MCNP-POLIMI MODELING FOR THE ACTIVE WELL COINCIDENCE COUNTER

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ABSTRACT

This paper presents the use of the MCNP-PoliMi code and a specifically designed post-processor to simulate the measurements performed with the active well coincidence counter. The code simulates the operation of the shift register, and can be used to perform time interval analysis. The code is described and preliminary results are discussed.

Key Words: Monte Carlo, Neutron detection, He-3 counters, MCNP-PoliMi, Active well counters

1. INTRODUCTION

Active well coincidence counters (AWCCs) are widely used for nondestructive assay applications in nuclear safeguards and nuclear waste characterization [1]. The method relies on the detection of correlated neutrons from fission by He-3 detectors embedded in a polyethylene moderator. In the assay of uranium, an active measurement must be performed to induce fission in the material. Typically, Am/Li neutron sources are used as the active source.

Monte Carlo studies of the measurement setup are useful in the design, optimization, and analysis of the detection system. The simulation must take into account factors such as the Am/Li neutron spectrum, the multiplicity of neutron emission in induced fission events, and the detection of thermalized neutrons by the He-3 counters. The present study addresses these issues and presents a post-processing code, which is used to analyze the MCNP-PoliMi output. The post-processing option gives the user great flexibility on the desired output parameters.

2. MONTE CARLO APPROACH

The MCNP-PoliMi code [2] is a modification of the MCNP4c code. The main features of the code can be summarized as follows. The simulation of spontaneous and induced fission includes the correct multiplicity of prompt neutrons and gamma rays from individual fission events. In addition, the algorithm generating secondary gamma rays from neutron interactions has been changed substantially from the standard MCNP algorithm. Neutron interaction and secondary gamma ray production are now linked to provide a better representation of the physics of individual particle interactions. A number of spontaneous fission and alpha/n neutron and gamma ray sources have been added. The output format of MCNP-PoliMi is in the form of a collision output table, which allows for the simulation of detector response with more flexibility. Further details on the modifications are given in reference 2.

In the simulation of the active well coincidence counter, the most important feature of MCNP-PoliMi is the sampling of the correct multiplicity of neutrons in induced fission, and the simulation of the Am/Li neutron source spectrum.

An excerpt of the MCNP-PoliMi data output file is shown in Table 1. Each line in the output file corresponds to an interaction event occurring in the He-3 detectors. Because the capture cross-section is predominant, all events reported in Table 1 correspond to capture events in the detectors (see column labeled “interaction type”). The column labeled “Time” gives the time at which the capture occurred, following a source event from the Am/Li neutron source. The column labeled “Generation number” gives the generation number of the neutron that is being captured (successive generations originate by induced fission in fissile isotopes). The column labeled “Number of scatterings” gives the total number of scatterings in the neutron’s history preceding its capture. The column labeled “Energy” gives the energy of the neutron prior to being captured in the He-3 detector. The data output of the code must be post-processed using a specifically developed code, which is written in Matlab™. This code is described in Section 3.

Table 1. Excerpt from MCNP-PoliMi data output file.

History number	Particle number	Projectile type*	Interaction type^Δ	Target nucleus•	Cell number of collision event	Time (shakes)	Generation number	Number of scatterings	Energy (MeV)
1	6	1	0	2003	1	32401.6	3	221	8.15E-09
3	1	1	0	2003	1	18052.97	0	156	7.61E-08
7	1	1	0	2003	1	11250.02	0	107	3.34E-08
9	4	1	0	2003	1	1894.416	1	17	5.72E-06
10	1	1	0	2003	1	5845.552	0	62	9.25E-08
16	3	1	0	2003	1	4388.813	1	73	4.58E-08
18	7	1	0	2003	1	19023.05	2	153	8.27E-08
18	4	1	0	2003	1	8990.743	2	81	4.27E-08
32	1	1	0	2003	1	7732.952	0	67	1.31E-08
35	1	1	0	2003	1	6319.3	0	49	1.53E-08
...									

* 1 = neutron
Δ 0 = absorption
• 2003=helium-3

The active well coincidence counter (AWCC) that was modeled in this study is in use at the Y-12 National Security Complex and was manufactured by JOMAR (model JOMAR 51). It consists of 42 He-3 tubes arranged in two concentric rings embedded in a polyethylene moderator. The sample cavity has a diameter of 22.9 cm. The He-3 tubes are placed in two rings having inner radius 30.8 cm and outer radius 38.1 cm. The tubes are operated at a pressure of 4 atm and their active length is 55.8 cm. Two Am/Li neutron sources are placed above and below the sample cavity. Figure 1 shows two views of the geometry of the MCNP simulation.

