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《Original》

Calculation of Neutron and Gamma-Ray Flux-to-Dose-Rate Conversion Factors

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Abstract

This paper presentss flux-to-dose conversion factors for neutrons and gamma-rays based on the concept of the maximum absorbed dose.

Neutron flux-to-does-rate conversion factors for energies from 2.5×10^{-8} to 20 MeV are presented while the conversion factors for gamma-rays are given in the energy range of 0.01 to 15MeV.

Flux-to-does-rate conversion factors, which were calculated under the assumption that the radiation energy distribution has nonlinearity in phantom, are different from those values obtained by monoenergetic radiation. Especially, these values obtained here were determined for the cross section libray such as DLC-23, DLC-27, and DLC-31.

The flux-to-dose-rate conversion factors obtained in this work are in a good agreement with the values presented by American National Standard Institute (ANSI) N666.

These results are used to calculate the dose rate distribution of neutron and gamma-ray in any radiation fields, and will be useful for the radiation shielding analysis, radiation protection and raadiation dosimetry concerned with problems of continuous energy distribution.

요 약

중성자 및 감마선에 대한 선량율 환산인자(flux-to-dose-rate conversion factors)를 최대흡수선량 개념을 근거로 하여 계산하였다.

중성자 및 감자선에 대한 선량율 환산인자는 에너지 범위가 각각 2.5×10^{-8}–20MeV 및 0.01–15 MeV에 대하여 계산하였다. 이제까지 선량율 환산인자는 단일에너지에 대한 값이 였었는데 본 연구에서는 유사인체조직(phantom)내에서 방사선의 에너지 분포가 직선적이 아니라고 가정하여 계산되었다. 특히 DLC-23, DLC-27, DLC-31 등 핵정수 자료의 각 군에 적합한 선량율 환산인자를 결정하였다는 점이 특색이다.

결과적으로 ANSI N666에 있는 값과 본 연구에서 계산된 값이 잘 일치된다는 것을 확인하였고, 본 결과는 어떤 방사선장에서도 중성자나 감마선 선량율 분포를 계산하는데 이용될 수 있고, 방사선 차폐해석, 방사선방어, radiation dosimetry 등에 필요한 값이 될 것이다.

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I. Introduction

The response function or flux-to-dose-rate conversion factors based on the concept of the maximum absorbed does are calculated. It is assumed that the important parameter is the maximum dose rate in tissue irradiated by a normal incidence beam. The conversion factors are used for converting a calculated neutron and gamma-ray flux to a dose rate in the radiation shielding analysis, radiation protection and radiation dosimetry.

The flux-to-dose-rate conversion factors used for neutron flux of a particular energy is the maximum dose rate in the dose-rate distribution calculated by Snyder and Neufeld[1] for a unit flux (or current) of monoenergetic neutrons that are incident normally on the phantom represented by a slab of tissue. Although these values have been used for many years in nuclear reactor and similar design work, the International Commission on Radiation Unit and Measurements gives no definitions which fit the dose or dose rate calculated with them. The Snyder-Neufeld conversion factors are stipulated for use by the U.S. Federal Register[2] for use in reactor design.

Jones[3] measured the dose delivered by gamma-rays between 0.027 and 1.25MeV to various parts of a manlike phantom and later reported the results in terms of calibration factors[4] for dosimeters. These factors are recommended for use in the design and calibration of gamma-ray exposure meters to permit direct estimation of the maximum dose in rads delivered to the critical organ for the ambient condition and hazard under consideration. Recently Claiborne and Trubey[5], using the discrete ordinates code ANISN[6] and the Monte Carlo code OGRE[7], calculated flux-to-dose-rate conversion factors for gamma-rays with energies between 0.01 and 16MeV incident on a slab phantom.

In the past, air kerma and sometimes the tissue kerma in a free field as calculated by Henderson[8] have been used in design work. It must be pointed out that use of these response functions yields a lower value than the maximum does that could be delivered to a body since multiple collisions are not considered. In addition, the practice is inconsistent with that adopted for neutrons.

