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THULIUM-170 OXIDE HEAT SOURCE EXPERIMENTAL AND ANALYTICAL RADIATION AND SHIELDING STUDY

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ABSTRACT

Radiation dose rates from three thulium-170 oxide sources (20.7, 10.0 and 5.0 thermal watts) were measured through three thicknesses ($\frac{1}{4}$, $\frac{1}{2}$ and 1 inch) of absorber by thermoluminescent dosimetry techniques. Absorber materials used were aluminum, stainless steel, lead, tungsten and depleted uranium. Resultant radiation doses were measured at two source - to - dose point separation distances - (19 cm and 100 cm). Comparison of theoretical dose rates calculated by computer with measured dose rates validated the calculation technique for lead, tungsten and uranium absorbers but not for aluminum and stainless steel.

Use of infinite medium build-up factors (B_{∞}) was thus validated in computation of dose rates for lead, tungsten and uranium absorbers; use of B_{∞} in computation of dose rates for aluminum and stainless steel absorbers overestimated dose rates vis-a-vis experimentally determined dose rates by an approximate factor of 2.

SECTION 1

INTRODUCTION

Thulium-170 in the sesquioxide form is a potential isotopic heat source for thermoelectric and thermionic applications. The overall objective of this program is to establish the feasibility of application of the oxide as a heat source by assessing and analyzing the properties of the oxide; several of these properties have already been discussed in previous reports (1, 2). The discussion herein relates to the application of several types of shielding materials to a representative pressed, sintered and activated Tm_2O_3 source, and the development of techniques and methods of analysis that will allow calculated dose rates to predict reliably those dose rates to be experienced experimentally.

The radioactive sources were in the form of Tm_2O_3 wafers of sufficient activity to obtain dose rates through the absorbers that would yield reliable data. Three source strengths were supplied - 20.7, 10.0 and 5.0 watts (thermal), respectively.

Five absorber materials selected were aluminum, stainless steel, lead, tungsten and depleted uranium; absorbers with thickness of $\frac{1}{4}$, $\frac{1}{2}$ and 1 inch were used to determine the dose rates at two source-to-dose point distances; 19 and 100 cm.

Computer programs were developed to generate the theoretical dose rate. The 19 centimeter source-to-dose point distance was selected as a theoretical dose point as it closely conformed to the finite boundary established by the absorber. A build-up factor for infinite medium (B_∞) was used throughout the calculations

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to determine the scattered flux component. A more rigorous treatment of the build-up factor is necessarily highly complex and was beyond the scope of this work.

Doses were measured using lithium fluoride powder as a phosphor and subsequently reading the absorbed dose with a thermoluminescent phosphor readout system.

A comparison of experimental results with theoretical values is presented in Section 3.

All of the experimental data, development of mathematical relationships, calculation procedures including discussion where appropriate, and error analyses are found in the Appendices.

SECTION 2

SUMMARY

Dose rates from 5, 10 and 20.7-watt Tm_2O_3 sources at two different source-to-dose point distances (19 and 100 cm) were experimentally determined using thermoluminescent dosimetry techniques. Five different absorber materials (aluminum, stainless steel, lead, tungsten and depleted uranium) of three different thickness ($\frac{1}{4}$, $\frac{1}{2}$ and 1 inch) were used. The overall uncertainty (experimental error) in the measured dose rate was $\pm 9\%$.

A computer code (FORTRAN) was developed to generate theoretical dose rates for the uncollided flux; a buildup factor for infinite medium was used to obtain the total flux.

Predicted-to-measured dose rate ratios of 0.58 to 1.85 characterized results for the three most dense absorbers; ratios of 0.84 to 3.0 characterized aluminum and stainless steel. Deviations of these ratios from 1 are primarily attributable to the assumption of an infinite medium for the buildup factor in generating the theoretical dose rates. Departure of dose rates at 100 cm from the inverse square law is also attributable to the build-up factor used in the analytical predictions.

The applicability to kilowatt sources of the dose rate computational method used in this study was analyzed. It was concluded that an infinite medium build-up factor should be used and a 22 energy group computation made; dose rates accurate to 10 to 20% will result. Of equal importance to system design, required shield thicknesses (Pb, W or U) will be overestimated by only 2 to 3%.

SECTION 3

COMPARISON OF EXPERIMENTAL AND ANALYTICAL DOSE RATES

3.1 RADIATION CHARACTERISTICS OF Tm-170

The radiation characteristics of Tm-170 have been measured and reported in the literature. The most current data are to be found in the Table of Isotopes⁽⁶⁾ and the NBS Nuclear Data Sheets⁽⁷⁾. The decay scheme of Tm-170 is presented in Figure 3-1. Table 3-1 identifies the radiations emitted by radioactive decay of Tm-170.

In addition to the beta, gamma and X-radiation, bremsstrahlung is produced as the beta particles slow down within the source medium (a more detailed discussion can be found in references 35,37 and 38). This bremsstrahlung radiation, combined with gamma and x-rays, constitutes the entire electromagnetic radiation spectrum of the source. Bremsstrahlung production for Tm-170 source has been calculated by Arnold⁽⁸⁾. The number of bremsstrahlung photons, within 50 Kev energy intervals, per 0.968 Mev and 0.884 Mev beta particles are tabulated in Appendix C. The total bremsstrahlung production for Tm-170 is the sum of that for each beta group weighted by the abundance of each group. All photons in each energy interval are assumed to have energy at the midpoint of that interval.

Another important radiation property is the ratio of activity in curies to power in watts. This factor for Tm-170 has recently been reported by Cornman⁽⁹⁾ as 445 ± 22 curies per watt. This value is used throughout this report.

Figure 3-1. Decay Scheme of Tm-170

TABLE 3-1
RADIATION CHARACTERISTICS OF Tm-170

<u>Radiation Emitted</u>	<u>Energy (Mev)</u>	<u>Abundance (%)</u>
Beta	0.968	78
Beta	0.884	22
Gamma	0.084	3.46
X-ray	0.052	5.05
X-ray	0.009	13.49

3.2 CALCULATED DOSE RATES

Equations for the uncollided flux were developed for Tm_2O_3 sources that allowed an approximate solution to be used rather than a more complex exact solution. Analysis indicates that the deviation from the true flux using this approximate solution is negligible.

A computer code was developed using the equations of the approximate solution. 19 cm was selected for the calculated source-to-dose point distance. FORTRAN was used to develop the code and a total of 22 photon energy groups were used to calculate the doses. The main program input data are given in the appendices.

Several factors were considered for inclusion in the computation:

- Attenuation by air
- "Tunnelling" effect
- Mass attenuation uncertainty
- Scattering effects other than those attributable directly to build-up
- Contribution of thulium-171.

The attenuation by air is very small compared to that of the encapsulation material and could be excluded from the computation. The "tunnelling effect" was also evaluated for this computation and found to be insignificant. The uncertainty for mass attenuation coefficients is taken as 5% (see appendices).

Other scattering effects are possible and are discussed in Appendix C, paragraph C.7. For these calculations, analysis shows the scattering to be either insignificant or to have been eliminated by the experimental design.

The contribution of thulium-171 to the dose rate is negligible because of the small percentage of the total activity that it represents and its low energy radiation compared to that of thulium-170. From mass analysis measurements made on 10 July 1968, the isotope distribution for Tm-170 and Tm-171 were 17.7 and 5.5%,

respectively. However, the activity of Tm-171 was only 5.7% that of the Tm-170 activity because Tm-171 has much longer half-life (1.9 years), compared to that of Tm-170 (128 days). The maximum beta energy of Tm-171 (~ 0.1 Mev) is considerably lower than that of Tm-170 (~ 0.9 Mev); therefore, the attenuation through the encapsulation material and absorbers will virtually eliminate any contribution to the dose rate due to Tm-171 atoms. For these reasons, the radiation from thulium-171 was not considered to be contributory to the dose rate and was excluded in the analytical dose rate computations.

Verification of the computer results was accomplished by comparing theoretical half-value thicknesses ($0.693/\mu$) with those generated by the computer program (HVT). (See Appendix C, paragraph C.4).

The total dose is obtained by adjusting the uncollided flux for the scatter component, the build-up factor. In these calculations we have assumed a build-up factor for an infinite homogeneous medium because the analyses of Compton scattering for a finite medium is very complex and beyond the scope of this program.

This assumption is reasonably sound for lead, tungsten and uranium, but would be expected to introduce a larger error for aluminum and stainless steel (See Appendix C, paragraph C.5).

To extend the uncollided dose rates to 100 cm, the second dose to source point, the dose rates for 19 cm were used. Using the $1/r^2$ relationship, each dose at 19 cm was multiplied by $\frac{(19)^2}{(100)^2} = 3.61 \times 10^{-2}$ and the corresponding dose at 100 cm determined.

The calculated dose rates are presented in Appendix A.

3.3 EXPERIMENTAL

A detailed description of the experimental apparatus and experiments is presented in Appendix B and the results in Appendix A. Tables A-1 through A-5 present the ratio of calculated to measured dose

rates for the various source strengths, absorber materials, etc.

Five separate TLD dosimeters were used to absorb the integrated dose in each experimental exposure ⁽¹¹⁾. Subsequently, each of the dosimeter LiF powder charges was loaded onto three heating elements so that a total of 15 readings of integrated dose was obtained. These data were then averaged and statistically analyzed. A sample case is given in Appendix B, paragraph B.5 to show how the measured data is averaged and analyzed.

After average measured doses and probable errors were obtained, the background doses were subtracted. The net dose rates (rad/hr) were then determined by dividing the net doses (rad) by the exposure time in hours. These values are the net dose rates at the time of measurement. Correction for decay must be made to obtain the dose rate at initial activity. All dose rates were normalized to August 2, 1968 (date of assay by calorimetric techniques) by the following equation:

$$D = D_m / e^{-\lambda t}$$

where D = normalized dose rate (August 2, 1968)

D_m = dose rate at the time of measurement

t = number of days between August 2, 1968 and the time of measurement

λ = decay constant = $0.693/T_{1/2}$

$T_{1/2}$ = half-life of ^{170}Tm = 128 days

A decay curve was plotted ($e^{-\lambda t}$ vs. t) and the decay correction factors were obtained from the curve. The normalized dose rates were then determined by dividing the net dose rates at the time of measurement by this decay correction factor.

For the uranium absorber, another correction is required because uranium is itself radioactive. Experiments were performed (as shown in Appendix B, paragraph B.6) with three uranium absorbers to obtain

the uranium dose rates. These results were then subtracted from the normalized dose rates.

The overall uncertainty for the measured data is approximately $\pm 9\%$.

3.4 COMPARISON OF RESULTS

Figures 3-2 through 3-6 show the comparison between analytical and experimental dose rates. The experimental results differ from the analytical predictions in two ways. One, the theoretical dose rates are generally greater for a given source-to-dose point distance than the actual dose rates and two, the measured dose rates at 100 cm are not those expected if the inverse square law were operating.

In the first case, the differences are generally outside the experimental error ($\pm 9\%$) and are largely due to the use of B_∞ in the analytical predictions. The error results, because the uncertainty in the calculated values is very high and for the following reasons:

- A build-up factor for an infinite medium has been assumed.
- No data were available to assess B_∞ for energies less than 0.5 Mev.
- No data are available for a $\mu t < 1$.

Although the last source of error might be small, the curves of Figures C-7 and C-8 clearly show the difficulty in extrapolating from the 0.5 Mev point back to zero and the resulting large uncertainty that exists, especially for an energy grouping that would be the most contributory. This uncertainty is also clearly greater for aluminum and stainless steel than for the high Z materials.

Further, the assumption of a build-up factor for an infinite medium will overestimate the dose rate to an extent that depends on the photon energy, absorbing medium and the geometry. As discussed in Appendix C, paragraph C.5, the dose rates for aluminum and stainless steel might be overestimated by as much as 50% whereas that of the heavy materials only 10 to 20%.

The second difference is the departure from the inverse square law. The doses from an unshielded source followed well the $1/r^2$ re-

Figure 3-2. Comparison of Computed and Experimental Dose Rate Data for Thulium -170 using Aluminum Absorbers

Figure 3-3. Comparison of Computed and Experimental Dose Rate Data for Thulium-170 using Stainless Steel Absorbers

Figure 3-4. Comparison of Computed and Experimental Dose Rate Data for Thulium-170 using Lead Absorbers

Figure 3-5. Comparison of Computed and Experimental Dose Rate Data for Thulium-170 using Tungsten Absorbers

Figure 3-6. Comparison of Computed and Experimental Dose Rate Data for Thulium-170 using Uranium Absorbers

lationship and therefore, the difference must lie with the absorber.

