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NEW BUILDUP FACTOR DATA FOR POINT KERNEL CALCULATIONS

D. K. Trubey
NRC Technical Data Management Center*
Oak Ridge National Laboratory
P.O. Box X
Oak Ridge, TN 37831

Y. Harima
Research Laboratory for Nuclear Reactors
Tokyo Institute of Technology
Tokyo 152, Japan

ABSTRACT

An American Nuclear Society Standards Committee Working Group, identified as ANS-6.4.3, is developing a set of evaluated gamma-ray isotropic point-source buildup factors and attenuation coefficients for a standard reference data base. As a first step, a largely unpublished set of buildup factors calculated with the moments method has been evaluated by recalculating key values with Monte Carlo, integral transport, and discrete ordinates methods. Attention is being given to frequently-neglected processes such as bremsstrahlung and the effect of introducing a tissue phantom behind the shield. The proposed standard contains data for a source energy range from 15 keV to 15 MeV and for approximately 19 elements and 3 mixtures (water, air, and concrete). The data will also be represented as coefficients for the G-P fitting function. The 1985 data base was released as part of the CCC-493B/QAD-CGGP code package available from the Radiation Shielding Information Center (RSIC).

INTRODUCTION

Since the 1979 accident at Three Mile Island, it has become increasingly apparent that a comprehensive and reliable set of gamma-ray attenuation data would be very useful to engineers involved in postaccident analysis and control. At Three Mile Island, for example, the on-site engineers were asked on day two or three to assist in the evaluation of the activity of ^{133}Xe in the waste gas decay tanks (WGDT).[1] Xenon-133 emits several low-energy gamma rays, the most important at 80 keV. Dose rates had been measured by health physicists in the WGDT room, and U.S. Nuclear Regulatory Commission (NRC) personnel needed to know the activity of the ^{133}Xe in the tanks that caused those dose rates in order to estimate the dose rates to the offsite populations when the ^{133}Xe was vented. However, to determine the activity in the tanks, they needed gamma-ray attenuation data that would predict the penetration of the ^{133}Xe gamma rays through the tank walls.

Another need for such data occurred about 60 days after the accident when gamma-ray dose rates and energy spectra were measured outside the 3.8-cm-thick steel containment equipment hatch. The gamma rays were due to noble gases inside the containment vessel, and if gamma-ray attenuation data describing the penetration of the noble gas gamma rays through the steel had been available, the activity of the noble gases within the containment vessel could be determined. Again, the offsite dose rates resulting from a purging of the noble gases from the containment vessel could have been readily determined.

Presently, under the "Lessons Learned from TMI" programs[2] instituted by NRC, plants must monitor releases from steam dump valves. Given the design constraints, attenuation data for low-energy gamma rays would be useful here also.

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The above TMI incidents have only served to emphasize the need for such data. Over the years, the Radiation Shielding Information Center at Oak Ridge National Laboratory (ORNL) has had numerous requests for similar data, but no reliable and comprehensive compilations have been available to satisfy the requests. As a result, shielding calculations in the nuclear industry have continued to rely on point-kernel methods incorporating interpolated or unpublished buildup factors. Various computations of attenuation data exist, but most of the data used by engineers are data calculated with the moments method by Goldstein and Wilkins and published in 1954.[3] The Goldstein-Wilkins document, undoubtedly the most cited reference in shielding literature, has been a *de facto* standard for more than 30 years, but it lacks low-energy data and data from many needed materials.

Following the TMI accident, it was suggested to the relevant standards subcommittee of the American Nuclear Society that the subcommittee assume responsibility for obtaining and/or generating a set of gamma-ray isotropic point-source buildup factors and attenuation coefficients for various engineering materials, evaluating them, and publishing them as a standard reference data base. ANS is a standards-writing organization member of the American National Standards Institute (ANSI), and the subcommittee to which the proposal was addressed was the Radiation Protection and Shielding Subcommittee (ANS-6) set up by the ANS Standards Committee. ANS-6 is charged with the responsibility for developing standards for radiation protection and shield design, providing shielding information to other standards-working groups, and developing standard reference shielding data and test problems. As is the case for all standards, the subcommittee's purpose in developing standards is to set forth acceptable practices, procedures, dimensions, material properties, specifications, etc. that have been agreed upon by representatives of a broad segment of the subject activity. Ideally, because of the standardization process, a standard is a high-quality, highly reliable, comprehensive summary of the state of the art.

