Original Article

Design and Optimization of A Shield for Am-Be Source to Reduce Radiation Hazards Using MCNPX Code

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Abstract

Aim: Protection against nuclear radiation is one of the most important issues in nuclear technology and industries that use this technology. Among the types of radiation emitted from radioactive sources, neutron and gamma rays are among the most dangerous radiations due to lack of electrical charge and serious damage to living tissues. The principal challenge in radiation protection is the proper design of a shield against neutron and gamma radiations. Hence, this study has investigated the improvement of the protection against these radiations. **Materials and Methods:** This study is of applied-developmental and quantitative type. Calculations have been performed using the MCNPX code in this study, which is one of the strongest nuclear calculation codes. The data were analyzed using quantitative statistics and ORIGIN software (OriginLab company, 1992, Northampton, Massachusetts, USA). **Results:** Based on the results, utilizing the spherical geometry had a better performance to increase the neutron fluxes in comparison with the cylindrical and cubic geometries. Moreover, polyethylene with high density was selected as the best moderator. Ultimately, it was dealt with the comparison and selection of the best protection to minimize the produced gamma rays due to the absorption of neutrons in different materials used by the source and neutrons that run away from the outer surface of the source configuration. **Conclusion:** Using the composite sphere of paraffin and polyethylene with high density up to a radius of 12 cm and tungsten with a thickness of 1 cm was suggested as the final configuration for the aim of this study. In comparison to the no-protection mode, this protection is effective to 74% in reducing the neuron dosage and 55% in reducing the primary gamma-ray, while the mentioned protection is about 72%–73% effective in reducing the general dose.

Keywords: Am-be source, mixed radiation field, Monte-Carlo computations, protection

Introduction

Nuclear technology in today's world has become an integral part of our life in various fields such as medicine, energy production, agriculture, and industry. Protection against nuclear radiation to prevent its harmful effects is one of the fundamental issues in this technology.[1-5] The basic principles in radiation protection are time and distance, and among them, protection is known as the most effective method. Due to the different nature of the beams, the importance of protection against the beams is also different.

Neutron and gamma radiations are the most important and dangerous nuclear radiations due to their very high penetration into tissues and bodies, and if these rays are not attenuated with proper protection, they can cause irreparable live tissue damage and radiation-sensitive electronic equipment. Proper protection for these radiations also reduces the effect of other radiations.[6,7] Neutron radiation protection is based on the acceleration of fast neutrons and the absorption of thermal neutrons.

For this reason, neutron protection shields are usually composed of two layers. The first layer consists of materials containing light elements with a high absorption cross-section through which neutrons are slowed down due to elastic scattering, and the second layer is of high atomic weight materials to absorb slow neutrons as well as secondary gammas produced by the reaction (n-γ). Heavy elements are also used to protect gamma radiation.

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A neutron is an unstable particle and is produced in different ways. Nuclear reactors are known as the most abundant producing neutron sources. Nuclear reactors are very large, and concerns about their safety and high maintenance costs have made their use impossible in many cases.[9] Isotopic sources have replaced nuclear reactors as another appropriate method for neutron production, the most common of which is the ^{241}Am-Be source.[10] The design of an effective radiation protection shield depends on the choice of material type, its thickness as well as its arrangement.[11] Reducing the exposure of people or sensitive devices to radioactive radiation is the most important goal of protection. In the protection shield, computations, determining the thickness and composition of materials in its construction to reduce the dose of radiation at certain points, are proposed to significantly reduce their risks.[12]

The MCNPX code is one of the strongest nuclear computing codes based on the Monte-Carlo method.[13] This computational code is used to analyze the transport of neutron, photon, electron, proton, and other particles alone and together. This code is used in almost all cases that are somehow related to different types of beams.[14]

Materials and Methods

MCNPX code has been used for simulation in this research. Three spherical, cylindrical, and cubic geometries in the same volumes have been used, which surrounded the ^{241}Am-Be cylindrical source to investigate the effect of the geometric shape of the protection shield. The flux of thermal and nonthermal neutrons per unit volume of these geometries was assessed. After selecting the geometric shape of the protection shield, the types of moderator, reflectors, and neutron and gamma attenuators [Table 1], as well as the appropriate thickness of each layer, were examined and compared, respectively. Finally, the types of layer arrangements (layer displacements) as well as their composites were studied to reduce the total neutron and gamma dose rates as much as possible.

