



Experimental/Computational Results for Shielding Neutron & Photon Radiations

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Abstract: This report demonstrates the results of the experiment that was carried out in the Radiation Laboratory in University of Liverpool. For the Photon experiment, two sources were used Cesium-137(661.5eV) and Sodium-22(1275eV) and absorber materials were Aluminum, Lead and Copper. The detector used for the photon experiment was Sodium Iodide Scintillator. For the attenuation of gamma radiation, the origin of these radiations is not significant; it is only their energy which determines the attenuation in a given medium. The degree to which gamma radiation is attenuated is dependent upon the energy of the incident gamma radiation, the atomic number and density of the elements in the shielding material, and the thickness of the shielding^[1]. All substances will attenuate gamma rays to some extent, although the linear attenuation coefficient generally increases with the density, for photons of given energy. This coefficient is very roughly proportional to the density and so, the thickness of different materials required to attenuate gamma radiation, of specified energy, to the same extent are inversely proportional to their respective densities. Consequently, where thickness of the shield is an important consideration, a material of high densities would be used to attenuate the gamma rays^[5].

I. Introduction

The most significant radiations for which shielding is required are the primary neutrons and photons. However, shield designed involves more than choosing a suitable shielding material (or materials) and determining the thickness required to decrease the total radiation effectiveness, i.e, the sum of the neutrons and gamma-ray does equivalents to an acceptable value. Absorption of the radiations in the shield is accompanied by the liberation of energy which appears as heat. In order to ensure the physical integrity of the shield material, the temperature distribution must be known. Thus a study of heat generation is an essential aspect of shield design, and this requires a knowledge of the distribution and absorption of neutrons and gamma rays throughout the thickness of the shield^[4].

Neutrons interact in matter via nuclear reactions, which depends strongly on the energy and the particular nuclei with which the neutrons collide. Neutrons may be scattered in which case energy is transferred to recoiling nuclei. Alternatively, they may be absorbed in the variety of different processes. At energies of a few MeV, fusion is the main contributor to the absorption cross section. In fusion, a neutron is first captured to form a compound nucleus excited to energy approximately equal to the initial kinetic energy plus the neutron's binding energy which is, typically, 7 to 8 MeV. This energy is subsequently released in the form of reaction products, which may be gamma rays, charge particles or neutrons. Reactions that result in the emission of the energetic charged particles are important because they enable the presence of neutrons to be detected^[2].

Photons transfer their energy to electrons via Compton scattering, photoelectric effect and pair production. The attenuation coefficient of photons depends strongly on energy and the atomic number of the material. Compton scattering is the most important interaction process for photon energies above 40KeV to several tens of MeV. The scattered photons may interact further by a second Compton scattering or by photo electric absorption, depending on the energy^[4].

II. Aim & Objective

The aim of this experiment was to establish the relationship between the source strength and shield thickness. Monte Carlo simulations using MCNP was also employed to compare the experimental results with the computational.

