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# Monte Carlo Simulation of an Active Neutron Counter for Fissile Material Accounting

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## 1. Introduction

Passive neutron coincidence counters have been developed for the measurement of special nuclear materials by the Korea Atomic Energy Research Institute (KAERI) [1, 2]. Those passive-mode counters are based on spontaneous fission from plutonium or curium in special nuclear materials. Therefore, uranium and other fissile materials can not be assayed by a passive mode because of its very low spontaneous fission yield. An active neutron counting method is one of the possible ways to measure fissile material, in which a neutron interrogation source is adapted for induced fission. Passive neutron counter could be used in an active-mode with some appropriate modifications and interrogation sources [3]. Preliminary research had been performed for an active-mode operation of a DUPIC safeguards neutron counter, which was developed as a passive counter, using a cadmium shutter and total neutron counting [4].

In this paper, MCNP simulation result for active neutron coincidence counting has been described and discussed. The result could be applied to determine the possibility and necessary modification for an active-mode operation of a developed neutron counter.

## 2. Methods and Results

In this section, a model of a neutron counter and a sample material used in the simulation are described. The simulation results are discussed for several important aspects with regard to an active-mode operation.

### *2.1 Description of the Simulation*

The design parameter of the ACP Safeguards Neutron Counter [2] was used in all the simulations in this work. The counter consists of 24 $^{3}$He tubes and appropriate moderating and shielding parts. A top reflector of graphite was removed from the original structure and replaced by an interrogation neutron source. A general neutron source for active neutron counting is an AmLi source (0.3-0.4 MeV of mean energy). In this study, however, relatively high energy sources were considered for a future experimental plan using a neutron generator. The source has $10^{6}$ n/s of intensity in isotropic direction and 2.5 MeV of neutron energy. This high-energy interrogation source means the counter needs a polyethylene moderator (refer to the 'source-moderator' in this paper) between the interrogation source and an assayed sample to slow down the source neutrons for a better absorption. Assayed samples were 1 kg of uranium metal disks of 5 cm-diameter with 0 to 20% enrichments of $^{235}$U. Other isotopes except for $^{238}$U and $^{235}$U were not considered because of their less effects and a simplification of the result. The sample was located at the center of the cavity and the interrogation source was positioned 25 cm above the sample.

### *2.2 Detection Efficiency and Die-Away Time*

Detection efficiency and a die-away time of a counting system depend on several factors, for example the neutron energy, position of the source and additional moderators or reflectors in the counter. Related to our purpose, interrogation neutrons and induced neutrons have a difference probability to be detected in a system mainly because of their position. Detection efficiency for interrogation neutrons emitted from the top cavity of the counter varies from 10% to 12% depending on the thickness of the source-moderator (7 to 0 cm). Efficiency for induced neutrons, which is a signature of fissile contents, is 20 to 23% depending on the top-moderator and enrichment of the sample. This higher efficiency than for the interrogation neutrons is because the counter is designed originally for a maximum efficiency for fission neutrons from the sample. Lower efficiency for interrogation neutrons is better for active neutron counting. The die-away time is 85-90 μs with the thickness of the source-moderator. It corresponds to 0.47- 0.50 of double gate fraction in the case of 4.5 μs of pre-delay and 64 μs of gate width in coincidence electronics.

### *2.3 Fission Neutrons without Interrogation Source*

Spontaneous fission of uranium yields neutrons and these neutrons induce further fission in the sample. The number of fission neutron from 1 kg of $^{238}$U is 13.6 n/s by spontaneous fission and less than 1 n/s by induced fission. Increasing the $^{235}$U ratio in the sample rapidly reduces the spontaneous fission rate and slightly promotes induced fission. Therefore the number of fission neutron without an interrogation source is always less than 15 n/s for the 1 kg-sample with any enrichment in this study. This value corresponds to 3 counts/sec in total counting and less than 1 event/sec in coincidence counting by the neutron coincidence counter.

### *2.4 Induced Fission and Coincidence Count Rate*

Using an interrogation neutron source, induced fission increases and a large number of fission neutrons can serve as a signature of fissile contents. The neutron energy used in the simulation is 2.5 MeV and the resultant values are conversed for $10^6$ n/s of interrogation intensity. Figure 1 shows the induced fission rate for the 1 kg uranium sample with an enrichment of 0 to 20%. Even for the 1% enriched sample, more than 1,350 fission/sec could be possible. Average fission multiplicity is 2.45, which means almost 3,300 neutrons are emitted per second from the 1% enriched sample. With 4.5 μs of pre-delay and 64 μs of gate width, the coincidence count rate is about 7% of the induced fission rate. Rate increment of the fission slows down with a higher enrichment.

