Experimental and theoretical cross sections of $^{115}In(n, n')$ reaction at 19 and 16 MeV using quasi-monoenergetic neutrons

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Introduction

Indium(In) is used in an alloy that is popular for making control rods in the nuclear reactor. The control rod is the key component of a reactor which controls the fission rate of the fuel to sustain the optimal operation of the reactor. Therefore the knowledge of composition material of the control rod and interaction probability of neutron is quite important. In is also widely being used for the measurement of neutron flux for the calculation of neutron-induced reaction cross sections. So the accurate determination of neutron-induced reaction data is very crucial in nuclear reactor technology. The EXchange FORmat (EXFOR) [1] data repository shows that there is a considerable amount of cross-section data available for the $^{115}In(n, n')^{115m}In$ reaction upto 20 MeV neutron energy, all of which have high mutual disparity. As a result, precise cross-section measurements for various neutron-induced reactions for the In isotopes are critical. The measurement of (n, n') reaction cross section for the ^{115}In is presented in this paper using quasi-monoenergetic neutrons produced via $^7Li(p, n)$ reaction at an average neutron energy of 19 and 16 MeV. The reaction cross-section stated above was also compared to literature data taken from EXFOR [1] and with the results of theoretical model codes, TALYS-1.95 [2] and EMPIRE-3.2.3 [3]. The effect of different level density models was also evaluated for a better representation of the current and literature data.

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Experimental Details

The experiment was performed using the 14UD Pelletron accelerator facility at BARC-TIFR, Mumbai, India using the method of activation followed by offline γ-ray spectroscopy. Quasi-monoenergetic neutrons were generated by using the $^7Li(p, n)$ *reaction* (E_{th} = 1.88 MeV) with a proton beam of 21 and 18 MeV energy. A Lithium foil of thickness $\approx 8.0\ mg/cm^2$ was sandwiched between two Tantalum (Ta) foils having a thickness of $\approx 4.0\ mg/cm^2$ at the front side and $\approx 0.1\ mm$ at the backside of the Li to stop the proton beam. The natural In foil of thickness $\approx 0.1\ mm$ was used for the irradiation. The stack of $Ta - Li - Ta$ and In sample were wrapped in thin aluminum foils and irradiated for about six hours at the 6 meter port of the main beam line of the pelletron to build up sufficient activity. After the irradiation, the sample was allowed to cool for adequate time and then after counted using the pre-calibrated single-crystal HPGe detector coupled with a PC-based 4K multi-channel analyzer. Standard multi-gamma ^{152}Eu source was utilized for the energy and efficiency calibration of the detector and the resolution of the detector was measured to be 1.88 keV at

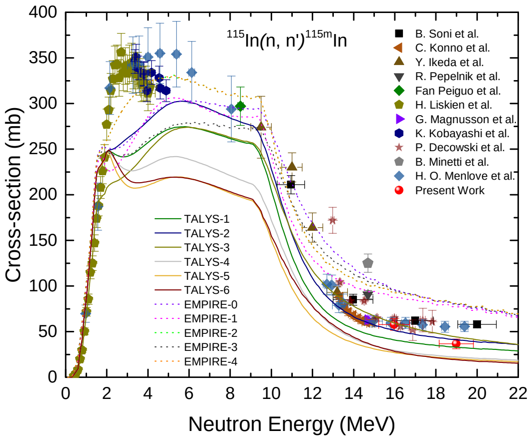

FIG. 1: Comparison of our present experimental result with the literature data and the theoretical calculations using the TALYS-1.95 [2] and EMPIRE-3.2.3 [3].

1332.50 keV γ of ^{60}Co. The sample was placed at a relevant distance from the detector head so that the dead time of the detector is $< 2\%$.

Data Analysis

The production of neutron spectra using the $^{7}Li(p,n)$ reaction has been discussed broadly elsewhere [4]. The $^{113}In(n,n')^{113m}In$ reaction was used for the evaluation of neutron flux. The neutron flux was calculated using the weighted reaction cross-sections from the EXFOR [1] utilizing the decay data of the 391.70 keV γ-line of ^{113m}In ($t_{1/2}$ = 99.48 $minutes$). The cross section for the reaction $^{115}In(n,n')^{115m}In$ has been calculated using this neutron flux and decay data of the 336.24 keV γ-line of the ^{115m}In ($t_{1/2}$ = 4.49 $hours$) in the following relation,

$$\sigma = \frac{A_\gamma \lambda(\frac{t_c}{t_r})}{N_0 \phi I_\gamma \epsilon (1-e^{\lambda t_i})(1-e^{-\lambda t_c})e^{-\lambda t_w}} \quad (1)$$

where all the symbols have their usual meanings. The contribution from lower energy neutrons to the reaction cross section was eradicated using a method of tailing correction [4]. The error propagation method [5, 6] was availed to calculate the uncertainty in this measurement, which came out to be 11.08 % and 14.46 % at neutron energy of 15.97 $\pm$ 0.75 MeV and 18.99 $\pm$ 0.83 MeV respectively.

Result and Discussion

The reaction cross section for $^{115}In(n,n')^{115m}In$ at average neutron energy of 18.99 $\pm$ 0.83 MeV and 15.97 $\pm$ 0.75 MeV was measured as 36.60 $\pm$ 5.29 (mb) and 57.74 $\pm$ 6.39 (mb) respectively. The present result has been compared with the literature data from EXFOR [1] and also with the results of theoretical codes using TALYS-1.95 [2] and EMPIRE-3.2.3 [3] as shown in Fig. 1. It is clear from the Fig. 1 that the measured cross sections follow the trend of literature data. Different level density (LD) models of TALYS-1.95 and EMPIRE-3.2.3 were used to explain the present result. It can be seen that the TALYS-1 and TALYS-2, LD model explains both the experimental results within experimental error whereas other TALYS-LD models underestimate the value in comparison with the present work. The results from different LD models of EMPIRE are overestimated as compared with the present result and also do not agree with the literature data at higher energies.

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