Flux-to-dose-rate conversion factors calculated in this work are for the cross section library, such as DLC-23 (CASK, 22 neutrons+18 gammas)[9], DLC-27 (104 neutrons+22gammas)[10], and DLC-31 (37 neutrons+21 gammas)[11] etc.

II. Theoretical Background

1. Neutron Flux-to-Dose-Rate Conversion Factors

The neutron flux-to-dose-rate conversion factors presented in this work are based on recommendations of the National Council on Radiation Protection and Measurements (NCRP)[12] for neutron energies between 2.5×10^{-8} and 400MeV. The data given in NCRP are expressed in terms of the neutron flux density ($cm^{-2}\cdot sec^{-1}$), at selected energies, which in a 40-hr period results in a maximum dose equivalent of 100mrem. The NCRP recommendations were based on calculations of absorbed dose and dose equivalent, as a function of depth, in cylinderical and slab phantoms. The flux densities quoted were derived using values of maximum dose equivalent per unit neutron fluence obtained from the discrete ordinates or Monte Carlo calculation. In the ANSI N666[13], only the energy range between 2.5×10^{-8} and 20 MeV was considered, and flux-to-dose-rate conversion factors for neutron energies between the values given in Table 1.

Table 1. Neutron Flux-to-Dose-Rate Conversion Factors and Mean Quality Factors (QF)

Neutron Energy (MeV)	QF*	(Rem/h)/ (Neut/cm²-s)
2.5−08⁺	2	3.67−06
1.0−07	2	3.67−06
1.0−06	2	4.46−06
1.0−05	2	4.54−06
1.0−04	2	4.18−06
1.0−03	2	3.76−06
1.0−02	2.5	3.56−06
1.0−01	7.6	2.17−05
5.0−01	11	9.26−05
1.0	11	1.32−04
2.5	9	1.25−04
5.0	8	1.56−04
7.0	7	1.47−04
10.0	6.5	1.47−04
14.0	7.5	2.08−04
20.0	8	2.27−04

* Maximum value of QF in a 30-cm phantom.
⁺ Read as 2.5×10^{-8}.

Conversion factors for neutron energies not given in Table 1 should be compute using the analytic equation given below. This analytic form is also useful for the calculation of energy-averaged dose rates. In the ANSI N666, for the energy range from 10^{-7} to 10^{-2} MeV, a cubic fit was made which reproduced the error range to within ±3%, and for all other energies, the data were represented as linear segments between adjacent points. The general form of the analytic function is

$$\ln DF_n = A + Bx + Cx^2 + Dx^3 \quad \cdots\cdots\cdots\cdots (1)$$

where DF_n = flux-to-dose-rate factors, (rem/hr) / (n/cm²·sec)

E = neutron energy in MeV

$x = \ln E$

The coefficients in the polynomial expression are given in Table 2.

For an energy band which spans one of the common energy boundaries, the dose rate is computed as

$$\int_{EL}^{El} DF_n(E)\phi(E)\,dE + \int_{El}^{Eu} DF_n(E)\phi(E)\,dE \quad \cdots\cdots\cdots\cdots (2)$$

where E_L = lower energy boundary,

E_u = upper energy boundary, and

E_l = common ennrgy boundary

2. Gamma-Ray Flux-to-Dose-Rate Conversion Factors.

Except for the point at 0.01MeV, the flux-to-dose-rate factors are based on calculations made by Claiborne and Trubey. The calcul-

Table 2. Polynomial Coefficients For Neutron Flux-to-Dose-Rate Conversion Factors.

