ADJOINT ACCELERATION OF MONTE CARLO SIMULATIONS USING TORT/MCNP COUPLING APPROACH: A CASE STUDY ON THE SHielding IMPROVEMENT FOR THE CYCLOTRON ROOM OF THE BUDDHIST TZU CHI GENERAL HOSPITAL

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Full-scale Monte Carlo simulations of the cyclotron room of the Buddhist Tzu Chi General Hospital were carried out to improve the original inadequate maze design. Variance reduction techniques are indispensable in this study to facilitate the simulations for testing a variety of configurations of shielding modification. The TORT/MCNP manual coupling approach based on the Consistent Adjoint Driven Importance Sampling (CADIS) methodology has been used throughout this study. The CADIS utilises the source and transport biasing in a consistent manner. With this method, the computational efficiency was increased significantly by more than two orders of magnitude and the statistical convergence was also improved compared to the unbiased Monte Carlo run. This paper describes the shielding problem encountered, the procedure for coupling the TORT and MCNP codes to accelerate the calculations and the calculation results for the original and improved shielding designs. In order to verify the calculation results and seek additional accelerations, sensitivity studies on the space-dependent and energy-dependent parameters were also conducted.

INTRODUCTION

Taiwan is located in the Circum-Pacific seismic zone and the Buddhist Tzu Chi General Hospital was built in the county where earthquakes happen most frequently. They installed a compact cyclotron to produce short half-life radioisotopes for positron emission tomography. The cyclotron room was originally constructed according to the shielding design based on the methodology and data provided in the NCRP Report 51(1). A plug door was normally applied on the wall of the cyclotron room to simplify the shielding design. Considering the possible damage to the drive mechanism of the heavy plug door by an earthquake, a maze structure was designed in the shielding room of the PETtrace cyclotron(2) installed in the Tzu Chi General Hospital to allow the application of a swing door with reduced thickness and weight, as shown in Figure 1a. During machine commissioning a high-radiation dose rate outside the swing door was observed, which could be attributed to the inadequate maze design and had to be fixed before normal operation.

In this study, an improved shielding design for reducing the dose rate outside the shielding room was performed by applying a full-scale Monte Carlo simulation of the cyclotron room. The Monte Carlo method is considered as the most accurate method available for solving radiation transport problems. The major disadvantage is its excessive computational expense for the simulation of real-world problems. An efficient variance reduction technique had to be used in this study to overcome this shortcoming for testing a variety of configurations of shielding modification, including covering the maze walls with polyethylene plates, reducing the width of maze entrance and installing an additional sliding door across the maze entrance. Although many variance reduction techniques are available, effective use of them for a specific problem is not straightforward and can be very time consuming or error-prone. In principle, all the variance reduction techniques require problem-specific parameters that are dependent on the importance of particles with respect to the objective function. Several importance estimation techniques have been developed and demonstrated to be effective for different types of problems(3–8). Among them, the Consistent Adjoint Driven Importance Sampling (CADIS) methodology provides a solid foundation for the variance reduction technique based on the importance sampling and has proven to be very effective for large/complex real-world shielding problems(9,10). Since the source and transport

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biasing are used consistently and straightforwardly, based only on the deterministic adjoint function, the method is intrinsically easy to understand and features superior computational performance and general applicability. The major difficulty in this method is the requirement of an appropriate adjoint function provided by the discrete ordinates (SN) calculation. The CADIS methodology was implemented in this work by manual coupling of the TORT and MCNP codes(11,12). The TORT run provides deterministic adjoint functions over phase space that could be used to generate source and transport biasing parameters for the MCNP calculation. The biased MCNP calculations result in significant speed-ups over the unbiased case. Owing to the significant computational speed-up, a variety of shielding modifications could be tested and evaluated to improve the original shielding design. A reduction in the width of the maze entrance and an additional sliding door installed across the maze entrance were finally suggested, as shown in Figure 1b, to comply with the stringent dose limit of 10\(\mu\)Sv h\(^{-1}\) for the controlled area.

