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MEASUREMENTS OF NEUTRON INDUCED CROSS SECTIONS AT THE OAK RIDGE ELECTRON LINEAR ACCELERATOR *

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Abstract

We have used the Oak Ridge Electron Linear Accelerator (ORELA) to measure neutron total and the fission cross sections of ^{233}U in the energy range from 0.36 eV to ~700 keV. We report average fission and total cross sections. Also, we measured the neutron total cross sections of ^{27}Al and natural chlorine as well as the capture cross section of Al over an energy range from 100 eV up to about 400 keV.

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DISCLAIMER

Introduction

To support the Nuclear Criticality Predictability Program, neutron cross section measurements have been initiated at the Oak Ridge Electron Linear Accelerator (ORELA). We measured the total and fission cross sections of ^{233}U in the neutron energy range from 0.36 eV to several hundred keV. This will not only help to resolve inconsistencies between different data sets but will also improve the representation of the cross sections since most of the available evaluated data rely only on old measurements. Usually these were done with poor experimental resolution or only over a very limited energy range which is insufficient for current application. To clarify inconsistencies in the criticality calculation for systems including Al and ^{235}U, the total and capture cross sections of Al in the energy range from about 100 eV to several hundred keV have been measured. To understand systems including PVC the total cross section of natural chlorine has been measured with much higher resolution than has been done previously. Evaluated data files based on these measurements will be utilized for both criticality analyses and benchmark data testing. Finally the data will be submitted for inclusion in ENDF/B.

Description

The Oak Ridge Electron Linear Accelerator (ORELA) was used as the neutron source for measurements of neutron induced fission cross sections. For the present fission measurements a ^{233}U fission chamber and a ^{6}Li glass scintillation detector were aligned in the neutron beam from ORELA at the 80-meter station on Flight Path No. 1. The center of the fission chamber was at 81.237 m from the neutron source and the ^{6}Li glass flux monitor was located 1.423 m in front of the fission chamber.

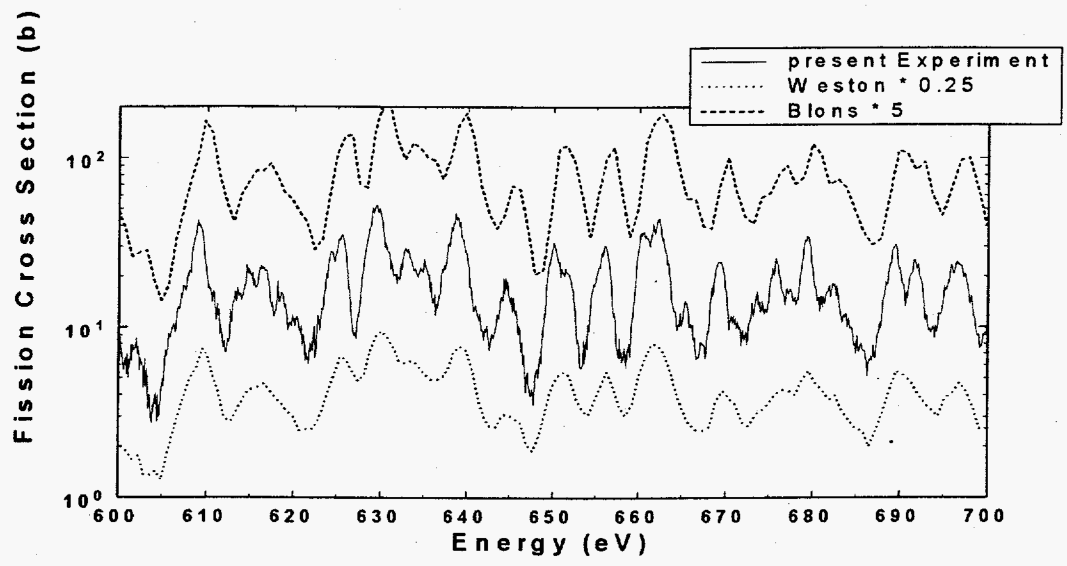

Figure 1: ORELA ^{233}U fission cross section compared to previous measurements. For a better display the data of Weston et al.[1] and Blons[8] are multiplied by different factors. The higher resolution of the present experiment can be clearly seen.

