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Dose evaluation in a portable D–T neutron generator facility by Monte Carlo method

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Abstract
A portable NG-9 D–T generator can produce fast neutrons with 14 MeV energy in a maximum yield of 4×10^8 n/s. Shielding around the neutron emitter is necessary to protect the operators from radiation exposure. MCNP5 was used to evaluate the dose rates in a generic building with an activation room and corresponding six surrounding rooms. Several designs of shielding structure were evaluated to ensure that the total dose rates in surrounding rooms are all less than 5 μSv/h. The annual operating time can reach to at least 4081 h around the activation room for the final design.

Keywords D–T neutron generator · Monte Carlo method · Dose estimation · Neutron and gamma shielding

Introduction

Electronic neutron generators are widely used in the fields of medical research [1], environmental monitoring [2], explosive and drug detection [3, 4], landmine detection [5], neutron radiography [6], industry [7, 8] and biophysical applications [9]. Compared with nuclear reactors, accelerator neutron sources are portable and easy to operate. Furthermore, electronic neutron generators can be controlled by electric power supply making them safer than nuclear reactors and radioactive sources. D–T (deuterium–tritium) neutron generator is a kind of accelerator neutron source which produces isotropic, monoenergetic neutrons at 14 MeV by $^{2}H + ^{3}H \rightarrow ^{4}He + n$ fusion reaction. During operation, the doses for neutrons and gamma rays generating from fast neutron (n, n′ γ) and thermal neutron (n_{th}, γ) reactions can cause radiation damage to staff. It is necessary to provide suitable radiation shielding and to assess the doses protecting workers from radiation exposure. Generally, hydrogen-containing materials are placed around the generator as moderators to slow down fast neutrons, and boron-containing materials will be used for thermal neutron shielding. The lead will be selected for gamma-ray shielding on the external side of the boron-containing materials. Many kinds of researches have reported the shielding structure around different neutron sources and generators [10, 11]. Does rate profiling studies around neutron generators have been studied earlier [12–14]. Combinations of different materials have been performed to improve the shielding performance [15, 16].

Model NG-9 portable D–T neutron generator (manufactured by Northeast Normal University, Changchun City, Jilin Province, China) has been applied in the field of coal quality analysis [17] and neutron therapy [18]. While in operation, appropriate shielding around neutron emitter must be set up to maintain the total dose within the limit of 20 mSv per year for operators recommended by the International Commission on Radiological Protection (ICRP) publication 116 [19] and the dose rates limit 5 μSv/h for workers according to the recommendation of ICRP publication 60 [20]. In general, operating neutron generators in expansive space and keeping a long distance between neutron generators and operators are better ways for radiation protection than shielding [21]. However, neutron generator equipment is often not in an independent space and covers a limited area. MCNP5 [22] was used to construct a generic building and was used for simulating the transport of neutrons and photons in the facility. The activation room in which the neutron generator is installed is likely to be on the middle floor of the building, usually with offices in the upstairs and downstairs rooms and

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surrounding rooms on the same floor. Necessary shielding structure around the neutron emitter must be designed to make sure the dose rates to operators in surrounding rooms would be within the prescribed limits.

Our work aims to provide operators a safe neutron generator facility used for educational or scientific researches based on model NG-9 neutron generator. The neutron and photon dose rates were calculated in the surrounding rooms under several shielding designs around the neutron emitter so that the total dose rate to operators in each surrounding room would be within 5 μSv/h recommended by ICRP 60. The annual operating time based on the final design is calculated to keep the annual effective dose to operators under 20 mSv recommended by the ICRP 116.

