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EVALUATION OF SOME ACTIVATION CROSS SECTIONS MEASURED BY MONOENERGETIC AND FISSION NEUTRONS

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Abstract: Neutron-induced activation cross sections applied in neutron dosimetry must be evaluated as accurate as possible and also their evaluated covariances are demanded in computer codes to unfold neutron spectra from dosimeter activities. The six activation cross sections, $^{27}Al(n,p)$, $^{27}Al(n,\alpha)$, $^{54}Fe(n,p)$, $^{56}Fe(n,p)$, $^{59}Co(n,\alpha)$, and $^{58}Ni(n,p)$ and their covariances have been simultaneously evaluated from differential experiments in which samples are activated with monoenergetic neutron sources and the integral experiments with $^{235}U(n,f)$ and ^{252}Cf(spontaneous) fission neutron spectra. The evaluated cross sections are smaller than those only with the differential data.

(Bayesian Method, Neutron Dosimetry, Fission Neutron Spectra, Sections, Simultaneous Evaluation)

Introduction

Neutron dosimetry is an important technique in a field of nuclear application. An activation method is mostly used in the dosimetry. Neutron spectra unfolding from activity measurements requires covariance matrices of the activation cross sections for the dosimeters not only concerning neutron energies but also respecting different reactions in connection with a development of sophisticated data procedure in neutron dosimetry.

Activation cross sections have been measured both differentially by monoenergetic neutrons and integrally by fission neutrons. Most of evaluated cross sections have been estimated from the differential experiments and the integral data have frequently been utilized to confirm the results of the evaluation. The integral measurements have not been used as basic data but as supplementary ones in traditional evaluations. The experiments in the fission neutron fields of $^{235}U(n,f)$ and ^{252}Cf (spontaneous), however, are intrinsically valuable for the cross section evaluation: they are strongly correlated owing to measurements in a well-defined neutron spectrum, they can give information in an energy range where intense chromatic neutron sources are scarce and the neutron energy is near the thresholds of interesting reactions as dosimeters. Therefore, both kinds of experiment should be used simultaneously to evaluate the activation cross sections for neutron dosimeters. The method applicable to this issue is addressed in the present work. It is an extension of the evaluation method which has been used to estimate fission and capture cross sections of heavy nuclides in our laboratory[1,2]. Similar method is presented in the other work[3].

In the present work, six activation reactions, $^{27}Al(n,p)$, $^{27}Al(n,\alpha)$, $^{54}Fe(n,p)$, $^{56}Fe(n,p)$, $^{59}Co(n,\alpha)$, and $^{58}Ni(n,p)$ are chosen as to include reactions of different threshold energies and of important dosimeters. As the differential experimental data, the absolute measurements of these six reactions, the cross section ratios of $^{27}Al(n,p)$, $^{54}Fe(n,p)$, $^{56}Fe(n,p)$, $^{59}Co(n,\alpha)$, and $^{58}Ni(n,p)$ to $^{27}Al(n,\alpha)$, and the cross section ratios of $^{54}Fe(n,p)$ and $^{59}Co(n,\alpha)$ to $^{56}Fe(n,p)$ were used. As the integral data, the ^{235}U thermal fission neutrons and ^{252}Cf spontaneous fission neutron spectrum averaged absolute cross sections of these six reactions were done. In order to study the effects of neutron spectra on the present evaluations, the Maxwellian type and evaluated spectra were employed in the calculation of spectrum averaged cross sections.

Estimation of Activation Cross Sections

The formulae for estimation of the cross sections and their covariances are similar to those in the early works[1,2]. The parameter vector θ and its covariance matrix M are given as

$$\theta = \theta_0 + M_0\Phi^t(\Phi M_0\Phi^t + V)^{-1}(y - \Phi\theta_0),$$

and

$$M = M_0 - M_0\Phi^t(\Phi M_0\Phi^t + V)^{-1}\Phi M_0,$$

respectively. The experimental data vector y can be approximately expressed as $y=\Phi\theta$, where Φ is a design matrix. The matrix V is a covariance matrix for the experimental data. The vector θ_0 and matrix M_0 are a prior parameter vector and its covariance matrix, respectively. The superscripts t and -1 denote the transposed and inverse, respectively.

