Neutron-induced activation measurements and EXFOR compilations in the energy range up to 20 MeV

V. Semkova, N. Otuka
[1]*Nuclear Data Section, International Atomic Energy Agency, A-1400 Vienna, Austria*

Accurate neutron-induced activation cross-section data are of interest in many fields of science and applications. Such data are needed for calculations and analysis of neutron transport, activation of materials, gas production and radiation damage, dose rates etc. Experimental data provide bases for the parameterization of reaction cross section calculations, and for the assessment of nuclear models and evaluated data libraries. Activation technique in combination with gamma spectrometry is well known and widely used method for neutron-induced reaction cross-section measurements. However, in some cases considerable differences exist between the results from different experiments. A careful consideration of the all factors that may affect each particular measurement is needed in order to obtain reliable data. Measured data are of little value until they are made conveniently available for users and evaluators. The International Network of Nuclear Reaction Data Centres (NRDC) collaborates in collection, compilation and dissemination of experimental nuclear reaction data in the EXFOR data library. In the present work some aspects of the ^{58}Ni(n,p)^{58}Co activation cross-section measurements at two different experimental facilities [1,2] and EXFOR compilation files will be presented [3].

Irradiation Procedure

At Institute for Reference Materials and Measurements (IRMM, Geel, Belgium) the ^{3}H(d,n)^{4}He reaction was used for the production of neutrons in the energy range from 14 to 21 MeV [1] and ^{3}H(p,n)^{3}He reaction for the production of neutrons from 0.5 to 6 MeV and at the Institute of Experimental Physics (Kossuth University, Debrecen, Hungary) the neutrons in the energy range 13.5 -14.8 MeV were produced via ^{3}H(d,n)^{4}He reaction [2]. Both reactions as well as ^{2}H(d,n)^{3}He are considered as sources of monoenetgetic neutrons, however low energy neutrons are present above certain energy threshold and also due to neutron production and scattering near the target area. A neutron spectrum unfolding with time-of-flight spectrum as a prior is a solution to make correction for the contribution of the background neutrons to the total measured reaction rate. The method of "spectral indexing" was applied at IRMM for the determination of neutron flux density distribution [4]. It is based on the measurements of the reaction rates induced by the neutrons in monitor reactions with different energy thresholds. Using time-of-flight measurement of neutron spectra the whole energy range is presented in group representation. The flux corresponding to each energy group is determined by method of generalized least squares using the known response characteristics of spectral-index reactions.

The information regarding neutron source reaction and neutron spectrum is included in EXFOR compilations under the keywords INC-SOURCE and INC-SPECT. The experimentalists are also advised to provide numerical data for neutron source spectra [5].

The fluctuations of the neutron source intensity during the irradiations in both experiments were registered by long counter operating in a multichannel acquisition mode and corrections were applied to the studied and monitor reactions

