REVISTA MEXICANA DE FÍSICA S **53** (3) 1–7 FEBRERO 2007

Neutron spectra re-binning and dose calculation using Monte Carlo methods

H.R. Vega-Carrillo[a,b], E. Manzanares Acuña[a], J.M. Ortiz Rodríguez[a,b], and T. Arteaga Arteaga[a,c]
Unidades Académicas:
[a]Estudios Nucleares, [b]Ing. Eléctrica, Universidad Autónoma de Zacatecas,
Apartado Postal 336, 98000 Zacatecas, Zac. México,
e-mail: fermineutron@yahoo.com,
[c]Envases de Zacatecas, SA de CV,
Parque Industrial de Calera de Víctor Rosales, Zac. México

Recibido el 2 de marzo de 2006; aceptado el 18 de agosto de 2006

One hundred thirty lethargy neutron spectra in 60 energy groups were converted to energy spectra and re-binned to 31 energy groups. Original spectra were taken form the compilation published by the IAEA and covers neutron spectra from isotopic neutron sources, nuclear reactors, medical and physical application accelerators, cosmic rays, etc. The 31 energy groups were taken from the BUNKIUT unfolding code, which is utilized to obtain the neutron spectrum from a multisphere neutron spectrometer. Re-binned spectra were utilized to calculate the ambient, personal and effective doses covering 13 types of doses. This calculation was carried out through the ICRP74 dose-to-fluence conversion coefficients. This procedure was performed using Monte Carlo methods using the MCNP 4C code. Two experiments were carried out to obtain, with a Bonner sphere spectrometer, the BUNKIUT code and the UTA4 response matrix, the neutron spectra of ^{252}Cf and ^{252}Cf/D_2O. For both sources and at the same locations the equivalent ambient dose were measured using a rem meter. Measured H*(10) and neutron spectra were compared with those obtained in the Monte Carlo calculations.

Keywords: Neutron spectrometry; dose; Monte Carlo.

Ciento treinta espectros por unidad de letargia definidos en 60 grupos de energía se convirtieron en espectros por unidad de energía y se re-estructuraron a 31 grupos de energía. Los espectros originales se tomaron de la colección publicada por el OIEA y comprenden espectros producidos por fuentes isotópicas de neutrones, reactores nucleares, aceleradores de uso médico y de investigación en física, rayos cósmicos, etc. Los 31 grupos de energía se tomaron del código de reconstrucción de espectros BUNKIUT que se utiliza para obtener los espectros de neutrones a partir de las tasas de conteo de un espectrómetro de esferas de Bonner. Los espectros re-estructurados se utilizaron para calcular 13 dosis que incluyen la dosis equivalente ambiental, la dosis equivalente personal y la dosis efectiva. Este cálculo se realizó utilizando los coeficientes de conversión de fluencia a dosis del ICRP74. Los cálculos se realizaron utilizando métodos Monte Carlo mediante el código MCNP 4C. Mediante el uso de un espectrómetro de esferas de Bonner, el código BUNKIUT y la matriz de respuesta UTA4 se realizaron dos experimentos donde se determinaron los espectros de dos fuentes de neutrones: ^{252}Cf y ^{252}Cf/D_2O. Además de la determinación de los espectros de neutrones se midió el equivalente de dosis ambiental utilizando un dosímetro moderado de neutrones. Los valores de H*(10) y los espectros de neutrones se compararon con los espectros y las dosis obtenidas con los cálculos Monte Carlo.

Descriptores: Espectrometría de neutrones; dosis; Monte Carlo.

PACS: 29.30.Hs; 87.58.Sp; 87.53.Wz

1. Introduction

Measurements and calculations of neutron fluence spectra are a key factor in radiation protection dosimetry of neutrons. There is a constant need for the determination of neutron dose equivalent because there are different neutron sources that can impact the working conditions. There are also requirements for selecting the devices used to measure the dose equivalent quantities in the workplace. This is difficult due, in part, to the fact that there are not neutron-induced reaction mechanisms in sensors that exactly match those in tissue [1].

