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MODELING OF HARDWARE SPECTRUMS OF CdZnTe- DETECTORS

This article has developed a technique for modeling the instrumental spectrum obtained by measuring the intrinsic gamma radiation of spent nuclear fuel with different burnup depths and the degree of leakage fuel cladding. The spectrum model of the studied fuel assembly made it possible to determine the sensitivity of measurements and choose the optimal algorithm for processing the spectra. Due to this, costs of developing hardware and software components of the nuclear fuel condition monitoring system have been reduced.

The difference between this technique and the known ones is that they did not use the simulation of the distribution of electric field strength in the volume of the sensor crystal and did not use the Monte Carlo method to simulate electric charge induced during the initial interaction of gamma radiation with the crystal. An experimental verification of the technique using the example of modeling the radiation spectrum of ^{137}Cs source confirmed the effectiveness of the use of the spectrometer based on the CdZnTe detector created in this work: compliance with the requirements for monitoring the state of nuclear fuel in real time; identification of fuel assemblies containing fuel elements with an unpressurized cladding; determination of fuel burnup based on the activity of fission products.

For the first time, a method has been proposed for processing large packets of the spectra of the own gamma radiation of spent nuclear fuel. A distinctive feature of the method is the two-stage processing of the spectra: at the first stage, parameters are determined that describe the measured spectra and are necessary for the operation of the algorithms for their processing by methods of numerical differentiation; at the second stage, using these methods, large arrays of spectra are automatically processed in real time.This method increases the accuracy of measurements and the completeness of the distribution pattern of burnup over the fuel assembly volume. This is the technological basis of the tomography method created in this work.

Keywords: parametric reliability, photodetectors, performance criterion, ionizing radiation, crystal structure

Formulation of the problem. The aim of this work is to develop new improved methods and means of controlling nuclear materials and the state protective barriers at nuclear power plants. Creation of new generation radiation sensors and measuring systems based on them.

Analysis of recent research. Methods for tomographic analysis objects of different physical nature, i.e. the restoration of the physical structure an object from physical fields measured outside the object, as a rule, on a closed surface, originated in the 70s of the 20th century. in connection with the construction of x-ray tomographic images of human organs [1]. In the 80s tomographic methods were widely used in industry for flaw detection [2, 3]. Most of the methods developed to date, as a rule, use active tomography, which assumes the presence of a radiation source passing through the object being examined, and a receiver (or group of receivers) recording the radiation transmitted through the object. To analyze the state of nuclear fuel (NF), in particular, fuel assemblies, it is advisable to use passive emission tomography, based on the registration of its own gamma radiation from fission products (FP) NF, followed by determination of their activity inside the studied fuel assembly.

A rather limited number of papers are devoted to the actual tasks passive emission tomography of nuclear fuel. All of them are made at Uppsala University (Department of Radiation Sciences, Uppsala, Sweden). In particular, in the most thorough pioneering work, as in the subsequent works of Swedish researchers, task was to substantiate theoretically passive emission tomography of the BWR fuel assembly produced by ABB Atom, Sweden of a square form containing 8x8 = 64 round fuel rods [4]. The goal of this work was to establish the absence of one or more fuel rods in the fuel assembly, since the task of nuclear fuel imaging was solved within the framework program for ensuring the nonproliferation of nuclear materials.

The analysis of state questions led to the following conclusions:

– there is a fundamental possibility of using emission gamma tomography of nuclear fuel in order to restore the distribution of fission products by the example of a fuel assembly of a BWR reactor with 64 fuel rods;

– for a fuel assembly of WWPR-1000 reactor containing much more structural elements, development of a new, more efficient tomography algorithm is required;

– from point view of real-time tomography implementation when performing routine operations with nuclear fuel, in particular, overloading, is required sufficiently high computational efficiency of the algorithm.

Main part. In this work, the main principle of constructing a spent nuclear fuel (SNF) control system was chosen to increase the efficiency of measuring the spectra of our own gamma radiation from the spent fuel assembly (SFA). At the same time, from the point of view of minimizing the time spent, it is optimal to carry out such a measurement directly in the process of SNF overload: during vertical movement from the rack of the holding pool [5-7].

