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AUTHORITY: AOC ADC [✓ ADD]
NAME: [illegible]
ORG: PNNL
2nd REVIEW-DATE: 7/27/99
NAME: JBriggs
ORG: PNNL-ADD

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SURFACE DOSAGE RATE STUDIES OF TASK III FEED MATERIAL

Introduction

During an investigation of exposures involved in the 234-5 processing cycle, the current estimate of the dosage rate from plutonium fluoride powder was questioned as possibly being too low. In that exposure time of hands during manual manipulation of plutonium fluoride powder must be limited by the neutron and gamma radiations present, a study was made of the surface dosage rate problems involving Task III feed material.

Summary

From the results of this study it appears that the best value for the surface dosage rate from 450 grams of pink colored plutonium fluoride powder contained in a Plexiglass jar with one-fourth inch thick walls is 4.5 rem/hr, of which 3.0 rem/hr is due to fast neutrons from the reaction F^{19}(alpha, neutron)Na^{22} with an average energy of 0.75 Mev. The surface dosage rate from 450 grams of blue colored plutonium fluoride powder contained in a Plexiglass jar with one-fourth inch thick walls appears to be 3.5 rem/hr of which 2.0 rem/hr is due to fast neutrons from the above reaction.

The 1.5 r/hr gamma radiation was measured through one-fourth inch of Plexiglass. Three effective energies were found:

680 kev	50%
50 kev	8%
17 kev	42%

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From this it has been possible to calculate the surface dosage rate through neoprene and surgical gloves (110 and 14 mg/cm^2) and through plastic gloves, or bags, and surgical gloves (40 and 14 mg/cm^2). These are 5.22 rem/hr and 5.34 rem/hr, respectively.

It is strongly recommended that techniques used in handling plutonium fluoride powder be carefully evaluated.

Discussion of Plutonium Fluoride Dosimetry

The penetrating radiations from plutonium fluoride powder consist primarily of fast neutrons and X-Ray and gamma photons. The neutrons from plutonium fluoride arise almost entirely from the F^{19}(alpha, neutron)Na^{22} reaction, however, one would expect a few neutrons from spontaneous fission(1). In one of the first reports on the dosimetry of plutonium fluoride powder Whipple (2) reported that the maximum neutron energy from the above reaction is about 4.5 Mev, however, further work has indicated that the maximum is slightly over 2.0 Mev with an average energy of about 0.75 Mev (3). In HW-20785 (1) Roesch gives the neutron energy spectrum from polonium-boron and polonium-fluorine sources.

From studies made with a recoil proton counter and a BF_3 proportional counter Whipple (concluded that the exposure rate handling a RG model boat of plutonium fluoride powder was from 8 to 24 mrem/hr "plus incidental gamma", while the hand exposure rate during the shaking of a mixed charge was 110 mrem/hr "plus incidental gamma". Using the equation found on page 7 of this report the author approximated a maximum exposure rate of 2000 mrem/hr for the manual manipulation of a RG model boat. The calculations used in this estimation are found in Appendix I, page 13. Since the mixed charge was sometimes handled in plastic jars similar to those used in this study it is estimated that the exposure rate for this operation was 4.5 rem/hr.

At a later date Reddie and Whipple made further studies around hoods on the RG line with a proton recoil counter and BF_3, however, these do not include surface dosage rates (4).

Gamma and X-Ray photons from plutonium fluoride powder arise from six principal sources (2, 3, 5, 6, 7, 8, 9, & 10):

1. Uranium L X-Rays produced by plutonium atoms which have just undergone decay and have been left in an excited electronic state due to absorption of gamma rays given off during decay.

2. L X-Rays are expected from alpha particle excitation of the plutonium atoms, although they have not been observed or identified as yet.

3. Gamma photons associated with the disintegration of plutonium. Energies ranging from 35 kev to 400 kev have been reported.

4. Inelastic alpha scattering denoted by the reaction (alpha, alpha, gamma). Gamma energies of 0.5 Mev and higher have been reported from alpha scattering with fluorine.

5. Gamma photons emitted from fission product contaminants.

6. Transuranic elements such as Americium. The concentration of these elements is a function of reactor ambients, cooling time, and decontamination factor of the separations process. Unfortunately, there is insufficient historical surface dosage rate data available which would enable one to predict possible gamma radiation levels from this source at pile exposures other than the present 600 MWD/T.

The number of gamma photons due to the first four sources is directly proportional to the amount of plutonium present. The contributions of the last two sources are variable, however, and will depend upon upstream processing conditions (3,11).

