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COMPARISON OF COMPUTER CODES' APPLICABILITY IN SHIELDING DESIGN FOR HADRON THERAPY FACILITIES*

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Abstract. Various computer codes have already been developed and applied for computation of the dose produced in the clinical environments due to proton and carbon ion therapy procedures. Computer codes BULK-I, BULK-II, PHITS, SHIELD-HIT, MARS15, FLUKA and GEANT4 have been utilized to obtain the secondary radiation dose or the required thickness of the shield around such facilities. Differences in the angular distribution of secondary neutrons and transmitted dose from the shield are observed among above mentioned codes. In the present work, a practical approach is undertaken to compute the shield thickness with the use of semi-empirical formulae along with the results of computer codes. The results are compared and the differences are explained. For this purpose, the spectra of secondary neutrons that have been simulated by various codes are studied. The discrepancies in outcomes are attributed to the nuclear models and approximations employed in each code. Then the shield thickness is evaluated from engineering point of view. One goal of this study is to develop engineering expertise for shielding design of newly established hadron therapy centers. Another goal is to compare these codes and choose the most proper one on cost-benefit basis. Finally, the proposed analytical calculations will be compared against the results of above codes.

Key words: secondary neutrons, neutron shielding, Monte Carlo codes.

1. INTRODUCTION

Interaction of high energy protons or Carbon ions with patient's tissue generates variety of secondary charged particles and neutrons. The importance of secondary neutrons in hadron therapy is twofold. Neutrons emanated from the patient's tissue deposits undesirable dose in healthy tissue. The generation of secondary neutrons increases the risk of developing a secondary radiation-induced cancer. Neutron dose in healthy tissue for large and medium target volumes is approximately 0.004 Sv and 0.002 Sv per treatment Gy respectively [1]. Besides,

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after passing through a thick layer of tissue, neutrons are free in the environment of clinic and are harmful for the personnel. Neutrons have a large quality factor, and thus even a small physical dose can result in essential biologic effects. Hadron therapy centers require shielding barriers and stringent controls in order to protect personnel from stray neutrons [2]. Therefore, Proton centers require shielding barriers and stringent controls in order to protect personnel from stray neutrons.

Technically, magnitude of neutron flux depends on the method of proton beam scanning on the tumor volume. The spot scanning method is expected to have a much lower neutron background than passive scattering [1]. The attenuation of neutron dose equivalent through a slab-type shield could be estimated as an exponential function of slab thickness [3].

For complex slab and maze shielding geometries semi-empirical techniques have been developed. In such methods many aspects of neutron transport physics and, in many important practical cases, the shielding geometry under consideration are simplified. In complex geometries, this may lead to significant errors in the predicted neutron dose equivalent values. The overall uncertainty in a complex shielding design is large and difficult to estimate. In principle, the most accurate way to predict neutron dose equivalent values in a highly complex proton therapy facility is the Monte Carlo technique.

The uncertainties in prediction of neutron dose using the Monte Carlo technique remains a matter of controversy and are large. The causes of discrepancy among results are poorly known. Improved Monte Carlo results contain significantly lower statistical uncertainties compared with the previous reports [4].

The structure of shield contains ordinary poured concrete shielding walls and ceilings; a maze for access to each gantry room; a maze with a sliding steel-and-concrete door for the cyclotron vault; steel and borated-polyethylene shields mounted on the gantries; and various other hatches, covers, and blocks [3, 5].

2. ESTIMATION OF NEUTRON DOSE

After the computation of neutron flux, the dose imposed on the environment might be calculated with the use of internationally accepted guidelines and equations *i.e.* KERMA concept or fluence-to-dose conversion factors. Computer codes based on Monte Carlo method are strong tools to produce a scope of the neutron flux and dose distribution shape, especially in complex geometries [6]. Comparative studies of a proton therapy facility revealed that analytical calculations overestimated the neutron dose equivalent in almost all cases, in some instances by more than two orders of magnitude [4].

As implicitly mentioned above, the magnitude of neutron flux is function of proton beam intensity, proton energy plus abundance and material of objects intercepting the proton beam. To a lesser extent, the anatomy of organ undergoing treatment affects neutron energy distribution and flux. Monte Carlo simulation of neutron doses in an eye treatment facility (with 65-MeV protons) using the scatter technique shows that a neutron dose of 10^{-4} Gy per therapy Gy exists in the healthy tissue behind the eye [1]. Simulating Hadron therapy facilities which utilizes protons in energy range of 200–250 MeV it was found that neutron dose for scatter technique is of order 10^{-2} Gy per therapy Gy while this figure is 10^{-3} for spot scanning. These simulations are supported by measurement in surface of a phantom or around the Bragg peak [1, 3]. The neutron dose in patient treatments with carbon ions was estimated to be 8 mGy per treatment-Gy based on the measured neutron yield [7].

3. NEUTRON SPECTRUM

There is no unique energy spectrum for secondary neutrons, but it shows angular functionality. In a swift glance, energy spectrum of neutrons smoothly falls as energy increases. This reduction happens more sharply as the spectrum approaches its high energy end [8].

At small angles, the energy distributions have a broad maximum at about half of the energy per nucleon. This maximum is related to the beam energy since the neutrons emitted at small angles are mostly produced by projectile abrasion, *i.e.* these neutrons have energies close to the beam energy at the moment of the nuclear interaction and are therefore sharply forward peaked. The maximum neutron energy extends to almost twice the energy of the projectile per nucleon.

Neutron energy spectra were simulated with PHITS for different angles down to very low energies. From these calculations, the fraction of neutrons below 20 MeV was found to be 2.3% at 0^0 and 14.3% at 30^0 [9]. The gross shape of the neutron spectra shown is well reproduced by the PHITS calculations. At 0^0 the calculated neutron yields are about 40% lower than the experimental ones, while at larger angles the broad peak seems to be more pronounced and the calculated yields slightly overestimate the data. Neutron flux in backward angles are 5 to 7 orders of magnitude lower.

