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Reduction of the Gamma Dose Equivalent due to ^{252}Cf and ^{241}Am-Be Neutron Sources in the Patients Soft Tissues When Using Body Chemical Composition Analyzer Bed

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Abstract : The ^{252}Cf radioisotope and ^{241}Am-Be are routinely compact and portable encapsulated neutron sources that have wide range of applications in laboratories, industries and medicine because of their high flux and reliable neutron spectrum. They are cost-effective neutron sources for Prompt Gamma Neutron Activation Analysis (PGNAA). PGNAA method can be used in medicine for neutron radiography and body chemical composition analysis. Unfortunately ^{252}Cf and ^{241}Am-Be sources generate not only neutrons but also emit high-energy and undesirable gamma-rays that are useless when using PGNAA method. Also, since the sample in medical treatments is a human body, it will be under the harmful bombardments of undesirable gamma-rays. In addition, the existence of high-rate gamma rays eventuate simultaneous pulses in the detector that can be piled up and causes a significant background and distorts spectra in the region of interest (ROI). All of these restrictions forced us to attenuate these gamma-rays in a practical way without being concerned about losing the neutron flux or significant alteration in the neutron spectrum. In order to solve these problems, a relatively safe Body Chemical Composition Analyzer was designed that uses an optimal spherical gamma-ray shield, enclosing neutron source. Gamma-ray shielding effects and optimum radius of spherical Pb shield have been investigated and compared with the unfiltered case, bare source, using MCNP4C code. At the end the gamma ray dose equivalent per source neutron rate (user defined parameter) in the soft tissue for several radius of spherical Pb shield, for both neutron sources are calculated. Results show how using an appropriate gamma-ray filter can reduce the risk of exposure to the ^{252}Cf and ^{241}Am-Be neutron sources when using them in a Body Chemical Composition Analyzer.

Keywords: PGNAA. Chemical Composition, Filter material, ^{241}Am-Be, ^{252}Cf, MCNP, Absorbed dose, Dose equivalent.

Introduction :

Determination of elemental compositions for a human body is one of the useful tools for understanding general physiology relationships, diagnosing some diseases and cancers, radiotherapy treatments, nutritional disorders, planning treatments for those same diseases and disorders and in quantifying the efficacy of some medical therapies. Therefore, a lot of techniques have been applied in order to determine the precise amount of body chemical compositions. An ideal approach for conducting these measurements is to use neutrons; depending upon the approach, it can be categorized into three general areas, (i) Fast Neutron Analysis (FNA), (ii) Prompt

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Gamma Neutron Activation Analysis (PGNAA) and (iii) delayed gamma Neutron Activation Analysis (NAA).

Prompt Gamma Neutron Activation Analysis (PGNAA) technique is a powerful elemental analysis method (Chichester, 2004). PGNAA has a wide range of applications, such as bulk material analysis and oil well logging, hidden or buried explosive materials detection (Marshall, 1989; Ellis, 1987; Pickrell, 2001).

The PGNAA method is based on the following simple treatment. An atom's nucleus absorbs a neutron, achieving an excited state with almost instantaneous de-excitation by emitting a high-energy (multi-MeV) prompt gamma ray. There are several prompt gamma rays with various intensions for a special element which is active. Ordinarily the most intensive gamma ray in the spectrum is a characteristic sign for each element. So a gamma spectrum of an active sample can be assumed as a footprint of the existence of special elements in the sample. Since most of nuclei show an increasing in the neutron absorption cross section as the neutron energy decreases, therefore PGNAA is a useful technique when using neutrons with low or thermal energy.

Radioisotope ^{252}Cf and ^{241}Am-Be neutron sources are used for many applications in which compact, portable and reliable neutron sources are required. They are commonly used in the PGNAA method. They provide a high flux and reliable neutron spectrum from a very small assembly. Unfortunately these neutron sources generate not only neutrons but also high-intensive gamma-rays, 4.438 MeV for ^{241}Am-Be and a continuous spectrum for ^{252}Cf from 0 to 10 MeV.

When using PGNAA method for medical purposes, the sample is the soft tissues of a human body and so these gamma-rays can have destructive effects on it. Moreover, the accumulations of these high-rate gamma-rays in the detector volume eventuate simultaneous pulses that can be piled up and distort spectra in the region of interest (ROI).These limitations forced us to attenuate these gamma-rays in a reasonable way without any side effect such as losing neutron flux in the vicinity of the sample or production of high-energy and high-rate prompt gamma-ray via neutron interaction by the shield material.

According to the latest recommendations of international institutes of radioprotection, an increasing attention must be paid to the patient protection during cancer radiotherapy and exclusive applications of neutron sources such as ^{252}Cf and ^{241}Am-Be. Therefore, the primary purpose of this paper is to design a relatively safe Body Chemical Composition Analyzer to use in cancer therapy. In this way a new approach, that is, employing a proper shield for gamma-rays, poor one for neutrons, between the sample and neutron source is developed and evaluated its effects. Since, between gamma filter materials, Pb is taken into account as a safe and proper gamma shield, therefore a spherical Pb shield (enclosing neutron source) was chosen and modelled in the calculations.

