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A prototype neutron dose measuring instrument based on prompt gamma detection

Ashwini Udupi, Priyada Panikkath, and P.K. Sarkar[a]

Manipal Centre for Natural Sciences, Manipal Academy of Higher Education, Manipal - 576104, Karnataka, India

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Abstract. Neutron detection and the measurement of ambient dose equivalent are essential to ensure radiation safety in the workplace neutron environment existing in neutron-generating facilities, like particle accelerators, nuclear reactors and other neutron-generating facilities. Based on the measurement of the prompt gammas emitted from elements like hydrogen, boron and carbon, consequent to neutron interactions, a prototype neutron dose measuring instrument has been fabricated using high-density polyethylene and borated high-density polyethylene. This instrument provides an optimized response that closely approximates the fluence to dose conversion coefficients recommended by the International Commission on Radiological Protection (ICRP) to estimate the ambient neutron dose equivalent. Experimental evaluation of the performance of the instrument has been done using neutrons emitted from a ^{241}Am-Be source, kept inside a concrete enclosure with a square opening on one side. The present system can measure neutron dose with a match of about 5% with theoretical estimates and an existing rem-meter. The measured minimum detection level (MDL) of the instrument is $8.35\,\mu\mathrm{Svh}^{-1}$ in the presence of significant gamma background and can be improved using a larger NaI(Tl) detector.

1 Introduction

Generally, for radiological surveillance in and around any neutron-generating facility, like accelerators, nuclear reactors, etc., the neutron ambient dose equivalent $H^*(10)$ [1–3] is measured using rem-meters, usually consisting of polyethylene cylinders or spheres to moderate neutrons to thermal energies (0.025 eV) to be detected subsequently by a thermal neutron detector. The neutron ambient dose equivalent is estimated using the simulated response matrix of the detection system for different mono-energetic neutrons approximating the energy-dependent neutron fluence to dose conversion co-efficients [4] (DCC) as provided by the International Commission on Radiological Protection (ICRP). The neutron energy range for acceptable responses for these rem-meters is from thermal (0.025 eV) to about 10 MeV. However, these monitors underestimate $H^*(10)$ in the energy range from thermal to 1 eV and overestimate in the energy range 1 eV to 100 keV. Above 10 MeV the response reduces significantly which leads to the underestimation of $H^*(10)$ [1–3].

It may be noted that estimation of $H^*(10)$ depends on the DCC provided by the ICRP, which has been changed in the past and might change in future as well. Any such change will render the measurements done now useless in future if the response of the instrument is explicitly dependent on the DCC as is true for most of the commercially available rem-counters presently. It is therefore profitable to make the measured data independent of the ICRP provided DCC.

To resolve these difficulties, various methods have been introduced for neutron dose measurement. The aim of the present work is to develop a field-deployable instrument, which depends on the measurement of prompt gamma intensities emitted following neutron interaction with a cylindrical system consisting of high-density polyethylene (HDPE) and borated high-density polyethylene (BHDPE). Peak intensities corresponding to each prompt gamma emission from the BHDPE+HDPE cylinder for mono-energetic incident neutrons are considered as the response of that particular gamma emissions. The responses of all such emitted gammas are combined linearly to fit the DCC provided by the ICRP (DCC-ICRP) in a least square sense to obtain the coefficients of optimal fitting [4]. The estimated dose conversion coefficients thus obtained are termed as DCC-estimated. Prompt gamma intensities can then be related to neutron ambient dose equivalent $H^*(10)$, by summing the peak areas after multiplying with the

[a] e-mail: pksarkar02@gmail.com (corresponding author)

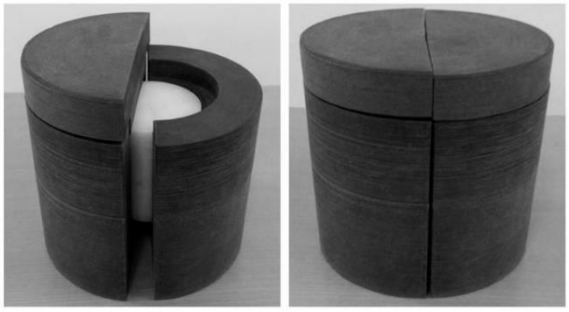

Fig. 1. Fabricated HDPE and BHDPE optimized system. The white inner cylinder is HDPE and the grey outer cylinder is BHDPE.