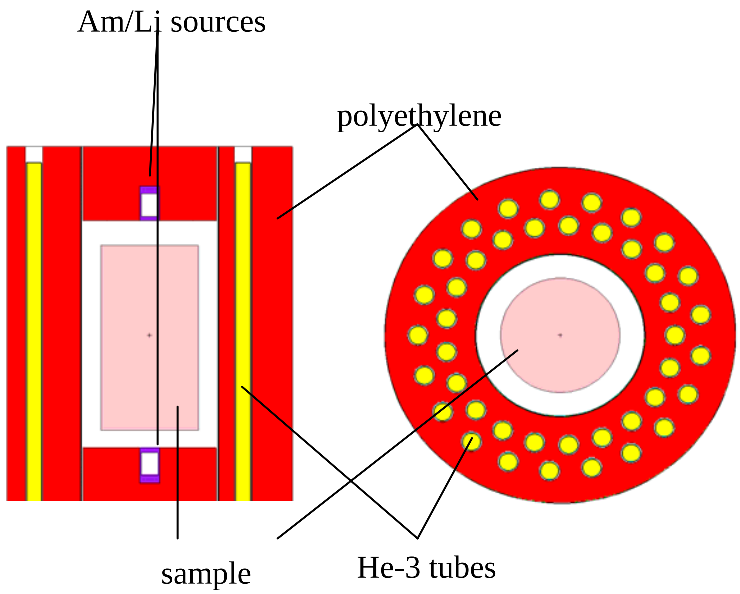


Figure 1. Geometry of MCNP simulation.

3. POSTPROCESSING CODE FOR THE AWCC

The information in the data output file was post-processed using a specifically developed post-processing code, written in Matlab™. Each history is initiated by an Am/Li source neutron, which is emitted at time zero. The detections in the He-3 counters are tallied and the following information is analyzed by the postprocessing code:

- Time of detection
- Generation number
- Energy of the neutron before capture
- Number of scattering events prior to capture

The post-processing code simulates the operation of the shift register to tally the number of captures in the counters. All counts occurring in the counters are placed together in a time-dependent sequence. This count sequence is analyzed by the shift register. Figure 2 shows a schematic diagram of the operation of the shift register. The first pulse to pass through the shift register triggers the pre-delay. The shift register tallies the pulses in the time window. The shift register's time window and pre-delay are adjustable input parameters in the code. In the case of Figure 2, four pulses would be tallied. The procedure is repeated as the pulses pass through the shift register. Because each Am/Li source neutrons are emitted independently and all the progeny is tracked before beginning the next history, the simulated multiplicity includes only real coincidences and no accidental coincidences. This corresponds to the measurement of the R gate in the AWCC.

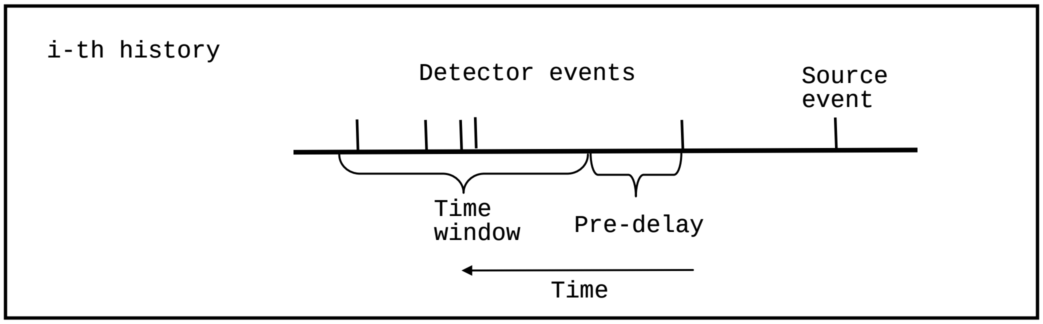
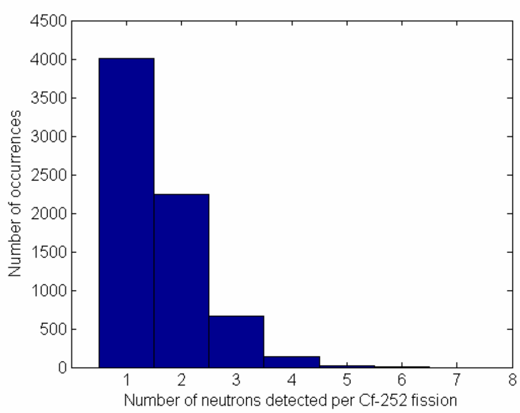
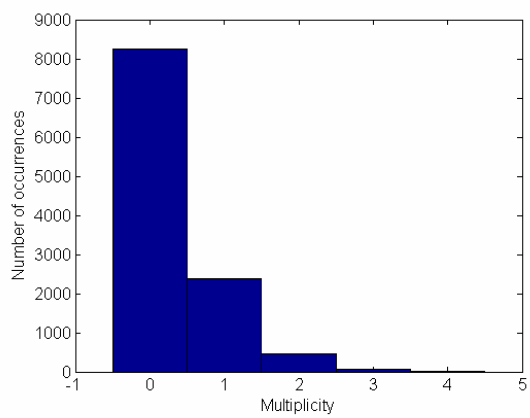


