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Health and Safety
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^{252}Cf SHIELDING GUIDE

by

D. H. Stoddard
H. E. Hootman

Approved by

C. M. Patterson, Research Manager
Radiological Sciences Division

and

P. L. Roggenkamp, Research Manager
Theoretical Physics Division

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E. I. DU PONT DE NEMOURS & COMPANY
SAVANNAH RIVER LABORATORY
AIKEN, S. C. 29801

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Larry E. Williams
Authorizing Official

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ABSTRACT

Neutron and gamma ray dose rates have been calculated for ^{252}Cf source interaction with most common shielding materials. Information has been compiled to assist ^{252}Cf evaluators and prospective users in determining:

- Radiations involved with the use of ^{252}Cf .
- Personnel dose guides and shipping dose rate levels for ^{252}Cf .
- Comparative thicknesses of different shield materials required for a specific ^{252}Cf dose reduction.
- Approximate flux levels for neutron activation and background radiation levels in applications.

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INTRODUCTION

A neutron-emitting radioisotope such as ^{252}Cf poses new and unique shielding problems both with respect to the nature of the radiations to be shielded and the portability requirements in some of its proposed applications.¹ Californium-252 shielding is similar to nuclear reactor shielding in that the spontaneously emitted neutrons react with the shield to produce secondary radiation sources distributed throughout the shield. In many cases this secondary radiation is the dominant contributor to the penetrating dose rate.

Modern methods of shielding calculation use large, high-speed computers to solve for neutron and gamma dose rates. The complex multi-energy transport equations used require detailed coupled neutron-gamma cross section sets to ensure better than a factor of two accuracy for the required shield dose reductions of 10^4 to 10^5 .

An aim of this report is to supply the ^{252}Cf user a series of attenuation curves with which particular shielding needs may be evaluated.

THE RADIATION PROPERTIES OF ^{252}Cf

Californium-252 decays 97% by the emission of alpha particles, and 3% by the spontaneous fission process which causes the emission of 2.4×10^{12} neutrons/sec from one gram of this material.

Although isotopic compositions will vary somewhat depending on the mode of production and decay time, the penetrating* radiation from californium sources will primarily be neutrons and gamma rays from the isotope ^{252}Cf and from its fission products.

The general nuclear properties of the californium isotopes are presented in Table I.² The penetrating radiations of ^{252}Cf are listed in the following sections.

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* Penetrating radiation is that which will penetrate the horny layer of the skin.

TABLE I

General Nuclear Properties of Californium Isotopes

Isotope	Spontaneous Fission Half-Life	Neutrons Per Fission of Pure Isotope	Alpha or Beta Decay Half-Life	Specific Activity of Pure Isotope, Ci/g
^{249}Cf	$1.5 \times 10^9 \text{ y}$	3	360 y	4
^{250}Cf	$(1.73 \pm 0.06) \times 10^4 \text{ y}$	3	11 y	1.31×10^2
^{251}Cf	-	-	$\sim 1500 \text{ y}$	0.95
^{252}Cf	$85.5 \pm 0.5 \text{ y}$	3.80 ± 0.035	$2.646 \pm 0.004 \text{ y}$	5.37×10^2
^{253}Cf	-	-	$18 \pm 3 \text{ d}$	2.87×10^4
^{254}Cf	$61.9 \pm 1.1 \text{ d}$	3.9 ± 0.14	-	-

GAMMA ACTIVITY OF ^{252}Cf

The gamma activity of ^{252}Cf consists of gamma rays from:

- The alpha decay process.
- Prompt spontaneous fission.
- Fission products continuously produced by spontaneous fission.

Gamma radiations from each of these sources are described in the following sections.

Alpha Decay

Gamma rays reported for the alpha decay of ^{252}Cf are listed in Table II.

TABLE II

Gamma Rays from ^{252}Cf Alpha Decay

Energy, Mev	Abundance, photons/(sec)(g of nuclide)
0.043	2.78×10^9
0.100	2.0×10^9

Spontaneous Fission

The energy and abundance of prompt gamma rays from spontaneous fission are listed in Table III.

Equilibrium Fission Products

The fission products formed from spontaneous fission approach equilibrium within a few hours after separation. Equilibrium fission product gamma activities are listed in Table III.

TABLE III

Gamma Rays from Spontaneous Fission of ^{252}Cf

Energy, Mev	Abundance, photons/(sec)(g of nuclide)		
	Prompt Gammas	Equilibrium Fission Product Gammas	Total
0 - 0.5	3.3×10^{12}	1.3×10^{12}	4.6×10^{12}
0.5 - 1.0	1.7×10^{12}	4.0×10^{12}	5.7×10^{12}
1.0 - 1.5	7.7×10^{11}	9.1×10^{11}	1.7×10^{12}
1.5 - 2.0	4.2×10^{11}	3.5×10^{11}	7.7×10^{11}
2.0 - 2.5	2.2×10^{11}		2.2×10^{11}
2.5 - 3.0	1.1×10^{11}		1.1×10^{11}
3.0 - 3.5	5.6×10^{10}		5.6×10^{10}
3.5 - 4.0	3.0×10^{10}		3.0×10^{10}
4.0 - 4.5	1.7×10^{10}		1.7×10^{10}
4.5 - 5.0	8.2×10^9		8.2×10^9
5.0 - 5.5	4.9×10^9		4.9×10^9
5.5 - 6.0	1.8×10^9		1.8×10^9
6.0 - 6.5	1.0×10^9		1.0×10^9

X-rays

Although X-rays are relatively high in abundance, those produced are very low in energy (most are <40 keV) and are not included in this report.

BETA ACTIVITY OF ^{252}Cf

No beta radiation has been reported from the decay process. The beta radiation associated with the equilibrium fission products during spontaneous fission is easily absorbed and is not included in this report.

NEUTRON ACTIVITY OF ^{252}Cf

The neutron radiation from ^{252}Cf consists principally of neutrons from spontaneous fission. A second source of neutrons is the α, n neutrons from the reaction of alpha particles with light elements. Table IV lists the neutrons present from spontaneous fission.

