A passive method for absolute dose evaluation of photoneutrons in radiotherapy

K.R. Rajesh[1*], R. Ganapathi Raman[1], M.M. Musthafa[2], C.V. Midhun[2], N. Joseph[2]

[1]*Department of Physics, Noorul Islam Centre for Higher Education,Kumaracoil,Thuckalay,TamilNadu India*
[2]*Department of Physics, University of Calicut, Kerala, India*

► *Technical note*

***Corresponding authors:**
Kalian R. Rajesh, Ms.C.,
E-mail:
rajeshkaliyan77@gmail.com

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ABSTRACT

Background: Photoneutrons are produced during the radiotherapy treatment, when high energy X rays interacts with structures of the head of the linear accelerator (linac). The present day TPS are not taking into account the photo-neutron dose and the biological effects associated with it. The late induction of cancer and recurrence of the disease in old cancer patients are being frequently reported. ***Materials and Methods:*** Patients undergoing radiotherapy treatment with 15 MV X rays from a Siemens Primus Plus linear accelerator was considered for the study. In the present work, photoneutron spectrum from the linac head is measured using CR 39 SSNTD and the corresponding dose is calculated using Geant4. The composite photoneutron spectrum from the linac head and the corresponding dose was calculated using the kerma evaluation method in a human equivalent tissue phantom. The repeated calculation outcomes and the covariance error analysis in the nuclear data give consistency and an accuracy of 2 % in the results. ***Results:*** The result shows that significant amount of photoneutron dose was deposited during radiotherapy treatment when high energy X rays are used. The photoneutron production from the patient itself is yet another major issue which will cause out off field dose. ***Conclusion:*** This work gives importance in considering the photoneutron dose during radiotherapy planning and protection. This extra dose might be a factor that contributes to the induction of cancer and also to the recurrence of cancer to previously cured patients.

Keywords: *Medical linear accelerator, CR 39, photoneutron tracks, cancer treatment, Geant4.*

INTRODUCTION

High energy medical linear accelerators produce significant amount of photoneutrons along with the primary photon beam through the photonuclear interactions [(1)]. The photoneutrons can be produced by the interaction of high energy bremsstrahlung photons with the treatment head (collimators, flattening filter (FF), Beryllium window if any, cooling water etc) and even from the patient itself. The neutrons interacts with major isotopes present in the human body through different reaction channels such as (n, elastic), (n,g), (n,a), (n,p) etc and therefore deposit an additional dose of photoneutrons in the patient. This dose has to be quantified accurately in order to minimise the unwanted exposure of the patients.

The behaviour of the neutron spectrum produced by the photonuclear interaction inside the patient is unpredictable because of the complexity in human composition. This leads to failure of the relative neutron dosimetry, normalized to fission neutron spectrum. The neutron dose is a function of interaction cross-section and Q value. The neutron dosimetry is highly sensitive to neutron energy. Therefore an accurate knowledge on neutron spectrum is essential for accounting these parameters and determines the dose with high precision. The spectrum based analysis can be performed through the kerma based approach which involves large number of matrices of nuclear data that cannot be done manually. Hence a montecarlo based approach is used for dose estimation using Geant4 (2).Though neutron detectors like Bonner sphere, ^{3}He proportional counters, which are inactive for photons, can provide quantitative information about the neutrons produced, they fail to give qualitative information. Due to the higher intensities of primary photon, the ordinary neutron spectroscopy method such as liquid scintillator based techniques, also, usually fails. Presently available method for neutron spectroscopy is the Time of Flight (TOF) method which is intricate and not suitable for routine measurements. Spectrum unfolding method also fails due to the presence of large photon flux (3). All these factors lead to the importance of passive detectors for neutron spectroscopy. CR-39 SSNTD is well-established methods for the neutron detection, basically in bulk etch modes. Generally bulk etch method gives information regarding track density rather than information on neutron energy (4, 5, 6).