In this research, the simulation results are divided into two parts: the first part is the computation related to neutron flux per unit area and unit volume, which are calculated by the F2 and F4 tallies, respectively, and the second part is the calculations related to the equivalent dose rates for which the flux-to-dose functions in MCNPX code are used. International Commission on Radiation Units and Measurements (ICRU) reference sphere was used to compare different protections. The ICRU sphere was placed at a radius of 15 cm at a distance of 130 cm from the source and filled with a substance made of body tissue, and finally, the equivalent dose was computed.

Results

In selecting the geometric shape of the shield, spherical geometry was used as the geometric shape of the protection shield due to the higher flux of thermal and nonthermal neutrons in spherical, cylindrical, and cubic geometries, respectively. According to the results of Figure 1 the amount of thermal neutron flux for the high-density polyethylene (HDPE) attenuator has the highest value compared to other attenuators, which is the result of further attenuation of nonthermal neutrons of the high energy sources and their conversion into thermal neutrons with lower-energy. The highest flux of thermal neutrons is observed at the surface of the sphere with a radius of 6 cm.

Paraffin with a radius of 6 cm and zirconium hydride with a radius of 7 cm is in the next ranks. For this reason, HDPE is

Table 1: Materials used in the protective shield

Type of protective shields					
Attenuators					
Polypropylene (C_3H_6)	Polystyrene (C_8H_8)	Polyethylene (C_2H_4)	Water (H_2O)	Melamine ($C_5H_7N_6$)	Polycarbonate ($C_{16}H_{14}O_3$)
Heavy water (D_2O)	Beryllium oxide (BeO)	Polyvinyl acetate ($C_4H_6O_2$)	Rubber, natural (C_5H_8)	Carbon, graphite (C)	Polyethylene terephthalate (PET)
Polymethyl methacrylate ($C_5O_2H_8$)	Polyethylene, borated (B_4C in C_2H_4)	High density polyethylene (HDPE) C_2H_4	Rubber, butyl	Zirconium hydride (Zr_5H_8)	Zirconium hydride (ZrH_2)
Bakelite	Paraffin $C_{25}H_{52}$				
Reflectors					
Water (H_2O)	Heavy water (D_2O)	Carbon, graphite (C)	Polystyrene (C_8H_8)	Zirconium hydride (Zr_5H_8)	Zirconium hydride (ZrH_2)
Paraffin ($C_{25}H_{52}$)	Polymethyl methacrylate ($C_5O_2H_8$)	Boron (B)	Boron carbide (B_4C)	Boric acid (H_3BO_3)	Beryllium (Be)
Beryllium carbide (Be_2C)	Beryllium oxide (BeO)	Lithium (Li)	Lithium hydride (LiH)		
Neutron and gamma absorbers					
Tungsten (w)	Lead (Pb)	Bismuth (Bi)	Gadolinium (Gd)	Stainless steel	Cadmium (Cd)
Vanadium (V)	Titanium (Ti)	Tin (Sn)	Nickel (Ni)	Concrete, portland	Kennertium
Cadmium tungstate ($CdWO_4$)					

