Comparison of MCNP4C and Experimental Results on Neutron and Gamma Ray Shielding Effects for Materials

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1. Introduction

MCNP code [1] is a general-purpose Monte Carlo radiation transport code that can numerically simulate neutron, photon, and electron transport. Increasing the speed of computing machine is making numerical transport simulation more attractive and has led to the widespread use of such code.

This code can be used for general radiation shielding and criticality accident alarm system related dose calculations, so that the version 4C2 of this code was used to evaluate the shielding effect against neutron and gamma ray experiments. The Ueki experiments [2] were used for neutron shielding effects for materials, and the Kansas State University (KSU) photon skyshine experiments [3] of 1977 were tested for gamma ray shielding effects.

2. Experiments and Model Descriptions

In Ueki experiments, a series of dose rate measurements have been performed on a number of common shielding materials using neutron sources. These materials included steel, cast iron, graphite, titanium hydride, polyethylene, titanium boride, and several ceramics. Of these, cast iron shield, graphite shield, and polyethylene shield were modeled using MCNP4C2 for neutron shielding effects. The neuron measurements were performed using a Cf-252 source. The measurement configuration for the neutron measurements is given in Figure 1. The shielding material was set between the source and the point detector. The thickness of the shield was increased toward the source. The MCNP neutron calculations were performed with the ENDF/B-V library (RMCCS) and a Cf-252 fission spectrum defined in the MCNP manual as

$$f(E) = e^{-E/1.025} \times \sinh(2.926E)^{1/2} \quad (E; MeV)$$

Figure 2 shows a cross sectional view of the silo walls parallel to the silo axis for the KSU photon skyshine experiment. This cross sectional view was modeled using MCNP4C2 for gamma ray shielding effect. In the model, a Co-60 source was placed at 6-ft 6-in. above grade on the axis of a 3-ft thick annular concrete silo. The bottom cask plate was modeled as 9-in. thick concrete. Ground was modeled as 1-ft thick soil layer. The ground and concrete compositions recommended by ANSI/ANS-6.6.1-1987 [4] were used in the simulation. Three Co-60 sources used were 10.33, 229.1, and 3804Ci. Taking advantage of radial symmetry, ring detectors were located a number of distances from the source. The ring detectors were centered on the photon source at a height of 1.0 m above and parallel to the ground plane. A 50 cm radius averaging volume was used to average the fluence around the rings.

3. Results

For neutrons, MCNP calculations were performed for each thickness of each shielding material given in the Ueki experiment, using 10^9 histories per case. No variance reduction techniques were applied. To convert the MCNP solution ($n/cm^2/n$) to the neutron dose rate ($\mu Sv/hr$), the ANSI/ANS-6.1.1-1977 [5] flux-to-dose conversion factors were used. Table 1 shows the final MCNP outcomes for the total 18 Ueki experiments along with the ratio relative to the corresponding measured values (C/E). The agreements between calculated and measured values for graphite and iron shields are within 33% and 16% respectively. However, the polyethylene results are up to 60% higher than the measurements.

For photons, MCNP calculations were performed for total eight experimental cases with 10^8 histories per case. Also no variance reduction techniques were applied. Because the results of the skyshine experiments were given in $\mu R/hr/Ci$, we used special tally options in MCNP for the gamma ray absorbed dose in air. The comparisons of the MCNP skyshine exposure calculations with the experimental data are summarized in Table 2. One can see the good agreement between them. The maximum C/E is just within in 14%.

4. Conclusion

According to both the neutron and the photon experiment simulations, the agreement between the MCNP simulation results and the measured values was reasonably good with the exception of polyethylene neutron shield.

References

[1] J.F. Briesmeister, "MCNP; Monte Carlo N-Particle Transport Code," LA-12625-M, Version 4C2, Los Alamos National Laboratory, June 2001.

