DOI: 10.15669/pnst.7.91

Progress in Nuclear Science and Technology
Volume 7 (2025) pp. 91-96

ARTICLE

Measurement system requirements for photofission signal detection with coincidence neutron counting method

Kim Wei Chin [a], Hiroshi Sagara [a,*], Jun-ichi Hori [b] and Yoshiyuki Takahashi [b]

[a] *Dept. of Transdisciplinary Sci. and Eng., Tokyo Institute of Technology, N1-19 2-12-1 Ookayama, Meguro-ku, Tokyo 152-8550, Japan;* [b] *Institute for Integrated Radiation and Nuclear Science (KURNS), Kyoto University, 2 Asashiro-Nishi, Kumatori-cho, Sennan-gun, Osaka 590-0494, Japan*

Photofission signal detection based on neutron measurement encounters challenges posed by the background neutrons created from the bremsstrahlung converter target, photon-induced neutrons from target and surrounding materials, and accidental neutron injection to the detector. These background neutrons include direct injection into the detectors, scattering in the system and neutron-induced fission or photonuclear (γ, 2n) reactions from the surrounding materials and uranium target itself. The neutron measurement system requirement was assessed using coincidence neutron counting method by employing multiple neutron detectors. The photofission-induced neutron signal was evaluated as two order of magnitude higher than the background neutron-induced fission signal, and it was suggested photofission reactions could be quantitatively measured by assuming ideal bremsstrahlung beam collimators to avoid the direct injection of photons or neutrons to detectors. The impact analysis was performed on the measurement possibility with accidental injections of single neutrons coming from mainly background neutron collision in nuclear material, and background neutron reduction requirement and possible neutron absorber designing were finally deduced. The existence of the neutron absorber such as pure ^{10}B and High Density PolyEthylene (HDPE) with 10% of fully enriched ^{10}B with certain length of thicknesses have successfully reduced the accidental injection noise down to the measurement goal defined as 1% of the total doubles of photofission neutrons, also improved the distinguishability of photofission neutrons from the background neutrons by about four orders magnitude difference. Aside from that, the study of the possible neutron absorbers discovered the higher effectiveness of HDPE+10%^{10}B in reducing background neutrons with less thickness required to achieve the same measurement goal, but encounters difficulty in lessening the high energy neutrons, with less than 1/3 shielding of >1 MeV neutrons.

Keywords: ***photofission signal detection; non-destructive assay; coincidence neutron counting; Uranium detection***

1. Introduction

This work focuses on numerical simulation using Monte Carlo N-Particle (MCNP) on design of coincidence neutron counting system to detect photofission signal coming from uranium target consisting of ^{235}U and ^{238}U. The information of photofission by two different energy beam can lead to the different enrichment of these two isotopes or multi-nuclide of nuclear material using the Photofission Reaction Ratio (PFRR) methodology [1,2]. Photofission signal detection via neutron detection encounters difficulties of background by not only the bremsstrahlung converter target, the surrounding structural material and collimator which could contribute photo-neutrons as background. Another challenge is the separation of neutrons signal by photonuclear reactions from photofission reactions. As such, this paper proposes coincidence counting method for the neutron signal separation, and qualitative assessment on background reduction of neutrons. The coincidence counting method is a technique for separating neutron signal from photonuclear reactions and extracting information on photofission reactions from counts measured by neutron detectors. As two neutrons are detected simultaneously by the two neutron detectors in the defined time gate, this signal is evaluated as fission and other neutron multiple emission reactions. In principle, the coincidence counting method can be operated with at least two detectors, and there are some existing multiplicity counters with reliable detectability developed by Los Alamos National Laboratory [3]. Provided the beam travels in a point source shape, the neutrons counting by neutron detectors are mostly from photofission or photonuclear reactions. Nevertheless, the neutron beam typically has a huge cross-sectional area, and many neutrons will directly inject into the detectors or scattered around the measurement area, as illustrated by

