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$(n, 2n)$ AND $(n, 3n)$ CROSS-SECTION MEASUREMENTS BETWEEN THRESHOLD AND 15 MeV[1]

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The $(n, 2n)$ and $(n, 3n)$ cross-sections for several nuclei have been measured from threshold up to 15 MeV incident neutron energy, using the large liquid scintillator method and the 12 MeV Tandem Van de Graaff accelerator of Bruyères-le-Châtel as a pulsed neutron source.

The $(n, 2n)$ and $(n, 3n)$ cross-sections have been normalized to the fission cross-section of ^{238}U and are obtained with a relative accuracy of 5 % to 10 %.

A knowledge of the shape and magnitude of $(n, 2n)$ cross-sections as functions of neutron energy is of interest from the point of view of nuclear reaction theory and of the use of certain materials as threshold detectors.

In the present experiment, $(n, 2n)$ and $(n, 3n)$ cross-sections have been determined by a direct counting of the reaction neutrons, using the large liquid Gd-loaded scintillator previously developed for precise $\bar{\nu}_p$ measurements [1].

Contrary to the more generally used activation method, this technique requires a relatively low neutron flux, which can generally be produced using conventional Van de Graaff accelerators.

The neutron detector (Fig. 1a) is a large sphecial liquid scintillator counter — 76 cm in diameter — containing a gadolinium loaded scintilator and surrounded by 12 photomultiplier tubes. Neutrons originating from a sample located at the centre of the counter, enter the liquid, where they are moderated and then are captured by the gadolinium nuclei with a mean life time of 11μs. The resulting capture γ-rays cause scintillations viewed by photomultiplier tubes. The time distribution of the neutron capture events after neutron emission is shown on Fig. 1b; 95 % of the capture events occur within 30 μs. In this manner, though the neutrons in the sample are emitted at the same time, separate delayed capture are pulses obtained. With an efficientcy of about 80 %, this detector is particularly well adapted for counting multiple neutron emission reactions such as $(n, 2n)$ or $(n, 3n)$ reactions: The efficiency is determined by comparing the number of neutron emitted per fission, measured for the spontaneous fission of ^{252}Cf to the reference value $\bar{\nu}_p = 3.732$ [2].

Incident neutrons in the energy range of 6—15 MeV were produced by the (d, D) reaction, using the pulsed beam of the tandem Van de Graaff accelerator of Bruyères-le-Châtel and a deuterium gaseous target. With the pulsing system adopted lower it is possible, using the time-of-flight technique, to separate the monokinetic neutrons from

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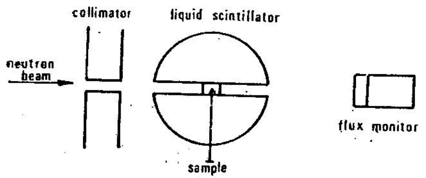

Fig. 1a. Experimental set up.

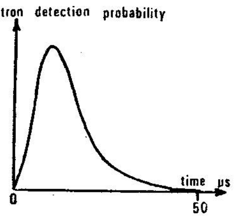

Fig. 1b. Time distribution of neutron capture.

deuteron break-up neutrons as well as from neutrons induced by (*d*, *n*) reactions on the target materials (Fig. 1c). A small liquid scintillator, associated with a fast photomultiplier tube located behind the sphere (Fig. 1a) was used as a neutron monitor (flight path ~ 4 meters). The relative efficiency of this monitor was experimentally determined in the energy range of 0.4—7 MeV by the time-of-flight technique, using the prompt neutron fission spectrum of a ^{252}Cf source as the standard. This calibration was extended up to 15 MeV using the results of a Monte Carlo code calculation.

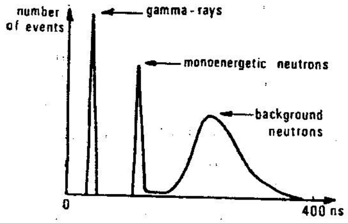

Fig. 1c. Incident neutron time-of-flight spectrum.

(*n*, 2*n*) and (*n*, 3*n*) cross-section measurements [3] were carried out as follows. About 10 to 15 grams of the sample under study are placed at the centre of the large liquid scintillator counter where they are irradiated by a collimated neutron beam. The tandem accelerator beam is pulsed at a ferquency of 2.5 MHz. An electrostatic beam sweeper is used to keep only 3 bursts each 60 µs. After each group of 3 bursts, a 30 µs counting gate (Fig. 1d) releases the output of the photomultiplier tubes. In this manner it is possible to obtain the number of events with 0, 1, 2... pulses counted. These data are corrected for background in 2 steps. Firstly, the accelerator non dependent background is subtracted. This background is measured for the same number of counting gates with the accelerator beam off. Then the accelerator dependent background is taken into account. This background is obtained with the beam on and the sample out, for the same incident neutron flux. The data are then corected for the detection dead time and for the detector efficiency. Another correction is also necessary, to take into account the possibility of two

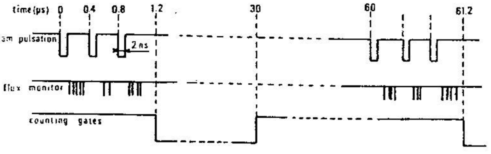

Fig. 1d. Time sequence of gated data collection.

