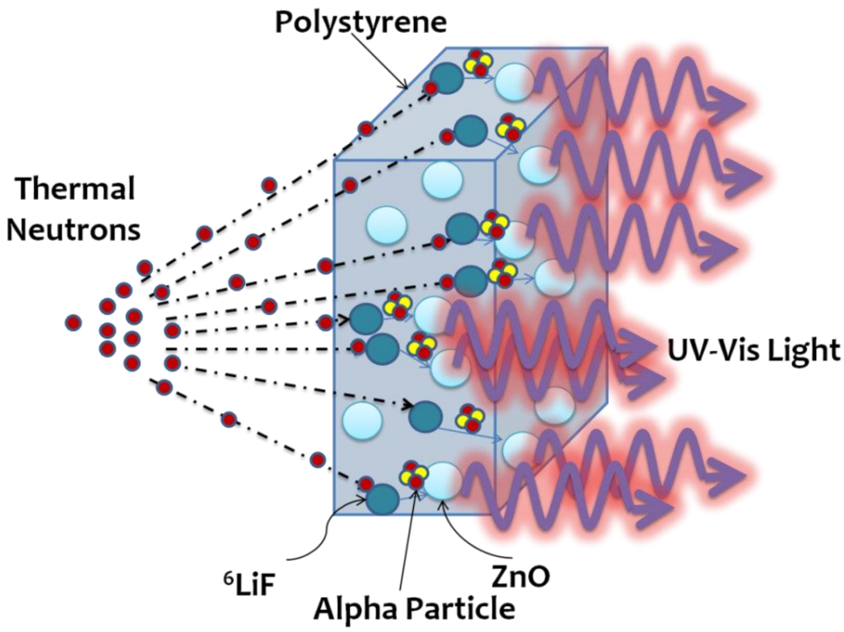


ZnO-^{6}LiF/Polystyrene Composite Scintillator for Neutron Detection

Chapter 7 ZnO-^{6}LiF/Polystyrene Composite Scintillator for Neutron Detection

7.1 Introduction

Neutrons are uncharged particles in various processes such as spontaneous fission of nuclei such as ^{252}Cf, Neutron-induced fission and (Alpha, n) reactions. Neutron Radiation is used in Radio-graphic, Artificial Radioactivity, Detection of hidden explosive/Nuclear materials, Gauges, Scattering and Diffraction experiments, Boron neutron capture therapy (BNCT), etc. Being an electrically neutral particle, it does not interact with the atomic electrons, rather interacts with atomic nuclei, making it difficult to detect compared to charged particles. Neutron interactions with target nuclei are dependent on the energy of the neutron. High energy neutrons usually scatter in-elastically making the target nuclei radioactive called neutron activation. Relatively lower energy neutrons elastically scattered by generating recoiled nuclei that ionize the matter. Passive Detectors such as P-I-N diode and superheated emulsion devices utilize this mechanism for measuring neutron dose. Slow or thermal neutrons are normally absorbed in the target of high capture cross-sections and emit charged particles which further ionize the target medium. This mechanism is the basis of indirect detection of neutrons by ionization process using ^{3}He, BF_3 counters, ^{6}Li Glasses and other detectors. (n, α) the reaction has been a popular reaction for the detection of a thermal neutron. ^{6}Li and ^{10}B offer a very high capture cross-section and are thus used in detectors based on n,α reaction. Alpha particle generated by the neutron capture by these materials is allowed to interact with alpha sensitive materials such as Glass scintillator, ZnS(Ag) and other detectors and generated light is measured. Various alpha sensitive scintillators such as ZnS (Ag), Glass and Plastic scintillator are used in combination of ^{6}Li or ^{10}B materials. Zinc oxide, because of its fast response, is preferred over ZnS(Ag).

