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PERFORMANCE OF PORTABLE STORAGE CONTAINERS IN REDUCING NEUTRON RADIATION

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Abstract—*The principal investigators designed portable shielded containers that reduce neutron doses to workers from fissile materials contained in storage canisters. We studied the shielding characteristics of several common shielding materials, such as polyethylene, borated polyethylene, B_4C, and cadmium. From these studies, we found and successfully demonstrated that by using a combination of CH_2 and B_4C, we could reduce neutron dose by a factor of four or better. In addition, the containers we designed with the new materials are of reasonable size and weight.*

I. INTRODUCTION

For many years, Los Alamos National Laboratory (LANL, the Laboratory) management has been concerned about neutron doses received by workers at its plutonium facility. In part, workers receive neutron doses as a consequence of fissionable material stored in rooms located at their worksites. They also receive doses when stored fissionable materials are transported from one worksite to another. One of the reasons for exposure is that transported materials are typically stored in stainless steel containers that lack proper radiation shielding. A portable shielded storage container designed for fissionable materials is a high priority for a facility that has the goal of reducing worker neutron exposure .

During this study, the principal investigators established the following design criteria for an effective portable neutron-shielded storage container:

- reasonable size and weight,
- reduced neutron dose by a factor of three or four, and
- limited effect on the criticality safety margin of the contents.

II. MONTE CARLO CALCULATIONS OF COMMON SHIELDING MATERIALS

The most commonly used neutron-shielding materials that reduce neutron dose to workers are polyethylene, borated polyethylene, B_4C, and cadmium. Each of these materials has a different neutron reaction cross section. The neutron fluence-to-dose conversion function (Fig.1) shows that higher energy neutrons carry higher dose factors; therefore, reducing the number of higher energy neutrons is a first and important objective toward achieving a goal of lower worker neutron doses.

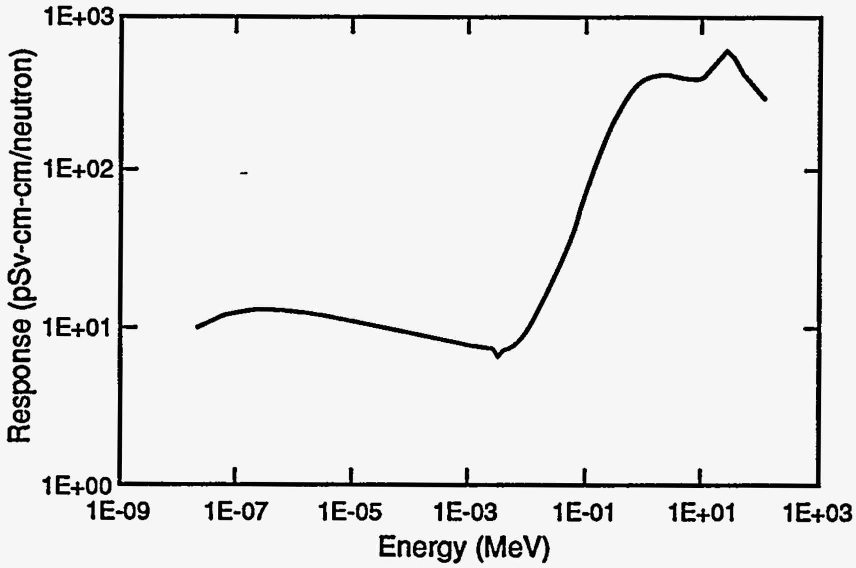

Figure 1. Fluence-to-dose function, Siebert and Schuhmacher (RPD v58)

The principal investigators used Monte Carlo code MCNP[1] to calculate neutron dose under the following conditions:

Geometry. A 1-cm spherical cavity with ten 2.54-cm incremental spherical shells filled with different shielding materials.

Shielding material. Polyethylene, borated (30% by weight) polyethylene, B_4C.

Neutron source. Isotropic ^{252}Cf fission source at the center.