Materials and methods

NG-9 neutron generator

The NG-9 neutron emitter with 4.3 cm in radius and 89 cm length is consisted of a neutron tube and corresponding high voltage multiplier circuit, which are enclosed in a stainless steel cylinder filled with sulfur fluoride insulating gases (as shown in Fig. 1). The weight of the neutron emitter is about 8.8 kg. Reservoir, Penning ion source, accelerator, and target enclosed within a vacuum-tight enclosure construct the neutron tube (as shown in Fig. 2). The neutron tube is a cylinder with 2.5 cm in radius and 10 cm length, and the total weight does not exceed 0.6 kg. The Penning ion source in the neutron tube is a low gas pressure, cold cathode ion source with crossed electric and magnetic fields and it is used to generate deuterium (D^+) and tritium (T^+) ions. The ion source voltage is normally between 2 and 7 kV. Positive charged mixed plasma of D^+ and T^+ irons generated from Penning ion source are accelerated by the acceleration system and

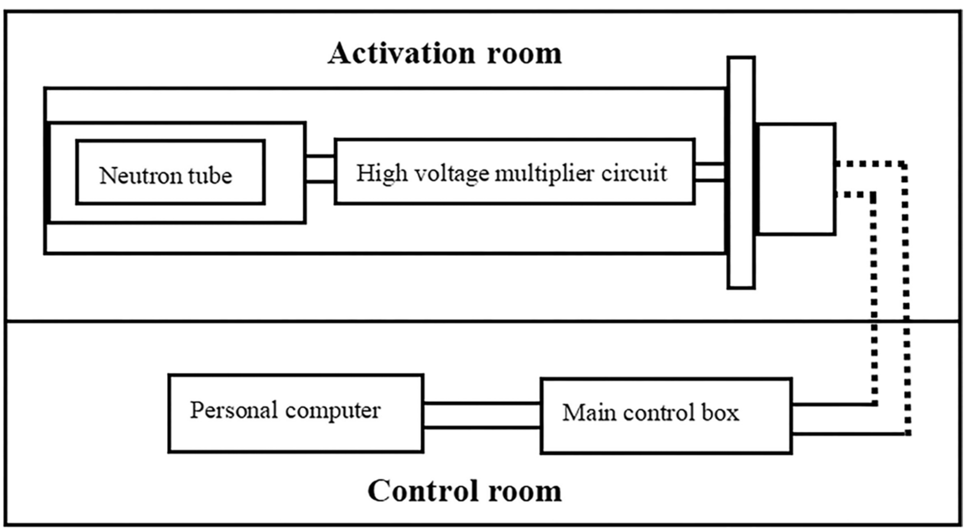

Fig. 1 Schematic of the neutron generator control circuit

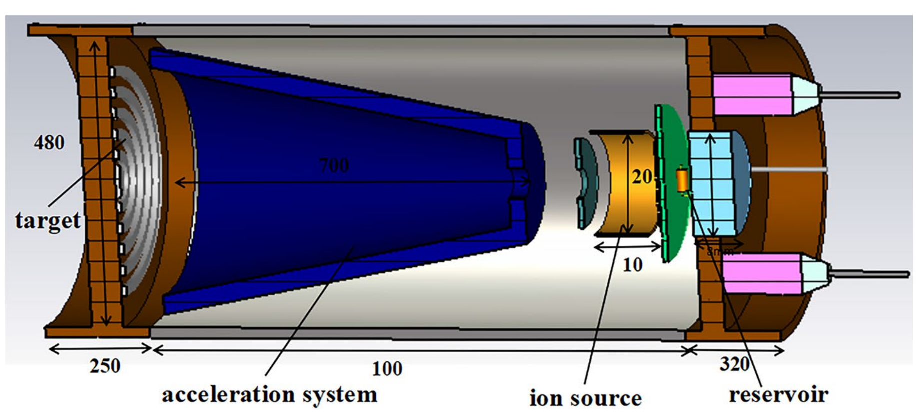

Fig. 2 Main components of NG-9 neutron tube (units: mm)

collide with a titanium target coated with aluminum oxide, where occurred the D–T reactions producing 14 MeV neutrons. The accelerator voltage is normally between 100 and 120 kV.

