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Determination of Neutron Dose-Equivalent Buildup Factors for Infinite Slabs Irradiated by Point Isotropic Neutron Sources Using the MCNP Code

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Abstract

Neutron dose-equivalent buildup factors were calculated for point isotropic neutron sources irradiating infinite slablike shields of lead, iron and water using the MCNP code. The factors are presented for some source neutron energies in the range from 0.025 eV to 14 MeV, and for shield thicknesses from 0.5 to 10 mfp. Since the MCNP code considers all kinds of neutron interactions with matter, and the variance reduction techniques available in this code allows one to reach correct answers, one can say that the buildup factors presented here are accurate enough to be used in neutron shielding and dosimetry calculations. Comparison of the results obtained here with some previously calculated buildup factors, shows good agreements.

Keywords: Monte Carlo method; Neutron buildup factors; MCNP code; Radiation shielding

1. Introduction

Buildup factors are the basic data for point kernel methods that are used in radiation shielding and dosimetry calculations. In the point kernel formulation, a desired quantity such as flux, dose, or dose equivalent is expressed as the product of the portion of that quantity due to uncollided radiation and a buildup factor. Thus, the buildup factor can be defined (e.g., see Ref. 1) as the ratio of the correct value of a desired quantity to the uncollided component of that quantity, which is fairly easy to estimate even in complex geometries. The buildup factor can be obtained, in principle, by experiment; but since the attenuation coefficients and the scattering cross sections are known with reasonable accuracy, buildup factors are customarily obtained either by solution of the transport equation or by Monte-Carlo method. A detailed historical review on buildup factor calculation and use is given by Harima [2]. Although buildup factors have been widely calculated and used for gamma rays [3-7], they have also been applied (in a limited number of cases) to neutron transport simulations [8-10].

In this work neutron dose-equivalent buildup factors were calculated for three common shielding materials (lead, iron and water) in a point-source, infinite-slab, point-detector geometry, using the MCNP(version 4c) code [11]. The buildup factors were calculated at some source neutron energy points in the range from 0.025 eV to 14 MeV and for shield thicknesses from 0.5 to 10 mfp. Since the MCNP code considers all kinds of interactions of neutrons with matter, one can say that the buildup factors calculated and presented here are accurate enough to be used in neutron shielding and

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dosimetry calculations in this energy range.

2. Calculation

The MCNP is a multipurpose code that simulates the transport of photons, neutrons and electrons through matter by Monte-Carlo method, and with the different variance reduction techniques available in this code one can obtain results with reasonable accuracy even in very complex systems.

In the geometry of this work, a neutron point isotropic source of definite energy E_0 is assumed to be located on the one side of an infinite slablike shield of definite thickness and a point detector on the other side of the shield with the line of sight between source and detector normal to the slab surfaces. The MCNP code simulates the transport of neutrons (emerging from the source) through the shielding medium (lead, iron or water) according to the various types of interactions and their corresponding cross-sections taken from libraries available in this code. The code uses the cross-sections from the ENDF/B-VI files. The average dose-equivalent (per source neutron) at the detector point (H) is then printed in the output of this code. This quantity is obtained provided that relevant information is given in the input program file of the code to convert flux to dose-equivalent. The code also prints the average uncollided dose-equivalent (again per source neutron) at the detector point (H_0), this is simply:

$$H_0 = \frac{h(E_0)e^{-\Sigma_t T}}{4\pi d^2} \qquad (1)$$

Where:

E_0= Source neutron energy

$h(E_0)$ = Flux-to-dose-equivalent conversion factor

Σ_t = Total macroscopic cross-section of the shield material at E_0

T=Shield thickness

d=Source- detector separation

[In this work T≈ d. In fact d/T was taken to be slightly greater than unity, 1≤d/T<1.1, to avoid technical problems arising when running the program].

The dose-equivalent buildup factor (B) is then calculated from:

$$B = \frac{H}{H_0} \qquad (2)$$

3. Results

The neutron dose-equivalent buildup factors calculated in this work are presented in Tables 1 to 3 for infinite slablike shields of lead, iron and water, respectively. For each material, the factors have been calculated at ten source neutron energy points in the range from 0.025 eV to 14 MeV. The energy points have been chosen to, nearly, represent the average or peak neutron energy of some common neutron sources in this range. At each source neutron energy point, the factors are presented for shield thicknesses from 0.5 to 10 mean free paths (mfp) of the source energy E_0 (1 mean free path $= \Sigma_t^{-1}$). To obtain each factor, on average, 500,000 histories were run. For shield thicknesses of > 4 mfp, the variance reduction methods available in the MCNP code, such as DXTRAN SPHERE, GEOMETRY SPLITTING and RUSSIAN ROULETTE were used and (as it is usual in this kind of investigations) much effort was spent to reach correct answers for which all program statistical checks program were passed. For the buildup factors presented here, maximum relative error which occurs at 10 mfp is about 2 percent.