Side effects of using this gamma-ray shield on the neutron flux and neutron energy spectrum are evaluated. At the end,

this shield in a usual Body Chemical Composition Analyzer was applied and its consequent results were calculated. The Absorbed Dose as a valid criterion to confirm using this gamma shield was measured in a phantom of soft tissue, in the presence and absence of the gamma-ray shield**.** The Monte Carlo N-particle general code (MCNP) (Briesmeister, 2000) that is usually employed for optimization studies in PGNAA was used comprehensive for simulation.

Configuration of the PGNAA Facility :

To protect personnel from biological effects of neutrons and to reduce background counts, neutron shielding must be considered, but high-speed neutrons are more difficult to shield, because absorption cross sections are much lower at higher energies. So at first, neutrons must be moderated in a hydrogenous material such as paraffin wax (14.86 % H, 85.14 % C). also because of Hydrogen has a great absorption Cross Section for thermal neutrons, the risk of neutrons for personnel vanishes. Although in the capture process (n, γ) by the Hydrogen target, potentially hazardous gamma-rays (2.224 MeV) will be produced, but totally it will be better, since it is possible to cover the moderator with a thin layer of Pb or tungsten to filter these secondary gamma-rays.

Figure 1 shows a cross-sectional view of the Body Chemical Composition Analyzer. Sheets of 2 cm thickness of Lead surround the paraffin wax to provide radiation shielding for personnel.

A sphere of Lead with optimum radius is centered at the source position to reduce gamma-ray component of the neutron source. Another part of this configuration is an invert, rectangular, cuneus void cast within a paraffin wax block ($40_{cm} \times 50_{cm} \times 60_{cm}$). To protect patient body from 2.224 MeV gamma-rays, produced by hydrogenous moderator, the inner wall of the valley-shaped (Figure 1) was lined by Pb sheet with 2cm thickness. By this way a rectangular neutron- beam aperture measuring 40 cm length (perpendicular to the paper sheet) and 20 cm (width) at the bed level is defined.

The patient is positioned supine, 40 cm above the neutron source, on an aluminum bed which moves directly over the neutron source and through the upwardly collimated neutron beam. Since the final results of the absorbed dose will be similar for most parts of human body and for more simplicity, just the absorbed dose in a soft tissue equivalent (TE) $10_{cm} \times 20_{cm} \times 20_{cm}$ as described in table 1 (URL) of density 1 gm/cm^3, was evaluated by MCNP code.

Table 1 : Chemical composition of a soft tissue

Element	Fraction by Weight
Hydrogen	0.10117
Oxygen	0.76183
Carbon	0.11100
Nitrogen	0.02600

Source Modelling

In the simulation of the optimum shielding investigation and Body Chemical Composition Analyzer, ^{241}Am-

Be and ^{252}Cf neutron sources were positioned one by one at the centre of a spherical Pb filter.

A ^{252}Cf neutron source with 1.61Ci of activity contained in standard X.35 capsules format (code CVN353) and an ^{241}Am-Be source with 5Ci of activity contained in standard Amershan X.14 capsules format (code AMN24), were used in the PGNAA configuration system. Even though, the final results are calculated in a way to be independent from the source activity.

The gamma energy spectrum of ^{252}Cf was simulated by using the following empirical energy distribution function :

$$SCf_{\gamma}(E_{\gamma}) = \begin{cases} 375E^{2}e^{\frac{-E}{0.109}} + 0.468e^{\frac{-E}{1.457}} \rightarrow E \leq 1.5MeV \\ e^{\frac{-E}{0.851}} \rightarrow E > 1.5MeV \end{cases}$$

This function is based on the experimental data reported by Glässel (Glässel, *et al.*, 1989).

The fission neutron spectrum $SCf_n(E_n)$ of the ^{252}Cf was simulated by using the watt fission spectrum using coefficients provided with the MCNP-4C code [6, Appendix H]. That is,

$$SCf_{n}(E_{n}) = 0.30033e^{\frac{-E_{n}}{1.025}} \sinh(2.926E_{n})^{\frac{1}{2}}$$

where E is the neutron energy in MeV.

This equation was used as the input neutron spectrum of ^{252}Cf source in the calculations.

Figures 2 and 3 show energy spectra of both gamma and neutron particles respectively for the ^{252}Cf neutron source as described above. The multiplicity ratio per fission event, $R = S_{\gamma}/S_n$ for the ^{252}Cf source is about 2.132 (Lanzanò, *et al.*, 1997). Also the ^{241}Am-Be neutron energy spectrum, Figure 4, extracted from experimental data reported by kluge and Weise (Kluge and Weise, 1982).

