# The research methods of energy release distribution in the IVG.1M Research Reactor after conversion

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**Abstract**

Within the framework of international initiatives on the nuclear weapon non-proliferation in the Republic of Kazakhstan, conversion of research reactors to low-enriched fuel is implemented. This work aims at determining the spatial power distribution of the IVG.1M reactor with low enrichment fuel. The focused attention to this parameter is explained by the fact that energy release (or power density) determines the power, composition of technological systems, and efficiency of a nuclear reactor. In order to assess the change in the parameters of the reactor after conversion and to check the reliability of the simulation, the method of experimental study of the distribution of power density at the stage of physical start-up of the IVG.1M reactor is considered. The results of MCNP6 calculation of the power distribution in fuel elements, fuel assemblies and in technological channels are presented.

***Keywords**:* Energy release; power density; FA; Fuel element; Power; Simulation; Gamma-spectrometry.

# 1.- INTRODUCTION

Currently, research reactors of Kazakhstan are being converted to low-enriched fuel, which entails changes in reactor core designs and the development of new fuel compositions. In this regard, it becomes necessary to assess the change in the operating characteristics of the reactor, in particular, the energy release (power density).

Studies of power distribution can be conducted both experimentally, based on direct and indirect measurements of physical quantities, and by calculation methods using neutron-physical / thermohydraulic calculations.

According to the method of power measuring, the following methods are distinguished:

– methods based on flux density registration (indicator activation, fission chamber, electron emission neutron detectors);

– methods based on measuring the intensity of gamma radiation (spectrometric method);

– methods of thermal control of power in fuel channels or cassettes (thermal balance method).

The most common methods for experimental studies of power distribution are presented in more detail in the next chapter.

## 1.1.- Research methods

At the stage of physical startup, one of the effective experimental methods to study the power density in the fuel assemblies and in the core is the **neutron activation method**. The method is based on the principles of correspondence between the power density and the measured activity of fission products or the activity of indicators (foil, wire). The power density of the reactor is determined by the total number of fissions occurring in the reactor core per pulse or during operation in a static mode. The number of fissions is defined as:

$$N = \Sigma_f V \bar{\phi}_c \qquad (1)$$

where $\Sigma_f$ – macroscopic fission cross section in the reactor core, $m^{-1}$;

V – reactor core volume, $m^3$;

$\bar{\phi}_c$ – average neutron fluence over the core, determined by the activation indicators method, $m^{-2}$.

The results of determining the absolute number of fissions in the indicators are transferred to fuel assemblies, taking into account the spatial correlation between the thermal neutron flux and the specific power density by fuel volume [Giot M. *et al.*, 2017]. NaI crystals and pure germanium semiconductor detectors are used as detectors for measuring gamma radiation activity.

**Gamma-scanning** is used to study the power distribution over the fuel assembly height. The method consists in determining the relative gamma activity of the La-140 fission product, which is proportional to the fission reaction rate in the fuel assembly (FA). It should be noted that the detector counting intensity for each measurement point is determined by the activity of fission products not only in the measured FA section, but also by the activity of these products in its adjacent sections [Izotov, V.V. *et al.*, 2006 , Cherepnin Y. S. *et al.,*1995].

In [Loving J. J. Nath R. J., 1970] it is pointed out that studies of energy distributions in cassettes can be carried out by in-core **miniature fission chambers**. Due to the need to take into account the deformation of the neutron field by the detector, such fission chambers are unsuitable for accurate measurements in contrast to thin activation detectors. Nevertheless, this method is useful for measurements in places where there are no fuel elements.

The use of such miniature detectors as boron counters and fission ionization chambers with U-235, U-238, Pu-239, Pu-240 is described in [Filliatre P., *et al.*, 2008; Geslot B. *et al.*, 2014]. Corrections related to the change in the neutron spectrum by the chamber shell are 1÷3%.

Measurements of only a part of irradiated fuel elements and fuel assemblies under study can reduce the time and labor costs of research. The values of power in the remaining fuel assemblies are reconstructed by **numerical methods**. The most used methods are least squares and regression analysis [Cherepnin Y. S. *et al.,*1995]. The accuracy of the reconstructed power distribution depends on the number and method of placing the experimental node points, the type and number of approximating functions, and the measurement errors of the gamma activity of fuel elements.

