

IRPA-10
10th International Congress of
The International Radiation Protection Association

Eye-Opener n.12

Mixed Field Dosimetry

A. Delgado.

CIEMAT, Avda Complutense 22, 28040 Madrid. SPAIN

E-mail: Antonio.Delgado@ciemat.es

INDEX

1.- Introduction.	1
2.- Principles of Measurements.	2
3.- Dosimetry of Mixed Beta-Gamma fields.	3
Relevant Quantities	
Instruments for Area Monitoring	
Dosemeters for Individual Monitoring	
4.- Dosimetry of mixed Neutron-Gamma Fields.	10
Relevant Quantities	
Instruments for Area Monitoring	
Dosemeters for Individual Monitoring	
5.- Summary.	26
6.- References.	26

INTRODUCTION

Mixed fields are those composed by radiation of different types and/or energy, e.g. photons and electrons, photons and neutrons or even neutrons with sufficiently different energy share some of the characteristics of a mixed field. Mixed fields are rather common and in fact more the rule than the exception, although very often in practice the doses caused by one of the field components are preponderant so that the contribution of the others can be neglected. When this is not the case, dosimetry of mixed fields for radiation protection purposes, i.e. for determination of the dose equivalent, presents some difficulty caused by the necessity of properly identifying the contribution of each field component.

The International Commission on Radiological Protection (ICRP) in its report 60 (ICRP 60, 1990), immediately after proposing the Absorbed Dose (D) as the fundamental dosimetric quantity in radiological protection, introducing the radiation weighting factors (w_R) and defining the Equivalent Dose (H) as the product of both, D and w_R , states:

When the radiation field is composed of types and energies with different values of w_R , the absorbed dose must be subdivided in blocks, each with its own value of w_R and summed to give the total equivalent dose.

Therefore, a first requirement for mixed field dosimetry is of a conceptual nature and independent determinations of the absorbed dose caused by each field component with different w_R , are necessary. This requirement seems to call for complex dosimetric systems, including specific dosimeters for each radiation component or alternatively a single dosimeter but with analysis capability of the mixed field and its components.

Mixed fields can as well be composed by radiation of different nature but with the same weighting factors, beta-photon fields for example. Considering the experimental measurement of the operational dose equivalent quantities in those mixed fields, some practical problems appear as it is usually difficult to measure the dose equivalent with a single detector. This difficulty is due to the different sensitivities, implying different calibration factors for each field component, or to the different measurement conditions required, such as in measurement of penetrating and non-penetrating radiation.

Among the situations and activities in which mixed radiation fields are found, specific workplaces in nuclear power plants can be cited, as well as other activities related to the nuclear fuel cycle. Mixed fields can be found around medical and research high energy accelerators, in civil and military flights at high altitude and in conditions which exist in the exploration of outer space.

In general, there are technically more or less acceptable solutions for area dosimetry for the most common mixed fields, so that the actual situation could be considered as relatively comfortable. However, new refinements and simplifications of presently existing devices will be welcome. For some applications the methods of personal dosimetry are in a comparatively lower development stage. One example is neutron personal dosimetry, where the situation is rather unsatisfactory.

The characteristic features of the methods and techniques employed for the dosimetry of mixed fields will be described in this lecture, with particular reference to the dosimetry

of beta-gamma and neutron-gamma fields, the most frequently found and also the best studied mixed fields.

PRINCIPLES OF MEASUREMENT

For the experimental determination of the ICRU operational dose equivalent quantities, for area and individual monitoring, two basic approaches have been followed, employing either the absorbed dose or the particle fluence as the measurable physical quantity. The definition of the dose equivalent as an absorbed dose modified by a dimensionless quantity, the quality factor, renders natural the choice of the absorbed dose as the most directly related measurable quantity. Nevertheless, sometimes it is easier to find detector materials presenting a more simple and efficient response to fluence than to absorbed dose, thus justifying fluence as a convenient choice.

In mixed fields absorbed dose detectors can be employed including a detector for each radiation component. Ideally these detectors should provide independent measurements in order to use the appropriate quality factor. Alternatively they could be directly calibrated in terms of the operational dose equivalent quantity for the type of radiation each one detects. The situation is simplified if all the field components have the same quality factors, for example in beta-photon fields. More interestingly, some absorbed dose detectors can provide information on the field composition, as Recombination Chambers and Tissue Equivalent Proportional Counters (TEPCs). TEPCs, can operate as LET spectrometers, providing information on the distribution of the absorbed dose in L (Linear Energy Transfer), D_L . This kind of detectors is particularly useful for the dosimetry of neutron-gamma fields, as they can separate the contribution of the low-LET and high-LET components. The distribution D_L is approximated by the distribution in lineal energy D_y , measured by the detector:

$$H = Q \cdot D = \int Q(y) \cdot D_y \cdot dy$$

Alternatively, the dose equivalent, H, can be also obtained starting from the measurement of the energy distribution of the neutron fluence Φ_E , making use of calculated conversion coefficients from fluence to dose equivalent, either ambient or personal, $h_\phi(E)$. H is then obtained by the equation:

$$H = \int h_\phi(E) \cdot \Phi_E \cdot dE$$

Geiger-Müller detectors are fluence detectors, as are some spectrometer systems employed for beta and neutron dosimetry.

Frequently, instead of measuring Φ_E , the neutron fluence distribution, it has been found advantageous to construct devices whose response, $R_\phi(E)$, in terms of the neutron energy matches that of the conversion coefficients, $h_\phi(E)$. If this is possible, then the reading of such instrument will be simply proportional to the dose equivalent independently of energy, what is a highly convenient characteristic for a dosimeter. The instrument reading, M, is given by:

$$M = \int R_\phi(E) \cdot \Phi_E \cdot dE$$

If N is the calibration factor, determined by exposing the device to known values of H in a metrology laboratory, then:

$$H=N \cdot M$$

It is difficult to find a detector material which by itself has the same energy dependence as the conversion coefficients $h_{\phi}(E)$. In practice, the starting device is usually a fluence detector showing a convenient sensitivity in an energy range. Adding structures, such as absorbers, moderators, converters or radiators to the detector makes it possible to modify its initial energy response, making it approach the desired one. Sometimes this can be achieved even extending the workable energy range. This approach has been followed for the design of many neutron dosimeters for area and personal dosimetry, and also of some photon and beta detectors, such as the compensated Geiger-Müller counters.

DOSIMETRY OF MIXED BETA-GAMMA FIELDS

a) Relevant Quantities

Beta rays can be generally considered as weakly penetrating radiations, although a few and not very common emitters can produce high-energy beta rays (^{106}Rh , 3.5 MeV or ^{16}N , 10.4 MeV) with not-so-weak penetration in tissue. For beta exposures the organs of concern are mostly the skin and, in some cases, the eyes or the testes. The pertinent operational quantities are the directional dose equivalent, $H(d, \Omega)$, for area monitoring and the personal dose equivalent, $H_p(d)$ for individual monitoring. The depth of tissue recommended by the ICRU (ICRU 39, 1985; ICRU 56, 1993) for weakly penetrating radiation is 0.070 mm for the skin and 3 mm for the eyes. In most cases, protection of the skin ensures sufficient protection also for the eyes and testes.

In a mixed beta-photon field, if the photon energies are sufficiently low (<20 keV approximately), the photon component is also weakly penetrating and therefore the same quantities as for the beta component, $H(0.07, \Omega)$ and $H_p(0.07)$, will suffice. If photons with higher energies are present, then the ambient dose equivalent $H^*(10)$ and the personal dose equivalent $H_p(10)$ should also be measured.

The instruments for area monitoring of weakly penetrating beta-photon fields should ideally consist of a very thin, planar, tissue equivalent detector, placed under a tissue equivalent absorber 0.07 mm thick. The detector should be backed by a plastic block some cm thick to ensure both the adequate radiation backscattering and also the effective attenuation of radiations incident from the back hemisphere, just to obtain an angle dependence close to that of the directional dose equivalent $H(0.07, \Omega)$. If high energy photons are present, then the situation is more complex because of the different requirements for the angle dependence and due to the different absorption and backscattering conditions required for the measurement of $H(0.07, \Omega)$ and $H^*(10)$. In practical devices intended for the measurement of penetrating and non-penetrating radiations, two different interchangeable absorbers, usually of 0.07 and 10mm thickness, can be placed surrounding the detector. In order to ensure adequate backscattering conditions for penetrating radiation also, the detector should be backed by a thicker block of material than for non-penetrating radiation. This block is sometimes made of

high Z materials producing the required effect with a lower thickness. Obviously the ICRU sphere itself is too bulky for a portable area monitor.

Individual dosimeters for beta-photon fields usually have different detectors for the penetrating and non-penetrating field components; each placed under the required thickness of a tissue equivalent absorber. Thin detectors are in principle required for non-penetrating radiations, either photons or betas. Individual dosimeters worn on the torso are intended for the measurement of fields with a reasonable uniformity over the whole human body. Beta radiation fields and also low energy photons are frequently rather non-uniform, so that it is advisable to measure the personal dose equivalent in the region presumably most exposed, usually the worker hands or fingers. In those circumstances where the dose measured on the torso cannot be considered representative, special extremity dosimeters for the measurement of $H_p(0.07)$, to be worn on the wrist or in the fingers, are required.

b) Instruments for area surveys

Ionometry

Survey instruments for area dosimetry can employ gas ionisation detectors such as ionisation chambers, proportional counters or Geiger-Müller detectors. Scintillation and semiconductor detectors can also be employed. At present the instruments based on ionisation chambers are perhaps the most widely employed although semiconductor detectors, presenting some attractive features, have a good potential for the development of competitive area monitors.

Ionisation chambers designed for beta-gamma measurements usually have a removable cap about 0.5 to 1 g/cm^2 thick to allow separation of penetrating and non-penetrating radiation, presenting a thin window of about $7\text{mg}/\text{cm}^2$ with the cap removed so that $H(0.07)$ can be measured (see upper figure 1b). The difference of the readings in the two situations, $H(0.07)-H^*(10)$, is estimated to be the contribution of the weakly penetrating field component. The side walls of a chamber for mixed field dosimetry should be thick enough to stop betas and low energy photons, thus detecting only over the front hemisphere, as required by the $H(0.07, \Omega)$ definition. With the cap in position, the chamber can measure $H^*(d)$, assuring electronic equilibrium for high-energy photons coming from all directions. Values of d can range between 3 and 10 mm, depending on the organ at whose approximate depth the equivalent dose is measured.

Some of the beta-gamma dosimeters using ionisation chambers, particularly older designs, have limitations for the accurate determination of beta and low energy photons, as they do not provide adequate backscattering conditions for low penetrating radiations. This in part is caused by the rather big and deep collection volumes employed, convenient for the measurement of low doses of strongly penetrating radiations, but severely underestimating low energy beta doses. A review of the characteristics of ionisation chambers employed for the measurement of low doses in mixed beta-gamma fields can be seen in the article of Swinth and Sisk (Swinth and Sisk, 1991) and in the ICRU report n. 56 (ICRU 56, 1997).

Figure 1a, reproduced from the paper by Swinth and Sisk, presents the variation of the response for the measurement of H (0.07) of various chamber designs for different beta ray sources. Instruments A, C and I include cylindrical chambers designed considering primarily the requirements of penetrating radiation. The large variation in response with the beta particle maximum energy can be seen. Chamber B is chamber A after decreasing the depth of the sensitive volume to only 1cm (initially it was 4.56 cm), a modification which improves performance for beta dosimetry. Model D, the one showing the better characteristics, is a dual chamber, especially designed for beta-gamma dosimetry, (lower fig. 1b). It includes in concentric geometry one thin chamber for non-penetrating radiation and a thick chamber of higher volume for high-energy photons.