The control circuit of the whole neutron generator is shown in Fig. 1. The main control box with a weight of 5.0 kg is connected with the neutron emitter. Users can adjust the ion source current, reservoir current and accelerator voltage of the neutron emitter through the control box. Besides, the neutron tube can be operated in continuous wave (CW) or pulse mode by adjusting the parameters in the control box. In our work, the neutron tube is working under CW condition. A personal computer is connected to the main control box through a cable. By adjusting the parameters of the control panel on the computer, the main control box is remotely controlled and then the neutron tube is operated. The output of the neutron generator depends on the adjustment of the ion source, reservoir, and accelerator when the neutron generator is turned on each time. A maximum yield can reach to 4×10^8 n/s when the acceleration voltage is 120 kV and the ion source current is 480 μA. The weight of the total system (except for the computer) is less than 14.4 kg. The cost of the whole system is less than $100,000.

MCNP model

In this study, we used MCNP5 to model the geometric structure of the activation room and surrounding rooms to estimate the radiological dose rates. As shown in Fig. 3, the activation room was in the center of the whole arrangement. The dimension of the activation room is $600 \times 350 \times 300$ cm, which is a general structure of an ordinary building in China. The NG-9 neutron emitter was surrounded by high-density polyethylene (HDPE) moderator with a density of 0.950 g/cm^3, 5% borated polyethylene (BPE) with a density of 0.955 g/cm^3 neutron shielding and Pb gamma shielding ($\rho = 11.340$ g/cm^3), respectively. The NG-9 with shielding was placed in the center of the activation room. The coordinate of the target was (0, 0, 0) in the simulation. The X-axis and Y-axis were along the axis and the radial direction of the neutron emitter, respectively. As shown in Fig. 3, a cavity ($20 \times 40 \times 10$ cm) filled with air was installed along the Y-axial in the right direction to the target from the surface of the neutron emitter. The sample can be put in the cavity for fast neutron therapy and material science. Users can adjust the ratio of thermal neutrons by adding a moderator with appropriate thickness in the cavity for neutron activation analysis. In our simulation, a 0 cm moderator between neutron emitter and the cavity was considered to obtain a maximum dose during operation. Another six rooms were placed along the X-axis (Z1, Z2), Y-axis (Z3, Z4), Z-axis (Z5, Z6) having the same geometry of $600 \times 350 \times 300$ cm. Additionally, the thickness of load-bearing walls and ceilings (made of ordinary concrete with a density of 2.300 g/cm^3) around the rooms are 18 cm and 12 cm respectively.

Monte Carlo calculations

Three different shielding constructions were tested using MCNP5 simulation to obtain the best design making the dose rates in the rooms around the activation room admissible for operators. In each construction, the shielding around the NG-9 emitter was modified, while the position and dimension of neutron emitter (with 4.3 cm in radius and 89 cm length) were unchanged. Calculations were performed by setting up the neutron source at Ti-target as a mono-energetic point source of 14 MeV. Neutron and secondary gamma-ray doses were produced after neutron passing through the shielding structure. The flux values in each room around the activation room, due to the isotropic point source of 14 MeV at origin (0, 0, 0), were calculated using the F4 card (average flux over a cell). DE and DF tally cards were used to transform particle flux into Ambient dose equivalents, H*(10) [23]. The flux-to-dose rate conversion factors for neutrons and photons were obtained using the

