

Neutron and gamma ray fluences measurement at radial Beam Port 1 of TRIGA MARK II PUSPATI research reactor

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Abstract. Neutron and gamma ray fluences are the important criteria to be looked at in a neutron diffractometer system facility design. The fluence is defined as the number of neutrons and gamma rays travel through a unit area. Currently, the facility for neutron diffractometer system are planned to be developed at radial beam port 1 of TRIGA MARK II PUSPATI research reactor (RTP). The aim of this research is to determine the value of neutron and gamma ray fluences produced at the end of radial beam port 1 and to identify the shielding materials suitable for neutron and gamma ray. In order to achieve this aim, an experiment has been designed to obtain the neutron and gamma dose rates by using TLD-600 and TLD-700. The results from this experiments are converted into neutron and gamma ray fluences and are then compared with the results from simulation. The comparison shows that both results meet an agreement on the feasibility of shielding material for neutron diffractometer system. Our research results may be of help in the design of shielding material for neutron diffractometer facility at RTP.

1. Introduction

Radiation shielding is a physical barrier that is able to lower radiation dose rate by reflecting or absorbing radiation particles when dealing with ionizing radiation such as neutrons, gamma, and other charge particles. Shielding plays a vital role in all radiation practices especially in nuclear applications. In neutron diffractometer system, shielding is one of the essential instruments that needs to be emphasized. Aside from being used as biological shielding for human, the shielding is also important to protect sensitive instruments from radiation [1]. There are several considerations need to be taken when choosing the suitable materials for shielding such as strength of shielding, radiation particle attenuation factor, damage resistance, mechanical properties and cost. Generally, high density materials are more practical to shield gamma radiations compare to their low density counterparts. The thickness of the shielding materials can also influence the efficiency of the shielding materials. As for neutron radiation, materials with low atomic number such as water and hydrogen based materials are known to be more effective. However, one significant drawback of low atomic number of materials is



the formation of gamma radiation upon their interactions with neutron. The common materials which have been used as radiation shielding materials are lead, boron, paraffin, cadmium, polyethylene and concrete [2][3][4].

In this paper, the performance of the shielding materials has been tested in order to identify most suitable material to shield ionizing radiations and to determine the fluences obtained at radial beam port 1 of RTP. An experimental investigation has been performed for neutron and gamma radiations emitted from radial beam port to identify the shielding properties of each materials. The neutron and gamma dose rate was obtained by using Thermoluminescence Detector (TLD). Monte Carlo N-Particle (MCNP) simulation also has been conducted for the same shielding arrangement from source of $^{241}\text{Am}/\text{Be}$. Both results from experiment and simulation were compared and thus obtained the best suitable shielding materials to be used for neutron diffractometer system applications.

2. Methodology

2.1. Materials

Total of eight materials have been chosen such as boron block, lead sheet, pure polyethylene block, ordinary concrete block, aluminium plate, cadmium sheet, steel slab, and pure paraffin block, to be irradiated. The thickness of all samples are set to be constant. Table 1 shows the details of shielding materials used in this experiment.

Table 1. Characteristics of shielding materials.

No	Material	Thickness, cm	Molecular weight, g/mol	Density, g/cm ³
1	Boron	5	10.81	2.50
2	Lead	5	207.2	11.4
3	Polyethylene	5	84.16	0.94
4	Concrete (Ordinary)	5	13.92	2.30
5	Aluminium	5	26.98	2.70
6	Cadmium	5	112.41	8.70
7	Steel	5	55.84	8.05
8	Paraffin	5	226.44	0.90

2.2. Experimental arrangement

The experiment was performed at the TRIGA MARK II PUSPATI research reactor (RTP), Malaysian Nuclear Agency (MNA), Kajang, Malaysia. Among four beam ports available, radial beam port 1 was chosen because the neutron diffractometer system was proposed to be developed there [6]. The experimental arrangement consist of neutron and gamma source directly from reactor core with the total flux intensity of $5.9 \times 10^{10} \text{cm}^{-2} \text{s}^{-1}$ at maximum power of 750 kW [7], two type of detectors (TLD-600 and TLD-700), and shielding materials. According to simulation by Farhi in 2015, $2.0 \times 10^8 \text{cm}^{-2} \text{s}^{-1}$ flux yielded from the end of the beam port 1 [6]. Figure 1 shows the setup of the experiment. The sample material was placed inside of a shielding box at a distance of 30cm away from the end of beam tube. Then, a detector was located in front of the box at distance of 5cm away from sample. The TLD-600 was used to measure dose rate of neutron and gamma ray while TLD-700 to measure gamma particles only. The detectors were found to be the most suitable method to determine the neutron and gamma ray dose rate [8].