Figure 2. Illustration of the shift register operation.

4. RESULTS

The results obtained using the MCNP-PoliMi code and its post-processor are discussed in this section. The first case studied was a Cf-252 source placed at the center of the sample cavity. No fissile material was modeled. The number of counts per history recorded by the He-3 detectors is shown in Figure 3 (a). Figure 3 (b) shows the number of neutrons tallied by the shift register, assuming a time window of 64 microseconds and a pre-delay of 4 microseconds. The values for the time window and pre-delay can be varied in the post-processing code.

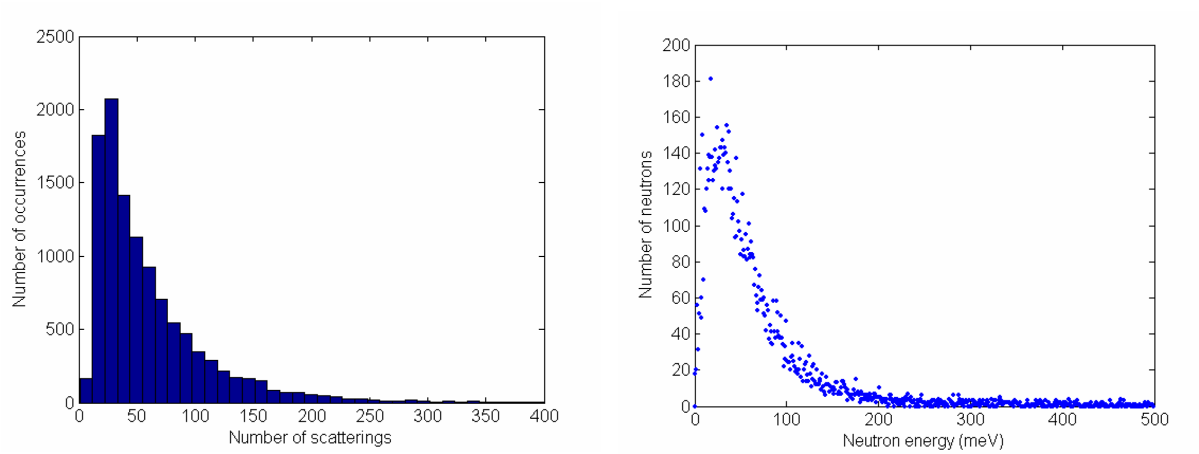




(a) (b)

Figure 3. Number of neutrons detected per Cf-252 fission in histories having at least one detection (a), and neutron multiplicity measured by shift register (b).

Figure 4 (a) shows the number of scatterings in the neutron history prior to capture. The energy of the neutrons before capture in the He-3 is given in Figure 4 (b).



