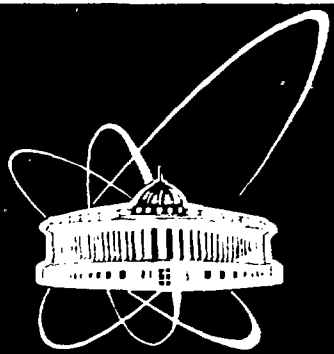

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THE DYNAMIC METHOD FOR TIME-OF-FLIGHT MEASUREMENT OF THERMAL NEUTRON SPECTRA FROM PULSED SOUPCES

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1. Introduction

The time-of-flight method is traditionally a measurement of the neutron reaction rate at different instants in time. Neutrons spreading with the most probable wavelength λ_m have the most intensive reaction rate, but the counting rate of individual interactions is limited to the dead time of the detector. When the wave length increases to the length $\lambda > \lambda_m$ the measurement time of neutron yield should increase as $(\lambda/\lambda_m)^4$. Unfortunately, there is a lack of possibilities to follow the thermal neutron spectrum, when the temperature is near 300 K, to count separate acts of interactions up to λ = 10 Å. The increase in efficiency of at least 10^4 times involves the electric current in the initial transformer that measures the thermal neutron flux yielded from the high-intensity pulsed sources based on the IBR-2 nuclear reactor [1], as well as the proton accelerators LAMPF [2], ISIS [3], IPNS etc., in a dynamic range of $\simeq 10^5$ within a time interval 0.2 – 150 ms of pulsed current. In this paper we state the dynamic method and examine the main problems connected with the electric current technique for time-of-flight measurement of the thermal neutron spectra emitted from pulsed sources. Then, we describe the construction of a system with an initial transformer based on a gas-filled ionization fission chamber and analyze the background current components some of which might create noise 10^{-2} – 10^{-4} relative to the useful signal. The results of this work are measured up to $\lambda \approx 24$ Å thermal neutron spectra emitted from water and cryogenic methane moderators set in the IBR-2 pulsed reactor.

1. Method

The time-of-flight spectrum M of thermal neutrons yielded from the moderator surface is related to the time dependence $I(t)$ of the electric current in the measuring system as,

$$I(t) = Q \cdot [(\dot{W} * G_M * M \cdot \eta_T) * G_D * G_E] + I_\gamma + I_D + I_E, \qquad (1)$$

where Q is the average charge per single act of nucleus fission in the sensitive volume of the chamber; η_T is the neutron sensitivity; W is the time dependence of the fast neutron flux of a power pulse; G_M, G_D, G_E are pulse characteristics of the moderator, initial transformer and measurement circuits, respectively; I_γ, I_D, I_E are background current components accordingly from γ-radiation, α-activity of neutron sensitive material inside the

chamber and noise of the measurement circuits, and * is the convolution operator. Let us assume W is a sum of the fast neutron flux within the power pulse W_θ and the background flux between adjacent pulses $W_{\theta b}$, and include current components due to fast neutrons I_f and neutrons from the intermediate range of the spectrum I_F (Fermi range). In this assumption expression (1) may be written as,

$$I(t) = [W_\theta * M_1] + [W_{\theta b} * M_1] + (I_f + I_F + I_\gamma + I_D) * M_2 + I_E, \qquad (2)$$

where

$$M_1 = (G_M * M \cdot \eta_T) * G_D * G_E \cdot Q; \quad M_2 = G_D * G_E.$$

The typical contribution of background components to I(t) would be :

$I_{\theta b} = [W_{\theta b} * M_1] \simeq (10^{-4} - 10^{-3}) \cdot I(t)$;
$I_\gamma \simeq I_D \simeq (10^{-6} - 10^{-5}) \cdot I(t)$;
$I_f \simeq (3 - 10^{-4}) \cdot I(t)$;
$I_E \simeq (10^{-3} - 10^{-5}) \cdot I(t)$.