We made two distinct runs under different experimental conditions in order to cover a broad neutron energy range. The first was made with a Cd filter in the beam and was carried out at a low repetition pulse rate of ~78 Hz. In this way we were able to cover the low neutron energies region (0.36eV to 100eV). The second run, with a ^{10}B filter, was carried out at ~400 Hz which covered the neutron energy region from about 100 eV to almost 700 keV. The Cd-filtered run was used to normalize[1,2] all the obtained data to previous experimental data in the low eV region. The high-resolution data obtained with the ^{10}B filtered run will allow the extension of the resolved resonance analysis to about 1 keV.

A run with several centimeters of polyethylene in the beam was used for background determination. Since all neutrons were blocked, we were able to determine the residual alpha particle and noise rates detected by the chamber electronics. The fission chamber used in this experiment contained a total of 2.11 grams of uranium with an isotopic composition of 99.997% ^{233}U distributed in 40 coatings onto 21 aluminum plates of 0.127 mm thickness. Each plate was coated to a diameter of 76 mm. In this arrangement the total thickness of ^{233}U was 0.00011956 atom/barn. The plates were connected electrically to form ten identical sections with a separate fast amplifier for each section. The electronic bias levels on the discriminators for each section were individually set such that the residual alpha detection rates with no neutrons present were about one count per second. The 10 individual discriminator outputs were combined via a logic OR unit to give the experimental fission chamber rate. The ^{6}Li glass detector used in this experiment was 1 mm thick and had a diameter of 10 cm. With this detector we determined the relative neutron flux versus energy. The efficiency of this flux monitor was calculated by means of a computer program used previously in other measurements by Macklin et al.[3] Since the attenuation of the neutron beam by the glass was not negligible, and the absolute amount of ^{6}Li in the glass was not accurately known, the transmission of the glass was measured.

Table I : Average experimental ^{233}U fission cross section data (barn)

Energy interval (eV)	Present work	Derrien Ref. 2 (Evaluation)	Weston Ref. 1[a]	Wagemans Ref. 4
0.4-1.0	129.86±0.54	128.7	129.2	130.7
1.0-2.1	385.19±0.49	387.9	391.7	388.8
2.1-2.75	205.39±0.53	204.6	207.5	204.4
2.75-3.0	56.77±0.53	50.2	51.9	50.1
3.0-15.0	105.65±0.15	104.3	104.9	106.2
15.0-30.0	94.72±0.13	95.51	95.03	
30.0-50.0	42.48±0.08	40.27	40.13	
50.0-75.0	42.75±0.07	40.53	40.66	
75-100	38.08±0.07	36.03	35.57	
100-125	39.32±0.08	36.97	36.84	
125-150	23.05±0.06	20.78	21.29	
15-150	43.34±0.03	41.45	41.39	

[a] renormalized by 2.5%

For the transmission measurements we used extremely high purity (99.999%) aluminum and natural chlorine samples. These were mounted in the holder of a sample changer positioned at about 10 meters from the neutron target in the beam of ORELA. For the aluminum measurements we used a thin (0.01894 a/b) and a thick sample (0.1513 a/b). A pre-sample collimation limited the beam size to about 2.54 cm on the samples and allowed only neutrons from the water moderator part of the neutron source to be used. The neutron detector was a 11.1cm diameter, 1.25 cm thick ^{6}Li glass scintillator viewed on edge by two 12.7 cm diameter photomultipliers and positioned in the beam at 79.815 meters from the neutron source. For the chlorine transmission sample we used a natural CCl_4 (thickness for chlorine 0.2075 a/b) sample as well as a "compensator" sample containing an equivalent amount of carbon (graphite disk) and sample-holder window material (brass). Additional measurements were made in both experiments for the open beam, and measurements with a thick polyethylene sample were used to determine the gamma-ray background from the neutron source.

