Chapter 4

Monte Carlo Simulation of Correction Factors for Neutron Activation Foils

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Additional information is available at the end of the chapter

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Abstract

In the conventional neutron activation analysis, the elemental concentrations are normally determined from the comparison ratios between the measured specific activities of the sample and the standard reference material. An advantage in the comparison ratio method is that the systematic error due to neutron self-shielding and multi-scattering effects is canceled out, and the correction factors can be ignored but the preparation of reference standards to match the same conditions with those of various samples is the main difficulty. In the modern trend of neutron activation analysis, the K_0-standardization method has been developed and applied in almost all the NAA laboratories. An important research work in the procedure under this method is the characteristic information regarding the neutron source, such as thermal and epithermal neutron fluxes, and epithermal spectrum shape-factor. These neutron spectrum parameters are experimentally determined by using the activation foils, in which the corrections for all neutron effects cause systematic errors should be taken into account. Using the MCNP5 code, a well-known Monte Carlo simulation program, the results of correction factors of thermal, epithermal and resonance neutron self-shielding factors for Au, Co, Mn, W activation foils are presented in this chapter.

Keywords: neutron activation analysis, self-shielding factor, Monte Carlo simulation

1. Introduction

Neutron activation analysis is often performed with a reactor neutron spectrum. When the size of the irradiation sample or activation foils are not thin enough for ignoring the variance of neutron flux distribution, the information of the thermal, epithermal and resonance self-shielding effects should be considered for correction. The neutron capture reaction rate R(E),

at neutron energy E per one atom of the irradiating sample in a neutron flux distribution $\phi(E)$, is defined as the following expression:

$$R(E) = \phi(E)\sigma_{\gamma}(E), \tag{1}$$

where σ_{γ} (E) is capture cross section. During the irradiation processes, strong resonance reactions deplete the neutron spectrum at the resonance energy due to absorption and scattering. For the energy regions just lower than the resonance energies, the neutron distribution is raised due to multiple scatterings from the resonances. Therefore, the capture reaction rate is affected because of the perturbation in the neutron spectrum. It is assumed that, for a sample or monitor with similar thickness but infinitely diluted, the neutron distribution inside this sample or monitor is a none-perturbed spectrum. The self-shielding correction factors can be estimated as the ratio of the reaction rate by the none-perturbed to that by the perturbed spectra.

The problem of calculations for determining the neutron self-shielding correction factors has been considered in neutron capture experiments, but in neutron activation analysis it is still less of information and numerical data. These correction factors are always needed to be taken into account in the data analysis of experiments such as neutron reaction cross sections measurements, neutron flux and spectrum measurements, and neutron activation analysis, and so on. In these experimental studies, the thickness of the irradiated sample or standard monitor is frequently not thin enough for ignoring the variance of neutron flux distribution inside the sample space. Therefore, the correction factors for the self-shielding and multi-scattering effects of thermal, epithermal and resonance neutrons should be determined exactly. This research topic had been previously carried out, and reported in case by case, which can be briefly discussed as follows. Lopes et al. [1] calculated the values of epithermal neutron self-shielding factors, including isotopic scattering, for foil of Au-197 and Co-59. Eastwood et al. [2] reported experimental values of resonance neutron self-shielding factors for foils and wires of Co-59, by the activation technique. Brose [3] measured the resonance neutron absorption factors for Gold foils with different thickness. Hisashi Yamamoto and Kazuko Yamamoto [4] reported their calculated values of resonance neutron self-shielding correction factors for foils of Au-197, W-186, Mn-55 and In-115. The effects of Doppler broadening and potential scattering were taken into account, considering only main individual resonances. Senoo et al. [5] introduced a Monte Carlo code, TIME-MULTI, for neutron multiple scattering calculations with time-of-flight spectra. Shcherbakov and Harada [6] proposed a fast analysis method for calculations of epithermal neutron self-shielding factors, which was made used the Padé approximation for Doppler broadening function. Trkov et al. [7] introduced a computer program for self-shielding factors in neutron activation analysis, in which the calculation method is based on the neutron slowing-down equation. The program can be used for calculations with multi-element samples and applicable only for reactor isotropic neutron field. Gongalves et al. [8, 9] performed resonance neutron self-shielding factors for foils and wires of different materials by using the MCNP code and proposed universal curves for a number of neutron source geometries, and applicable only for single element samples.