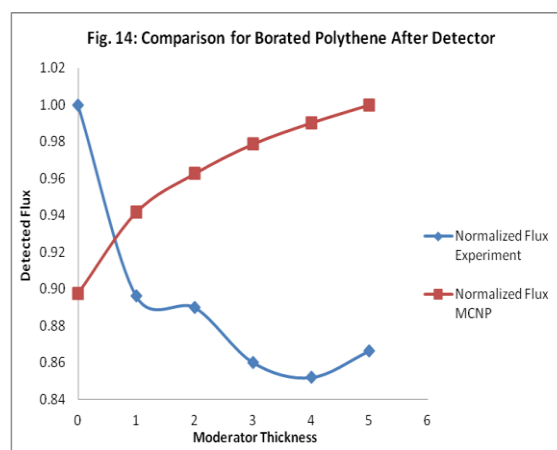
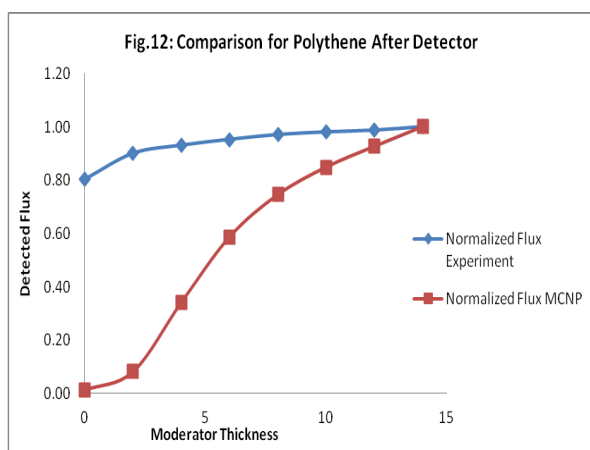
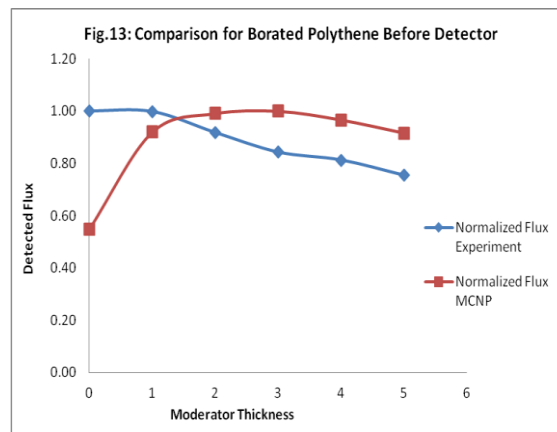
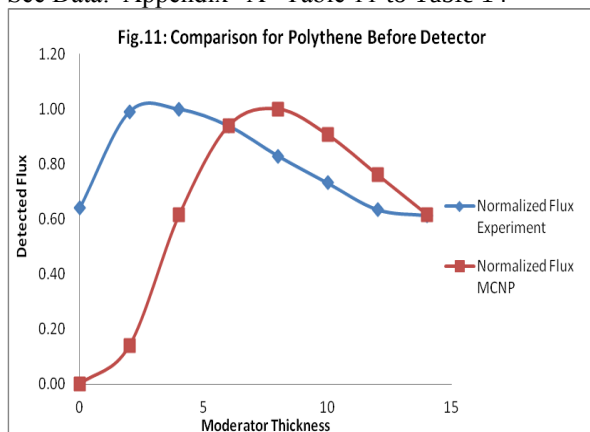
III. Methodology & Results

A. Comparison between Laboratory & Computational Simulations for Neutron

Comment: the comparison of the experimental result and the MCNP simulations as shown in Fig.11, are both in agreement, although the number of neutrons detected in the experimental result when there is no shield and at

small moderator thickness of 2cm & 4cm is greater than that detected using MCNP simulations, whereas the reverse is the case for moderator thickness greater than 4cm. From the graph below, I can conclude that the MCNP simulations results was a bit more accurate than the experimental results because it shows that more neutrons are being captured and some scattered of at smaller thickness than that observed in the experiment. But when the moderator is placed far from the detector as shown in Fig.12, the number of neutrons detected from the experiment point of view is greater than that of MCNP simulations from when there is no shield to the last shield thickness. This result still confirms the fact that MCNP simulations is a bit more reliable than the experimental results I got in the laboratory.