Kaigler (12) made some radiation measurements on plutonium fluoride powder in an attempt to determine a rule of thumb for the calculation of the neutron surface dosage rate by field personnel on a routine basis. By extrapolation of dosage rate versus distance data to zero distance he obtained a graphically determined surface dosage rate. For the specific cases studied he found that the surface dosage rate was about nine-fourths times the dosage rate measured at the nearest approach to the source. However, it later became evident that this nine-fourths rule was unreliable and, therefore, was discarded.

Jech (13) attempted three methods to determine the gamma surface dosage rate from plutonium fluoride powder:

1. Measure the dosage rate at one inch intervals, plot, and extrapolate to zero distance. This proved unreliable as the results depended so much upon the manner in which the curve was drawn (as in Kaigler's method).

2. From a factor determined by knowing the dosage rate at "half distances", calculate the dosage rate at one-fourth inch (assumed to be surface). This does not hold well, however, due to geometrical considerations in the calibration of the detecting instrument and in the behaviour of radiation from a source of finite dimensions.

3. Use film to determine the surface dosage rate. However, film studies are misleading unless the energy dependence of the film is known and the energy distribution of the source is known.

   Roesch has reported that the correction factor used in measuring exposure to the gamma radiations from plutonium varies from one lot of film to the next (14). In HW-27486 (15) Roesch reports the energy dependence of Hanford film badges (Dupont 502 film). It is of interest to note that the film sensitivity changes by a factor of two between 17 and 27 kev.

As a result of his studies Mr. Jech made the following recommendations:

1. More work should be done on this project,

2. Neutron measurements obtained in this study were of no value due to lack of correlation, and

3. For the time being multiply C.P. dosage rates at 2 inches by 6 to obtain the surface dosage rate.

Both Mr. Jech and Mr. Kaigler recognized the possibility of higher dosage rates than were previously considered to be the case and initiated rule of thumb methods for determining these dosage rates. However, they also recommended further study of methods for monitoring surface dosage rates.

Until recently the dosage rate for close approach to plutonium fluoride powder was estimated as 200 mr/hr plus nine-fourths of the neutron dosage rate at closest approach. Shortly after starting work on HW-28918(11) in early June 1953 it was recognized that the nine-fourths rule was deficient and that 800 mrem/hr would be more nearly correct. In September 1953 this was changed to the present 4.5 rem/hr based on the preliminary results of this report.

As the results of a literature search conducted by personnel of the Technical Information Group it appears that there is no published information on plutonium fluoride powder surface dosage rate studies conducted at other AEC installations (16).

Discussion of Survey Equipment and Methods

The basic neutron data for this report was obtained by the use of a standard moderated BF_3 proportional counter(17). This instrument is normally used for the quantitative determination of slow and intermediate neutron fluxes. Since the average neutron energy from plutonium fluoride powder is 0.75 Mev, it was necessary to provide a correction in the slow neutron calibration of the BF_3. Rossch and Glascow (18) found that a BF_3 with a sensitivity of 8.47 c/m per slow $n/cm^2/sec$, also had a sensitivity of 7.0 c/m per fast $n/cm^2/sec$. Since a flux of 1 fast $neutron/cm^2/sec$ from plutonium fluoride neutrons is equivalent to 0.074 mrem/hr, then 7.0 divided by 0.074 or 94.6 c/m is equivalent to 1 mrem/hr of plutonium fluoride neutrons. Therefore, a BF_3 whose sensitivity to slow neutrons is known will have a sensitivity to plutonium fluoride neutrons in the ratio of 94.6 to 8.47, or 11.2. The BF_3 used in this study had a sensitivity of 6.74 c/m per $sn/cm^2/sec$. Therefore, its calibration for plutonium fluoride neutrons is

$$\frac{(94.6)(6.74)}{(8.47)} = (6.74)(11.2) = 75.3 \text{ c/m per mrem/hr.}$$

The physical orientation of the source and detector with respect to scattering objects was found to affect the calculated surface dosage rate.

Generally speaking, however, most of the determinations were made under conditions where the scattering of neutrons was avoided as much as possible. In every case the moderator was kept broadside to the source to avoid the errors of assymetry as reported by Whipple (2). Seven of the eleven runs were made with the moderated BF_3 tube remaining fixed and the source moving vertically up and down beneath it. Two of the runs were made under the opposite conditions, and two of the runs were made with the BF_3 "seeing" the side of the source. It was found that dosage rates on the side of the cylinder calculated by the Dr^2 method (see below) would vary considerably, and therefore, this method of obtaining data was discontinued. The problem is discussed further under "Analytical Methods", page 5.

