NEUTRON SPECTROMETRY AND DOSIMETRY WITH ANNs

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ABSTRACT

Artificial Neural Networks technology has been applied to unfold the neutron spectra and to calculate the effective dose, the ambient equivalent dose, and the personal dose equivalent for $^{252}$Cf and $^{241}$AmBe neutron sources. A Bonner sphere spectrometry with a $^6$Li(Eu) scintillator was utilized to measure the count rates of the spheres that were utilized as input in two artificial neural networks, one for spectrometry and another for dosimetry. Spectra and the ambient dose equivalent were also obtained with BUNKIUT code and the UTA4 response matrix. With both procedures spectra and ambient dose equivalent agrees in less than 10%.

Key words: Artificial Neural Networks; Bonner sphere spectrometer; neutron spectrum; dose

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INTRODUCTION

Neutron spectrometry

In radiation spectrometry the key factor is the definition of the intensity of a radiation field expressed in terms of certain quantity like energy, position, angle, resulting in the determination of radiation spectrum; that helps to characterize the radiation field. (Thomas and Klein 2003) Mostly all neutron sources produce neutrons in a wide energy range extending from few thousandths of eV to several hundreds of MeV named neutron-fluence spectrum or simply neutron spectrum, $\Phi_E(E)$. (Alevra 1999, ICRU 2001) The Bonner sphere spectrometer, BSS, is a set of high-density polyethylene spheres with a thermal neutron detector that is utilized to obtain the $\Phi_E(E)$.

Each sphere-detector combination has a unique response called response function, the whole set of response functions is the response matrix, $R_\Phi(E)$, that is shown in Figure 1.

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<th>Neutron energy [MeV]</th>
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**Figure 1.- Response matrix of a BSS$^{6}$/LiI(Eu)**
Inside a neutron field the BSS produce a set of count rates, $C_i$, with the matrix response, $R_{\Phi_i}$, the $\Phi_{E}(E)$ is unfolded. Unfolding process implies to solve the Equation 1.

$$C_i = \int_{E} R_{\Phi_i}(E) \Phi_{E}(E) \, dE \quad (1)$$

Equation 1 this is a Fredholm integral equation of the first kind, $C_i$ is the count rate measured with the $i$-th sphere-detector combination, $R_{\Phi_i}$ is its response function. As can be noticed in Figure 1 response functions have poor energy resolution and are not fully linearly independent of each other. On the other hand, the spectrum is defined in a larger number of energy groups in comparison to the number of spheres used in the spectrometer; therefore Equation 1 is an ill-conditioned problem. (Vega-Carrillo et al., 2006a)

Total fluence is obtained integrating the $\Phi_{E}(E)$ for all energies as shown in Equation 2.

$$\phi = \int_{E} \Phi_{E}(E) \, dE \quad (2)$$

Once the $\Phi_{E}(E)$ is obtained the required dose, $\Delta$, can be calculated using Equation 3.

$$\Delta = \int_{E} \delta_{\Phi}(E) \Phi_{E}(E) \, dE \quad (3)$$
In Equation 3, $\delta \Phi(E)$ are the fluence-to-required dose conversion coefficients. (Vega-Carrillo et al., 2006b) Therefore, the main problem is to solve Equation 1, to do this several procedures are utilized. (Vega-Carrillo et al., 2006a)

**Artificial Neural Networks**

In the human brain, a neural network is a massively parallel distributed processor with a natural propensity for saving experiential knowledge, previously acquired through a learning process, making it available for latter use. The Artificial Neural Networks, ANNs, have been proposed to emulate this feature from brain’s behaviour. (Haykin 1994) The mathematical models used to design an ANN have at least three layers, the input, hidden and the output. Each layer has several process units, the neurons, which are connected through synaptic weights where the knowledge is stored. The activation of a neuron depends on the given weight of the synapses that are appraised through the activation functions. (Cordes et al., 1998, Kardan et al., 2003)

The ANN learning is the adaptation process to the training data. The training has the purpose of selecting the weights that adapts better to the network in relation to training data. During ANN training a set of data, with input and output information, is utilized and the synaptic weights are adjusted until, at some point, the pattern between input and output data is accomplished.

The ANN technology has been utilized to unfold the neutron spectrum, and with this the neutron dose has been estimated. (Cordes et al., 1998, Vega-Carrillo et al., 2006a, Kardan et al., 2003) Also, The ANNs have been applied to obtain directly the
neutron doses without the need of neutron spectrum information. (Vega-Carrillo et al., 2006b) In all these applications only the count rates measured with a BSS are required.

In this work the ANN technology has been utilized to unfold the neutron spectrum and to determine the neutron doses of $^{252}$Cf and $^{241}$AmBe isotopic neutron sources. In both ANNs only the input was the count rates measured with a BSS. These results were compared with spectra and the ambient dose equivalent, $H^*(10)$, obtained with BUNKIUT code.

MATERIALS AND METHODS

Artificial Neural Networks

Two different ANNs were designed and trained, the first to perform the neutron spectrometry and the second to calculate three dosimetric quantities: the Ambient dose equivalent, $H^*(10)$, the Effective dose, $E$, and the Personal dose equivalent $H_{p,s}(10,\theta)$.

