

Thermal neutron radiative capture cross-section of $^{186}\text{W}(n, \gamma)^{187}\text{W}$ reaction

V H Tan¹ and P N Son²

¹ Agency for Radiation and Nuclear Safety, 113-Tran Duy Hung, Hanoi, Vietnam

² Nuclear Research Institute, 01-Nguyen Tu Luc, Dalat, Vietnam

E-mail: pnson.nri@gmail.com

Abstract. The thermal neutron radiative capture cross section for $^{186}\text{W}(n, \gamma)^{187}\text{W}$ reaction was measured by the activation method using the filtered neutron beam at the Dalat research reactor. An optimal composition of Si and Bi, in single crystal form, has been used as neutron filters to create the high-purity filtered neutron beam with Cadmium ratio of $R_{\text{cd}} = 420$ and peak energy $E_n = 0.025$ eV. The induced activities in the irradiated samples were measured by a high resolution HPGe digital gamma-ray spectrometer. The present result of cross section has been determined relatively to the reference value of the standard reaction $^{197}\text{Au}(n, \gamma)^{198}\text{Au}$. The necessary correction factors for gamma-ray true coincidence summing, and thermal neutron self-shielding effects were taken into account in this experiment by Monte Carlo simulations.

1. Introduction

Thermal neutron radiative capture cross sections are important data for safety analysis and design of reactors as well as in the studies of nuclear physics, and the nucleosynthesis process in astrophysics [1]. The previous measured values are quite discrepant from different experiments, and still not completely satisfied in quality and quantity for related applications. That is the reason why new experimental data and theoretical calculations to improve the neutron induced reaction cross-sections are necessary. In this study, the thermal neutron capture cross section for the reaction of $^{186}\text{W}(n, \gamma)^{187}\text{W}$ has been measured by the activation method with the filtered neutron beam at the Dalat nuclear research reactor. A modern gamma-ray spectroscopy (ORTEC DSPEC) in compact with a high-efficiency HPGe detector was used to measure the specific radioactivity of the irradiated samples. The necessary correction factors for gamma true coincidence summing, and neutron self-shielding effects were taken into account in this experiment by Monte-Carlo simulations using the well know MCNP5 code. The new measured value of thermal neutron capture cross section for ^{186}W nucleus was obtained relatively to the reference value of $\sigma_0 = 98.65 \pm 0.09$ (barn) for the $^{197}\text{Au}(n, \gamma)^{198}\text{Au}$ standard reaction [2].

2. Experimental

The neutron filter technique [3] has been applied to derive a pure thermal neutron beam at the horizontal channel No.2 of the Dalat research reactor (DRR), Dalat Nuclear Research Institute. The single-crystals of 80 cm silicon and 6 cm bismuth were used as neutron filters, for which the epithermal component is absorbed or scattered out of the beam line. The measured thermal neutron flux at outer position of the beam line is 1.6×10^6 n/cm²/s, and the value of Cadmium ratio $R_{\text{cd}}(\text{Au})$ is



420. Accordingly, the contribution of reaction rates by epithermal neutron component to the total reaction rate was less than 0.25%, which is much smaller than other sources of uncertainties, and may be negligible. The physical parameters of the thermal neutron beam line in comparison with others facilities are given in Table 1.

Table 1. Characteristics of the thermal neutron beam at the radial channel No.2 of DRR, in comparison with others facilities

Parameters	Thermal neutron beams		
	Channel No.2 of DRR (<i>this work</i>)	Channel No.4 of DRR	Ohio Reactor [3]
Thermal neutron flux ϕ_{th}	1.6×10^6 n/cm ² .s	1.7×10^6 n/cm ² .s	8.5×10^6 n/cm ² .s
Cadmium ratio $R_{Cd}(Au)$	420	112	266
Filter materials	6cm Bi + 80cm Si	98cm Si + 1cm Ti + 35g/cm ² S	10.16cm Bi + 12.7cm Sapphire
Beam diameter	3cm	3cm	3cm

