The (n, γ) and (n, 2n) cross-sections for ^{238}U at neutron energy of 15.5 MeV

S.V. Surayanarayana[a], V. K. Mulik[b], Sadhana Mukerji[c], H. Naik[d], P. M. Prajapati[e], B. S. Sivashankar[f], A.K. Mohanty[a], A. Goswami[d], S. D. Dhole[a], V.N. Bhoraskar[a]

[d]Nuclear Physics Divisio Bhabha Atomic Research Centren, [b]Department of Physics, University of Pune - 411 007, India,

.[c]Reactor Physics Ddesign Division [d]Radiochemistry Division, Bhabha Atomic Research Centre, Mumbai -400 085, India

[e]Physics Department, Faculty of Science, The M. S. University of Baroda, Vadodara-390 002, India

[f]Department of Statistics, Manipal University, Manipal - 576 104, India.

Presently, the reactors operating over the globe are light or heavy water reactors that use enriched or natural uranium as a fuel. However, recent innovations in reactor technology are aiming for nuclear power based on the fast reactor and advanced heavy water reactor (AHWR) [1,2] in order to meet the increased demand of power production, In the fast reactor, the fissile material ^{239}Pu along with ^{238}U is used as a fuel for power generation. Therefore isotope ^{238}U is very important in this regard, as ^{238}U is used as the breeding material to regenerate the fissile material ^{239}Pu. The required initial stock of ^{239}Pu is first generated in a thermal reactor from ^{238}U(n, γ)^{239}U reaction followed by two successive β decays. Therefore, in the present work the ^{238}U(n, γ) and ^{38}U(n, 2n) reaction cross-sections have been studied. These reactions were determined at average neutron energy of 15.5±0.7 MeV using the neutron beam from ^{7}Li(p, n) reaction and by activation technique followed by off-line γ-ray spectrometry.

The neutron beam for the present work was obtained from the ^{7}Li(p, n) reaction, utilizing the proton beam of energy 18 MeV and current of 250 nA at 14UD BARC-TIFR Pelletron facility, Mumbai, India. The natural ^{238}U metal foil of 571.5 mg/cm^2 thickness was wrapped with 0.025 mm thick aluminum foil and the sample was irradiated for 5 h with neutrons. After sufficient cooling, 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.

The half-life of ^{239}U is 23.54 min., which decays 99.6% to ^{239}Np within 3 h. In view of this, the ^{238}U(n, γ) reaction cross-section was calculated from the observed photo-peak activity of ^{239}Np and was identified through the γ-lines of 103.73, and 277.85 keV. On the other hand the ^{238}U(n, 2n) reaction was calculated from the 208.0 keV γ-line of ^{237}U. 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)

$$A_i = N\sigma\Phi a\varepsilon Y\,(1\text{-}e^{-\lambda t})\, e^{-\lambda T}(1\text{-}e^{-\lambda\Delta T})/\lambda \qquad (1)$$

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 18 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 corresponding to quasi mono energetic peak was obtained as 15.5±0.7 MeV after removing the tailing distribution of the neutron spectrum below 13 MeV. The neutron flux (Φ) of 3.78x10^9 n cm^{-2} h^{-1} was used to calculate the ^{238}U(n,γ) reaction cross-section, which is 2.389±0.075 mb. Similarly, the ^{238}U(n,2n) cross-section was calculated using the neutron flux of 2.15x10^9 n cm^{-2} h^{-1}, which is 1094.4±75.6 mb.

For ^{238}U(n, γ) reaction, the low energy neutrons below 13 MeV 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 1.912 and 1.476 from ENDF/B-VII.0 and JENDL-4.0 respectively. The actual cross-section was obtained as 0.623±0.075 mb, which are given in Table 1. Similarly, the tailing contribution for ^{238}U(n,2n) reaction was calculated from 6.16 to 13 MeV, which are 565.36 mb from ENDF/B-VII.0 and 550.23 mb from JENDL-4.0. The actual cross-section was obtained as 536.58±75.67 mb, which is given in Table 1.The evaluated ^{238}U(n,γ) and (n,2n) reaction cross-sections from ENDF/B-VII.0 and JENDL-4.0

are also given in Table 1 for comparison. It can be seen from Table 1 that the measured ^{238}U(n, γ) and (n,2n) reaction cross-sections are almost within the range of the evaluated data of ENDF/B-VII.0 and JENDL 4.0.