Siji *et al.* measured the photo-neutron dose equivalents for different depths in Phantom using CR 39 (7).The track density data was used to calculate the dose. The fast neutron dose equivalent measured at the phantom surface was 0.57 mSv/Gy.Cheol-Soo Park *et al.* measured the neutron dose using CR 39 from a medical linear accelerator inside the 10 × 10 field size for 1 Gy, 2 Gy and 5 Gy of photon irradiation (8). According to the experimental results, 0.35 mSv, 0.65 mSv, 1.82 mSv of fast neutrons on average were generated from 1 Gy, 2 Gy and 5 Gy of photon irradiation, respectively. The number of tracks generated per unit area was calculated using an automated counter with a microscope or microfilm reader, which aimed to determine the neutron dose.

Since neutron dose is energy sensitive, bulk etch rate will not give accurate information on neutron dosimetry. With this perspective, the current method focuses on the spectrum based neutron kerma evaluation, accounting the propagated uncertainties from benchmarked nuclear data sets.

MATERIALS AND METHODS

Neutron KERMA measurement

The most accepted method for the measurement of neutron dose deposition is through the neutron kerma (9, 10). So in a neutron dosimeter, the dose deposited on it can be expressed as shown in equation 1:

$$K = 1.602 \times 10^{-8} \sum_{ij} N_i m^{-1} \int_0^{Emax} \Phi(E)\sigma_{ij}(E)[E_{tr}]_{ij} dE \qquad (1)$$

Here '*i*' refers to the type of material and 'j' for the reaction channel. $N_i m^{-1}$ refers to the number of '*i*' type of atoms per gram of the material sample. Ø(E) will be the fluence matrix,σ_{ij}(E)is the cross section matrix for '*j*' th ejectile from '*i*' th target and $[E_{tr}]_{ij}$ is the energy released in the medium, corresponding to the particular reaction channel. Since the radiative loss component is absent in the neutron, the kerma can be directly equated as the dose. In practice, a neutron detector records the matrix Ø(E) and calculates the kerma using equation 2:

$$K = \sum_E \Phi(E) F_n(E) \qquad (2)$$

Where F_n(E) is the kerma factor tabulated. Through this method, the kerma from detector system can be easily converted to the dose in the interesting system.

Experimental setup

250 micrometers thick CR 39 SSNTD films with an area of 1 cm^2 and physical density of 1.3 g/cc have been used for neutron spectrum acquisition. CR- 39 SSNTD film was placed on the patient in the field centre during the treatment with 15 MV X rays from Siemens Primus Plus medical linac. The irradiation set up is shown in figure 1. After the treatment, the

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CR-39 film was taken for processing. The photo-neutrons create tracks in the CR-39 film by recoil protons. The film was then chemically etched at standard etching conditions, such as 6N NaOH solution at a constant temperature of 60^0C. The etching time was 6 hours with a constant stirring of 30 rotations per minute. The image of the chemically etched film was obtained from an optical microscope with a magnification of 40X. This is recorded with an acceptable resolution of 250keV bin. The image of the etched film is shown in figure 2.

The image was then analyzed using the program, TRIAC-2, a Mat Lab programme based on the Hough transform method [11]. The length of the major axis, minor axis, orientation and brightness of each elliptical track in the image was obtained. The recoiled proton energy corresponding to each track was calculated from the track diameter by using a calibration graph. The calibration graph data was taken from the combined work of Matiullah Tufail M *et al.*, N. Sinenian *et al.* and Antony Joseph [12, 13, 14]. The equation to best fit data is calculated and is given by $E_p = (0.0295956*(d^2)) + (-1.16821*d) +12.674$, where d is the track diameter. Then the corresponding neutron energy is calculated using the equation $E_P = E_N\cos^2\Theta$, where E_p is the energy of the proton, evaluated in the previous step, E_n is the neutron energy and Θ is the recoil angle.

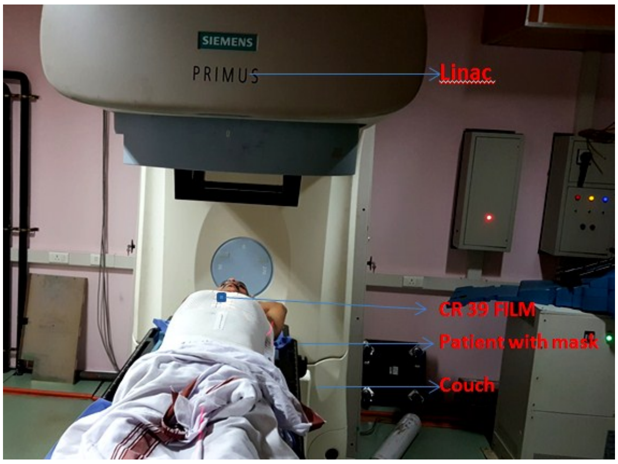

Figure 1. Irradiation set up of Siemens Primus Plus medical linac.