Neutron Energy (MeV)	A	B	C	D
2.5−08 to 1.0−07	−1.2514+01			
1.0−07 to 1.0−02	−1.2210+01	1.7165−01	2.6034−02	1.0273−03
0.01 to 0.1	−8.9302	7.8440−01		
0.1 to 0.5	−8.6632	9.0037−01		
0.5 to 1.0	−8.9359	5.0696−01		
1.0 to 2.5	−8.9359	−5.5979−02		
2.5 to 5.0	−9.2822	3.2193−01		
5.0 to 7.0	−8.4741	−1.8018−01		
7.0 to 10.0	−8.8247			
10.0 to 14.0	−1.1208+01	1.0352		
14.0 to 20.0	−9.1202	2.4395−01		

ational procedure used is briefly as follows. Multigroup calculations, with the discrete ordinates code ANISN and Monte Carlo code OGRE, are made to determine the dose-rate distributions in a 30cm-thick slab phantom for photon energies between 0.02 and 16 MeV. Using these data, the maximum gamma-ray dose rates in the dose-rate spatial distributions for various incident photon energies were obtained. In all cases, these maxima occurred within the first 2cm of the phantom. These maximum dose rates were the ones recommened by Claiborne and Trubey as a standard response function in the design of shields for reactors and similar radiation sources.

The conversion factors were extended to 0.01MeV using the tissue kerma factors calculated by Wells and Liversay[14] for 10 KeV x-rays. At this energy, the difference between the maximum dose in tissue and the tissue kerma is negligible. The recommended flux-to-dose-rate factors for various photon energies are tabulated in Table 3.

Conversion factors for photon energies not given in Table 3 should be computed using the analytic equation given below. This analytic form is also useful for the calculation of energy-averaged dose rates. The general form of the analytic function is

$$\ln DF_g = A + Bx + Cx^2 + Dx^3 \quad \cdots\cdots\cdots\cdots (3)$$

where DF_g = flux-to-dose-rate conversion factor (rem/hr)/(photons/cm^2·sec)

E = gamma-ray energy in MeV

$x = \ln E$

The coefficients of the polynomial expression are given in Table 4. For an energy band which spans one of the common energy boundaries, the does rate is computed in the same way as in the neutron case.

Table 3. Gamma-ray Flux-to-Dose-Rate Conversion Factors.

Photon Energy (MeV)	(rem/h)/(Photon/cm^2-s)	mrem/h/(MeV/cm^2-sec)
0.01	3.96−06	3.96−1
0.03	5.82−07	1.94−2
0.05	2.90−07	5.80−3
0.07	2.58−07	3.69−3
0.1	2.83−07	2.83−3
0.15	3.79−07	2.53−3
0.2	5.01−07	2.51−3
0.25	6.31−07	2.52−3
0.3	7.59−07	2.53−3
0.35	8.78−07	2.51−3
0.4	9.85−07	2.46−3
0.45	1.08−06	2.40−3
0.5	1.17−06	2.34−3
0.55	1.27−06	2.31−3
0.6	1.36−06	2.27−3
0.65	1,44−06	2.22−3
0.7	1.52−06	2.17−3
0.8	1.68−06	2.10−3
1.0	1.98−06	1.98−3
1.4	2.51−06	1.79−3
1.8	2.99−06	1.66−3
2.2	3.42−06	1.55−3
2.6	3.82−06	1.47−3
2.8	4.01−06	1.43−3
3.25	4.41−06	1.36−3
3.75	4.83−06	1.29−3
4.25	5.23−06	1.23−3
4.75	5.60−06	1.18−3
5.0	5.80−06	1.16−3
5.25	6.01−06	1.14−3
5.75	6.38−06	1.11−3
6.25	6.74−06	1.08−3
6.75	7.11−06	1.05−3
7.5	7.66−06	1.02−3
9.0	8.77−06	9.74−4
11.0	1.03−05	9.36−4
13.0	1.18−05	9.08−4
15.0	1.33−05	8.87−4

Ⅱ. Calculation of The Flux-to-Dose-Rate Conversion Factors for the Continuous Energy Distribution.

The flux-to-dose-rate conversion factors

Table 4. Polynomial Coefficients For Gamma-Ray Flux-to-Dose-Rate Conversion Factors.

