of 100 millirem for one-to-one (1:1) mixtures of moderations.

field tests is that the TLD neutron system for both two-pairs and three-pairs of dosimeters performed adequately under the wide range of situations, encountered in plutonium processing operations.

CONCLUSIONS

The personnel neutron dosimetry system, based on LiF thermoluminescent dosimeters, which has been developed, adequately covers the range of neutron exposure situations routinely encountered in plutonium-processing operations. The system holder is the approximate size of a neutron film packet and contains either two or three pairs of TLD-600 and TLD-700 Harshaw LiF dosimeters. Since no significant difference exists in the response between the two-pair and the three-pair system, the two-pair system appears the practical choice for routine applications. A practical lower neutron dose limit of 10 to 20 mrem can be measured in the presence of up to 100 mrem photon dose.

The system response can be made independent of the distribution of neutron incidence through a wide range of possible distributions. With the appropriate choice of values of *f* and *C* in Equation 1 or 2 and the use of the *BCd/FCd* index to identify the system response to unmoderated neutron fluences, the system response can be made essentially independent of the amount of moderation for various ranges of moderator thicknesses.

The data presented here indicate that the system can be used as a personnel neutron dosimeter for neutron spectra from plutonium fission and from the plutonium *(a, n)* reaction on oxygen, calcium, and fluorine for moderations up to the equivalent of ten centimeters of polyethylene.

Use as a personnel neutron dosimeter in other neutron exposure situations requires a calibration to the total spectrum from that particular neutron source over the range of moderations which would be encountered.

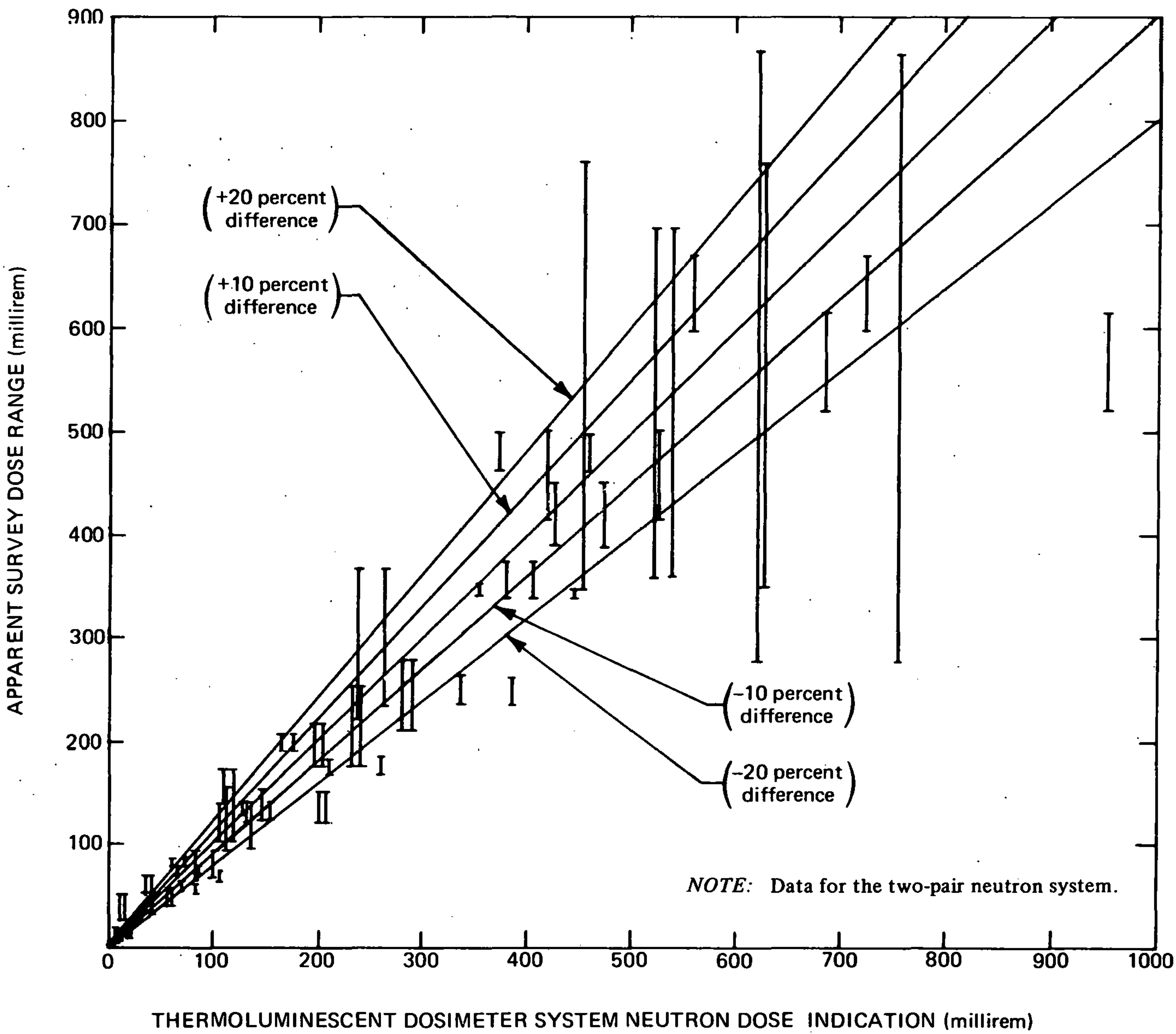


FIGURE 14. Comparison of thermoluminescent dosimeter dose indication with the apparent survey dose range for phantom exposures simulating routine working conditions.

