CHARACTERIZATION OF A RADIATION DETECTOR FOR AIRCRAFT MEASUREMENTS

Leonardo de Holanda Mencarini[1,2], Claudio A. Federico[1,2] and Linda V. E. Caldas[1]

[1] Instituto de Pesquisas Energéticas e Nucleares – IPEN, Av. Lineu Prestes 2242 – Cidade Universitária – CEP 05508-000, São Paulo – SP – Brasil. (lcaldas@ipen.br)

[2] Instituto de Estudos Avançados – IEAv, Trevo Coronel Aviador José Alberto Albano do Amarante, no 1 – Putim, CEP - 12.228-001, São José dos Campos – SP – Brasil.

(mencarini@ieav.cta.br, Claudio.federico.40@gmail.com)

Abstract

Aircrews, as pilots and flight attendants, are subjected to cosmic ray doses which can be higher than the average doses on workers from the nuclear industry. The diversity of particles of high energies present in the radiation field on board of aircrafts turns the determination of the incident dose difficult, and requires special care regarding dosimetric systems to be used in this kind of radiation field. The Brazilian Air Force, through its Institute for Advanced Studies (Instituto de Estudos Avançados, IEAv/DCTA) in conjunction with the Institute of Energetic and Nuclear Research (Instituto de Pesquisas Energéticas e Nucleares – IPEN/CNEN-SP) are working on this subject since 2008. A prototype of a radiation detector for aircraft measurements was previously built and tested in flight and laboratory conditions. The detector is able of measuring a quantity known as absorbed dose (using passive dosimeters), which will subsequently be correlated to the ambient dose equivalent and the effective dose received by aircrews. In this context, a theoretical approach through Monte Carlo simulations with the computational codes MCNP5 and MCNPX was used to model and characterize the detector response at such experimental conditions. This work presents the preliminary results of the computational modeling, with special emphasis on the comparison between the absorbed doses measured and simulated, and its relationship with the ambient dose equivalent and the effective dose for this detector.

Keywords: Ambient dose equivalent, MCNPX, passive detector.

1.- INTRODUCTION

Monte Carlo N-Particle eXtended MCNPX 2.7.0 (Pelowitz, 2011) is a powerful Monte Carlo program developed for radiation transport by Los Alamos National Laboratory (LANL) and distributed by the Radiation Safety Information Computational Center (RSICC). After some years of development, it has been applied in many different fields of physics, as radiation protection, medical or reactor physics. Neutrons in an energy range between 10^{-5} MeV and 100 MeV are important for dose assessment in radiation protection of aircrews. Thus, the Brazilian Air Force, through its Institute for Advanced Studies (Instituto de Estudos Avançados, IEAv/DCTA) in conjunction with the Institute of Energetic and Nuclear Research (Instituto de Pesquisas Energéticas e Nucleares – IPEN/CNEN-SP) are working on this subject since 2008. A prototype of a radiation detector for aircraft measurements was previously built and tested in flight and laboratory conditions (Federico, 2011). The detector is able of measuring a quantity known as absorbed dose (using passive dosimeters), which will subsequently be correlated to the ambient dose equivalent and the effective dose received by aircrews. In this context, a theoretical approach through Monte Carlo simulations with the computational code MCNPX was used to model and characterize the detector response at such experimental conditions. The ambient dose equivalent, H*(10), is a weighted radiation dose, taking the quality factor of the particles depositing the energy in matter into account. It was thoroughly described and analyzed in the report ICRU 57 (1998). The MCNPX dose calculation procedure for neutrons was tested by calculating the ambient dose equivalent for neutrons and the ambient dose, D*(10), in the ICRU sphere, and compared with results published earlier (Leuthold *et al*., 1992).

According to the ICRU 57 (1998), the ambient dose equivalent, H*(10), is the dose equivalent at a point in a radiation field that would be produced by the corresponding expanded and aligned field in the ICRU sphere at a depth of 10 mm on the radius vector opposing the direction of the aligned field. According to the definition, the dose is to be calculated at a depth of 1 cm from the upstream surface of a sphere with radius of 15 cm made of tissue equivalent (TE) material. This was represented in the MCNPX code by placing a 30 cm-sphere of TE-material (10.1% H; 11.1%C; 2.6% N and 76.2% O) in the whole volume, which was filled with almost a "vacuum" condition (air with very low density). To score the ambient dose equivalent, a small sphere volume (1 mm diameter) made of TE was placed in 1 cm depth of the sphere, the cross section facing the incoming beam. The neutron beam irradiates the whole sphere homogeneously, parallel to the source-sphere axis (aligned and expanded). The absorbed dose in a 10 mm depth in the same geometry without any quality-weighting factor will be denoted by ambient dose, D*(10).