TABLE IV

Neutrons from Spontaneous Fission of ^{252}Cf

Energy, Mev	Neutrons/(sec) (g of nuclide)
0 - 0.5	2.8×10^{11}
0.5 - 1.0	3.7×10^{11}
1.0 - 2.0	7.6×10^{11}
2.0 - 3.0	4.6×10^{11}
3.0 - 4.0	2.8×10^{11}
4.0 - 5.0	1.6×10^{11}
5.0 - 6.0	5.6×10^{10}
6.0 - 7.0	4.0×10^{10}
7.0 - 8.0	1.3×10^{10}
8.0 - 10.0	9.9×10^9
10.0 - 13.0	2.2×10^9
Total	2.4×10^{12}

Neutrons associated with ^{252}Cf can also be represented by the Watt formula.³

$$N(E) = 0.373 \exp(-0.88E) \sinh(2.0E)^{\frac{1}{2}}$$

where $N(E)$ is the fraction of neutrons per unit energy range, and E is the neutron energy in Mev.

UNSHIELDED DOSE RATE

The neutron and gamma dose rate in air from ^{252}Cf may be determined by multiplying the unshielded neutron flux by the $4\pi r^2 D(r)$ factors listed below:

Type Radiation	Dose Rate $4\pi r^2 D(r)$	
	$\left(\frac{\text{mrad-hr}^{-1}}{\text{neutron-sec}^{-1}} \right) \text{cm}^2$	$\left(\frac{\text{mrem-hr}^{-1}}{\text{neutron-sec}^{-1}} \right) \text{cm}^2$
Neutron	1.43×10^{-2}	1.22×10^{-1}
Gamma	8.58×10^{-3}	8.58×10^{-3}
Total	2.29×10^{-2}	1.31×10^{-1}

For example, the total dose rate (D_T) at 100 cm from a 100-mg source of ^{252}Cf is

$$D_T = \frac{1 \times 10^2 \text{ mg} \times 2.4 \times 10^9 \text{ neutrons/sec-mg}}{4\pi(100)^2 \text{ cm}^2} \times 1.31 \times 10^{-1} \left(\frac{\text{mrem-hr}^{-1}}{\text{neutron/sec}^{-1}} \right) \text{cm}^2 = 2.5 \times 10^5 \text{ mrem/hr}$$

REGULATIONS

PERSONNEL DOSE GUIDES⁴

Federal regulations, effective January 1, 1961, and in effect when this document was published, permit no more than the following radiation doses to any person 18 years of age or older:

- 5 rems multiplied by the number of years beyond age 18, and the dose in any consecutive 13 weeks not to exceed 3 rems for the critical organs (whole body, head and trunk, active blood forming organs, eyes, or gonads).
- 30 rems/yr to the skin (not to exceed 7.5 rems in any 13 consecutive weeks).
- 75 rems/yr and the dose in any 13 consecutive weeks shall not exceed 18.75 rems for the hands, forearms, feet, and ankles.

Any person less than 18 years old may receive no more than 10% of the above doses.

Federal regulations urge that every reasonable effort be made to maintain radiation exposures as far below these limits as practicable, that is, "as low as is practicably achievable taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety and in relation to the utilization of atomic energy in the public interest."⁴ At Savannah River, continuously occupied areas are shielded to <1 mrem/hr dose rate. This assures personnel exposures of <2 rem/yr when working 250 days per year. Other AEC sites shield to 2.5 mrem/hr, which assures doses within the regulations.

SHIPPING REGULATIONS⁵

At the time of publication of this document, shielded shipping casks for general use require shielding to dose rate levels of <200 mrem/hr at the accessible surface of the container and/or <10 mrem/hr at 3 feet from the accessible external surface.

Packages exceeding these levels may be transported by exclusive use of the vehicle (except aircraft) if the following regulations are not exceeded:

- 1,000 mrem/hr at 3 ft from the external surface of the package (closed transport vehicle only).
- 200 mrem/hr at any point on the external surface of the car or vehicle (closed transport vehicle only).

- 10 mrem/hr at 6 ft from the external surface of the car or vehicle.
- 2 mrem/hr or equivalent in any normally occupied position in the car or vehicle except this does not apply to private motor carriers.

SHIELD MATERIAL CHARACTERISTICS

ATTENUATION CALCULATIONS

The collection of dose attenuation calculations presented in Figures 4 through 48 provides a handy and compact shield reference for ^{252}Cf users and potential evaluators. In these figures dose rates are normalized to 1 fission neutron/sec, and multiplied by $4\pi r^2$ to remove the effect of spherical divergence.

The dose rate listed is the surface dose rate to tissue surrounding the shielding material. In most hydrogenous materials, this dose rate is almost the same as if it were in an infinite thickness of the shielding material. Some suggested uses are:

- Comparisons of Shield Materials

Relative effectiveness of different shield materials may be evaluated by summing the dose constituents at a common thickness for the materials being compared. For example, such a comparison between "Benelex"* 70 and water indicates the former material to be four times as effective in dose reduction at thicknesses of 120 cm, but essentially the same at thicknesses less than 30 cm.

- Estimations of Shield Thickness

^{252}Cf may usually be assumed to be a point source; however, the cask or shield may be any of a number of shapes. The effect of shield configuration on the presented dose data follows.

Applications

Dose Rates at the Surface of a Spherical Shield

Total dose rate at the surface of a spherical shield (Figure 1) may be determined by adding the neutron dose rates and the gamma dose rates at the common shield thickness and multiplying the ^{252}Cf source weight in grams times $1/4\pi r^2 \times 2.4 \times 10^{12}$ neutrons/sec-g

$$D_T \text{ (mrem/hr)} = (D_{n_T} + D_{\gamma_T}) \times \frac{1}{4\pi r^2} \times 2.4 \times 10^{12} \times \text{grams}$$

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* Registered trademark of Masonite Corp.

Dose Rates at the Surface of a Cylindrical Shield

Total dose rate at the surface of a cylindrical shield (Figure 2) will always be less than at the surface of a sphere that can be inscribed within that cylinder. Direct use of the spherical shield dose rate curves presented for cylindrical shields may be expected to yield slightly conservative results (or a higher dose rate).