TABLE VI. Results of field tests for the thermoluminescent dosimeter neutron system.

Description of Exposure	Survey Dose Range (mrem)		Thermoluminescent Dosimeter Dose Indication (mrem)		
			Neutron		
	Neutron	Gamma	Two-Pair	Three-Pair	Gamma
Plutonium fluoride processing	93-129	22-23	113	99	20
Four-inch Lucite shielding			137	119	23
Port shield closed	188-204	28	176	158	27
			164	159	28
Plutonium fluoride processing	116-150	12	207	200	13
Four-inch Lucite shielding			202	194	14
Port shield open	236-262	21-25	335	321	25
			383	361	26
	415-500	36-40	419	430	37
			523	508	37
	64-93	8-10	99	91	13
			82	78	11
	61-73	11-12	106	98	14
			107	103	14
	277-865	48-87	754	759	60
			622	638	66
Plutonium calcine processing	348-760	63	453	479	68
Four-inch Lucite and Benelex shielding			626	608	74
Port shield open	52-58	17-18	67	71	22
			82	80	21
	388-450	69	425	430	62
			473	483	63
	232-365	52-65	262	250	57
			237	221	58
Plutonium metal processing	100-139	39-44	117	124	53
No neutron shielding			104	105	49
Plutonium nitrate processing	52-68	33-37	33	34	36
No neutron shielding			39	38	34
	9-14	82-97	16	18	149
			17	18	176
Americium processing	8-12	49-65	10	11	109
No neutron shielding			10	10	117
	73	792	64	48	1029
			84	66	1058
	23-50	460-479	10	13	431
			14	8	473
	6-13	101-158	4	5	171
			10	10	177
	36-50	275-1976	40	44	457
			10	10	526

(continued) →

Description of Exposure	Survey Dose Range (mrem)		Thermoluminescent Dosimeter Dose Indication (mrem)		
			Neutron		
	Neutton	Gamma	Two-Pair	Three-Pair	Gamma
Plutonium calcine storage line	122-140	22-25	130	129	22
Two- to four-inch Lucite and Benelex shielding			153	146	23
Port shield open	126-172	40-42	117	118	45
			109	99	47
General background between U-shaped lines	358-697	137-150	523	500	128
Fluoride, calcine, and green cake			539	553	123
Two- to four-inch Lucite and Benelex shielding	38-54	13-14	57	54	15
Phantom, 8 to 10 feet from lines in			56	55	17
U-shape	120-153	35-40	146	140	33
			177	118	33
Storage Area Exposures – Canisters in arrays along walls, shielded with two-inch Lucite:					
General room background	2490-3142	391-456	3774	3600	406
Three to five feet from canister rows			4061	3880	391
Primarily edge exposures					
Plutonium fluoride storage	341-349	35-36	355	329	41
Phantom, two feet from array facing canisters			446	431	40
	208-278	24-26	290	263	28
			280	259	29
L-shaped fluoride storage area	597-669	56-58	558	544	47
Front and edge exposures			724	704	48
	460-497	43	459	434	37
			372	388	35
Plutonium oxide storage	339-372	32	380	356	32
Phantom facing array of canisters			405	391	34
	126-139	15-16	137	130	16
			127	116	16
Plutonium metal storage	518-614	98-102	955	869	87
Phantom facing array of canisters			686	668	96
Plutonium metal storage	175-252	22-26	240	238	27
Edge exposure, three feet from array			233	219	27
Metal and calcine storage	167-183	17	210	213	11
Front exposure, six feet from array			260	263	10
letal and calcine storage	136	--	172	161	9
Back exposure, six feet from array			196	186	12

(continued) ⟶

Description of Exposure	Survey Dose Range (mrem) Neutron	Survey Dose Range (mrem) Gamma	Thermoluminescent Dosimeter Dose Indication (mrem) Neutron Two-Pair	Neutron Three-Pair	Gamma
Barrel waste storage	78-83	17	60	65	24
No neutron shielding			72	76	22
Phantom, seven feet from line of barrels	175-216	41	197	203	48
			204	209	48
Response to materials in unshielded and shielded (two-inch Benelex) conditions. Phantom, two feet from material. Frontal exposure:					
Plutonium fluoride, unshielded	151-158	23	115	125	31
			107	123	31
	1054-1105	164	836	903	204
			1115	1213	194
	76-114	26-29	77	85	26
			86	83	30
Plutonium fluoride, unshielded	518-525	113	646	640	121
			463	488	120
	2275	428	1770	1825	432
			1982	2060	420
Plutonium fluoride, moderated	93	21	68	70	27
			106	111	26
	91-116	27-36	102	105	29
			110	113	32
Plutonium calcine, unshielded	51-69	44	34	40	47
			37	33	49
	473-563	376-415	415	408	380
			435	428	417
Plutonium oxide, unshielded	81-83	86	59	65	62
			18	24	65
Plutonium metal unshielded	40	85	51	58	65
			23	25	74
	50-67	77	51	60	84
			66	68	61
Plutonium fluoride					
unshielded	383	113	443	443	120
moderated	74	18	72	75	23
combined unshielded and moderated	457	131	555	568	155
Plutonium fluoride					
unshielded	830-835	123	795	798	1[illegible]
moderated	94-111	23	114	116	19
combined unshielded and moderated	924-946	146	906	917	163

→ (Table VI)