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Optimization of Body Composition Analyzer Facility, Considering Operator Dosimetry

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Abstract

Background: Changes in body composition may be used for monitoring progression/regression of a disease. Prompt γ-rays in vivo neutron activation analysis (IVNAA) has been widely used for the measurement of body composition in recent years.

Objective: In this paper, we tried to improve the safety of IVNAA operator.

Methods: The most important factor for reducing the operator receiving dose is the optimization of shields. An appropriate shield should not only reduce the operator receiving dose, but it also must have the least effect on the detected spectrum. Because all parts of setup, including the operator shield, can be activated, the emitted γ-rays may be counted in detectors and increase the background level. In this research, several shields have been considered for an IVNAA setup. 4 different shields—concrete, epoxy colemanite resin, paraffin borated with bismuth layer (PE-Bi layer), and paraffin borated uniformly mixed with bismuth (PE-Bi)—were simulated by MC-NPX code.

Results: We found that the PE-Bi shield decreases the absorbed dose to 77% compared with "no shield" and 74% compared to concrete. Also, the reduction rate of dose equivalent was 95% compared to "no shield" and 91% compared to colemanite resin. The neutron flux decreased almost 400 times in the presence of PE-Bi; it had less background in γ-spectrum compared to other suggested shields.

Conclusion: Among the tested shields, PE-Bi would be the best one.

Keywords

Operator shield; Absorbed dose; Dose equivalent

Introduction

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Neutron activation analysis (NAA) is a useful technique for analyzing elemental concentrations by measuring γ-rays emitted following neutron capture [1]. When NAA is applied to human body, it can provide quantitative information on trace as well as main constituent elements. Since sampling is usually painful and difficult in studies of human body composition and trace elements accumulated in specific organs, *in vivo* analysis is preferred. One of the most important aspects of wide application of *in vivo* NAA [2-4], is its safety for the operator. While there are concerns about the safety of patients, the dose that operators receive is usually neglected.

In recent years, with increased utilization of radioactive materials and development of radiation sources with exceedingly high intensities,

the problem of radiation protection has become extremely important. The conventional method of affording protection to personnel and equipment against radiation is to provide a shield or barrier substantially opaque to the radioactive emanations, between them and the source of the radiation.

Although there is no need for operator to be in the vicinity of the setup during the measurement, the operator should be near the setup during the preparation of patient and after that for monitoring the patient in the control room. So in the case of no shielding, the operator would receive a significant dose. The most common method used for shielding a neutron radiation system was concrete and paraffin [5]. However, paraffin is a significant moderator due to great amount of hydrogen which can decelerate neutron flux; it does not have the ability of neutron capture, so we use paraffin borated instead. The ordinary concrete is also replaced with new heavy concretes which have the more ability of shielding [6-9]. However, these kinds of concretes have lots of advantages in capturing penetrating particles. Unfortunately, it is not suitable to be used in PGNAA because they make a high background in the final detected spectrum.

Therefore, to protect the operator against exposure, various options for the shield is studied in this research and the optimized shield is discussed.

Materials and Methods

Facility Description

The setup used in this research was extracted from the facility which previously optimized in FUM Neutron Activation Research Centre (NARC) for prompt γ-rays uniformity (Fig. 1) [10]. The phantom was a cubic soft tissue element which was irradiated by two pairs of ^{241}Am-Be, each containing 4.2 Ci. The collimator was a $40 \times 40 \times 60$ cm^3 cubic shell of graphite which was surrounded by a 20-cm thick piece of concrete as neuron and γ shield.

As the purpose of this paper was to improve the safety of operator, the exterior shields could be changed. Two lead layers were used under 3"×3" NaI detectors for protecting them against background. The NaI (Tl) detector used to obtain all spectra in this work was contained in a thin-walled aluminum can (wall thickness: 0.005") to reduce absorption of low-energy photons and to prevent excessive Compton scattering from the packaging material. The *in vivo* NAA setup was modeled with great detail in Monte Carlo simulations. The detector structure was modeled as precisely as possible, except details of photomultipliers.

Monte Carlo Simulation

Simulation was done by MCNPX™ 2.4.0 code. While the neutron energy spectrum of ^{241}Am-Be source was extracted from the IAEA report 403 [11]; the γ-rays of ^{241}Am-Be was chosen from the previous study of Miri, *et al*. [12]. The neutron and γ-ray flux were assessed by the use of F4 tally. Also the F6 tally was used for dosimetry calculations, assuming Kerma approximation [13]. To simulate a full shield state, photon weight (PWT) card was employed with no shield around detector. The PWT card allowed turning off photon production in cells. To consider detector resolution, the initial responses of the MCNP calculation were broadened with FT4 card together with GEB (Gaussian Energy Broadening) option using appropriate indices of experimentally applied detector [14]. Molecular effects and scattering treatment, $S(\alpha,\beta)$, were used for neutrons below 4 keV (MTm card).

