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Sensitivity Study of PADC Track Detector with External Radiators

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Abstract: The aim of this study was to investigate the sensitivity and reliability of PADC as a neutron dosemeter suitable for large scale routine personnel monitor. A personal neutron dosemeter is being developed in our laboratory by using PADC as a detector and hydrogenated materials as proton converters for fast neutron detection. To increase the PADC response and sensitivity to thermal neutrons, boron converter was added to the dosemeter. Several dosemeters with these configurations were mounted on a water-filled phantom and exposed normal to: ^{241}Am-Be, ^{241}Am-Be softened with a 20 cm radius of polyethylene sphere moderator, ^{252}Cf neutron spectra. The irradiated detectors were electrochemically etched and evaluated in order to determine their dose equivalent response in terms of H (10, α). The results obtained were compared to those obtained from Monte Carlo simulations using the MCNPX code in order to improve, the sensitivity of the PADC response. The detector configuration can be adapted to neutron spectra in different practical situations. It would be able to provide dose equivalent response in a large range of energies.

Key words: Neutron dosimetery, PADC etched track, MCNPX, External radiator, personal dosimetery, simulation, boron converter

INTRODUCTION

Around the reactors and particle accelerators which used in research, industry and radiation therapy, there is possibility to detect thermal neutrons as secondary radiation, which is produced in the interactions with the structural materials of the accelerator and/ or with targets, beam dumps and collimators influence the shielding design. For low and intermediate energy accelerators, neutrons are the particles that contribute mainly to these secondary radiation fields (Silari, 2001). Thermal neutrons can be detected via (n,α) reactions in ^{10}B, because low energy α particles ≤2 MeV could be recorded. The Poly-Allyl Diglycol Carbonate (PADC) detector is of particular interest for the development of a fast neutron dosemeter (Griffith *et al*., 1981). Neutron elastic interactions with CR-39 plastic leave latent recoil charged particle tracks which can be brought out by chemical (EC) or electrochemical etching (ECE). The track density and the geometrical parameters of tracks depend on the track formation process, which is strongly determined by the type of particles and the etching conditions applied to develop the latent tracks (Tommasino and Harrison, 2004). Above a threshold of ~7.9 MeV, inelastic alpha particle breakup of ^{12}C is also possible with the energy lost by the neutron partitioned among the three alpha particles (Fleischer *et al*., 1965). Fast neutrons interact with the constituents of the CR-39 detector and produce H, C and O recoils as well as (neutron (n), alpha (α)) reactions. These neutron- induced charged particles contribute towards the response of CR-39 detectors (Khan *et al*., 2000). Study of its application to neutron dosimetry has shown that this material can be used in personnel neutron dosimetry. Using combination of chemical pre-etching and electrochemical etching may lead to detect low energy neutrons (Azimi-Garakani *et al*., 1987). So there is the need to apply especial electrochemical etching condition. ECE in combination with automatic image processing systems is widely used in neutron dosimetry (Tommasino *et al*., 1984). The size and characteristics of the tracks depend upon the charged particle mass, energy and direction. Energy dependence has been reported previously on the distribution of track sizes for monoenergetic neutrons (Hankins and Westermark, 1987) and for broad spectrum

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neutrons (Jakes *et al.*, 1997). Phillips *et al.* (2006) have investigated the dependence of the track density with energy, fluence and with direction. PADC in contact with boron and polyethylene have been studied to provide a thermal sensitive neutron detector. The aim of this research is to study the high sensitivity and reliability of PADC as a neutron dosemeter suitable for large scale routine personnel monitor. This study has been performed with new configuration of the thermal converter, which were simulated by MCNPX code (MCNPX, 2002) and validated with a series of irradiations to realistic neutron fields.

MATERIALS AND METHODS

The study was conducted from Sep - 2008 to May - 2010 in our neutron dosimetry laboratory.

Dosemeter arrangement: It is composed, following the incidence direction of the neutron beam, of a 3 mm thick layer of polyethylene acting as fast neutron converter via elastic scattering with hydrogen, on a 1 mm thick layer of PADC used as a detector and on a 5 mm thick methacrylate holder. To increase the sensitivity of this configuration to thermal neutrons, 250 μm of boron layer was added next to the polyethylene layer. The cadmium absorber with 250 μm thick was added rear surface of the PADC. The whole configuration can be seen in Fig. 1.

Neutron irradiation: Thirty irradiation cards of dosemeter approximately 1.8×1.8 cm^2 were irradiated for ^{241}Am-Be, ^{241}Am-Be softened with a 20 cm radius of polyethylene sphere moderator and ^{252}Cf neutron spectra. Irradiations were performed with the dosemeters mounted on water-filled phantoms of 15×30×30 cm^3 at normal neutron incidence.