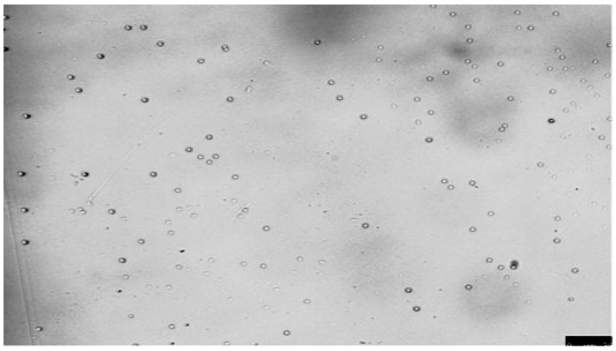

Figure 2. Tracts in etched film after irradiation.

Spectroscopy

The neutron energy is binned with 250 KeV and the average energy for each bin is calculated. The number of photo-neutrons for each average energy is counted and corrected with the efficiency calculated using equation 3:

$$Y=n\sigma\varphi \tag{3}$$

Here n is the number of hydrogen atoms present in a $1cm^2$ unit of CR39 film, σ is the cross section for (γ, n) reaction obtained from EXFOR ENDF, φ is the efficiency corrected number of photo-neutrons, from the elastic scattering cross -section, and Y is the number of tracks formed in the CR39 film in 1 cm^2 area. The spectrum is then plotted with energy along X-axis and corrected counts along Y-axis. The photoneutron spectrum is shown in figure 3.

This spectrum is then loaded into Geant4 for the further dose evaluation. The Geant4 simulation programme consists of total Monte Carlo simulation codes with kerma based dose calculation equations and an ICRU-A-150 tissue equivalent spherical phantom with 30cm diameter [15, 16]. The programme will account for the whole behaviour of the spectrum with different reaction channels like (n, n), (n, 2n), (n, alpha), (n, elastic) etc in the ICRU-A-150 phantom. The kerma and hence the dose for

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each energy and intensity of photoneutron was calculated separately. Then the integral dose was calculated for the whole spectrum. Covariance error analysis is considered to reduce the error in nuclear data. About 40 patients undergoing radiotherapy treatment for the pelvis case (ca prostate, ca cervix) treated with 15 MV X rays from a Siemens Primus Machine were examined. Photoneutron dose calculated for about 10 patients is given in table 1, which describes the minimum to maximum measured. The calculation is repeated in 40 patients for consistency. The value will be almost same because the percentage of photoneutron production from the treatment head depends on the type of material present, interaction crossection, neutron separation energy etc. Some differences exist only among different accelerator models. Average Photoneutron energy ($\frac{\int_{E=0}^{Emax}\Phi E dE}{\int_{E=0}^{Emax}\Phi dE}$) was found to be around 2 MeV by considering 40 patients.

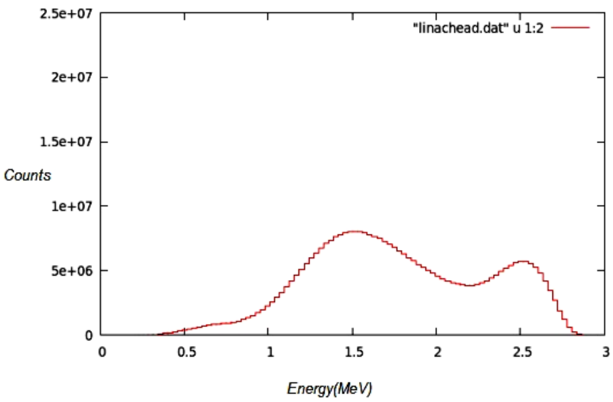

Figure 3. Resultant photoneutron spectrum.