Results and Discussion

To protect the operator from exposure, the whole facility should be covered with a material which shields neutron and γ-rays properly. An appropriate composition to capture γ-rays must have high density and atomic number; because γ capture is occurred in photoelectric process and its probability is proportional to atomic number. Also high density will in-
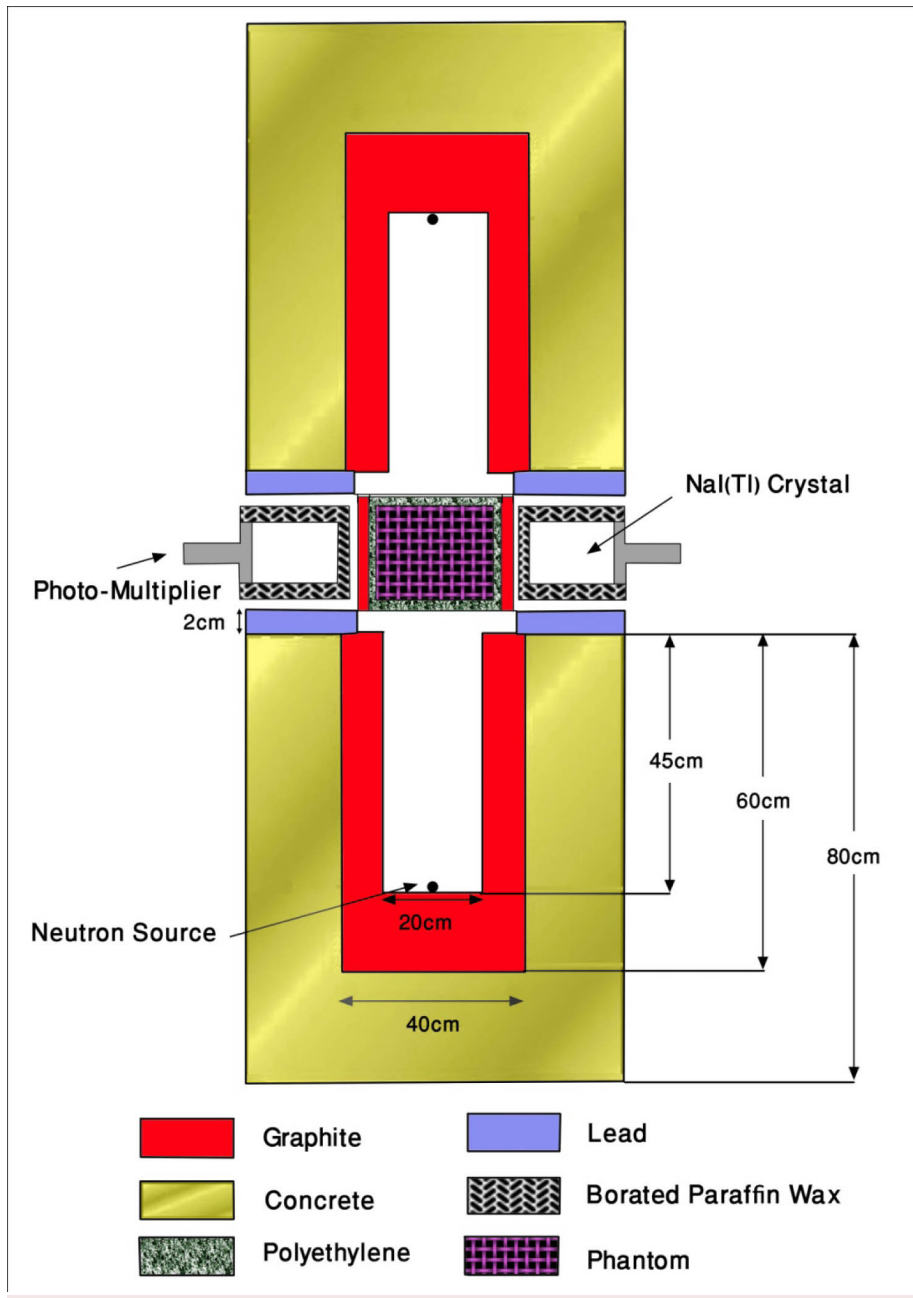
Figure 1: The layout of in vivo NAA setup

crease the number of interactions which result in capture. So the higher atomic number leads to greater probability of γ-ray capture. To stop neutron in the shield, the selected composition should have an acceptable ability to absorb neutron. With a high absorption cross section, neutrons will be stopped in the material. So the selected material for shield must contain heavy elements together with compositions with high neutron capture cross section.

