^{238}U(n,γ) reaction cross-section at the neutron energy 8.96 MeV

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Introduction

Uranium has two isotopes ^{235}U with 0.71% and ^{238}U with 99.3% abundance. ^{238}U being the fertile material needs to be acted upon to convert it into fissile material. So it can be used as fuel in nuclear power plants for the generation of electricity. This is one of the new concepts accepted worldwide for the nuclear power generation and major efforts are going on in this direction. Among them, the accelerator driven sub-critical system (ADSs), fast reactor, compact and high temperature reactors and advanced heavy water reactor (AHWR) are the most important for power production. Besides power production, ADSs is important for incineration of the long-lived actinides and transmutation of the long-lived fission products as well. For the design of such reactors nuclear data such as reaction and fission cross-sections of structural materials, cladding materials and fuel elements with medium to fast neutron energies are important [1-3]. The cross-section for capture of thermal neutrons in ^{238}U is typically 2.47 times that in ^{232}Th. Thus uranium offers greater competition for the capture of the neutrons and lower losses to structural and other parasitic materials leading to an improvement in conversion of ^{238}U to ^{239}Pu.

In the present work, we have determined the ^{238}U(n,γ) reaction cross-section at the neutron energy of 8.96 MeV by using the off-line γ-ray spectrometric technique. Present study also describes the excitation function ^{238}U(n,γ) reaction calculated using theoretical model code TALYS-1.8 [4-5]. Both experimental and theoretical results were compared with the existing data from the data libraries; ENDF/B-VII-1 and JENDL-4.0 and show a good agreement and are discussed in detail in the results and discussions section [2].

Current Status of present work

An examination of International Atomic Energy Agency-Exchange Format (IAEA-EXFOR) database shows that a significant discrepancy exits in the measured experimental data for many neutron threshold reactions in MeV region and at thermal energies. Furthermore, literatures available in IAEA-EXFOR database show that most of the thermal neutron activation cross-sections were made in reactors with neutron spectra and therefore these data are not precise thermal cross-section measurements [2].

The detailed literature survey on cross-section measurements related to U-Pu fuel cycles show that the exhaustive experimental work has been carried out in low energy neutron induced fission of actinides [3,4]. However, the experimental data in medium to high energy neutron induced fission of actinides, which has immense importance for the design of advanced reactors and ADSs are rare and very much limited [2-3].

Experimental Details

The experimental part of the present work was carried out using the 14UD BARC-TIFR Pelletron facility at Mumbai, India. The neutron beam was obtained from the ^{7}Li(p, n) reaction. The energy spread for the proton at 6m beam line was nearly 50–90 keV. Further, we use a collimator of 6mm diameter before the target. The size of the ^{238}U metal foil was 1.0×1.0 cm^2 and weight of 0.4473gm. The γ-ray activity of fission monitor reaction was used to measure the neutron flux. The Ta-Li-Ta and U was irradiated for 16 hours and 5 minutes. The proton beam energy was 11 MeV. The proton current during the irradiations was 120 nA. After irradiation, the samples were cooled. Then, the irradiated target along with the Al wrapper was mounted on Perspex plate and taken for γ-ray spectrometry. The γ-rays of fission/reaction products from the irradiated U sample was counted in an energy- and efficiency-calibrated 80 c.c. HPGe detector coupled to a PC-based 4K channel analyzer [6-7].

Calculations

From the photo-peak activity of 277 keV γ-ray of ^{238}Np, the ^{238}U(n,γ) reaction cross-section (σ) was calculated by using the usual decay equation [6-7]. The fission monitor was used for the neutron flux calculation.

Results and Discussion

^{238}U(n,γ) reaction cross-section obtained from the present work at the neutron energies of 8.96 MeV are shown in Table I. The ^{238}U(n,γ) reaction cross-section at average neutron energy of 8.96 MeV in the present work is determined for the first time and it falls well within the range of available experimental as well as theoretical data (Fig. 1).

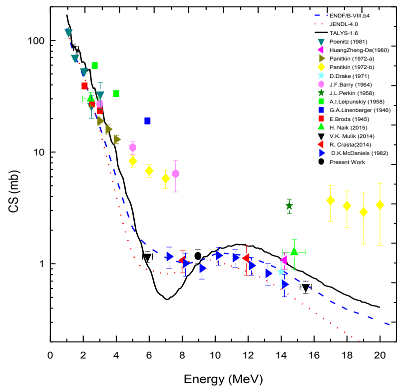

Fig.1 Comparison of present experimental ^{238}U(n,γ) reaction cross-section with the EXFOR, ENDF, JENDL and TALYS data [2-5].

References

[1] S. S. Kapoor, Pramana-J.Phys **59,** 941 (2002).

[2] IAEA-EXFOR Data base, at http://www.nds.iaea.org/exfor

[3] H. Naik, et al., Eur. Phys. J. **A47,** 51 (2011).

[4] A. J. Koning et al., Talys User Manual, A nuclear reaction program, User Manual, NRG-1755 ZG Petten, The Netherlands (2011).

[5] R.Capoteet al., RIPL–3 Reference input parameter library for calculation of nuclear reactions and nuclear data evaluations, Nuclear Data Sheets, **110,** 3107–3214 (2009).

[6] V. Vansola et al. Radiochimica Acta **103(12),** 817 (2015).

[7] V. Vansola et al. Radiochim. Acta 2016; **104(11)**, 749 (2016).

Table I: Present experimental data

Fission Monitor				$^{238}_{92}U\ (n,\gamma)^{239}_{93}Np$			
Proton Energy (MeV)	En (MeV)	Cross Section (mb)	Flux n/cm^{2}sec	Cross Section (mb)			
				Present Work (mb)	TALYS-1.6 (mb)	ENDF (mb)	JENDL (mb)
11	8.96±1.3	220.0	4.2E+06	1.17±0.17	1.00	0.6525	1.096