PHYSICAL DEVICES FOR ECOLOGY, MEDICINE, BIOLOGY

MONITORING THE FAST NEUTRON FLUENCE

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Abstract. The experimental technique and methods of processing experimental data during monitoring of controlled neutron fluence with energy greater than 0.1 MeV are described. Approximation of the fast neutron fluence value based on experimental values of neutron fluence with energies greater than 1 and 3 MeV, directly measured by Nb and Fe monitors, is necessary forassessment of radiation damage to materials to justify the safe operation of nuclear reactors. The reliability of determining the values of fast neutron fluence at controlled points of the irradiation device is confirmed by a comparative analysis of calculated and experimental data.

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1. INTRODUCTION

The article presents an analytical description of the experimental technique and methods of processing experimental data when monitoring the controlled fluence of fast neutrons. Determination of neutron-physical characteristics is an important aspect in planning and conducting experiments at research reactors. Structural materials are subject to radiation damage when irradiated. Of greatest interest is the experimentally obtained information on the fluence of fast neutrons used to assess the radiation damage of materials to justify the safe operation of nuclear reactors.

The experimental technique is based on the neutron activation method [1, 2], which most fully satisfies the specific measurement conditions (wide range of energy and intensity of neutron radiation, high radiation levels, elevated temperatures, small dimensions and inaccessibility of measurement sites, and much more). JSC SRC RIAR has developed a methodology and

implemented technical means for neutron dosimetric supportreactor experiments in a wide range of irradiation conditions at a fast neutron flux density of up to 5 1015 $cm^{-2}\cdot With^{-1}$with a relative expanded uncertainty of no worse than 11.5% [3].

Experiments on reactor testing of materials are accompanied by periodic intermediate inspections of the irradiation conditions in the reactor. The program of neutron-dosimetric support of reactor tests provides for a regulation that includes monitoring of neutron fluence based on the readings of fluence monitors [4, 5]. Setneutron activation detectors (DND) is exposed to simultaneous irradiation at a controlled point. Predictors of the value of the controlled fluence of neutrons with an energy greater than 0.1 MeV are the threshold values of the fluence of neutrons with energies greater than 1 and 3 MeV, directly measured by Nb- and Fe-monitors taking into account the effective cross-sections, respectively.

The aim of the article is to describe the technique of an experiment for monitoring the fluence of fast neutrons with energies greater than 0.1 MeV.

2. NEUTRON-DOSIMETRIC SUPPORT OF REACTOR EXPERIMENTS

The detector accumulates information about the fast neutron flux during the entire irradiation time, typically over several reactor campaigns. The mechanism of neutron detection in the NAD substance is based on an indirect method, which refers to the passive method of neutron analysis. The neutron detection process begins when they interact with the NAD nuclei based on nuclear reactions, accompanied by the formation of nuclide-monitor products, in the form of gamma quanta and X-rays, which are registered by the detector.

Materials from iron and niobium most fully meet the requirements for DNA activation in monitoring fast neutron fluence. The condition for conducting such nuclear reactions, like $^{93}\mathrm{Nb}(\mathrm{n},\mathrm{n}')^{93\mathrm{m}}\mathrm{Nb}$ – on a niobium monitor and $^{54}\mathrm{Fe}(\mathrm{n},\mathrm{p})^{54}\mathrm{Mn}$ – on an iron monitor, is the presence of an energy threshold, while a nuclear reaction such as $^{58}\mathrm{Fe}(\mathrm{n},\gamma)^{59}\mathrm{Fe}$ occurs with thermal neutrons. The registration mechanism is based on neutron capture by $^{93}\mathrm{Nb}$ nuclei with the formation of $^{93\mathrm{m}}\mathrm{Nb}$, which decays back to $^{93}\mathrm{Nb}+\gamma$ with a period of about 16.13 years. For the second monitor, the registration mechanism is based on neutron capture by $^{54}\mathrm{Fe}$ nuclei with the formation of $^{54}\mathrm{Mn}$, which is characterized by positron activity and decays to $^{54}\mathrm{Cr}+\mathrm{e}^{+}$ with a period of approximately 312.05 days.

3 TECHNOLOGY AND TOOLS OF THE METHOD

The method for determining the controlled neutron fluence with energy greater than 0.1 MeV is based on parameterization of the spectral function with the selection of threshold values of neutron fluences with energies greater than 1 and 3 MeV as predictors and is divided into two parts:

– the experimental part of the work includes activation of detectors, measurement of absolute activities of nuclear reaction products, and obtaining activation integrals;

– the calculation part represents the application of mathematical and computational methods for obtaining information about spectral coefficients from activation integrals.

3.1 Experimental part

The DNA activation method has an advantage over other methods in sensitivity to neutron field parameters in a wide energy range (from thermal neutrons to neutrons with energy of 20 MeV), which makes it possible to obtain detailed information about the spectral function and spectral coefficients of the neutron field [6] (shown below in section 3.2).

3.1.1 Preparation for the experiment. The experiment to determine the neutron flux density begins with planning according to the structure of the technological process as shown in Fig. 1, including the selection of composition and size of samples during the development of activation detectors, which must meet the following requirements.

1. The half-life of activation products should be comparable to the irradiation time.
2. After irradiation, the activity of monitors should be sufficient for registration by the spectrometric system, but should not exceed the permissible radiation safety standards at the workplace.

Fig. 1 Structure of the technological process of neutron-dosimetric support for reactor experiments

When installed for one or several campaigns in a research reactor or for one campaign in a power reactor, the most effective fluence monitors are made of iron-54 and niobium-93 (Tables 1, 2). When using DNA with a thickness of about 0.1 mm and a diameter of about 1 mm, the induced activity will meet the requirements described above.