The objective of this work was to obtain preliminary results of the computational modeling, with special emphasis on the comparison between the absorbed dose and its relationship with the ambient dose equivalent for this detector.

2.- MATERIALS AND METHODS

The materials used in this work were the computational codes (MCNPX and “Star codes”) and a prototype of a radiation detector for aircraft measurements, based on the ICRU sphere (1980), with cavities to insert a pair of thermoluminescent dosimeters (TLD-600 and TLD-700) conveniently distributed in the volume of the sphere, in order to facilitate the assessment of the quantity H*(10) and allow to obtain additional information over the ionizing radiation field, such as its directionality and depth of the estimated maximum absorbed dose. The positioning of the inserts inside the detector is shown as a schematic drawing of Figure 1.

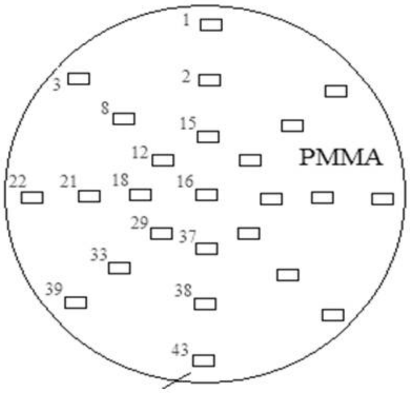

Figure 1. Schematic representation of a passive detector and the cavities to insert thermoluminescent dosimeters (not to scale).

The methodology was the same adopted by Garny *et al* (2009), but in this work, the MCNPX code was used to calculate the energy deposited by particles and ions inside the control volume (sphere with 1 mm diameter), in 20 groups of energy, from 10^{-5} MeV to 201 MeV. At the energy range below 15 MeV the whole energy of a proton or another ion accelerated after the collision with a neutron is deposited in one step instead of in very small steps, which would be the case without the multiple scattering option. Therefore, the tabulated quality factor Q depending on the linear energy transfer (LET), which is given in ICRU 57 (1998), cannot be used pointwise, because the LET varies strongly during the slowing-down of the proton or ion. For this purpose, a mean quality factor, simulating the slowing-down of a proton or ion, was defined. Therefore, the Q (LET) versus LET(E) dependence was integrated from E_{START} to E=0 over all energies below E_{START}:

$$Q_{mean}(E_{start}) = \frac{\int_{E_{start}}^{0} Q(LET(E))dE}{\int_{E_{start}}^{0} dE} \qquad (1)$$

using the LET-relationships from ICRU 49 (1993) (protons and alpha particles) and ICRU 73 (2005) (heavier ions: C, N and O). Including this mean quality factor in the routine, the ambient dose equivalent H*(10) was calculated using the following equation:

$$H^{*}(10) = \sum_{particles} \sum_{steps} Q_{mean}\,(E_{start}) \times D \qquad (2)$$

where $Q_{mean}(E_{start})$ is the mean quality factor of the particle with starting energy E_{start} (1) and D is the absorbed dose deposited by the particle during the relevant step (for protons and heavy ions $D = E_{start}/m$ with m being the mass of the scoring volume).

3.- RESULTS

In order to demonstrate the usefulness of the MCNPX code for neutron calculations, the ambient dose equivalent was calculated for neutrons using the quality factor Q from report ICRP 60 (1991). The results of the MCNPX calculations are compared to those given in ICRP 74 (1997) that are based on MCNP calculations performed by various laboratories.

Two basic quantities to be compared between two neutron transport calculations (ICRU sphere and passive detector) are the ambient dose and the ambient dose equivalent as a function of the neutron energy inside a defined volume, as shown in Table 1, where:

- A : incident neutron energy;
- B : effective quality factor obtained in this work;
- C : absorbed dose per unit neutron fluence obtained in this work;
- D : ambient dose equivalent per unit neutron fluence, by ICRP 74 (ICRP, 1997);
- E : ambient dose equivalent per unit neutron fluence obtained in this work.

Table 1 - Results of simulations involving the ICRU sphere.