*Corresponding author. E-mail: sagara.h.aa@jim.titech.ac.jp

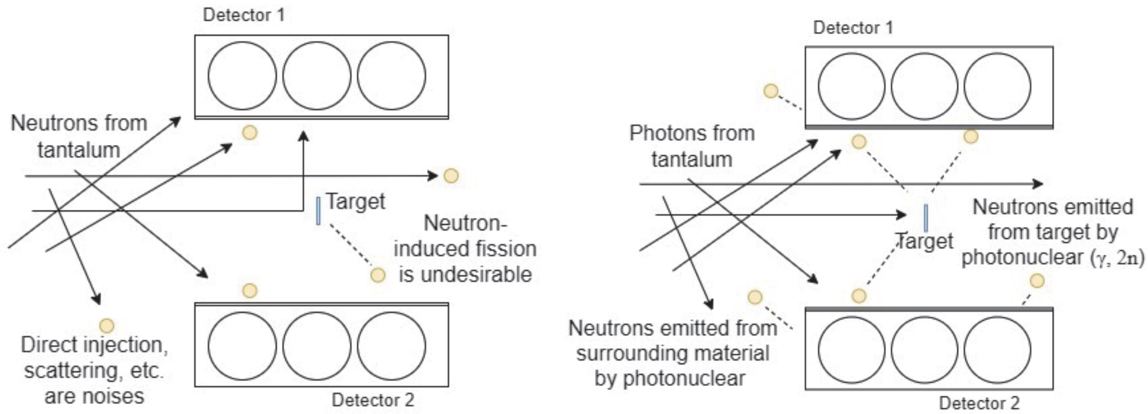

Figure 1. background neutrons coming from (a) tantalum directly, (b) photons created by tantalum.

Figure 1(a). Not mentioning the bremsstrahlung photons from tantalum, one of the bremsstrahlung beam generators which is considered in this study, induce multiple neutrons emission by photonuclear reaction of (γ, 2n), as can be visualized in **Figure 1(b)**. These are the possible sources of background affecting desirable neutrons counting only from photofission reaction.

2. Methodology

2.1. The coincidence neutron counting method and photofission model in MCNP

The coincidence counting method is dependent heavily on the variable neutron multiplicity. The number of neutrons emitted in spontaneous fission can vary from zero to eight [3]. The distribution of the number of neutrons is known as the multiplicity distribution. In practice, the analysis of multiplicity data is usually not based directly on the observed multiplicity distribution, but rather the factorial moments, where the first moment is the “singles”, or “totals”, the second is called the “doubles” or the “reals”, and the third is known as “triples” [3]. The design of coincidence detector in this chapter will be focused on achieving the goal of reducing background of accidental coincidences by the background neutron source down to below 1% of the total singles rate of photofission, and appropriate thickness of different combinations of moderator and shielding materials were investigated to achieve the goal. The time after trigger for the doubles rate was tallied with a 2μs interval, ranging from 0 up to 200μs.

The MCNP v.6.2 cloud, a Monte Carlo code coupled with the ENDF/B-VII.1 nuclear cross-section library, was used for the tantalum bremsstrahlung and photofission simulations [4]. The utilization of photofission model is available in the phys:p option, and it is provided in MCNP6 and subsequent versions for characterizing the photofission process. MCNP uses LLNL fission model to create prompt photofission gamma rays at low energy, and associated with photofission neutrons with suitable multiplicities. PHT code coupled with event generators create photons from fission fragments at greater energy [4].

The coincidence counting approach is used to quantify photofission secondaries that are only sampled when a photofission event happens. The bremsstrahlung photon flux was examined at different target distance from the tantalum photon source, from 0 up to 898.026 cm. Due to the lack of cross-sectional data as well as usable theoretical model for photofission, the photofission library used is mainly based on neutron-induced data [5]. According to this reference, we used the cross-section of the ^{235}U(n,f) reaction up to 2 MeV and the ^{238}U(n,f) reaction up to 9 MeV as references to validate their analysis procedure. We calculated the ^{238}U(n,f) cross-section and compared it to the recommended standard values in the neutron energy range. The results generally agreed with the ENDF evaluation, which gave confidence in the normalization factors used. However, the paper also notes that the ^{238}U(n,f) cross-sectional data obtained with the big diameter targets slightly overestimated the evaluated cross-sections for neutron energies in the energy range 0.6 – 4 MeV. In conclusion, while the photofission library based on neutron-induced data was used and validated against standard reactions, there were some discrepancies observed in certain energy ranges. This suggests that while the approach is generally valid, it may not be entirely accurate in all situations. The photonuclear data library used in this work is the ENDF/B-VII.0. The conventional neutron-induced fission model (PAR=*sf*) in MCNP for both ^{235}U and ^{238}U (spontaneous fission) is assumed to behave similarly Gaussian-distribution-wise as the photofission reaction, where the difference between them lies mainly on the number of neutrons generated on 0th polynomial order since the nucleus captures a neutron before fissioning in neutron-induced fission (Verbeke et al., 2016).