4. Comparison with Other Calculations

Dunn *et al.* [8] have calculated and presented neutron (and gamma-ray) dose-equivalent buildup factors at some energy points in the range from 10^{-7} to 14 MeV for some common shielding materials and for a similar geometry as considered in our case. In their method, they have used a combination of Boltzmann transport equation (for single scattering of neutrons) and a Monte-Carlo model (for multiple-scattering of neutrons). Their results are consistent with the results of this work, although there are small discrepancies at thicknesses of less than 5 mfp (specially in the case of water at some energy points) which could be due to the fact that the MCNP code considers all kinds of interactions of neutrons with matter and follows the history of each neutron down to 10^{-5} eV and uses the appropriate molecular cross-sections at very low energies. There are, however, considerable discrepancies at thicknesses of > 5 mfp which may be accounted for the fact that authors of Ref. 8 have admitted standard deviations of > 10% for thick shields of >5 mfp, because (as they stated in their article) their code was not specially designed for deep penetration transport calculations. Results of this work are compared with those of Ref. 8 in Figures 1 and 2 at example energies of 0.1 eV and 10 MeV for water. It is seen that while good agreements exist for thicknesses of < 5 mfp, the results of this work show increasing behavior of buildup factors at higher thicknesses (Ref. 8 has calculated the buildup factors up to 7 mfp).

Table 1. Neutron dose-equivalent buildup factors for lead

Energy (MeV)	Thickness (mfp)										
	0.5	*1*	*2*	*3*	*4*	*5*	*6*	*7*	*8*	*9*	*10*
2.5e-8	1.726	2.829	7.127	17.42	41.05	95.24	223.7	583.7	1219	2877	6691
10^{-7}	1.735	2.85	7.28	18.36	43.42	101.1	262.4	608.9	1437	3505	8510
10^{-5}	1.74	2.903	7.616	18.81	47.33	120.4	286.3	732.5	1832	4894	11696
10^{-3}	1.748	2.855	7.61	18.58	47.66	119.1	308.0	755.8	1895	4964	12797
0.1	1.730	2.802	7.091	17.34	42.93	100.6	233.5	563.1	1316	3247	7608
1.0	1.693	2.729	6.006	15.54	36.16	88.08	206.7	466.7	1083	2487	5771
2.5	1.743	2.842	7.627	19.91	49.8	127.1	318.2	749.1	2027	5123	12420
5.0	1.7	2.7	6.906	17.62	44.38	108.2	279.3	687.5	1690	4387	10425
10.0	1.745	3.093	8.518	21.94	52.37	130.6	330.3	764.5	1844	4593	10606
14.0	1.762	2.955	7.81	21.32	52.04	125.6	318	734.5	1799	4520	10190

Table 2. Neutron dose-equivalent buildup factors for iron

Energy (MeV)	Thickness (mfp)										
	0.5	*1*	*2*	*3*	*4*	*5*	*6*	*7*	*8*	*9*	*10*
2.5e-8	1.554	2.264	4.276	7.572	12.71	21.25	34.05	55.4	87.1	135.5	214.5
10^{-7}	1.629	2.515	5.35	10.4	20.08	36.95	65.82	116	198.8	339.3	562.2
10^{-5}	1.732	2.846	7.273	17.12	40.79	97.42	221.2	507.1	1162	2657	5881
10^{-3}	1.545	2.296	4.956	10.54	22.11	43.56	93	189.4	388.6	804.1	1609
0.1	1.705	2.746	6.433	15.16	34.18	79.17	201.0	454.3	1169	2874	7097
1.0	1.586	2.363	5.52	15.37	29.45	70.72	169.9	418.6	1048	2629	6614
2.5	1.759	2.906	7.566	18.67	45.3	106.2	246.3	578.9	1365	3314	7973
5.0	1.668	2.599	6.514	15.9	37.11	87.24	204.1	477.5	1175	2858	6684
10.0	1.614	2.574	6.45	15.26	34.98	78.53	195.5	445.0	966.4	2558	5896
14.0	1.577	2.373	5.57	11.99	28.83	64.76	136.4	328.9	733.5	1689	3888

