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RADIOISOTOPE-POWERED CARDIAC PACEMAKER

REPORT TO:
UNITED STATES ATOMIC ENERGY COMMISSION
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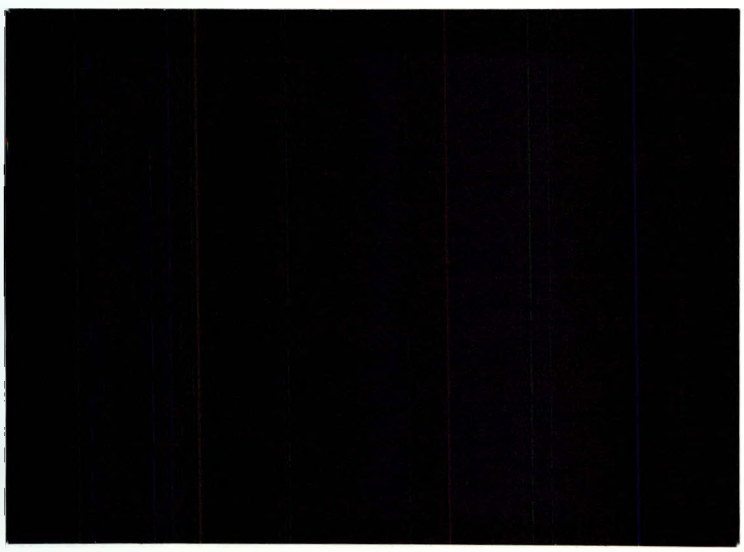

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SAFETY ANALYSIS

PRELIMINARY DOSE RATE EVALUATION

By Roger D. Caldwell

RADIOISOTOPE POWERED
CARDIAC PACEMAKER PROGRAM

Contract AT(30-1)-3731

September 30, 1966

Prepared For
UNITED STATES ATOMIC ENERGY COMMISSION

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Energy Conversion Division
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SAFETY ANALYSIS
PRELIMINARY DOSE RATE EVALUATION

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SAFETY ANALYSIS

PRELIMINARY DOSE RATE EVALUATION

by

Roger D. Caldwell

Radioisotope Powered Cardiac Pacemaker
Contract AT(30-1)-3731

MOTIVATION

The Pu-238 fueled cardiac pacemaker is designed to be inserted into the patient's body. Because Pu-238 in the form and purity available emits gamma and neutron radiations and the pacemaker is in intimate contact with tissue, the magnitude and possible effects of this penetrating radiation must be fully considered. Consequently, a preliminary radiation analysis was scheduled early in the RPCP program plan.

This preliminary analysis serves two purposes: (1) It will guide the design of the external radiation experimental study; (2) it will indicate and permit inclusion of additional shielding if necessary before the prototype pacemaker design has been firmly committed.

PACEMAKER DESCRIPTION AND GENERAL ANALYTICAL APPROACH

The NUMEC radioisotope powered cardiac pacemaker (RPCP) is a very small (6 cm x 5 cm x 2.8 cm) electronic pulser. Power is supplied by a thermoelectric generator which consists of a triply encapsulated Pu-238 metal heat source with base-metal quartz thermocouple tapes wrapped around it. The pacemaker electronics and electrode attachment are integrated into the generator case. NUMEC drawing 15-D-174, Figure 1 gives the layout details of the reference design.

The heat source consists of three Pu metal rods each separately encapsulated in an inner Ta-10W capsule and a secondary Hastelloy C sheath. The three capsules are combined in the third titanium encapsulation.

The general approach to estimating the dose to tissues surrounding the pacemaker includes:

1. Determining the source intensity and spectra for neutron and gamma radiations.

2. Calculating the radiation flux at points of interest. This also includes taking into account the shielding effect of the pacemaker materials and surrounding tissues.