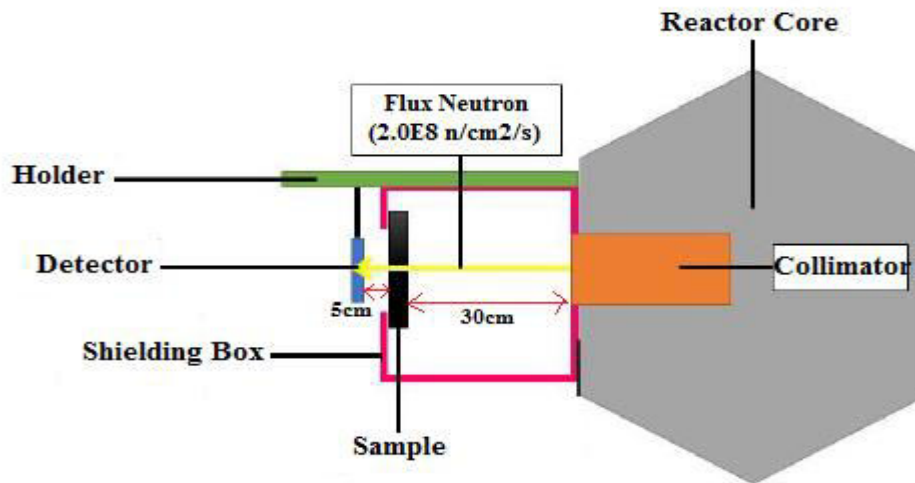


Figure 1. Experimental setup

Initially, the power of reactor is set to 15 kW. The reason of choosing low thermal power is because of the safety issue where insufficient shielding blocks were available around the experimental site. Then, TLD detectors from both types are placed in front of the shielding box without any sample for a minute in order to measure background radiation dose emitted from radial beam port 1 of RTP. Next, first sample of experiment is put inside the shielding box. The reactor need to be scrammed before the following step as for radiation precaution. The steps was followed by placing the TLDs 5cm away from sample for a minute to measure radiation dose rate produced after the sample. The experimental procedures were repeated for the other 7 samples.

Radiation dose rate of neutron and gamma measured from the TLD-600 and TLD-700 was analyzed at Secondary Standard Dosimetry Laboratory (SSDL). The neutron and gamma radiation dose was calculated using the following equation [10]:

$$H_p(10)_n = (Q4(nC) - Q3(nC)) \times CF_n \left(\frac{mSv}{nC} \right) \quad (2.1)$$

$$H_p(10)_\gamma = (Q2(nC)) \times CF_\gamma \left(\frac{mSv}{nC} \right) \quad (2.2)$$

where equation 2.1 is used to calculate neutron dose rate and equation 2.2 to calculate gamma dose rate respectively. The neutron and gamma dose rate obtained are then converted into fluence using the following formula [5]:

$$P = \frac{H}{\phi} \quad (2.3)$$

where P is dose equivalent per fluence ($9.1pSvcm^2$), H is radiation quantity dose equivalent in Sv, and ϕ is neutron fluence in cm^{-2} .

2.3. Monte Carlo simulation code MCNP

The results from the experiment were compared with the simulation results of the general purpose MCNP code MCNPX. The MCNPX code is a Fortran90 radiation transport code developed in 1994 that can transport particles with various energy ranges. Basically, the interaction of neutrons and photons with matter can be described through statistical means which best simulated using Monte Carlo method. This method is a computational algorithm that can provide approximate solutions to a variety of nuclear by the simulation of random quantities [9].