This choice is due to the significantly higher information content measurements of the own gamma radiation of SFA. Such measurements make it possible to determine the burnup, holding time and initial enrichment of a controlled SFA without using additional information. In contrast, the data obtained using the existing methodology — measuring the total neutron count rate — make it possible to estimate SNF burnup with the use of additional information on the exposure time and initial enrichment [8, 9]. However, since the overload time is strictly regulated, operations to control burnup depth of nuclear fuel, the exposure time and the initial enrichment should be coordinated in time with the schedule of overload process. Therefore, the main criterion in determining the structure of burnup depth control system should be chosen to work in real time.

As a result of passage gamma radiations through the detector and subsequent processing of the received signal, it is possible to obtain an instrumental spectrum, which is a direct reflection of the interactions of gamma rays with a detecting medium. It provides the primary information used for further analysis of gamma radiation.

The hardware spectrum is complex due to the peculiarities of registering gamma radiation with proportional detectors. In addition, there are both natural and technological limitations on how accurately a detecting system can detect gamma radiation energy. A natural limitation arises mainly due to statistical fluctuations associated with the processes of charge formation in the detector. The positions of the total absorption peaks can also be distorted by such electronic effects as noise, pulse superposition, incorrect installation of the pole-zero circuit, etc. In addition, the real spectrum of gamma radiation of the sample can differ significantly from the spectra obtained in laboratory conditions.

Therefore, in this work, to analyze the possibility of using the manufactured spectrometer in a radiation-technological monitoring system for the state of nuclear fuel, to refine and debug algorithms and programs for processing the measured spectra, we developed a technique for modeling instrument spectra when measuring the own gamma radiation of spent nuclear fuel at various burnup depths and leaks fuel cladding [10]. This significantly reduced the development of hardware and software components of the radiation technology control system, since the theoretical spectrum model of fuel assembly under study allows one to determine measurement sensitivity and select the optimal spectral processing algorithm.

When developing the methodology, the statement was used that the changes in the measurement conditions correspond to linear irreversible transformations of the space of the instrument spectra, and the spectrum of *i*-th component under arbitrary measurement conditions can be represented as:

$$\varphi(a) = \sum_{i=0}^{L} a_i \varphi_i(a), \quad \sum a_i = 1, \tag{1}$$

where $\varphi_i(a)$ – linearly independent spectrum obtained by preliminary measurements, and the coefficients a_i are ame for all components.

The technique for simulating instrumental spectra is based on the following procedures:

1. The spectrum of isotopes with a large number lines is represented as a linear combination of monoenergetic spectra taking into account the quantum yield in geometry of the "narrow beam"; for each isotope, self-absorption in fuel assemblies is not taken into account.

2. The change in the monoenergetic spectrum is simulated due to interaction with the material of surrounding technological environment and the fuel matrix.

3. The first and second procedures are used to form the instrumental spectrum of a mixture isotopes.

The instrumental spectrum of monoenergetic gamma radiation in the “narrow beam” geometry consists of three main components — the peak of total absorption, the peak of emission, and the continuous distribution on the left side due to Compton scattering.

The peculiarity of the detector used is that the analogous representation of the total absorption peak is not a simple Gaussian due to the physics of the process of collecting charge arising from the interaction of the gamma quantum with the detector material.

There are several methods for modeling the hardware gamma-ray detection spectra with detectors based on CdZnTe and CdTe [11, 12]. This method differs from the known methods in that it did not use the simulation of the distribution of electric field strength in the detector volume and did not use the Monte Carlo method to simulate electric charge induced during initial interaction of gamma radiation with detector material [10]. This is due to the fact that applied system of electrodes uses an electrode system that creates spherical geometry of the electric field. Verification of the fulfillment of the quasisphericity conditions for the collection of charge carriers is a key link in the manufacture of the detector. In addition, the peak of departure is modeled, and this does not provide for any of the mentioned methods.

Common to all techniques for modeling the total absorption peak in CdZnTe-based detectors are taking into account the “pulling” of left half total absorption peak due to the contribution of the slower “hole” component, as well as the use of experimental parameters. This is due to poor knowledge processes of growing homogeneous defect-free single crystals of CdZnTe and CdTe and, as a result, the inability to describe band structure of a specific sample from which the detector will be made from generalized electrophysical characteristics. In addition, presence and quantitative characteristics of defects, differences in the mobility and configuration of electric field, and influence of the site primary interaction gamma quantum with the crystal cannot be foreseen. At the same time, general view of the hardware spectrum for detectors made using the same technology is same.