Processing and reading: The PADC plates were etched using an electrochemical etching in optimized condition as specified in Table 1. The etched PADC detectors were digitized using a high resolution scanner (2000 dpi) with transparency adaptor (UMAX scanner power look III), which is connected to a PC-based image analyzer and counting program. The program has been developed in MATLAB 6.5.1 environment (Taheri, 2005). After implementing the image processing, the tracks were automatically counted inside the Region of Interest (ROI) and the track density was calculated. The ROI includes all the tracks inside the scanning area (3 cm^2) whose major axis ranges greater than 20 μm. The background track density was measured over 18 samples and the average value obtained was 40±5 cm^{-2}.

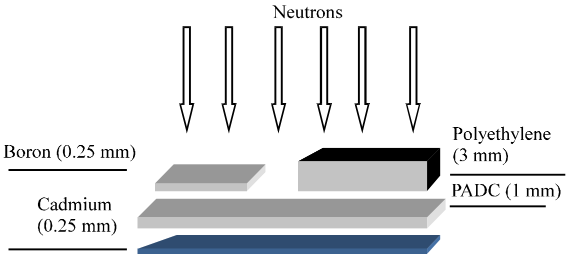

Fig. 1: Arrangement of the dosemeter configuration

Table 1: The four steps of the electrochemical etching applied

Step	Chemical solution	Time	Temperature	Field strength	Frequency
Pre-etching	6M KOH	1 h	60°C	-	-
2nd step	6M KOH	4 h	60°C	15 kV cm^{-1}	50 Hz
3rd step	6M KOH	1 h	60°C	15 kV cm^{-1}	2000 Hz
Post-etching	6M KOH	10 min	60°C	-	-
Final	Tap water	10 min	-	-	-

Monte carlo simulations: The response function of detector with each converter was simulated with the MCNPX code. The code has been used to calculate the number of protons crossing the front detector surface and classify them according to their energy and angle of incidence. For this purpose, the Particle Track Output Card (PTRAC) was used for calculating recoils for the whole history. The PTRAC generates an output file of user-filtered particle events. This option of MCNPX code registers the whole history, such as interaction type, energy, direction and position of particles. The simulated source was a broad parallel beam of monoenergetic neutrons normal to detector surface with 33 energies ranging from 0.025 eV to 30 MeV. For all calculations, the code was run for 10^7 histories.

RESULTS AND DISCUSSION

To simulate the response of dosemeter configuration, the MCNPX code (MCNPX, 2002) has been used to calculate the number of protons, carbons and oxygen crossing the front detector surface and produce in the PADC. To simulate the real condition, recoils classify according to their energy and angle of incidence. Improvement on thermal neutron up-scatter in polyethylene is performed by using the S (α, β) model treatment (MCNPX, 2002). To improve the sensitivity of detector, boron layer has been used in contact with front surface of the PADC as shown in Fig. 1. The detector response is obtained by considering protons, carbon, oxygen recoils and a particles with incidence angles smaller than the critical angle of 45° and with energies within the PADC energy windows response 0.1-2.4 MeV (Fernandez *et al.*, 1996). The simulated response function of the PADC is compared with the result which is presented in the IAEA technical reports series No. 318 (IAEA, 1990). The shape of the responses in Fig. 2 shows a good agreement between simulation and the results have been processed electrochemically (IAEA, 1990). The experimental and calculated average responses for all neutron sources which have been used in this study also show good agreement. The response function curves of the PADC in contact with polyethylene and boron convertor are plotted in Fig. 3 and 4, respectively. For each shape, the response is simulated with MCNPX shows that the sensitivity of the detector to epithermal and thermal neutron spectra is increased. These figures shows that the sensitivity of the detector for low energies of the incident neutron is increased in compare to results presented (Phillips *et al*., 2006; IAEA, 1990). The Y axis of the Fig. 2 to 4 are expressed in arbitrary units, but the magnitude of the units is the same for all graphs. The integrated PADC detector with different converter which has been presented in this work shows good improvement of the PADC response to thermal neutron. The code can be used to define the most adequate dosemeter configuration adapted to neutron spectra in practical situations encountered at protection level. In order to compare the simulated result with experiment, the energy used was the same as that for the experimental irradiations.

The experimental neutron response in terms of H_p (10, α) has been evaluated from the net measured track densities (measured track density minus average background) taking into account the corresponding reference values of each source corrected for the effect of source-to-detector distance. The experimental and simulated dose equivalent response and normalized per