Table 1. Photoneutron dose for patients

Sl.No	Position	Dose (mSv/Gy)
1	Anterior on field	0.177
2		0.34
3		0.55
4		0.86
5		0.64
6		0.44
7		0.51
8		0.36
9		0.69
10		0.66

RESULTS

The image of the CR39 SSNTD shows a very large number of tracks. The TRIAC II programme is very much effective in terms of the time consumed for counting. The neutron energy corresponding to each track is calculated and the photoneutron spectrum was measured. Each energy bin and corresponding counts are given as an input to the GEANT4 and the corresponding dose was calculated in ICRU A 150 phantom using montecarlo simulations. Then the integral dose was calculated for the whole spectrum. Since the neutron dose is very much dependent on energy, the spectroscopic information gives accuracy in dose estimation. The photo-neutron spectrum clearly shows the heterogeneous photo-neutron energy produced during the treatment. The incident bremsstrahlung photons have energy ranging from 0 to 15 MeV. The spectrum shows energy range from 0.7 MeV to 2.6 MeV. The tracks due to scattered and secondary neutrons having low energy (<0.7 MeV) are lost on etching. This leads to loss of information about the low energy of neutrons produced. Alternative mechanism is to be employed for recording low energy primary neutrons. The average equivalent dose calculated was 0.53 mSv/Gy, by assigning a quality factor of 10 for neutrons of energy up to 2 MeV and 20 for above 2 MeV. This dose includes the contribution from the head only. The thermal neutron energy part and the neutron production from the patient were not considered here. Therefore, the inclusion of all contributions will result in a significant amount of photoneutron dose for a prescription dose of 50 Gy or more considering the high LET nature of neutrons [17, 18, 19].

The materials used in the collimators, shielding blocks, flattening filter (lead, tungsten etc), cooling water assembly etc are the sources of photoneutrons from the treatment head. The production channels responsible for the photoneutron production can be identified using En = *Eresonance* - *Qreaction* - *Erecoil* where En,

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Eresonance, Qreaction and Erecoil are energy of neutron, photon resonance energy, Q value for the reaction and the recoil energy of the residual nucleus respectively. Q-value and recoil energy for each reactions are calculated using NNDC Q-value calculator with Eresonance taken from EXFOR-ENDF data library, as an input. Contributing neutrons due to lead and tungsten are visible in the form of broadening of peaks from 1 MeV to 2.6 MeV. The yield of neutrons due to each channel depends upon natural abundance of the isotope and the magnitude of cross section at each energy.

DISCUSSION

There are a lot of works related to photoneutron dosimetry using CR 39 SSNTD in radiotherapy treatment. The main drawback associated with neutron dosimetry is that the real interaction of photoneutron with the patient will be different from the detector media. The real time estimation of photoneutron transport and the dose deposition in the human body is difficult to measure. Present CR 39 SSNTD uses passive methods in measuring photoneutron dose based on bulk etching. The average dose is measured here and the spectroscopic information is missing. The photoneutron dose measured by Siji *et al.* and Park *et al.* is based on the neutron track density rather than the spectroscopic information [7, 8]. Since the neutron produced equivalent dose is very much dependent upon energy, the spectroscopic based integral dose can give accurate results. The photonuclear reaction channels cannot be identified in bulk etching method.

The presently available calculation algorithms used in the treatment planning systems do not consider the photoneutron dose and the associated biological effects. Even small amount of neutron dose can cause severe biological effects due to its high LET nature. Even the critical organs away from the treatment site can receive significant amount of dose due to the neutrons produced from the patient and this has already been reported in several literatures. The induced activity and the production of short lived radioactive isotopes due to photoneutron can cause severe health hazards.

CONCLUSION

The photoneutron dose is significant in radiotherapy treatment and it has to be accounted with great importance in order to avoid the late effects. The dose calculation algorithms employed in the treatment planning systems have to be programmed to include the photoneutron contribution. The passive method developed using CR 39 SSNTD based on the photoneutron spectrum information accurately measures the dose and it also makes the reaction channel identification possible which is very much informative in designing the linac head in order to reduce the photoneutron contribution.

Conflicts of interest*: *Declared none.

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