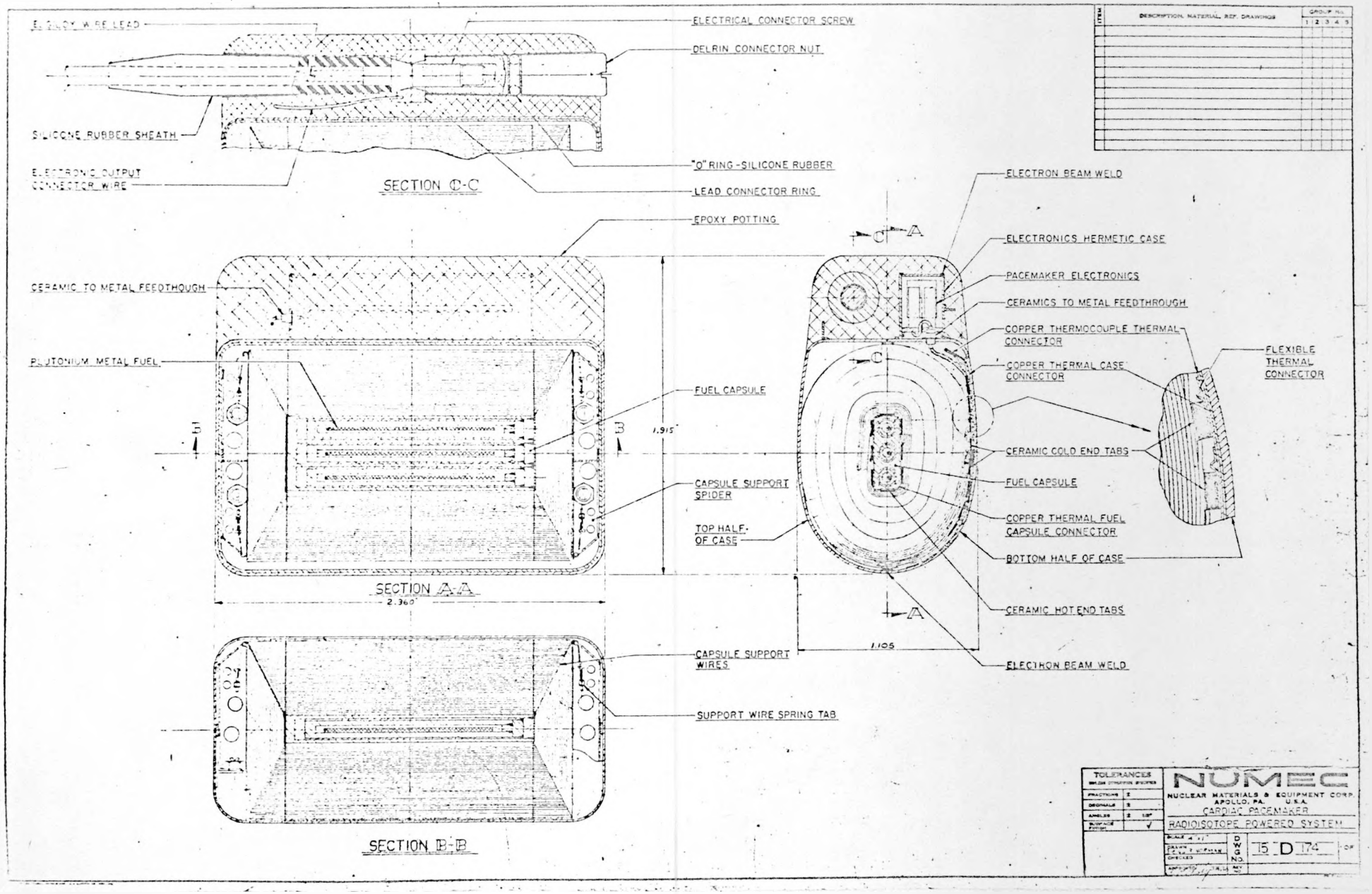

EL 2 LOY W RE LEAD
ELECTRICAL CONNECTOR SCREW
DELRIN CONNECTOR NUT
SILICONE RUBBER SHEATH
ELECTRONIC OUTPUT CONNECTOR WIRE
"O" RING - SILICONE RUBBER
LEAD CONNECTOR RING
SECTION D-C
EPOXY POTTING
ELECTRON BEAM WELD
ELECTRONICS HERMETIC CASE
PACEMAKER ELECTRONICS
CERAMICS TO METAL FEEDTHROUGH
COPPER THERMOCOUPLE THERMAL CONNECTOR
COPPER THERMAL CASE CONNECTOR
FLEXIBLE THERMAL CONNECTOR
CERAMIC TO METAL FEEDTHOUGH
PLUTONIUM METAL FUEL
FUEL CAPSULE
1.915
CERAMIC COLD END TABS
FUEL CAPSULE
CAPSULE SUPPORT SPIDER
COPPER THERMAL FUEL CAPSULE CONNECTOR
TOP HALF OF CASE
BOTTOM HALF OF CASE
CERAMIC HOT END TABS
SECTION A-A
2.360
1.105
CAPSULE SUPPORT WIRES
ELECTRON BEAM WELD
SUPPORT WIRE SPRING TAB
SECTION B-B
ITEM
DESCRIPTION, MATERIAL, REF. DRAWINGS
GROUP NO.
TOLERANCES
NUMEC
NUCLEAR MATERIALS & EQUIPMENT CORP.
APOLLO, PA.
U.S.A.
CARDIAC PACEMAKER
RADIOISOTOPE POWERED SYSTEM
DWG NO. 15 D 174

Figure 1

3. Converting the calculated flux to dose rate.

4. Calculating the decrease in dose rate with depth in tissue.

5. Calculating the increase in gamma dose rate with time resulting from "grow-in" via daughters of Pu-236.

In this preliminary evaluation only the contact points closest to the source and the fall off of dose rate with depth in a perpendicular line away from the pacemaker case were calculated. These dose points are illustrated in Figure 2. All other points will have lower dose rates because they are more distant from the source and/or more shielding material intervenes.

As much detail as are available was included in the calculations. For instance all of the elements present in Hastelloy C were used in calculating gamma absorption. However, certain assumptions employed in the calculations such as the purity of the Pu-238 metal available or radiation quanta per group may be subject to error. Consequently the calculated dose rates presented herein are preliminary estimates and should be expected to give an order of magnitude indication of the potential hazards associated with pacemaker implantation. The dosimetry measurements, which are detailed later in this report, eventually must provide the basis for a correct assessment of the possible hazards.

RADIATIONS FROM Pu-238 METAL

The data used for the dose calculations comes principally from the Savannah River Laboratory Report DP-984, "Radiation Properties Of Pu-238 Produced For Isotopic Power Generators". This appears to be the only unclassified report currently available with sufficient detail for detailed calculations. It should be noted, however, that there are apparent discrepancies between these data and those found in the classified literature.

Pu-238 metal was chosen as the RPCP fuel because of its expected lower radiation levels. Compared to the radiation properties of PuO_2, Pu-238 metal has a much smaller neutron emision rate and a slightly lower gamma photon emission in the 1-3 Mev range.

The isotopic composition of the fuel reported in DP-984 is:

Pu-236	120 ppm
Pu-238	81%
Pu-239	15%
Pu-240	2.9%
Pu-241	0.8%
Pu-242	0.1%

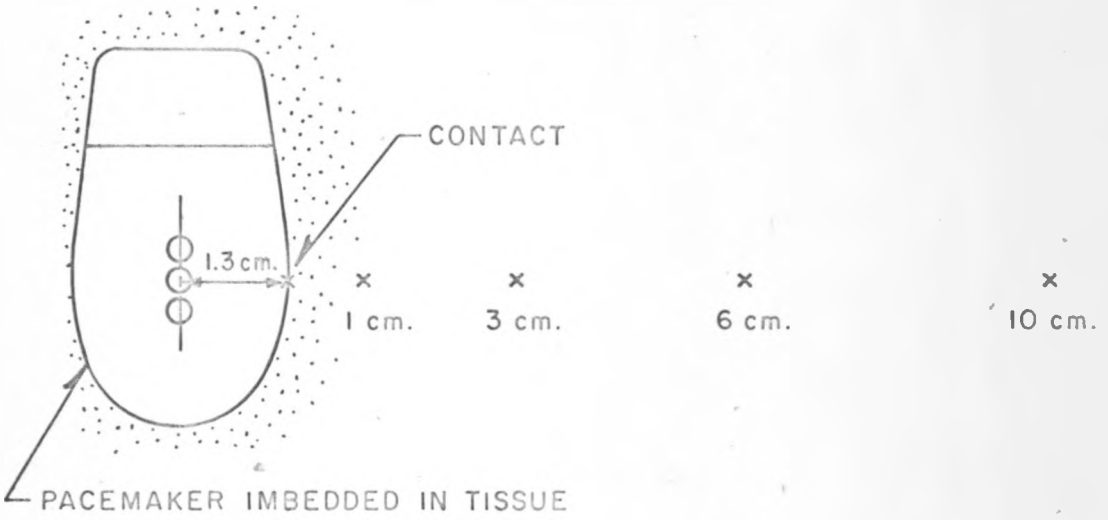


Figure 2

LOCATION OF DOSE POINTS
CARDIAC PACEMAKER DOSIMETRY CALCULATION

This is typical of most Pu-238 fuel available although significant variations in the Pu-236 content are possible. This variation is of significance in that Pu-236 causes an increase in the penetrating gamma dose because Tl-208 and Bi-212 daughters "grow in".