Thus, in response to the above suggestion, the chairman of Subcommittee ANS-6.4, Shielding Materials, led an organizational meeting in early 1980 to initiate an effort to develop an appropriate and comprehensive standard. Later, one of the authors was appointed chairman of a new working group, designated as Working Group ANS-6.4.3, and members with special expertise were recruited from the following institutions: Bettis Atomic Power Laboratory, GA Technologies, Inc., Los Alamos National Laboratory, Pennsylvania State University, Indira Gandhi Centre for Atomic Research (India), and Tokyo Institute of Technology (Japan). (Additional organizational details are included in the appendix to this paper.)

CHARTER OF WORKING GROUP ANS-6.4.3

After the first meeting of Working Group ANS-6.4.3 in November 1980, a charter was approved by the ANS Standards Steering Committee at its February 1981 meeting. The scope reads as follows:

This standard presents evaluated gamma-ray elemental attenuation coefficients and pure-material buildup factors for selected engineering materials for use in shielding calculations of structures at power plants and other nuclear facilities. The data cover the energy range 0.01–10 MeV and up to 20 mean free paths. These data are intended to be standard reference data for use in radiation analyses employing point-kernel methods.

Notification was subsequently received, in May 1982, that the charter had also been approved by the ANSI Nuclear Standards Management Board.

ACTIVITIES OF WORKING GROUP ANS-6.4.3

Before Working Group ANS-6.4.3 was organized, a data base of buildup factors and absorption coefficients based on moments method calculations was already in existence at the National Bureau of Standards (NBS) for a number of materials: $_4$Be, $_5$B, $_6$C, $_7$N, $_8$O, $_{11}$Na, $_{12}$Mg, $_{13}$Al, $_{14}$Si, $_{15}$P, $_{16}$S, $_{18}$Ar, $_{19}$K, $_{20}$Ca, $_{26}$Fe, $_{29}$Ca, $_{42}$Mo, $_{82}$Pb, $_{92}$U, water, air, and concrete. Data on $_{50}$Sn, $_{74}$W, and $_{64}$Gd are being obtained by other methods.

Though the NBS data base is comprehensive, only the results for air, water, and iron have been published,[4] and thus most of the data are not yet generally available. Also, some outstanding questions concerning the data base needed to be resolved. For example, the moments method calculations did not take into account coherent scattering or bremsstrahlung, and the importance of these effects needed to be determined. In addition, there was an unknown uncertainty arising from the reconstructions of the moments.

With the large NBS data base available, it was decided that the first effort of the Working Group ANS-6.4.3 should concentrate on validating the buildup factors included in the data base for water, lead, and iron, at the same time investigating the importance of the neglected effects. The validation effort consisted of recalculations of the buildup factors for these materials by Monte Carlo, integral transport, and discrete ordinates methods. The Monte Carlo calculations, employing the MCNP code,[5] were carried out at Los Alamos National Laboratory (LANL). The integral transport calculations were done at the Indira Gandhi Centre for Atomic Research in India with the ASFIT code,[6] and the discrete ordinates calculations were performed at GA Technologies, Inc., with the DTFX code[7] and at Japan Atomic Energy Research Institute with the PALLAS code.[8]

TECHNICAL ISSUES BEING ADDRESSED

Effect of Coherent Scattering

As noted above, the moments method calculations do not account for coherent scattering. Our own Monte Carlo calculations confirm statements in the literature that coherent scattering can be neglected for most shielding calculations with the proviso that the total attenuation coefficient used in the calculations does not include it. Several MCNP calculations were performed to evaluate the effect of neglecting coherent scattering. Table 1 shows the effect for a 100 keV source in iron with three response functions. The effect generally increases with shield thickness and can be of the order of 20–40%.