In this chapter, the preparation of ZnO-^{6}LiF composite with polystyrene polymer using solution mixing is presented. An alpha-induced scintillation study shows the alpha sensitivity of the composite scintillator. Thus, it can be used to detect neutrons using (n, alpha) reaction. For measurement of neutron radiation, ^{241}Am-Be (1Ci) source was used. ZnO-^{6}LiF /PS film coupled PMT and MCA, placed in Polyethylene container exposed to a neutron source for different time durations and pulse height spectra recorded simultaneously. ZnO-^{6}LiF thermal neutron response was studied and compared to ZnO and natural LiF. A significant increase in pulse height confirms the (n,α) due to the interaction of thermal neutrons with composite scintillator. ZnO and LiF in different weight proportions 2:1, 1:1 and 1:2, keeping total loading 50% (w/w) of polystyrene is prepared. Scintillation pulse heights show that composite with ZnO:^{6}LiF in 1:2 is higher than the other two.

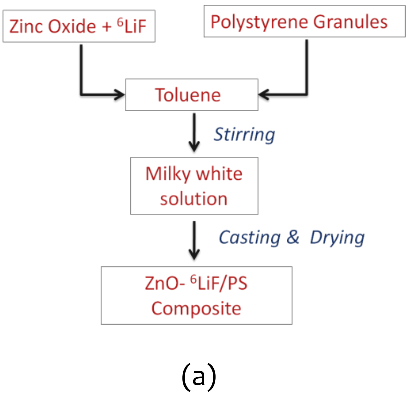
Further repetitive measurements were performed with ZnO:^{6}LiF (1:2) composite and ~1% variation was observed. The background-subtracted scintillation counts for different time duration are computed and plotted with neutron doses, measured using solid-state dosimeter. The response was found to increase linearly with the neutron dose. Similarly, neutron measurement at different source to detector distances is carried out and counts plotted against the distance, showing a similar trend from solid-state dosimeter. This study showed that ZnO-^{6}LiF/PS could be used for the measurement of neutron radiation.

7.2 Experimental Procedure

Zinc Oxide (99.999 %, NOAH), ^{6}LiF (Sigma Aldrich, 95% ^{6}Li enriched) mixed in different ratios 2:1, 1:1, 1:2 in separate vials. Polystyrene fine granules are added in the same quantity in all three vials and shacked to get a uniform mixture with ZnO and ^{6}LiF. Toluene is then added and mechanically stirred until a uniform viscous milky solution is obtained. The solution is casted on quartz substrate and allowed to dry for 24 hours to get ~150 µm ZnO-^{6}LiF/Polystyrene composite scintillator. The process is schematically summarized in figure 7.1 a and the optical photograph in figure 7.1 b. The scintillator is then coupled to PMT and MCA as in alpha measurement reported in the previous chapter. The scintillator coupled PMT is then placed in high-density polyethylene (HDPE) cylinder. Am-Be (1Ci) neutron source is used for this study. Neutron pass through ~4.5 mm thick HDPE before reaching the scintillator. HDPE absorbs the neutron energy making them slow, suitable for capture due to ^{6}Li. To reduce the gamma-radiation emitted from Am-Be source, the neutron beam is covered by 3 cm thick lead, reducing the gamma radiation considerably. Thermal neutron incident on the composite scintillator interacts with ^{6}Li isotope and generates alpha particles, which then interacts with Zinc Oxide generating UV-Vis scintillation photons measured by the PMT (figure 7.2a).

$$ {}_{0}^{1}n + {}_{5}^{10}Li \rightarrow {}_{1}^{3}H + {}_{2}^{4}\alpha $$
$$ {}_{2}^{4}\alpha + ZnO \rightarrow ZnO^{*} \rightarrow ZnO + hv $$

The actual experimental set-up is shown in figure 7.2 b and the Gamma-neutron dosimeter used in neutron measurement is shown in 7.2 c. Pulse height distribution recorded from 1k MCA with in-built voltage divider for PMT, preamplifier and spectroscopic amplifier for 0.5-5 hrs. of neutron exposure and compared with PMT without scintillator.