We then tallied results for the following:

- transmitted neutron spectra with different kinds of shielding materials and with various thicknesses to show how the shielding material changes neutron spectra;
- neutron doses at the outer surface and at 1 m from the source by folding the neutron spectra to neutron fluence-to-dose function;
- gamma dose, due to gamma rays produced by (n,gamma) reaction, at the outer surface by folding gamma spectra to gamma-fluence-to-dose function; and
- neutron spectra reflected from shielding back into the cavity and then folded to the Pu fission cross section to show the criticality effect.

We obtained the following results:

Neutron fluence (neutron/cm^2/source neutron)

Table 1.

Sphere Radius (cm)	Material			
	None	CH_2	$CH_2 + B$†	B_4C
1.0	7.96e–2	7.96e–2	7.96e–2	7.96e–2
3.54	6.35e–3	6.34e–3	6.04e–3	6.08e–3
6.08	2.15e–3	1.98e–3	1.67e–3	1.88e–3
8.62	1.07e–3	7.85e–4	6.11e–4	8.08e–4
11.16	6.39e–4	3.35e–4	2.55e–4	3.95e–4
13.70	4.24e–4	1.50e–4	1.16e–4	2.06e–4
16.24	3.02e–4	6.93e–5	5.65e–5	1.11e–4
18.78	2.26e–4	3.65e–5	2.88e–5	6.14e–5
21.32	1.75e–4	1.81e–5	1.53e–5	3.45e–5
23.86	1.40e–4	9.27e–6	8.46e–6	1.96e–5
26.40	1.14e–4	4.89e–6	4.78e–6	1.13e–5

† $CH_2 + B$ is borated (30% by weight) polyethylene.

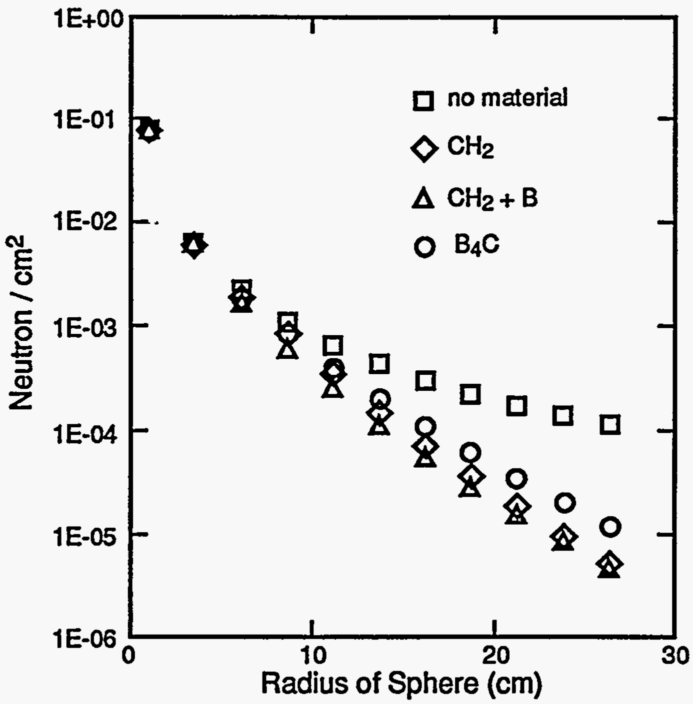

Figure 2. Neutron leakage

When there is no material, the fluence change is $1/R^2$. Boron has very high-reaction cross section for low- energy neutron; however, it is not as effective as hydrogen in slowing down neutrons. Changing the neutron spectra (see also Figure 4) also plays an important role in decreasing neutron fluence.

Neutron dose at surface: (pSv–cm^2 / neutron)

Table 2.

