A Deterministic Dose Calculation Algorithm for Boron Neutron Capture Therapy

Hyegang Chang [a], Sangmin Lee [a], and Sung-Joon Ye [a,b,*]

[a]*Program in Biomedical Radiation Sciences, Department of Transdisciplinary Studies, Graduate School of Convergence Science and Technology, Seoul National University, Seoul, Korea*

[b]*Biomedical Research Institute, Seoul National University Hospital, Seoul, Korea*

[*]*Corresponding author: sye@snu.ac.kr*

1. Introduction

Boron neutron capture therapy (BNCT) is a targeted radiotherapy method. After boron drug injection, boron-10 (B-10) compounds are accumulated in the tumor cells, while the healthy tissue has a lower boron concentration. Neutrons are irradiated at the tumor site that thermalized through the tissues and interact with the B-10 nuclei. B-10, especially having large absorption cross section for thermal neutron, may cause significant decrease in thermal neutron flux in tumor cells. As a result, high linear energy transfer (LET) alpha and lithium particles are produced and destroy the tumor cells selectively.

In contrast to photon radiotherapy, the transport of epithermal neutrons involves radiation components having different relative biological effectiveness (RBE) which therefore need to be considered separately. There are four dose components in BNCT which are boron, nitrogen, fast neutron and gamma doses. Boron dose generated from $^{10}B(\mathrm{n},\alpha)\ ^{7}Li$ reaction, nitrogen dose from $^{14}N(\mathrm{n},\mathrm{p})^{14}C$ reaction, fast neutron dose from $^{1}H(\mathrm{n},\mathrm{n}')\ ^{1}H$ reaction and gamma dose from $^{1}H(n,\gamma)\ ^{2}H$, $^{10}B(\mathrm{n},\alpha)\ ^{7}Li$ and some contaminated gamma rays come from the beam shaping assembly (BSA). An algorithm is required to calculate the influences of each of this dose component in BNCT for computerized treatment planning. [1]

The uniform international dosimetry guidance for BNCT radiotherapy modalities is not exist. Only Monte Carlo (MC) based treatment planning systems (TPS) have been applied in clinical BNCT because of the complex and scatter-dominated nature of the thermal neutron transport. [2] However, dose calculation using the MC stochastic simulation methods are often very time consuming and require huge computing power, which has led to the development of the faster deterministic methods.

Convolution/superposition (CS) is a deterministic dose calculation method using MC simulation results. Usually applied in dose calculation for photon beam radiation therapy. CS calculation speed is faster than the MC method. Furthermore, the MC method result is only slightly more accurate for photon beams than the CS. In this paper, the CS method is applied for BNCT dose calculation algorithm both in gamma and neutron dose components. To do this, 'TEGMA' is suggested first in this paper, the abbreviation of 'Total Energy Generated per unit MAss'. It is a same concept of TERMA - Total Energy Released per unit MAss - used in CS methods, the difference is that it indicates the energy generated by nuclear reaction. TEGMA convoluted with neutron energy deposition kernel and calculate the neutron dose. After dose calculation using CS algorithm, the results are compared with MC simulations in aspect of precision and speed.

2. Materials and Methods

2.1 Convolution/Superposition algorithm

As mentioned above, the CS calculation of radiotherapy dose distributions are traditionally performed for photon beams. Calculation is done in each voxel of the irradiated phantom by convolving precomputed poly-energetic energy deposition kernels with TERMA. [3] The concepts of kernel and TERMA will be explained more in detail on the next section. In this paper, CS algorithm is applied in both gamma and neutron dose calculation for BNCT. The unit sizes of kernel, TERMA, internal gamma fluence and neutron reaction rate used for CS method are $61\times61\times120$, $1\times1\times60$, $61\times61\times60$, and $41\times41\times60$ voxels respectively. The voxel size is 2.5mm × 2.5mm × 2.5mm, composed of brain composition (ICRU-44) with 10ppm boron-10. Convolution is done along z axis, which is parallel to beam direction and superposition is done along x and y axis of beam field. (Fig.1) The dose result comes out after doing convolution and superposition of product and

