RADIATION SAFETY ASPECTS OF ^{90}Y BREMSSTRAHLUNG RADIATION PRODUCED FROM RADIATION SHIELDING APPARATUS USING THE MONTE CARLO SIMULATION

by

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Selective internal radiation therapy using an ^{90}Y labelled microsphere is increasingly used to treat hepatocellular carcinoma. Based on its properties, ^{90}Y can produce bremsstrahlung radiation which is essential for post-treatment localisation and dosimetry. However, bremsstrahlung radiation could lead to an increase of radiation exposure of radiation workers. The aim of this work was to determine the ^{90}Y bremsstrahlung radiation produced from the polymethyl methacrylate radiation shielding apparatus using the Monte Carlo simulation. A scintillation detector with a ^{137}Cs standard source was used to validate the Monte Carlo simulation. After validation, the ^{90}Y bremsstrahlung photons spectrum produced from the radiation shielding apparatus was simulated. The radiation equivalent dose rates to the head, neck, body, lower extremities at a distance of 30 centimeters, and finger (contact with the knob) were estimated to be 4.9   0.6, 6.2   0.1, 18.9   0.4, 13.1   0.6, and 3900   50 μSvh^{-1}, respectively. The corresponding annual doses exceeded the limit when radiation workers performed 2631, 1563, 769, and 515 cases per year with contact the knob 3, 5, 10, and 15 minutes per case, respectively. The simulation result showed that radiation exposure of radiation workers and the number of selective internal radiation therapy procedures performed should be considered.

Key words: Monte Carlo, SIRT, ^{90}Y, bremsstrahlung, radiation safety, radiation equivalent dose rate

INTRODUCTION

Selective internal radiation therapy (SIRT) using the ^{90}Y labelled microsphere is increasingly used to treat hepatocellular carcinoma (HCC) and metastatic hepatic tumours. Nowadays, two types of microspheres are available for clinical practices: ^{90}Y labelled glass-microspheres (TheraSphere; MDS Nordion, Ottawa, Canada) and ^{90}Y labelled resin microspheres (SIR-sphere; SIRTEX, Sydney, Australia) [1-3].

Practically, ^{90}Y is a high energy beta particle emitter with the maximum beta energy of 2.27 MeV and average energy of 0.93 MeV. Based on its radiating properties, ^{90}Y can generate bremsstrahlung radiation which is essential for post- treatment localisation and dosimetry [4, 5]. The ^{90}Y bremsstrahlung radiation is produced as a broad spectrum ranging from zero to the maximum beta energy it is emitting. The radiation yield of bremsstrahlung is dependent on the beta energy and the atomic number of the interacting material. Thus, the low atomic number material is typically chosen for shielding of ^{90}Y to reduce the bremsstrahlung photons production [6, 7].

For the ^{90}Y SIRT procedure, the treatment requires processing by an intervention radiologist, radiation technologist and nurse. The ^{90}Y microsphere manufacturer recommends indirectly handling the radiotracer by using the radiation shielding apparatus produced from polymethyl methacrylate (PMMA) also known as acrylic. However, many studies have reported that bremsstrahlung radiation can still be produced from the low atomic number materials while

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performing the ^{90}Y SIRT procedure [6, 8-11]. This might lead to an increase of radiation exposure of radiation workers to ^{90}Y SIRT.

In several studies, shielding properties and bremsstrahlung radiation produced from shielding materials were investigated by using different Monte Carlo simulations such as GEANT4 and MCNP (Monte Carlo N-Particle) [6, 12]. Boukhris et al used the GEANT4 code to study the radiation shielding properties of tellurite-lead-tungsten glasses for gamma and beta radiation [12]. Furthermore, the MCNP were used by many research groups for designing novel shielding materials for photons and neutrons [13, 14].

The primary aim of this work was to determine the extent of the ^{90}Y bremsstrahlung radiation produced from the PMMA radiation shielding apparatus. In addition, the radiation safety aspect of ^{90}Y SIRT was investigated by the Monte Carlo simulation.