Photon Energy (MeV)	A	B	C	D
0.01 to 0.03	−2.0477+01	−1.7454		
0.03 to 0.5	−1.3626+01	−5.7117−01	−1.0954	−2.4897−01
0.5 to 5.0	−1.3133+01	7.2008−01	−3.3603−02	
5.0 to 15.0	−1.2791+01	2.8309−01	1.0873−01	

presented here are calculated, under the assumption that radiation energy distribution has nonlinearity in the phantom and have different meaning from those values obtained by monoenergetic radiation. These values are determined for the cross section library used in radiation shielding analysis, radiation protection, radiation dosimetry etc.

The assumption is made that the important parameter is the maximum dose rate in tissue irradiated by a beam having normal incidence. This assumption gives results which are conservative for assumed whole-body exposures.

The calculational procedure used is as follows. The flux spectrum inside each group span is assumed to be $\frac{1}{E}$ for neutrons which energies are greater than 0.414 eV, and to be constant for thermal neutrons and gamma-rays.

$$\text{Dose rate}=\int_E DF(E)\cdot\phi(E)\,dE \cdots\cdots\cdots (4)$$

where $DF(E)=e^{a+bx+cx2+dx3}$

$x=\ln E$

For neutron energy $(E)>0.414\text{eV}$

$$\phi(E)\propto\frac{1}{E}(E_g{}^L\sim E_g{}^U) \cdots\cdots\cdots (5)$$

then

$$\text{Dose rate}=\sum_g DF_g\cdot\phi_g \cdots\cdots\cdots (6)$$

where

$$DF_g=\int_{Eg}DF(E)\cdot\frac{1}{E}dE\Big/\int_{Eg}\frac{1}{E}dE$$

$$=\left[\ln\left(\frac{E_g{}^U}{E_g{}^L}\right)\right]^{-1}\cdot$$

$$\int_{Eg}DF(E)\cdot\frac{1}{E}dE \cdots\cdots\cdots (7)$$

For neutron energy $(E)\leq 0.414\text{eV}$ and gamma-ray $\phi(E)$ is assumed to be constant in each group.

then

$$\text{Dose rate}=\sum_g DF_g\cdot\phi_g \cdots\cdots\cdots (8)$$

where

$$DF_g=\int_{Eg}DF(E)\,dE \cdots\cdots\cdots (9)$$

In equation (7);

let

$$I_g=\int_{Eg}DF(E)\frac{1}{E}dE$$

$$=\int_{Eg}e^{a+bx+cx2+dx3}\cdot\frac{1}{E}dE$$

$$=\int_{Eg}e^{a+bx+cx2+dx3}\cdot dx \cdots\cdots\cdots (10)$$

If both c and d are nonzero, an analytic solution is obtained.

$$I_g=e^a\cdot\frac{1}{b}(e^{bx_g^u}-e^{bx_g^L}) \cdots\cdots\cdots (11)$$

(for $b\neq 0$)

$$I_g=e^a(x_g{}^u-x_g{}^L) \cdots\cdots\cdots (12)$$

(for $b=0$)

If either c and d is nonzero, a numerical integration is required.

1) Divide X_g into N subintervals (N=even)
2) Apply Simpson's formular.

$$I_g=\frac{h}{3}(f_0+4f_1+2f_2+4f_3+2f_4+\cdots\cdots$$

$$+2f_{N-2}+4f_{N-1}+f_N)$$

$$=\frac{h}{3}(f_0+f_N+2\text{Even}+4\text{Odd}) \cdots\cdots (13)$$

where

f=integrand, $h=X_g/N$

$\text{Even}=f_2+f_4+\cdots\cdots+f_{N-2}$

$\text{Odd}=f_1+f_3+\cdots\cdots+f_{N-1}$

3) Divide X_g into $2N$ subintervals

4) Apply Simpson's formular again.