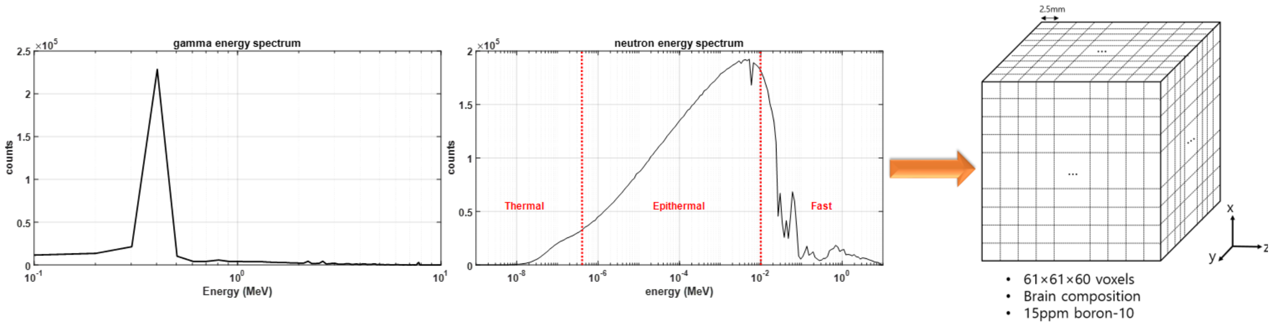

Figure 1. Convolution phantom irradiated with neutron and gamma energy spectra

sum voxel by voxel.

2.2 Dose components

To calculate the BNCT dose by using CS methods, the dose components in BNCT are categorized into five types in this paper. First, they are divided into two broad categories which are gamma dose and neutron dose. Then gamma dose split into two, external and internal gamma dose. The external gamma dose is a gamma ray comes from the BSA which is made during the neutron source production. The internal gamma is generated during the therapy, by nuclear reaction with neutron and materials in human body. The last one is neutron dose and it includes boron, nitrogen, and fast neutron doses. Each of the dose component will be explained in detail on the sections below.

2.2.1. External gamma dose

External gamma dose comes to the patient due to the incident photon from the BSA. The external gamma dose is calculated with traditional CS method. The absorbed dose, D(i,j,k), is expressed as the convolution of the total energy released per unit mass, known as TERMA, with a dose spread kernel. [4]

Here, TERMA is defined as the total energy released during the photon interaction with a material per unit mass, i.e., the kinetic energy of resultant secondary electrons also including energy liberated from other interactions such as excitations. This radiant energy loss by photon interaction can be absorbed locally or at a distant. TERMA depends on the primary photon energy fluence.

The dose spread kernel describes the probability per unit volume of the energy deposited in the vicinity site of the primary photon interaction. The kernel is position dependent. The dose spread kernel results from the sum of energy deposition of a mono-energetic photon at an interaction point (i', j', k') to a volume element at a point (i,j,k).

$$\boldsymbol{D}_{\gamma ex}^{i,j,k} = \sum_{i',j',k'} \sum_{E_g} \boldsymbol{w}_{E_g,\gamma} \cdot \boldsymbol{TER}_{E_g}^{i',j',k'} \cdot \boldsymbol{H}_{E_g}^{i-i',j-j',k-k'}$$

$w_{E_g,\gamma}$: weighting factor

$TER_{E_g}^{i',j',k'} = \Psi_{E_g}^{i',j',k'} \cdot \left(\frac{\mu}{\rho}\right)_{E_g}$: Total Energy Released per unit MAss, TERMA

($\Psi_{E_g}^{i',j',k\prime}$ *:photon energy fluence,* $(\frac{\mu}{\rho})_{E_g}$*: mass attenuation coefficient*)

$H_{E_g}^{i-i',j-j',k-k'}$: kernel

E_g: *Energy group*, 0.1~0.5 MeV ($\Delta E = 10$ keV), 0.5~2.3, 6.5~7.0, 7.5~8.0MeV ($\Delta E = 100$keV)

To calculate the external gamma dose, three parameters are required, weighting factor, TERMA, and kernel. The weighting factor can be obtained from the gamma-ray energy spectrum in figure 1 comes from the BSA of the accelerator. It is made by Monte Carlo simulation, using MCNP6.1 code. This spectrum is divided into 71 energy groups and weighting factors for each energy groups are obtained.

TERMA is calculated manually along the beam direction by multiplying photon energy fluence and mass attenuation coefficient in brain material (ICRU-44) of each energy group. The intervals are 2.5mm up to 15cm.

The kernels are pre-calculated with MC simulations using edknrc, which is EGSnrc code for calculating energy deposition kernel of photon interaction. They were generated in a homogeneous brain composition material with radius of 15cm sphere. The energy deposition kernels are presented in a 2D polar grid. It consists of cones and spheres and the intervals of them are chosen evenly for 1° and 2.5mm respectively. After generating the kernels, their coordinate is transformed from spherical to Cartesian.