Flux and Absorbed Dose Calculations :

The dose equivalent rate in a soft tissue, $\dot{H}$ is written as $\dot{H} = Q.\dot{D}$ where Q = quality factor and $\dot{D} =$ absorbed dose rate. Absorbed dose rate in a tissue is the rate of energy absorbed per unit mass in the tissue, and its SI unit is the Gray (G*y*), defined as 1 G*y* = 1 *J*/Kg. Because of biological effects and absorbed dose don't always have one-to-one correspondence, so a factor called quality factor is introduced.

In this work the calculation of the total gamma dose equivalent rate is divided to three parts.

1. $\dot{H}_{S,\gamma}$ that is the gamma dose equivalent rate due to the gamma-ray component of neutron Source, 4.438 MeV for ^{241}Am-Be and a continuous spectrum from 0 to 10 MeV (Figure 2) for ^{252}Cf.
2. $\dot{H}_{S,\gamma}$ the gamma dose equivalent rate arising from prompt gamma-rays generated by the interaction between neutrons and Filter material, here Pb, as well as the source encapsulation and surrounded materials.
3. $\dot{H}_{S,\gamma}$ the gamma dose equivalent rate due to prompt gamma-rays generated by the interaction between neutrons and the soft Tissue.

The total gamma dose equivalent rate in sivert per second (Sv/s), is then

$$\dot{H}_{\gamma} = \dot{H}_{s,\gamma} + \dot{H}_{f,\gamma} + \dot{H}_{t,\gamma}.$$

In order to be independent from the source activity, a user defined parameter that is gamma dose equivalent per source neutron rate was calculated. Dose equivalent per source neutron rate is the dose equivalent rate in Sv/s divided by the value of emitted neutrons rate (neutron source activity).

When thermal neutrons are captured, in addition to the prompt gamma-ray, residual elements may be active with various Half-Lives and intensities. In this work the total dose equivalent rate due to delayed gamma-rays are not considered.

In the gamma-ray flux calculations, two separate input files were written. One is to track gamma-rays which originally originate from the source position and the other is to track those prompt gamma-rays which are produced via interaction of neutrons by the source capsule or filter material.

The intensity of 4.438 MeV gamma-rays to neutrons, $R = S_\gamma / S_n$, for the ^{241}Am-Be source is about 0.596 (Croft, 1989) and the multiplicity ratio per fission event, $R = S_\gamma / S_n$, for the ^{252}Cf source is about 2.132 (Glässel, *et al*., 1989).

Final results in the gamma-ray flux calculation are the summation of two parts.

1. Multiply the gamma-ray flux obtained from tracking the neutron source gamma-ray component by R.
2. Gamma-ray flux obtained from tracking prompt gamma-rays.

Results and discussion :

Results are classified into two sections; first section assesses spherical Pb shield effects and the second section evaluates the effects of employing this spherical shield in the Body Chemical Composition Analyzer.

1. The total flux of gamma-rays that originate from neutron sources and those are produced in the inelastic or capture process in the source capsule and filter material are measured for ^{241}Am-Be and ^{252}Cf sources separately. For more simplicity the relative flux to the unfiltered case, bare, are evaluated instead of absolute flux. Figure 5 shows a decreasing flux of gamma-ray when the radius of the spherical Pb shields increases. In according to the Figure 5, a Lead spherical shield of radius 4 cm reduces the gamma-ray flux of the ^{252}Cf source up to 94% relative to the unfiltered neutron source while this value for ^{241}Am-Be source varies up to 50% with a same radius for spherical shield.

The neutron flux passing through the spherical shield is shown in Figure 6 for both kinds of neutron sources. Under these conditions, the total neutron flux shows an increasing treatment relative to the “bare” neutron source while gamma-ray flux decreases as described above (Figure 5). This increasing arises from (n, Xn) interactions in the Lead filter. Where X can be 2 or 3 depending upon the neutron energy. Table 2 (JEF-PC Software, 1998) lists (n, Xn) interaction Cross Sections for the natural Pb. this fact can eliminate the concerning about losing the neutron flux due to the (n, γ) interactions in the filter material and strengthen the idea to use a gamma-ray shield for neutron sources.

Table 2 : (n, Xn) Cross Section for natural Pb

Neutron Energy (MeV)		Cross Section (Barns)
(n, 2n)	6.31	4.35231E-03
	7.08	5.85478E-02
	7.90	2.97119E-01
	8.90	7.45645E-01
	10	1.74400E+00
	20	0.00000E+00
(n, 3n)	10	2.60488E-01
	20	0.00000E+00

Comparisons in the neutron energy spectrum between filtered (4 cm radius) and unfiltered source were shown in Figures 3 and 4 for both neutron sources. These show that using a Lead gamma shield not only increase the neutron flux, but also drifts the whole neutron spectrum to the low or thermal energy region which in its turn plays a role in slowing down neutrons. Practicality a spherical Pb shield of radius 4 cm is the optimum size, even though spherical shields with greater size can be effective.

