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LAWRENCE LIVERMORE LABORATORY
University of California/Livermore, California

THE USE OF ^{10}B -LOADED PLASTIC IN PERSONNEL
NEUTRON DOSIMETRY

Richard V. Griffith

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THE USE OF ^{10}B -LOADED PLASTIC IN PERSONNEL NEUTRON DOSIMETRY*

Richard V. Griffith

Lawrence Livermore Laboratory, University of California
Livermore, California 94550

REVISED ABSTRACT

At the Lawrence Livermore Laboratory, various plastics loaded with enriched ^{10}B have been used as small shields for personnel dosimeters, in environmental neutron monitors, and for special dosimetry application. Unlike other forms of ^{10}B shielding, the borated plastic is inexpensive to fabricate, easy to handle, and can be shaped for fabricating special shields.

Small discs of plastic filled with ^{10}B to a density of 1.40 g/cm^3 are now used as shields for copper and indium foils in nuclear-accident dosimeters worn by Laboratory personnel. The use of borated plastic in place of cadmium for low-energy neutron shielding increases the effective threshold for the $^{63}\text{Cu}(\eta, \gamma)^{64}\text{Cu}$ reaction by nearly two orders of magnitude. Also, by suppressing the low-energy activation, the copper response is more representative of the neutron flux for 0.1 keV to 1 MeV, which enhances the usefulness of copper as an accident dosimeter. The indium foil is covered with borated plastic to suppress the $^{115}\text{In}(\eta, \gamma)^{116\text{m}}\text{In}$ reaction. Thus, the activation of the more useful $^{115}\text{In}(\eta, n')^{115\text{m}}\text{In}$ threshold reaction can be almost immediately determined, without the long delay usually required for decay of the short-lived $^{116\text{m}}\text{In}$ activity. We are also using borated plastic in the development of a TLD albedo dosimeter to reduce the low-energy neutron response of the $^6\text{Li}(\eta, \alpha)^4\text{T}$ reaction in TLD-600.

In addition to these applications, we have taken advantage of the $^{10}\text{B}(\eta, \alpha)^7\text{Li}$ reaction in the plastic to locate and define low-energy neutron beams from various neutron-generating facilities. For this application, the borated material is placed against a sheet of cellulose nitrate in a neutron beam. When the cellulose nitrate is chemically etched, the pattern made by the alpha tracks accurately defines the neutron field.

*Work performed under the auspices of the U. S. Atomic Energy Commission.

INTRODUCTION

Measurement of neutron dose in the region from 1 eV to 1 MeV (intermediate energy) is particularly difficult in personnel neutron dosimetry, partially because of the magnitude of the energy range. Detectors normally used in this region involve reactions with response functions having the typical "1/v" shape; reactions of this type include the $^{63}\text{Cu}(\eta, \gamma)^{64}\text{Cu}$ reaction commonly used in nuclear accident dosimetry [1] and the $^6\text{Li}(\eta, \alpha)^3\text{H}$ reaction, which is finding increasing use in albedo-type personnel dosimetry [2,3,4]. The unfortunate aspect of the 1/v detector is that it is used to determine dose in a region in which the dose response increases for energies greater than about 400 eV. As a result, the 1/v detector gives serious overemphasis to the low-energy neutrons; and considerable errors in dose estimation can result if the spectrum is unknown, as it usually is.

In radiation-protection work, the 1 eV-1 MeV region can be responsible for a significant portion of the dose. For a fission source, the 5-cm-depth dose fraction in the region from 1 eV to 1 MeV ranges from about 12% for an uncollided spectrum to as much as 40 to 50% for a heavily moderated configuration. It is clear that the dose component in this energy range can neither be ignored nor safely inferred from higher-energy detectors. At present, there does not seem to be a universally acceptable alternative to the 1/v detector for measuring intermediate-energy neutrons.

When we are faced with the problem of using detectors whose responses vary by as much as four orders of magnitude, the question is, "What can we do about the low-energy overresponse?" Generally, the solution has been to wrap the detectors, copper, TLD-600, etc., in cadmium, which essentially eliminates response to neutrons below about 0.5 eV. However, there remains significant overresponse to neutrons with energies from 0.5 to 100 eV, which caused us to search for another cover material that would reduce detector response in this range.