Although the research topic on neutron self-shielding has been considered for a long history, the available resources for neutron activation analysis (INAA) applications are still limited in

both respects of numerical data and computational tools. The previous published values, calculations and measurements are mainly for the cases of resonance neutron self-shielding, and information is still lack for the cases of thermal and epithermal neutron self-shielding that are need in INAA with different standard foil and sample materials. In this chapter, the Monte Carlo code MCNP-5 has been applied for calculation of the thermal, epithermal and resonance neutron self-shielding factors for several standard neutron activation monitors that are often used in neutron activation measurements.

2. K_0-standardization method in neutron activation analysis

In the absolute standardization method, the concentration of the nuclide in a given sample can be determined as the following formula [10]:

$$\rho(\mu g/g) = N_p\left[\Delta_k.\frac{N_A.W.\theta.\gamma_k}{M}.\varepsilon_p.\Phi_e.\sigma_0.\left(G_{th}.f + G_{epi}.Q_0(\alpha)\right)\right].10^6 \tag{2}$$

In the K_0-standardization method, the concentration of the nuclide in a given sample can be determined as the following formula [10]:

$$\rho_a(\mu g/g) = \left[\frac{\left(\frac{N_p/t_m}{w.S.D.C}\right)_a}{A_{sp,Au}}\right].\frac{1}{k_{0,Au}(a)}.\frac{\left[G_{th}.f + G_{epi}.Q_0(\alpha)\right]_{Au}}{\left[G_{th}.f + G_{epi}.^{epi}Q_0(\alpha)\right]_a}.\frac{\varepsilon_{p,Au}}{\varepsilon_{p,a}}.10^6 \tag{3}$$

in which G_{th} and G_{epi} are the thermal and epithermal self-shielding correction factors and the k0-factor is defined as [10]:

$$k_{0,c}(s) = \frac{M_c.\theta_s.\sigma_{0,s}.\gamma_s}{M_s.\theta_c.\sigma_{0,c}.\gamma_c} \tag{4}$$

3. Determination of α in the $E^{-(1+\alpha)}$ epithermal neutron spectrum

In a nuclear research reactor, the distribution of epithermal neutrons per unit energy interval is considered inversely proportional to the neutron energy. However, this assumption is only valid if the following conditions are satisfied [11].

- The medium in which fast neutrons are being slow down is homogeneous and infinite.
- The fast neutron sources are homogeneously distributed.
- The slowing down power is energy independent.
- There is no absorption during the moderation processes.
- The moderating atoms have the same mass as the neutron and behave as free particles.

In practical situations, these conditions are almost not satisfied in a nuclear reactor. Accordingly, the deviations from the 1/E distribution of epithermal neutron can occur in irradiation channels, and errors could be induced in the resonance integral defined as

$$I = \int_{E_{Cd}}^{\infty} \frac{\sigma(E)}{E} dE \tag{5}$$

where: E_{Cd} = 0.55 eV is the Cadmium cut off energy.

In order to take into account, the correction for deviation effect in the expression of the resonance integral, the $1/E^{1+\alpha}$ distribution has been introduced [11] where α is an energy-independent coefficient but its values dependent on the neutron source configuration. Therefore, the effective resonance integral is rewritten as follows:

$$I(\alpha) = \int_{E_{Cd}}^{\infty} \frac{\sigma(E)}{E^{1+\alpha}} dE \tag{6}$$

Experimental determination of the coefficient α for a specific neutron irradiation channel or facility is required for exact estimation of the corresponding effective resonance integral. The relationship between resonance integral I and effective resonance integral I(α) is expressed as the following equation [11]

$$I(\alpha) = (I - 0.426\sigma_0)\left(\overline{E_r}\right)^{-\alpha} + \frac{0.426\sigma_0}{2\alpha + 1} E_{Cd}^{-\alpha}, \tag{7}$$

where $\overline{E_r}$ is the effective resonance energy, and σ_0 is the 2200 m/s neutron capture cross section. The method for experimental instantaneous determination of α value, based on co-irradiation of three suitable resonance monitors, has been introduced by F. De CORTE [11], which represent as follows.