Several attempts were made to use neutron sensitive film to determine the surface dosage rate. The first run was made on the outside of a jar of plutonium fluoride powder. These results were considerably lower than the dosage rate calculated by the Dr^2 method or by formula (see page 7). The second run was a failure also, as the source material did not appear to be uniformly fluorinated. Results of this run presented no identifiable pattern of consistency. No attempts to continue the film study are planned for the future, since, under present conditions, the statistical reliability of film measurements is inadequate.

The gamma radiation studies were made using the Trent (19) as the detector. However, due to production schedules, it was not possible to obtain gamma dosage rate and energy data on all the runs observed. As reported elsewhere (3,11) the intensity of the gamma components does not appear to be constant from batch to batch. It is hoped that the gamma dosage rate reported in this document will be accurate for a major portion of the runs encountered in the future. However, this point is open to question, since the filter boat data obtained on the particular runs studied shows them to be about average. (Gamma dosage rates on filter boats of unfluorinated plutonium oxalate average around 13 mr/hr (11).) Runs with higher than normal filter boat dosage rates could conceivably have a higher surface dosage rate than that reported here. It would be likely to assume that most of this would be due to higher energy fission products (ca. 680 kev effective energy), although the concentration of transuranic elements in the feed stream could also affect this.

The Trent was used to determine the variation of the gamma dosage rate versus distance and also the effective gamma energies of the source. The effective energies were determined by an absorption study using the method and materials described in HW-28918 (

Discussion of Analytical Methods

Initial attempts by the writer to determine the surface dosage rate from plutonium fluoride powder by extrapolation of dosage rate data to zero distance were essentially worthless. It became evident after a few trials that the answer depended considerably on how the curve was drawn. Answers varying by a factor of 10 were possible from the same data. In a discussion of the problem Dr. W. C. Roesch, Physics, Biophysics Section, Radiological Sciences, suggested a new mathematical tool which had recently been proposed. This method is known as the Dr^2 method and is known to work quite well for flat surfaces and thick sources. Wende (20) describes a thick source as being one which acts nearly as though the activity were being emitted from one mean free path, L, where L is equal to the reciprocal of the linear absorption coefficient.

Briefly, the Dr^2 method involves the following steps:

1. Obtain data showing the variation of dosage rate versus the distance between reference points on the source and receptor.
2. Determine the "effective center" by plotting the reciprocal of the square root of the dosage rate versus the distance as measured in step 1. (The definition of "effective center" is somewhat elusive: If the receptor is very small compared to the source depth, the "effective center" becomes that of the source, and vice versa. However, when both the source and receptor have approximately the same depth, this term becomes a correction factor to be added to the measured distances.)

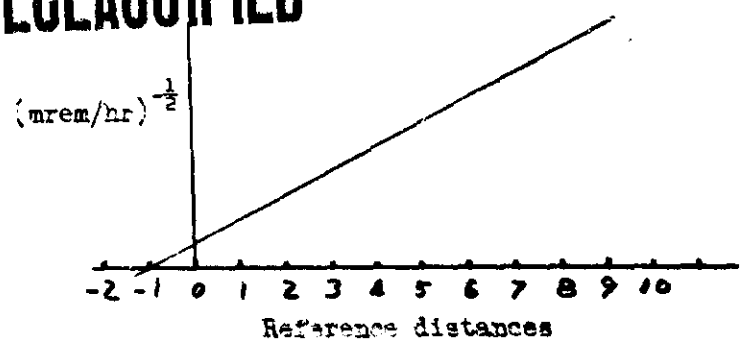

Figure 1 Determination of "effective center"

For instance, in this example the distances as measured in step 1 are to be corrected by adding one unit distance to the reference distances.

3. Using the corrected distances plot the product of the dosage rate and the distance squared (Dr^2) versus the distance. The initial portion of the curve shows a rather rapid variation in Dr^2 as r increases. This is the range in which the geometry of the source and receptor is poor. The last portion of the curve represents the range in which scattering is taking place. The middle portion of the curve represents the range in which the inverse square law holds, i.e., where Dr^2 is a constant. Extend this flat portion of the curve back to $r = 0$ and determine the value of Dr^2.