MATERIALS AND METHODS

Problem description and tools

The PETtrace cyclotron is designed to accelerate protons up to 16.5 MeV and deuterons up to 8.4 MeV. The main products and reactions by bombarding protons and deuterons on targets are \(^{18}\)O(p,n)\(^{18}\)F, \(^{14}\)N(p, )\(^{14}\)C, \(^{16}\)O(p, )\(^{15}\)N, \(^{14}\)N(d,\(n\))\(^{15}\)O and \(^{20}\)Ne(d,\(z\))\(^{18}\)F. The main concern for the shielding design is that the neutron production through \(^{18}\)O(p,n)\(^{18}\)F dominates over other reactions. The neutron production rate, which is also the production rate of \(^{18}\)F, can be obtained by integrating the proton stopping power and the reaction cross section in the IAEA-TECDOC-1211 report\(^{(13)}\). The neutron source intensity of the \(^{18}\)O(p,n)\(^{18}\)F reaction at 16.5 MeV was calculated to be \(8.03 \times 10^{11}\) neutrons s\(^{-1}\) at 100 \(\mu\)A proton beam, which is consistent with that provided by the GE company\(^{(14)}\). Owing to the lack of data availability, the energy and angular distributions of the neutron source were taken from the double differential data of the nearby reaction \(^{14}\)N(p,n)\(^{14}\)O at 17 MeV, as shown in Figure 2, presented in the ICRU Report 63\(^{(15)}\). This substitution is not a key issue since the initial energy and angular distributions of a neutron source are not
so critical to the final result for a shielding problem of deep penetration or multiple scattering.

Figure 1a shows the schematic layout of the cyclotron room at the Buddhist Tzu Chi General Hospital. The cyclotron was considered to be a point neutron source for simplicity. Concrete barriers, including walls and roof, against neutrons and induced gamma rays have a thickness of 2 m. The ceiling is 4 m height. Instead of a thick plug door, an inadequate maze associated with an outlet swing door, consisting of 20 cm polyethylene and 1.5 cm lead, was constructed. The geometrical size of the problem is ~10 m in each axis. Figure 1 also shows the positions of seven scoring detectors used in the Monte Carlo simulations. These detectors could be divided into two groups, one consisting of detectors 1–5 and the other consisting of detectors 6 and 7. Detectors 1–5 were used to examine the characteristics of the radiation transport through the labyrinth and the detectors 6 and 7 were used to evaluate the neutron deep-penetration through the 2-m-thick concrete wall. Neutrons were induced with energy and angular distributions according to those described in Figure 2 for a collimated proton beam with an incident direction towards detector 7. The Monte Carlo code MCNP version 4C with continuous-energy cross section library ENDF60 was used for the calculation of the radiation dose rates including neutron and induced gamma-ray components. The track length estimation was applied to tally the particle flux. The effective dose rates were obtained by multiplying the flux with the dose conversion factors of the AP irradiation. The three-dimensional (3-D) discrete ordinates code TORT was used to calculate the scalar adjoint flux for importance estimation. The multigroup cross section library applied in the SN transport calculation is problem dependent. In this study, since the adjoint function is used only for the variance reduction, it need not be very accurate. Therefore, the choice of the multigroup library is not so important. BUGLE-96, a coupled 47 neutron and 20 gamma-ray group cross section library, was applied in this study and the GIP code was used for pre-mixing material cross section sets for the SN calculations. All the computer programs were run on the off-the-shelf personal computers with the Linux operation system.

**Adjoint acceleration of Monte Carlo simulations**

Deterministic and Monte Carlo methods are two distinct approaches that have been developed to solve the problem of radiation transport for several decades. Each method has its own advantages and disadvantages. They are fundamentally different in many aspects but could be used together in a complementary manner. A review article by Haghighat and Wagner pointed out that the recent trends in advanced Monte Carlo code development have reflected recognition of the benefits of using deterministic adjoint functions for Monte Carlo variance reduction. The adjoint function refers to a particle property, which is the expected contribution or ‘importance’ of a particle with respect to a user-defined objective. This physical meaning makes the adjoint function well suited to the variance reduction of Monte Carlo simulations. Based on deterministic adjoint functions, Wagner and Haghighat developed the CADIS methodology and the Automated Adjoint Accelerated MCNP (A3MCNP) code for complete automation of variance reduction for the MCNP shielding/fixed-source calculations. However, with limitations on single adjoint source and the restricting input of double differential source distribution, the A3MCNP code cannot be applied here directly. We, therefore, adopted a manual coupling approach of the TORT and MCNP codes to accelerate the Monte Carlo simulations by using the deterministic adjoint function. The following steps have been taken to implement the CADIS methodology in this study:

1. Based on the original shielding design, an MCNP input describing this fixed-source/shielding problem in the unbiased mode was prepared. The source, geometry and detectors were modelled accordingly without further simplification. The energy-angle double differential source distribution was integrated with respect to the angle to give the energy distribution first. The angular dependence was then provided for each energy bin by using the dependent source card in the MCNP input. Although it takes a lot of computer time and the statistical uncertainty is still high, this unbiased answer will provide the basis of comparison for checking the convergence and measuring the speed-up of various biased runs afterwards.