Table 1

E_n MeV	σ_F barns ^{238}U	$\sigma(n, 2n)$ barns					$\sigma(n, 3n)$ barns ^{238}U
		^{169}Tm	^{181}Ta	^{197}Au	^{209}Bi	^{238}U	
7.93 ± 0.150	0.956				0.132 ± 0.015	1.091 ± 0.110	
8.94 ± 0.125	0.957	0.374 ± 0.027	0.802 ± 0.047	0.286 ± 0.020	0.804 ± 0.051	1.303 ± 0.112	
9.93 ± 0.110	0.952	1.232 ± 0.069	1.560 ± 0.084	1.027 ± 0.057	1.563 ± 0.086	1.378 ± 0.114	
10.42 ± 0.100	0.952	1.574 ± 0.087	1.803 ± 0.097	1.348 ± 0.075	1.845 ± 0.100	1.413 ± 0.092	
10.91 ± 0.095	0.953	1.663 ± 0.107	1.852 ± 0.115	1.591 ± 0.101	1.950 ± 0.124	1.390 ± 0.142	
11.40 ± 0.090	0.959	1.719 ± 0.079	1.864 ± 0.103	1.695 ± 0.076	1.940 ± 0.110	1.481 ± 0.111	
11.88 ± 0.085	0.964	1.797 ± 0.096	1.875 ± 0.096	1.814 ± 0.094	1.933 ± 0.101	1.359 ± 0.125	0.004 ± 0.046
12.36 ± 0.085	0.973	1.943 ± 0.084	2.041 ± 0.112	2.002 ± 0.083	2.120 ± 0.118	1.442 ± 0.112	0.108 ± 0.043
12.85 ± 0.080	0.998	2.091 ± 0.086	2.141 ± 0.117	2.164 ± 0.087	2.193 ± 0.121	1.373 ± 0.105	0.209 ± 0.068
13.33 ± 0.075	1.045	2.160 ± 0.125	2.166 ± 0.121	2.227 ± 0.125	2.194 ± 0.124	1.223 ± 0.110	0.342 ± 0.061
13.80 ± 0.075	1.106	2.212 ± 0.090	2.260 ± 0.121	2.329 ± 0.092	2.393 ± 0.132	0.923 ± 0.117	0.516 ± 0.093
14.28 ± 0.070	1.165					0.910 ± 0.184	0.503 ± 0.116
14.76 ± 0.065	1.224					0.754 ± 0.183	0.651 ± 0.129

E_n MeV	σ_F barns ^{238}U	$\sigma(n, 2n)$ barns					
		^{56}Fe	Fe nat.	^{59}Co	^{89}Y	^{175}Lu	Lu nat.
8.94 ± 0.125	0.957					0.681 ± 0.050	0.708 ± 0.048
9.93 ± 0.110	0.952					1.482 ± 0.083	1.493 ± 0.083
10.42 ± 0.100	0.952					1.716 ± 0.096	1.723 ± 0.095
10.91 ± 0.095	0.953			0.019 ± 0.006		1.874 ± 0.125	1.879 ± 0.125
11.40 ± 0.090	0.959			0.127 ± 0.010		1.806 ± 0.105	1.814 ± 0.105
11.88 ± 0.085	0.964	0.039 ± 0.008	0.052 ± 0.006	0.232 ± 0.015	0.053 ± 0.008	1.876 ± 0.103	1.884 ± 0.103
12.36 ± 0.085	0.973	0.107 ± 0.010	0.116 ± 0.009	0.366 ± 0.024	0.214 ± 0.012	2.050 ± 0.119	2.055 ± 0.119
12.85 ± 0.080	0.998	0.220 ± 0.016	0.221 ± 0.015	0.507 ± 0.030	0.450 ± 0.021	2.099 ± 0.124	2.104 ± 0.124
13.33 ± 0.075	1.045	0.330 ± 0.023	0.323 ± 0.022	0.609 ± 0.037	0.677 ± 0.042	2.163 ± 0.110	2.168 ± 0.126
13.80 ± 0.075	1.106	0.423 ± 0.026	0.409 ± 0.026	0.724 ± 0.042	0.862 ± 0.036	2.235 ± 0.117	2.238 ± 0.125

(n, n') events occuring in the same counting gate, being then indistinguishable from a $(n, 2n)$ event.

Associated with the relative flux measurement, this method gives relative $(n, 2n)$ cross-sections for non-fissionable samples. This technique can be extended to fissionable materials, providing a cross-section relative to the fission cross section value.

Fission neutron multiplicities have already been measured for monoenergetic neutrons [1]. Fission neutron multiplicities can therefore be calculated for the incident neutron energy spectrum given by the flux monitor detector. By comparing calculated and measured multiplicities for events with more than 3 neutrons detected, the number of fission events can be deduced. The number of fission events with 2 and 3 neutrons, and subsequently the number of $(n, 2n)$ and $(n, 3n)$ events can be obtained. In this manner $(n, 2n)$ and $(n, 3n)$ cross-sections relative to fission cross-sections are obtained. In this study, all the $(n, 2n)$ and $(n, 3n)$ cross-sections have been normalized to the ^{238}U fission cross-section, using the relative neutron flux measurement.

The $(n, 2n)$ cross-sections for Fe, ^{59}Co, ^{89}Y, ^{169}Tm, Lu, ^{181}Ta, ^{197}Au, ^{209}Bi, ^{238}U and the $(n, 3n)$ cross-section for ^{238}U have been measured from the threshold up to a 15 MeV incident neutron energy. Preliminary values are given in Table 1. The $(n, 2n)$ cross-sections for ^{56}Fe and ^{175}Lu have been deduced from measurements on natural elements by correcting the measurements for the contribution of the other isotopes using the results of Pearlstein's calculations [4]. The ^{238}U fission cross-section used for normalization has been taken from Sowerby's evaluation [5]. The errors quoted are only of a statistical origin and do not include the error on the fission cross-section of ^{238}U.

For most of the investigated nuclei, the present results are the only ones covering the energy range from the reaction threshold up to 15 MeV and are of interest to determine the shape of $(n, 2n)$ cross-sections as a function of neutron energy.

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