See Data: Appendix "A" Table 11 to Table 14



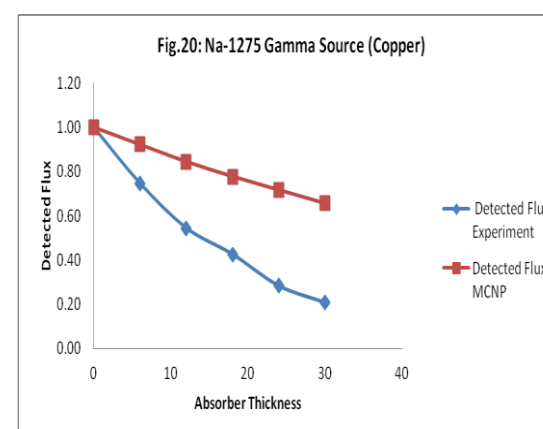
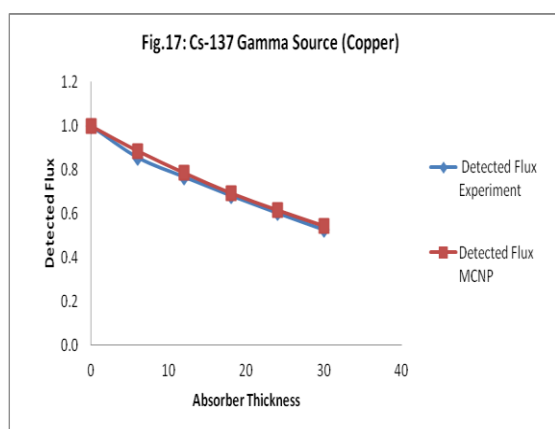
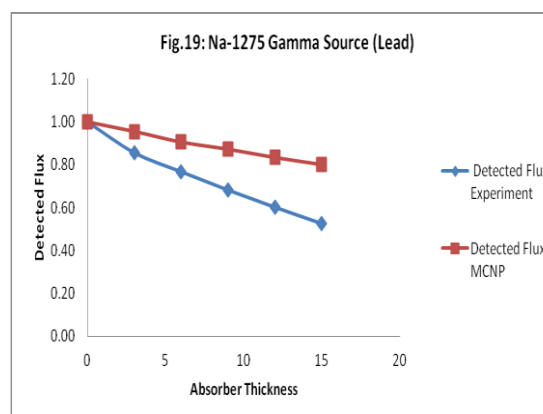
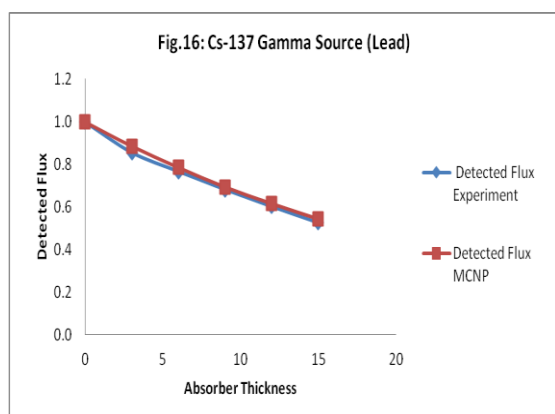
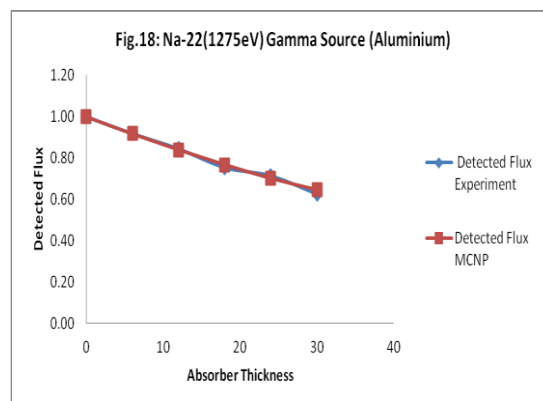
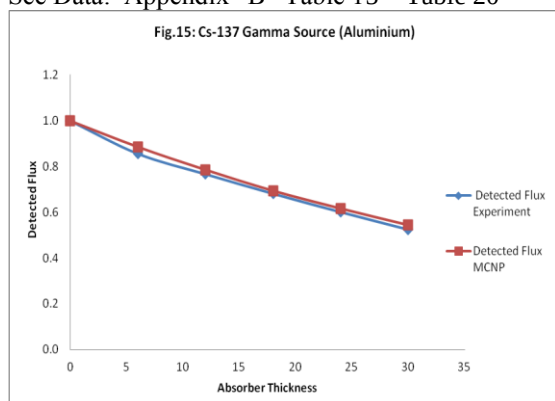
B. Comparison between Laboratory & MCNP Simulation for the Photon Shielding

Comment: the graphs below shows the correlations between the experimental results and the MCNP simulations for the photon experiment using cesium-137 source and sodium-22(1275eV) source. The comparison was carried out for Aluminium, Lead and Copper. According to the result I got (Fig. 15-17), it shows that for Cesium source of the three different materials used, the number of photons detected decreases with increasing thickness for both the experiment and the MCNP simulations. Therefore it can be concluded that the results gotten in both cases where in agreements.

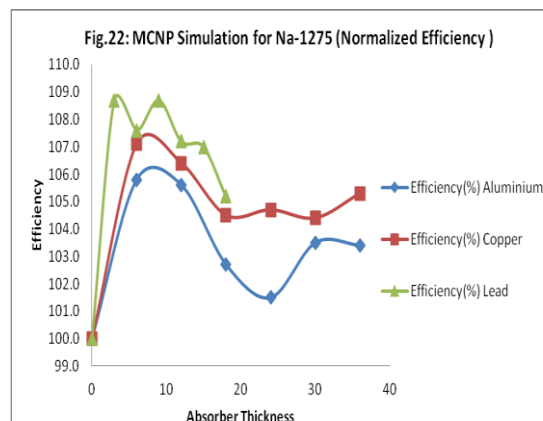
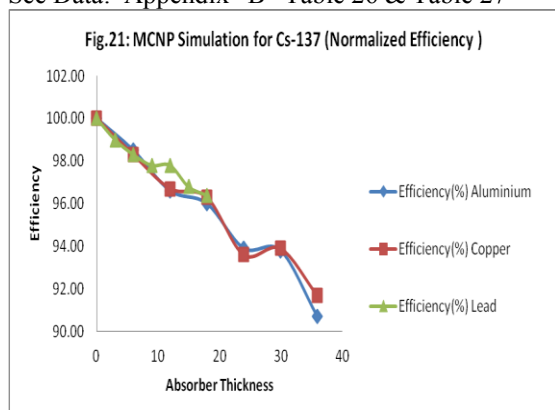
For sodium-22 source, we can observed that for the case of aluminium (Fig.18) the detected flux decreases with increasing thickness as expected without any significant change in the number of detected flux for both cases (experiment & MCNP). But for the case of Lead (Fig.19) we observed that the number of photons detected by the scintillator for the MCNP simulations was higher than that recorded in the experiment. This shows that the experimental results obtained was more accurate than the MCNP simulation. Also, a large observable change is seen in the case of copper (Fig.20), the detected flux for experiment reduces drastically more than the simulations for different absorber thickness. This result implies that the experimental result for using a Sodium-Iodide scintillator is more reliable than MCNP simulations.

