Comparison of Criticality Dose Calculations Using Various Standards

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1. Introduction

A series of MCNP runs have been performed to compare the results of radiation dose calculated from several standards. A point source and Watt fission energy spectrum were defined for the calculations.

For conversion from flux to rem/rad-in-tissue, ANSI/ANS-6.1.1-1977 [1] has been used. The new standards, ANSI/ANS-6.1.1-1991 [2] and ICRU-57 [3] are introduced to compare the converted dose/dose rate.

For dose/dose rate in rad-in-air, F5 option in MCNP has been used. ICRU-44 [4] and ICRP-74 [5] standards are introduced to check the MCNP results.

2. Standards for Dose Calculations

ANSI/ANS-6.1.1-1977 standard has been used to date for radiation shielding calculations and intended for shield designer to calculate whole body dose rates to radiation workers and the general public. The neutron flux-to-dose rate conversion factors are given in units of rem/hr/(n/cm^2-sec) in tissue. MCNP results are given in n/cm^2/n. To convert the MCNP result to a dose in a unit of rem, the conversion factors in this standard are used.

ANSI/ANS-6.1.1-1991 standard is the most recent revision. The neutron and gamma ray conversion factors are given in sievert (100rem) per unit fluence (Sv-cm^2) in tissue. The fluence-to-dose factors are presented for four modes of exposure; frontal exposure, rear exposure, side exposure, and normal to the length of the phantom and rotationally symmetric. The standard states that, “If orientation is unknown, those corresponding to the most conservative dose estimate (frontal exposure) should be used, except in the case of a subject immersed in a cloud of gamma ray emitting isotopes, in which case those values corresponding to isotopic exposure.” So among these modes, the frontal exposure mode was used in this study.

ICRU-57 is the latest publication, which was prepared jointly by ICRP and ICRU and published in 1998. This report gives the tissue-absorbed dose for neutrons, and air KERMA factors for photon. Using conversion factors of this standard, the frontal exposure mode was used.

KERMA factors for various tissue substitutes are tabulated in ICRU-44, which was published in 1989. Data in this report provides a means of checking MCNP calculated neutron dose in rad-in-air. The neutron KERMA factors of air are specified in units of Gy-m^2 and are used in calculating the neutron dose in rad-in-air (rad-cm^2).

ICRP-74 gives the data set for photon KERMA factors for air, which was published in 1997. Conversion factors are given for air KERMA per unit fluence in units of pGy-cm^2, and are converted to rad-cm^2.

3. Criticality Dose Calculations

The criticality dose evaluation involves a series of calculations. First, the fission source needs to be defined. With the defined source, radiation dose calculations for neutrons and gamma rays are then performed; neutron dose, neutron induced secondary gamma ray dose, prompt gamma ray dose, and fission product gamma ray dose.

The composition and geometry of the fission source should represent a critical or near critical system of interest. Critical dimensions and compositions for the system of interest can be found in Ref. 6. A spherical source was defined using critical mass and dimension data for an unreflected, water moderated U-235 system, and this configuration was fine tuned by a series of MCNP calculations. KCODE calculations showed that a critical bare sphere containing 1,856g of U-235 mixed with full-density water has a radius of 16cm.

This critical volume source was converted into a point source for the subsequent calculations using MCNP KCODE run. The results showed that the leakage fraction of neutrons was 0.44875, leakage fraction of gamma rays was 2.0605, and the number of neutrons produced per fission was 2.52756. Also the neutron and gamma ray spectra were generated.

Dose calculations involve defining the neutron and gamma ray spectra of the fission source, and performing a series of dose calculations to estimate the criticality dose. Two different methods were presented for ca1culating neutron and gamma ray spectra; 1) using the Watts fission spectrum for neutrons, and the prompt and fission product gamma ray spectra from fission, 2) making MCNP generated neutron and gamma ray spectra.

For the fission spectrum, the neutron spectrum is approximately by a Watt fission energy spectrum. [6]

$$0.4527 \times e^{-E/0.965} \times \sinh(2.29E)^{1/2} \ (E; MeV) \qquad (1)$$

The prompt gamma ray spectrum is approximated in photons/fission/MeV by the segmented fit shown below.

$$\begin{array}{ll} 6.6 & 0.1 < \mathrm{E} < 0.6\mathrm{MeV} \\ 20.2 \times e^{-1.78} & 0.6 < \mathrm{E} < 1.5\mathrm{MeV} \\ 7.2 \times e^{-1.09} & 1.5 < \mathrm{E} < 10.5\mathrm{MeV} \end{array} \qquad (2)$$

And, the shape of the fission product spectrum is as follows with E being the energy in MeV. [7]

$$7.4 \times e^{-1.1E} \quad photons / fission / Mev \tag{3}$$

With the spectra calculated from MCNP, doses from neutrons, neutron induced secondary gamma rays, and prompt gamma rays are calcu1ated. However, spectrum for the fission product gamma rays cannot be calculated from MCNP. It is conservative to assume that the dose from the fission product gamma rays is half of the prompt gamma ray dose.