The specific count rate for an interesting γ-peak, emission from an irradiated sample, is defined as

$$A_{sp} = \frac{1}{m} \frac{C\lambda}{(1 - e^{-\lambda t_1})(e^{-\lambda t_2})(1 - e^{-\lambda t_3})}, \tag{8}$$

where t_1, t_2, t_3 are the irradiation time, decay time, measurement time, respectively, C the number of counts under γ-peak, and m the weight of irradiated sample. The specific count rate, A_{sp}, can be also calculated from the following expression:

$$A_{sp} = [f + Q(\alpha)]\phi_{epi}\sigma_0\varepsilon\gamma\theta C/M \tag{9}$$

where M, θ, γ, ε are atomic weight, isotope abundance, γ-ray absolute intensity, and the efficiency of the detector used in the gamma-ray spectrum measurement, respectively. Q(α) is the ratio of the resonance integral in the $1/E^{(1+\alpha)}$ epithermal neutron spectrum to the (n,γ) reaction cross section σ_0; f is the ratio of thermal to epithermal neutron flux [11].

$$Q(\alpha) = \frac{I(\alpha)}{\sigma_0} = (Q - 0.426)\overline{E_r}^{-\alpha} + \frac{0.426}{2\alpha + 1} E_{Cd}^{-\alpha} \tag{10}$$

$$\mathrm{Q} = \mathrm{I}/\sigma_0 \tag{11}$$

By using the co-irradiation of two suitable standard foils, denoted as 1 and 2, the flux ratio f can be determined by the following equation [11].

$$f = \left(\frac{\phi_{th}}{\phi_{epi}}\right)_{1,2} = \left[\frac{k_1\varepsilon_1}{k_2\varepsilon_2} Q_1(\alpha) - \frac{A_{sp,1}}{A_{sp,2}} Q_2(\alpha)\right]\left[\frac{A_{sp,1}}{A_{sp,2}} - \frac{k_1\varepsilon_1}{k_2\varepsilon_2}\right]^{-1}, \tag{12}$$

$$\mathrm{k} = \gamma\sigma_0\theta/\mathrm{M}. \tag{13}$$

When three resonance detectors (foils), denoted as 1, 2 and 3, are irradiated under the same experimental conditions, Eq. (12) can be rewritten for detector couples 1–2 and 1–3. Making equality between the quantities $f_{1,2}$ and $f_{1,3}$ leads to the following equation:

$$F(\alpha) = (a - b)Q_1(\alpha) - (a + 1)Q_2(\alpha) + (b + 1)Q_3(\alpha) = 0 \tag{14}$$

where

$$a = \left[\left(A_{sp,1}k_2\varepsilon_2\right)/\left(A_{sp,2}k_1\varepsilon_1\right) - 1\right]^{-1} \tag{15}$$

$$b = \left[\left(A_{sp,1}k_3\varepsilon_3\right)/\left(A_{sp,3}k_1\varepsilon_1\right) - 1\right]^{-1} \tag{16}$$

The coefficient α would be experimentally determined by solving the Eqs. (12) and (14).

In an ideal 1/E epithermal neutron spectrum, the resonance integral cross section for a neutron capture reaction is defined as follows:

$$I_0 = \int_{E_{Cd}}^{\infty} \frac{\sigma(E)}{E} dE, \tag{17}$$

where σ (E) is the neutron capture cross section as a function of neutron energy E, and E_{Cd} is the cadmium cut-off energy to be 0.55 eV when Cd shield thickness is 1 mm. In a non-ideal epithermal neutron spectrum which can be approximated by $1/E^{1+\alpha}$ distribution, the resonance integral is defined as:

$$I_0(\alpha) = \int_{E_{Cd}}^{\infty} \frac{\sigma(E)(1eV)^{\alpha}}{E^{1+\alpha}} dE, \tag{18}$$

The relation between I_0 and $I_0(\alpha)$ is defined as the following expression:

$$I_0(\alpha) = (1eV)^{\alpha}\left[\frac{I_0 - 0.426\sigma_0}{E_r^{\alpha}}\right] + \frac{0.426\sigma_0}{(2\alpha + 1)(E_{Cd})^{\alpha}}. \tag{19}$$

