International Atomic Energy Agency

INDC(IND)-043
Distr.: GI

INDC INTERNATIONAL NUCLEAR DATA COMMITTEE

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December 1990

IAEA NUCLEAR DATA SECTION, WAGRAMERSTRASSE 5, A-1400 VIENNA

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Lecture presented at the IAEA Interregional Training Course
on Nuclear Measurements and Methods in Reactor and
Personal Neutron Dosimetry, 18 May-9 June 1989, Riga, USSR

December 1990

Reproduced by the IAEA in Austria
January 1991

91-00131

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ABSTRACT

Personnel Neutron Dosimetry aims at providing a method to evaluate the magnitude of the detrimental effects on the personnel exposed to neutrons. Neutron Dosimetry is done for a small though growing number of personnel working with neutrons in a wide range of situations. A review of currently available Personnel Neutron Dosimetry Systems is presented in this paper.

INTRODUCTION:

Personnel Neutron Dosimetry is done for a small though growing number of personnel working with neutrons in a wide range of situations. The main sources of neutrons are radioactive neutron sources, nuclear reactors, particle accelerators, neutron generators and fuel reprocessing plants. Neutron sources are being widely used in medicine, industry, research and agriculture. Growth in nuclear power production and development of fusion power technology are likely to increase the number of radiation workers to be monitored for neutron exposures. Neutrons are also produced in the atmosphere by cosmic rays and add a small amount to the general exposure of man.

Personnel Neutron Dosimetry aims at providing a method to evaluate the magnitude of the detrimental effects on the personnel exposed to neutrons. This requires two things - (a) a quantity which is a reasonably accurate measure of the biological detriment caused by neutron irradiation and this quantity is dose equivalent for low level chronic exposure and (b) to provide suitable methods for measurement of this quantity for routine use in the working environment for the personnel. Information from personnel neutron dosimeter must be augmented by information using neutron area monitors and spectrometers in the radiation environment to provide a complete picture of the personnel neutron exposure (1).

Unlike X or gamma personnel dosimetry, personnel neutron dosimetry presents several problems. Main problem is the fact that neutrons have a biological effect which depends upon the energy of neutrons and can be greater than the effect produced by an equal dose of gamma rays by a factor of as much as ten and an upward revision of quality factor by a factor of two for neutrons is recommended by International Commission on Radiological Protection (ICRP) recently. Also moderated neutron spectra cover nine decades of energy and the dosimeter response must follow the fluence to dose

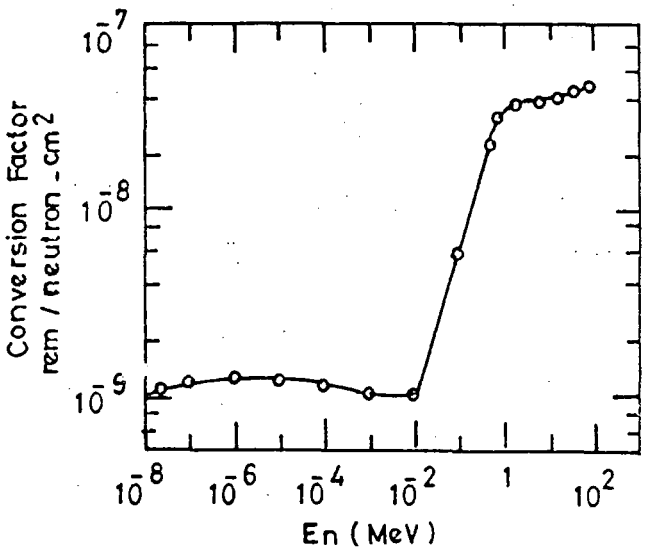

Fig. 1. Neutron Dose equivalent Conversion Factor (unidirectional broad beam, normal incidence). Data from (ICRP 21)

equivalent conversion factor which varies by a factor of forty over this range as shown in Fig. 1. Knowledge of the neutron spectrum in the working environment is also essential. The interference of associated gamma dose can also present problems in personnel neutron dosimetry.

FLUENCE TO DOSE EQUIVALENT CONVERSION FACTORS

The dose equivalent (H) used in neutron dosimetry is most commonly based on the fluence to dose equivalent conversion factors given by the National Council on Radiation Protection and Measurement of U.S.A. (2) and International Commission on Radiological Protection (3). These conversion factors for evaluating H have been established mainly by Monte Carlo calculations. In essence, these calculations simulated neutron behaviour in a model of the human body and determined the statistical distribution of absorbed dose D and dose equivalent H for mono-energetic neutrons upto 14 MeV incident

normally to the surface on one side of the body. For reasons of conservatism, the maximum value of H per unit fluence, for both neutrons and gamma rays generated in the body, was selected for the recommended conversion factors for radiation protection purpose.

There are several problems with the conversion factors used in connection with the evaluation of H that have created considerable confusion. One is that these factors depend on the particular model of the body phantom. Results for conversion factors have been published for a variety of models including semi-infinite slabs, spheres, circular cylinders and eliptical cylinders of unit density materials with a mass composition of 76.2 % oxygen, 11.1% carbon, 10.1% hydrogen and 2.62% nitrogen (4). Thus, the computed dose equivalent for a particular situation varied depending upon the particular set of conversion factors used.