2.2.2. Internal gamma dose

Internal gamma dose produced by photons which is induced from neutron capture reaction in tissues. There are two types of internal gamma, one is generated by ${}^{1}H(n,\gamma)\ {}^{2}H$ hydrogen capture. Hydrogen in tissue absorbs thermal neutron and emits 2.2 MeV photons. Other is from ${}^{10}B(\mathrm{n},\alpha)\ {}^{7}Li$ reaction, i.e. n-alpha reaction. From this reaction, about 94% of the recoiled Li-7 ion is produced in an excited state and emits a 0.48 MeV gamma ray during de-excitation.

The phase-space file of neutron source from BSA obtained by MC simulation, using MCNP6.1 code.

$$\boldsymbol{D_{\gamma in}^{i,j,k}} = \sum_{E_g} \boldsymbol{w_{E_g,n}} \cdot \boldsymbol{\Psi_{E_g}^{i,j,k}} \cdot \left(\frac{\mu_{en}}{\rho}\right)_{E_g} \quad (2)$$

$w_{E_g,n}$: neutron source energy weighting factor

$\Psi_{E_g}^{i,j,k}$: photon energy fluence

$\left(\frac{\mu_{en}}{\rho}\right)_{E_g}$: mass energy absorption coefficient

E_g: energy group, 0.1~10 MeV ($\triangle\mathrm{E} = 100$ keV)

The equation of internal gamma dose calculation above requires three parameters, weighting factor, photon energy fluence, and mass absorption coefficient in brain material according to ICRU-44 report. Photon energy fluence times mass absorption coefficient represents the energy deposited per unit mass.

First, dividing internal gamma energy from 0.1 MeV to 10 MeV into 100 groups, interval is 100 keV, then the weighting factors of each energy bin are obtained from the gamma energy spectrum.

The photon energy fluence is a product of photon fluence and energy of the fluence. The photon fluence of each energy group is obtained from MC simulation using MCNP6.1 code with F4 tally. The medium energy of the energy bins is used as an energy for multiplying with photon fluence.

2.2.3. Neutron dose

Neutron dose is caused by the nuclear reaction of a neutron and an atomic nucleus. There are three representative nuclear reactions in BNCT, neutron capture reaction of ${}^{10}B(\mathrm{n},\alpha)\ {}^{7}Li$, ${}^{14}N(\mathrm{n,p}){}^{14}C$, contribute to boron and nitrogen dose respectively, and elastic scattering reaction of ${}^{1}H(\mathrm{n,n'})\ {}^{1}H$, which becomes fast neutron dose.

The neutron source of phase-space file from BSA also obtained by MC simulation using MCNP6.1 code.

$$\boldsymbol{D^{i,j,k}} = \sum_{E_g} \boldsymbol{w_{E_g,n}} \cdot \boldsymbol{TEG_{E_g}^{i,j,k}} \cdot \boldsymbol{k} \cdot \frac{\mathbf{1}}{\boldsymbol{\rho}} \quad (3)$$

k: kernel, k=1 (local deposit)

$\frac{1}{\rho}$: brain density, 1.03 g/㎤

$TEG_{E_g}^{i,j,k} = Q \cdot \sum_{E_g} \cdot\ \phi_{E_g}^{i,j,k}$: Total Energy Generated per MAss, TEGMA

$w_{E_g,n}$: neutron source energy weighting factor

Q: Q-value

$\sum_{E_g}$: macroscopic cross section

$\phi_{E_g}^{i,j,k}$: neutron flux

E_g: energy group, $10^{-9}{\sim}10$ MeV, 199 log scale bins

To calculate the neutron dose, three parameters are required, kernel, brain density, and TEGMA. The concept of equation is adopted from CS equation used for photon dose calculation. The energy is divided into 199 energy groups in log scale. The brain density ρ is required to fit the unit, 1.03 g/㎤ from the ICRP 60 report.

The kernel, same concept of photon energy deposition kernel, signifies the spread of energy generated from the nuclear reaction. The kernel value is assumed to be 1, because the energy is deposited locally. It can be confirmed by the stopping powers of alpha and Li particles generated from the n-alpha reaction. Below 10 MeV alpha and Li particles have a big linear energy transfer of 150 keV/μm and 175 keV/μm respectively, goes a short distance of in-flight only about 4.5 to 10 μm in a tissue. These distances are in the magnitude of the cell diameter, i.e. local deposit.