Table II : Average experimental ^{233}U total cross section data (barn)

Energy interval (eV)	Present work	Pattenden Ref. 9
100-125	58.06	51.08
125-150	37.83	36.58
150-200	36.93	35.80
200-250	39.35	37.32
250-300	39.03	34.48
300-350	32.63	32.21
350-400	34.32	31.31
400-450	23.24	24.76
450-500	25.98	26.63
500-600	27.44	27.20
600-700	30.89	30.54
700-800	25.54	26.39
800-900	27.62	27.30
900-1000	23.79	25.14
1000-1200	23.24	23.96
1200-1400	22.76	23.06
1400-1600	21.51	21.53
1600-1800	20.51	21.67
1800-2000	20.83	21.24
2000-3000	19.87	20.06
3000-4000	17.93	18.46
4000-5000	17.29	17.69
5000-6000	16.65	17.49
6000-7000	16.55	18.66
7000-8000	15.71	17.40

For neutron total cross section measurement of ^{233}U we used high purity metallic samples with an enrichment of 99.76%. The samples were mounted in a cryogenic device in the sample changer at 10 meters and cooled down to 11K. The cooling to these low temperatures enabled us to reduce the Doppler broadening of the individual resonances and to improve the experimental resolution significantly. Again the measurements were carried out with two different samples and under two different experimental conditions. The thin sample (0.00291 a/b of ^{233}U) was used during a low repetition rate run at 78 Hz and with a Cd overlap filter in the beam. Under these conditions we were able to cover the energy range from about 0.36 eV to roughly 100 eV. A ^{10}B overlap filter was used in the high-energy runs with repetition rate of 525 pps and a thick sample (0.01186 a/b of ^{233}U). This run covers the energy range from 100 eV to 300 keV. Additional measurements were made with thick polyethylene and ^{238}U filters, in order to determine the backgrounds. A special run at a low repetition rate with the ^{10}B filter in the beam was used to determine the effect of overlap neutrons from the previous pulse in the high-energy run. The data were later corrected for this experimental effect. The neutron detector used in these measurements was a 11.1 cm diameter, 1.25 cm thick ^{6}Li glass scintillator positioned in the beam at 79.827 meters.

For the neutron capture measurements[5] we used two (0.01520 a/b and 0.04573 a/b) rectangular aluminum samples. The samples and the C_6D_6 detectors were located at a distance of 40 meters from the neutron target. This capture system has been re-engineered to minimize structural material surrounding the sample and detectors. A 0.5-mm thick ^{6}Li-glass scintillator served as the neutron flux monitor. Pulse-height weighting was employed with the C_6D_6 detectors; normalization of the capture efficiency was carried out in a separate measurement using the "black resonance" technique by means of the 4.9-eV resonance from a gold sample.[6]

Results

The raw data were first corrected for dead-time effects due to the data acquisition electronics. Then corrections for the measured time-independent and -dependent backgrounds were applied. Additionally the ^{10}B-filtered runs for the fission and transmission measurements of ^{233}U, Al and Cl were corrected for overlap neutrons. For normalizing our data we used the best available set of resonance parameters[2] and determined with SAMMY[7] the normalization factor in the neutron energy region from 3.1 to 20.0 eV. In both the Cd and ^{10}B filtered runs we used the same normalization factors. The data are corrected for self-shielding and multiple scattering in the neutron energy range up to 150 eV where resonance parameters were available. The maximum correction applied to the data was 8%. The average fission cross sections are compiled in Table I and compared with previous data sets and the most recent evaluation.