Table 1. Nuclear-physical characteristics of detector materials

Detector material	Type of nuclear reaction	Effective reaction cross-section σ^{eff}, mb	Neutron threshold energy, MeV	Mass content, %
Iron	$^{54}Fe(n,p)^{54}Mn$	348	3	5.8(^{54}Fe)
Niobium	$^{93}Nb(n,n')^{93m}Nb$	209	1	100(^{93}Nb)

Table 2. Characteristics of activation reaction products [7]

Radionuclide	$T_{1/2}$, days	Decay constant □, 1/c	Absolute radiation intensity □, %	Radiation energy, keV
^{93m}Nb	$(5.73 \pm 0.22) \cdot 10^3$	$1.36 \cdot 10^{-9}$	$I_{K□} = 9.33$	$E_{K□} = 16.6$
^{54}Mn	312.3 ± 0.26	$2.57 \cdot 10^{-8}$	$I_{□} = 99.98$	$E_{\gamma} = 834.8$
^{59}Fe	44.5 ± 0.01	$1.80 \cdot 10^{-7}$	$I_{□} = 56.5$	$E_{\gamma} = 1099.2$
^{182}Ta	114.74 ± 0.12	$6.99 \cdot 10^{-8}$	$I_{□} = 35.24$	$E_{\gamma} = 1121.3$

At the DNA preparation stage, they are weighed and hermetically encapsulated in quartz, which provides thermal stability. The typical monitor mass is 0.3-0.9 mg, therefore it is recommended to use scales with accuracy class not worse than Sartorius S4 Supermicro with a maximum permissible error of ±8 μg.

3.1.2 Preparation for DNA activity measurements. To determine DNA activity, spectrometric systems are used - complexes consisting of a detection unit, analyzer, and personal computer for data visualization and processing. The directly measured value is the number of photon radiation pulses during the exposure time. To convert the number of pulses into radioisotope activity, it is necessary to perform calibration by energy and radiation detection efficiency.

For calibration, sets of standard spectrometric sources with well-identifiable full absorption peaks (FAP) in the spectrum are used. The calibration establishes the relationship between channel number and energy, as well as integral nonlinearity. For the Eco-PAK-02-3 spectrometer, the relationship between channel number and energy is linear, with an integral nonlinearity value of $1.24 \cdot 10^{-2}$ % at a maximum allowable value of 0.04%.

Efficiency calibration [1]is performed to determine the absolute activity value of the DNA monitor nuclide. During calibration, the number of pulses in the FAP of each radioisotope from the set of standard spectrometric sources is measured. The detection efficiency is determined by formula (1).

$$\varepsilon_i = \frac{N_{\text{ср}}}{\tau A_0 \exp(-\lambda t)\eta}, \qquad (1)$$

where ε_i [pulse/photon] is the gamma quantum detection efficiency by the detector for a given energy value, A_0 [Bq] is the certified value of external radiation of this energy into a 4π angle, λ $[1/\text{c}]$ is the decay constant of the monitor nuclide - reaction product, according to the data in Table 2; t [days] is the time elapsed since the source certification, τ [s] is the spectrum acquisition time ("live time" of the analyzer), η [rel.units] is the quantum yield for the corresponding gamma quantum, $N_{\text{ср}}$ [pulses] is the average number of pulses registered in the FAP of the corresponding energy, based on the results of n measurements.

The formula of the analytical dependence for approximating the quantum detection efficiency $\varepsilon(E)$ in the FAP is accepted in the form

$$\varepsilon(E) = e^{P_n(\ln E)},$$

where

$$P_n(\ln E) = \sum_{i=0}^{5} p_i (\ln E)^i$$

– orthogonal polynomial of degree $n = 5$, $\varepsilon(E)$ [pulse/photon] is the gamma quantum detection efficiency by the detector for a given energy value, p_i – i -th approximation coefficient at $i = \overline{0 \ldots 5}$, E [кэВ] is the γ-quantum energy.

Considering that irradiated DNA may have different activities, efficiency calibration is performed in various measurement geometries $L - R$, providing the nominal distance from the source to the upper wall of the cryostat $R[\text{см}] \in [5; 40]$. Due to the small size of the source (typical DNA size: thickness about 0.1 mm and diameter about 1 mm) when measuring in geometry $L - 10$ (98.5 mm from the detector cryostat), no correction for source geometry is introduced – the source is considered point-like.

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[1] Efficiency is the ratio of the number of registered pulses to the number of emitted pulses. This relationship is not linear.

An important stage preceding the measurement of induced DNA activity is the cooling down of the irradiation device. This process allows reducing the activity of short-lived radioisotopes from the capsule surface to decrease the dose load on personnel, as well as to eliminate unwanted lines in the γ -spectrum from the irradiated monitor. This process is especially important for niobium monitors: in addition to the target radioisotope ^{93m}Nb, a number of other radioisotopes are formed, which have peaks in the spectrum with similar or identical energies of emitted photons (^{95m}Nb and ^{92m}Nb). Furthermore, the isotope ^{95}Nb creates a significant dose load on personnel and equipment, and also increases the dead time (signal processing time) of the spectrometric path. Characteristics of activation products of stable niobium are presented in Table 3 [8].

Table 3. Characteristics of activation products ^{93}Nb

Radionuclide	$T_{1/2}$, days	Radiation energy E_γ , keV	Absolute radiation intensity I_γ , %	Reaction	Neutron energy, MeV; Cross-section, barn
^{93m}Nb	(5.89 ± 0.05)·10 ^{3}	X $_\alpha$=16.48–16.58 X $_\beta$=18.57–18.94	9.33 ± 0.98 1.79 ± 0.18	^{93}Nb(n, n')	Fast, E >1.0
^{94}Nb	(7.41 ± 0.58)·10 ^{6}	702.65 ± 0.06 871.09 ± 0.02	99.82 ± 0.002 99.89 ± 0.001	^{93}Nb(n,γ) ^{94}Nb	Thermal, σ = 1.15
^{92m}Nb	10.15 ± 0.02	912.60 ± 0.20 934.44 ± 0.10 X $_\alpha$=15.65–15.74 X $_\beta$=17.62–17.96	1.78 ± 0.10 99.07 ± 0.04 54.41 ± 5.50 9.68 ± 0.98	^{93}Nb(n,2n) ^{92m}Nb	Fast, E >1.0
^{95}Nb	34.99 ± 0.01	765.8 ± 0.01	99.81 ± 0.01	^{94}Nb(n,γ) ^{95}Nb	Thermal, σ = 15
^{95m}Nb	3.61 ± 0.03	235.69 ± 0.02 X $_\alpha$=16.48–16.58 X $_\beta$=18.57–18.94	24.83 ± 0.77 35.38 ± 3.62 6.49 ± 0.66	^{94}Nb(n,γ) ^{95m}Nb	Thermal, σ = 0.6

Thus, before starting activity measurements, it is necessary to hold for 10 half-lives of the most significant isotopes ^{95m}Nb and ^{92m}Nb, which is about 3 months.