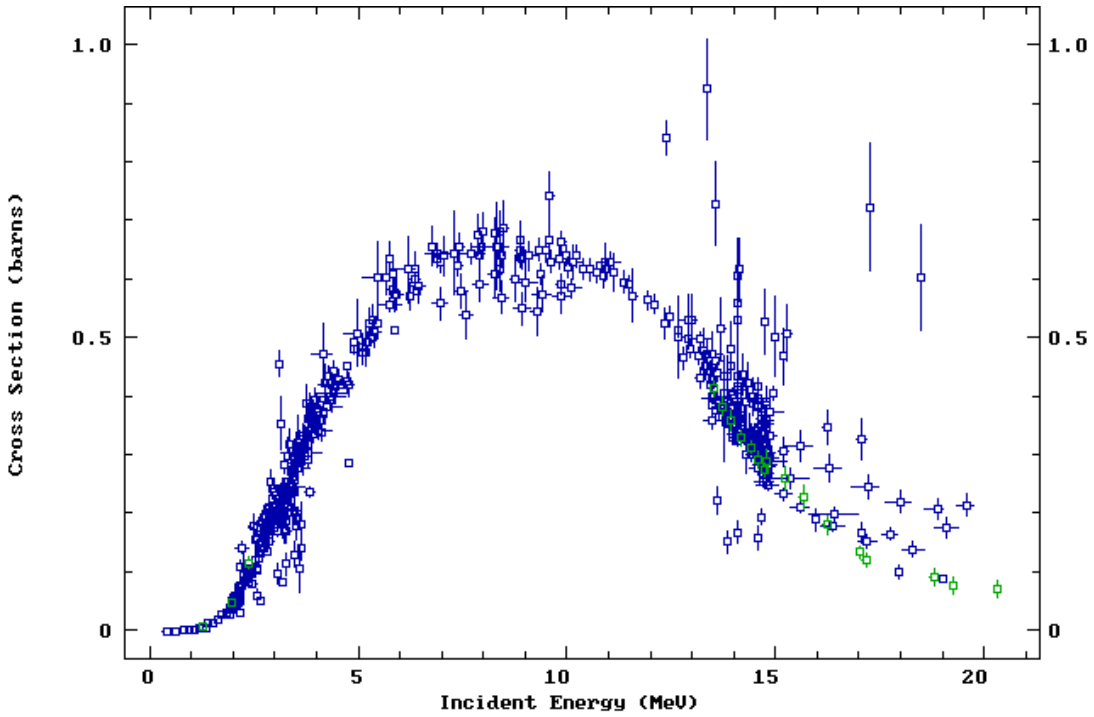

Figure 1. ^{58}Ni(n,p)^{58}Co reaction cross section. Blue symbols are EXFOR data, green symbols data from [1,2].

Radioactivity Measurement

HPGe detectors are conventionally used for gamma-rays spectroscopy nowadays. A set of monoenergetic gamma sources was used for the efficiency calibration of the detector in order to avoid coincidence summing effects since most of the measurements was carried out at closed geometry. In addition a model of the detectors was developed for Monte Carlo simulation of the detector response, which allows taking into account the detailed characteristics of the detector and samples (complex shape, sample matrix, γ-ray self-attenuation, volume activity distribution, coincidence summing effects, etc.) and in consequence increases flexibility and accuracy of the measurements [6]. Comparison of detector efficiencies determined by standard sources and MCNP calculations for tungsten sample with 0.25 mm thickness (both for total and peak efficiencies) are presented on Figure 2. Due to the large gamma-ray attenuation in tungsten and the limitation of the self-absorption correction it cannot properly account for the gamma-ray attenuation in the sample and calculated detector response increasingly deviates from the simulations with decreasing of gamma-ray energy.

The decay data and their uncertainties reported by authors are included in EXFOR compilation. Uncertainty propagation of the branching ratios is rather straightforward, however regarding the half-live it depends also on irradiation, cooling and counting time intervals. The compilers are advised to compile this information under the keyword METOD and code ACTIV if available in the reference. IAEA-NDS also provides and extensive and up to data nuclear structure and decay database through https://www-nds.iaea.org/livechart/.

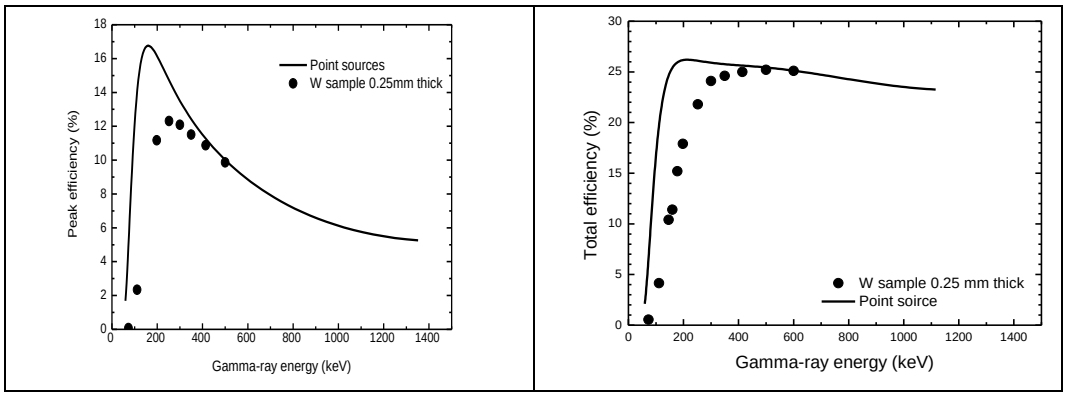

Figure 2. Full energy peak (left) and total (right) efficiencies obtained by fitting of data from point standard sources (line) and MCNP calculations for 0.25 thick W sample.

