DEVELOPMENT OF COMPUTER SOFTWARE FOR NEUTRON ENERGY SPECTRUM ADJUSTMENT IN RESEARCH REACTORS

by

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Received on December 18, 2008; accepted in revised form on March 27, 2009

A computer program has been developed for neutron energy spectrum adjustment using the deconvolution method. The BUNKI-based algorithm has been implemented to converge quickly yielding calculated neutron energy spectrum which is in good agreement with theoretical predictions. The foil activation data have been used as an input for each unfolding technique and various activation foils including Au-197, Al-27, Ni-58, Co-59, and Mg-24 covering thermal to fast energy range have been utilized. The group cross-section values were derived from the data given in the pre-processed cross-section libraries in ENDF-6 format of IRDF-90/NMF-G. Firstly, virtual approach was used for neutron energy spectrum adjustment. In this case, the activity of foils before and after the adjustment was almost the same but the flux had the maximum error of 14%. Secondly, the experimental measured activity of the threshold foils was then used for a real system. The activity of the threshold foils before and after the neutron energy adjustment had the maximum error of 33%.

Key words: neutron spectrum, BUNKI, PARR-2, spectrum adjustment, research reactors

INTRODUCTION

The neutron spatial distribution and energy spectrum provide the key information for the calculation of radiation damage, neutron dosimetry, fast neutron physics, isotope production, and safety analysis. The distribution of neutron energies in a reactor differs from the fission neutron spectrum due to the slowing down of neutrons in elastic and inelastic collisions with fuel, coolant, and construction materials. The determination of neutron energy spectrum in a nuclear reactor is essential for analyzing its irradiation characteristics [1-3].

The identification of neutron sources by unfolding the initial neutron energy spectrum is a very promising method, based on the observable differences in the spectra of different sources. The knowledge of neutron spectral distribution produced by neutron sources also plays an important role in neutron dosimetry studies, neutron-induced nuclear data measurements, radiation analysis of bulk samples, and fusion reactor technology. Neutron spectra can be measured normally by conventional spectrometers and indirectly by the activation unfolding technique. This knowledge is also necessary for core neutronics, radioisotope production, and for the estimation of dosimetric values such as material damage and dose equivalent.

The radio-activation technique is one of the earliest methods of neutron detection that still provides a basic means for monitoring the neutron flux density in a radiation field [1]. An activation detector may be defined as any piece of material used to detect neutrons by measuring induced activity following exposure in a neutron environment. Detectors sensitive to various neutron energies are available.

Therefore, by selecting the appropriate set of detectors, neutrons spectra can be unfolded in thermal, intermediate, and fast energy regions.

The determination of dosimetric parameters on the basis of false information about the neutron energy spectrum can result in substantial economic penalties and it may bring harm to health. Furthermore, the uncertainties in the determination of the neutron spectrum are also important in estimating the error in the

Nuclear Technology & Radiation Protection
Scientific paper
UDC: 539.125.52:519.254:621.039.572
BIBLID: 1451-3994, *24* (2009), 1, pp. 13-17
DOI: 10.2298/NTRP0901013I

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dosimetric values. Therefore, it is important to determine the neutron energy spectrum.

For fission neutrons, present in a reactor, the entire neutron energy spectrum can be considered to contain three different energy distributions of neutrons at the given time [4].

– the first is a fission neutron spectrum, which is likely to be the Watt distribution,
– the second is a slowing down spectrum, which is characterized by a $1/E^{(1+\alpha)}$distribution where α is the shape parameter, and
– the third is a thermal neutron spectrum, which is almost a Maxwellian distribution characterized by the temperature of the medium.