In the present work, the evaluated cross sections are estimated by two steps. In the first step, both the differential data and the integral data are used to estimate the respective evaluated values and their covariances of the six reactions. The elements of the design matrix Φ for the differential data are expressed with B-spline functions. The elements of the vector y are logarithmics of the experimental values. The matrix V is obtained from the errors given by authors and the assumption that correlation factors are 50% except for the experimental data whose covariances are given by the authors. The experiments are not investigated so severely as to screen their availability. Absolute and relative measurements are discriminated as possible as and they are classified in the design matrix Φ, though there are not so many data which can be assigned as relative experiments because of scanty information. The evaluated cross sections in JENDL-2 are also used as one of prior data in order to interpolate and extrapolate smoothly cross section curves in the energy regions where the differential experiments are not available.

In the second step, the estimated values and covariance obtained at the previous step are used as the prior vector θ_0 and covariance matrix M_0 to evaluate the cross sections and their covariances of the reactions. The elements of the design matrix Φ for the integral data are expressed so as to give the spectrum averaged cross sections for ^{235}U and ^{252}Cf fission neutron spectra, respectively. In order to study the effects of neutron spectrum on the present evaluations, the Maxwellian type and evaluated spectra were employed in the calculation of spectrum averaged cross sections. The evaluated neutron spectra are taken from JENDL-3T[4] and Mannhart[5], respectively. The values of 1.29[MeV] and 1.42[Mev] were taken as the temperatures of Maxwellian spectra , respectively. The fission neutron spectra were not adjusted in the present work.

Results and Discussion

The results and the differential experimental data are compared in Figs.1 to 6. The solid lines in these figures are obtained without taking account of the integral experiments. The dashed and long-dashed lines are the results simultaneously evaluated with both the differential and integral data. The evaluated cross sections of the six reactions agree with those of JENDL-2 and ENDF/B-IV. The result for $^{27}Al(n,\alpha)$ without the integral data agrees very well with Vonach's[6]. As seen in Figs.1 to 6, the results with both the differential and integral data are smaller in the energy region below approximately 13 MeV than those without the integral data. It comes from the fact that almost the integral data are smaller than the values calculated from the cross sections estimated with the differential data. The changes of cross sections estimated using the spectrum of Mannhart and JENDL-3T are smaller than those done using the Maxwellian spectrum. The former were evaluated within about five percent of the results evaluated without the integral experiments. The latter were done within ten percent. These phenomena are seen in the estimations of cross sections whose reaction thresholds are high. The ^{252}Cf fission neutron spectrum of Mannhart and ^{235}U thermal fission neutron spectrum are shown in Fig. 9 as the ratios to Maxwellian spectra. The spectra of JENDL-3T and Mannhart are smaller than the Maxwellian spectra in the high neutron energy region. Therefore, the small changes resulted from the spectra of JENDL-3T and Mannhart.

In Figs.7 and 8, the spectrum averaged cross sections calculated using the cross sections adjusted to experimental differential and integral data are compared with ones calculated using cross sections evaluated with only differential data. The spectrum averaged cross sections figured by the solid and dash-dotted lines were calculated from the cross sections evaluated without the integral data. The former were averaged using the fission neutron spectra of Mannhart and JENDL-3T. The latter done using the Maxwellian spectra. In the simultaneous evaluation with differential and integral experimental data, they are revised to the dashed and dash-two-dotted lines, respectively.

