Monte Carlo Simulations for Shielding Analysis of the TR-24 Cyclotron at INRNE-BAS

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Abstract. The Institute for Nuclear Research and Nuclear Energy at the Bulgarian Academy of Sciences is building an accelerator laboratory to operate a TR-24 cyclotron. Thus, a preliminary radiation shielding analysis of the accelerator bunker is required. For this purpose Monte-Carlo simulations have been performed and dosimetric (Dose equivalent) quantities have been estimated in two model geometries - simplified spherical geometry and full-scale bunker, respectively during operation and after the end of the cyclotron life. Our current efforts are directed to the production of $^{18}$F thus in all the conducted simulations a water target enriched with $^{18}$O is considered. NiGa$_3$ as a target for production of $^{68}$Ge has also been simulated and a comparison of the radiation shielding characteristics of the bunker for the two targets has been made.

1. Introduction

The task undertaken by the Institute for Nuclear Research and Nuclear Energy at the Bulgarian Academy of Sciences (INRNE-BAS), the construction of a cyclotron centre, is complex and has many aspects [1, 2]. One of these is the radiation shielding analysis of the facility. For this purpose numerical simulations are performed, based on the Monte Carlo method, by employing the FLUKA code [3, 4]. This approach is known to be reliable for shielding analysis of low and high energy accelerators [5–8]. INRNE-BAS has acquired a TR-24 cyclotron, produced by Advanced Cyclotron Systems Inc., which is capable of generating a proton beam with energy in the interval of 15 to 24 MeV and current up to 400 $\mu$A. An accelerator with such characteristics can be employed for the production of a large variety of positron-emission tomography (PET) and single-photon emission computed tomography (SPECT) radioisotopes.

In the current paper, we consider two model geometries representing the cyclotron bunker - simplified spherical geometry and full-scale bunker. We also take into account two types of concrete as a material for the bunker walls - ordinary concrete with Portland cement (CPC) and low activation concrete (LAC) with limestone.

Since the number of imaging procedures in Bulgaria using fluorodeoxyglucose ($^{18}$F-FDG) keeps increasing over the past ten years [9], the first objective of the future cyclotron centre is the production of $^{18}$F-FDG. The nuclide $^{18}$F is produced with a cyclotron primarily by proton irradiation of an $^{18}$O enriched water target, and therefore the generated secondary neutrons and gamma-rays should be taken into account in the radiation shielding analysis of the facility. We use the full-scale bunker geometry to make estimations of the neutron and the gamma ambient...
dose equivalent during the target irradiation. In this case, we simplify the target to a cube of 2.5 ml of enriched water. In these simulations, the target is directly irradiated with a proton beam of 24 MeV and the results obtained are with acceptable accuracy. However, studying the gamma ambient dose equivalent distribution a day or a month after the 20 years of cyclotron operation is more challenging in terms of obtaining results with reasonable accuracy. The usual solution is either to increase the number of primary particles (in this case protons) or the number of runs which increase simulation run time. In our case we employ a two-step approach, i.e. two simulations run sequentially. In the first simulation, by direct irradiation solely of the target, the fluence energy spectrum of the secondary particles is obtained. Then, in the second step, the target is replaced by a point source of secondary particles (such as neutrons or gamma-rays) with the respective, previously obtained fluence energy spectrum. We employed the two-step approach to make estimations of the gamma ambient dose equivalent distribution, for different local shielding materials, after the cyclotron is permanently shut down. These simulations are performed in the spherical model geometry. A neutron point source is used for the second step because the secondary neutrons are the main cause for activation of the bunker walls and the local shielding.

Another objective of the cyclotron centre is the production of radioisotopes that emerge as new alternatives to the currently used ones in nuclear medicine. A good example is $^{68}$Ge which is produced by $(p, x)$ reaction on natural gallium. $^{68}$Ge decays to $^{68}$Ga, which is a possible replacement of $^{111}$In [13]. Here, our first step is to check the expected $^{68}$Ge yield by simulating NiGa$_3$ target irradiation with FLUKA. Next, a comparison of the full-scale bunker shielding characteristics, for the NiGa$_3$ and $^{18}$O enriched water targets, is performed.

2. Description of the models

In Figures 1 and 2, the two studied model geometries are schematically presented. Figure 1 shows a "melon slice" of the simplified spherical model geometry. In its centre, it has an air-filled sphere with a radius of 2 m. This sphere is surrounded by a spherical shell (in red) with a thickness of 250 cm.