coefficients C_i obtained [5,6]. The uncertainties associated with the emitted prompt gamma intensities depend on the emission cross-sections of elements present in the BHDPE+HDPE cylinder, incident neutron energy and neutron fluence. It may be noted that the measured prompt gamma intensities do not depend on the DCC, only a fitting of the responses with the DCC curve is done consequent to the measurement. Thus with any change in DCC values the measured data remains invariant, only an offline refitting need to be done to get new coefficients. The design of the presently fabricated instrument to estimate $H^*(10)$ for neutrons was optimized theoretically in a previous work [6] by the authors. In the earlier done optimization process the outer BHDPE layer was varied. Similarly, the inside HDPE diameter has also been varied to find an optimal combination of the present configuration. Also we have exchanged the position of HDPE and BHDPE to find that HDPE cylinder with diameter 10.5 cm and 11.5 cm in length surrounded by a BHDPE layer of 5.5 cm on all sides approximates the DCC-ICRP more closely.

In a workplace radiation environment, neutrons are always associated with significant gamma background. It is required to discriminate neutron response against gamma response, particularly when the measurement system employed contains a gamma detector. This discrimination is inherent in the characteristic prompt gamma emission mechanism which is activated only by neutrons and not gammas. The prompt gammas are of discrete energies and are more or less uniquely dependent on the elements emitting them. The elements are so chosen that the intensities of these prompt gammas are much larger than the ambient gamma background where the measurement is carried out. A gamma spectrum measured without the BHDPE+HDPE cylinder gives an estimate of the BHDPE+HDPE cylinder independent background spectra.

Present work aims to experimentally evaluate the performance of the earlier designed BHDPE+HDPE configuration [6]. To do this the combined BHDPE+HDPE cylinder is exposed to neutrons from a ^{241}Am-Be source placed inside a thick concrete enclosure with an opening on one side. The resulting prompt gamma emissions from the BHDPE+HDPE cylinder is measured using a 7.6 cm × 7.6 cm NaI(Tl) detector.

2 Materials and methods

A hollow cylinder of BHDPE having 5% boron content covering a HDPE cylinder is the system used for the estimation of neutron dose. The system details are as follows, density of HDPE is 0.92 g cm^{-3} and the weight fraction of hydrogen is 0.144 while for carbon it is 0.856. The density of BHDPE is 1.01 g cm^{-3} with 0.95 weight fraction for HDPE and 0.05 for boron. The inside HDPE cylinder has radius of 5.25 cm, height 10.5 cm surrounded by a cylindrical outer BHDPE of thickness 5.5 cm and height 21.5 cm. Figure 1 shows the fabricated HDPE and BHDPE system. The gammas emitted from constituent atoms (H, B and C) are due to the reactions ^{10}B (n,α) ^{7}Li (γ-energy 0.478 MeV), ^{1}H (n,γ) ^{2}H (γ-energy 2.223 MeV) and ^{12}C $(n,n'\gamma)$ ^{12}C (γ-energy 4.439 MeV), respectively. It is expected that the contributions from the boron, hydrogen and carbon to the prompt gamma spectrum can be identified through well-resolved peaks in the spectrum.

A schematic view of the experimental setup is given in fig. 2(a) along with the neutron energy spectrum (fig. 2(b)) emerging from the ^{241}Am-Be source placed inside the concrete bunker (enclosure) through the opening and incident on the detector system. The emergent neutron spectrum from the concrete bunker was calculated using the FLUKA Monte Carlo simulations [6–8] considering the geometry shown in fig. 2(a).

A neutron source of 592 GBq ^{241}Am-Be is used for the measurements. The source is doubly encapsulated in stainless steel and is placed inside a solid concrete enclosure of 140 cm × 140 cm × 175 cm having an opening of 40 cm × 40 cm on one side. The high-intensity neutron source is kept inside the concrete bunker to ensure radiological safety. Moreover, the emergent spectrum from the concrete bunker, due to multiple scattering inside the bunker, has large number of low-energy neutrons, which is similar to the spectral shape of a workplace neutron environment. Figure 2(b) gives the energy distribution of emergent neutrons from the bunker.