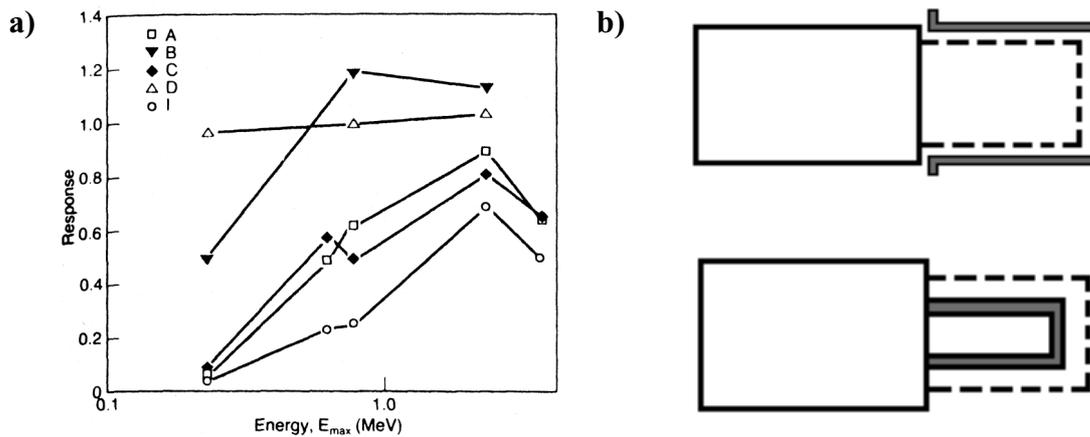


Figure 1. Energy response (H (0,07))of several ionisation chamber survey meters for various beta-ray sources. b) Upper: conventional chamber with a removable cap. Lower: dual chamber in concentric geometry. (Swinth and Sisk, 1991)

In general chambers with a comparable depth and diameter are convenient for penetrating radiation ($H^*(10)$) while thinner chambers with a large diameter, to ensure a convenient detection volume, are required for the correct measurement of H (0.07) caused by non-penetrating radiation. Nevertheless, as explained before for measurements in mixed fields, large volume and deep ionisation chambers are used very often, because of their good sensitivity. In that case geometric correction factors should be applied for the measurement of H (0.07) if the source to detector distance is lower than approximately 20cm (Swinth and Sisk,1991). In these circumstances care should be taken for the field uniformity over the detector dimensions, if not H (0.07) could be underestimated. Cylindrical chambers, 300-500 cc in volume with dimensions around 12 cm depth and 7 cm diameter, and thick walls made from a tissue equivalent plastic, are easy to construct. These detectors permit to measure $H^*(10)$ in the range 10 μ Sv/h to 1Sv/h with a nearly energy independent response from 40 keV to several MeV. Some of these chambers have been modified to improve their measurement characteristics for low energy photons and betas, an example can be seen in (Chabot et al.,1988).

Proportional counters operating with gas amplification present a higher sensitivity than ionisation chambers with reasonably good energy dependence for photons. Nevertheless, even proportional detectors with thin beta windows are not accurate enough for the dosimetry of low energy betas, which is inconvenient for a dosimetric survey instrument in mixed fields (ICRU 56,1997). Geiger-Müller counters and proportional counters are

used mainly for the detection of surface contamination. GM detectors tend to present poor energy and directional responses for dosimetric measurements. The possibility of using GM detectors for beta dosimetry, improving the low energy response by the use of two detectors with windows of different thickness has been recently proposed by Borg and Christensen (Borg and Christensen,1995). From the signals of the two detectors an energy dependent correction factor can be derived, improving the accuracy of beta doses.

Semiconductor diodes can also be operated as direct dosimeters measuring the radiation induced current or charge pulses in different working modes to estimate H (0.07) or H*(10) with very good sensitivity, adequate for very low doses. Some of the most recent developments of semiconductor based dosimeters (Marshall et al., 1994) have been promoted for personal monitoring, and so their readings are expressed in terms of Hp(0.07) and Hp(10), but could also be employed to estimate the area ICRU quantities.

Spectrometry

Scintillators can be employed in two different modes for the dosimetry of photons and beta fields. As a direct dosimeter, calibrating the current or the pulse frequency, produced by the photomultiplier connected to the scintillator, in terms of dose quantities, and using removable caps to separate penetrating and non-penetrating radiations. Organic scintillators are the best choice due to their nearly-tissue equivalence, thus favouring good energy dependence. Alternatively, scintillators can be operated as beta spectrometers, providing the energy distribution of the beta radiation with excellent discrimination of the photon component, whose contribution has to be determined by a different instrument. The spectroscopic mode is convenient when beta doses are important. The equivalent dose can be accurately computed starting from the spectral distribution by the use of the appropriate values for the mass stopping power. In addition to doses, beta spectrometry also provides useful information, such as the maximum beta energy, which can be used to optimise the protection of exposed workers. A description of the use of different types of scintillators can be found in (Swinth and Sisk , 1991) and (ICRU 56,1997).

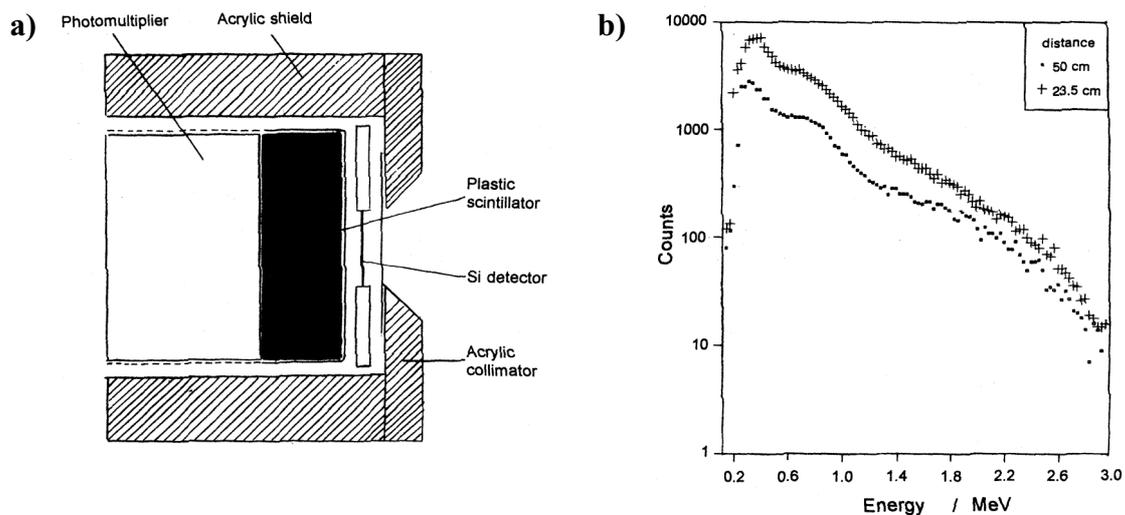


Figure 2. a) Schematic diagram of a telescope with a silicon detector and a plastic scintillator. b) Beta-ray spectra measured with this telescope at two distances from a contaminated component of a reactor fuelling system. (Horowitz, et al. 1994)

Beta spectrometry can also be achieved with semiconductor detectors or by combining both scintillators and semiconductors, assembling thin and thick detectors forming a two-element telescope. However, the superior energy resolution of semiconductors makes them preferable over scintillators for some applications. Descriptions of these systems can be found in references (Horowitz et al., 1994) and (Horowitz et al., 1995). The operational principle of the telescope type spectrometers is based on the different way photons and betas interact with matter. Betas, interacting via coulombian forces, lose energy in a continuous way, while photons lose energy by discrete interactions in specific places. A typical telescope is presented in figure 2a, reproduced from (Horowitz et al., 1994). The thin detector (silicon) is placed at the beam entrance with the thick detector (plastic scintillator) behind. Beta radiation will lose some energy in traversing the thin detector producing a pulse while photons likely not if the detector is thin enough. Operating in coincidence the pulses produced simultaneously in both detectors will only be caused by betas, discriminating in this way the photon contribution. The scintillator detector should be thick enough so as to stop all the betas and measure their energy distribution. Figure 2b shows the beta ray spectra measured at two distances from a burnt fuel element from a nuclear reactor.

b) Individual Dosimetry

Thermoluminescent (TL) detectors are at present the most widely employed for individual monitoring. As it is known, TL materials with good tissue equivalence with an adequate shape and thickness and also excellent sensitivity are available. Modern TL personal dosimeters, for whole body control, can estimate the personal dose equivalent at $d=0.07$ and 10 mm, the recommended depths for non-penetrant and penetrant radiation. Therefore these dosimeters are adequate for personal monitoring in beta-photon fields. Usually, different TL detectors of adequate thickness, and placed behind different absorbers are employed, combining their readings with a mathematical algorithm producing the shallow and the depth doses. For beta-photon fields the use of tissue equivalent TL materials is strongly recommended to avoid the over-response to low energy photons typical of TL materials with higher atomic number than the average of soft tissue. An excessive over-response to photons will complicate the estimation of the beta contribution to $H_p(0.07)$.

Figure 3 represents the energy dependence of several TL dosimeters for the measurement of both $H_p(10)$, figure a) and $H_p(0.07)$ figure b). The figures are reproduced from a recent report published by an EURADOS action group (Bartlett et al., 2000) in which personal dosimetry laboratories from the members states of the European Union (and Switzerland) have participated. Figure 3a shows that for most of the systems the $H_p(10)$ response lies within $\pm 20\%$ for the energy range 10keV to 2.0 MeV, which is satisfactory. However, some systems present higher departures, over or underestimating the dose at the lower energies. Systems based on non-tissue equivalent TL materials are prone to present these departures because of excessive or insufficient correction of their intrinsic higher efficiency for low energy photons. Approximately the same tendency can be seen in figure 3b for $H_p(0.07)$.

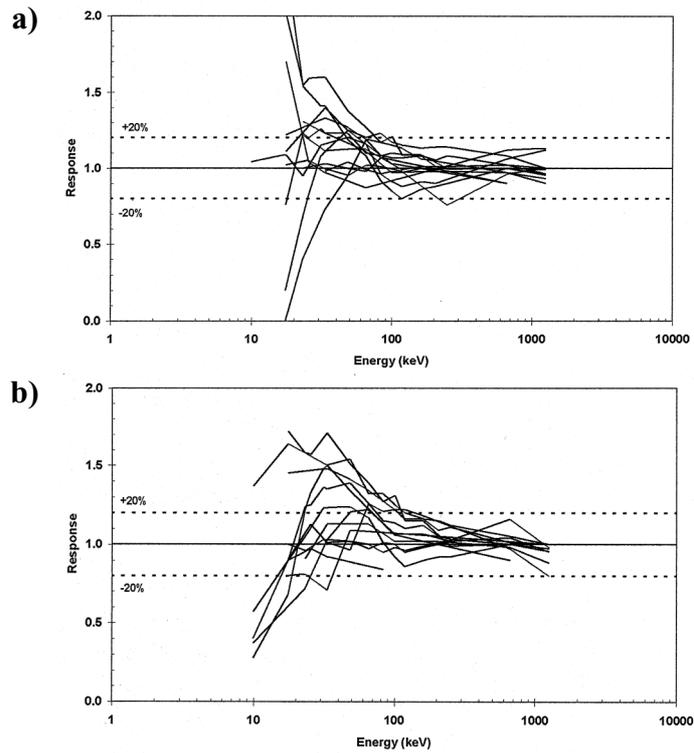


Figure 3. a) Response, $H_p(10)_{\text{measured}}/H_p(10)_{\text{true}}$ as a function of energy for whole body photon TL-dosimeters for normal incidence. b) The same for $H_p(0.07)_{\text{measured}}/H_p(0.07)_{\text{true}}$ (Bartlett et al. 2000).