MATERIALS AND METHODS

Monte Carlo simulation with MCNP

The MCNP5, Monte Carlo N-Particle 5, is a three-dimensional computational transport code that can be used practically for all particles and all energies [15]. Applications for the code are quite board, including detector design, radiation dosimetry and radiation safety. In this work, MCNP5 is relied on to simulate ^{90}Y bremsstrahlung radiation produced from the shielding apparatus through the user-defined geometry and material compositions. The MCNP is set up in the photon and electron mode with default physics parameters to enable MCNP to track down all source electrons and photons produced from subsequent bremsstrahlung radiation. The discrete energy spectrum of the electrons source defined in the MCNP simulation was based on beta decay of ^{90}Y that is composed of 92.4 keV (0.0000014 %), 518.0 keV (0.017 %) and 2278.7 keV (99.983 %) [16]. Both photons and electrons were simulated and followed when their energies were above 1 keV. The ENDF/B-IV library is used for all atomic and cross section data [17]. Several tally options are available in the MCNP5. The F8 tally was used in this study because it provides the bremsstrahlung energy distribution with the pulse unit [18]. Relative errors of MCNP5 results in this work were limited within 10 % by sufficiently increasing the numbers of simulating particles.

The MCNP validation with experimental measurement

The MCNP is so versatile that it provides a the user with an ability to capture effects from energy broadening and to describe the full width at half maximum (FWHM) of the detector through specifying several parameters in the Gaussian energy broadening (GEB) and special treatment for tallies of FT input cards, respectively. The GEB option was used to simulate a physical radiation detector which is broadening the tallies energy into Gaussian distribution.

$$f(E) = Ce^{-\left(\frac{E-E_0}{A}\right)^2} \tag{1}$$

where E, E_0, C, and A are the broadened energy, unbroadened energy of the tally, normalization constant and Gaussian width. The latter parameter is related to the full FWHM that the desired FWHM is specified by the user-provided constants, a, b, and c as illustrated in eq. (2) [19]

$$\mathrm{FWHM} = a + b\sqrt{E + cE^2} \tag{2}$$

where E is the photon energy in MeV. In this study, three gamma sources (^{241}Am, ^{137}Cs, and ^{60}Co) were used to obtain the FWHM data for determining a, b, and c parameters [19]. For our work, the constants a, b, and c used in equations were $-9.978\ 10^{-3}$ MeV, $8.9944\ 10^{-2}$ MeV$^{1/2}$, and $-3.87858\ 10^{-1}$, respectively.

In order to obtain appropriate parameters for MCNP to subsequently provide reliable assessments on the equivalent dose rates (EDR) on the medical staff, MCNP must be validated against experimental measurements. The experimental setup had a 2-inch × 2-inch thallium-doped sodium iodide (NaI(Tl)) scintillation detector (Mirion Technologies Inc., California, USA) placed in front of a ^{137}Cs standard point source fig. 1.

The detector was previously calibrated before performing all measurements. The MCNP5 input for the NaI(Tl) scintillation detector including the crystal,

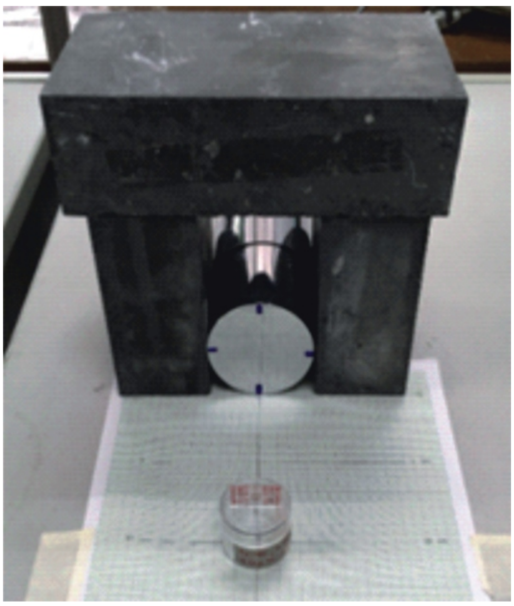

Figure 1. Experimental set-up of the scintillation detector with the ^{137}Cs standard source

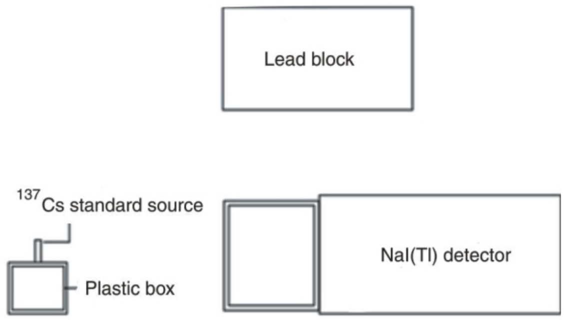

Figure 2. The MCNP5 geometry of ^{137}Cs measured with the NaI(Tl) detector for code simulation

housing and photomultiplier tube (obtained from the manufacturer specification), ^{137}Cs point source and the related geometry were created to resemble the experiment fig. 2.