(a) (b)

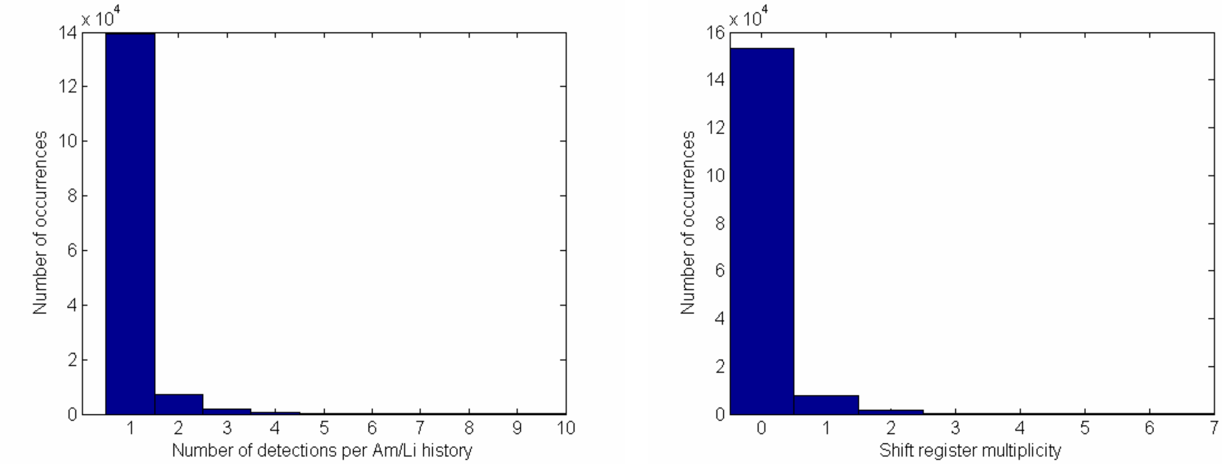
Figure 4. Number of scatterings in neutron's life before detection (a) and neutron energy before detection (b).

The detection efficiency of the AWCC can be determined by considering the number of neutrons from the Cf-252 source and the total number of detections in the He-3 counters. The efficiency is defined as

$$\varepsilon = \frac{n_{\text{detected}}}{n_{\text{emitted}}} \qquad (1)$$

where n_{detected} is the total number of neutrons detected and n_{emitted} is the total number of neutrons emitted by the Cf-252 source. The calculated efficiency for the AWCC in this study is 29%.

The second case studied was the simulation of a measurement performed on a sample of uranium metal having mass equal to 6 kg and enrichment equal to 92 wt% U-235. Am/Li neutron sources were modeled and placed in the positions indicated in Figure 1. Figure 5 shows the number of detections per history and the multiplicity recorded by the shift register for this case.



(a) (b)

Figure 5. Number of neutrons detected per Am/Li neutron in histories having at least one detection (a), and neutron multiplicity measured by shift register (b).

Figure 6 shows the number of neutrons detected as a function of their generation number. A generation number equal to 0 refers to a neutron from the Am/Li neutron source. A generation number equal to 1 refers to a neutron from the first generation of fissions induced in the uranium sample, a generation number equal to 2 to the second generation of fissions, and so on. It is interesting to note that only approximately 35% of the detections are given by induced fission neutrons. Most of the detections (~65%) are given by the Am/Li source neutrons. This type of analysis may be used to evaluate the capability of a given source to induce fission in the sample.

We calculated the efficiency of induced fission neutron detection. This efficiency is defined similarly to the one given in Equation 1, but it refers to induced fission neutrons only (i.e. neutrons with a generation number greater than 0). This efficiency turned out to be equal to 25.6%. This efficiency is lower by approximately 12% when compared to the efficiency of detecting neutrons from the bare Cf-252 source. This difference can be explained by considering the presence of the fissile material, which shields the induced fission neutrons, and possibly by considering the different energy spectra of the neutrons.

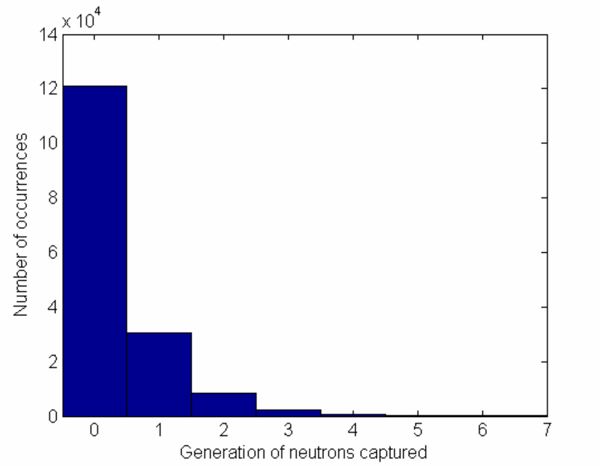


Figure 6. Generation of neutrons detected by the AWCC in the configuration with 6 kg uranium metal, 92% enriched and Am/Li neutron sources.