In figure 4 the energy dependence of whole body TL personal dosimeters for beta rays can be seen. A severe dependence with the average beta energy can be appreciated, with a very low response for the low energy betas from ^{147}Pm (70 keV average energy). This rather poor performance of whole body personal dosimeters for beta radiation is counterbalanced by the few and rare applications in which beta fields may produce significant whole body doses. More typically, beta doses are of localised type affecting only parts of the body, such as extremities: hands or fingers, for which, as stated previously, special dosimeters are required.

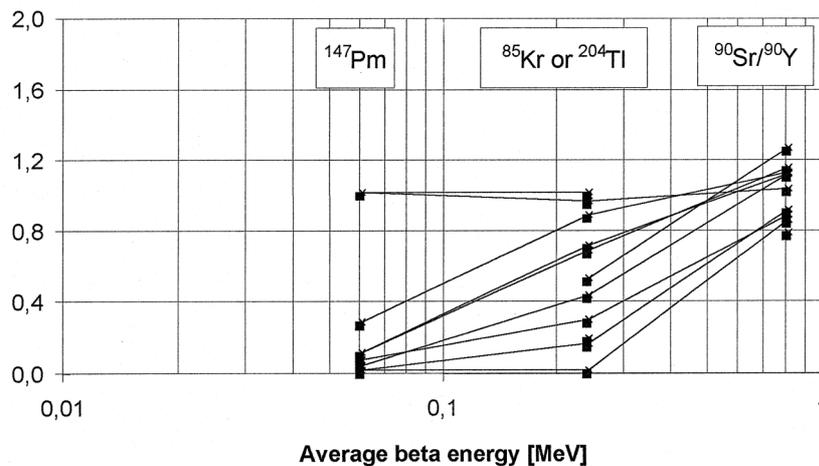


Figure 4. Response, $H_p(0.07)_{\text{measured}}/H_p(0.07)_{\text{true}}$ as a function of average beta energy (MeV) for whole body beta TL-dosimeters at normal incidence. (Bartlett et al. 2000).

Extremity dosimeters employed for skin dose assessment should ideally consist of a thin detector, on the order of 5 mg/cm², behind a layer of tissue equivalent material of the same thickness, and placed in an also thin and tissue equivalent holder. In this way the dosimeter can adequately estimate Hp(0.07). Reviews of the requirements for extremity dosimeters can be found in (Fellinger, 1991) and (Goldfinch, 1993). Descriptions of commercial systems for extremity dosimetry are in (Tawil et al., 1993) and (Yuen et al., 1999). Table 1 presents the energy and the angle dependence for betas of the dosimeter presented in the last reference (Yuen et al., 1999).

Table 1: Beta angular response of an extremity dosimeter.
(Yuen et al., 1999)

	Angle of incidence (deg)				Average
	0	20	40	60	
⁹⁰ Sr- ⁹⁰ Y					
Measured dose (mGy)	13.03±0.61	12.96±0.25	10.77±0.75	7.57±0.61	
True delivered dose (mGy)	10.01	10.21	11.01	11.91	
Response	1.30±0.06	1.27±0.02	0.98±0.07	0.64±0.05	1.04±0.03
²⁰⁴ Tl					
Measured dose (mGy)	4.80±0.21	4.93±0.26	3.33±0.18	2.74±0.24	
True delivered dose (mGy)	5.00	5.00	5.00	5.00	
Response	0.96±0.04	1.02±0.05	0.72±0.04	0.75±0.07	0.86±0.03
¹⁴⁷ Pm					
Measured dose (mGy)	2.30±0.19	1.87±0.11	1.29±0.11		
True delivered dose (mGy)	2.14	2.13	2.13		
Response	1.07±0.09	0.92±0.05	0.86±0.07		0.95±0.04

The very thin detectors required for TL extremity dosimetry have intrinsically low sensitivity because of their low mass, and thus relatively high practical dose thresholds. This has been considered a difficulty, despite the not very low equivalent dose limit recommended for protection of the skin, 500 mSv/year. The new TL material LiF:Mg,Cu,P, a high sensitivity, nearly tissue equivalent material, has shown to be adequate for extremity dosimetry. It has allowed to extend the workable range to substantially lower doses, improving the reproducibility of the measurements as well. The improvement mainly comes from the more intense signals always obtained with LiF:Mg,Cu,P than with more conventional materials, e.g. the traditional LiF:Mg,Ti (Christensen, 1993).

Physically thin detectors are really the most convenient for the determination of skin doses, but there are other approaches. Thick detectors have been occasionally employed. Several detectors of normal thickness (10 to 20 mg/cm²), are placed under different absorbers so that the shallow dose can be obtained by extrapolation. Alternatively, surface only sensitive detectors can be prepared loading a thick TL detector with carbon in order to reduce the light transmission to only the surface of the detector, i.e. the region where non-penetrating radiation deposits energy. Thick detectors are more robust and stable and are simpler to handle. Nevertheless thin detectors are preferable as they present inherently better angle and energy response for the measurement of non-penetrating radiation, particularly beta rays.

DOSIMETRY OF NEUTRON-GAMMA FIELDS

a) Relevant Quantities.

In neutron-gamma fields only the quantities for penetrant radiation need to be usually determined. Accompanying photons in most cases have sufficiently high energy and neutrons are always penetrating radiation. Therefore, the quantities of interest are the ambient dose equivalent $H^*(10)$ for area monitoring and $H_p(10)$ for individual monitoring. Needless to say, neutrons also impart energy at shallow depths in tissue, but usually the values of $H^*(10)$ and $H_p(10)$ are higher than $H(0.07)$ and $H_p(0.07)$, so that an effective protection is attained by controlling the quantities at 10mm.

As with other radiations, neutron or gamma-neutron area dosimeters employed for the measurement of $H^*(10)$ should ideally have a dose equivalent response independent of the angle of incidence of radiation. The energy dependence of the dosimeter should mimic that of $H^*(10)$ in terms of the physical quantity directly measured by the detector, absorbed dose or particle fluence. Figure 5 presents the energy dependence of the conversion coefficients from fluence to absorbed dose (vertical axis on the right), and to $H^*(10)$ and $H_p(10)$ in ICRU tissue (vertical axis on the left), after ICRP 60. Personal dosimeters should have an angle and energy dependence as close as possible to the dependence of $H_p(10)$ respect to the physical quantity measured by the detector.

A distinctive characteristic of neutron fields is the very different weighting or quality factors defined for neutrons depending on their energy. If gamma radiation is present in the external field, another component with unity quality factor should be added. The multiplicity of quality factors is a severe complication for the determination of the dose equivalents, ambient or personal, as the sole measurement of the absorbed dose is not sufficient, as it was for example in beta-photon fields with $Q = 1$ for all components.

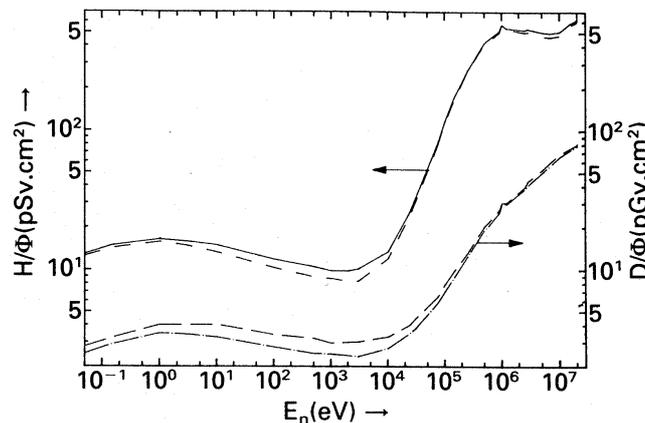


Figure 5. Fluence-to-dose equivalent conversion functions (upper curves, left ordinate for $H_{si}(10)$ and $H^*(10)$ (dashed line). Lower curves, right ordinate: conversion functions for the absorbed dose $D_{si}(10)$ (dashed line) and $D^*(10)$ (—·—). (Hollnagel, 1994)

In addition, neutrons can interact with matter in many different ways, including nuclear reactions and scattering processes in tissue producing protons of different energy, and also high energy photons, each one giving its specific contribution to the dose equivalent. As a result, the general requirement of tissue equivalence for a material to conform a suitable dosimeter in the case of neutrons has to be taken literally.

Equivalence in principle should be in the exact atomic composition of the material and not only equivalence in the effective atomic number, as in the case of photons. This is a rather restrictive requirement, nearly impossible to fulfil in practice.

Except in those circumstances in which microdosimetric techniques (TEPCs) can be applied, identifying the contribution of low and high LET components with a single detector, the dosimetry of mixed neutron-photon fields requires the independent determination of the dose equivalents of neutrons and photons, employing different dosimeters. The photon and neutron dose equivalents are added to determine the total dose equivalent. A problem arises with photons generated by neutrons in the body tissues. In principle they are included in the fluence to dose equivalent conversion coefficients, and thus their contribution accounted without need of specific measurements. However, it is not possible to avoid doubly counting those photons which leave the body and reach the external photon detector, although the error caused is usually not important.

The dosimetry of photons is well developed and free of important problems and it is possible to find materials insensitive to neutrons. The dosimetry of neutrons is considerably more difficult, and discrimination against photons is frequently a problem in mixed fields.

b) Instruments for area survey

The only active instrument providing simultaneously information on the photon and neutron dose equivalents is the Tissue Equivalent Proportional Counter (TEPC). As noted before, this detector operates along the principles of microdosimetry, providing the distribution of the absorbed dose in terms of the imparted lineal energy, and thus permitting to separate the contribution of the weakly ionising and densely ionising radiation. If this type of instrument is not employed, then separate measurements of the neutron and photon components have to be performed. Active detectors for photons using ionisation chambers, proportional counters or Geiger-Müller detectors are well known and have reached a high performance and reliability. Descriptions of these instruments can be found in basic books of radiation dosimetry as *Fundamentals of Radiation Dosimetry* (Greening, 1981) or *Introduction to Radiological Physics and Radiation Dosimetry* (Attix, 1986), for example. Therefore, in addition to TEPCs, only neutron related instrumentation is treated here.

Tissue Equivalent Proportional Counters

A recent and complete description of TEPCs including operational principles, design and their use can be found in (Schmitz et al., 1995), a special volume of *Radiation Protection Dosimetry* reporting the activities of the EURADOS working group on microdosimetry. A TEPC consists of a cylindrical or spherical cavity with walls made of a tissue equivalent plastic filled with an also tissue equivalent gas. The gas pressure is of the order of a few hundred Pascals, so that with a cavity diameter of a few centimetres, simulation of a few micrometers of tissue is obtained. To define the proportional counting region, a structure with thin electrical wires is employed, with a central collecting wire held at a positive potential, typically of a few hundred volts. Figure 6, taken from (Kliuga et al., 1995), represents a cross section of a spherical proportional counter. An auxiliary electrode in helical form placed around the collecting

wire is employed in this design for field shaping. The detector is 2.54 cm in diameter. Although spherical detectors are those presenting the better response isotropy, cylindrical detectors are also employed because they are more robust and perhaps better adapted to field work. They are also easier to construct.