The reason for the deviation can be seen by again considering the scattered component of the dose rate. In the calculation of total dose rate, build-up factor for infinite medium (B_{∞}) has been used. This B_{∞} is obtained by assuming both the radiation source and the dose point are imbedded in an infinite homogeneous medium. For 19 cm geometry, the dose point is very close to the shield surface (~2 to 3 cm), while for 100 cm geometry, the separation distance between the shield surface and the dose point is approximately 80 cm. The solid angle subtended by the detector is nearly 2π for 19 cm geometry while it is much smaller for 100 cm geometry. Therefore, compared to the 100 cm geometry, the 19 cm geometry more closely approximates the infinite medium case. Consequently, B_{∞} is more applicable to dose point located close to the shield surface. If B_{∞} is applied to dose point far away from shield surface, it overestimates the total dose. The degree of overestimation depends on the shielding material and its thickness, or more specifically that scattering for 100 cm is a different case than scattering for 19 cm. Because of the complexity of the problem, it is very difficult to evaluate analytically. Therefore, to extend the total dose rate correctly from 19 cm to 100 cm, experimental ratios (D_{100}/D_{19}) have to be used. These ratios are listed in Table B-9. (see Appendix B, paragraph B.6.2). For practical shielding design, this overestimation increases the margin of safety.

3.5 EXTENSION TO LARGE SCALE APPLICATIONS

The sources used in this study had a maximum power of 20 watts. In actuality, sources in the kilowatt range would be more the rule. We will extend the results then to larger thulium-170 sources and will discuss only lead, tungsten and uranium shield materials as aluminum and stainless steel only attenuate radiation in a relatively minor way.

A change in source strength only means an increase in volumetric source strength. Equations show that the dose rates are

increased proportionally with the volumetric source strength. However, a change of this nature without change in source dimension and source geometry is unlikely and these changes may result in considerable difference in resultant dose rates.

Since the gamma spectrum for thulium-170 consists of mainly low energy photons, the self-absorption becomes important. Calculations must be made to re-evaluate the dose rates. The methods have been described in Appendix C and the reader is cautioned against using less than 22 energy groups in the analysis as dose rates can be increased by as much as a factor of 2 if only four groups are used.

B_{∞} should be used to obtain the dose rates. This will overestimate the dose rates by 10 to 20% but will result in only a 2 to 3% overestimation in shielding thickness (based on a 1 kilowatt thulium-170 source). This conservative margin is desirable because an uncertainty of 3 to 5% is involved in the absorption coefficients.

Axial dose rates were used throughout this study. The readers are directed to the work of Arnold⁽⁸⁾ for radial dose rate information.

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APPENDIX A

EXPERIMENTAL DATA

TABLE A-1
DATA SUMMARY FOR ALUMINUM ABSORBERS

Initial Source Strength (watts)	Absorber Thickness (inches)	Exposure Time (hours)	Measured Dose (rads)	Net Dose (rads)	Net Dose Rate (rads/hr)	Measured* Dose Rate (rads/hr)	Calculated Dose Rate (rads/hr)	Ratio Calculated to Measured
<u>1. 19 cm</u>								
20.7	1/4	0.25	21.7 ± .3	21.7	86.8	178 ± 2	161	0.90
	1/2	0.25	20.9 ± .2	20.9	83.6	171 ± 2	160	0.94
	1	0.25	16.2 ± .2	16.2	64.7	132 ± 2	156	1.18
10	1/4	0.25	17.8 ± .3	17.8	71.0	141 ± 2	119	0.84
	1/2	0.25	14.3 ± .1	14.2	57.0	113 ± 1	118	1.04
	1	0.25	12.5 ± .1	12.4	49.7	98.5 ± .9	115	1.17
5	1/4	0.25	11.7 ± .1	11.7	46.8	94.7 ± 1	81.6	0.86
	1/2	0.25	9.10 ± .03	9.06	36.2	73.2 ± .3	81.1	1.11
	1	0.25	6.50 ± .04	6.46	25.8	52.2 ± .3	79.0	1.51
<u>2. 100 cm</u>								
20.7	1/4	0.5	2.18 ± .01	2.05	4.10	5.70 ± .03	5.80	1.02
	1/2	0.5	1.86 ± .01	1.74	3.47	4.82 ± .02	5.80	1.20
	1	0.5	1.35 ± .01	1.23	2.46	3.42 ± .02	5.60	1.64
10	1/4	0.5	1.22 ± .01	1.19	2.37	4.70 ± .03	4.30	0.92
	1/2	0.5	1.06 ± .01	1.03	2.05	4.06 ± .01	4.30	1.06
	1	0.5	.697 ± .003	.668	1.34	2.64 ± .01	4.20	1.59
5	1/4	1	2.17 ± .01	2.05	2.05	2.58 ± .01	2.96	1.15
	1/2	1	1.81 ± .01	1.69	1.69	2.13 ± .01	2.92	1.37
	1	4	4.77 ± .02	4.65	1.16	1.47 ± .01	2.85	1.94

*Dose rate is normalized to correspond to the initial source strength.

TABLE A-2

DATA SUMMARY FOR STAINLESS STEEL ABSORBERS

Initial Source Strength (watts)	Absorber Thickness (inches)	Exposure Time (hours)	Measured Dose (rads)	Net Dose (rads)	Net Dose Rate (rads/hr)	Measured* Dose Rate (rads/hr)	Calculated Dose Rate (rads/hr)	Ratio Calculated to Measured
<u>1. 19 cm</u>								
20.7	$\frac{1}{4}$	.25	16.1 $\pm$.2	16.1	64.2	131 $\pm$ 2	132	1.01
	$\frac{1}{2}$	.25	9.01 $\pm$.05	8.99	36.0	73.6 $\pm$.4	102	1.38
	1	.25	3.95 $\pm$.03	3.94	15.7	32.2 $\pm$.2	59	1.83
10	$\frac{1}{4}$	.25	10.80 $\pm$.05	10.7	43	87.6 $\pm$.4	95.4	1.09
	$\frac{1}{2}$	.25	6.81 $\pm$.05	6.80	27.2	55.4 $\pm$.4	73.4	1.32
	1	.25	3.08 $\pm$.04	3.07	12	24.4 $\pm$.3	41.2	1.68
5	$\frac{1}{4}$	.25	7.15 $\pm$.05	7.11	28.4	57.2 $\pm$.4	63.6	1.11
	$\frac{1}{2}$	.25	4.35 $\pm$.04	4.32	17.3	34.8 $\pm$.3	48.1	1.38
	1	.25	1.90 $\pm$.02	1.86	7.45	15.0 $\pm$.2	26.3	1.75
<u>2. 100 cm</u>								
20.7	$\frac{1}{4}$	.5	1.66 $\pm$.01	1.53	3.06	4.20 $\pm$.02	4.77	1.13
	$\frac{1}{2}$	.5	.996 $\pm$.003	.869	1.74	2.38 $\pm$.01	3.72	1.56
	1	.5	.460 $\pm$.001	.333	.666	.911 $\pm$.004	2.13	2.34
10	$\frac{1}{4}$	.75	1.26 $\pm$.01	1.24	1.65	3.25 $\pm$.01	3.43	1.05
	$\frac{1}{2}$	.75	.717 $\pm$.002	.689	.918	1.81 $\pm$.01	2.64	1.46
	1	1.5	.492 $\pm$.002	.464	.31	.610 $\pm$.002	1.48	2.42
5	$\frac{1}{4}$	1.5	2.01 $\pm$.01	1.89	1.26	1.62 $\pm$.01	2.30	1.42
	$\frac{1}{2}$	1.5	1.16 $\pm$.01	1.04	.695	.862 $\pm$.005	1.73	1.94
	1	1.5	.482 $\pm$.003	.366	.244	.313 $\pm$.002	0.94	3.00

*Dose rate is normalized to correspond to the initial source strength.

TABLE A-3

DATA SUMMARY FOR LEAD ABSORBERS

Initial Source Strength (watts)	Absorber Thickness (inches)	Exposure Time (hours)	Measured Dose (rads)	Net Dose (rads)	Net Dose Rate (rads/hr)	Measured* Dose Rate (rads/hr)	Calculated Dose Rate (rads/hr)	Ratio Calculated to Measured
<u>1. 19 cm</u>								
20.7	1/4	.25	3.09 $\pm$.02	3.07	12.3	25.2 $\pm$.2	27.3	1.08
	1/2	.25	.850 $\pm$.003	.837	3.34	6.85 $\pm$.02	8.72	1.27
	1	1.5	.751 $\pm$.004	.738	.492	1.01 $\pm$.01	1.32	1.31
10	1/4	.25	1.96 $\pm$.03	1.94	7.77	15.8 $\pm$.2	17.2	1.09
	1/2	.5	1.23 $\pm$.01	1.21	2.42	4.93 $\pm$.03	5.27	1.07
	1	3	.973 $\pm$.003	.956	.319	.650 $\pm$.002	.774	1.19
5	1/4	.25	1.24 $\pm$.01	1.20	4.79	9.65 $\pm$.11	9.97	1.03
	1/2	.5	.708 $\pm$.003	.671	1.34	2.71 $\pm$.01	2.98	1.1
	1	3.5	.735 $\pm$.003	.698	.200	.402 $\pm$.002	.43	1.07
<u>2. 100 cm</u>								
20.7	1/4	1	.772 $\pm$.003	.643	.643	.886 $\pm$.004	.970	1.10
	1/2	2	.473 $\pm$.001	.344	.172	.237 $\pm$.001	.310	1.31
	1	15	.499 $\pm$.002	.370	.0247	.0341 $\pm$.0002	.047	1.38
10	1/4	1	.486 $\pm$.002	.37	.37	.493 $\pm$.002	.614	1.24
	1/2	3	.420 $\pm$.001	.304	.101	.135 $\pm$.001	.190	1.41
	1	20	.376 $\pm$.001	.26	.013	.0174 $\pm$.0001	.028	1.61
5	1/4	4	1.22 $\pm$.01	1.07	.268	.348 $\pm$.001	.360	1.04
	1/2	15	.959 $\pm$.004	.814	.0542	.0705 $\pm$.0003	.110	1.56
	1	74	.660 $\pm$.004	.515	.00696	.00916 $\pm$.00006	.0155	1.70

*Dose rate is normalized to correspond to the initial source strength.

TABLE A-4

DATA SUMMARY FOR TUNGSTEN ABSORBERS

Initial Source Strength (watts)	Absorber Thickness (inches)	Exposure Time (hours)	Measured Dose (rads)	Net Dose (rads)	Net Dose Rate (rads/hr)	Measured* Dose Rate (rads/hr)	Calculated Dose Rate (rads/hr)	Ratio Calculated to Measured
<u>1. 19 cm</u>								
20.7	$\frac{1}{8}$	.25	1.93 $\pm$.02	1.90	7.61	14.9 $\pm$.2	16.6	1.11
	$\frac{1}{2}$	.5	.802 $\pm$.004	.78	1.56	3.06 $\pm$.02	3.7	1.21
	1	6.25	.842 $\pm$.005	.82	.131	.257 $\pm$.002	.288	1.12
10	$\frac{1}{8}$	.25	1.22 $\pm$.01	1.18	4.73	9.22 $\pm$.1	10.2	1.11
	$\frac{1}{2}$	.75	.755 $\pm$.003	.721	.96	1.87 $\pm$.01	2.21	1.18
	1	22.5	1.65 $\pm$.02	1.62	.072	.139 $\pm$.002	.167	1.19
5	$\frac{1}{8}$	1	2.62 $\pm$.03	2.6	2.6	5.04 $\pm$.05	5.85	1.16
	$\frac{1}{2}$	2	1.10 $\pm$.01	1.08	.539	1.05 $\pm$.01	1.23	1.17
	1	18	.765 $\pm$.006	.744	.0413	.0801 $\pm$.0006	.092	1.15
<u>2. 100 cm</u>								
20.7	$\frac{1}{8}$	1.25	.380 $\pm$.002	.311	.249	.433 $\pm$.003	.600	1.38
	$\frac{1}{2}$	4.5	.274 $\pm$.002	.205	.0456	.0793 $\pm$.0006	.134	1.69
	1	72.5	.299 $\pm$.002	.230	.00318	.00559 $\pm$.00003	.010	1.79
10	$\frac{1}{8}$	2	.362 $\pm$.002	.303	.152	.256 $\pm$.001	.370	1.45
	$\frac{1}{2}$	7	.257 $\pm$.001	.198	.0283	.0478 $\pm$.0002	.079	1.65
	1	120	.286 $\pm$.001	.227	.0019	.00325 $\pm$.00004	.006	1.85
5	$\frac{1}{8}$	3	.419 $\pm$.002	.268	.0893	.144 $\pm$.001	.210	1.46
	$\frac{1}{2}$	18	.451 $\pm$.005	.3	.0167	.0269 $\pm$.0005	.044	1.63
	1	168	.350 $\pm$.001	.199	.00118	.00194 $\pm$.00002	.0033	1.70

*Dose rate is normalized to correspond to the initial source strength.