The relationship between the maximum dose equivalent H_{MADE} and neutron fluence is given in Table 1 for monoenergetic neutrons and in Table 2 for some radioactive neutron sources and moderated neutron spectra. These currently accepted factors are evaluated at the maximum of the depth dose equivalent curves in a 30 cm thick slab of tissue equivalent material (3).

NEUTRON FIELD DOSIMETRY

For carrying out dosimetry in working areas where neutron fields exist, neutron area monitors are used. These are dynamic monitors which can be used to obtain an immediate indication of neutron dose equivalent or fluence. These can be categorised into four types: thermal neutron monitors, fast neutron monitors, moderator type dose equivalent monitors and spectrometers. The detection of thermal neutrons is straight forward and depending on the intensity of the thermal fluence, many techniques can be used. Activation detectors like gold, indium, dysprosium, manganese, cobalt and copper can be used for high thermal neutron fluence measurement for example in criticality accident dosimetry. For low thermal neutron fluence, detectors utilising ^{10}B (n, alpha), ^{3}He (n, p), ^{6}Li (n, alpha) or ^{235}U (n, f) reactions can be used. Such detectors are gas proportional counters or scintillators and are readily available.

For the detection of fast neutrons, the elastic scattering of neutrons by hydrogen in ionisation chambers, proportional counters and scintillators is used. Dennis and Loosemore counter (5) which has a lining of tissue equivalent material and is filled with

TABLE 1

RELATIONSHIP BETWEEN THE MAXIMUM DOSE EQUIVALENT, H_{MADE} AND THE NEUTRON FLUENCE (3)

Neutron Energy (MeV)	Quality factor Q	Conversion factors	
		$(\frac{cm^{-2}\ s^{-1}}{\mu Sv/h})$	$(\frac{cm^{-2}}{Sv})$
Thermal	2.3	26	9.36×10^{10}
1×10^{-7}	2.0	24	8.64×10^{10}
1×10^{-6}	2.0	22	7.92×10^{10}
1×10^{-5}	2.0	23	8.28×10^{10}
1×10^{-4}	2.0	24	8.64×10^{10}
1×10^{-3}	2.0	27	9.27×10^{10}
1×10^{-2}	2.0	28	1.01×10^{11}
1×10^{-1}	7.4	4.8	1.73×10^{10}
5×10^{-1}	11.0	1.4	5.04×10^{9}
1	10.6	0.85	3.06×10^{9}
2	9.3	0.70	2.52×10^{9}
5	7.8	0.68	2.45×10^{9}
10	6.8	0.68	2.45×10^{9}
20	6.0	0.65	2.34×10^{9}
50	5.0	0.61	2.20×10^{9}
1×10^{2}	4.4	0.56	2.02×10^{9}

TABLE 2

RELATIONSHIP BETWEEN THE MAXIMUM DOSE EQUIVALENT, H_{MADE} AND NEUTRON FLUENCE (21)

SOURCE	Average Energy (MeV)	Conversion Factors cm^{-2}/Sv
^{241}Am-Be	4.1	2.72×10^{9}
^{239}Pu-Be	2.3	3.00×10^{9}
^{252}Cf	2.1	3.17×10^{9}
$^{239}PuF_4$	1.4	3.09×10^{9}
H_2O Moderated Fission Neutrons through concrete of thickness 60 cm.		6.71×10^{9}
D_2O Moderated Fission Neutrons through concrete of thickness 60 cm		7.80×10^{9}

methane and argon gas can be used for fast neutron radiation survey work and is commercially available. The most commonly used scintillators are stilbene crystals, plastic scintillators and the NE 213 liquid scintillator.

Moderator type dose equivalent meters which are popularly known as Rem Counters consist of a spherical or a cylindrical polyethylene moderator assembly surrounding a thermal neutron detector like BF_3 or ^3He proportional counter or ^6LiF scintillator. The energy response of the monitor approximates as closely as possible to the variation with neutron energy of dose equivalent per unit fluence. These instruments are commercially available and are extensively used for neutron field survey in the working environment. They directly give the neutron dose equivalent rate (6,7).

For carrying out accurate neutron dosimetry, knowledge of the neutron spectrum in the working environment is also essential. The usual methods of neutron spectrum measurements are activation detectors, proton recoil methods, scintillation spectrometers and multisphere detectors or Bonner spheres. As neutron fluxes in the working invironment are usually low, Bonner spheres are used for neutron spectrum measurements in radiation protection. These spheres consist of thermal neutron detectors surrounded by a set of polyethylene spheres of 2,3,5,8,10,12 and 15 inches. Measured counts from the thermal neutron detector at the centre, usually ^3He, ^6LiI or ^6LiF TLDs, are unfolded using the response matrix to yield the neutron spectrum (8,9,10). A cadmium covered detector and a bare one are usually included in the spectrometer for measuring the thermal and epithermal fluxes. With the advent of microprocessors, the neutron spectrometers have a reduced size and more elaborate treatment of data can be carried out to arrive at dosimetric quantities with high accuracy.

PERSONNEL NEUTRON DOSIMETRY

One can broadly identify the requirements of an ideal personnel neutron dosimeter as follows (11).