The neutron absorption cross section in high energy region is very insignificant for even proper absorbers, so it is recommended to reduce neutron energy and then they can be captured in the shield materials. The best compositions for decelerating neutrons are hydric compositions in order to the greatest lethargy of hydrogen. Boron is of great interest in shielding technology because of its excellent shielding property. Boron compositions have a
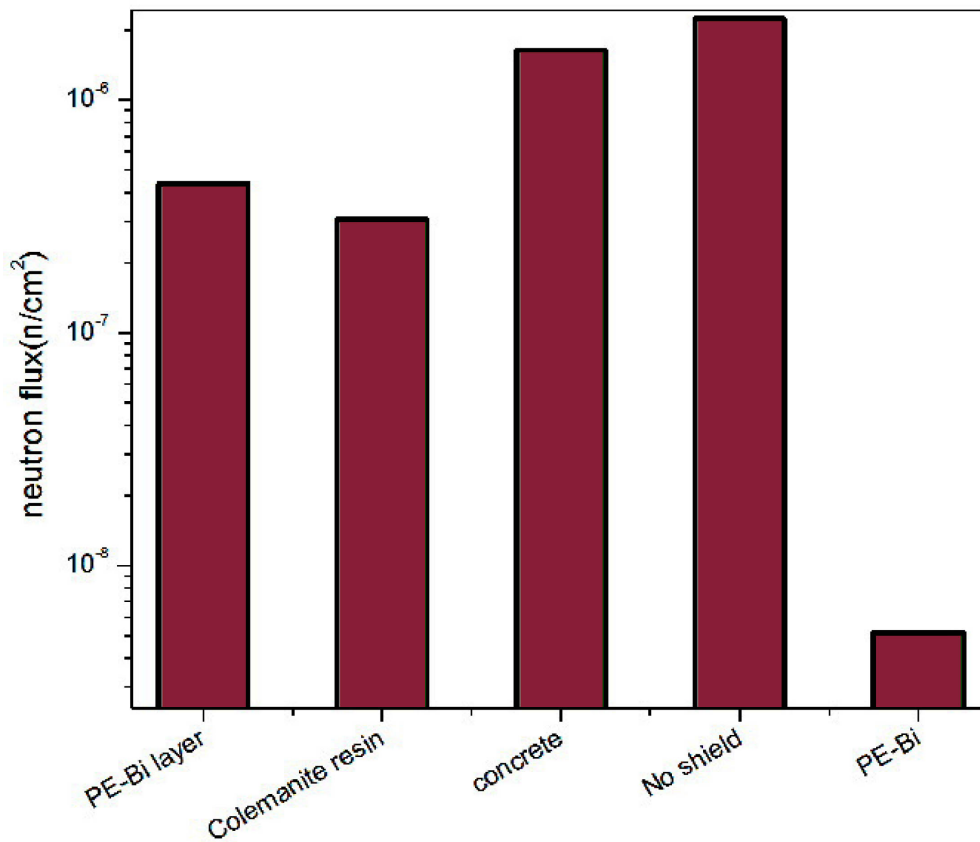
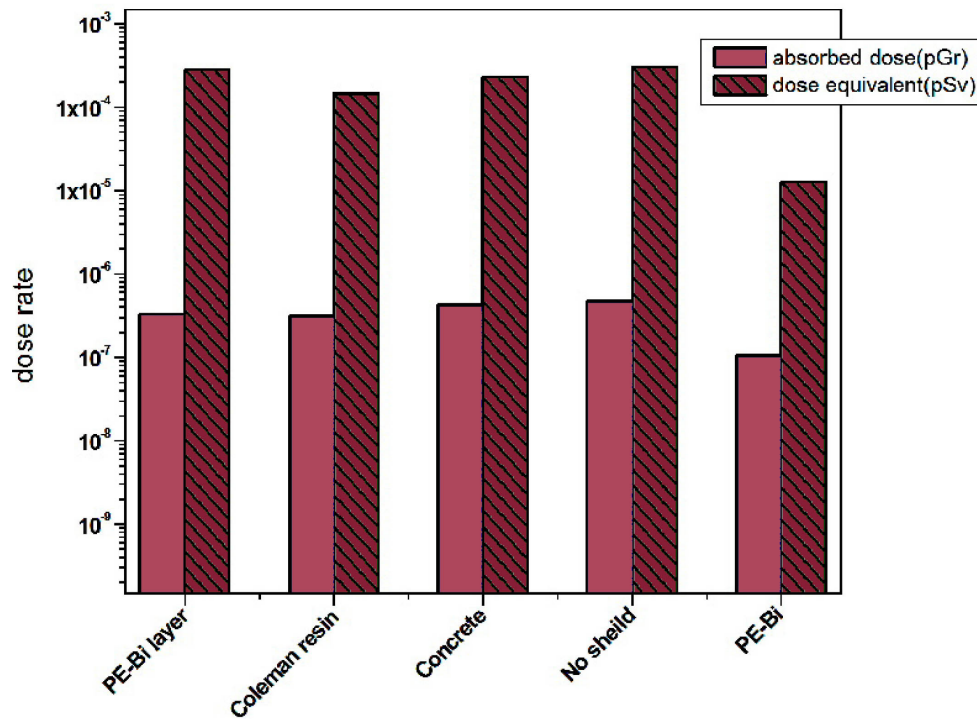**Figure2:** The surface neutron flux emitted to operator