[2] K. Ueki, et al., "Systematic Evaluation of Neutron Shielding Effects for Materials," *Nucl. Sci. Eng.*, 124, 455(1996)

[3] R.H.Olsher, et. Al., "Benchmarking the MCNP Monte Carlo Code with a Photon Skyshine Experiment," *Nucl. Sci. Eng.*, 219, 114(1993)

[4] ANSI/ANS-6.6.1-1987, "Calculation and Measurement of Direct and Scattered Gamma Radiation from LWR Nuclear Power Plants," June 1987

[5] ANSI/ANS-6.1.1-1977, "Neutron and Gamma- Ray Flux-to-Dose Factors," August 1991

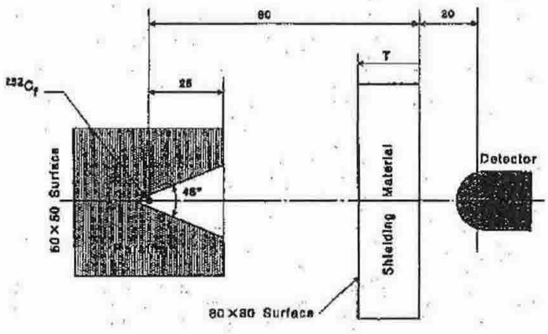

Fig. 1: Schematic arrangement of source, shield, and detector

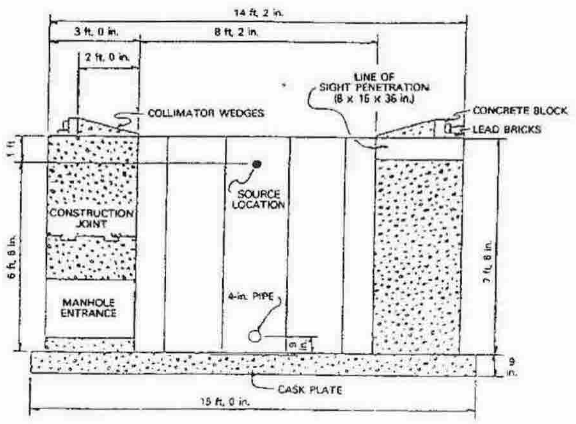

Fig. 2: Cross-sectional view of the silo walls parallel to the silo axis

Table 1. Comparison of measured and calculated neutron dose rates for various shielding materials

Thickness (cm)	Experimental Result	MCNP Result		C/E
		μSv/hr	Relative Error (%)	
Graphite Shield				
0	176.2	200.82	0.04	1.14
5	131.3	152.38	0.13	1.16
15	66.4	81.30	0.23	1.22
20	45.5	56.78	0.29	1.25
25	30.8	39.07	0.33	1.27
35	13.5	17.95	0.48	1.33
Iron Shield				
0	165.3	185.95	0.04	1.12
5	117.2	136.41	0.19	1.16
15	62.5	71.54	0.27	1.14
20	46.3	50.36	0.33	1.09
25	34.6	34.86	0.36	1.01
35	19.0	16.13	0.50	0.85
Polyethylene Shield				
0	683.0	779.03	0.10	1.14
5	288.0	363.37	0.17	1.26
15	42.6	66.18	0.43	1.55
20	18.3	29.35	0.66	1.60
25	8.3	13.25	0.99	1.60
35	2.25	3.09	2.02	1.37

Table 2. Comparison between KSU photon skyshine experiment and MCNP results

Distance from Source to Det. (m)	Co-60 Source Strength (Ci)	Experimental Result (μR/hr/Ci)	MCNP Result		C/E
			μR/hr/Ci	Relative Error (%)	
50	10.33	24.24	26.3	0.09	1.08
100	10.33	9.66	9.89	0.13	1.02
200	10.33	2.425	2.58	0.11	1.06
300	229.1	0.76	0.87	0.16	1.14
400	229.1	0.31	0.33	0.20	1.06
500	229.1	0.117	0.132	0.29	1.13
600	3804	0.0542	0.0556	0.51	1.03
700	3804	0.0244	0.0238	0.86	0.98