Table 1. Response ratios (without coherent/with coherent) for a 0.1 MeV source in Fe.

mfp	ϕ	Fe	Air
1/2	0.994	0.647	0.840
1	1.013	0.694	0.869
2	1.021	0.566	0.793
3	1.051	0.513	0.758
4	1.070	0.605	0.834
5	1.105	0.791	0.962
6	1.089	0.731	0.928
7	1.146	0.729	0.942
8	1.162	0.840	1.017
10	1.199	0.643	0.909
15	1.406	0.859	1.144
20	1.626	1.133	1.397

Effect of Secondary Sources

The Goldstein-Wilkins data do not include secondary sources such as annihilation, fluorescence, and bremsstrahlung. Our own ASFIT calculations[3] confirm the importance of all three sources in general, although of varying importance for different materials and source energies.

The NBS moments method data do not include bremsstrahlung, but ASFIT and PALLAS calculations do. Such data are important for high-Z materials and for the higher source energies.[9] Typical results of the PALLAS code are given in Table 2.[10]

Table 2. Bremsstrahlung effect on exposure buildup factor for a point isotropic source in lead

Distance (mfp)	Energy (MeV)			
	4	8	10	15
10	5.46	4.97	4.7	4.3
	5.80	8.97	13.8	32.8
20	15.4	30.0	40	81
	16.1	53.6	128	783
40	61.5	817	2620	34,800
	63.7	1310	7510	300,000

Upper entry: No bremsstrahlung; lower entry: bremsstrahlung included. Data of Tekeuchi and Tanaka, PALLAS Code.

Selection of Response Function

There is no universally recognized standard response function for the determination of buildup factors. Traditionally, the response medium has been air, which implies exposure as the endpoint, and hence exposure buildup factors are frequently found in the literature. It would be better if the endpoint were dose equivalent in a reasonably realistic geometry, which implies a tissue phantom geometry. An ANSI standard issued in 1977[11] gives guidance for the estimation of dose equivalent in tissue when one knows the spectrum impinging upon the tissue phantom. This is not logically applicable to the point-kernel method, however, since it is based on a knowledge of the source spectrum rather than the impinging spectrum, and the buildup factor used is derived from an infinite-medium calculation.

On the other hand, the estimated dose for radiation protection purposes should include the effect of multiple scattering in tissue (as the ANSI standard does), rather than in an infinite medium of shielding material alone. In the case of a water shield, the effect of replacing some of the water with tissue is not great, but in the case of a lead shield, the effect is quite large. For a source energy of 0.2 MeV, the lead correction factor for radiation protection purposes is as large as 1.45.[12] This correction factor, and similar correction factors for other cases can be deduced from the ratio of the column 4 values to the column 1 values shown in Table 3. The table shows buildup factors in slab geometry for an infinite medium of lead (column 1), a finite slab of lead (column 2), and lead-tissue sequence (columns 3 and 4). The buildup factor given in column 4 is the maximum in the tissue.

Table 3. Plane collimated buildup factors for lead[a]

E (MeV)	Thickness (mfp)	Buildup factor[b]			
		1	2	3	4
0.05	1	1.02	1.01	1.69	1.74
	2	1.02	1.01	1.70	1.75
	5	1.03	1.02	1.71	1.84
	10	1.03	1.03	1.71	1.77
	20	1.04	1.04	1.74	1.78
0.20	1	1.23	1.16	1.74	1.76
	2	1.26	1.20	1.80	1.83
	5	1.36	1.29	1.94	1.98
	10	1.51	1.41	2.14	2.16
	20	1.81	1.64	2.47	2.48
1.0	1	1.38	1.36	1.53	1.53
	2	1.65	1.63	1.84	1.84
	5	2.29	2.24	2.56	2.56
	10	3.06	3.01	3.46	3.46
	20	4.18	4.09	4.75	4.75
3.0	1	1.49	1.40	1.51	1.51
	2	1.85	1.76	1.89	1.89
	5	2.95	2.78	3.05	3.05
	10	4.80	4.56	5.03	5.03
	20	8.68	8.15	9.16	9.16

[a]From Ref. 12.

[b]1 = infinite medium; 2 = finite medium, 3 = interface; and 4 = corrected (tissue maximum).