The system employs a gas-filled ionization fission chamber which contains a set of parallel plate electrodes forming two neutron sensitive volumes separated by a cadmium plate [4]. Electrodes in this chamber are connected to form a differential circuit so that electric currents I_f, I_γ and I_D in one volume are subtracted from the same currents in the other volume because both volumes have a common external circuit for the signal electrode. If the chamber is irradiated by a collimated beam directed normally to the electrode plates, the current I_F in the first volume is compensated by the current from neutrons transferred through the cadmium plate to the second volume. As a result, the current flowing in the external circuit is caused by those neutrons which have an energy lower than the threshold energy of the cadmium filter and which interact with fission nuclei in the first volume. The noise component I_E does not depend on the output chamber current and has been fixed in another measurement. Thus, for the completely compensated chamber, expression (2) assumes the form

$$I(t) - I_E \simeq [W_\theta * M_1] + [W_{\Theta b} * M_1].$$

The background component of the current from thermal neutrons $I_{tb}=W_{\theta b}*M_1$ is measured in the time interval between pulses and can be extrapolated with

sufficient precision within the time interval of the pulse by means of the fast neutron flux time dependence. When I_{tb} is known, M is defined as

$$W_\theta * G_M * G_D * G_E * M = I_c(t)/(\eta_T \cdot Q), \tag{3}$$

where $I_c(t) = [I(t) - I_{tb}(t) - I_E]$ is the chamber current from the thermal neutron flux yielded from the moderator surface. Parameters that define the form of functions G_D, G_E, G_M and M may be varied within a certain interval and selected so that a duration of functions G_D, G_E, G_M is negligible compared to M. For this case M is expressed analytically as follows,

$$M(t) = I_c(t)/(\eta_T \cdot Q). \tag{4}$$

2. Thermal-neutron detection efficiency and the issue of background current compensation in the ionization chamber

The detection efficiencies ϵ and η_T are related as $\eta_T = \epsilon \cdot S$, where S is the fission coating area. For a chamber containing a set of N parallel plates fixed one after another normally to a collimated neutron beam and held at a high voltage sufficient to provide the saturation ionization current, the value η_T is defined as

$$\eta_T = S_0 \cdot \exp(-C) \cdot [1 - \exp(-a_u)] \cdot [1 - \exp(-bN)]/[\exp(b) - 1],$$

where $a_u = \Sigma_u X$; $b = a_u + a_F$; $a_F = \Sigma_F d$; Σ_u, Σ_F are the fission macroscopic cross sections of nuclei in the chamber and the capture cross sections of the plate material, respectively; and S_0 is the plate area. The coefficient C accounts for attenuation of the neutron beam by the chamber wall. The calculated specific sensitivity $\eta^* = \eta_T/S_0$ of the chamber assembly vs. λ and the N number of stainless steel plates of $d = 0.4$ mm thickness covered by layers of U_3O_8 with a thickness of 1 mg/cm^2 enriched to 90% (mass) by ^{235}U nuclei is shown in Figs. 1 and 2. Obviously, in order to accurately measure the neutron flux at λ up to 20Å, it makes little sense to increase the number of plates beyond 10 – 20 plates. The value of significance S_0 may be calculated using the maximum chamber current I_m as follows,

$$S_0 = I_m/(QF_m\eta^*),$$

where F_m is maximum flux neutron density on a detector. For the sensitive chamber volume formed with $N = 20$ plates, we have measured

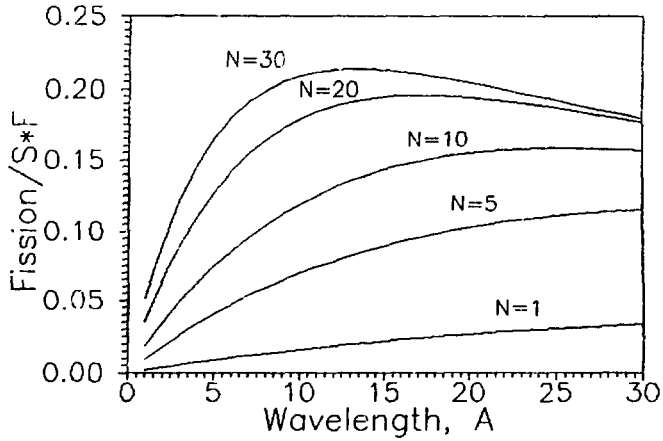

Fig.1 The estimated dependence of sensitivity, normalized on a square unit upon the neutron wavelength (for the set of N plates). S – fission coating area, F – neutron flux dencity.