During neutron transport in matter they deposit energy in the nuclei producing complex spectra of secondary charged particles. An accurate measurement of deposited energy depends upon knowledge of the neutron fluence spectrum. Any instrument utilized to determine the neutron spectrum or to measure the dose must be calibrated at regular basis utilizing standard sources. To characterize the reference calibration field the neutron spectrometry is necessary. Spectrometric measurements in the workplace may also be needed to provide a basis for correction factors if the workplace spectrum is found to be significant different from that of the reference radiation. If a simulated workplace radiation field is developed for calibration purposes, it will also be necessary to perform spectrometric measurements to verify the simulation of the workplace field [1,2].

There is a wide range of different devices utilized for neutron spectrometry; in the last decade several efforts has been carried out to develop instruments and methods to increase the performance of neutron dosimeters. Olsher *et al.* [3], improved the design of moderated dosimeter named WENDI. Wiegel and Alevra [4], based upon BSS did included Cu and Pb shells inside the polyethylene moderators to produce a new spectrometer called NEMUS; that allows to measure high-energy neutrons. Recently Olsher *et al.* [5], developed a new dosimeter named PRESCILA. In Japan several studies has been carried out to evaluate the quality of their neutron dosimeters [6]. The need to have better dosimeters, with larger liability and capability, had motivated the development of devices that measure neutron and photon direction and energy [7,8], as well to calculate response matrices [9].

The basic physical quantity in radiation protection and neutron spectrometry is the neutron fluence and its related differential distributions in energy and direction. The fundamental dosimetric quantity in radiological protection is the absorbed dose, D, defined as the energy absorbed per unit mass and its unit is the joule per kilogram (J-kg^{-1}), which is given the special name gray (Gy). More specifically, D is defined as the quotient of the mean energy imparted, dε, to matter of mass dm. The quantity kerma, K, relates to the kinetic energy of the charged particles released in matter by uncharged particles. The K is the quotient of the sum of the initial kinetic energies of all the charged particles liberated by uncharged particles, dE_{tr}, in the mass, dm, of material [10].

For specific application to the radiation protection of individuals two types of quantities have been defined: Protection and Operational quantities. The current protection quantities are recommended by the International Commission on Radiological Protection (ICRP) [11] for dose limitation and dose control purposes; while the current operational quantities were introduced by the International Commission on Radiation Units and Measurements (ICRU) [12]. Both quantities are expressed in J-kg^{-1} (Sv) however there are important conceptual differences between the types of quantities, including the weighting method which is applied to account for the different biological effectiveness of radiation (RBE).

1.1. Protection quantities

The ICRP 60 includes three main quantities: Mean absorbed dose ($D_{T;R}$), Equivalent dose ($H_{T,R}$) and the Effective dose (E). $D_{T,R}$, in a specific tissue or organ T of the human body exposed to radiation R, and equivalent dose, $H_{T,R}$, in a specific tissue or organ T exposed to radiation R are related through Eq. (1).

$$H_{T,R} = w_R D_{T,R} \tag{1}$$

where w_R is the radiation weighting factor related to the RBE of the external radiation field.

When the radiation field is composed of different types of radiation with different energies, must be used as many values of w_R as required. Thus, several components of the absorbed dose must be considered and multiplied by their own value of w_R, and the total equivalent dose H_T is given by Eq. (2).

$$H_T = \sum_R w_R D_{T,R} \tag{2}$$

The E is obtained through the summation of equivalent doses, as shown in Eq. (3).

$$E = \sum_T w_T H_T \tag{3}$$

here, w_T is the tissue weighting factor for tissue or organ T.

The absorbed dose distribution and the dose-related quantities in a human body depend on the energy and direction distributions of the incident radiation and the orientation of the body in the radiation field. Calculations of E have been performed for several standardized simplified irradiation geometries that reasonably simulate, singly or in combination, practical situations. These includes antero-posterior (AP), postero-anterior (PA), lateral (LAT), rotational (ROT), and isotropic (ISO). For radiological protection purposes the ICRP recommends a limit on effective dose of 20 mSv per annum averaged over defined period of five years, with the further provision that E should not exceed 50 mSv in a single year.

1.2. Operational quantities

Operational quantities were proposed by ICRU to provide appropriate estimates of the protection quantities and to serve as calibration quantities for dosimetric devices. For the definition of operational quantities the concept of dose equivalent has been applied and defined in specific points in phantoms that simulate the human body.