Gamma Radiation

The gamma radiation properties of typical Pu-238 metal as presented in DP-984 are shown in Table 1.

Table 1

Gamma Activity Of Pu-238 Metal (81% Pu-238)
Abundance, photons/(sec) (gm Pu-238 Metal)

Energy, Mev	$t^{(a)}$ = 1 day	t = 1 yr	t = 2.5 yr	t = 5.0 yr
0.04 - 0.5	2.4×10^{8}	2.4×10^{8}	2.4×10^{8}	2.3×10^{8}
0.5 - 1.0	3.7×10^{5}	3.8×10^{5}	4.4×10^{5}	5.5×10^{5}
1.0 - 2.0	2.4×10^{3}	2.8×10^{3}	4.6×10^{3}	8.3×10^{3}
2.0 - 3.0	8.1×10^{2}	9.8×10^{3}	5.2×10^{4}	1.4×10^{5}
3.0 - 5.0	1.6×10^{2}	1.6×10^{2}	1.6×10^{2}	1.5×10^{2}
5.0 - 7.0	2.6×10^{1}	2.6×10^{1}	2.6×10^{1}	2.5×10^{1}

(a) time since chemical separation

Absorption of photons below 1.0 Mev varies markedly with energy level for all materials, especially for heavy metals. Consequently, in order to refine the calculations the energy groups, .04-.5 and .5-1 Mev, were broken down further. This was done by considering the relative abundance of various gamma rays within these energy ranges. The resulting spectrum used with t = 1 day is shown in Table 2.

Table 2

Gamma Activity Of Pu-238 Metal And Pacemaker

t = 1 day

Mev	Photons/sec gm	Photons/sec For [a] .097 gm Pu-238 wire	Photons/sec For [a] .29 gm Pu-238
.044	1.95×10^{8}	1.89×10^{7}	5.67×10^{7}
.100	$.41 \times 10^{8}$	$.40 \times 10^{7}$	1.20×10^{7}
.152	5.0×10^{6}	4.85×10^{5}	1.46×10^{6}
.203	1.8×10^{4}	1.75×10^{3}	5.25×10^{3}
.760	2.66×10^{5}	2.58×10^{4}	7.74×10^{4}
.875	1.04×10^{5}	1.01×10^{4}	3.03×10^{4}
1.5	2.4×10^{3}	2.33×10^{2}	6.99×10^{2}
2.5	8.0×10^{2}	7.8×10^{1}	2.34×10^{2}
4.0	1.6×10^{2}	1.6×10^{1}	4.80×10^{1}
6.0	2.6×10^{1}	2.5	7.5

a. Source for reference design device contains three wires (one in each primary capsule) comprising .29 gm total Pu-238.

Neutron Radiation

Neutrons in pure Pu-238 metal originate mostly from spontaneous fissioning of the 238 isotope. With metal of presently attainable purity, however, there appears to be an increase in the neutron spectrum below 2.5 Mev. Unclassified Mound Laboratory data[1] indicate that the increase is probably due to an (alpha, neutron) reaction from low levels of fluoride impurity.

The addition of 30 per cent more neutrons, distributed according to the Po-α-F neutron spectrum in UCRL-3839[2] results in a spectrum which closely approximates the measured spectrum Coffey cites for the Mound Laboratory data, as shown in Table 3.

1. Letter of August 31, 1966 from D. L. Coffey to A. H. Kasberg.
2. UCRL-3839 "Neutrons From Small (α,n) Sources", W. N. Hess, July 1957.

Table 3

Calculated Neutron Activity Of Typical Pu-238 Metal

Mev	Spontaneous Fission	Pu-238- α-F (N/sec gm Pu-238 metal)	Total N/sec gm	N/sec Pacemaker
0 - .5	250	38	288	83.5
.5 -1	330	132	462	134.0
1 - 2	670	335	1005	291.5
2 - 3	410	103	513	148.8
3 - 4	200	-	200	58.0
4 - 5	150	-	150	43.5
5 - 6	81	-	81	23.5
6 - 7	27	-	27	7.8
7 - 8	19	-	19	5.5
8 - 10	7	-	7	2.0
10 - 13	3.5	-	3.5	1.0
	2147	608	2755	799.1

A significant increase in neutron dose rate is caused by about 5 ppm of F. Similar trace quantities of other low atomic weight elements will cause additional increases in neutron output. Because trace impurities are variable from batch to batch, it is likely that the neutron output could also vary. Specifications on impurity level will keep this variable within bounds but cannot eliminate it, because sensitivities of commonly used analytical techniques are limited. Perhaps atomic absorption spectroscopy (AAS), which will be operational at Mound in 1967, will permit control of this variable. AAS should yield detection limits of fractions of ppm for light elements. The current status of Pu-238 analytical capability at Mound Laboratory is summarized in Table 4.

Table 4

Mound Laboratory Capability Status For Analytical Chemistry Of Pu-238

Element	Present Detectability Limit, ppm	% Accuracy	Procedure	Require Method Development	Anticipated AAS Capability: Detection Limits ppm	Anticipated AAS Capability: Accuracy %
Fluorine	5	5	Spectrophotometry	No	-	-
Oxygen	1 (estimated)	20	Gas Chromatography	Yes	-	-
Beryllium	.5	20	Emission Spectroscopy	No	.03	2
Aluminum	5	20	Emission Spectroscopy	No	1	2
Sodium	100 (estimated)	20	Emission Spectroscopy	Yes	.04	2
Iron	5	3	Spectrophotometry	No	-	-
Nickel	5	20	Emission Spectroscopy	No	.2	2
Cobalt	5	20	Emission Spectroscopy	No	.2	2
Silicon	50 (estimated)	3	Spectrophotometry	Yes	-	-

a. From letter of 8-30-66 of D. L. Coffey to A. H. Kasberg.
b. Methods development work anticipated completion 10-1-66.
c. Atomic absorption spectroscopy expected to be in operation in early 1967.

DOSE RATE CALCULATIONS

Calculation Of The Radiation Flux

The intensity of radiation at a given point (radiation flux) is usually expressed as the number of particles or photons passing through a unit area in unit time. This quantity must be known in order to compute the dose rate at a given point.