3. Results and discussion

3.1. Neutron and gamma dose rate measurement.

The result from the calculations was then interpolated to the power of 750 kW and shown in table 2.

Table 2. Neutron and gamma dose rate.

Material	Thickness, cm	Hp(10) η , mSv	Hp(10) γ , mSv
Background	5	19.3×10^{-1}	6.88×10^{-1}
Boron	5	3.36×10^{-5}	14.2×10^{-1}
Lead	5	9.46×10^{-1}	3.43×10^{-1}
Polyethylene	5	12.8×10^{-1}	9.83×10^{-1}
Concrete	5	15.4×10^{-1}	10.4×10^{-1}
Aluminium	5	18.9×10^{-1}	11.8×10^{-1}
Cadmium	5	4.83×10^{-5}	22.8×10^{-1}
Steel	5	2.70×10^{-2}	11.9×10^{-1}
Paraffin	5	15.3×10^{-1}	8.40×10^{-1}

3.2. Neutron and gamma fluence measurement

Fluence is defined as the number of radiation particles that intersect at a unit area of material. The unit of fluence is cm^{-2} . Based on the neutron and gamma dose rate obtained from TLDs, fluence can be calculated by using the equation 2.3 as stated before. The fluence of neutron and gamma measured in this experiment are tabulated and plotted in table 3 and figure 2.

Table 3. Neutron and gamma fluence

Material	Thickness, cm	Absorption Cross Section, σ_a	Fluence, cm^{-2}	
			Neutron	Gamma ray
Background	5	-	2.12×10^8	7.56×10^7
Boron	5	767	3.69×10^3	1.56×10^8
Lead	5	0.171	1.04×10^8	3.77×10^7
Polyethylene	5	0.336	1.41×10^8	1.08×10^8
Concrete	5	6.4	1.69×10^8	1.14×10^8
Aluminium	5	0.231	2.08×10^8	1.30×10^8
Cadmium	5	2520	5.30×10^3	2.51×10^8
Steel	5	2.56	2.91×10^6	1.31×10^8
Paraffin	5	0.337	1.68×10^8	9.23×10^7

