ВІЙСЬКОВА ТЕХНІКА І ТЕХНОЛОГІЇ ПОДВІЙНОГО ПРИЗНАЧЕННЯ

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RESEARCH PROCESSES OF GAMMA RADIATION DETECTOR FOR DEVELOPING A PORTABLE DIGITAL SPECTROMETER

When considering methods of combating the illicit circulation of nuclear materials, it is necessary to detect trace amounts of materials, and in many cases not to seize them immediately, but to establish the place of storage, processing, routes of movement, etc. As a result, there is a new demand for isotope identification measurements to meet a wide range of different requirements. Measurements should be carried out in the field in a short time, when results need to be obtained within tens of seconds. The devices with which the personnel work should be small and low-background. Such requirements appear when working to identify cases of illegal trade in nuclear materials and radioactive sources, as well as when solving radiation protection problems and when handling radioactive devices and waste.

In this work, new generation radiation sensors and measuring systems based on them have been created, which open up previously unknown possibilities in solving problems of nuclear fuel analysis, increasing the accuracy and efficiency of monitoring technological parameters and the state of protective barriers in nuclear power plants, and creating means for IAEA inspections. For the first time a portable digital gamma-ray spectrometer for radiation reconnaissance in the field was developed and created. Distinctive features of such devices are:

The analysis showed that the required value of error due to energy dependence of the sensitivity can be achieved using, for example, Analog Devices 10-bit AD9411 ADCs with a sampling rate of 170 MHz. The number of quantization levels is determined by the requirement to measure the dose rate of gamma radiation with an energy of at least 10 keV. This minimum energy corresponds to the use of 10-bit ADCs.

On the basis of the developed model, an ionizing radiation detector for dosimetry was created. Its fundamental difference from known devices is the use of CdZnTe crystals as a primary gamma-ray converter (sensor). The advantages of such a solution, proved by previous studies, made it possible to create a detector with: high resolution, no more than 40 keV; a wider dynamic range of values of the recorded radiation dose rate - from background to emergency operating modes of the reactor; lower value of the energy equivalent of noise.

Keywords: energy dependence of sensitivity, gamma-ray spectrometer, frequency sampling, ionizing radiation detector

Introduction. The key problem of nuclear power - radiation safety - is solved by ensuring the reliability of protective barriers for the main objects of the technological process of NES functioning: fuel elements, fuel assemblies (FA), coolant transfer circuits, etc.

The new generation radiation sensors and measuring systems created in this work open up previously unknown possibilities in solving problems of nuclear fuel analysis, increasing the accuracy and efficiency of monitoring technological parameters and the state of protective barriers in nuclear power plants, creating means for IAEA inspections.

When considering methods of combating the illicit circulation of nuclear materials, it is necessary to reconsider approaches to the organization of control: today it is necessary to detect trace amounts of materials, and in many cases not to seize them immediately, but to establish the place of storage, processing, routes of movement, etc.

Analysis of previous studies. The level of development and application radiation technologies is largely determined by the state of nuclear instrumentation. In a relatively short period of time, this industry went through several stages of development, and each of them was marked by the emergence of various devices that register and measure parameters of ionizing radiation: gas-discharge counters, scintillators, semiconductor detectors, and others. Their emergence and further widespread use was provided in the past by works from Crookes, Rutherford, Geiger and Müller to the works of A.B. Dmitriev, S.N. Perelman, V.G. Tchaikovsky, and V.G. Baranov, which are closer to us in time. I., Golbek G.R., Nemirovsky B.V., Yakubovich A.L. and many others. The basis of progress in nuclear instrumentation was the simultaneous development of two directions - nuclear physics research and electronics. However, both directions at that time developed independently, without proper mutual connection.

Turning on the CdZnTe detector in the mode of operation pulsed proportional ionization chamber makes it possible to significantly increase its sensitivity and expand the dynamic range of values of recorded dose rate from background values to those caused by emergency operating modes of reactor facility. The use of pulsed mode makes it possible to practically implement other possibilities and, first of all, compensation energy dependence of the sensitivity (EDS), so-called "stroke with rigidity".

Currently, dosimetry devices with semiconductor Si-based detection units are mass-produced [1-5]. Advantages of CdZnTe over Si – higher sensitivity and lower energy equivalent of noise [6-8]. However, a large effective atomic number also determines a larger (more than 10) value of the energy dependence of sensitivity (EDS).

