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A Study of Shielding Design for the Radioisotope Store and Dispensing Room of Nuclear Medicine Centers in Bangladesh

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ABSTRACT

The gamma ray attenuation of different locally available shielding materials was measured for ^{99m}Tc and ^{131}I radioisotopes which are frequently used radioactive sources in nuclear medicine center. Existing protective arrangements were also studied and required to be partly changed for further improved protection in a cost effective way. On the basic of the measurements on the locally available shielding materials a modified plan of shielding design has been suggested for the radio-isotope store and dispensing room of future possible nuclear medicine center in Bangladesh.

INTRODUCTION

Gamma ray interaction with matter yield useful information for designing experimental arrangements for handling the radioactive sources decaying by gamma emission. The most important parameter characterizing the penetration and diffusion of gamma radiation in extended media is the attenuation co-efficient. This quantity depends on the photon energy and the nature of the absorbing medium [1]. The co-efficient of attenuation is also very useful at the time of building and shielding a nuclear reactor, accelerator and any other nuclear radiation establishment. For this purposes, the attenuation co-efficient for constructional materials are most essential. The construction of a nuclear reactor and some other radiation laboratories or areas demands the study of gamma ray absorption of the required materials [2].

In nuclear medicine center, gamma emitting as well as beta emitting radionuclides are being widely applied for both diagnostic and therapeutic purposes. Gamma radiation is highly penetrating than beta radiation and even causes serious biological effects as well [3]. Because of its harmful consequences, this gamma sources when not being used, should be properly stored in a well shielded area so as to make the surround zone safe from the radioactive contamination.

The existing nuclear medicine centers of Bangladesh are observed to have shielding arrangements from the radiation safety consideration. But, some other new centers are also expected to be set up very soon in different place of the country because of its growing demands day bye day and there must have also proper protective arrangements no doubt.

Although the existing centers have adequate protective devices but slight change in the geometrical arrangement or shielding materials, better working arrangement can be set up in the future station.

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However, in recent years, many research works have already been performed in different field of nuclear medicine center but no significant work on radiation protection like shielding have yet to be done [5,6]. Therefore this present plan was assigned to measure the gamma ray attenuation of different locally available materials in order to design and suggest a shielding arrangement for the running and newly approaching centers so as to ensure more protection in a cost-effective way.

Methodology

In this study, eight types of locally available materials were used to access its radiation protecting capabilities effectively. The materials used for this present purpose are Concrete, Normal Brick, Wood, Lead and Glass. The apparatus was arranged as shown in fig.1. The instrument was turned on and allowed it to warm up for about 10 minutes. The voltage of the detector was set at its operation value (1000 V) and the average back ground count rate from a series of five-minute counting was recorded. The radiation source was placed before the window of the detector at a distance about 40 cm and three five-minute activity readings were recorded. The average value I_o of these readings was taken. A slab of absorber was placed between the source and the detector window and the three-five minutes activity readings were recorded. The average value of I of these readings was also taken. In this way further absorbers were placed and its sequent readings were recorded for I-131 and Tc-99m [7].

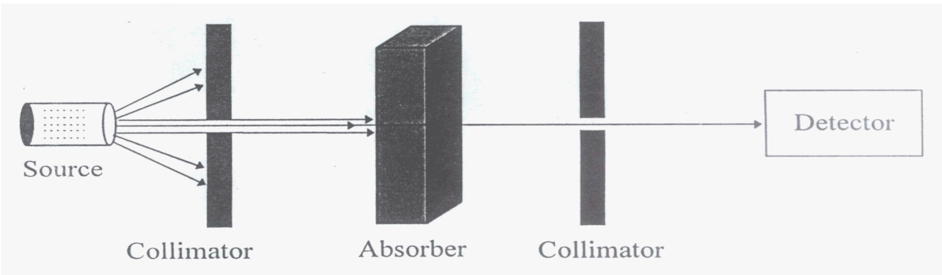

Fig. 1 Experimental Arrangement

The attenuation co-efficient was calculated by the following formula, $I=I_0e^{-\mu X}$. Where, I and I_0 are the number of counts measured without and with absorber respectively, μ and X are the linear attenuation co-efficient and the thickness of the absorber material respectively [8]. The linear attenuation co-efficient (μ) also converted to mass attenuation co-efficient (μ_m). Furthermore, half-value thickness (HVT) and even tenth-value thickness (TVT) [6, 7] were also determined and these are presented in table 1 and table 2.