The spectrum averaged cross sections calculated using ^{252}Cf fission neutron spectrum of Mannhart agree with the experimental integral data. The both of the spectrum averaged cross sections calculated using ^{235}U thermal fission neutron spectrum of JENDL-3T and Maxwellian spectrum disagree with the experimental data. In the $^{27}Al(n,p)$, $^{54}Fe(n,p)$, and $^{58}Ni(n,p)$ reactions, the ^{235}U thermal fission neutron spectrum averaged cross sections using spectrum of JENDL-3T from the cross sections evaluated with only differential data are larger than those done using the Maxwellian spectrum. These reactions have low reaction thresholds. In other reactions, the cross sections averaged using the spectrum of JENDL-3T are smaller than those done using the Maxwellian spectrum. These reactions have high reaction theresholds. In the low thresholds reactions, the Maxwellian spectrum reproduces the experimental integral data more than the spectrum of JENDL-3T. In the high thereshold reactions, the spectrum of JENDL-3T reproduces the experimental integral data. The adjusted spectrum averaged cross sections agree with the experimental data. However, the bad reprducibility was found compare to the ^{252}Cf fission neutron spectrum averaged cross sections. From these results, it is found that the evaluated cross sections with the present method strongly depend on the neutron spectra.

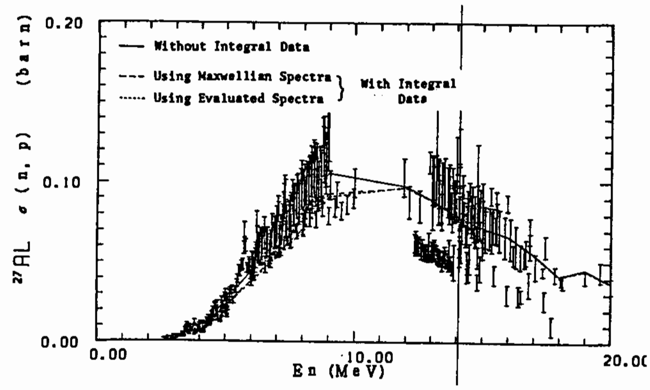

Fig.1 Comparison of evaluated cross sections and differential experiments of $^{27}Al(n,p)^{27}Mg$.

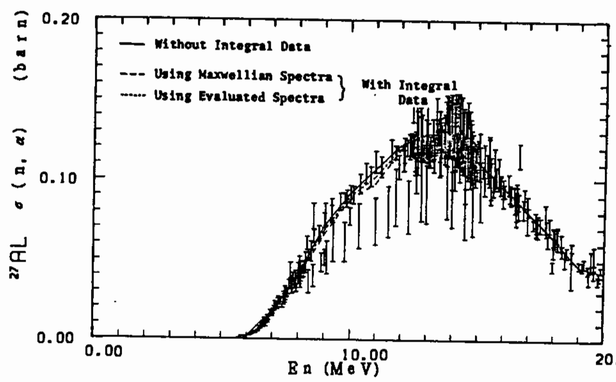

Fig.2 Comparison of evaluated cross sections and differential experiments of $^{27}Al(n,\alpha)^{24}Na$.

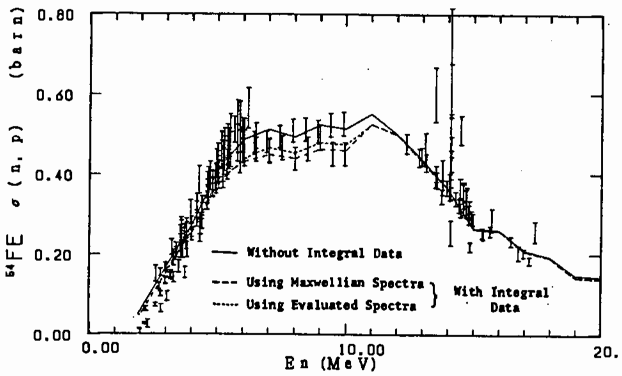

Fig.3 Comparison of evaluated cross sections and differential experiments of $^{54}Fe(n,p)^{54}Mn$.