Fig 22. Shows the efficiencies for different materials used for the photon experiments. According the graph in Fig22., we can see that the Lead material is the best absorbing material compare to Aluminium and Copper. Fig. 23 also confirm by showing that the number of neutrons detected using a lead material is less than that detected using aluminium and copper, and as such we can conclude that a Lead material is better for shielding of gamma radiation than Aluminium or Copper.

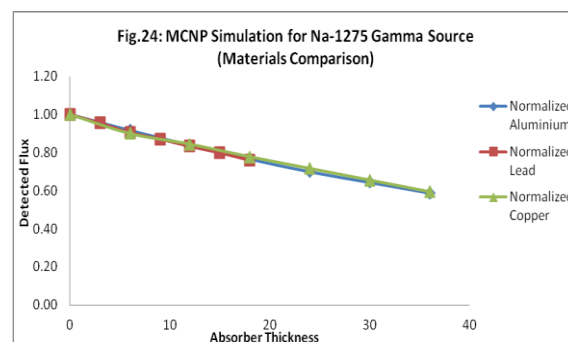
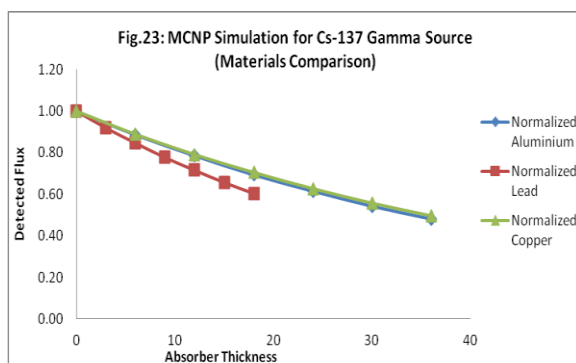
See Data: Appendix “B” Table 15 – Table 20



See Data: Appendix “B” Table 26 & Table 27



See Data: Appendix “B” Table 28 & Table 29



C. Photon Shield Calculations

Na-22 decay scheme spectrum show a gamma ray at 1274.5keV, an annihilation peak at 511keV (from the beta-plus), and probably x-ray from the electron capture. For the shielding scenario, let us assume that all gamma-rays are 1274.5keV and the rest are ignored for the shielding calculations.

Count rate from Na-22 source ($A_{1274.5\text{keV}} = 1 \text{ mCi} = 3.7 \times 10^7 \text{ Bq} = 37 \text{ MBq} = 37 \times 10^6 \text{ cps}$)

Count rate required after shielding $= 1.625 \text{ MBq}$ ($A_{1274\text{keV}} = 0.325 \text{ MBq}$ and $A_{511\text{keV}} = 1.30 \text{ MBq}$) $= 1.625 \times 10^6 \text{ cps}$

The dose received from exposure to an external source emitting radiation in all direction depends on its distance and any shielding which may be in place. Approximate formula for estimating the rate at which a dose is accumulated in tissue at a distance r meters from a radioactive source emitting A photons per second of energy (E_γ) in MeV is given as (Lamarsh & Baratta 2001)^[3]:

$$\text{Dose rate } \frac{dD}{dt} (\mu\text{Svh}^{-1}) \approx \frac{A(\text{MBq}) * E_\gamma(\text{MeV})}{6 * [r(\text{m})]^2}$$

$$A(\text{MBq}) \approx \frac{\frac{dD}{dt} * 6 * [r(\text{m})]^2}{E_\gamma(\text{MeV})}$$

