Neutron deep-penetration calculations and shielding design of proton therapy accelerators

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Abstract

Proton therapy accelerators have become increasingly popular for cancer treatment in recent years. In Taiwan, the first proton treatment center equipped with a 235 MeV proton cyclotron in Linkou Chang Gung Memorial Hospital is ready for beam commissioning. Proton therapy accelerators in the energy range could potentially produce intense secondary neutrons, which must be carefully evaluated and shielded for the purpose of radiation safety in a densely populated hospital. Monte Carlo simulations are generally the most accurate method for accelerator shielding design. On the other hand, simplified approaches such as the commonly used point-source line-of-sight model are usually preferable in many practical occasions. Understanding the appropriateness or uncertainties associated with these methods is critical to the quality of a shielding design. Through a systematic comparison between the FLUKA and MCNPX calculations, this study examined an important problem in multigroup neutron deep-penetration calculations. Based on continuous-energy MCNPX calculations, this work also provides a set of reliable shielding data with reasonable coverage of common target and shielding materials for 100-300 MeV proton accelerators. The shielding data including source terms and attenuation lengths were derived from a consistent curve fitting process of a number of depth-dose distributions in shield corresponding to various beam-target-shield configurations. A practical application of the data set for proton accelerator shielding is demonstrated.

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I. Introduction

Proton therapy accelerators in the energy range of approximately 100-300 MeV have become increasingly popular for cancer treatment in recent years due to its distinctive physical and biological advantages. As of 2012, a total of about 40 proton therapy centers are in routine operation and many more are under construction or proposed [1]. In Taiwan, the first proton treatment center equipped with a 235 MeV proton cyclotron in Linkou Chang Gung Memorial Hospital is near completion and beam commissioning will be started soon. Proton therapy accelerators in the energy range could potentially produce intense secondary radiation, which must be carefully evaluated and shielded for the purpose of radiation safety in a densely populated hospital. For the proton therapy accelerators, neutrons produced from hadronic cascade are the dominant dose component. Therefore, accurate estimation of neutron production from proton bombardment and reliable neutron deep-penetration calculations are key issues in accelerator shielding analysis.

In this study, a benchmark calculation of neutron production from proton bombardment was first carried out and verified through a comparison with experiment. Then, we examined the problem of multigroup neutron deep-penetration calculations and possible effects on the estimation of important shielding parameters. Based on a series of rigorous radiation transport calculations, this work has generated a set of reliable shielding data with reasonable coverage of common target and shielding materials for 100-300 MeV proton accelerators. This paper presents some general characteristics of this data set and demonstrates its use in the shielding design or dose assessment for a proton therapy accelerator.

II. Calculation methods and models

Source term estimation and radiation attenuation in shield are two important aspects in a shielding design for proton therapy accelerators. The source term estimation involves a simulation of complicated nuclear phenomena including the development of intranuclear cascades initiated by primary protons and subsequent nuclear evaporation of excited nuclei. These are the production mechanisms of secondary radiation, including neutrons, photons and a number of sub-atomic particles. The second part is to estimate the attenuation of various secondary particles in shield. In cases of complicated geometry or demand for accurate results, the shielding design usually requires a high-fidelity radiation transport simulation.

Two well-known multi-particle interaction and transport codes were used in this study: FLUKA [2,3] and MCNPX [4]. FLUKA is a fully integrated particle physics Monte Carlo simulation package. MCNPX is also a general-purpose Monte Carlo radiation transport code. Both codes show a capability of simulating radiation interaction with matter for nearly all particles and nearly all energies. They have many applications in high energy physics, medical physics, health physics, dosimetry and radiobiology.