The relative uncertainty of activity values, according to the certified methodology, is 6%. The relative uncertainty of neutron fluence values is 9.3% [9].

3.1.3 Measurement of DND activity. After irradiation and holding at the measurement site, the capsules are opened, unless the irradiation mode is high-temperature. Under high-temperature conditions, measurements of ^{54}Mn activity are carried out in capsules [10, 11] to account for radiation heating and sublimation. Identification of extracted fluence monitors after opening the capsules is performed by gamma rays (^{54}Mn, ^{60}Co, etc.).

For Nb-DNA, there are also specifics of activity measurement related to the neutron-physical characteristics of reactor irradiation, which affect the production of impurity tantalum on the niobium monitor. Tantalum is a metal whose traces are inseparably present in niobium, thus requiring a correction for tantalum activity. The stable isotope ^{181}Ta, interacting with thermal neutrons, forms radioactive ^{182}Ta, which can cause fluorescence of stable ^{93}Nb. This process emits electromagnetic radiation quanta with energies of 16.6 and 18.6 keV, i.e., the same quanta that are emitted by ^{93m}Nb of interest to us. To exclude the influence of this fluorescence, a correction factor is introduced, taking into account the measured absolute activity of ^{182}Ta, increased by K_{Ta} , according to the data in Table 4.

Table 4. Correction factor K_{Ta} for fluorescence of ^{93}Nb

Reactor	K_{Ta}
SM-3	2.36 ± 0.20
RBT-6	2.33 ± 0.18
BOR-60	negligibly small

Measurement of the activity of irradiated monitors is recorded by pulses in the SSS with decoding of the instrumental spectrum and is provided by a set of technical characteristics that are not worse than the parameters of gamma-spectrometric installations Eco-PAK-02-3 with a coaxial HPGe detector [2]model GCD-25185 [3]and with a multichannel analyzer BOSON from Baltic

[2]HPGe – High Purity Germanium crystal that does not require liquid nitrogen for storage.

[3]Solid-state detector of coaxial configuration, providing a radial electric field with an axial component for charge collection, which affects the speed, and with a large sensitive volume, ensuring high detection efficiency for high-energy gamma rays.

Scientific Instruments Ltd for gamma sources and ORTEC GMX10P4-70 [4]with a beryllium window and with an ORTEC DSPEC 50 analyzer for X-ray sources.

In gamma spectrometry, the multichannel analyzer from the detector forms output pulses whose amplitudes are proportional to the energy lost by photons in the detecting medium. The measuring system includes a method for analyzing all input pulses by amplitude and presenting their spectrum.

The end result of multichannel analysis is a histogram (spectrum) of registered output pulses, sorted by amplitude. The pulse amplitude spectrum is a direct reflection of the energy spectrum of gamma-ray interactions in the detecting medium and constitutes the spectrometric information used in determining the DMA activity.

When a gamma quantum enters the detecting medium, it transfers part or all of its energy to an atomic electron, releasing the electron from its atomic shell. Usually, this freed electron then transfers its kinetic energy in a series of collisions in the detecting medium to other atomic electrons. The amount of energy required to form an electron-ion pair in the detecting medium determines the total charge that forms the output pulse.

Various programs are used for spectrum processing: for example, ASW-2 is used for Eco-PAK-02-3, and Gamma Vision for the Ortec spectrometer. The functionality of these programs includes spectrum processing, efficiency and energy calibrations, and scripting for measurement series. According to the data in Fig. 2, the Gamma Vision program provides information about peak areas, background values of these peaks, as well as energy resolution: for energies of 16.6 keV and 18.6 keV, it is 0.67 keV and 0.69 keV respectively.

Fig. 2. Spectrum processing ^{93m}Nb in GvPlot (Gamma Vision) program

Fig. 3 and 4 show examples of processing FHP (full energy peaks) with energies of 835 keV (energy resolution is 1.44 keV) and 1099 keV respectively of isotopes ^{54}Mn and ^{59}Fe using the ASW-2 program.

Fig. 3. Operation mode with ^{54}Mn peak in ASW-2 program

Fig. 4. Operating mode with the peak ^{59}Fe in the ASW-2 program

[4]Planar detector consisting of a crystal with a circular cross-section and a sensitive layer thickness of 10 mm. The electric field is perpendicular to the plane of the crystal's cross-section. The crystal thickness is selected taking into account measurements of low energies in the X-ray spectrum range.

The absolute activity of the monitor nuclide, which is a product of the nuclear reaction on the neutron-sensitive nuclide of DNA, is calculated using the formula

$$A = \frac{S_\gamma \lambda e^{\lambda t_{\text{выд}}}}{\varepsilon(E)\eta\left(1-e^{-\lambda\tau}\right)}, \quad (2)$$

where A [Bq] - activity of the indicator at the end of irradiation, E [eV] - energy of γ -quanta according to Table 2 (column 5), η [rel.units] - quantum yield with energy E according to Table 2 (column 4), ε(E) [count/quantum] - energy dependence of the registration efficiency of γ -quanta, $t_{\text{выд}}$ [c] - time from the end of irradiation to the beginning of measurement, τ [s] - exposure duration, λ $[1/\text{c}]$- decay constant of the monitor nuclide - product of the reaction according to Table 2 (column 3), S_γ [pulse/photon] - photopeak area with corrections for decay during measurement time and geometric factors:

$$S_\gamma = S_{\text{фп}} k_{\text{сп}} k_{\text{сумм}} k_{\text{геом}} k\,, \quad (3)$$

where $S_{\text{фп}}$ [имп./фотон] - measured photopeak area minus background.