Boron-10 has been used for many years to reduce the low-energy response of the $^{239}\text{Pu}(\eta, \text{fission})$ reaction for use in area criticality monitoring [5]. In that capacity it has been used as boron carbide in sintered spheres. The use of 1.65 g/cm² of ^{10}B gives the plutonium reaction an effective threshold of 1 to 10 keV. It was this fact that caused us to become interested in the possibilities for ^{10}B as a shielding material for personnel dosimeters. Figure 1 shows the relative cross sections for cadmium and ^{10}B absorption [6]. We wished to take advantage of the significantly higher ^{10}B cross section in the region from 0.5 eV to 1 keV.

Plastic

The fabrication cost for sintered boron carbide would make it prohibitive for use in small shields in personnel dosimetry. However, we found that the plastics shop at our Laboratory could provide sheets of plastic such as polyethylene or Viton A¹ with very high ^{10}B loading, using enriched crystalline boron.

¹Reference to a company or product name does not imply approval or recommendation of the product by the University of California or the U. S. Atomic Energy Commission to the exclusion of others that may be suitable.

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Fig. 1. Relative cross sections for cadmium and boron. The cross-section data were taken from Ref. 6.

The material used in most of our work was a mixture of Viton A and enriched boron, formed in a rubber mill into sheets 0.16 cm thick. The finished material, which had 82% enriched ^{10}B , contained 81% total boron by weight and was formed to a density 2.1 g/cm^3 , yielding an overall ^{10}B density of 1.4 g/cm^3 . In the thickness we used, the ^{10}B thickness was 0.22 g/cm^2 . The finished plastic sheet is brittle, but it can be bent and is quite durable. It can be punched or machined.

APPLICATION

Nuclear Accident Dosimetry

The first use of our borated plastic was as a low-energy neutron shield for the copper foil in personnel nuclear-accident dosimeters. To estimate the effect of the plastic on detector response, we made calculations for bare copper, copper covered with 0.05 and 0.1 cm (0.43 and 0.86 g/cm^2) of cadmium, and copper covered with 0.16 and 0.32 cm (0.23 and 0.47 g/cm^2 ^{10}B) of borated plastic. Using a high-speed computer, we calculated the relative response of these detector configurations to 35 different neutron spectra used as part of the SAND II computer code [6]. As we had a ^{252}Cf neutron source with a strength of about 10^8 n/sec available, we also made response calculations using the ^{252}Cf neutron spectrum and computer-calculated transmission spectra for ^{252}Cf centered in 12-30-, and 50-cm-diam spheres of D_2O , graphite, steel, and lucite. All calculations were made to determine the effective response for the particular detector-shield configuration over the energy range from 1 eV to 1 MeV. Figure 2 shows the effective-response function curves for the $^{63}\text{Cu}(\eta, \gamma)^{64}\text{Cu}$ reaction with and without shields of cadmium and borated plastic. For simplicity, the curve for 0.1 cm of cadmium has been omitted, since it differs little from the curve for 0.05 cm of cadmium, except at 0.025 eV, where the extra material suppresses the response significantly.

Data for the SAND II neutron spectra are shown in Table I, and the ^{252}Cf -moderator data are shown in Table II. The effective-response data for the uncovered detector demonstrates the need for cadmium shielding, which will be a surprise to no one. Note, however, that the use of 0.16 cm of borated plastic reduces the expected variation in weighted cross section by a factor of 2.2 when compared with cadmium for all spectra considered. There seems to be little further improvement in using 0.32 cm of borated plastic.

We made irradiations with moderated and unmoderated ^{252}Cf neutrons in our calibration facility, to estimate the accuracy of the computer calculations. The moderators used for these irradiations were 12-, 30-, and 50-cm-diam spheres of D_2O ; we calculated the spectra for these spheres (Fig. 3) with a Monte Carlo computer code. Copper foil, covered with 0.05 cm of cadmium or 0.16 cm of borated plastic, was exposed at 50 cm from the center source for a period of 16 hr. With a sodium iodide crystal, we assayed the ^{64}Cu activity, and we determined the effective reaction cross section from the computer-calculated fluence from the moderating spheres in the 1 eV to 1 MeV region. With a cadmium cover, the average cross section was determined empirically to be 0.236 ± 0.035 barns, whereas the calculation predicted 0.242 barns. With a borated plastic cover, the average cross section was determined empirically to be 0.127 ± 0.025 barns, compared with a calculated value of 0.155 barn.

Fig. 2. Relative $^{63}\text{Cu}(\eta, \gamma)^{64}\text{Cu}$ cross section with cadmium or ^{10}B covers.