The radiation flux from the pacemaker source is reduced by two mechanisms: (1) distance; (2) interaction of the particles or photons with matter between the source and the dose point. The first effect, strictly geometrical in origin, is very important in the pacemaker because tissue is only 1.3 centimeters away from the source. As the dose point moves away from the pacemaker the flux will be greatly reduced because of the decreasing solid angle between the dose point and the edges of the source. The pacemaker materials also attenuate the gamma portion of the radiation flux. This is critically important in reducing the more abundant low energy photons. The neutron shielding effect of the pacemaker materials is negligible.

The equations used to make these dosimetry calculations are found in "Reactor Shielding Design Manual", T. Rockwell (Macmillan And Co., Ltd.). The equations combine both effects of geometry and shielding.

Neutron Fluxes At Contact, 1 cm, 3 cm, 6 cm, and 10 cm

There is relatively little neutron self absorption in the Pu metal wires. Hence, each wire may be considered as a line source of neutrons and the flux is given by (Rockwell, page 348):

$$\phi = B \frac{S_L}{2\pi a} F(\theta, b)$$

where the dose point is equidistant from both source ends and where:

B = Buildup factors = 1, inasmuch as only first collision dose is considered.

S_L = Source strength of line source, neutrons/cm sec.

a = Perpendicular distance from line source to reference point, cm.

$$F(\theta, b) = \int_0^{\theta} e^{-b} \sec \beta \, d\beta$$

β = Dummy reference angle

θ = Angle subtended by the source end and the reference point location on the perpendicular bisector of the line source.

$$b = \sum \mu_i x_i + \mu_s Z$$

where μ = Macroscopic cross section of shields 1,2.....n, cm^{-1}.

μ_s = Macroscopic cross section of source material, cm^{-1}.

Z = Effective self-attenuation distance, cm

This expression for flux at a point includes the combined effects of distance and shielding. Values of $F(\theta,b)$ for various θ, the angle subtended, and b, the sum of the relaxation lengths of absorbers, are listed as a series of plots by Rockwell (see pages 385-390).

Table 5 gives the results of the neutron flux calculations for all energy groups at contact, 1 cm, 3 cm, 6 cm, and 10 cm for the reference design RPCP.

Calculated Gamma Fluxes At Contact, 1 cm, 3 cm, 6 cm, and 10 cm

Pu metal has a high degree of self shielding for gamma radiation because of its high electron density and relatively tightly bound k shell electrons. Consequently, the equation for a infinite cylindrical source (see Rockwell, page 360) was chosen for gamma flux calculation:

$$\phi = \frac{B\ S_v\ Ro^2}{2\ (a + Z)}\ F(\theta,b)$$

where the dose point is equidistant from both ends and

S_v = The Source strength of a volume source, photons/cm^2 sec

Ro = Radius of cylindrical source, cm

a = Distance from reference point to source outside, cm.

θ = Angle subtended by the source end and the reference point located on the perpendicular bisector of the cylindrical source.

In this case in order to evaluate $F(\theta,b)$ for gamma radiation, b, the sum of the shield relaxation length for various energies must be determined. The first step involves listing the thickness of various absorbing materials of the RPCP reference design as follows:

1. Capsule	cm	Element
Primary - .030 inch Ta-10W	.0686	Ta
	.0076	W
Secondary - .009 inch Hastelloy C	.0036	Cr
	.0009	W
	.0037	Mo
	.0006	Co
	.0013	Fe
	.0129	Ni
Pt Coating .001 inch Pt	.0025	Pt
Tertiary .0165 inch Ti (least thickness)	.0420	Ti

Table 5

Calculated Neutron Fluxes, RPCP

Energy Level Mev	S_L neutron/sec cm	ϕ, Neutron Flux in neutron/cm^2 sec At				
		Contact	1 cm	3 cm	6 cm	10 cm
0 - .5	11	3.05	1.20	.350	.097	.023
.5 - 1	17.6	4.80	1.95	.552	.154	.038
1 - 2	38.2	10.50	4.20	1.208	.336	.084
2 - 3	19.5	5.33	2.070	.613	.171	.043
3 - 4	7.6	2.20	.860	.253	.070	.018
4 - 5	5.7	1.65	.640	.190	.053	.013
5 - 6	3.1	.90	.350	.104	.029	.007
6 - 7	1.03	.30	.120	.035	.010	.002
7 - 8	.72	.21	.082	.024	.007	.002
8 - 10	.27	.09	.035	.010	.003	.001
10 - 13	.13	.04	.016	.004	.001	-

Parameters

Location	a	θ	b	$F(\theta,b)$	Tissue Neutron Attenuation Factor [a]
Contact	1.3	45°	0	.76	1.0
1 cm	2.3	30°	0	.53	1.0
3 cm	4.3	17°	0	.29	.96
6 cm	7.3	10°	0	.175	.77
10 cm	11.3	6.5°	0	.1	.48

a. HW-21168, "Shields of Neutron Sources", F. R. Jones, General Electric Company, Richland, Washington, 1958.

2. Thermocouple Assembly

Single Tape (inches)	40 layers	
.007 quartz cloth	.711 cm	glass
.001 Cupron wire (20% of Area)	.020 cm	Cu
.001 Tophel wire (20% of Area)	.020 cm	Ni
.0005 Titanium foil	.051 cm	Ti

3. Case .020 inch Titanium .051 cm Ti

The total thickness of elements are then:

Ta	.0686 cm
W	.0085 cm
Cr	.0036 cm
Mo	.0037 cm
Co	.0006 cm
Fe	.0013 cm
Ni	.0329 cm
Pt	.0025 cm
Ti	.1440 cm
Quartz	.7110 cm
Cu	.0200 cm

Table 6 lists the number of relaxation lengths (μx) for each absorber and energy level. The values for μ, the linear absorption coefficient, were computed by multiplying the mass absorption coefficient as listed in ORNL-421[3] by the element density.

Applicable values for the function $F(\theta,b)$ are listed in Table 7. Values for 1.5, 2.5 and 4.0 Mev were omitted at 3 cm distance or greater, because the associated geometric attenuation leads to negligible contribution from from those energy groups.

The flux calculations are shown in Table 8.