2.2. Bremsstrahlung beam creation and background neutron spectrum

To evaluate the feasibility of the photofission signal detection, a perfectly collimated beam from an electron linear accelerator is used to interrogate with the uranium target. To be as close to reality as possible in engineering manner, the beam from the Kyoto University electron linear accelerator facility which is known as KURRI

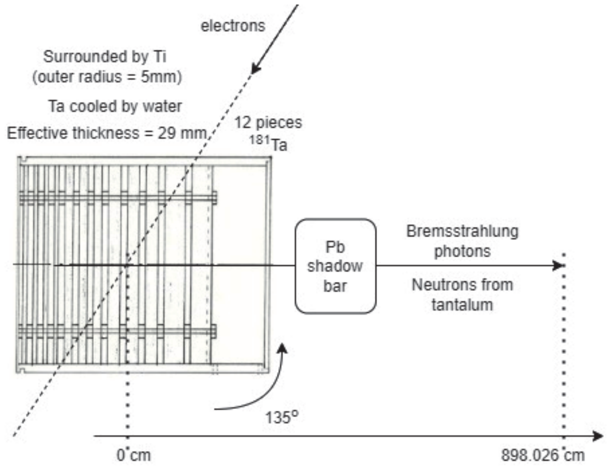

Figure 2. bremsstrahlung photons and background neutrons generation from tantalum.

LINAC is employed as the source. KURRI LINAC has a maximum operating condition of 46 MeV and 130μA. The tantalum converter target in KURRI LINAC as illustrated in **Figure 2**. has an effective thickness of 29 mm, 50 mm effective diameter and 60 mm diameter surrounded by titanium. There is a total of 12 pieces of tantalum plates, with each plate thickness ranges from 1 to 5 mm. The neutrons created from the tantalum target at few meters away from the target, approximately 898.026 cm are defined as the background neutron source, as can be seen in Figure 2. The incident electron energy of bremsstrahlung beam simulated is 18 MeV. The histories of particles given in the simulation work is 10^9.

2.3. Detection measurement system

The detector considered to measure the photofission signal using coincidence counting method needs to satisfy the requirement for neutron counting. Potential candidate of detector is the ^{3}He neutron proportional counter. Two main cases were simulated to detect photofission signal: background neutron source and the photofission mode of uranium target. For the background neutrons coming from the tantalum converter, the bremsstrahlung creation by the tantalum was pre-simulated with 18 MeV electrons and at the location of 898.026 cm away from tantalum the neutron flux is recorded as can be seen in **Figure 3**. For the photofission mode, a 1 mm thin natural uranium undergoing photofission passively is tallied as shown in **Figure 4**.

The uranium target in this model has a thickness of 1 mm and a height of 1 cm. There are two perfect absorbers on top and below of the background neutron beam assuming to be perfect collimator surrounding the beam with a small 2 cm gap. At x = -80, a point source of neutrons travels along the x-axis in the mono-direction and impinges on the target of interest at x=20, either inducing fission reaction, (n, xn) reaction, neutron capture or simply scattered away. This work focuses more on fission and (n, xn) reaction instead of neutron capture since the latter is more likely occur with slow or thermal neutrons while the neutrons in this study are mostly fast neutrons. The neutrons, prior to reaching ^{3}He counter, passed through a cadmium lining. This Cd lining is very thin, approximately 0.5 mm, and it can effectively protect the detector system from avalanche of thermal neutrons. While the high energy neutrons are moderated to thermal energy region by HDPE which is more sensitive for the ^{3}He neutron detector. The ^{3}He neutron detector is conceptualized to be multiple counters blending homogenously into the HDPE. In practical applications, these ^{3}He tubes are typically arranged in a specific configuration in the HDPE matrix to optimize neutron detection efficiency. For simplicity of simulation, instead of explicitly modeling individual ^{3}He tubes, the ^{3}He detectors were conceptually treated as a homogeneous mixture within the HDPE. The detector system was designed such that a hole is opened behind the target because most of the scattered neutrons directly from the background neutron beam goes into 180º direction. Two sides of multiple detectors were arranged such that the uranium target is positioned near the front instead of in the middle as to avoid possible direct accidental injections from the neutron beam or any photoneutrons originated from the possible neutron absorber.