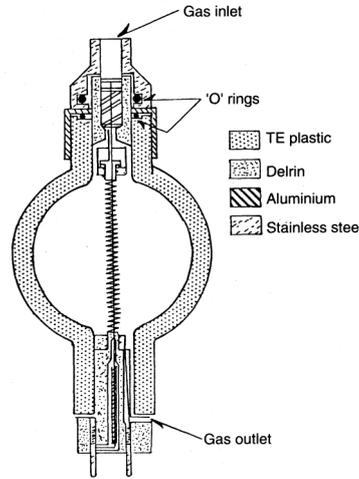


Figure 6: A typical 2,54 cm spherical proportional counter with helix, (Rossi counter).

Instead of measuring the average current as in an ionisation chamber or counting the pulses as in a proportional counter employed in a conventional mode, microdosimeters detect and classify by amplitude the individual charge pulses produced in the detector by individual particles. In this way, the distribution of the ionisation events in terms of the energy deposited is obtained. The deposited energy can conveniently be expressed in terms of the lineal energy, y , in $\text{keV}/\mu\text{m}$ assuming a mean cord length inside the cavity. This distribution, $f(y)$, is represented dividing the whole range of lineal energy in channels or intervals of equal width and representing the number of events (counts) recorded for every interval (see figure 7).

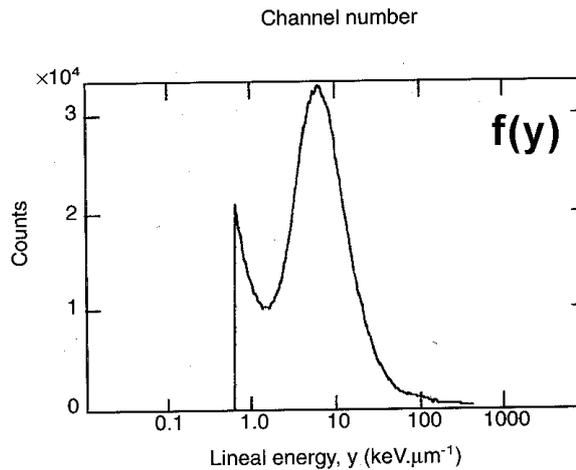


Figure 7. Dose Equivalent distribution in y , $f(y)$.

A measure of the total energy absorbed from ionisation events of size y is given by the product of the number of events, c , and their size, y : $c \cdot y$. Representing the product $c \cdot y$ for each y value, produces the so called absorbed dose distribution in y , D_y :

$$D_y = y \cdot f(y)$$

For practical purposes, the lineal energy, y , can be considered to approximate sufficiently well the lineal energy transfer, L , and then the known Q versus L , (or y), relationship, can be employed to convert the absorbed distribution into a dose equivalent distribution:

$$H_y = Q(y) \cdot D_y.$$

The total dose equivalent obtained by integration over the whole range of y values:

$$H = \int Q(y) \cdot D_y \cdot dy.$$

This process is very well illustrated in figure 8 obtained from (Waker, 1995). In the figure, the conversion of the microdosimetric dose distribution, D_y in μGy , into a dose equivalent distribution, H_y in μSv , using the $Q(y)$ function defined in ICRP 26 (ICRP 26, 1977) can be seen. The higher quality factor at higher y values determines the appreciable enhancement of the contribution to the dose equivalent of the high LET component of the radiation field in detriment of the low LET component.

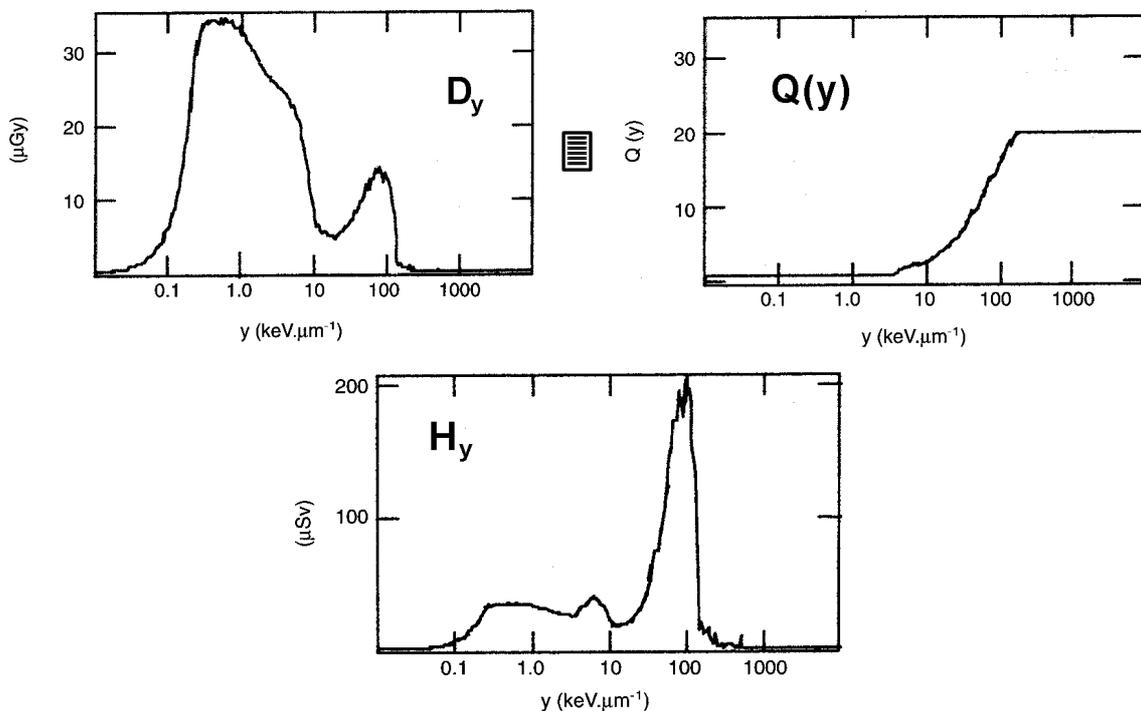


Figure 8. Conversion of a microdosimetric dose distribution into a dose equivalent distribution using a quality factor-lineal energy function equivalent to the ICRP26 quality factor-LET relationship. (Waker, 1995).

TEPCs as every dosimetric method have limitations. They tend to be rather complex instruments requiring multichannel analysers to store the pulse amplitude distribution. The construction of the detector itself is not simple. Nevertheless, developments in microelectronics and machinery have permitted to construct portable instruments for radioprotection field work. Problems arise because neither the walls nor the cavity gas are perfectly equivalent to ICRU tissue, causing departures from the desired energy dependence. The spectrometric capability of practical TEPCs is limited because of variations in the real range of the particles in the active volume of the detector and this limitation can be appreciated in the rather broad distributions experimentally found even with monoenergetic neutron fields. At intermediate neutron energies, below 100 keV, TEPCs tend to underestimate the dose equivalent, mainly because the range of the recoil protons is appreciably shorter than the cavity diameter. In addition, TEPCs tend to be somewhat inefficient detectors, useful only for relatively high doses. At the same time, dose rates cannot be very high in order to avoid pulse pile-up distorting the measured spectrum. TEPCs employed in the so-called variance mode can avoid that dose rate limitation (Lindborg et al.,1988).

All these facts do not undermine the exceptional possibilities of TEPCs for the dosimetry of complex n- γ fields, such as those found in nuclear power plants and civil aviation for example, activities in which TEPCs are progressively being considered as the reference instrument.

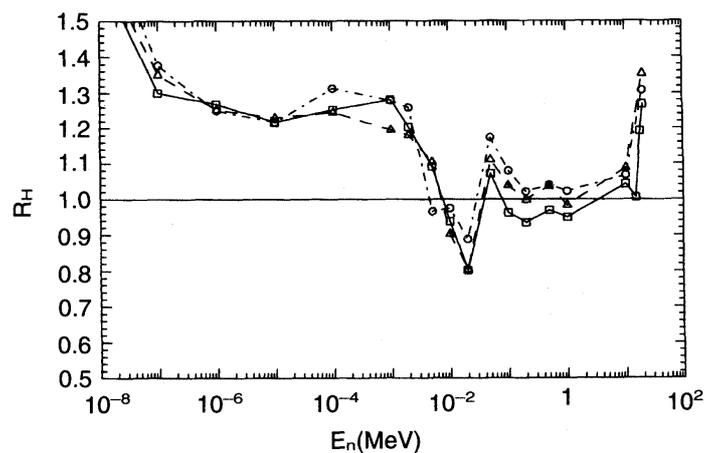


Figure 9: Dose equivalent response, R_H , of a TEPC in a $15 \times 30 \times 30 \text{ cm}^3$ Perplex slab phantom for different angles of incidence as a function of the energy of the incident neutrons, E_n . (Schuhmacher, 1995).

The ability of a small TEPC to measure $H^*(10)$ and $H_p(10)$ has been tested when exposed to neutron-gamma fields using different phantoms (Schuhmacher,1995). Dose equivalent seems to be correctly measured $\pm 25\%$ for photons and neutrons coming from the front hemisphere of the phantom. Figure 9 represent the dose equivalent response versus neutron energy in a slab phantom. Several intercomparisons of the TEPCs performance in different n- γ fields have demonstrated a systematic and progressive under-response for neutron energies $< 5\text{MeV}$, when different TEPCs were exposed free in air (without a phantom) to monoenergetic neutron beams (Alberts et al.,1989), (Menzel et al.,1989).

Recombination chambers

Recombination ionization chambers are interesting instruments based on the dependence of the initial recombination of pairs of ions on the ionisation density. The ionisation density is related to the linear energy transfer of the field components and thus, the recombination characteristics can provide information on the quality of the radiation detected. Specially designed plane-parallel, high-pressure ionisation chambers filled with a tissue equivalent gas have been employed (Golnik,1997) in the recombination mode for the estimation of $H^*(10)$ in mixed fields.

The determination of the recombination characteristics of a mixed field requires the operation of the chamber at several collecting voltages and the determination of the dose equivalent is obtained by numerical fitting of the saturation curve. Therefore, the use of recombination chambers is not simple, requiring long measurement times and expertise in numerical data analysis. The best studied recombination chamber is the REM-2 chamber (Golnik,1997) which consists of 25 parallel-plate tissue equivalent circular electrodes, 25 mm in diameter, separated 7mm. The total weight of the chamber is 6.5 kg. The chamber is filled with a mixture of methane and nitrogen at a pressure of 1 MPa. The collection voltage is varied sequentially between 5 and 1000 V. The operation of the chamber and the associated electrometer is computer controlled. The neutron energy response reported for the REM-2 is reasonably (and comparatively) good, -20 % to +55% within the range 0.07 to 20 MeV.

Recombination chambers do not provide the LET spectra of a mixed field as the TEPCs do, but permit to estimate a recombination index close to the average quality factor, from which $H^*(10)$ can be estimated. Recombination chambers are promising instruments for the area monitoring of mixed fields.

Moderator based instruments

Moderator based instruments, also known as rem-counters, are the most frequently used instruments for neutron area surveys. They consist of a central thermal neutron detector, surrounded by a moderator made of a hydrogenous material, usually polyethylene. The dimensions and shape of the moderator are chosen trying to optimise the energy and angle response, but with the constraint of the weight, which cannot be excessive for a hand-held instrument.

This kind of instruments pertains to the range of instruments not measuring any physical quantity, such as fluence or absorbed dose, but presenting an energy response approaching that of the fluence to ambient dose conversion coefficient, $h_{\phi}(E)$ (see figure 6), in a certain energy range. In these conditions, the readings of the instruments are proportional to the dose equivalent, $H^*(10)$, in that energy range.

