AVERAGE NEUTRON CROSS-SECTION OF ^{27}AL USING ^{241}AM/BE NEUTRON SOURCE

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Abstract

The average neutron cross sections of ^{27}Al(n,tot), ^{27}Al(n,n)^{27}Al and ^{27}Al(n,p)^{27}Mg were computed with the numerical graphical method of neutron source utilizing theoretical values of neutron cross sections. These values were obtained using two codes, SCAT2000 for total and elastic cross section calculations and CINDY code for ^{27}Al(n,p) ^{27}Mg cross section calculation. The global neutron OMPs were obtained using the microscopic optical model code MOM. The average ^{27}Al(n,p) ^{27}Mg cross section was measured with ^{241}Am/Be neutron source and NaI(Tl)"3×3" well type detector. The mean measured value is 25 ± 0.48 mb, which in agreement with the calculated values.

1. Introduction

Neutron Activation Analysis (NAA) is a quantitative and qualitative method, it's based on the nuclear reaction between neutrons and target nuclei. NAA is used to investigate the trace elements in of part per million (ppm) range, although, it's useful method for determination of geological, environmental, biological samples without chemical separation[1].

The importance of NAA in applied physics and technology makes it necessary to have detailed knowledge of the cross-sections for these reactions[2]. The ^{241}Am/Be neutron source is widely used in many fields of applied nuclear physics, and the one the most important in geophysics applications techniques[3].

In recent years ^{241}Am/Be or ^{241}Am-^{242}Cm-Be sources are getting more popular for process control[4].

There are sufficient experimental data in the literature for the average neutron cross-sections of ^{252}Cf source while the data for the ^{241}Am/Be source and other isotopic neutron sources are rather less. So, this work can contribute some data on average neutron cross-sections especially for ^{241}Am/Be source, and may give help in the case of fast neutron activation analysis.

The optical model (OM) is used to describe the interaction of particles with target nuclei by means of potential consisting of real and imaginary parts. The optical model parameter (OMP) provides the basis for many theoretical analyzes and evaluations of nuclear cross sections. The optical model potentials are widely used

in quantum-mechanical pre-equilibrium and direct-reaction theory of neutron shape elastic and total cross sections calculations, and (most importantly) in supplying particle transmission coefficients for Hauser-Feshbach statistical-theory as used in nuclear data evaluations.

The importance of optical model parameter is made even more apparent by the worldwide diminution of experimental facilities for low-energy nuclear physics measurements and the consequent increased reliance on theoretical methods for providing nuclear data for applications. The Reference Input Parameter Library-2[5] accumulate a large body of information that will be useful in optical model calculations and to provide computer codes for processing the information into inputs for commonly used optical model codes.

2. Calculation and Measurement

2.1 Calculation of Total and scattering cross sections

The calculations of total and elastic scattering neutron cross sections for the energy region (according to the neutron source range) of 1 up to 11 MeV in step of 0.4 MeV is based on the modern nuclear models and mechanisms of nuclear reactions, which are performed using the spherical optical model realized in the SCAT2000 code[6].

The inputs OMPs for the SCAT2000 are obtained using the microscopic optical model code (MOM)[7] which is recommended in RIPL-2 OMP library

An optimal set of optical model parameters has been estimated from the analysis of MOM code of neutron total and elastic scattering cross sections. These parameters are shown in Table(1).

Table (1): The Volume Real Potential as Woods-Saxon and other OM Potentials.

POTENTIAL	STRENGTH	r (fm)*	RE. (fm)#	a (fm)$	VOLUME INTEG.
REAL VOLUME V_r	$V_r = -0.32E_n + 55.4269$	3.5092	1.17	.75	$V_r = 535.97-9.85163E_n +0.106414\ E_n^2$
IMAGINARY VOLUME W_i	$W_i = 0.22E_n - 1.56$	3.7791	1.26	.58	$W_i = 2.27177\ E_n - 16.1093$
REAL SURFACE V_s	0.0	0.0	0.0	1.0	0.0
IMAGINARY SURFACE W_s	$W_s = -0.25\ E_n + 12.5635$	3.7791	1.26	.58	$W_s = -4.1568\ E_n + 208.9$
REAL SPIN-ORBIT V_{so}	6.2	3.0293	1.01	0.75	0.0
IMAGINARY SPIN-ORBIT W_{so}	0.0	0.0	0.0	1.0	0.0

* r=radius (fm)

#RE.= linear energy dependence of the radius

$a = diffuseness (fm)

2.2. Calculation of $^{27}Al(n,p)^{27}Mg$ Cross Section

The $^{27}Al(n,p)^{27}Mg$ Neutron Cross Section is calculated using CINDY code[8] for the energy region 1 up to 11 MeV in step of 0.4 MeV .