TABLE I

Calculated Effective Cross Sections for $^{63}\text{Cu}(\eta, \gamma)^{64}\text{Cu}$
 from 1 eV to 1 MeV when Exposed to SAND II Neutron Spectra

Cover material	Bare	Cadmium	^{10}B	^{10}B
Thickness, g/cm^2	0	0.43	0.23	0.47
Average cross section, σ , barns	1.71	0.235	0.132	0.108
σ minimum, barns	0.0236	0.0232	0.0234	0.0231
σ maximum, barns	16.0	0.565	0.251	0.212
σ max/ σ min	678.	24.3	10.7	9.2

TABLE II

Calculated Effective Cross Sections for $^{63}\text{Cu}(\eta, \gamma)^{64}\text{Cu}$
from 1 eV to 1 MeV when Exposed to ^{252}Cf and Moderated ^{252}Cf Spectra

Cover material	Bare	Cadmium	^{10}B	^{10}B
Thickness, g/cm ²	0	0.43	0.23	0.47
Average cross section, σ , barns	0.334	0.183	0.120	0.099
σ minimum, barns	0.0298	0.0293	0.0289	0.0281
σ maximum, barns	0.918	0.422	0.245	0.195
σ max/ σ min	30.8	14.4	8.5	6.9

Fig. 3. Spectra of ^{252}Cf in D_2O spheres normalized to 10 MeV.

We had an opportunity to test the use of borated plastic coverings at the Seventh Intercomparison of Nuclear Accident Dosimetry Systems held at the Oak Ridge National Laboratory (ORNL) in 1970. We used our conventional personnel accident dosimeters, which include bare and cadmium-covered gold foils, for measurement of thermal neutrons; a cadmium-covered copper foil for measurement of "intermediate-energy neutrons"; a cadmium-covered indium foil, which measures the neutrons in excess of 1 MeV through the $^{115}\text{In}(\eta, \eta')^{115}\text{In}$ reaction; a sulfur pellet for neutrons above 3 MeV; and a sodium chloride pellet, which helps establish body orientation through comparison with body sodium activation. In addition to the normal badge, we used a copper foil covered with 0.16 cm of borated plastic.

Detectors were placed on the front of a water-filled human phantom for each of three pulses from the Health Physics Research Reactor (HPRR) used to simulate accident conditions. The phantom was exposed to a bare reactor spectrum and to two moderated pulses through a 12-cm hydrogenous plastic shield and a 13-cm steel shield. For the unmoderated burst, the total dose estimate from the data for boron-covered copper agreed with the dose estimate from the standard badge to within 7%. However, for the two moderated bursts, use of the data from the boron-covered copper instead of the data from the cadmium-covered copper reduced the difference in the dose estimate from the dosimeter by an average of 33%, when compared with Hurst proportional-counter data provided by the ORNL staff.

At the Oak Ridge intercomparison, we also exposed indium foils in borated plastic, and we found two extra benefits to be gained from use of the plastic.

First, we found that the boron suppresses the $^{115}\text{In}(\eta, \gamma)^{116\text{m}}\text{In}$ activation quite drastically: a factor of 40 or more. With a cadmium cover, the $^{116\text{m}}\text{In}$ activity completely overwhelms the $^{115\text{m}}\text{In}$ activity from the 1-MeV threshold reaction, so that a matter of hours must normally be allowed before the $^{115\text{m}}\text{In}$ can be analyzed. This is not desirable, following an accident when early data are important. Use of the borated plastic to cover the indium precludes the long waiting time and further decay of the $^{115\text{m}}\text{In}$ activity, which is often low to start with. Figure 4 shows the computer-calculated spectrum of an indium foil with a sodium iodide crystal 1 hr after irradiation by a fission source through 15 cm of lucite. The 0.335-MeV gamma activity from $^{115\text{m}}\text{In}$ is barely suggested in the cadmium-covered foil spectrum, but is well formed in the spectrum from the boron-covered foil. A decay time of 7.9 hr would be required before the cadmium-covered foil spectrum would have the same relative shape as that covered with boron, and during that time the $^{115\text{m}}\text{In}$ activity would be reduced by 70%.