The assumptions for the basic model were that a point source was positioned at the center of a sphere, and the dose in rem at 100m was calculated. Total number of fissions was 1.0×10^{18}.

Using conversion factors of ANSI/ANS-6.1.1-77, the dose results in rem for fission spectrum and MCNP generated spectrum are shown in Table 1. The neutron dose from Eq. (1) is three times higher than that of MCNP generated spectrum. This is due to conservative point source modeling that assumed that the neutron leakage fraction is 100% (for MCNP spectrum, this fraction is 44.875%).

Using conversion factors in ANSI/ANS-6.1.1-77 and ANSI/ANS-6.1.1-91, the dose results in rem for fission spectrum are shown in Table 2. The results indicate that neutron dose calculated from ANSI/ANS-6.1.1-91 is about half of the dose calculated from ANSI/ANS-6.1.1-77. The gamma ray dose values from ANSI/ANS-6.1.1-91 are about 30% less than those calculated from ANSI/ANS-6.1.1-77. This may be due to the difference between the whole body exposure and frontal exposure.

The basic model using Watt spectrum was run using the conversion factors of ANSI/ANS-6.1.1-77, ANSI/ANS-6.1.1-91, and ICRU-57. Table 3 compares these results. The results indicate that the neutron dose in tissue (in rad) calculated from ANSI/ANS-6.1.1-91 and ICRU-57 are about 30% of the dose calculated from ANSI/ANS-6.1.1-77. This may be also due to the difference between the whole body exposure and frontal exposure.

And the basic model was run to calculate neutron dose in rad-in-air for F5 option of MCNP and KERMA factors from ICRU-44, (Table 4) and also was run to calculate gamma ray dose in rad-in-air for F5 option and KERMA factors from ICRP-74. (Table 5)

4. Conclusion

From the results of the neutron dose and gamma ray dose in rem-in-tissue calculated from ANSI/ANS-6.1.1-77 and ANSI/ANS-6.1.1-91, for exposure to personnel, shield designer may use ANSI/ANS-6.1.1-77 for higher dose prediction.

Neutron and gamma dose in rad-in-air calculated by using F5 option in MCNP were very close to those of ICRU-44 and ICRP-74 conversion factors were used, respectively. It is recommended that ICRU-44 and ICRP-74 be used to check the accuracy of the dose/dose rate by MCNP where F5 option is used.

References

[1] ANSI/ANS-6.1.1-1977, "Neutron and Gamma-Ray Flux-to-Dose Factors," 1977.

[2] ANSI/ANS-6.1.1-1991, "Neutron and Gamma-Ray Fluence-to-Dose Factors," August 1991.

[3] ICRU Publication 57, "Conversion Coefficients for Use in Radiological Protection against External Radiation," 1998

[4] ICRU Report 44, "Tissue Substitutes in Radiation Dosimetry and Measurement," 1989

[5] ICRP Publication 74, "Conversion Coefficients for Use in Radiological Protection against External Radiation," 1997

[6] H.C. Paxton and N.L. Pruvost, "Critical Dimensions of System Containing U-235, Pu-239, and U-233," LA-10860-MS, LANL, July 1987

[7] N.M. Schaeffer, Reactor Shielding for Nuclear Engineers," TID-25952, U.S. Atomic Energy Commission Office of Information Services, May 1973

Table 1. Dose results for two methods (rem)

Type	Using Fission Spectrum	Using MCNP Generated Spectrum
Neutron	113.25±0.0014	33.237±0.0025
Secondary Gamma Ray	0.028±0.0083	0.0345±0.0053
Prompt Gamma Ray	2.032±0.0013	1.9728±0.0020
Fission Product Gamma Ray	0.855±0.0017	0.9864

Table 2. Dose results for ANSI/ANS-6.1.1 (rem)

Type	ANSI/ANS-6.1.1 -1977	ANSI/ANS-6.1.1 -1991
Neutron	113.25±0.0014	55.743±0.0013
Secondary Gamma Ray	0.028±0.0083	0.024±0.0085
Prompt Gamma Ray	2.032±0.0013	1.467±0.0012
Fission Product Gamma Ray	0.855±0.0017	0.615±0.0012

Table 3. Dose results in rad-in-tissue

Type	ANSI/ANS-6.1.1 -1977	ANSI/ANS-6.1.1 -1991	ICRU-57
Neutron	16.908±0.0019	5.843±0.0013	6.866±0.0013

Table 4. Dose results in rad-in-air

Type	F5 Option	ICRU-44
Neutron	0.40707±0.0024	0.42354±0.0024

Table 5. Dose results in rad-in-air

Type	F5 Option	ICRP-74
Gamma Ray	1.4293±0.0022	1.4325±0.0022