5. TIME INTERVAL ANALYSIS

Time interval analysis (TIA) was first proposed by Bruggeman and colleagues a few years ago [3]. The idea is to preserve the information of the time of detections by measuring time correlations among detectors and detector autocorrelations. This approach has the potential to lead to a more complete analysis compared to the measurement of multiplicity alone. One analysis is performed by measuring the time interval between pairs of detections in the AWCC. If the detections were random, the distribution of these time intervals would be exponential. If fissile material is present and fission chains are initiated, the distribution of time intervals is no longer random, and deviates from an exponential distribution. Figure 7 shows the number of detections as a function of time of detection for neutrons of generations 0 to 3. Generation 0 neutrons are neutrons from the Am/Li source, generation 1 neutrons are neutrons from the first generation of induced fission, and so on. Inspection of Figure 7 shows that the time constant for generation 0 neutrons is different from the time constant of induced fission neutrons (generations 1 to 3). The die away time of the different generation neutrons are given in Table 1.

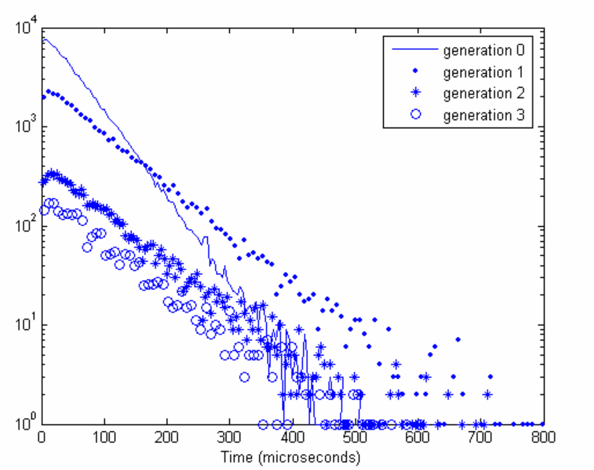
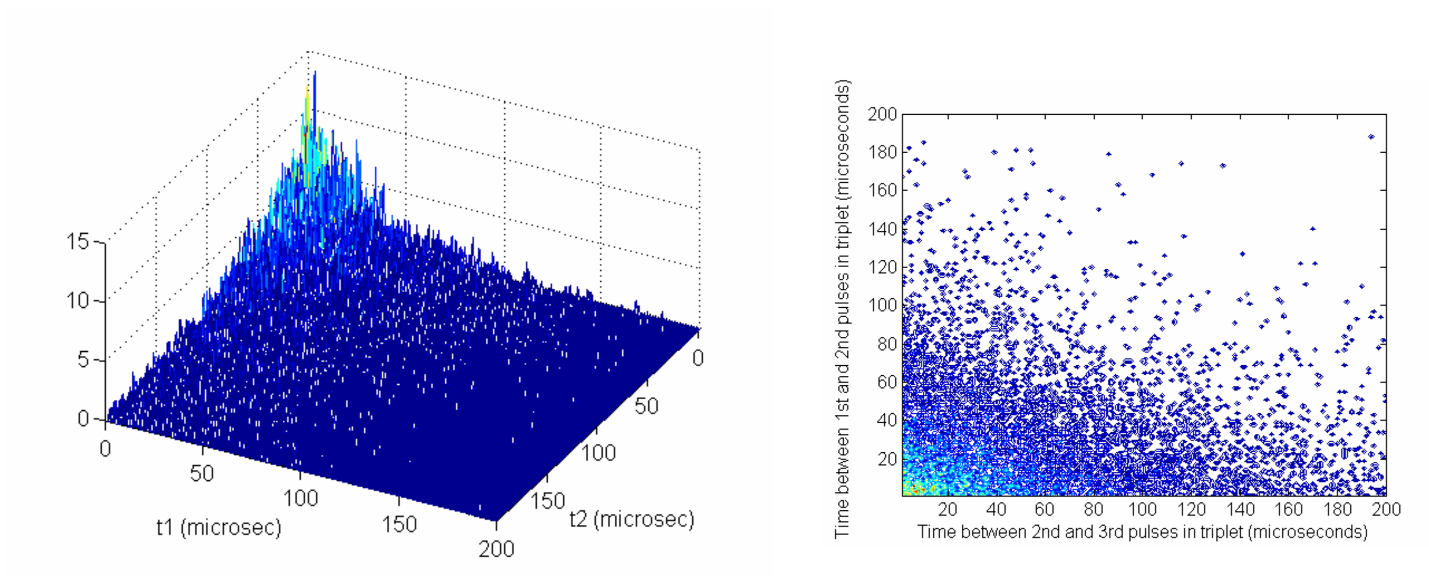


Figure 7. Time of detection after source event for neutrons of the source (generation 0) and induced fission neutrons of successive generations (generation 1 to 3).