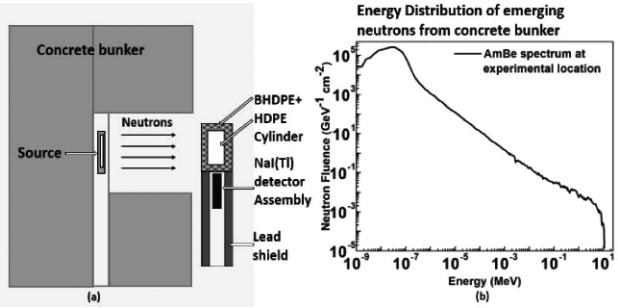

Fig. 2. (a) Cross-sectional view of the experimental setup used in the FLUKA simulations and (b) emergent neutron spectrum of ^{241}Am-Be from the concrete bunker obtained using Monte Carlo simulations.

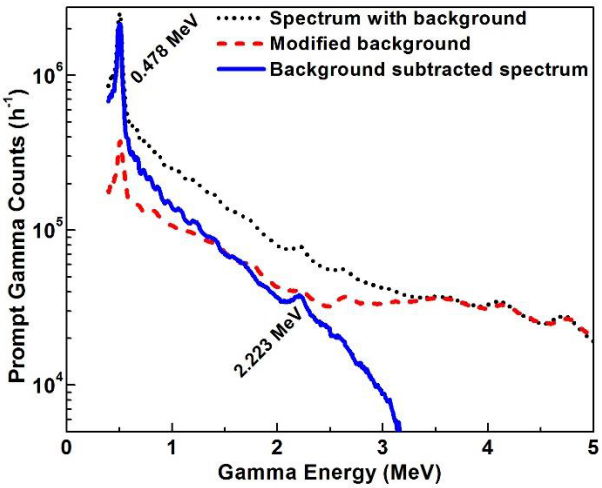

Fig. 3. Prompt gamma spectra from the BHDPE+HDPE cylinder before (dotted line) and after background subtraction (solid line). The background spectrum used here is the attenuation corrected background (dash). Energy calibration is about 8 keV per channel.

The BHDPE+HDPE cylinder is placed vertically in front of the opening in the concrete at a distance of 85 cm from the center of the source. The emitted gammas are detected by the NaI(Tl) detector, which is placed directly below the BHDPE+HDPE cylinder, covered with an annular lead shield of 20 cm outer diameter and 10 cm inner diameter. An USB based 1024 channel analyzer coupled to a 7.6 cm × 7.6 cm NaI(Tl) detector is used as gamma spectroscopy system. Measurements were done for 3600 s live time and the measured spectrum is shown in fig. 3.

2.1 Dose estimation

It is expected that the contributions from the boron, hydrogen and carbon to the prompt gamma spectrum can be identified through well resolved peaks in the spectrum. For dose estimation the technique requires estimation of the areas under the prompt gamma peaks excluding the continuum gammas present in the spectrum. With such requirements fulfilled the dose estimation using the area under the peak of the emitted prompt gammas can be mathematically described as follows.

The net area under the peak A_i can be related with the incident neutron energy distribution φ_j and response matrix R_{ij} as

$$A_i = \sum_j R_{ij}\varphi_j, \tag{1}$$

where i and j indicate the constituent elements in the BHDPE+HDPE cylinder and energy bin of incident neutron energy distribution, respectively. The response R_{ij} is the probability that an emitted neutron from the source in energy bin j interacts with the nuclei present in the system, undergoes possible change in the energy and produces a gamma photon detectable by the NaI(Tl) detector and contributes to the peak area count A_i.

The constituents of the matrix R_{ij} for different elements are combined and fitted with the DCC-ICRP [4] which is denoted by F_j. It is done by obtaining the best fit parameters C_i estimated using the weighted least square minimization techniques:

$$F_j = \sum_i C_i R_{ij} \pm \varepsilon_j. \tag{2}$$

Here ε_j is the fitting error in the j-th energy bin. The optimal values of the coefficients (C_i, $i = 1, 2, 3$) are obtained by minimizing the magnitude of ε_j.

The neutron ambient dose and neutron fluence can be related as

$$H^*(10) = \sum_j F_j \varphi_j. \tag{3}$$

Using eqs. (1) and (2) in eq. (3),

$$H^*(10) \cong H_A^*(10) = \sum_i C_i \sum_j R_{ij} \varphi_j = \sum_i C_i A_i. \tag{4}$$

Here $H_A^*(10)$ gives the estimated value of neutron ambient dose equivalent using the measured area under the peak from the obtained prompt gamma spectrum and closely approximates the actual dose $H^*(10)$.