A second advantage of covering the indium foil with borated plastic is that it appears that the $^{115}\text{In}(\eta, \gamma)^{116\text{m}}\text{In}$ reaction could also be used as a detector for 1 eV-1 MeV neutrons. Figure 5 shows the suppression of the low-energy response region in the $^{115}\text{In}(\eta, \gamma)^{116\text{m}}\text{In}$ reaction through use of a 0.32-cm borated plastic cover. The result is that the response of the boron-covered indium is less variant than the copper cross section in the 1 eV-1 MeV region. Table III shows the comparison of the computer-calculated responses for boron-covered copper and indium to the 35 SAND II and 13 ^{252}Cf -moderated spectra discussed earlier. The variation in response for the indium foil is about 1/3 that of the copper foil. Irradiations

Fig. 4. Calculated sodium iodide spectra of indium foils 1 hr after exposure.

Fig. 5. Relative $^{115}\text{In}(\eta, \gamma)^{116\text{m}}\text{In}$ cross section with cadmium or ^{10}B cover. The data for the cross section without cover were derived at LLL.

TABLE III

Calculated Effective Cross Sections for $^{115}\text{In}(\eta, \gamma)^{116\text{m}}\text{In}$ and
 $^{63}\text{Cu}(\eta, \gamma)^{64}\text{Cu}$ Reaction from 1 eV to 1 MeV

Neutron spectra	SAND II		Moderated ^{252}Cf	
	Copper	Indium	Copper	Indium
Foil material	^{10}B	^{10}B	^{10}B	^{10}B
Cover material	^{10}B	^{10}B	^{10}B	^{10}B
Thickness, g/cm^2	0.23	0.47	0.23	0.47
Average cross section, σ , barns	0.132	0.663	0.120	0.629
σ minimum, barns	0.251	0.932	0.245	0.936
σ maximum, barns	0.0234	0.296	0.0289	0.301
σ max/ σ min	10.7	3.1	8.5	3.1

similar to those for copper were made with the ^{252}Cf source in D_2O moderators to determine the accuracy of the calculated response function. The measured average cross section was 0.75 ± 0.15 barns for the foil covered with 0.32 cm of borated plastic, compared with a calculated value of 0.75 barns. The agreement seems adequate.

Although tests of the indium foil used in this way during the Oak Ridge intercomparison were incomplete, they indicated that a reduction in the average dosimeter error by as much as a factor of 2 could be achieved using a boron-covered indium foil to measure the 1 eV-1 MeV neutrons. In addition, the sensitivity of the indium foil in disintegrations per gram is greater than that of the copper for over 5 hr after irradiation.

As a result of these tests, we have now issued the dosimeter shown in Fig. 6 to 110 employees who work in areas of potential criticality throughout our Laboratory. Our routine TLD gamma dosimeter disc is placed inside the large hole. In this badge, foils of copper and indium are covered with 1.1-cm-diam discs of 0.16-cm-thick borated plastic. A spacer ring of borated plastic is used to contain the foils and shield the edges. The other dosimeter materials are those cited previously with the addition of an uncovered indium foil for postaccident personnel screening.

Albedo Dosimetry

A number of laboratories are now using or developing TLD albedo dosimeters for routine personnel neutron dosimetry [2, 3, 4]. Many of these involve pairs of TLD containing ^6Li : one is covered on the incident side with cadmium, and the other is uncovered or covered on the body side. Cadmium is used in this way to provide a crude two-point spectrometer to give spectral quality information that can be compared with detector calibrations to help arrive at a dose estimate.

We are involved in albedo-dosimeter development studies, using borated plastic to do the same thing. The plastic used is similar to that described previously, except that the boron enrichment is lower and the plastic is thicker than that used in our nuclear-accident dosimeter. By looking at Fig. 7, we can see the relative responses for bare, cadmium-covered, and boron-covered $^6\text{Li}(\eta, \alpha)^3\text{H}$ detectors. The increased suppression at low energies should give a better estimate of the low-energy/high-energy spectral quality. One uncovered ^6Li TLD (either TLD 100 or TLD 600) and one covered only on the front with borated plastic will have the same albedo response. However, the uncovered dosimeter also responds to incident low-energy neutrons, whereas the boron-covered dosimeter does not. The spectral quality can then be estimated by examining the ratio of difference in detector responses (low-energy incident neutrons) to the boron-covered detector response (high-energy incident plus albedo neutrons).

In addition to the potential improvement in spectral information with the use of borated plastic, the use of boron largely avoids the problem of capture gamma from thermal neutrons experienced with cadmium. These gammas only serve to add confusion to the gamma-detecting TLD system, even with ^6Li -depleted TLD-700 used for gamma compensation. A further benefit of borated plastic is that it is lower in density than cadmium, potentially yielding a lighter dosimeter.

Fig. 6. Personnel nuclear-accident dosimeter.