An acceptable difference between the MCNP result and its associated experimental measurement (*i. e.* corresponding simulated and experimental count rates) is defined to be 2 % or less, therefore, the parameters must be adjusted in a number of MCNP runs during the validation process until an acceptable difference is acquired [20].

Simulation of the ^{90}Y bremsstrahlung photons spectrum and safety aspect in the treatment procedure

After a successful MCNP5 validation, the PMMA shielding apparatus used in SIRT was simulated together with the PMMA vial similar as in a clinical setting. The ^{90}Y bremsstrahlung photons spectrum produced from the radiation shielding apparatus with the NaI(Tl) detector was simulated fig. 3.

To estimate the safety aspects for radiation workers during SIRT, the 3 GBq activity was simulated. This selected activity was based on the maximum activity according to the empiric planning of the ^{90}Y resin microsphere [21]. The simulation was based on the ^{90}Y source placed in the radiation shielding apparatus and then positioned at a 30 cm distance from the simplified geometry phantom model (from the information given in the ICRP (International Commission on Radiological Protection) publication 89 on the basic anatomical and physiological data for use in radiological protection: reference values) [22]. The simulated reference phantom was a homogeneous tissue equivalent material (MS20) with a density of 1.00 gcm^{-3}. The finger equivalent dose was simulated when the finger model (diameter of 1 cm. and length of 5 cm) made contact with the knob of the apparatus fig. 4.

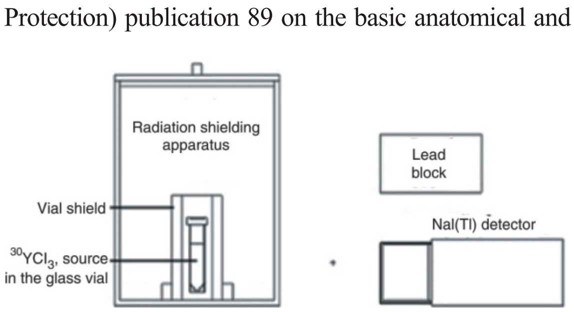

Figure 3. The MCNP5 geometry (midplane of the side view) of ^{90}Y with the radiation shielding apparatus with the NaI(Tl) detector

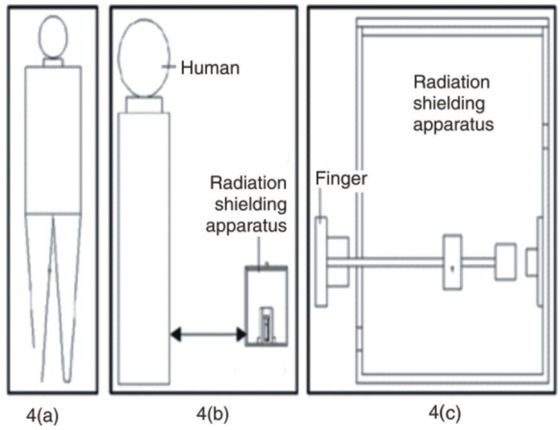

Figure 4. Simplified geometry model of the clinical SIRT condition using data given in the ICRP publication 89 fig. 4(a), the reference phantom positioned at 30 cm from the radiation shielding apparatus fig. 4(b), and the finger model on the knob fig. 4(c)

The EDR to the head, neck, body, and lower extremity from bremsstrahlung radiation produced from the PMMA radiation shielding apparatus with the simplified geometry model were simulated in the Monte Carlo simulations. The EDR is calculated by multiplying the tally particle fluence with the photons flux to the dose rate conversion factor from the ICRP publication 21 and the source activity [23].

RESULTS AND DISCUSSIONS

The NaI(Tl) scintillation detector with the ^{137}Cs standard source was used to validate the MCNP5 code. The GEB option was applied to better simulate a physical radiation detector in which the energy peak exhibits GEB.

Figure 5 shows the count rates obtained from the experimental NaI(Tl) detector and the MCNP simulation with GEB treatment. The total count rates and simulated count rates were aligned in good agreement with a difference of 0.14 %. The ^{137}Cs standard source has been chosen to verify the Monte Carlo simulations compared with the radiation detector measurement in a number of studies due to its wide used to calibrate the NaI(Tl) scintillation detector [24, 25].