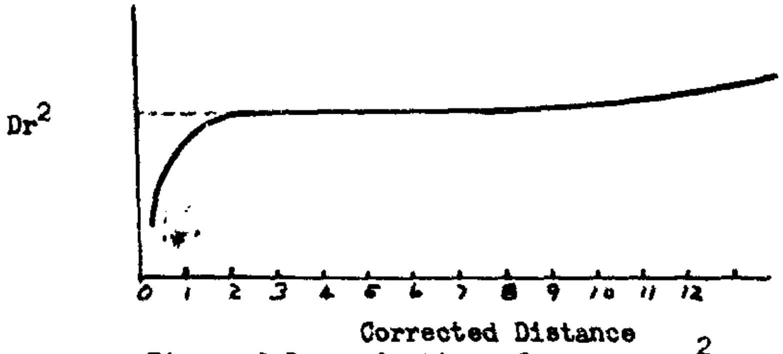

Figure 2 Determination of average Dr^2 value

4. Knowing the projected area of the source, A, the surface dosage rate is found by the following formula:

$$D_{surface} = \frac{2\pi}{A} \times Dr^2$$

It should be noted that the average Dr^2 value may be found from the curve of the reciprocal of the square root of the dosage rate, since the reciprocal slope is r divided by $(D)^{-\frac{1}{2}}$, or $(D)^{+\frac{1}{2}}r$, or $(Dr^2)^{+\frac{1}{2}}$. This eliminates one step and was used in this study rather than the longer method.

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Dosage rates obtained from the data in which the BF_3 "saw" the side of the source are in error due to the constantly changing "projected area" as the source to receptor distance changes. Note that the average height of powder in the jar was about one inch while the BF_3 moderator has a diameter greater than five inches.

During the course of analyzing the experimental data, attempts were made to develop some sort of mathematical relationships which would lead to a calculated surface dosage rate. Two equations were derived which give solutions for (a) the maximum surface dosage rate, I_b, at the center of the base of a right cylinder, and (b), the dosage rate, I_s, at the side of a right cylinder at the base.

$$I_b = 2\pi I_0^* h\left[\frac{R}{h}\cot^{-1}\frac{R}{h} + \frac{1}{2}\log_e\left(1+\frac{R^2}{h^2}\right)\right] = 2\pi I_0^* h\left[f_b(D,h)\right],$$

$$I_s = 2I_0^* h\int_0^{\frac{\pi}{2}}\left[\frac{D\cos\theta}{h}\cot^{-1}\frac{D\cos\theta}{h} + \frac{1}{2}\log_e\left(1+\frac{D^2\cos^2\theta}{h^2}\right)\right]d\theta$$

$$= 2I_0^* h\left[f_s(D,h)\right],$$

where D is the diameter of the cylinder
R is the radius of the cylinder
h is the height of the cylinder
I_0^* is the intensity at unit distance from the activity in unit volume of the source.
$f_b(D,h)$ and $f_s(D,h)$ are function of D and h. See Fig. 5, page 20

To calculate I_0^* it is necessary to know dosage rate versus distance data, corrected for the difference in measured distance and "effective center", and the volume of the source. Therefore,

$$I_0^* = \frac{I_r \cdot r^2}{V}$$

It should be noted that the dosage rate equations given above do not contain a correction for self absorption. In the case of neutrons these equations should prove quite reliable, however, they are of little value for determining gamma surface dosage rates from plutonium fluoride powder. The derivation of these equations is given in a separate report (21).

Discussion of Results

Appendix II, page 14 lists the detailed information on each batch of powder studied. It will be noted that, in general, the mathematical calculations for the neutron surface dosage rate are consistently higher than the Dr^2 values by a factor varying from 1.4 to 1.8.

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There are several possible reasons for this such as:

1. Self absorption may be appreciable, however, this is not what one would expect.

2. The Dr^2 value may be in error.

3. The mathematical value is the maximum dosage rate at the center of the base rather than the average.

On the basis of the data collected, it appears that the best value for the neutron surface dosage rate from 450 grams of pink colored plutonium fluoride powder contained in a Plexiglass jar with one-fourth inch thick walls is 3.0 rem/hr, while the corresponding dosage rate from blue powder is 2.0 rem/hr. The gamma surface dosage rate through one-fourth inch of Plexiglass (0.747 grams/cm^2) appears to be 1.5 r/hr. From studies of the effective energy distribution of the gamma radiations of plutonium fluoride powder it appears that the following energies and percentages may be taken as representative:

680 kev	50%
50 kev	8%
17 kev	42%

It is therefore recommended that under present process conditions that the surface dosage rate from one batch of pink plutonium fluoride powder be taken as 4.5 rem/hr through one-fourth inch of Plexiglass. Present process conditions may be roughly defined by stating that the gamma activity from a filter boat of plutonium oxalate does not exceed 20 mr/hr as measured with a C.P. through the carrier, and secondly, that the pile exposure is about 600 MWD/T. Exposure of personnel to powder in containers where the shielding is less than this amount will be greater than 1.5 r/hr. It will depend upon the effective energy distribution and the thickness of the material.