1. It should record neutron dose equivalent from a fraction to multiples of maximum permissible dose equivalent limits.
2. Its neutron energy response should be proportional to dose equivalent response i.e. it should be useful over a considerable range of neutron energies.

3. It should be insensitive to other types of radiations like beta, x-rays and gamma rays.
4. It should have a good signal stability with little fading.
5. It should not be toxic to the wearer.
6. It should be rugged, easy to handle for a large scale of monitoring and be reasonably inexpensive.

There are mainly three types of personnel neutron dosimeters which have found acceptance for general use. These are nuclear emulsion KODAK NTA, thermoluminescent albedo dosimeters and solid state nuclear track detector. Each of these personnel neutron dosimeters have different detection mechanism, advantages and limitations. These are passive dosimeters which require some form of development after a use of certain period to provide the dose equivalent. Pocket neutron dosimeters which are under development in many laboratories are dynamic personnel neutron dosimeters. Bubble damage polymer detectors proposed by Ing and Birboim (12) are new personnel neutron dosimeters coming on the scence. In these, microscopic droplets of superheated liquid are distributed in a firm elastic polymer in a glass tube. When a neutron strikes a droplet or its immediate vicinity, the energy of the charged particle causes the droplet to explode resulting in a gas bubble which can be visually counted and correlated with neutron dose equivalent. These detectors are now commercially available from Chalk-river, Canada. They are insensitive to gammas and are neutron energy independent from 100 keV to 14 MeV and can be detect about 100 uSv from ^{241}Am-Be neutrons, Of course, these detectors are susceptible to temperature and impact and are quite expensive also (13).

KODAK NTA PERSONNEL NEUTRON DOSIMETER

One of the earliest methods employed for personnel neutron dosimetry has been the use of special photographic emulsions (14, 15, 16, 17). The nuclear emulsion type A (NTA) produced by KODAK is the most commonly used photographic emulsion for personnel neutron dosimetry. It is the oldest and still the most popular system in use. It is a 30 micron thick photographic emulsion coated on one side of the cellulose acetate base and has fine grains of 0.3 micron size. The elastic collisions of neutrons with hydrogen nuclei in the emulsion, cellulose acetate base, packaging and the film holder produce recoil proton tracks. These are counted on a high magnification ($\times$ 750) microscope and

correlated with neutron dose equivalent. The proton tracks of varying lengths are produced for neutrons of different energies.

However, to recognise a proton track, only those tracks which consist of at least three to four developed grains are taken to represent a proton track. Thus the minimum proton energy that can be recognised in the emulsion is 300 keV and correspondingly the minimum energy for neutrons must be at least 300 keV. In practice, easy recognition of tracks is possible only from neutron energies of 0.5 MeV and above. Thus Kodak NTA is essentially a fast neutron personnel dosimeter. However, thermal neutrons can also produce protons of 0.58 MeV in the nitrogen of the emulsion by ^{14}N (n, p) reaction. Cadmium filters are used in the film holder in the front and at the back to cut off the incident as well as backscattered thermal neutrons from body.

Lot of studies have been carried out with Kodak NTA personnel neutron dosimeter. Main studies have been on its energy response and post-irradiation latent image fading. Knowledge of its energy response is very important for its use in different neutron fields. Generally it responds to neutrons from 300 keV to 20 MeV and by using proton radiators in front its response can be extended upto 50 MeV. Its energy response has been determined experimentally to various radioactive neutron source like ^{241}Am-Be and ^{252}Cf and monoenergetic neutrons from accelerators. Its theoretical energy response is shown in Fig. 2. Curve 1 is based on the calculations of Cheka (14) and is for Kodak NTA film covered by a proton radiator of 100 mg/cm^2 and curve 3 is the response of the film in the factory wrapping and these two curves are based on the calculations of Piesch (15). Thus it must be clearly noted that the response of Kodak NTA dosimeter is affected by the radiator thickness in front of film at higher neutron energies. Usually, the calibration of this dosimeter is done using a calibrated ^{241}Am-Be neutron source. But for its use in different neutron environments, the energy response must be clearly known. Extensive computations have been done for different moderated neutrons spectra chosen from the Compendium of Neutron Spectra for Criticality Accident Dosimetry (18). We have also determined the experimental as well as computed response for a number of radioactive neutron sources and twenty two selected moderated neutron spectra (19,20,21). Based on these studies, correction factors can be applied for using the Kodak NTA dosimeter in different neutron environments.