![Figure 1. Melon slice of the simplified spherical model geometry.](image1.png)

![Figure 2. Full-scale bunker model geometry (seen from above).](image2.png)

The two-step approach is employed for this geometry. In its geometrical centre, enclosed by a local shielding (in blue in Figure 1), the neutron point source is positioned. The neutron fluence energy spectrum (Figure 3) is calculated in the first step, where we simulated the irradiation of a simplified target representing a cube of 2.5 ml of enriched water. The target is surrounded by air and the energy spectrum (Figure 3) of the secondary particles is scored on the surface of a sphere with a radius of 2 cm enclosing the target. The geometry represented in Figure 1 is used to study the effectiveness of five types of local shieldings with thickness of 20 cm, respectively:
a mixture (M1) of 40% paraffin and 60% carbon steel (CS); a mixture (M2) of 40% paraffin and 60% ferro-boron (FeB); paraffin (P); borated polyethylene (BP); a sandwich shielding structure (SSS) consisting of two 10 cm layers - a layer of paraffin and a layer of M2. The local shielding material recipes that we used are: for P [10] - 14.9% H, 85.1% C with density $\rho = 0.93 \text{ g/cm}^3$; for BP [10] - 12.5% H, 10% B, 77.5% C with density $\rho = 1 \text{ g/cm}^3$; for CS - 98.6% Fe, 0.18% Si, 0.031% S, 0.031% P, 1% Mn with density $\rho = 7.8 \text{ g/cm}^3$; FeB [11] - 72.2% Fe, 21.9% B, 3.4% C, 2.4% O, 0.1% Si, 0.1% Al with density $\rho = 6.32 \text{ g/cm}^3$. As it was already mentioned, in these simulations we have considered two types of concrete - CPC and LAC. The chemical composition of the CPC is: 1% H, 0.1% C, 52.9% O, 1.6% Na, 0.2% Mg, 3.4% Al, 33.7% Si, 1.3% K, 4.4% Ca, 1.4% Fe and its density is $\rho = 2.3 \text{ g/cm}^3$ [10]. For the LAC, it is, respectively: 0.721% H, 8.915% C, 47.772% O, 0.076% Na, 0.24% Mg, 0.275% Al, 1.241% Si, 0.033% K, 40.514% Ca, 0.063% Fe, 0.088% S, 0.008% Cu, 0.034% Sr, and its density is $\rho = 2.2 \text{ g/cm}^3$ [12]. One important difference, between these two types of concrete, is the concentration of hydrogen, since it is a good moderator of the neutron flux.

Figure 3. Neutron fluence energy spectrum of $^{18}$O enriched water target for $^{18}$F production.

Figure 2 shows the full-scale bunker geometry with internal dimensions: 17 m (length), 9 m (width) and 3.65 m (height). Its outer wall thickness (walls, ceiling, floor) is 2.5 m, and the internal walls are 60 cm thick. The arrow indicates the proton beam direction and the target position.

3. Results and discussion

In the current paper, we present our results for the neutron and gamma ambient dose equivalent ($H^*$(10)) distribution and attenuation during irradiation (full-scale bunker) and after 20 years of operation (spherical geometry). The ambient equivalent doses are calculated using the fluence-to-dose coefficients provided by the ICRP74 [14].

3.1. Simplified spherical geometry

This simple geometry is useful when one needs to study the effectiveness of local shielding around the target. Its advantage is that it gives a clear physical picture without the unnecessary complications caused by the complexity of the real bunker geometry.

In Figure 4, the attenuation profiles of the gamma ambient dose inside the local shielding and in the spherical shell consisting of CPC are presented. It is clearly seen, that without local shielding (——) the gamma ambient dose, inside the air-filled sphere, is high - 700 $\mu$Sv/h after one day (Figure 4 (a)) and 60 $\mu$Sv/h after one month (Figure 4 (b)). On the other hand, when there is local shielding, the higher gamma ambient dose equivalent is caused by the activation of the local shielding itself when materials containing high Z elements as Fe, Mg are used, respectively, M1 (-----), M2 (--), SSS (○), as shown in Figure 4. Not only will the personnel conducting manipulations around the target receive a higher dose, but also the local
Figure 4. Gamma ambient dose equivalent attenuation profile (spherical geometry) for (a) 1 day and (b) 1 month of cooling after 20 years of cyclotron operation. No local shielding CPC (——); P (---); BP (······); M1 (—-·---); M2 (—-·---); SSS (◊)

Figure 5. Gamma ambient dose equivalent attenuation profile (spherical geometry) for (a) 1 day and (b) 1 month of cooling after 20 years of cyclotron operation. No local shielding CPC (——); no local shielding LAC (---); CPC and P (······); LAC and P (—-·---); CPC and BP (—-·---); LAC and BP (◊)

shielding will become additional radioactive waste. The standard radiation shielding materials P (---) and BP (······) show the best result, as the overall gamma ambient dose is lower, which is presented in Figure 4. In the case of P shielding the gamma ambient equivalent doses inside the air-filled sphere are 80 µSv/h (Figure 4 (a)) and 7 µSv/h (Figure 4 (b)), respectively, after one day and one month of cooling. For BP the respective values are even lower - 50 µSv/h after one day (Figure 4 (a)) and 3 µSv/h after one month (Figure 4 (b)) of cooling.