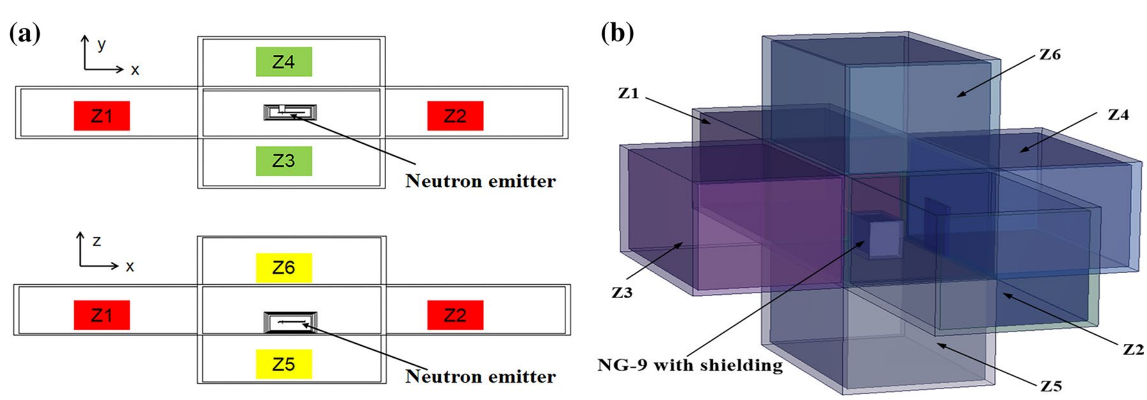

Fig. 3 **a** General arrangement of the activation room and surrounding rooms, **b** 3D view

ICRP publication 74 [24]. In our simulation, the particle history was set to 1×10^7 and the statistical uncertainty of each output result was less than 5%. Finally, H*(10) per history were multiplied by 4×10^8 n/s, which is the maximum yield for NG-9 neutron generator.

Case #1

This design of the neutron emitter with shielding materials is shown in Fig. 4. In the first simulation of this study, the neutron emitter was surrounded by HDPE. The thickness of HDPE along the X-axis was 30 cm and 20 cm, along the Y-axis was 30 cm and 30 cm and along the Z-axis was 50 cm and 35 cm, respectively. The HDPE was surrounded by BPE and Pb with a thickness of 5 cm, respectively.

A higher thickness of 30 cm HDPE moderator is needed on the side close to the Z1 room because of the position of the target inside the neutron emitter. Additionally, the distance between the neutron emitter and Z5 room is relatively short compared with the Z6 room due to the Z5 room is directly on the lower floor of the activation room. Therefore, a thicker thickness of 50 cm HDPE is required between the neutron emitter and the lower floor on the side close to the Z5 room.

Case #2

This design of the neutron emitter with double shielding materials is shown in Fig. 5. In the second simulation of this study, another cover made of 5 cm-thick of BPE and 5 cm-thick lead was used to cover the shielding structure, while