1.3. Area monitoring

Area monitoring is defined to determine *a priori* the radiation levels in the aim to control occupational exposure. For area monitoring a 30 cm-diameter sphere made with a tissue equivalent material (ρ = 1 g-cm^{-3}) composed with 76.2% oxygen, 11.1% carbon, 10.1% hydrogen and 2.6% nitrogen is used as phantom, these is known as the ICRU sphere. [10]

For area monitoring the ICRU recommends the directional dose equivalent H'(d, Ω), that includes a point located at certain depth in body (d) as well as the direction of radiation incident in the body (Ω). For weakly penetrating radiation, where skin and lens of eye is included d is 0.07 mm while d is 10 mm for strong penetrating radiation [10].

Ambient dose equivalent H*(d) is defined for strong penetrating radiation, H*(d) is the dose equivalent that would be produced by the corresponding aligned and expanded field, in the ICRU sphere at a depth d, on the radius opposing the direction of the aligned field. Once defined the value of d H*(d) is written as H*(10); this should provide an estimate of the effective dose [10] E.

1.4. Individual monitoring

For individual monitoring the ICRU recommends the personal dose equivalent H_p(d), that includes weakly and strong penetrating radiations, depending upon the value of d. H_p(d) is the dose equivalent in soft tissue, at certain depth d, below a specific point in the body. For strongly penetrating radiation a depth of 10 mm is recommended, then H_p(10) is expected to provide an estimate of E. For calibration purposes the definition of H_p(10) is extended to include the dose equivalent in a phantom whose elemental composition is like the ICRU sphere, but the geometry is a 30×30×15 cm^3 slab, in whose case H_p(d) is approximated to $H_{p,slab}$(10).

Spectrometry techniques are used to measure the physical features of radiation fields, such as the fluence, energy and direction distributions Φ(E, Ω) or $\Phi_{E,\Omega}$. The link between this basic field quantity and the radiation protection quantity, either operational or protection dose Δ is given by Eq. (4).

$$\Delta = \int_E \int_\Omega \delta_\Phi \left(E, \bar{\Omega}\right) \Phi_{E.\Omega} d\Omega dE \tag{4}$$

where Δ is the value of the operational or protection quantity, $\Phi_{E,\Omega}$ is the distribution of the fluence with respect to energy E and direction Ω of radiation, and δ_Φ(E, Ω) is the fluence-to-operational, or the fluence-to-protection, quantity conversion coefficient. $\Phi_{E,\Omega}$ is also called the spectrum, if this is integrated for all directions gives $\Phi_{E,g}$r Φ_E(E)u

Detectors containing ^{6}Li, ^{10}B, ^{3}He or ^{197}Au have a large efficiency for the detection of thermal neutrons, to make them sensitive to higher-energy neutrons the detectors are located inside a moderator rich in hydrogen nuclei. In 1960 Bramblett *et al*. [13] did proposed to use spherical polyethylene moderators with different diameters, this is named Bonner spheres spectrometer or multisphere spectrometer (BSS). The set of efficiencies of thermal neutron detector located at the center of moderators in known as response matrix. With the count rates obtained with the detector in each sphere, including the bare detector, the neutron spectrum is calculated by using a procedure named unfolding where the discrete version of first kind integral Fredholm equation is solved [14].

Neutron spectrum unfolding is carried out using several procedures like Monte Carlo, Regularization, Maximum Entropy, Genetic algorithms, Artificial Neural Networks, etc. BUNKIUT code, together with the appropriated response matrix, is utilized to perform the unfolding task; this code takes the BSS count rates and gives out the associated neutron spectrum distributed in 31 energy groups ranging from thermal to 400 MeV. With the spectrum and the fluence-to-dose conversion coefficients the desired dose quantity can be estimated; for this calculation it is required that the fluence-to-dose conversion coefficients must be defined in the same energy group's structure as the spectrum which is not a trivial task [15,16].