2. To prepare the GIP and TORT inputs in the direct (forward) calculation mode, the geometry was discretised into a spatial mesh distribution. A uniform mesh size with dimensions of $50 \times 50 \times 50$ cm$^3$ was applied initially. The geometry of the outlet swing door in Figure 1a was approximated using a single layer of 50-cm-thick polyethylene. The CADIS methodology requires only an appropriate adjoint solution rather than a forward one. This step is not absolutely necessary; the intention was to make it easier to switch to the next step and check the consistency between forward and adjoint calculations.

3. The GIP and TORT inputs were prepared in adjoint mode, by modifying the source and detector description in the previous step. The source in an adjoint problem is equivalent to the detector in the corresponding forward problem. In this case, two adjoint sources were placed at the positions where...
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detectors 5 and 6 were located with source strength given by the flux-to-dose conversion factors. The single adjoint detector was located at the position of the neutron source in the forward mode and its response function was expected to be the same as that of the original source spectrum. Note that the energy-dependent data prepared in the adjoint mode has to be supplied in the adjoint form, i.e. orders reversed with respect to the energy.

(4) A small program was written to manipulate the scalar adjoint flux from the output of the Step (3) and to generate input parameters for the MCNP biased run. Here, we briefly describe the CADIS methodology excerpted from the original papers (9,10). The source particles are sampled from the biased source distribution \( q(\vec{r}, E) \) rather than the original one \( q(\vec{r}, E) \). The relationship between them is

\[
q(\vec{r}, E) = \frac{\phi^+(\vec{r}, E)q(\vec{r}, E)}{R},
\]

where \( \phi^+(\vec{r}, E) \) refers to the space-dependent and energy-dependent adjoint flux and \( R \) is the total detector response. \( R \) could be obtained by two equivalent ways

\[
R = \int_E \int_V \phi^+(\vec{r}, E)q(\vec{r}, E)\,d\vec{r}dE = \int_E \int_V \phi(\vec{r}, E)R(\vec{r}, E)\,d\vec{r}dE,
\]

where \( \phi(\vec{r}, E) \) and \( R(\vec{r}, E) \) are the forward flux and the detector response function, respectively. The statistical weight of the source particle \( w \) must also be corrected accordingly

\[
w(\vec{r}, E) = \frac{R}{\phi(\vec{r}, E)}.
\]

To consider the particle transport process, the transport kernel \( K(\vec{r}', E' \rightarrow \vec{r}, E) \) is altered to a biased one \( \hat{K}(\vec{r}', E' \rightarrow \vec{r}, E) \) given by

\[
\hat{K}(\vec{r}', E' \rightarrow \vec{r}, E) = K(\vec{r}', E' \rightarrow \vec{r}, E) \frac{\phi^+(\vec{r}, E)}{\phi^+(\vec{r}', E')},
\]

and the associated particle weight is modified according to

\[
w(\vec{r}, E) = w(\vec{r}', E') \frac{\phi^+(\vec{r}', E')}{\phi^+(\vec{r}, E)}.
\]

The source biasing of Equation 1 was brought into the MCNP by using the SB input card and the corresponding weight adjustment of Equation 3 was taken care of by the MCNP automatically. The transport biasing of Equations 4 and 5 could easily be implemented by using the superimposed mesh-based weight-window facility in the MCNP after version 4C without any modification of source code. This new feature enables users to set up geometries without the addition of extra cells and surfaces for geometric importance or cell-based weight windows (12). To use the weight-window facility within the MCNP, we provided space-dependent and energy-dependent weight-window lower bounds such that the statistical weights defined in Equation 3 were at the centre of the weight intervals. The default width of the weight interval was employed throughout this work.

