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FLUENCE-TO-DOSE EQUIVALENT CONVERSION FACTORS FOR POLYETHYLENE-MODERATED ^{252}CF

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ABSTRACT

Neutron measurements and calculations were conducted to characterize the polyethylene-moderated ^{252}Cf source at Oak Ridge National Laboratory's Radiation Calibration Laboratory (RADCAL). The 12-inch-diameter polyethylene sphere produces a highly scattered neutron spectrum which is more representative of most radiation fields found in the workplace than the D_2O-moderated ^{252}Cf neutron spectrum typically used for dosimeter calibration. However, the energy-dependent fluence and dose equivalent must be well known before using such a source for radiation protection purposes. The measurements and calculations were performed as independent checks of the desired quantities which were the flux, the absorbed dose rate, the dose equivalent rate, and the average energy. These quantities were determined for the polyethylene sphere with and without an outer cadmium shell and compared with a D_2O-moderated ^{252}Cf source.

MEASUREMENTS

A 5-inch-diameter tissue-equivalent proportional counter (TEPC) was used to measure the neutron dose and dose equivalent rates and a multisphere spectrometer system was used to determine the energy-dependent neutron flux, the total neutron flux, and the average neutron energy. The multisphere system also calculates the neutron dose and dose equivalent rates. The reference distance was chosen to be 1 meter from the center of the source. In addition to the reference distance, the neutron dose equivalent rates were also measured at 0.75, 1.5, and 2 meters with the TEPC to provide an estimate of the room scatter contribution at 1 meter for both source configurations using the National Bureau of Standards technique from Publication 633.[1]

The data used to determine the room return for each source configuration are given in Table 1. The TEPC estimated a neutron dose equivalent rate at 1 meter of 55.0 mrem/h for the polyethylene-moderated source without the cadmium, 53.3 mrem/h with the cadmium shell, and 341 mrem/h for the D_2O-moderated source without the cadmium shell. Two different ^{252}Cf sources were used with the polyethylene and D_2O moderators. The room return was estimated to be 8.1%, 6.8%, and 6.7% for the polyethylene moderator without the cadmium shell, the polyethylene moderator with cadmium and D_2O without cadmium, respectively.

The multisphere measurements were taken with the bare Li(I) detector and the Li(I) detector covered with cadmium, and 3-, 5-, 8-, 10-, and 12-inch polyethylene spheres. Only the 1-meter distance was measured with the multispheres for each source configuration. The measured count rates were very high and typical counting times were 100-200 seconds. The results of the count rate analysis for each detector configuration were a total neutron flux, dose rate, dose equivalent rate, and average neutron energy. These results are given in Table 2.

Table 1. TEPC data for room return calculation

	Distance, r (m)	Dose Equivalent, H (mrem/h)	r^2	Hr^2
^{252}Cf Source				
Poly-moderated	0.75	93.3	0.56	52.2
	1.00	55.0	1.00	55.5
	1.50	26.3	2.25	59.2
	2.00	16.6	4.00	66.4
Poly-mod. w/Cd	0.75	92.8	0.56	52.0
	1.00	53.3	1.00	53.3
	1.50	25.7	2.25	57.8
	2.00	15.9	4.00	63.6
D_2O-moderated	0.75	619	0.56	347
	1.00	341	1.00	341
	1.50	169	2.25	380
	2.00	104	4.00	416

Table 2. Multisphere results at 1 meter from source

	Total Neutron Flux (n/cm^2-s)	Absorbed Dose Rate (mrad/h)	Dose Equivalent Rate (mrem/h)	Average Neutron Energy (MeV)
^{252}Cf Source				
Poly-moderated	1.176E+3	6.05	47.7	0.577
Poly-mod. w/Cd	7.610E+2	6.29	51.8	1.595
D^2O-moderated	1.425E+4	62.0	415.	0.626

CALCULATIONS

The neutron spectra were calculated using the monte carlo transport code, MCNP Version 3B,[2] along with nuclear data from ENDF/B-V. The exact polyethylene sphere was modeled, including the varying diameter hole through the middle that accommodated the source capsule. The detachable cadmium shell that fit around the outside of the polyethylene sphere was also modeled. MCNP was used to calculate the energy-dependent fluences at 1 meter away from the center of the sphere, neglecting room return. The shape of the spectra and the total fluence rates (fluxes) are compared to the multisphere analysis results and the derived neutron dose equivalent rates are compared to the TEPC for verification purposes.