The experimental values of resonance integral $I_0(\alpha)$ for target x can be determined relative to that of $^{197}Au(n,\gamma)^{198}Au$ as a standard reaction by the following relation [11]:

$$I_0(\alpha)_x = I_0(\alpha)_{Au} \frac{(g\sigma_0)_x}{(g\sigma_0)_{Au}} \frac{(R_{Cd} - F_{Cd})_{Au}}{(R_{Cd} - F_{Cd})_x} \frac{G_{epi,Au}}{G_{epi,x}} \frac{G_{th,x}}{G_{th,Au}}, \tag{20}$$

The deviation of the epithermal neutron spectrum from the 1/E shape parameter can be experimentally determined by the 'Cd-ratio for multi-monitor' method, using the monitors of ^{197}Au, ^{59}Co, ^{186}W and ^{55}Mn. When a set of n monitors are irradiated with and without Cd-cover, the α-parameter can be obtained as the slope $(-\alpha)$ of the straight-line $\log T_i$ versus $\log(E_{r,i})$ [11].

$$T_i = \frac{E_{r,i}^{-\alpha}}{(F_{Cd,i}R_{Cd,i} - 1)Q_{0,i}(\alpha)G_{e,i}G_{th,i}}, \tag{21}$$

where i denotes the i^{th} isotope, F_{Cd} the cadmium transmission factor, R_{Cd} the Cd-ratio, Er the effective resonance energy in eV, G_{th} and G_{epi} the self-shielding factor for thermal and epithermal neutrons, and $Q_0(\alpha)$ is the ratio of the resonance integral in $1/E^{1+\alpha}$ epithermal neutron spectrum to the capture cross section σ_0 for 2200 m/s neutrons. The parameter T_i is a function of the α parameter, which can be determined by an iterative least square fit to the regression line.

4. Monte Carlo simulation method for neutron self-shielding calculations

Monte Carlo (MC) simulation is known as an essential numerical method for performing the statistical process of radiation interaction with material. The principle MC simulation in this subject is random selection of particle properties and its interaction behaviours from their probability distribution functions. By tracking the history of each particle during the interaction process, the information of particle fluxes, energy spectra and energy deposition in a specific cell of the simulating model can be obtained. Accordingly, the radiation dose rate at any position in the environment of the experiment can be estimated with statistical uncertainty. A typical block diagram for process of neutron particle transport are shown in the **Figure 1** [5].

5. Neutron self-shielding correction factors

In the neutron activation analysis experiments, the thickness of the samples and monitors may not be thin enough for ignoring the variance of neutron flux distribution and should be considered for correction. Therefore, the correction factors for thermal (G_{th}), epithermal (G_{epi}) and resonance (G_{res}) neutron self-shielding effects should be determined exactly. In this work, the Monte-Carlo code MCNP5 was used for calculations with specified case of irradiation foils with different thickness from 1×10^{-5} to 2 mm.

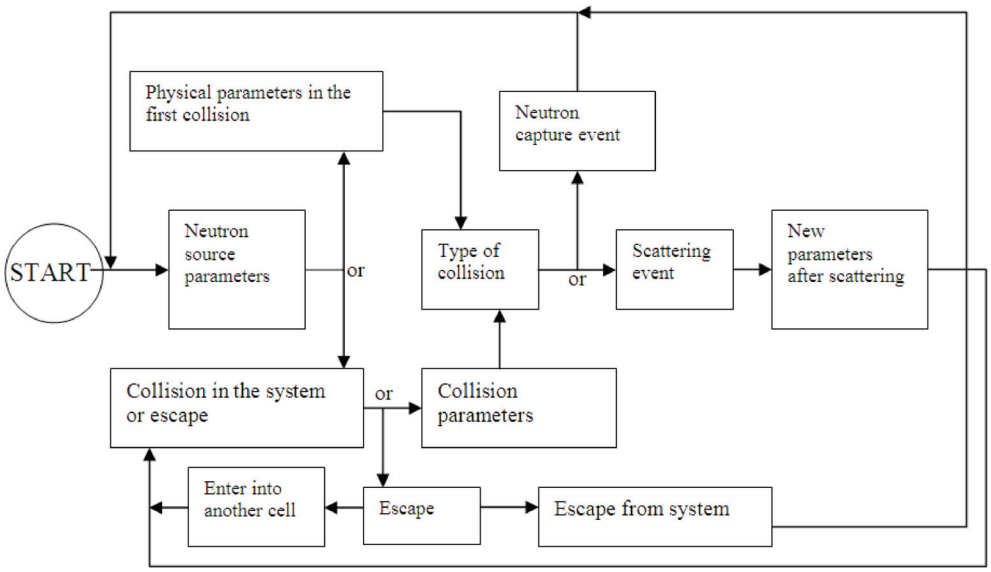

Figure 1. A typical block diagram for Monte-Carlo simulation of neutron transport process.