2. With increasing the radius of shield, we have a quite little decreasing in the total neutron flux colliding to the phantom of soft tissue, Figure 7. This fact shows that although by employing a proper gamma-shield the outgoing neutron flux from the shield increases (Figure 6), but in practical conditions when the spherical Pb shield is not the only material (Figure 1) the neutron flux in the vicinity under the soft tissue can diminish.

According to this fact that just part of source neutrons meets the sample, so results in Figure 7 are presented per source neutron rate.

Figures 8 and 9 show the energy spectrum of neutrons which collide with the phantom of soft tissue. These present that the neutron flux won't have any considerable changing if the spherical shield is used or not in the PGNAA setup. Note that 2 cm thickness of Pb shield surrounding the paraffin wax is fixed in all over evaluations.

Finally Figure 10 shows a decreasing in the gamma dose equivalent per source neutron rate in the soft tissue equivalent for several radius of the spherical Pb shield. As mentioned before, final results are presented independently from the source activity, hence Figure 10 is also included into this regular, normalized to the one neutron emitting from the source.

For a spherical Pb shield of radius 4 cm, reduction of the gamma dose equivalent per source neutron rate is about 8.44×10^{-17} (Sv per source neutron rate) when the neutron source is ^{252}Cf and is about 1.24×10^{-16} (Sv per source neutron rate) when the neutron source is ^{241}Am-Be.

Since this setup can reduce effectively the gamma dose equivalent in the soft tissue, the primary purpose of this paper is reached, although this is associated with very small alteration in the neutron flux colliding to the sample.

Summary :

Simulation results on the performance comparison obtained from the ^{252}Cf and ^{241}AmBe sources as well as overall assessments were presented.

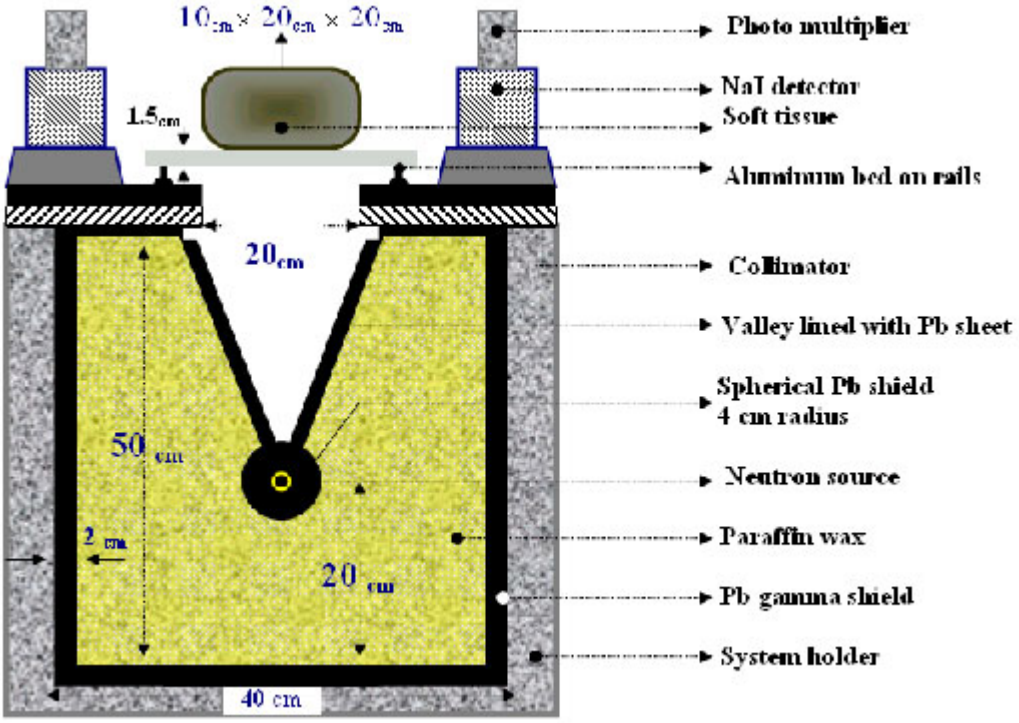

Fig. 1 : Schematic diagram of the Body Chemical Composition Analyzer.