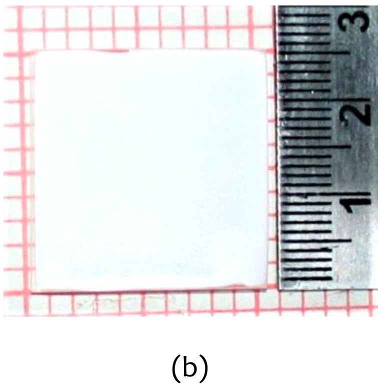
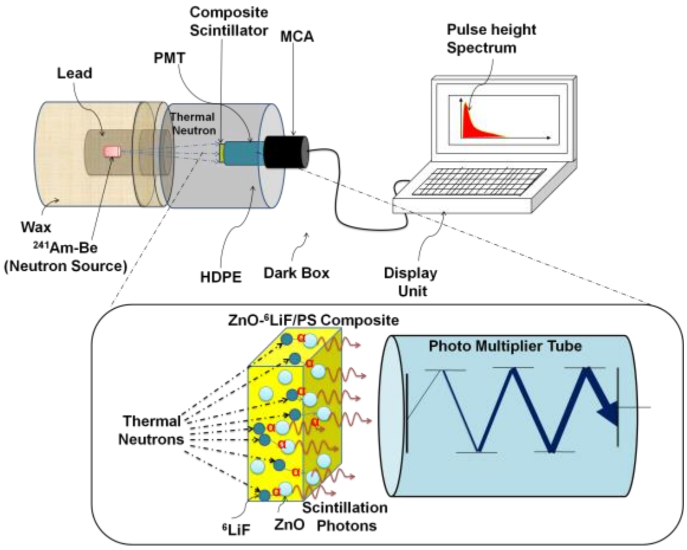


Figure 7.1: (a) Schematic for preparation of ZnO-^{6}LiF/PS Composite (b) Optical photograph of developed composite



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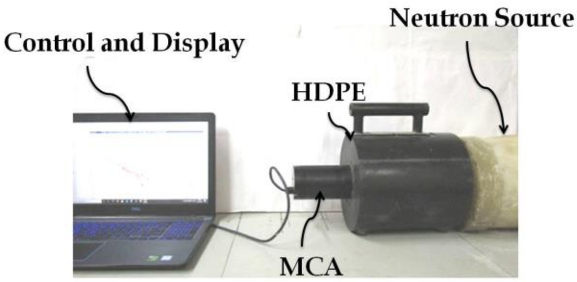

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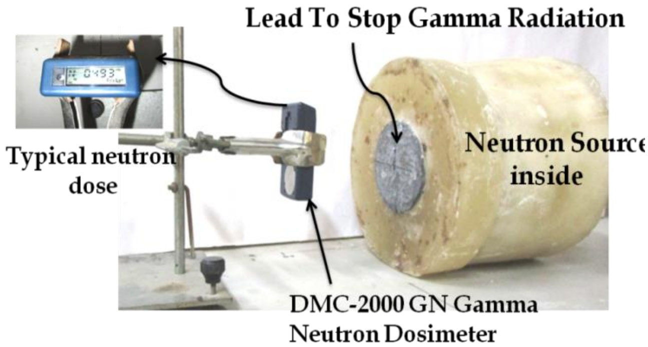


Figure 7.2: (a) Schematic diagram of the experimental set up (b) Actual Measurement set-up (c) Radiation Measurement using DMC 2000GN Gamma Neutron Dosimeter

7.3 Results and Discussions

7.3.1 Neutron Radiation Response

Pulse height spectra recorded using ZnO-LiF/PS composite with ZnO and 95% ^{6}Li enriched LiF in ratio 1:2 along with blank PMT without any scintillator is shown in figure 7.3. To confirm the neutrons, the alpha reaction is happening in the ZnO-^{6}LiF/PS composite; pulse height is recorded using a sample containing natural LiF in the same ratio as the enriched one. It is evident neutron response for enriched LiF is significantly higher as compared with that of natural LiF.