CINDY program treats the angular momenta according to the statistical model, which is based upon the Hauser – Feshbach – Moldaur formalism for j-j coupling of angular momenta.

The angular momenta of $^{27}Al(n,p)^{27}Mg$ reaction are obtained from table of isotopes[9], and are listed in Table(2).

Table (2): The Angular Momenta Data

Reaction	$\pi_0 J_0$	$\pi_2 J_2$	$\pi_3 J_3$	$\pi_4 J_4$	L_2	L'_2
$^{27}Al(n,p)^{27}Mg$	+2.5	+1.5	+0.5	99.9	1.0	2.0

The optical potentials well-shape parameters employed in Cindy Program are those obtained by Becchetti and Greenlees[10] of neutron and proton as following:

The radius parameters of real, imaginary and spin-orbit are:

r_r = 1.17 fm, r_i = 1.32 fm for neutron , r_i = 0.32 fm for proton and r_{so}=1.01 fm.

The diffuseness parameters of real Woods- Saxon potential and of imaginary and spin-orbit optical potential are:

a_r = 0.75 fm , a_i = 0.51+ 0.7 ξ fm and a_{so} = 0.75 fm

where ξ = (N–Z)/A and E = incident lab energy.

The spin-orbit potential is:

V_{so} = 6.2 MeV

The real and imaginary neutron and proton OMPs are as follows[11]:

$$V_r=50+e^{(2.322458-0.02017 E_n -0.0012 E_n^2)} \quad (1)$$

$$W_r=2.5-12\,\zeta+e^{(0.212535+0.2545322 E_n -0.0070354 E_n^2 -0.000362 E_n^3)} \quad (2)$$

for neutron OMPs, and:

$$V_r=50+e^{(2.2225-0.032 E_p -0.00148 E_p^2)} \quad (3)$$

$$W_r=0.5+12\,\zeta+e^{(0.38737+0.33235 E_p -0.02974 E_p^2 +0.0006023 E_p^3)} \quad (4)$$

for proton.

The Competing Exit Channel Data of Residual Nucleus were as follows:

The parameters of level density energy E_0=-0.35 MeV, level density temperature T=2.08 MeV, level density employed in the high energy expression a=3.45, spin cut-off parameter σ=2.0, and pairing correction like Gilbert-Cameron[12] are used(P=1.8).

2.3 Cross Section Measurement

The choice of isotopic neutron source is partly dictated by the threshold energies of the required reactions. For fast neutron there are three types of reactions that predominate:(n,p),(n,α),(n,2n) the reaction produced by (n,p) generally have thresholds in the range 1 to 3 MeV and therefore cannot produced with thermal neutrons, the thresholds for (n,α) and (n,2n) are even higher and are usually in the range of 10 to 20 MeV[13],so the $^{27}Al(n,p)^{27}Mg$ reaction is measured.

The importance of investigated element appears in using the $^{27}Al(n,p)^{27}Mg$ reaction as monitor reactions for most researchers [see Refs. 14, 15 and 16], and is the one of elements of geochemical interest determined by the fast neutrons[17], and an important in analysis of natural materials in nuclear geophysics[3].

The neutron activation analyses method was used in this work to determine the neutron cross section of $^{27}Al(n,p)^{27}Mg$ by measuring delayed γ-rays. The sample that had been irradiated may reduce the γ-ray intensity by self-absorption. The absorption depends on the sample thickness, therefore, the real sample activity must be calculated. The self-absorption coefficient calculations were performed by using DSAC program[11].