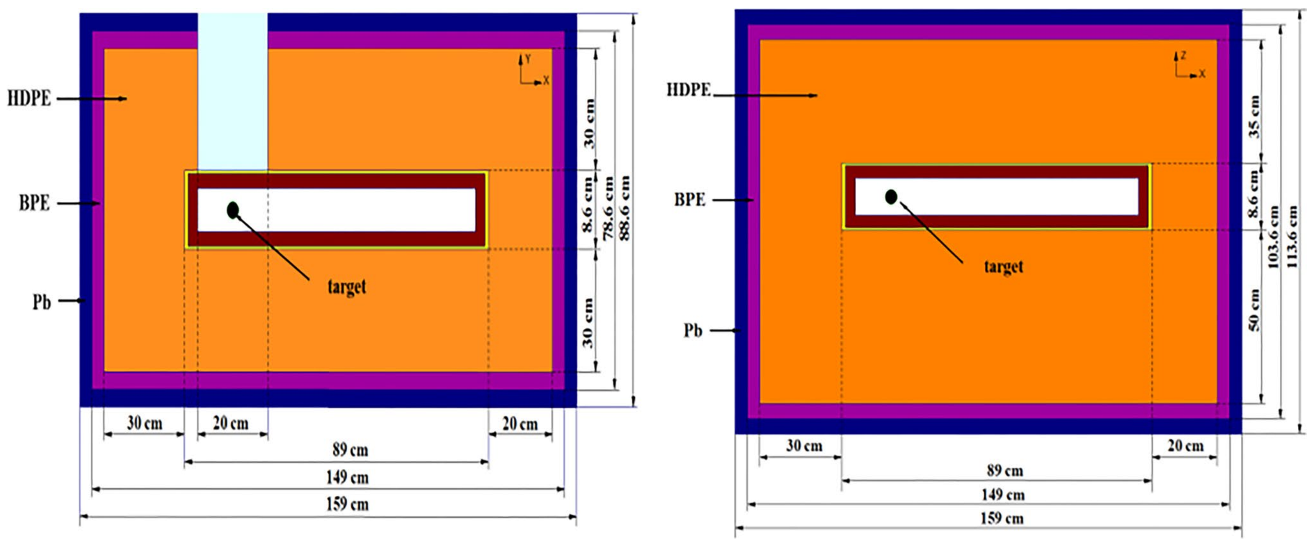

Fig. 4 Case #1: MCNP5 dimensions of NG-9 neutron emitter with shielding materials [X–Y cross-section (left), X–Z cross-section (right)]

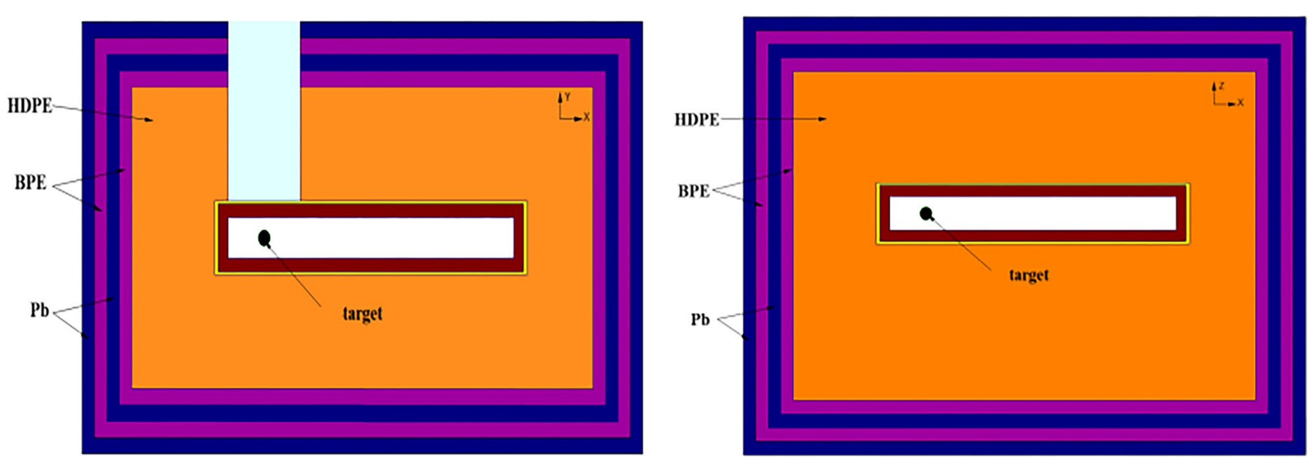

Fig. 5 Case #2: MCNP5 model of NG-9 neutron emitter with double shielding materials [X–Y cross-section (left), X–Z cross-section (right)]

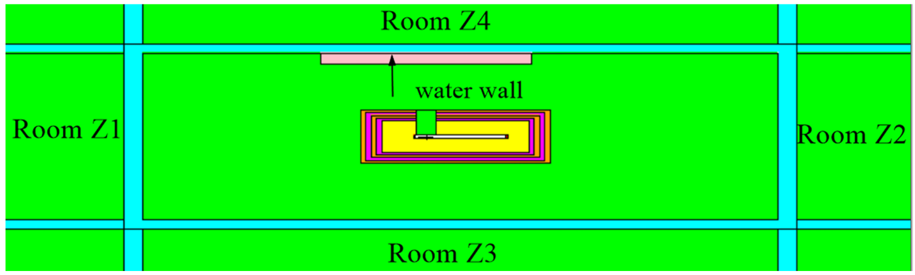

Fig. 6 Case #3: MCNP5 model of NG-9 neutron emitter with shielding material lined with the wall in face of the cavity in the activation room (for scale)