The input model for the MCNP calculations consisted of a 12-inch-diameter sphere of polyethylene with a density of 0.95 g/cc. The cadmium shell was modeled as a 30-mil-thick layer around the polyethylene. The hole through the middle of the sphere was modeled as a cylinder with a 2-inch diameter for the bottom 2.5 inches, a cylinder with a diameter of 0.5 inch from 2.5 inches up to 7 inches, and a cylinder with a diameter of 0.25 inch the rest of the way through (Figure 1). The source term was an isotropic point source in the center of the sphere with the built-in neutron energy distribution given in MCNP for the spontaneous fission of ^{252}Cf.

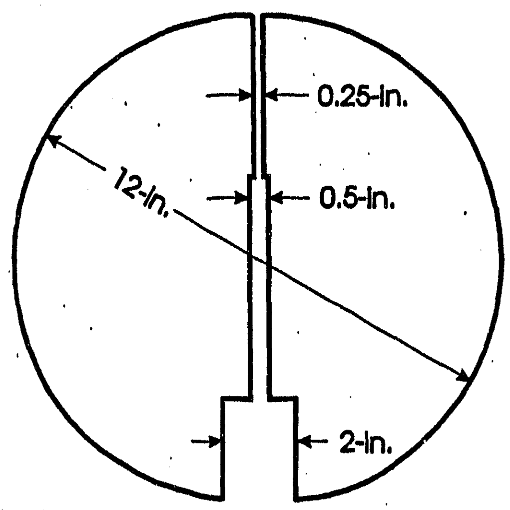

Figure 1. MCNP Geometry Model of Polyethylene Moderated

The ring detector tally in MCNP was used to calculate the energy-dependent fluences at 1 meter, taking advantage of the two-dimensional symmetry of the model. The calculated neutron energy spectra for polyethylene-moderated ^{252}Cf are shown in Figure 2. As expected, the cadmium cover affects the magnitude of the neutron fluence only below about 1 eV. The spectrum for polyethylene-moderated ^{252}Cf without the cadmium cover is compared to a D_2O-moderated ^{252}Cf spectrum, also calculated by MCNP, in Figure 3. The polyethylene is more effective at moderating the higher energy neutrons but also absorbs more intermediate energy neutrons and thus does not build in a large intermediate component like the D_2O moderator. The MCNP-calculated quantities had statistical errors of less than 5% for the differential fluences and less than 0.4% for the integral values.

Since the fluences calculated by MCNP are normalized to the number of source neutrons transported, the results were multiplied by the source strength (neutrons/second) to give the energy-dependent and total neutron fluxes. The source strength used for the polyethylene-moderated configurations was 5.248×10^8 neutrons/second on April 26, 1993, based on the National Institute of Standards and Technology calibrated source strength of 2.51×10^9 neutrons/second on May 6, 1987, and a 2.645-year half-life. A different ^{252}Cf source was used with the D_2O sphere, which had a calibrated source strength of 7.63×10^9 neutrons/second on April 30, 1987. The decayed value used was 1.59×10^9 neutrons/second. The tissue kerma factors from Caswell and Coyne[3] and the quality factors from Cross and Ing[4] were used to convert the energy-dependent fluxes to absorbed dose and dose equivalent for each source calculation. The MCNP-derived quantities are given in Table 3.

DISCUSSION

The TEPC estimated a neutron dose equivalent rate of 55.0 mrem/h for the polyethylene-moderated source without the cadmium and 53.3 mrem/h with the cadmium shell. The room return was estimated to be 8.1% for the case without the cadmium shell and 6.8% for the case with the cadmium shell. The neutron dose equivalent rates determined using MCNP were 49.3 mrem/h and 48.7 mrem/h for the polyethylene sphere without the cadmium shell and with the cadmium shell, respectively. When the contribution from room return is added to these results, the values become 53.3 mrem/h and 52.0 mrem/h, which are within 3% of the values measured by the TEPC. Both the measured data and the calculated data showed that the cadmium cover had no effect on the spectrum-averaged quality factor.