Monte Carlo simulations are generally the most accurate method for accelerator shielding design. On the other hand, simplified approaches such as the commonly used point-source line-of-sight model are usually preferable in many practical occasions. The model assumes radiation is coming from a point-like source and exhibits simple exponential attenuation in a shield, as shown in Eq. (1). $H$ is the transmitted dose at position of interest, $E_p$ is the energy of primary proton beam, $r$ is the distance between the target and the detector, $d$ is the thickness of shield. $H_0$ and $\lambda$ are the parameters representing a pseudo-source term and corresponding attenuation length under the specified conditions.

$$H(E_p, \theta, d / \lambda) = \frac{H_0(E_p, \theta)}{r^2} \exp(-\frac{d}{\lambda(E_p, \theta)}),$$  

The key for a successful use of this simplified shielding model relies on selecting a set of source term and attenuation length most suitable to the problem in consideration. Agoste et al. [5,6] presented a comprehensive review on the shielding data for 100–250 MeV proton accelerators and proposed their estimates based on the calculated double differential neutron distributions and attenuation in thick concrete and iron. Their calculation models and results are considerably valuable. This work begins by repeating their calculations for verification. To conduct a consistent comparison, we used the same geometry and materials modeling as that described in their papers [5,6]. As shown in Fig. 1, the model is simple by launching a monoenergetic proton pencil beam to a target located at the center of a large spherical shield. The target dimension was set to be slightly larger than the stopping length of protons in that material to ensure
producing maximum amount of secondary radiation. Secondary radiation consisting of neutrons, photons, protons, and electrons emerging from the target were subsequently incident on the inner surface of the spherical shield. Subsequent radiation transport in the surrounded shield was limited within interested angular bins with respect to the direction of incident proton beam. The spherical shield was divided into several consecutive layers for one-way boundary crossing flux scoring and variance reduction purposes. The energy-dependent flux at various depths in concrete or iron shield were scored and converted into ambient dose equivalents H*(10).

Fig. 1: A schematic view of the geometry model for calculating dose attenuation profiles in shield.

III. Results and discussion

III.1 Validation of neutron yield calculation

Both FLUKA and MCNPX are capable of simulating hadronic and electromagnetic cascades in considerable detail. In order to validate our neutron production calculations, a bombardment experiment of the same particle type and similar energy was selected from literature for benchmark calculations. The experiment performed by Meier et al. in 1990 [7] measures absolute neutron yields at angles of 30°, 60°, 120°, and 150° for the 256 MeV proton bombardment of natural iron.

We performed precise benchmark calculations using both FLUKA and MCNPX according to their measurements to validate the calculation tools and models. Fig. 2 shows an absolute comparison between the predictions and their measured results for the double differential neutron yields from stopping-length iron target bombarded by 256 MeV protons. Both FLUKA and MCNPX simulations mostly reproduces the experimental results over the entire neutron energy range and for various emission angles. However, minor discrepancies could be observed for neutron yields above 20 MeV at large angles for MCNPX, where the calculation slightly overestimates the neutron yields. This is acceptable to the study since it tends to predict more conservative estimates. In general, the good consistency with the measurement validates both calculation models for neutron production and largely gives confidence on the results of subsequent neutron transport in thick shield.
III.2 Depth-dose distributions in shield calculated by FLUKA and MCNPX

The point-source line-of-sight model is quite common and preferable in practical shielding design because of its simplicity. The key for a successful use of this simplified model relies on selecting a set of source term and attenuation length most suitable to the problem. Relevant literature contains a large amount of such shielding data for various accelerator energies, beam targets, and shielding materials. However, you will find a wide range of variability in the published data, which may lead to very different dose estimates and different shielding requirement. Some of the discrepancies are believed to be related to the nature of neutron deep-penetration. Neutron deep-penetration calculations have been well known a challenging and troublesome problem in terms of the computation efficiencies and accuracies, especially for multigroup neutron transport calculations.

