THE RESPONSE OF SOME THERMOLUMINESCENT MATERIALS TO NEUTRONS

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1. INTRODUCTION

Thermoluminescent materials are very often used for gamma dosimetry. However, they exhibit a response to neutrons and in mixed fields therefore, the results given by these dosimeters must be corrected.

This correction is a function of the detector, of its environment and varies with the neutron energy (E_n). We present here the results obtained recently in our laboratory for monoenergetic neutrons and combine with these those we obtained with polyenergetic neutrons (1).

2. NEUTRON SOURCES AND DETECTORS STUDIED

2.1. Neutron sources

- polyenergetic sources :
  - thermal neutrons (pool reactor TRITON)
  - Pu-Be source (10 Ci)
  - ^{252}Cf source.

The fluence rate of these sources has been determined by activation measurements or by the manganese bath method.

- monoenergetic neutrons :

Experiments were carried out near a number of installations : Van de Graaff generators (5 MeV at Cadarache (Cad), 4 MeV at Bruyères-le-Chatel (B 3) and 3 MeV at the G.S.F. in Munich (G.S.F.)), the neutron generator of the R.D.I. in Prague (R.D.I.) and the SAMES accelerator of STEP at Fontenay-aux-Roses (F.A.R.).

In the first two cases the neutron fluence has been determined with the aid of a directional counter; the energy and fluence at the irradiation point have been recalculated according to known data (2), (3). At Prague the reaction $^{56}Fe(n,p)^{56}Mn$ has been used. For calculating the kerma value we have referred to (4).

The irradiations at Munich were made during the ENDIP experiments organized in 1975 by EURATOM and the neutron doses considered here are the preliminary results obtained by the Munich team (5)(6). At Fontenay many methods have been used, among others a T.E. chamber.

2.2. Thermoluminescent detectors studied

These were powdered forms of the following :

- $CaSO_4$: Dy
- ^7LiF : Mg (99,95 % ^7Li)
- ^{nat}LiF : Mg
- Al_2O_3

En (MeV)	Place	$CaSO_4$		^7LiF		^{nat}LiF		Al_2O_3	
		Pl	Al	Pl	Al	Pl	Al	Pl	Al
0.252	Cad			1.3 ± 0.4	0.9 ± 0.3	6.1 ± 0.8	5.9 ± 0.6		
0.350	"					5.9 ± 1.6	6.2 ± 1.8		
0.716	"	0.5 ± 0.2	0.2 ± 0.1	1.6 ± 0.2	1.2 ± 0.3	3.0 ± 0.4	1.6 ± 0.2		
1.4	B 3			0.9 ± 0.1	0.8 ± 0.1	1.7 ± 0.2	1.7 ± 0.2	0.9 ± 0.2	0.8 ± 0.1
2.1	GSF		0.4 ± 0.2	1.4 ± 0.3	1.5 ± 0.4	2.1 ± 0.4	2.3 ± 0.5		
2.2	Cad	1.5 ± 0.2	0.4 ± 0.1	2.2 ± 0.3	1.3 ± 0.4	2.5 ± 0.4	1.7 ± 0.3		
3	B 3	1.8 ± 0.3		2.3 ± 0.3		2.2 ± 0.5		1.8 ± 0.3	
3.3	FAR	1.8 ± 0.3	1.1 ± 0.2	2.1 ± 0.3	1.5 ± 0.3	2.8 ± 0.3	1.9 ± 0.2	1.9 ± 0.4	1.1 ± 0.3
5.25	GSF	5.1 ± 1.3	3.3 ± 1.0	3.9 ± 1.2	2.9 ± 1.1	4.3 ± 1.3	2.4 ± 0.9	5.0 ± 2.5	2.2 ± 0.9
7	B 3	7.3 ± 1.7	3.0 ± 1.4	7.3 ± 1.3	3.6 ± 1.8	7.5 ± 1.5	3.0 ± 1.1	7.5 ± 1.4	2.5 ± 1.1
14.7	RDI	24.2 ± 1.2	9.8 ± 0.8	25.2 ± 1.2	10.2 ± 0.5	32.6 ± 2.3	13.5 ± 0.4	37.6 ± 2.3	17.1 ± 0.5
15.1	GSF	27.6 ± 2.9	10.4 ± 1.4	25.3 ± 3.3	10.6 ± 1.6	28.6 ± 3.2	11.2 ± 1.4	38.6 ± 4.1	19.8 ± 2.0
^{252}Cf	FAR	3.1 ± 2.4	$1.1 ^{+1.4}_{-1.1}$	5.3 ± 0.9	2.6 ± 1.4	9.3 ± 1.7	5.7 ± 1.4	3.4 ± 0.8	$1.3 ^{+1.4}_{-1.3}$
PuBe	"	5.6	3.3 ± 1.4	6.4	5.6 ± 1.4	8.7	7.5 ± 1.4	5.4	$1.1 ^{+1.4}_{-1.1}$
* th	"		$2.1 ^{+2.8}_{-2.1}$		8.8 ± 1.4		1580 ± 140		1.7 ± 1.3

* For thermal neutrons the tabulated value is k.

TABLE 1 - RELATIVE RESPONSE OF TLDs TO NEUTRONS ($k \times 10^2$)*

They were irradiated in polyethylene tubes (3.9 mm internal diam.; wall thickness 1.1 mm; referred to as Pl) and aluminium tubes (3.1 mm internal diam.; wall thickness 0.9 mm; referred to as Al).

2.3. Dosimeter evaluation

The detectors are evaluated using a LDT 20 (Saphymo Srat) reader standardized in our laboratory using a ^{60}Co source. The emitted signal is translated in terms of dose of gamma ^{60}Co in a unit tissue volume.

2.4. Gamma exposure measurements

Two methods have been used to determine the gamma exposure associated with the neutron beam at the irradiation point.

- photographic emulsion : according to the method described by Bewley (7) the emulsion is enclosed in a lead sheath.

- Geiger-Muller counter (Philips 18509) under tin and lead shields (8) with an additional ^{6}LiF shield.

The neutron response of these dosimeters is very slight and has been considered here as negligible.

3. THERMOLUMINESCENT DOSIMETER NEUTRON RESPONSE.

3.1. Results

We define the relative response (k) by : $k = \frac{L - X}{D_N}$

where L is the dosimeter reading, X is the dose in a unit tissue volume due to gamma only, D_N is the neutron dose (tissue kerma) and k expresses the neutron response in terms of gamma ^{60}Co.

Table 1 presents the results obtained. The powders irradiated under aluminium yield results approaching the intrinsic response. However the relative response obtained under polyethylene more closely represents current practice.

3.2. Discussion

In general, the relative response of TLD's increases with the neutron energy (E_n); the calculated kerma values for the same materials show a similar trend. Comparison of these two results enables interesting conclusions to be drawn regarding the thermoluminescent yield (9).

In the case $E_n < 3$ MeV the intrinsic responses are inferior to 0.02. All the detectors, except of course ^{nat}LiF, can therefore be used for gamma measurements in a mixed field almost without correction. For higher energies and polyenergetic sources (Cf, Pu-Be) corrections are necessary and become increasingly important with increasing values of E_n. This trend is most marked in Al_2O_3 and has suggested an application (10) for total dose measurements in mixed gamma and 14.7 MeV neutron fields.

Table 1 also shows that the influence of recoil protons (i.e. the difference between the responses obtained using the polyethylene and aluminium containers) is not negligible especially for high values of E_n. In this case therefore, it would be desirable to use non-hydrogenous containers.

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