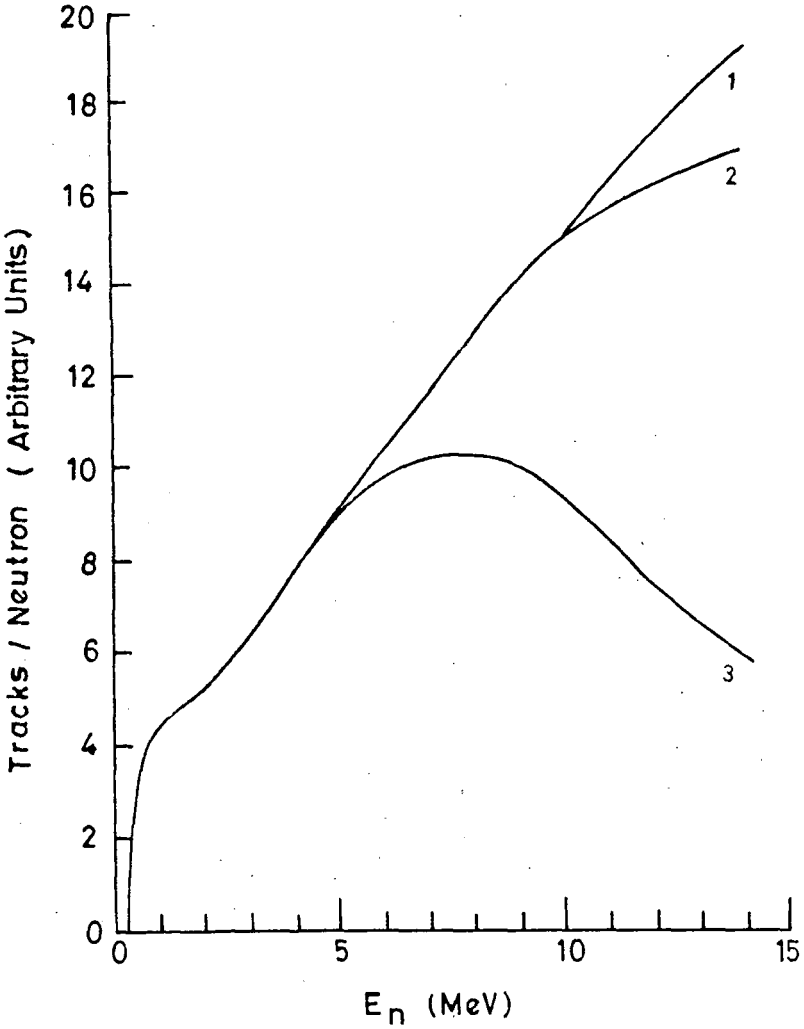

Fig. 2. Neutron Response of Kodak NTA Monitor

One of the drawbacks of Kodak NTA film is the post irradiation latent image fading. In factory wrapping, all information is lost within one week at 30°c and 80% relative humidity. Thus unless carefully controlled conditions are ensured or moisture proof packaging is provided, it is difficult to use these films for a reasonable service period. Lot of studies have been carried out under different temperature and humidity conditions. The fading in these films is caused due to the oxidation of the silver of the latent image turning it back into silver ions in the presence of oxygen and water vapour. A great improvement can be achieved by dessication of the films in dry nitrogen and subsequent sealing in aluminised polythene pouches (22). We are also sealing the films in triple laminate pouches of paper, twelve micron aluminium foil and polythene and find no fading for a period of four months (23). Thus, the Kodak NTA film can be easily used for a service period of two to three months.

Other characteristics of this dosimeter are its gamma sensitivity being a photographic emulsion, particularly to low energy photons. But it can be used with 30 mSv gamma background. Evaluation of dose equivalent from these dosimeters is done by manual counting of proton tracks on a microscope which is quite tedious. The dose range covered by Kodak NTA dosimeter is from 100 uSv to 100 mSv. It gives a permanent record of dose. It is easy to handle, rugged and not very expensive for large scale personnel dosimetry for fast neutrons.

ALBEDO NEUTRON PERSONNEL DOSIMETER

These dosimeters rely upon neutron reflected from the body of the wearer. These neutrons are called albedo neutrons and can be detected by a dosimeter placed on the body. The neutron albedo factor defined as the ratio of the neutron fluence scattered from the body to the total incident neutron fluence entering the body varies between 0.8 for thermal neutrons and 0.1 for neutrons of 1 MeV. It is possible to estimate the dose equivalent in the body due to the original incident neutrons from the dosimeter worn on the body and such a dosimeter is known as albedo neutron personnel dosimeter.

The detecting material in the albedo dosimeter can be any type of thermal neutron detector but in practical applications LiF thermoluminescent dosimeters (TLD) are most frequently used (11). The neutron detector mechanism involves ^{6}Li (n, alpha) ^{3}H reaction. The alpha particle and the triton are absorbed by the LiF detector with a deposition of the energy that can be detected by common TLD readout techniques. Natural Lithium TLDs are sensitive to thermal neutrons and this sensitivity can be increased by making the TLD out of Lithium enriched upto 95.62% of ^{6}Li. Similarly TLD enriched in ^{7}Li upto 99.93% is sensitive to gammas only. Both TLDs-enriched in ^{6}Li or ^{7}Li have comparable gamma sensitivities. When used in pairs, the reading from ^{7}LiF TLD gives gamma response and ^{6}LiF TLD gives neutron and gamma response and the difference between the two gives the neutron response of ^{6}LiF TLD. The basic albedo dosimeter design is shown in Fig. 3. It consists of a pair of ^{6}LiF and ^{7}LiF TLDs on each side of a cadmium disc. The TLD pair on the bottom is used to measure the albedo neutrons and the top pair is used to measure the incident thermal neutron fluence. The thermal neutron fluence response of ^{6}LiF (TLD-600) is 1500 times that of ^{7}LiF (TLD-700). There are nearly a dozen of albedo neutron