2.2 Monte Carlo simulations

The estimation of the elements of the response matrix R_{ij} is done using the Monte Carlo simulation with the FLUKA code (version 2011.2c) [7,8]. The simulation involves a plane parallel beam of mono-energetic neutrons or neutrons having energy distributions. For response matrix estimations a mono-energetic neutron of j-th energy bin is made to fall on the BHDPE+HDPE cylinder which interacts with the constituent nuclei resulting in the emission of a prompt gamma which is then detected by a NaI(Tl) detector covered by lead shield. The processes involved in this are chosen randomly, such as collision point of neutron inside the BHDPE+HDPE cylinder which depends on the macroscopic neutron interaction cross-section, reaction type (elastic or inelastic) and interacting nucleus (boron, hydrogen and carbon). Once the neutron is incident it is followed until either it is absorbed inside or escapes the BHDPE+HDPE cylinder. The prompt gammas emitted during this process are followed until they are detected by the detector. To track these neutrons and gammas a track length estimator known as USRTRACK is used which is an option available within the FLUKA code. The range of incident neutron energy varies from 1×10^{-9} MeV to 20 MeV and the emitted gammas are estimated in the energy range 0.1 MeV to 8 MeV with a width of 0.1 MeV. The simulation is carried out for 10^7 run histories which is divided into 5 batches.

3 Results and discussions

Prior to measuring the gamma spectrum emitted from the BHDPE+HDPE cylinder, background gamma spectrum is measured without the BHDPE+HDPE cylinder in position. By subtracting the background without the cylinder in position from the original spectra one can eliminate the background gammas from contributing to the peak counts. However it is observed that the background counts as observed without the BHDPE+HDPE cylinder is attenuated by the cylinder when it is in position and hence do not represent the actual background when the cylinder is in position.

To account for this discrepancy we have calculated the gamma attenuation through the BHDPE+HDPE cylinder using Monte Carlo simulations and have modified the background spectrum accordingly. Figure 3 shows the prompt gamma spectrum before and after background subtraction. From the plot one can observe the two well separated prompt gamma peaks from boron and hydrogen. The carbon peak is not decipherable in the present measurement because the number of neutrons above 5 MeV is rather small (about 6% of the total incident neutrons).

As can be seen from fig. 3, the total absorption peaks of boron and hydrogen are located on top of an almost linear (or piecewise linear) decreasing continuous background. To eliminate this background gamma contribution to the peak area, a straight line fit is made to the end points of the peak and the estimated contribution of background to the peak area is subtracted.

Figure 4 gives a comparison of energy-dependent responses $\sum_i C_i R_{i,j}$ (DCC-estimated) with the DCC-ICRP for only HDPE, only BHDPE and the present system of BHDPE+HDPE combination. It can be observed that the DCC-estimated for the combination system gives much better agreement with the DCC-ICRP. It might appear surprising that the DCC-estimated values of only BHDPE are higher compared to those of BHDPE+HDPE at low energies. This is because the responses plotted in fig. 4 are results of least square fittings (regression) of the individual responses

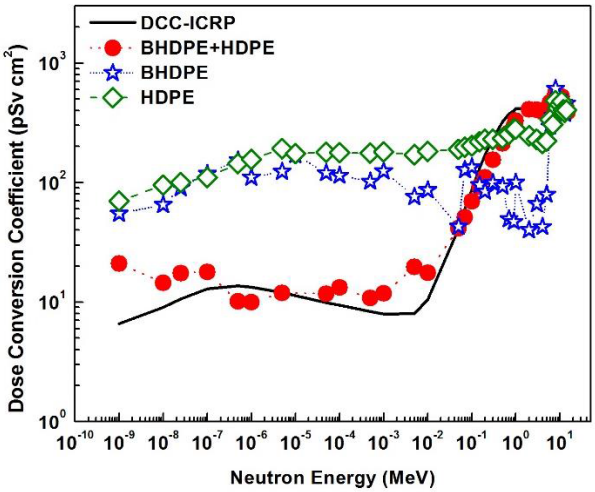

Fig. 4. Comparison of DCC_estimated for cylindrical systems of only HDPE, only BHDPE and the BHDPE+HDPE combination (present system) with the DCC-ICRP.