$$I'_g=\frac{h'}{3}(f'_0+4f'_1+2f'_2+4f'_3+\cdots\cdots$$
$$+2f'_{2N-2}+4f'_{2N-1}+f'_{2N})$$
$$=\frac{h'}{3}(f'_0+f'_{2N}+2\text{Even}'+4\text{Odd}')\cdots(14)$$

where

$$h'=\frac{1}{2}h,\quad f'_0=f_0,\quad f'_{2N}=f_N$$
$$\text{Even}'=f'_2+f'_4+\cdots\cdots+f'_{2N-2}$$
$$=f_1+f_2+f_3+\cdots\cdots+f_{N-1}$$
$$=\text{Even}+\text{Odd}$$
$$\text{Odd}'=f'_1+f'_3+f'_5+\cdots\cdots+f'_{2N-1}$$

5) Test convergence by comparing I_g/I'_g.

If not, divide x_g into $4N$ subintervals and repeat 3), 4) and 5) until convergence is reached.

In equation (9);

let

$$I_g=\int_{Eg}DF(E)dE=\int_{Eg}e^{a+bx+cx2+dx3}\cdot e^x dx$$
$$={}_{Eg}e^{a+b'x+cx2+dx3}\cdot dx \cdots\cdots\cdots\cdots\cdots\cdots\cdots(15)$$

where

$$b'=b+1$$

The same integration formular is applied. Those equation are calculated using computer and the results are shown in Table 5, 6 and 7.

Table 5. The Group Dose Rate Conversion Factors Based on Ansi-N666 For Cask 22 Neutron +18 Gamma (DLC-23) Group Structure.

Group	Upper Energy (MeV)	Lower Energy (MeV)	Conversion Factors (rem/h/Unit Flux)
1	1.490E 01	1.220E 01	1.991E −04
2	1.220E 01	1.000E 01	1.633E −04
3	1.000E 01	8.180E 00	1.471E −04
4	8.180E 00	6.360E 00	1.475E −04
5	6.360E 00	4.960E 00	1.530E −04
6	4.960E 00	4.060E 00	1.509E −04
7	4.060E 00	3.010E 00	1.393E −04
8	3.010E 00	2.460E 00	1.285E −04
9	2.460E 00	2.350E 00	1.253E −04
10	2.350E 00	1.830E 00	1.263E −04
11	1.830E 00	1.110E 00	1.290E −04
12	1.110E 00	5.500E −01	1.161E −04
13	5.500E −01	1.110E −01	5.334E −05
14	1.110E −01	3.350E −03	8.412E −06
15	3.350E −03	5.830E −04	3.713E −06
16	5.830E −04	1.010E −04	4.009E −06
17	1.010E −04	2.900E −05	4.295E −06
18	2.900E −05	1.010E −05	4.476E −06
19	1.010E −05	3.060E −06	4.567E −06
20	3.060E −06	1.120E −06	4.535E −06
21	1.120E −06	4.140E −07	4.370E −06
22	4.140E −07	1.000E −08	3.961E −06
23	1.000E 01	8.000E 00	8.772E −06
24	8.000E 00	6.500E 00	7.478E −06
25	6.500E 00	5.000E 00	6.375E −06
26	5.000E 00	4.000E 00	5.414E −06
27	4.000E 00	3.000E 00	4.622E −06
28	3.000E 00	2.500E 00	3.960E −06
29	2.500E 00	2.000E 00	3.469E −06
30	2.000E 00	1.660E 00	3.019E −06
31	1.660E 00	1.330E 00	2.628E −06
32	1.330E 00	1.000E 00	2.205E −06
33	1.000E 00	8.000E −01	1.833E −06
34	8.000E −01	6.000E −01	1.523E −06
35	6.000E −01	4.000E −01	1.185E −06
36	4.000E −01	3.000E −01	9.114E −07
37	3.000E −01	2.000E −01	6.757E −07
38	2.000E −01	1.000E −01	4.364E −07
39	1.000E −01	5.000E −02	3.418E −07
40	5.000E −02	1.000E −02	1.025E −06

Table 6. The Group Dose Rate Conversion Factors Based on Ansi-N666 For Dlc-27 104 Neutron+22 Gamma Group Structure.

