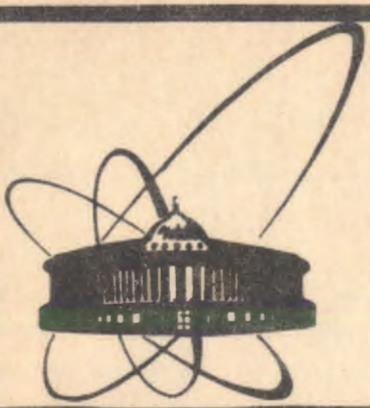

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ANS - THE ANALYSIS OF THE NEUTRON SPECTRA

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1. Nature of physical problems

In the past years in the Joint Institute for Nuclear Research at large significance were found group methods of neutrons calculation [3]. The neutron spectrum is divided into a constant number groups. In each group there are functionals and parameters, which described interaction neutrons and structure of experimental materials. The group's parameters depended on neutron's energy and therefore we compute the average resonance parameters in the energy partition of adequate group.

It is known what the statistics of group cross sections and functionals in the unresolved energy region is facilitated and applied to the fitting of average resonance parameters to the poorly resolved experimental data. In the resolved energy region all neutron sections can be parameterized by one-level approximate theory, by means of small number resonance parameters: E_0, Γ_n, Γ_x, Γ.

In the unresolved energy region one-level approximate theory is unfit and the problem parameterization of resonance speed section is strongly complicated. Usually the resonance structure is computed by an extrapolation of resonance parameters in resolved region [1], in this method significance of detailed resonance structure in averaged section is not necessary and sufficiently computing average resonance parameters: Γ_f, Γ_n, Γ_γ, D, distribution of width near average significance. All the group constants in the unresolved region, can be computed from experiment by reading resonance structure.

The experiments intended to calculate the energy of reactors on fast neutrons in 25-groups system [2], have been performed at Laboratory of Neutron Physics of JINR. Neutron

transmission and self-indication functions in the U-238 capture cross section were measured on the 120m, 500m and 1000m flight paths of the IBR-30 booster at neutron energies $E_n = 1\text{-}10^7$ eB. A 16-section scintillation gamma-detector was used to measure seft-indication.

The necessity of the graphical interpretation and modeling of neutron transmission experiments is large. The analysis of the existing programs on computers IBM XT/AT shows, that most of them are not used for the considerated interpretation of experiments.
In this paper is described the program ANS, which is intended to be a helpful tool for these problems.

2. Description of the ANS

The Analysis of Neutron Spectra (ANS) is the graphical user friendly program to process evaluated neutron data files for interpretation of transmission experiments. By this program one may calculate the neutron transmission, self-indication functions, the areas under peaks and spectra normalization. It is possible to carry out the work with two spectra simultaneously.

The ANS program was written in the Turbo Pascal v.5 language and may work on the IBM AT computer with Math CoProcessor. ANS consists of six main modules, and requires 160K bytes memory. To use this program there must be created subdirectory: /ans/data - for neutron data files and /ans/result - for results of processing operations. The ANS program profits from the ans.hlp file, which must be in the same directory as the ans.exe file.

The results of all the operations in the program may be written onto files for illustration by other graphical programs (as Grapher) or for view on the screen. This program consists of the following main options, which are called by user in the graphical form:

1. Reading of one or two files, containing the neutron data.
2. Selection of one or two sections, containing the files of neutron data.
3. Editing of the files of neutron data.
4. Definition and change of parameters used for description of an experiment.
5. Processing operations for each of the spectrum or the section. These are:
   (a) Dead time correction,
   (b) Stable and variable background calculation,
   (c) Channel's transmission calculation,
   (d) Group's transmission calculation,
   (e) Group's cross-section calculation,
   (f) Arithmetical operations:
   - Summarize of spectra,
   - Divide of spectra,
   - Multiple of the spectrum by a number,
   - Computing of an area in the specified intervals,
   - Normalization of spectra.
6. Graphical operations , these are view of the results of the processing operations.

Reading of the neutron data

At the beginning of the using, the ANS is reading one or two files (File Mode), containing the neutron data (8096 bytes - 4096 numbers: two bytes for one number), or selecting one or two sections, containing the files of the neutron data (Section Mode).

In both modes the user can select the files from the displayed subdirectory: /ans/data/*.* , this directory can be changed.

For each spectrum user must define the monitor. In Section Mode for each section after the selection of files is

generated the file containing the arithmetical average of all spectra in this section. This file consists of the new spectrum, which can be used to the next analysis.

The results of processing the operations may be generated in ASCII format and can be written in the subdirectory: /ans/result/*.*.

Editing of the neutron data

After reading of the specified file the user can edit the neutron data by the small screen editor of the ANS program. For each channel the number of neutrons can be changed and printed in the ASCII format or written onto an new file in input format.