4. COMPUTER CODES REVIEW

4.1. PHITS

In PHITS, neutrons can be transported over a vast energy range 0.1 meV up to 200 GeV. Below 20 MeV down to thermal energies, neutrons are described in the same manner as in the MCNP4C code based on the Evaluated Nuclear Data such as the ENDF-B/VI6, JENDL-3.37, and LA150 libraries [10]. The physical processes included in PHITS can be schematically divided into collision and transport processes. Transport calculations are supported for neutrons (10^{-5} eV to 200 GeV), protons, nuclei (0–3AGeV) and photons and electrons (1 keV to 1 GeV). PHITS can treat both neutron scattering and transport processes for a shielding design by simulating neutron interactions with various nuclei in parallel with production of secondary particles.

This computer code can generate the neutron spectra that are produced by charged particle interaction with various materials. At 100 AMeV of ^{12}C, code over-estimates the neutron spectra, unless at 90 degree which coincides very well down to 10 MeV neutrons, below which the code under-estimates the practice. At 180 AMeV the results agree within 15 to 60 degrees. Outside this angular range the code under-estimates the neutron flux.

4.2. BULK-II

BULK-II was developed for proton and carbon accelerator facilities at incident energies of 50–400 MeV/nucleon. The basic concept of BULK-II is the same to the previous version of BULK-I, such that the build-up effect, which plays an important role to radiation protection calculation in recent high energy accelerator facilities, included to the Moyer formula.

The computational results of BULK-II have been compared with experimental outcomes at GSI. For angles from 10 to 130 degrees this code over-estimates the dose compared with GSI experimental data. The difference is remarkable in the region 10 to 30 degrees [11].

The neutron dose computed by BULK-II is more than PHITS for concrete thickness of more than 100 cm. For less than 100 cm is Vice-versa. Both computer codes usually produce doses higher than the experimental results. It has been shown that BULK-II results agrees well with 3D full Monte Carlo calculations for deep penetration of ^{12}C induced secondary neutron dose in concrete shield and angular distribution of ^{12}C induced secondary neutron dose.

4.3. FLUKA

In FLUKA, hadrons are transported from 1 keV (thermal energies for neutrons) up to 20 TeV [1].

Because FLUKA does not discriminate between the energy deposited by primary and secondary protons (*i.e.*, recoils from neutron scattering), separate calculations were performed, with and without secondary neutron production. In a general look, FLUKA and PHITS are in better agreement with observations than MNCPX [8].

5. SHIELD CALCULATION APPROACHES

Traditional methods to design neutron shielding are based on semi-empirical and analytical methods that simplify many aspects of neutron transport physics and in some cases, the shielding geometry. In complex geometries, this is the cause of remarkable errors in the predicted neutron flux and dose. In principle, the most accurate way to predict neutron dose equivalent values in a highly complex proton therapy facility is the Monte Carlo technique. With the use of new Monte Carlo programs results with significantly lower statistical uncertainties compared with the previous reports have been achieved [3, 4].

The attenuation of neutron dose equivalent through a slab-type shield was estimated as an exponential function of slab thickness. In the literature, the dose equivalent of neutron flux is given for mono-energetic neutron as a function of the concrete wall thickness and incident. These data have been combined with the Monte Carlo generated neutron spectra to calculate dose equivalent transmission curves as a function of shielding wall thickness for cases of interest.

The neutron shielding at 235 MeV proton therapy facility was investigated with measurements, analytical calculations, and realistic three-dimensional Monte Carlo simulations. In the majority of cases, the analytical calculations predicted higher neutron dose equivalent rates outside the shielding than the measured, typically by more than a factor of 10, and in some cases more than 100. Except at one location, all of the analytical model predictions and Monte Carlo simulations overestimate neutron dose equivalent.

For slab and maze shielding geometries assumptions for estimation of neutron dose attenuation with semi-empirical techniques are based on the assumptions that (1) high-energy neutrons determine the neutron dose equivalent attenuation behind thick shields, and (2) deep in the shield, the low-energy neutrons are in radiation equilibrium with the high-energy neutrons [3].

6. CONCLUSIONS

We conclude from the measurements and simulations that the dose deposited by secondary neutrons during proton radiotherapy using the spot scanning technique can be neglected in the treatment region.

- For concentration of fragments, there is remarkable difference between FLUKA and GEANT4.
- Special purpose codes such as BULK-2 are much easier to use than MCNP, FLUKA, and GEANT4.

The shield design for a hadron therapy facility is different than a nuclear reactor in the fact that in a reactor high flux of neutrons with average 2 MeV energy are to reduced while in Hadron therapy low flux of neutrons with 100 MeV energy.

A general guideline for designing a shield for hadron therapy facility is:

1. The energy of proton or carbon ion plus beam irradiation time in week and its current are required as part of input data.

2. The type beam scanning over tumor volume must be specified *i.e.* spot scanning or passive scanning. Measurements shows that spot scanning technique has a dose advantage of at least 10 for the scatter foil technique.

3. Using computer codes the flux of secondary neutrons is approximated at specified direction and distance.

4. Neutron flux at any point of interest is converted to dose by given relationships or published tables.

5. Attenuation of neutron flux in the shield might be estimated by exponential approximation or computer codes or improvised semi-empirical methods.

The shield thickness is a matter of risk-benefit analysis. If someone designs the shield for longest possible time of operation per week, the shield would be heavy. If the working hours are restricted the shield is reduced. Often, the computed dose is much higher than the measured dose in clinic.

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