(5) The unbiased MCNP input used in Step (1) was modified to accommodate the source and transport biasing techniques mentioned above. It is because the source biasing and transport biasing are treated in a consistent manner, the warning message ‘weight of source particle is above/below window’ does not pop-up during the MCNP execution. The computational efficiency was compared using the figure-of-merit (FOM),

\[
\text{FOM} = \frac{1}{\sigma^2 T},
\]

where \( \sigma \) is the relative statistical error and \( T \) is the computer time (min).

RESULTS AND DISCUSSION

Speed-up of biased run

The full-scale Monte Carlo simulation of the cyclotron room, as shown in Figure 1, consists of two different radiation transport processes: one is the deep-penetration of ~2-m-thick concrete wall and the other is the multiple scattering/attenuation along the rudimentary maze. In the absence of an appropriate \( S_N \) adjoint flux, it is almost impossible to manually construct a 3-D space-dependent and energy-dependent importance distribution for the effective use of variance reduction. Without applying efficient variance reduction techniques, it is impractical to perform repeated Monte Carlo simulations for a variety of different shielding modifications. The CADIS methodology has been proven to be very effective for the variance reduction of deep-penetration problems especially with a large distributed source (10). In this study, we decided to use it as the unique variance reduction method for the shielding problem of a cyclotron room with a point-like neutron source. Two regions of interest at detectors 5 and 6 were selected simultaneously to be the objectives of acceleration. Following the calculation steps (1)–(3) described in the previous section, the detector responses from three different approaches were obtained: unbiased Monte Carlo simulation and deterministic \( S_N \) forward and adjoint calculations. Table 1 lists the total dose rates at detectors 5 and 6 for the original shielding design calculated by
the deterministic TORT and Monte Carlo MCNP codes, respectively, and their computer time for comparison. It is of no surprise that the unbiased Monte Carlo simulation required a computer time of several orders of magnitude longer than the SN calculations to achieve reasonable statistical accuracy. The obvious discrepancies between the Monte Carlo and SN results could be attributed to a rather huge mesh size used in the TORT calculations. However, the aim of the adjoint calculation is not to obtain a precise answer, but rather to generate a function with approximately the correct shape. Thus, very coarse mesh size could be used for saving computer time. In addition, the consistency between forward and adjoint answers of $52 \times 46 \times 34$ mesh lattice in Table 1 makes us quite confident about the deterministic adjoint function that will be used later. As expected, when the mesh number was doubled for each axis the SN results approached the Monte Carlo results at the expense of computer time.

By using the adjoint flux with the correct shape over the whole-phase space, the CADIS methodology can assist the Monte Carlo code in distinguishing where to spend more time in tracking particles in the right direction and where to terminate those in wrong directions in order to achieve a better FOM. Note that, although an approximate $S_N$ adjoint flux is used for implementing variance reduction, the method does not put any restriction on the intrinsic accuracy of the MCNP simulation, e.g. detailed 3-D geometry, continuous energy and angular treatment. According to Step (4) in the previous section, the biased source distribution and the weight-window lower bounds for subsequent biased Monte Carlo simulations were determined. Figures 3 and 4 show the calculated biased source and the spatial distribution of the weight-window lower bounds for the lowest neutron energy group, respectively. The calculated dose rates of detectors 5 and 6 for the original shielding design and the corresponding speed-ups of the CADIS-biased MCNP simulations are listed in Table 2. Comparing the results between biased and unbiased calculations, speed-ups of more than hundred times for neutron tallies and speed-ups of up to several tens of times for induced-gamma ray have been achieved. Although the CADIS performs both source biasing and transport biasing, further investigations indicate that the effect of acceleration almost comes from transport biasing rather than source biasing. It is attributed to two reasons: a point source was assumed and the energy dependence of neutron transport in the neutron source range $(0.1–10$ MeV$)$ was not so crucial in this problem. This also reflects on the spectrum of the biased source that looks almost similar to the original, as shown in Figure 3. In the case of transport biasing, Figure 4 shows an $x–y$ profile at the source level for the distribution of the weight-window lower bounds for the lowest energy group. The lower the weight-window lower bound, the more important is the region’s contribution to the objectives. The weight-window profile in Figure 4 clearly indicates those important pathways for neutron transport from source to the detectors. For detector 5, the dose contribution results from the multiple scattering through the labyrinth, while for detector 6 it comes directly from deep-penetration through the 2-m-thick concrete wall. Accordingly, the MCNP simulation will be spending more time on sampling more important phase spaces and less time for less important ones to improve the overall computational efficiency.