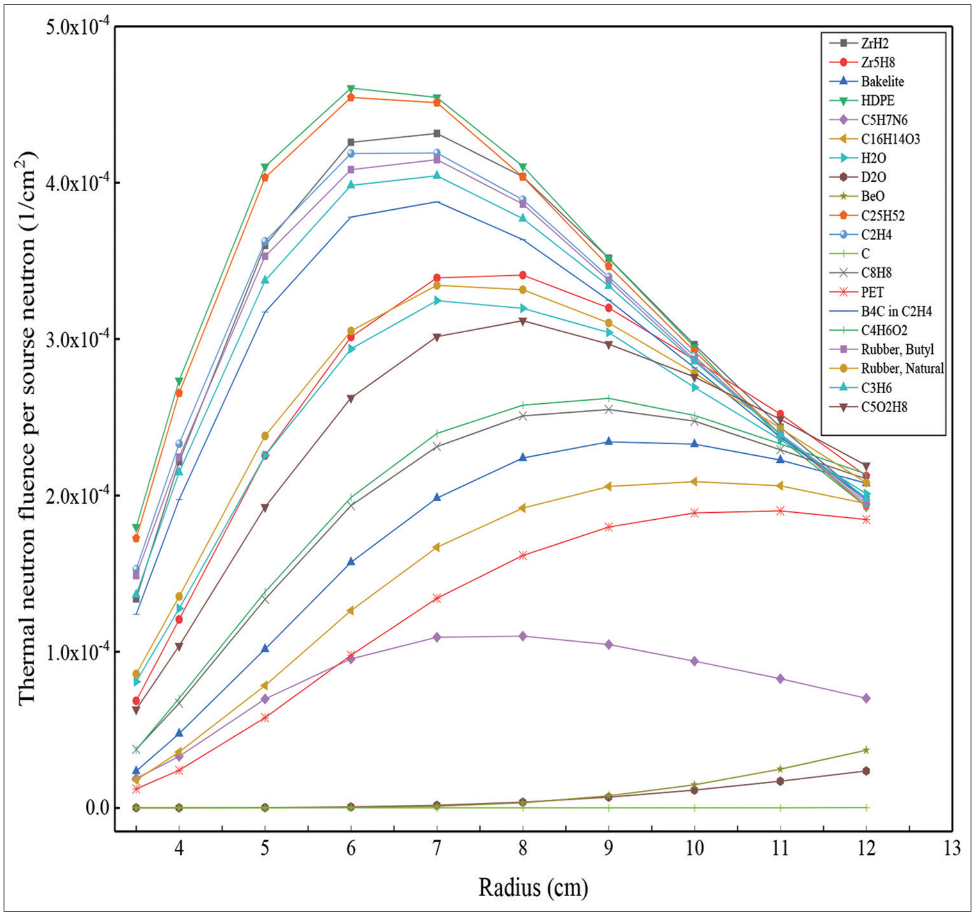

Figure 1: The flow of thermal neutrons per unit volume for different moderators

used as a moderator. The radius of 6 cm was also considered as the moderator radius as the place that showed the highest thermal neutron flux.

Also, the addition of each of the reflectors to the Am-Be source configuration increased the neutron flux, especially the thermal neutron flux. Paraffin has a higher efficiency in reflecting neutrons emitted from the source and increasing the flux of thermal neutrons at the deceleration surface, and as shown in Figure 2, it increases the flux of thermal neutrons up to a radius of 12 cm. The further increase of the reflecting radius will not have a significant effect on increasing the flux of thermal neutrons. For this purpose, a radius of 12 cm was chosen as the reflecting radius.

In the next step, tungsten and Kennertium alloy (76% tungsten, 15% nickel, and 9% copper and has a density of 16.8 g/cm^3) have the best performance in reducing neutron and gamma dose and increasing thickness of each of the protections further reduces the neutron and gamma dose. Based on the results, the gamma dose can be reduced to a minimum by adding higher thicknesses, while for neutrons, a thicker amount of Kennertium or tungsten should be used, which will be very heavy. Furthermore, by adding 1 cm of Kennertium or tungsten, the total dose rate can be reduced below the allowable limit.

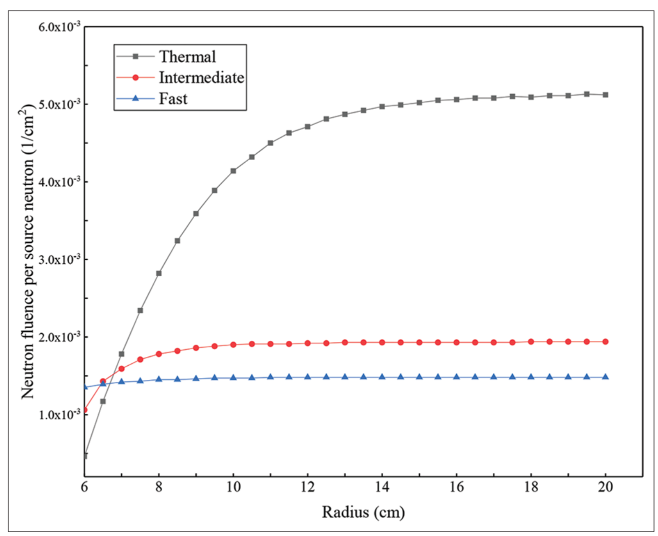

Figure 2: Neutron flux due to the addition of paraffin reflector thicknesses to the moderator

In the last step, while maintaining the total thickness of the shield, the layers were arranged and the neutron and gamma dose rates (instantaneous and delayed) were measured in the ICRU reference sphere.

Based on Table 2, the percentage of each of the Kennertium, tungsten, paraffin, and polyethylene materials in the types of composites is specified.

According to Table 3 and Figure 3, the Kennertium and tungsten alloys perform best in reducing neutron and gamma doses, and increasing the thickness of each shield reduces the neutron and gamma doses as much as possible. In the ultimate conclusion, it was found that the use of tungsten and Kennertium in the same volume has better efficiency compared to other materials, which, of course, are not recommended due to the high cost and weight of the protection. The next alternative suggestion is made of Paraffin composite sphere and high-density Polyethylene with a radius of 12 cm and Tungsten with a thickness of 1 cm are proposed as the final configuration for this study.