The neutron energy distribution for the polyethylene-moderated ^{252}Cf source with the cadmium shell measured by the multispheres and calculated with MCNP is shown in Figure 4. The shapes of the two spectra differ in the thermal region and the location of the uncollided peak. The larger multisphere thermal flux is attributed to the additional room scatter component, which is measured but not included in the calculations. However, the total neutron flux differs by only 9% for the polyethylene sphere with the cadmium cover and 16% for the sphere without the cadmium cover. The fluence-weighted average energies calculated by MCNP were 0.91 MeV for the polyethylene-moderated source, 1.78 MeV for the cadmium-covered polyethylene-moderated source, and 0.75 MeV for the D_2O-moderated source. The average energies determined by the multisphere system were 10-40% lower than the MCNP values.

Based on the results calculated by MCNP and verified by the TEPC measurements, the neutron fluence-to-dose equivalent conversion factors for the polyethylene-moderated ^{252}Cf source, with and without the cadmium shell, are 1.96×10^{-5} mrem-cm^2/neutron and 9.78×10^{-6} mrem-cm^2/neutron, respectively. These conversion factors are for the source neutrons only. Both polyethylene-moderated sources produce

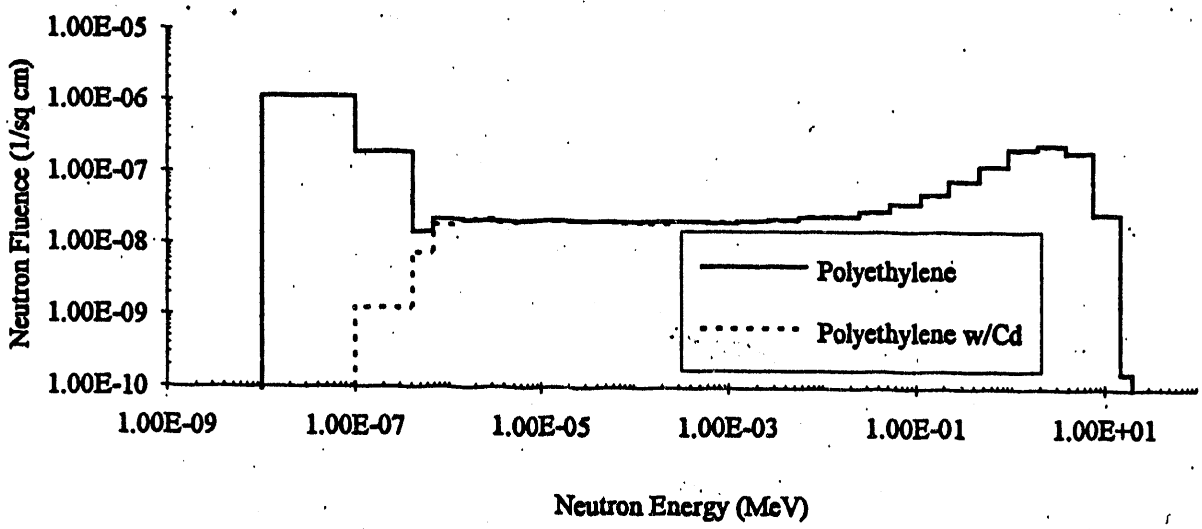

Figure 2. MCNP Neutron Spectra for Polyethylene-Moderated ^{252}Cf, With and Without Cadmium Shell