When modeling the spectrum in this work, we used the analytical representation of the peak of total absorption:

$$n(E_i) = n_0 \exp\left[-\frac{(E_i - E_0)^2}{2\sigma^2}\right] + n_0 F_t(E_i), \qquad (2)$$

where $F_t(E_i) = \{A \cdot \exp[B(E_i - E_0)]\}\left\{1 - \exp\left[\frac{C(E_i - E_0)^2}{2\sigma^2}\right]\right\}\delta$ is a function describing the left “tail” of peak total absorption formed due to later collection of charge by holes (F_t – symbol *t* from “tail” – remainder); $n(E_i)$ – number of samples in the channel corresponding to energy of gamma rays E_i; n_0 – peak amplitude; E_0 – centroid peak; σ^2 – dispersion of the Gaussian distribution; $FWHM = 2\sigma\sqrt{\ln 2}$ – full width of photopeak at half its height; A – parameter determining amplitude of the function F_t; B – parameter that determines decline of the function F_t; C – parameter defining “cutoff” of the function F_t; $\delta = 1$ at $E_i < E_0$ and $\delta = 0$ at $E_i > E_0$.

In this case, the peak of emission is described by a Gaussian distribution similar to the first term of equation 2, shifted by the value of the experimentally determined parameter [11].

The dependences of the values of parameters A and B obtained from laboratory energy with exemplary sources are shown in Figures 1 and 2, respectively. The values for the two types of radiation detectors based on CdZnTe and CdTe are given. The parameters were determined by comparing the experimentally measured spectrum with the calculated model spectrum for various values of the studied parameters. The obtained dependences are in good agreement with the data presented in [12]. This is because the detectors are made from raw materials produced by eV PRODUCTS. The literature indicates that the value of the parameter *C*, which determines "cutoff" of the function F_t, is 0.7 for a large set of investigated detectors [12].

To represent Compton distribution $\mu(E, E_i)$, we used the statistical test method (Monte Carlo method), the most effective when considering radiation transfer in matter due to the statistical nature of this process. This is explained by the fact that elementary scattering events occur on a free electron, and therefore the properties of the crystal are not significant. In addition, scattering has a continuous spectrum of secondary particles, and more statistics are needed for its correct presentation. Therefore, it is optimal to use the Monte Carlo method, in which random motion of a particle is considered as a certain trajectory, and its state at each nodal point is played using random numbers from the corresponding distributions. It has been shown that modern implementation algorithms for this method make it possible to achieve modeling accuracy of up to 1-2% [6, 11].

The distribution $\mu(E, E_i)$ is the probability of scattering secondary electrons in the energy interval and is determined by the Klein-Nishina formula:

$$W(\alpha_1)d\alpha_1 = A(\alpha)f(\alpha,\alpha_1)d\alpha_1, \quad \frac{\alpha}{1+2\alpha} \le \alpha_1 \le \alpha, \qquad f(\alpha,\alpha_1) = \frac{\alpha_1}{\alpha} + \frac{\alpha}{\alpha_1} + \left(\frac{1}{\alpha} - \frac{1}{\alpha_1}\right)\left(2 + \frac{1}{\alpha} - \frac{1}{\alpha_1}\right), \tag{3}$$

where α, α_1 – energy before and after scattering in units of electron mass; $A(\alpha)$ – normalization constant.

