(n, γ) cross-section for ^{238}U at neutron energy of 11.90 MeV

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Improved and accurate neutron reaction cross-section of actinides are required for design of innovative reactor systems including fast breeder reactors and advanced heavy water reactor (AHWR) [1, 2]. In AHWR ^{232}Th-^{233}U in the oxide form is the primary fuel, whereas in the fast reactor ^{238}U-^{239}Pu in the form of carbide is used as the primary fuel. The ^{239}Pu is first generated in a thermal reactor from ^{238}U(n, γ)^{239}U reaction followed by two successive β decays. Then the fissile material ^{239}Pu along with ^{238}U is used as a fuel in fast reactor for power generation. Thus for the production of ^{239}Pu, it is necessary to have knowledge about ^{238}U(n, γ) reaction cross section at various neutron energies. In the present work ^{238}U(n, γ) reaction cross-section was determined at average neutron energy of 11.90±0.35 MeV using neutron beam from ^{7}Li(p, n) reaction and by employing activation technique followed with off-line γ-ray spectrometry.

The experiment was carried out using 14UD BARC-TIFR Pelletron facility at Mumbai, India. Neutron beam required for the study was obtained using ^{7}Li(p, n) reaction. The natural uranium metal foil weighing about 0.3441 g was irradiated for 7 h with proton beam energy of 14 MeV and a proton current of 300 nA. After irradiation and sufficient cooling, the γ-rays of fission/reaction products from the irradiated U sample were counted using energy and efficiency calibrated 80 c.c. HPGe detector coupled to a PC-based 4K channel analyzer.

The half-life of ^{239}U is 23.54 min., which decays 99.6% to ^{239}Np within 3 h. In view of this, ^{238}U(n, γ) reaction cross-section was calculated from the observed photo-peak activity of ^{239}Np and was identified through the characteristic γ-lines of 103.73, and 277.85 keV. The fission products from ^{238}U have varying half-lives. The net photo-peak area (A_{net}) for γ –lines of ^{239}Np and for different γ-lines of fission products (e.g. 743.3 keV of ^{97}Zr) were obtained from their total peak area after subtracting the linear background due to Compton effects. From the A_{net} of a particular fission products (e.g. ^{97}Zr), neutron flux (φ) was obtained using decay equation (1)

$$A_{net} = \frac{N\sigma\varphi Y\varepsilon a(1-e^{-\lambda t})e^{-\lambda T}(1-e^{-\lambda \Delta T})}{\lambda} \quad (1)$$

where N is the number of atoms of the isotope of the element and σ is the fission cross-section of ^{238}U. Y is the cumulative yield of ^{97}Zr. 'ε' is the detection efficiency, 'a' is the γ-ray abundance and λ is the decay constant of the product nuclide. 't', T and ΔT are irradiation, cooling and counting time respectively.

The incident proton energy in the present experiment was 14 MeV. The neutrons from ^{7}Li(p, n) reaction are not mono-energetic, and their energy spectra were obtained from literature [3]. The average neutron energy was obtained as 11.90±0.35 MeV after removing the tailing distribution of the neutron spectrum below 10.5 MeV. The neutron flux (Φ) of (1.30±0.02) X10^{7} n cm^{-2} s^{-1} was used to calculate the ^{238}U(n, γ) reaction cross-section ,which is 3.50±0.18 mb.

For ^{238}U(n, γ) reaction, the low energy neutrons also contribute to the cross-section. This contribution from the tail was estimated using the ENDF/B-VII.0 [4] and JENDL-4.0 [5] by folding the cross-sections with neutron flux distributions of Ref. [3]. The contributions from the above evaluations are 2.38 mb and 1.81 mb from ENDF/B-VII.0 and JENDL-4.0 respectively. The actual cross-section was obtained as 1.12±0.18 mb, which are given in Table 1. The evaluated ^{238}U(n, γ) and reaction cross-sections from ENDF/B-VII.0, JENDL-4.0 JEFF 3.1/A [6] and CENDL-3.1[7] are also given in Table 1 for comparison. It can be seen from Table 1 that the measured ^{238}U(n, γ) reaction cross-section is almost within the range of the evaluated data of ENDF/B-VII.0 and JENDL 4.0 and JEFF 3.1/A. But, the evaluated ^{238}U(n, γ) reaction cross-section from

CENDL-3.1 are not in good agreement with the present experimental value.

Table 1. ^{238}U (n, γ) reaction cross-section at 11.90 MeV neutron energy in mb.

Expt	ENDF/B -VII.0	JENDL-4.0	JEFF3.1 /A	CENDL -3.1
1.12± 0.18	1.24-1.05	1.0-0.79	1.23-1.02	0.87-0.97

To examine this, ^{238}U(n, γ) reaction cross-sections from the present work and similar data from literature given in EXFOR [8 are plotted in Fig. 1. It can be seen from Fig. 1 that the ^{238}U(n, γ) reaction cross-section at neutron energy of 11.90 MeV is in agreement with the value of Mc Daniels et al [9] at 11.2-12.2 MeV.

Theoretically the ^{238}U(n, γ) reaction cross-sections at different neutron energy beyond 100 keV were also calculated using the nuclear model based computer code TALYS 1.2 [10] and is plotted in Fig. 1. It can be seen from the figure that the trend of the experimental and evaluated ^{238}U(n, γ) reaction cross-section are well produced by the TALYS 1.2. However, The theoretical ^{238}U(n, γ) reaction cross-section from TALYS are slightly higher than the experimental and evaluated values for neutron energy from 100 keV to 2 MeV. Further, it can be seen from Fig. 1 that the experimental, evaluated and the theoretical ^{238}U(n, γ) reaction cross-sections decrease from 100 keV to 7 MeV and predict a dip around 7.3-8.0 MeV. Beyond 8.0 MeV, it increases up to neutron energy of 14 MeV and then again decreases. The dip in the ^{238}U(n, γ) reaction cross-section around neutron energy of 7.5-8.5 MeV indicates the opening of (n, 2n) reaction channel besides (n, nf) channel.

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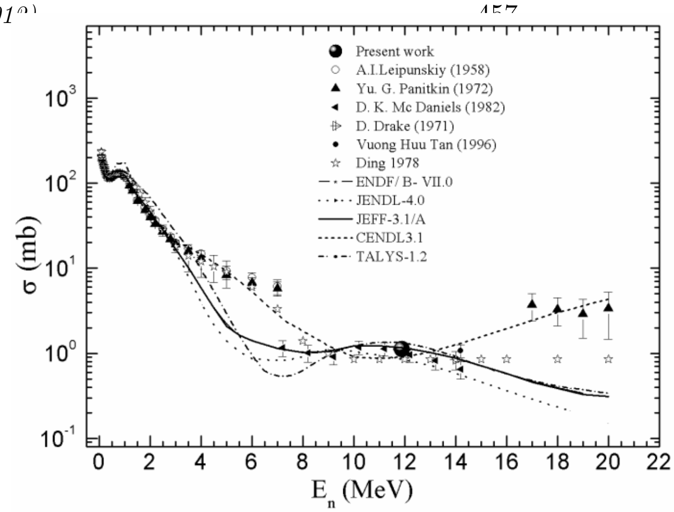

Figure 1: Plot of ^{238}U(n, γ) reaction cross-section as a function of neutron energy

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