Dose Rates for a Slab Shield

The dose attenuation curves used for a slab shield (Figure 3) will yield a conservative dose rate ($\sim 2\times$ depending on the shielding material) at short ranges. The data become less conservative, and will approach the spherical dose rate as the distance between the shield and receptor increases.

Dose Rates for Laminated Shields

The curves presented may be used to approximate⁶ the effects of using different materials to make a composite shield. The limitation on such use includes the assumption that the neutron and gamma energy spectrum does not vary significantly where the materials join.

A typical cask might be a 40-in.-OD (20-in.-radius) paraffin sphere with 10 mg/cm³ natural lithium mixed uniformly in the paraffin. A 3-in.-OD inner lead shield surrounds a 1-in.-OD source cavity. Assuming a 50- μ g ²⁵²Cf source, the dose rate at the surface of the cask is estimated by the following:

<u>Dose Constituents</u>	<u>$4\pi r^2 D(r)^a$</u>	<u>Correction</u>	<u>Total Dose Rate, $\frac{\text{mrem-hr}^{-1}}{\text{neutron-sec}^{-1}} \text{ cm}^2$</u>
Neutron			
Fast	9×10^{-5}	-	9×10^{-5}
Thermal	1×10^{-5}	-	1×10^{-5}
Gamma			
Primary	1.6×10^{-3}	0.17 ^b	2.7×10^{-4}
Secondary	1.2×10^{-3}	0.24 ^c	2.9×10^{-4}
			6.6×10^{-4}

^a. From paraffin curve Figure 7.

^b. From iron-lead curve Figure 5.

^c. From lithium curve Figure 35.

Therefore, the total dose rate D_T is:

$$D_T = 6.6 \times 10^{-4} \frac{\frac{\text{mrem-hr}^{-1}}{\text{neutron-sec}^{-1}} \text{ cm}^2 \quad 50 \times 10^{-6} \text{ g} \quad 2.4 \times 10^{12} \frac{\text{neutron-sec}^{-1}}{\text{g}}}{4\pi(20 \times 2.5)^2 \text{ cm}^2}$$

$$= 2.4 \text{ mrem/hr}$$

Other Applications

Data plotted on curves for different materials may be used to estimate unperturbed flux levels for activation and to estimate background radiation levels in detector applications. For example, the thermal neutron flux may be calculated from the thermal neutron dose rate by dividing the thermal neutron dose rate by the thermal dose rate conversion factor 0.00373 from Table VII.

Calculations

Dose rate attenuation curves shown in this report have been calculated using the ANISN⁷ code.

ANISN is an acronym for the discrete ordinates or S_n method of approximating the solution of the energy dependent linear Boltzmann transport equation with anisotropic scattering. The one-dimensional equation may be applied to slab, spherical, or cylindrical geometry.

In every case, the source was assumed to be in an aluminum matrix, spherical in shape, having a volume of 1 cm^3 , and a radius of 0.62 cm.

The angular flux was approximated by a sixteenth order quadrature (S_{16}) and the anisotropy of elastically scattered neutrons and of Compton-scattered photons was approximated by a third order Legendre expansion (P_3).

Energy Group Structure

The energy group structure has been derived using the multi-group theory of neutron and gamma-ray transmission. The groups have been divided into 13 neutron energy groups and 9 gamma energy groups. The neutron groups have been coupled with the gamma groups to calculate the inelastic and capture gamma activity that arises from neutron scatter or absorption.

Source

The source used in the calculations includes the gamma rays shown in the previous section distributed within their energy intervals and the neutrons as distributed by the Watt formula. Low energy X-rays and beta radiation are not significant for personnel shielding because of their inability to penetrate materials in which sources are usually encapsulated.

Table V lists the distribution of neutrons and gamma rays as a function of energy for each neutron emitted from ^{252}Cf .

TABLE V

Twenty-Two Group Source Distribution for ^{252}Cf

Group	Energy Interval, Mev	Type Radiation	Source Distribution
1	10.0 - 14.92	Neutron ↓	0.0036
2	6.70 - 10.00		0.0279
3	5.49 - 6.70		0.0356
4	4.49 - 5.49		0.0547
5	3.68 - 4.49		0.0730
6	3.01 - 3.68		0.0869
7	2.02 - 3.01		0.1894
8	0.91 - 2.02		0.3033
9	0.41 - 0.91		0.1462
10	0.11 - 0.41		0.0676
11	0.015 - 0.11		0.0119
12	0.0000041 - 0.015		0.0
13*	0.0 - 0.0000041	Gamma ↓	0.0
14	6.0 - 10.0		0.0012
15	5.0 - 6.0		0.0056
16	4.0 - 5.0		0.020
17	3.0 - 4.0		0.071
18	2.0 - 3.0		0.094
19	0.9 - 2.0		1.06
20	0.4 - 0.9		2.4
21	0.12 - 0.4		2.0
22	0.01 - 0.12		0.0

* Called thermal neutrons.

Cross Sections

Cross sections for neutron-induced reactions and transitions between groups were taken from a 100-group ENDF/B library. The ANISN code was used to collapse the cross sections from 100 to 13 neutron energy groups. Gamma-ray group-to-group transfer cross-sections were generated by the MUG⁸ code which includes photo-electric and pair production absorption as well as Compton effects. Nine gamma-ray energy groups were used. Cross sections for neutron-induced secondary and inelastic scattering gamma rays were coupled in the manner of POPOP4⁹ between the neutron and photon downscatter matrices.

Flux-to-Dose Conversion

Dose rates are given in terms of the biological dose rate (multiplied by $4\pi r^2$) computed at the surface of tissue at some point. This is accomplished by immediately surrounding this point with a material of composition approximating that of man. The dose rate then at that point contains the contributions from tissue backscatter. The composition of tissue used for the standard man is shown in Table VI. Conversion factors used to convert to biological dose rate are given in Table VII.