The literature describes the method of hardware correction of EDS using a device assembled on discrete elements with a low degree of integration [9]. Such a device is unreliable, expensive, and has a low dynamic range. In this work, a device for digital correction of EDS when the crystal is operating in counting mode is proposed and manufactured on basis of a modern element base.

Main part. Thus, there is a new need for isotope identification measurements to meet a wide range of different requirements. Measurements should be carried out in the field in a short time, when results need to be obtained within tens of seconds. The devices with which the personnel work should be small and low-background. Such requirements appear when cases of illegal trade in nuclear materials and radioactive sources are detected, as well as when solving problems of radiation protection and when handling radioactive waste.

Correction is carried out by changing the pulse frequency at the output of the detecting unit depending on the energy of the registered radiation E_γ:

$$n_{out} = n_{inp} \cdot K(x), \tag{1}$$

where n_{out} – is the pulse frequency at the output of the detecting unit, n_{inp} – is the pulse frequency at the output of the detector preamplifier; $K(x)$ — coefficient of pulse frequency change at the output of the detecting unit; x – number of the channel corresponding to energy E_γ.

The numerical value of the coefficient of change in the pulse frequency at the output of the detecting unit is determined on the basis of the analytical dependence of the ratio of the detector's sensitivity to the recorded gamma radiation $S(E_\gamma, x)$ and the sensitivity to gamma radiation with the energy at which its calibration was carried out $S(E_{\gamma k}, x)$:

$$\varepsilon\left(E_\gamma\right) = \frac{\int\limits_{x_{\min}}^{x_{\max}} S\left(E_{\gamma,} x\right) \cdot K\left(x\right) dx}{\int\limits_{x_{\min}}^{x_{\max}} S\left(E_{\gamma k,} x\right)\left(x\right) dx}, \tag{2}$$

where $x_{\min}$ – is the number of channel corresponding to the noise level; $\varepsilon\left(E_\gamma\right)$ – preset relative dependence of detector sensitivity on energy.

For the practical implementation of correction output signal of the detecting unit, it is proposed to use piecewise linear interpolation of the given relative analytical dependence of the detector's sensitivity on energy:

$$\varepsilon\left(E_\gamma\right)=\left[\frac{\begin{matrix}S\left(E_{\gamma_1},x_1\right)K\left(x_1\right)\\ S\left(E_{\gamma_1},x_1\right)K\left(x_1\right)+S\left(E_{\gamma_2},x_2\right)K\left(x_2\right)\end{matrix}}{S\left(E_{\gamma_1},x_1\right)K\left(x_1\right)+S\left(E_{\gamma_2},x_2\right)K\left(x_2\right)+...+S\left(E_{\gamma_j},x_j\right)K\left(x_j\right)}\right]\times\frac{1}{\int\limits_{N_{\min}}^{N_{\max}}S\left(E_{\gamma k},x\right)dx}. \quad (3)$$

Thus, the task of correcting the energy dependence of the sensitivity is to obtain the value of *K(x)* for a certain energy range of registered photons E_{γ_j}.

At the first stage, when creating an algorithm for digital correction of the energy dependence of sensitivity, the data on the coefficient *K(x)*, given in table 1 [10] were used.

Table 1

Δx, keV	40÷80	80÷170	170÷350	350÷450	450÷1100	1100÷1500
$K(x)$	0,015625	0,039	0,625	3,875	4,5	22

The relative dependence of the detector's sensitivity on energy is shown in Table 2.

Table 2

E_γ, keV	59	122	166	279	392	662	835	1250
$\varepsilon(E_\gamma)$	1,03	1,00	1,05	0,99	1,04	1,02	0,93	1,00

Thus, it follows from table 2 that the maximum error caused by dependence of the sensitivity of CdZnTe sensor on the energy of gamma radiation (“stroke with stiffness”) is 7% for an energy of 835 keV.

Figure 1 shows the block diagram of first variant of the dosimetric detection unit with digital correction of “stroke with stiffness”. In terms of dimensions and output signals, the developed detector is compatible with the BDMG-41 type detection unit used today [11]. Let's consider the principle of operation one of the variants such a detector.