A-5

TABLE A-5

DATA SUMMARY FOR DEPLETED URANIUM ABSORBERS

Initial Source Strength (watts)	Absorber Thickness (inches)	Exposure Time (hours)	Measured Dose (rads)	Net Dose (rads)	Net Dose Rate (rads/hr)	Measured*+ Dose Rate (rads/hr)	Calculated Dose Rate (rads/hr)	Ratio Calculated to Measured
<u>1. 19 cm</u>								
20.7	$\frac{1}{8}$	1	3.12 $\pm$.03	3.03	3.03	5.61 $\pm$.06	7.71	1.37
	$\frac{1}{2}$	2	1.09 $\pm$.01	1.01	.503	.888 $\pm$.009	1.17	1.32
	1	48	3.84 $\pm$.02	3.76	.0704	.0548 $\pm$.001	.05	0.91
10	$\frac{1}{8}$	1	1.96 $\pm$.01	1.88	1.875	3.38 $\pm$.02	4.65	1.37
	$\frac{1}{2}$	3	1.01 $\pm$.01	.927	.309	.513 $\pm$.005	.68	1.32
	1	88	5.97 $\pm$.04	5.89	.067	.0318 $\pm$.001	.029	0.91
5	$\frac{1}{8}$	1	1.23 $\pm$.02	1.15	1.15	1.98 $\pm$.03	2.62	1.32
	$\frac{1}{2}$	4	.932 $\pm$.004	.848	.212	.324 $\pm$.002	.397	1.23
	1	108	6.98 $\pm$.05	6.89	.0636	.0226 $\pm$.001	.016	0.71
<u>2. 100 cm</u>								
20.7	$\frac{1}{8}$	2	.422 $\pm$.001	.262	.141	.218 $\pm$.001	.280	1.28
	$\frac{1}{2}$	20	.511 $\pm$.001	.371	.0185	.0286 $\pm$.0001	.042	1.47
	1	169	.406 $\pm$.001	.226	.00157	.0025 $\pm$.00002	.0018	0.72
10	$\frac{1}{8}$	3	.499 $\pm$.002	.356	.119	.165 $\pm$.001	.170	1.03
	$\frac{1}{2}$	24	.441 $\pm$.002	.298	.0124	.0174 $\pm$.0001	.025	1.44
	1	168	.345 $\pm$.001	.202	.0012	.00172 $\pm$.00001	.001	0.58
5	$\frac{1}{8}$	6	.437 $\pm$.001	.298	.05	.0733 $\pm$.0002	.094	1.28
	$\frac{1}{2}$	65	.554 $\pm$.001	.415	.00638	.00946 $\pm$.00003	.014	1.48
	1	168	.241 $\pm$.001	.102	.0006	.00091 $\pm$.000006	.00053	0.64

*Dose rate is normalized to correspond to the initial source strength.

+Uranium background is subtracted.

APPENDIX B

EXPERIMENTAL APPARATUS, PROCEDURES AND RESULTS

B.1 DESCRIPTION AND CALIBRATION OF THULIUM OXIDE SOURCE

The individual radiation sources used in these measurements consisted of doubly encapsulated stacks of thulium-170 oxide wafers as depicted in Figure B-1. Both the exterior capsule and liner were Hastelloy C.

The thermal and radiation strengths of the sources are given in Table B-1. These thermal strengths were determined calorimetrically on 2 August 1968 by ORNL⁽¹⁰⁾. The calorimeter and the procedures used are described in references 17 and 18. The calorimeter was calibrated in a standard cell just prior to use. The calibration was made using a precision heater and a calibrated wattmeter and recording the thermocouple potentiometer reading as a function of power input to the heater. The same potentiometer and standard cell were then used to measure the heat output of the radiation sources. Since the calorimeter used was a less refined model than the type described in the references, the accuracy is estimated at 2% at the 95% confidence level.

One irradiated wafer from these sources was dissolved by ORNL for analysis and assumed to be representative of the batch. A mass spectrographic analysis made on 10 July 1968 gave the following thulium isotopic distribution: Tm-169 (76.8%), Tm-170 (17.7%), Tm-171 (5.5%). A chemical analysis of the solution performed on 25 June 1968 determined the ytterbium content to be 17% by weight.

TABLE B-1

THERMAL AND RADIATION STRENGTHS OF THULIUM-170 OXIDE

<u>Source</u>	<u>Thermal Strength (watts)</u>	<u>Radiation Strength* (curies)</u>
No. 1	5.0 $\pm$ 0.1	2225 $\pm$ 119
No. 2	10.0 $\pm$ 0.1	4450 $\pm$ 225
No. 3	20.7 $\pm$ 0.1	9211 $\pm$ 455

*Conversion factor = 445 $\pm$ 22 curies/watt⁽⁹⁾

Figure B-1. Representative Thulium Source and Capsule

The physical characteristics of individual Tm_2O_3 wafers used in the sources are described in Table B-2. The distribution of these wafers in the sources was as follows: Source No. 1 (Wafer No. 6), Source No. 2 (Wafers No. 5, 8), and Source No. 3 (Wafers No. 1, 2, 3, 7). Table B-3 gives the precise dimensions (in inches) of the exterior capsules and their liners.

TABLE B-2

PHYSICAL CHARACTERISTICS OF THULIUM-170 OXIDE WAFERS

<u>Wafer No.</u>	<u>Weight (gms)</u>	<u>Diameter (inches)</u>	<u>Thickness (mils)</u>	<u>Theoretical Density (%)</u>
1	4.8768	0.736	98.5	80.0
2	4.8500	0.730	100.1	79.6
3	5.0450	0.732	99.9	83.7
5	4.5846	0.742	95.5	76.2
6	4.8380	0.735	99.4	78.7
7	4.9690	0.738	100.0	79.9
8	4.9243	0.734	99.5	79.7

TABLE B-3

DIMENSIONS OF THULIUM-170 OXIDE SOURCES (INCHES)

<u>Capsule No.</u>	<u>Bottom Thickness</u>	<u>Wall Thickness</u>	<u>Capsule OD</u>	<u>Capsule ID</u>	<u>Capsule Height</u>
<u>Exterior Capsule</u>					
1	0.0513-0.0518	0.0330-0.0352	0.9210	0.8520	0.9600
2	0.0511-0.0514	0.0327-0.0340	0.9200	0.8530	0.9600
3	0.0512-0.0515	0.0326-0.0335	0.9190	0.8525	0.9590
<u>Capsule Liners</u>					
1	0.0495-0.0503	0.0388-0.0405	0.842	0.760	0.6715
2	0.0495-0.0505	0.0365-0.0380	0.844	0.7674	0.6700
3	0.0495-0.0505	0.0385-0.0405	0.842	0.761	0.6684

B.2 EXPERIMENTAL APPARATUS

The equipment used in this experiment to perform thulium-170 dose rate measurements is shown in Figures B-2 through B-5. They are also schematically depicted in Figure B-6. Figure B-2 is an axial photograph of the measurement fixture showing the source shield cask with the beam tunnel shutter open and the one-quarter inch lead absorber in the absorber holder. Also shown is the radiation scatter shield (see Appendix C, paragraph C.7) and the radiation dosimeters in their holder at the one meter irradiation position. Figure B-3 is a disassembled view of the source shield cask and the absorber holder showing the shield cask plug and the aluminum source holder. Also shown is a model fuel capsule. Figure B-4 is a top view of the partially assembled source shield cask and absorber holder showing the source holder in position inside the cask. Figure B-5 shows the thermoluminescent dosimetry (TLD) reader (Con-Rad Model 5100B) used in the experiment to evaluate the dosimeters. Included is a dosimeter holder loaded with five black polyethylene dosimeter capsules and the vibratory instrument (see paragraph B.3 for a detailed discussion) which dispenses precise quantities of TLD phosphor powder (LiF) into them.

The measurement fixture consisted of a rectangular base platform with the following mounted upon it: (1) a shielded enclosure for the radiation sources, (2) a fixed holder for the various absorber specimens and (3) a second fixed holder for the thermoluminescent dosimeters. This fixture is schematically depicted in Figure B-7. The shield enclosure incorporated lead walls of such thickness (approximately 3 inches) that with the strongest anticipated thulium-170 source contained within it, the dose rate at one meter from the enclosure was no greater than 10 mr/hr. In addition, the enclosure had a removeable lead cover over the source cavity of likewise appropriate thickness. The cavity itself was so constructed and aligned with a radiation beam "tunnel" that with the thulium capsule in place, only the axial radiation emission was allowed to exit through the tunnel and be incident upon the absorber

68-1198-5
 Figure B-2. Dose Rate Measure-
 ment Fixture

68-1198-4
 Figure B-3. Source Shield
 Cask

68-1198-3
 Figure B-4. Top View of a Par-
 tially Assembled
 Source Shield Cask

68-1198-2
 Figure B-5. Thermoluminescent
 Dosimetry (TLD)
 Reader

Figure B-6. Experimental Apparatus

B-8

Figure B-7. Dose Rate Measurements Schematic Arrangement

specimen. The source shield enclosure and the absorber holder are depicted in Figure B-8. Both the absorber and dosimeter holders were axially aligned with the source capsule and beam tunnel centerline. In addition, the dosimeter holder (shown in Figure B-9) was located so that the dosimeters could be positioned at various distances from the vertical source centerline.

B.3 THERMOLUMINESCENT DOSIMETRY (TLD) SYSTEM

B.3.1 THERMOLUMINESCENT PHOSPHOR

The thermoluminescent phosphor, TLD-700, is lithium fluoride (LiF) powder which has been refined so that it is essentially free of the Lithium-6 isotope (i.e., it consists almost entirely of Lithium-7). The range of radiation dosage measurable with this phosphor extends from 10 mrad to 100,000 rad for gamma, x-ray, beta and electron radiation. The radiation dose measurement precision attainable with this phosphor as a function of dose is given in Table B-4⁽¹⁹⁾. The overall accuracy of dose measurement depends not only upon dosimeter precision, but also upon the accuracy of calibration of the readout instrument and dosimeter system. The expected overall accuracy was determined to be $\pm 9\%$ for this experiment, based on large dose measurement (>1 rad), as discussed in paragraph B.3.3.

The response of LiF phosphor vs photon energy is shown in Figure B-10. For all practical purposes, the response is independent of incident photon energy down to about 100 keV.

B.3.2 TLD READOUT SYSTEM

The thermoluminescence dosimetry system used in obtaining the radiation dose rates in this experiment consisted of the following:

- Con-Rad Model 5100B Readout Instrument.
- Con-Rad Model PDAV Vibratory Phosphor Dispenser.
- Thermoluminescent Dosimetry Powder Type TLD-700 (LiF-7).

In the operation of the readout instrument, the phosphor

B-10

Figure B-8. Source Shield Enclosure Arrangement

Figure B-9. Dosimeter Mounting Arrangement

B-12

Figure B-10. Response of Con-Rad Lithium Fluoride TLD Dosimeters to Radiation Energy

TABLE B-4
MEASUREMENT PRECISION OF LiF PHOSPHOR

<u>Measured Dose</u>	<u>Measurement Precision</u>
10 mrad	$\pm$ 20% S.D.
100 mrad	$\pm$ 5% S.D.
$1-10^5$ rad	$\pm$ 2% S.D.

sample or dosimeter is heated and the amount of light emitted within a fixed time period is integrated and is presented digitally. The instrument can be adjusted in such a manner that the dose can be read directly in dose units over a broad dose range. The concept of the instrument is indicated in block diagram as shown in Figure B-11.

The system employs reusable and exchangeable heating elements. When measuring loose powder dosimeters, high-resistivity heating elements are used onto which the powder is poured. These heating elements are inserted by a slide into the instrument where the edges becomes gripped by copper-beryllium alloy contacts during heating. The Constant Current Heater Control module provides a constant current through the heating element and ensures a reproducible heating process.

A photomultiplier tube selected for a low dark current, a large range of measurable light fluxes, and a high stability is the first stage in the measuring system. A Teflon tube socket effectively prevents leakage currents. The High Voltage Power Supply provides the photomultiplier with voltage. It is continuously adjustable, and the setting can be controlled to $\pm$ 0.3 volts, which simplifies exact adjustment of the overall light sensitivity of the instrument.

The anode current of the photomultiplier is integrated on a polystyrene capacitor in the integrator subunit. The voltage on this capacitor can be divided by successive factors of 10 by pressing a scale factor switch-button. An electrometer circuit, also placed in

Figure B-11 Block Diagram of Readout Instrument

the integrator, is used as an intermediate stage when the voltage on the capacitor is measured, to provide the instrument with a long time constant in the retention of the signal. A Low Voltage Power Supply provides this part of the circuitry with the necessary voltages.

The final measurement of the collected charge from the photo-multiplier and the display of the result is made with a Digital Voltmeter.

The readout cycle is controlled by a synchronous motor drive Cam Timer, which is programmed to operate a series of relays once the "Start Bar" is actuated. This feature makes routine operation simple and reliable.

A Regulating Transformer compensates effects of primary voltage fluctuations on all components of the apparatus. Among other things, it assists in keeping the heating current constant.