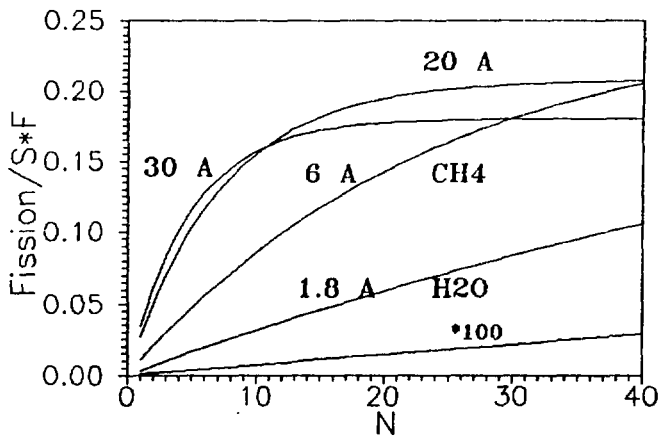

Fig. 2 The estimated dependence of sensitivity, normalized on a square unit upon number N plates in the set for thermal neutrons with wavelength 1.8 A (water moderator at temperature of 300 K), 6 A (solid methan moderator at a temperature of 20 K), 20, 30 A and fast neutrons. (*100 means the result multiplied by 100.) S – fission coating area, F – neutron flux dencity.

$\eta^* = 0.05$, $Q = 1.7 \cdot 10^{-13}$ C. When the maximum neutron flux density is $F_m = 2 \cdot 10^{10}$ s^{-1}·cm^{-2} ($\lambda = 1.8$ Å) and I_m is limited, for example to 20 mA, the significance is found to be $S_0 = 70$ cm^2. A critical compensation criterion is the balance of the background current within the sensitive volume placed in the neutron beam in front of the cadmium plate filter and the current within second sensitive volume accommodated behind the first one and the cadmium plate. Taking into account the capture of neutrons with an energy higher than the threshold capture energy of the cadmium filter and the ettenuation of γ-rays in the collimated beam, the number of fission material coated plates inside the first N and second $K > N$ sensitive volumes relate as :

for neutrons

$$\exp(bN) + \exp(-bK) = 2,$$

for γ-radiation

$$\exp(\mu dN) + \exp(-\mu dK) = 2,$$

and for α-activity of fission material inside the chamber

$$K = N.$$

Here μ is the linear attenuation coefficient for photon decrease within the material of the chamber assembly. Fig. 3 shows the correlation dependencies of the ratio K/N with N calculated on the assumptions that the background current within the first sensitive volume is equal to the current within the other volume under the incident flux of fast neutrons, that the neutrons have the cadmium filter threshold energy of 0.45 eV. As shown, the ratio K/N strongly influences the compensation coefficient and the increase in specific sensitivity of thermal neutrons by enlarging the number N involves a nonlinear increase in the number K vs. the neutron λ. This is why it is deemed impossible to provide an exact compensation for both background current components I_f and I_F. In this situation we consider it sufficient to achieve the exact compensation for only the main item I_F from neutrons with the energy close to the threshold capture energy of the cadmium plate. The compensation coefficient for the background current component from fast neutrons and neutrons with an energy above the threshold energy of the cadmium plate is close to 100. If $N < 20$ the compensation coefficient for I_α is around 10. The background current component I_γ may be efficiently reduced up to 100 times by placing additional plates without a fission coating

to the set of plates in the compensation volume.

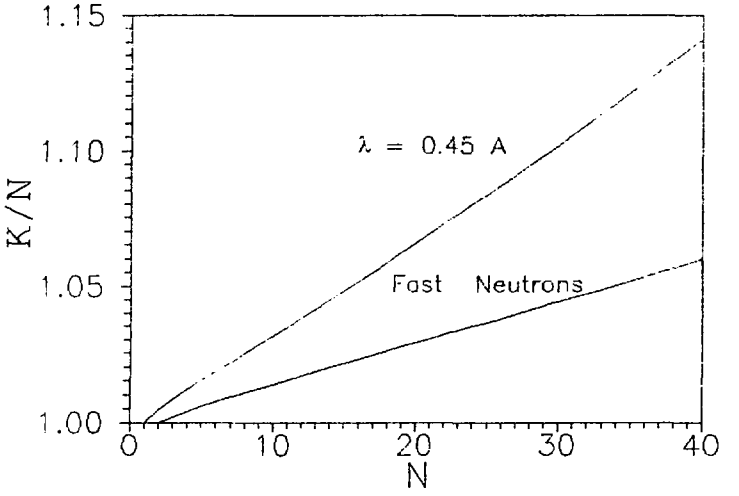

Fig.3 Estimation of a ratio of the chamber plates number (K) in a compensation volume to the number of plates in a sensitive one (N) upon a number of plates N.