To approximate the surface dosage rate without the benefit of the Plexiglass shielding, let us make the following assumptions:

1. That the powder is still in the same shape as in the original study, but that the walls are regular vinylite plastic such as that used in routine 234-5 RMA hood operations. This material has a thickness of approximately 40 mg/cm^2.

2. That the operator is wearing surgical rubber gloves. On an <u>unstretched</u> sample a thickness of 15.5 mg/cm^2 was determined. Let us assume that the thickness when stretched over the hand is about 14 mg/cm^2.

3. That the energy distribution holds as quoted above.

4. That the absorption coefficients of plastic, plexiglass, and rubber gloves are the same as for human tissue. As shown in HW-27194 (22) the effective absorption coefficients are larger than the tabulated ones for transmission through absorbers in contact with a thick source. Unfortunately, the analysis there does not apply to curved absorbers at a short distance from a thick source. Neglect of this will result in a slightly higher calculated dosage rate (23).

Further, let us define I as the dosage rate through Plexiglass and I_o as the dosage rate through plastic and rubber gloves. Therefore,

$$I = I_o \exp -u(t_{Plexiglass} - t_{plastic} - t_{gloves}) = I_o e^{-0.693u}.$$

Energy	Percent	u	net thickness	ut	e^{ut}	I	I_o
kev	%	cm^2/gram	grams/cm^2			mr/hr	mr/hr
680	50	0.077	0.693	0.0534	1.054	750	790
50	8	0.217	0.693	1.1504	1.163	120	140
17	42	1.16	0.693	0.8039	2.233	630	1410
					Total		2340

Therefore, the total dosage rate (neutron plus gamma components) is 5.34 rem/hr. By the same process the dosage rate through neoprene hood gloves is found to be 5.22 rem/hr. This will approach the exposure rate one might receive at Station 10, Hood 9, Task II, although the shape of the source is somewhat different.

During a portion of the Task III work the plutonium fluoride powder is mixed with calcium and iodine and the mixture is placed in a magnesium oxide crucible, which has previously been charged with a small quantity of plutonium metal turnings. From discussions with Operations personnel the level of the mixture is generally from three to four inches below the crucible top. This unit is called a can pack. Operators must handle the can pack by the sides and occasionally by the top. Based on the data presented above and in Appendix III, page 15 , the dosage rate on the sides has been approximated at 1.6 rem/hr and 300 mrem/hr on the top. The calculations for these approximations are given in Appendix IV, pages 16 through 18 . In the past a three-sixteenths inch thick magnesium oxide cap had been placed over the crucible. With this in place the total dosage rate at the top of the can pack is reduced to 275 mrem/hr. However, for uniformity it is recommended that 300 mrem/hr be used.

The dosage rate at the top of a jar of powder had been calculated by the subtraction method shown on page 7 of HW-30185 (21). For a typical run this is approximately 100 mrem/hr. Calculation of this dosage rate is found in Appendix V, page 19 . It is readily recognized that it is much safer to carry a jar of powder by the top rather than by the bottom.

Conclusions

1. Surface dosage rates presently encountered in the Task II and III operations involving plutonium fluoride appear to be:

top surface of filter boat	5.3 rem/hr
through ¼" Plexiglass jar (about 450 grams of powder)	4.5 rem/hr
through side of can pack	2.4 rem/hr
at top of can pack	300 mrem/hr
at top of Plexiglass jar	100 mrem/hr

2. In view of the many process variables, such as pile ambients, pile exposure, and decontamination factor, which influence the neutron and gamma dosage rates, it will be necessary to periodically re-examine these surface dosage rate problems.

3. High surface dosage rates involved in handling plutonium fluoride powder make it highly desireable to limit total hand exposure by either eliminating or minimizing present manual manipulations.

The writer wishes to thank Dr. W. C. Roesch for his assistance in the evaluation of methods and data.

Geo. Lewis Helgeson

Geo. Lewis Helgeson
Seaparations, Radiation Monitoring

GLH:jj

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APPENDIX I

APPROXIMATION OF THE DOSAGE RATE FROM AN RG MODEL BOAT

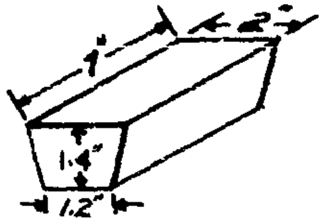


Figure 3 RG Model Boat

Approximate weight of powder per boat was 190 to 200 grams.

Assume uniform depth of powder. Assume that the density of the powder was the same as present powder, or 2.14 grams/cc (average).