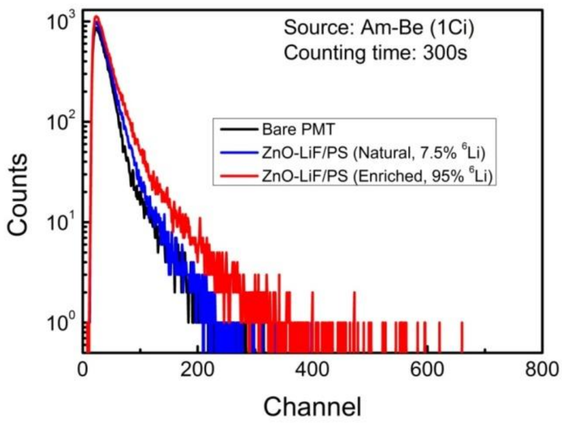


Figure 7.3: Neutron response of ZnO-LiF/PS composite with Natural LiF and ^{6}Li enriched LiF

7.4 Neutron Response with Different Ratios of ZnO and ^{6}LiF

To optimize the ratio of ZnO and LiF, Pulse height spectra recorded using different composite samples with ZnO:LiF as 2:1, 1:1 and 1:2 along with the blank PMT are shown in figure 7.4. The Net counts obtained by subtracting blank PMT counts with the counts with composite are shown as a bar diagram in the inset of figure 7.4. The lowest response is obtained from ZnO-LiF (1:2) and the maximum for ZnO-LiF (1:1). Also, to see the variation of net counts in multiple runs, each experiment is performed five times and net counts for all the three composites is plotted in figure 7.5. The variations are within ±5%.

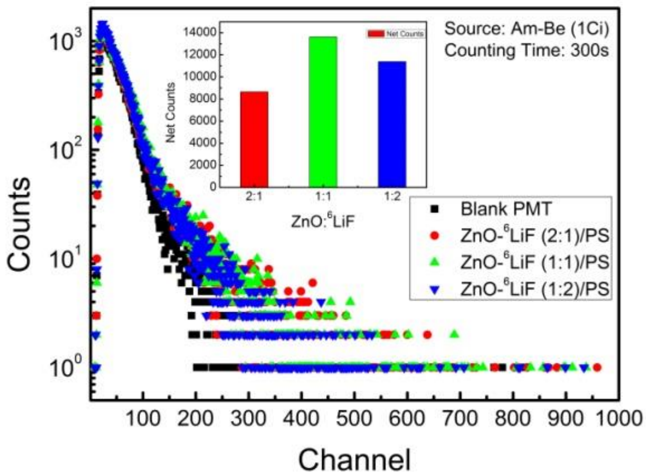


Figure 7.4: Neutron radiation response of ZnO-^{6}LiF/PS composite with three different concentration ratios

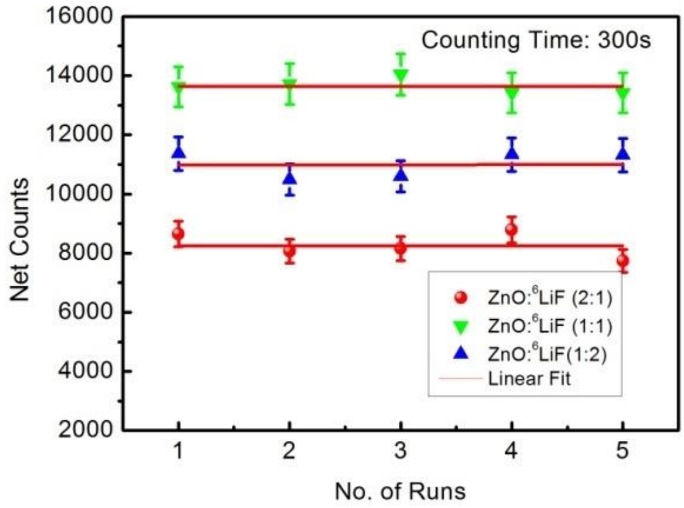
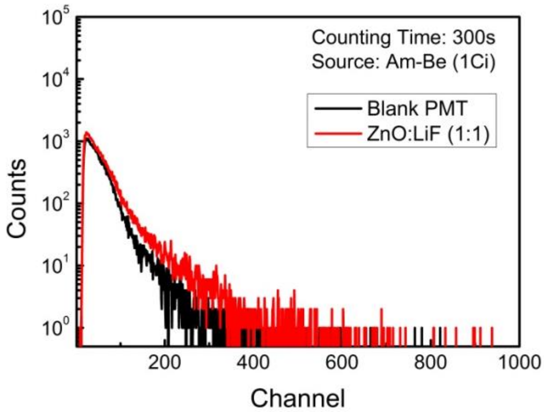