Effective (whole body) dose limit, per year, for radiation workers is;

$$\frac{dD}{dt} = D(x) = 20 \text{ mSvy}^{-1}$$

Assuming a working year is $8 * 5 * 50 = 2000 \text{ hrs}$ during which the worker remains 30cm

away from the source;

$$\frac{dD}{dt} = D(x) = \left(\frac{20 * 10^3}{2000} \right) \mu\text{Svh}^{-1} = 10 \mu\text{Svh}^{-1}$$

Hence, the maximum allowed activity,

$$A(\text{MBq}) \approx \frac{10 \mu\text{Svh}^{-1} * 6 * (0.3)^2 \text{ m}^2}{1.274} = 4.24 \text{ MBq}$$

The dose equivalent rate outside a shield of thickness x is given as;

$$D(x) = D_o e^{-\mu x}$$

Re – arranging we have;

$$x = -\frac{1}{\mu} * \ln\left(\frac{D(x)}{D_o}\right)$$

For Lead thickness where $\mu = 0.793 \text{ cm}^{-1}$ (1274.5keV)

$$x_{Pb} = -\frac{\ln\left(\frac{4.24}{37}\right)}{0.793} = 2.73 \text{ cm}$$

For Aluminium thickness where $\mu = 0.142\text{cm}^{-1}(1274.5\text{keV})$

$$x_{Al} = -\frac{\ln\left(\frac{4.24}{37}\right)}{0.142} = 15.26\text{cm}$$

For Copper thickness where $\mu = 0.492\text{cm}^{-1}(1274.5\text{keV})$

$$x_{Cu} = -\frac{\ln\left(\frac{4.24}{37}\right)}{0.492} = 4.40\text{cm}$$

Let us consider the case where all the gamma-ray photons travel towards the radiation worker.

The Effective dose,

$$E = \sum_T w_T \sum_R w_R D_{T,R}$$

Where; $D_{T,R} = \text{Dose rate to tissue} \approx \frac{dD}{dt} = 10\mu\text{Sv.h}^{-1}$

$w_T = \text{Whole body tissue weighting factor} = 1$

$w_R = \text{Gamma radiation weighting factor} = 1$

$$\therefore E = \left(\frac{10E-6}{3600}\right)\text{Sv.s}^{-1} = 2.78 * 10^{-9}\text{Sv.s}^{-1}$$

The Absorbed dose $D_T = 2.78 * 10^{-9}\text{Gy.s}^{-1}$

For radiation protection purposes, an average absorbed dose for a tissue or organ

$$D_T = \frac{E_T}{m_T}$$

$$E_T = D_T * M_T$$

Assuming mass of a whole body to be 70kg

$$E_T = 2.78 * 10^{-9}\text{Gy.s}^{-1} * 70\text{kg} = 1.94 * 10^{-7}\text{J.s}^{-1}$$

$$\therefore \text{Activity } A(\text{MBq}) = \frac{E_T(\text{Js}^{-1})}{E_\gamma(\text{J})}$$

$$E_\gamma = 1274.5\text{keV} \approx 1274.5 * 10^3\text{eV}$$

$$E_\gamma = (1274.5 * 10^3 * 1.6 * 10^{-19})\text{J} = 2.0392 * 10^{-13}\text{J}$$

$$A = \left(\frac{1.94 * 10^{-7}}{2.0392 * 10^{-13}}\right) = 951353.4719\text{Bq}$$

$$A = 0.951\text{MBq}$$

For Lead thickness where $\mu = 0.793\text{cm}^{-1}(1274.5\text{keV})$

$$x_{Pb} = -\frac{\ln\left(\frac{0.951}{37}\right)}{0.793} = 4.62\text{cm}$$

For Aluminium thickness where $\mu = 0.142\text{cm}^{-1}(1274.5\text{keV})$

$$x_{Al} = -\frac{\ln\left(\frac{0.951}{37}\right)}{0.142} = 25.78\text{cm}$$

For Copper thickness where $\mu = 0.492\text{cm}^{-1}(1274.5\text{keV})$

$$x_{Cu} = - \frac{\ln\left(\frac{0.951}{37}\right)}{0.492} = 7.44 \text{ cm}$$

D. Neutron Shield Calculations

The dose equivalent rate outside a shield of thickness x is given as;

$$D(x) = D_o e^{-\Sigma x}$$

Re – arranging we have;

$$x = - \frac{1}{\Sigma} * \ln\left(\frac{D(x)}{D_o}\right)$$

Effective (whole body) dose limit, per year, for radiation workers is;