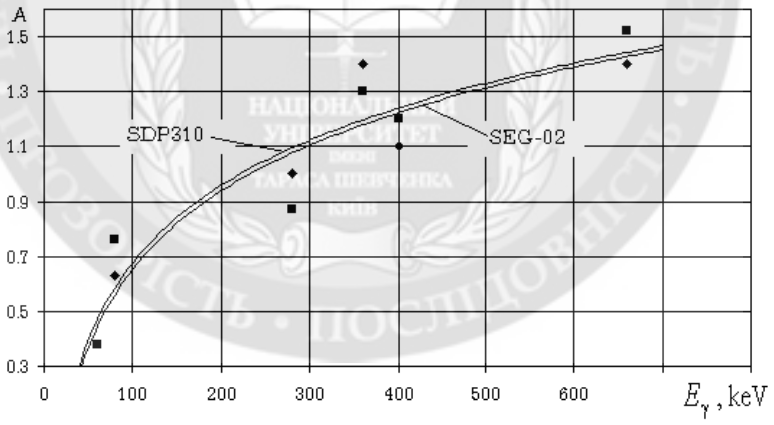

Figure 1 – The dependence of parameter *A*, which determines the amplitude of the function F_t, on the energy of detected radiation: 1 – prototype crystal with a quasispherical contact design; 2 – prototype crystal with planar contact design

In this case, maximum energy of a Compton electron is determined by the formula:

$$E_{k\max} = \frac{E}{1 + \frac{mc^2}{2 \cdot E}}. \tag{4}$$

For monoenergetic gamma radiation, the maximum energy E of the Compton electron is calculated by the formula (4), and then the Monte Carlo model is used to construct a model of the Compton distribution. When simulating the required portion of the instrument spectrum, the resulting distribution is multiplied by a correction factor, which is the detector constant and determines its characteristic as "compton / peak" ratio.

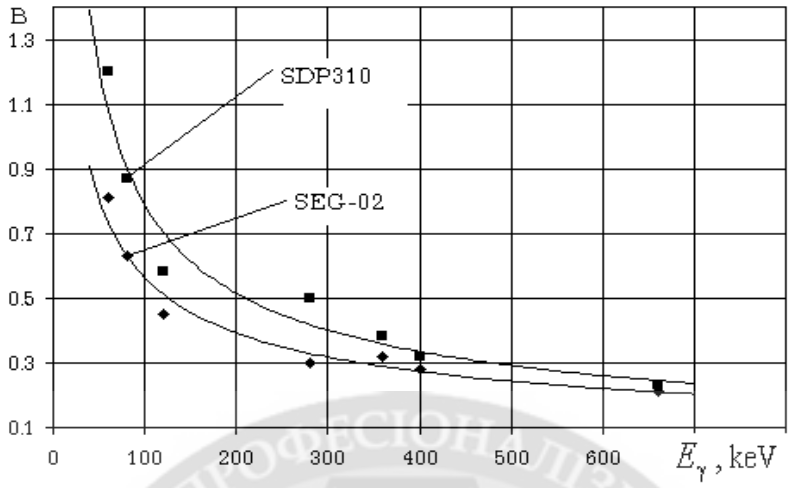

Figure 2 – The dependence of parameter *B*, which determines the decay of the function F_t, on the energy of detected radiation: 1 – prototype crystal with a quasispherical contact design; 2 – prototype crystal with planar contact design

Then differential energy spectrum $d\varphi(E)/dE$ for a large number of gamma rays in the decay scheme has the form:

$$d\varphi(E)/dE = \sum_{i=1}^{M}\left[n_i(E)+n_{esc}(E)+\mu(E,E_i)\right], \quad (5)$$

where *M* is the number of gamma lines in the decay scheme of the studied isotope.

The experimental spectrum $S_э(E)$ has a discrete character, therefore, the theoretical spectrum is converted into a discrete form with a given energy window ΔE:

$$\varphi(E) = (d\varphi(E)/dE)\Delta E\,. \quad (6)$$

A model of instrumental spectrum obtained by measuring the intrinsic radiation of a fuel assembly can be obtained using expression (5) based on simulation results of the instrumental spectra for each isotope present under the cladding of a fuel rod. In addition, in order to take into account influence of isotopes present in small quantities, a correction in the form of a continuous distribution was introduced during modeling. In order to ensure that the spectrum obtained on the basis of (5) corresponds to the instrumental spectrum when measuring gamma radiation of spent nuclear fuel, contribution of scattering in the fuel matrix and environment was additionally taken into account. The resulting distribution is normalized to determine the absolute values; therefore, reference data on differential cross sections for the interaction of gamma radiation with matter are used [10].

Based on the developed methodology and experimentally determined characteristics of spectrometer, instrumental spectrum were simulated under various measurement conditions with subsequent processing of obtained model spectra.

