Neutron and Gamma Radiation Level Estimation and Biological Shielding Evaluation of a Medical Cyclotron Using a Monte Carlo Simulation Method

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1. Introduction

Medical cyclotrons are used extensively worldwide for the production of radionuclides, which are widely utilized in medical fields like nuclear medicine for the diagnostic imaging purposes [1]. Flourine-18, a positron-emitting radionuclide produced by medical cyclotrons, is one of the most commonly used radionuclide in the Positron Emission Tomography (PET), a quickly emerging state-of-the-art imaging technology in nuclear medicine [2].

Cyclotrons produce a high neutron and gamma field as secondary radiation during normal operation. The accurate estimation of radiation field is of utmost importance and necessary for the cyclotron shielding design and radiation protection of worker, public and the environment [3]. Nonetheless, various international documents and scientific researches demonstrating the analytical methods for evaluation of cyclotron shielding are mostly based on approximate or ideal geometry set ups or use the generalised results or source terms[3][4]. Monte Carlo simulation code in general is equipped with accurate and up to date radiation transport and interaction libraries. Moreover, the actual cyclotron geometry, source term and site specific characteristics are used in simulations. Therefore, use of the Monte Carlo simulation results in a more accurate estimation of the radiation level and thereby enabling optimal shielding evaluation of the cyclotron.

In this study, a medical cyclotron was simulated using a Monte Carlo simulation code. Then, an accurate estimation of neutron and gamma fluence and radiation levels was carried out. Furthermore, the biological shielding furnished to the medical cyclotron was evaluated based on the acceptable radiation level recommended by a current international standard [5].

2. Methods and materials

In this work, the simulation code used is the Monte Carlo N-Particle Transport Code, MCNP6 version 1 [6**]**. MCNP is a general purpose code for modelling particle transport and interaction with matter. Simulation was carried out on the non-self shielded cyclotron model PETtrace 800, GE Medical system [7]. This equipment is capable of accelerating protons up to a maximum energy of 16.5MeV. The proton current used in the simulation was 40 μA which amounts to an emission rate of 2.5×10^{14} protons/second.

2.1 Medical cyclotron geometry

The simulation was carried out on an Oxygen-18 enriched water target used for the production of F-18 radionuclide by (p, n) reaction. The target system was simulated as per the manufacturer's specifications [7]. The target was a silver chamber filled with 95% O-18 enriched water and at the front end it is sealed from the vacuum by 75µm thick Havar foil (an alloy of cobalt (42.5%), chromium (20%), nickel (13%) and traces of manganese, molybdenum and iron). The geometry of the target system simulated is shown in Figure 1.

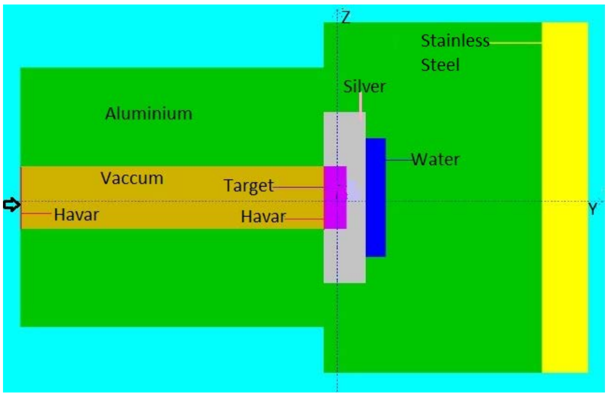

Fig. 1. Cross-sectional geometry of the target system.

All components of the target described in Figure 1 are concentric cylinders centred at 110cm above the floor. The cyclotron is furnished with a radiation shield made of 5 cm thick lead placed on the target side at a distance of 40cm from the target.

2.2 Shielding system geometry

The cyclotron simulated was placed in a concrete bunker wherein the inner room dimension was 4.5m x 4.5 m in breadth and length and 3.5m in height. Regular concrete of density 2.3g/cc and a thickness of 160cm were used in the simulation [8]. The schematic diagram of the concrete bunker and the target system are shown in Figure 2. The cyclotron target was positioned at the centre of the room in the simulation.

2.3 Tally and conversion factors

Point detector method F5 tally was used for the estimation of neutron and gamma fluence wherein the detector is approximately taken as a point and then calculates the point flux or fluence. The point detectors were placed at 11 locations in different axial directions inside the bunker at a distance of 1 and 2 meters from the target as shown in Figure 2.

ICRP74 was used for the neutron fluence to ambient dose equivalent conversion factor. For the gamma fluence rate to effective dose equivalent conversion factor, ICRP116 was used.