$$\frac{dD}{dt} = D(x) = 20 \text{ mSv y}^{-1}$$

Assuming a working year is $8 * 5 * 50 = 2000 \text{ hrs}$ during which the worker remains 30 cm

away from the source;

$$\frac{dD}{dt} = D(x) = \left(\frac{20 * 10^3}{2000}\right) \mu\text{Sv h}^{-1} = 10 \mu\text{Sv h}^{-1}$$

$$\text{but } D_o = \frac{S}{4\pi r^2}$$

Where S = Neutron count rate * Dose factor

Neutron count rate = $2.8 \text{ E}06 \text{ (neutrons/s)}$

Dose factor = $0.156 \text{ (mrem/hr) / (neutrons/cm}^2 \cdot \text{s)}$

$$\therefore D_o = \frac{2.8 \text{ E}6 * 0.156}{4 * \pi * 30^2} = 38.62 \text{ mrem/hr} \approx 386.2 \mu\text{Sv/hr}$$

For Polyethylene thickness where $\Sigma = 0.111 \text{ cm}^{-1}$

$$x_{poly} = - 9 * \ln\left(\frac{10}{386.2}\right) = 32.88 \text{ cm}$$

References

- [1] M. El-Wakil, Nuclear Power Engineering, McGraw-Hill, 1962.
- [2] Glassione and Sesonske, Nuclear Reactor Engineering, Van Nostrand Reinhold Company, 1981.
- [3] J. Lamarsh and A. Baratta, Introduction to Nuclear Engineering, Prentice Hall Inc., 2001.
- [4] J. Lilley, Nuclear Physics Principles and Applications, John Wiley and Sons, Ltd, 2001.
- [5] N. Todreas and M. Kazimi, Nuclear Systems 1 Thermal Hydraulic Fundamentals, Thermisphere Publishing Corporation, 1990.

Table 11:

POLYTHENE BEFORE DETECTOR				
Moderator Thickness (cm)	Detected Flux Experiment	Detected Flux MCNP	Normalized Flux Experiment	Normalized Flux MCNP
0	14163	3.60E-10	0.64	0.00
2	21887	2.97E-08	0.99	0.14
4	22094	1.29E-07	1.00	0.61
6	20752	1.98E-07	0.94	0.94
8	18333	2.10E-07	0.83	1.00
10	16166	1.91E-07	0.73	0.91
12	14040	1.60E-07	0.64	0.76

Table 12:

POLYTHENE AFTER DETECTOR				
Moderator Thickness (cm)	Detected Flux Experiment	Detected Flux MCNP	Normalized Flux Experiment	Normalized Flux MCNP
0	14163	3.60E-10	0.81	0.01
2	15855	2.38E-09	0.90	0.08
4	16384	1.00E-08	0.93	0.34
6	16762	1.73E-08	0.95	0.59
8	17079	2.19E-08	0.97	0.75
10	17245	2.49E-08	0.98	0.85
12	17360	2.72E-08	0.99	0.93

Table 13:

BORATED POLYTHENE BEFORE DETECTOR				
Moderator Thickness (cm)	Detected Flux Experiment	Detected Flux MCNP	Normalized Flux Experiment	Normalized Flux MCNP
0	14196	3.60E-10	1.00	0.55
1	14163	6.05E-10	1.00	0.92
2	13027	6.53E-10	0.92	0.99
3	11965	6.58E-10	0.84	1.00
4	11537	6.36E-10	0.81	0.97
5	10711	6.03E-10	0.75	0.92

Table 14:

BORATED POLYTHENE AFTER DETECTOR				
Moderator Thickness (cm)	Detected Flux Experiment	Detected Flux MCNP	Normalized Flux Experiment	Normalized Flux MCNP
0	14163	3.60E-10	1.00	0.90
1	12692	3.78E-10	0.90	0.94
2	12606	3.86E-10	0.89	0.96
3	12183	3.93E-10	0.86	0.98
4	12066	3.98E-10	0.85	0.99
5	12269	4.01E-10	0.87	1.00

APPENDIX B

Table 15:

Cs-137 Gamma Source (Aluminium)				
Absorber Thickness (mm)	Detected Flux Experiment	Detected Flux MCNP	Normalized Experiment	Normalized MCNP
0	25593.20	3.59E-03	1.0	1.0
6	21911.80	3.18E-03	0.9	0.9
12	19627.80	2.82E-03	0.8	0.8
18	17442.80	2.49E-03	0.7	0.7
24	15389.60	2.21E-03	0.6	0.6
30	13431.00	1.95E-03	0.5	0.5