A	B	C	D	E
[MeV]	[Sv/Gy]	[pGy×cm^2]	[pSv×cm^2]	[pSv×cm^2]
1.00×10^{-5}	2.43	3.20 ± 0.60	11.3	7.76 ± 1.46
1.00×10^{-4}	2.45	2.69 ± 0.07	9.40	6.58 ± 0.18
1.00×10^{-3}	2.00	3.63 ± 1.06	7.90	7.25 ± 2.00
1.00×10^{-2}	3.48	2.76 ± 0.07	10.5	9.61 ± 0.25
3.00×10^{-2}	5.15	3.92 ± 0.80	23.7	20.2 ± 4.10
5.00×10^{-2}	6.81	5.71 ± 1.20	41.1	38.9 ± 5.81
1.00×10^{-1}	10.2	8.65 ± 1.68	88.0	88.5 ± 7.20
2.00×10^{-1}	15.9	11.2 ± 0.60	170	178 ± 9.77
3.00×10^{-1}	17.3	14.6 ± 0.60	233	253 ± 11.2
5.00×10^{-1}	17.7	18.5 ± 0.10	322	328 ± 1.83
7.00×10^{-1}	19.3	23.1 ± 1.49	375	445 ± 28.8
1.00×10^{0}	18.1	29.6 ± 0.66	416	537 ± 12.0
2.00×10^{0}	15.7	35.5 ± 0.70	420	557 ± 11.7
5.00×10^{0}	6.48	54.7 ± 3.70	405	354 ± 24.0
1.00×10^{1}	6.43	67.9 ± 3.24	440	437 ± 20.8
1.60×10^{1}	7.11	76.3 ± 3.64	555	543 ± 25.9
2.00×10^{1}	7.42	81.8 ± 8.23	600	607 ± 61.0
3.00×10^{1}	4.90	81 ± 13.3	515	453 ± 74.1
1.00×10^{2}	4.01	76.7 ± 5.91	285	307 ± 23.7
2.01×10^{2}	5.35	35.5 ± 5.48	260	190 ± 29.3

In Figure 2, these results are compared with the ICRP 74 values:

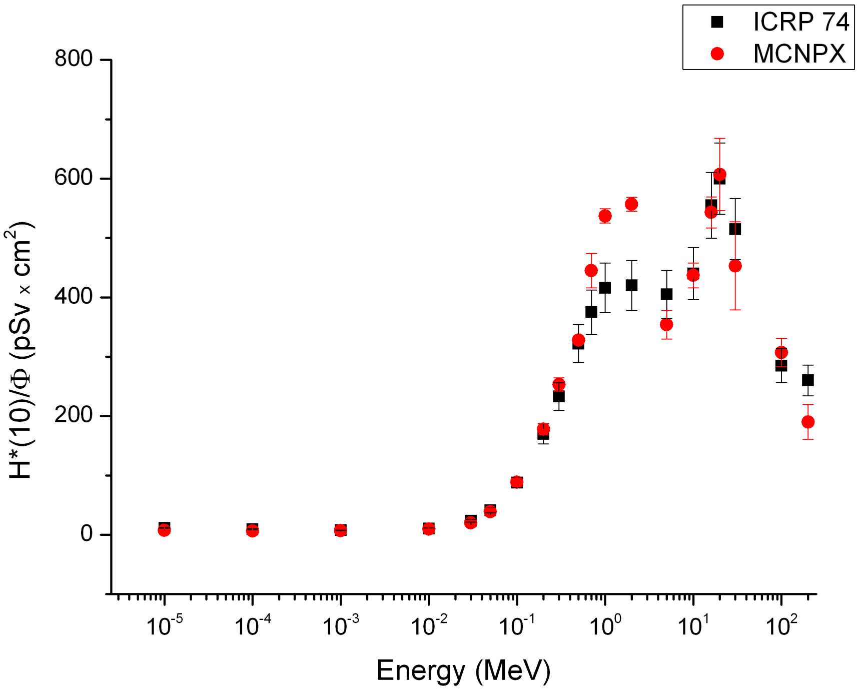

Figure 2. Comparison of the ambient dose equivalent H*(10) per unit neutron fluence obtained in this work and the ICRP 74 values.

The results of the simulations obtained with the passive detector are presented in Table 2, where:

- E : incident neutron energy;
- Qeff : effective quality factor obtained in this work;
- D(10)/ϕ : absorbed dose per unit neutron fluence obtained in this work;
- H(10)/ϕ : ambient dose equivalent per unit neutron fluence obtained in this work.