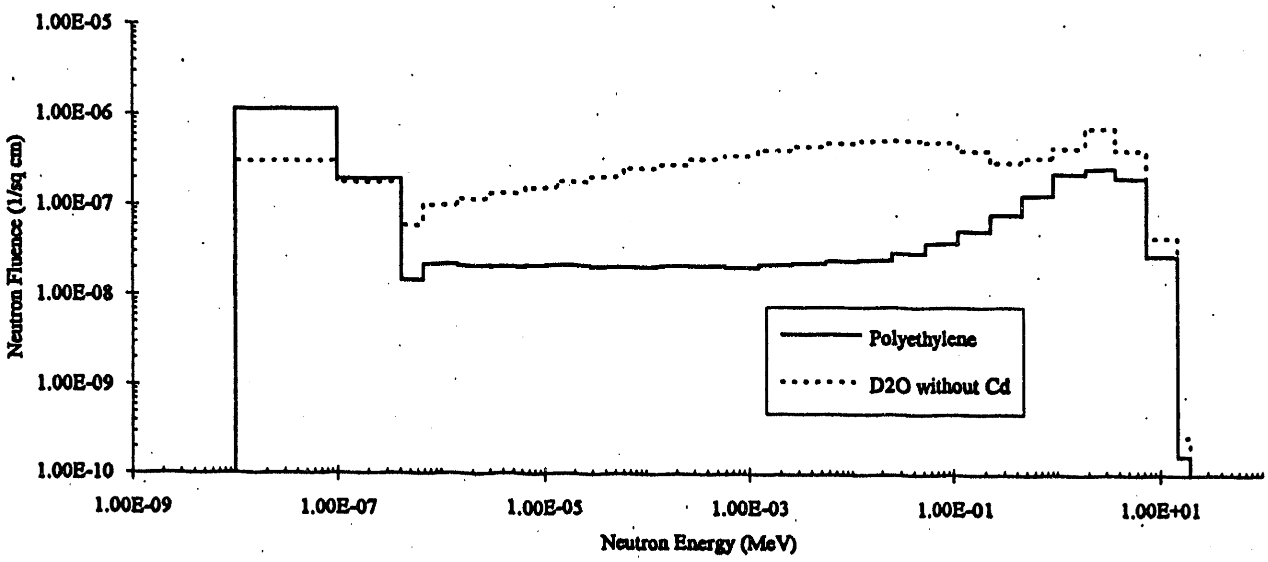

Figure 3. MCNP Neutron Spectra for Polyethylene- and D_2O-Moderated ^{252}Cf

Table 3. MCNP results at 1 meter from source

	Total Neutron Flux (n/cm^2-s)	Absorbed Dose Rate (mrad/h)	Dose Equivalent Rate (mrem/h)	Average Neutron Energy (MeV)
Poly-moderated	1.406E+3	5.46	49.3	0.911
Poly-mod. w/Cd	6.900E+2	5.39	48.7	1.782
D^2O-moderated	1.285E+4	46.5	435.	0.752

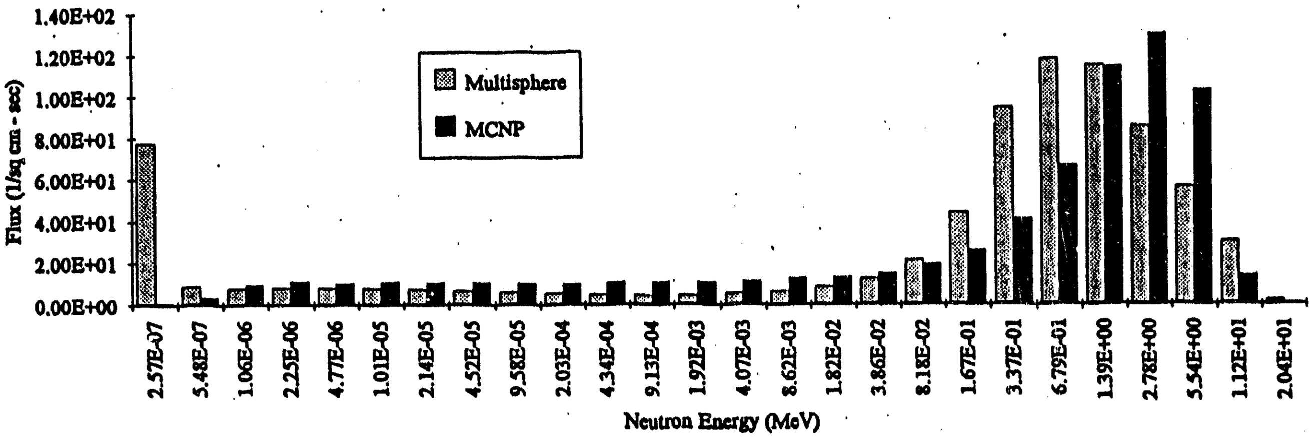

Figure 4. Polyethylene-Moderated ^{252}Cf with Cadmium Shell

higher conversion factors than the standard D_2O-moderated source with cadmium that has a value of 9.08×10^{-6} mrem-cm^2/neutron. However, dosimeter response, which is highly energy-dependent, may vary much more than is indicated by the integral quantities.

ACKNOWLEDGMENTS

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