



Gamma Ray Spectrum by Software Methods for Radioactive Waste

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Abstract

The requirements of NTD (Neglected Tropical Diseases) and technological regulations for the operation of NPP (Nuclear Power Plant) power units (NP-001-97 (OPB-88/97), NP-082-07) define the requirements for monitoring the specific activity of iodine-131 (the amount of iodine) in the NPP primary circuit coolants. The advantages of laboratory control include accuracy of measurement and the radionuclide composition of the primary coolant, measured using high-precision laboratory equipment. Instrumental spectra were obtained for the detection units BDKG-205m with various options for the placement of waste in a container, their composition, mass of waste, average density, and various activity levels of waste. The basic idea behind gamma-ray spectroscopy is to detect and analyze the energy of incident gamma rays. Gamma rays of varying energy and intensity are emitted from radioactive sources. The gamma-ray energy spectrum is produced when gamma rays are detected and examined using a spectroscopy instrument. The initial stage in gamma-ray spectroscopy is to detect gamma rays using a suitable detector. The detector captures and measures the energy of incoming gamma rays. Scintillation detectors, semiconductor detectors, and gas-filled detectors are among the detectors used in gamma-ray spectroscopy. The incoming gamma-ray energy is converted into electrical signals that can be processed and studied by these detectors. The spectroscopic system measures and records the energy of gamma rays when they are detected. The derived energy spectrum depicts the intensity distribution of gamma rays as a function of energy. The spectrum is a visual representation of the different energy levels found in gamma-ray emission.

1. Introduction

To increase the efficiency of control and to eliminate the shortcomings of laboratory control, a number of companies have proposed automated control based on a semiconductor detector made of high-purity germanium (HPGe). HPGe-monitoring of the radionuclide composition of the coolant is, in fact, laboratory equipment, which entails some features in its operation and maintenance, as well as imposes a number of restrictions on the

qualifications of the personnel serving it. There are many manufacturers on the market for automated control systems. But, despite this, the choice of components for installations based on HPGe detectors is commonly limited to two manufacturers: Ortec and Canberra (USA), which practically do not have fundamental differences in the operation and design of spectrometric equipment. The main disadvantages of installations created on the basis of HPGe detectors:

- Difficult to use equipment;
- Limited performance (laboratory placement required).
- Qualified maintenance personnel is required to work with the equipment (spectrometric engineer).

The operating experience of such installations has shown that in the event of equipment failure, repair on site or even at the equipment dealer in the territory of the Russian Federation is difficult and often impossible, due to the complexity of the equipment. The equipment has to be sent to the manufacturer, and in this case, the repair can last up to a year or more. As a result, the NPP remains without operational control. Thus, we can highlight the main tasks that need to be solved:

- Increase the reliability of equipment;
- Increase its maintainability, radically reduce the repair time in case of equipment failure;
- Reduce the cost of the system.

Proposals to eliminate the shortcomings of the existing systems for monitoring the radionuclide composition of the coolant are presented below.

1. Implementation in an automated system of two independent measuring channels operating in parallel, which, in turn, will lead to an increase in the frequency of obtaining measured data and an increase in the reliability of the system.
2. Use of a domestic-made detector in the installation, which will reduce the cost of the installation, increase maintainability, reduces the repair time and supply of components.

Of all types of detectors, detectors based on a crystal of lanthanum bromide ($\text{LaBr}_3(\text{Ce})$), which are gaining popularity all over the world, are optimal for solving these problems. These detectors have a better resolution compared to detectors based on NaI , CsI , etc. crystals, which makes it possible to provide selective detection of gamma radiation of iodine radionuclides (^{131}I - ^{135}I) in the NPP primary coolant. Detectors based on the $\text{LaBr}_3(\text{Ce})$ crystal also have improved performance characteristics (operating temperature from -30°C to $+100^\circ\text{C}$, maximum load of the spectrometric path is not less than 200,000 pulses/s), especially in case of violations of normal NPP operation conditions. It should also be noted that the use of a detector based on a $\text{LaBr}_3(\text{Ce})$ crystal made it possible to increase the safety class of the installation to ^3H , which is impossible when using HPG semiconductor detectors. At present, LLC NPP RADICO has completed the manufacture of a prototype of the SGZh-101 unit, which has undergone pilot operation at the Kursk NPP, and has also completed testing and certification procedures for approval of the SI type.

A Monte Carlo model of the active zone of a VVER-1200 reactor with radial and axial homogeneous reflectors has been developed for performing calculations in the MCNP code [1]. The model specifies core profiling by various types of TVS-2M based on a cartogram of a stationary load, which is supposed to be implemented at the Belarusian NPP [2]. A program has been created in the Mathematic package that allows preparing an input file for the MCNP code when calculating the neutron-physical characteristics of the reactor core. The description of the periodic filling of the core with fuel assemblies (FA) and the fuel assemblies themselves with fuel rods in MCNP is performed using a special "lattice language", where the coordinate system is determined (for this core, it is oblique with an angle of 60° between the axes). Profiling is set by specifying the type of fuel assembly for a particular cell. The program developed in the Mathematic package provides the ability to change the number of fuel assemblies loaded into the core by setting the range of changes in the indices of the oblique coordinate system.