The count rate in the full energy γ-ray peak, at a delayed time t_d after an irradiation of an element j placed in a neutron flux φ_n for an irradiation time t_i is given by[18]:

$$A(t_d)= I_\gamma \varepsilon \varphi_n \sigma \frac{m_j f N_A}{M_j}(1-e^{-\lambda t_i})e^{\lambda t_d} \quad (5)$$

Where I_γ, ε, σ, m_j, f, N_A, and M_j are the γ-ray branching ratio, the detecting efficiency, the cross section of the reaction (in mb), the mass of element j, the target isotopic abundance (%), Avogadro's number, and the atomic weight of the element j respectively.

The umber of counts D, collected during the time interval t_c is given by:

$$D = \frac{A(t_d)}{\lambda}(1-e^{-\lambda t_c}) \quad (6)$$

Then, from Eqs.(5) and (6), the following relationship obtained:

$$D=\frac{m_j f N_A}{M\lambda_j} I_\gamma \varepsilon \varphi_n \sigma \kappa (1-e^{-\lambda t_i})(1-e^{-\lambda t_c})e^{\lambda t_d} \quad (7)$$

Where λ is the decay constant of the formed radionuclides and κ is the correction term of self-absorption coefficient and equal to the self-absorption function $F_p(\mu)$. The relationship between the self-absorption function and self-absorption coefficient μ is obtained as follows[11]:

$$F_1(\mu)= 0.0845302 + e^{-0.0969-0.869\times\mu+0.061336\times(\mu)^2} \quad (8)$$

where p=1 for cylindrical sources (see Ref.[19]).

2.4 Irradiation

The samples were irradiated by ^{241}Am/Be source with neutron yield about 4×10^{7} n/sec. The irradiation was carried out with the sample attached to the source at a distance of 6 cm from the upper point of the cylindrical neutron source parallel to the cylindrical axis (i.e. at the optimal distance).

The scattering effects were avoided by placing the source on a thin rod piece of wood kept about one meter above the floor and 3 meter from the walls. The influence of back scattered thermal neutron component yielding from neutron source was reduced by placing the sample inside a cadmium box with a wall thickness of 0.5 mm during neutron irradiation.

Cadmium, which has large cross-section with thermal neutron (about 2520±50 barn for natural Cd)[20] is expected to absorb most of thermal neutrons. The irradiation period was not less than three times the isotope half-life.

In case of irradiation by neutron source the detection dead time was very small, so, the irradiated samples were measured directly (i.e. the delay time was as short as possible).

Because ^{241}Am/Be neutron source has a long half-life, its flux intensity was assumed to be constant.

The measurement of the spectra were performed with 3″×3″ NaI(Tl) well-type with 1.8 cm diameter × 6 cm deep well, the pulses were recorded using SPECTRA-2000 MCA which permits 4096 channels of stored data.

2.5 Average Cross-Section Calculation

When using isotopic neutron sources it is necessary to take into account variations of the neutron emission with time and modifications in the spectrum caused by interactions of neutrons inside the source[21]. Therefore, the neutron source averaged cross-sections are the effective values of reaction cross-sections for a given neutron source spectrum[3].

The calculations of average neutron cross-section for ^{241}Am/Be Source were performed using the intensity distribution of ^{241}Am/Be neutron source as a function of neutron energy, which is described by De Guarrini & Malaroda[22] and Radiochemical Center, Amersham,[23] as shown in Figures (1-a,b). Also, the corresponding neutron cross-section values at certain neutron energies were taken from the calculated neutron cross-section curves.

The average cross-section σ_{ave} of a reaction in a continuous neutron spectrum is defined by [22]:

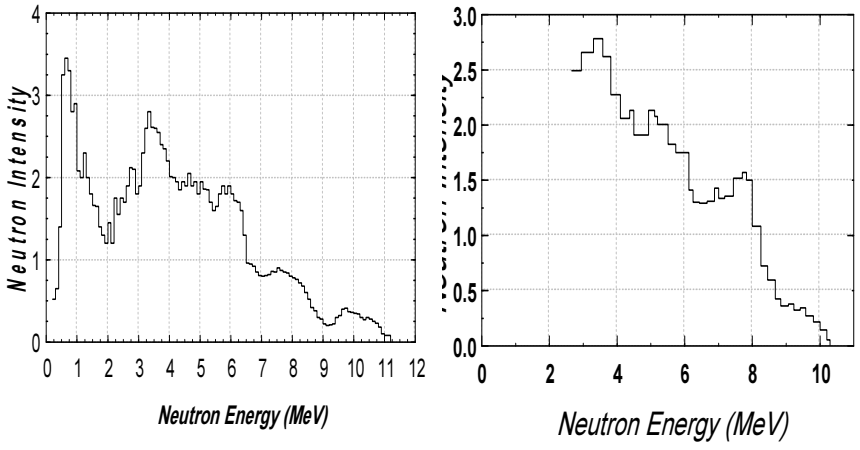
$$\sigma_{ave} = \frac{\int_{0}^{E\max} \sigma(E)\, N(E)\, dE}{\int_{0}^{E\max} N(E)\, dE} \qquad (9)$$