RESULTS AND DISCUSSION

Measured values of the linear attenuation co-efficient (μ) and the mass attenuation co-efficient (μ_m) are presented in table-I and table-II. Half-value thickness (HVT) and tenth value thickness (TVT) are presented in the same tables. Comparison of results of present study with data received by Hungerford from Hogerton Grass Reactor Handbook [4] is presented in table-3.

Table 1 Gamma ray attenuation co-efficient, half value thickness (HVT) and tenth-value thickness (TVT) for source I-131

Materials	Linear attenuation co-efficient, μ (cm^{-1})	Mass attenuation co-efficient, μ_m (cm^{-2}/gm)	Half-value thickness (HVT) $X_{1/2}$ (cm)	Tenth-value thickness (TVT) $X_{1/10}$ (cm)
Concrete-1	0.1353	0.07325	5.122	17.018
Concrete-2	0.1469	0.06813	4.717	15.675
Concrete-3	0.1323	0.06774	5.238	17.404
Normal Brick	0.1324	0.07274	5.234	17.391
Wood (Mango)	0.04538	0.06525	15.237	50.629
Wood (Jack-fruit)	0.03583	0.05723	19.341	64.264
Lead	0.81972	0.08477	0.845	2.809
Glass	0.1460	0.05877	4.746	15.771

Table 2 Gamma ray attenuation co-efficient, half value thickness (HVT) and tenth-value thickness (TVT) for source Tc-99m

Materials	Linear attenuation co-efficient, μ (cm^{-1})	Mass attenuation co-efficient, μ_m (cm^{-2}/gm)	Half-value thickness (HVT) $X_{1/2}$ (cm)	Tenth-value thickness (TVT) $X_{1/10}$ (cm)
Concrete-1	0.5261	0.2848	1.317	4.377
Concrete-2	0.5369	0.2490	1.291	4.289
Concrete-3	0.5168	0.2646	1.341	4.455
Normal Brick	0.5175	0.2843	1.339	4.449
Wood (Mango)	0.1811	0.2598	3.827	12.714
Wood (Jack-fruit)	0.1437	0.2295	4.822	16.024
Lead	20.3	2.0992	0.034	0.113
Glass	0.7136	0.2872	0.971	3.227

Table 3 Comparison of our experimental results with other references [4]

Materials	Density (gm/cm^3)	Energy (MeV)	Linear attenuation Co-efficient, μ (cm^{-1})	Reference
Concretes	2.2	1	0.141	Hungford [4]
	2.4	1	0.154	
Bricks	1.78	1	0.113	
	2.20	1	0.130	
Wood	0.51	1	0.0345	
	0.67	1	0.0452	
Lead	11.34	1	0.797	
Glass	2.23	1	0.141	
	2.40	1	0.152	
Concretes	1.84	0.36	0.135	Present work
	2.15	0.36	0.146	
	1.95	0.36	0.132	
Bricks	1.82	0.36	0.132	
Wood	0.697	0.36	0.045	
	0.626	0.36	0.035	
Lead	9.67	0.36	0.819	
Glass	2.484	0.36	0.014	

Attenuation through lead

The linear absorption co-efficient of lead for I-131 source was found to be 0.8191 cm^{-1} indicates that 81.97 % of photons of the incident beam would be attenuated in each 1 cm layer of this material. The measured values of half-value thickness and tenth-value thickness are 0.84 cm & 2.809 cm for this material for I-131 source. Lead was observed to have significant radiation absorption capability in case of Tc-99m source. In this case 20.3 % of intensity of radiation would be disappeared due to the interaction occurred in each 0.01 cm of lead and thus 0.034 cm (HVT) of lead is enough to decrease the photon beam to 50 %. Again for 90 % reduction in the intensity of radiation 0.113 cm (TVT) of this material is sufficient.