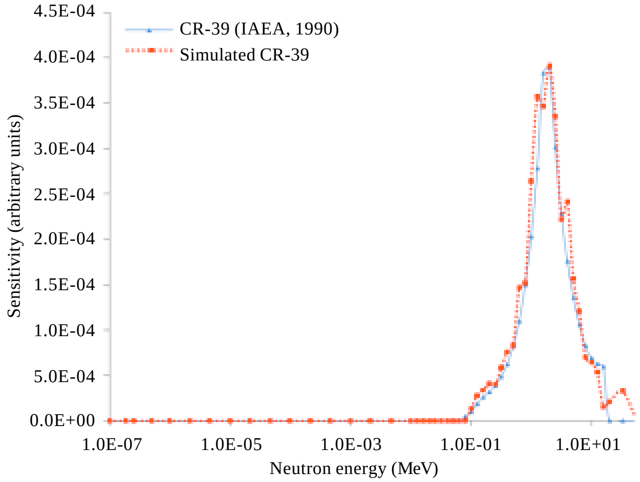

Fig. 2: Comparison of the response function curve of the PADC detector

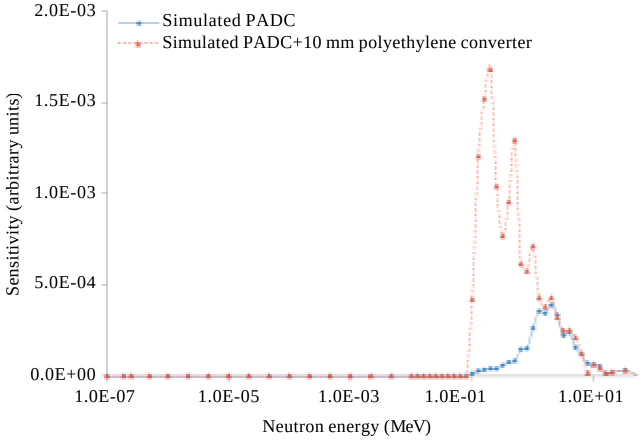

Fig. 3: Simulated response function curve of the PADC + Polyethylene converter

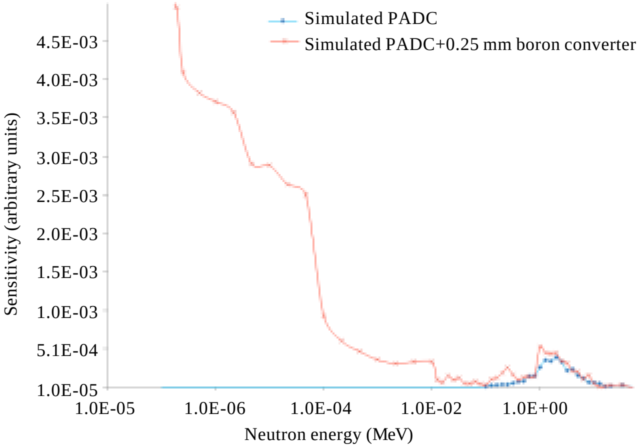

Fig. 4: Simulated response function curve of the PADC + Boron converter

Table 2: Experimental and simulated dose equivalent response and normalized per unit personal equivalent dose

Source	Experimental average response (cm^{-2} mSv^{-1})	Calculated average response (cm^{-2} mSv^{-1})	Dosemeter reading per unit H_p (10,α)
Am-Be	485±29	563±37	0.79
Am-Be with 20 cm Polyethylene	572±45	627±41	0.87
^{252}Cf	581±41	630±48	0.91

unit personal equivalent dose at normal incidence and for the calibration sources are shown in Table 2, which also displays the uncertainties corresponding to one standard deviation. A very good agreement can be seen between the simulated and the experimental results of the dosemeter configuration for all sources. The response value is the lowest for ^{241}Am-Be and increases as energy of neutrons moderated and for ^{252}Cf was reached to the highest value. This fact can be attributed to a detection efficiency increase when the energy of the source decreases. In order to use this dosemeter in routine measurements, a calibration factor, (2.7±0.9) μSv.cm^2, for all sources and neutron incidences has been calculated as the reciprocal of the mean dose equivalent response to 0° neutron incidence. The percent deviations between the values obtained using the calibration factor and real doses in all the cases range from +49% and -11%, which fulfill the International Atomic Energy Agency (IAEA) requirement. The corresponding limit (-33%) was recommended by the IAEA (1999). The results of detector configuration which is presented in this study, is comparable with the data presented by Garcia *et al.* (2005). The minimum detectable dose equivalent (MDDE) is defined with a level of confidence of 97.5% according to Harvey *et al.* (1998). A mean MDDE value of 62±25 μSv has been found for all sources and neutron incidence with 0° angle. This value is smaller than the limit of 80 μSv recommended by the International Commission on Radiological Protection (Annals of the ICRP 60, 1990).

CONCLUSIONS

Monte Carlo simulations showed that the shape of the response function of the PADC is very similar to that is presented in IAEA Safety Guides, No. RS-G-1.3. The small discrepancies can be ascribed to actual critical angle and geometrical effects. From the results of this work, it can be stated that the MCNPX computer code reproduces the experimental results with good agreement. The code can also be used to define the most adequate dosemeter configuration for thermal and intermediate neutrons if the cross sections of the appropriate nuclear reactions are introduced in the code. The detector configuration can be adapted to neutron spectra in different practical situations. It would be able to provide dose equivalent response in a large range of energies, as well as for intermediate neutrons.

Deviations of the dose equivalent evaluated using the mean calibration factor obtained in this work from its true value range from -11 to +42%, which fulfill the IAEA requirement. The minimum detectable dose equivalent of the dosemeter which is presented in this work has been found to be 62±25 μSv, which is smaller than the 80 μSv limit recommended by the ICRP 60. This good value is probably due to having the PADC with good condition, converter and electrochemical etching method, which is known as an optimized technique.

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