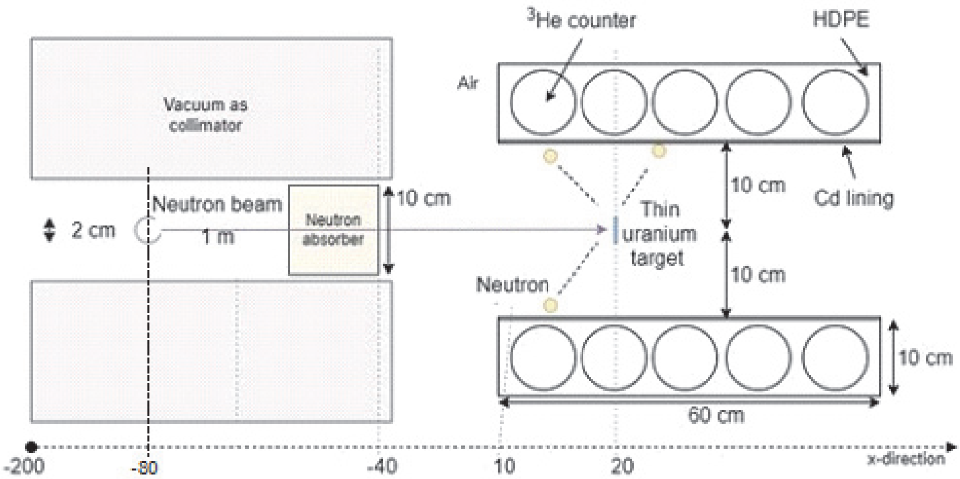

Figure 3. MCNP calculation model of background neutrons colliding on a 1mm thin uranium target.

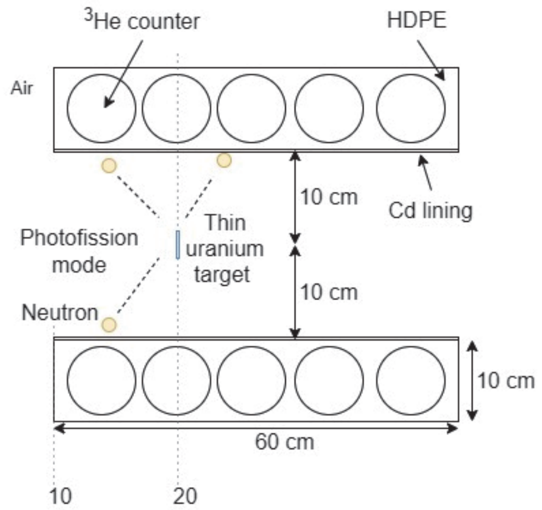

Figure 4. MCNP calculation model of photofission mode.

Candidates of the neutron absorber for study are pure ^{10}B and HDPE merged with 10% fully enriched ^{10}B homogenously. Increment of boron composition in the latter was also investigated. Parameter survey on the thickness of neutron absorber was executed to figure out adequate thickness to shield the incoming neutrons from tantalum, but at the same time would not induce too many photoneutrons on the neutron absorber material. ^{10}B is superb in moderating neutrons in thermal energy region through (n, α) reaction, however it is quite inefficient in moderating high-energy neutrons.

Aside from a perfect collimator, another imperative assumption is made here too. According to the simulation results by MCNP, there is approximately 29.37 photons and 1.3×10^{-3} neutrons created for each electron in the whole calculation system. According to the MCNP output of bombardment of the created photon flux by 18 MeV electron onto a thin natural uranium target, where out of these 29.37 photons, there is a probability of 1.16×10^{-5} for photofission to occur (loss to photofis). The assumption here made is that all 29.37 photons are directed to the target without escaping out the system.

In MCNP6.2, F8 tally can be used to record the neutron pulses in the ^{3}He neutron detectors. In conjunction with the FT8 tally card, the tally neutron coincidence capture tally (CAP) is used. Following the FT8 card is the definition of the trigger time, which was given as 2μs interval in this study. This pulse-height tally scoring is done at the end of each history, and the scoring is reasonably easy to describe in the absence of variance reduction.