For measurement of small doses, the instrument is equipped with a gas inlet so that the readout chamber can be filled with appropriate gas, normally prepurified nitrogen. The use of this feature makes it possible to measure with LiF down to 10 mRad with a standard error of less than $\pm 20\%$. Among the accessory equipment is a recorder for glow curve registration, which can be attached without disturbance on the normal integration procedure.

B.3.3 TLD READOUT INSTRUMENT CALIBRATION

Field calibration of Con-Rad TLD Readout Instruments was performed by means of two light sources provided with the instrument. These sources were used to set the instrument sensitivity for dosimetry phosphor obtained from Con-Rad. They were fabricated to emit the same light intensity within $\pm 2\%$ of their corresponding standards at the Con-Rad Laboratory. The TLD phosphor obtained for the dose rate measurements along with the instrument was accompanied by an Instrument Calibration Form which provided the recommended light source readings to permit calibration of the instrument so that readout sensitivities of 0.001, 0.10 and 10 Rad per digit could be obtained with the phosphor. The initial calibration procedure employed upon receipt of the instrument is given below.

The instrument was energized and permitted to warm up for two hours to achieve stability. Then, light source A was inserted into the readout chamber and output readings taken for photomultiplier high voltages of 400, 600, 800 and 1000 volts. These readings were found to be in excellent agreement with corresponding curves provided by the manufacturer for the instrument. Similarly, light source B was next inserted and readout values obtained at 1000, 1200, 1400, 1600 and 1800 volts. Again, the agreement with the specification curves was very good. Also, as part of these readings, stability of the zero point of the digital voltmeter, as well as the light source readout values, was confirmed, thereby indicating instrument stability. Finally, the instrument was calibrated to obtain readout sensitivity of 0.001 Rad per digit by reinsertion of light source B and adjustment of the photomultiplier high voltage to obtain digital readout of 185, as specified on the Instrument Calibration Form. Subsequently, 10 additional readings were performed and the average value confirmed to be the required value. In addition, the standard deviation of these readouts was calculated and confirmed as not exceeding $\pm 1\%$, thereby assuring reproducibility of the measurements.

Although this light source technique of readout instrument calibration is simple and effective, the measurement accuracy obtained thereby is limited by the following factors:

- The light source readings for the desired sensitivity are valid only for the particular batch of phosphor with which they are supplied.
- The readings are valid only for dosimeter phosphor which has not been annealed (for annealed powder, see paragraph B.3.4)
- The calibration is based upon exposures to Cobalt-60 gamma radiation. For photon energies less than about 100 keV, corrections should be applied for phosphor energy dependence.
- The fundamental accuracy of this calibration techniques is determined by:

- a. The accuracy of calibration by Con-Rad of batches of dosimetry phosphor which is about $\pm 4\%$.
- b. The accuracy of the light source used compared with that of the light source standard maintained at Con-Rad Laboratories to establish dosimeter sensitivities which is $\pm 2\%$.
- c. Inaccuracies created by the influence of ambient temperature on photomultiplier spectral sensitivity, variations in positioning of light source and in setting of heater current in both Con-Rad Laboratory and field instruments which should contribute less than $\pm 3\%$

As a result of these sources of error, an overall accuracy of about $\pm 9\%$ or better was expected for the thulium-170 dose rate measurements subsequently performed.

B.3.4 TLD PHOSPHOR CALIBRATION

As mentioned previously, the accuracy of the TLD system calibration is partially limited by the fact that after the phosphor has been used and subsequently annealed, its sensitivity to radiation changes. Consequently, repeated exposures of a given quantity of phosphor to identical irradiation configurations would yield varying dose indications. Accordingly, since it was anticipated that it would be necessary to conduct several annealing operations during the course of these dose rate measurements, a small quantity of unirradiated phosphor was initially set aside as a control. After each annealing operation, the zero dose reading of a small sample of "clean" phosphor was compared with that of the freshly annealed stock. This assured that annealing had been complete.

B.3.5 TLD POWDER DISPENSER CALIBRATION

An investigation was conducted regarding the reproducibility of

measurement of thermoluminescent dosimetry (TLD) phosphor powder into the dosimeter capsules being used in the dose rate measurement exposures. The instrument being used to measure calibrated quantities of TLD phosphor (LiF-7) into the capsules is the Con-Rad Phosphor Dispenser - Vibrator (Model PDAV-1). (See Figure B-12). This instrument is claimed to measure 43.7 milligrams of phosphor with a reproducibility of ± 0.1 percent standard deviation. The dispenser consists of a horizontal cylindrical plunger with a small vertical hole, i.e., a metering volume, normal to its axis. This plunger is closely fitted into a hole in a block of dissimilar metal which forms the dispenser body. The top of the body is machined to provide a reservoir for the powder incorporating a conical bottom with a small outlet hole which is aligned with the plunger metering volume. When the plunger is pushed, the metered volume is moved under the reservoir. A simultaneously actuated vibrator assures thorough packing of a calibrated quantity of phosphor into the cavity. After a certain vibratory packing interval, the vibrator ceases and the plunger is then pulled out against its stop whereupon the metering volume is aligned with the dispenser tube under which is placed the receptacle for the powder. Again, the vibrator is actuated during the dispensing cycle ensuring thorough removal of the calibrated phosphor quantity.

To check the reproducibility of this instrument under actual conditions, a series of 30 measurements were taken of the weights of clean, dry phosphor powder dispensed each time. These measurements were made with an Ainsworth analytical balance (Model 21 N). Table B-5 presents the measured data along with associated statistical evaluation calculations. The net result of these measurements is that the average weight of phosphor dispensed by the instrument is:

$$\text{average weight} = 43.89 \pm 0.22\% \text{ S.D. mg.}$$

This value appeared in good agreement with the manufacturer's claimed reproducibility of phosphor measurement, and was therefore assumed to be satisfactory. Subsequent examination of the literature (20, 21) also revealed that owing to this comparatively large nominal phosphor

Figure B-12. Phosphor Dispenser Vibrator

TABLE B-5

REPRODUCIBILITY OF TLD PHOSPHOR DISPENSER VIBRATOR

Experimental Data

<u>Test No.</u>	<u>Sample Weight (mg)</u>	<u>Test No.</u>	<u>Sample Weight (mg)</u>
1	43.9	16	43.9
2	44.0	17	44.1
3	44.0	18	43.8
4	43.9	19	43.9
5	43.9	20	43.9
6	44.0	21	43.8
7	43.9	22	43.7
8	43.9	23	43.8
9	43.8	24	43.7
10	43.8	25	43.9
11	43.8	26	43.9
12	44.0	27	43.9
13	44.0	28	43.8
14	44.1	29	43.9
15	43.9	30	43.9

Statistical Analysis of Data

$$\bar{X} = 43.89$$

$$\sum (x - \bar{X})^2 = 0.279$$

$$\sigma = \sqrt{\frac{1}{N-1} \sum (x - \bar{X})^2} = 0.098$$

$$\text{or } \sigma = \pm \frac{0.098}{43.89} \times 100 = \pm 0.22\%$$

weight per dosimeter (i.e., 43.7 mg) the slightly higher experimental value obtained herein represented only a few additional grains of phosphor per dosimeter and any radiation dose measurement error introduced thereby is insignificant.

B.4 EXPERIMENTAL PROCEDURES*

The radiation dose rates from thulium-170 sources through lead, stainless steel, tungsten, aluminum and depleted uranium absorbers were determined by measurement of the luminescence of lithium fluoride phosphors in a TLD readout instrument. The energy imparted to this material as a result of interactions with incident radiation is released as luminescence at elevated temperatures. The particular dosimeter used for these measurements consisted of precise quantities of LiF powder metered into special holders. Three thulium-170 sources in strengths of 5, 10 and 20.7 thermal watts were utilized. Three different thicknesses ($\frac{1}{4}$, $\frac{1}{2}$ and 1 inch) of each shield material (lead, tungsten, stainless steel, aluminum and depleted uranium) were used. The dose rate measurements were made at 19 and 100 cm from the source. Each measurement utilized five separate dosimeters in accordance with the matrix outlined in Table B-6. The use of five separate TLD dosimeters for each measurement assured high reliability of test results.

After exposure, the thermoluminescent dosimeters were read by placing them in a specially designed heating device which raised them to a specified temperature at which light is emitted proportional to the amount of radiation absorbed by the dosimeter. This light emission is detected by a photomultiplier tube, the integrated output

*A complete test procedure: (SNP100017) may be found in SNC-3693-5, Part III "Test Procedures for Tm_2O_3 Material Properties Study"⁽³⁶⁾

TABLE B-6

DOSE RATE MEASUREMENT MATRIX

current of which is electronically converted into a dose reading.

B.4.1 PROCEDURES PRIOR TO MEASUREMENTS

Each of the various absorbers to be used in the measurements was visually inspected for cleanliness, freedom from surface defects and blemishes. These absorbers were then arranged in special trays (equipped with hinged covers capable of being opened and closed with the hot cell remote manipulators) which were placed next to the measurement fixture inside the hot cell.

Each dosimeter to be used in the measurement experiment was visually inspected for cleanliness, freedom from surface defects and blemishes. These dosimeters were also arranged in special trays (equipped with opaque covers) in groups of eight dosimeters per tray. Three of these trays (24 dosimeters, total) were placed within easy reach of the remote manipulators for each dosimeter exposure sequence with a particular absorber material. In addition, the dosimeters were located as far as practicable from the stored radiation sources and were protected by sufficient auxiliary radiation shielding that they received minimal spurious radiation exposure during source transfer operations and during purposeful source exposure procedures.

The TLD dosimeter reader was calibrated against a vendor-supplied standard. This calibration was repeated at appropriate intervals throughout the performance of measurements.

B.4.2 DOSE RATE MEASUREMENT PROCEDURES

- a. The first radiation source to be measured was removed from its storage cask and transferred into the uncovered shield enclosure, the beam-hole shutter of which was closed. The shield enclosure cover was then replaced.
- b. The particular thickness of absorber to be utilized was removed from its storage tray and positioned in its special holder in front of the beam-hole shutter.

- c. The first tray of dosimeters to be exposed was opened and each dosimeter was individually subjected to the following procedures:
- each dosimeter was placed in position in the dosimeter holder, the dosimeter holder was then positioned in the configuration being measured.
 - the shutter in front of the radiation beam-hole was opened for a predetermined length of time.
 - the shutter was closed and the dosimeters were removed from their holder; the dosimeters were replaced into their original positions in the tray. (At all times the tray was positioned and shielded such as to minimize nonintentional exposure of dosimeters).
- d. Succeeding trays of dosimeters were opened and the sequence repeated for the successive absorber thickness until the transmitted radiation had been monitored with five dosimeters at each thickness for a total of 15 exposure readings.
- e. The radiation source was replaced into storage and the exposed dosimeters were removed from the hot cell. Additional dosimeters required for the next exposure were brought in.
- f. The source was returned to the shield enclosure and the first thickness of the next absorber material placed into the special holder. The complete exposure sequence of steps c. and d. was repeated for a total of 15 exposures, after which the procedure of step e. was repeated.
- g. The sequence described in step f. was repeated for the remaining absorber materials, resulting in a total of 75 exposures for the five absorber materials.
- h. Steps c. through f. were repeated for the remaining two sources for a total of 225 exposures.

- i. Having completed the procedures of steps c. through h. for each of the three thulium oxide sources and for all absorber specimens, these procedures were then repeated for each of the remaining geometrical configurations for a total of 450 exposures.
- j. The exposed dosimeters were inserted into the TLD reader and their absorbed doses measured. Three readings were made for each exposed dosimeter. The resulting data were recorded on data sheets.
- k. These dosimeters were annealed after reading to eliminate any thermoluminescence remaining after the irradiation and readout process. For loose LiF powder, this consists of heating it to 400°C for one hour, followed by exposure at 80°C for 24 hours.

B.5 EXPERIMENTAL DATA REDUCTION

The data obtained by experiment were statistically analyzed⁽²²⁻²⁵⁾ to yield a single representative value of absorbed dose from readings of each dosimeter. Comparison of this value of integrated dose with the exposure time in each case will yield the dose rate at a given separation distance due to a known strength of thulium-170 with a known thickness of absorber material.

To illustrate the procedures of experimental data analyses, the following example is presented:

Source #1. (5 watt at Aug. 2, 1968, 3.86 watts at the time of measurement).

Shielding Material: Pb

Shielding Thickness: 1 inch

Separation Distance: 100 cm

Exposure Time: 74 hours

The dose measurements obtained for the above case and the statistical analyses of these data are shown in Table B-7.

The total dose is 9901mRad (sum of X observed dose, 1-15).

The arithmetical average is: $\bar{X} = \frac{9901}{15} = 660.06$

The sum of the square of the deviations is 6237.68. Therefore, one standard deviation is $\sigma = \sqrt{\frac{6237.68}{15-1}} = 21.11$.