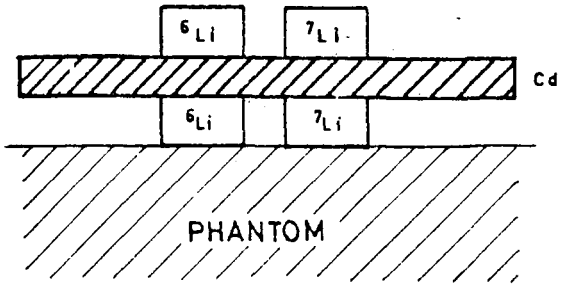

Fig. 3a Basic Albedo Dosimeter Design

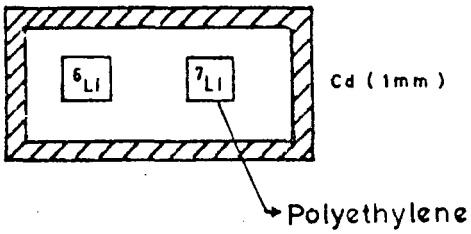

Fig. 3b Hankins-type Albedo Dosimeter

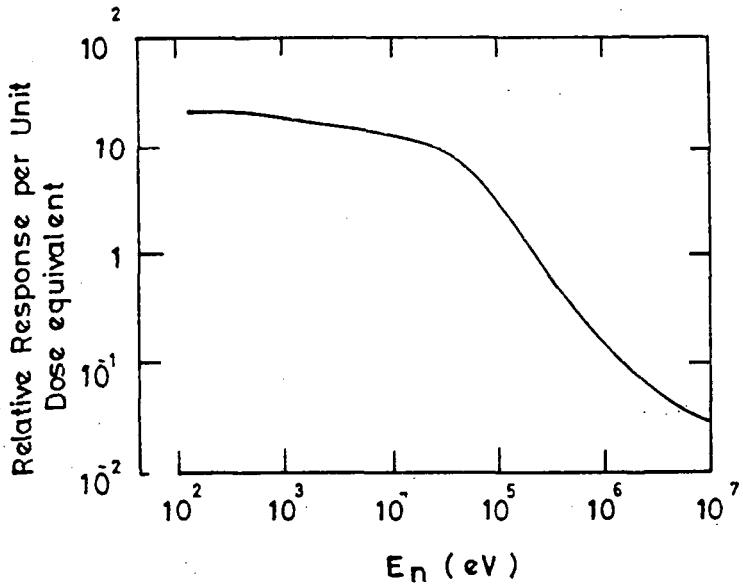

Fig. 4. Dose-equivalent Response of Hankins type Albedo Dosimeter

dosimeter designs described in literature using from one pair to three pairs of TLDs (24). Fig. 3b shows one of the earliest designs i.e. Hankins type albedo dosimeter. It has been found that the albedo response of all the dosimeter types are similar in energy response and differ mainly for thermal and epithermal neutrons. Fig.4 shows the response of Hankins type albedo neutron dosimeter (25). The large decrease of the albedo response curve beyond 10 keV shows the difficulty of applying albedo

dosimeters in personnel neutron dosimetry. It requires a very elaborate calibration in different neutron fields. The technique used is ratio of readings from two types of neutron detectors usually a nine inch polyethelene sphere to a three inch diameter polyethylene sphers covered with cadmium (4).

Albedo neutron dosimeters have high sensitivity and can be subjected to automation. Due to their gamma sensitivity, these can be used in fields where neutron to gamma ratio is high. These dosimeters show high neutron energy dependence and wide variations with detector to body distance and orientation. Albedo personnel neutron dosimeters are useful for personnel working in neutron fields with moderated spectra. A detailed discussion of albedo personnel neutron dosimeters is given by Piesch andBurgkhardt (26).

SSNTD FOR PERSONNEL NEUTRON DOSIMETRY

Solid State nuclear track detectors (SSNTD) have found application in neutron dosimetry for over more than fifteen years. Since neutrons themselves are uncharged, they can produce etchable tracks in detectors via the nuclear reactions involving the production of charged particles having significant kinetic energy. Therefore, SSNTDs for personnel neutron dosimetry can be broadly of two types (a) used with radiators placed in contact and registring fission fragments or charged particles emitted by these radiators or (b) directly registring recoil particles including alphas and protons produced within the detector itself from nuclear reactions occuring there due to incident neutrons. In recent years, it is second category of SSNTDs, which have shown promise for personnel neutron dosimetry. These are organic materials. The common constituents of these organic polymers are H, C, O and occasionally N and when irradiated with neutrons, elastic recoils of H, C, O and N give rise to tracks in the detector materials. Also non-elastic (n, p) and (n, alpha) reactions can occur and these protons and alpha particles can produce tracks depending upon the sensitivity of the polymer. A survey of current techniques for neutron dosimetry is provided by Spurny and Turek (27), Griffith et al (11) and Fleischer et al (28). Attempts were made to detect heavy elastic recoils of carbon and oxygen produced by neutrons, within the polycarbonate detector itself. It was found that cellulose nitrate has some sensitivity to protons but it was not until 1978 that the proton track detecting properties of the polymer CR-39 were discovered by Cartwright et al (29). The registration of protons in track detectors has long been recognised

as the basic requirement for a successful neutron personnel dosimeter and CR-39 offers the best prospect for personnel neutron dosimetry in the foreseeable future. The interest in detectors like polycarbonate or cellulose nitrate which do not detect protons is declining rapidly.