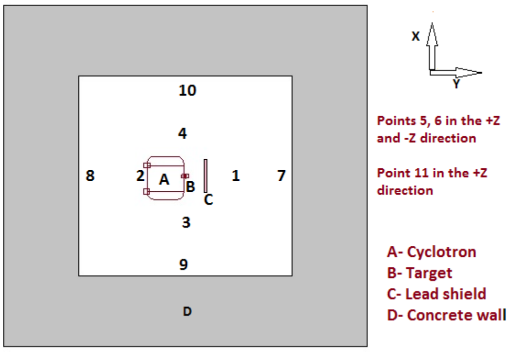

Fig. 2. Cyclotron and concrete bunker (Y-X view).

3. Results and discussions

As a result of proton interaction with the target, high levels of neutron and gamma radiation are generated. The resultant neutron and gamma radiation fluence was estimated at various locations indicated in Figure 2 and is shown in Table 1.

Table 1: Neutron and gamma fluence at different locations.

Location	Distance from target	Neutron Fluence neutrons/cm^2 Sec	Gamma Fluence rate photons/cm^2 Sec
1	1m	8.23E+05	2.43E+06
2	1m	1.57E+06	2.15E+07
3	1m	1.32E+06	1.49E+07
4	1m	1.34E+06	1.51E+07
5	1m	1.34E+06	1.50E+07
6	1m	1.71E+06	2.03E+07
7	2m	5.93E+05	1.94E+06
8	2m	8.54E+05	8.48E+06
9	2m	7.36E+05	6.29E+06
10	2m	8.67E+05	6.65E+06
11	2m	7.62E+05	6.05E+06

From Table 1 it can be seen that neutron fluence ranges from 8.2×10^5 to 15.7×10^5 at 1 meter from the target and ranges from 5.9×10^5 to 8.5×10^5 at 2 meter from the target. But in case of gamma, the fluence rate at 1meter from the target ranges from 2.4×10^6 to 21.5×10^6 and at 2 meter from the target it ranges from 1.9×10^6 to 8.5×10^6 at 2m.

In Figures 3 and 4, the resultant neutron and gamma fluence and fluence rate as a function of energy at location 2, which is at a distance of 1 meter from target in the -Y direction, is shown as a sample case for deliberation. The neutron fluence was observed to be higher at both the thermal energy range and at an energy range of 0.05MeV to 8MeV. Whereas, gamma fluence rate was moderately higher both at the lower energy range of 0.1MeV to 1.2MeV and at the higher energy range of 6MeV to 9MeV. But, a highest range of gamma fluence rate was observed in the energy range of 2MeV to 5MeV.

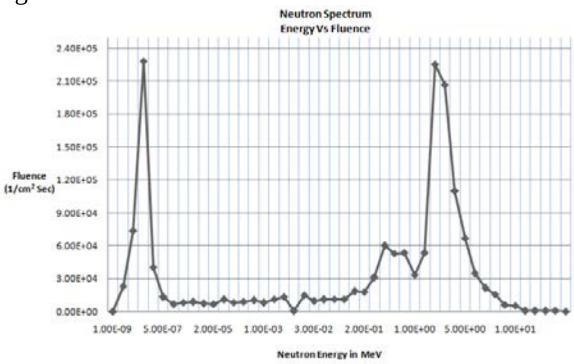

Fig. 3. Neutron fluence as a function of energy at location 2 (1 meter from target).

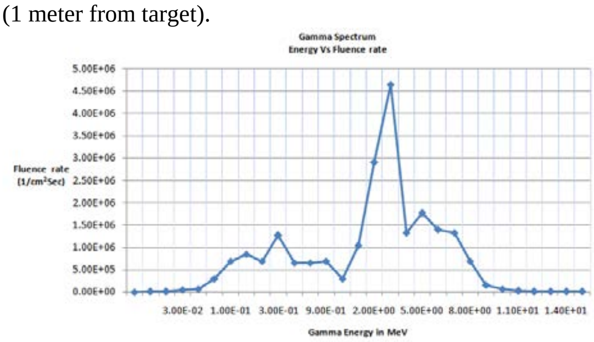

Fig. 4. Gamma fluence rate as a function of energy at location 2 (1 meter from target).

Also, neutron ambient dose equivalent rate and gamma effective dose equivalent rate was calculated at various locations indicated in Figure 2 and is shown in Table 2. In case of neutron, the highest dose rate observed at 1 meter from the target was 1520.5mSv/hr, whereas, for gamma it was 590.4mSv/hr. Furthermore, at 2 meter from the target, the estimated highest dose rate of neutron was 544.5mSv/hr and of gamma was 179.2 mSv/hr.