To introduce corrections in (3), the self-absorption coefficient is calculated using the formula $k_{\text{сп}} = \frac{\mu_\gamma x \ln(\cos\alpha_0)}{\left(1-e^{-\frac{\mu_\gamma x \cdot \ln(\cos\alpha_0)}{\cos\alpha_0 - 1}}\right)(\cos\alpha_0 - 1)}$,

where $k_{\text{сп}}$ - self-absorption coefficient, x [g/cm ^{2}] - mass thickness of the detector, μ_γ [cm ^{2} /g] - mass attenuation coefficient of γ -radiation by the detector material,

$$\cos\alpha_0 = \frac{H_{\text{эфф}}}{\sqrt{\frac{D^2}{4} + H_{\text{эфф}}^2}}, \quad H_{\text{эфф}} = h_0 + h\,,$$

h_0 - distance from the spectrometer surface to the effective registration center, h - distance from the source to the crystal surface of the spectrometer, D - diameter of the detector crystal, cm. Usually $\mu_\gamma x \le 0.1$.

If γ -quanta are emitted in a cascade, there is a probability of registering them in the sum of amplitudes created by each quantum. This leads to leakage of pulses from the FEP (Full Energy Peak). When registering γ -quantum with energy E_1 , emitted in a cascade with γ -quantum of energy E_2 , the correction for summation has the form

$$k_{\text{сумм}} = \left(1 - p_\gamma(E_2)\bar{W}\frac{\varepsilon(E_2)}{\Re(E_2)}\right)^{-1},$$

where $p_\gamma(E_2)$ is the fraction of γ -quanta with energy E_2 per decay, $\square(E_2)$ is the ratio of the number of pulses in the FEP to all registered pulses in the spectrum, $\bar{W}$ is the build-up factor of γ -radiation by the detector material, accounting for the angular correlation of γ -quanta ($\bar{W} = 1$ for characteristic radiation, $\bar{W} = 2$ for annihilation radiation). For relatively large distances $k_{\text{сумм}} = 1$.

To bring all measured activation indicators of one assembly to the irradiation conditions at a single point in space, where the main indicator was placed, a geometric correction is introduced for the i -th indicator:

$$k_{\text{геом}\,i} = \frac{S_0}{S_i},$$

where S_i is the pulse count rate in the FEP of the energy of measured photons near the i -th indicator and the main indicator, normalized per nucleus. One of the threshold indicators should be chosen as the main indicator (for example, ^{54}Mn).

The correction is automatically taken into account if the recording equipment has a "live" time mode , k – correction for the spectrometer's "dead" time:

$$k = \frac{1}{1 - S_{\text{фп}} \cdot \frac{\theta}{\tau}},$$

where θ [μs] is the spectrometer's "dead" time, τ [μs] is the measurement duration.

Monitor reaction rate q_i per target nuclide nucleus is calculated based on the results of $n \square 1$ activity measurements of the monitor nuclide DNA:

$$q_i = \frac{\frac{1}{n}\sum_{j=1}^{n} A_j}{N\lambda} C_{\lambda,P}, \quad (4)$$

where A_j [Bq]– indicator activity at the end of irradiation for i -th measurement, N – number of target nuclide nuclei in the monitor (according to the material certificate for DNA sets), λ [1/s] – decay constant of the monitor nuclide – reaction product according to Table 2 (column 3), $C_{\square,P}$ – correction for reactor power history and local power, which accounts for formation and decay of activation reaction product during changes in neutron flux density during detector irradiation time,

$$C_{\lambda,P} = \frac{\int_0^\tau P(t)dt}{\int_0^\tau P(t)e^{-\lambda(\tau - t)}dt},$$

$P(t)$ [МВт] – reactor power history in chronological order from calendar date and time of day (chronological dependence of reactor power on time),.

Integral neutron flux density is determined by the formula

$$\varphi_{E_{эфф}} = \frac{R}{\sigma_{эфф}},$$

where $E_{эфф}$ – effective activation reaction threshold, R – activation reaction rate, $\sigma_{эфф}$ – effective cross-section of neutron interaction with target nuclei of target nuclides according to Table 1 data.

3.2 Calculation Part

Experimental value of activation reaction rate:

$$\mathcal{R}_i = \frac{q_i}{\tau_{эфф}},$$

where q_i [rel.units] – threshold activation reaction rate of DNA per target nuclide nucleus according to (4), $\tau_{эфф}$ [c] – effective irradiation time, determined by the formula

$$\tau_{эфф} = \frac{\int_0^{\tau} P(t)\mathrm{d}t}{P_{ном}},$$

where $P(t)$ – history of reactor power changes during irradiation time, P_{nom} – declared nominal reactor power level, MW; τ [s] – calendar duration of irradiation.

The calculated value of the activation reaction rate R_i [s^{-1}]:

$$R_i = \int_{E_{пор}^i}^{\infty} \varphi(E)\sigma_i(E)\mathrm{d}E,$$

where $i = 1, 2, \ldots, n$ is the threshold reaction index, $\sigma_i(E)$ [bn] is the energy dependence of the cross-section i -th threshold reaction, $\varphi(E)$ [$s^{-1} \cdot cm^{-2} \cdot eV^{-1}$] is the differential neutron flux of energy E , determined by the formula

$$\varphi(E) = {}^{*}\varphi f(E),$$

${}^{*}\varphi$ [$cm^{-2} \cdot s^{-1}$] is a scale factor, E [эВ] is neutron energy, $E_{пор}^i$ [eV] is the energy threshold for neutrons interacting with target nuclide nuclei for LDA by threshold activation reaction.