3. Results

3.1. Feasibility of photofission signal detection with collimated beam by coincidence neutron counting method

The feasible study of photofission signal detection was carried out with a perfectly-collimated beam coupled with coincidence neutron counting method. Due to statistical capability, triples rate is often not enough to represent the data, thus only singles and doubles rate were studied in depth. There is a total of seven factorial moments for the multiple neutrons injection, and since the average neutron multiplicity is often between 2 and 3, the doubles rate solely can already satisfy the representation of multiple neutrons emission from the target. The singles rate for photofission of a thin natural uranium per source photon is in the order of 10^{-1} while the total doubles rate is 9×10^{-3}. Converting these rates to per source electron (multiplies 29.37) and per fission probability (multiplies 1.16×10^{-5}) earn the total singles rate of 3.82×10^{-5} and total doubles of 3.11×10^{-6}, respectively. On the other hand, the total singles rate for the background neutron source case per source neutron is around 5.21×10^{-3}, and total doubles rate as 1.3×10^{-5}. Converting them to per source electron (multiplies 1.3×10^{-3}) yield total singles of 6.75×10^{-6} and total doubles of 1.69×10^{-8}. On the same unit of per source electron, the doubles rate of both cases could be discerned by 2 orders magnitude difference. The emergent findings deduce that under idealized conditions, our simulation suggests a potential for discerning photofission signals with a 2-orders-of-magnitude difference compared to background signals. The result is represented by **Figure 5**.

3.2. Direct neutron noise reduction by neutron absorber

Apart from the background neutrons coming from the tantalum or surrounding materials, the accidental injection into the ^{3}He counters is an unignorable neutron background despite the assumed ideal conditions. 10% of the total singles by background neutrons is assumed to be the neutron accidental injections, and the goal of neutron background reduction is set to be the 1% of the total doubles by photofission mode. The 10% figure is an estimated value used for planning and analytical purposes. This estimate serves as a benchmark for the system's performance in reducing background noise, rather than a precise measurement. Based on this assumption, the current 10% of the total singles by background neutrons of accidental neutron noise is estimated around 20% of the goal ($0.1\times6.75\times10^{-6}$ / $0.01\times3.11\times10^{-6}$). The accidental counting and the goal of the background neutron reduction can be visualised in **Figure 6**. In accordance with the goal of the background neutron reduction, different material and thicknesses of neutron absorber was explored and discussed here.

The possible neutron absorber material considered for study are pure ^{10}B plate and HDPE + 10%^{10}B (mixed homogenously) for their excellent performance in moderating neutrons especially in thermal energy region. The neutrons created from tantalum peak at around 1 MeV and its effective spectrum spreads across energies from 0 up to 4 or 5 MeV. The observed discontinuity in the neutron energy spectrum below 0.1 MeV is attributable to the specified tally energy range, which does not encompass this particular energy interval. Pure ^{10}B plate has the advantage over HDPE + 10%^{10}B of not emitting photoneutrons as addition to the background neutrons, nonetheless ^{10}B has a very low neutron absorption cross section of around

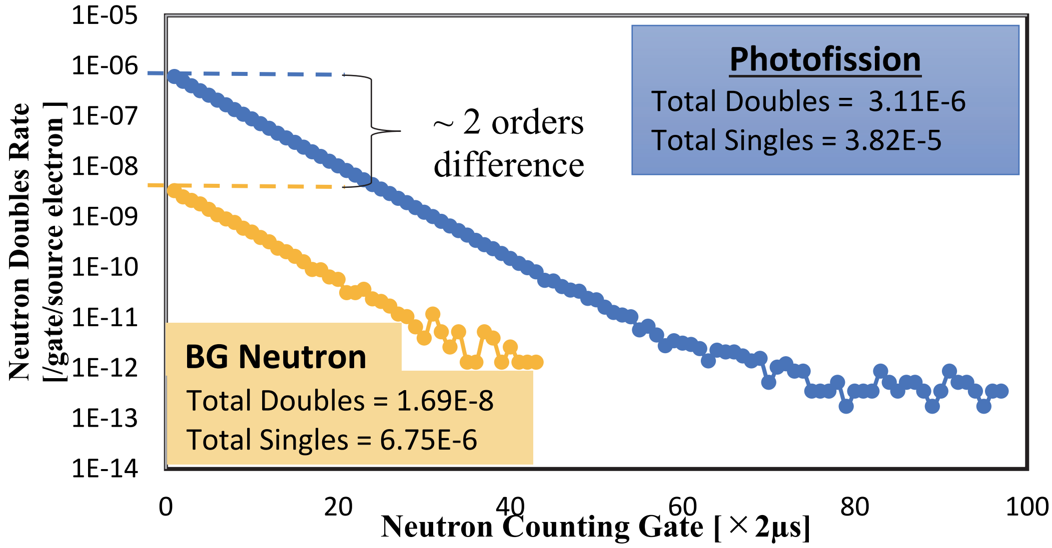

Figure 5. total doubles by photofission and background neutron source with Singles included.