Spherical or cylindrical moderators are the usual geometries employed in commercially available instruments, following the initial designs of Andersson-Braun (Andersson and Braun,1964) and Leake (Leake,1968). The Andersson-Braun instrument consists of a right circular, cylindrical polyethylene moderator, 47 cm long and 20 diameter, with the outer end rounded to improve energy and angular response. It also incorporates a perforated thin layer of boron-loaded plastic placed surrounding the detector. The

detector is a BF_3 proportional counter. The instrument weights approximately 10 kg. The Leake survey instrument consists of a spherical polyethylene moderator of 20.8 cm diameter, allocating in the central position a ^3He proportional counter surrounded by a perforated cadmium layer. The total weight is 6.6 kg. The use of the inner absorber of cadmium or cadmium loaded plastic, transparent for thermal neutrons but absorbing fast neutrons, permits to obtain a good energy response with a smaller hydrogenous moderator, i.e. reducing weight.

There have been more modern designs, following the former ideas of Leake and Andersson-Braun, and also new approaches, as the Dineutron (Mourges et al., 1985). This is an instrument consisting of two moderating spheres of different diameter (6.2 and 10.7 cm), each enclosing a ^3He proportional counter. A microprocessor incorporated to the device combines the signals from the two detectors with different moderation, using an algorithm producing a single estimate of the dose equivalent. This instrument is lighter than others, only 3.2 kg. Another example is the SWENDI neutron detector, with improved response above 2 MeV, with a ^3He detector. Weight is 13 kg.

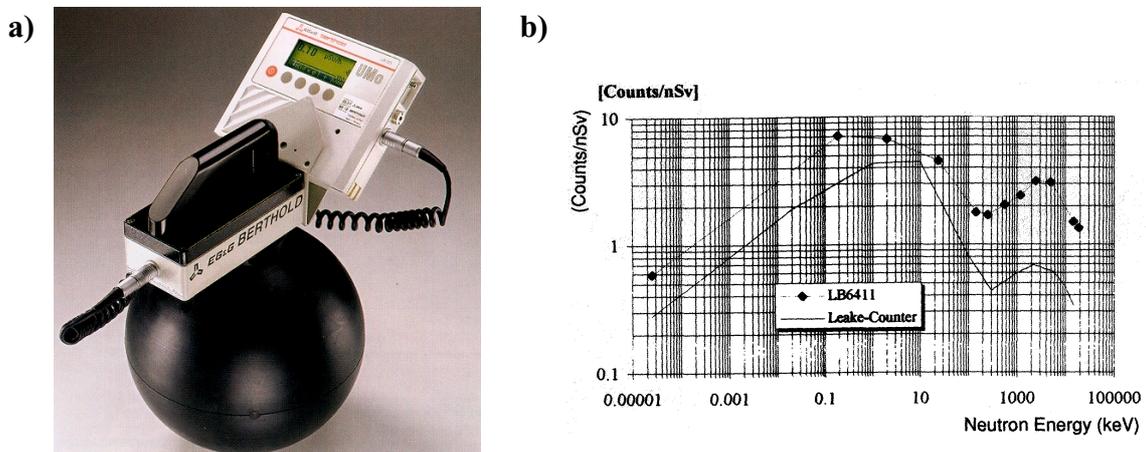


Figure 10. a) LB 6411 neutron probe, a portable measuring system connected to the basic unit LB 1230 of the Universal Radiation Protection Monitor LB123 UMo. b) Comparison of sensitivity and response of different detector types.

Figure 10 shows a recent commercial instrument using also a spherical moderator, the LB 6411, (Burgkhardt et al., 1997). The energy response of this instrument compared with the response of the Leake counter. All of the moderator type neutron dosimeters have energy responses with similar general features, presenting a clear over-response at intermediate neutron energies, in the keV range, and a comparable under-response in the thermal energy region.

Bubble Detectors

Bubble detectors are often considered an exotic dosimetric method. In reality they are based on the same physical principles as the famous bubble chambers employed for a rather long time as particle detectors in High Energy Physics: boiling in a superheated liquid can be initiated by charged particles interacting with the medium.

Neutron bubble dosimeters are devices in which small droplets of a superheated liquid are suspended or dispersed in a viscous medium, all this enclosed in a transparent tube.

The superheated state is intrinsically unstable but in the absence of nucleation sites and if the temperature and pressure at which the superheated drops were formed are not altered in excess, the superheated state can remain, metastably, but for sufficient time in the liquid state. When such a system is exposed to neutrons, the energy carried by the secondary charged particles can be transferred to a droplet inducing vaporisation, forming finally macroscopic bubbles that can be seen simply by eye. For an effective vaporisation, it is necessary that a minimum energy be transferred, so that the initial bubble can reach a critical dimension. Otherwise, the bubble instead of growing, will recondense back to the liquid state. A description of the characteristics of bubble detectors can be found in the original work of Apfel (Apfel, 1979) and in more recent articles such as (d Errico et al.,1995) and (d Errico,1999).

Current bubble detectors are completely insensitive to photons. This is a good characteristic for applications in which the photon contribution is high but where the neutron doses need to be determined. In principle, the number of bubbles formed is proportional to the neutron fluence, and with a reasonably constant energy dependence and good isotropy. Bubble detectors are free of background and so the minimum dose is that corresponding to one single bubble, which seems to be around $0.5 \mu\text{Sv}$, a rather low dose. The weak points of bubble detectors are related to their problematic temperature dependence, the poor reproducibility and uniformity of the operational characteristics and the dependence of the response with storage time.

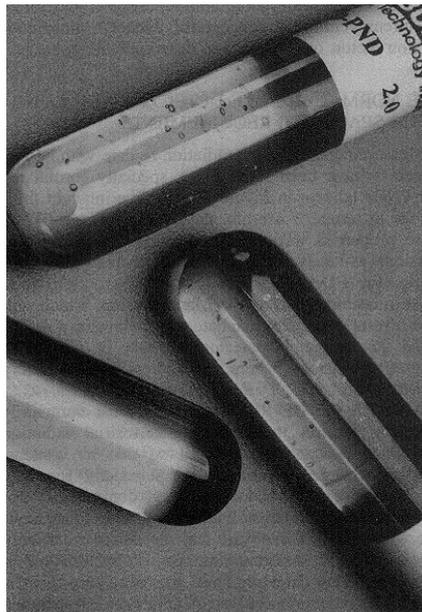


Figure 11: Bubble Detectors for neutron monitoring

Two families of bubble detectors have reached commercial stage, the so-called Bubble Damage Detectors (BDD) and the Superheated Drop Detectors (SDD). BDDs employ a very viscous medium forcing the bubbles to remain where they are produced, allowing to count them even by eye. In some models temperature compensation is applied, so the variation in response is limited to $\pm 20\%$ in $15-40^\circ\text{C}$ range. BDDs can be resetted by re-pressurisation. The dosimeters have a piston cap for this purpose. The re-pressurisation process can be repeated a limited number of cycles (10-20). SSDs employ a less viscous gel permitting to move the bubbles to the surface; the gas accumulates on top of the dosimeter displacing a piston with a cursor indicating the dose. More refined systems

count the bubbles employing the noise produced when they are formed, discriminating external noise by anticoincidence methods. The dose and the dose rate are calculated by a computer and displayed in real time, hence operating as an active dosimeter. Figure 11 presents a bubble detector of the type with re-pressurisation cap. Bubble detectors can be employed as personal dosimeters as well.

c) Dosimeters for Individual Monitoring

Individual dosimeters for mixed neutron-gamma fields usually employ different detectors for each radiation component, often enclosed in the same badge. Although the most frequently used detectors, are sensible to photons and neutrons, their interaction processes and the exposure conditions are different and consequently the use of separate detectors is preferable. Ideally, the readings of the two types of detectors should correspond independently to the neutron and the photon doses. However, this is not always possible and a common situation is that one of the detectors is sensible to n and γ while the other only to γ . In this situation a less accurate difference method must be applied to separate the neutron and photon primary signals, from which the dose equivalent can be derived.

The current techniques employed for photon dosimetry enables accurate measurement of very low doses without major difficulty. Neutron individual dosimetry is more complicated and in fact nowadays a sufficiently reliable method, covering the whole range of the neutron energies, does not exist. While the dosimetry of thermal and epithermal neutrons is feasible, the dosimetry of fast neutrons remains problematic. This situation is mainly due to the different processes induced by neutrons in tissue in these energy intervals and the difficulty of finding adequate tissue surrogates for dosimeters.

Capture reactions of neutrons by some nuclei, ${}^6\text{Li}$, ${}^{10}\text{B}$, producing charged particles, present the same $1/v$ cross section dependence (v , neutron velocity) that with nitrogen or hydrogen, constituents of the ICRU tissue. Therefore, these nuclei incorporated to the dosimeter will help to obtain an energy dependence similar to the dependence of the personal dose equivalent in the energy range where capture reactions are effective, from thermal to a few keV. Beyond 10 keV the cross section for capture reactions is very low and the main effect is the elastic scattering of neutrons producing recoil protons and heavy nuclei. Hydrogen is very efficient in producing recoil protons, but is not normally a constituent of the common dosimeter materials. An approach to improve the dosimetry of fast neutrons is to cover the dosimeter material, insensitive by itself, with a hydrogen-rich material, producing protons in response to the fast neutrons, to which the dosimeter is sensible.

Neutron dosimetry is one of the sectors where the need for active personal dosimeters is felt more intensely. This sentiment is due to the possibility of receiving appreciable doses in short periods, characteristic of some applications, such as nuclear fuel reprocessing (Tokaimura) among others. Active personal detectors would permit to identify the activities and circumstances producing such doses, helping to apply ALARA corrective actions. While TLDs are passive dosimeters, semiconductor based electronic dosimeters can be designed and operated in the active mode. This is the main reason for the intense effort dedicated to study of different types of semiconductor based devices.

Thermoluminescence Dosimetry

TL materials well characterised for dosimetry, such as LiF:Mg,Ti and Li₂B₄O₇:Mn, can be adapted for neutron dosimetry by altering the natural isotope concentration of Li or B, enriching the content of ⁶Li or ¹⁰B isotopes, which both have a high cross section for neutron capture reactions: ⁶Li(n,α)³H and ¹⁰B(n,α)⁷Li. As explained before, the cross section of these reactions presents the same energy dependence (1/v) as the capture by N and H components of the ICRU tissue. However, the cross section, decreasing with neutron energy, presents acceptable values for the thermal region but decreases to vanishingly low values for a few keV. Therefore, ⁶LiF:Mg,Ti or ⁶Li₂¹⁰B₄O₇:Mn in principle can be employed for the dosimetry of thermal and intermediate energies with a reasonably good energy response for the measurement of Hp(10); however, the efficiency of these materials for fast neutrons is very low: capture reactions cannot ensure an efficient interaction mechanism and the elements forming these TL dosimeters do not provide comparable kerma factors to those of the elements of ICRU tissue for fast neutrons. In that region, the more efficient energy transfer mechanism in ICRU tissue is the elastic scattering of neutrons by hydrogen producing recoil protons. As stated before, hydrogen is not a constituent of the mentioned TL dosimeters.