Figure 7.5: Repeatability of Net Counts obtained from ZnO-6LiF/PS composites

7.4.1 Neutron Dose measurement with at different counting time

Pulse height spectra recorded using ZnO-LiF (1:1)/PS composite for different counting times ~300s to 4 hrs. at constant source to detector distance of 24.5 cm. Figure 7.6 e-f shows all these recorded spectra along with blank PMT. As the counting time is increasing, the gap between blank PMT and due to composite is increasing. Net counts are obtained by subtracting the empty PMT response from the composite response and are tabulated in table 7.1.



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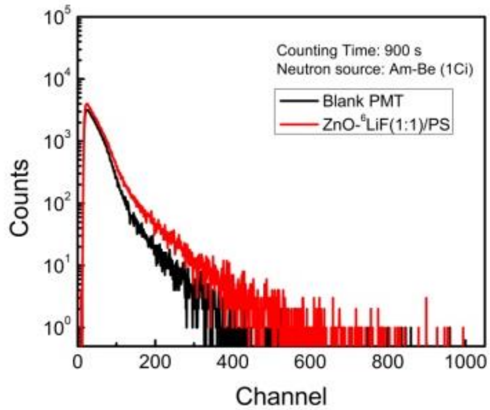

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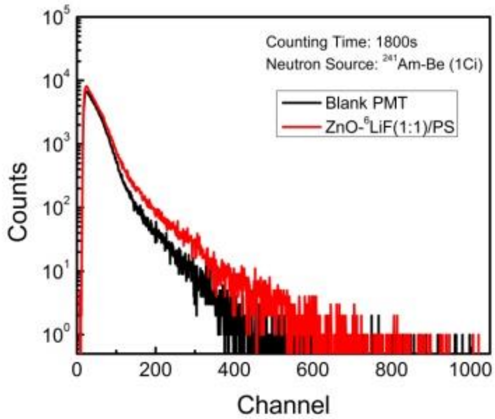

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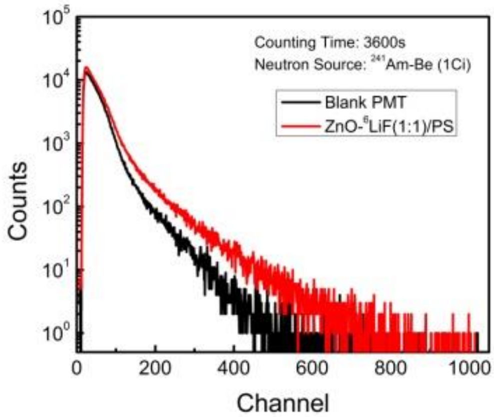