The analytical spectral function at a specific point in the reactor is represented by a set of mathematical expressions with four parameters [12, 13]:

$$f(E) = \begin{cases} P/E^{(1+\alpha)} & E_m < E_f \\ P\ F\ {E_m}^{(0.5+\gamma)}\ \mathrm{e}^{-(E_m/\mathrm{T})} \cdot 10^{-6} & E_f \le E_m \end{cases}, \quad (5)$$

where $f(E)$ [MeV $^{-1}$] is the differential spectral function, E [eV] and E_m [MeV] are neutron energies ($E = E_m \cdot 10^{-6}$) , E_f[MeV] is the fission energy of the neutron source, determined by the formula

$$E_f = 0.63\ (0.5 + \gamma)\ ,$$

$T = 1.290$MeV is the Maxwell distribution parameter for ^{235} U, P is the power coefficient corresponding to the order of the scale factor $^{*}\varphi$ [rel.units], characterizes the value of irradiation power, □ [rel.units] is the correction factor for deviation from the Fermi spectrum, F [rel.units] is the proportionality coefficient for the fast spectrum, γ [rel.units] is the coefficient of deviation of the fast spectrum from the Maxwell fission spectrum that determines the dependence on T in the form

$$\gamma = \frac{T}{1.26} - 0.5.$$

Parameters for restoring spectral coefficients are strictly determined from activation integrals according to the block diagram in Fig. 5 [14].

The spectral function at a specific point in the reactor is represented by a set of mathematical expressions (5). For the calculated determination of spectral parameters $\{P, \alpha, F, \gamma\}$ a target function of the form

$$S(\mathbf{x}) = \sum_{i=1}^{M}(1 - \mathrm{R}_i\ (\mathbf{x})/\mathcal{R}_i)^2 \text{ is used,}$$

where $M = 3$ – the number of dosimetric reactions, A_i – the reaction rate of the i -th detector, measured by neutron activation analysis, $\mathbf{x}$ – the parameter vector $\mathbf{x} = [x_1, x_2, x_3, x_4]^T$ is dynamically recalculated into physical parameters of the spectral function approximation according to the following set of formulas:

$$P = 10^7 \cdot \frac{1}{1+e^{-x_1}}\ ,\ \alpha = 0.6 \cdot \frac{1}{1+e^{-x_2}}\ ,\quad F = 2 \cdot 10^6 \cdot \frac{1}{1+e^{-x_3}}\ , \gamma = 0.2 \cdot \frac{1}{1+e^{-x_4}},$$

n =4 – the number of parameters to be determined, $R_i(\mathbf{x})$ – theoretically calculated reaction rate of the i -th detector based on analytical approximation (12) in the energy
range 2.15·10 $^{-1}$ –1.05·10 $^{-7}$ eV according to the formula

$$\mathrm{R}_i(\mathbf{x})\ ,$$

$L = 30$ – the number of all energy groups, σ_{ij} [бн] – the average activation cross-section of the i -th detector in the j -th energy group according to the nuclear data library, ΔE_j – the interval of the j -th energy group, determined by the formula

$$\Delta E_j = E_{j+1} - E_j.$$

The set of parameters $P, \alpha, F, \gamma$are determined algorithmically [12] in accordance with the block diagram in Fig. 5. The spectral coefficients of the neutron field $g_{1\to0,1}$ and $g_{3\to0,1}$ are determined by the formulas

$$g_{1\to0,1} = \varphi_{0,1}/\varphi_1\ ,\ g_{3\to0,1} = \varphi_{0,1}/\varphi_3,$$

where φ_1 [см$^{-2}$·с$^{-1}$] – the neutron flux density with energy greater than $E = 1$MeV,

$$\varphi_1 = \int_1^\infty f(E)dE\ ,$$

φ_3 [см$^{-2}$·с$^{-1}$] – the neutron flux density with energy greater than $E = 3$MeV,

$$\varphi_3 = \int_3^\infty f(E)dE\ ,$$

$\varphi_{0,1}$ – the neutron flux density with energy greater than $E = 0.1$MeV, determined by the formula

$$\varphi_{0,1} = \int_{0,1}^\infty f(E)dE\ .$$

Fig. 5. Block diagram of spectral function parameterization by optimization method

After determining the two values $\varphi_{0,1}$ the result is generalized as a weighted average:

$$\varphi_{0,1} = \frac{\omega_1 g_{1\to0,1}\cdot\varphi_1^э + \omega_2 g_{3\to0,1}\cdot\varphi_3^э}{\omega_1+\omega_2},$$

where ω_1 , ω_2 – weight coefficients [9, 10], determined by the formulas

$$\omega_1 = 1/\theta^2(\varphi_1),\quad \omega_2 = 1/\theta^2(\varphi_3),$$

where $\theta(\varphi_i)$ – error in determining the neutron flux density φ_E with energy greater than 1 MeV and 3 MeV respectively , $\varphi_1^э$ and $\varphi_3^э$ – experimental values of neutron flux density with energy greater than 1 MeV and 3 MeV respectively, determined by the formulas

$$\varphi_1^э = \frac{\mathcal{R}_1}{\sigma_1^э},\ \varphi_3^э = \frac{\mathcal{R}_2}{\sigma_2^э}.$$

The neutron fluence with energy greater than 0.1 MeV is determined by the formula

$$\mathrm{Ft}_{0.1}\ [\text{см}^{-2}]\ .$$

4. CONCLUSIONS

The experimental technique and gamma-spectrometric equipment, methods of analysis and processing of experimental data for monitoring the fast neutron fluence with energy greater than 0.1 MeV are described.

Annually, sets of LDN from iron and niobium are successfully used in monitoring the fast neutron fluence for key assessment of radiation damage to materials to ensure safe operation of

nuclear reactors at JSC "SSC RIAR". The reliability of determining the values of fast neutron fluence with energy greater than 0.1 MeV at monitored points of the irradiation device is confirmed by comparative analysis of calculated and experimental data.