<table>
<thead>
<tr>
<th>Total dose rate ($\mu$Sv h$^{-1}$)</th>
<th>Detector 5</th>
<th>Detector 6</th>
<th>Time (min)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Forward TORT $(26 \times 23 \times 17)^{(a)}$</td>
<td>188.8</td>
<td>2.1</td>
<td>8.5</td>
</tr>
<tr>
<td>Adjoint TORT $(26 \times 23 \times 17)$</td>
<td>82.5</td>
<td>0.7</td>
<td>10.6</td>
</tr>
<tr>
<td>Forward TORT $(52 \times 46 \times 34)$</td>
<td>283.2</td>
<td>0.4</td>
<td>70.8</td>
</tr>
<tr>
<td>Adjoint TORT $(52 \times 46 \times 34)$</td>
<td>209.3</td>
<td>0.3</td>
<td>127.0</td>
</tr>
<tr>
<td>Unbiased MCNP</td>
<td>$331.5 \pm 3.6%$</td>
<td>$1.0 \pm 24.2%$</td>
<td>2341.2</td>
</tr>
</tbody>
</table>

*(a)*The numbers in parentheses are the number of meshes used for TORT $x–y–z$ geometry description. The TORT calculations were performed with P3 Legendre order and S8 symmetric quadrature.
Sensitivity study

Using $S_N$ method to solve radiation transport problems, the geometry of interest must be discretised into spatial meshes to enable the approximation of spatial derivatives with finite differences. The size of these spatial meshes will definitely affect the accuracy of the $S_N$ result. In order to investigate the effect of the mesh size on the adjoint results and the influence of the accuracy of the adjoint results on the variance reduction, two kinds of uniform mesh configurations were tested in this study. As expected, the $S_N$ answers approach to the Monte Carlo result when the mesh number increases and are presented in Table 1. However, the size of the $S_N$ output file has grown by eight times proportionally. The computer time required for the $S_N$ calculation also increases by $\approx 10$ times, but still is much faster than the unbiased Monte Carlo calculation.

Comparing the Monte Carlo results biased by using these two sets of adjoint fluxes with different mesh configurations as shown in the rows labelled as CADIS-biased and double mesh, respectively, in Table 2, we can examine the effect of the accuracy of the adjoint flux on the variance reduction. From this comparison, it is evident that the effectiveness of the variance reduction is not sensitive to the mesh configuration used to generate the adjoint flux.

Similar to discretised spatial variables, the entire energy range of the $S_N$ calculation is divided into many consecutive sub-ranges or groups—a process called the multigroup approximation. The accuracy of the $S_N$ solution is usually dictated by the appropriateness of the group structure and problem-dependent multigroup cross section set used in the $S_N$ calculation. In order to investigate the effect of the group structure of adjoint fluxes used in the biasing process, the original adjoint flux with 47-neutron and 20-gamma-ray groups was collapsed to fewer groups weighted by the corresponding forward fluxes. With these collapsed adjoint fluxes, the weight-window lower bounds were derived in the same way. The results calculated by using a broad series of collapsed adjoint fluxes are also listed in Table 2. From this table, it can be seen that all the dose rates calculated by using different coarse-group adjoint fluxes converge to the same result as that of the calculation with the fine-group adjoint flux (the case denoted as CADIS-biased) if taking the associated statistical uncertainties into account. The calculations with the coarse-group adjoint flux, however, clearly improves the calculation efficiency as reflected from the FOM and Figure 4.

Figure 4. An $x$–$y$ profile of the weight-window lower bounds at the source level for the neutron energy range ($1.000 \times 10^{-11}$ to $1.068 \times 10^{-5}$ MeV).
speed-up figures in Table 2. Figure 5 also shows the normalised FOM for each calculation biased using few-group adjoint flux compared to the reference case with fine-group adjoint flux. It shows that there is an additional speed-up by a factor of 2 or 3 when using the collapsed adjoint fluxes except for the extreme case (1n+1g). One single energy group means completely without energy dependence, which is equivalent to only the geometric importance used for variance reduction. In principle, the more energy groups used, the more accurate is the adjoint flux. However, the most accurate adjoint flux does not guarantee the best computation efficiency. The overall computation efficiency must also consider the computational overhead associated with the process of weight checking when particles are crossing space/energy boundaries and undergoing interactions. If more energy groups are involved, more computer time has to be spent on weight checking between energy group boundaries. In this case, the dependence of the FOM on the number of adjoint energy groups was rather insensitive and we adopted 8-neutron and 4-gamma-ray group structure as our standard configuration used for the variance reduction.