Therefore, the average depth of powder in the boat was

$$\frac{(200\ \text{grams})}{(2.14\ \text{grams/cc})} \times \frac{(1\ \text{cubic inch})}{(16.387\ \text{cc})} \times \frac{1}{(1.2\ \text{inches})(7\ \text{inches})} = \frac{(5.7\ \text{in}^3)_2}{(1.2)(7)\text{in}}$$

equals 0.68 inches deep.

Assume the material is in the form of half of a cylinder 7 inches long. Then the diameter is

$$\frac{5.7\ \text{in}^3}{7\ \text{in}} = 0.814\ \text{in}^2, \quad A = \frac{\pi D^2}{8}, \quad D = \sqrt{\frac{(8)(0.814)}{\pi}} = 1.44\ \text{inches}$$

From discussions with Operations personnel it was learned that the general practice was to move the boats two at a time in a scow, which was 14 inches long. The scow was held by the ends. Therefore, let us approximate the maximum dosage rate at the end of a cylinder 1.44 inches in diameter and 14 inches long. The actual exposure rate would be somewhere in the vicinity of half of this.

$D/h = 1.44/14 = 0.103$, $f_b = 0.08$

I_o^* (neutrons) = 350 mrem/hr, I_o^* (gamma) = 175 mr/hr

$I_n = (2\pi)(350)(14)(0.08) = 2462$ mrem/hr,

$I_g = (2\pi)(175)(14)(0.08) = 1231$ mr/hr, Total = 3693 mrem/hr

Therefore, the approximate exposure rate is in the vicinity of 1850 mrem/hr. Assume a value of 2.0 rem/hr.

APPENDIX II

234-5 Run Number	Color	Weight grams	depth inches	inside diameter inches	volume cc	density grams/cc	I^*_o (neutrons)	Surface Dosage Rates				
								Neutrons mrem/hr			Gamma mr/hr	
								I_{Dr^2}	I_b	I_s min	I_{Dr^2}	I_b/I_{Dr^2}
RMX-13-6-8 &9*	pink	841.0	1.875	3.75	339	2.48	301	4860	4019	2179	1290	0.85
RMX-13-7-146 & 148	pink	836.4	2.25	4.0	463	1.806	262	5930	3846	2154	1390	0.65
RMX-13-8-84	pink	503.9	1.188	3.688	208	2.424	365	2880	4136	2260	1360	1.44
RMX-13-9-16	pink	484.2	1.313	4.0	270	1.791	341	2965	4218	2256	--	1.44
RMX-13-9-18	pink-blue	484.7	1.25	3.875	242	2.006	382	2910	4556	2443	--	1.57
RMX-13-9-19	blue	339.4	0.688	3.875	133	2.554	456	1960	4061	2146	--	2.07
RMX-13-9-20	blue	332.1	1.25	3.938	249	1.331	241	1782	2909	1545	--	1.63
RMX-13-9-35	green	510.4	1.125	4.125	246	2.147	321	2270	4177	2204	--	1.84
RMX-13-9-39	purplish-blue	459.9	0.938	4.0	193	2.382	404	2400	4285	2500	--	1.79
RMX-13-9-40	pink	524.4	1.0	4.0	206	2.547	448	2870	4871	2614	--	1.70
Hood 10 Sweepings* (from 8-28-53)	pink	998	3.5	3.25	458	2.100	362	6050	4975	2914	1390	0.82

*These measurements were made at the side of the jar and were not used in determining the average surface dosage rate.

APPENDIX III

Can Pack Data

Powder	
Average weight	450 grams
Average density	2.14 grams/cc
Average volume	210 cc
Iodine	
Average weight	108.7 grams
Density	4.93 grams
Volume	22 cc
Absorption Coefficients	(cm^2/gram)
680 kev	0.071
50 kev	10.
17 kev	34.
Calcium	
Average weight	164.6 grams
Density	1.55 grams/cc
Volume	106.2 cc
Absorption Coefficients	(cm^2/gram)
680 kev	0.070
50 kev	0.8
17 kev	16.5

Crucible, lid, and sand		
Composition	MgO	
Density of pure MgO	3.63 grams/cc	
Porosity of crucible	21%	
Effective crucible density	3.0 grams/cc	
Dimensions	See sketch	
Absorption coefficients	(cm^2/gram)	
	Magnesium	Oxygen
680 kev	0.074	0.076
50 kev	0.38	0.21
17 kev	5.8	1.4

Can pack steel	
Thickness	0.017 inches
Density	7.7 grams/cc
Composition	Mild steel, (assume to be Fe)
Absorption Coefficients	(cm^2/gram),
680 kev	0.070
50 kev	2.0
17 kev	42.0