In [Mener H., 1977], the energy release (power density) in fuel cassettes is calculated using two-dimensional interpolation under the assumption that the distribution of energy release over the section of the cassette is represented in the form of two components:

1. Component characterizing the type, internal properties of the cassette;
2. Component that takes into account the influence of the macrodistribution of energy release in the reactor.

As a solution, a modified zero-order Bessel function of the first kind was proposed, which, during interpolation, is refined using a term that quadratically depends on the distance R of the fuel element to the cassette axis. As a result, the power distribution is as follows:

$$\beta(x,y) = P_0 + P_1 R + P_2 R^2 + P_3(x \cdot \cos P_4 + y \cdot \sin P_4), \qquad (2)$$

where $R = \sqrt{x^2 + y^2}$;

x,y - fuel element coordinates;

$P_0$–$P_4$ – parameters determined by the least squares method in an iterative process.

The root-mean-square deviation of the calculated values from the experimental data is ~ 2.1%.

The paper [Walter L. *et al.,* 1976] describes a technique for measuring radial power distribution by a non-stationary temperature method. The temperature field is monitored by 14 thermocouples placed at selected points along the radius of the fuel elements.

The authors of [Birri A *et al.,* 2020] argue that the power distribution in the reactor core can be determined using the response of an array of **optical fiber gamma thermometers**.

It should be noted that at present, methods of three-dimensional monitoring of reactor power, based on an **artificial neural network**, are being intensively developed [Xia H. *et al*., 2014]. In [Pirouzmand A., Morteza K. D., 2015], to predict in real time the relative spatial power distribution of VVER-1000 reactor, the neutron network was trained by obtaining calculated and experimental data on 200 reactor operating states with different power density distributions at different control rod positions. In [Peng X. *et al.*, 2018], a Bayesian inference was used to reconstruct the core power distribution based on neutron-physical modeling and measurements of in-core neutron detectors. However, in the study of [Peng X. *et al.*, 2018], it is applicable only to the radial power distribution, and the axial power distribution can be reconstructed using the cubic spline synthesis method.

The purpose of this work was to determine the spatial power distribution of the IVG.1M reactor with low enrichment fuel.

# 2.- METHODS

### 2.1.- THE EXPERIMENTAL METHOD OF RESEARCHING POWER DENSITY YIELD USED AT THE IVG.1M REACTOR

The IVG.1M research reactor is a heterogeneous thermal neutron reactor with a light water coolant and moderator. Beryllium is used as a neutron reflector. The reactor core contains 30 water-cooled process channels, each containing a fuel assembly. Reactor conversion allows the reactor design to remain unchanged. Computational studies to justify the thermophysical characteristics during the conversion of the IVG.1M reactor core to low-enriched fuel [Prozorova I.V., *et al.*, 2013] showed that the thermal state of the reactor practically does not depend on the fuel enrichment with the same design features of the fuel assemblies. This means that there is no need to significantly revise the methods for determining the power distribution in the core and fuel assemblies at the stage of the physical start-up of the reactor. However, it should be noted that during the 30-year operation of the IVG.1M reactor, new methods of studying power distributions and numerical methods for reconstructing power density field have appeared, which can be useful in improving the existing technique.

At the IVG.1M reactor, the method used to study the power density yield over the volume of fuel assemblies is the neutron-activation method using the mockup of technological channels, in which they are installed:

– activation detectors (copper and gold wires) for studying the axial power distribution;