### Shielding modification

For the original shielding design of the cyclotron room, the dose rate at detector 5 was far exceeding the dose-rate limit of 10 μSv h⁻¹ for the radiation working area required by the regulatory authority. It was obviously caused by the inadequate maze design, because large areas directly illuminated by neutrons from the cyclotron were visible from the outlet door. The resulting high-dose rate was

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**Table 2. Calculated dose rates of detectors 5 and 6 for the original shielding design and the speed-ups of the CADIS-biased Monte Carlo simulations for different cases.**

<table>
<thead>
<tr>
<th>Detector 5</th>
<th>Neutron tally</th>
<th>Gamma-ray tally</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Dose rate (μSv h⁻¹)</td>
<td>Error (%)</td>
</tr>
<tr>
<td>Unbiased</td>
<td>6.79E+01</td>
<td>15.99</td>
</tr>
<tr>
<td>CADIS-biased</td>
<td>6.33E+01</td>
<td>1.63</td>
</tr>
<tr>
<td>SB only</td>
<td>4.98E+01</td>
<td>34.69</td>
</tr>
<tr>
<td>WW only</td>
<td>6.42E+01</td>
<td>1.69</td>
</tr>
<tr>
<td>C(1n+1g)</td>
<td>6.17E+01</td>
<td>2.96</td>
</tr>
<tr>
<td>C(2n+1g)</td>
<td>6.20E+01</td>
<td>3.55</td>
</tr>
<tr>
<td>C(5n+2g)</td>
<td>6.47E+01</td>
<td>3.95</td>
</tr>
<tr>
<td>C(6n+3g)</td>
<td>6.23E+01</td>
<td>3.42</td>
</tr>
<tr>
<td>C(8n+4g)</td>
<td>6.83E+01</td>
<td>3.07</td>
</tr>
<tr>
<td>C(12n+5g)</td>
<td>6.70E+01</td>
<td>3.80</td>
</tr>
<tr>
<td>C(24n+10g)</td>
<td>6.00E+01</td>
<td>3.02</td>
</tr>
<tr>
<td>Double mesh</td>
<td>6.48E+01</td>
<td>4.10</td>
</tr>
</tbody>
</table>

**Detector 6**

|            | Dose rate (μSv h⁻¹) | Error (%) | FOM | Speed-up |
| Unbiased   | 1.55E−01           | 45.46      | 2.10E−03 | 1.0 |
| CADIS-biased | 8.98E−01           | 4.30       | 2.60E−01 | 123.8 |
| SB only    | 3.66E−01           | 100.00     | 3.90E−03 | 1.9  |
| WW only    | 9.05E−01           | 4.21       | 2.70E−01 | 128.6 |
| C(1n+1g)   | 1.05E+00           | 9.67       | 6.90E−02 | 32.9  |
| C(2n+1g)   | 9.33E−01           | 9.28       | 4.90E−01 | 233.3 |
| C(5n+2g)   | 1.02E+00           | 11.50      | 3.80E−01 | 181.0 |
| C(6n+3g)   | 8.07E−01           | 10.24      | 4.70E−01 | 223.8 |
| C(8n+4g)   | 8.76E−01           | 11.21      | 3.80E−01 | 181.0 |
| C(12n+5g)  | 9.61E−01           | 9.81       | 4.70E−01 | 223.8 |
| C(24n+10g) | 7.62E−01           | 8.87       | 4.50E−01 | 214.3 |
| Double mesh | 8.81E−01           | 7.91       | 3.10E−01 | 147.6 |

(a) CADIS-biased Monte Carlo simulations consist of both source biasing and transport biasing. ‘SB only’ means only the source biasing was applied while ‘WW only’ means only transport biasing through weight-window technique was used.

(β) C(xn+ yg) refers to the case of CADIS-biased Monte Carlo simulations by using the collapsed adjoint flux (x neutron groups and y gamma-ray groups).