TABLE VI

Atomic Composition of the "Standard Man"

Element	wt %	Atoms/g
H	10	5.98×10^{22}
C	18	9.03×10^{21}
N	3	1.29×10^{21}
O	65	2.45×10^{22}
P	1.0	1.94×10^{20}
Ca	1.5	2.25×10^{20}

TABLE VII

Dose Rate Conversion Factors

Group ^a	Factor, $\left[\frac{\text{mrem-hr}^{-1}}{\text{particle-sec}^{-1}} \right] \text{ cm}^2$
1	0.147
2	0.147
3	0.142
4	0.139
5	0.134
6	0.130
7	0.125
8	0.130
9	0.105
10	0.056
11	0.020
12	0.0056
13	0.00373
14	0.0081
15	0.0061
16	0.0054
17	0.0045
18	0.0036
19	0.0031
20	0.0014
21	0.00065
22	0.00015

a. Dose rate conversion factors for
Groups 1-13 were determined from data
in Reference 10; those for Groups 14-22
were determined from data in Reference 11.

Shield Optimization

Many applications for ^{252}Cf use will require shields which are highly mobile; therefore, weight is likely to be the most sensitive design parameter for optimization. In many of the applications of ^{252}Cf , such as geological well-logging, the source is transported over rough terrain to remote locations for use. In medical or educational uses the source is stored near its place of use. Obviously, the weight and size of shields for these different applications have different priority. Generally, the more dense gamma shielding materials should be kept close to the source where they occupy less volume for a given thickness of absorber. Secondary gamma radiations originating in the shield complicate the design and usually require lamination of the dense and lighter shield materials. Additives such as boron and lithium can be used to suppress the secondary gamma dose from hydrogenous shields. (A more mathematical approach to spherical shield weight optimization may be found in Reference 12).

Shield Heating

^{252}Cf generates about 38 watts of energy per gram of pure isotope: half of which comes from an alpha decay reaction; the remainder, from spontaneous fission. Of the fission energy released, 86% is deposited in the source capsule; the remainder, in the form of neutrons, gamma rays, and neutrinos, is dissipated in the shield and beyond. Thus, 93% of the 38 watts/g, or 35 watts/g, appears as sensible heat at the source.

An approximation can be made by assuming that all the heat is generated at the source, then one can use a simple heat conduction equation to estimate maximum shield internal temperatures. For example, assume a 1-mg ^{252}Cf source in a paraffin spherical cask of 40-in. OD and 4-in. ID. The temperature difference through the paraffin shield would be 0.4°F from

$$\Delta T = \frac{3.41 Q}{K} \frac{r_o - r_i}{4\pi r_o r_i}$$

where

K = thermal conductivity (0.14 BTU-ft/hr-ft²-°F)

Q = thermal power, ^{252}Cf source heat = 0.038 watts

r_o = outer radius, 20 inches = 1.67 ft

r_i = inner radius, 2 inches = 0.167 ft

This result is conservative because it ignores the effect of heat leakage along the source rod and sheath.

The above example illustrates that heating is not a major design concern for moderately sized ^{252}Cf sources. A more detailed analysis of the thermal considerations in both finned and unfinned casks is given in Reference 5A.

Shield Damage from Radiation

Extended exposure to neutron and gamma radiation will possibly alter the structure and physical properties of shield materials. This is caused in part by collision of fast neutrons with shield nuclei, displacing them from metallic lattices or from molecules. Ion formation or excited atoms also contribute to shield damage. Fragments of disrupted molecules in some materials may form new chemical compounds, some of which may be gaseous (i.e., CH_4 , H_2).

Radiation damage from ^{252}Cf sources in casks and shields would be anticipated to be much lower than reported effects in nuclear reactors because of the lower doses and less critical operating conditions. Nevertheless, for large ^{252}Cf source containment, provision for gas venting and cask component replacement may be deemed necessary. There has been a large amount of work done on radiation damage effects.^{13,14}

Induced Activities

Neutron irradiation leads to induced activity which can be estimated using the equation:

$$A = N\sigma\phi(1 - e^{-\lambda T})$$

where N = number of atoms

σ = cross section, cm^2/atom

ϕ = flux, neutron/ $\text{cm}^2\text{-sec}$

T = time, sec

λ = decay constant of activated material

Size and Weight

For many applications, both the physical size and the weight of the source shield may be important considerations.

Since very few shields are constructed of only one material, and because their shape may vary as well, the shield designer must compute separately the various volumes and weights of the shields. Figures 49 and 50 showing spherical and cylindrical shield volumes and Figures 51 and 52 showing some spherical and cylindrical shield weights for some common material densities are included and may be used as a guide for first approximation data. Shield data for spherical geometries can usually be used for cylindrical shields as well.

SHIELD MATERIAL SELECTION

Several factors must be considered when selecting a shield material. They are cost, safety, reliability, weight, and maintenance requirements. Properties to be considered are density, fabricability, durability, and heat transfer properties. The following sections of this report are designed to provide only neutron and gamma attenuation properties calculated for a ^{252}Cf source spectrum. No attempt has been made to catalogue material costs, merits, properties, or construction methods.

Single Component Shields

"Benelex"

"Benelex" (Figure 4) is made from pulverized wood chips. Unwanted elements are driven off, leaving only cellulose fibers and lignin, wood's natural bonding agent. Fibers and lignin are rewelded by pressing to form panels of controlled densities, thicknesses, and sizes.

This material is a good neutron shielding material because of its relatively high hydrogen atom concentration. The average density of "Benelex" is 1.4 g/cm^3 .

"Benelex" has good structural strength and rigidity, is easily machined.

Iron and Lead

Iron and/or lead by itself provide almost no fast neutron attenuation ($\rho_{\text{Pb}} = 11.3 \text{ g/cm}^3$, $\rho_{\text{Fe}} = 7.9 \text{ g/cm}^3$). Both of these materials do provide gamma attenuation, however, and when used with hydrogenous materials become an important part of californium shields. Figure 5 shows the ^{252}Cf primary gamma attenuation for iron, lead, and normal concrete. Iron and lead will attenuate most capture gammas in a similar manner. For comparison of the gamma attenuation properties, normal concrete ($\rho = 2.35 \text{ g/cm}^3$) is also shown.