For the measurement of neutron energy spectra, the method of multiple foil activation combined with the neutron spectrum adjustment code is frequently used. Various unfolding codes and methods have been developed for solving the inverse problem for particle detection and spectrum determination. In reactor dosimetry for "few-channel" unfolding the codes based on linear least-squares methods have been successfully applied in the STAY'SL, LEPRICON , DIFBAS, DIFMAZ of the HEPRO package, and MSITER and MINCHI are the representatives of a large number of available least-squares adjustment codes using a priori information on the fluence. These codes are also successfully applied to the unfolding of measured data obtained by Bonner spheres [4-7].

As the second group, the codes excluding negative fluence values, *e. g*. SAND-II, GRAVEL (based on an improved SAND-II algorithm), LSL-M2 17, LOUHI 25, BUNKI 26, RADAK 27, and Genetic Algorithms (GA) have to be mentioned. These codes in principle perform a non-linear least-squares adjustment, with the constraint of non-negative particle fluence [2].

The emphasis has been laid on how to obtain a unique solution through the proper selection and treatment of the problem parameters. It aims on establishing confidence in the procedure used for the calculation of the neutron energy spectrum for use in routine measurements as well as for the prediction of the neutron flux spectrum at other locations which were not measured.

In this work, a number of threshold detectors and the 1/v activation detector in the form of thin foils were used for spectrum measurements. A high efficiency high purity germanium (HPGe) detector was used for the activity measurement of the foils. The measured activity values of foils were used as the input to various unfolding algorithms.

THEORY

The methodology for the BUNKI approach for the neutron spectrum adjustment can be written as based on the basic equation of the threshold detector activity

$$A \quad R\Phi \tag{1}$$

where A is the vector of activities in all the detectors, Φ is the unknown neutron spectrum, and R (reaction rates) is the response matrix ($\Sigma = \sigma N$). As we had a number of threshold detectors less as compared to neutron flux groups, the problem became undetermined. To solve it, we have multiplied both sides of eq. (1) by the transposed response matrix

$$R^T A \quad R^T R\Phi \tag{2}$$

or

$$Y \quad X\Phi \tag{3}$$

where and $X = R^T R$ and $Y = R^T A$

After necessary manipulations, we get

$$\Phi_{n\ 1}(E) \quad \frac{\Phi_n(E)}{\sum_{j\ 1}^{m} \sigma_j(E)} \sum_{i\ 1}^{n} \sigma_i(E) \frac{N_i^0}{N_i^n} \tag{4}$$

where $\Phi_n(E)$ is the initial neutron spectrum, $\Phi_{n+1}(E)$ is the unfolded spectrum after n iterations, N_i^0 is the initial detector readings, N_i^n is the calculated detector readings after n-iterations and m is the number of detectors (foils). $\sigma_i(E)$ means the group cross-section.

The algorithm of the solution of eq. (1) shows that the neutron spectrum after $n + 1$ iterations is determined by the spectrum after n-iterations and by the ratio of the initial and calculated detector readings, with the statistical weight being equal to the i^{th} detector's reading with energy E. It is possible, however, to increase the speed of the iterating process. If the preference in the unfolding spectra for each energy E is given to the detectors that are most sensitive to this energy, then we can obtain it by using the following algorithm *i. e*.

$$\Phi_{n\ 1}(E) \quad \frac{\Phi_{\text{n}}(E)}{\sum_{j\ 1}^{m} \frac{\sigma_{\text{j}}(E)}{N_j^0}} \sum_{i\ 1}^{n} \frac{\sigma_i(E)}{N_i^0} \frac{N_i^0}{N_i^n} \tag{5}$$

In this expression, the statistical weight with which the ratio N_i^0/N_i^n enters the neutron spectra after $n + 1$ iterations is proportional not to the cross-section but to the relative contribution of the energy E in the i^{th} detector's reading.

UNFOLDING PROCEDURE

We used two approaches for the neutron spectrum adjustment: (1) virtual, and (2) original. Twenty neutron energy group structure is used in these calculation as tabulated in tab. 1.

