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Numerical simulation of coolant flow in the IBR-2M pulsed reactor fuel assembly

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Contents

Abstract

The purpose of this work is to carry out numerical simulation of the coolant flow in the IBR-2M fast neutron reactor fuel assembly.

The research included following problems: familiarization with the ANSYS software package; determination of assumptions and approximations for the calculation model; calculation model development; simulation of the normal operation conditions; performing additional evaluation calculations for validation of the principles and approaches, used in the calculation model creation; analysis of the obtained results.

To solve the technical problems, a theoretical study was carried out using computation methods and mathematical models of the ANSYS software package.

Passport values of the IBR-2M reactor core parameters were used as input data.

As a result, conservative evaluations of the thermal-hydraulic characteristics of the system, using a three-dimensional mathematical model, are obtained.

Introduction

Currently, one of the important world nuclear community activity aspects is to ensure failure free operation of existing nuclear facilities. Achieving the required reliability level is impossible without having a deep knowledge of the processes, occurring in the nuclear reactor core. Such processes should include the coolant flow through the core, as it is accompanied by the presence of phenomena that cannot be fully taken into account using standard approaches. In this case, there is a need to create more accurate calculation models, based on numerical methods and to evaluate thermal-hydraulic installation parameters with different visualization degrees.

With the development of information technologies, computer methods are being improved, allowing to simulate various processes during the operation of the installation in normal and abnormal operation conditions, including design and non-design basis accidents. Today, there are many methods and programs for the calculation of nuclear reactors, one of which is ANSYS Discovery Live Student software package (hereinafter – ANSYS SP), used in this work.

In this paper, the evaluations of the IBR-2M reactor fuel assembly's thermal-hydraulic parameters were carry out using the average power approximation, the results of which can be used as input data for further studies of processes, occurring in the nuclear reactor core. For these purposes, the geometry of the reactor installation was analyzed, a list of the necessary technological equipment, taken into account in the model, was selected, assumptions and approximations for the calculation model creation were determined, and a calculation model was developed, using the ANSYS SP. As a result, conservative evaluations of the thermal-hydraulic parameters of the system, characterizing the reactor installation normal operation conditions, are obtained.

1. General characteristic of the IBR-2M reactor

1.1 General description of the IBR-2M reactor installation

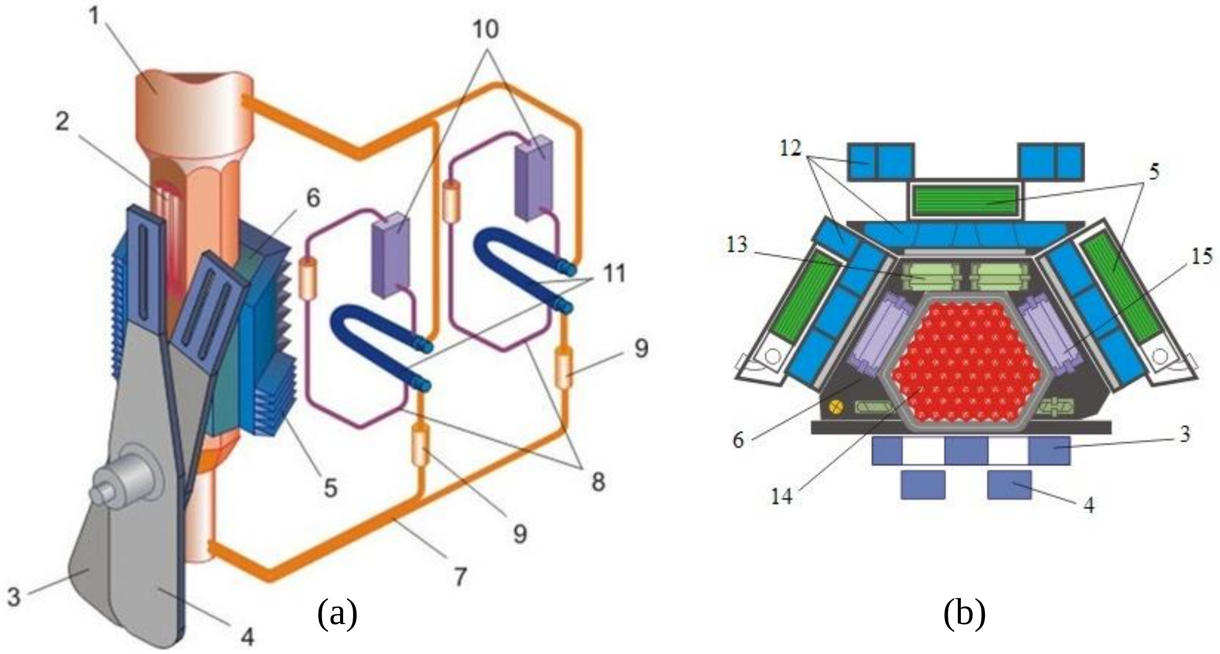
The IBR-2M research nuclear reactor is a pulsed fast reactor, which is intended for use as a source of neutrons for beam researches in various scientific fields. The reactor has a three-circuit loop composition and a special unit for generating neutron flashes – a reactivity modulator. The reactivity modulator is made in the form of two coaxially arranged steel rotors, rotating at different speeds in a plane parallel to one of the hexagonal reactor vessel's six sides.