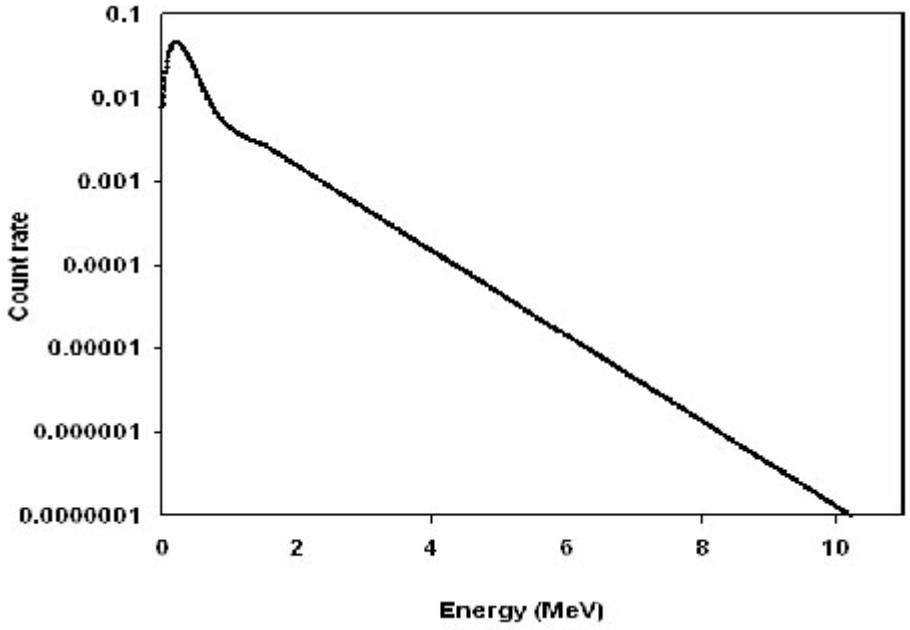

Fig. 2 : ^{252}Cf gamma-ray spectrum (per source neutron rate).

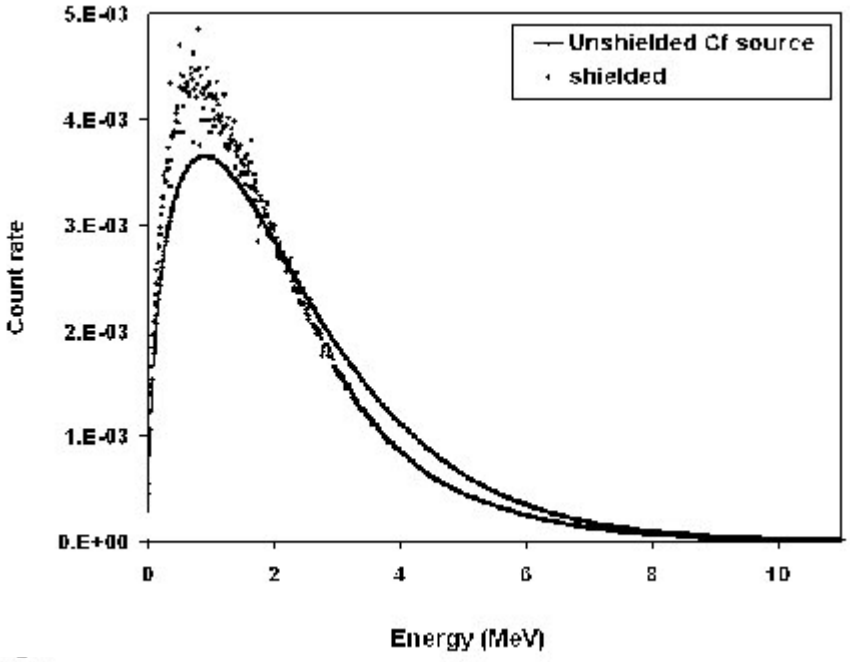

Fig. 3 : ^{252}Cf neutron spectrum (per source neutron rate) and the effect of a spherical Pb shield of radius 4 cm on this spectrum.

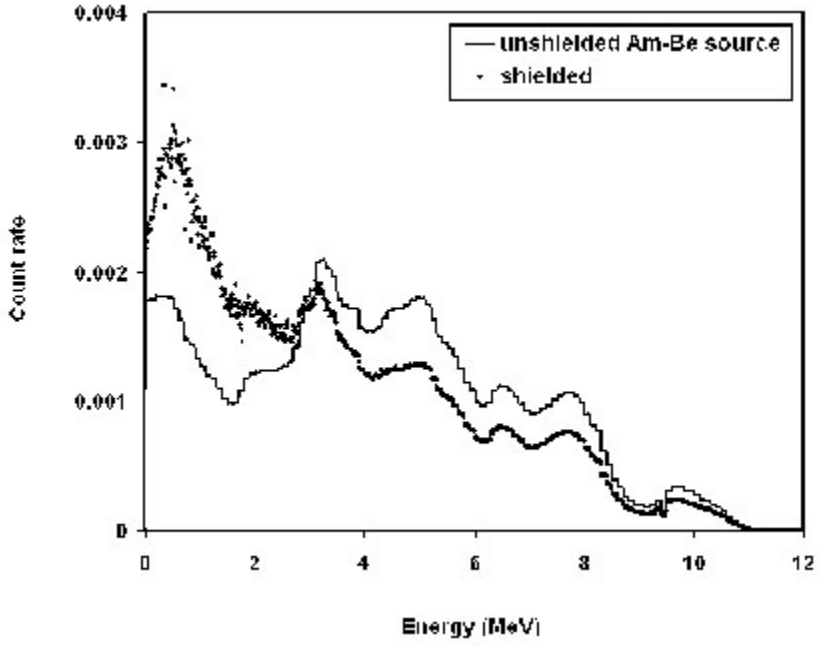

Fig. 4 : ^{241}Am-Be neutron spectrum (per source neutron rate) and the effect of a spherical Pb shield of radius 4 cm on this spectrum.