In fig. 6 illustrates the simulated spectrum of the ^{90}Y bremsstrahlung radiation with the PMMA radiation shielding apparatus. The spectrum showed a continuous and broad photon spectrum with a peak at 40 keV. Our finding was in line with the previous result which performed the Monte Carlo simulation (MCNP6) of the ^{90}Sr/^{90}Y source with Plexiglas (another trade name of

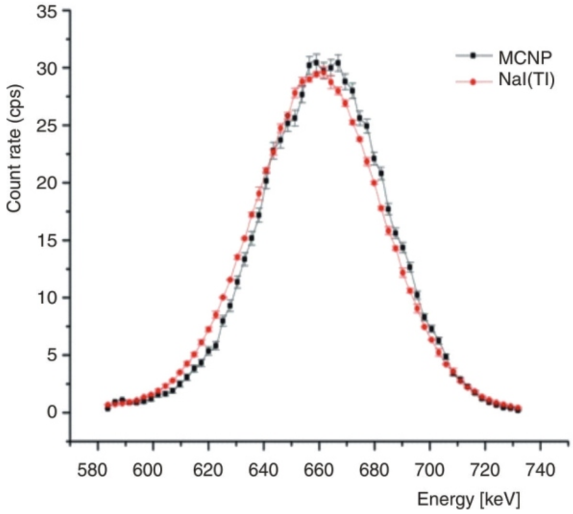

Figure 5. The ^{137}Cs count rates obtained from the experimental measurement with the NaI(Tl) detector and MCNP simulation with GEB; cps – counts per second

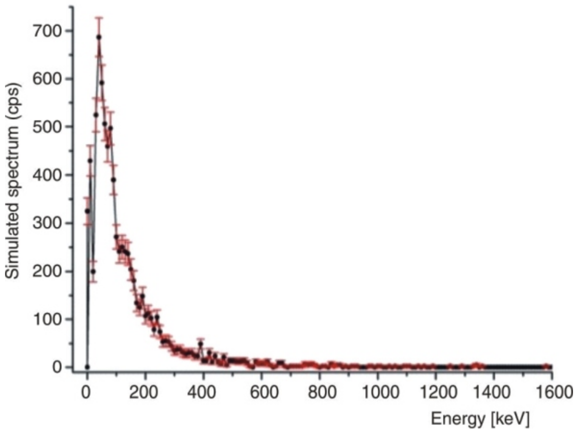

Figure 6. Simulated spectrum of the ^{90}Y bremsstrahlung radiation with the PMMA radiation shielding apparatus; cps – counts per second

PMMA) [20]. The reported peak from the previous study was at 38 keV in the case of Plexiglas. This study also indicated that the production of bremsstrahlung radiation from Plexiglas shielding of ^{90}Sr/^{90}Y was much greater than bremsstrahlung radiation that was produced by ^{32}P and ^{89}Sr. Hence, the bremsstrahlung radiation dose resulting from the usage of ^{90}Y for SIRT should be considered. Also, there is room for improvement on the materials for ^{90}Y shielding, for example barium borate glasses shielding [14, 26].

To obtain further information on the radiation safety aspect, the EDR to head, neck, body, and lower extremities to the radiation worker using the simplified geometry model using data given in the ICRP publication 89 was simulated with the ^{90}Y activity of 3 GBq, at a distance of 30 cm. Consequently, the finger dose was simulated based on the assumption that the intervention radiologist's finger is connected to the knob of the radiation shielding apparatus. The EDR are tabulated in tab. 1.

Table 1. The EDR (Mean SD) to the radiation worker using the simplified geometry model at a distance of 30 cm. from the PMMA shielding apparatus and finger EDR (connected) to the knob)

Organ	EDR [μSvh^{-1}]
Head	4.9 0.6
Neck	6.2 0.1
Body	18.9 0.4
Lower extremity	13.1 0.6
Finger	3900 50

The maximum EDR was at the finger (3900 50 μSvh^{-1}), which touched the knob. According to the 500 mSv ICRP finger dose limit of the radiation worker, the calculated EDR limited the SIRT to 128 patients per year assuming that the radiation worker's procedure lasted 1 hour performed 1 hour per procedure [27]. However, the knob contact time was much lower in the clinical routine. The contact times of 3, 5, 10, and 15 minutes resulted in the equivalent finger doses of 0.19, 0.32, 0.65, and 0.97 mSv per procedure, respectively. These were correlated to 2631, 1563, 769, and 515 cases per year, respectively. The EDR obtained from MC simulations can be reduced in practice by decreasing the contact time.