The neutron capture reaction rate R(E), at neutron energy E per one atom of the irradiating sample in a neutron flux distribution ϕ(E), is defined as the following expression:

$$\mathrm{R(E)} = \phi(\mathrm{E})\sigma_\gamma(\mathrm{E}), \tag{22}$$

where σ_γ (E) is capture cross section. During the irradiation processes, strong resonance reactions deplete the neutron spectrum at the resonance energy due to absorption and scattering. For the energy regions just lower than the resonance energies, the neutron distribution is raised due to multiple scatterings from the resonances. Therefore, the capture reaction rate is affected because of the perturbation in the neutron spectrum. It is assumed that, for a sample or monitor with finite thickness but infinitely diluted, the neutron distribution inside this sample or monitor is a non-perturbed spectrum. The correction factor a real sample with thickness 't' can be calculated as the following ratio.

$$G(t) = \int_{E_1}^{E_2} \varphi(E)\sigma_\gamma(E)dE / \int_{E_1}^{E_2} \varphi_0(E)\sigma_\gamma(E)dE, \tag{23}$$

where ϕ_0(E) is the original or non-perturbed neutron spectrum; ϕ(E) represents the perturbed neutron spectrum inside the real irradiating sample; E1 and E2 are, respectively, the lower and the upper limits of the neutron spectrum; for G_{th}: E1 = $1e^{-5}$ eV and E2 = 0.5 eV; for G_{epi}: E1 = 0.5 eV and E2 = 2e-1 MeV. The non-perturbed and perturbed neutron spectrum inside a real sample can be calculated by Monte-Carlo simulation using the MCNP5 code based on the ENDF/B-VII nuclear data library.

The results of MCNP5 simulations for perturbed and none-perturbed neutron spectra, $\phi(E)$ and $\phi_0(E)$, inside the real irradiating samples of Au-197 and Co-60 in different thickness is represented in **Figures 2** and **3**.

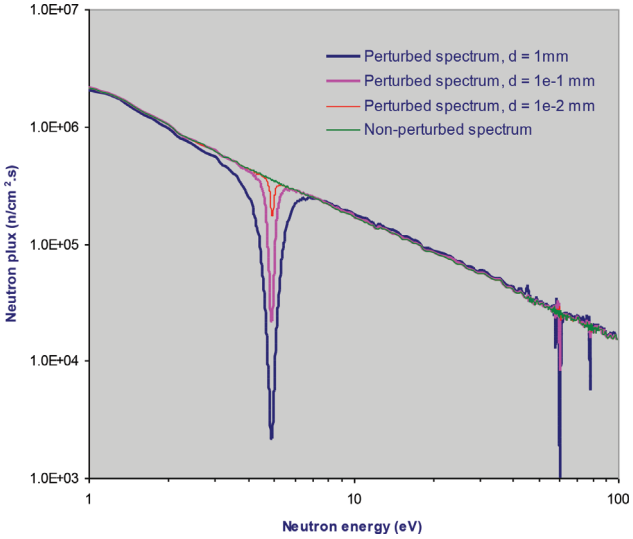

Figure 2. MCNP5 simulated results for perturbed and none-perturbed epithermal neutron spectrum in Gold foils of different thickness.