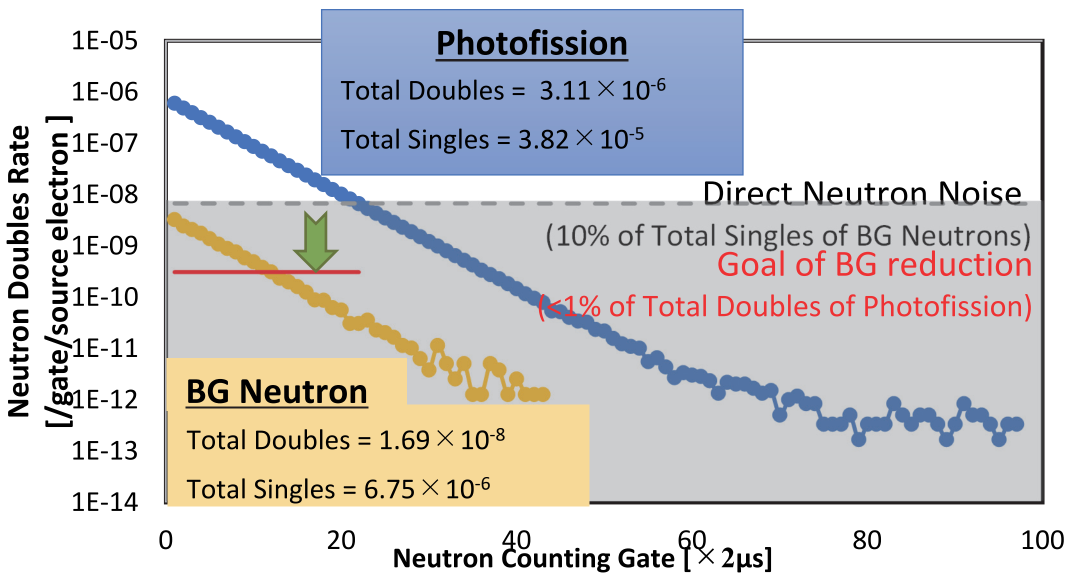

Figure 6. accidental neutron injection noise reduction goal.

2×10^{-1} at 1 MeV which is the peak energy of neutrons coming from tantalum. In this sense of comparison, HDPE + 10%^{10}B is more effective due to its hydrogen content for neutron moderation. The main criteria in selection of the appropriate thickness of each material is the parameter of total singles, which 10% of it is considered to be accidental neutron noise. The neutron flux and energy after the beam passed through the neutron absorber were tallied too. A 3 cm thick box after the neutron absorber with the following dimensions: x = 10.1 ~ 13.1; y = -5 ~ 5; z = -5 ~ 5 was tallied for the neutron flux and neutron surface (surface on x=10.1). The neutron moderation outcome by the different thickness of ^{10}B and HDPE + 10%^{10}B is apparent in Fig. 7, where 5 cm of HDPE + 10%^{10}B as well as 15 cm of pure ^{10}B plate are adequate to meet the goal of requirement in the numerical environment with perfect collimated beam. The results suggest the better capability of HDPE + 10%^{10}B than pure ^{10}B plate in weakening the neutrons background coming directly from tantalum. The reduction degree of neutrons spectrum in different energy regions by HDPE + 10%^{10}B is tabulated in **Table 1**. This material works incredibly for thermal and epi-thermal neutrons, however fast neutrons with energy above 1 MeV remain strong. This suggests the requirement consideration of longer thickness for engineering applicability since the fast neutrons are quite large portion of the whole flux, about 24.2% dominance. Also note the singles used in this simulation is per induced electron to tantalum. To calculate the real accidental coincidence rate, the facility beam current and ^{3}He tube gate width are required:

Table 1. Examination of reduction effect of HDPE+10%^{10}B on neutron flux ratio with different energy region.