The samples were prepared from the high-purity metal foils of Au (99.99%, 1.27 cm in diameter, 0.05 mm in thickness) as a standard for neutron flux monitor and W (99.98%, 1.27 cm in diameter, 0.05 mm in thickness) as the target, by the SHIELDWERX company. Each W sample was sandwiched between two Au foils. The irradiation time was five hours for every sample group. The specific activities of the targets and the gold foils were measured by using the DSP-base gamma-ray spectrometer, in compacting with a HPGe detector (58% efficiency, 2.5keV energy resolution at 1.332 MeV of ⁶⁰Co). The detector was calibrated by using a set of standard radioisotope sources: ¹⁵²Eu, ⁶⁰Co, ⁵⁷Co, ¹³⁷Cs, ²⁴¹Am, ¹¹³Sn and ⁸⁸Y. The absolute detection efficiency function for 5 cm distance from the end-cap of the detector was determined by Monte Carlo simulation method using the Geant4 toolkit [4]. The results of simulation and measurement of the efficiency curve are illustrated in Figure 1. The correction factors for neutron multiple scattering and self-shielding in the samples were also calculated by the Monte Carlo method, using the well-know MCNP5 code.

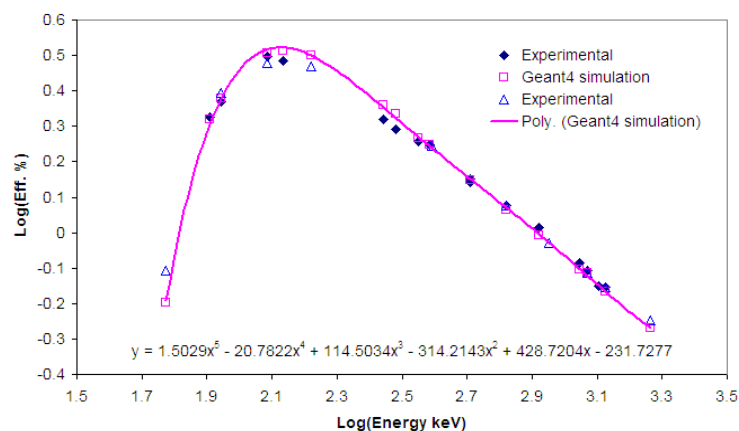


Figure 1. The absolute detection efficiency curve of the HPGe detector for foil sample at 5 cm from the detector end-cap

3. Data analysis

During irradiation in a neutron beam with energy spectrum $\phi(E)$, the capture reaction rate, R , of samples with neutron is defined as follows:

$$R = N \int \phi(E) \sigma_a(E) dE, \quad (1)$$

where N is the number of nuclei in the sample, and $\sigma_a(E)$ is the neutron capture cross section as a function of incident neutron energy E . The average thermal neutron capture cross section, σ_0 , and neutron flux, Φ , are defined as following:

$$\sigma_0 = \frac{\int \sigma_a(E) \phi(E) dE}{\int \phi(E) dE}; \quad \Phi = \int \phi(E) dE. \quad (2)$$

Applying these average quantities, the integral equation (1) can be rewritten as follows:

$$R = N \sigma_0 \Phi \quad (3)$$

The activity, A , of the irradiated sample at the end of neutron irradiation is calculated by expressions:

$$A = R(1 - e^{-\lambda t_1}), \quad \text{and} \quad A = \frac{C f_c \lambda}{\epsilon_\gamma I_\gamma e^{-\lambda t_2} (1 - e^{-\lambda t_3})}, \quad (4)$$

where C denotes the net counts of the corresponding gamma peak; t_1 , t_2 and t_3 are irradiating, cooling and measuring times, respectively; λ is the decay constant of the product nucleus; ϵ_γ is the detection efficiency of the detector; I_γ is the intensity of interesting gamma peak; f_c is the correction factor for the effects of neutron multiple scattering and self-shielding in the irradiated samples. Finally, from equations (2-4), the thermal neutron capture cross sections σ_0 can be determined relative to that of ^{197}Au standard by the following relations:

$$\sigma_0^x = \frac{C^x f(\lambda, t)^x \int_c^x I_\gamma^{Au} \epsilon_\gamma^{Au} N^{Au} \sigma_0^{Au}}{C^{Au} f(\lambda, t)^{Au} \int_c^{Au} I_\gamma^x \epsilon_\gamma^x N^x}, \quad (5)$$

$$f(\lambda, t) = \frac{\lambda}{(1 - e^{-\lambda t_1}) e^{-\lambda t_2} (1 - e^{-\lambda t_3})}, \quad (6)$$

where the superscript 'x' denotes the nucleus of sample. The standard thermal capture cross section of ^{197}Au is extracted from the reference [2]. The relevant decay data of product nuclei used in this work are given in Table 2.