(c) Double mesh represents the numbers of meshes in each axis (x, y, z) were doubled in mesh description; therefore, the total number of meshes was increased from the original 26 × 23 × 17 to 52 × 46 × 34.
confirmed by the Monte Carlo simulation at detector 5 that exceeds a level of 300 μSv h⁻¹ (Table 1). The neutron-induced gamma rays are the dominating component, with a fraction of ~80% of the total dose rate, as shown in Table 2. This shielding weakness must be improved to comply with the regulation requirement before normal operation. Although the Monte Carlo method usually gives the most accurate answer, it takes far more computer time without proper use of variance reduction techniques. However, with orders of magnitude improvement in the calculation efficiency by applying the variance reduction using \(S_N\) adjoint flux, it makes possible to carry out repeated Monte Carlo simulations to test a variety of ideas for modification of the original shielding design. We have investigated several possible solutions, e.g. covering the maze walls with polyethylene plates, increasing the thickness of the outlet swing door, enhancing the shielding effectiveness of the 1-m-thick maze wall, reducing the width of the maze entrance and installing an additional sliding door across the maze entrance. These shielding modifications were expected to reduce the dose rate outside the swing door to different extents. The lower the dose rate the fewer the particles that travel from the source to detectors. It thus makes the Monte Carlo simulations of these modified shielding designs more difficult than those of the original case. Therefore, the variance reduction is indispensable for Monte Carlo simulations of these shielding modifications. Since the adjoint flux used in the variance reduction is not sensitive to the space-mesh configuration and the energy-group structure as studied in the previous section, the same adjoint flux used in the calculations of the original shielding design was used directly in the variance reduction applied to the Monte Carlo simulations of the modified shielding designs. The details of these calculations are in another paper by the authors²⁰. It was found that the most effective measure of the shielding improvement was to reduce the width of the maze entrance to 1.5 m and install an additional sliding door with 25-cm-thick polyethylene and 5-cm-thick lead layers across the maze entrance, as shown in Figure 1b. This modification effectively shielded the areas directly illuminated by source neutrons that were visible from the outlet swing door. Consequently, both neutron and induced gamma-ray dose rates at detector 5 were reduced by a factor of a great many tens and the total dose rate was then below the dose limit of 10 μSv h⁻¹. The dose rates at detector 6 originally being below the dose limit were almost unaffected by this shielding modification. All the biasing schemes used for the original shielding design were applied again to this case. The calculated results are listed in Table 3 using the same format of Table 2 for convenient comparison. Apparently, the efficiencies of the CADIS-biased Monte Carlo simulations for the improved shielding design are as good as those of the original case, although the biasing scheme was not developed for it. It thus demonstrates again the general applicability and effectiveness of the CADIS methodology for the Monte Carlo variance reduction in this category of problems.

To further investigate the effectiveness of shielding enhancement on the characteristics of the radiation field, neutron energy spectra of detectors 1, 5, 6 and 7 were scored and the results are shown in Figures 6 and 7 for the original and the improved shielding design, respectively. For the convenience of interpretation, the neutron spectra may be divided into two regions: the source neutron region (0.1–10 MeV) and the thermal neutron region (<0.4 MeV). Comparing both spectra at detectors 1 and 5 in Figures 6 and 7, it is observed that the maze and the outlet swing door play a dual role of providing an attenuation of the neutron flux by roughly four orders of magnitude and degrading the neutron spectra to the ones dominated by thermal neutrons. It can also be found that the shielding enhancement resulted in a depression of neutron fluxes at detectors 1 and 5 by roughly two orders of magnitude. The spectrum at detector 7, which locates just opposite the source, consists dominantly of source neutrons and a small amount of scattered neutrons. The spectrum at detector 6, which locates behind the 2-m-thick concrete wall, on the contrary, consists dominantly of thermal neutrons and a small fraction of source neutrons. Note that the very sharp peak at 2.35 MeV in the spectrum at detector 6 obviously resulted from the cross section window of \(^{16}\text{O}\) at that energy. From the comparison of neutron spectra at detectors
6 and 7, it is observed that the 2-m-thick concrete wall provides an attenuation of neutron flux by roughly seven orders of magnitude. It can also be found from the comparison of neutron spectra at detector 6, between Figures 6 and 7, that the shielding enhancement has also resulted in a small fraction of depression of thermal neutrons at detector 6. This depression may be attributed to the reduction of the scattering areas.