Figure 3: The dose equivalent and absorbed dose of operator for different shields

Table 1: The composition and density of colemanite resin (%)

C	H	N	O	SiO_2	Al_2O_3	Fe_2O_3	CaO
25.5	3.8	1.2	15.8	2.23	0.6	0.3	16.6
MgO	Na_2O	K_2O	TiO_2	P_2O_5	Li	B_2O_3	Density
0.41	0.029	0.088	0.035	0.016	0.005	30.4	1.74 g/cm³

great cross section of thermal neutron capture [15]. The B(n,α)Li interaction leads to release α-particles which stopped in shield material. For shielding γ-rays the heavy elements like Pb, Bi, W, ... can be utilized. So due to the mentioned items, four different options are suggested for operator shield.

Different shield options
Colemanite resin

Among the minerals, colemanite (2CaO . 3B_2O_3, 5H_2O) and borax (Na$_2$O. 2B_2O_3. 10H_2O) have commercial importance. Because of the Na content, borax is not suitable for shielding purposes. Due to the high B_2O_3 content, up to 51%, and low price, colemanite is more attractive for neutron shielding [16]. In the past, colemanite was used, with a maximum of 7 wt.%, together with some heavy aggregates such as barite, steel punching or with some special cements such as lumnite, and magnesium oxychloride [17]. As operator shield, it must be set like bricks around the setup. Because of its low mechanical strength and solubility in water, it is difficult to make concrete from it. So one should make the bricks in a resin form. The composition and density of colemanite resin is indicated in Table 1.

Paraffin borated with Bismuth layer (PE-Bi layer)

Another option would be paraffin borated with bismuth layer (PE-Bi layer). Because of

high amount of hydrogen in paraffin, the neutrons will slow down and captured by boron; also the bismuth layer has a great density and atomic number to prevent photons.

Paraffin borated uniformly mixed with bismuth (PE-Bi)

It is possible to mix bismuth powder in paraffin uniformly. Mixing uniformly of paraffin and bismuth is not simple due to their different densities; we can pour the bismuth powder in the liquid paraffin and blending them until getting solid.

Concrete

The other option is concrete which is famous for shielding neutron and γ-rays; it was in the primary setup. Table 2 shows the density and elements of concrete.

Dosimetry Calculation

Using the shields for operator, it is expected that the flux of neutrons emitted to the operator will decrease comparing to having no shield. As shown in Figure 2, the neutron flux is reduced in the operator location and the highest reduction of neutron flux is in the case of using PE-Bi. The decrease of neutron encounter consequently causes less absorbed dose of the operator. This fact was obtained by simulating the setup in MCNP code. The calculated absorbed dose and dose equivalent is presented in Table 3 and Figure 3. The dose equivalent and absorbed dose per second were

Table 2: The density and elements of concrete

Element	H	C	O	Na	Mg	Al	Si	K	Ca	Fe	Density (g/cm³)
Quantity (%)	0.76	0.07	40	1.2	0.15	25.9	25.8	0.99	3.37	1.07	1.4

derived for one neutron when the operator was standing two meters away from the setup. As it is clear in Figure 3, by applying PE-Bi shield, the absorbed dose and dose equivalent had the least values of 1.06E7 PGr and 1.27E5 PSv, thus, the best shield would be PE-Bi. The annual dose equivalent for operator of this setup would be about 4.6 mSv which is less than the maximum permissible dose equivalent for occupational exposure of 50 mSv [18].

Background Checking

Since the aim of using this (or any other similar) *in vivo* NAA setup is analyzing the final spectrum of patient, it should be considered that the utilized operator shield does not change the final spectrum. So the γ-spectrum was calculated in the presence of different shield (Fig. 4). As shown in Figure 4, by applying the PE-Bi shield, the background is less than the time when using concrete. Although the PE-Bi attenuated the spectrum, it maintained the diagnosing peaks the estimation of body elements are based on.

In conclusion, the shield of paraffin borated uniformly mixed with bismuth would be the best choice for it provides the lowest neutron exposure to the operator.

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