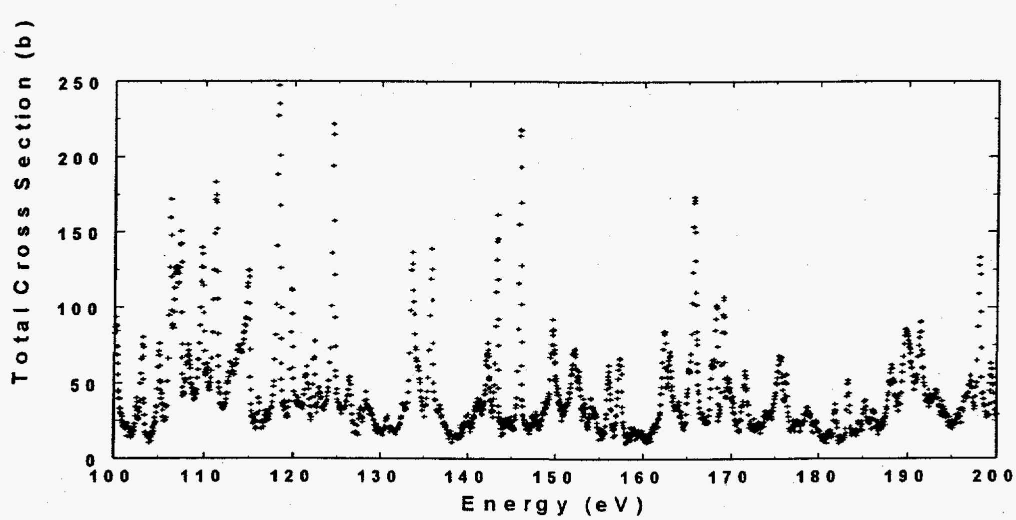

Figure 2: ^{233}U total cross section data, taken with a cooled sample(T=11K). For a better display not all data point are plotted.

The $^{233}U(n,f)$ data from 600 to 700 eV are plotted in Figure 1 and compared to the measurements for Ref. 1 and Ref. 8. Due to the better experimental resolution the new data set describes the fission cross section more precisely, i.e. there are more data points over the individual resonances. This will improve the determination of the correction to the experimental data by the data analysis programs. With the present data set, it will be possible to obtain good resonance parameters to energies as high as 1.0 keV.

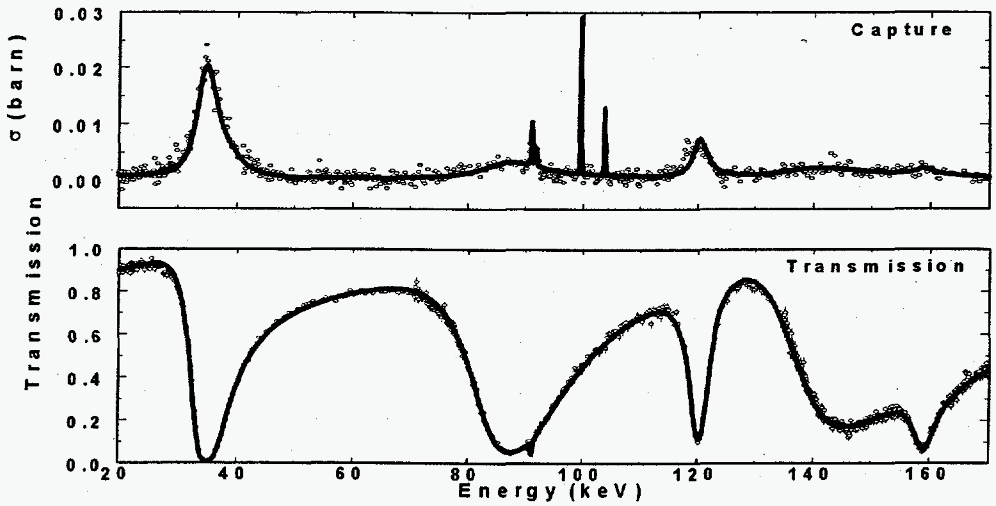

Figure 3: Neutron capture and transmission data for ^{27}Al in the energy range from 20 keV to 170 keV. The solid line represents the SAMMY fit to the data (circle).