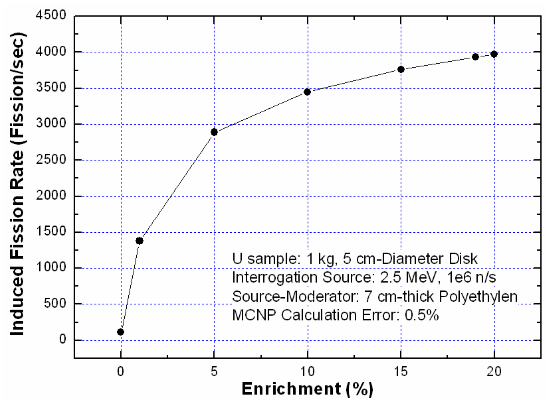


Fig. 1. Induced fission rate as a function of the enrichment. Distance from the interrogation source is 3.8 cm to the top of the source-moderator and 25 cm to the top of the sample.

### *2.5 Effect of Moderation of Interrogation Source*

Because of the high energy of interrogation neutrons, the induced fission rate is affected by the degree of moderation by circumstances. To evaluate this effect, a set of simulations with varying a thickness and location of a source-moderator was performed. As shown in Figure 2, the 7 cm-thick source-moderator is the best for coincidence counting even though the highest induced fission rate was obtained for the 8 cm-thick one. It could be explained by a greater moderation and a longer die-away time of the fission neutrons by the thicker source-moderator. For the location of the source-moderator, closer to the interrogation source provides more fission and coincidence counts as shown in Figure 3. A larger solid angle with a closer moderator produces more moderated interrogation neutrons that enter to a sample cavity.

## 3. Conclusions

MCNP simulation has been used to evaluate active neutron counting to measure fissile uranium contents in a sample. Even with a high-energy interrogation source and a low enriched uranium sample, more than 100 coincidence events/sec can be counted by the neutron counter with 20% of detection efficiency. The efforts to optimize the source-moderator geometry can enhance the performance. In a real situation, accidental coincident events by a high intensity of the interrogation source may affect real coincidence counting. Effects by additional components to shield interrogation neutrons will be evaluated as a further work followed by an experimental verification.

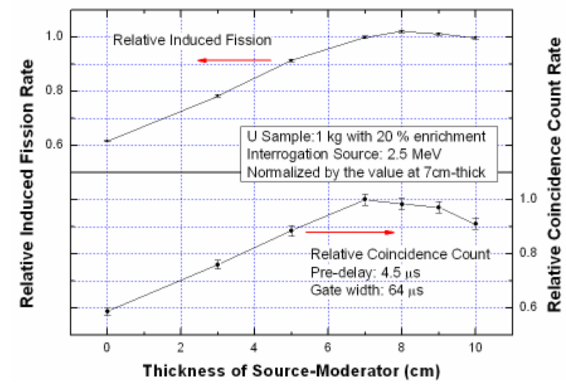


Fig. 2. Relative induced fission rate and coincidence count rate as a function of the thickness of the source-moderator. As the thicknesses are changing, the distance from the bottom of the moderator to the top of the sample was fixed as 13 cm.

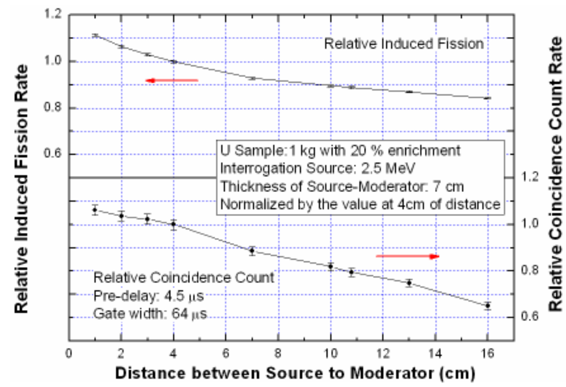


Fig. 3. Relative induced fission rate and coincidence count rate as a function of the distance from the interrogation source to the source-moderator. Source-to-sample distance was fixed as 25 cm.

## REFERENCES

[1] Y. G. Lee, Development of DUPIC Safeguards Neutron Counter, Korea Atomic Energy Research Institute Report KAERI-TR-1381-99, 1999.

[2] T. H. Lee, H. D. Kim, K. J. Jung, S. W. Park, Development of a Neutron Coincidence Counter for the Advanced Spent Fuel Conditioning Process, J. Korean Phys. Soc., Vol. 48(2), p. 218, 2006.

[3] H. O. Menlove, Description and Operation Manual for the Active Well Coincidence Counter, Los Alamos Scientific Laboratory Report LA-7823-M, 1979.

[4] H. Y. Kang, H. R. Cha, Y. G. Lee, G. M. Koo, H. D. Kim, W. I. Ko, D. Y. Song, J. S. Hong, M. S. Yang, Active NDA Neutron Measurement Method to Determine Fissile Contents of Fuel Material, Korean Nucl. Soc. Autumn Meeting, Seoul, Korea, October 1999.