TLDs are in general very efficient gamma dosimeters. Thus, in mixed fields ⁶LiF:Mg,Ti and ⁶Li₂¹⁰B₄O₇:Mn will detect both neutrons and gammas. Altering the isotope content is an efficient way to achieve neutron discrimination. ⁷Li or ¹¹B isotopes have substantially lower capture cross section and so ⁷LiF:Mg,Ti or ⁷Li₂¹¹B₄O₇:Mn, are only sensitive to photons. Pairing both types of isotope enriched detectors, ⁶LiF:Mg,Ti and ⁷LiF:Mg,Ti, or ⁶Li₂¹⁰B₄O₇:Mn and ⁷Li₂¹¹B₄O₇:Mn, will permit to detect the gamma component with similar sensitivity by each element of the pair, while the dosimeters with ⁶Li or ¹⁰B will additionally detect neutrons. Applying a difference method enables the separation of each contribution and, through the application of the corresponding calibration factors, estimate the total personal dose equivalent.

A different approach to separate neutron and gamma doses is based on the different LET sensitivity of different glow peaks in some TL materials, notably LiF:Mg,Ti. (Pradhan et al., 1985). The determination of the intensity or the area under low and high temperature peaks permits to make an estimation of the relative contribution of the two types of radiations. The application of computerised glow curve analysis has permitted to improve the two-peak method (Youssin and Horowitz, 1998). However, this method is not free of difficulties, which generally arise from the usually very different sensitivity of the peaks to be compared and their different dose dependence.

TLDs are frequently employed for neutron or neutron-photon dosimetry as albedo dosimeters. Albedo techniques were developed to try to overcome the difficulty of TLDs (and other 1/v detectors) to detect fast neutrons. Neutrons incident on the human body are scattered and moderated by the nuclei of the body and as a consequence of these interactions part of them leave the front surface of the body. The spectra of the outgoing neutrons, called albedo neutrons, is shifted towards lower energies with respect to the incident neutrons as a result of the body moderation processes, converting fast into thermal neutrons and permitting their measurement by 1/v detectors. Albedo dosimetry measures the neutrons emitted by the body instead of (or in addition to) the incident neutrons. This can be achieved placing the TL detectors behind a thermal neutron absorber, thus eliminating the incident thermal neutrons, but close to the body

surface to detect the reflected neutrons. A description of the features of albedo dosimeters can be found in (Piech and Burgkhardt, 1985). Different dosimeter designs using the albedo concept are described in this reference.

A problem of albedo dosimetry is that the fraction of backscattered neutrons depends appreciably on the energy of the incident neutrons, changing from around 0.8 for thermal to 0,2 for 1 MeV. This is an important shortcoming, determining a really poor energy response. Figure 12 reproduced from (Bartlett et al., 1999) represents the energy dependence in terms of $H_p(10)$ of an albedo dosimeter. The good constancy of the response up to 10 keV can be appreciated. Nevertheless, for higher neutron energies the response is very poor, being typically around 100 times lower at 1 MeV than in the thermal region, and around a factor of 15 between 0.1 MeV and 1 MeV.

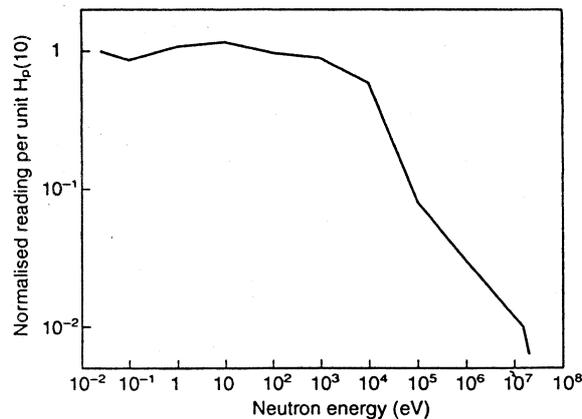


Figure 12: Albedo dosimeter response characteristics. Normalised dosimeter reading per $H_p(10)$. (Bartlett et al., 1999).

Attempts have been made to palliate the poor energy response inherent to albedo dosimeters, by including additional detectors so that incident thermal neutrons were also measured. The ratio of the albedo to the directly exposed detectors provides information on the spectrum of the measured neutrons, deriving correction factors (Douglas and Marshall, 1978). Information on the neutron spectrum is essential to improve the accuracy of the equivalent dose measured by albedo techniques by using appropriate sensitivity factors. The information on the neutron field characteristics provided by extra TL detectors is however, limited and thus the required spectral information should be obtained by independent means, such as genuine spectrometric methods. Frequently, the neutron spectrum to be monitored is constant or nearly constant and thus, a field calibration providing measured or calculated correction factors is a practical and convenient procedure. It seems that albedo dosimeters are reasonably well suited for the measurement of stray fields with degraded spectra (Piech, 1988).

The sensitivity of TL materials for photons is very good, and nearly tissue equivalent materials like LiF:Mg,Ti have detection limits of the order of 20-50 μSv . LiF:Mg,Cu,P , also with good tissue equivalence, is 30 to 40 times more sensitive to photons than LiF:Mg,Ti and can thus measure doses even lower than 1 μSv (Delgado, 1996). On the contrary, the sensitivity to neutrons of the ^6Li enriched varieties of these TL materials is appreciably lower than for photons, with practical detection limits of around 100 μSv in albedo dosimeters (Piech, 1988). $^6\text{LiF:Mg,Cu,P}$ has a lower sensitivity to neutrons than $^6\text{LiF:Mg,Ti}$. The dose dependence of these TL materials is linear for doses well above

the usual radioprotection levels, both for photons and neutrons. The angle dependence for the measurement of Hp(10) obviously depends on the particular design of the dosimeter badge, but it is not normally a problem. A recent paper on this characteristic (Tanner, 1997) reports satisfactory results for an albedo dosimeter, comparing the experimentally determined values for Hp(10), exposing to moderated and unmoderated ^{252}Cf neutron sources, with the calculated values using an MCNP Monte Carlo code.

Semiconductor Detectors

Dosimeters based on semiconductor radiation detectors are in a privileged situation due to the increasing need for active personal dosimeters. Semiconductors behave like solid state ionisation chambers with the advantage of a lower energy for the creation of an electron-hole pair in Si (around 3 eV), than for creating a pair of ions in air (around 33 eV). The higher density, providing higher detection efficiency than air in ion chambers, and a straightforward interconnection with the electronics circuits for control and measurement, are all advantages which make semiconductors very attractive for the construction of small and sensitive personal dosimeters. Nevertheless, much in the same way as with other systems, as TLDs for example, while there are good photon semiconductor based dosimeters, problems appear for the development of entirely satisfactory neutron personal dosimeters.

The usual approach followed for the design of neutron dosimeters with semiconductor detectors is to try to produce a response vs. neutron energy as close as possible to the dependence of the fluence to Hp(10) conversion coefficient with the energy. This dependence is shown in figure 5. To achieve this behaviour, the semiconductor poor intrinsic response is modified by the use of neutron converters. Converters are materials with an atomic composition favouring neutron interactions, nuclear reactions or scattering processes, producing charged particles, that can be detected with high efficiency by semiconductors. As with TLDs, for the low and intermediate energy regions, up to about 10 keV, converters rich in ^6Li or ^{10}B , are employed, to take advantage of the relatively high cross section and the convenient energy variation ($1/v$), of reactions as $^6\text{Li}(n,\alpha)^3\text{H}$ and $^{10}\text{B}(n,\alpha)^7\text{Li}$. For the detection of fast neutrons the converters (called sometimes radiators) are made of a hydrogen rich material in order to produce recoil protons caused by neutron elastic scattering.

Many different designs with variations around these basic ideas have been developed. A description of them can be found in recently published review articles (Alberts, 1999) (Bartlett et al., 1999). Different approaches have been employed with a single diode or two diodes for improving photon discrimination and/or performance at high and low neutron energies. Other recently proposed systems employ several diodes under different converters, or converters together with absorbers and moderators, trying to achieve a better response for the intermediate energy region (Luszk-Bhadra et al., 1999). Figure 13, reproduced from a recent article (Nakamura et al., 1999), presents a cross sectional view of a two detector system. The slow neutrons detector is in contact with a natural boron radiator, to detect α and Li ions in response to low energy neutrons. The two detectors are in contact with an 80 μm thick polyethylene radiator to

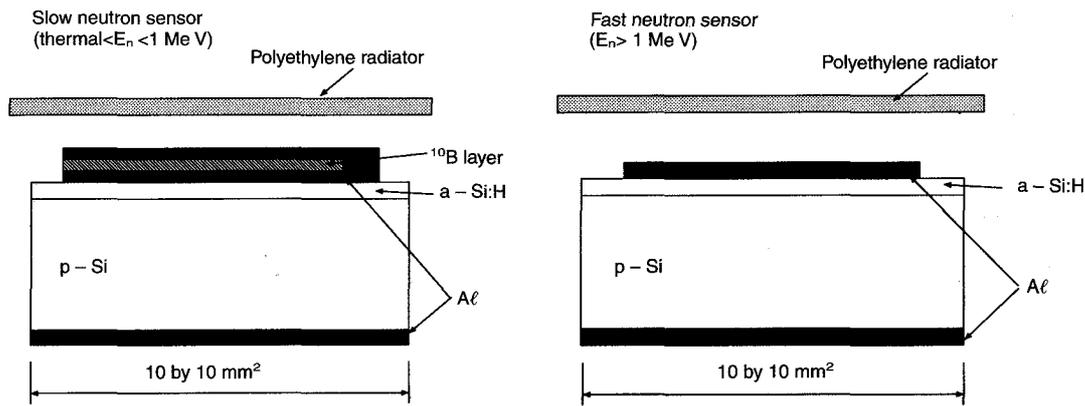


Figure 13. Cross-sectional view of the slow and fast neutron sensors. (Nakamura et al, 1999).

produce recoil protons in response to high-energy neutrons. In this configuration the slow neutron sensor has also some sensitivity for fast neutrons. The dosimeter response as a function of the neutron energy can be seen in figure 14. Curve A in the figure represents the fast neutron sensor response, curve C the slow neutron sensor and the compound response is represented by curve B. Curve D is the energy variation of the fluence to personal dose equivalent conversion coefficient, i.e. the reference curve for the dosimeter response.

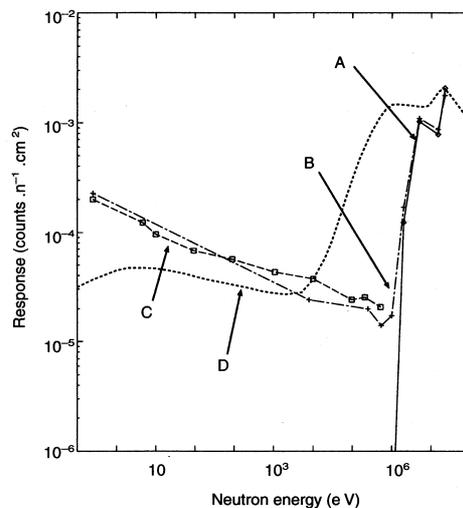


Figure 14. Dosimeter response to fluence as a function of neutron energy with the ambient dose equivalent conversion factor described in ICRP 74. (Nakamura et al. 1999).

Figure 14 illustrates well the difficulty of obtaining the desired energy response in a sufficiently wide range of energies. In principle the response below a few keV is not very problematic. Perhaps by increasing moderation of absorption of incident neutrons the response of the slow neutron sensor (curve C) would be closer to the reference curve (curve D). This solution was not attempted in the design now commented, meaning to compensate the clear under-response for energies between 10 keV and 1.0 MeV, the more problematic region. Above 1 MeV the response of the high energy sensor (curve A) is well approaching the reference curve D.