To verify the correct implementation, the spectrum of a ^{137}Cs point source with random noise was superimposed (Fig. 3). For comparison, Figure 4 shows the measured 137Cs spectrum from a set of gamma-ray spectra. A characteristic feature of model is a more pronounced peak in the region of maximum energy of Compton electrons.

In addition, a peak in region of 210 keV and peaks in the X-ray region should be noted in the experimentally obtained spectrum. The peak in the region of 210 keV is due to gamma rays, which were subjected to Compton scattering in one of the materials surrounding the detector. Gamma rays scattered by more than 110-120 °, will have almost the same energy values in the range from 200 to 250 keV. Consequently, the contribution of the monoenergetic source will be represented by many scattered gamma rays whose energy is close to this minimum value.

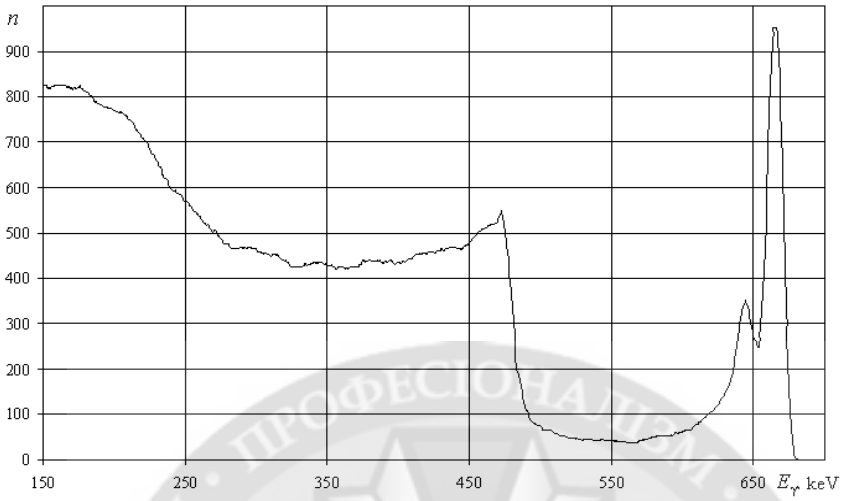

Figure 3 – Model of instrumental spectrum when measuring ^{137}Cs point source

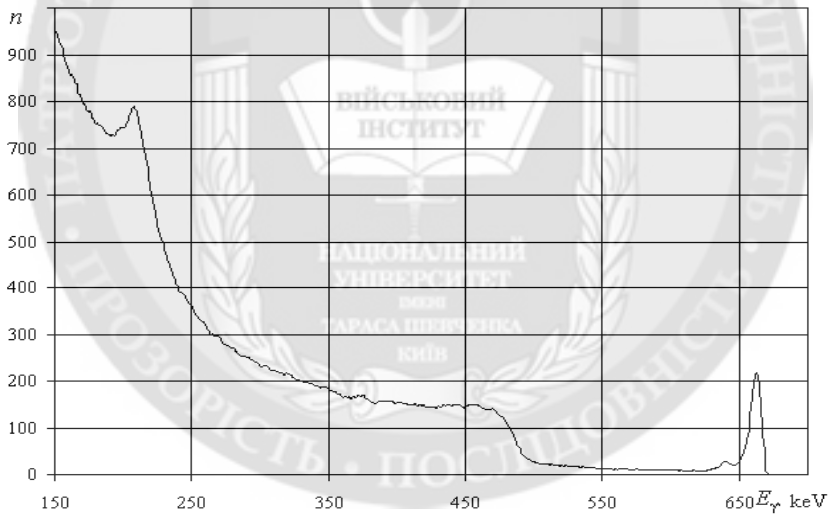

Figure 4 – Hardware spectrum for measuring ^{137}Cs point source

Conclusion. X-ray peaks are associated with the peculiarity of the organization of measurements – location of the source, detector and protection, and are caused by the characteristic radiation of lead from which the protection is made. Unfortunately, such peaks are rather difficult to model, but this is not necessary, since in real conditions a collimator is used and these peaks are absent.

The results of modeling the instrumental spectra during measurements of intrinsic gamma radiation of SNF are close to those obtained previously. This confirms the conclusion that the CdZnTe-based spectrometer made in this work satisfies the requirements for monitoring the state of SNF in real time, identifying fuel assemblies containing fuel elements with an unpressurized cladding, and determining the burnup of nuclear fuel using analysis of the activity of fission products.