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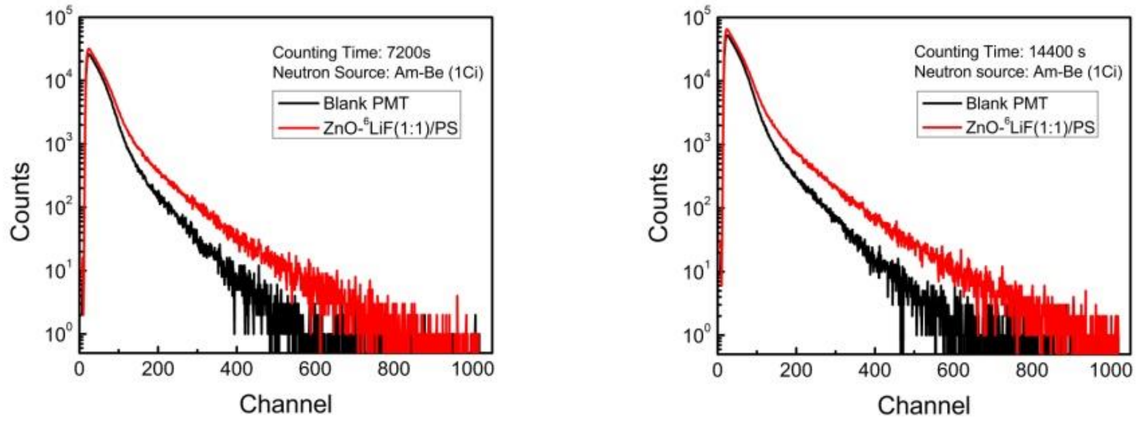


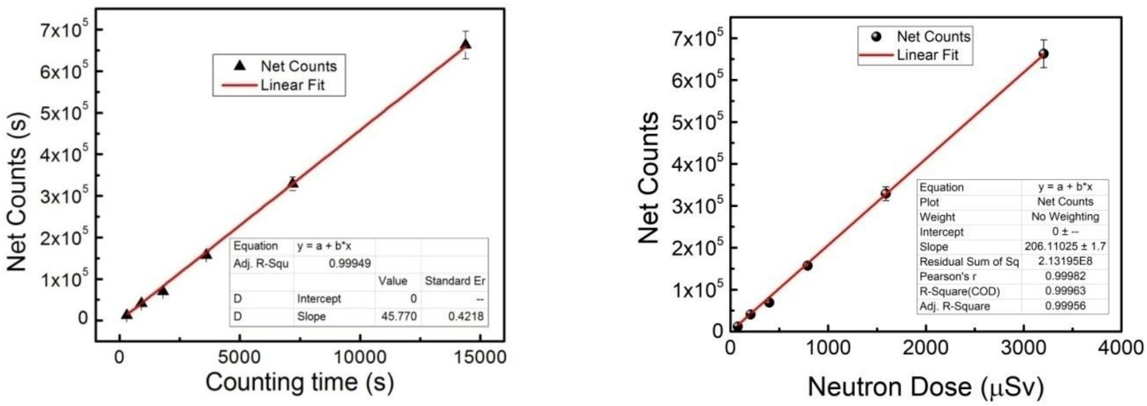
(e) (f)

Figure 7.6: Neutron Radiation Measurement at different time durations

Net counts are linearly increasing with counting time (figure 7.7a). Neutron dose is measured using a solid-state Gamma-Neutron dosimeter (Mirion, Model: DMC 2000 GN) and at the same counting times measured dose is also tabulated in table 7.2. Net counts plotted vs. neutron dose in figure 7.7b. The sensitivity of ZnO-^{6}LiF/PS is obtained by the slope of the line and is ~206 counts/µSv

Table 7.1: Net Counts measured from ZnO-^{6}LiF/PS composite and Fast neutron dose measured using solid-state dosimeter

Counting Time (s)	Net Counts	Measured Dose (µSv)
300	12000	76
900	40981	206
1800	69445	398
3600	157030	789
7200	328997	1589
14400	662967	3205

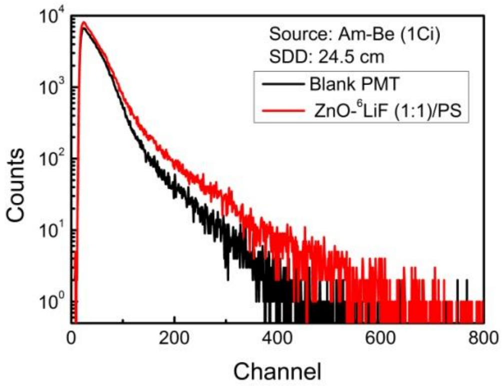


(a) (b)