For the ^{233}U transmission experiment we applied the usual corrections for dead-time, and subtracted the measured time-dependent and –independent and overlap neutron backgrounds. During the initial analysis of the data, it was discovered that there was additional 0.127 mm Al in the sample-in data not accounted for by the sample-out runs. This extra Al was thought to be due to improperly machined sample holders. We used our aluminum resonance parameters to calculate the transmission for this thickness and corrected our data. Some preliminary average total cross sections from 100 eV to 8 keV are compiled in Table II and compared to previous data in this region. We find that above 400 eV our data is lower than the measurement from Ref. 9. The average total cross section in the energy range from 1 keV to 8 keV is 5% lower than the value from Pattenden and Harvey[9]. For the high energy region between 50 keV and 200 keV our results are 4% lower than the data of Poenitz[10]. Figure 2 shows an example of the experimental data for the thick ^{233}U sample.

In Figure 3 we show the capture yield and transmission in the energy range from 20 to 170 keV obtained from our aluminum measurements. The solid line represents the fit to the data generated by the computer code SAMMY. With these new determined resonance parameters Γ_n and Γ_γ, which will be reported elsewhere, we are able to describe the cross section up to 400 keV. We find very good agreement in the region of overlap with the most recent high resolution transmission experiment done at Geel[11]. But this data set has a low energy cut off at about 200 keV, whereas our data extend down to 100 eV. For the present data without any neutron sensitivity correction we obtain an average of 15% reduction in the gamma width compared to previous measurements.

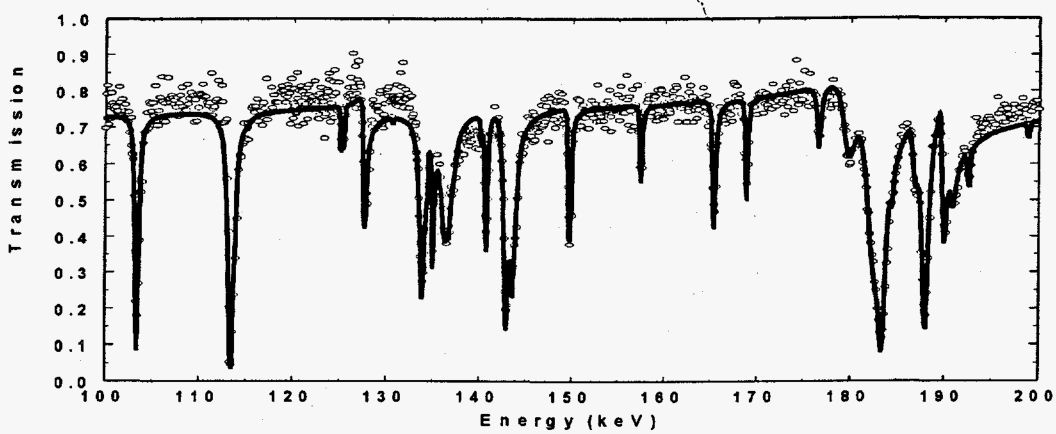

Figure 4: Neutron transmission of natural chlorine from 100 keV to 200 keV. The solid line is the preliminary SAMMY fit.

In Figure 4 the transmission data for the natural chlorine sample are plotted over the energy range from 100 to 200 keV. We found several new resonances compared to ENDF/B-6 and JENDL 3.2. With the help of previous measurements[12] on enriched chlorine isotopes as well as the fitting code SAMMY we were able to identify the new resonances and assign them to an isotope. From preliminary fits we found several inconsistent spin and neutron energy assignments. The resonance parameters will be reported in a forthcoming publication.

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