The Effective dose was calculated for the different irradiation geometries: Anteroposterior, $E_{AP}$, posteroanterior, $E_{PA}$, right lateral, $E_{RLAT}$, left lateral, $E_{LLAT}$, rotational, $E_{ROT}$, and isotropic, $E_{ISO}$. The Personal dose equivalent was calculated for different angle of incidence of neutrons for $0^\circ$, $15^\circ$, $30^\circ$, $45^\circ$, $60^\circ$ and $75^\circ$: $H_{p,s}(10,0^\circ)$, $H_{p,s}(10,15^\circ)$, $H_{p,s}(10,30^\circ)$, $H_{p,s}(10,45^\circ)$, $H_{p,s}(10,60^\circ)$, $H_{p,s}(10,75^\circ)$. 
The ANN for spectrometry was trained using neutron spectra compiled by the International Atomic Energy Agency (IAEA 1990, IAEA 2001). In the compilation neutron spectra are defined in terms of lethargy and in different energy groups. Lethargy spectra were converted to energy spectra and were re-binned from their original energy groups to 31 energy groups. (Vega-Carrillo et al., 2007a) Final energy groups are the same utilized in BUNKIUT code. (Lowry and Johnson 1984) The re-binned was performed using Monte Carlo methods with the MCNP 4C code. (Briesmeister 2000)

For each spectrum, using equation 3, were calculated the $E_{AP}$, $E_{PA}$, $E_{RLAT}$, $E_{LLAT}$, $E_{ROT}$, $E_{ISO}$, $H^*(10)$, $H_{p,s}(10,0^\circ)$, $H_{p,s}(10,15^\circ)$, $H_{p,s}(10,30^\circ)$, $H_{p,s}(10,45^\circ)$, $H_{p,s}(10,60^\circ)$, $H_{p,s}(10,75^\circ)$. Calculations were performed using the fluence-to-dose conversion coefficients from ICRP 74 (ICRP 1996) that were input in the MCNP4C code.

The 31 energy groups of re-binned spectra, $\Phi_i$, for the j-th sphere were utilized, together with UTA4 matrix response, (Hertel and Davidson 1985) $M_{i,j}$, to calculate their respective count rates produced in the Bonner spheres. Calculations were realized for Ball 0, Ball 2, Ball 3, Ball 5, Ball 8, Ball 10, and Ball 12. The polyethylene sphere diameters are 0, 5.08, 7.62, 12.7, 20.32, 25.4 and 30.48 cm respectively. The count rate for the j-th sphere-detector combination, $C_j$, was carried out with equation 4.

$$C_j = \sum_{i=1}^{31} M_{i,j} \Phi_{i,j}$$

One hundred seventy seven neutron spectra, their doses and their respective BSS count rates were utilized to train and test both ANNs. From the available set of data the
10% was randomly selected and isolated from the whole set to test the ANN, the remainder was left for training purposes.

Both ANNs were trained and tested using MATLAB® software. Topology of ANN for spectrometry was 7:140:500:140:70:27, where each number is the amount of neurons in each layer. For dosimetry the ANN topology was 7:140:200:401:140:13.

Both ANNs were feed-forward with backpropagation algorithm and variable learning rate. (Vega-Carrillo et al., 2007a) After training, the ANNs’ performance was verified using the data set that was not utilized for training purposes.

**Experiments**

The BSS was utilized to obtain the count rates produced by $^{252}$Cf and $^{241}$AmBe isotopic neutron sources. Their respective activities were $4.34 \times 10^8$ and $3.7 \times 10^9$ Bq. During BSS measurements the source-to-spectrometer distance was 120 and 50 cm for $^{252}$Cf and $^{241}$AmBe respectively. Measurements for $^{252}$Cf were realized with the bare source inside a low scattering room; source and BSS were located at the center of the room located at 3 m from the nearest wall and at 200 cm above floor. The $^{241}$AmBe source is enclosed in 2.54 cm-thick polyethylene cylinder. BSS measurements were done in a small brick-walls room at 150 cm above floor and at 100 cm from the nearest wall.

For both sources the count rates were utilized to unfold the neutron spectra using BUNKIUT code and UTA4 response matrix. Neutron spectra were utilized to estimate the $H*(10)$.

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1 MATLAB is a trade mark of The MathWorks, Inc.
Neutron spectra were also unfolded with the ANN designed for spectrometry, and the dosimetric quantities were calculated using the ANN designed for dosimetry. Resulting doses were normalized to the total neutron fluence for both spectra to report the dose factors.

RESULTS AND DISCUSSION

Neutron spectrometry

In Figure 2 is shown the neutron spectrum of $^{252}$Cf obtained with BUNKIUT code and the ANN for spectrometry.