Table 3. Neutron dose-equivalent buildup factors for water

Energy (MeV)	Thickness (mfp)										
	0.5	*1*	*2*	*3*	*4*	*5*	*6*	*7*	*8*	*9*	*10*
2.5e-8	1.756	2.948	7.695	18.99	47.76	117.1	251.5	711.5	1818	4551	11501
10^{-7}	1.763	2.99	8.103	20.65	53.10	130	313.7	758.1	1880	4094	10090
10^{-5}	1.711	2.935	8.165	23.27	52.88	133.3	330.8	760.3	1783	3708	9673
10^{-3}	1.808	3.123	9.13	22.99	71.44	177.2	448.3	1083	2707	6530	15094
0.1	1.375	1.845	3.489	7.051	14.9	33.3	75.62	151.3	333.7	733.4	1594
1.0	1.44	2.033	3.700	6.394	11.14	17.85	28.14	48.83	78.02	125.5	202.5
2.5	1.613	2.289	3.947	5.59	6.953	8.527	10.09	12.02	14.89	16.68	18.2
5.0	1.508	2.148	3.521	4.823	6.292	7.623	8.738	10.53	12.01	13.05	14.46
10.0	1.56	2.196	3.897	5.517	7.325	8.363	10.48	13.99	17.43	21.2	25.09
14.0	1.504	2.093	3.981	5.58	7.833	11.25	15.51	21.15	26.59	34.15	42.9

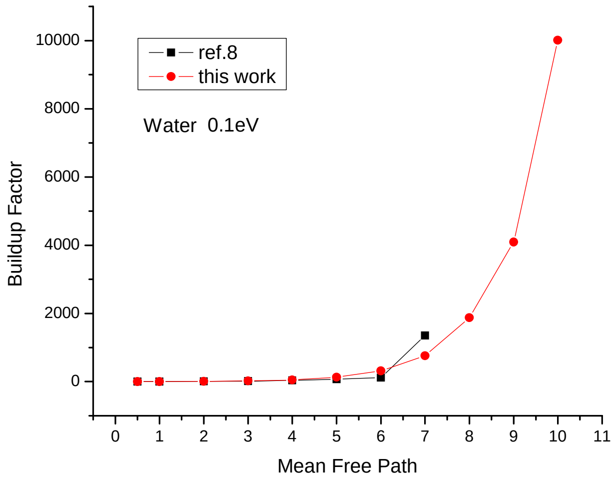

Figure 1. Comparison of buildup factors with those of Ref. 8 for water at 0.1 eV.

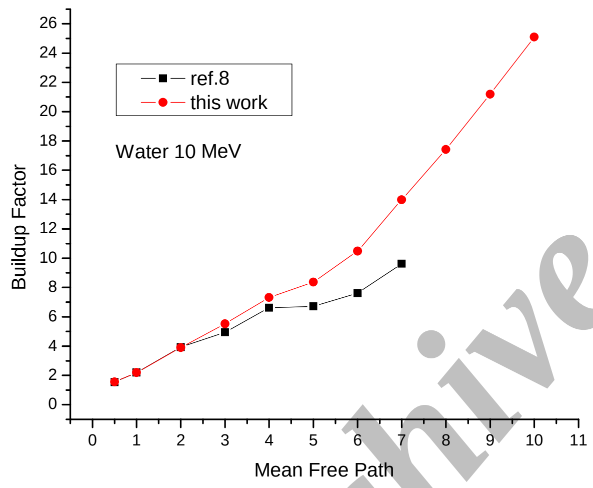

Figure 2. Comparison of buildup factors with those of Ref. 8 for water at 10 MeV.

5. Conclusion

Dose-equivalent buildup factors were calculated for uniform infinite slabs of lead, iron and water irradiated by neutron point isotropic sources in the energy range from thermal to fast 14 MeV neutrons. The MCNP code was employed in these calculations and variance reduction techniques available in this code (such as DXTRAN SPHERE, GEOMETRY SPLITTING and RUSSIAN ROULETTE) were used to obtain relatively accurate results. The buildup factor values presented here have maximum relative error of 2 percent which occurs for thick shields of 10 mfp. For thinner shields, the relative error is less than 1%.

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