Group	Upper Energy (MeV)	Lower Energy (MeV)	Conversion factors (rem/h/Unit Flux)
1	1.500E 01	1.350E 01	2.082E −04
2	1.350E 01	1.221E 01	1.906E −04
3	1.221E 01	1.105E 01	1.718E −04
4	1.105E 01	1.000E 01	1.550E −04
5	1.000E 01	9.048E 00	1.471E −04
6	9.048E 00	8.187E 00	1.471E −04
7	8.187E 00	7.408E 00	1.471E −04
8	7.408E 00	7.000E 00	1.471E −04
9	7.000E 00	6.708E 00	1.476E −04
10	6.708E 00	6.360E 00	1.489E −04
11	6.360E 00	6.065E 00	1.503E −04

12	6.065E 00	5.488E 00	1.523E −04	61	1.930E −02	1.503E −02	5.432E −06
13	5.488E 00	4.966E 00	1.550E −04	62	1.503E −02	1.171E −02	4.465E −06
14	4.966E 00	4.750E 00	1.548E −04	63	1.171E −02	9.119E −03	3.709E −06
15	4.750E 00	4.493E 00	1.523E −04	64	9.119E −03	7.102E −03	3.554E −06
16	4.493E 00	4.066E 00	1.486E −04	65	7.102E −03	5.531E −03	3.563E −06
17	4.066E 00	3.679E 00	1.439E −04	66	5.531E −03	4.307E −03	3.577E −06
18	3.679E 00	3.329E 00	1.393E −04	67	4.307E −03	3.355E −03	3.596E −06
19	3.329E 00	3.012E 00	1.349E −04	68	3.355E −03	2.613E −03	3.618E −06
20	3.012E 00	2.725E 00	1.306E −04	69	2.613E −03	2.035E −03	3.644E −06
21	2.725E 00	2.466E 00	1.265E −04	70	2.035E −03	1.585E −03	3.674E −06
22	2.466E 00	2.350E 00	1.253E −04	71	1.585E −03	1.234E −03	3.707E −06
23	2.350E 00	2.231E 00	1.256E −04	72	1.234E −03	9.611E −04	3.743E −06
24	2.231E 00	2.019E 00	1.262E −04	73	9.611E −04	7.485E −04	3.782E −06
25	2.019E 00	1.827E 00	1.269E −04	74	7.485E −04	5.829E −04	3.824E −06
26	1.827E 00	1.653E 00	1.276E −04	75	5.829E −04	4.540E −04	3.867E −06
27	1.653E 00	1.496E 00	1.283E −04	76	4.540E −04	3.536E −04	3.912E −06
28	1.496E 00	1.353E 00	1.290E −04	77	3.536E −04	2.754E −04	3.959E −06
29	1.353E 00	1.225E 00	1.297E −04	78	2.754E −04	2.145E −04	4.007E −06
30	1.225E 00	1.108E 00	1.305E −04	79	2.145E −04	1.670E −04	4.055E −06
31	1.108E 00	1.003E 00	1.312E −04	80	1.670E −04	1.301E −04	4.104E −06
32	1.003E 00	9.072E −01	1.285E −04	81	1.301E −04	1.013E −04	4.153E −06
33	9.072E −01	8.209E −01	1.221E −04	82	1.013E −04	7.889E −05	4.202E −06