When ionizing radiation interacts with the sensor material, pulses of negative polarity appear at its output with an amplitude proportional to the energy absorbed in the crystal. The signal taken from the sensor crystal goes to the preliminary charge-sensitive amplifier. Then the signal is amplified in main amplifier (MA) to voltages sufficient for the operation of ADC and the pulse normalizer. A single-chip microcomputer (SCM) and an ADC form a spectrum analyzer (in nuclear spectrometry, the term “multichannel pulse analyzer” is often used) with subsequent processing to determine the pulse frequency change factor at the output of the detecting unit according to the algorithm described above. Also, pulses from the normalizer are sent SCM to determine the count rate. The SCM, in turn, by controlling the counter with a variable division ratio (SPKD), corrects the energy dependence of the sensitivity of the CdZnTe sensor [12-14]. From the counter output, the corrected pulses of standard amplitude corresponding to the levels of logical “0” and “1” of the used element base are sent to the interface node. The interface node converts indicated pulses in terms of the duration and amplitude required for the operation of ARSMS channel, which does not comply with modern information transfer standards.

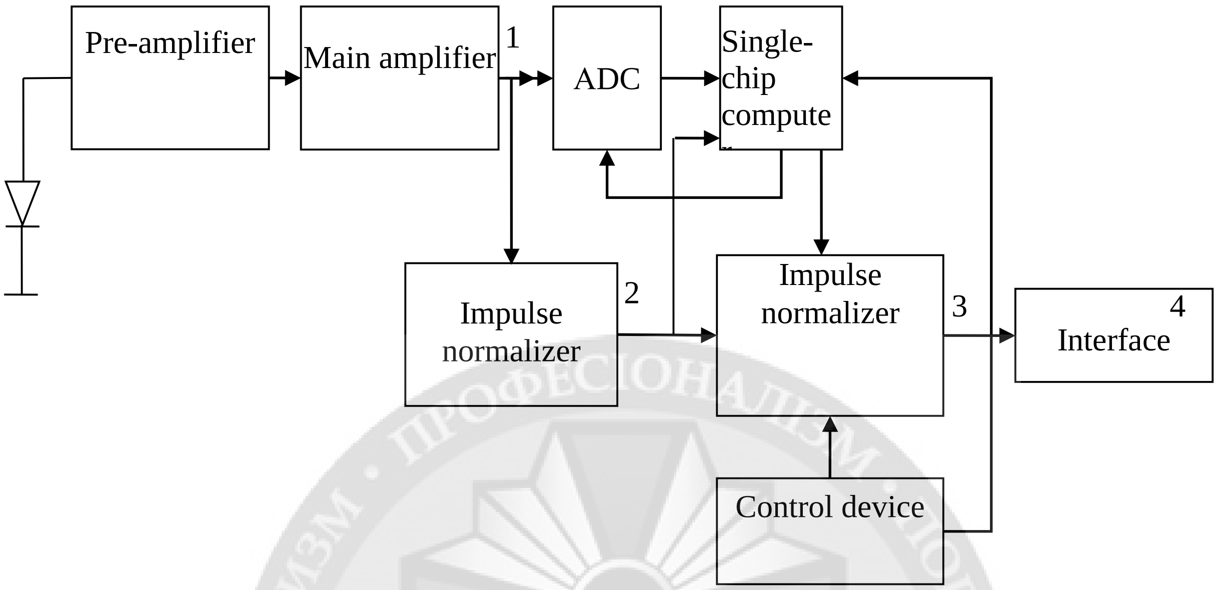

Figure 1 – Block diagram of a detecting unit based on a CdZnTe sensor

Fig. 2 illustrates signal processing in the detecting unit when measuring the same dose rate for gamma radiation of different energies (digital designations according to fig. 1). Due to the high effective atomic number of CdZnTe, at the output of main amplifier (CdZnTe-sensor and preamplifier, respectively), the number of detected gamma quanta is greater for lower energies (fig. 2 a, b).

Fig. 2 shows the signals obtained at control points 1 - 4 of Figure 1. At point 1a, the original analog signal was received after the op-amp, showing the shape of the pulse after registration of gamma radiation (Figure 2 a). Plot 2a shows an amplitude-normalized rectangular signal, in which the number of pulses corresponds to the number of registered radiation photons. Plot 3a represents the number of pulses corrected according to formula (2) based on the analysis of the pulse amplitude in point 1a. When the energy changes, the number of pulses changes (plots 2b, 3b). Such signals are not yet suitable for processing by the automated radiation safety monitoring system (ARSMS). Therefore, the signals are normalized in amplitude and duration (plots 4a, b).

The disadvantage of the above technical solution is the small dynamic range of measured values dose rate.