3. The methodical error

The transition condition from (3) to (4) can be expressed by dispersions Di, $i = \theta, M, D, E$ of functions included in (3) and dispersion of a discrete D_t as follows :

$$t_0^2/2 >> D_\theta + D_M + D_D + D_E + D_t, \qquad (5)$$

where $t_0 = L/v_0$ is the time of flight of neutrons with the velocity $v_0 = \sqrt{2kT/m}$ of the distance L between the moderator surface and the detector; T is the temperature of the neutrons; k is Boltzmann constant and m is the neutron mass. For a fast neutron pulse with a duration of 0.1–220 mks in a periodical neutron source, the power dependence is like a Gaussian or exponential distribution. The moderator response G_M can be approximately described by an exponential distribution. Its duration depends on the moderator type and equals 50–200 mks. The ionization chamber

with parallel-plane electrodes, the pulse characteristic for positive ions in a one- velocity approximation can be represented as the sum of the triangular distributions of both positive and negative charge carriers. The dispersion value D_D is basically defined by the positive ion assembling time. The characteristic response time $\sqrt{D_D}$ of the chamber filled with an argon-based gas mixture may be about 40 mks. The value of D_E as of a low frequency filter is small and equals 4mks. In this case, the measuring channel influence on the odd error of the measurements is more than on the spectrum form. So, expression (5) can be presented as follows :

$$t_0^2/2 >> \tau_{\theta P}^2 + \tau_M^2 + a/6 \cdot T_+^2 + \Delta t^2,$$

where $\tau_{\theta P} = \theta_{1/2}/2.36$ for a pulsed reactor, $\tau_{\theta P}^2 = \tau_\theta^2 + t_p^2/12$ for a booster; $\theta_{1/2}$ is the power pulse width at half maximum; t_p is the injected neutron pulse duration; τ_θ is the first harmonic of the fast neutron flux density decay constant; τ_M is the first harmonic of thermal neutron flux density decay constant in a moderator; a=1/2; T_+ is the assembling time of positive ions in the sensitive volume of chamber, and Δt is the sampling period (30–50 mks). For example, the IBR–2 reactor must fulfill the condition $t_0 \gg 200$ mks, and for t_0 >2 ms, the thermal neutron time-of-flight spectrum is described by expression (4) with an error of $\simeq 1\%$.

4. The odd error

Let us evaluate the odd error of the ionization chamber current measurement, and the limitations caused by the value of the measurement range. The principle sources of the odd error are fluctuations in intensity of neutrons registered σ_n, neutron-sensitive chamber materials α-decay background current σ_α, measuring channel noise σ_E and analog-to-digital conversion discretization errors σ_c. Other error sources, e.g., the number of charge carrier fluctuations in the sensitive volume of the chamber for one neutron or α-particle registration, can be neglected in this case. A summary dispersion of odd noises

$$\sigma_\Sigma^2 = \sigma_n^2 + \sigma_E^2 + \sigma_\alpha^2 + \sigma_c^2$$

can be evaluated by the spectral density of each component reduced to the measuring channel input:

$$\sigma_n^2 = 2Q \int\limits_{\Delta f_D} I_c \cdot G_D^2 df, \quad \sigma_\alpha^2 = 2Q_\alpha \int\limits_{\Delta f_E} I_\alpha \cdot G_D^2 df, \quad \sigma_E^2 = \int\limits_{\Delta f_E} \nu^2 \cdot df,$$

$$\sigma_c^2 = (0.29\Delta)^2 = (0.29 \cdot I_m/2^n G_E)^2,$$

where Q_α is the average charge emitted in the detector for one α–event; I_α is the mean background current caused by α–particles; ν^2 is the specific intensity of the measuring channel fluctuation noise; Δ is the level discrete quantum; $\Delta f_D, \Delta f_E$ are the detector and the measuring channel bandwidths, respectively; n is the ADC word length; and I_m is the maximum current measured. The total dispersion of quasi–flat noises can be presented as follows:

$$\sigma_\Sigma^2 \simeq 2QI_c[\Delta f_+ + \Delta f_-] + 2Q_\alpha I_\alpha[\Delta f_+ + \Delta f_-] + \nu^2 \Delta f_E + (0.29 \cdot I_m/2^n G_E)^2,$$

where $\Delta f_+ = 1/(2\pi T_+)$, $\Delta f_- = 1/(2\pi T_-)$, and T_- is the assembling time of electrons in the sensitive volume of the chamber. The relative error can be reduced by averaging the complete signal by the number of neutron pulses, and by optimal filtration of the useful signal can be obtained by effective coherence of the filter bandwidth Δf_F with the signal. Under the condition $\Delta f_F < \Delta f_+, \Delta f_-, \Delta f_E < I_c\ /2\pi Q$, the relation

$$\sigma_\Sigma^2/I_c^2 \simeq 4\Delta f_f Q/(I_c m)[1 + Q_\alpha I_\alpha/(QI_c) + \nu^2/(QI_c)] + \\ + (0.29 I_m/G_E 2^n)^2/p I_c^2 \qquad (6)$$

is obtained. The factor p in (6) depends on the technical realization of the measuring channel and input noise level. Thus, a filter-dependent component noise dispersion decreases in approximately $(\delta f_+ + \delta f_-)m/(2\Delta f_F)$ times, and a discretization dispersion in $1 \leq p \ll m$ times. For small currents (meaning $I_c \leq 2\pi Q \Delta f_F$) the frequency filtration has practically no influence on the neutron noise dispersion, which can be reduced only by averaging the signal over a number of neutron pulses. When $Q = 1.7 \cdot 10^{-13}$ C, $Q_\alpha \simeq 10^{-14}$ C and $\Delta f_F = 20$ kHz, a threshold current value, below of which the frequency filtration is not effective, equals about 10^{-8}A for neutrons and 10^{-9} A for α–particles. Note that in gas–filled chambers 90% enriched by ^{235}U nuclei, the α–current level is usually a little higher than the threshold and equals about $6 \cdot 10^{-9}$ A per 1 g of ^{235}U. Therefore, a current value of 10^{-8}A can be taken as a lower measurement limit.

At considerably large values of $I_c > 2\pi Q \Delta f_F$ a real noise reduction, without taking into account the discretization noises for the argon–based gas mixture filled chamber, equals about 13 · m ($T_+ = 50$ mks, $T_- = 30$ ns). From here it follows that at the IBR–2 reactor pulse frequency equals

5 s^{-1} relative rms errors can be reduced by 500 times for about one hour of measurements. It is necessary to note that under the same conditions, the current technique neutron registration efficiency is at least 10^4 times higher, and the rms error at optimal filtration is 10^3 times smaller than for single neutron transaction counting. The basic error of the lower measurement range is brought by a discretization error. Practically, with signal averaging the discretization error and related dynamic measurement range is altered essentially less than in $\sqrt{p}$ times, as would follow from (6). It is because (6) does not take into account the multitude of small errors, which totality depend on the value of the dynamic range of the lower limit. Examples are: a round-off and integer-to-real conversion error, the deviation of a real noise spectrum from a flat one, and so on. Therefore, if one defines the value of the dynamic range by its lower limit as 2^n, then for 10^5 range realizations an ADC word length n should be not less than 17. Since a measuring system of this kind is difficult to realize, the following approach was used. At small values of current, a basic noise is caused by discretization error, which could be minimized if the measurement is carried out at a large gain factor in comparison with the large current range. In this case at a gain factor relation of $2^{(n-1)}$ the measurement range is defined as $2^{2\cdot(n-1)}$ and is equal to $2.6 \cdot 10^5$ for $n = 10$. The discretization error in this case does not exceed 1.5%.

5. Electronics equipment and program support.

The method described is on the basis of a measuring-computing system, which consists of a initial transformer, an electronic unit and an IBM-compatible personal computer. The KNK-21A ionization chamber was used as the primary converter. It consists of a row parallel plates 44 mm in diameter. These plates form two neutron sensitive volumes, separated by a cadmium plate with a diameter of 46.4 mm and a thickness of about 0.4 mm. Both signal electrodes are electrically connected. Plate surfaces in each volume are covered by layers of ^{235}U nuclei enriched uranium oxide with a thickness of 1 mg/cm^2 over 1000 cm^2 area. The chamber is filled with an argon-based gas mixture at a pressure of 0.3 MPa. The chamber sensitivity to thermal neutrons is $\nu_T = 0.3$ cm^2 at a temperature of 300 K, Q $= 1.7 \cdot 10^{-13}$C. The I_α and I_γ current compensation factor is about 70. The chamber with a cylindrical cadmium collimator was located in the channel behind the biological shielding at a distance of 5.55 m from the surface of a grooved water moderator, which surrounds the core of the IBR-2 reactor.