From the standard deviation, the standard error can be obtained as follows:

$$\sigma_{\bar{X}} = \frac{21.11}{\sqrt{15}} = 5.45$$

The probable error then, is

$$r_{\bar{X}} = 0.6745 \times 5.45 = 3.67$$

Applying the Chi-square test:

$$\chi^2 = \frac{6237.68}{660.06} = 9.45$$

TABLE B-7
STATISTICAL ANALYSES OF MEASURED DOSE RATES

Observation Number	Observed Dose, X (mRad)	$X - \bar{X}$	$(X - \bar{X})^2$
1	632	28.06	787.36
2	628	32.06	1027.84
3	645	15.06	226.80
4	634	16.06	257.92
5	640	20.06	402.40
6	638	22.06	486.64
7	686	25.94	672.88
8	676	15.94	254.08
9	667	6.94	48.16
10	688	27.94	780.64
11	666	5.94	35.28
12	665	4.94	24.40
13	690	29.94	896.40
14	672	11.94	142.56
15	<u>674</u>	13.94	<u>194.32</u>
TOTAL	9901		6237.68

The number of degrees of freedom is:

$$F = 15 - 1 = 14$$

Therefore, the probability P from the tables (26-27) is:

$$P = 0.8$$

With this probability value, we conclude that the distribution follows Poisson distribution.

The analyzed results are as follows:

Measured dose	=	660.06 $\pm$ 3.67 mRad/74 hours
Background	=	145.0 $\pm$ 0.7 mRad/74 hours
Net Dose	=	515.06 $\pm$ 3.71 mRad/74 hours
Dose rate	=	6.96 $\pm$ 0.05 mRad/hr (at time of measurement)
Normalized dose rate	=	9.16 $\pm$ 0.06 mRad/hr (August 2, 1968).

B.6 SUPPLEMENTARY MEASUREMENTS

B.6.1 HALF-LIFE DETERMINATION

A half-life determination was made to show that the measurement system works properly by comparing the measured decay rate to the expected decay rate. The half-life of Tm-170 is reported as 128 days. Any significant deviation from this expected decay rate would indicate some possible faults in the measurement technique.

Unshielded dose rates at 19 cm were taken at different time intervals. The results are shown in Table B-8. Note that the two decay rates are in good agreement within the experimental error ($\sim 10\%$). Figure B-13 shows the observed and expected decay rates for all three Tm-170 sources.

B.6.2 RATIO OF DOSE AT 19 CM TO 100 CM SEPARATION DISTANCES

As discussed in Appendix C, paragraph C.4, the inverse square relationship used to extend doses at 19 cm separation distance to 100 cm only holds for the unshielded case. For the shielded case the deviation from inverse square depends on the shielding material

TABLE B-8
UNSHIELDED DOSE RATES

		<u>Unshielded dose rates (rad/hr) at 100 cm</u>		
<u>Power (watts)</u>		<u>Sept. 27, 1968</u>	<u>Dec. 5, 1968</u>	<u>Dec. 13 1968</u>
20.7	measured	5115	3670	3365
	expected*	-	3459	3063
	ratio	-	1.06	1.1
10	measured	3677	2640	2420
	expected*	-	2434	2148
	ratio	-	1.08	1.12
5	measured	2318	1665	1525
	expected*	-	1566	1448
	ratio	-	1.06	1.05
Time from September 27, 1968		-	61 days	77 days
decay correction factor		-	0.7185	0.6585

*Expected dose rate = $\frac{\text{measure dose rate at September 27, 1968}}{\text{decay correction factor}}$

Figure B-13. Decay Rate for Tm-170 Sources

and its thickness. To extend the results from 19 cm separation distance to 100 cm experimental ratios must be used.

To obtain the experimental ratio, another dosimeter capsule holder was fabricated to permit simultaneous exposures at both 19 cm and 100 cm dose points, the 20.7-watt source (initial strength) was used as the radiation source.

Table B-9 shows the experimental ratios obtained for various materials at different thickness. These results were also plotted in Figure B-14. Note that for 1 inch absorbers of W and U, the ratios were obtained from Figure B-14 by extrapolation.

B.6.3 DOSE RATE FROM DEPLETED URANIUM ABSORBER

Uranium - 238 is a naturally occurring alpha emitter. It is the first member of uranium series decay chain. This decay series is shown below:

Therefore, ${}^{234}\text{Pa}$ and ${}^{234}\text{Th}$ are in equilibrium with ${}^{238}\text{U}$, that is, the activity of ${}^{234}\text{Pa}$ and ${}^{234}\text{Th}$ are approximately the same as the activity of ${}^{238}\text{U}$. The maximum beta energy of ${}^{234}\text{Pa}$ and ${}^{234}\text{Th}$ are 2.29 Mev and 0.191 Mev, respectively. Alphas and weak gammas are also present.

When a thermoluminescent dosimeter is placed near a uranium absorber in ${}^{170}\text{Tm}$ dose measurement, in addition to the dose received from ${}^{170}\text{Tm}$, it also records the dose from uranium absorber. To obtain true dose from ${}^{170}\text{Tm}$ source, this additional dose contributed by uranium absorber must be subtracted from the total dose.

Measurements were performed to determine the uranium dose rates. Survey meter was used to measure both gamma and beta dose rates at 1 inch from the surface of the uranium absorbers ($\frac{1}{4}$, $\frac{1}{2}$ and

TABLE B-9
EXPERIMENTAL RATIO - $(D)_{100 \text{ CM}} / (D)_{19 \text{ CM}}$

	$\frac{1}{4}''$	$\frac{1}{2}''$	$1''$
Al	3.44×10^{-2}	3.17×10^{-2}	3×10^{-2}
SS	3.35×10^{-2}	3.02×10^{-2}	2.74×10^{-2}
Pb	3.12×10^{-2}	2.88×10^{-2}	2.6×10^{-2}
W	2.9×10^{-2}	2.66×10^{-2}	2.37×10^{-2}
U	2.72×10^{-2}	2.53×10^{-2}	2.25×10^{-2}

Figure B-14. Experimental Ratio $(D)_{100 \text{ cm}} / (D)_{19 \text{ cm}}$

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1 inch). All three measurements gave the same dose rates - 45 mrad/hr at 1 inch. This is expected because almost all of the dose was delivered by beta radiation. The maximum range of 1 Mev beta is approximately 0.25 mm in uranium. Thus, all of the dose was produced by the betas at the surface of the uranium absorber, i.e., the dose rates are independent of the thickness of the absorber.

Table B-10 shows the calculation of uranium dose rates corresponding to the geometry of Tm-170 experiment. The summary data tabulations for depleted uranium (as presented in Appendix A) have been corrected using these results. Note that the corrections are only made for 19 cm geometry. The uranium background for 100 cm geometry is so small that a correction is not necessary.

TABLE B-10
DOSE RATES FOR DEPLETED URANIUM ABSORBERS

<u>Thickness of U Absorber (inches)</u>	<u>Distance Between Surface of U to Dosimeter (inches)</u>	<u>Dose Rate at 1" (mrads/hr)</u>	<u>Dose Rate at Dosimeter (mrads/hr)</u>
$\frac{1}{4}$	1.075	45	39
$\frac{1}{2}$	0.95	45	50
1	0.7	45	92

APPENDIX C

ANALYTICAL TECHNIQUES, PROCEDURES AND RESULTS

C.1 UNCOLLIDED DOSE RATE EQUATIONS

C.1.1 DOSE RATES AND GAMMA FLUXES

The gamma dose rate in tissue can be determined if one knows the gamma flux at the point of interest. For a shielded radiation source, the gamma flux consists of two components: uncollided and scattered. The first component can be readily evaluated by using the exponential attenuation through source medium and shield material. The attenuation through source medium is called self-attenuation. The scattered component requires sophisticated and lengthy computations. In this section, only the uncollided portion is treated. The scattered portion is discussed in paragraph C.5.

For a multigroup gamma spectrum, dose rates are calculated for each gamma energy group and then summed over all the energy groups to obtain the total gamma dose rate. Since 22 energy groups have been used for Tm-170 (see Table C-1), 22 separate dose rate calculations are required.

The uncollided dose rate in tissue from gamma rays of initial energy E_0 can be calculated from uncollided gamma flux by the following equation⁽⁸⁾:

$$D_{\text{unc.}}(E_0) = 5.767 \times 10^{-2} (\mu a / \rho)_{\text{tissue}} E_0 \phi(E_0) \quad (\text{C-1})$$

where $D_{\text{unc.}}(E_0)$ = uncollided dose rate, mrads/hr

$(\mu a / \rho)_{\text{tissue}}$ = energy absorption mass attenuation
coefficient for tissue, cm^2/gm

E_0 = initial photon energy, Mev

$\phi(E_0)$ = uncollided gamma flux, $\text{photons}/\text{cm}^2\text{-sec.}$

C.1.2 EXACT EQUATION FOR COMPUTING UNCOLLIDED FLUX

The configuration of source and the relationship between source, shield, and dose point are shown in Figure C-1. The source is assumed to be a right circular cylinder with diameter R and height h . The dose point, C , is located on the axis of the cylinder at a distance, a , away from the surface of the source. The thickness

TABLE C-1
GAMMA SPECTRUM OF Tm_2O_3 SOURCE

<u>Types of Radiation</u>	<u>Photon Energy (Mev)</u>	<u>Photons per Disintegration</u>
Gamma	0.084	3.459×10^{-2}
X-ray	0.052	5.050×10^{-2}
X-ray	0.009	1.349×10^{-1}
Bremsstrahlung	0.05	5.574×10^{-2}
	0.1	2.014×10^{-2}
	0.015	9.857×10^{-3}
	0.2	5.430×10^{-3}
	0.25	3.169×10^{-3}
	0.3	1.903×10^{-3}
	0.35	1.155×10^{-3}
	0.4	6.992×10^{-4}
	0.45	4.180×10^{-4}
	0.5	2.441×10^{-4}
	0.55	1.377×10^{-4}
	0.6	7.386×10^{-5}
	0.65	3.700×10^{-5}
	0.7	1.683×10^{-5}
	0.75	6.669×10^{-6}
	0.8	2.136×10^{-6}
	0.85	4.778×10^{-7}
	0.9	4.897×10^{-8}
	0.95	1.713×10^{-10}

Figure C-1. Spatial Relationships Between Source, Shield and Dose Point

Figure C-2. Attenuation Distances for Self-Absorption and Shielding Materials

of shield material is t . μ_1 , μ_2 and μ_3 are linear absorption coefficients of shield, source and air, respectively. Figure C-2 shows the triangle ABC.

Let S_v = volumetric source strength, photons/cm³-sec, the volume element is:

$$dV = 2 \pi r dr dx$$

the photon strength from the volume element is:

$$dA = S_v 2 \pi r dr dx$$

and the attenuation factor is equal to:

$$e^{-\mu_1 \rho_1 - \mu_2 \rho_2 - \mu_3 (\rho_3 + \rho_4)}$$

Thus, at dose point C, the uncollided gamma flux due to the element volume source dV through various attenuations is:

$$d\phi = \frac{S_v 2 \pi r dr dx}{4 \pi \rho^2} e^{-\mu_1 \rho_1 - \mu_2 \rho_2 - \mu_3 (\rho_3 + \rho_4)} \quad (C-2)$$

Upon integration, one obtains the total uncollided flux:

$$\phi = \frac{S_v}{2} \int_0^h dx \int_0^R dr \frac{r}{\rho^2} e^{-\mu_1 \rho_1 - \mu_2 \rho_2 - \mu_3 (\rho_3 + \rho_4)} \quad (C-3)$$

ρ_1 , through ρ_4 can be readily obtained by inspection of Figure C-2.

$$\begin{aligned} \cos \theta &= \frac{x+a}{\rho} = \frac{x+a}{\sqrt{(x+a)^2 + r^2}} \\ &= \frac{1}{\sqrt{1 + \frac{r^2}{(x+a)^2}}} = \frac{1}{L} \end{aligned}$$

$$\text{where } L = \sqrt{1 + \frac{r^2}{(x+a)^2}}$$

$$\begin{aligned} \text{then } \rho_1 &= t L \\ \rho_2 &= x L \\ \rho_3 &= e L \\ \rho_4 &= c L \end{aligned}$$

Substitute all the ρ 's into equation (C-3). The uncollided flux becomes:

$$\phi = \frac{S_v}{2} \int_0^h dx \int_0^R dr \frac{r}{r^2 + (X + a)^2} \exp \left\{ - \left[\mu_1 t + \mu_2 x + \mu_3 (e + c) \right] L \right\} \quad (C-4)$$

where $L = \sqrt{1 + \frac{r^2}{(X + a)^2}}$

To evaluate the above expression, one requires double numerical integration which is more time consuming than the numerical integration with a single independent variable. In the following section, an expression is given that approximates the results that would be achieved if the exact equation (C-4) for uncollided flux were used. It is this approximate solution which is subsequently used to compute the uncollided dose rate, (see Equation C-1).