Radius of sphere (cm)	Material: None	CH_2	$CH_2 + B$	B_4C
1.00	3.05e+1	3.05e+1	3.05e+1	3.05e+1
3.54	2.43e+0	1.86e+0	1.98e+0	2.26e+0
6.08	8.25e–1	4.46e–1	5.02e–1	6.70e–1
8.62	4.11e–1	1.49e–1	1.78e–1	2.77e–1
11.16	2.45e–1	5.90e–2	7.46e–2	1.31e–1
13.70	1.63e–1	2.56e–2	3.44e–2	6.62e–2
16.24	1.16e–1	1.20e–2	1.70e–2	3.50e–2
18.78	8.65e–2	5.95e–3	8.81e–3	1.91e–2
21.32	6.71e–2	3.06e–3	4.76e–3	1.07e–2
23.86	5.36e–2	1.64e–3	2.66e–3	6.00e–3
26.40	4.38e–2	9.04e–4	1.51e–3	3.45e–3

We obtained neutron doses by folding neutron fluences into the fluence–to–dose function calculated by Siebert and Schuhmacher[2]. Notice that in this calculation that neutron fluences with CH_2 as shielding are higher than those with borated CH_2, but the neutron doses are lower. This is due to neutron spectral differences. Figure 4 shows three transmitted neutron spectra of 4-in- thick shielding materials. There are FEWER high-energy neutrons when CH_2 is used as shielding.

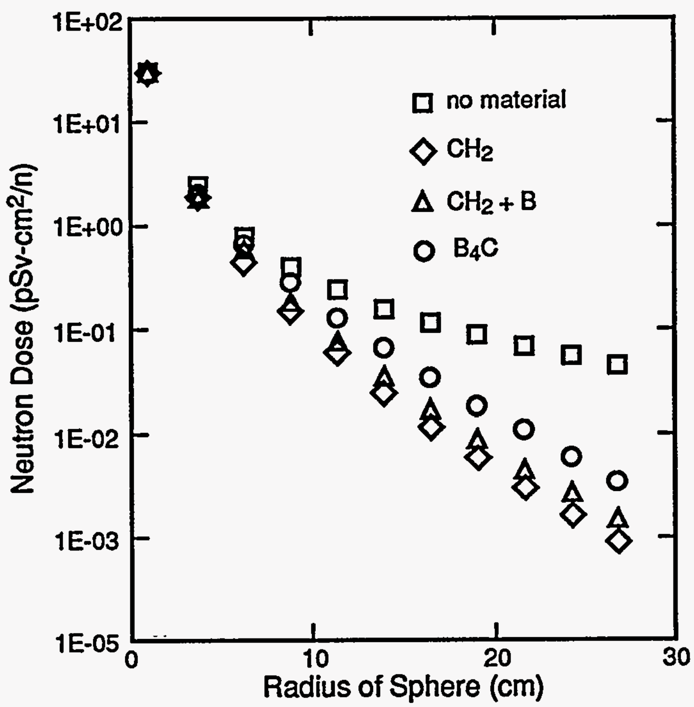

Figure 3. Neutron dose at surface

Neutron dose at 1 m from the source: ($pSv-cm^2$ / neutron)

Table 3.

Radius of sphere (cm)	Material			
	None	CH_2	$CH_2 + B$	B_4C
3.54	3.05e-3	2.34e-3	2.48e-3	2.83e-3
6.08	3.05e-3	1.65e-3	1.86e-3	2.48e-3
8.62	3.05e-3	1.11e-3	1.33e-3	2.06e-3
11.16	3.05e-3	7.34e-4	9.30e-4	1.63e-3
13.70	3.05e-3	4.81e-4	6.45e-4	1.24e-3
16.24	3.05e-3	3.17e-4	4.48e-4	9.25e-4
18.78	3.05e-3	2.10e-4	3.11e-4	6.75e-4
21.32	3.05e-3	1.39e-4	2.17e-4	4.84e-4
23.86	3.05e-3	9.31e-5	1.51e-4	3.42e-4
26.40	3.05e-3	6.30e-5	1.06e-4	2.40e-4