The collimator aperture angle was chosen to cover the entire surface of the moderator as viewed through the neutron channel. A compensation factor for the current caused by the fast neutron pulse is equal to 4.

The thermal neutron current signal from the chamber, synchronized with the IBR-2 control system's "Start" signal, was included in the electronics unit input. The signal is pre-amplified, filtered, and digitized by a 10-bit ADC. ADC data are transmitted to the computer memory by a DMA channel. All the measurement and acquisition parameter control and data processing is accomplished by specialized software. The user interface is presented in the form of a branched menu. The measuring program was written on Turbo Pascal and includes a set of low-level hardware controlling routines. It allows one to turn the low-frequency filter on/off, to set the optimal ADC discretization period from 30 mks to 35 ms, to choose the period and number of points for one measurement, to select one of four input signal ranges (2, 20, 200, 2000 mkA), to set the extra gain factor (from 1 up to 64) in any of them, and to define the number of measurement cycles for averaging. This program also produces a normalization of the measurement data, graphical visualization and saves the data in a disk file. All measurement parameters can be written in a configuration file, allowing the experimental conditions to be reproduced. The processing program is written in FORTRAN-77. It performs the data processing in accordance with the algorithms described in the present paper. The final result of this program operation is a calculated thermal neutron flux density distribution within wave length. The interaction between the measuring and processing programs is realized on a files exchange level.

6. Experimental results

The chamber current time dependence, averaged for approximately one hour of thermal neutron flow measurement, is shown in Fig.4. The neutrons were coming out of the water moderator at a mean reactor power of 2 MWt and a pulse frequency of 5 s^{-1}. It can be seen that the neutron current fast component in an IBR-2 power pulse in instant 2.64 ms is compensated 4 times. In the same figure is shown an I_c current time dependence as well. This is the result after subtraction of the background thermal neutron radiation and pulsed background current, which was caused by a power emission at the time the counterbalance of the main moving reflector blade passed near the core.

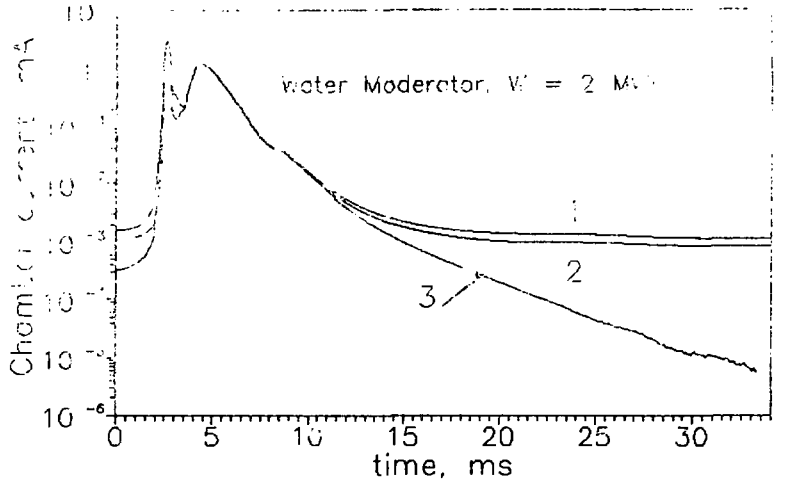

Fig.4 Detector current time dependence averaged for one hour of measurements at the IBR-2 reactor. Pulses frequency is 5 s^{-1}.

1. Sum of the section currents;
2. Section current caused by neutrons with an energy lower than the cadmium filter threshold;
3. Section current component after subtraction of the background neutron radiation between pulses.