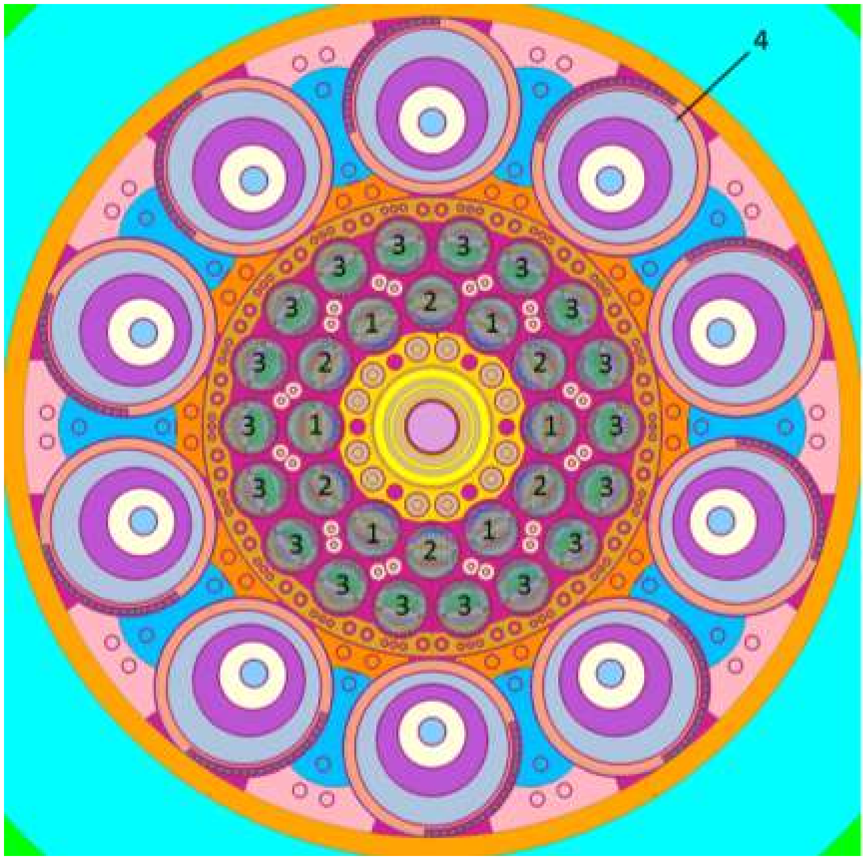
– fuel element motitors for studying radial power distributions.

The material composition of the mockups corresponds to the composition of the studied technological channels. The fission products formed during irradiation are distributed in the materials of the mockups in proportion to the density of the number of fissions of uranium-235. To increase the accuracy of determining the power density, the fuel element monitors are pre-calibrated for uranium-235 concentration.

It should be noted that remains one of the reliable methods for obtaining experimental information used for verifying computer codes and for direct experimental assessment of the efficiency of fuel assemblies [Cherepnin Y. S. *et al.,*1995].

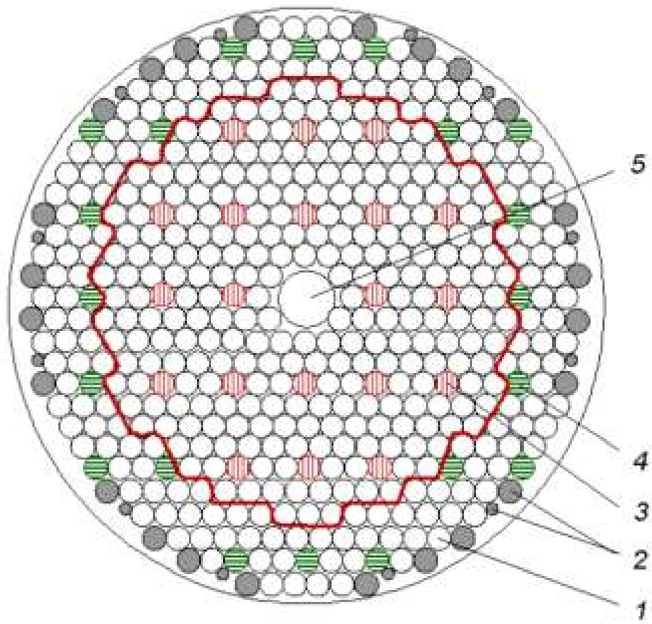
The symmetrical arrangement of water-cooled channels in relation to the center of the core and to the control drums makes it possible to determine the power density of the whole core by investigating only one of the channels of each row of the core (Figure 1).