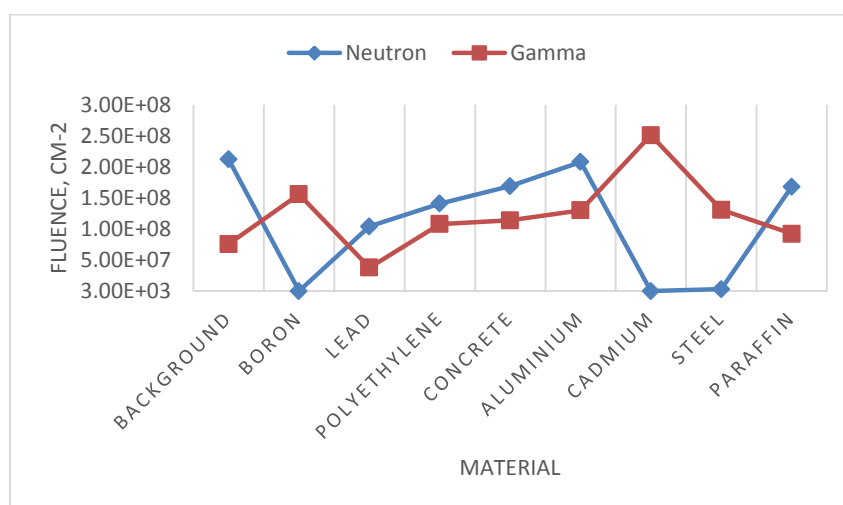


Figure 2. Neutron and gamma fluence

Based on the experimental results, average fluence measured at the end of the beam port 1 at 750 kW power is $1.44 \times 10^{10} \text{ cm}^{-2}$ which is in the acceptable range as reported in the previous study [6]. Specifically, neutron fluence and gamma fluence measured from the experiment at the end window of radial beam port 1 was $2.12 \times 10^8 \text{ ncm}^{-2}$ and $7.56 \times 10^7 \gamma \text{ cm}^{-2}$ respectively. The neutron fluence decreased after a shielding sample was placed in front of the beam port. Based on figure 2, the lowest neutron flux was obtained with boron sample block which is $3.69 \times 10^3 \text{ ncm}^{-2}$ followed by cadmium sheets, steel slabs, lead sheets, pure polyethylene block, paraffin block, ordinary concrete block and aluminium plates. Boron has significantly reduced the neutron fluence due to the absorption of neutron particle as boron is known to be effective in capturing neutron, especially thermal neutron as it has high neutron absorption cross section [11]. Boron (^{10}B) based element is one of the well-known materials to be used in nuclear applications because of its strong neutron absorption properties. Similarly, cadmium also has high neutron absorption cross section which enables it to absorb thermal and epithermal neutrons [11]. As for steel slabs, iron based materials are known to exhibit high neutron scattering cross section. The reduced neutron fluence was mainly attributed to the scattering of neutron particles by the materials. However, iron based materials is not commonly used to shield neutron as its can be easily activated upon interaction with ionizing radiations [12]. In case of pure polyethylene, paraffin and ordinary concrete, the neutron fluence was reduced owing to the presence of light hydrogen element in these materials which are very effective to slow down and stop the neutron particles [13]. Aluminium was found to be neutron transparent as the neutron fluence was only slightly reduced. The low neutron activation properties of aluminium has prompted the utilization of aluminium based materials for instrumentation inside the reactor core [14].

On the other hand, the increase in gamma fluence was observed when the beam passed through all the shielding samples except when lead was used, in which the fluence was reduced to $3.77 \times 10^7 \gamma \text{ cm}^{-2}$. The gamma fluence measured was in the following increasing order: lead sheets, paraffin block, pure polyethylene block, ordinary concrete block, aluminium plate, steel slabs, boron block and cadmium sheets. The increase in gamma fluence was mainly due to the interaction of these materials with neutron emitted secondary gamma radiation, particularly for boron and cadmium. Lead was a very common shielding material used to shield gamma radiation as it can reduce gamma particle. High gamma absorption cross section and high atomic number properties of lead enable it to be very effective in shielding gamma and x-ray radiations [15].

Results from experiment was compared with the results obtained from MCNPX code simulation as shows in table 4. Figure 3 shows the comparison of the neutron fluence measured and figure 4 compare the gamma fluence obtained. Most of the experiment results was measured a bit higher than the simulation. This was affected by the different incident fluence yielding from the reactor core and

the source used in simulation code. However, the results obtained agreed well as both shows the identical trend. The peak produced in Figure 4 at cadmium sample indicates that the cadmium emitted high energy of secondary gamma radiation compare to secondary gamma radiation emitted by boron. In addition, higher measurement of gamma fluence at cadmium during experiment compared to simulation was recorded because of the strong emission of secondary gamma radiations that been detected by detectors.

Table 4. Comparison between simulation and experimental results.

Material	Thicknes s, cm	Neutron			Gamma		
		Simulation ncm ⁻²	Experimental ncm ⁻²	Error %	Simulation γcm ⁻²	Experimental γcm ⁻²	Error %
Background	5	1.97x10 ⁸	2.12x10 ⁸	7	6.06x10 ⁷	7.56x10 ⁷	20
Boron	5	9.35x10 ²	3.69x10 ³	75	1.60x10 ⁸	1.56x10 ⁸	3
Lead	5	8.13x10 ⁷	1.04x10 ⁸	22	3.36x10 ⁷	3.77x10 ⁷	11
Polyethylene	5	1.13x10 ⁸	1.41x10 ⁸	20	7.48x10 ⁷	1.08x10 ⁸	31
Ordinary Concrete	5	1.28x10 ⁸	1.69x10 ⁸	24	9.49x10 ⁷	1.14x10 ⁸	17
Aluminium	5	1.39x10 ⁸	2.08x10 ⁸	33	1.12x10 ⁸	1.30x10 ⁸	14
Cadmium	5	1.54x10 ³	5.30x10 ³	71	1.38x10 ⁸	2.51x10 ⁸	45
Steel	5	1.26x10 ⁶	2.91x10 ⁶	57	1.28x10 ⁸	1.31x10 ⁸	2
Paraffin	5	1.14x10 ⁸	1.68x10 ⁸	32	7.51x10 ⁷	9.23x10 ⁷	19