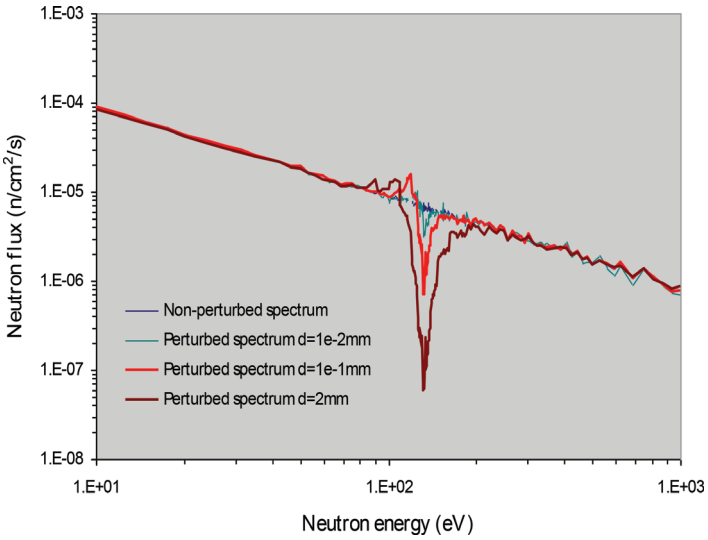

Figure 3. MCNP5 simulated results for perturbed and none-perturbed epithermal neutron spectrum in Cobalt foils of different thickness.

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In order to make validation for the simulation procedure, the comparison between the results Gres factors with the literature data for foil materials of Au-197 and Co-59 have been carried out with different sample thicknesses as presented in **Figures 2** and **3**. Up to present, from the literature overview, the available data of neutron self-shielding factors are almost for the study of resonance neutron self-shielding correction G_{res} which is considered only in the energy region of primary resonance peak of reaction cross-section, but in NAA the thermal and epithermal neutron (or effective) self-shielding correction factor should be taken into account. As shown in **Figures 4** and **5**, the result of present simulated G_{res} factors for Gold and Cobalt foils has reasonable agreement with the experimental and calculated data by Gongalves [9], Brose [12] and Eastwood [2].

In order to provide effective information of neutron self-shielding correction factors in NAA, the validated simulation procedure has been applied for obtaining G_{th}, G_{epi} and G_{epi} factors for the activation nuclides of Au-97, Co-59, Mn-55 and W-186. In this work, the foil samples with 1.3 cm in diameter and thickness varying from 10^{-5} mm to 2 mm were used for simulations. The experimental configurations were simulated for the real condition of a reactor-based neutron activation experiment in which the irradiation channel can be described as a cylindrical isotropic neutron sources with sample flat to the channel axis. The dimensions of the irradiating channel are 30 cm length and 2.4 cm in diameter. These simulations were conducted for three case studies of neutron energy spectrum: (i) the Maxwellian distribution with average energy of 0.025 eV was

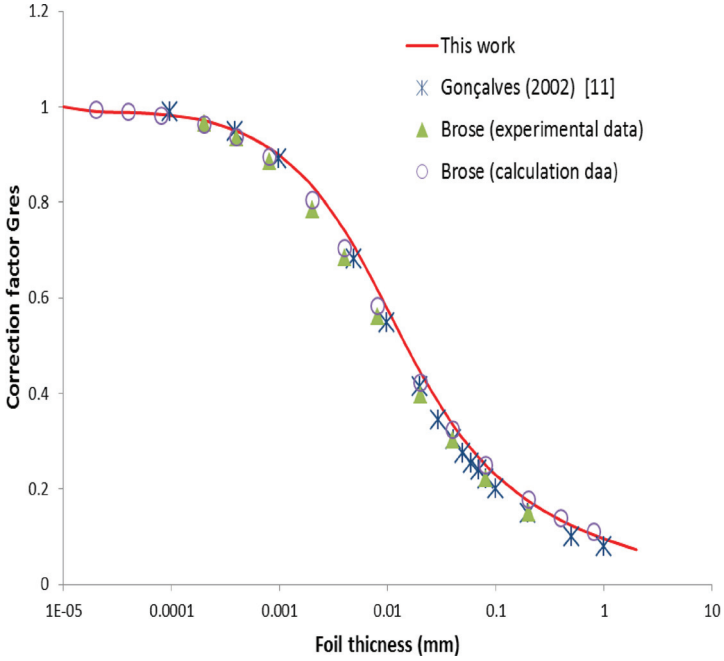

Figure 4. The calculated resonance neutron self-shielding correction factors for Gold foils of different thickness, in comparison with published data by Gongalves [9] and Brose [3].

