Measurement of Neutron Spectrum in an Irradiation Chamber of a Research Reactor Using Activation Method (Postprint)

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Abstract

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Full Text

Preamble

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Using Activation Method to Measure Neutron Spectrum in an Irradiation Chamber of a Research Reactor

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Abstract: The neutron spectrum should be measured before test samples are irradiated. The neutron spectrum in an irradiation chamber of a research reactor was measured using the activation method during normal reactor operation at 2 MW. Sixteen kinds of non-fission foils (19 reaction channels) were selected, of which 10 were sensitive to thermal and intermediate energy regions, while the others had different threshold energies and were sensitive to fast energy regions. By measuring the foil radioactivity, the neutron spectrum was unfolded using the iterative methods SAND-II and MSIT. Finally, shielding corrections of group cross-sections and main factors affecting calculation accuracy were studied, and the uncertainty of the solution was analyzed using the Monte Carlo method in the SAND-II process.

Keywords: Neutron spectrum, Iterative method, SAND-II, MSIT, Group cross-section
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II. Theoretical Analysis

Most neutrons in Irradiation Chamber 3 of the research reactor are intermediate and fast neutrons, because most thermal neutrons are absorbed by the lead box. Many samples can be irradiated simultaneously in the irradiation chamber, and measuring the neutron spectrum at the irradiation point provides important information for irradiation experiments [**?**].

The activation method is a simple and widely used technique for measuring neutron energy spectra [**?**, **?**, **?**]. The foil size is small enough to ensure a uniform neutron field at the measuring point, and different foil materials can measure neutrons in different energy ranges. The activation method depends greatly on spectrum unfolding technology. In recent years, many unfolding methods have been developed that can be divided into three classes [**?**, **?**]: iteration methods, methods of undetermined coefficients, and Monte Carlo methods. The method of undetermined coefficients was once widely used but produces rough results. The Monte Carlo method dealing with a large number of neutron energy groups requires a great deal of computation time, while the reliability of the results is poor when a small number of activation foils is used. In recent years, with the development of computer technology and improvements in cross-section libraries, iteration methods have been widely used and produce much more accurate results. Both SAND-II and MSIT are iteration methods. In this work, MC uncertainty analysis was added to SAND-II to obtain even better results.

A. The Fundamental Procedures of Activation Method

The fundamental principle of the activation method for measuring neutron spectrum is as follows: a group of foils with known activation cross sections of the nuclides are irradiated at the measuring point, and the relationship between the single reaction rate and neutron energy spectrum is given by Eq. (1) [**?**]:

$$A_i = \int_0^\infty \phi(E)\sigma_i(E)dE \quad (i = 1, 2 \dots n),$$

where A_i is the single nuclear reaction rate of foil i, $\phi(E)$ is the neutron energy spectrum, $\sigma_i(E)$ is the nuclear reaction cross section of foil i at neutron energy E, and n is the number of adopted nuclear reaction channels. Stable nuclei are activated by neutron irradiation, and A_i is obtained by measuring γ-rays emitted from the activated nucleus. $\sigma_i(E)$ can be obtained from evaluated databases. $\phi(E)$ can be obtained by solving Eq. (1) in theory. The process of finding the solution $\phi(E)$ from Eq. (1) is called spectrum unfolding.

B. Basic Principle of Spectrum Unfolding

Due to the limited number of activation foils in the measurement, one can use the A_i data to obtain an approximate $\phi(E)$, rather than a continuous spectrum, by applying certain assumptions about the neutron field and restriction conditions.

In the iterative methods SAND-II and MSIT, neutron spectrum unfolding is performed by dividing the energy region into many energy groups, transforming Eq. (1) into Eq. (2):

$$A_i = \sum_{j=1}^{m} \phi_j(E)\sigma_{i,j}(E)\Delta E_j \quad (i = 1, 2 \dots n),$$

where $\phi_j(E)$ is the average neutron flux density of energy group j; $\sigma_{i,j}(E)$ is the average nuclear reaction cross section of energy group j for activation foil i; ΔE_j is the energy interval of energy group j; and m stands for the number of energy groups.