*"Lucite"**

"Lucite" ($\text{C}_5\text{H}_8\text{O}_2$)_n with a density of ~ 1.18 is a good neutron shield like almost all organic materials (Figure 6). Its low density indicates it must (in most cases) be combined with a more dense material to shield the gamma rays. "Lucite" and iron or lead laminations will perform similarly in shielding effectiveness to iron or lead and water.

Paraffin

Because of its high hydrogen density, paraffin ($\text{C}_{30}\text{H}_{62}$) (density $\sim 0.952 \text{ g/cm}^3$) provides an excellent neutron shield (Figure 7). Paraffin, in combination with a thermal neutron absorber, such as boron or lithium, and with a more dense gamma attenuator, such as iron or lead, provides an excellent shield for californium. It is easily formed, but because of its low melting and flash points, it must be used with care.

Polyethylene

Polyethylene ($\rho = 0.961 \text{ g/cm}^3$) has a hydrogen density greater than that of water; therefore, its neutron attenuation properties are better than those of water (Figure 8). It is easily formed to shape, and can be purchased in many preformed shapes. Its low density, however, requires that it be used in conjunction with some gamma attenuation materials. Because it softens at about 220°F and burns readily, it requires careful use. It can be borated by adding boron carbide (B_4C).

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* Registered trademark of Du Pont.

Water

Because of its high density of hydrogen atoms, cost, and availability, water is often used as a shielding material (Figure 9). Because the mass density is low, however, it is a poor gamma shield. Capture gamma radiation produced in thick shields also may be the most dominant type of radiation contributing to the dose rate. The addition of a thermal neutron absorber such as boron will help reduce the capture gammas. Boron-containing compounds, however, are known to corrode carbon steel, primarily at the air surface interface. For this reason, stainless steel should be considered for at least the air-liquid interface.

The primary cost of water shields is containment. Plastic water-containing tanks appear feasible in conjunction with some gamma shielding materials.

Water also has the disadvantages of evaporating or leaking which could render the shield useless.

Water-Extended Polyester

A water-extended polyester resin used as a shield by G. D. Oliver, Jr. and E. Bailey Moore¹⁵ of the M. D. Anderson Hospital and Tumor Institute has the appealing qualities of easy preparation, high water content, solid form, and resistance to fire (Figure 10).

The water-extended polyester with 65% water (called WEP-65) has a density of 1.1 g/cm³.

*Lithium-6 Stearate*¹⁶

Lithium-6 Stearate, with a melting point around 220°C, attenuates neutrons similarly to water (Figure 11). It can be cast and recast into shape, carved, machined, or drilled. When used with a more dense material, such as iron or lead to attenuate the primary gamma rays, it appears to be a good shielding material for ²⁵²Cf.

Glycerin

Glycerin (C₃H₈O₃) with a density of ~1.26 g/cm³ is a good neutron shield (Figure 12). Its low density indicates it may require the addition of a more dense material to shield the gamma rays. Glycerin will mix easily with boric acid solution to reduce the capture gamma dose rate.

Multicomponent Shields

Concrete

Concrete is a commonly used shielding material that is inexpensive, structurally useful, and easily formed. Concrete can vary widely in its elemental composition, and because of this will vary considerably in shielding effectiveness for californium. Figures 13 through 16 show the effect of varying the elemental composition of concretes and the density. The curing time and temperature, and the aggregate used¹⁷ most severely affect the water content of the concrete. Aggregate size also may be important to ensure that the mixture is free from large voids and is of sufficiently high density.

^{252}Cf shielding is similar to nuclear reactor shielding in that both neutron and gamma rays must be attenuated. Concrete has been widely used as an inexpensive reactor shield because it is a mixture of light nuclides, which slow down neutrons, and heavy nuclides, which absorb gamma rays. Although not as efficient a neutron shield as paraffin, polyethylene, or water, concrete casks provide a safety factor because they will not melt, burn, or leak.

The concretes chosen represent both the best and worst attenuation to be expected from any concrete shield. The concrete types 01, 03, and 02A are classified as ordinary concretes consisting of a water-cured cement and an aggregate yielding a density of 2.3 g/cm^3 . Heavy concretes [e.g., LS-a (Limonite and Steel Punchings, type a)] are made by adding iron punchings to the aggregate to obtain a density of 4.6 g/cm^3 . A density increase improves gamma-ray attenuation (Figure 16).

Water content plays a significant role in the effectiveness of concrete as a neutron shield because hydrogen is the most effective light element for slowing down neutrons. A factor of 8 in the difference in hydrogen content between concrete types 01 and LS-a is reflected in their relative neutron shielding abilities.

Soil

The major constituents of Nevada Test Site (N.T.S.) soil with varying degrees of water content are given in Reference 18. Soil may vary considerably in its elemental composition and its water content for different locations as well as climates around the world. This data may be useful, however, for those considering underground storage of californium sources, or for those users making tests such as activation analyses or well logging in the soil (Figures 17-19.)

Iron-Water and Lead-Water Combinations

Combinations of water and dense materials such as iron or lead are good overall shield materials for ^{252}Cf . The iron or lead and water combinations can be made like laminations or like a homogeneous solution. Water shields containing 20 vol % lead and 60 vol % iron, uniformly distributed throughout the shield, will attenuate neutrons at about the same rate as the gamma rays. This nearly homogeneous shield can be obtained by using thin multiple slabs, packed lead or steel wool, or balls (shot). The addition of a highly absorbing thermal neutron absorber, such as boron or lithium, will reduce the capture gammas so that the primary dose rate will be from the fast neutrons even for shields 100 cm thick in a 40 vol % Fe - H_2O shield. Although laminations do cause increasing and decreasing dose rates from the various energy groups, an approximate final dose rate can be estimated with the assumed homogeneous mixtures shown in Figures 20-25.

If there is some question as to what arrangement may be considered homogeneous, then the shield should be evaluated on the basis of its actual materials arrangement. All the iron or lead cannot be placed close to the source for this interpretation, because the capture gamma radiation produced in the shield will easily penetrate the remaining water. Placement of all the lead or iron on the outside of the shield may not provide the neutron attenuation necessary, and will increase the overall weight of the shield significantly.