Therefore, variant of the detecting unit has been created in which the algorithm for correcting the signal at unit output is implemented using a signal processor. In this case, a digital code is fed to the input of signal processor, proportional to the amplitude of pulse that occurs when registering a gamma quantum. Such a pulse will be obtained as a result of conversion by a high-speed ADC installed immediately after the output of charge-sensitive preamplifier. This approach was developed in detail when creating gamma-ray spectrometers [10, 11, 12].

The analysis showed that the required value of the error due to the energy dependence of the sensitivity can be achieved using, for example, Analog Devices 10-bit AD9411 ADCs with a sampling rate of 170 MHz. The number of quantization levels is determined by the requirement to measure the dose rate of gamma radiation with an energy of at least 10 keV. This minimum energy corresponds to the use of 10-bit ADCs.

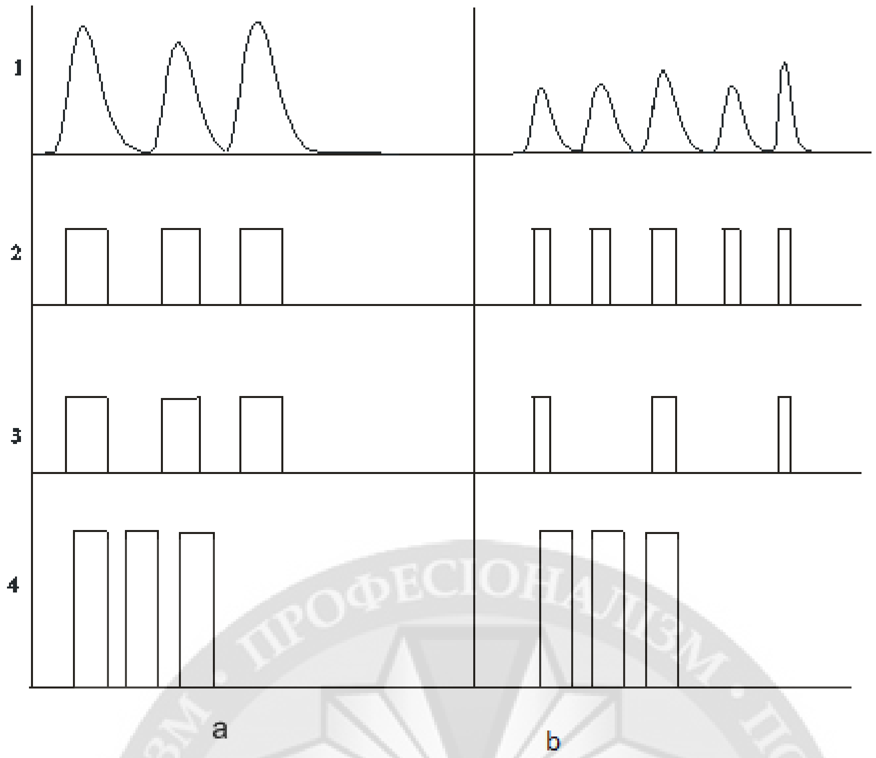

Figure 2 – Time diagram of the operation detecting unit based on CdZnTe-sensor when measuring the same dose rate for gamma radiation of different energies: a – energy is higher; b – energy is less than for a (1-4 – according to fig. 1)

The structural diagram of the spectrometer is similar to that shown in fig. 2. In contrast, this device has two sensors. The second one is placed on a telescopic rod, which allows monitoring in hard-to-reach places with increased radiation hazard.

Structurally, BDPG-CZT consists of two blocks:

- detector unit (DU) designed to register gamma-radiation flux pulses and convert them into voltage pulses;
- a signal processing unit (SPU) designed to convert the voltage pulses issued by the detector into voltage pulses of constant amplitude, constant duration and with a repetition rate proportional to the radiation dose rate.

The database consists of following nodes:

- a sensor made on the basis of a semiconductor crystal CdZnTe, which receives pulses of gamma radiation and converts them into charge pulses;
- a preamplifier (PP), which is a charge-sensitive amplifier (CSA), which converts the sensor charge pulses into voltage pulses;
- high-resistance resistor of the sensor power supply circuit.

