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INTEGRATION TECHNIQUE FOR NEUTRON SOURCE STANDARDIZATION INSIDE A CYLINDRICAL WATER TANK

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معايرة المصادر النيوترونية بطريقة المسح الاشعاعي الكلي داخل وعاء اسطواني مملوء بالماء

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يتناول هذا البحث تقدير الانبعاث النيوتروني من المصادر النيوترونية - ولقد استخدمت في هذه القياسات أربعة مصادر مختلفة الطاقة - كما استخدم في المسح النيوتروني الكلي الكاشف (ثلاثي فلوريد البورون – BF_3) .

ولقد قدرت كمية النشاط الاشعاعي لهذه المصادر كما حسبت نسبة الخطأ الكلي التي وجد أنها لا تزيد عن ± ١٠٪ ، وتتميز هذه الطريقة بأنها لا تشكل أية خطورة من ناحية التعرض الاشعاعي ، بالإضافة إلى امكانية استخدامها في أعمال التنشيط الاشعاعي ، وكذلك يمكن تطبيقها في أعمال اشعاعية أخرى .

Key Words: Neutron source, Standardization, Boron trifluoride

ABSTRACT

This work deals with the evaluation of neutron irradiation facility. Two (Pu-Be) sources and two (Ra-Be) sources are used. Integration technique by a BF_3 detector for neutron source standardization is used. The necessary calibrations have been performed, and all the possible known errors have been taken into consideration. The total error using this technique does not exceed ± 10%. This assembly is capable of safely accommodating the neutron sources and of permitting work inside the assembly without any hazard. Moreover, the availability of such neutron irradiation facilities with isotropic neutron sources provide a promising tool for applications to neutron activation analysis.

INTRODUCTION

Irradiation facilities may be designed and constructed by applying one of the most frequently used moderating media, e.g. water, paraffin wax or graphite. Sources of slow neutrons are derived from high energy neutrons by allowing the neutron to diffuse through one of these moderating materials. Many authors have attempted to determine the neutron emission rates by different types of techniques as shown by Valente (1978), Curtiss (1959) and El-Kolaly (1985). Seidle and Harris (1947) determined the neutron flux out of Ra-Be source by activation of boric acid bath. Anderson Feld (1977) derived an empirical formula for the computation of a neutron output from a mixture of $RaBr_2$ grains with beryllium powder due to (α,n) reaction. Noyce and Mosburge (1963) determined the neutron emission rate by relative comparison with a standard source using heavy water and activation technique for Mn foils and 4 π β–γ coincidence counting technique. El-Kolaly (1970) used a $MnSO_4$ solution in different cylindrical tanks by varying both the solution concentration as well as the tank size; he reported the optimum geometrical size of the container and the solution concentration.

EXPERIMENTAL

The calibration system consisted of a stainless steel cylindrical tank filled with a distilled water. The water content was $1m^3$. The height of the tank was 110 cm. A BF_3 gas-filled counter was used. This was fitted with appropriate amplifier and discriminator in order to cut down the gamma background. Scaller, timer and power supply were used for this work. Neutron sources were available for use as a reference as well as for calibration purposes. A special shielding was applied to accommodate the sources during the

non-working hours.

The source under investigation was placed at the middle of water tank so that only a negligible fraction of the neutrons might escape through the boundary. The BF_3 tube was placed against the side of the cylindrical source. The neutron measurements were carried out at different positions apart from each other by equal distances. The active length of the BF_3 tube was placed against the side of the cylindrical source. The radial readings were taken at 5 cm intervals until no more leakage could be detected about 50 cm away of the source center.

THEORETICAL BASIS

The source of strength Q is placed at the centre of the water tank which is large enough (one cubic meter) that only a negligible fraction of neutrons escape through the outer boundary. The neutrons are slowed down in their collisions with the hydrogen of the water. At the steady state where there is no leakage from the system Q is assumed to be

$$Q = 4\ \pi\ N \int_o^\infty \sigma_{(a)} \phi_{(r)}\ r^2\ dr \qquad (1)$$

where N = the number of absorbing atoms (the hydrogen atoms of the water molecules/c.c. in the medium).

$\sigma(a)$ = the average absorption cross section.

$\Phi(r)$ = the flux density at (r) cm from the source.

The area under the curve (S) of the thermal neutron distribution in the water from the neutron source is performed with Simpson's formula:

$$S_o = \frac{h_i}{3}(N_1 + 4N_2 + 2N_3 + 4N_4 + \ldots + 2N_{n-2} + 4N_{n-1} + N_n) \qquad (2)$$

where S = Area under the curve for the unknown neutron source.

h_i = Distance between 2 successive measurements (cm).

N_i = Count rate (CMP) at i = 1 up to i = n.