Table 1. Twenty groups of neutron energy structure in calculation [4]

Group No.	Upper energy boundaries [MeV]
1	$1.00 \cdot 10^{-10}$
2	$5.00 \cdot 10^{-9}$
3	$1.00 \cdot 10^{-8}$
4	$2.50 \cdot 10^{-8}$
5	$5.00 \cdot 10^{-8}$
6	$1.40 \cdot 10^{-7}$
7	$3.50 \cdot 10^{-7}$
8	$6.25 \cdot 10^{-7}$
9	$9.72 \cdot 10^{-7}$
10	$1.15 \cdot 10^{-6}$
11	$4.00 \cdot 10^{-6}$
12	$7.55 \cdot 10^{-5}$
13	$1.43 \cdot 10^{-3}$
14	$9.11 \cdot 10^{-3}$
15	$1.83 \cdot 10^{-1}$
16	$3.03 \cdot 10^{-1}$
17	$8.21 \cdot 10^{-1}$
18	$2.23 \cdot 10^{0}$
19	$3.68 \cdot 10^{0}$
20	$6.07 \cdot 10^{0}$

Implementation with virtual approach

First of all, the activities of the threshold detectors were calculated by multiplying the known fluxes with the cross-section. This technique is called the virtual approach. By this technique, we find out how true the results can be discovered with this algorithm using different guess neutron flux. After applying the developed software, the activities and flux values were generated. The recalculated activities are shown in tab. 2 with their relative error percentages. As shown in fig. 1, it is clear that the algorithm works perfectly as the relative percentage error in the activities is equal to 0. So we can apply this algorithm on the true measured values of the activities to find out the fluxes. The unfolded spectrum obtained from these values of regenerated activities is shown in fig. 2 with comparison with original neutron spectrum.

Table 2. Activities with relative error for virtual approach

Materials	Assumed saturation activities (A_{sat}) [s^{-1}]	Calculated activities (A_{sat}) [s^{-1}]	Relative error [%]
Au-197	$2.82\ 10^{12}$	$2.82\ 10^{12}$	–0.0006
Au-197a	$1.99\ 10^{10}$	$1.99\ 10^{10}$	0.0004
Al-27	$1.42\ 10^{14}$	$1.42\ 10^{14}$	–0.0014
Ni-58	$3.96\ 10^{13}$	$3.96\ 10^{13}$	0.002
Co-59a*	$1.32\ 10^{8}$	$1.32\ 10^{8}$	–0.1225
Mg-24	$3.04\ 10^{8}$	$3.04\ 10^{8}$	0.122

*a – cadmium covered

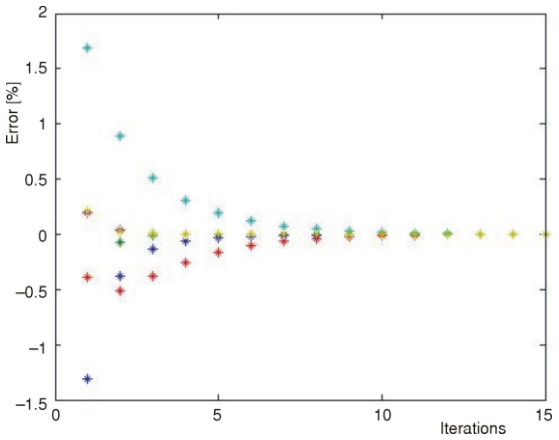

Figure 1. Variation of activity error with iteration in virtual approach

Implementation with original

Different threshold thin foil (Au, Ni, Co, Mg, Al) detectors' activities were measured in Pakistan Research Reactor-2 (PARR-2) [4]. The measured activities with their errors in measurement are shown in tab. 3.

These original values of the activities were then used to find out the flux values as the algorithm fitted on the virtual approach. Then the relative error in percentage was found out with the original measured activities. The graph obtained for the relative error percentage is shown in fig. 3.