The initial neutron energy spectrum $\phi^{[0]}(E)$ can be calculated if $\phi_j^{[0]}(E)$ is obtained based on the conditions of the neutron field, the single nuclear reaction rate A_i^0, and group cross sections are known. The corrected spectrum of iteration $[K+1]$ can be obtained by using a correction factor and the spectrum of iteration $[K]$ for both SAND-II and MSIT. Eqs. (3) and (4) [**?**] are the modifier formulas for SAND-II and MSIT, respectively.

For SAND-II:

$$\phi^{[K+1]}(E) = \phi^{[K]}(E)\exp(C^{[K]}),$$

where $\phi^{[K]}$ and $\phi^{[K+1]}$ are energy spectra of iteration $[K]$ and $[K+1]$, respectively; $C^{[K]} = \sum_i W_i^{[K]} \ln R_i^{[K]} / \sum_i W_i^{[K]}$ is the correction factor, $W_i^{[K]} = \sigma_{i,j}/A_i^{[K]}$ is the weighting factor, and $R_i^{[K]} = A_i/A_i^{[K]}$ is the ratio of measured value to calculated value.

For MSIT:

$$\phi^{[K+1]}(E) = \phi^{[K]}(E)C^{[K]},$$

where $C^{[K]} = \sum_i(\sigma_{i,j}/A_i^{[K]}) / \sum_i(\sigma_{i,j}/A_i)$ is the correction factor of energy interval j.

The iterative process stops when one of three termination conditions is met: (1) the standard deviation between the calculated reaction rate and the measured reaction rate is less than a given value; (2) all differences between two adjacent iterative differential spectra for each energy group are less than a given value; or (3) the number of iterations reaches a given value.

III. Experimental Measurement of Reaction Rates

Sixteen kinds of activation foils were adopted in the experiment. The diameter of Mg and Mo foils is 8 mm, while the other foils are 20 mm in diameter. The characteristic parameters of activation foils and measured single nuclear reaction rates are shown in Table 1 . After proper cooling, γ-ray spectra of the irradiated foils were measured using high-purity Ge detectors with detection efficiency calibrated in advance. Eq. (5) is used to calculate the single reaction rate [**?**]:

$$A_i = \frac{C}{\gamma_d \varepsilon N_i (1 - e^{-\lambda_i t_0}) e^{-\lambda_i (t_1 - t_0)} (1 - e^{-\lambda_i (t_2 - t_1)})},$$

where C is the net peak area of the γ-ray spectrum; γ_d is the branching ratio of the γ-ray; ε is the detection efficiency; m_i is the mass of foil i; $N_i = (m_i/M_i)\theta_i \times 6.023 \times 10^{23}$ is the number of nuclei for foil i, M_i is the isotopic atomic weight of nucleus i, and θ_i is the isotopic abundance of nucleus i; λ_i is the decay constant of nucleus i; t_0 is the irradiation time; $t_1 - t_0$ is the cooling time; and $t_2 - t_1$ is the measuring time.

IV. Results

A. Processing of Group Cross Sections

The cross-section data for spectrum unfolding are from ENDF/B-VII. They are processed through the following procedures. First, the complete raw data is selected from ENDF using PREPRO and transformed to linearized form. The next procedure is to reconstruct the resonances. Then, Doppler broadening is performed based on the desired temperature. Finally, multi-group cross sections are generated. At this stage, a cross section is used as one of infinite dilution,

though in practical situations the foil has certain thickness, and sometimes thick foils are adopted.

When a foil of finite thickness is placed in the neutron field, the foil surface quickly absorbs neutrons with large absorption cross sections. The shielding effect reduces the average effective cross section of the foil. This directly affects the accuracy of spectrum results; therefore, a shielding correction should be applied to the group cross-section for foils of finite thickness.