Table 2 - Results of simulations involving the passive detector

E [MeV]	D(10)/ϕ MCNPX [pGy×cm^2]	H(10)/ϕ MCNPX [pSv×cm^2]	Qeff [Sv/Gy]
1.00×10^{-5}	2.40 ± 0.59	5.21 ± 0.60	2.17
1.00×10^{-4}	2.35 ± 0.73	4.86 ± 0.75	2.07
1.00×10^{-3}	1.21 ± 0.71	3.31 ± 0.72	2.75
1.00×10^{-2}	0.90 ± 0.90	6.15 ± 0.39	6.83
3.00×10^{-2}	4.45 ± 0.78	21.3 ± 0.9	4.79
5.00×10^{-2}	5.12 ± 0.73	40.0 ± 1.3	7.82
1.00×10^{-1}	7.87 ± 0.75	97.1 ± 2.6	12.3
2.00×10^{-1}	11.5 ± 0.6	196 ± 4	17.0
3.00×10^{-1}	15.8 ± 0.5	296 ± 6	18.8
5.00×10^{-1}	21.8 ± 0.6	395 ± 7	18.1
7.00×10^{-1}	24.9 ± 0.5	500 ± 7	20.1
1.00×10^{0}	34 ± 1	664 ± 17	19.5
2.00×10^{0}	42.4 ± 0.9	690 ± 12	16.3
5.00×10^{0}	57.9 ± 1.0	700 ± 12	12.1
1.00×10^{1}	73.6 ± 1.4	613 ± 11	8.33
1.60×10^{1}	92.6 ± 1.5	678 ± 11	7.33
2.00×10^{1}	99.5 ± 2.1	668 ± 13	6.71
3.00×10^{1}	98.0 ±14.8	856 ± 129	8.74
1.00×10^{2}	80.4 ± 7.04	386 ± 33.8	4.81
2.01×10^{2}	48.6 ± 6.86	209 ± 29.5	4.30

A calibration curve was obtained for the results of simulations involving the ICRU sphere and the results of simulations involving the passive detector, from 10 eV to 700 keV, using a linear curve fit in values about dose equivalent for neutrons per unit neutron fluence, as shown in Figure 3.

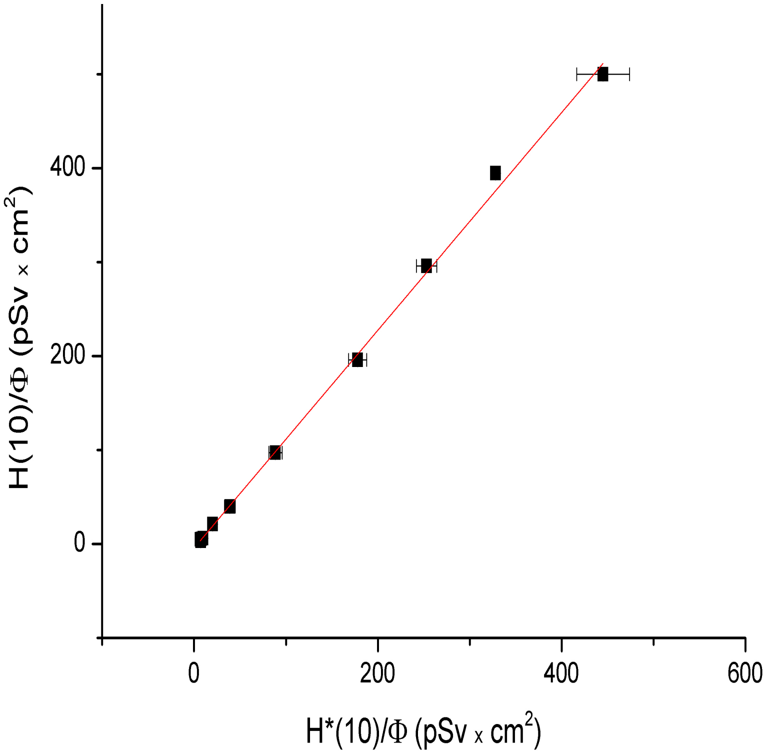

Figure 3. Calibration curve between ICRU sphere and passive detector (Mencarini, 2013).

4.- DISCUSSION

As can be seen in Table 1, the results of the present MCNPX-calculations are in accordance with those obtained by report ICRP 74 (1997). Thus, the methodology adopted by Garny *et al.*(2009) and in this work is valid to this type of problem.

The detector simulation using the MCNPX code shows that both the detector and the ICRU sphere have different peak maxima, thus the possible energy range to calibrate them is from 10 eV to 700 keV, as seen in Tables 1 and 2.

As the correlation coefficient (R^2) of the linear curve fit of Figure 3 is equal to 0.9977, it can be considered that there is a proportionality in terms of dose equivalent in the energy range of 10 eV to 700 keV.

5.- CONCLUSIONS

MCNPX calculations were performed for neutrons energies of 10^{-5} MeV to 201 MeV. The neutron fluence, as the neutron ambient dose and the ambient dose equivalent were calculated as a function of energy and a good agreement with published results was obtained. Therefore, the MCNPX code is able to calculate absorbed doses correctly for the neutron energy range from 10^{-5} MeV to 201 MeV, if the methodology which is used here is applied, both the detector and in ICRU sphere. In addition, it demonstrates that in the MCNPX code the use of a weighting function is possible to convert absorbed dose to any weighted dose, which might be relevant for applications in radiation biology or other fields.

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