Another problem in selecting a response function is that there is no standard composition or geometry for tissue. Because of the lack of standardization, Hubbell[13] has given energy deposition coefficients for several tissues reported by the ICRP and ICRU.

In view of the above problems, Working Group ANS-6.4.3 plans to provide exposure buildup factor data and also a table of values to correct for the shield-tissue interface. This problem is being addressed by using the ASFIT and PALLAS codes.

Fitting Functions

Although some analysts prefer to use data tables directly in computer calculations, others prefer parameterized forms. Over the years a number of functional forms have been used to parameterize attenuation data, in particular, the Taylor and Berger forms. For the usual Taylor form and for the Berger form, accuracy is a problem for low-Z materials and low energies. It is now clear that the best available form is the geometric progression (G-P) form.[14–16]

This formula can accurately reproduce the data over the full range of energy and atomic number within a few percent. For example, the maximum deviation to 40 mean free paths of water for the the exposure buildup factor is 3%. In Table 4, Max. dev. is the maximum deviation from the moments method results for $0 \leq x \leq 40$ mfp. The maximum occurs at X_{max}. Table 4 shows the standard deviation in the last column. The G-P function is:

$$B(E,x) = 1 + (b-1)\ (K^x - 1)/(K-1) \text{ for } K \neq 1 \tag{1}$$

$$= 1 + (b-1)x \qquad \text{for } K = 1\ ,$$

Table 4. G-P Coefficients for Water

Water Kerma								
E(MEV)	b	c	a	X_K	d	Max. dev.(%)	X_{max}	St. dev.(%)
0.015	1.188	0.464	0.172	14.00	-0.0829	1.37	0.5	0.508
0.020	1.449	0.532	0.152	14.61	-0.0764	1.91	0.5	0.730
0.030	2.411	0.741	0.084	14.62	-0.0452	2.29	0.5	1.065
0.040	3.587	1.114	-0.018	12.48	0.0013	0.68	0.5	0.248
0.050	4.554	1.457	-0.084	13.69	0.0341	1.47	35.0	0.977
0.060	5.018	1.735	-0.127	13.70	0.0676	2.37	35.0	1.564
0.080	5.030	2.054	-0.167	13.84	0.0763	3.00	35.0	1.982
0.100	4.627	2.207	-0.184	13.27	0.0799	2.98	10.0	1.885
0.150	3.888	2.206	-0.180	14.27	0.0738	2.37	0.5	1.478
0.200	3.462	2.132	-0.173	14.51	0.0750	2.36	0.5	1.335
0.300	2.897	2.008	-0.162	14.18	0.0641	2.40	35.0	1.412
0.400	2.646	1.874	-0.148	14.16	0.0591	2.41	35.0	1.377
0.500	2.499	1.749	-0.132	14.36	0.0517	2.00	1.0	1.195
0.600	2.383	1.662	-0.121	14.19	0.0482	1.97	35.0	1.224
0.800	2.223	1.524	-0.101	14.31	0.0403	1.97	1.0	1.078
1.000	2.106	1.436	-0.088	14.19	0.0367	1.68	0.5	0.970
1.500	1.948	1.265	-0.057	14.98	0.0245	1.51	0.5	0.616
2.000	1.843	1.169	-0.038	14.22	0.0157	1.61	0.5	0.551
3.000	1.716	1.050	-0.011	13.63	0.0027	0.97	0.5	0.314
4.000	1.633	0.979	0.007	14.23	-0.0060	0.65	0.5	0.252
5.000	1.571	0.928	0.022	13.20	-0.0157	0.62	4.0	0.336
6.000	1.521	0.893	0.033	11.92	-0.0208	1.87	1.0	0.636
8.000	1.432	0.873	0.038	11.56	-0.0204	0.59	3.0	0.355
10.000	1.378	0.849	0.045	14.34	-0.0280	0.98	0.5	0.574
15.000	1.280	0.829	0.052	14.85	-0.0367	1.23	3.0	0.688