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FIGURE CAPTIONS

Fig. 1. Structure of the technological process for neutron-dosimetric support of a reactor experiment.

Fig. 2. Processing of the ^{93m}Nb spectrum in the GvPlot program (Gamma Vision).

Fig. 3. Working mode with the ^{54}Mn peak in the ASW-2 program.

Fig. 4. Working mode with the ^{59}Fe peak in the ASW-2 program.

Fig. 5. Block diagram of the spectral function parameterization by optimization method.

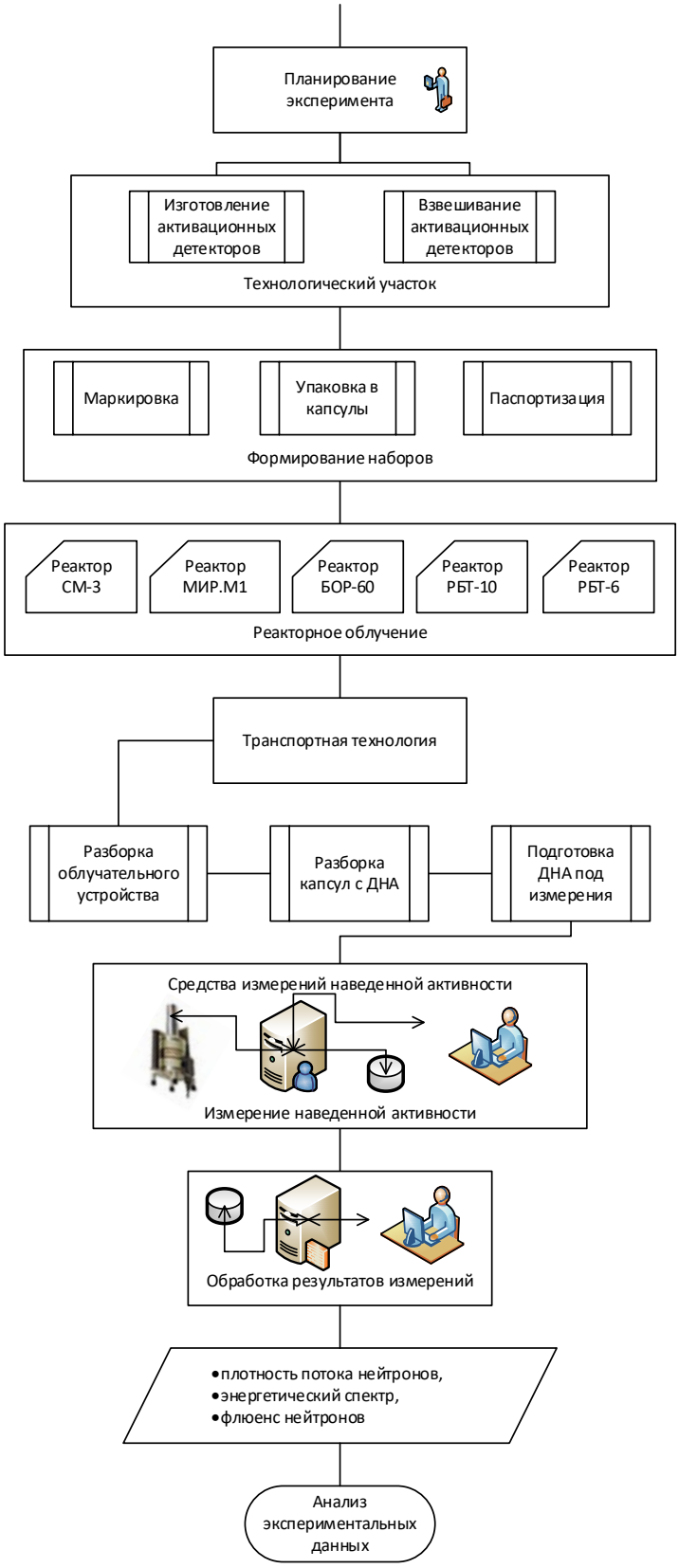
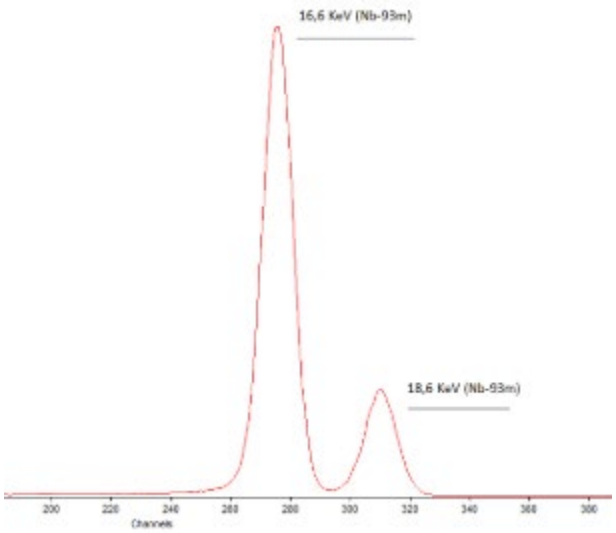

Планирование эксперимента
Изготовление активационных детекторов
Взвешивание активационных детекторов
Технологический участок
Маркировка
Упаковка в капсулы
Паспортизация
Формирование наборов
Реактор СМ-3
Реактор МИР.М1
Реактор БОР-60
Реактор РБТ-10
Реактор РБТ-6
Реакторное облучение
Транспортная технология
Разборка облучательного устройства
Разборка капсул с ДНА
Подготовка ДНА под измерения
Средства измерений наведенной активности
Измерение наведенной активности
Обработка результатов измерений
•плотность потока нейтронов,
•энергетический спектр,
•флюенс нейтронов
Анализ экспериментальных данных

Fig. 1



Analysis Results Table

Energy	FWHM (keV)	Area	Background
5,86	0,58	5327	10140
7,14	0,59	31816	8209
9,05	0,67	5151	4413
16,59	0,67	314096	15332
18,58	0,69	71237	7987
22,16	0,83	289	1135
32,19	0,84	386	384

Fig. 2.