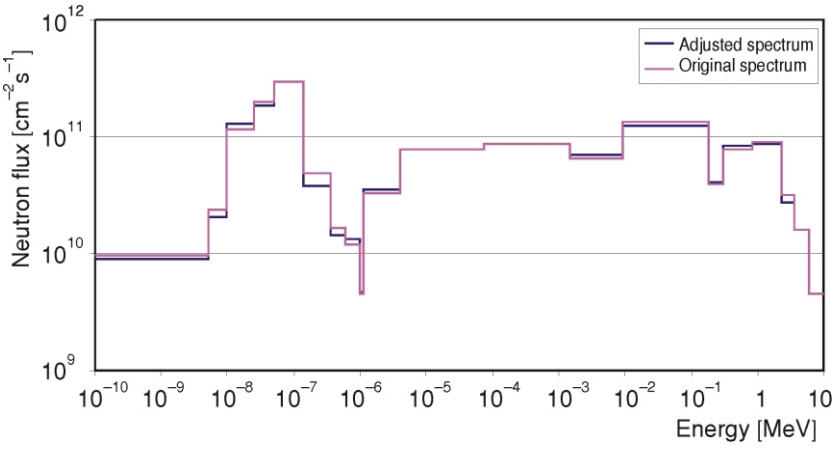

Figure 2. Adjusted and original neutron spectra in virtual approach

Table 3. Measured saturation activities with errors [4]

Materials	Measured saturation activities (A_{sat}/nuclei) [s^{-1}]	Variances [%]
Au-197	$2.82\ 10^{-12}$	6.28
Au-197a*	$1.99\ 10^{-14}$	7.67
Al-27	$1.42\ 10^{-10}$	5.66
Ni-58	$3.96\ 10^{-11}$	5.56
Co-59a	$1.99\ 10^{-16}$	9.9
Mg-24	$2.28\ 10^{-16}$	5.96

*a – cadmium covered

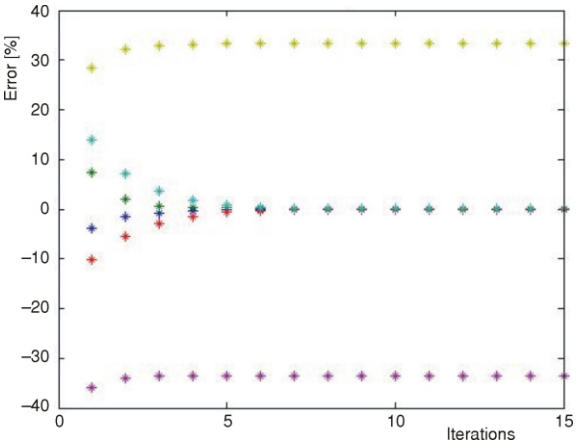

Figure 3. Variation of activity error with iteration in real experimental data analysis

RESULTS AND CONCLUSIONS

A computer program is developed using MATLAB. The BUNKI approach based on deconvolution method was employed. As the first step, the virtual approach was used to check the quality of the computer program. Then the neutron spectrum was adjusted in a research reactor. In the case of underdetermined problems, the unfolding programs generally make use of 'a-priori' information that constitutes the best approximation to the output to be determined. In this case, the calculated spectrum was taken as the priori spectrum. The group cross-section values were derived from the data given in the pre-processed cross-section libraries in ENDF–6 format of IRDF-90/NMF-G. In the BUNKI method, the accuracy of the unfolding results is not strongly dependent on the input spectrum.

In the virtual approach, the program adjusted the activities of the foils in high precision and there was no error in the input and output activities as shown in tab. 2, whereas the output neutron spectrum has some error in one group as high as 14% maximum. The error in the activities decreases with iterations and can be seen in fig. 1. The input and output neutron spectrum for the virtual approach is shown in fig. 2. The neutron adjustment program was used for a real system in Pakistan Research Reactor-2 (PARR-2) [8]. The activity of the threshold activation foils irradiated in the core is taken from the reference [4]. In this case, the calculated neutron spectrum was used as the input/guess spectrum. Also, it can easily be seen that the error percentage is high in the last two foils. The results are shown in tab. 4. This may be due to the error in the measurements. The second reason is the number of energy groups. For the last two foils only two groups were available. If in the fast region the number of the energy groups became increased, this error would also be decreased. The adjusted and calculated neutron spectra are shown in fig. 4.