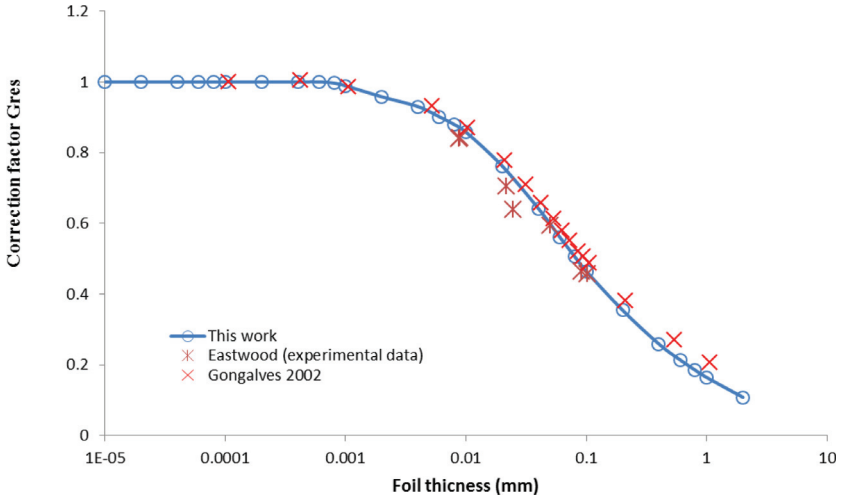

Figure 5. The calculated resonance neutron self-shielding correction factors for Cobalt foils of different thickness, in comparison with published data by Gongalves [9] and Eastwood [2].

applied for thermal neutrons, (ii) the pure 1/E distribution in energy range from 0.55 eV to 0.2 MeV was applied for epithermal neutrons, and (iii) the primary resonance energy peaks for resonance neutron spectrum. The results of thermal and epithermal neutron self-shielding correction factors for the activation foils of Au-197, W-186, Co-60 and Mn-55 in different thickness are presented in **Tables 1** and **2**.

Thickness (mm)	Neutron self-shielding correction factors for Au-197 foils			Thickness (mm)	Neutron self-shielding correction factors for Co-60 foils		
	Gth	Gepi	Gres		Gth	Gepi	Gres
0.00001	1	1	1	0.00001	1.000	1.001	1.000
0.00002	1	0.991	0.99	0.00002	1.000		1.000
0.00004	0.999	0.989	0.988	0.00004	1.000		1.000
0.00006	0.999	0.987	0.986	0.00006	1.000		1.000
0.00008	0.999	0.985	0.984	0.00008	0.999		1.000
0.0001	0.999	0.983	0.982	0.0001	0.999	1.001	1.000
0.0002	0.999	0.973	0.971	0.0002	0.999		1.000
0.0004	0.998	0.953	0.949	0.0004	0.998		1.000
0.0006	0.997	0.935	0.93	0.0006	0.997		1.000
0.0008	0.996	0.92	0.913	0.0008	0.997		0.997
0.001	0.996	0.906	0.898	0.001	0.996	0.998	0.989
0.002	0.993	0.846	0.834	0.002	0.995	0.996	0.958
0.004	0.988	0.76	0.74	0.004	0.992		0.930

Thickness (mm)	Neutron self-shielding correction factors for Au-197 foils			Thickness (mm)	Neutron self-shielding correction factors for Co-60 foils		
	Gth	Gepi	Gres		Gth	Gepi	Gres
0.006	0.984	0.696	0.671	0.006	0.989		0.902
0.008	0.981	0.646	0.618	0.008	0.987		0.880
0.01	0.977	0.607	0.575	0.01	0.985	0.992	0.858
0.02	0.961	0.486	0.445	0.02	0.974	0.988	0.762
0.04	0.934	0.384	0.334	0.04	0.957	0.981	0.642
0.06	0.912	0.335	0.282	0.06	0.941	0.977	0.562
0.08	0.893	0.306	0.251	0.08	0.928	0.974	0.506
0.1	0.876	0.286	0.229	0.1	0.916	0.971	0.465
0.2	0.805	0.234	0.175	0.2	0.865	0.961	0.355
0.4	0.707	0.194	0.135	0.4	0.792	0.946	0.259
0.6	0.636	0.175	0.116	0.6	0.736	0.935	0.214
0.8	0.58	0.162	0.104	0.8	0.690	0.925	0.186
1	0.535	0.152	0.096	1	0.651	0.915	0.164
2	0.388	0.124	0.073	2	0.516	0.873	0.108

Table 1. The results of neutron self-shielding correction factors foils of Au-197 and Co-60.