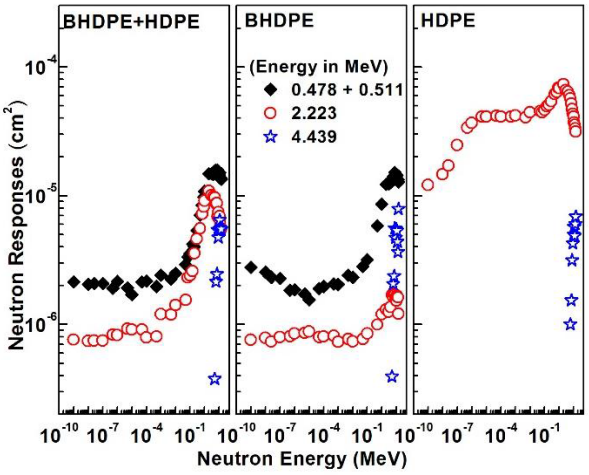

Fig. 5. Plots of individual responses (R) of 0.478 MeV, 2.223 MeV and 4.439 MeV in the case of BHDPE+HDPE, BHDPE and HDPE.

of elements with respect to DCC-ICRP. In the case of BHDPE the high-energy response also gets reduced due to high absorption cross-section of boron and to make the overall fit match better (in the least square sense) the low energy part of the response becomes higher. For the combined system, there is no underestimation of the response from thermal to 1 eV and no significant overestimation in the energy range between 1 eV and 100 keV.

To further investigate this trend the individual responses (R) from boron (0.478 MeV + 0.511 MeV), hydrogen (2.223 MeV) and carbon (4.439 MeV) are plotted in fig. 5 for only HDPE, only BHDPE and BHDPE+HDPE combination. The y-axis scale is same for all the plots. Since, the two peaks 0.478 MeV and 0.511 MeV cannot be separated by the NaI(Tl) detector, a combined response of these two gammas are considered in the present work.

It can be observed that for only HDPE the response of 2.223 gammas from hydrogen is very high, whereas for only BHDPE it is very small. This is because most of the neutron is captured by boron which has a very high capture cross-section (3844 barns at thermal neutron energies). As mentioned earlier, this causes an upward shift in the combined response at low energies during regression fitting with DCC-ICRP. In the case of BHDPE+HDPE combination the response of 2.223 MeV gammas are higher than that in only BHDPE making the final response of the system closer to DCC-ICRP after the regression. Moreover, the shape of the energy-dependent response from 2.223 MeV gammas changes between BHDPE and BHDPE+HDPE making the shape closer to the DCC-ICRP curve for the latter. The response of 4.439 MeV gammas from carbon remains more or less the same in all the three cases since this gamma is produced in the initial few centimeters in the system from neutrons above 5 MeV.

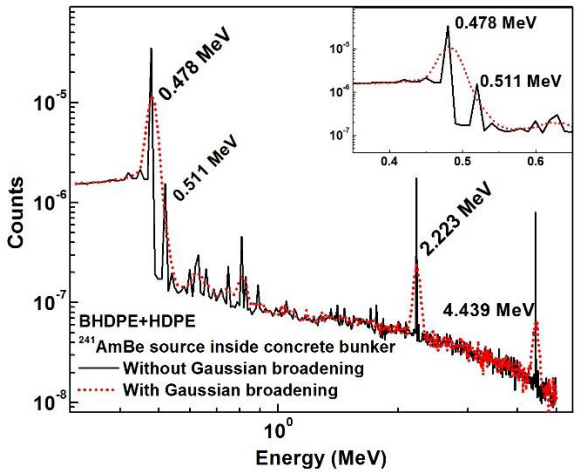

Fig. 6. Plot of discrete prompt gamma energies from the BHDPE+HDPE system along with a Gaussian-broadened energy distribution as observed by a NaI(Tl) detector for the neutrons from ^{241}Am-Be placed inside the concrete bunker. The inset in figure shows an expanded view of 0.478 MeV and 0.511 MeV gamma peaks.

There are possible practical difficulties while estimating the experimental dose. This difficulty mainly involves the boron peak area calculations. The boron peak which is at 0.478 MeV has significant interference from the 0.511 MeV gammas which are the results of annihilation gammas originating in the lead shield or the surrounding material due to the pair production from high energy gammas. The NaI(Tl) detector used in the present scenario cannot separate these two peaks due to its poor resolving power. These peaks may be resolved by using a high-resolution HpGe detector.