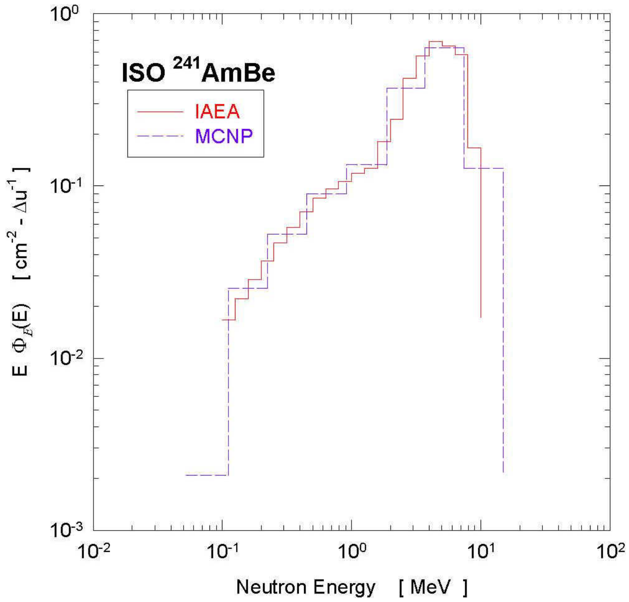

FIGURE 1. MCNP calculated and IAEA reported lethargy spectrum of ^{241}AmBe.

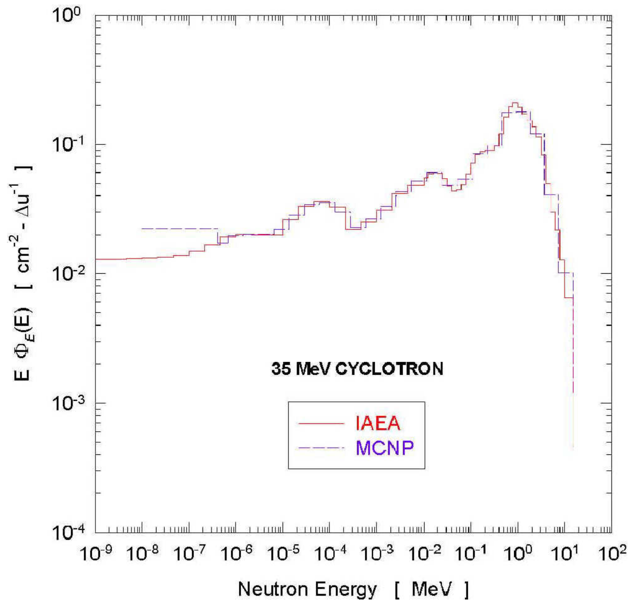

FIGURE 2. MCNP calculated and IAEA reported lethargy spectrum of 35 MeV Tohoko University cyclotron.

In this study 130 neutron spectra, taken from the IAEA compilation, were re-binned from 60 to 31 energy groups. For each spectrum thirteen doses, including Effective dose, Ambient dose equivalent and Personal dose equivalent, were calculated. This procedure was carried out using, for first time, Monte Carlo methods. To compare the calculated results, the spectra of ^{252}Cf and ^{252}Cf/D_2O were obtained with a BSS with a ^{6}LiI(Eu), the UTA4 response matrix and the BUNKIUT code. The Ambient dose equivalent of these sources was also measured using a rem meter.

2. Materials and methods

2.1. Calculations

From the IAEA compilation [17,18] one hundred thirty neutron spectra were taken and converted from lethargy to energy spectra. Resulting energy fluence spectra were normalized and used as point-like source term in Monte Carlo code MCNP 4C [19]. Neutrons were isotropically transported from the source to a detector located at 10 cm in vacuum. The energy bin structure in the detector was the 31 energy groups defined in BUNKIUT code. For each spectrum the average neutron energy was calculated. During Monte Carlo calculations the Effective dose antero-posterior (E_{AP}), postero-

TABLE I. Dose-to-fluence conversion factor for bare ^{241}AmBe.