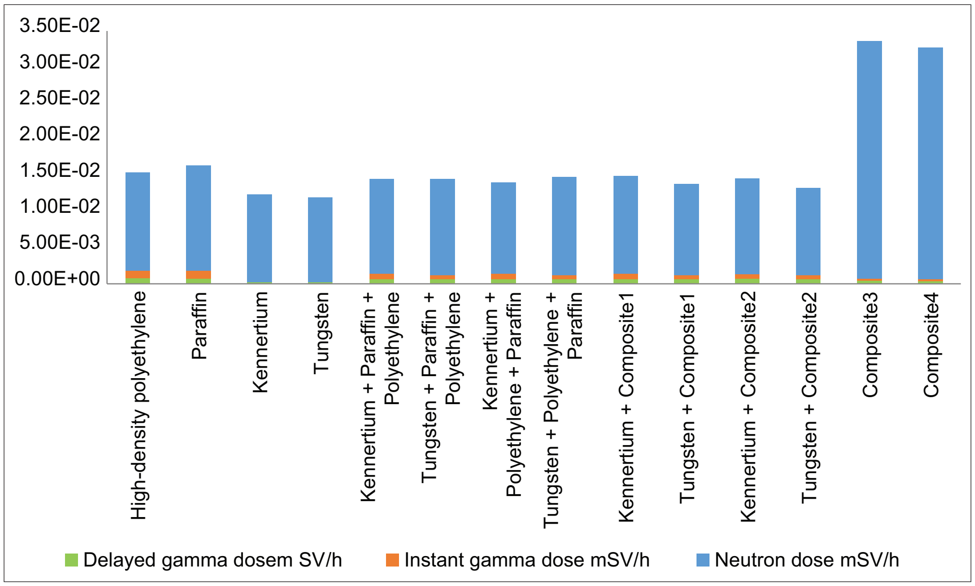

Figure 3: Total dose rate for different protections

Table 2: Percentage of materials in different types of composites

	Polyethylene	Paraffin	Tungsten	Kennertium	Density (g/cm³)
Composite 1	12.4	87.6	0	0	0.934194
Composite 2	88.4	11.6	0	0	0.960940
Composite 3	2.1	14.8	0	83.1	4.336777
Composite 4	1.9	13.2	84.9	0	4.872938

Table 3: Dosage rate in different arrangements

Materials	Neutron dose (mSV/h)	Delayed gamma dose (mSV/h)	Instant gamma dose (mSV/h)	Total dose (mSV/h)
High density polyethylene	1.3533×10^{-2}	7.8878×10^{-4}	1.0780×10^{-3}	1.5400×10^{-2}
Paraffin	1.4504×10^{-2}	7.4075×10^{-4}	1.0838×10^{-3}	1.6329×10^{-2}
Kennertium	1.2116×10^{-2}	2.3053×10^{-4}	3.0305×10^{-6}	1.2350×10^{-2}
Tungsten	1.1723×10^{-2}	2.1659×10^{-4}	9.4285×10^{-7}	1.1941×10^{-2}
Kennertium + paraffin + polyethylene	1.3102×10^{-2}	6.8157×10^{-4}	7.0192×10^{-4}	1.4485×10^{-2}
Tungsten + paraffin + polyethylene	1.3279×10^{-2}	6.2481×10^{-4}	6.1963×10^{-4}	1.4523×10^{-2}
Kennertium + polyethylene + paraffin	1.2630×10^{-2}	6.9217×10^{-4}	6.9601×10^{-4}	1.4018×10^{-2}
Tungsten + polyethylene + paraffin	1.3525×10^{-2}	6.3450×10^{-4}	6.1945×10^{-4}	1.4779×10^{-2}
Kennertium + composite 1	1.3512×10^{-2}	7.0670×10^{-4}	6.7765×10^{-4}	1.4896×10^{-2}
Tungsten + composite 1	1.2601×10^{-2}	6.1631×10^{-4}	6.1367×10^{-4}	1.3831×10^{-2}
Kennertium + composite 2	1.3230×10^{-2}	6.8674×10^{-4}	6.7032×10^{-4}	1.4587×10^{-2}
Tungsten + composite 2	1.2054×10^{-2}	6.0924×10^{-4}	6.0851×10^{-4}	1.3272×10^{-2}
Composite 3	3.2886×10^{-2}	4.0701×10^{-4}	3.2632×10^{-4}	3.3619×10^{-2}
Composite 4	3.2049×10^{-2}	3.7985×10^{-4}	2.5194×10^{-4}	3.2681×10^{-2}