Table 1. ^{238}U(n, γ)^{239}U and ^{238}U(n, 2n)^{237}U reaction cross-sections at neutron energy of 15.5±0.7 MeV.

Neutron Flux $(n\ cm^{-2}h^{-1})$	Cross-section (mb) Expt.	ENDF/B-VII	JENDL-4.0
	^{238}U(n, γ)^{239}		
3.78×10^{9}	0.623±0.075	0.710-0.580	0.415
	^{238}U (n, 2n)^{237}U		
2.15×10^{9}	536..58±75.67	508.3	483.4

To examine this, the ^{238}U(n, γ) and (n,2n) reaction cross-sections from the present work and evaluated data from ENDF/B-VII.0 and JENDL-4.0 JEFF 3.1/A [6] and CENDL-3.1[7] along with data given in EXFOR [8] along with are plotted in Fig. 1 and Fig.2, respectively. It can be seen from Fig. 1 that the ^{238}U(n, γ) reaction cross-section at neutron energy of 15.47 MeV is in agreement with the value of Mc Daniels et al [9] at 11.2-12.2 MeV but not with CENDL-3.1 [7] evaluation. The ^{238}U(n,2n) reaction cross-sections from the present work are in agreement with evaluated data (Fig. 2).

Theoretically, the ^{238}U(n,γ) reaction cross-sections at different neutron energy beyond 100 keV and ^{238}U(n,2n) above threshold were also calculated using the nuclear model based computer code TALYS 1.2 [10] and are plotted in the Fig. 1 and Fig.2. It can be seen that the trend of the experimental and evaluated ^{238}U(n,γ) and (n,2n) reaction cross-sections are well produced by the TALYS 1.2. However, 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-section decreases from 100 keV to 7 MeV and predict a dip in 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.

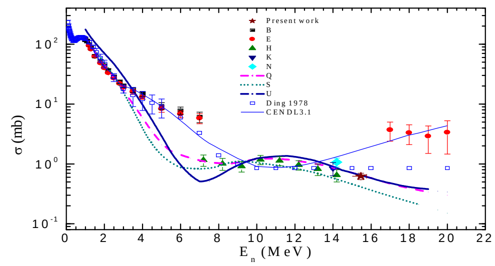

Fig. 1: Plot of ^{238}U(n, γ) reaction cross-section as a function of neutron energy

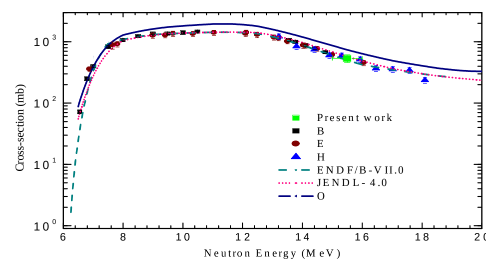

Fig. 2. Plot of ^{238}U(n,2n) reaction cross-section as a function of neutron energy

The authors are thankful to the TIFR-BARC Pelletron facility to provide the proton beam to carry out the experiment. We are also thankful to Mr. Ajit Mahadkar and Mrs. Dipa Thapa of target lab, TIFR for providing Li and Ta targets.

Reference

[1] Fast Reactors and Accelerator Driven Systems Knowledge Base, IAEA-TECDOC-1319: Thorium fuel utilization: Options and Trends
[2] R.K. Sinha and A. Kakodkar, Nucl. Eng. Des, **236**, 683 (2006).
[3] S.G. Mashnik, et al., Los Alamos National Laboratory, Los Alamos, NM 87545, USA, February 8 (2008)
[4] M. B. Chadwick et al., Nucl. Data Sheets, **107,** 2931 (2006).
[5] K. Shibata et al., J. Nucl. Sci. Technol, **48**, 1 (2011).
[6] A.J. Koning, et al., Proceeding of the International Conference on Nuclear Data for Science and Technology, Nice, (2007).
[7] Y. Tang et al., China Evaluated Nuclear Data Library CENDL-3.1, (2009).
[8] IAEA-EXFOR Database, at http://www- nds.iaea.org/exfor.
[9] D.K. McDaniels et al., Nucl. Phys. A, **384**, 88 (1982).
[10] A.J. Koning et al., Proc. International Conf. Nucl. Data for science and Tech ND, (ed. R.C. Haightand et al.) AIP **769**, 1154 (2004).