Neutron energy region	Neutron flux ratio without absorber	Neutron flux ratio with 5 cm HDPE+10%^{10}B	*Degree of reduction, R*
Thermal to Intermediate (0 eV ~ 0.01MeV)	0.149	0.00035	R=0.149/0.0035=42.57
Epi-thermal (0.01MeV ~ 1 MeV)	0.609	0.027	R=0.609/0.027=22.56
Fast (> 1MeV)	0.242	0.973	R=0.242/0.973=0.249

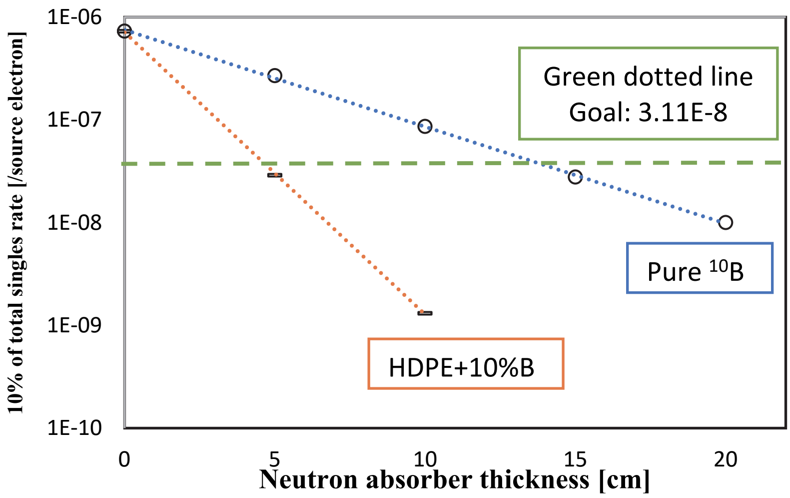

Figure 7. required thickness to reduce accidental neutron injection noise.

$$Accidental\ CR = \left\{\frac{Total\ events}{electrons\ number * Beam\ current * gate\ width}\right\}^2$$

where CR refers to Coincidence Rate without considering the time resolution of ^{3}He detectors, electrons number ($6.75\times10^{-6} + 3.82\times10^{-5}$), Beam current with the unit of electron/s while gate width in seconds. KURRI LINAC has max. current of 130 μA and typical gate width is few μs.

4. Conclusion

Photofission signal detection based on neutron measurement encounters challenges posed by the background neutrons created from the bremsstrahlung converter target, photon-induced neutrons from target and surrounding materials, and accidental neutron injection to the detector. The neutron measurement system requirement was assessed using coincidence neutron counting method by employing multiple neutron detectors. The photofission-induced neutron signal was evaluated as two-order of magnitude higher than the background neutron-induced fission signal, and it was suggested photofission reactions could be quantitatively measured by assuming ideal bremsstrahlung beam collimators to avoid the direct injection of photons or neutrons to detectors. The impact analysis was performed on the measurement possibility with accidental injections of single neutrons coming from mainly background neutron collision in nuclear material, and background neutron reduction requirement and possible neutron absorber designing were finally deduced.

References

[1] K.W. Chin, H. Sagara and C.Y. Han, Application of Photofission Reaction to Identify High-Enriched Uranium by Bremsstrahlung Photons, *Annals of Nuclear Energy* 158 (2021), 108295.

[2] K.W. Chin, H. Sagara, R. Kimura and K. Tanabe, On the Numerical Method for Photofission-Based Nuclear Material Isotopic Composition Estimation in Thorium-Uranium Systems, *Nuclear Science and Engineering* 196 (2022), pp. 852-872.

[3] N. Ensslin, M.S. Krick, D.G. Langner, M.M. Pickrell, T.D. Reilly and J.E. Stewart, Passive Neutron Multiplicity Counting, *Los Alamos National Laboratory* (2011), LA-UR-07-1402.

[4] OAK RIDGE NATIONAL LABORATORY RSICC COMPUTER CODE COLLECTION, MCNP6.2-EXE Monte Carlo N-Particle® Transport Code System Version 6.2 EXE-ONLY, *Oak Ridge National Laboratory* (2017), LA-UR-17-29981.

[5] J.M. Verbeke, C. Hagmann and D. Wright, Simulation of Neutron and Gamma Ray Emission from Fission and Photofission. LLNL Fission Library 2.0.2., *Lawrence Livermore National Security*, LLC.UCRL-AR-228518-REV-1.