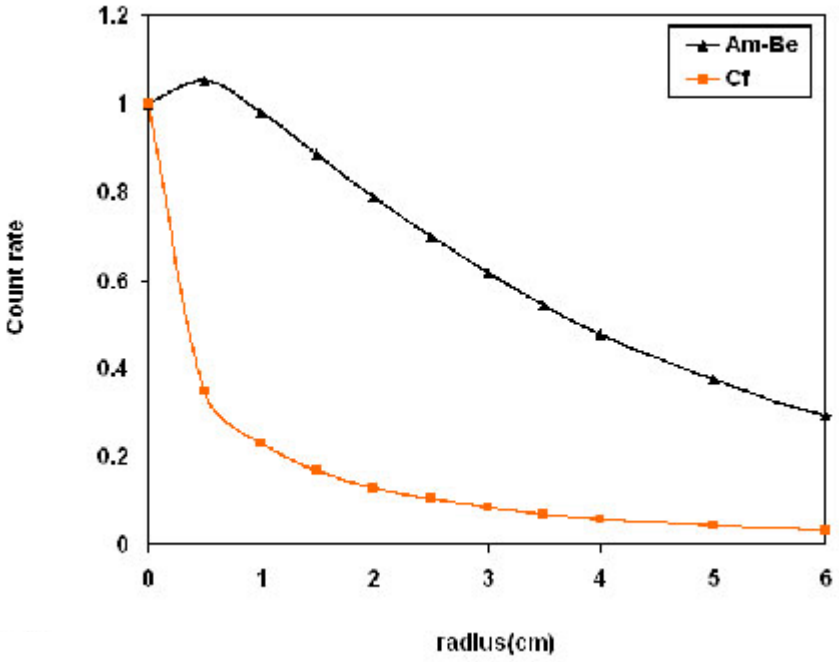

Fig. 5 : ^{241}Am-Be neutron spectrum (per source neutron rate) and the effect of a spherical Pb shield of radius 4 cm on this spectrum.

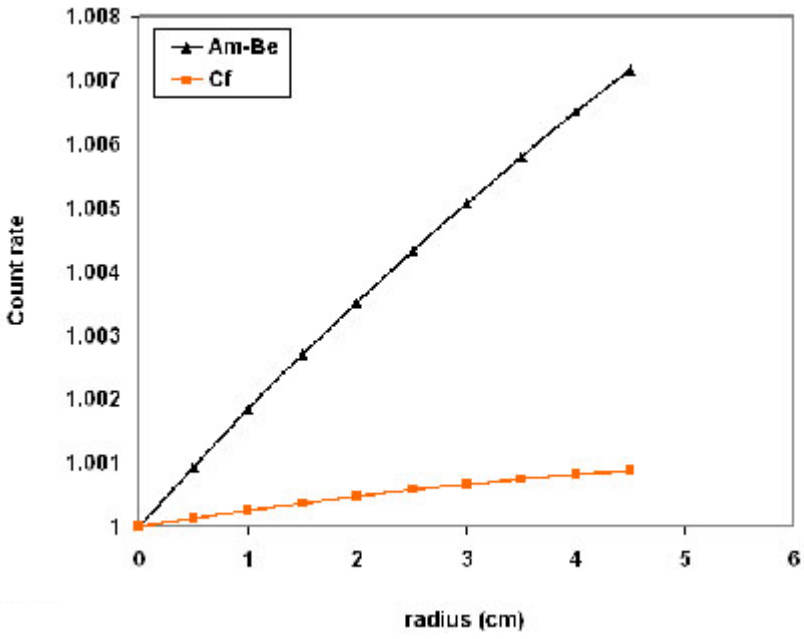

Fig. 6 : Neutrons count rate passing through the spherical shield with several radius (Normalized to an unfiltered neutron source) for both ^{252}Cf and ^{241}Am-Be sources.