Table 1 Dose rates in the surrounding rooms in Case #1 (see Fig. 4) for a 4×10^8 n/s neutron yield (a relative precision at 1 sigma is about 10%)

Location	Neutron dose rate (μSv/h)	Photon dose rate (μSv/h)	Total dose rate (μSv/h)
Z1	3.7 ± 0.4	0.3 ± 0.1	4.0 ± 0.4
Z2	0.8 ± 0.1	< 0.1	1.0 ± 0.1
Z3	6.8 ± 0.7	0.5 ± 0.1	7.2 ± 0.7
Z4	32 ± 3	1.1 ± 0.1	33 ± 3
Z5	12 ± 1	0.8 ± 0.1	13 ± 1
Z6	10 ± 1	0.7 ± 0.1	11 ± 1

Table 2 Dose rates in the surrounding rooms in Case #2 (see Fig. 5) for a 4×10^8 n/s neutron yield (a relative precision at 1 sigma is about 10%)

Location	Neutron dose rate (μSv/h)	Photon dose rate (μSv/h)	Total dose rate (μSv/h)
Z1	1.6 ± 0.2	< 0.1	1.7 ± 0.2
Z2	0.4 ± 0.1	< 0.1	0.5 ± 0.1
Z3	2.7 ± 0.3	0.2 ± 0.1	2.9 ± 0.3
Z4	20 ± 2	0.7 ± 0.1	20 ± 2
Z5	5.0 ± 0.5	0.4 ± 0.1	5.4 ± 0.5
Z6	4.3 ± 0.4	0.3 ± 0.1	4.6 ± 0.4

the shape and dimensions of the HDPE moderator, and the shielding layer of 5 cm BPE and 5 cm Pb in Case #1 were unchanged.

Case #3

This design of shielding material lined with the wall [15] near the cavity in the activation room is shown in Fig. 6. In the third simulation of this study, H_2O (200 × 24 × 180 cm) was selected as the shielding material because of its economy and ease of access, which is usually injected into iron or plastic containers to shield neutrons. The "water wall" was in face of the cavity with 200 cm length along the X-axial, 24 cm width along the Y-axial and the height from the floor to top of the "water wall" along the Z-axial was 180 cm. In Case #3, the shape of the shielding structure in Case #2 kept constant.

Results and discussion

Monte Carlo simulations were performed for the design of the best shielding structure around the NG-9 neutron emitter. The neutron, photon and total dose rates obtained in the six surrounding rooms (Z1–Z6) based on Case #1 are shown in Table 1. The HDPE around the neutron emitter has the ability to moderate fast neutrons. BPE is an ideal shielding material for thermal neutrons because of the absorption cross-section of boron is large (767 barns) [12], which can absorb neutrons leaking out of the device and reduce the neutron dose around the neutron emitter. The gamma rays produced in HDPE and BPE can be absorbed by adding lead to the outermost layer of the shielding structure. In all rooms, the dose rates due to neutrons are larger than the dose rates due to photons. Most of the total dose rates are higher than the accepted limit recommended by ICRP publication 60 (5 μSv/h). The neutron and photon dose rates reach the maximum at the Z4 room which is probably due to the existence of the cavity. The total dose rates in Z1 and Z2 room are relatively low because of the distance from NG-9 neutron emitter to Z1 and Z2 rooms are further (the distance along X-axis in the direction of Z1 and Z2 is 600 cm in the activation room).

To reduce the dose rates in the surrounding rooms, another layer of 5 cm BPE and 5 cm Pb was added to the outer surface of lead in Case #1. The simulation results for Case #2 are shown in Table 2. The dose rates for neutrons and photons decrease due to the increase of shielding materials. The dose rates in Z3, Z5, and Z6 rooms show a significant reduction compared to Case #1.

