Neutron Spectrum Measurement in a Medical Cyclotron

V. Sathian1, Deepa. S.2, U.V. Phadnis1, P.S. Soni3 and D.N. Sharma1
1Radiation Safety Systems Division, 2Radiological Physics and Advisory Division 3Lab Nuclear Medicine Section
Bhabha Atomic Research Centre (BARC) Trombay, Mumbai, India, 400 085,

Abstract

Radio nuclides are being used in diagnostic purpose as well as therapeutic purpose under medical category for early detection of malignancies. The diagnosis using radio nuclides are playing very important role. PET scan is a milestones in medical history. These scan is done using a positron emitter $^{18}$F. This radio nuclide is produced by the reaction $^{18}$O(p, n)$^{18}$F in Medical cyclotron of Radiation Medicine center (RMC), India. The neutron production is an unwanted by product. These neutrons will make lot of radiation protection problem as neutron itself and also these neutrons will produce induced activity in the different elements of cyclotron shielding material as well as the wall materials in long run. So the accurate neutron dose measurement is very important and this depends on the accurate neutron spectrum. The unfolding of neutron spectrum and measurement of neutron dose in cyclotron of RMC using activation foil technique is presented in this paper.

KEYWORDS: Medical Cyclotron, Neutron Spectrum, Activation Foils, Positron Emission Tomography (PET), Reaction Cross-Section, Induced Activity

1.Introduction:

High energy accelerators are being used more and more in different applications like Accelerator Driven Systems (ADS), spallation neutron sources, intense synchrotron radiation sources, for basic research in high energy physics and also in medical applications. In these accelerator facilities, neutrons constitute the major radiation field along with gammas. Depending upon the beam energy, these neutrons span a very wide energy range starting right from thermal energies going up to several hundreds of MeV. The origins of these neutrons are different for different accelerated particles. In electron accelerators above 10 MeV, mostly photo neutrons are produced by the bremsstrahlung photons generated through electron beam interaction with high Z materials. In high energy proton accelerators the major contribution comes through spallation neutrons. With the growing advent of high energy accelerators to be used in different research and applications, assessment of radiation environment in these facilities has opened an entirely new area of research since it is completely different from the conventional radiation safety. In this paper, we have discussed the neutron measurement in a high energy medical cyclotron accelerator used for Positron Emission Tomography (PET) and Single Photon Emission Computed Tomography (SPECT) Cyclotron at Radiation Medicine Centre (RMC), India, using foil activation technique. The cyclotron at RMC, India is a vertical one with horizontal beam direction and accelerates negatively charged hydrogen ions to 16.5 MeV or negatively charged deuterium ions to 8.4 MeV energies. The beam current is 75μA for protons in single port and 80μA in dual port. For deuteron the maximum current is 60μA. This is equipped with 6 targets at fixed locations where radio nuclides C-11, N-13, O-15 and F-18 are produced. Neutrons are generated when proton beam hits the target or container due to (p, xn) reactions[1-3]. Among the various targets $^{18}$O(p, n)$^{18}$F gives the highest neutron yield. Al is
used as a container for the targets. In the present study, the neutron spectrum and hence the neutron dose has been measured at 5 cm from the target at two different angles (0 degree and 90 degree). The neutron spectrum measurement has been carried out during the normal operation with $^{18}\text{O}(p, n)^{18}\text{F}$ reaction using various activation foils such as Au, Cu, Rh, In, S, Ni, Mg, Al etc. and also these foils under cadmium cover. Different reactions in the same foil also was used. These foils were irradiated for one hour during the normal operation of the machine and then removed after four hours of cooling time. The induced activity was measured using the $4\pi\beta-\gamma$ coincidence system, the primary standard for activity measurement and an HP(Ge) system. These activities were used to unfold the neutron spectrum using SAND11 code. The unfolded spectra was used to convert in to the neutron dose using the fluence to dose equivalent conversion factor averaged over the spectrum.