C.1.3 APPROXIMATE EQUATION FOR COMPUTING UNCOLLIDED FLUX

The uncollided flux reaching C from a truncated right-circular cone source (as shown in Figure C-3) is given by the following equation⁽¹²⁾.

$$\phi = \frac{S_v}{2\mu_s} \left[E_2(b_1) - E_2(b_3) + \frac{E_2(b_3 \sec \theta)}{\sec \theta} - \frac{E_2(b_1 \sec \theta)}{\sec \theta} \right] \quad (C-5)$$

$$\text{where } b_1 = \sum_i^n \mu_i t_i$$

$$b_3 = b_1 + \mu_s h$$

$$E_2(b) = b \int_b^\infty \frac{e^{-t}}{t^2} dt$$

S_v = volumetric source strength, photons/cm³-sec

μ_s = linear absorption coefficient of source material, cm⁻¹.

Figure C-3. Geometry For A Truncated Right-Circular Cone Source

Figure C-4. Simulation Of A Cylindrical Source By A Truncated Right-Circular Cone Source

For a right-circular cylindrical source of radius R and height h (see Figure C-4), the uncollided flux at point C on the axis of the cylinder can be evaluated approximately by using equation (C-5). By replacing the cylinder with a truncated cone, the apex of which is at C , an upper limit can be obtained from equation (C-5) with $\theta = \theta_1 = \tan^{-1} \left(\frac{R}{a} \right)$. A lower limit can be similarly obtained with $\theta = \theta_2 = \tan^{-1} \left(\frac{R}{a+h} \right)$. It is obvious that the upper limit overestimates while the lower limit underestimates the flux. The degree of error involved depends on the geometry of the source and the distance between the source and the dose point. If $\theta_1 \approx \theta_2$, one can use this approximation to obtain uncollided gamma flux without significant deviation from the true flux.

C.1.4 JUSTIFICATION FOR USE OF THE APPROXIMATION EQUATION

The source geometry tested favors the use of the approximate equation for determination of the uncollided flux because: (1) the source diameter is greater than the height of the source and (2) the separation distance between source and dose point is much larger than the dimensions of the source. These conditions assure $\theta_1 \approx \theta_2$.

Specifically, consider the case of the 20.7-watt source (see Appendix B, paragraph B.1, for a detail discussion on source geometry) with separation distance of 19 cm, as shown in Figure C-5. The angles for upper limit θ_1 , and lower limit θ_2 , can be evaluated as follows:

$$\begin{aligned}\tan \theta_1 &= \frac{0.94}{19} = 0.0495 \\ \tan \theta_2 &= \frac{0.94}{19 + 1.016} = 0.0470 \\ \text{and } \theta_1 &= 2^\circ 50' \\ \theta_2 &= 2^\circ 41.5'\end{aligned}$$

Since the only functions of θ involved in the approximate equation (see eq. C-5) are in the form of $\sec \theta$ and $E_2(b \sec \theta)$ the deviation of the approximate solution from the true solution is small when $\sec \theta_1 \approx \sec \theta_2$,

$$\text{and } E_2(b \sec \theta_1) \approx E_2(b \sec \theta_2)$$

Figure C-5. Spatial Dimensions For The Source and Dose Point Geometry

From mathematical tables:

$$\text{Sec } \theta_1 = 1.0012$$

$$\text{Sec } \theta_2 = 1.0011$$

Therefore, the argument of E_2 function only changes 0.01% which indicates that the change in E_2 function is also small.

The error involved will be smaller still for other sources (i.e., 10 watt and 5 watt) and for the 100 cm separation distance. Thus, it is concluded that the approximate equation can be used, with either θ_1 or θ_2 to obtain uncollided flux with negligible deviation from the true flux.. θ_1 is selected and is used throughout this report.

C.2 COMPUTER CODE DEVELOPMENT

The basic equations for the uncollided dose rate calculation has been developed in the previous paragraph. Equations (C-5) and (C-1) give the gamma flux and the uncollided dose rate, respectively.

FORTTRAN was used to develop a computer code for evaluating the set of equations. For each photon energy group (total of 22 groups), dose rates were calculated each for 5 different materials (Al, SS, Pb, W, U), at the thicknesses ($\frac{1}{4}$, $\frac{1}{2}$ and 1 inch), for three source strengths (5, 10, 20 watts). The dose rates obtained at 20 watts were corrected to dose rates at 20.7 watts (see paragraph C.6).

To calculate the E_2 function, a subroutine was obtained⁽²⁸⁾ to first evaluate E_1 ; E_2 was then obtained from the following equation⁽²⁹⁾:

$$E_2(b) = \text{EXP}(b) - bE_1(b) \quad (\text{C-6})$$

19 cm was used as the source-to-dose point separation distance in the computation. The dose rates at 100 cm separation distance were not evaluated by the above program due to the extremely small angle ($\theta = 0^\circ 32'$, $\text{Sec } \theta = 1.00004$) involved. By considering the terms:

$$E_2(b) - \frac{E_2(b \text{ Sec } \theta)}{\text{Sec } \theta}$$

in equation (C-5) and the nature of the subroutine for the E_2 function, i.e., terms like $0.010416662 \times b \sec \theta$, and if $\sec \theta$ is very small, one observes immediately the difficulties involved to subtract two very close numbers. The truncation error from the finite digits carried in the computer will be significant if $b \sec \theta$ is very small. This is indeed what would happen were the uncollided dose rate for 100 cm separation distance to be calculated by the program used to perform the calculation at 19 cm for the specific dose point/source geometry under consideration.

To obtain the uncollided dose rates at 100 cm, a $1/r^2$ relationship between the values at 19 cm and 100 cm was used.

The main program and the subroutine for calculation of the uncollided dose rate at 19 cm is presented below.

C.3 MAIN PROGRAM INPUT DATA

C.3.1 VOLUMETRIC SOURCE STRENGTH

Source dimension (per wafer) = 0.1" x 0.74" diameter

Volume per wafer = $\frac{\pi}{4} (0.74 \times 2.54)^2 \times 0.254 = 0.705 \text{ cm}^3$

Power per wafer = 5W

Activity/power = 445 ± 22 ci/watt (see Appendix B, paragraph B.1)

No. of betas/wafer = $(5 \times 445)(3.7 \times 10^{10}) = 8.233 \times 10^{13} \text{ } \beta/\text{sec}$

Photons per beta = (γ / β) (see Table C-2)

$$S_v = \left(\frac{\gamma}{\beta} \right) \frac{\# \text{ of } \beta}{\text{volume}} = \left(\frac{\gamma}{\beta} \right) \frac{8.233 \times 10^{13}}{0.705}$$

$$= 1.168 \times 10^{14} \left(\frac{\gamma}{\beta} \right) \text{ photons/cm}^3 - \text{sec.}$$

Note that the error involved in source strength is about $\pm 5\%$.

C.3.2 PHOTON SPECTRUM

The total photon spectrum, i.e., gamma, x-ray, and bremsstrahlung, is presented in Table C-2.

C.3.3 LINEAR ABSORPTION COEFFICIENTS

The absorption coefficients (without coherence) for shielding

COMPUTER CODE FOR DOSE RATE CALCULATIONS

```
//
OPEN(3, INPUT, /DATA/)
OPEN(4, INPUT, /DTAA/)
DIMENSION AMUS(30), GMA(30), AMUA(30), E(30), SV(30), C(30)
DIMENSION A(300), B(300), D(300), DR(300), ENC(30), AMU(30)
TYPE 100
100 FORMAT(/////$TM-170 DOSE RATE CALCULATIONS$/)
TYPE 950
950 FORMAT(//$SHIELDING MATERIALS: 1=AL, 2=SS, 3=PB, 4=W, 5=U$/)
ASEC=SQRT(1.+(.94/19.38)**2)
READ 3, 110, (E(L), GMA(L), AMUS(L), AMUA(L), ENC(L), L=1, 22)
110 FORMAT(1F, 1E, 3F)
DO 2 L=1, 22
SV(L)=GMA(L)*1.168E14
C(L)=AMUA(L)*E(L)*5.767E-2
2 CONTINUE
DO 24 JJ=1, 5
READ 4, 115, (AMU(L), L=1, 24)
115 FORMAT(8F)
READ 4, 117
117 FORMAT(/)
TYPE 220, JJ
220 FORMAT(/$SHIELDING MATERIAL:$, I2)
AH=.127
DO 16 J=1, 3
AH=AH*2.
TYPE 120, AH
120 FORMAT(/$SOURCE THICKNESS = $, F6.4)
DO 22 LL=1, 4
IF(LL-3) 31, 22, 31
31 AL=LL
T=.635*AL
TYPE 125, T
125 FORMAT(//$SHIELD THICKNESS = $, F6.4//)
DO 8 I=1, 22
A(I)=AMU(I)*T+ENC(I)
A(I+22)=A(I)+AMUS(I)*AH
A(I+44)=A(I)*ASEC
A(I+66)=A(I+22)*ASEC
8 CONTINUE
DO 12 M=1, 88
AUG=A(M)
CALL EXPI(RES, AUG, IER)
B(M)=EXP(-AUG)-RES*AUG
12 CONTINUE
```

COMPUTER CODE FOR DOSE RATE CALCULATIONS (CONT)

```

DR(0)=0
DO 14 K=1,22
D(K)= C(K)*(SV(K)/(2.*AMUS(K)))*(B(K)-B(K+22)+(B(K+66)
+      -B(K+44))/ASEC)
TYPE 13, D(K)
13, FORMAT(/5(2X,E12.5))
DR(K)=DR(K-1)+D(K)
14 CONTINUE
TYPE 14, DR(22)
14, FORMAT(/,20X,$TOTAL DOSE RATE = $, E12.5///)
22 CONTINUE
16 CONTINUE
24 CONTINUE
END
SUBROUTINE EXPI(RES,X,IER)
IER=0
IF(X-4.) 10,10,20
10 IF(X+4.) 55,30,30
30 IF(X) 40,50,40
40 RES=-ALOG(ABS(X))-((((((((((((((1.03176E-11*X-.15798675E-10)
+ *X+.16826592E-9)*X-.21915699E-8)*X+.2763583E-7)*X-.30726221
+ E-6)*X+.3099604E-5)*X-.2833759E-4)*X+.23148392E-3)*X-
+ .00166669E-6)*X+.01416662)*X-.05555552)*X+.25)*X-1.0)*X-
+ .57721566
RETURN
20 ARG=4./X
RES=EXP(-X)*((((((((((0.0009442761*ARG-.0049362007)*ARG+
+ .011723272)*ARG-.017555779)*ARG+.020412099)*ARG-
+ .022951979)*ARG+.031208561)*ARG-.062498588)*ARG+
+ .24999999)*ARG
RETURN
50 RES=1.E75
RETURN
55 IER=1
RETURN
END
*
```

TABLE C-2
GAMMA SPECTRUM AND LINEAR ABSORPTION COEFFICIENTS

Photon Energy (Mev)	Photons Per Beta (γ / β)	<u>Linear Absorption Coefficients (cm⁻¹)</u>		
		Source <u>Tm₂O₃</u>	<u>Tissue</u>	Encapsulation (Hastelloy C)
.084	3.459 x 10 ⁻²	56	.025	6.53
.052	5.05 x 10 ⁻²	40	.041	21.7
.009	1.349 x 10 ⁻¹	100	6.	1950
.05	5.574 x 10 ⁻²	42.6	.0409	21.7
.1	2.014 x 10 ⁻²	32.2	.0252	3.98
.15	9.857 x 10 ⁻³	11.0	.0276	1.97
.2	5.43 x 10 ⁻³	5.39	.0297	1.4
.25	3.169 x 10 ⁻³	3.25	.031	1.13
.3	1.903 x 10 ⁻³	2.22	.0317	1.03
.35	1.155 x 10 ⁻³	1.65	.032	.94
.4	6.992 x 10 ⁻⁴	1.31	.0325	.87
.45	4.18 x 10 ⁻⁴	1.1	.0325	.804
.5	2.441 x 10 ⁻⁴	.935	.0327	.775
.55	1.377 x 10 ⁻⁴	.82	.0327	.733
.6	7.386 x 10 ⁻⁵	.752	.0326	.708
.65	3.7 x 10 ⁻⁵	.695	.0325	.68
.7	1.683 x 10 ⁻⁵	.642	.0322	.652
.75	6.669 x 10 ⁻⁶	.6	.032	.625
.8	2.136 x 10 ⁻⁶	.566	.0318	.614
.85	4.778 x 10 ⁻⁷	.54	.0318	.59
.9	4.897 x 10 ⁻⁸	.513	.0312	.572
.95	1.713 x 10 ⁻¹⁰	.49	.031	.554

materials (Al, SS, Pb, W, U) are obtained from NBS Circular 583^(3,5). These coefficients are tabulated in Table C-3.

The coefficients for tissue are obtained from NBS Handbook 85⁽¹³⁾. For the source material, Tm_2O_3 , the absorption coefficients are obtained by the following equation:

$$(\mu)_{Tm_2O_3} = \frac{2A(Tm) \frac{\mu_o}{\rho}(Tm) + 3A(O) \frac{\mu_o}{\rho}(O)}{2A(Tm) + 3A(O)} \rho(Tm_2O_3) \quad (C-7)$$

The density of Tm_2O_3 was assumed to be 8.6 gm/cm^3 ⁽⁸⁾. The values for tissue and Tm_2O_3 are given in Table C-2.