Based on the simplified beam-target-shield configuration as shown in Fig. 1, FLUKA and MCNPX were used to simulate the production and transport of secondary radiation from proton bombardment. The resulting depth dose distributions in shield can be used to estimate the wanted shielding parameters through a curve fitting process. Frist of all, we repeated the FLUKA calculations in Ref. [6] for concrete shielding against 200 MeV protons impinging on a thick iron target. The derived source terms and attenuation lengths at various angles are consistent with those in Ref. [6]. As shown in Fig. 3, dose attenuation in forward angles exhibits longer attenuation lengths than those in backward directions because of the angular dependence of source neutron energies. Based on the classical two-parameter attenuation formula of Eq. (1), an equivalent source term and attenuation length can then be derived from a curve fitting process of the depth-dose distribution in shield. In practice, experiences indicate that the dose attenuation in thick shield should be better approximated by a double-exponential expression, from which two source terms, $H_1$ and $H_2$, and two attenuation lengths, $\lambda_1$ and $\lambda_2$, can be obtained through independent curve fitting for shallow and deep regions in the shield, respectively. The shallow region was defined in this study between the depths of 20-150 cm in concrete and 20-75 cm in heavy metals; while the deep region was referred to those deeper than 200 cm in concrete and 100 cm in heavy metals.
For the case of iron shielding, the resulting attenuation lengths were found to be much larger than those in the concrete case and seem to be angular independent. Compared with the data in Ref. [6], their differences are quite evident, particularly for the attenuation lengths in deep region. Not to mention the 10-50% deviation in their absolute values, their variations with respect to angles are also completely different. Our result indicates no obvious angular dependence of the attenuation length whereas the reference result indicates that the larger the neutron incident angles, the shorter the attenuation lengths will be, similar to that observed in the concrete case. These inconsistencies in both the absolute values and general trend were unexpected because we assumed that the same calculation tool and modeling were used. After further investigation, we figured out it is pertinent to the deep-penetration calculations in iron and the effects of multigroup neutron cross-sections.

In order to find out the root cause of discrepancies between our calculations and that in Ref. [6], we have performed a careful comparison of neutron deep-penetration calculations with FLUKA and MCNPX. An important difference between FLUKA and MCNPX is their treatments for low-energy neutrons. Transport of low-energy neutrons is performed in FLUKA by a multigroup framework, whereas they are explicitly simulated based on continuous-energy cross sections in MCNPX. Therefore, neutron transport in MCNPX is immune from multigroup approximations and self-shielding effects. A series of repeated FLUKA simulations with various multigroup cross-section sets were performed to quantitatively evaluate the effects of multigroup cross sections on the neutron deep penetration in iron. Six iron cross-section sets in combination of two group structures (72 and 260 groups) and three degrees of self-shielding corrections (infinite dilute, partially and fully self-shielded) were considered in this comparison. Most of the FLUKA calculations underestimate the results in varying degrees compared with the MCNPX prediction. As shown in Fig. 4, the resulting attenuation lengths for deep penetration varies significantly from 130 up to 260 g/cm² in forward direction. The comparison indicates that orders of magnitude difference in dose estimation for deep penetration in iron is possible among the FLUKA calculations depending on which cross-section set is used. In this particular case, the FLUKA calculation with 260-group self-shielded cross sections provides the most consistent results with that of continuous-energy MCNPX. This example clearly demonstrates the importance of selecting a cross-section set with the degree of self-shielding most suitable to your problem. Without explicitly specified, FLUKA will choose infinite dilute neutron cross sections by default and it is not always suitable.
III.3 A shielding database for proton therapy accelerators

The effectiveness of a point-source line-of-sight model in radiation shielding analysis strongly depends on the availability and appropriateness of associated parameters. The aim of this work is to generate a set of source terms and attenuation lengths suitable for shielding against proton therapy accelerators with reasonable coverage of common beam, target and shielding materials. Validation of source neutron production model and using continuous-energy neutron cross sections make MCNPX an accurate and reliable method for this purpose. A series of MCNPX calculations corresponding to the combinations of these selected parameters were conducted. The shielding data were then derived from a curve fitting process of the resulting depth-dose distributions. Note that the dose attenuation in thick shield can be better approximated by a double-exponential expression, from which two source terms, $H_1$ and $H_2$, and two attenuation lengths, $\lambda_1$ and $\lambda_2$, can be obtained through independent curve fitting for shallow and deep regions in the shield, respectively.