Changing of the parameters

The ANS program uses the anssys.sys file, containing beginning of experiment's parameters: number of channels, dead time coefficient, channel's width, reactor's pulses, widths of energy groups, channel's interval, monitors and definition of energy groups. If this file is not in the actual directory, it is generated by the ANS program. All these parameters can be changed and written onto an actual anssys.sys file.

Processing operations

The formulae of some processing operations of the ANS system have been presented in below. The formulae are not complicated, but they may not be obvious to the first-time user of ANS. Presented operations are performed in the described succession.

Dead time correction of the neutron data

The neutron data can be corrected in the mathematical terms as follows:

$$Nc_i = \frac{N_i}{1 - \frac{a * N_i}{t * N_o}}$$

where: Nc_i - corrected number of neutrons on the channel i,
N_i - number of neutrons on the channel i,
a - dead time coefficient,
t - channel's width, N_o - reactor's pulses.

This module created the file: *xxx*dc.dat, where *xxx* is the initial three chars in the name of the specified file with the neutron data.

Stable background calculate

The stable background for the spectrum is calculated in the channel's interval [ch1,ch2] according to the formulae:

$$N_{av} = \frac{\sum_{i=ch1}^{ch2} p_i * N_i}{\sum_{i=ch1}^{ch2} p_i} \qquad @N_{av} = \frac{1}{\sqrt{\sum_{i=ch1}^{ch2} p_i}}$$

where: N_{av} - an average number of neutrons in interval [ch1,ch2],
$@N_{av}$ - statistical error for N_{av},
p_i - weight [$p_i = 1/(@N_i)^2$; $@N_i = \sqrt{N_i}$].

The next step in this module is calculation of the new N_i'' and $@N_i''$:

$$N_i'' = N_i - N_{av}; \quad @N_i'' = \sqrt{(@N_i)^2 + (@N_{av})^2}, \quad i=1..4096.$$

This module created the file: *xxx*sb.dat, where *xxx* is the initial three chars in the name of the specified file with the neutron data.

Variable background calculate

This module calculates coefficients: a_j in the formula:

$$f(x_i) = \sum_{j=-1}^{m} a_j * (x_i)^{-j}, \quad m = 1..3,$$

by the least squares metod.
The upper formula interpolates the neutron spectrum in the specified by user points: (x_i-channel,Nx_i) for i=1..n. Subsequently for the spectrum is calculated the new $N_i\tilde{}$ for each i=1..4096 as follows: $N_i\tilde{} = N_i - f(x_i)$.

This module created the file: *xxx*vb.dat, where *xxx* is the initial three chars in the name of the specified file with the neutron data.

Channel transmission

The channel transmission for the two spectra is calculated as follows:

$$T_i = \frac{N_i'(1)}{N_i'(2)}; \qquad N_i'(1) = N_i\tilde{}(1)/Mt_1 \text{ (first spectrum)}$$
$$N_i'(2) = N_i\tilde{}(2)/Mt_2 \text{ (second spectrum)}$$

$$@T_i = T_i * \sqrt{[@N_i'(1)/N_i'(1)]^2 + [@N_i'(2)/N_i'(2)]^2}$$

where: T_i - the transmission in channels i=1..4096,
$@T_i$ - statistical error for T_i,
Mt_1 - the monitor for the first spectrum,
Mt_2 - the monitor for the second spectrum.

This module created the file: *xxxyyy*ct.dat, where *xxx* and *yyy* are the initial three chars in the names of the specified files with the neutron data.

Group transmission

This module first of all calculates for each spectrum the average number of neutrons for the described by user energy groups according to the formulae:

$$N_l(s) = \frac{\sum_{i=lch(l)}^{rch(l)} p_i * N_i}{\sum_{i=lch(l)}^{rch(l)} p_i}; \qquad @N_l(s) = \frac{1}{\sum_{i=lch(l)}^{rch(l)} p_l}$$

where:
l - group's number (l=0..24),
s - spectrum number (s=1 v s=2),
lch(l),rch(l) - left and right channel of l-group.

Subsequently is calculated the transmission Tl for each group and the statistical error for Tl:

$$T_l = \frac{N_l(1)}{N_l(2)}, \qquad l = 0..24,$$

$$@T_1 = T_1 * \sqrt{[@N_1'(1)/N_1'(1)]^2 + [@N_1'(2)/N_1'(2)]^2}\ .$$

This module created the file: *xxxyyy*gt.dat, where *xxx* and *yyy* are the initial three chars in the names of the specified files with the neutron data.