CR-39 SSNTD PERSONNEL NEUTRON DOSIMETER

CR-39 is a trade name where CR stands for Columbia Resin. It finds great applications as a major component in copolymers used for eye glass lenses and thus is an optically clear and commercially available plastic. Its chemical name is polyallyl diglycol carbonate (PADC) and its chemical formula is $C_{12}H_{18}O_7$.

CR-39 is proton-sensitive over a wide range of energies and is the best prospect for personnel neutron dosimetry due to the following three reasons (i) the (n, p) scattering cross-section is large and smoothly varying (ii) the efficiency of energy transfer from neutrons is the greatest for protons and (iii) the ranges of recoil protons are much greater than for alpha particles and heavier recoils. It also satisfies most of the criteria for an ideal polymer track detector. It is a highly cross-linked and totally amorphous polymer having a closely packed and uniform molecular structure. It is having a non-solvent chemical etchant being a thermoset material and is optically transparent. These properties indicate that potential exists for high sensitivity, a low energy neutron threshold and a good dose equivalent response for CR-39.

The damage tracks produced in the organic polymers on irradiation can be etched chemically or electrochemically to enlarge the tracks and make them detectable. A large number of studies of experimental etching parameters for the more common polymer detectors are available in the literature (30, 31). The influence of composition of the etchant (usually alkalies), its concentration and temperature, the degree of liquid agitation and various pre-etching treatments have been investigated. In spite of all these studies, the optimum etching conditions cannot be predicted and must be determined empirically for a specific detector. In practice, the most important parameters for the control of the etching speed of the detector are the temperature and the concentration of the etchant.

The charged particle tracks produced by the neutrons in the polymers are short and close to or below the limits of resolution of optical microscope. Electrochemical etching (ECE) makes it

possible to enlarge all types of damage tracks upto macroscopic sizes so that automatic counting or easy viewing becomes possible. ECE is obtained whenever damage tracks are stressed by AC electric fields during the chemical etching (32,33). For this type of etching, the irradiated polymer foil is sandwiched between two separate volumes of electrolyte of which at least one is etchant. The alternating high voltage is applied across the polymer foil by two electrodes which are immersed in the electrolyte solution. The ECE conditions often used are electric fields of the order of a few tens of KV per cm at frequencies of one to ten KHz. The electrical breakdowns at the tips of the tracks by applying electrical fields are induced locally and result in treeing phenomenon. The resulting tracks look like trees and are large enough to be viewed with a microfiche reader.

Electro-chemical etching involves the use of elaborate and special equipments namely multi-foil etching cell, high voltage programmable waveform generator and a microfiche reader or an image analyser system to view the electrochemically etched tracks and also an oven to carry out the electrochemical etching at an elevated temperature. CR-39 polymer of dosimetry grade is now available from Pershore Mouldings, UK and also in USA and Japan.

For the application of CR-39 SSNTD to personnel neutron dosimetry a knowledge of its various characteristics is essential. One of the most important characteristic is the neutron energy response of the dosimeter. In case of CR-39 dosimeter, the energy response depends upon the etching conditions used (34). Two sets of etching conditions are favoured (i) chemical etching at 60°c followed by electrochemical etching at room temperature and (ii) electrochemical etching at 60°c. Energy response obtained in both the cases is quite different. Griffith et al (35) used 5 hour chemical etching at 60° c in 6N KOH followed by 5 hour ECE at room temperature in 6N KOH, 2KHz and 31.5 KV/cm of CR-39 foil and the energy response to mono-energetic neutrons as measured by them is shown in Fig. 5. The threshold energy is 200 keV and the response peaks at 2.0 MeV and thereafter it falls. The chemical pre-etching step reduces the background. The second procedure of carrying out ECE at an elevated temperature of 60°c has been developed by Tomassino et al (36, 37). They have demonstrated that ECE at high temperatures reveals even tracks produced by lower energy protons as can be seen in Fig. 6. Because surface removal then occurs during ECE, more tracks are revealed while those already