Where $\sigma(E)$ is the excitation function of the reaction, and N(E) is the energy distribution of neutrons.

An approximation of σ_{Ave} can be derived from:

$$\sigma_{ave} = \frac{\sum_{j=1}^{\gamma} \sigma_j N_j}{\sum_{j=1}^{\gamma} N_j} \qquad (10)$$

Where N_i is the relative neutron intensity integrated over the j^{th} group and σ_i



(a) (b)

Figure (1): Neutron spectra of ^{241}Am/Be source were investigated by:
(a) De Guarrini & Malaroda (1971).
(b) Radiochemical Center, Amersham 1976).

is an appropriate average value of the excitation function in this group.
The neutron spectra shown in Figs. (1-a, b) were fitted using the statistical model of exponential growth[24] and dividing spectra into various neutron energy regions as follows:

1-Spectrum (a):

$I(n)=0.412443+\exp(1.590191-0.8941592E_n)$ for $0.2 \le E_n \ge 2.0$ MeV
$I(n)=-60.11663+\exp(4.1374405-0.0017717E_n)$ for $2.0 \le E_n \ge 5.0$ MeV
$I(n)=-121.366+\exp(4.846641-0.005891E_n)$ for $5.0 \le E_n \ge 7.0$ MeV
$I(n)=-52.197+\exp(4.034331-0.008225E_n)$ for $7.0 \le E_n \ge 9.0$ MeV
$I(n)=-24.8261+\exp(3.2896725-0.00657E_n)$ for $9.0 \le E_n \ge 11.1$ MeV

2-Spectrum (b):

$I(n)=-4.455524+\exp(0.5064+0.925E_n-0.145343E_n^2)$ for $2.66 \le E_n \ge 4.0$ MeV
$I(n)=0.5051+\exp(-4.2667+2.0924E_n-0.23E_n^2)$ for $4.0 \le E_n \ge 6.5$ MeV
$I(n)=1.272+\exp(-1652.683+478.88E_n-34.7312E_n^2)$ for $6.5 \le E_n \ge 7.0$ MeV
$I(n)=1.39+\exp(-510.336+130.95533E_n-8.4221E_n^2)$ for $7.0 \le E_n \ge 8.0$ MeV
$I(n)=0.07+\exp(34.673-6.60116E_n+0.29E_n^2)$ for $8.2 \le E_n \ge 10.33$ MeV

where I(n) and E_n are neutron intensity and neutron energy respectively.

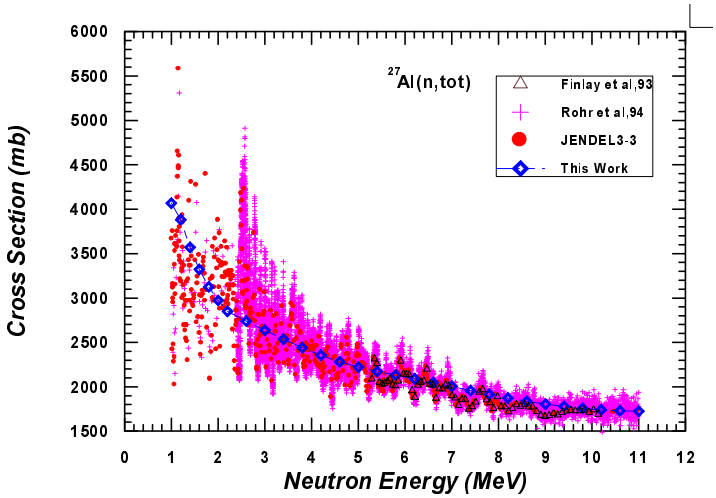
The calculations of the average neutron cross-sections were performed using ANCSC program, which is built according to the average numerical graphical of neutron energy spectrum of the source[see Ref. 11].