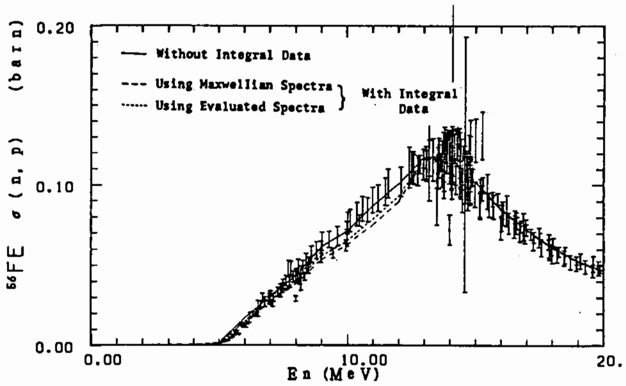

Fig.4 Comparison of evaluated cross sections and differential experiments of $^{56}Fe(n,p)^{56}Mn$.

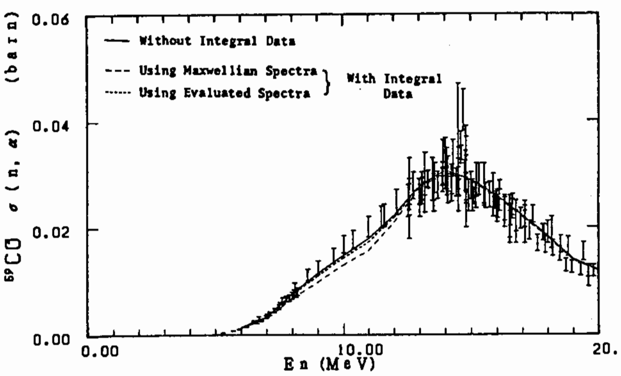

Fig.5 Comparison of evaluated cross sections and differential experiments of $^{59}Co(n,\alpha)^{59}Fe$.

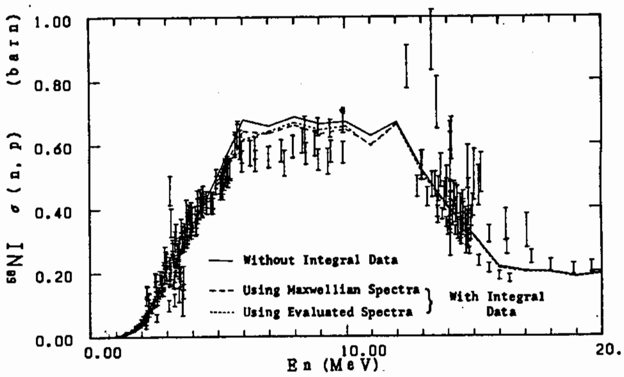

Fig.6 Comparison of evaluated cross sections and differential experiments of $^{58}Ni(n,p)^{58}Co$.

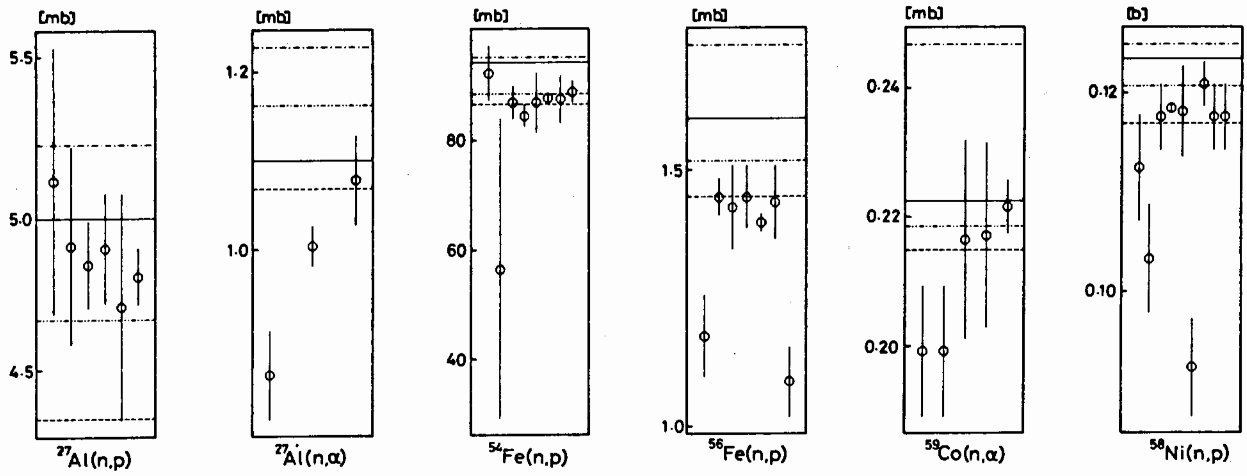

Fig.7 Comparison of evaluated and measured ^{252}Cf fission neutron spectrum averaged cross sections. Symbols: Experiments. Lines: Evaluations.