Dosimetric quantity	Calculated dose per unit fluence rate [pSv – cm^2]	IAEA reported dose per unit fluence rate [pSv – cm^2]
E_{AP}	413	NR
E_{PA}	304	NR
E_{RLAT}	176	NR
E_{LLATP}	200	NR
E_{ROT}	280	NR
E_{ISO}	223	NR
H*(10)	396	395
$H_{p,slab}(10, 0^\circ)$	415	415
$H_{p,slab}(10, 15^\circ)$	413	NR
$H_{p,slab}(10, 30^\circ)$	429	NR
$H_{p,slab}(10, 45^\circ)$	418	NR
$H_{p,slab}(10, 60^\circ)$	386	NR
$H_{p,slab}(10, 75^\circ)$	292	NR

NR means Not Reported

TABLE II. Dose-to-fluence conversion factor for AVF, Tohoko University, 35 MeV Cyclotron.

Dosimetric quantity	Dose per unit fluence rate [pSv – cm^2]	Reported IAEA reported dose per unit fluence rate [pSv – cm^2]
E_{AP}	143	NR
E_{PA}	90.3	NR
E_{RLAT}	45.6	NR
E_{LLATP}	52.7	NR
E_{ROT}	89.4	NR
E_{ISO}	65.1	NR
H*(10)	181	180
$H_{p,slab}(10, 0^\circ)$	188	188
$H_{p,slab}(10, 15^\circ)$	185	NR
$H_{p,slab}(10, 30^\circ)$	187	NR
$H_{p,slab}(10, 45^\circ)$	173	NR
$H_{p,slab}(10, 60^\circ)$	147	NR
$H_{p,slab}(10, 75^\circ)$	87.7	NR

NR means Not Reported

anterior (E_{PA}), right lateral (E_{RLAT}), left lateral (E_{LLAT}), rotational (E_{ROT}), isotropic (E_{ISO}), Ambient dose equivalent (H*(10)), and Personal dose equivalent, for different orientations ($H_{p,slab}$(10,0°), $H_{p,slab}$(10,15°), $H_{p,slab}$(10,30°), $H_{p,slab}$(10,45°), $H_{p,slab}$(10,60°), $H_{p,slab}$(10,75°)) were calculated using the ICRP 74 neutron fluence-to-dose conversion coefficients [20].

2.2. Measurements

Using a BSS with 0 (bare detector), 2, 3, 5, 8, 10 and 12 inches-diameter polyethylene spheres the spectra of a ^{252}Cf/D_2O neutron sources was measured, for the case of bare ^{252}Cf neutron source an extra sphere, 18 inches-diameter, was included. The BSS count rates were input in BUNKIUT code and with the response matrix UTA4 both spectra were unfolded.

Using a single-sphere rem meter Eberline model ASP-1 the H*(10) was measured at the same location were the spectra were determined. Measured spectra and ambient dose equivalent are compared with those obtained with MCNP 4C calculations.

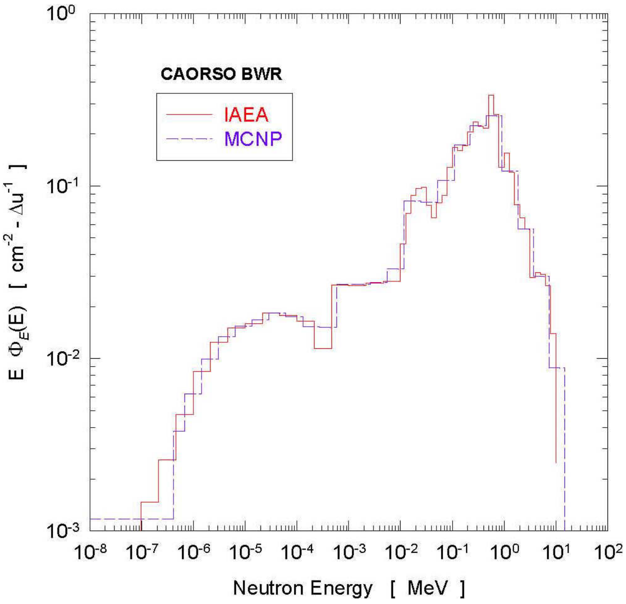

FIGURE 3. MCNP calculated and IAEA reported lethargy spectrum of BWR Caorso nuclear reactor.