Table 2. Decay properties of the activation product nuclei [5]

Activation products	Half-life	γ -ray (keV)	Intensity (%)
^{198}Au	2.6948 ± 0.0012 (d)	411.8	95.62
^{187}W	24.000 ± 0.004 (h)	479.53	26.6 ± 0.4

4. Results and Discussion

The thermal neutron radiative capture cross-section of $^{186}\text{W}(n, \gamma)^{187}\text{W}$ reaction was obtained by the neutron filter technique and activation method. The Monte-Carlo simulation has been also performed for exact determinations of detector efficiency curve and correction parameters of neutron multiple scattering and self-shielding inside the irradiating samples. The new measured value of cross section is

given in the Table 3, which is almost agreement with the previous experimental and evaluated values. In this experiment, the total uncertainty of cross-section value is about 3.5%, mainly due to the statistical errors (0.1-0.5%), the uncertainties of gamma-ray detection efficiency (1.5%), the reference cross-section of the monitor sample (1.5%) and the correction factors for self-shielding and multiple scattering effects (1.0%).

Table 3. The result of thermal neutron capture cross section for the $^{186}\text{W}(n, \gamma)^{187}\text{W}$ reaction, in comparison with previous measured and evaluated values

Reaction	Neutron energy (eV)	Authors	σ_0 (barn)	Year
$^{186}\text{W}(n, \gamma)^{187}\text{Mo}$	0.0253	This work	37.5 ± 1.3	2015
$^{186}\text{W}(n, \gamma)^{187}\text{Mo}$	0.0536	M.S.Uddin [6]	26.6 ± 1.6	2008
$^{186}\text{W}(n, \gamma)^{187}\text{Mo}$	0.0253	S.J.Friesenhahn [7]	37.0	1966
$^{186}\text{W}(n, \gamma)^{187}\text{Mo}$	0.0253	JENDL 4.0 [8]	38.099	2011
$^{186}\text{W}(n, \gamma)^{187}\text{Mo}$	0.0253	ENDF/B-VII.1 [9]	38.0949	2011

5. Conclusion

The thermal neutron capture cross-section of $^{186}\text{W}(n, \gamma)^{187}\text{W}$ reaction at incident neutron energy of 0.0253 eV was determined to be 37.5 ± 1.3 (barn) by activation method using the new filtered thermal neutron beam at the radial channel No.2 of the Dalat research reactor. The Monte-Carlo simulation has been also performed for the experimental setups, detector efficiency determination, and correction factor calculations.

Acknowledgments

This research is funded by Vietnam National Foundation for Science and Technology Development (NAFOSTED) under grant number “103.04-2012.59”.

References

- [1] Agbemava S E, Nyarko B J B, Fletcher J J, Sogbadji R B M, Mensimah E and Asamoah M 2011 *Annals of Nuclear Energy* **38** 1616
- [2] IAEA 2007 “*International Evaluation of Neutron Cross-Section Standards*” ISBN 92-0-100807-4
- [3] Simpson O D and Miller L G 1986 *Nucl. Instr. and Meth.* **61** 245
- [4] Nikolic J, Puzovic J, Todorovic D and Rajacic M 2015 *Nucl. Instrum. Methods Phys. Res. Sect. A* **799** 159
- [5] Nudat 2.2 Database, <http://www.nndc.bnl.gov/nudat2/>
- [6] Uddin M S, Chowdhury M H, Hossain S M, Latif S A, Hafiz M A, Islam M A, Zakaria A K M and Azharul I S M 2008 *Applied Radiation and Isotopes* **66** 1235
- [7] Friesenhahn S J, Haddad E, Froehner F H and Lopez W M 1966 *Nuclear Science and Engineering* **26** 487
- [8] Keiichi S et al 2011 *Journal of Nuclear Science and Technology* **48** 1
- [9] Chadwick M B et al 2011 *Nuclear Data Sheets* **112** 2887