3. Results and Discussion

3.1 The Theoretical Neutron Cross Sections

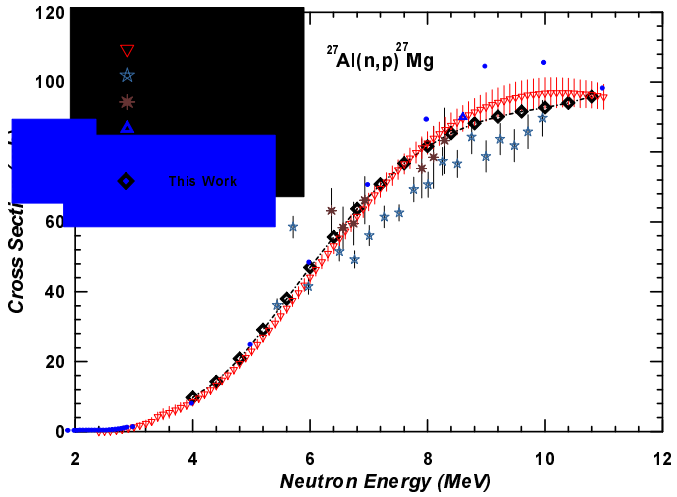
The calculated values for total neutron cross sections are shown in figure (2).

The present result for the ^{27}Al(n,tot) reaction are in agreement with that measured by Finlay et al[1993][25] and in good agreement with the evaluated values of Rohr et al [1994][26] and JENDL-3.3[27].

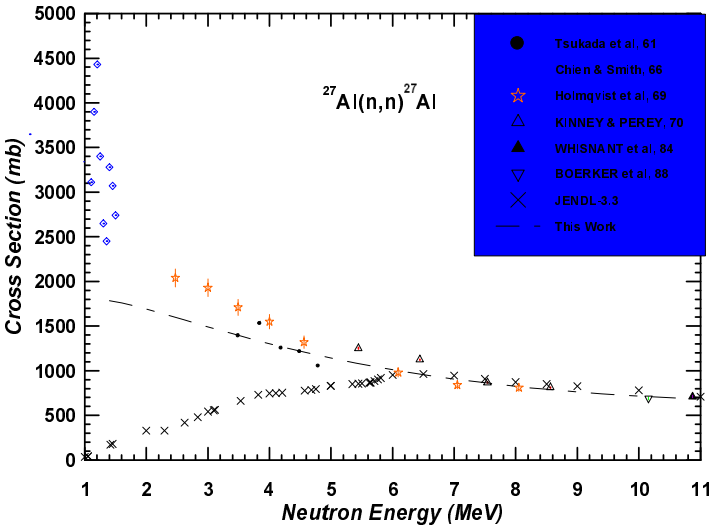


Figure(2): The calculated Total neutron cross section values compared with other works.

The obtained elastic cross section values for ^{27}Al(n,n)^{27}Al reaction are compared with experimental and evaluated data as shown in figure(3). The present results are comparable to some extent with the measurements of Tsukada et al [1961][28], and are in good agreement with the measurements of Holmqvist et al [1969][29], Kinney and Perey [1970][30], Whisnant et al [1984][31] and Boerker et al [1988][32] at energy range greater than 6 MeV, but disagrees with the measurements of Chien & Smith [1966][33], it's also observed that the results are comparable to that of JENDL-3.3[27].



Figure(4): The neutron cross section in mb versus neutron energy in MeV for ^{27}Al(n,p)^{27}Mg reaction.



Figure(3): The calculated Elastic neutron cross section values compared with other works.

The calculated cross section values of ^{27}Al(n,p)^{27}Mg reaction is shown in figure(4), the theoretical values are compared with the experimental data which appear a good agreement with the values obtained by Lapena et al(1975)[34] and are in

acceptation agreement with measurements of Smith et al(1975)[35], Dbradley et al(1985)[36] and Enz et al(1985)[37].

The present values are in good agreement with the evaluated values of JENDL-3.3[27] at energies less than 8 MeV.