Hastelloy C is used for encapsulation materials. There are two layers of encapsulation with 0.05 inch thickness for each layer. The density of Hastelloy C is 8.94 gm/cm^3 ⁽³⁰⁾. The absorption coefficients for Hastelloy C are taken from the coefficients of nickel⁽⁴⁾ (see Table C-2) because Hastelloy C is a nickel based alloy.

The attenuation in air is very small compared to that of the encapsulation and therefore, was not included in the computation.

The so called "tunnelling effect", that arises because of a discontinuous change in the coefficient at K edge, has also been analyzed. The dominant energy groups for this shield calculation are somewhere between 0.15 to 0.5 Mev for low Z materials and between 0.35 to 0.6 Mev for high Z materials. For example, the K edge for stainless steel is about 0.007 Kev and the corresponding mass attenuation coefficient is approximately 50⁽³¹⁾⁽³²⁾. This value is much greater than the coefficient at 0.030 Mev, which is 8. Therefore, the "tunnelling effect" would not be important in this calculation.

The uncertainty for mass attenuation coefficients is reported⁽³⁾ as:

$$\begin{array}{ll} 0.1 \text{ Mev} < E < 1 \text{ Mev} & \text{about 3 to 5\%} \\ E < 0.1 \text{ Mev} & \text{about 10\%} \end{array}$$

TABLE C-3

LINEAR ABSORPTION COEFFICIENTS FOR SHIELDING MATERIALS

Photon Energy (Mev)	<u>Linear Absorption Coefficients (cm⁻¹)</u>				
	<u>Al</u>	<u>SS</u>	<u>PB</u>	<u>W</u>	<u>U</u>
.084	.456	4.38	13.61	144.6	36.1
.052	.883	14.46	58.8	76	134
.009	71.3	140	908	1120	2113
.05	.883	14.46	58.85	76	134
.1	.436	2.718	60	81.2	19.82
.15	.363	1.446	20.86	27.79	45.25
.2	.325	1.09	10.16	13.66	21.88
.25	.298	.964	6.384	8.685	14.21
.3	.279	.837	4.037	5.655	8.452
.35	.263	.774	3.016	4.246	6.265
.4	.25	.726	2.359	3.358	4.843
.45	.237	.687	1.928	2.799	3.927
.5	.228	.654	1.644	2.413	3.291
.55	.22	.632	1.44	2.123	2.899
.6	.211	.602	1.293	1.949	2.543
.65	.203	.581	1.179	1.795	2.3
.7	.198	.561	1.083	1.66	2.113
.75	.191	.544	1.009	1.563	1.926
.8	.185	.525	.948	1.473	1.777
.85	.180	.51	.896	1.4	1.664
.9	.179	.498	.851	1.332	1.571
.95	.171	.482	.812	1.28	1.487

Since the photon spectrum for thulium-170 source consists mainly of low energy photons, the uncertainty for mass attenuation coefficients is taken as 5%.

C.4 UNCOLLIDED DOSE RATES

The uncollided dose rates for 19 cm separation distance have been obtained from the computer with the input data presented in the last section. A typical output is given in Table C-4. Note that the sequence of photon energy for each dose rate group is the same as input data (see Table C-3). Complete tabulation is not given here because these results are to be modified by a build-up factor as shown in paragraph C.6.

To obtain the uncollided dose rates at 100 cm, a $1/r^2$ relationship can be used as follows:

$$(D)_{100 \text{ cm}} = \frac{(19)^2}{(100)^2} (D)_{19 \text{ cm}} = 3.61 \times 10^{-2} (D)_{19 \text{ cm}} \quad (C-8)$$

The total dose rate can then be obtained by multiplying the uncollided dose rates by a build-up factor. However, experimental results show that the ratio, $(D)_{100 \text{ cm}} / (D)_{19 \text{ cm}}$, is equal to 3.61×10^{-2} only for unshielded case. For shielded dose rates, this ratio is not a constant value but varies depending on shielding materials and thicknesses. To extend the results from 19 cm to 100 cm properly, experiments would have to be performed to obtain proper experimental ratios. These experimental ratios would be used to multiply total dose rates at 19 cm to obtain total dose rates at 100 cm. The calculation procedures are presented in paragraph C.6.

To verify the computer results, half-value thickness studies have been made. Typical examples are shown in Table C-5, and the uncollided dose rates are plotted in Figure C-6. Comparison between theoretical $(0.693/\mu)$ and calculated half-value-thickness (HVT) indicates good agreement.

TABLE C-4
TYPICAL UNCOLLIDED DOSE RATES FROM COMPUTER CALCULATIONS

SHIELDING MATERIAL: 3
SOURCE THICKNESS = .2540

SHIELD THICKNESS = .6350

SHIELD THICKNESS = 1.2700

.16015E 0
 .21628E-14
 .00000E 0
 .18178E-14
 .60840E-13
 .14588E- 1
 .19413E 2
 .21742E 3
 .83098E 3
 .12390E 4
 .14027E 4
 .12890E 4
 .10377E 4
 .75014E 3
 .48885E 3
 .28779E 3
 .15048E 3
 .67366E 2
 .23971E 2
 .59372E 1
 .65529E 0
 .24838E- 2

.28120E- 4
 .12880E-30
 .00000E 0
 .10487E-30
 .16903E-29
 .25582E- 7
 .30518E- 1
 .37645E 1
 .63917E 2
 .18232E 3
 .31335E 3
 .37866E 3
 .36511E 3
 .30046E 3
 .21497E 3
 .13606E 3
 .75622E 2
 .35483E 2
 .13125E 2
 .33600E 1
 .38160E 0
 .14827E- 2

TOTAL DOSE RATE = .78116E 4

TOTAL DOSE RATE = .20866E 4

TABLE C-5
HALF-VALUE-THICKNESS STUDIES

<u>Selected Group</u>	<u>Uncollided Dose Rates (mrad/hr)</u>			<u>Half-Value-Thickness</u>		
	<u>$\frac{1}{4}$"</u>	<u>$\frac{1}{2}$"</u>	<u>1"</u>	<u>μ</u>	<u>$0.693/\mu$</u>	<u>HVT (Fig. C-6)</u>
Al, 5W, 0.2 Mev	1 x 10 ⁴	8.17 x 10 ³	5.4 x 10 ³	0.325	2.13	2.15
Pb, 20W, 0.6 Mev	1.5 x 10 ³	6.6 x 10 ²	1.28 x 10 ²	1.3	0.533	0.54
W, 10W, 0.4 Mev	7.44 x 10 ²	8.8 x 10 ¹	1.23	3.36	0.206	0.2

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Figure C-6. Determination of Half-Value Thickness

C.5 BUILD-UP FACTOR CONSIDERATION

The dose rates from uncollided photons have been obtained and are presented in the previous sections. The collided photons, in addition, may also reach the dose point after one or more scatterings in the shield medium. Therefore, the total dose rates at the dose point consists of two components:

$$(D)_{\text{total}} = (D)_{\text{uncollided}} + (D)_{\text{scattered}} \quad (\text{C-9})$$

In practice, a buildup factor is defined by the following equation:

$$B = \frac{(D)_{\text{total}}}{(D)_{\text{uncollided}}} \quad (\text{C-10})$$

and the total dose rate can be obtained by multiplying the uncollided dose rate by B, the build-up factor

$$(D)_{\text{total}} = B \times (D)_{\text{uncollided}} \quad (\text{C-11})$$

The calculation of a build-up factor is a difficult problem because of the complexity of the Compton scattering. An exact solution is almost impossible to obtain. However, various approximation methods have been developed. By using the "moments method" introduced by Spencer and Fano^(33,34), an extensive series of calculations of dose rate build-up factors of gamma rays in infinite homogeneous media has been carried out by Goldstein and Wilkins⁽¹⁴⁾. B_{∞} is defined as a build-up factor in an infinite medium. Table C-6 shows a typical set of B_{∞} tabulated by Goldstein and Wilkins.

For medium geometry there are boundary conditions that exist. Different techniques, such as Monte Carlo, albedo matrix approach, etc., have to be used. These procedures are more complicated and more time consuming than is worthwhile for most problems and beyond the scope of this report. Therefore, B_{∞} is used to obtain the total dose rates throughout this report.

It is obvious that the use of B_{∞} overestimates the total dose rate for a finite geometry because the photons escaping from the

TABLE C-6
DOSE BUILD-UP FACTOR FOR LEAD
(for point isotropic source in infinite medium)

E_o , Mev	$\mu_o r$						
	1	2	4	7	10	15	20
0.5	1.24	1.42	1.69	2.00	2.27	2.65	(2.73)
1.0	1.37	1.69	2.26	3.02	3.74	4.81	5.86
2.0	1.39	1.76	2.51	3.66	4.84	6.87	9.00
3.0	1.34	1.68	2.43	3.75	5.30	8.44	12.3
4.0	1.27	1.56	2.25	3.61	5.44	9.80	16.3
5.1097	1.21	1.46	2.08	3.44	5.55	11.7	23.6
6.0	1.18	1.40	1.97	3.34	5.69	13.8	32.7
8.0	1.14	1.30	1.74	2.89	5.07	14.1	44.6
10.0	1.11	1.23	1.58	2.52	4.34	12.5	39.2

boundary can no longer contribute to scattered dose. The magnitude of error involved depends on photon energy, absorbing medium and its geometry.

For example, suppose one replaces the medium by air at some distance from the source. At the boundary, the dose for heavy material ($Z > 50$) should be close to that expected for an infinite medium. For light material, however, backscattering is important and therefore the build-up factor should be considerably less for finite medium than that for an infinite medium. This decrease in build-up factor is only observed in the low energy range.

Experiments at NBS⁽¹⁵⁾ indicate that for initial energy around 1 Mev and low Z materials (e.g., Al, SS), the boundary effect reduces the buildup factors by about 30%. For lower energies the reduction is expected to be much greater. For high Z materials (e.g., Pb, W, U), the reduction is small. For example, for Pb at 1 Mev, the reduction is about 2 to 3%⁽¹⁶⁾. If B_{∞} is used in the calculation, the total dose rates for Al and SS will be overestimated by about 50%. Fortunately, these light materials are unlikely to be chosen as shielding materials. For heavy materials (Pb, W, U), the deviation is usually small.

B_{∞} for Al, SS, Pb, W and U are obtained from Goldstein and Wilkins⁽¹⁴⁾. A typical set of B_{∞} for Pb is given in Table C-6. The values of B_{∞} for intermediate energy E_0 are obtained from linear interpolation. Figure C-7 shows a typical set of curves to obtain intermediate values of B_{∞} for Pb (similar curves for W and U absorbers). Figure C-8 shows the same curve for Al (similar curve for SS). Note that in Figure C-7 the convergence of B_{∞} at low energies makes extrapolation of B_{∞} more accurate compared to that in Figure C-8. Therefore, the extrapolation of B_{∞} in low energy for Al and SS introduces a much larger uncertainty. This may explain partially why there are large discrepancies between measured and calculated dose rates for Al and SS.

Figure C-7. Dose Build-up Factor - Pb (Point Isotropic Source)

Figure C-8. Dose Build-up Factor - Al (Point Isotropic Source)

In calculating uncollided dose rates, 22 energy groups have been used. In obtaining B_{∞} , however, only four energy groups are used: 0.1, 0.2, 0.475 and 0.775 Mev. This simplifies the work considerably while still retaining accuracy. Pb has been used to test the validity of this method. The total dose rates obtained from 22 groups of B_{∞} and four groups of B_{∞} are in good agreement.

Therefore, B_{∞} obtained for energies of 0.1, 0.2, 0.475 and 0.775 Mev are tabulated in Table C-7. Note that the absence of B_{∞} in a particular energy group indicates that it is unnecessary to have B_{∞} because the uncollided dose rate is negligible in this particular group. The uncollided dose rates obtained from computer calculations are also summed over the four groups (i.e., 0.009 to 0.1, 0.15 to 0.3, 0.35 to 0.6, and 0.65 to 0.95 Mev). The product of B_{∞} and the uncollided dose rate is obtained for each group and the sum of the product of all four groups gives the total dose rates.

C.6 TOTAL DOSE RATES

The total dose rates at 19 cm separation distance are obtained by multiplying B_{∞} to the uncollided dose rates. Table C-8 shows a typical set of calculations for evaluating total dose rate. Note that the actual source strength of three sources are: 5, 10 and 20.7 watts (see Appendix B.1), while for convenience 5, 10 and 20 watts have been used in calculations. Therefore, the results obtained from calculation for 20 watt source have to be multiplied by $(\frac{20.7}{20})$ to give correct dose rates for 20.7 watt source. Table C-9 presents the uncollided and total dose rates at 19 cm.