Table 4. Activities with error for original data

Materials	Measured saturation activities (A_{sat}/nuclei) [s^{-1}]	Calculated activities (A_{sat}) [s^{-1}]	Relative error [%]
Au-197	$2.818\ 10^{-12}$	$2.818\ 10^{-12}$	0.18
Au-197a*	$1.989\ 10^{-14}$	$1.993\ 10^{-14}$	0.4713
Al-27	$1.417\ 10^{-10}$	$1.417\ 10^{-10}$	0.18
Ni-58	$3.96\ 10^{-11}$	$3.96\ 10^{-11}$	0.1799
Co-59a	$1.989\ 10^{-16}$	$1.323\ 10^{-16}$	–31.3398
Mg-2	$2.278\ 10^{-16}$	$3.035\ 10^{-16}$	31.4086

*a – cadmium covered

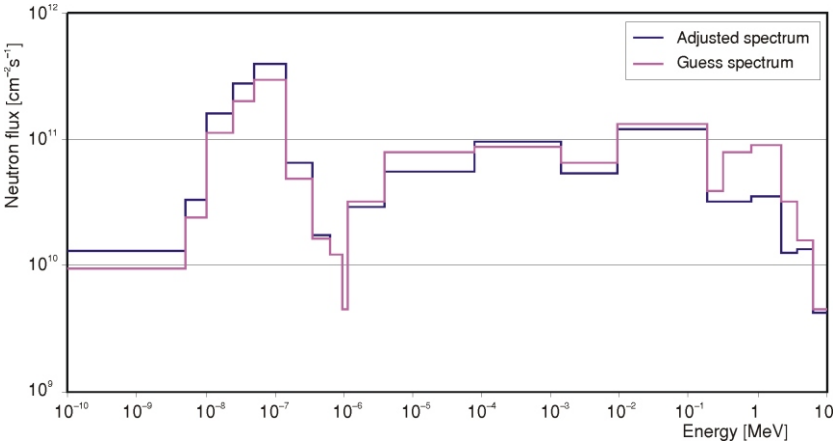

Figure 4. Adjusted and calculated/guess neutron spectra in real experimental approach

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РАЗВОЈ РАЧУНАРСКОГ СОФТВЕРА ЗА ПОДЕШАВАЊЕ НЕУТРОНСКОГ СПЕКТРА У ИСТРАЖИВАЧКИМ РЕАКТОРИМА

Коришћењем деконволуционе методе развијен је рачунарски програм за подешавање неутронског спектра. Употребљен је алгоритам заснован на BUNKI процедури, који брзо конвергира и израчунава неутронски спектар у доброј сагласности са теоријским предвиђањима. Подаци о активацији фолија коришћени су у сваком деконволуционом поступку, а употребљене су различите активационе фолије, укључујући Au-197, Al-27, Ni-58, Co-59 и Mg-24, да покрију подручје од термичких до високих енергија. Вредности групних пресека изведене су из података садржаних у библиотеци пресека ENDF-6, у формату IRDF-90/NMF-G. За подешавање неутронског спектра најпре је коришћен виртуелан приступ. У овом случају, активност фолија пре и после подешавања била је скоро иста, али са максималним одступањем флукса од 14%. Затим је, за реални систем, коришћена експериментално мерена активност праг фолија. Пре и после подешавања неутронске енергије, активност праг фолија максимално је одступала 33%.

Кључне речи: неутронски спектар, BUNKI, PARR-2, подешавање спектра, истраживачки реактор