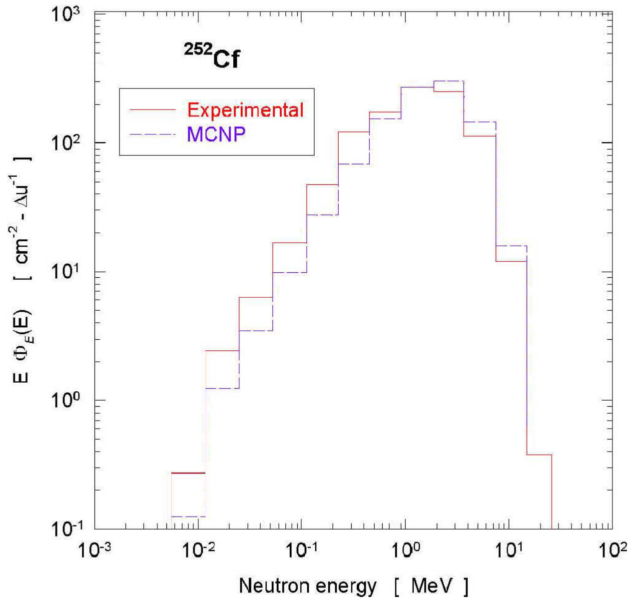

FIGURE 5. Experimental and MCNP re-binned lethargy spectrum of ^{252}Cf neutron source.

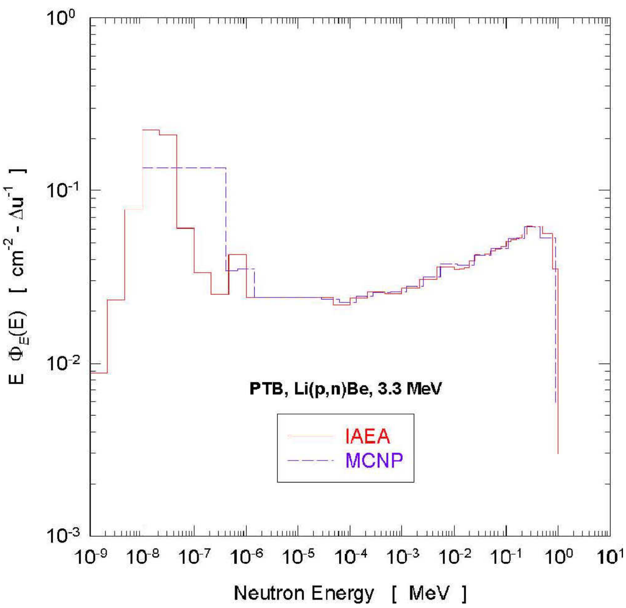

FIGURE 4. MCNP calculated and IAEA reported lethargy spectrum of neutrons produced during Li(p, n)Be nuclear reactions.

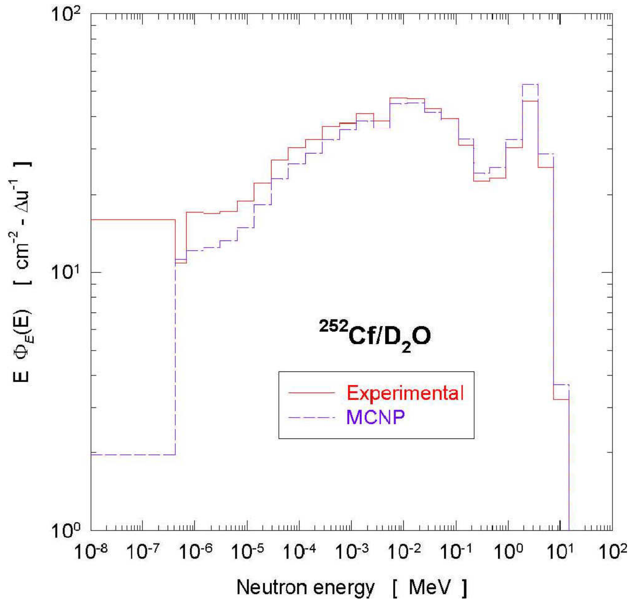

FIGURE 6. Experimental and MCNP re-binned lethargy spectrum of ^{252}Cf/D_2O neutron source.