Data Analysis

Cross sections were calculated using the well-known activation formula. The count rates were corrected for coincidence effects, for γ -ray abundance, γ -ray self-absorption, efficiency of the detector, and measurement geometry, neutron flux fluctuations during the irradiations, and the background neutrons.

Proper reporting of the uncertainties and correlations between partial uncertainties currently are of great interests. EXFOR rules allow compilation of the partial uncertainties and correlations between that in order to provide sufficient information for the construction of covariance matrices.

Example of compilation of discussed experimental details from Ref. 1 in EXFOR entry 22820 is included below.

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INC-SOURCE (D-T) 3H(d,n)4He. Deuterons incident on a metal titanium-tritide target (approx.
                 2 mg/cm2 thick on a silver backing (0.5 mm thick). This reaction generated
                 the primary neutrons with energies in the range 14 to 20.5 MeV. Samples were
                 placed at angles from 0 to 75 degrees. The energy scale uncertainty is
                 5 keV. Deuteron energies of 1, 2, 3, and 4 MeV were used.
           (P-T) 3H (p,n)3He reaction was used to produce neutrons with energies 1 - 3.4 MeV
MONITOR    All relative measurements were ultimately normalized to the primary standard.
           (13-AL-27(N,A)11-NA-24,,SIG) Primary standard used directly or indirectly for all
           the cross section determinations.
           (49-IN-115(N,INL)49-IN-115-M,,SIG) Spectral index component. Very sensitive to
           the lowest-energy neutrons.
           (13-AL-27(N,P)12-MG-27,,SIG) Secondary standard and spectral index component.
           (26-FE-56(N,P)25-MN-56,,SIG) Secondary standard and spectral index component.
           (41-NB-93(N,2N)41-NB-92-M,,SIG) Secondary standard and spectral index component.
MONIT-REF  (,H.Conde+,R,INDC(SEC)-101,1992)  - for the 27Al(n,a)24Na
           (,M.Wagner+,B,PH-DAT,13,5,1990)  - for 93Nb(n,2n)92mNb reactions
           (,A.B.Smith+,R,ANL-NDM-115,1990) - for the 115In(n,n')115mIn reaction
           (,P.L.Rose+,R,BNL-201,1991)  ENDF/B-VI - for the 27Al(n,p)27Mg and 56Fe(n,p)56Mn
ERR-ANALYS (ERR-T) Total errors for the cross sections were deduced by combining partial
            errors from various sources in quadrature. Error sources considered were-
           (ERR-S,1.,15.) statistical counting errors;
           (ERR-1,2.,3.) uncertainty in detector efficiencies;
           (ERR-2,0.01,3.) uncertainties in decay branching factors;
           (ERR-3,0.,7.) uncertainty in coincidence summing corrections;
           (ERR-4,0.1,5.) uncertainty in the absorption of gamma rays;
           (ERR-5,,1.) sample mass error;
           (ERR-6,,1.) counting time error;
           (ERR-7,0.,12.) uncertainty in low-energy neutron corrections;
           (MONIT-ERR,0.5,2.) uncertainties in monitor reactions;
           (ERR-8,0.5,2.) neutron irradiation geometry error;
           (ERR-9,0.01,4.) uncertainty in decay half-lives
ANALYSIS    Cross sections obtained by allowing for complete decay of the isomeric state.
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Summary

Careful consideration of all factors that may affect particular activation reaction cross section measurement is ensures high quality and accuracy of the results. Comprehensive reporting and EXFOR compilations are important for comparison and further analysis of the data from different experiments.

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