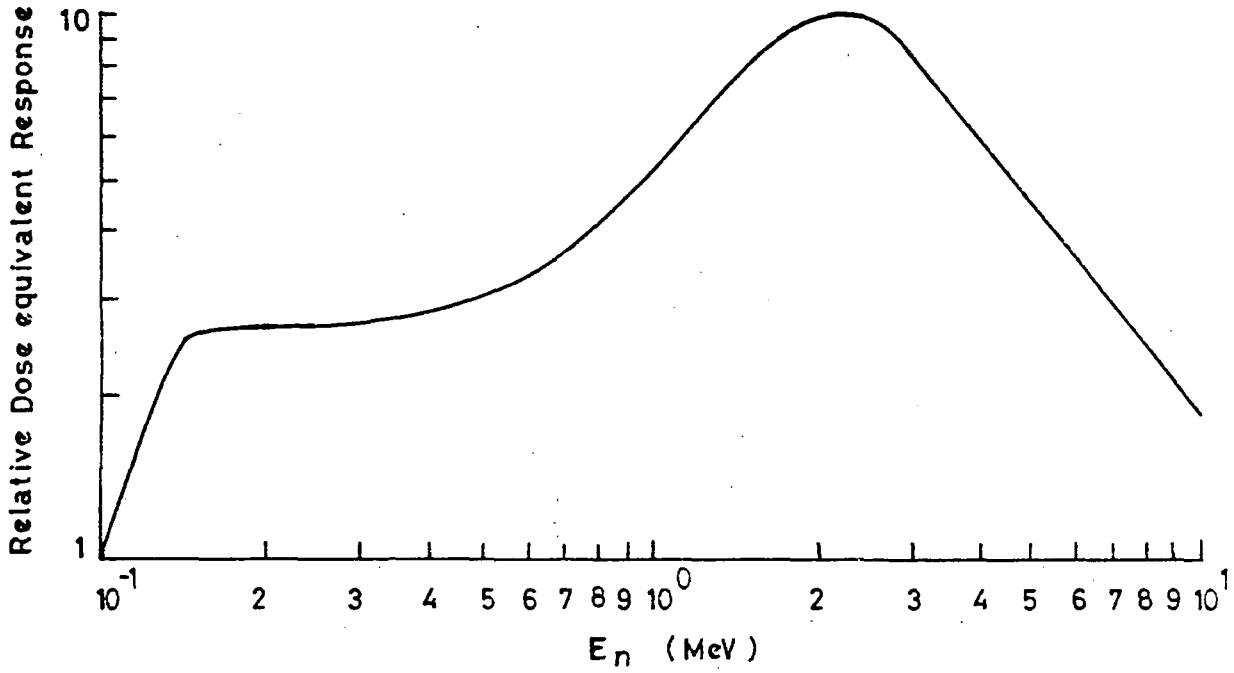

Fig. 5. Energy Response of Electrochemically etched CR-39 following Chemical pre-etching

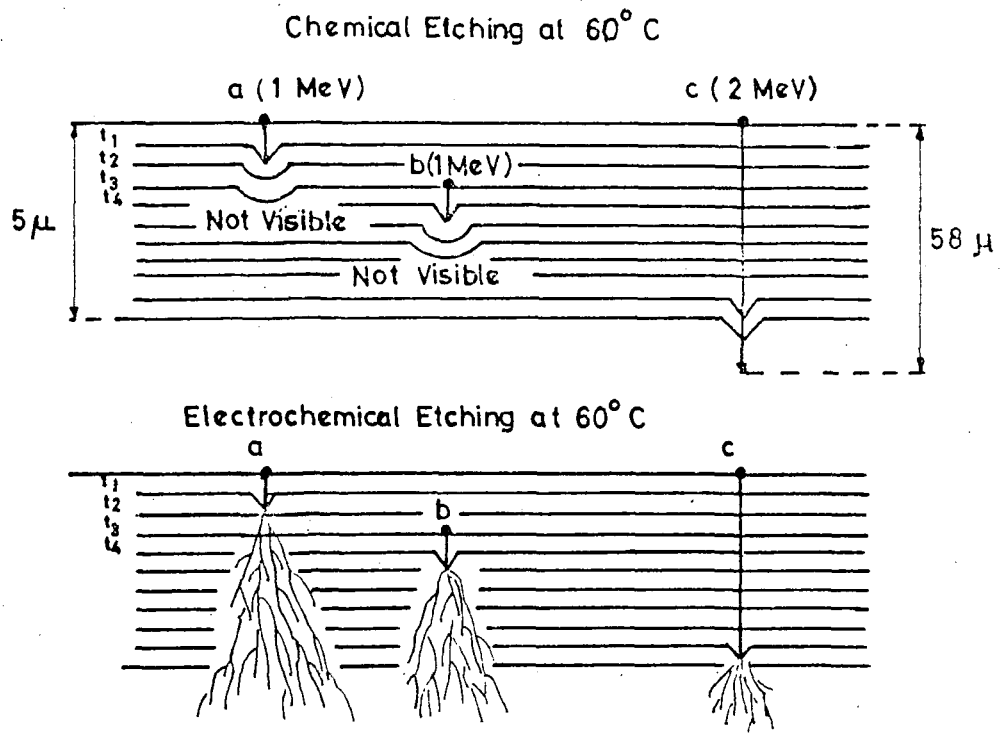

Fig. 6. Schematic diagrams for track formation at different etching times for Chemical and Electrochemical Processes