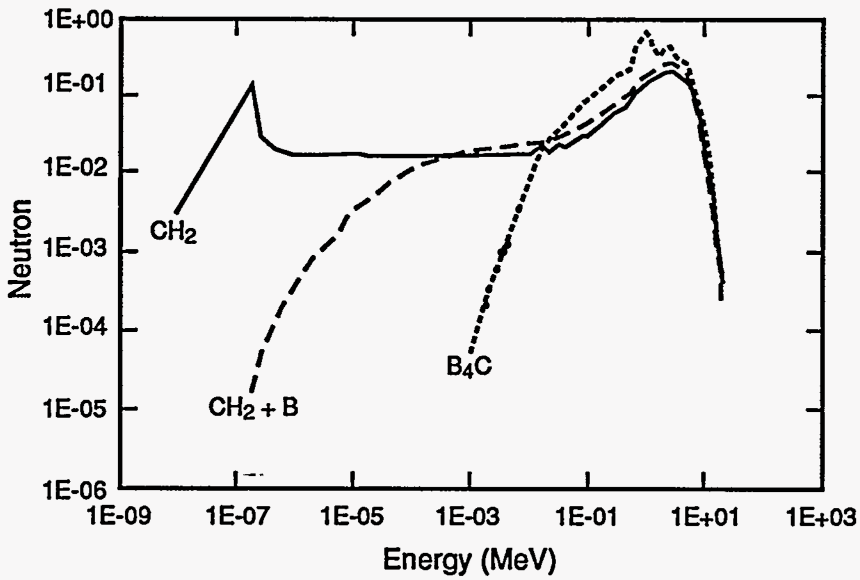

Figure 4. Neutron spectra with 4-inch shielding

Gamma dose at surface: (pSv–cm^2 / neutron)

Table 4.

Radius of sphere (cm)	Material CH_2	$CH_2 + B$	B_4C
3.54	1.10e–3	1.06e–3	1.15e–3
6.08	1.62e–3	1.10e–3	7.00e–4
8.62	1.75e–3	9.14e–4	4.96e–4
11.16	1.58e–3	6.82e–4	3.61e–4
13.70	1.29e–3	4.93e–4	2.62e–4
16.24	1.01e–3	3.54e–4	1.85e–4
18.78	8.58e–4	2.54e–4	1.28e–4
21.32	6.65e–4	1.83e–4	8.83e–5
23.86	5.17e–4	1.34e–4	6.10e–5
26.40	4.03e–4	9.81e–5	4.18e–5

We obtained the gamma dose by folding gamma spectra into the gamma–fluence–to–dose function[3]. These gamma rays are from (n, gamma) reaction with shielding materials. In this study, the principal investigators do not consider the intrinsic gamma rays from the fissionable material and its decay products. Note that when CH_2 or borated CH_2 ~~are~~ is used as shielding, we have a greater gamma dose from the 2.2 MeV gamma ray produced by H(n,gamma) reactions, and 4.4 MeV gamma rays produced by C(n,n'gamma) reactions.