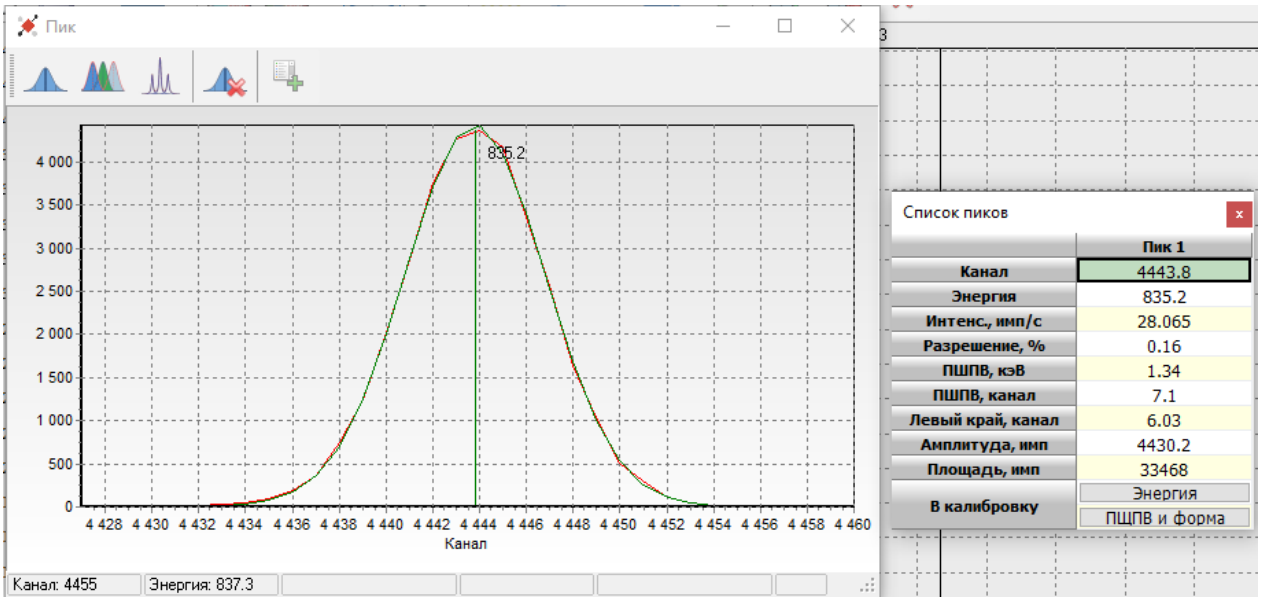

Пик
835.2
4 000
3 500
3 000
2 500
2 000
1 500
1 000
500
0
4 428 4 430 4 432 4 434 4 436 4 438 4 440 4 442 4 444 4 446 4 448 4 450 4 452 4 454 4 456 4 458 4 460
Канал
Канал: 4455
Энергия: 837.3
Список пиков
Пик 1
Канал 4443.8
Энергия 835.2
Интенс., имп/с 28.065
Разрешение, % 0.16
ПШПВ, кэВ 1.34
ПШПВ, канал 7.1
Левый край, канал 6.03
Амплитуда, имп 4430.2
Площадь, имп 33468
В калибровку
Энергия
ПЩПВ и форма

Fig. 3.

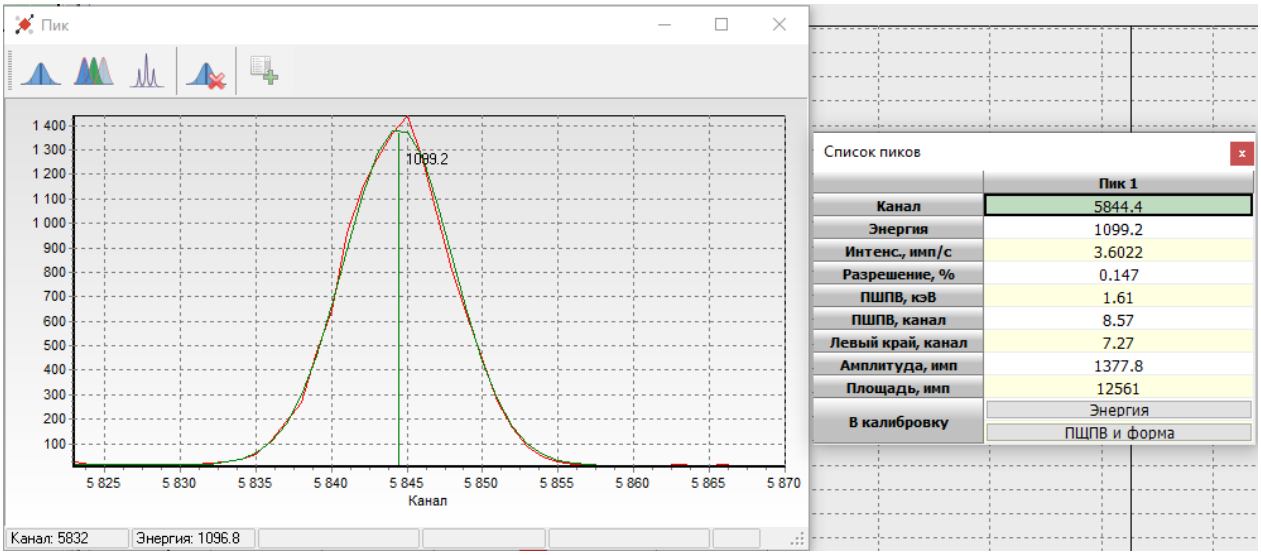

Пик
1099.2
1 400
1 300
1 200
1 100
1 000
900
800
700
600
500
400
300
200
100
5 825 5 830 5 835 5 840 5 845 5 850 5 855 5 860 5 865 5 870
Канал
Канал: 5832
Энергия: 1096.8
Список пиков
Пик 1
Канал 5844.4
Энергия 1099.2
Интенс., имп/с 3.6022
Разрешение, % 0.147
ПШПВ, кэВ 1.61
ПШПВ, канал 8.57
Левый край, канал 7.27
Амплитуда, имп 1377.8
Площадь, имп 12561
В калибровку
Энергия
ПЩПВ и форма

Fig. 4.