3. ORNL-421, "Absorption of Gamma Rays", W. S. Synder and J. L. Powell March 14, 1950.

Table 6

Gamma Absorbers In The Reference Design RPCP

Element	Equivalent Thickness x (cm)	Energy Level, Mev																	
		.044		.100		.152		.203		.760		.875		1.5		2.5		4.0	
		μ	μx	μ	μx	μ	μx	μ	μx	μ	μx	μ	μx	μ	μx	μ	μx	μ	μx
Ta	.0686	78.02	5.352	59.76	4.100	23.24	1.590	12.12	.831	1.39	.095	1.21	.083	.797	.055	.631	.043	.65	.044
W	.0085	90.07	.766	69.48	.590	27.02	.230	14.01	.119	1.62	.014	1.41	.012	.926	.008	.733	.006	.75	.006
Cr	.0036	17.75	.064	2.70	.010	1.42	.005	1.01	.004	.48	.002	.45	.002	.326	.001	.263	.001	.23	.001
Mo	.0037	25.50	.094	3.88	.014	2.04	.007	1.53	.006	.69	.003	.64	.002	.469	.002	.377	.001	.34	.001
Co	.0006	22.25	.013	3.38	.002	1.78	-	1.34	.001	.61	-	.56	-	.409	-	.329	-	.29	-
Fe	.0013	19.50	.025	2.96	.004	1.56	-	1.17	.002	.53	-	.49	.001	.359	-	.289	-	.26	-
Ni	.0329	22.25	.732	3.38	.111	1.78	.059	1.34	.044	.61	.020	.56	.018	.409	.013	.329	.011	.29	.010
Pt	.0025	121.9	.305	96.30	.240	35.31	.088	18.19	.045	1.93	.005	1.65	.004	1.07	.003	.856	.002	.86	.002
Ti	.1440	11.25	1.620	1.71	.246	.90	.130	.68	.098	.31	.045	.28	.040	.207	.030	.167	.024	.15	.021
Quartz	.7110	1.17	.832	.44	.313	.36	.256	.31	.220	.18	.128	.17	.121	.130	.092	.099	.070	.08	.058
Cu	.0200	29.37	.587	4.00	.080	1.93	.039	1.42	.028	.61	.012	.53	.011	.392	.008	.329	.006	.29	.006
Totals μx			10.390		5.710		2.404		1.398		.325		.294		.284		.164		.149

Self Absorbtion In Plutonium Source

Pu	.0287	183.4	5.26	25.1	.72	46.3	1.33	23.2	.67	2.22	.06	1.81	.05	1.08	.03	.849	.024	.85	.024
μx			2.2		.38		.70		.36		~0		~0		~0		~0		~0

Table 7

Values Of The Function F(θ,b) For RPCP Gamma Dose Calculated Contribution
(From Rockwell, "Reactor Shielding Design Manual", p. 385-390)

1. At Contact, a = 1.3, θ = 45°

Gamma Energy, Kev	44	100	152	203	760	875	1500	2500	4000
$b = \sum \mu_i x_i + \mu_s Z$	12.590	6.09	3.104	1.758	.325	.294	.284	.164	.149
$F(45^\circ,b)$	1.1×10^{-6}	10^{-3}	2.5×10^{-2}	.11	.50	.52	.52	.64	.65

2. At 1 cm, a = 2.3 θ = 29.50°

μx, 1 cm H_2O	.235	.165	.145	.133	.078	.073	.055	.044	.033
$b = \sum \mu_i x_i + \mu_x Z$	12.825	6.255	3.249	1.891	.403	.367	.339	.208	.182
$F(30^\circ,b)$	8×10^{-7}	7.6×10^{-4}	1.7×10^{-2}	.07	.35	.36	.37	.43	.44

3. At 3 cm, a = 4.3, θ = 17°

μx, 3 cm H_2O	.705	.495	.435	.394	.234	.219	-	-	-
$b = \sum \mu_i x_i + \mu_s Z$	13.295	6.585	3.539	2.152	.559	.513	-	-	-
$F(17^\circ,b)$	4×10^{-7}	3.7×10^{-4}	8×10^{-3}	3.5×10^{-2}	.17	.18	-	-	-

4. At 6 cm, a = 7.3, θ = 10°

μx, 6 cm H_2O	1.410	.990	.870	.788	.468	.438	-	-	-
$b = \sum \mu_i x_i + \mu_s Z$	14.000	7.08	3.974	2.546	.793	.732	-	-	-
$F(10^\circ,b)$	1.4×10^{-7}	1.4×10^{-4}	3.3×10^{-3}	1.3×10^{-2}	7.8×10^{-2}	8.4×10^{-2}	-	-	-

5. At 10 cm, a = 11.3, θ = 6.5°

μx, 10 cm H_2O	2.35	1.65	.145	1.33	.78	.73	-	-	-
$b = \sum \mu_i x_i + \mu_s Z$	14.94	7.74	4.554	3.088	1.105	1.024	-	-	-
$F(6.5^\circ,b)$	3.5×10^{-8}	5×10^{-5}	1.2×10^{-3}	5.2×10^{-3}	2.2×10^{-2}	4×10^{-2}	-	-	-

Table 8

Calculated Gamma Fluxes, RPCP

Energy Level Mev	Photons/ sec gm	Photons/ sec .097 gm	S_v, photons/ sec cm^3	Photons/sec cm^2 At				
				Contact	1 cm	3 cm	6 cm	10 cm
.044	1.95×10^8	$.19 \times 10^8$	2.9×10^9	1.0	-	-	-	-
.100	$.41 \times 10^8$	3.98×10^6	6.1×10^8	192.0	82.5	21.5	4.8	1.2
.152	5×10^6	4.85×10^5	7.4×10^7	581.0	224.0	56.2	13.7	3.4
.203	1.8×10^4	1.7×10^3	2.6×10^5	9.0	-	-	-	-
.760	2.67×10^5	2.59×10^4	4×10^6	628.0	249.0	64.6	17.5	3.3
.875	1.04×10^5	1×10^4	1.5×10^6	245.0	96.0	25.7	7.1	2.3
1.500	2.4×10^3	2.3×10^2	3.5×10^4	5.7	2.3	-	-	-
2.500	8×10^2	77.6	1.2×10^4	2.4	.9	-	-	-
4.00	1.6×10^2	15.5	2.4×10^3	.49	.2	-	-	-

Gamma Buildup

Gamma attenuation is made up of three distinct processes commonly called photoelectric capture, pair production and Compton scattering. The first two of these are annihilation processes in which the gamma ray is destroyed. In the third process the gamma photon interacts with an electron and is deflected to another direction, losing only part of its original energy. As a result, some gammas that were counted as lost in evaluating the linear absorption coefficient, μ, do penetrate the shield under consideration and thus the net reduction is not as great as predicted by the exponential term $e^{-\mu x}$. This leads to the use of the use of the buildup factor, B in dealing with Compton scattering effects.