Figure 7.7: Net counts at different counting time (b) Net counts from ZnO-^{6}LiF vs. dose

7.5 Response against a variation of Source to detector distance

The detector's response against the variation of the source to detector distance was studied at 24.5 to 104.5 cm in different steps. Pulse height spectrum recorded at these distances are plotted in figure 7.8 and net counts are plotted with distance in figure 7.9. As the distance is increasing, the net count is rapidly decreasing. Neutron doses also measured using gamma neutron dosimeter shows a nearly similar trend (figure 6.8)



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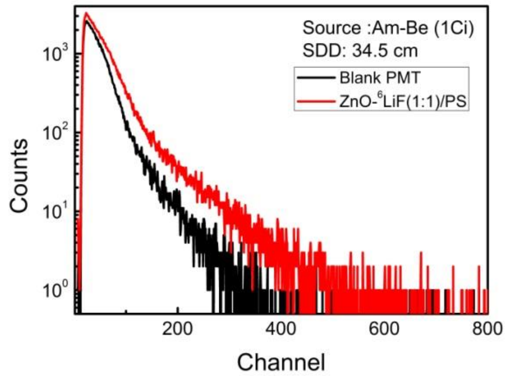

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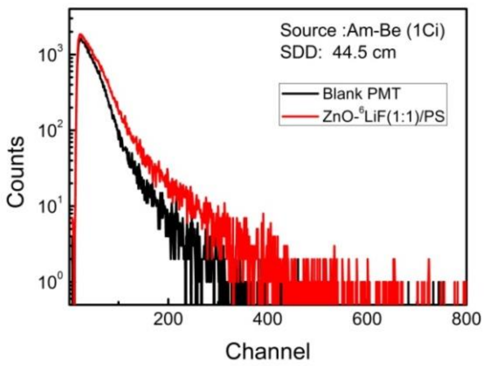

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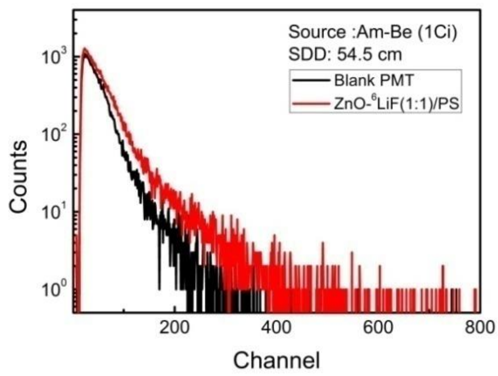

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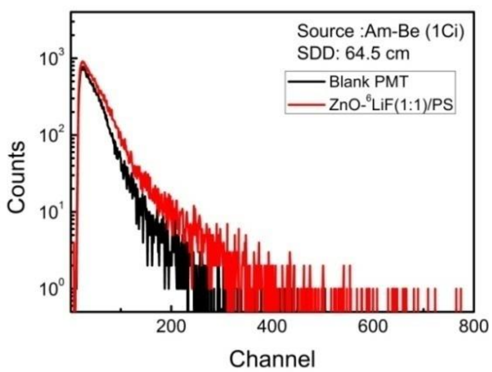

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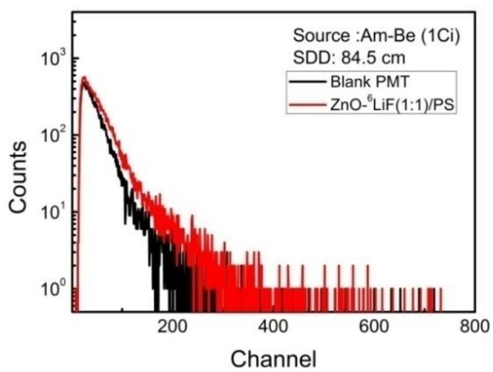