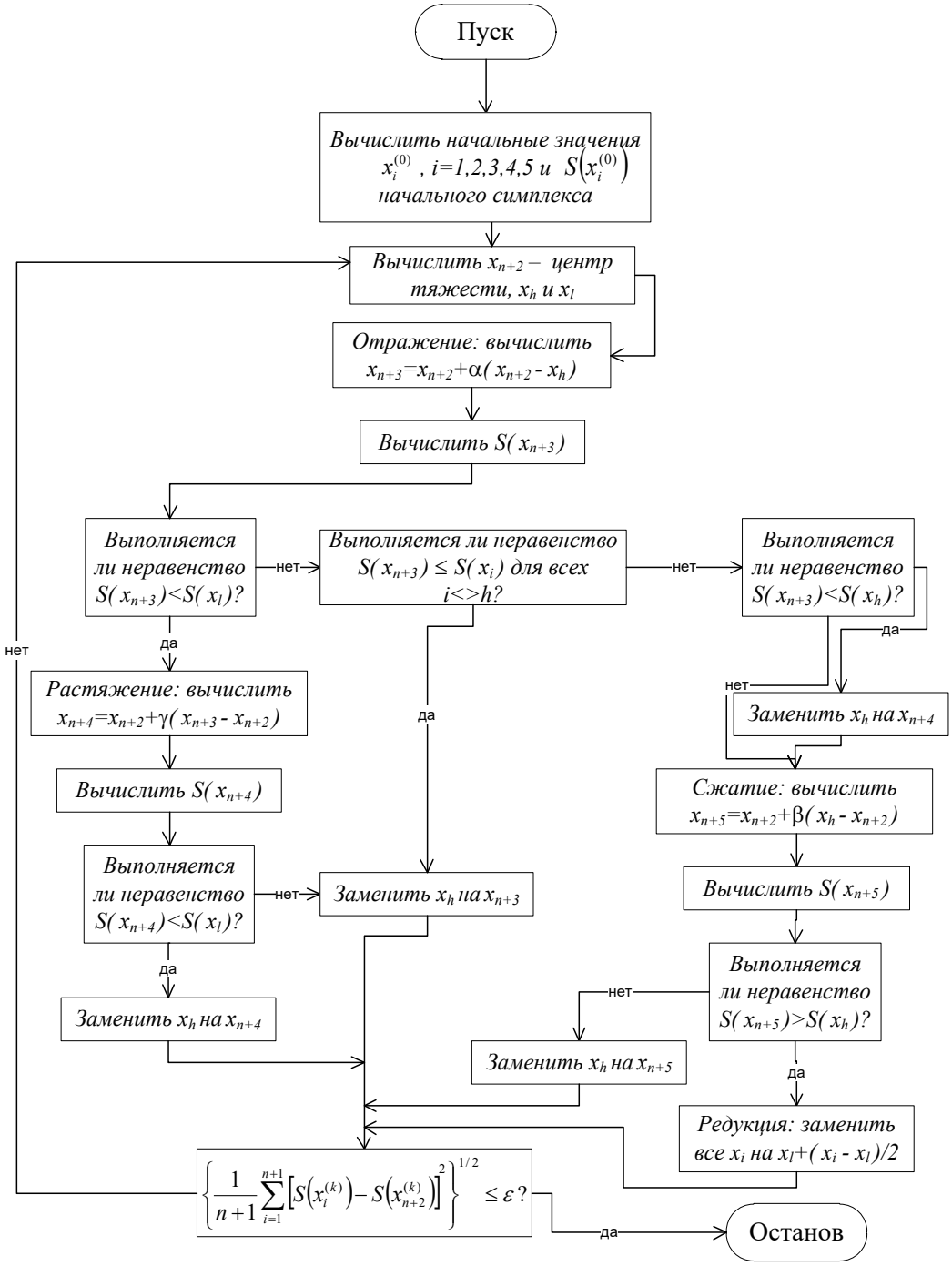

Пуск
Вычислить начальные значения $x_i^{(0)}$, i=1,2,3,4,5 и $S(x_i^{(0)})$ начального симплекса
Вычислить x_{n+2} – центр тяжести, x_h и x_l
Отражение: вычислить $x_{n+3}=x_{n+2}+\alpha(x_{n+2}-x_h)$
Вычислить $S(x_{n+3})$
Выполняется ли неравенство $S(x_{n+3})<S(x_l)$?
нет
Выполняется ли неравенство $S(x_{n+3}) \le S(x_i)$ для всех $i<>h$?
нет
Выполняется ли неравенство $S(x_{n+3})<S(x_h)$?
да
нет
да
Растяжение: вычислить $x_{n+4}=x_{n+2}+\gamma(x_{n+3}-x_{n+2})$
Заменить x_h на x_{n+4}
Вычислить $S(x_{n+4})$
Сжатие: вычислить $x_{n+5}=x_{n+2}+\beta(x_h-x_{n+2})$
Выполняется ли неравенство $S(x_{n+4})<S(x_l)$?
нет
Заменить x_h на x_{n+3}
Вычислить $S(x_{n+5})$
да
Выполняется ли неравенство $S(x_{n+5})>S(x_h)$?
Заменить x_h на x_{n+4}
нет
Заменить x_h на x_{n+5}
да
Редукция: заменить все x_i на $x_l+(x_i-x_l)/2$
$\left\{\frac{1}{n+1}\sum_{i=1}^{n+1}\left[S(x_i^{(k)})-S(x_{n+2}^{(k)})\right]^2\right\}^{1/2} \le \varepsilon$?
нет
да
Останов

Fig. 5.