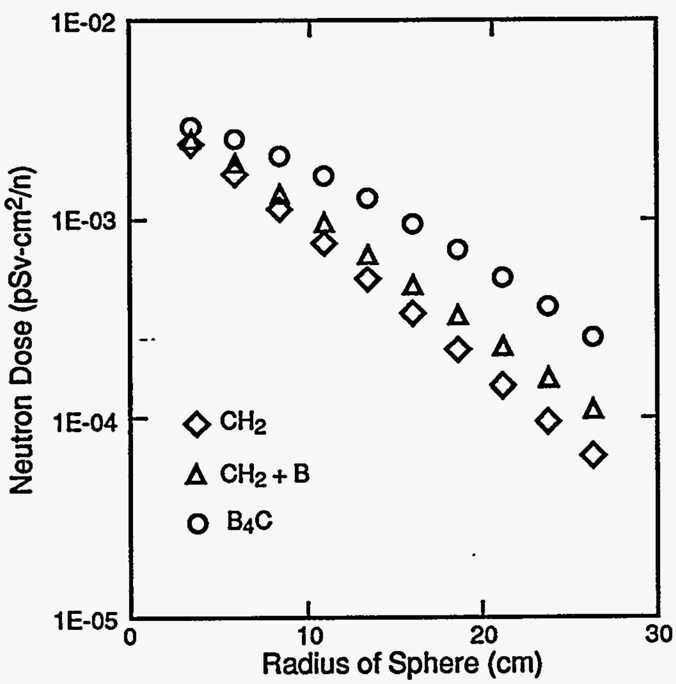

Figure 5. Neutron dose at 1 meter

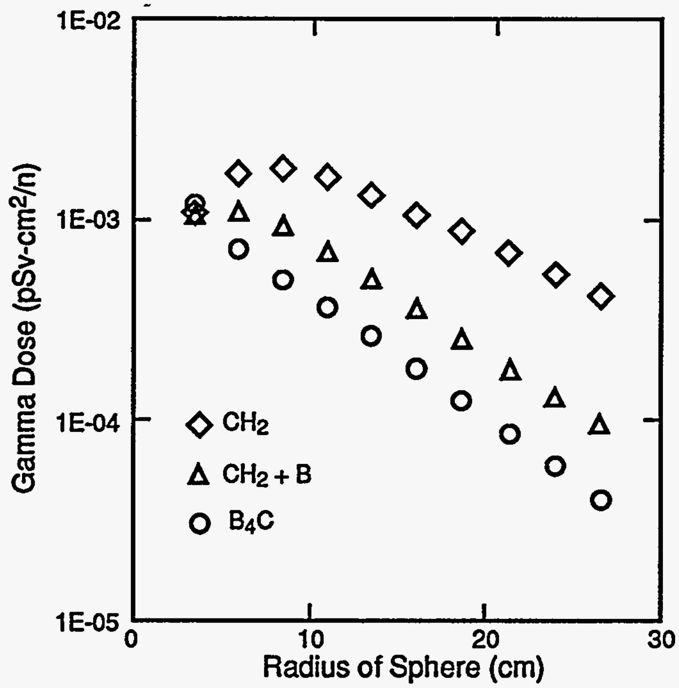

Figure 6. Gamma dose at surface

Effect on Criticality

To calculate the increase of ^{239}Pu fissions, we placed a very small amount of ^{239}Pu in the cavity. We folded the ^{239}Pu fission cross section in the cavity with neutron spectra, which was reflected back by the different shielding materials. Figure 7 shows reflected neutron spectra from the different 1-in-thick shielding materials.