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МОДЕЛЮВАННЯ АПАРАТУРНИХ СПЕКТРІВ CdZnTe-ДЕТЕКТОРІВ

У даній статті розроблена методика моделювання апаратурних спектрів, одержуваних при вимірюванні власного гамма-випромінювання відпрацьованого ядерного палива з різною глибиною вигоряння і ступенем негерметичності оболонки твела. Модель спектру досліджуваної тепловиділяючої збірки дозволила визначити чутливість вимірювань і вибрати оптимальний алгоритм обробки спектрів. За рахунок цього скорочені витрати на розробку апаратної і програмної складових системи контролю стану ядерного палива.

Відмінність даної методики від відомих сотоит в тому, що не використовувалося моделювання розподілу напруженості електричного поля в об'ємі кристала датчика і не застосовувався метод Монте-Карло для моделювання електричного заряду, індукованого при первинному взаємодії гамма-випромінювання з кристалом.

Експериментальна перевірка методики на прикладі моделювання спектра випромінювання джерела ^{137}Cs підтвердила ефективність застосування створеного в даній роботі спектрометра на основі CdZnTe-детектора: відповідність вимогам контролю стану ядерного палива в режимі реального часу; ідентифікації ТВС, що містить ТВЕЛ з негерметичной оболонкою; визначення вигоряння палива на основі активності продуктів поділу.

Вперше запропоновано метод обробки великих пакетів спектрів власного гамма-випромінювання відпрацьованого ядерного палива. Відмінною особливістю методу є двоетапна обробка спектрів: на першому етапі визначаються параметри, що описують виміряні спектри і необхідні для роботи алгоритмів їх обробки методами чисельного диференціювання; на другому етапі за допомогою цих методів автоматизовано обробляються великі масиви спектрів в режимі реального часу.

Даний метод збільшує точність вимірювань і повноту картини розподілу вигоряння за обсягом ТВС. Це є технологічною основою створеного в даній роботі методу томографії.

Ключові слова: параметрична надійність, фотоприймальні пристрої, критерій працездатності, іонізуюче випромінювання, кристалічна структура

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МОДЕЛИРОВАНИЕ АППАРАТУРНЫХ СПЕКТРОВ CDZNTE-ДЕТЕКТОРОВ

В данной статье разработана методика моделирования аппаратурных спектров, получаемых при измерении собственного гамма-излучения отработавшего ядерного топлива с разной глубиной выгорания и степенью негерметичности оболочки твэла. Модель спектра исследуемой тепловыделяющей сборки позволила определить чувствительность измерений и выбрать оптимальный алгоритм обработки спектров. За счет этого сокращены затраты на разработку аппаратной и программной составляющих системы контроля состояния ядерного топлива.

Отличие данной методики от известных сотоит в том, что не использовалось моделирование распределения напряженности электрического поля в объеме кристалла датчика и не применялся метод Монте-Карло для моделирования электрического заряда, индуцированного при первичном взаимодействии гамма-излучения с кристаллом.

Экспериментальная проверка методики на примере моделирования спектра излучения источника ^{137}Cs подтвердила эффективность применения, созданного в данной работе спектрометра на основе CdZnTe-детектора: соответствие требованиям контроля состояния ядерного топлива в режиме реального времени; идентификации ТВС, содержащей твэл с негерметичной оболочкой; определения выгорания топлива на основе активности продуктов деления.

Впервые предложен метод обработки больших пакетов спектров собственного гамма-излучения отработавшего ядерного топлива. Отличительной особенностью метода является двухэтапная обработка спектров: на первом этапе определяются параметры, описывающие измеренные спектры и необходимые для работы алгоритмов их обработки методами чисельного дифференцирования; на втором этапе с помощью этих методов автоматизировано обрабатываются большие массивы спектров в режиме реального времени.

Данный метод увеличивает точность измерений и полноту картины распределения выгорания по объему ТВС. Это является технологической основой созданного в данной работе метода томографии.

Ключевые слова: параметрическая надежность, фотоприемные устройства, критерий работоспособности, ионизирующее излучение, кристаллическая структура.