Fig. 7. Relative ${}^6\text{Li}(\eta, \alpha){}^3\text{H}$ cross section with cadmium or ${}^{10}\text{B}$ cover. The data for ${}^6\text{Li}(\eta, \alpha){}^3\text{H}$ were derived at LLL.

Other Applications

The borated plastic can be used to fabricate special dosimeter shields for personnel protection and for experimentation. In one such situation, we needed fission foil dosimeters, but did not have room for the traditional 5-cm boron carbide shields [5]. We fabricated special cans with liners made from borated plastic rings (Fig. 8). The final thickness of ^{10}B was 0.82 g/cm^2 , compared with the 1.65 g/cm^2 used in the boron carbide spheres. Figure 9 shows the relative response curve for a ^{239}Pu (η , fission) foil in these two containers. A computer comparison using the SAND II and ^{252}Cf neutron spectra showed that, by using a threshold of 0.1 keV , we actually suffer less potential variation in the effective cross section with a 0.82-g/cm^2 borated plastic shield than with the 1.65-g/cm^2 boron carbide sphere with an assumed threshold of 10 keV [6] (Table IV).

We have used the borated plastic as an alpha radiator, by using the $^{10}\text{B}(\eta, \alpha)^7\text{Li}$ reaction. With cellulose nitrate registration material, one can use sheets of borated plastic in this way for neutron radiography. In one experiment with reactor-produced neutrons, using a common electronic circuit board as a sample, we were able to get image resolutions in the cellulose nitrate sufficient for us to see the fine fiberglass patterns in the board. It is also possible, particularly with plastic containing enriched ^{10}B , to do radiography by using our ^{252}Cf source in a D_2O moderating sphere.

CONCLUSIONS

Sheets of plastic with a high loading of enriched ^{10}B can be fabricated with relative ease and have wide utility in neutron dosimetry, both as a low-energy shielding material and as a neutron detector through the $^{10}\text{B}(\eta, \alpha)^7\text{Li}$ reaction. For some shielding applications, it is superior to cadmium. It can be used in nuclear-accident dosimetry to reduce the spectral response variations for η, γ detection in the 1 eV - 1 MeV energy range, resulting in more uniform and accurate dosimeter response. The borated plastic can also be used to suppress undesirable η, γ activity in some threshold detector applications. It can be used to improve the performance of albedo dosimeters and, at the same time, eliminate unwanted gamma response from neutron capture in cadmium. The relative ease of handling makes it useful for fabrication of neutron shields for special dosimetric and experimental applications, and it can be used for image transfer in neutron radiography.

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Fig. 8. Threshold detector neutron shields, showing liners made of ^{10}B -loaded plastic.

Fig. 9. Relative $^{239}\text{Pu}(\eta, \text{fission})$ cross section in ^{10}B -loaded shields. The data for the $^{239}\text{Pu}(\eta, \text{fission})$ cross section were taken from Ref. 6.

TABLE IV

Calculated Effective Cross Sections for $^{239}\text{Pu}(\eta, \text{fission})$
in 5 cm ^{10}B Carbide Sphere and Lawrence Livermore Laboratory
 ^{10}B Plastic Shields

Shield	Boron carbide sphere		Borated plastic cylinder	
^{10}B thickness	1.65 g/cm ²		0.82 g/cm ²	
Effective threshold	10 keV		0.1 keV	
Neutron spectra	SAND II	Moderated ^{252}Cf	SAND II	Moderated ^{252}Cf
Average cross section, σ , barns	1.83	1.81	1.75	1.75
Standard deviation, barns	0.17	0.18	0.07	0.07
σ minimum, barns	1.54	1.57	1.61	1.6
σ maximum, barns	2.24	2.29	1.88	1.84
σ max/ σ min	1.45	1.46	1.17	1.15

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FIGURE CAPTIONS

- Fig. 1. Relative cross sections for cadmium and boron. The cross-section data were taken from Ref. 6.
- Fig. 2. Relative $^{63}\text{Cu}(\eta, \gamma)^{64}\text{Cu}$ cross section with cadmium or ^{10}B covers.
- Fig. 3. Spectra of ^{252}Cf in D_2O spheres normalized to 10 MeV.
- Fig. 4. Calculated sodium iodide spectra of indium foils 1 hr after exposure.
- Fig. 5. Relative $^{115}\text{In}(\eta, \gamma)^{116\text{m}}\text{In}$ cross section with cadmium or ^{10}B cover. The data for the cross section without cover were derived at LLL.
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