34	8.209E −01	7.427E −01	1.161E −04	83	7.889E −05	6.144E −05	4.250E −06
35	7.427E −01	6.721E −01	1.103E −04	84	6.144E −05	4.785E −05	4.296E −06
36	6.721E −01	6.081E −01	1.049E −04	85	4.785E −05	3.727E −05	4.341E −06
37	6.081E −01	5.502E −01	9.970E −05	86	3.727E −05	2.902E −05	4.383E −06
38	5.502E −01	4.979E −01	9.477E −05	87	2.902E −05	2.260E −05	4.422E −06
39	4.979E −01	4.505E −01	8.821E −05	88	2.260E −05	1.760E −05	4.459E −06
40	4.505E −01	4.076E −01	8.061E −05	89	1.760E −05	1.371E −05	4.491E −06
41	4.076E −01	3.688E −01	7.367E −05	90	1.371E −05	1.068E −05	4.519E −06
42	3.688E −01	3.337E −01	6.732E −05	91	1.068E −05	8.315E −06	4.542E −06
43	3.337E −01	3.020E −01	6.153E −05	92	8.315E −06	6.476E −06	4.560E −06
44	3.020E −01	2.732E −01	5.623E −05	93	6.476E −06	5.043E −06	4.572E −06
45	2.732E −01	2.472E −01	5.139E −05	94	5.043E −06	3.928E −06	4.578E −06
46	2.472E −01	2.237E −01	4.696E −05	95	3.928E −06	3.059E −06	4.577E −06
47	2.237E −01	2.024E −01	4.292E −05	96	3.059E −06	2.382E −06	4.568E −06
48	2.024E −01	1.832E −01	3.923E −05	97	2.382E −06	1.855E −06	4.552E −06
49	1.832E −01	1.657E −01	3.585E −05	98	1.855E −06	1.445E −06	4.528E −06
50	1.657E −01	1.500E −01	3.277E −05	99	1.445E −06	1.125E −06	4.495E −06
51	1.500E −01	1.357E −01	2.995E −05	100	1.125E −06	8.864E −07	4.455E −06
52	1.357E −01	1.228E −01	2.737E −05	101	8.864E −07	6.826E −07	4.406E −06
53	1.228E −01	1.111E −01	2.501E −05	102	6.826E −07	5.316E −07	4.346E −06
54	1.111E −01	8.652E −02	2.150E −05	103	5.316E −07	4.140E −07	4.278E −06
55	8.652E −02	6.738E −02	1.762E −05	104	4.140E −07	1.000E −10	3.954E −06
56	6.738E −02	5.248E −02	1.448E −05	105	1.000E 00	8.000E 00	8.772E −06
57	5.248E −02	4.087E −02	1.191E −05	106	8.000E 00	6.500E 00	7.478E −06
58	4.087E −02	3.183E −02	9.785E −06	107	6.500E 00	5.000E 00	6.375E −06
59	3.183E −02	2.479E −02	8.043E −06	108	5.000E 00	4.000E 00	5.414E −06
60	2.479E −02	1.930E −02	6.610E −06	109	4.000E 00	3.000E 00	4.622E −06