Boron Addition to Hydrogenous Shields

Because most hydrogenous shields have a density $\sim 1 \text{ g/cm}^3$, they are poor gamma radiation shields. However, because the hydrogen density is high they are excellent shields for fast neutrons. A substantial reduction in capture gamma dose rates can be obtained by adding boron to the hydrogenous ^{252}Cf shield. This addition of boron can either reduce the shield weight and size, or increase the ^{252}Cf capacity of the shield.

Boron has a high cross section (3840 barns) for the neutron absorption reaction $^{10}\text{B}(n,\alpha)^7\text{Li}$, which is accompanied by a 0.48-Mev gamma ray from the excited ^7Li nuclide. Neutron absorption in hydrogen (0.33 barn) produces a 2.23-Mev capture gamma ray that is more difficult to attenuate than the 0.48-Mev boron reaction gamma ray. The addition of boron reduces neutron captures in hydrogen in proportion to the reduced macroscopic cross section for hydrogen in the mixture and produces a change in the spectral distribution of the neutrons and capture gamma rays.

Thermal neutron dose rates are reduced by ~90% with the addition of boron; however, the total neutron dose rates in the shield remain relatively unaffected because thermal neutrons normally constitute less than 10% of the total neutron dose rate.

Figure 26 shows the effect of ^{10}B on the capture gamma dose rate for a spherical water shield. The capture gamma dose rate decreases rapidly and then levels off at low ^{10}B concentrations.

Figure 27 shows the effect of natural boron addition to spherical paraffin shields.

Figures 28 through 34 show the effect of the addition of boron carbide (B_4C) to paraffin. At thicknesses up to about 30 cm, paraffin by itself rapidly reduces the total dose rate because of its attenuation of the dominant fast neutron radiation (Figure 7). At thicknesses greater than 30 cm, the gamma radiation is the predominant contributor to the dose rate. This gamma radiation is from two sources.

- 1) Primary radiation which can be shielded by surrounding the source with a dense material.
- 2) Secondary or capture gamma radiation resulting in the shield material.

Two methods are used to reduce the capture gamma dose rate: one is to provide sufficient dense material in the outermost part of the shield, and the other is to provide a thermal neutron absorber such as ^{10}B . It is sometimes necessary to use both methods. Care should be taken, however, to assure that boron additives do not contain large quantities of elements that result in significantly more energetic gamma rays than those produced by hydrogen.

Figure 34 shows the total neutron and gamma dose rate (less the primary gamma radiation) in paraffin with varying quantities of B_4C added. For shields greater than 30 cm in thickness, the addition of boron carbide significantly reduces the total dose rate in the shield. Reference 19 lists many potential shield additives containing boron. The effect of boron addition to water for windows is discussed on page 25.

Lithium Addition to Hydrogenous Shields

Substantial reduction in capture gamma dose rates can be obtained by adding natural lithium to hydrogenous ^{252}Cf shields. This reduction is caused by the $^6\text{Li}(n,\alpha)\text{T}$ reaction, which has a 945-barn cross section for thermal neutrons and produces no capture gamma radiation. The lithium absorption overrides the $\text{H}(n,\gamma)\text{D}$ reaction in the hydrogenous material, which is the predominant source of the secondary gamma. Care should be taken to assure that lithium compounds used for this purpose do not result in significantly more energetic gamma rays than those produced by the hydrogen.

The results shown in Figure 35 were calculated with the ANISN code for different concentrations of natural (7.65% ^6Li) lithium and shield thicknesses from 14 to 50 in. The capture gamma dose decreases rapidly on addition of relatively small amounts of lithium.

Previous sections have discussed the similar effect of boron addition to reduce capture gamma activity. Cask designers may find lithium more attractive than boron as a moderator addition because it is easily distributed uniformly in the shield material.

To estimate the effect of lithium addition on a specific cask design, the capture gamma dose determined from curves is multiplied by the appropriate reduction factor for the desired lithium concentration. Neutron dose rate increased only 10% over the range of lithium concentrations shown.

Window Attenuation

Large sources that must be handled routinely at short distances often require the use of a shielding window. In addition to its shielding effectiveness, windows must have good transparency for viewing, allow sufficient viewing area, and have a reasonable lifetime.

Although in some instances hydrogenous materials such as water or oil, contained by glass or "Lucite," may be warranted, these low density materials offer such poor gamma attenuation by themselves, it is generally necessary to make the window much larger than the shielding wall. For this reason, these materials are often combined with reasonably transparent materials with higher density, such as lead-containing glass.

The attenuation properties for "Lucite" or water windows are shown in Figures 6 and 9.

The attenuation properties of laminations of various types of materials, however, become much more complicated. Figure 36 shows the attenuation properties for one specific design with three types of glass combined with oil. The side of the window nearest the source has a double thickness of nonbrowning glass separated by an air space, which acts as a protective barrier to the window from radiation damage, chemical damage, shock, and contamination*. The glass pane nearest the source is usually

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*Contamination is referred to here as the presence of an unwanted material, radioactive or otherwise.

designed so that it might be changed if damaged without affecting the rest of the window.

Because spectral changes occur as the thickness of the shield changes, each window design with these laminations should be separately calculated for these changes.

The compositions of the materials used in the calculations are listed in Table VIII.

Zinc bromide (ZnBr_2) solutions can be used as shielding windows for ^{252}Cf sources. Figures 37, 38, and 39 show the effect of thickness on dose rate for three different concentrations of ZnBr_2 . These solutions have good transparency and good resistance to the effects of radiation. Reported data²⁰ suggest the addition of hydroxylamine hydrochloride ($\text{H}_2\text{NOH}\cdot\text{HCl}$) to inhibit discoloration. Venting and careful selection of materials (because of corrosion) may be required.

Boric acid (H_3BO_4) solutions can be used in shielding windows. Figure 40 shows the effect of thickness on dose rate for saturated boric acid.