Table 4. Continued

Air kerma (exposure)								
E(MEV)	b	c	a	X_K	d	Max. dev.(%)	X_{max}	St. dev.(%)
0.015	1.182	0.463	0.175	14.23	-0.0908	1.69	0.5	0.634
0.020	1.427	0.549	0.143	14.86	-0.0707	2.31	0.5	0.780
0.030	2.335	0.736	0.087	13.28	-0.0419	2.34	0.5	1.180
0.040	3.477	1.117	-0.019	11.67	0.0026	0.89	0.5	0.298
0.050	4.461	1.457	-0.084	13.62	0.0341	1.26	10.0	0.850
0.060	4.983	1.730	-0.126	13.64	0.0561	2.22	10.0	1.563
0.080	5.059	2.059	-0.168	13.67	0.0770	2.94	35.0	1.946
0.100	4.663	2.221	-0.186	13.33	0.0826	3.02	10.0	1.925
0.150	3.897	2.242	-0.185	14.19	0.0777	2.55	15.0	1.584
0.200	3.478	2.154	-0.176	14.50	0.0774	2.50	0.5	1.492
0.300	2.920	2.022	-0.164	14.21	0.0655	2.46	1.0	1.471
0.400	2.660	1.882	-0.149	14.24	0.0595	2.17	35.0	1.372
0.500	2.500	1.766	-0.135	14.33	0.0546	2.46	1.0	1.380
0.600	2.377	1.679	-0.124	14.23	0.0503	2.02	1.0	1.265
0.800	2.212	1.544	-0.105	14.36	0.0437	1.97	0.5	1.186
1.000	2.103	1.441	-0.089	14.22	0.0378	1.55	0.5	0.895
1.500	1.939	1.269	-0.058	14.52	0.0246	1.91	0.5	0.757
2.000	1.839	1.173	-0.039	14.07	0.0161	1.45	0.5	0.522
3.000	1.710	1.056	-0.013	11.82	0.0047	0.70	0.5	0.239
4.000	1.621	0.989	0.004	13.45	-0.0041	0.55	1.0	0.211
5.000	1.554	0.939	0.018	13.55	-0.0122	0.51	30.0	0.333
6.000	1.507	0.903	0.029	16.13	-0.0272	0.85	25.0	0.475
8.000	1.422	0.879	0.035	13.36	-0.0191	0.89	0.5	0.445
10.000	1.362	0.859	0.042	13.37	-0.0247	0.90	0.5	0.453
15.000	1.267	0.843	0.047	15.08	-0.0336	1.02	1.0	0.604

$$K(x) = cx^a + d \cdot [\tanh(x/X_k - 2) - \tanh(-2)]/[1 - \tanh(-2)] \ , \quad (2)$$

where x is the source-detector distance in mean free paths (mfp), b is the value of the buildup factor at 1 mfp, and K is the multiplication per mfp. Equation (2) represents the dependence of K on x; a, c, d, and X_k are fitting parameters which depend on source energy.

The G-P fitting function has been implemented in the CCC-493/QAD-CGGP and CCC-494/G33-GP codes[17] available from RSIC at Oak Ridge National Laboratory. Fitting function parameters and attenuation coefficient data for 22 materials and 2 response functions are utilized in the form of data files. This will allow easy modification when the final standard buildup factors become available from ANS-6.4.3. If these data are incorporated in other codes, it is recommended that the same format be used so that revised data can be easily incorporated. The two FORTRAN 77 codes have been tested at RSIC on CRAY X-MP, IBM 3033, Data General Eclipse MV/4000, and IBM PC computers.

In summary, modern buildup factor data are now available across the whole range of atomic number and source energies, and an accurate fitting function is available to reproduce these data in point kernel calculations on computers of all sizes.

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APPENDIX

The American Nuclear Society Standards Steering Committee is chaired by John H. Crowley, United Engineers and Constructors, Inc., who also represents the society on the Nuclear Standards Management Board of ANSI. Marilyn D. Weber is the Committee Administrative Secretary and ANS Standards Manager.

Subcommittee ANS-6, Radiation Protection and Shielding, is chaired by D. K. Trubey. E. Boeing, chairs ANS-6.4, Materials.

Working Group ANS-6.4.3 is chaired by D. K. Trubey. Group members are:

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A. Foderaro, Pennsylvania State University, State College, Penn.,
D. V. Gopinath, Indira Gandhi Centre for Atomic Research, Kalpakkam, India,
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