Table 16:

Cs-137 Gamma Source (Lead)				
Absorber Thickness (mm)	Detected Flux Experiment	Detected Flux MCNP	Normalized Experiment	Normalized MCNP
0	25593.20	3.59E-03	1.00	1.00
3	18605.80	3.30E-03	0.73	0.92
6	13610.00	3.04E-03	0.53	0.85
9	10074.00	2.79E-03	0.39	0.78
12	7469.60	2.57E-03	0.29	0.72
15	5844.00	2.36E-03	0.23	0.66

Table 17:

Cs-137 Gamma Source (Copper)				
Absorber Thickness (mm)	Detected Flux Experiment	Detected Flux MCNP	Normalized Experiment	Normalized MCNP
0	25593.20	3.59E-03	1.00	1.00
6	16265.80	3.19E-03	0.64	0.89
12	10337.00	2.84E-03	0.40	0.79
18	6741.00	2.53E-03	0.26	0.70
24	4496.20	2.25E-03	0.18	0.63
30	3328.20	2.00E-03	0.13	0.56

Table 18:

Na-1275 Gamma Source (Aluminium)				
Absorber Thickness (mm)	Detected Flux Experiment	Detected Flux MCNP	Normalized Experiment	Normalized MCNP
0	5862.97	1.80E-03	1.00	1.00
6	5381.03	1.65E-03	0.92	0.92
12	4951.48	1.51E-03	0.84	0.84
18	4415.73	1.38E-03	0.75	0.77
24	4182.61	1.26E-03	0.71	0.70
30	3662.15	1.16E-03	0.62	0.64

Table 19:

Na-1275 Gamma Source (Lead)				
Absorber Thickness (mm)	Detected Flux Experiment	Detected Flux MCNP	Normalized Experiment	Normalized MCNP
0	25593.20	1.80E-03	1.00	1.00
3	21911.80	1.72E-03	0.86	0.96
6	19627.80	1.63E-03	0.77	0.91
9	17442.80	1.57E-03	0.68	0.87
12	15389.60	1.50E-03	0.60	0.83
15	13431.00	1.44E-03	0.52	0.80

Table 20:

Na-1275 Gamma Source (Copper)				
Absorber Thickness (mm)	Detected Flux Experiment	Detected Flux MCNP	Normalized Experiment	Normalized MCNP
0	5862.97	1.80E-03	1.00	1.00
6	4378.09	1.62E-03	0.75	0.92
12	3191.58	1.52E-03	0.54	0.84
18	2494.06	1.40E-03	0.43	0.78
24	1656.3	1.29E-03	0.28	0.72
30	1210.42	1.18E-03	0.21	0.66

Table 21:

MCNP RESULTS-Aluminium			Cs-137 Gamma Source		
Absorber Thickness (mm)	Incident Flux	Detected Flux	Normalized Incident	Normalized Detected	Efficiency %
0	2.09E-05	3.59E-03	1.000	1.000	100.0
6	1.88E-05	3.18E-03	0.900	0.886	98.5
12	1.70E-05	2.82E-03	0.813	0.786	96.6
18	1.51E-05	2.49E-03	0.722	0.694	96.0
24	1.37E-05	2.21E-03	0.656	0.616	93.9
30	1.21E-05	1.95E-03	0.579	0.543	93.8
36	1.11E-05	1.73E-03	0.531	0.482	90.7

Table 22:

MCNP RESULTS-Lead			Cs-137 Gamma Source		
Absorber Thickness (mm)	Incident Flux	Detected Flux	Normalized Incident	Normalized Detected	Efficiency %
0	2.09E-05	3.59E-03	1.000	1.000	100.0
3	1.94E-05	3.30E-03	0.928	0.919	99.0
6	1.80E-05	3.04E-03	0.861	0.847	98.3
9	1.66E-05	2.79E-03	0.794	0.777	97.8
12	1.53E-05	2.57E-03	0.732	0.716	97.8
15	1.42E-05	2.36E-03	0.679	0.657	96.8
18	1.31E-05	2.17E-03	0.627	0.604	96.4

Table 23:

MCNP RESULTS-Copper			Cs-137 Gamma Source		
Absorber Thickness (mm)	Incident Flux	Detected Flux	Normalized Incident	Normalized Detected	Efficiency %
0	2.09E-05	3.59E-03	1.000	1.000	100.0
6	1.89E-05	3.19E-03	0.904	0.889	98.3
12	1.71E-05	2.84E-03	0.818	0.791	96.7
18	1.53E-05	2.53E-03	0.732	0.705	96.3
24	1.40E-05	2.25E-03	0.670	0.627	93.6
30	1.24E-05	2.00E-03	0.593	0.557	93.9
36	1.13E-05	1.78E-03	0.541	0.496	91.7