110	3.000E 00	2.500E 00	3.960E −06
111	2.500E 00	2.000E 00	3.469E −06
112	2.000E 00	1.660E 00	3.019E −06
113	1.660E 00	1.330E 00	2.628E −06
114	1.330E 00	1.000E 00	2.205E −06
115	1.000E 00	8.000E −01	1.833E −06
116	8.000E −01	6.000E −01	1.523E −06
117	6.000E −01	5.200E −01	1.289E −06
118	5.200E −01	5.000E −01	1.200E −06
119	5.000E −01	4.000E −01	1.099E −06
120	4.000E −01	3.000E −01	9.114E −07
121	3.000E −01	2.600E −01	7.507E −07
122	2.600E −01	2.500E −01	6.887E −07
123	2.500E −01	2.000E −01	6.131E −07
124	2.000E −01	1.000E −01	4.364E −07
125	1.000E −01	5.000E −02	3.418E −07
126	5.000E −02	2.000E −02	6.516E −07

Table 7. The Group Dose Rate Conversion Factors Based On Asin-N666 For Dlc-31 37 Neutron+21 Gamma Group Structure.

Group	Upper Energy (MeV)	Lower Energy (MeV)	Conversion Factors (rem/h/Unit Flux)
1	1.960E 01	1.690E 01	2.221E −04
2	1.690E 01	1.490E 01	2.148E −04
3	1.490E 01	1.420E 01	2.103E −04
4	1.420E 01	1.380E 01	2.078E −04
5	1.380E 01	1.280E 01	1.975E −04
6	1.280E 01	1.220E 01	1.853E −04
7	1.220E 01	1.110E 01	1.722E −04
8	1.110E 01	1.000E 01	1.553E −04
9	1.000E 01	9.050E 00	1.471E −04
10	9.050E 00	8.190E 00	1.471E −04
11	8.190E 00	7.410E 00	1.471E −04
12	7.410E 00	6.380E 00	1.478E −04
13	6.380E 00	4.970E 00	1.529E −04
14	4.970E 00	4.720E 00	1.547E −04
15	4.720E 00	4.070E 00	1.498E −04
16	4.070E 00	3.010E 00	1.394E −04
17	3.010E 00	2.390E 00	1.281E −04
18	2.390E 00	2.310E 00	1.254E −04
19	2.310E 00	1.830E 00	1.264E −04
20	1.830E 00	1.110E 00	1.290E −04
21	1.110E 00	5.500E −01	1.161E −04
22	5.500E −01	1.580E −01	6.047E −05
23	1.580E −01	1.110E −01	2.811E −05
24	1.110E −01	5.250E −02	1.786E −05
25	5.250E −02	2.480E −02	9.914E −06
26	2.480E −02	2.190E −02	6.938E −06
27	2.190E −02	1.030E −02	4.986E −06
28	1.030E −02	3.350E −03	3.572E −06
29	3.350E −03	1.230E −03	3.661E −06
30	1.230E −03	5.830E −04	3.783E −06
31	5.830E −04	1.010E −04	4.009E −06
32	1.010E −04	2.900E −05	4.295E −06
33	2.900E −05	1.070E −05	4.473E −06
34	1.070E −05	3.060E −06	4.566E −06
35	3.060E −06	1.130E −06	4.536E −06
36	1.130E −06	4.140E −07	4.371E −06
37	4.140E −07	1.000E −11	3.954E −06
38	1.400E 01	1.000E 01	1.102E −05
39	1.000E 01	8.000E 00	8.772E −06
40	8.000E 00	7.000E 00	7.663E −06
41	7.000E 00	6.000E 00	6.926E −06
42	6.000E 00	5.000E 00	6.191E −06
43	5.000E 00	4.000E 00	5.414E −06
44	4.000E 00	3.000E 00	4.622E −06
45	3.000E 00	2.500E 00	3.960E −06
46	2.500E 00	2.000E 00	3.469E −06
47	2.000E 00	1.500E 00	2.927E −06
48	1.500E 00	1.000E 00	2.316E −06
49	1.000E 00	7.000E −01	1.766E −06
50	7.000E −01	4.500E −01	1.313E −06
51	4.500E −01	3.000E −01	9.609E −07
52	3.000E −01	1.500E −01	1.137E −07
53	1.500E −01	1.000E −01	3.829E −07
54	1.000E −01	7.000E −02	3.319E −07
55	7.000E −02	4.500E −02	3.699E −07
56	4.500E −02	3.000E −02	6.112E −07
57	3.000E −02	2.000E −02	8.267E −07
58	2.000E −02	1.000E −02	2.144E −06

IV. Conclusions

1) The flux-to-does-rate conversion factors have been calculated using the concept of kerma and maximum absorbed dose.
2) Differences between the ANSI N666 values and the values obtained here are as follows.
   (1) For all energies in the ANSI N666, the conversion factors were represented as linear segments between adjacent points.

(2) In this work, the flux spectra inside each group span are assumed to be $\frac{1}{E}$ for neutrons of which energies are greater than. 0.414eV, and to be constant for thermal neutrons and gamma-rays.

3) These data will be useful for the radiation shielding analysis, radiation protection, and radiation dosimetry etc. for the continuous energy distribution.

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