2. Material and Methods

One of the main factors ensuring safety in the management of radioactive waste (RW), including their storage, is the reliable determination of such waste accounting characteristics as the radionuclide composition, total activity, specific activity of each radionuclide, RW mass RW subject to certification is fragmented solid

radioactive waste, the main part of which is metal fragments of equipment and metal structures (stainless steel, aluminum alloys) and construction waste (concrete, wood). All wastes have surface contamination with a half-life of 2 years due to the long exposure period. The main contamination is determined by radionuclides Cs-137 and Co-60. Taking into account the heterogeneity of placement and high self-absorption of radioactive waste in such a package, it is necessary to measure it at several points.

The passportizer software has been developed (Program for Upgrading Nuclear Materials Protection, in Russia) which provides: control of the movement of the measuring part along the container, collection and processing of gamma radiation spectra from all detection units of the measuring part of the passportizer, identification of radionuclides and determination of their specific activity, saving measurements in the database and generating a container passport with radioactive waste. In the measuring part of the passportizer, intelligent spectrometric units for detecting gamma radiation BDKG-205m, specially developed at UE "ATOMTEH", are used. Overall dimensions of packing containers: 1320×1032×740 mm. The container is a steel frame lined with 2 mm thick steel sheets. On Figure (1) is a photograph of the packaging container [3, 4, 5].



Figure (1). Packing container.

To calibrate the measuring part of the passportizer and debug the algorithm for determining the specific activity, the instrumental spectra of the BDKG-205m detection unit, which were obtained by Monte Carlo simulation [Monte Carlo Simulation (or Method) is a probabilistic numerical technique used to estimate the outcome of a given, uncertain (stochastic) process. This means it's a method for simulating events that cannot be modelled implicitly. This is usually a case when we have random variables in our processes, using Monte Carlo in Python programming language. Were used, the model developed by Monte Carlo is shown in Figure (2)].

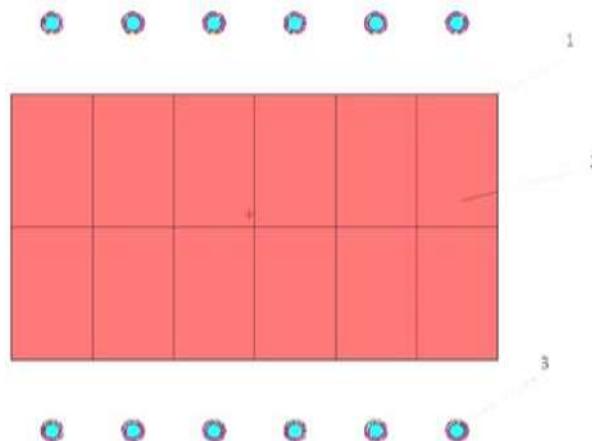


Figure (2). Model developed by Monte Carlo, where 1 is a container; 2 – filling the RW container; 3 – detection units BDKG-205m.

3. Results and Discussion

With the help of modeling, instrumental spectra were obtained for the detection units BDKG-205m with various options for the placement of waste in a container, their composition, mass of waste, average density, and various activity of waste. On Figure 3 shows the model spectra at a container filling density of 0.2 g/cm^3 (A) and 2.0 g/cm^3 (B) with iron and a total container activity of 1 GBq in both cases. In this model, it was assumed that the container is evenly filled with iron and that the contamination is determined only by the Cs-137 radionuclide.

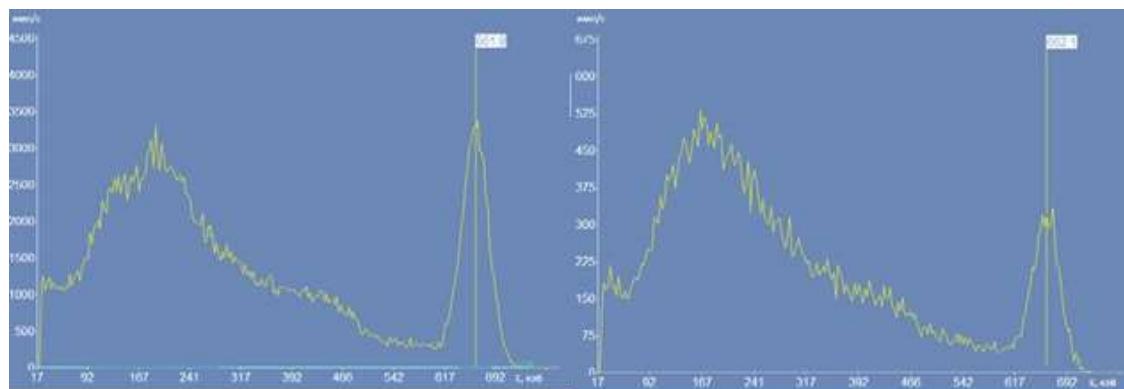


Figure (3). Model spectra at an average iron density of 0.2 g/cm^3 and 2.0 g/cm^3 with uniform contamination with Cs-137 radionuclide.