![Figure 2.- Neutron spectrum of $^{252}$Cf source.](image)
Here, can be noticed that both results have good agreement. A bare $^{252}$Cf do not produce epithermal neither thermal neutrons. The neutron spectrum shown in Figure 2 show epithermal and thermal neutrons due the experimental conditions used during measurements. (Vega-Carrillo et al., 2007c)

The total neutron fluence is $193 \pm 6 \text{ cm}^{-2}\cdot\text{s}^{-1}$ and $191 \pm 6 \text{ cm}^{-2}\cdot\text{s}^{-1}$ for spectrum obtained with BUNKIUT and the ANN respectively. The uncertainty was taken from the uncertainty of count rates, therefore in both cases no uncertainties were assigned to the unfolding process.

The $^{241}$AmBe neutron spectra obtained with BUNKIUT and the ANN for spectrometry is shown in Figure 3. It can be noticed that both are similar.

![Figure 3.- Neutron spectrum of $^{241}$AmBe source.](image)
A bare $^{241}$AmBe produce neutrons from 0.5 to approximately 10 MeV, the $^{241}$AmBe source used in the measurements is located inside a small polyethylene cylinder; besides measurements were carried out inside a small brick-walls rooms, this condition induce the presence of epithermal and thermal neutrons due to room return. (Vega-Carrillo et al., 2007c) The total fluence for both spectra is $8 \pm 5\% \text{ cm}^{-2}\text{-s}^{-1}$. As in both previous cases no uncertainties were assigned to the unfolding process and the 5% comes from the uncertainties obtained with the BSS count rates.

**Neutron dosimetry**

The neutron fluence-normalized dosimetric quantities, obtained with the ANN for dosimetry, are shown in Figure 4. These values were estimated dividing the dose by the total neutron fluence.

The Effective dose factor has the same value for both sources, except for the anteroposterior irradiation. The ambient dose equivalent and the personal dose equivalent factors due to $^{252}$Cf source are larger to factors produced by $^{241}$AmBe; probable explanation is as follow: For bare and point-like neutron sources in vacuum the mean energy and the doses for $^{252}$Cf are smaller than $^{241}$AmBe respective values. Here, measurements were done inside rooms, $^{241}$AmBe size and geometry are far from be point-like and is located inside a polyethylene enclosure that produces a softer spectrum. On the other hand, the $^{252}$Cf is small that is alike a point-like source. Experimental conditions modifies the spectra produced by bare sources producing softer spectra.
The neutron fluence-to-ambient dose equivalent factors, $h^*(10)$, for $^{241}\text{AmBe}$ and $^{252}\text{Cf}$ obtained with the ANN, were 270 and 322 pSv-cm$^2$ respectively. With BUNKIUT these values were 265 and 314 pSv-cm$^2$ respectively. The differences in these results are 1.7 to 5% respectively.

According to IAEA (1990) the $h^*(10)$ for $^{241}\text{AmBe}$ is 395 pSv-cm$^2$ while for $^{252}\text{Cf}$ is 380 pSv-cm$^2$, these values are assuming point-like sources in vacuum. In this
work \( h^{*}(10) \) was larger for \(^{252}\text{Cf}\) than \(^{241}\text{AmBe}\) due to the changes in the neutron spectra due to experimental conditions.

The \( H^{*}(10) \) calculated with the ANN were 2.3 and 61.5 nSv, with BUNKIUT these values are 2.2 and 61.0 nSv for \(^{241}\text{AmBe}\) and \(^{252}\text{Cf}\) respectively. The differences between those values are less than 5%.

Considering the uncertainties of the BSS count rates, the differences noticed in \( h^{*}(10) \) and \( H^{*}(10) \) are not significant. The ANN for dosimetry was able to calculate the neutron doses using only the measured BSS count rates.

**CONCLUSIONS**

Two ANNs, one for spectrometry and another for dosimetry, were utilized to unfold the neutron spectra and to estimate the dosimetric features of \(^{252}\text{Cf}\) and \(^{241}\text{AmBe}\) neutron sources. In these calculations both ANNs were feed only with the count rates that both sources produced in a BSS. Spectra and doses were also calculated using the BUNKIUT code. Results obtained with both procedures were in agreement.

The ANN technology is an alternative procedure for neutron spectrometry and dosimetry because both networks can be used independently. Another unfolding
procedures requires an unfolding code and an initial guess, as close as possible of searched spectrum, once the spectrum is obtained the dose can be calculated. In actual situations suggesting the initial guess spectrum is not always achievable. For dosimetry another option is to use the neutron spectra’s mean energy and with this the dose is estimated using the fluence-to-dose conversion coefficients; in this option the estimation of the mean energy in not easy.

The use of unfolding codes for neutron spectrometry requires of certain level of expertise because several data are required to obtain the solution. To apply the ANNs for spectrometry and dosimetry the user only needs the count rates.

So far a drawback of ANN technology is that a different network must be designed and trained for each set of spheres and detector in the spectrometer. In this case both ANNs works are useful for BSS with 0, 2, 3, 5, 8, 10, and 12 inches-diameter spheres with a $0.4 \times 0.4$ cm $^6$LiI(Eu) scintillator. As a work in progress a user-friendly code has been developed to apply this technology.

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