Group cross section

The group cross-section functionals are calculated from evaluated neutron data on the required accuracy, on the wj- width of energy groups according to the formulae:

$$G_l = \frac{\ln(1/T_l)}{w_j}\ , \qquad @G_l = \frac{T_l * @T_l}{w_j}\ , \qquad l = 0..24,$$

where: G_l - cross-section functional for l-group,
$@G_l$ - statistical errors for G_l,
w_j - width of energy group (j=1..10).

This module created the file: *xxxyyy*cs.dat, where *xxx* and *yyy* are the initial three chars in the names of the specified files with the neutron data.

Normalization of spectra

This operation is performed in the marked interval channels according to the formulae:

$$N = \frac{Ar(1)}{Ar(2)}\ , \qquad @N = N * \left(1 / \sqrt{Ar(1)} + 1 / \sqrt{Ar(2)} \right)\ ,$$

where: N - the norm in [ch1,ch2]-interval,

@N - statistical errors for N,
Ar(1) - the area under first spectrum in [ch1,ch2]-interval,
Ar(2) - the area under second spectrum in [ch1,ch2]-interval.

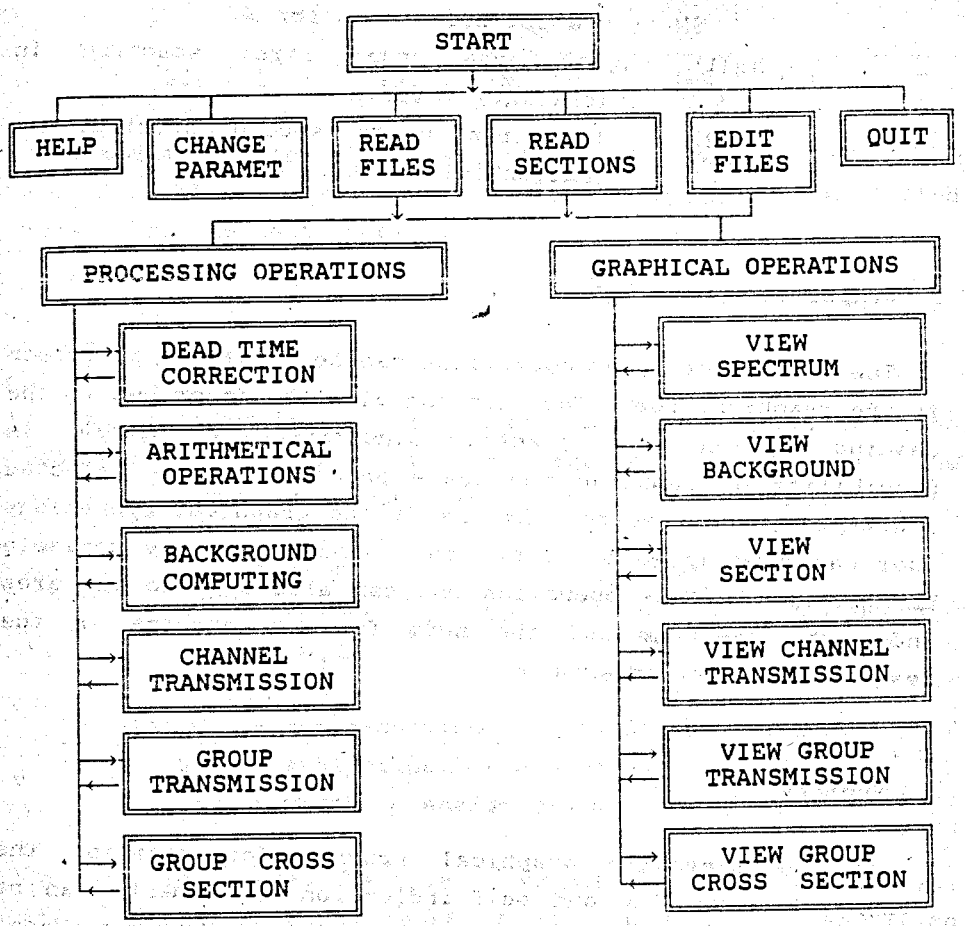
3. Graphical operations

The results of upper operations can be displayed on screen in the graphical form. The user can view the described in the reading option two spectra simultaneously. There is possibility of changing the scale and viewing the selected fragment of the spectrum. By one of the graphical operations user can mark up to 64 points for computing of the variable background. In this operation one can also compute the area under the spectrum and the norm for two spectra in the described by user intervals.

4. Summary

By the presently graphical program for modeling the neutron transmission and self-indication experiments can be efficiently evaluated neutron data files. Group-averaged functionals are calculated from evaluated neutron data. The statistical errors of functionals are calculated in the unresolved energy region and used in the fitting of average resonance parameters to the measured group-averaged cross-section.

5. Appendix - the flow diagram of the ANS



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