Table 2: Neutron and gamma dose rates at different locations.

Location	Distance from target	Neutron Ambient dose rate (mSv/hr)	Gamma Ambient dose rate (mSv/hr)
1	1m	449.00	33.22
2	1m	1520.49	590.43
3	1m	1219.63	390.89
4	1m	1165.98	398.26
5	1m	1180.93	391.88
6	1m	1429.64	442.05
7	2m	185.51	22.67
8	2m	544.45	179.20
9	2m	426.33	127.85
10	2m	439.43	127.20
11	2m	416.48	138.34

Further, in order to carry out the shielding evaluation, neutron and gamma dose rates in different axial directions as a function of the concrete shielding thickness were calculated and are depicted in Figures 5 and 6. It is observed that in the case of neutron, a concrete shielding thickness of 120 to 140cm was required to reach at a dose rate level of less than 1μSv/hr. However, in the case of gamma, the concrete shielding thickness required to reduce the dose rate level to less than 1μSv/hr was found to be 150 to 160cm. Moderately higher dose rate was observed in –Y and X direction, which may be attributed to the axial difference in geometry, shielding and scattering contribution and might require further deliberation.

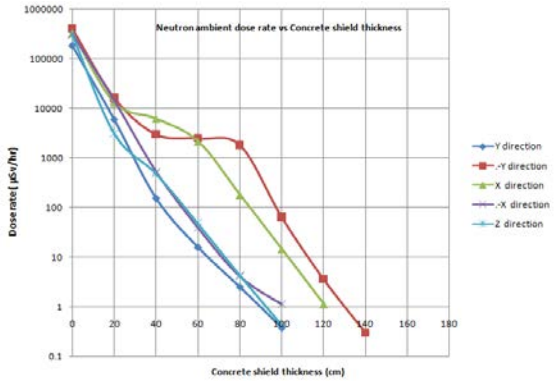

Fig. 5. Neutron dose rate Vs concrete thickness.

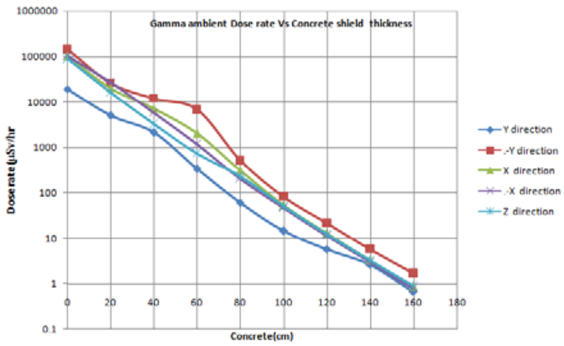

Fig. 6. Gamma dose rate Vs concrete thickness.

4. Conclusions

It was evident from this study that high levels of neutron and gamma radiation are generated during the operation of medical cyclotrons and an accurate estimation of those levels are vital from radiation safety view point. In this study, using a Monte Carlo simulation code, an accurate estimation of neutron and gamma fluence and radiation levels was arrived at. Furthermore, the biological shielding evaluation was performed and found that 160cm of concrete is adequate to achieve the acceptable radiation level for the specific machine and site characteristics used in this study. Future work will focus to evaluate the optimization of shielding by using various types of concrete having higher density than the regular concrete used in this study. The assessment of induced activation of shielding material by the secondary radiations will also be the focus of future research.

REFERENCES

[1] International atomic energy agency, Cyclotron produced radionuclides: Principles and practice, Technical report series no. 465, IAEA, Vienna, 2008.

[2] International atomic energy agency, Standard operating procedures for PET/CT: A practical approach for use in adult oncology, Human health series no. 26, IAEA, Vienna, 2013.

[3] International atomic energy agency, Cyclotron produced radionuclides: Guidelines for setting up a facility, report series no. 471, IAEA, Vienna, 2009.

[4] National Council on Radiation Protection and Measurements, Radiation protection for particle accelerator facilities, Report No. 144, NCRP, 2005.

[5] International Commission On Radiological Protection, Annals of the ICRP, Publication 103, 2007.

[6] D. Pelowitz (ed.), MCNP6 User's Manual, Version 1.0, Los Alamos National Laboratory, LA-CP-13-00634, Rev. 0 , 2013.

[7] J. O. Bergstrom, PETTRACE - unshielded machine: Summary of source terms, radiation fields and radwaste production, GE Healthcare document ref: DOC0100224, 2005.

[8] Pacific Northwest National Laboratory, Compendium of Material Composition Data for Radiation Transport Modeling, 2011.