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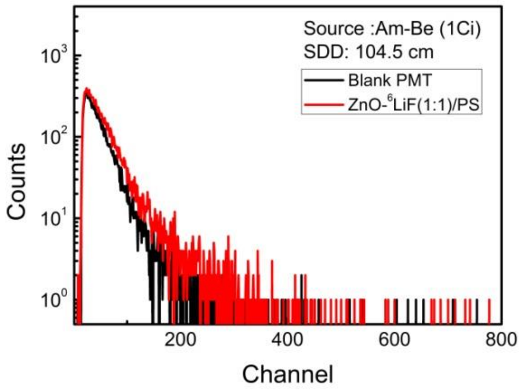

(g)

Figure 7.8: Pulse height spectra obtained at different source to detector distance (a-e)

Table 7.2: Neutron radiation measurement using ZnO-^{6}LiF (1:1)/PS composite and Gamma neutron dosimeter

Source to Distance (cm)	Net Counts	Radiation dose Measured by Gamma Neutron Dosimeter
24.5	69466	398
34.5	42347	262
44.5	20493	160
54.5	14636	90
64.5	9190	52
84.5	6468	45
100.5	3870	18

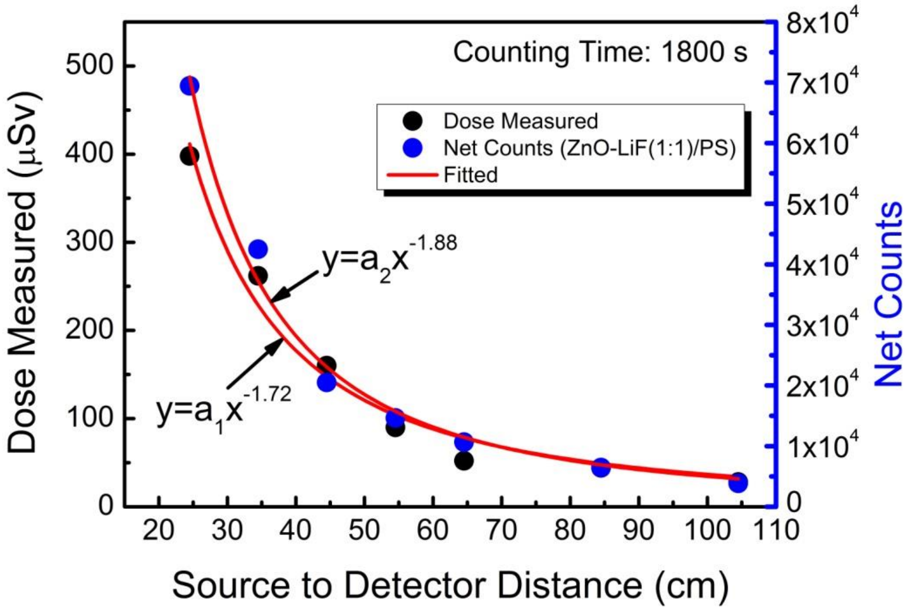

Figure 7.9: Neutron Response of ZnO-^{6}LiF (1:1)/PS composite at various source to detector distance

7.6 Conclusion

ZnO-^{6}LiF/Polystyrene composite prepared using the cost-effective method. Comparison of Neutron response with natural and ^{6}Li enriched LiF based composite confirm the n, alpha reaction as increased pulse height obtained in 95% ^{6}Li enriched LiF. The ratio of ZnO and LiF affects the performance of the composite scintillator and the 1:1 ratio is found optimum for highest response. Net counts are repeatable within ± 5% error and are linear with counting time. The sensitivity of neutron radiation measurement is obtained to ~150 counts/µSv. The response of composite with a source to detector distance has similar variation as obtained from standard dosimeter. The promising response of the ZnO-^{6}LiF/Polystyrene composite detector makes it a suitable material for neutron radiation measurement.