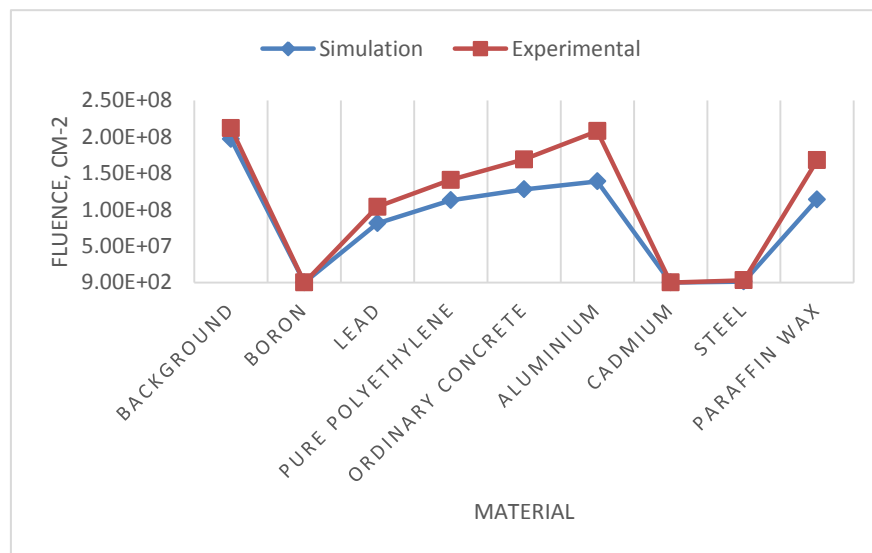


Figure 3. Comparison of neutron fluence between simulation and experiment.

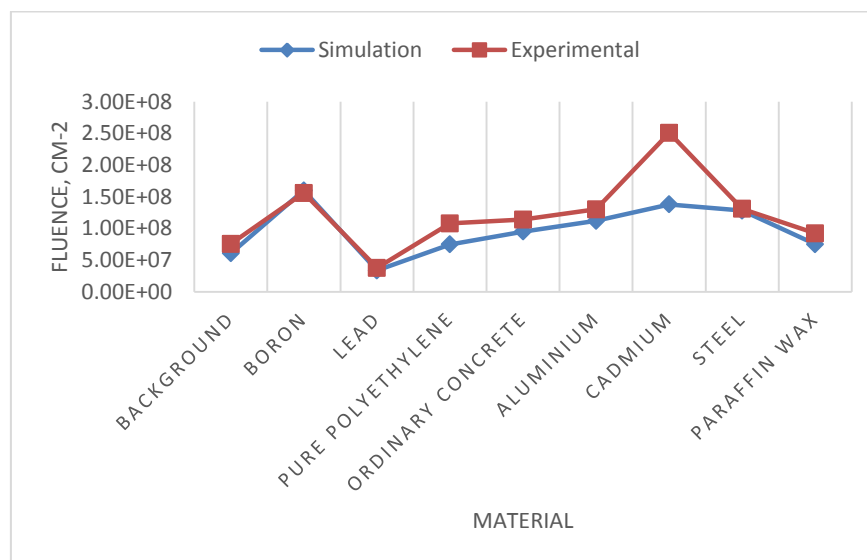


Figure 4. Comparison of gamma fluence between simulation and experiment.

4. Conclusion

Comparisons between simulation and experimental parameters were carried out in this paper. As result, the average fluence produced from the radial beam port 1 at maximum power of 750 kW is $\sim 10^8 \text{ cm}^{-2}$. Besides, the appropriate biological shielding materials for neutrons and gamma rays also has been identified. Boron shielding sample has been found to be the best material to shield neutron particles while lead shielding material is good for gamma radiation particles. Both results from simulation and experimental agreed well in term of trend with average error of 38% and 18% for neutron and gamma respectively.

5. References

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