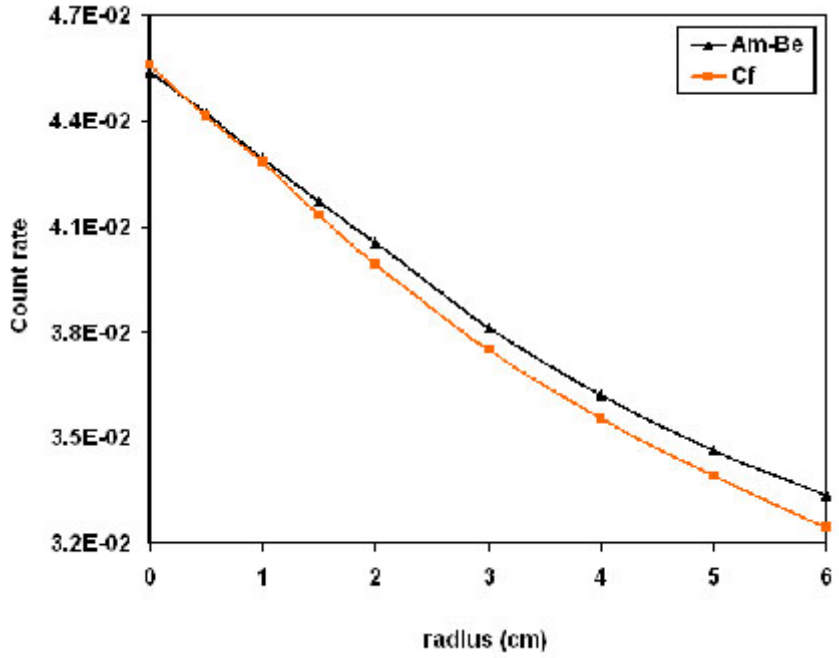

Fig. 7 : Neutron flux (per source neutron rate) in the vicinity under the sample for several radius of spherical Pb shield for both ^{252}Cf and ^{241}Am-Be.

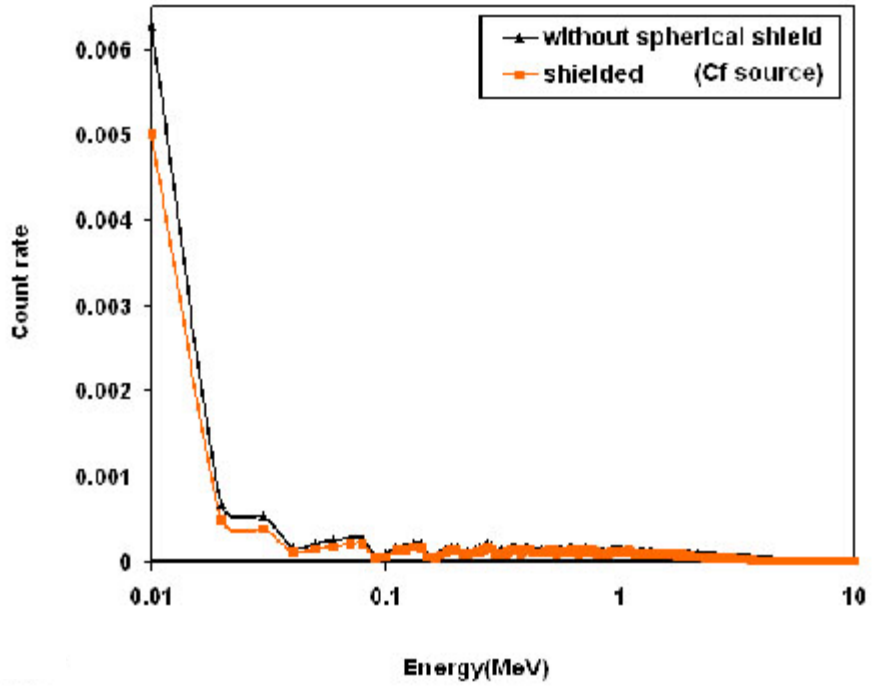

Fig. 8 : ^{252}Cf neutron spectrum (per source neutron rate) in the vicinity under the sample.