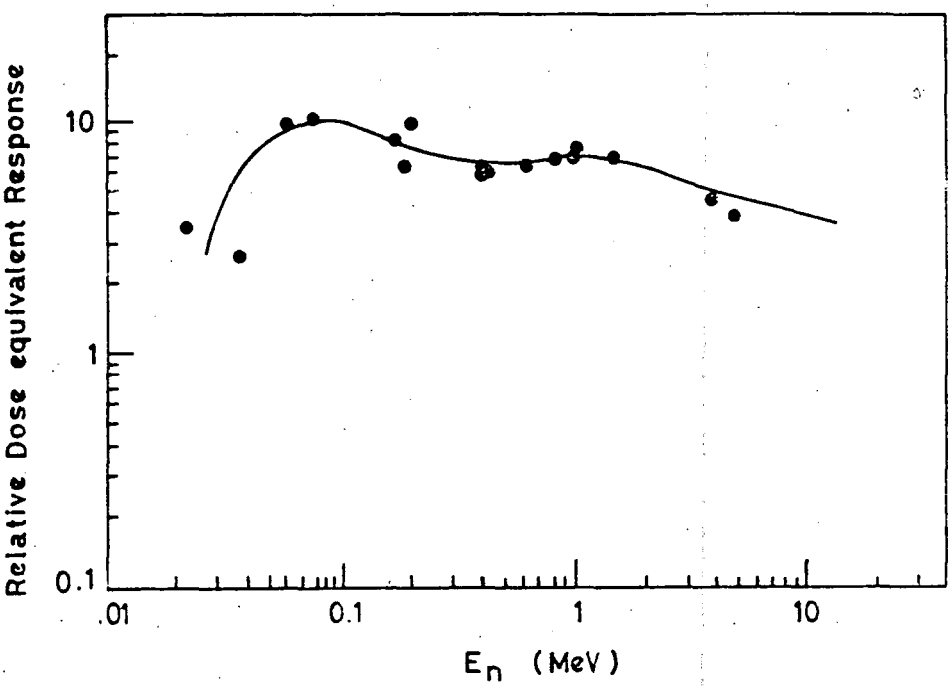

Fig. 7. Neutron Energy Response of CR-39 Monitor Etched only Electrochemically at 60 °C for 5 h with 30 kV cm^{-1} r.m.s at 50 Hz

treeing continue to enlarge. This has the effect of increasing the response to low and high energy neutrons more than to those around 2 MeV and therefore flattens the dose equivalent response and threshold is lowered to 100 KeV as shown in Fig. 7. Elevated temperature ECE (ETECE) is the preferred technique now and some laboratories in USA and UK are now using this dosimeter as a personnel neutron dosimeter routinely (38, 39). We have also developed this dosimeter in our laboratory and done studies with it (40, 41). Sensitivities of the order of 100 to 800 tracks cm^{-2} mSv^{-1} have been obtained depending upon the grade of the CR-39 material and etching conditions used. Thus it is a very sensitive dosimeter.

Advantages of CR-39 as personnel neutron dosimeter are its lower neutron energy threshold and better energy response. It is insensitive to beta, x-rays and gamma rays. There is no fading of tracks and counting is easy and can be automated. Disadvantages are its inherent background, dependence of sensitivity and energy response on etching conditions used, ageing of the foil i.e. decrease in sensitivity and increase in background with time. It is a brittle material and needs careful handling and large scale processing is tedious. However, because of its lower energy threshold, it is suitable for personnel neutron dosimetry in reactor

TABLE 3

A METHOD OF SELECTING THE APPROPRIATE DOSIMETRY SYSTEM (4)

Category	Dose equivalent rate $H_n + H_r$ (mSv/a)	(μSv/h)*	$\frac{H_n}{H_n+H_r}$	Neutron spectrum	Personnel monitoring for neutrons	Survey Instruments for neutrons
A	< 1.5	< 0.75	< 1	Variable	Gamma-ray dosimeter	DE meter
B	< 15	< 7.5	< 0.2	Variable	Gamma-ray dosimeter	DE meter
C	< 15	< 7.5	> 0.2	Constant	Simple albedo or track-etch detector	DE meter sphere ratio
D	< 50	< 25	> 0.2	Variable, $\bar{E}_n$ < 1MeV, correlation with ratio e.g. H_{th}/H_{albedo} or H_{th}/H_f	Analyser albedo or track-etch detector system	DE meter, sphere ratio, thermal detector
E	< 50	< 25	> 0.2	Variable $\bar{E}_n$ < 1MeV, no correlation	Recoil track etch detector and analyser albedo	DE meter and multi-sphere ratio or single sphere albedo technique
F	< 50	< 25	> 0.2	Variable, 1 < $\bar{E}n$ < 20 MeV	NTA film or track-etch detector	DE meter, recoil proportional counter.
G	< 50	< 25	> 0.2	Variable $\bar{E}n$ > 20 MeV	Nuclear emulsion, activation threshold detectors	Ion chamber.

*Assuming 2000 working hours/annum.

environments and fuel reprocessing facilities. The development problems associated with implementation of CR-39 dosimetry are magnified by the small number needed for neutron dosimetry and its development cost to banefit ratio is relatively high (42, 43).

CONCLUSION

Personnel neutron dosimetry must meet severe requirements. No currently available dosimeter can meet all of these requirements. A manual prepared by IAEA (4) provides a method of selection of the appropriate personnel neutron dosimetry system based on the neutron dose equivalent rate in the working environment and the energy of neutrons to which a person is likely to be exposed. Table 3 from the above manual presents the neutron field categories and criteria for selection of the appropriate personnel neutron dosimetry system. Some general and useful references in the field of Neutron Dosimetry are also given (44, 45, 46, 47).

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