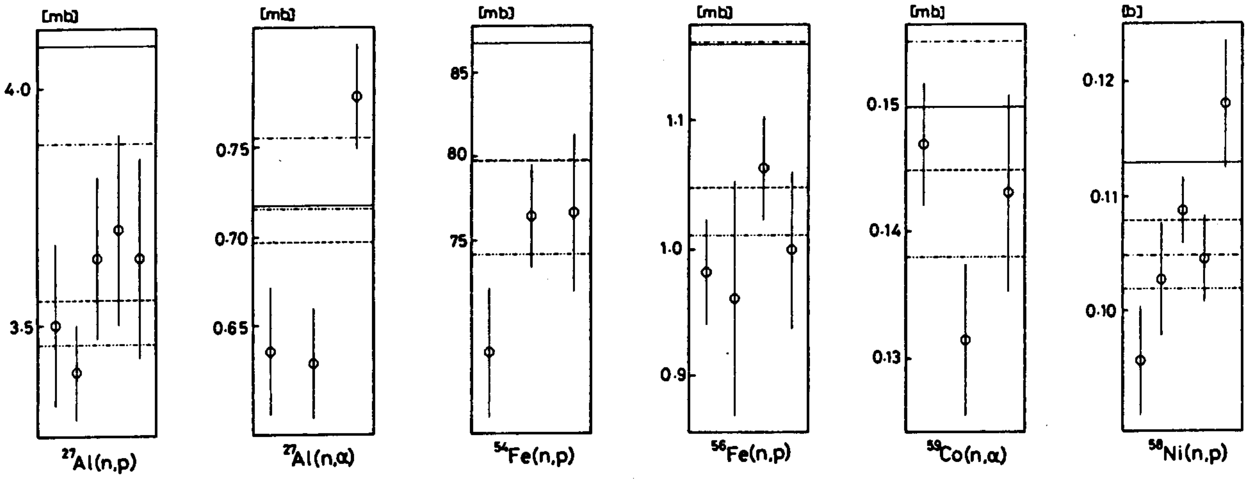

Fig.8 Comparison of evaluated and measured ^{235}U fission neutron spectrum averaged cross sections. Symbols: Experiments. Lines: Evaluations.

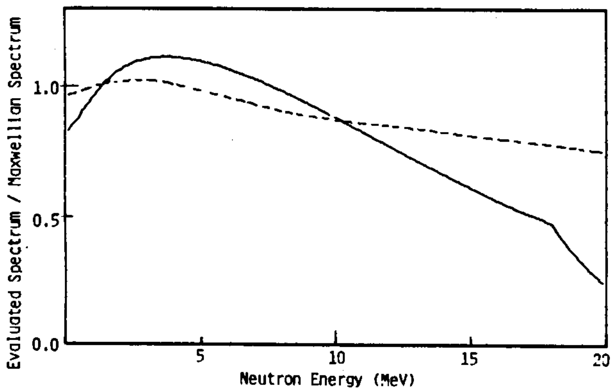

Fig.9 Ratios of evaluated fission neutron spectra and Maxwellian spectra.
Solid line: ^{235}U thermal fission neutron of JENDL-3T[4]. Dashed line: ^{252}Cf spontaneous fission neutron of Mannhart[5].

Conclusion

The simultaneous evaluation method developed in our laboratory[1,2] can be applied to estimate reaction cross sections using both differential and integral data. It is very valuable for the evaluation of the activation cross sections used in neutron dosimetry, since covariance matrices respecting neutron energy and reactions can be given. The cross sections evaluated by the present method are sensitive to the neutron spectra. The accurate neutron spectra are required.

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