**Comparison with measurements**

This work aims to present an effective method for the shielding calculation of an actual medical cyclotron facility by using full-scale Monte Carlo simulations with variance reduction. The emphasis was on the detailed description of the variance reduction technique applying the deterministic adjoint flux. A variety of sensitivity studies carried out in the previous sections has verified the calculation method. However, in order to check the accuracies of many physical parameters adopted in the calculation process, a comparison of the final computed results with measurement data has also been made. The dose rates at detector 5 were measured using conventional gamma-ray and neutron survey meters at the proton current of 35 μA. After normalising to 100 μA proton current, the gamma-ray and neutron dose rates were determined to be 171 and 34 μSv h⁻¹, respectively, for the original shielding design. Compared with the corresponding best estimate of the computed results of 263 and 63.3 μSv h⁻¹ as shown in Table 2 (CADIS-biased), the discrepancies were 54 and 86 % for gamma-ray and neutron dose rates, respectively. The agreement is satisfactory, taking into account that various physical parameters were respectively. The agreement is satisfactory, taking into account that various physical parameters were respectively. The agreement is satisfactory, taking into account that various physical parameters were respectively.
involved, among others, the source strength, the source characteristics and the flux-to-dose rate conversion factors; and that the self-shielding of the cyclotron body has been ignored in the calculation. The actual engineering implementation of the shielding improvement has made some modifications to the improved shielding design. The 1-m-thick maze wall was extended to narrow down the maze entrance with a 70-cm-thick wall consisting of 40 cm polyethylene, 8 cm lead and 22 cm concrete. At the end of the extended wall an opening with dimensions of 90 cm width and 190 cm height was reserved and a sliding door consisting of 20 cm polyethylene and 2.5 cm lead was installed. The gamma-ray dose rate at detector 5 after shielding improvement was measured to be 6.8 mSv h$^{-1}$ normalised to 100 μA proton current, which happened to be the same as the computed result as shown in Table 3 (CADIS-biased) for
the improved shielding design. The neutron dose rate was reduced to below the minimum detectable level of the neutron rem counter.

SUMMARY

Despite continuous and significant increase in available computational resources, unbiased Monte Carlo simulations are still impractical for some real-world applications. Choosing effective variance reduction techniques is very important. Full-scale Monte Carlo simulations of the cyclotron room of the Buddhist Tzu Chi General Hospital were carried out to investigate the most effective way to improve the inadequate maze design. We adopted a manual coupling approach of the TORT and MCNP codes for accelerating the Monte Carlo simulations by using deterministic adjoint flux. The TORT/MCNP manual coupling approach based on the CADIS methodology has been explored to provide significant achievement on computation efficiency. Basically, the CADIS utilises an $S_N$ adjoint flux for the variance reduction through the source biasing and consistent transport biasing. With the use of this method, the computational efficiency was increased significantly by more than two orders of magnitude compared to the unbiased run and the statistical convergence was also very reliable. In order to verify the calculation results and seek for additional accelerations, sensitivity studies on the space-dependent and energy-dependent parameters used in this approach were also performed.

With orders of magnitude improvement on the computation efficiency by using the variance reduction, it is possible to carry out repeated Monte Carlo simulations to test a variety of ideas for modification of the original shielding design. It was found that the most effective measure of the shielding improvement was to reduce the width of the maze entrance and install an additional sliding door across the maze entrance. Both neutron and induced gamma-ray dose rates outside the outlet swing door were effectively reduced to comply with the dose limits set by the regulation. Neutron energy spectra at different locations were also investigated for the original and improved shielding designs. This study has demonstrated the high effectiveness and flexibility of the TORT/MCNP manual coupling approach for the Monte Carlo simulations in problems of the cyclotron room shielding. The only drawback of this approach is that it requires the user to be highly knowledgeable in the Monte Carlo and the deterministic methods as well as the codes. However, owing to its significant improvement on Monte Carlo simulations and great flexibility to meet user's needs, it is worth the effort for a large/complex shielding problem. A comparison has also been made of the final computed results with the measurement data.

The good agreement ensured the propriety of the various parameters and assumptions adopted in the calculation process.

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REFERENCES