Since the NaI(Tl) detector cannot separate the above two peaks, the 0.511 gammas were also included in the response (R) of boron while estimating the dose. It may be noted that these annihilation gammas are produced in the system as a consequence of neutron interaction (high energy prompt gammas leading to pair production and consequent annihilation), the response due to both these gammas in combination can be utilized to estimate the neutron dose. This is not expected to affect the dose calculation since both are generated in the detection system. Therefore, what is termed here as boron response is actually a combined response due to 0.478 MeV and 0.511 MeV gammas. Hence, there is no need to estimate any contribution from the annihilation gammas. Contributions from outside the detection system will automatically be eliminated during background subtraction.

A prior idea of the range of energy spanned by the ambient neutrons will be helpful. In any case, the absence or the presence of the carbon prompt gamma peak indicates whether this upper limit is below or above 5 MeV of incident neutron energy since the ^{12}C $(\mathrm{n}, \mathrm{n}'\gamma)$ ^{12}C reaction has a threshold of about 5 MeV.

Inclusion of the prompt gamma peak area from carbon in the analysis would be more profitable since carbon prompt gammas are generally not expected to be affected by the background gammas. This, however, is not true in the present case, where capture gammas from silicon (3.54 MeV and 4.93 MeV) and the first escape peak from the 4.93 MeV gammas (4.42 MeV) can interfere with the gamma peak from carbon (4.439 MeV). Possible detection of the gamma peak from carbon with larger counting time is discussed later.

To investigate the contribution of the 0.511 MeV gammas to the 0.478 MeV gamma peak from boron, as well as to estimate the contribution of 4.439 MeV gammas from carbon in the measured spectrum, Monte Carlo simulations have been done for the combined BHDPE+HDPE and for only HDPE when neutrons from the ^{241}Am-Be placed inside the concrete bunker and from a bare ^{241}Am-Be are incident on the systems. These simulated results do not include background gammas generated in the concrete bunker due to neutron capture. Discrete gamma peak as can be observed in a high resolution Hp-Ge detector are shown along with the Gaussian-broadened gamma spectrum as observed in a NaI(Tl) detector are shown. Figure 6 shows the plot of discrete gamma energies (counts per incident neutron) due to prompt gammas emitted from the BHDPE+HDPE system along with a Gaussian-broadened energy distribution as observed by a NaI(Tl) detector. As can be clearly seen, 0.478 MeV and 0.511 MeV cannot be separated by the NaI(Tl) detector. The inset in figure shows an expanded view of these two peaks. Prompt gamma peaks of 2.223 MeV (hydrogen) and 4.439 MeV (carbon) are also observed in the plot. There are peaks observed at around 0.6 MeV and 0.8 MeV due to prompt gammas emitted from Pb-shield exposed to the neutrons.

Similarly, plot for the BHDPE+HDPE system is given in fig. 7 (counts per incident neutron) for a bare ^{241}Am-Be source and the spectra appear similar to the earlier one. The carbon peak (4.439 MeV) is more than 2 times higher than source kept inside concrete bunker, since the number of neurons above 5 MeV is 30 percent of the total.

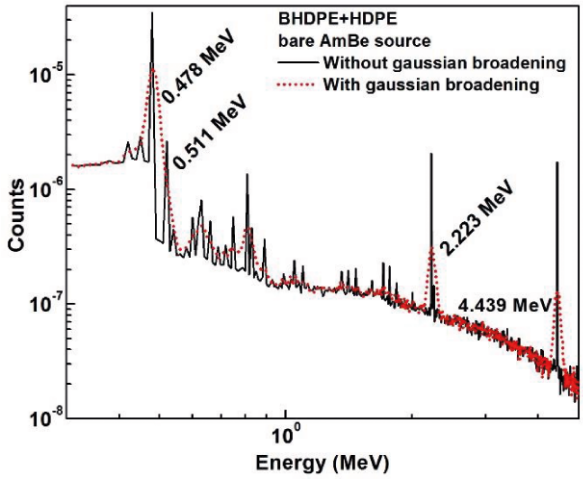

Fig. 7. Plot of discrete prompt gamma energies from the BHDPE+HDPE system along with a Gaussian-broadened energy distribution as observed by a NaI(Tl) detector for the neutrons from a bare ^{241}Am-Be source.