The comparison of the gamma ambient dose equivalent attenuation profiles of LAC and CPC without local shielding and, respectively, when local shielding of P or BP is applied, is presented in Figure 5. For LAC without local shielding (---) the gamma ambient dose rate is 20 to 30 times lower than in the case of CPC without local shielding (——), respectively 35 µSv/h after one day (Figure 5 (a)) and 2 µSv/h after a month (Figure 5 (b)) of cooling. For the combination of LAC and P (—-·---) the gamma ambient equivalent dose is - 4µSv/h after one day (Figure 5 (a)) and 0.20µSv/h after a month (Figure 5 (b)). For LAC and BP (◊) the gamma ambient doses are 2µSv/h after one day (Figure 5 (a)) and 0.09µSv/h after a month (Figure 5 (b)) of cooling. Overall, the results in Figures 4 and 5 show that LAC without local shielding is a better solution than the combinations of CPC and local shielding considered in this paper.

3.2. Full-scale bunker
The assessment of the gamma and neutron dose equivalent distributions inside and outside the bunker is an important task when shielding analysis of the accelerator facility is done. In the
next simulations we irradiate directly an enriched $^{18}\text{O}$ water target, positioned in the full-scale bunker, with a proton beam with energy 24 MeV and 100$\mu$A current.

**Figure 6.** Distribution of (a) gamma and (b) neutron ambient dose equivalent in the full-scale bunker geometry and CPC walls.

**Figure 7.** Distribution of (a) gamma and (b) neutron ambient dose equivalent in the full-scale bunker geometry and LAC walls.

**Figure 8.** Attenuation profiles of (a) gamma and (b) neutron ambient dose equivalent in the outer bunker wall, behind the target. In blue for CPC wall, in red for LAC wall.

The gamma (Figures 6 (a) and 7 (a)) and neutron (Figures 6 (b) and 7 (b)) ambient dose equivalent distributions, respectively, in the case of CPC (Figure 6) and LAC (Figure 7) are shown in Figures 6 and 7. There is no substantial difference in the distribution of the gamma ambient dose equivalent (Figures 6 (a) and 7 (a)) and in its attenuation profile in the wall behind the target (Figure 8 (a)). Since the hydrogen concentration is lower in LAC, the neutron ambient dose equivalent outside the bunker wall is higher (Figure 7 (b)) than in the case of CPC (Figure 6 (b)). The attenuation profile shown in Figure 8 (b) demonstrates that it is about 13
times higher, respectively 0.10 µSv/h (for LAC) and 0.008 µSv/h (for CPC). However, in both cases it is lower than the usual background radiation in Sofia, Bulgaria (0.14-0.2 µSv/h). Based on these results we can state that the 250 cm wall thickness is enough to provide radiation protection for the personnel outside the bunker.

3.3. NiGa$_3$ target for production of $^{68}$Ge

The production of newly emerging radioisotopes is the next milestone of the future cyclotron centre at INRNE-BAS. One of the options for a TR-24 cyclotron is $^{68}$Ge. Our first step in this direction was to compare the $^{68}$Ge yield, from a thick target, obtained by FLUKA simulations with results from tabulated recommended data, published in IAEA report [15], for the $^{nat}$Ga(p, x)$^{68}$Ge reaction. The comparison shows good agreement (Figure 9).

![Figure 9. Comparison of $^{68}$Ge yield, from $^{nat}$Ga: FLUKA simulations (in red) and tabulated recommended data published in IAEA report (in blue) [15].](image)

![Figure 10. Distribution of (a) gamma and (b) neutron ambient dose equivalent in the full-scale bunker geometry.](image)

![Figure 11. Attenuation profiles of (a) gamma and (b) neutron ambient dose equivalent in the outer bunker wall, behind the target. In blue enriched $^{18}$O water target, NiGa$_3$ target in red.](image)
In Figure 10 (a) gamma and (b) neutron ambient dose equivalent distributions are shown. In this case, a NiGa\(_3\) target, for production of \(^{68}\text{Ge}\), is irradiated with a 24 MeV proton beam with 100 \(\mu\)A current and the material of the bunker walls is CPC. Also, a comparison between the attenuation profiles in the case of an \(^{18}\text{O}\) enriched water target and a NiGa\(_3\) target is made (Figure 11). As expected, in the case of NiGa\(_3\) the ambient equivalent doses outside the bunker walls are lower, respectively, 2.5 times for the gamma-rays (Figure 11 (a)) and 5 times for the neutrons (Figure 11 (b)).

4. Conclusion
The FLUKA Monte Carlo simulations in the simplified spherical geometry have shown that the low activation concrete without local shielding is a better solution for the TR-24 cyclotron at INRNE-BAS than any combination of ordinary concrete and local shielding. The presented studies suggest that the standard radiation shielding materials (paraffin and borated polyethylene) are a good choice for local target shielding. A proper combination of concrete and local shielding reduces the activation of the bunker walls substantially.

The simulations conducted with the full-scale bunker model show that 250 cm thick outer walls provide substantial radiation protection for the personnel working outside the bunker.

The results of our FLUKA simulations for the \(^{68}\text{Ge}\) yield are in good agreement with the tabulated recommended data from the IAEA report. The comparison of the ambient dose equivalent attenuation profiles during the irradiation of H\(_2\)O (enriched with \(^{18}\text{O}\) for production of \(^{18}\text{F}\)) and NiGa\(_3\) targets shows that the dose equivalent is higher for the first case.

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