The last equation can be put as follow, for the standard source:

$$S_o = \frac{h_i}{3} \sum_{i=1}^{n} K N_i^\circ \ \ldots\ldots \qquad (3)$$

where K = coefficient from Simpson's formula

K = 1 when i = 1 and = n, and k = 4 when i = even numbers

K = 2 when i = odd numbers.

i.e. K = 1, 4, 2, 4, 2, 4, 2, 4, 1 respectively for 9 readings. The neutron yield Q (n/sec) from the unknown neutron source

$$Q = Q_o \frac{S}{S_o} \ (n/sec) \qquad (4)$$

where Q_o is the neutron yield from the standard source.

The relative error from the unknown source δ Q.

$$\delta Q = \sqrt{\delta^2 Q_o + \delta^2 S^\circ + \delta^2 S} \qquad (5)$$

where $\delta\ Q_o$ = relative root mean square error of the neutron yield from the standard source (taken from its certificate).

δS_o, δS = relative root mean square error of measurements of areas under the curves of the thermal neutron distribution in water from the standard and unknown neutron sources.

$$\delta^2 S_o = \frac{\sum_{i=1}^{n} N_i^\circ\ \delta\ S_i^\circ}{\sum_{i=1}^{n} N_i^\circ} \qquad (6)$$

where N^o_i = counting rate (pulse/min) in (i) measurement without background

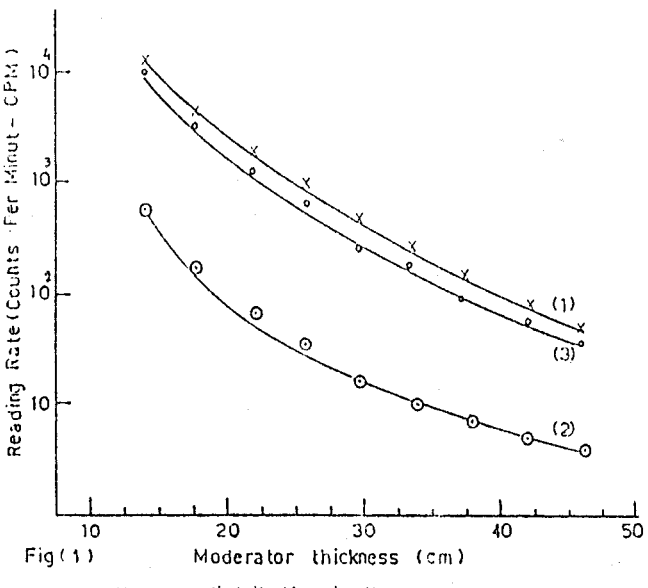
$$\delta^2 S = \sqrt{\frac{(\delta h_i)^2}{h_i} + \frac{(\delta N_i^\circ)^2}{N_i^\circ}} \qquad (7)$$

where δh_i = Absolute error of the interval hi measurement equal to half of the scale point of measuring ruler.

The area under the curve S_o for the standard neutron source can be evaluated by equation (3). It is necessary to determine the area under the curve S for the unknown source by repeating the previous steps using the unknown neutron source. To find out out the neutron yield Q (n/sec) for the unknown source equation (4) should be applied. To evaluate the relative error and the degree of uncertainly equations (5), (6) and (7) should be considered.

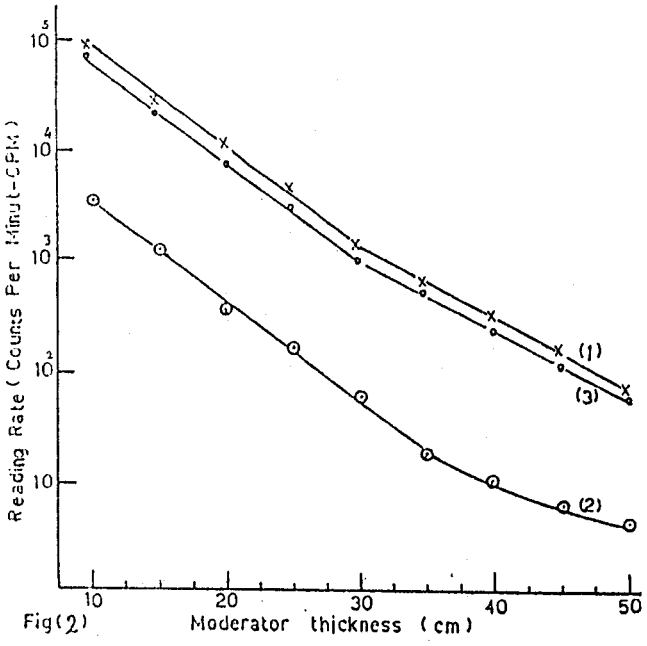
RESULTS AND DISCUSSION

In this work, radiation measurements by the BF_3 detector are taken at different locations for 10 minutes each. The interval between each two successive locations is 5 cm distance. The average counting rate per minute (CPM) at each positions is determined. Fig. (1) shows the response of these counting rates against their corresponding thicknesses. This figure represents the neutron distribution at different thicknesses of water moderator when the small (Ra-Be) neutron source is used. Curve (no. 1) shows the total neutron distribution when the BF_3 detector is bare, while curve (2) shows the detector response when it is covered with Cd of



Fig(1) Neutron distribution in the water tank using small (Ra-Be) neutron source.