Table 24:

MCNP RESULTS-Aluminium			Na-1275 Gamma Source		
Absorber Thickness (mm)	Incident Flux	Detected Flux	Normalized Incident	Normalized Detected	Efficiency %
0	7.34E-06	1.80E-03	1.000	1.000	100.0
6	6.36E-06	1.65E-03	0.866	0.917	105.8
12	5.83E-06	1.51E-03	0.794	0.839	105.6
18	5.48E-06	1.38E-03	0.747	0.767	102.7
24	5.06E-06	1.26E-03	0.689	0.700	101.5
30	4.57E-06	1.16E-03	0.623	0.644	103.5
36	4.18E-06	1.06E-03	0.569	0.589	103.4

Table 25:

MCNP RESULTS-Lead			Na-1275 Gamma Source		
Absorber Thickness (mm)	Incident Flux	Detected Flux	Normalized Incident	Normalized Detected	Efficiency %
0	7.45E-06	1.80E-03	1.000	1.000	100.0
3	6.55E-06	1.72E-03	0.879	0.956	108.7
6	6.27E-06	1.63E-03	0.842	0.906	107.6
9	5.98E-06	1.57E-03	0.803	0.872	108.7
12	5.79E-06	1.50E-03	0.777	0.833	107.2
15	5.57E-06	1.44E-03	0.748	0.800	107.0
18	5.39E-06	1.37E-03	0.723	0.761	105.2

Table 26:

MCNP Simulation for Cs-137 Gamma Source			(Normalized Efficiency Values)	
Absorber Thickness (mm)	Absorber Thickness (mm)	Efficiency(%) Aluminium	Efficiency(%) Lead	Efficiency(%) Copper
0	0	100.00	100.00	100.00
3	6	98.50	99.00	98.30
6	12	96.60	98.30	96.70
9	18	96.00	97.80	96.30
12	24	93.90	97.80	93.60
15	30	93.80	96.80	93.90
18	36	90.7	96.40	91.70

Table 27:

MCNP Simulation for Na-1275 Gamma Source			(Normalized Efficiency Values)	
Absorber Thickness (mm)	Absorber Thickness (mm)	Efficiency(%) Aluminium	Efficiency(%) Lead	Efficiency(%) Copper
0	0	100.0	100.0	100.0
3	6	105.8	108.7	107.1
6	12	105.6	107.6	106.4
9	18	102.7	108.7	104.5
12	24	101.5	107.2	104.7
15	30	103.5	107.0	104.4
18	36	103.4	105.2	107.2

Table 28:

MCNP Simulation for Cs-137 Gamma Source							
Absorber Thickness (mm)	Absorber Thickness (mm)	Detected Aluminium	Detected Lead	Detected Copper	Normalized Aluminium	Normalized Lead	Normalized Copper
0	0	3.59E-03	3.59E-03	3.59E-03	1.00	1.00	1.00
3	6	3.18E-03	3.30E-03	3.19E-03	0.89	0.92	0.89
6	12	2.82E-03	3.04E-03	2.84E-03	0.79	0.85	0.79
9	18	2.49E-03	2.79E-03	2.53E-03	0.69	0.78	0.70
12	24	2.21E-03	2.57E-03	2.25E-03	0.62	0.72	0.63
15	30	1.95E-03	2.36E-03	2.00E-03	0.54	0.66	0.56
18	36	1.73E-03	2.17E-03	1.78E-03	0.48	0.60	0.50

Table 29:

MCNP Simulation for Na-1275 Gamma Source							
Absorber Thickness (mm)	Absorber Thickness (mm)	Detected Aluminium	Detected Lead	Detected Copper	Normalized Aluminium	Normalized Lead	Normalized Copper
0	0	1.80E-03	1.80E-03	1.80E-03	1.00	1.00	1.00
3	6	1.65E-03	1.72E-03	1.62E-03	0.92	0.96	0.90
6	12	1.51E-03	1.63E-03	1.52E-03	0.84	0.91	0.84
9	18	1.38E-03	1.57E-03	1.40E-03	0.77	0.87	0.78
12	24	1.26E-03	1.50E-03	1.29E-03	0.70	0.83	0.72
15	30	1.16E-03	1.44E-03	1.18E-03	0.64	0.80	0.66
18	36	1.06E-03	1.37E-03	1.07E-03	0.59	0.76	0.59