Table 2. Neutron die away time

Neutron generation number	Die away time (μsec)
0	52.6
1	90.2
2	98.9
3	103.3

Figure 8 shows the time interval analysis for triplets of neutrons. The number of triplets is shown as a function of the time interval between the first and second pulses and the time interval between the second and third pulses. Figure 9 shows the integral of the result of Fig. 8 along the two axes t_1 and t_2.



(a) (b)

Figure 8. Number of neutron triplets detected as a function of the time interval between the first and second pulses in the triplet (t_1) and second and third pulses in the triplet (t_2) (a), contour plot (b).

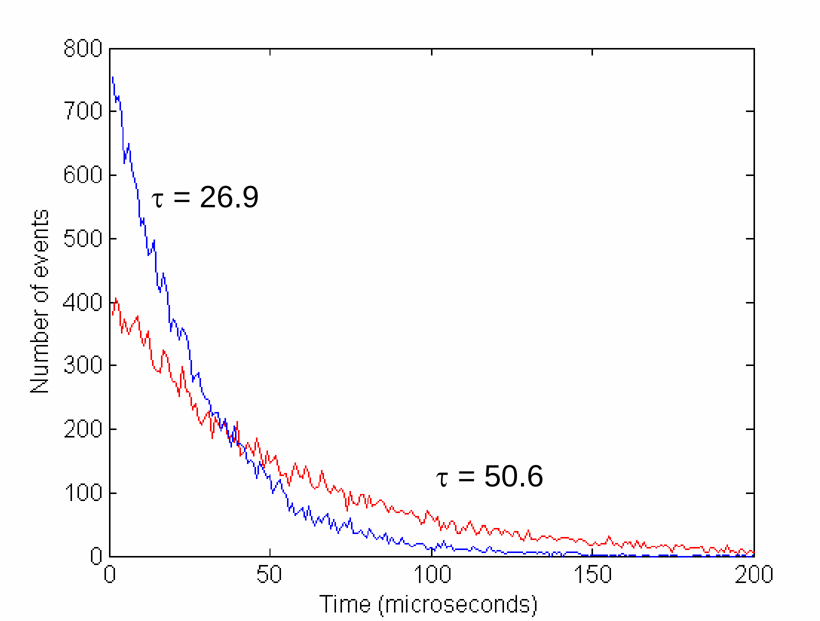


Figure 9. Integral of the result shown in Figure 8 along t1 (red curve) and along t2 (blue curve).

The result of Fig. 9 shows that the distributions of times occurring between the first and second pulses, and between the second and third pulses are not the same, as would be intuitively expected. The ratio of the decay constants in the simulation is approximately equal to 2. This result agrees with an analytical derivation by Bruggeman and colleagues [3]:

$$R_2(t_1,t_2) = Ae^{-2t_1/\tau} e^{-t_2/\tau}$$

where t_1 and t_2 are the times between the first and second pulse, and between the second and third pulse, and R_2 is the joint probability distribution of triplets in the t_1 - t_2 space.

6. CONCLUSIONS

This paper presented the use of the MCNP-PoliMi code and a new post-processing code to simulate the neutron measurements performed with an active well coincidence counter. The information given in the data output of the code can be used to gain an improved understanding of the physical processes that occur in the active well. In particular, it was shown that it is possible to simulate the operation of the shift register, to evaluate the energy degradation of the neutrons by analyzing the number of scatterings and the neutron energy prior to detection, and to determine the efficiency of the counter, both to source neutrons and to induced fission neutrons. Furthermore, time interval analysis measurements can be simulated. These are based on the knowledge of the times of neutron detection in the He-3 counters.

7. ACKNOWLEDGMENTS

The NIS-5 Safeguards Science and Technology group at Los Alamos National Laboratory provided the original MCNP input file for the active well coincidence counter.

8. REFERENCES

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