Dose build-up factors have been computed as a function of μx, the relaxation length, and are plotted as graphs in the Radiological Health Handbook, U.S. Department of Health, Education and Welfare, Public Health Service, Revised edition, Sept., 1960, (PB 121784R).

The number of relaxation lengths for the RPCP reference design material shielding is only greater than one for low energy gamma radiation. Applying a build up factor in these high atomic weight materials is not appropriate for low energy radiation. Most absorption occurs by photoelectric capture and there is relatively little scattered radiation. For the higher energy photons μx is less than one and the buildup factor is negligible.

Buildup in the surrounding tissue was considered. Table 9 shows the buildup with distances and energies where B is greater than one.

Table 9

Buildup Factors, B [a,b]

Distance	100 Kev	152 Kev	760 Kev	875 Kev
6 cm	4.5	3.0	1.0	1.0
10 cm	9.0	6.0	2.0	1.8

a. "Radiological Health Handbook", page 150.

b. Values listed are for point isotropic source in water.

These buildup factors were not used directly in computing the corrected flux in that a different approach was used as shown in the following section on Flux to Dose Conversion (also refer to note d of Table 11).

Conversion Of The Radiation Flux To Dose Rate

Radiation doses are expressed in terms of the energy imparted to matter by ionizing radiation per unit mass of irradiated material at the point of interest. In this connection one rad of absorbed dose is equivalent to the absorption of 100 ergs per gram of target material.

An RBE (Relative Biological Effect) factor, or for radiation protection purposes, a quality factor (QF), is applied to the absorbed dose obtained for radiation having high linear energy transfer (LET). This permits the use of a biologically equivalent dose unit, the rem defined by Rem = Rad x QF.

Although the linear energy transfer varies for different gamma energies, the QF is customarily taken as one for x and gamma rays.

Fast neutron irradiation of tissue results primarily in high LET proton recoils. For neutrons having energies up to 10 Mev, a QF of 10 is applied to the absorbed dose. This QF is based principally on the ability of chronic neutron irradiation to produce delayed effects such as cataracts in the eye lens. The applicability of this high QF to local fatty and muscular tissue irradiation, as in the case of the implanted RPCP, is questionable.

However, in order to present conservative numbers in this preliminary radiation analysis, the absorbed dose rate was computed and a factor of 10 applied to obtain the rem dose rate. The calculated neutron dose rates are listed in Table 10.

For the conversion of gamma flux to dose rates, the following relation was used:

$$\text{Dose Rate} = \frac{\phi\left[\frac{\text{photons}}{\text{cm}^2\ \text{sec}}\right]\ E\left[\frac{\text{Mev}}{\text{photon}}\right]\ \mu_t(\text{cm}^{-1})\ K\left[\frac{\text{ergs}}{\text{Mev}}\right] 3.6 \times 10^3 \left[\frac{\text{sec}}{\text{hr}}\right]}{\rho_t \left[\frac{\text{gm}}{\text{cm}^3}\right]\ 100\ (\text{ergs/gm/rad})\ (.001)\left[\frac{\text{rad}}{\text{mrad}}\right]}$$

$$= \text{mrad/hr}$$

μ_t = linear absorption coefficient in tissue
K = 1.6×10^{-6} ergs/Mev
ρ_t = density of tissue = 1.0 gm/cm^3

A dimensional check of the above relation shows that it represents the true absorbed dose rate.

Table 10

Conversion Of Neutron Flux To Dose Rate For Reference Design RPCP

En Mev	*Rad neutron/cm²sec	Contact		1 cm		3 cm		6 cm		10 cm	
		∅	mrads/hr	∅	mrads/hr	∅	mrads/hr	∅	mrads/hr	∅	mrads/hr
0 - .5	.0086	3.05	.026	1.20	.010	.350	.003	.097	.001	.023	-
.5 - 1	.0137	4.80	.066	1.95	.027	.552	.008	.154	.002	.038	.001
1 - 2	.0146	10.50	.153	4.20	.061	1.208	.018	.336	.005	.084	.001
2 - 3	.0155	5.33	.083	2.070	.032	.613	.010	.171	.003	.043	.001
3 - 4	.0180	2.20	.040	.860	.015	.253	.005	.070	.001	.108	-
4 - 5	.0210	1.65	.035	.640	.013	.190	.004	.053	.001	.013	-
5 - 6	.0225	.90	.020	.350	.008	.104	.002	.029	.001	.007	-
6 - 7	.0256	.30	.008	.120	.003	.035	.001	.010	-	.002	-
7 - 8	.0256	.21	.005	.082	.002	.024	.001	.007	-	.002	-
8 -10	.0252	.09	.002	.035	.001	.010	-	.003	-	.001	-
10 -13	.0252	.04	.001	.016	-	.004	-	.001	-	-	-
Total (mrads/hr)			.439		.172		.052		.014		.003
mrem/hr			4.39		1.72		.52		.14		.03

*W. S. Synder and C. Newfield "Conversion of Neutron Flux To Physical Dose", Radiation Research, 6(1):67(1953).

Table 11

*Conversion Of Gamma Flux To Dose Rate For Reference Design RPCP

E (gamma)	u_t	Contact		1 cm		3 cm		6 cm		10 cm	
		∅	mrads/hr	∅	mrads/hr	∅	mrads/hr	∅	mrads/hr	∅	mrads/hr
.044 Mev	.235	1.0	.017	-	-	-	-	-	-	-	-
.100	.165	192.0	.528	82.5	.230	21.5	.060	4.8	.068	1.2	.027
.152	.145	581.0	2.097	224.0	.838	56.2	.213	13.7	.153	3.4	.072
.203	.133	9.0	.003	-	-	-	-	-	-	-	-
.760	.078	628.0	6.215	249.0	2.505	64.6	.660	17.5	.180	3.3	.066
.875	.073	245.0	2.633	96.0	1.041	25.7	.285	7.1	.078	2.3	.043
1.500	.055	5.7	.078	2.3	.032	-	-	-	-	-	-
2.500	.044	2.4	.044	.9	.018	-	-	-	-	-	-
4.00	.033	.49	.012	.2	.006	-	-	-	-	-	-
	Total	11.627		4.670		1.218		.479		.208	

Notes

a. *Listed flux is for single wire.

b. Dose rate is for 3 wires at contact and 1 cm;2(.95) D was used for end wire dose.

c. At other distances, the dose for one wire was multiplied by 3.

d. Dose at 6 and 10 cm includes buildup, listed in Table 9.