1mm. thickness i.e. epi-cadmium response. Curve (3) represents the effect of thermal neutrons i.e. below cadmium cut-off. From this figure it is clear that the neutron distribution decreases with increasing water moderator thickness. In this case neutrons diffuse through the material losing energy as they suffer scattering. If the capture cross section is small compared to the scattering cross section (this depends upon neutron energy), the neutrons continue to lose energy until their energy is comparable to the energy of thermal agitation of the nuclei in the moderating material.

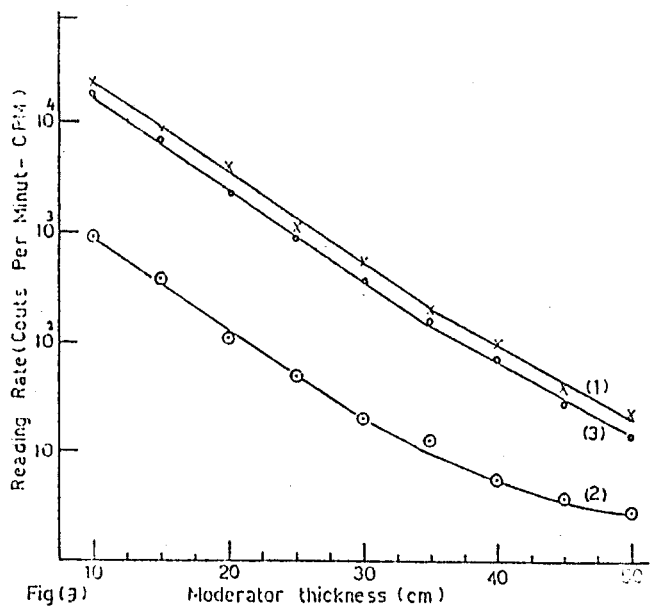


Fig(2) Neutron distribution in the tank filled by water and using big (Ra-Be) neutron source.

Fig. 2 shows the neutron distribution when using the big (Ra-Be) neutron source. It is more or less the same as Fig. (1). The counting rate in this case is about five times more at each similar location than the case of the small source. Also it is clear that the thermal neutron distribution (curve no. 3) is nearly equal to the total neutron distribution (curve 1) in this figure, while the epithermal neutrons (Curve 2) represent a percentage of about 5-10% of the total neutron yield. This is because boron (B-10) in the BF_3 counter gives (n, α) reaction which is useful for absolute determination of thermal neutron flux density, since boron cross section is assumed to be the most proper detector. This is based on the determination of the thermal neutron distribution in the water tank.

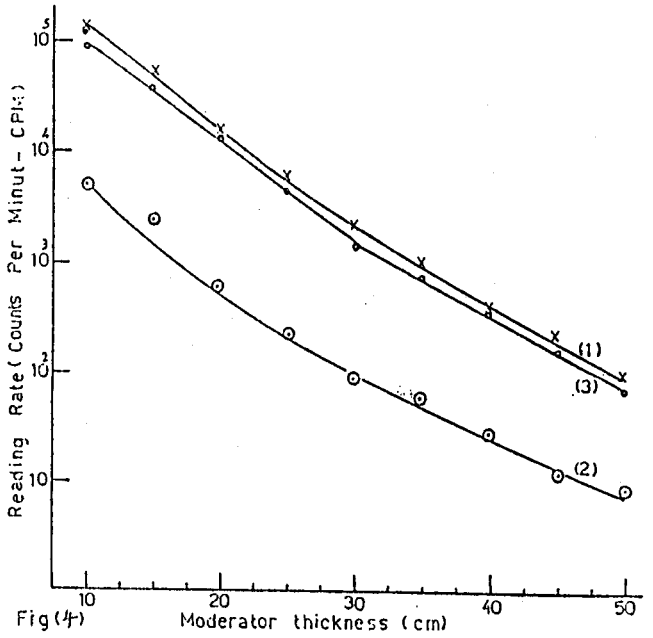
Fig. 3 gives the neutron distributions when the small (Pu-Be) neutron source is used. It is clear that thermal neutrons (curve 3) are nearly equally to the total neutron distribution (curve 1). The differences is less than 5%. The epithermal neutron distribution (curve 2) is a small ratio. It is found that the decrease in the distribution in all neutron energies is exponential in nature, except for the case of epithermal neutrons after about 30 cm water thickness which deviates from this trend.