2. Materials & Methods

Activation detectors are widely used for fluence rate and dose measurements in intense neutron fields. The activation method of neutron detection is based on the fact that many elements become radioactive when exposed to a neutron flux. Normally Gold and copper were used for the thermal neutron dose measurement and Indium, Sulphur, Nickel, Magnesium, Aluminum foils were used for fast neutron dose measurement when the neutron spectrum is known. In this case a spectrum averaged cross section is used for finding the neutron fluence rate. But when the neutron spectrum is not known, this has to be unfolded using any method. There are different methods available for unfolding neutron spectrum.[4-15] The foil activation method is used for unfolding the neutron spectrum in this study at a distance of 5 cm from the target of the RMC cyclotron accelerator with 16.5 MeV proton beam current of 60μA and a target pressure of 438 psi. The procedure involves exposing the sample serving as neutron detector to the neutron flux for a measured length of time and then removing it to a counting position and counting its induced radioactivity. The induced gamma activity is estimated using an HPGe and $4\pi\beta-\gamma$ detector and beta activity is estimated using a plastic scintillator based beta counter. The total no. of counts are first corrected for background. The net number of counts is then reduced to the activity which would have been obtained right at the end of exposure. From the specific activity, the neutron spectrum was unfolded using the computer code.
3. Spectrum Unfolding

The activation detectors provide a very convenient method to measure the neutron fluence and to give an idea of the neutron spectrum, provided the activities of various detectors can be interpreted appropriately and satisfactorily.

After irradiating the foils in the neutron field which has to be standardized, it is counted in the counting set up. The measured count rate is related to the neutron fluence level by the equation

\[ A_i = N_i \varepsilon_1 \varepsilon_2 \varepsilon_3 \left(1 - \exp\left(-\lambda t\right)\right) \exp\left(-\lambda T\right) f_s f_d \int \sigma_i(E) \phi(E) \, dE \]

Where

- \( A_i \) is the count rate of \( i \)th detector in the counting set up
- \( N_i \) is the number of the atoms
- \( \varepsilon_1 \) is the isotopic abundance
- \( \varepsilon_2 \) is the abundance of the radiation counted.
- \( \varepsilon_3 \) is the efficiency
- \( f_s \) is the self shielding factor for the foil
- \( f_d \) is the flux depression factor of the foil for that medium
- \( \sigma_i(E) \) is the activation cross section at the energy \( E \).
- \( \phi(E) \) is the neutron fluence rate of energy \( E \)
- \( t \) is the irradiation time
- \( T \) is the decay time

The activation integral is defined as:

\[ A_i = \int \phi(E) \sigma(E) \, dE \]

Where \( \phi(E) \) is the unknown spectrum. The estimation of \( \phi(E) \) is called unfolding the neutron spectrum. The correct information of \( \phi(E) \) with limited number of activation foils are very complicated and difficult. There are so many methods for getting the information on \( \phi(E) \) with limited number of foils. Every method has its own merit and demerits. The simplest method is
the matrix method. In this method the numbers of energy bins are very small equal to the number of activation foils used. Various unfolding codes are used for unfolding the neutron spectrum.

But in the unfolding codes a guess spectrum is taken first from the theoretical calculation or by experience, even mathematical formulations are possible for the guess spectrum. With an adjustment procedure we are looking for the deviation from the trial spectrum which is the first estimation of the actual spectrum. With the proper choice of detectors and suitable guess spectrum a good estimation of the unknown spectrum is possible. Starting from integral activation data, it is clear that the possibility to obtain much detail is very small. The detectors used are such that their responses overlap. The normal output is a smooth spectrum with little structure, depending on the number of detectors and overlapping of their response range. The measured activities and cross-section data used are always associated with some errors. Any adjustment procedure must take care of this. Most of the spectrum adjustment codes are iterative and stop the iteration procedure when it leads in general to oscillations which are not acceptable.

Any ideal spectral adjustment code should have the following attributes:

(a) Consistency: The spectra obtained from adjustment should be consistent with the measured activities within the limit of measuring uncertainties.
(b) Possitivity: Only positive solutions represent physically acceptable spectra.
(c) Smoothness: Large oscillations in the solutions are not acceptable.
(d) Flexibility in Accepting Prior Information: Such information is usually introduced in the form of trial spectra, uncertainties and co-variance matrices.