Table 11 list the calculated gamma dose rates for the reference design RPCP. Figure 3 shows the results of the calculations for the gamma and neutron depth rates for the reference design RPCP.

INCREASE OF GAMMA DOSE WITH TIME

The production of Pu-238 by irradiating Np-237 with reactor neutrons also produces some Pu-236 by the (n,2n) reaction. The RPCP fuel is assumed to have 120 ppm Pu-236. Pu-236 is the parent of a long decay chain that terminates in stable Pb-208. Most of the contribution of these members of the Pu-236 decay chain to the gamma activity of the Pu-238 fuel is insignificant when Pu-236 is at the ppm level. However two daughters, Bi-212 and Tl-208 contribute significant energetic gamma radiation. The increase with time for different energy groups was tabulated earlier. The calculated dose rate increase with time for the RPCP is shown on Figure 4. The contribution to the increase by energy group is shown in Table 12.

Table 12

Increase In Contact Dose Rate With Tissue For Reference Design RPCP

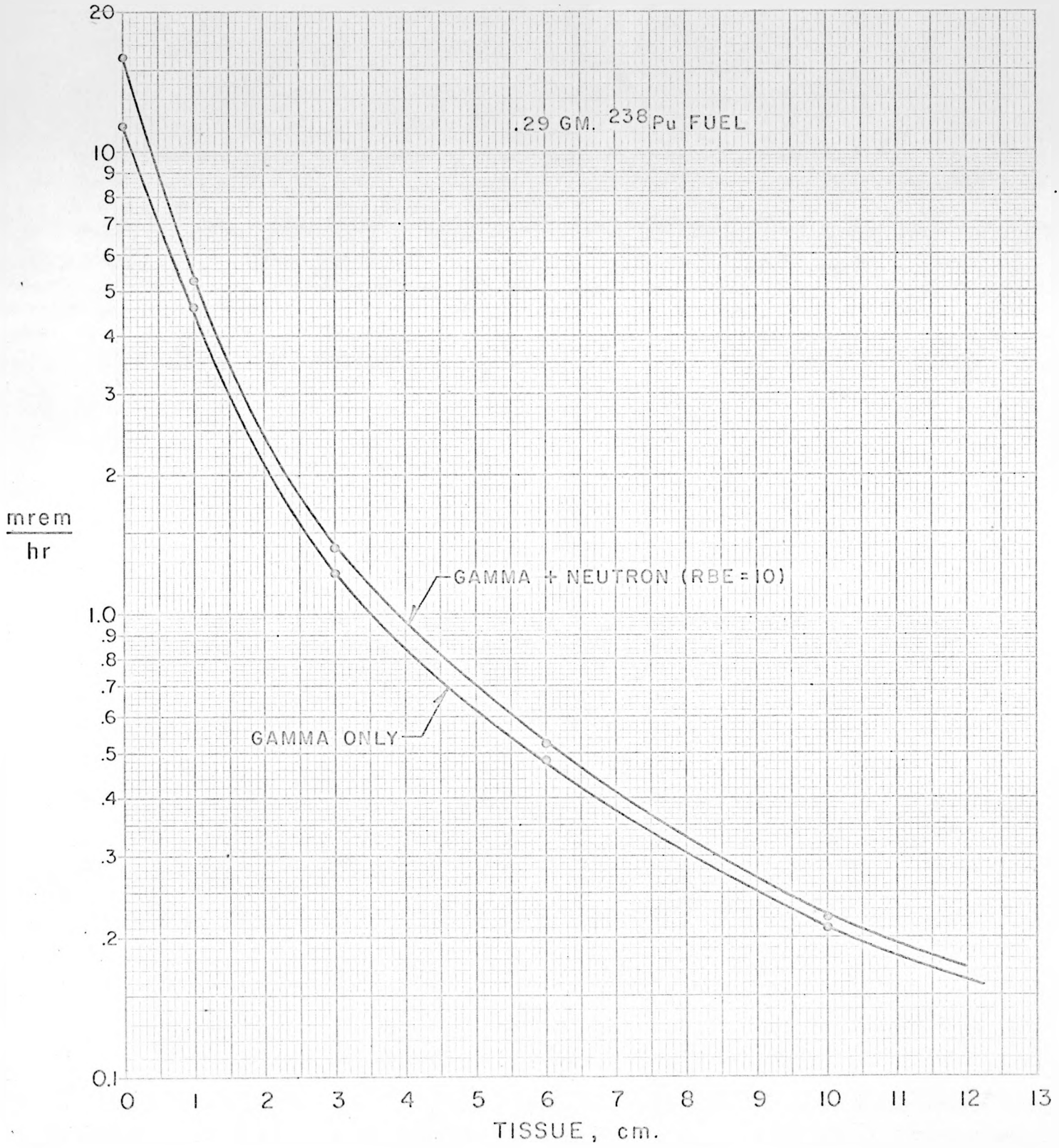
Dose Rate, mrad/hr

Energy Group, Mev	t = 1 day	t = 1 yr	t = 2.5 yr	t = 5 yr	t = ∞
.5 - 1	8.848	9.087	10.520	13.148	26.296
1 - 2	.078	.091	.150	.270	.540
2 - 3	.044	.530	2.820	7.600	15.200
	8.920	9.708	13.490	21.018	42.036
-8.920 dose correction back to t = 1 day					
Increase with time due to Pu-236	0	.738	4.520	12.048	33.066
+11.627 total dose at t = 0 at contact					
Total dose rate	11.627	12.365	16.147	23.675	44.693