3.2 The Computed Average Neutron Cross Sections

The computed average numerical graphical neutron cross–section values are presented in table(3), These are obtained according to calculated curvature cross–sections values of figures(2, 3 and 4).

Table(3): The Mean Numerical Graphical Cross–Section Values Corresponding to the Neutron Spectra Of Fig.(1. a, b) Compared With Other Works.

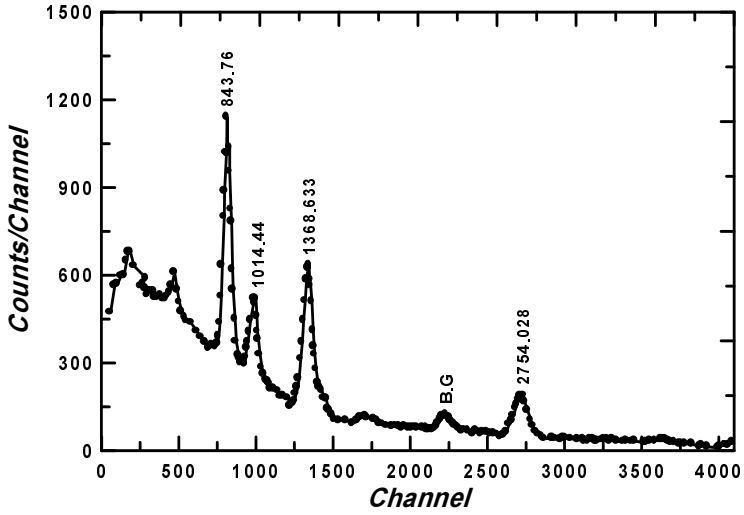
Reaction	Spectrum(a)		Spectrum(b)	
	This Work	Other work	This Work	Other work
	σ_{av1} (mb)		σ_{av2} (mb)	
$^{27}Al(n,tot)$	2455.51	-	2433.5	2507.55,Ref.[21]
$^{27}Al(n,n)^{27}Al$	1275.24	-	1262.8	1866.57,Ref.[21]
$^{27}Al(n,p)^{27}Mg$	25.68	25 ,Ref.[38]	26.89	28.46,Ref.[21], 28,Ref.[38]

The results of the computations show an acceptable differences between the results of this work and others.

The differences between the computed mean cross– section of the present study and those calculated by Riepo (1979)[38] could be due to the numerical graphical method of calculations employed in this work. Another reason could be the differences between the theoretical curvatures neutron cross–section data used in this work.

3.3 The Measured $^{27}Al(n,p)^{27}Mg$ Cross Section

The elemental pure powder of Al samples was used as a reference. The γ-spectrum was measured as shown in figure(5). The irradiation, waiting, and measuring times were 44h, 40sec, and one hour respectively. The photo peaks of $^{27}Al(n,p)^{27}Mg$ [Eγ = 843.76 keV, 1014.44 keV] and $^{27}Al(n.\alpha)^{24}Na$ [Eγ = 1368.83 keV, 2754 keV] are observed. The average cross–section of $^{27}Al(n.\alpha)^{24}Na$ reaction is calculated for tow spectra using the cross sections data of JENDL-3.3 libraray[25] it is 8.93 ± 0.6 mb and agrees with that 9.66 mb value which calculated by Kocherov[1993][21]. The average cross–section of $^{27}Al(n,p)^{27}Mg$ reaction was measured using the self reference method (i.e self monitor) by utilizing the 843.76 keV relative to $^{27}Al(n,\alpha)^{24}Na$ [Eγ = 1368.633 keV]. The value obtained is 25 ± 0.48 mb. This value is greater than 17.5±3.5 mb which was measured by Rieppo (1984)[39] and quite agrees with that calculated value



Figure(5) γ-ray spectrum of $^{27}Al(n,p)^{27}Mg$ and $^{27}Al(n,\alpha)^{24}Na$ reactions collected for one hour. The energies are in keV

4. Conclusion

The advantages of this work appear in using an updated programs to obtain the OMPs and to calculate cross section values, although, in using the self absorption at measuring the activation cross section.

In general, the agreements between the average cross–section values obtained by numerical graphical method and those obtained by activation method are fairly good apart and the numerical graphical method is a good process to estimate the average cross–section experimentally.

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