However, the dose rates in the Z4 room are 20 ± 2 μSv/h for neutrons and 0.7 ± 0.1 μSv/h for photons, which are still too high due to there are fewer materials over the cavity. A "water wall" was designed to line with the wall in the face of the cavity in the activation room to reduce the dose rates in the Z4 room in Case #3. As seen in Table 3, the dose rates delivered to staff can reach to an acceptable level with 4.6 ± 0.5 μSv/h for neutrons and 0.3 ± 0.1 μSv/h for photons in Z4 room, and the total dose rates obtained in all of the rooms are within the recommended limit 5 μSv/h. The

Table 3 Dose rates in the surrounding rooms in Case #3 (see Fig. 6) for a 4×10^8 n/s neutron yield (a relative precision at 1 sigma is about 10%)

Location	Neutron dose rate (μSv/h)	Photon dose rate (μSv/h)	Total dose rate (μSv/h)
Z1	1.5 ± 0.2	< 0.1	1.6 ± 0.2
Z2	0.3 ± 0.1	< 0.1	0.4 ± 0.1
Z3	2.6 ± 0.3	0.2 ± 0.1	2.8 ± 0.3
Z4	4.6 ± 0.5	0.3 ± 0.1	4.9 ± 0.5
Z5	4.6 ± 0.4	0.4 ± 0.1	4.9 ± 0.5
Z6	4.0 ± 0.4	0.3 ± 0.1	4.3 ± 0.4

shielding material lined with the wall in Case #3 has a significant impact on the dose rates in the Z4 room. Meanwhile, the dose rates in other rooms also decreased slightly.

In order to maintain a safe environment for operators in a portable neutron generator facility, the accumulated dose in the surrounding rooms also must be within acceptable limits. Annual effective doses recommended by the ICRP 116 [19] are 20 mSv per year for operators. As shown in Table 3, the maximum dose rate in Case #3 is 4.9 ± 0.5 μSv/h existing in rooms Z4 and Z5 for a 4×10^8 n/s neutron yield, and the corresponding annual operating time is 4081 h for operators. The annual operating time in room Z1, Z2, Z3, and Z6 are 12,500, 50,000, 7142 and 4651 h, respectively. Compared with the Adelphi DD-110 neutron generator for a 1×10^9 n/s neutron yield installed at National Centre of Nuclear Sciences and Technologies [25], the annual operating time in our work based on NG-9 neutron generator with a 4×10^8 n/s neutron yield is at least 4081 h higher than that of 2000 h. This is mainly due to the difference in neutron yield and the dose rates around the activation room. Anyway, both the shielding of NG-9 and DD-110 provide a shielding reference and a safe working environment for operators.

Conclusions

In this study, Monte Carlo simulations were carried out to estimate dose rates for neutrons and photons in a generic building installed an NG-9 neutron generator. The building was modeled as an activation room (600 × 350 × 300 cm) with six similar rooms on six sides of the cuboid. Three designs of shielding structures around the neutron emitter were investigated. Simulations were proposed to evaluate systemically the shielding effects to find the best shield to minimize the neutron and photon dose rates in surrounding rooms. According to the simulation results, the best shielding structure is obtained with Case #3, having HDPE around the neutron emitter with 30 cm and 20 cm thickness along the X-axis, with 30 cm and 30 cm thickness along the Y-axis, with 50 cm and 35 cm thickness along the Z-axis. Double layers of shielding with one layer of 5 cm BPE and 5 cm Pb and the other layer of 5 cm BPE and 5 cm Pb are laid around the HDPE moderator. A water wall (200 × 24 × 180 cm) is lined in face of the cavity. In this case, the total dose rates in all rooms are within the limit 5 μSv/h recommended by ICRP 60. The annual operating time can reach to 4081 h for operators for a maximum neutron yield 4×10^8 n/s under Case #3. In other words, operators can work in the building for at least 4081 h in a safe condition. This study provides useful information for users on the radiation protection of operators. It can also be used to optimize the shielding design of activation rooms for neutron activation analysis, educational researches and material science.

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