Thickness (mm)	Neutron self-shielding correction factors for Mn-55 foils			Thickness (mm)	Neutron self-shielding correction factors for W-186 foils		
	Gth	Gepi	Gres		Gth	Gepi	Gres
0.00001	1.000	1.000	1.000	0.00001	1.00	0.995	1.000
0.00002	1.000	1.000	1.000	0.00002	1.00	0.996	1.000
0.00004	1.000	1.000	1.000	0.00004	1.00	0.995	1.000
0.00006	1.000	1.000	1.000	0.00006	1.00	0.996	1.000
0.00008	1.000	1.000	1.000	0.00008	1.00	0.999	1.000
0.0001	1.000	1.000	1.000	0.0001	1.00	0.999	0.990
0.0002	1.000	1.000	1.000	0.0002	1.00	1.006	1.000
0.0004	1.000	1.000	1.000	0.0004	1.00	1.001	0.992
0.0006	1.000	1.000	1.000	0.0006	1.00	0.994	0.982
0.0008	1.000	1.000	1.000	0.0008	1.00	0.990	0.976
0.001	0.999	1.000	1.000	0.001	1.00	0.993	0.989
0.002	0.998	1.000	1.000	0.002	1.00	0.973	0.951
0.004	0.997	1.000	1.000	0.004	1.00	0.947	0.911
0.006	0.996	1.000	1.000	0.006	1.00	0.916	0.865
0.008	0.995	1.000	1.000	0.008	0.99	0.891	0.827

Thickness (mm)	Neutron self-shielding correction factors for Mn-55 foils			Thickness (mm)	Neutron self-shielding correction factors for W-186 foils		
	Gth	Gepi	Gres		Gth	Gepi	Gres
0.01	0.994	1.000	1.000	0.01	0.99	0.867	0.791
0.02	0.990	0.999	0.980	0.02	0.99	0.777	0.657
0.04	0.982	0.998	0.946	0.04	0.98	0.683	0.515
0.06	0.976	0.997	0.905	0.06	0.98	0.631	0.437
0.08	0.970	0.997	0.878	0.08	0.97	0.598	0.388
0.1	0.965	0.996	0.845	0.1	0.96	0.574	0.353
0.2	0.941	0.994	0.740	0.2	0.94	0.510	0.258
0.4	0.905	0.990	0.602	0.4	0.90	0.458	0.184
0.6	0.876	0.987	0.534	0.6	0.87	0.433	0.148
0.8	0.851	0.983	0.493	0.8	0.85	0.417	0.127
1	0.829	0.980	0.461	1	0.83	0.406	0.112
2	0.745	0.966	0.369	2	0.74	0.374	0.076

Table 2. The results of neutron self-shielding correction factors foils of Mn-55 and W-186.

6. Discussion and conclusion

Up to present, from the literature overview, the available data of neutron self-shielding factors are almost for the study of resonance neutron self-shielding correction G_{res} which is considered only in the energy region of primary resonance peak of reaction cross-section, but in NAA the thermal and epithermal neutron (or effective) self-shielding correction factor should be taken into account. Although the research topic on neutron self-shielding has been considered for a long history, the available resources for neutron activation analysis (INAA) applications are still limited in both respects of numerical data and computational tools. The previously published values, calculations and measurements, are mainly for the cases of resonance neutron self-shielding, and information is still lacking for the cases of thermal and epithermal neutron self-shielding that are needed in INAA with different standard foil and sample materials.

The MCNP5 code has been applied for exact simulation of neutron self-shielding correction factors of G_{th}, G_{res} and G_{epi} for activation foils of Au-97, Co-59, Mn-55 and W-186. The experimental configurations were simulated for the real condition of a reactor-based neutron activation experiment in which the irradiation channel can be described as a cylindrical isotropic neutron sources with sample flat to the channel axis. The three case studies of neutron energy spectrum: (1) the Maxwellian distribution with average energy of 0.025 eV was applied for thermal neutrons, (2) the pure 1/E distribution in energy range from 0.55 eV to 0.2 MeV was implemented for epithermal neutrons, and (3) the primary resonance energy peaks for resonance neutron spectrum. The results of simulated data with different foil thickness are presented in **Tables 1** and **2**.

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