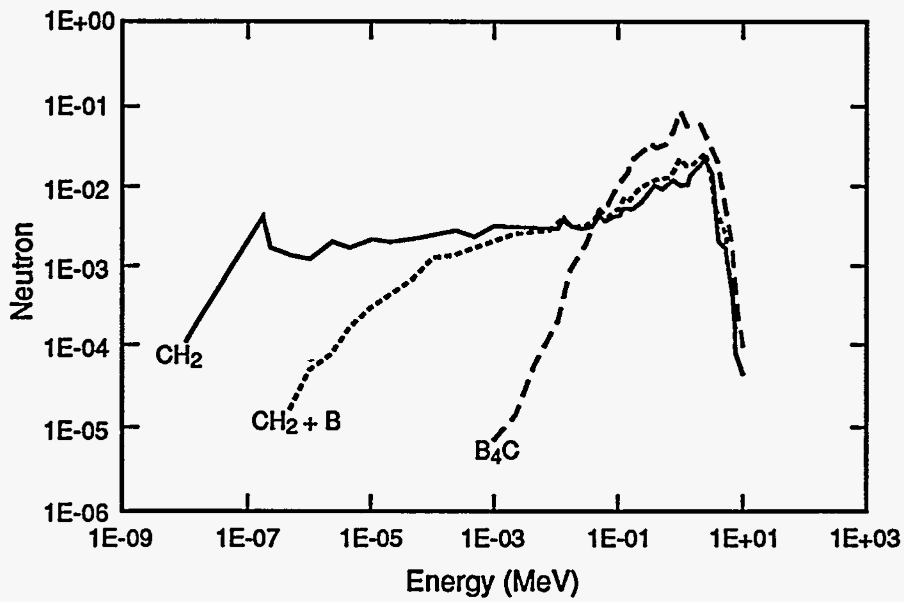

Figure 7. Neutron spectra reflected from 1-inch shielding

We observe that the CH_2 wall reflects many more low-energy neutrons back into the cavity. ^{239}Pu has a very high fission cross section at low-neutron energy. When contrasted with the CH_2 wall, we observe both the B_4C and borated CH_2 walls cause about 10 times fewer fissions in the cavity. When one considers questions of criticality, the conclusion is that B_4C and borated CH_2 are better materials.

Using these calculations, we conclude that a lining of combined CH_2 and B_4C provides good shielding.

III. PORTABLE SHIELDING CONTAINERS

Two of the most commonly used cylindrical stainless steel containers for the storage of fissionable materials at LANL are 12.7 cm x 17.78 cm and 20.32 cm x 25.4 cm (diameter x height). The shielding containers have cylindrical cavities of 15.24 cm x 18.10 cm and 22.86 cm x 25.72 cm. Lining the cavities are layers of 0.32-cm-thick B_4C or borated CH_2, 7.62-cm-thick-CH_2, and 0.16-cm stainless steel.

We performed the same Monte Carlo calculations for each of LANL's two designs. The following tables present a summary of results. All dose values are in pSv–cm^2 / neutron.

Small-sized container

	Without shielding	With shielding	
		B_4C	Borated CH_2
^{239}Pu fission	1.96e–6	9.95e–6	1.31e–5
Neutron dose at surface	4.22e–1	2.27e–2	2.28e–2
Neutron dose at 1 m	3.08e–3	8.11e–4	8.15e–4
Gamma dose at surface	2.30e–4	5.92e–4	6.55e–4
Gamma dose at 1 m	1.71e–6	2.18e–5	2.42e–5
Weight of container	1.33 Kg	29.7 Kg	29.1 Kg

Neutron dose at the surface is reduced by a factor of 18; at 1 m, dose is reduced by a factor of 3.8. The increased gamma dose is a small part of the total dose.

Large-sized container

	Without shielding	With shielding	
		B_4C	Borated CH_2
^{239}Pu fission	8.47e–7	1.24e–6	1.92e–6
Neutron dose at surface	1.81e–1	1.41e–2	1.41e–2
Neutron dose at 1 meter	3.08e–3	7.74e–4	7.76e–4
Gamma dose at surface	1.33e–4	3.37e–4	3.73e–4
Gamma dose at 1 meter	2.34e–6	1.91e–5	2.11e–5
Weight of container	3.90 Kg	50.1 Kg	48.9 Kg

The neutron dose at the surface is reduced by a factor of about 13; at 1 m, dose is reduced by a factor of approximately four. The increase of gamma dose is a small part of the total dose.

IV. CONCLUSION

The new designs for a portable shielded container, while not optimal, are very simple and achieve the goals set for this study:

- the unit is transportable and
- neutron dose is reduced by predicted factors.

If criticality were not an issue, we could further simplify the design by using CH_2 but only for shielding that reduces weight, complexity of construction , and cost. Our data show not much difference in exposure rates between B_4C and borated CH_2 used for the inner layer of the container; however, the hardness of B_4C is a positive characteristic that prevents wear and tear on the inner surface of the container.

V. ACKNOWLEDGMENTS

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