Activation detectors and adjustment techniques are a powerful means but one should not ask more information from them than it contain. Neutron spectra determination is usually not a goal in itself but a means to determine other integral quantity such as the equivalent dose, the protection quantity of radiation.

\[ H = \int C(E) \phi(E) \, dE \]

Where \( C(E) \) is the fluence to dose equivalent conversion factor for the neutron. The resolution required in the measurement depends on the final quantity of interest. For the dose equivalent evaluation the demand on the resolution is not that stringent. There are so many algorithms available for the adjustment of neutron spectra.

SANDII is a widely used spectrum adjustment code in iteration mode and it is used to determine the spectrum and the absolute magnitude of the neutron fluence rate from activity of the foils which is activated in that environment. This is accomplished by selection of an initial approximate spectrum and subsequent perturbation of that spectrum by interactive iterative adjustment, to yield a solution spectrum that produces calculated activities which agree with the measured activities. The energy range of the solution spectrum is from \( 10^{-10} \) MeV to 18 MeV.

In the code, all foil activities are taken to be adjusted to infinite dilution of target nuclei; i.e. the code does not calculate the effect of self shielding, so all input activities must first be corrected for self shielding. The unfolded spectrum is used to calculate the ambient dose equivalent using fluence to dose equivalent conversion factor averaged over the spectrum[16].
<table>
<thead>
<tr>
<th>Activation foils</th>
<th>Nuclear Reaction</th>
<th>Product</th>
<th>Half-life</th>
<th>Type and energy of radiation measured (MeV)</th>
<th>Abundance of measured radiation (Disintegration⁻¹)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Gold</td>
<td>197 Au (n, γ) 198 Au</td>
<td>2.7 days</td>
<td>γ = 0.412</td>
<td></td>
<td>0.96</td>
</tr>
<tr>
<td>Copper</td>
<td>63 Cu (n, γ) 64 Cu</td>
<td>12.7 hrs.</td>
<td>γ = 0.511</td>
<td></td>
<td>0.36</td>
</tr>
<tr>
<td>Indium</td>
<td>115 In (n, n’) 115mIn</td>
<td>4.5hrs.</td>
<td>γ = 0.336</td>
<td></td>
<td>0.46</td>
</tr>
<tr>
<td>Sulphur</td>
<td>32 S (n, p) 32 P</td>
<td>14.3 days</td>
<td>β = 1.711</td>
<td></td>
<td>1.0</td>
</tr>
<tr>
<td>Nickel</td>
<td>58Ni (n, p) 58Co</td>
<td>71 days</td>
<td>γ = 0.811</td>
<td></td>
<td>0.99</td>
</tr>
<tr>
<td>Magnesium</td>
<td>24Mg (n, p) 24Na</td>
<td>15hrs</td>
<td>γ = 1.369</td>
<td></td>
<td>1.0</td>
</tr>
<tr>
<td>Aluminium</td>
<td>27 Al (n, α) 24Na</td>
<td>15hrs</td>
<td>γ = 1.369</td>
<td></td>
<td>1.0</td>
</tr>
</tbody>
</table>
Unfolded Neutron Spectrum of RMC Cyclotron
(Angle = 0 degree)

Unfolded Neutron Spectrum of RMC Cyclotron
(Angle = 90 degree)
4. RESULTS AND CONCLUSION

The Neutron Dose-rate at 5 cm near the target in the Cyclotron Accelerator RMC at 60 μA current of 16.5 MeV proton beam is:

\[ H = \int C(E) \phi(E) \, dE \]

Total Neutron Dose-rate = \( 3.9 \times 10^3 \) Sv/hr.

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REFERENCES

[2]. Radiation Shielding Analysis For Medical Cyclotron At Radiation Medicine Centre, Parel' by R. A. Agrawal et. al. No: RPD/SS/02/2001

[4]. ‘Slow-Neutron Detection by Foils’ by C. W. TITTLE, Department of Physics, North Texas State College, Denton, Texas, NUCLEONICS, July 1951, Vol. 9, No. 1, Pages 60-67


[16]. ICRP ; Recommendations of the international commission on radiological protection, publication 74 (1997).