The thermal neutron spectra measurement results for the water and cryogenic methane moderators at corresponding temperature of 300 K and 60, 32 and 13 K are shown in Fig.5. The spectrum for the water moderator is approximated by the Maxwell distribution density at $\lambda_0 = 1.77$ Å:

$$M(\lambda) = N_0 \cdot (\lambda_0^4/\lambda^5) \cdot e^{-(\lambda_0/\lambda)^2},$$

where N_0 is the factor depending on the measurement conditions; $\lambda_0 = h \cdot t_0/m$; and The neutron temperature was calculated from

$$T = m \cdot (L/\Delta t_m)^2/(4k),$$

where $\Delta\ t_m$ is the time interval between maxima caused by thermal and fast neutron pulses. The corresponding value of T is 304 K. At increasing λ values, the actual spectra measurement result is deviated from the Maxwell distribution. Mainly, this is caused by flight-base air, flux weakening in

the wall of the moderator and partial absorption in the sensitive volume of the chamber. The spectrum for the cold methane moderator greatly differs from its Maxwell estimation. In this case the main differences are related to neutron capture in the thick aluminum wall of the moderator.

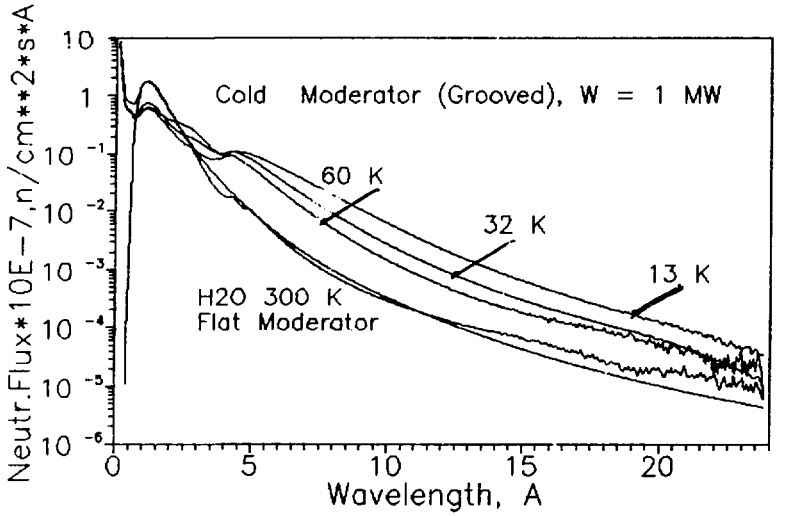

Fig. 5 The thermal neutron flux density spectrum upon λ and its Maxwell approximation for a water moderator, and the spectra for a grooved solid methane moderator at a temperature of 60, 32 and 13 K

Conclusion

Thermal neutron spectra and temperature measurements in pulsed sources were carried out successfully using an ionization chamber which operates in a current mode. In this case the statistical precision is increased, and the measurement time is shortened. It gives the ability for an experimentor to obtain experimental data which are inaccessible to traditional single neutron transaction counting techniques. The authors hope that by means of optimizing some of the ionization chamber parameters, low energy neutron registration efficiency can be increased and that the spectra measurement precision can reach up to $\lambda = 27$ Å.

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Динамический метод измерения времяпролетных спектров тепловых нейтронов в импульсных источниках

Рассмотрен высокоэффективный метод измерения спектров тепловых нейтронов по времени пролета в импульсных источниках. Метод позволяет реализовать эффективность регистрации нейтронов в 10^5 раз выше, чем в традиционном четном методе измерения. Рассмотрены также основные проблемы, связанные с техникой токового анализа при измерении времяпролетного спектра тепловых нейтронов. Представлены способы учета различного рода методических погрешностей и пути конструирования специального детектора нейтронов на основе ионизационной камеры деления. Приведены некоторые экспериментальные результаты по измерению спектров эмиссии тепловых нейтронов с поверхности водяного и криогенного метанового замедлителей импульсного реактора ИБР-2.

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The Dynamic Method for Time-of-Flight Measurement of Thermal Neutron Spectra from Pulsed Sources

The time-of-flight method for a measurement of thermal neutron spectra in the pulsed neutron sources with high efficiency of neutron registration, more than 10^5 times higher in comparison with traditional one, is discribed. The main problems connected with the electric current technique for time-of-flight spectra measurement are examined. The methodical errors, problems of a special neutron detector design and other questions are discussed. Some experimental results, spectra from surfaces of the water and solid methan moderators, obtained in the pilsed reactor IBR-2 (Dubna, Russia) are presented.

The investigation has been performed at the Frank Laboratory of Neutron Physics, JINR.

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