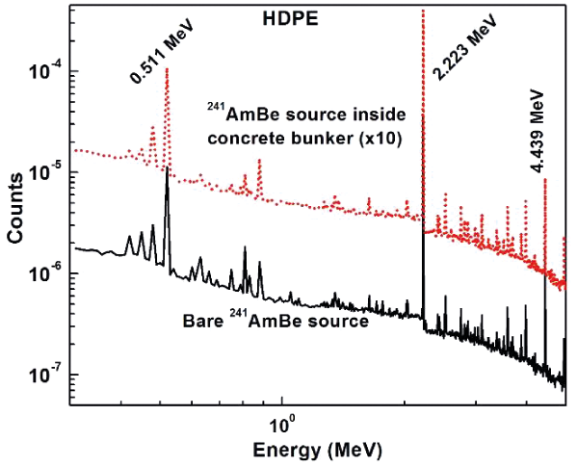

Fig. 8. Plot gives discrete energy spectra for only HDPE system for both bare ^{241}Am-Be and source inside a concrete bunker.

Figure 8 gives discrete energy spectra for only HDPE system. It is found that 0.511 MeV peak is higher in the case of only HDPE by a factor of about 7 compared to the BHDPE+HDPE system when the ^{241}Am-Be placed inside the concrete bunker is used and by a factor of about 5 when the bare ^{241}Am-Be is used. This is because of larger amount of 2.223 MeV gammas produced in the case of HDPE as can be seen from fig. 5.

For the case of ^{241}Am-Be placed inside the concrete bunker used in our measurements, the simulations (without the capture gamma background generated in the concrete bunker) reported in fig. 6 indicate that 4.439 MeV gamma peak from carbon is less by a factor of about 4 compared to the 2.223 MeV peak from hydrogen. Since, the hydrogen peak is detected in the present experiment and not the carbon peak for 1 h counting, it might require about 4–5 h counting with the present system to detect the carbon peak in a statistically significant manner. It can be observed from fig. 3, the 4.93 MeV capture gamma peak from silicon (present in concrete) is quite prominent. This peak interferes with the carbon peak in two ways: First, the first escape peak due to 4.93 MeV gammas is 4.42 MeV which is very close to the carbon peak (4.439 MeV); second, the capture gammas from silicon are produced deep inside the concrete and will scatter and consequently degrade in energy before being incident on the NaI detector. This will cause a long low-energy tail in the energy distribution contaminating carbon peak region.

To examine the possibility of detecting the carbon peak statistically over background fluctuations, the total background counts of 307300 can be considered around the expected carbon peak region where a broad hump is observed in fig. 3. These background counts as well as the expected peak counts 6612 of carbon (obtained by dividing the observed hydrogen peak counts by 4) are spread over 20 channels (8 keV per channel). If the counts are distributed uniformly over 20 channels (which is not very correct but may be accepted observing the broad hump in this region), then background counts per channel are 15365 having a 3 times standard deviation value (σ) of 372, while the expected carbon peak counts per channel are 330, which is less than the 3σ value of the background noise. The background noise obscures the carbon peak. The situation improves for a 4 h counting, when the 3σ value per channel becomes 744 and the corresponding peak counts per channel are 1320.

Table 1. Comparison of measured and theoretical estimations, $H_S^*(10)$ using FLUKA simulation of the experimental scenario.

Theoretical estimations ($mSvh^{-1}$)		Experimental estimations ($mSvh^{-1}$)	
$H_S^*(10)$	1.35 ± 0.05	Present system	1.38 ± 0.14
		Rem-meter	1.42 ± 0.13

The ambient dose equivalent is estimated in the present experimental measurement as (see eq. (4))

$$C_H A_H + C_B A_B, \tag{5}$$

where $C_H = 3.6728 \times 10^5$ pSv is the coefficient corresponding to hydrogen gamma (2.223 MeV) response, $C_B = -1.27 \times 10^4$ pSv is for boron gamma (0.478 MeV and 0.511 MeV) response, $A_H = 26450$ is the counts under the hydrogen peak in 1 hour and $A_B = 655960$ is the counts under the boron peak in 1 hour. The estimated measured dose rate is thus $1.38\,mSvh^{-1}$ and the estimated dose rate using the FLUKA [7,8] Monte Carlo simulations of the experimental scenario, $H_S^*(10)$ are given in table 1 for comparison in units of $mSvh^{-1}$ along with the dose measured using a rem-meter (Ludlum 4142L).