The problems encountered in the energy range 10 keV - 0.5 MeV are mainly caused by the low energy of the recoil protons produced by the hydrogen rich converters. These protons have limited ranges in silicon, only 1 μ m for 100 keV protons, and thus the presence of unavoidable dead layers in the semiconductor detectors (from .1 to 1 μ m) decreases the detector response. In addition, low energy protons produce pulses of comparable amplitude to that of photons, complicating the not simple task of discriminating photons, particularly in mixed fields with a high photon contribution. Attempts have been made to improve the energy response, and several related articles can be found in the proceedings of the 8th Neutron Dosimetry Symposium (Menzel et al., 1997) and of the 12th Solid State Dosimetry (Delgado et al., 1999). Energy response is not the sole problematic characteristic, as sensitivity is not very high, with detection limits of the order of 100 μ Sv, when the desirable limit is not higher than 10 μ Sv.

Recently, new designs have been proposed including up to four or six detectors, placed behind different absorbers/converters, recording the pulse height spectra and combining the individual measurements using mathematical algorithms (Luszk-Bhadra et al., 1999) and (Fehrenbacher et al., 1999). Figure 15, reproduced from the work of Luszk-Bhadra, presents the dose equivalent response of a four element dosimeter. In the figure S is the overall response, after combining the individual response of the four elements (A: albedo; C: closed; F: fast and O: open). Although the overall response is not flat, some improvement respect to simpler designs for the intermediate energy region can be appreciated. The price is a substantially higher complexity, both in the construction and in the analysis of the raw data. The low dose limit reported for this dosimeter is surprisingly good, of the order of 3 μ Sv for routine personal monitoring.

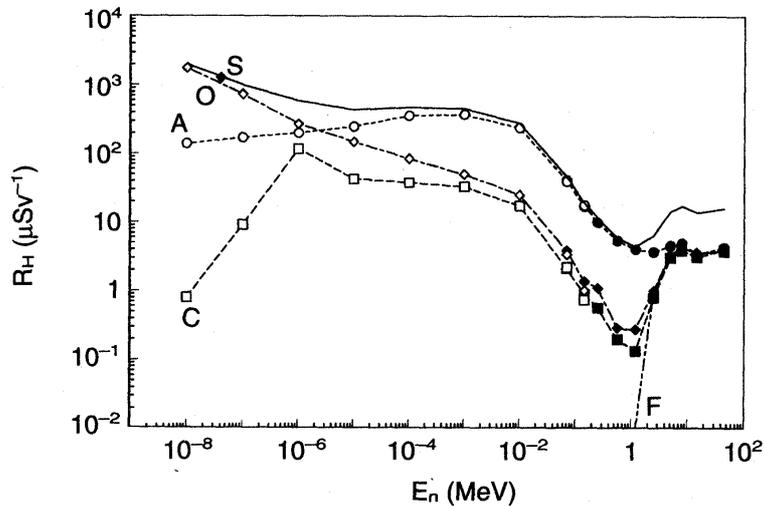


Figure 15: Dose equivalent response of the dosimeter components and the summed value (Sum) using normally incident neutrons and a threshold for charged particles at 1.3 MeV. The full symbols indicate measurements and the open symbols calculations. (Luszk-Bhadra et al., 1999).

Etched Track Detectors

Different materials have been employed for neutron detection by counting the tracks formed by secondary charged particles, namely polycarbonates, Markofol among them or cellulose nitrates, LR-115 for example. Nevertheless, the material that has aroused more interest has been Poly Allyl Diglycol Carbonate, PADC or CR-39. This rather common plastic presents a high sensitivity to protons (Cartwright et al., 1978) and thus to recoil protons produced by the scattering of fast neutrons by the hydrogen content of CR-39. Covering the detector with a suitable converter containing ^6Li or ^{10}B , permits to extend the working range also to thermal neutrons. CR-39 is also insensible to photons, thus permitting good discrimination in mixed fields.

Figure 16 presents the fluence response with the neutron energy of the PADC personal dosimeter employed at NRPB (UK), (Tanner et al., 1999). This response somewhat resembles the reference curve for the energy dependence of the fluence to Hp(10) conversion coefficients represented in figure 6. There is a blind or nearly blind region between 10 and 100 keV, where the efficiency of both neutron capture and neutron scattering is low. Thus, PADC, as many others neutron dosimeters, is a material whose utilisation requires some knowledge of the neutron field in order to apply sensitivity correction factors. Nevertheless, in view of the severe limitations of other methods to detect fast neutrons, the ability of CR-39 to work in that region is to be remarked. CR-39 in fact has been employed together with TLDs in albedo dosimeters, complementing their respective capabilities, TLD for the low and intermediate energy neutrons and CR-39 for fast neutrons (Piech and Burgkhardt, 1988).

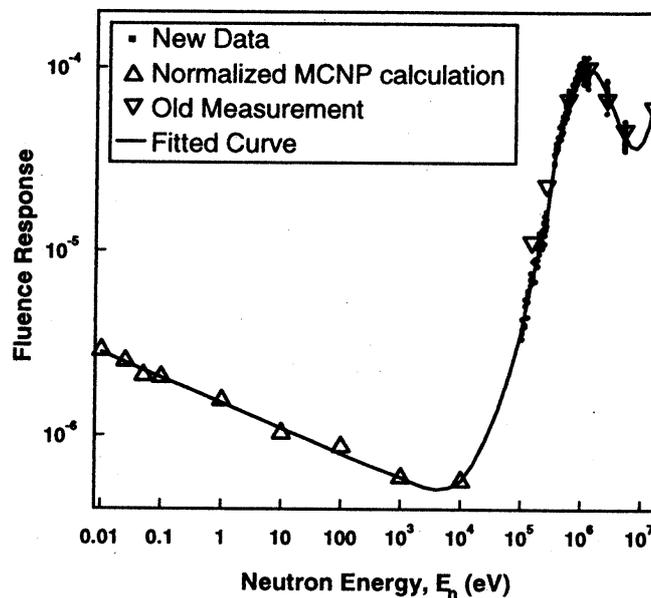


Figure 16: Fluence response of the PADC personal dosimeter employed at NRPB (UK). (Tanner et al., 1999).

Apart from the limited energy dependence, another important problem of CR-39 for dosimetry is the highly variable background, caused by the poor reproducibility of the characteristics of the material available in the market (Harvey et al., 1997). Undetected high background areas in a sheet, can lead to the occurrence of false positive readings. Several investigations have been conducted on the influence of the etching procedures

on the background characteristics. Chemical plus one or two steps of electrochemical etching have been employed (Tanner et al., 1997). Nevertheless, it seems that the most effective way to overcome or palliate this important drawback relies on the improvement of the manufacturing methods. The application of curing procedures seems to improve reproducibility (Ahmad and Stejny, 1991). Several types of quality control tests for the acceptance of CR-39 sheets have been implemented in dosimetry services (Jackson, 1997) improving the reliability of the dose estimations, and reducing the incidence of false positives to less than 1/1000.

The poor background reproducibility also determines an inconveniently high minimum detectable dose, typically of the order of 200 μSv . The angle dependence of the usually thin and plane CR-30 dosimeters is not very good, with the response decreasing as the angle of incidence increases. Configurations including several elements have been developed to improve the angular dependence (Harvey et al., 1997).

d) Spectrometric methods for neutron field characterisation.

Spectrometry can efficiently help dosimetry to obtain more accurate dose equivalent data in area and personal monitoring. As explained in the preceding sections, a problematic characteristic of whichever dosimetric method is the deficient energy response. Since usually in activities related with the protection of exposed workers neutron spectra can be rather broad, ranging from meV to several tens of MeV, and with different fluence distributions, the knowledge of the actual features of the measured neutron field is essential. This can allow to apply appropriate sensitivity correction factors, overcoming or at least palliating the detectors poor energy response. Spectrometric methods can provide the required information on the neutron field, for example in situations where the dose limits have been exceeded or simply approached the estimation of the effective dose should be required. Then, it would be convenient to re-estimate the values of the operational quantities taking into account the neutron spectra to improve accuracy. Therefore, spectrometry for many reasons is a convenient complement of neutron dosimetry.

Multisphere spectrometers (Bonner spheres) are perhaps the better known systems, in use since the work of Bramblett, Ewing and Bonner (Bramblett et al., 1969). These consist of a number of moderator spheres of different diameters with a $1/v$ type detector in their centre. Up to 6 spheres between 2 (5.1 cm) and 18 (45.7 cm) can be employed to obtain the energy distribution of the neutron fluence. Different detectors can be employed to count the thermal neutrons absorbed in the spheres. ^3He proportional counters, $^6\text{LiI}(\text{Eu})$ scintillator counters, or even passive detectors like $^6\text{LiF:Mg,Ti}$. The energy range covered is very ample, from meV to 20 MeV, or up to substantially higher energies, 10 GeV, using bigger spheres.

Different moderation, i.e. different diameter for a sphere j , implies different energy response $R_j(E)$. If the system of spheres is exposed to a neutron fluence with an energy distribution, Φ_E , then the reading of the detector j , is given by:

$$C_j = \int \Phi_E \cdot R_j(E) \cdot dE \quad (1)$$

The energy distribution of the neutron fluence, Φ_E is obtained unfolding the set of equations (1) for all the detectors by numerical methods. Recent improvements of the multi-sphere spectrometer have been published (Mares et al., 1994), (Aroua et al., 1997).

Recoil proton proportional counters filled with a hydrogenous gas can also be employed for neutron spectrometry. A system including four spherical proportional counters with different gas fillings has been recently proposed (Ing et al., 1997). It covers the range 50 keV to 4.5 MeV and permits a high-resolution spectrometry. As the proportional counters are mounted in a slowly rotating base, the system is called ROSPEC, ROTating SPECTrometer.

Regardless of the spectrometric method employed, once the neutron fluence distribution has been obtained the ambient dose equivalent can be derived through the known coefficients for fluence to dose equivalent $h_q(E)$, as explained in the Principles of Measurement section above.

SUMMARY

A description of the most important features and characteristics of the dosimetry of mixed fields has been presented. This description includes such aspects as the principles of measurement for determination of the dose equivalent for personal and area dosimetry, the operational quantities to be measured in each case, and an overview of the experimental methods employed.

The beta-gamma and the neutron-gamma mixed fields have been treated with some detail as they are the most frequently found in practice. The achievements and the limitations of the traditional, as well as of some recent and innovative methods proposed for the dosimetry of these mixed fields have been commented. Abundant references have been included, where interested readers can extend their knowledge of the complex topic of the dosimetry of mixed radiation fields.

REFERENCES

a) GENERAL

Attix, F.H. (1986). Introduction to Radiological Physics and Radiation Dosimetry. Wiley-Interscience Pub., New.York., USA.

Greening, J.R., (1981). Fundamentals of Radiation Dosimetry. Medical Physics Handbooks, Adam Hilger, Bristol, UK.

ICRP 26 (1977). International Commission on Radiological Protection, 1977 Recommendations of the International Commission on Radiological Protection, ICRP Publication 26. (Pergamon Press, Oxford).

ICRP 60 (1991). International Commission on Radiological Protection, 1990 Recommendations of the International Commission on Radiological Protection, ICRP Publication 60, Annals of the ICRP **21**, No.1/3 (Pergamon Press, Oxford).

ICRU 39 (1985). International Commission on Radiation Units and Measurements. Determination of Dose Equivalents Resulting from External Radiation Sources. ICRU report 39, (ICRU, Bethesda, MD).

ICRU 51 (1993). International Commission on Radiation Units and Measurements. Quantities and Units in Radiation Protection. ICRU report 51, (ICRU, Bethesda, MD).