BFB consists of the following units:

- resistive divider (RD), which sets the voltage on detector;
- the main amplifier (MA), which amplifies the pulses of detector up to 2V-amplitude;
- a comparator that converts the op-amp pulses with an amplitude higher than a given threshold into logical pulses with TTL levels;
- latch flip-flop, which is set to "1" by comparator pulses;
- an amplitude recording device (ARD), which selects and fixes the amplitude of pulses at the output of op-amp;
- an analog-to-digital converter (ADC) that converts the output voltage of the UZA into a 12-bit serial code;

- microcontroller (MC), which is the main control and computing device of the BDPG-CZT unit;
- a reprogrammable read-only memory (RROM), which stores the correction and calculation factors used by the MC program;
- interface, which is a device that generates output pulses BDPG-CZT with a frequency set by codes generated by MC;
- transceiver RS-232, which implements the exchange of information at the physical level between the MC and an external personal computer (PC) via RS-232 interface;
- a protective diode that protects the BDPG-CZT in case of incorrect polarity of the +12 V power supply circuit;
- a +5 V voltage stabilizer that generates a +5 V constant voltage to power biofeedback devices (ADC, MK, etc.).

The BDPG-CZT detecting unit operates as follows.

An external voltage of +400 V is applied to the BDPG-CZT, which creates an offset on the sensor through the RD and a high-resistance resistor. When gamma radiation enters the detector, charge pulses are formed on it, which are converted by the PU into voltage pulses. These pulses are amplified by the op-amp and fed to the comparator and UZA.

The logic pulses at the comparator output are set to "1" by the latch trigger, the state of which is read by the MC. After detecting a logic "1" at the output, MC resets this flip-flop to prepare for receiving the next comparator pulse. The choice of the comparator threshold allows you to set the level of suppression of the noise component op amp signal when detecting radiation pulses, i.e. this threshold actually determines the lower energy level of the detected radiation.

The BDPG-CZT detecting unit operates as follows.

An external voltage of +400 V is applied to the BDPG-CZT, which creates an offset on the sensor through the RD and a high-resistance resistor. When gamma radiation enters the detector, charge pulses are formed on it, which are converted by the PU into voltage pulses. These pulses are amplified by the op-amp and fed to the comparator and UZA.

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The output pulses are continuously delivered to the measuring channel. Even if the BDPG-CZT unit does not register radiation, its output always contains pulses with a frequency of 0.3 - 0.5 Hz. This is used to check the functionality of the battery measurement channel.

A voltage of + 6 V is applied to the "Blenker" input, due to this, the MC sets at the interface output pulses with a repetition rate of 1000 Hz.

In the calibration mode, initialized by PC, service information, correction and calculation coefficients are written into the microcontroller through the computer port interface. The microcontroller stores all this information in the non-volatile memory of EPROM. In the measurement mode, only this information is read.

MC conducts a series of measurements and analyzes the result. The first step is to measure the count rate of input pulses. The maximum counting rate is limited from above by a value of 65536 imp/s, so a preliminary measurement is made at an exposure of 0.1 s and MC compares the obtained value with the number 6500 (10% of the maximum channel load). If the count rate exceeds specified limit, MC automatically sets the output pulse repetition rate of 65000 imp/s.

If the count rate is in range of values from 0 to 65000 pulses/s, then the MC conducts a set of pulses with subsequent averaging. So, at a count rate from 0 to 3000 pulses/s, the dialing time is 16 s. At a count rate of 3000 to 10000 cps, the dialing time is 4 s. At the maximum counting rate, the dialing time is 1 s.