The present system estimates the dose of $1.38\,mSvh^{-1}$ when the corresponding total number of counts (above background) under the peak for hydrogen prompt gammas is 2.64×10^4.

3.1 Estimation of minimum detection level (MDL)

The minimum detection level (MDL) of neutron dose depends on the incident neutron spectrum and fluctuations in the gamma background. As is observed, the capture gammas produced in the concrete bunker contributes significantly to the background. The concrete has about 15% of water content generating large amount of capture gammas from hydrogen leading to background in the present measurement. Similarly, the capture gammas from silicon present in concrete obscures the carbon gamma peak from the BHDPE+HDPE system in 1 h counting. The MDL can be obtained by reducing the neutron source strength computationally such that the actual peak counts (reduced) are just above the background (reduced) fluctuations (3 times the standard deviation) in both hydrogen and boron peaks. The background, in fact, consists of two parts: the first is capture gammas produced in the surroundings because of neutron interactions and varies linearly with neutron source strength; the second is the gamma background in the experimental environment when there is no neutron source present and is independent of neutron source strength. In calculating the minimum detection level, the second part of the background cannot be considered, since it does not reduce when the neutron source strength is reduced. This implies that the second part has to be much smaller than the first part so that it can be neglected.

For the hydrogen peak, the backgrounds counts are 470700 in 1 h whereas the counts above the background are 26450 in 1 h. Three times the standard deviation of background counts is 2058, which is less by a factor 12.85 of the actual counts. Therefore the measured data is statistically valid. The minimum detectable dose rate can be obtained by dividing the measured dose ($1.38\,mSvh^{-1}$) by the square of this factor (165.14) and can be calculated as $8.35\,\mu Svh^{-1}$. This estimation requires that the background counts in the channels corresponding to the hydrogen peak to be much less than 2850 in 1 h when there is no neutron source. This is fulfilled since the total counts detected in these channels are less by a factor of 11.4 and are about 250 counts in 1 h.

For the boron peak (0.478 MeV + 0.511 MeV) the total number of counts above the background is 655960 while three times the standard deviation of background counts is 4647. The square of the ratio between counts and three times the standard deviation of background counts is 19924, which makes the minimum detection level $0.069\,\mu Svh^{-1}$. However, the required background in the boron channel for NaI(Tl) without the neutron source should be much less than 120 counts in 1 h. This is not fulfilled since the measured background without the neutron source is about 600 counts in 1 h. The reduced background then cannot be less than about 6000 (keeping a factor of 10). This increases the MDL by a factor of about 50 and is estimated to be about $3.45\,\mu Svh^{-1}$.

The MDL calculated using the hydrogen peak counts is $8.35\,\mu Svh^{-1}$ and is more than two times the MDL calculated using the boron peak counts. Since this value is higher of the two, it has to be accepted as a conservative estimate of MDL for the present BHDPE+HDPE system and for the present (high) background caused by the capture of neutrons in the concrete bunker. Moreover, the hydrogen peak alone can estimate the dose [5] (though less accurately) while the presence of boron can only improve the estimation but cannot estimate the dose alone. Therefore, the MDL ($8.35\,\mu Svh^{-1}$) calculated using the hydrogen peak is to be accepted.

The MDL of the present instrument is expected to improve if a NaI(Tl) detector of larger dimension is used and by taking into account the total pulse height counts including the Compton contribution instead of only the total absorption peak.

4 Conclusions

The fabricated cylindrical system of combined HDPE and BHDPE functions well as expected theoretically. Experimental measurements confirm accurate dose estimation in presence of high gamma background. The dose estimate from the Monte Carlo simulation of the experimental scenario gives a close estimate of the dose obtained from measurements using the present BHDPE+HDPE cylinder configuration.

The present method of dose estimation is well protected from any possible change in the dose conversion coefficients since the measured areas under the prompt gamma peaks are independent of the dose conversion coefficients, which if changed, can be taken care of using modified fitted coefficients.

Improvement in the sensitivity of the instrument can be achieved by taking into account the total pulse height counts including the Compton contribution and by using a larger volume gamma detector for the emitted prompt gammas.

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