CALCULATED DEPTH DOSE RATES, ^{238}Pu METAL FUELED

CARDIAC PACEMAKER IN TISSUE

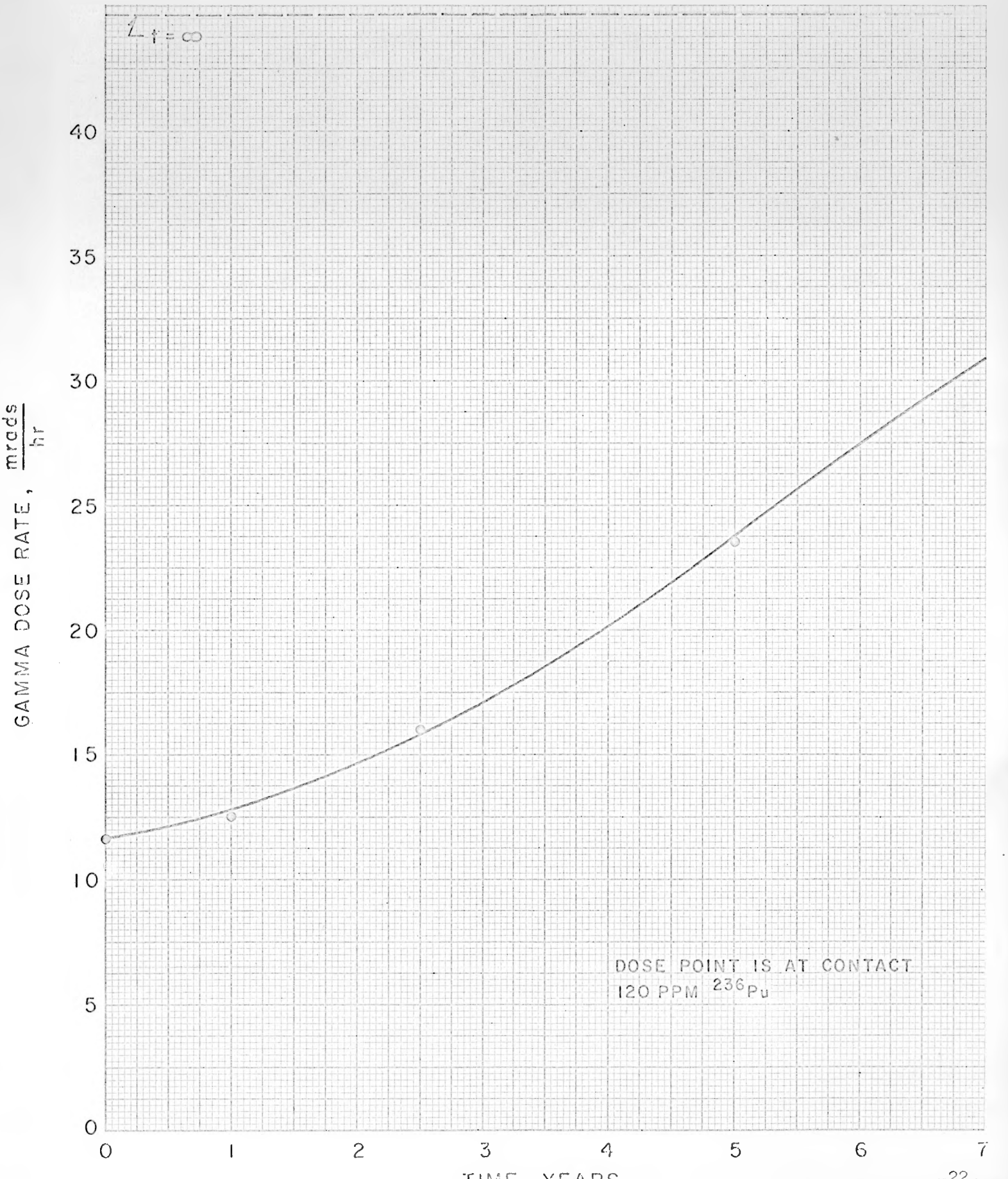
Figure 3



CALCULATION OF INCREASE IN GAMMA DOSE RATE WITH AGE OF ^{238}Pu SINCE CHEMICAL SEPARATION

CARDIAC PACEMAKER

Figure 4



PRELIMINARY INTERPRETATION OF CALCULATED DOSE RATES

The dose rates calculated in this report are somewhat higher than those presented in the RPCP proposal, (or the 5.0 mrem/hr on the surface specified as the maximum allowable operation dose rate in Section 1.a.A(10) of the Contract). This was due principally to calculating air dose and not including spectral effects for both neutron and gamma radiation in the proposal. However, the rapid fall off of total dose with distance should be noted. At 2 inches from the RPCP the maximum calculated dose rate is 0.7 mrem/hr. Thus the total tissue irradiated to levels higher than specified in the proposal is very small. In fact, it is probable that all such tissue will be fatty or scar tissue which is notably radioresistant.

The calculated dose rates assumed chemically pure (except for 5 ppm F) Pu-238 metal, 120 ppm Pu-236 and activity levels published in DP-984. Any deviation from these specifications will change the calculated dose rate. This consideration along with normal deviations of actual configuration from assumed model in the calculations makes a measurement program imperative. The study plan was outlined previously in the Program Plan NUMEC 3731-1.0-1 under the subtask 5.3.

DOSIMETRY PLAN FOR RPCP

In order to correctly evaluate the external radiation hazard from an implanted RPCP, detailed dosimetry of the neutron and gamma radiation is necessary. A brief summary of the intended measurement program is given below.

1. Gamma Measurements

   Absorbed energy measurements will be made with small thermoluminescent dosimetry (TLD). Gamma spectrometry will utilize Ge(Li activated) detectors for resolution of the high energy spectrum (100-3,000 Kev) and NaI (Tl activated) 1 mm crystals for the low energy spectrum (10-100 Kev).

   The gamma depth dose measurements testing program will proceed along the following lines:

   a. Set up and calibrate equipment during October, 1966.

   b. Inter-calibrating of instruments and standards with Brookhaven National Laboratory and the National Bureau of Standards. This will be carried out with BNL during November and with the NBS early in 1967.

   c. Scaling experiments will be carried out upon receipt of fueled capsules from Mound Laboratory, some time after December 1, 1966.

d. TLD measurements will be made along three axes up to 30 cm depth in water. LiF microdisimeters will be used for short depth dosimetry and for large and small diameter bone dosimetry. CaF hot press dosimeters will be used at depths where the dose rate is low. Liter polyethylene jars filled with hollow polyethylene spheres and water will be used to simulate lung density.

2. Neutron Measurements

Fast neutron fluxes will be measured with NTA fiber packets and poison fragment damage dosimeters. The FFD dosimeters will employ Np-237 foils in close contact with Lexan plastics. The damage locations will be etched with hot caustic solution and the fission fragment tracks counted under the microscope. The fast spectrum will be estimated by microscopic examination of the track lengths in NTA film.

Additional neutron measurements will be carried out with LiF dosimeters enriched in the Li-6 and Li-7 isotopes.

If possible, a stilbene organic crystal will be borrowed from Mound Laboratory to measure the neutron spectrum at depths in tissue. An alternate procedure will be to send an RPCP to Mound for these measurements.

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