ICRU 56 (1997). International Commission on Radiation Units and Measurements. Dosimetry of External Beta Rays. ICRU report 56, (ICRU, Bethesda, MD).

b) BETA-GAMMA DOSIMETRY

Bartlett, D.T., Ambrosi, P., VanDijk, J.W. and Bordy J.M. (Editors). (2000). Report of EURADOS action group 4.1 Harmonisation and Dosimetric Quality Assurance in Individual Monitoring of External Radiation . Part II, A Catalogue of European Dosimetric Services that Estimate Doses of External Radiation as Personal Dose Equivalent. To be Published in Radiat. Prot. Dosim.

Borg, J. and Christensen, P. (1995). Determination of Dose Rates in Beta Radiation Fields Using Extrapolation Chambers and GM Counter. Radiat. Prot. Dosim. **61**, 47-52.

Chabot, G.E., McWilliams, F.F. and Skrable, K.W. (1988). Modifications of the Victoreen Model 640-4A Ionisation Chamber for Beta Dose Measurements. Radiat. Prot. Dosim. **25**, 49-53.

Christensen, P. (1993). Study of LiF:Mg,Cu,P TL Detectors for Individual Monitoring of Weakly Penetrating Radiations. Radiat. Prot. Dosim. **47**, 425-430.

Fellinger, J. (1991). Present State and Problems of Monitoring Techniques for Individual Dosimetry of Weakly Penetrating Radiations. Radiat. Prot. Dosim. **39**, 127-130.

Goldfinch, E.P. (1993). Individual Thermoluminescence Dosimeters for Extremity and Eyes. Radiat. Prot. Dosim. **47**, 347,351.

Horowitz, Y.S., Hirning, C.R., Yuen, R. and Aitkens, M. (1994). Beta-ray Spectroscopy Based on a Plastic Scintillation Detector/Silicon Surface Barrier Detector Coincidence Telescope. Nucl. Instrum. Methods. **A338**, 522-533.

Horowitz, Y.S., Weizman, Y. and Hirning, C.R. (1995). A Beta-Ray Spectrometer Based on a Two-, or Three-Element Silicon Detector Coincidence Telescope. Radiat. Prot. Dosim. **61**, 53-56.

Marshall, T.O., Bartlett, D.T., Burgess, P.H., Cranston, C.S., Higginbottom, D.J. and Sutton, K.W. (1994). Electronic Personal Dosimeters. Radiat. Prot. Dosim. **34**, 93-97.

Swinth, K.L. and Sisk D.R. (1991). Recent Developments and Performance of Survey Instruments for the Monitoring of Weakly Penetrating Radiations. *Radiat. Prot. Dosim.* **39**, 149-156.

Tawil, R.A., Velbeck, K.J., Rotunda, J.E., Gooley, R., Moscovitch, M. and Wu, C.F. (1993). A System for Extremity Monitoring. *Radiat. Prot. Dosim.* **47**, 445-450.

Yuen, P.S., Freeman, N.O., Frketich, G. and Rotunda, J.E. (1999). Evaluation of Bicron NE MCP DXT-RAD Passive Extremity Dosemeter. *Radiat. Prot. Dosim.* **85**, 187-195.

c) NEUTRON-GAMMA DOSIMETRY

Ahmad, S. and Stejny, J. (1997). Polymerisation, Structure and Track Recording Properties of CR-39. *Nucl. Traks Radiat. Meas.* **19**, 11-16.

Alberts, W.G. (1999). Active Neutron Personal Dosimetry. *Radiat. Prot. Dosim.* **85**, 27-30.

Alberts, W.G., Dietze, G., Guldbakke, S., Kluge, H. and Schuhmacher H. (1989). International Intercomparison of TEPC Systems used for Radiation Protection. *Radiat. Prot. Dosim.* **29**, 47-53.

Anderson I.O. and Braun, J. (1964), A Neutron REM Counter. *Nukleonik* **6**, 237-241

Apfel, R.E. (1979). The Superheated Drop Detector. *Nucl. Instrum. Meth.* **162**, 603-608.

Aroua, A., Grecescu, M., Petre, S. and Valley, J.-F. (1997). Improved Neutron Spectrometer Based on Bonner Spheres. *Radiat. Prot. Dosim.* **70**, 285-290.

Bartlett, D.T., Tanner, R.J. and Thomas, D.J. (1999). Active Neutron Personal Dosimeters-A Review of Current Status. *Radiat. Prot. Dosim.* **86**, 107-122.

Bramblett, R.L., Ewing, R.J. and Bonner, T.W. (1960). A New Type of Neutron Spectrometer. *Nucl. Instrum. Meth.* **9**, 1-2.

Burgkhardt, B., Fieg, G., Klett, A., Plewina, A. and Siebert, B.R.L. (1997) The Neutron Fluence and H*(10) Response of the New LB 6411 Rem Counter. *Radiat. Prot. Dosim.* **70**, 361-364.

Cartwright, B.G., Shrik, E.K. and Price, P.B. (1986). A Nuclear-track Recording Polymer of Unique Sensitivity and Resolution. *Nucl. Instrum. Meth.* **153**, 457-460.

D Errico, F. Fundamental Properties of Superheated Drop (Bubble) Detectors. *Radiat. Prot. Dosim.* **84**, 55-62.

D Errico, F., Alberts, W.G., Curzio, G., Guldbakke, S., Kluge, H. and Matzke, M. (1995). Active Neutron Spectrometry with Superheated Drop (Bubble) Detectors. *Rad. Prot. Dosim.* **61**, 159-162.

Delgado A. and Gomez Ros, J.M. (Editors) (1999). Proceedings 12th International Conference on Solid State Dosimetry. (Burgos, July 1998). Radiat. Prot. Dosim. **84**, **85**.

Delgado, A. (1996). Recent Advances in LiF:Mg,Ti and LiF:Mg,Cu,P Based Environmental Dosimetry. Radiat. Prot. Dosim. **66**, 129-134.

Douglas, J.A. and Marshall, M. (1978). The Responses of Some TL Albedo Neutron Dosimeters. Health Phys. **35**, 315-324.

Fehrenbacher, G., Schutz, R., Hahn, K., Sprunk, M., Cordes, E., Biersack, J.P. and Wahl, W. (1999). Proposal of New Method for Neutron Dosimetry Based on Spectral Information Obtained by Application of Artificial Neural Networks. Radiat. Prot. Dosim. **83**, 293-301.

Golnik, N. (1997). Review of Recent Achievements of Recombination Methods. Radiat. Prot. Dosim. **70**, 211-214.

Harvey, J.R., Tanner, R.J., Alberts, W.G., Bartlett, D.T., Piech, E. and Schraube, H. (1997) The Present State of Etched Track Neutron Dosimetry in Europe and the Contribution of CENDOS and EURADOS. Radiat. Prot. Dosim. **70**, 127-132.

Hollnagel, R.A. (1994). Conversion Functions of the Dose Equivalent H_{sl} (10) in the ICRU Slab used for the Calibration of Personal Neutron Dosimeters. Radiat. Prot. Dosim. **54**, 227-230.

Ing, H., Clifford, T., McLean, T., Webb, W., Cousin, T. and Dhermain, J. (1997). ROSPEC- A simple Reliable High Resolution Neutron Spectrometer. Radiat. Prot. Dosim. **70**, 273-278.

Jackson, M. and French, A.P. (1977). Analysis of Acceptance testing Data for More Than 800 Sheets of CR-39 Plastic Assessed for the DRPS Approved Neutron Dosimetry Service. Radiat. Prot. Dosim. **70**, 139-142.

Kliauga, P., Waker, A.J. and Barthe, J. (1991). Design of Tissue-Equivalent Proportional Counters. Radiat. Prot. Dosim. **61**, 309-322.

Leake, J.W. (1968). An Improved Spherical Dose Equivalent Neutron Detector. Nucl. Instrum. Meth. **63**, 329-332.

Lindborg, L., Grindborg, J.E. and Tilikidis, A. (1988). The Variance Technique and Low Pressure Proportional Counters in Radiation Protection Measurements. Radiat. Prot. Dosim. **23**, 223-226.

Lusik-Bhadra, M., Matzke, M., Dietz, E., Guldbakke, S., Hecker, O., Sosaat, W. and Wigal, B. (1999). An Active Personal Dosimeter/Spectrometer for Neutrons. Radiat. Prot. Dosim. **84**, 375-380.

Mares, V. and Schraube, H. (1994). Evaluation of the Response Matrix of a Bonner Sphere Spectrometer with LiI Detector from Thermal Energy to 100 MeV. Nucl. Instrum. Meth. **A337**, 461-473.

- Menzel, H.G., Chartier, J.L., Jahr, R. and Rannou, A. (Editors) (1997). Proceedings 8th Symposium on Neutron Dosimetry (Paris, November 1995) . Radiat. Prot. Dosim. **70**.
- Menzel, H.G., Lindborg, L., Schmitz, Schuhmacher, H. and Waker, A.J. (1989). Intercomparison of Dose Equivalent Meters Based on Microdosimetric Techniques: Detailed Analysis and Conclusions. Radiat. Prot. Dosim. **29**, 5-68.
- Mourgues, M., Caross, J.C. and Portal, G. (1985). A Light Rem-counter of Advanced Technology. In Proc. Fifth Symp. On Neutron Dosimetry, Munchen, EUR-9762, CEC, Luxembourg).
- Nakamura, T., Sasaki, M., Ueda, O. and Suzuki, T. (1999). Characteristics of a Real-time Personal Neutron Dosimeter with Two Silicon Detectors. Radiat. Prot. Dosim. **85**, 45-48.
- Piech, E. and Burgkhardt, B. (1985). Albedo Neutron Dosimetry. Radiat. Prot. Dosim. **10**, 175-188.
- Piech, E. and Burgkhardt, B. (1988). Albedo Dosimetry System for Routine Personnel Monitoring. Radiat. Prot. Dosim. **23**, 117-120.
- Pradhan, A.S., Rassow, J. and Meissner, P. (1985). Dosimetry of d(14)+Be Neutrons with the Two Peak Method of LiF TLD-700. Phys. Med. Biol. **30**, 1349-1354.
- Schmitz, Th., Waker, A.J., Kliauga, P. and Zoetelief, H. (Editors) (1995) EURADOS Report Design, Construction and Use of Tissue Equivalent Proportional Counters . Radiat. Prot. Dosim. **61**.
- Schuhmacher, H. (1995). On the Achievements of Operational Radiation Protection Quantities for Mixed Photon-Neutron Radiation Fields. Radiat. Prot. Dosim. **61**, 63-66.
- Tanner, J.E. McDonald, R.D., Stewart, R.D. and Wernli, C. (1977). Dose Equivalent Response of Personal Neutron Dosimeters as a Function of Angle. Radiat. Prot. Dosim. **70**, 165-168.
- Tanner, R.J., Bartlett, D.T., Hager, L.G. and Lavelle, J. (1997). Future Developments in the NRPB PADC Neutron Personal Monitoring Service. Radiat. Prot. Dosim. **70**, 121-126.
- Tanner, R.J., Thomas, D.J., Bartlett, D.T. and Horwood, N. (1999). Systematic Errors in the Readings of Track Etch Neutron Dosimeters Caused by the Energy Dependence of Response. Radiat. Meas. **31**, 483-486.
- Waker, A.J. (1995). Principles of Experimental Microdosimetry. **61**, 297-308.
- Youssian, D. and Horowitz, Y.S. (1998). Estimation of Gamma Dose in Neutron Dosimetry using Peak 4 to Peak 5 Ratios. Radiat. Prot. Dosim. **77**, 151-158.