To extend the uncollided dose rates from 19 cm to 100 cm, the inverse square law, $1/r^2$ is used. The total dose rates at 100 cm are obtained by multiplying the dose rates obtained for 19 cm by 3.6×10^{-2} . Table C-10 gives the uncollided and total dose rates at 100 cm.

TABLE C-7
DOSE BUILD-UP FACTORS
(for point isotropic source in infinite medium)

<u>Material</u>	<u>Thickness (inch)</u>	<u>Energy (Mev)</u>			
		<u>0.1</u>	<u>0.2</u>	<u>0.475</u>	<u>0.775</u>
Al	$\frac{1}{8}$	1.7	1.4	1.21	1.14
	$\frac{1}{2}$	2.25	1.7	1.4	1.27
	1	3.9	2.5	1.8	1.51
SS	$\frac{1}{8}$	2.85	1.7	1.44	1.32
	$\frac{1}{2}$	5.3	2.4	1.83	1.6
	1	-	4.22	2.8	2.28
Pb	$\frac{1}{8}$	-	1.4	1.26	1.22
	$\frac{1}{2}$	-	1.6	1.43	1.4
	1	-	-	1.7	1.73
W	$\frac{1}{8}$	-	1.5	1.4	1.38
	$\frac{1}{2}$	-	1.6	1.66	1.7
	1	-	-	2.11	2.28
U	$\frac{1}{8}$	-	1.3	1.3	1.3
	$\frac{1}{2}$	-	-	1.48	1.53
	1	-	-	1.7	1.92

TABLE C-8

CALCULATION OF TOTAL DOSE RATE AT 19 cm FOR 20.7-WATT SOURCE
 WITH Pb SHIELDS OF $\frac{1}{4}$, $\frac{1}{2}$ and 1 INCH THICKNESSES

Total Dose Rate for 20.7-Watt Source (R/hr)

Energy Group	$\frac{1}{4}$ "			$\frac{1}{2}$ "			1"		
	(D) unc	B	(D) total	(D) unc	B	(D) total	(D) unc	B	(D) total
0.1	-	-	-	-	-	-	-	-	-
0.2	2.124	1.4	2.974	.139	1.6	.222	-	-	-
0.475	16.91	1.26	21.31	4.914	1.43	7.027	.544	1.7	.925
0.775	1.698	1.22	<u>2.072</u>	.837	1.4	<u>1.172</u>	.205	1.73	<u>.354</u>
Total Dose Rate For 20-Watt			26.36				8.421	1.279	
Total Dose Rate For 20.7-Watt			27.28				8.716	1.324	

TABLE C-9

CALCULATED DOSE RATES AT 19 CM (RAD/HR)

Shielding Material	Thickness (inches)	<u>Uncollided Dose Rates</u>			<u>Total Dose Rates (With B_{∞})</u>		
		<u>5W</u>	<u>10W</u>	<u>20.7W</u>	<u>5W</u>	<u>10W</u>	<u>20.7W</u>
Al	$\frac{1}{4}$	60.8	89.8	123	81.6	119	161
	$\frac{1}{2}$	50.6	75	103	81.1	118	160
	1	35	52.5	73	79	115	156
SS	$\frac{1}{4}$	39.8	60.3	84.4	63.6	95.4	132
	$\frac{1}{2}$	22.6	35	50	48.1	73.4	102
	1	7.73	12.4	18.2	26.3	41.2	59
Pb	$\frac{1}{4}$	7.81	13.4	21.5	9.97	17.2	27.3
	$\frac{1}{2}$	2.1	3.69	6.1	2.98	5.27	8.72
	1	0.252	.455	.78	.43	.774	1.32
W	$\frac{1}{4}$	4.16	7.27	11.8	5.85	10.2	16.6
	$\frac{1}{2}$	.741	1.32	2.14	1.23	2.21	3.7
	1	.042	.077	.128	.092	.167	.288
U	$\frac{1}{4}$	2.02	3.58	5.9	2.62	4.65	7.71
	$\frac{1}{2}$	.253	.456	.78	.397	.68	1.17
	1	.0086	.0158	.028	.016	.029	.05

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TABLE C-10
 CALCULATED DOSE RATES AT 100 CM (RAD/HR)

<u>Shielding Material</u>	<u>Thickness (inches)</u>	<u>Uncollided Dose Rates</u>			<u>Dose Rates With B_∞</u>		
		<u>5W</u>	<u>10W</u>	<u>20.7W</u>	<u>5W</u>	<u>10W</u>	<u>20.7W</u>
Al	$\frac{1}{4}$	2.19	3.24	4.45	2.96	4.30	5.80
	$\frac{1}{2}$	1.83	2.71	3.73	2.92	4.30	5.80
	1	1.26	2.00	2.64	2.85	4.20	5.60
SS	$\frac{1}{4}$	1.44	2.18	3.05	2.30	3.43	4.77
	$\frac{1}{2}$	.82	1.26	1.80	1.73	2.64	3.72
	1	.28	.45	.66	.94	1.48	2.13
Pb	$\frac{1}{4}$	.28	.49	.78	.360	.614	.97
	$\frac{1}{2}$	.076	.13	.22	.110	.190	.31
	1	.009	.016	.028	.0155	.028	.047
W	$\frac{1}{4}$	.15	.263	.43	.210	.370	.600
	$\frac{1}{2}$	.027	.048	.079	.044	.079	.134
	1	.0015	.0028	.0047	.0033	.006	.010
U	$\frac{1}{4}$	.0722	.13	.213	.094	.170	.280
	$\frac{1}{2}$	.009	.0166	.028	.014	.025	.042
	1	.00031	.00058	.001	.00058	.001	.0018

The estimated overall uncertainties for Pb, W and U are expected to be approximately 20%. However, for Al, and SS, the overall uncertainties are estimated to be much greater, perhaps 50% or more.

Radiation scattered from the experimental fixture and from the hot cell walls are considered below.

C.7 RADIATION SCATTERING FROM THE EXPERIMENTAL ENVIRONMENT

Investigations were conducted into the possible modes and effects of radiation scatter in the experimental geometry. Consideration of experimental setup (see Figure B-6) reveals that in operation, the thulium-170 sources were well shielded in their experimental cask, and that the only exit path for the emitted radiation was the beam tunnel which was designed to the same diameter as the source wafers themselves. Thus, the solid beam of radiation emerging from this tunnel was well defined, and the absorber specimens were designed to completely intercept this beam in their experimental positions. Since the scatter of these radiations in air is insignificant, the possible sources of such scattering become: (1) single and multiple scatter by the shielding material to dose point; (2) double scatter by the absorber and a part of the experimental fixture; (3) single backscatter off the rear wall of the hot cell to the dose point; and (4) double scatter by the absorber and a hot cell structural member.

C.7.1 SINGLE AND MULTIPLE SCATTER BY SHIELDING MATERIAL

Single and multiple scatter by the shielding material to the dose point constitutes the build-up factor and is discussed in paragraph C.5)

C.7.2 DOUBLE SCATTER BY THE ABSORBER

Examination of Figure B-6 and the diagrammatic sketch shown in Figure C-9 reveals that only the radiations scattered by the absorbers approximately straight ahead and in the downward direction

had any probability of encountering a part of the experimental fixture (notably the aluminum base channel) and thus had any real probability of second scatter to the dose point. Therefore, to prevent this from occurring, a lead scatter shield was designed and installed to intercept and attenuate all such scattered radiations (see Figure B-6).

Figure C-9. Radiation Scattering from Experimental Set-up

C.7.3 SINGLE BACKSCATTER

The calculation to determine the intensity of radiation backscattered off the rear wall of the hot cell to the dose point was performed with the aid of Figure C-10. The intensity of the radiation scattered within the unit volume of scattering medium through an angle (θ) is⁽¹²⁾:

$$dI = I_o \times (7.94 \times 10^{-26}) \eta F_1(E_o, \theta) \left(\frac{e^{-\mu r_2}}{4\pi r_1^2} \right) \left(\frac{e^{-\mu r_3}}{r_4^2} \right) dV$$

by integrating over volume, then

$$\frac{I}{I_o} = \frac{7.94 \times 10^{-26} \eta}{4\pi} \int_{V_{o1}} F_1(E_o, \theta) \frac{e^{-\mu(r_2 + r_3)}}{r_1^2 r_4^2} dV \quad (C-12)$$

where:

I = intensity of the scattered beam, Mev/cm²-sec

I_o = intensity of the initial beam, Mev/cm²-sec

η = number of electrons per cm³ of scattering material

r = path length of scattered radiation, cm

dV = volume element of scattering volume, cm³

μ = linear attenuation coefficient of scattering material,
cm⁻¹

$F_1(E_o, \theta)$ = scatter probability for particle energy E
(Mev) and scattering angle θ . It defines as:

$$F_1(E_o, \theta) = \frac{p^3}{2} \left(\frac{1}{p} + p - \sin^2 \theta \right) \quad (C-13)$$

and

$$p = \frac{E}{E_o} = \frac{1}{1 + \frac{E_o}{m_o c^2} (1 - \cos \theta)} \quad (C-14)$$

Figure C-10. Radiation Scattering from Hot Cell Wall

From the geometry of the experiment, as shown in Figure C-11, the approximate scattering intensity was calculated as follows:

$$\text{Assume } E_o = 0.5 \text{ Mev}$$

$$\eta = 2.2 \times 10^{24} \text{ e-/cm}^3 \text{ (for steel)}$$

$$r_1 = 3.4 \times 10^2 \text{ cm}$$

$$r_4 = 2.5 \times 10^2 \text{ cm}$$

$$\text{From Figure C-11, } \sin \theta = \sin (180 - \phi) = \sin \phi = \sin 4.50^\circ$$

and

$$\begin{aligned} \cos \theta &= \cos (180^\circ - \phi) \\ &= -\cos \phi \\ &= -\cos 4.5^\circ \end{aligned}$$

$$\text{Then, } p = \frac{1}{1 + (1 + \cos 4.5)} = \frac{1}{1 + 2} = \frac{1}{3} = 0.33$$

$$F_1 = \frac{1}{2} (p^2 + p^4 - p^3 \sin^2 \theta) = 6 \times 10^{-2}$$

Assuming all scattering to take place at this surface, then

$$e^{-\mu(r_2 + r_3)} = 1$$

For the maximum scattering case, a circular disc volume with radius = 75.1 cm and 1 cm thick was assumed.

$$\text{Then } \Delta V = 1.7 \times 10^4 \text{ cm}^3$$

Substitute all the quantities into equation (C-12).

$$\frac{I}{I_o} = \frac{7.94 \times 10^{-26}}{4 \pi} \eta F_1 (E_o, \theta) \frac{1}{(r_1^2)(r_4^2)} \Delta V = 10^{-9}$$

Therefore, the scattering contribution from the rear wall of the hot cell is so small that it can be neglected.

Figure C-11. Spatial Dimensions Between Source and Hot Cell Wall

C.7.4 DOUBLE SCATTER

The size of the solid angle of radiation beyond the absorber was such that no radiation could be directly incident upon any of the side walls, the ceiling or the hot cell floor. Thus, only double scatter by the absorber and one of these structural materials could possibly result in scattered radiation incident at the dose point as illustrated in Figure C-12.

Figure C-12. Double Scattering from Source and Hot Cell

To calculate the intensity of radiation scattered out of the original flux to the dose point (I/I_0), reference is made to the scattering equation⁽¹²⁾.

$$\begin{aligned} \frac{I}{I_0} = & 7.94 \times 10^{-26} \eta_1 F_1 (E_0, \theta_1) \frac{1}{4\pi r_0^2 r_1^2} \Delta V_1 \\ & \times 7.94 \times 10^{-26} \eta_2 F_2 (E_1, \theta_2) \frac{1}{r_2^2} \Delta V_2 \end{aligned} \quad (C-15)$$

where the quantities are previously defined.

It is assumed that all radiation scatter occurs at the surfaces of the scattered materials. The following are the pertinent dimensions of the assumed scattering system:

$$\begin{aligned} \eta_1 &= \eta_2 = 2.2 \times 10^{24} \text{ electrons/cm}^3 \text{ (for steel)} \\ r_0 &= 15 \text{ cm} \\ r_1 &= 90 \text{ cm} \\ r_2 &= 120 \text{ cm} \\ \Delta V_1 &= 24.6 \text{ cm}^3 \\ \Delta V_2 &= 1.7 \times 10^4 \text{ cm}^3 \\ E_0 &= 0.5 \text{ Mev} \end{aligned}$$

Upon insertion of these into equations (C-13), (C-14) and (C-15), the intensity of radiation doubly scattered to the dose point becomes:

$$\frac{I}{I_0} = 1.65 \times 10^{-10}$$

which is negligible except for very high fluxes incident over long irradiation intervals. It should be noted that the highest strength sources with the thinnest absorbers also had the briefest irradiation intervals. Therefore, this scattering mechanism was not thought to constitute a significant source of error. In light of these results, it was concluded that with the exception of buildup factor, no significant contributions to the radiation dose rates would arise due to scattering.