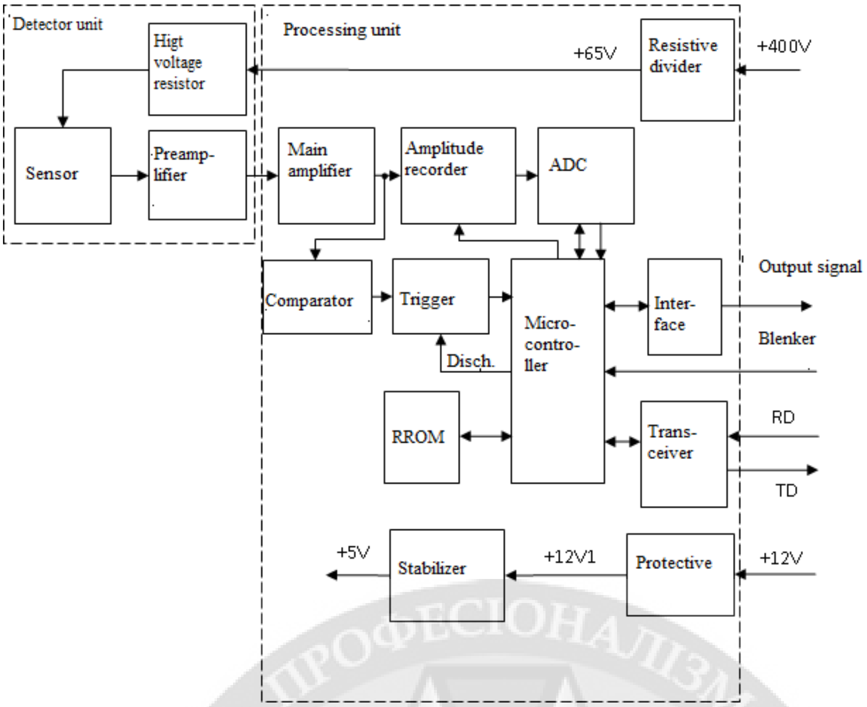

Figure 3 – Block diagram of the BDMG-CZT detecting unit

Conclusions. For the first time a portable digital gamma-ray spectrometer for radiation reconnaissance in the field was developed and created. Distinctive features of such devices are:

- application of CdZnTe detectors with coplanar and quasi-spherical crystal geometry;
- the use of digital methods of filtering by the pulse shape, implemented in a digital spectrometer.

The manufactured sets of such gamma spectrometers have an energy resolution of 6 keV, which meets the requirements for assessing the isotopic composition of nuclear fuel.

A prototype of a digital gamma spectrometer using a multielement CdZnTe sensor has been developed. On the basis of inexpensive CdZnTe-detectors, a prototype unit for detecting the power of air kerma with an average sensitivity of more than 120,000 s^{-1} at an absorbed power of 1 rad/h has been developed and manufactured. The range of the measured absorbed dose rates was from 50 mrad/hour to 10 rad/hour with a crystal size of 5x5x1 mm.

The developed spectrometer meets the basic requirements for application in the program of international safeguards for the non-proliferation of nuclear materials.

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ДОСЛІДЖЕННЯ ПРОЦЕСІВ ДЕТЕКТОРА ГАММА-ВИПРОМІНЮВАННЯ ДЛЯ РОЗРОБКИ ПОРТАТИВНОГО ЦИФРОВОГО СПЕКТРОМЕТРА

При розгляді методів боротьби з незаконним обігом ядерних матеріалів необхідно виявляти слідові кількості матеріалів, причому в багатьох випадках не вилучати їх відразу, а встановлювати місце зберігання, обробки, маршрути переміщення і т.д. Як результат, з'явилася нова потреба в вимірах із забезпеченням ідентифікації ізотопів з виконанням широкого набору різних вимог. Виміри повинні бути проведені в польових умовах за короткий термін, коли результати необхідно отримати в межах десятків секунд. Прилади, з якими при цьому працює персонал, повинні бути малого розміру і низькофонова. Такі вимоги з'являються при роботах з виявлення випадків незаконної торгівлі ядерними матеріалами та радіоактивними джерелами, а також при вирішенні завдань радіаційного захисту і при поводженні з радіоактивними приладами і відходами.

У даній роботі створено радіаційні датчики нового покоління і вимірювальні системи на їх основі відкривають раніше невідомі можливості в рішенні задач аналізу ядерного палива, збільшення точності і ефективності контролю технологічних параметрів і стану захисних бар'єрів в АЕС, створення засобів для інспекцій МАГАТЕ. Розроблено портативний цифровий спектрометр гамма-випромінювання для радіаційної розвідки в польових умовах. Основними особливостями таких приладів є застосування CdZnTe-детекторів з компланарності і квазісферіченою геометрією кристала і використання цифрових методів фільтрації за формою імпульсу, реалізованих в цифровому спектрометрі. Виготовлені комплекти таких гамма-спектрометрів мають енергетичну роздільну здатність 6 кеВ, що задовольняє вимогам оцінки ізотопного складу ядерного палива. Розроблено прототип цифрового гамма-спектрометра з застосуванням багатоелементного CdZnTe-датчика. На основі недорогих CdZnTe-детекторів розроблений і виготовлений макетний зразок блоку детектування потужності повітряної керма із середньою чутливістю понад 120000 з-1 при поглиненої потужності 1 рад / год. Діапазон вимірюваних потужностей поглинутої дози при цьому склав від 50 мкрад / год до 10 рад / год при розмірах кристала 5х5х1 мм.

Ключові слова: гамма-спектрометри, радіаційні датчики, цифровий спектрометр, ядерне паливо, поглинена доза, методи фільтрації, ефективності контролю.