Attenuation through concrete

The linear absorption co-efficient of concrete-1 (cement, sand & brick pieces) for I-131 source was found to be 0.1353 cm^{-1} i.e., the intensity of radiation will be reduced by 13.53 % in traversing a distance 1 cm in the absorber. Similarly, The linear absorption co-efficient of concrete-2 (cement, sand & stone pieces) and concrete-3 (cement, sand, stone & brick pieces) was found to be 0.1469 cm^{-1} and 0.1323 cm^{-1} i.e., the intensity of radiation will be reduced by 14.69 % and 13.23 % in traversing a distance 1 cm in concrete 2 and concrete 3 respectively. So far, all the three types of concretes have nearly the same attenuating capability and so can be frequently replaced by one-another. The measured values of half-value thickness and tenth-value thickness are 5.12 cm & 17.01 cm for concrete-1, 4.7 cm & 15.67 cm for concrete-2 and 5.23 cm and 17.40 cm for concrete-3 respectively. The linear absorption co-efficient of concrete-1 for Tc-99m source was found to be 0.5261 cm^{-1} i.e., the intensity of radiation will be reduced by 52.61 % in traversing a distance 1 cm in the absorber. Similarly, The linear absorption co-efficient of concrete-2 and concrete-3 was found to be 0.5369 cm^{-1} and 0.5168 cm^{-1} i.e., the intensity of radiation will be reduced by 53.69 % and 51.68 % in traversing a distance 1 cm in concrete 2 and concrete 3 respectively. All the three types of concretes have nearly the same attenuating capability for Tc-99m source. The measured values of half-value thickness and tenth-value thickness for Tc-99m source are 1.317 cm & 4.37 cm for concrete-1, 1.29 cm & 4.289 cm for concrete-2 and 1.34 cm and 4.45 cm for concrete-3 respectively.

Attenuation through brick

Bricks were seen almost the same interacting probability to gamma radiation as concrete used for I-131 source. 13.24% of gamma photons were observed to be attenuated in each 1 cm. The corresponding half-value thickness and tenth value thickness are 5.23 cm and 17.39 cm respectively. For Tc-99m source, 51.75% of gamma photons were observed to be attenuated in each 1 cm. The corresponding half-value thickness and tenth value thickness are 1.339 cm and 4.45 cm respectively.

Attenuation through wood

Wood of jack-fruit variety is mostly used for doors and windows in nuclear radiation laboratory. It has 3.58 % of gamma absorption capability in 1 cm of this material of wood for I-131 source because of the co-efficient of linear absorption measured in this case was 0.03583 cm^{-1}. Mango variety of wood was found to absorb 4.54 % of radiation intensity in 1 cm of its thickness. The attenuation co-efficient of this material was observed to have the value 0.04548 cm^{-1} for I-131 source. But this variety of wood is not widely used as like as the wood of jack-fruit. Similarly, the measured value of linear absorption co-efficient is 0.1437 cm^{-1} for wood of jack-fruit and for wood of mango is 0.1811 cm^{-1} for Tc-99m source.

Attenuation through glass

Attenuation co-efficient of glass material for I-131 source was measured and has the value 0.146 cm^{-1}. Distinctly, 14.60 % of the intensity of the incident radiation would be reduced by each 1 cm glass. In order to reduce the intensity of radiation to half of its original value, 4.74 cm (HVT) of glass material can be used to achieve the desired result. Its tenth-value thickness was also calculated to be 15.77 cm. Attenuation co-efficient for Tc-99m source was 0.7136

cm^{-1} i.e., 71.36 % of the intensity of the incident radiation would be reduced by each 1 cm glass. The corresponding half-value thickness and tenth value thickness are 0.97 cm and 3.22 cm respectively.

CONCLUSION

In the dispensing room, the solution of iodine (I-131) kept in glass containers should be placed with in a special arrangement. To keep the radiation worker safe from the radiation risk, there should be a well protective barrier in between the source and the worker concern. For this purpose a 5" thick wall of brick or concrete of dimension 1.5´×1.5´ along with inner lead layer of thickness about 0.6 mm can be set up on the laboratory bench. Further, a lead glass may be placed on the top of this wall for mirror viewing and to safe the eyes of the workers. If the desired sourced be placed under the arrangement along with lead bricks as used before, the arrangement so prepared will offer improved protection. During the milking of Tc-99m from its generating system, there is a risk of high exposure and so if the same type of barrier can be set up in between the generating system and the person involved, more protection can surely be established. However, instead of making a separate room for Tc-99m generator, Tc-99m generating system along with this protective device can also be placed in the dispensing room. This type of arrangement will ensure also improved protection with low cost.

In the isotope dispensing room, instead of the holes or cavities used as storage space for radioisotopes, a space should be made by building a 5" wall of bricks or concretes of height about 2.5 feet along with inner lead layer of 0.6 mm thick at a distance of about 2.5 feet from and parallel to the building wall. This space can also be used for the storage of isotopes. In fact, this type of design will offer higher safety with low cost.

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