3. Results and discussion

3.1. Calculations

Four of the 130 rebinned spectra calculated with MCNP 4C code are shown in Figs. 1, 2, 3 and 4, where has been included the original published by the IAEA. The average neutron energy of these spectra are 4.04 MeV for ^{241}AmBe, 0.72 MeV for the 35 MeV Tohoko University Cyclotron, 0.22 MeV for BWR Caorso nuclear reactor and 0.053 MeV for Li(p, n)Be 3.3 MeV nuclear reaction. The corresponding doses for each spectrum are shown in Tables I, II, III and IV.

Comparing the calculated spectra with those published by the IAEA can be noticed that both agree. In the IAEA compilation is included H*(10) and $H_{p,slab}$(10,0°); in the tables can be noticed that those doses agrees with the respective doses calculated with MCNP 4C. Relating the average neutron energy with the doses intensities can be observed that the doses intensities are bigger for those spectra with larger average neutron energy; this is in agreement with published results [21,22].

3.2. Measurements

In Fig. 5 the measured and MCNP calculated ^{252}Cf spectra are shown while in Fig. 6 are the ^{252}Cf/D_2O spectra. In both cases one can noticed that they are alike; the differences are attributed to the sky and ground shine neutrons as well to room return effect produced by the experimental conditions. This is particularly enhanced in the case of ^{252}Cf/D_2O where the experimental spectrum shows a large contribution of thermal neutrons. Total neutron fluence rate for ^{252}Cf is 309 cm^{-2}-s^{-1} and for ^{252}Cf/D_2O is 250 cm^{-2}-s^{-1} at 100 cm from the source.

From the H*(10) measurements the ^{252}Cf gives 9.4E(4) pSv-s-1, dividing the dose by the total neutron fluence rate gives a dose per unit fluence of 309 pSv-cm^2. This value is 383 pSv-cm^2 calculated using MCNP and to 385 pSv-cm^2 reported by the IAEA. In the case of ^{252}Cf/D2O the measured H*(10) is 1.9E(4) pSv-s^{-1} that divided by the total fluence gives a dose per unit fluence of 76 pSv-cm^2. According with MCNP calculation this value should be 107 pSv-cm2 which is close to 105 pSv-cm^2 reported by the IAEA. In both cases the differences are attributed to the experimental conditions that produce softer spectra due to skyshine, groundshine and room return of neutrons.

TABLE III. Dose-to-fluence conversion factor BWR Caorso Nuclear Reactor.

Dosimetric quantity	Calculated dose per unit fluence rate [pSv – cm^2]	Reported IAEA reported dose per unit fluence rate [pSv – cm^2]
E_{AP}	78.4	NR
E_{PA}	48.0	NR
E_{RLAT}	21.4	NR
E_{LLATP}	25.0	NR
E_{ROT}	43.9	NR
E_{ISO}	33.4	NR
H*(10)	113	113
$H_{p,slab}(10, 0^\circ)$	118	118
$H_{p,slab}(10, 15^\circ)$	116	NR
$H_{p,slab}(10, 30^\circ)$	113	NR
$H_{p,slab}(10, 45^\circ)$	99.5	NR
$H_{p,slab}(10, 60^\circ)$	76.8	NR
$H_{p,slab}(10, 75^\circ)$	36.3	NR

NR means Not Reported

TABLE IV. Dose-to-fluence conversion factor for PTB Li(p, n)Be, 3.3 MeV.

Dosimetric quantity	Calculated dose per unit fluence rate [pSv – cm^2]	IAEA reported dose per unit fluence rate [pSv – cm^2]
E_{AP}	29.9	NR
E_{PA}	18.6	NR
E_{RLAT}	7.78	NR
E_{LLATP}	9.09	NR
E_{ROT}	16.6	NR
E_{ISO}	12.5	NR
H*(10)	42.4	42.4
$H_{p,slab}(10, 0^\circ)$	44.4	44.5
$H_{p,slab}(10, 15^\circ)$	43.4	NR
$H_{p,slab}(10, 30^\circ)$	41.1	NR
$H_{p,slab}(10, 45^\circ)$	35.1	NR
$H_{p,slab}(10, 60^\circ)$	25.7	NR
$H_{p,slab}(10, 75^\circ)$	11.1	NR

NR means Not Reported

4. Conclusions

One hundred thirty spectra were taken from IAEA compilation and using MCNP code were re-binned from their original energy structure to that utilized by the BUNKIUT code utilized during neutron spectra unfolding from the count rates measured by a Bonner sphere spectrometer.