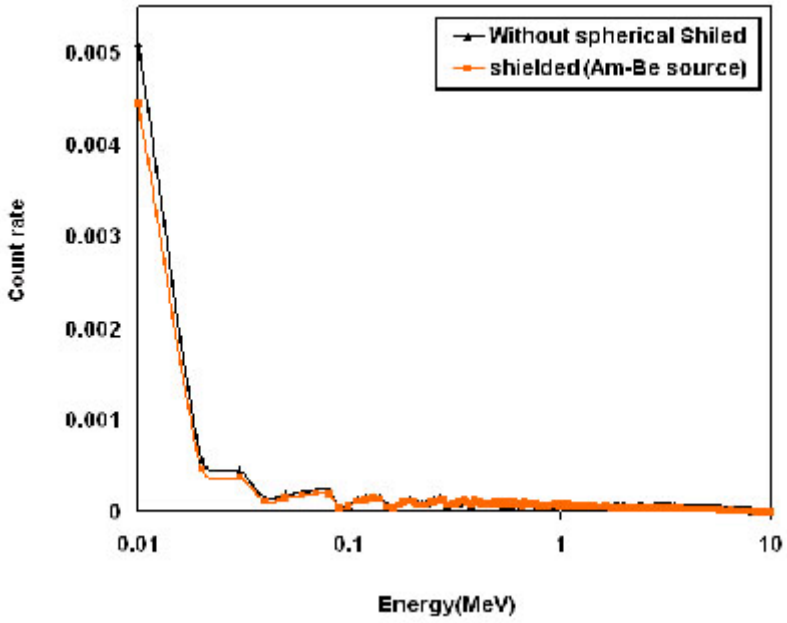

Fig. 9 : ^{241}Am-Be neutron spectrum (per source neutron rate) in the vicinity under the sample.

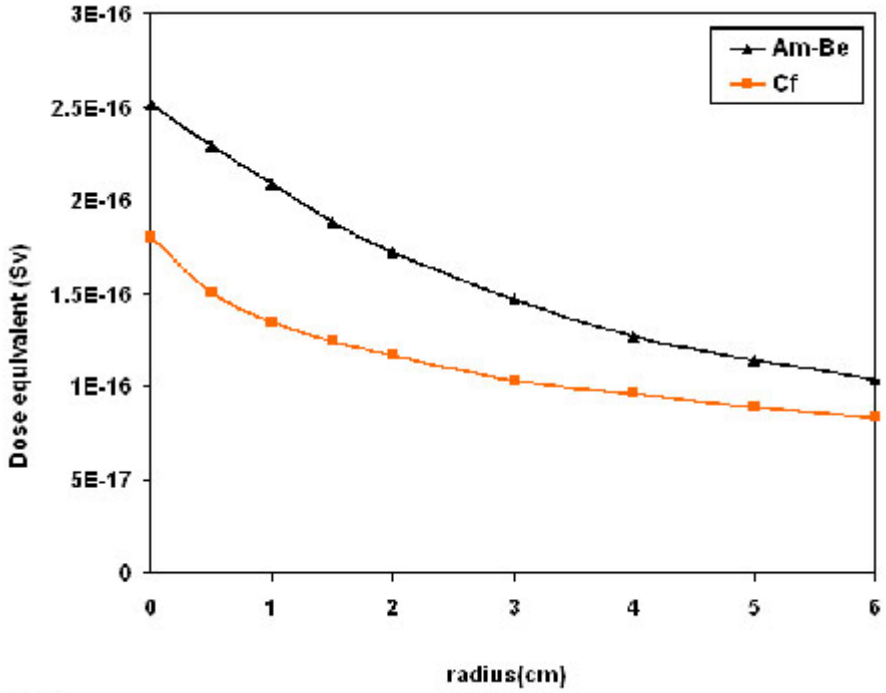

Fig. 10 : Gamma dose equivalent (in Sv per source neutron rate) measured in the soft tissue plotted against thickness of the shielding for the different sources.

According to obtained results inserting Pb shield as a barrier for gamma-rays has no distinguishable and considerable alteration in the neutron energy spectrum which is important for the PGNAA method.

Since two detectors are shielded from gamma-rays by Pb shield in the proposed setup, so this worry about accumulation of unnecessary and high-rate gamma-rays in the detector volume and consequent pulse pile up vanishes.

The proposed design of the Body Chemical Composition Analyzer, while is a safe one for personnel, is in agreement with the end goal that is reducing the gamma absorbed dose in the patient body.

References

Briesmeister J.F. (Ed.), MCNP (2000) : A General Monte Carlo N-particle Transport Code, Version 4C, Los Alamos National laboratory, LA–13709–M.

Chichester D.L. (2004) : E. Empey, Appl. Radiat. Isot. **60,** 55.

Croft S. (1989) : Nucl. Instr. and Meth. A 281 103-116.

Ellis D.V. (1987) : Well Logging for Earth Scientific, Elsevier, New York, 1987.

Glässel *et al*., P. (1989) : Nucl. Phys. A 502 315c.

JEF-PC Software (1998) : Version 2.0, OECD Nuclear Energy Agency.

Kluge H. and Weise K. (1982) : Radiation Protection Dosimetry, **2,** 85-93.

Lanzanò G. *et al*., (1997) : II Nuovo Cimento A **110(5),** 505.

Marshall III J.H. (1989) : J.F. Zumberge, Nucl. Geophys. **3,** 445.

NNDC, National Nuclear Data Center, http://www.nndc.bnl.gov/capgam

Pickrell M.M. : Novel Methods for Neutron-Induced-Gamma Detection of Land Mines and Buried Explosives, Los Alamos National Lab Report. http://www.uxocoe.brtrc.com/TechnicalReps/DoeReport/LosAlamosNatLab.pdf

URL address: http://atom.kaeri.re.kr/cgi-bin/w3xcom?m=262