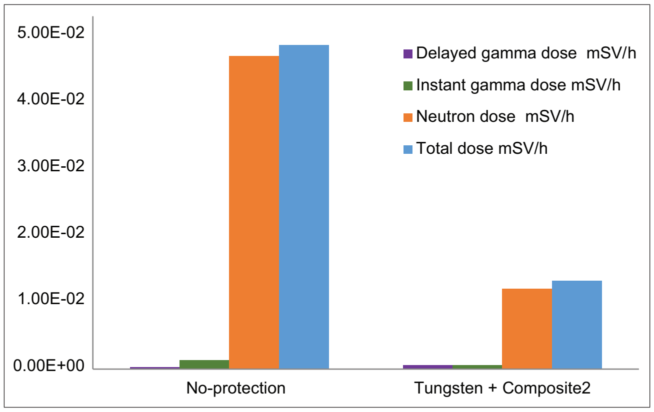

Figure 4: Comparison of the total dose rate in the absence and presence of protection

Figure 4 indicates that the protection is more effective than in the no-protection mode with the rate of 74% in reducing the neutron dose, 55% in the primary gamma radiation, and 72%–73% in the reduction of the general dose.

Discussion

In the topic of protection against nuclear radiation, the most efficient shields are composed of hydrogen-containing materials, heavy metals, and other neutron absorbers. However, carrying out such simulations allows us to provide the most appropriate protection in terms of thickness, weight, cost, resistance, maximum absorption, and production of minimum disturbing radiation, etc. with all types of radiation sources without performing any laboratory work. Also, if the laboratory work is done, the simulation results can be compared with the real results and their consistency can be ensured. In shielding against neutrons, the most effective method is to slow down the neutron and trap it in the shield. In this research, the effect of the shield geometry was investigated, which included the comparison of three geometries (spherical, cylindrical, and cubic).

The study conducted by Cai *et al*.[15] shows that the effect of spherical geometry is better than other types of geometry, which is also true in our study. In another study conducted by Didi *et al*. in 2016, Paraffin was a better moderator for neutron flux compared to Water.[16] Also, in 2018, Hila *et al*. pointed out in their research that Paraffin shows a better performance than Polyethylene in reducing the neutron dose.[17] In the current research, among the materials used, Paraffin and high-density Polyethylene have shown the best performance as neutron absorbers, and another reason for this choice is, in addition to the performance of these materials in reducing the neutron dose in the ICRU sphere, increasing the thermal neutron flux protection is in configuration. The cross-sectional area of thermal neutrons is higher than that of fast neutrons, which causes thermal neutrons to be absorbed around the source, and a smaller amount reaches the ICRU sphere compared to non-thermal neutrons. Due to the absorption of neutrons in their configuration, neutron-absorbing materials cause the production of delayed gamma, which increases the total gamma dose. One of the parts of the present study is the investigation of delayed gamma resulting from neutron absorption. The importance of this issue is the interaction between neutrons and the elements in the shielding layers, which leads to the production of this disturbing beam and choosing efficient shielding. A study conducted by Moadab *et al*. in 2019 showed that Kennertium had the best results in reducing gamma doses.[18] Due to the high gamma dose outside the shield, it is necessary to use a shield with heavy elements to reduce the gamma dose, then Tungsten and Kennertium were selected as gamma absorbers.

In 2018, Cai *et al*. proved in their research that the optimized composite had better performance than the multilayered state. In the present study, the use of high-density Polyethylene composite and Paraffin showed a higher performance than the multilayer mode to reduce the neutron dose.

According to the research background mentioned in the references,[19-20] it is possible to make multilayer and composite shields.

Considering that the main goal of this research is optimization in terms of safety and optimization in all dimensions (cost, weight, safety, volume, etc.) is not possible, it is suggested that in future studies, other protective features should be investigated.

Conclusion

According to the MCNPX simulation results, the properties of several materials in different layers of protection as mediators, reflectors, as well as neutron and gamma absorbers were investigated. By comparing the flux of thermal neutrons per unit volume of the moderator, it is concluded that HDPE has a higher potential for attenuation and moderation of neutrons than other moderators. In the next layer, the use of paraffin as a reflector had the greatest effect on reflecting the neutrons emitted from the source and increasing the flux of thermal neutrons at the mediator level compared to other reflectors. Tungsten also has the best performance as a neutron and gamma absorber.

The use of Tungsten and Kennertium in the same volumes compared to other materials has a better performance for protection, which is not recommended due to the high density of these materials and therefore the heavyweight of the protection. Finally, the composite sphere consisting of 88.4% HDPE and 11.6% paraffin with a radius of 12 cm showed the best results in slowing down the neutrons emitted from the ^{241}Am-Be source.

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Conflicts of interest

There are no conflicts of interest.

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