With the re-binned spectra thirteen doses, including Effective, Ambient and Personal equivalent doses, were calculated and compared with those reported by the IAEA. In all cases the doses were similar.

The spectra of a bare ^{252}Cf and ^{252}Cf/D_2O neutron sources were measured using a BSS; comparing the experimental spectra with those calculated with MCNP. Also using a rem meter the H*(10) of both sources was measured. Similarities between MCNP calculated and measured spectra were founded. By comparing the measured ambient dose equivalent per unit fluence with that calculated with MCNP differences were observed. Differences observed in the spectra and in the doses are attributed to the skyshine, groundshine and room effect produced by the experimental conditions.

Acknowledgments

This work was partially supported by CONACyT (Mexico) under contract SEP-2004-C01-46893.

1. J.C. McDonald, B.R.L. Siebert, and W.G. Alberts, *Physics Research A* **476** (2002) 347.
2. D.J. Thomas, *Radiation Protection Dosimetry* **110** (2004) 141.
3. R.H. Olsher *et al.*, *Health Physics* **79** (2000) 170.
4. B. Wiegel and A.V. Alevra, *Physics Research A* **476** (2002) 36.
5. R.H. Olsher *et al.*, *Health Physics* **86** (2004) 603.
6. J. Saegusa *et al.*, *Physics Research A* **516** (2004) 193.
7. J.L. Muñiz *et al.*, *Radiation Protection Dosimetry* **110** (2004) 243.
8. M. Luszik-Bhadra, M. Reginato and V. Lacoste, *Radiation Protection Dosimetry* **110** (2004) 237.
9. H.R. Vega-Carrillo, E. Manzanares-Acuña, V.M. Hernández-Dávila, and G.A. Mercado Sánchez, *Rev. Mex. Fís.* **51** (2005) 47.
10. D.T. Bartlett, J.-L. Chartier, M. Matzke, A. Rimper, and D.J. Thomas, *Radiation Protection Dosimetry* **107** (2003) 23.
11. ICRP, 1990 Recommendations of the ICRP, Publication 60, Annals of the ICRP 21 (1-3), International Commission on Radiological Protection (Pergamon Press, New York, 1991).
12. ICRU, Quantities and Units in Radiation Protection Dosimetry,ICRU Report 51, International Commission on Radiation Units and Measurements (Bethesda, Maryland, 1993).
13. R.L. Bramblett, R.I. Ewing, and T.W. Bonner, *Nuclear Instruments and Methods* **9** (1960) 1.
14. H.R. Vega-Carrillo and M.P. Iñiguez de la Torre, *Physics Research A* **476** (2002) 270.
15. H.R. Vega-Carrillo *et al.*, *Radiation Measurements* **41** (2006) 425.
16. H.R. Vega-Carrillo *et al.*, Artificial neural networks in neutron dosimetry. Radiation Protection Dosimetry. (in press. doi:10.1093/rpd/nci354).
17. R.V. Griffith, J. Palfalvi, and U. Madhvanath (editors), Compendium of Neutron Spectra and Detector Responses for Radiation Protection Purposes. International Atomic Energy Agency, Technical Reports Series No 318. Vienna (1990).
18. R.V. Griffith, J. Palfalvi, and B.R.L. Siebert (editors), Compendium of Neutron Spectra and Detector Responses for Radiation Protection Purposes. Supplement to Technical Reports Series No. 318. International Atomic Energy Agency, Technical Reports Series No 403. Vienna (2001).
19. J.F. Briesmeister (editor), MCNP^{TM}-A general Monte Carlo N-Particle Transport Code. Los Alamos National Laboratory Report LA-13709-M (2000).
20. ICRP, Conversion Coefficients for use in Radiological Protection against External Radiation, Publication 74, Annals of the ICRP 26 (3/4), International Commission on Radiological Protection, Pergamon Press, New York (1996).
21. H.R. Vega-Carrillo *et al.*, *Rev. Mex. Fís.* **51** (2005) 494.
22. A.N. Garg and R.J. Batra, *Journal of Radioanalytical and Nuclear Chemistry* **98** (1986) 167.