Fuel element monitors are placed in the measuring section of the mockup according to Figure 2.




1 – first row, 2 – second row, 3 – third row, 4 – control drums

Figure 1 – IVG.1M reactor core




1 – fuel element; 2 – filling aggregate; 3 – fuel element monitor of the central profiling zone; 4 – fuel element monitor of the peripheral profiling zone; 5 – zirconium central rod

Figure 2 – Placement of fuel element monitors

# 3.- RESULTS AND DISCUSSION

## 3.1.- Results of simulation of power distribution in the IVG.1M reactor

Calculation through simulation is one of the most effective methods for the study of complex systems for any field of engineering sciences. The most accurate spatial energy distribution of neutrons and power density can be obtained by programs based on the Monte Carlo method. To calculate the power density field, it is necessary to know the distribution of nuclear fission reaction rates (or neutron flux density) and the energy released during fission. In estimations, the total fission energy is usually assumed to be 200 MeV. In more accurate calculations, it should be taken into account that the energy yield of fission can vary by up to 3%, depending on the fissile isotopes in the fuel and the energy of the fissile neutrons [Ma X.B. *et. al*., 2013, Pirouzmand A., Morteza K. D., 2015].

The calculation of the power distribution in the technological channels (fuel assembly) of the IVG.1 M reactor was carried out in the MCNP6 using a model of a reactor [Irkimbekov R.A *et al*.,2019] with low enriched uranium fuel. The reactor temperature was set to 21°C and the position of the control drums was set to 81° (criticality position).

Figure 3 shows the linear power distribution of the fuel element (FE) with maximum load at 1 MW. The zero mark corresponds to the lower end of the channel. The maximum value of the linear power of a fuel element is 0.21 W/mm.

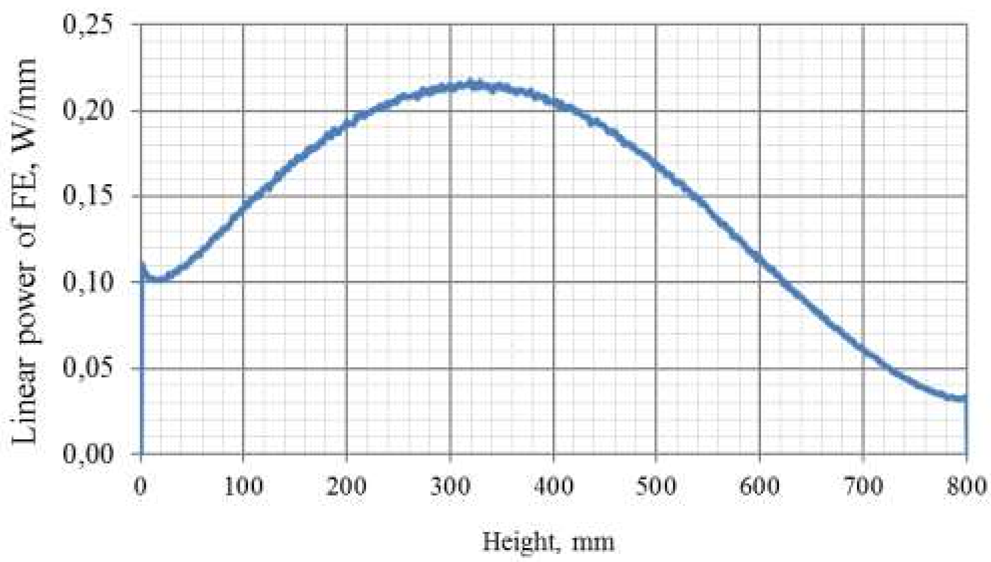



Figure 3 – Linear power distribution of fuel element

Figure 4 shows the steady-state spatial distribution of the linear power for the first, second, and third row of water-cooled technological channels for the same reactor power.

According to Figure 4, the power of the technological channel of the third row, which has a length of 600 mm, is lower, in contrast to the first two channels with a length of 800 mm. Since the reactor power is directly proportional to the average neutron density in the fuel volume in its core, the power distribution also obeys Bessel's law: the reactor power decreases from the “central” first row to the “peripheral” third row. The maximum values of the linear power for technological channels of 1, 2 and 3 rows are 69, 66 and 56 W/mm, respectively.