TABLE VIII
Composition of Window Materials

Material	Component	Elemental Density, g(component)/cm ³ (compound)
Oil	H	0.125
	C	0.751
Glass ($\rho=2.7 \text{ g/cm}^3$)	B_2O_3	0.081
	Na_2O	0.176
	SiO_2	1.750
	K_2O	0.243
Glass ($\rho=3.39 \text{ g/cm}^3$)	SiO_2	1.617
	PbO	1.148
	Na_2O	0.017
	K_2O	0.462
Glass ($\rho=6.29 \text{ g/cm}^3$)	SiO_2	0.341
	PbO	5.084
	B_2O_3	0.775
Concentrated zinc bromide solution	Zr	0.5588
	Br	1.3663
	H	0.0639
	O	0.5111
Saturated H_3BO_4	H	0.108
	O	0.884
	B	0.008

SPECIAL APPLICATIONS

MEDICAL APPLICATIONS

Medical applications are expected for californium-252 sources ranging from 0.5 μg to 50 μg . If a number of such sources are stored in one location, shielding requirements (for storage) may range from 0.5 μg to 200 μg . Shields, therefore, may range from a few centimeters to >100 cm depending on the allowable dose rate at the container surface.

Some examples which represent the dose rate in steel (Fe) followed by saturated boric acid solution or paraffin containing 1 wt % B_4C are given in Figures 41 through 44. Figures 45 and 46 show the effect of surrounding the hydrogenous materials with more steel. This added component of dense material adds considerably to the shield by further attenuating the primary gamma rays, and also reducing the secondary gamma contribution. In such shields, approximately 3 in. (7.62 cm) of steel on the outside of the shield will reduce both the primary gamma and the capture gamma dose rate by about a factor of 10. (See Figure 5 for an approximate attenuation factor.)

Portability may be desirable in many medical applications, necessitating size and weight considerations.

ACTIVATION ANALYSIS

Californium-252 sources used for activation analysis are expected to range from 0.1 to 100 mg. Figures 47 and 48 indicate the thermal flux predicted at various thicknesses of some commonly used moderating materials. Fast fluxes are also of interest because of the excitation produced in some elements in the inelastic scattering process.

SHIELDING EFFECTIVENESS OF COMMONLY USED CASKS

Most casks built for gamma shielding are inadequate for ^{252}Cf because they lack hydrogenous materials. Some casks, using water as a heat transfer mechanism, may be used as ^{252}Cf shipping casks. Concrete casks may also be used; however, because the emphasis has in the past primarily been placed on the density, the hydrogen content of the concrete is usually unknown. If this is the case, the best way to determine its shielding capacity is by measurement. A detailed chemical analysis of the concrete will allow it to be calculated with the ANISN code.

Paraffin-filled drums used to ship and store Po-Be and Pu-Be sources are usually available, or may be fabricated at low cost. As shown in Table IX, these drums are limited to less than 1 mg of ^{252}Cf .

TABLE IX

Calculated Effectiveness of Paraffin-Filled Drums²¹

Size, gal	Drum Dia, in.	Common Carrier Shipping Limit, $\mu\text{g } ^{252}\text{Cf}$	Wt, lb
15	15½	17	150
30	18	62	250
55	22½	93	450
70	27	180	600
100	30	240	900
130	33	313	1,200
170	36	400	1,500
230	40	525	2,000
300	44	700	2,600

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Fast Neutron, Thermal Neutron, Primary Gamma, and
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FIG. 1 SPHERICAL SHIELD GEOMETRY

FIG. 2 CYLINDRICAL SHIELD GEOMETRY

FIG. 3 SLAB SHIELD GEOMETRY

FIG. 4 BENELEX - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE TO TISSUE IN "BENELEX" FROM A POINT ISOTROPIC FISSION SOURCE OF ^{252}Cf

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FIG. 6 LUCITE - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE TO TISSUE IN "LUCITE" FROM A POINT ISOTROPIC FISSION SOURCE OF ^{252}Cf

FIG. 7 PARAFFIN - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE TO TISSUE IN PARAFFIN FROM A POINT ISOTROPIC FISSION SOURCE OF ^{252}Cf

FIG. 8 POLYETHYLENE - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE TO TISSUE IN POLYETHYLENE FROM A POINT ISOTROPIC FISSION SOURCE OF ^{252}Cf

FIG. 9 WATER - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN AN INFINITE MEDIUM OF WATER FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 10 WATER-EXTENDED POLYESTER - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN AN INFINITE MEDIUM OF WATER-EXTENDED POLYESTER (65%) FROM A POINT ISOTROPIC FISSION SOURCE OF ^{252}Cf

FIG. 11 LITHIUM-6 STEARATE - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CATTURE GAMMA DOSE RATE IN AN INFINITE MEDIUM OF LITHIUM-6 STEARATE FROM A POINT ISOTROPIC FISSION SOURCE OF ^{252}Cf

FIG. 12 GLYCERIN - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN GLYCERIN FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 13 TYPE 01 CONCRETE - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN TYPE 01 CONCRETE FROM A POINT ISOTROPIC FISSION SOURCE OF ^{252}Cf

FIG. 14 TYPE 02A CONCRETE - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN TYPE 02A CONCRETE FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 15 TYPE 03 CONCRETE - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN TYPE 03 CONCRETE FROM A POINT ISOTROPIC FISSION SOURCE OF ^{252}Cf

FIG. 16 TYPE LS-a CONCRETE - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN TYPE LS-a CONCRETE FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 17 N.T.S. SOIL (DRY) - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN AN INFINITE MEDIUM OF N.T.S. DRY SOIL FROM A POINT ISOTROPIC FISSION SOURCE OF ^{252}Cf

FIG. 18 N.T.S. SOIL (50% SATURATED) - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN AN INFINITE MEDIUM OF N.T.S. SOIL (50% SATURATED) FROM A POINT ISOTROPIC FISSION SOURCE OF ^{252}Cf

FIG. 19 N.T.S. SOIL (100% SATURATED) - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN AN INFINITE MEDIUM OF N.T.S. SOIL (100% SATURATED) FROM A POINT ISOTROPIC SOURCE OF ^{252}Cf