It can be seen from the spectra that, at a higher average density of iron in the container, the intensity of the Cs-137 line decreased by a factor of 10. The results of the work will be used to create a prototype passportizer and its experimental verification.

In the process of licensing during the construction of a nuclear power plant, in accordance with the law, it is necessary to carry out an examination of the safety documents submitted by the designer. This kind of expertise includes verification independent calculations, for which software tools that are not involved in the preparation of these documents should be used.

To determine the correspondence between the fuel assembly number (see Figure (1)) and the cell element index in the Mathematic file, an appropriate algorithm was created, which makes it possible to automate the process of filling the core of fuel assemblies of various types. The block of material parameters of the environment in the Mathematic file allows you to change the characteristics of materials throughout the MCNP file, effectively describe materials in a data map, and set the position of sources (nuclear fuel). The block of geometric data allows you to change the parameters of the core and reflectors. In the model of a fuel rod/fuel rod and fuel assemblies, the boundary of the upper axial reflector begins at the height of the fuel column, and the material medium is represented as a homogeneous mixture of substances that fill it (see Figure (2)).

Within the framework of this method, the upper reflector is presented in the form of three layers: the first layer extends from the upper boundary of the fuel column to the block of fuel assembly guide channels and has a height of 22.2 cm, the second layer 4.5 cm high occupies the distance from the top plugs of fuel rods/fuel rods to the head FA, the third layer from the FA baffle grid to the beginning of the shell with a lower length of 5.5 cm. In total, the upper reflector occupies 32.2 cm in height. On surface 1 (see Figure (2)), at the boundary of the upper reflector, flows are calculated in two directions, coinciding with the OZ axis. In [3], a group of authors used the same technique to obtain two-group diffusion constants of the corresponding reflectors using the NESSEL cell program and the additional DESET program for the VVER-1000 model (Figure(4, 5)).

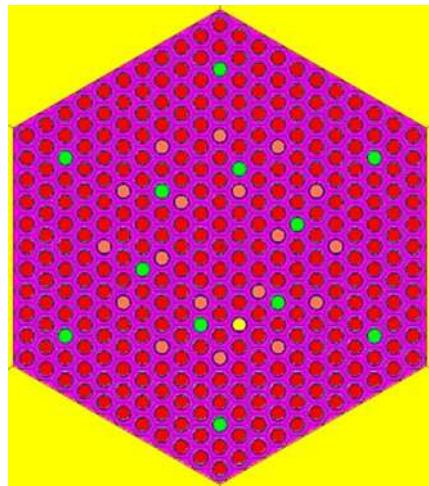


Figure (4). Cross-section by plane xy of fuel assemblies of types Z33Z2 and Z49A2.

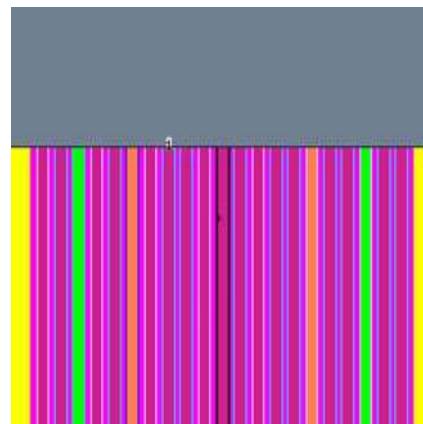


Figure (5). TVS Z40D, surrounded by water and the first layer of the upper axial homogeneous reflector.

4. Conclusions

It should be noted here that in order to calculate the albedo of thermal neutrons, the propagation of fast neutrons into the reflector region should be limited. Due to the fact that in a homogeneous mixture of the reflector, water occupies from 56% to 98.9%, and, therefore, effectively slows down neutrons, the vast majority of fast neutrons crossing surface 1 in the direction of the reflector will cross it in the opposite direction with thermal energies.

The calculation was performed for a model with one fuel assembly loaded into the central cell of the core. The ratio of fluxes passing in the direction of the inner and outer normal to surface 1 for the spectrum of fast neutrons is 0.641, which correlates well with the albedo for VVER-1000 TVSA [3]. The albedo value at the reflectors is used in the DIN3D code, which models the basic reactor behavior under steady state and transient processes after making changes to the system reactivity. In the future, the obtained values can be estimated in the DYN3D program by comparing the results of calculating the calculated field with the restored distribution in the peripheral regions of the core.

Conflict of Interest: The authors declare that there are no conflicts of interest associated with this research project. We have no financial or personal relationships that could potentially bias our work or influence the interpretation of the results.

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