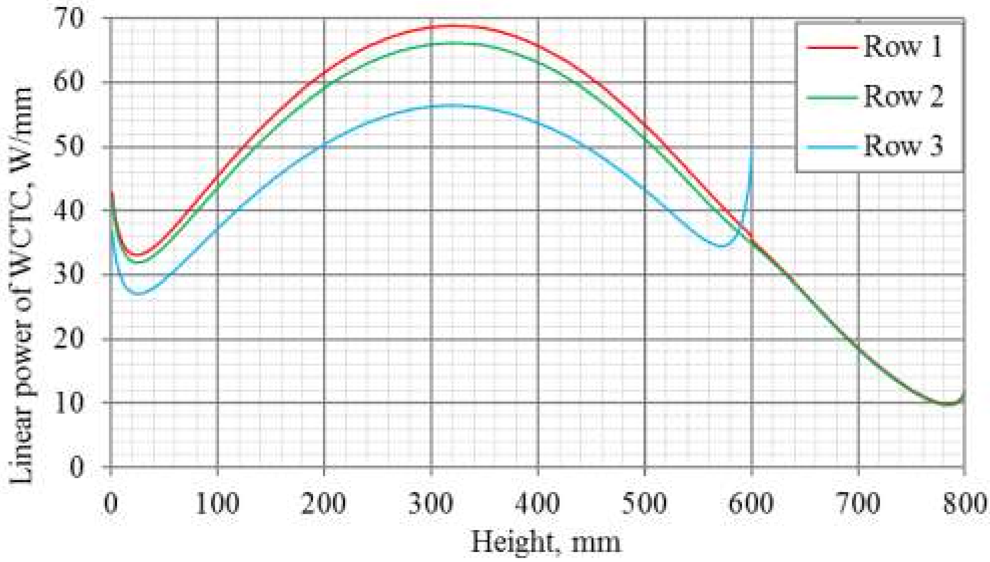
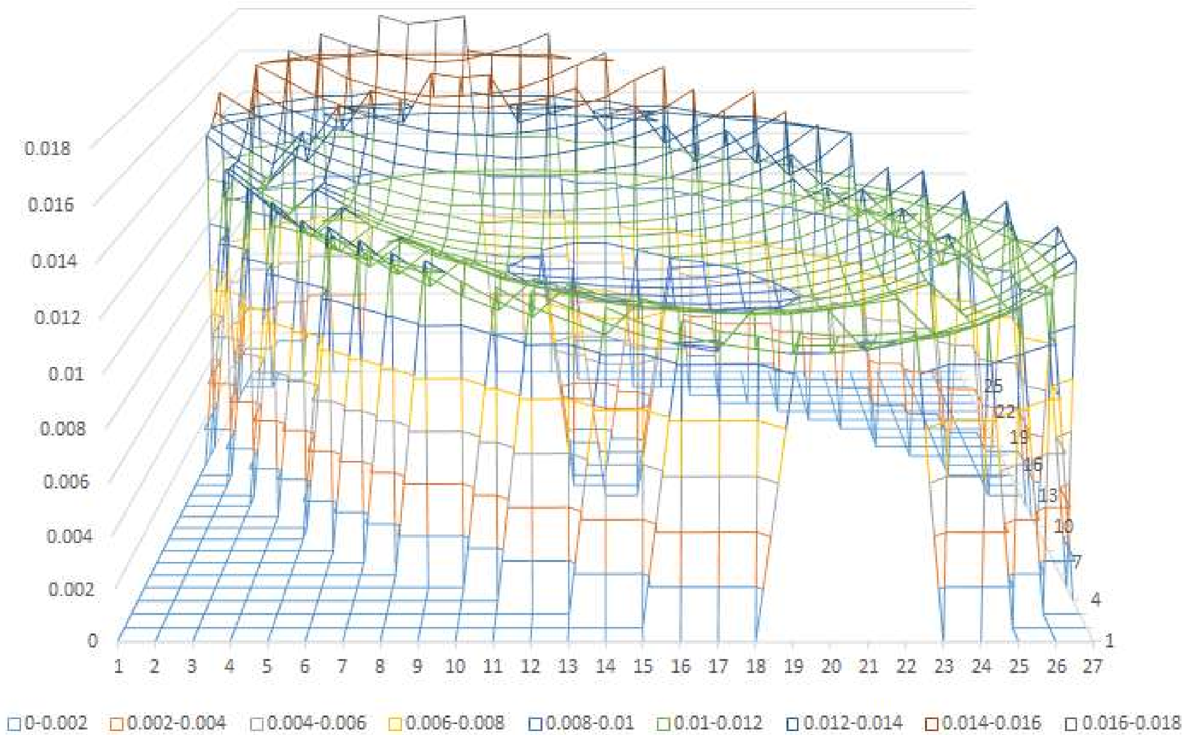