FIG. 20 10% LEAD IN WATER - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN AN INFINITE MEDIUM OF 10 VOL % Pb in H₂O FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 21 20% LEAD IN WATER - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN AN INFINITE MEDIUM OF 20 VOL % Pb IN H₂O FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 22 10% IRON IN WATER - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN AN INFINITE MEDIUM OF 10 VOL % Fe IN H₂O FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 23 20% IRON IN WATER - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN AN INFINITE MEDIUM OF 20 VOL % Fe IN H₂O FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 24 40% IRON IN WATER - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN AN INFINITE MEDIUM OF 40 VOL % Fe in H_2O FROM A POINT ISOTROPIC FISSION SOURCE OF ^{252}Cf

FIG. 25 60% IRON IN WATER - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN AN INFINITE MEDIUM OF 60 VOL % Fe IN H₂O FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 26 BORON IN WATER - EFFECT OF ADDING BORON TO A SPHERICAL WATER SHIELD ON THE CAPTURE GAMMA DOSE RATE

FIG. 27 BORON IN PARAFFIN - EFFECT OF ADDING BORON TO A SPHERICAL PARAFFIN SHIELD ON THE CAPTURE GAMMA DOSE RATE

FIG. 28 0.5 WT % B₄C IN PARAFFIN - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN AN INFINITE MEDIUM OF 0.05 WT % B₄C IN PARAFFIN FROM AN POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 29 0.1 WT % B₄C IN PARAFFIN - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN AN INFINITE MEDIUM OF 0.1 WT % B₄C IN PARAFFIN FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 30 0.5 WT % B₄C IN PARAFFIN - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN AN INFINITE MEDIUM OF 0.5 WT % B₄C IN PARAFFIN FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 31 1.0 WT % B₄C IN PARAFFIN - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN AN INFINITE MEDIUM OF 1.0 WT % B₄C IN PARAFFIN FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 32 2.0 WT % B₄C IN PARAFFIN - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN AN INFINITE MEDIUM OF 2.0 WT % B₄C IN PARAFFIN FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 33 5.0 WT % B₄C IN PARAFFIN - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN AN INFINITE MEDIUM OF 5.0 WT % B₄C IN PARAFFIN FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 34 VARYING AMOUNTS OF B₄C IN PARAFFIN - TOTAL DOSE RATE (WITHOUT PRIMARY GAMMA DOSE RATE) IN PARAFFIN WITH VARYING AMOUNTS OF B₄C FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 35 LITHIUM IN PARAFFIN - EFFECT OF ADDING LITHIUM TO HYDROGENOUS SHIELDS ON CAPTURE GAMMA DOSE RATE

FIG. 36 GLASS AND OIL WINDOW SHIELD - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN A LAMINATED GLASS AND OIL WINDOW SHIELD FROM A POINT ISOTROPIC FISSION OF ^{252}Cf

FIG. 37 50% ZnBr₂ WINDOW SHIELD - FAST NEUTRON, THERMAL NEUTRON
PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN A 50%
ZnBr₂ WINDOW FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 38 65% ZnBr₂ WINDOW SHIELD - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN A 65% ZnBr₂ WINDOW FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 39 SATURATED ZnBr_2 WINDOW SHIELD - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN A SATURATED ZnBr_2 WINDOW FROM A POINT ISOTROPIC FISSION SOURCE OF ^{252}Cf

FIG. 40 BORIC ACID WINDOW SHIELDS - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, CAPTURE GAMMA, AND TOTAL DOSE RATE IN A SATURATED BORIC ACID WINDOW FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 41 SATURATED BORIC ACID AND 1-IN. STEEL - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN A SHIELD CONTAINING 1 IN. OF STEEL AND A SATURATED BORIC ACID SOLUTION FROM A POINT ISOTROPIC FISSION SOURCE OF ^{252}Cf

FIG. 42 SATURATED BORIC ACID AND 2-IN. STEEL - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN A SHIELD CONTAINING 2 IN. OF STEEL AND A SATURATED BORIC ACID SOLUTION FROM A POINT ISOTROPIC FISSION SOURCE OF ^{252}Cf

FIG. 43 1.0 WT % B₄C IN PARAFFIN AND 2-IN. STEEL - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN A SHIELD CONTAINING 2 IN. OF STEEL AND 1.0 WT % B₄C IN PARAFFIN FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 44 1.0 WT % B₄C IN PARAFFIN AND 3-IN. STEEL - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN A SHIELD CONTAINING 3 IN. OF STEEL AND 1.0 WT % B₄C IN PARAFFIN FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 45 2-IN. STEEL, SATURATED BORIC ACID, AND 2-IN. STEEL - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN A SHIELD CONTAINING 2 IN. OF STEEL, SATURATED BORIC ACID SOLUTION, AND 2 IN. OF STEEL FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 46 3-IN. STEEL, 1 WT % B₄C IN PARAFFIN, AND 3-IN. STEEL - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN A SHIELD CONTAINING 3 IN. OF STEEL, 1 WT % B₄C IN PARAFFIN, AND 3 IN. OF STEEL FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 47 THERMAL NEUTRON FLUX IN Be, BeO, D₂O, H₂O, AND GRAPHITE AS A FUNCTION OF DISTANCE FROM A POINT ISOTROPIC FISSION SOURCE OF ²⁵²Cf

FIG. 48 THERMAL NEUTRON FLUX IN WATER, POLYETHYLENE, AND PARAFFIN AS A FUNCTION OF DISTANCE FROM A POINT ISOTROPIC FISSION SOURCE OF ^{252}Cf

FIG. 49 VOLUME OF A SPHERE AS A FUNCTION OF ITS RADIUS

FIG. 50 VOLUME OF A CYLINDER AS A FUNCTION OF ITS RADIUS

FIG. 51 WEIGHT OF A SPHERICAL SHIELD AS A FUNCTION OF ITS RADIUS FOR MATERIALS OF DIFFERENT DENSITY

FIG. 52 WEIGHT OF A CYLINDRICAL SHIELD AS A FUNCTION OF ITS RADIUS FOR MATERIALS OF DIFFERENT DENSITY WHEN $h=2r$

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