The total energy generated per unit mass (TEGMA), $TEG_{E_g}^{i,j,k}$is a synonym of TERMA from the CS method for photon, indicates the energy generated from the nuclear reaction. This concept is first proposed in this paper. Weighting factor, $w_{E_g,n}$, calculated from neutron source energy spectrum. Q-value is the amount of energy absorbed or released during the nuclear reaction. In ${}^{10}B(\mathrm{n},\alpha)\ {}^{7}Li$ reaction, Q-values are 2.33 MeV and 2.81 MeV accounts for 94% and 6% respectively. In

$^{14}N(\mathrm{n,p})^{14}C$ reaction Q-value is 0.63 MeV.

Macroscopic cross section is the effective target area of all of the nuclei contained in the volume of the material, for B-10, N-14, and H-1. Multiplying neutron flux to macroscopic cross section represents the reaction rate. It is obtained by MC simulation, using MCNP6.1 with FM tally and fmesh for each energy bins and reaction types.

$$TEG_{E_g}^{i,j,k} = E \cdot \Sigma_{E_g} \cdot \phi_{E_g}^{i,j,k}, E = \frac{1}{2}E_g \quad (4)$$

In case of $^{1}H(\mathrm{n,n'})\,^{1}H$ reaction, instead of Q-value, $\frac{1}{2}E_g$ is used. It assumes that the proton will take half of the incident neutron energy, (in this equation, E_g indicates the average energy of each of the incident neutron energy group) after elastic scattering with neutron.

3. Results and Discussions

CS dose calculation speeds are compared with MC results using MCNP and PHITS codes. With 48-core CPU processors, it took minimum 70 minutes using MCNP to calculate 61x61x60 voxels. PHITS took more time for calculation in same situation. However, with CS methods, using GPU CUDA cores instead of CPU multi-threads, it took only about 34 seconds. The uncertainties in table 1 indicate the stochastic precision error in each code. As CS code is deterministic calculation method, it is impossible to predict the uncertainty.

	CPU (Intel)		GPU		Calculation time	Uncertainty @ 2cm depth
	Cores	Processor base frequency	Cores			
Convolution /superposition	1	2.80 GHz	3,584 CUDA cores	1.5 GHz	34.9 sec	-
MCNP	48	2.67 GHz	-		70 min	Less than 1%
PHITS	4	2.80 GHz	-		24 h	3~5 %
	48	2.67 GHz	-		4 h	

Table 1. Calculation speed comparison between CS, MCNP and PHITS.

In Fig.2, the dose calculation results were calculated with CS, MCNP6.1, and PHITS for pencil beam and 3x3 field size of beam. Each beam field size is 2.5 ㎜ x2.5 ㎜. Using CS shows less than 1% difference with MCNP results. In PHITS, for external and internal gamma, the difference is huge. It reported that for evaluating gamma ray dose, PHITS approximates dose 5% lower than MCNP. [5] Although the difference in internal gamma dose is more than 10%, needs further verification.

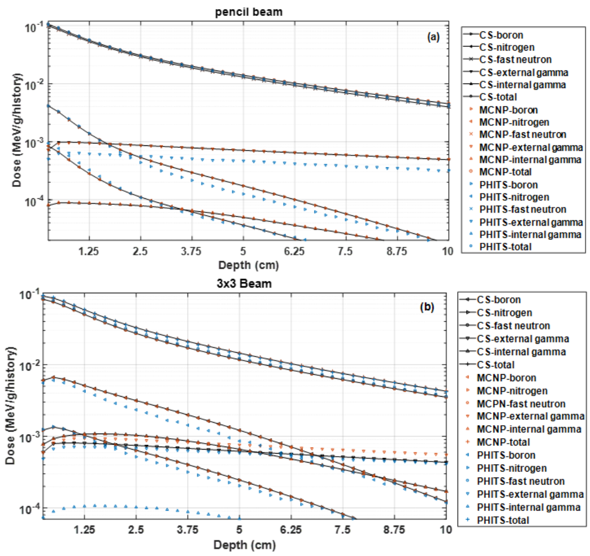

Figure 2. Dose calculation results of CS, MCNP, and PHITS for (a) pencil beam and (b) 3 × 3 cm^2 field beam

4. Conclusions

Neutron and photon dose calculations using a developed CS algorithm showed accuracy comparable to MC dose calculations. Further, the CS algorithm showed superior performance in terms of computation time. Since the developed algorithm seemed to be practical in terms of computational speed and accuracy in busy clinical environment, it would be a calculation engine in BNCT treatment planning system. A CS dose calculation algorithm in heterogeneous media is under development.

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