Figure 4 – Stationary linear power distribution of the technological channel

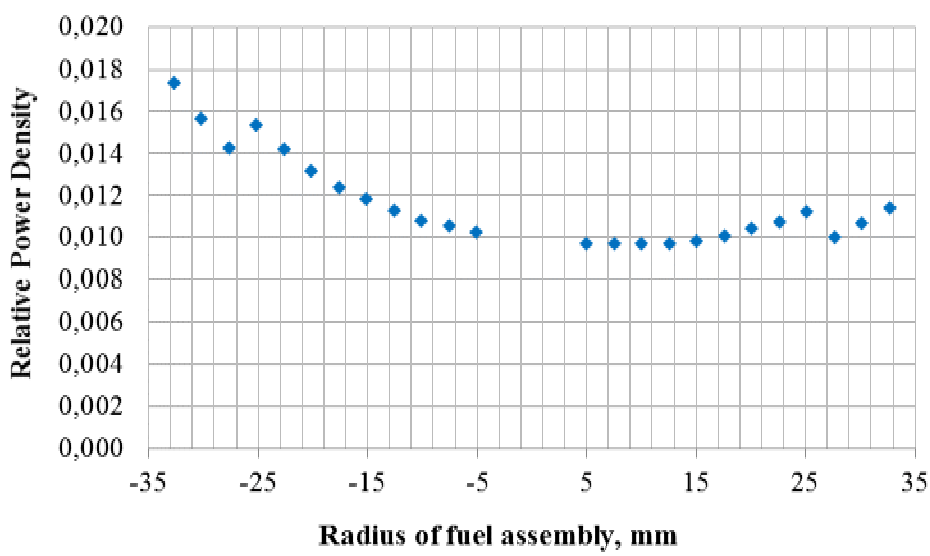
Figure 5 shows the relative radial power distribution for the fuel assembly in the first-row channel. The power density was calculated for all 468 fuel rods in the fuel assembly. During the calculation, the following reactor configuration was specified:

– assembly 72.000 (loop channel) is installed in the central experimental channel;

– all reactivity compensation rods are inserted into the reactor core.




a)




b)

Figure 5 – Relative radial power distribution of the fuel assembly

It should be noted that the drops in the power density at a fuel assembly radius of approximately 26 mm are due to the change in the profiling zones. The larger values of power density on one side (-35 to -5 mm in the fig.5) are explained by its closer location to the center of the core. The lack of power in the center is due to the location of the zirconium rod in the center of fuel assembly.

The power peaking factor along the radius of the fuel assembly is 1.62.

# 5.- CONCLUSIONS

In order to assess changes in the parameters of the reactor after conversion and to check the reliability of the simulation, the experimental method of study of the power distribution at the stage of physical start-up of the IVG.1M reactor is considered. The results of MCNP6 calculations of the axial power distribution in the fuel element and technological channels of the three rows of the IVG.1M reactor and the radial power distribution for fuel assembly are presented.

Analysis of the calculation results shows that the maximum linear power value of the fuel element is 0.21 W/mm, while for technological channels 1,2 and 3 rows this value is 69, 66 and 56 W/mm, respectively. The power peaking factor along the radius of the fuel assembly is 1.62.

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