The study of Mrdja, D., *et al.* [10] evaluated the finger dose from 1-3 GBq of ^{90}Y bremsstrahlung radiation received by the medical staff using Geant4 simulation. The finger dose of 5 mSv (for 10 minutes per procedure) was given which was estimated to 100 cases per year to reach the annual equivalent finger dose limit. The simulated finger equivalent dose from this study was higher than our results. Nevertheless, this further means that the annual finger dose should be taken into account for the SIRT procedure even when the ^{90}Y is well placed in the radiation shielding apparatus. Apart from the finger EDR estimation, our simulation was calculated the body EDR of 18.9 0.4 μSvh^{-1} for 3 GBq. Mrdja et al also reported the dose rate of 20 μSvh^{-1} for similar activity of ^{90}Y placed within shielding. It is important to note that the inconsistency in the finger dose might come from the dissimilarity of the finger model in the simulation, shielding material and simulation code.

CONCLUSION

This work demonstrated that ^{90}Y bremsstrahlung radiation can be produced from the PMMA radiation shielding apparatus. Monte Carlo with MCNP5 can be used as a tool for evaluation of the radiation safety aspect from bremsstrahlung photons in the SIRT procedure. Prior to simulation, the MCNP5 code was validated with the ^{137}Cs standard source and NaI(Tl) scintillation detector. The GEB was applied by using parameters from three gamma sources. The difference between simulation and experiment was well accepted.

From the radiation safety aspect, the simulation result showed that the EDR to head, neck, body, and lower extremities to the radiation worker using a simplified geometry model using data given in the ICRP publication 89 was simulated with ^{90}Y activity of 3 GBq, at a distance of 30 cm. The body EDR for the radiation worker was much lower than the whole-body annual dose limit. However, the finger EDR can possibly reach the annual dose limit when the radiation worker performed 2631, 1563, 769, and 515 cases per year with contact with the knob 3, 5, 10, and 15 minutes per case, respectively. Based on our findings, the finger EDR can be reduced in practice by decreasing the contact time. Future work with regard to the additional protective equipment for the SIRT procedure *e. g.*, novel material for the shielding apparatus should be examined.

AUTHORS' CONTRIBUTIONS

Chayanit Jumpee conceived and simulated the MCNP5, prepared the figures and drafted the manuscript. Chanakarn Onnomdee conducted the experimental procedure, ran the MCNP5, and prepared the tables and figures. Putthiporn Charoenphun supervised and participated in the study design, and coordination. Phiphat Phruksarojanakun supervised and validated the MCNP5 in this work. Krisanat Chuamsaamarkkee is the manuscript corresponding author and provided the research ideas and methods.

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АСПЕКТИ БЕЗБЕДНОСТИ ОД ЗАКОЧНОГ ЗРАЧЕЊА ^{90}Y ПРОИЗВЕДЕНОГ ОД АПАРАТА ЗА ЗАШТИТУ ОД ЗРАЧЕЊА, ПРОЦЕЊЕНИ МОНТЕ КАРЛО СИМУЛАЦИЈОМ

Селективна интерна терапија зрачењем коришћем микросфере обележене ^{90}Y све више се користи за лечење хепатоцелуларног карцинома. На основу особина ^{90}Y може се произвести закочно зрачење које је неопходно за локализацију и дозиметрију након третмана. Међутим, закочно зрачење може довести до повечања изложености радијацији професионалног особља. Циљ овог рада је да се помоћу Монте Карло симулације утврди закочно зрачење ^{90}Y произведено из полиметил метакрилата апарата за заштиту од зрачења. За потврду Монте Карло симулације коришћен је сцинтилациони детектор са стандардним извором ^{137}Cs. Након валидације, симулиран је спектрум фотона ^{90}Y закочног зрачења произведен из апарата за заштиту од зрачења. Еквивалентне дозе зрачења за главу, врат, тело, доње екстремитете на удаљености од 30 центиметара и прст (контакт са дугметом) процењене су на 4,9 0,6, 6,2 0,1, 18,9 0,4, 13,1 0. и 3900 50 μSvh^{-1}, респективно. Одговарајуће годишње дозе премашују дозвољену границу када радијационо особље обави 2631, 1563, 769 и 515 случајева годишње са контактом на дугме 3, 5, 10, и 15 минута по случају. Резултат симулације показао је да треба узети у обзир изложеност радијационих радника и број изведених селективних интерних терапијских процедура.

Кључне речи: Монте Карло, SIRT, ^{90}Y, закочно зрачење, радијациона сигурност, јачина еквивалентне дозе зрачења