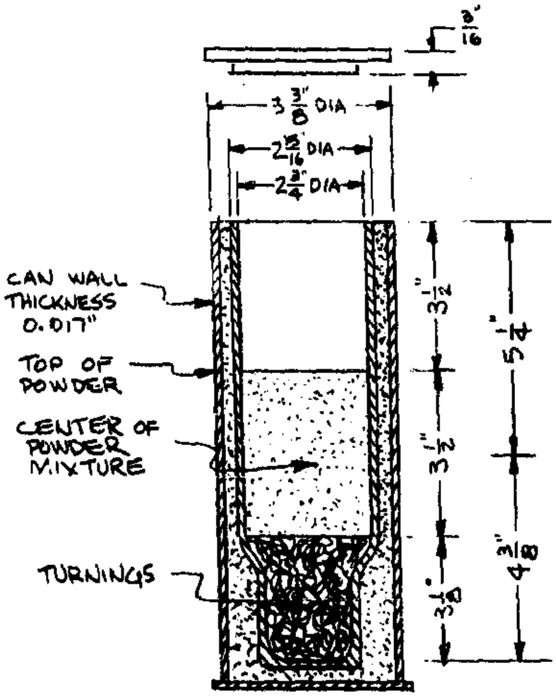


Figure 4
Makeup of Can Pack

[redacted]

APPENDIX IV

Approximation of Surface Dosage Rates on Can Pack

From the data in Appendix III the weight, volume, and density of the mixture is found.

Material	Weight grams	Density grams/cc	Volume cc
Powder	450	2.14	210.3
Iodine	108.7	4.93	22.0
Calcium	164.7	1.55	106.2
Mixture	723.4	2.14	338.5 or 20.66 cu. in.

Therefore, the depth of the powder in the crucible is

$$h = \frac{V}{\pi r^2} = \frac{(20.66)}{(\pi)(1.375)^2} = 3.5 \text{ inches}$$

From dosage rate versus distance data (17) average values of I_o^* for the neutron and gamma components have been found.

I_o^* (neutrons) = 350 mrem/hr at one inch from one cubic inch

I_o^* (gamma) = 175 mr/hr at one inch from one cubic inch

Note: These values of I_o^* were obtained through one-fourth inch of Plexiglass.

Since we now have a different volume, the I_o^* values will be changed to:

$$I_o^* \text{ (neutrons)} = \frac{(350)(210.3)}{(338.5)} = 217 \text{ mrem/hr (etc.)}$$

$$I_o^* \text{ (gamma)} = \frac{(175)(210.3)}{(338.5)} = 108.7 \text{ mr/hr (etc.)}$$

As shown in the neutron attenuation curves in HW-28913 (11) one-fourth inch of Plexiglass will have essentially no effect on the neutron dosage rate. However, it will reduce the gamma component. Therefore, it will be necessary to calculate values of I_o^* for each gamma energy on the basis of no Plexiglass shielding. Therefore,

$$680 \text{ kev} \quad I_o^* = (0.50)(108.7)e^{0.0575} = (54.4)(1.059) = 57.6 \text{ mr/hr (etc.)}$$

$$50 \text{ kev} \quad I_o^* = (0.08)(108.7)e^{0.162} = (8.7)(1.176) = 10.2 \text{ mr/hr (etc.)}$$

$$17 \text{ kev} \quad I_o^* = (0.42)(108.7)e^{0.867} = (45.6)(2.38) = 108.6 \text{ mr/hr (etc.)}$$

[redacted]

APPENDIX IV (Continued)

Case I, Dosage Rate at the Side of a Can Pack

Since the additives are assumed to be well mixed with the plutonium fluoride powder, it will be difficult to estimate the degree of shielding gained by these additives. Let us base the initial calculations on the assumption that the additives will provide no shielding whatsoever. Therefore, the only shielding will be due to the can pack. The following table presents the stepwise calculation of the I_o^* values for each energy correcting for the can pack shielding:

Energy kev	Component	cm^2 u/gram	t inches	t $grams/cm^2$	e^{-ut}	I_o^* uncorrected	I_o^* corrected
680	MgO Fe	0.075 0.070	0.3125 0.017	2.38 0.332	0.836 0.977	57.6	47.0
50	Mg O_2 Fe	0.38 0.21 2.0	0.1875 0.125 0.017	1.43 0.955 0.332	0.581 0.134 0.516	10.2	0.41
17	Mg O_2 Fe	5.8 1.4 42.0	0.1875 0.125 0.017	1.43 0.955 0.332	2.5×10^{-4} 0.263 8.7×10^{-7}	108.6	6.2×10^{-9}

Therefore, as far as the gamma component is concerned, one need only determine the dosage rate due to the 680 and 50 kev components, or $I_o^* = 47.4$ mr/hr (etc.).