Four target materials (iron, copper, carbon, and tissue) and three shield materials (concrete, iron, and lead) were evaluated in this work in combination with five proton energies from 100 to 300 MeV. Iron is the most common target material for conservative shielding design. Copper is usually the material of beam collimator for different shapes of tumors. The tissue target defined by the ICRU four-element composition could be used for dose evaluation outside the shielding during routine operation for patient irradiation. The carbon target was evaluated because a portion of beam power is expected to be lost at the proton energy degrader made by carbon material. For shielding material, concrete is the most important shielding material in accelerator facilities. Iron is the main material of many accelerator components or structures and lead might be installed for local shielding purpose. All the shielding parameters for various beam-target-shield combinations were collected to become a database for shielding applications. Fig. 5 is a part of the database showing the derived source terms and attenuation lengths as a function of angle for concrete shielding against 250 MeV protons impinging on a thick iron target.
III.4 A practical demonstration on the use of the shielding database

As how the shielding data were generated, their application in a single-layer shielding problem is straightforward by using the same Eq. (1). Fig. 6 shows a demonstration of using the data set for shielding design and dose analysis of a proton therapy accelerator. It is a typical proton therapy treatment room with a maze entrance design, the source term is 250 MeV proton beam hit a thick iron target from the direction of north to south. The dose map in the figure is calculated by a time-consuming MCNPX Monte Carlo simulation. The right-hand side shows a comparison between the Monte Carlo calculated result and the model prediction for a 360-degree dose distribution around the room perimeter. Except the two corners around 140° and 230° and near the maze entrance (~320°), the simple model of Eq. (1) can give quite accurate or conservative estimates on the transmitted dose rates. It proves that the effectiveness and advantage of the database in quick shielding design or dose evaluation for proton therapy accelerators. As to the dose underestimation at these two corners, they are not important in radiation safety because of very low dose rates compared with other area.

As shown in the case, the point-source line-of-sight model cannot predict good results near the maze entrance due to the effect of radiation streaming along the labyrinth. The model fails because the directly transmitted dose is not the major component near the maze entrance. Neutron spectra outside the bulk shielding are completely different from that near the maze entrance. High-energy neutrons dominate the transmitted neutron spectra after thick shielding. However, a lot of thermal neutrons resulting from scatterings along the maze are the major contributor to the dose near the maze entrance. The physics and calculation of radiation streaming is more complex and of course cannot be estimated using such a simple point-kernel method. Does the shielding data set obtained in this work have any help to this problem? The answer is yes! There are many empirical formulae that can be used to evaluate the radiation streaming and attenuation in labyrinths, such as the Cossairt formula in Ref. [8]. The basic principle is simple: the dose at the exit of a section of labyrinth can be estimated by multiplying an appropriate attenuation factor to the dose at the mouth of this section. One can use Monte Carlo simulation to estimate the dose at the maze mouth but it is relatively troublesome and of course time-consuming. In fact, this can be done by directly using the source term obtained in this work corresponding to appropriate beam-target configuration. Fig. 7 shows a comparison between the Monte Carlo calculated dose distribution and the model prediction (Cossairt formula in Ref. [8]) along the maze centerline. They are quite consistent and it proves the usefulness of the database in quick shielding design and dose analysis for proton therapy accelerators.
IV. Summary

Proton therapy accelerators in the energy range of 100 to 300 MeV can potentially produce significant amount of secondary radiation. Neutrons are the dominant dose component. A benchmark calculation for neutron production from proton bombardment with similar energy was performed and the result shows that both FLUKA and MCNPX can give satisfactory prediction on the neutron yield. MCNPX uses continuous-energy cross sections for low-energy neutron transport and therefore is immune from multigroup approximations and self-shielding effects. Transport of low-energy neutrons in FLUKA relies on multigroup cross sections. Selection of proper multigroup cross sections is important and critical for accurate neutron deep-penetration calculations. Based on a series of continuous-energy MCNPX simulations for various beam-target-shield configurations, this work has generated a set of reliable shielding data with reasonable coverage of common target and shielding materials for 100-300 MeV proton accelerators. For shielding analysis of a proton therapy accelerator, the usefulness and effectiveness of this database has been demonstrated through a realistic example.

References