Considering a collimated neutron beam of one-way incidence, the cross-section modification involves multiplying each group of infinite dilution cross sections with the shielding factor G_j for group j [**?**]:

$$G_j = \frac{1 - e^{-\tau_j}}{\tau_j},$$

where $\tau_j = t\Sigma_{a,j}$ is the thickness of the foil expressed in units of absorption length, t is the physical thickness of the foil, and $\Sigma_{a,j}$ is the macroscopic absorption cross section of group j. Using Co as an example, effective group cross sections for three conditions are compared in Fig. 1 [Figure 1: see original paper].

B. Spectrum Unfolding Results of SAND-II and MSIT

The neutron energy range from 10^{-4} to 2×10^7 eV was divided into 640 energy groups. The spectrum in the range 10^{-4}–3.6×10^{-3} eV is set to 0 because most neutrons in this range are shielded by the lead. The distribution of $\phi(E) \times E$ from 3.6×10^{-3} eV to 2×10^7 eV was simulated using MCNP, as shown in Fig. 2 [Figure 2: see original paper]. The simulation result of $\phi(E) \times E$ versus E was converted into $\phi(E)$ versus E and used as the initial spectrum for SAND-II and MSIT. The initial spectrum and the spectrum unfolding results are shown in Fig. 3 [Figure 3: see original paper]. The calculated single reaction rates are given in Table 2 , using the unfolding results and Eq. (2).

V. Discussion

Deviation in the neutron spectrum measured using the activation method includes the following sources: (1) measurement deviations of the single reaction rate. The deviation of any parameter in Eq. (5) affects the single reaction rate result, with deviations of approximately 3%–10%; (2) deviation caused by group cross-sections; (3) deviation caused by the initial spectrum; and (4) deviation caused by the spectrum unfolding process itself.

Selecting a well-collocated set of foils helps the results converge closer to the real neutron energy spectrum.

In Fig. 3, the spectrum unfolding results using SAND-II and MSIT are nearly identical. From Table 2, the difference between calculated and measured data

for ^{58}Ni(n,p)^{58}Co is large due to measurement error. The singular point in the resonance region in Fig. 3 is also due to the measurement error of ^{58}Ni(n,p)^{58}Co.

The uncertainty analysis [**?**, **?**] is performed by Monte Carlo in the following process [**?**]: (1) the error of the single reaction rate and the uncertainty of the cross section for each activation foil are specified; (2) new reaction rates and nuclear cross sections are generated using random sampling methods that extract deviations from the original reaction rates and nuclear reaction cross sections; and (3) the uncertainty of the iterative solution spectrum is calculated.

The relative deviation for each energy group in the spectrum unfolding by SAND-II is shown in Fig. 4 [Figure 4: see original paper]. The uncertainty is $±$5% for the energy region from 10^{-4} eV to 100 eV; $±$30% for the energy region from 100 eV to 5×10^4 eV; and $±$15% for the energy region from 5×10^4 eV to 2×10^7 eV. From this analysis, the deviation of group cross sections near the resonance peak is larger.

VI. Conclusion

Measurements made with the foil-activation technique provided valuable information about the neutron spectrum in Irradiation Chamber 3 of a research reactor. The group cross sections for foils of finite thickness were processed before unfolding.

Two spectrum unfolding methods based on iterative principles were used. The unfolding results are generally consistent. The uncertainty analysis of the unfolding result was performed by Monte Carlo, which was added to the SAND-II process. If the result were not dependent on the initial spectrum, SAND-II would be perfect. From the above analysis, the experiment is successful due to the suitable foils and spectrum unfolding methods. The neutron spectrum of Irradiation Chamber 3 can now be known, which is important for future irradiation tests.

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Note: Figure translations are in progress. See original paper for figures.