The big (Pu-Be) neutron source is used as a standard one, and taken as a reference for the other sources. Fig. 4 shows the neutron distributions for this source. These distributions are more or less as in Fig. 3. The only difference is that the counting rate at each position is about five times more than that in cases of Fig. 3. It is clear that by increasing the



Fig(3) Neutron distribution in the tank filled by water and using small (Pu-Be) neutron source.

penetration thicknesses of the water moderator, the neutrons are slowed down in their collision with the hydrogen of the water. The rate of neutron absorption is directly proportional to the original neutron yield. So the original distribution shape will maintain itself. For this reason, intercomparisons between the areas under the curves for the different neutron sources give good estimation for the neutron yield for each source regardless of its energies or its neutron spectrum. This is done by the help of the standard neutron source. In our case, the strong (Pu-Be) neutron source is taken as a reference source. If the size of the slowing down medium is not sufficient, the neutrons will escape before attaining thermal neutron equilibrium. So, it is recommended to use water tank large enough to prevent neutron escape. In our case the radius of the tank is 50 cm, and the percentage neutron leakage from the boundary of the tank does not exceed 1-2% of the total neutron yield.



Fig(4) Neutron distribution in the tank filled by water and using big (Pu-Be) neutron source.

Table 1
Evaluation of the factor (K N^{o}_{i}) for the different sources at different locations.

	Distance	Small (Ra-Be)	Big (Ra-Be)	Small (Ru-Be)	Big (Ru-Be)
1	10	15085	81105	20283	110141
2	15	21486	114980	33680	82564
3	20	3340	17960	5810	31558
4	25	2987	16064	5132	27888
5	30	458	2608	746	4050
6	35	467	2512	876	4776
7	40	164	562	206	1130
8	45	102	556	168	936
9	50	14	77	17	97

The obtained experimental data are tabulated as in Table 1. In this table the factor (K N^{o}_{i}) has been evaluated for the different neutron sources at different locations according to equation (3), where K is the coefficient of Simpson's formula and has the values between 1 and 4. The term N^{o}_{i} represents the net counting rate after subtracting the background at the position of interest.

Table (2) gives the summary of the neutron yield from the different neutron sources according to equations (3) and (4). The degree of accuracy and the total errors are also tabulated according to equations (5), (6) and (7) which are mentioned earlier. It is found that the total errors in the determination of the neutron yield from the different sources do no exceed ±(6-10)%, regardless its neutron spectrum.

Table 2
Neutron yield from different neutron sources

Factor	Small (Ra-Be)	Big (Ra-Be)	Small (Ru-Be)	Big (Ru-Be)
$\sum_{i=1}^{i=n} KN^{o}_{i}$	54830	236424	66918	363480
$S=\frac{hi}{3}\sum_{i=1}^{i=n} KN^{o}_{i}$	9.15×10^{4}	3.95×10^{5}	1.12×10^{5}	6.07×10^{5}
Q (n/sec) 10^{6}	0.85±0.076	3.67±0.33	1.04±0.07	5.65±0.28

The obtained result shows that the small (Ra-Be) neutron source of activity 90m Ci gives neutron yield (0.85±0.076) x 10^{6} n/sec. For the sake of intercomparison, Karamyan and Yaritsyna (1964) used 100 m Ci of (Ra-Be) neutron source in graphite moderator together with BF_3 detector. They found that the neutron yield equals 1.03×10^{6}n/sec. Also, Davy (1985) used (Ra-Be) source of acitivity 100 m Ci in MnSO4 solution and 4 π counter. he concluded that the neutron emission equal to 1.0 x 10^{6} n/sec.

So, the obtained data shows slight variation from the absolute numerical values of the neutron yield from similar sources. Such variation does not exceed 10%. This may be attributed to the difference in the methodology used in measurements as well as the efficiency of the different counting systems applied during the neutron source standardization.

CONCLUSIONS

It could be concluded from these results that the total errors do not exceed±10% regardless of the neutron sources of different spectrum. This facility has the following important features:

(1) The total and thermal neutron flux distribution within the region of neutron irradiation have constant symmetric features. This allows irradiation of samples at equi-distances.

(2) This facility proved to be a conventional tool for radioactive production. Also it is of low cost, easy to use and has possibility for activation techniques of trace elements.

(3) To get higher sensitivity for the present method a stronger neutron source is demanded.

Generally, the availability of such facility gives rise to the solution of many problems in the field of neutron research as well as neutron applications.

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