Since a distance correction is rather difficult to determine, let us assume that the radiation is uniformly distributed throughout the diameter of the can pack, 3.375 inches. Therefore, I_o^* will be reduced by the ratio of the volumes.

$$\frac{V_1}{V_2} \quad \frac{\pi D_1^2 h/4}{\pi D_2^2 h/4} = \frac{(2.75)^2}{(3.375)^2} = 0.664$$

$$I_o^* \text{ (corrected)} = (0.664)(47.4) = 31.5 \text{ mr/hr (etc.)}$$

$$D/h = 3.375/3.5 = 0.964, \quad f_{s_{max}} = 1.93$$

Therefore,

$$I_{s_{max}} = (2)(31.5)(3.5)(1.93) = 425 \text{ mr/hr}$$

This calculation assumes no absorption in the source itself, and therefore gives a maximum dosage rate.

Next, the neutron dosage rate must be estimated. Let us assume, once again, that the radiation is uniformly distributed throughout the diameter of the can pack. Therefore,

$$I_0^* = (0.664)(217) = 144 \text{ mrem/hr (etc.)}$$

and,

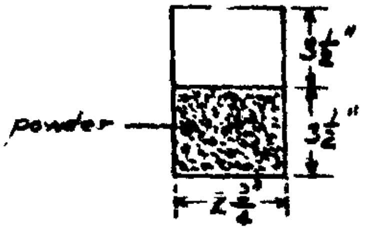
$$I = (2)(144)(3.5)(1.93) = 1945 \text{ mrem/hr}$$

Therefore, the total dosage rate at the side of a can pack will be approximately 1945 plus 425 or 2370 mrem/hr = 2.4 rem/hr.

Case II, Dosage Rate at the Top of a Can Pack

The dosage rate along the axis of a cylinder somewhat removed from the center of the base (or top) may be estimated by the subtraction method described on page 7 of HW-30285 (21).

The neutron component is estimated as follows:



$$D/h = 2.75/7 = 0.393; \quad f_b = 0.29$$

$$D/h_{3.5} = 2.75/3.5 = 0.786; \quad f_b = 0.54$$

$$I = (2)(217)(7)(0.29) = (3.5)(0.54)$$

$$= 191 \text{ mrem/hr}$$

It is difficult to correct for absorption in the source since there is considerable absorption of the lower energy components. Wende (20) has shown for a thick source that the radiation may be assumed to be emitted from a depth of one mean free path in the source. For instance, one mean free path of 17 kev gamma radiation passing through iodine is

$$\frac{1}{(34.0)(4.93)} = 0.006 \text{ cm} = 0.00236 \text{ inches.}$$

One can see that many complications could arise in trying to correct for absorption. Therefore, let us neglect absorption entirely, realizing that any answer so obtained will be on the high side.

Estimating the gamma dosage rate on the top of the can pack by the subtraction method:

$$I = (2)(108.7)\left[(7)(0.29) - (3.5)(0.54)\right] = 96 \text{ mr/hr.}$$

Therefore the total dosage rate at the top of a can pack is approximately 300 mrem/hr.

APPENDIX V

Calculation of the Dosage Rate at the Top of a Jar of Powder

The following data is taken as a typical run:

450 grams of plutonium fluoride powder

average density	2.14 grams/cc
average volume	210. cc
inside diameter of the jar	4.0 inches
average depth of powder	1.0 inch

distance from top of powder to top of jar (through the lid) $6\frac{1}{4}$ inches

jar cover is one-fourth inch thick Plexiglass

I_o^* (neutrons) is 350 mrem/hr at one inch from one cubic inch of powder

I_o^* (gamma) is 175 mr/hr at one inch from one cubic inch of powder

By the subtraction method:

$D/h = 4/7\frac{1}{4} = 0.552$ $\qquad f_b = 0.392$

$D/h_2 = 4/6\frac{1}{4} = 0.640$ $\qquad f_b = 0.450$

$$I_n = (2\pi)(350)\left[(7.25)(0.392) - (6.25)(0.45)\right] = 64 \text{ mrem/hr}$$

Neglecting absorption, as in Case II, Appendix IV,

$$I_g = (2\pi)(175)\left[(7.25)(0.392) - (6.25)(0.45)\right] = 32 \text{ mr/hr}$$

Therefore, the total dosage rate at the top of a jar of powder is about 96 mrem/hr. The best value to use would be 100 mrem/hr.

Figure

f(D,h) versus