The physical start-up of the IBR-2M reactor was carried out from December 10, 2010 to June 10, 2011. The IBR-2M physical start-up program included experiments to determine the actual values of the core critical parameters, the efficiency of the control elements, the reactor monitoring and protection, the duration of power pulses and the value of their amplitudes fluctuation, as well as a number of other physical reactor characteristics, required to confirm its design capabilities and to determine the safe operation limits. As a result of the IBR-2M physical start-up, the critical load of the reactor core was 64 fuel assemblies. The five cells remaining unloaded during operation, will ensure the reactor operation for at least 20 years. The efficiency of the reactor control and protection system (CPS) working elements and the speed of the emergency protection units meet the requirements of nuclear safety. Accidents imitation on disconnection of various technological systems, ensuring the normal operation of the reactor, showed that the reactivity insertion rate in all cases is significantly less than the permissible value and the emergency protection units in any situation will reduce the power during the time between pulses to a safe level [1].

Technological systems of the reactor during the physical start-up worked without any problems. The shape of the fast neutron pulse is close to the Gaussian one with the duration (220 ± 3) μs. Standard values of pulses energy fluctuation do not exceed 5 %. The work was performed in the JINR Frank Laboratory of neutron physics.

The reactor building, its biological protection and technological systems of the reactor are preserved. Thus, after the modernization, a completely new IBR-2M reactor was created, operating as part of the IBR-2 research nuclear installation.

The composition of the IBR-2M reactor installation is shown on figure 1.



(a) (b)

Figure 1 – The IBR-2M reactor installation composition:
a – general view of the IBR-2M reactor installation; b – IBR-2M reactor core cross-section; 1 – reactor vessel; 2 – core; 3 – main movable reflector (OPO); 4 – additional movable reflector (DPO); 5 – water comb-shaped moderators; 6 – stationary reflector; 7 – I cooling loop; 8 – II cooling loop; 9 – circulation pump; 10 – air heat exchangers; 11 – intermediate heat exchangers; 12 – water moderators; 13 – scram units; 14 – fuel assemblies; 15 – control units

The IBR-2M reactor works in the following operating regimes [2]:

– pulsed operation at a power of up to 2 MW with a pulse frequency of 5 Hz;

– stationary operation at a power up to 100 W without rotating reactivity modulator with the operation of the ASUZ-12P system in the "критсборка" regime.

The stationary operation regime of the reactor is designed for nuclear fuel loading, reloading and unloading from the reactor core, as well as for the efficiency evaluating of reactivity modulators, control elements and reactor shielding.

The IBR-2M reactor with a designed average power of 2 MW differs from the IBR-2 primarily in the smaller core size and the absence of a central channel. Cooling of the reactor core is provided by sodium liquid metal coolant. Circulation of the coolant is provided by electromagnetic pumps.

To create neutron pulses, a reactivity modulator is used, which consists of a main (OPO) and an additional (DPO) movable reflectors rotating in opposite directions at a speed of 600 rpm (OPO) and 300 rpm (DPO). The rotor of the OPO is a blade with three "teeth" made of a nickel alloy, and the DPO is a blade with two "teeth" also made of a nickel alloy. The level of IBR-2M reactivity is regulated by the CPS elements, which are movable tungsten blocks in the matrix of stationary steel reflectors: control elements CO – 2 pieces, scram units AZ – 2 pieces, the unit of the manual control (intermediate) regulator PP and the element of the automatic control regulator AR.

The reactor vessel, equipped with pressure and drain tubes, is designed for placement of fuel assemblies in it, technological operations implementation and organization of the primary coolant flow through the core.

Reactor core cooling system is three-circuit, two-loop (figure 1). Each loop is designed to take off 50 % of power.

1.2 Brief description of IBR-2M reactor fuel element

The IBR-2M reactor fuel assembly contains 7 fuel elements, the geometric configuration of which is shown on figure 2. As a fuel there are sleeve-type pellets (with a hole in the center) of sintered plutonium dioxide, the density of which is 10.1 g/cm^3. The outer diameter of the nuclear fuel rod is 7.5 mm, and the diameter of the inner hole is 1.5 mm. The height of the nuclear fuel rod is 444 mm. The pellets are placed in a stainless steel cladding and from above, through the plate, are pressed by a spring. The internal volume of fuel element is filled with helium. The height of the gas gap is 240 mm. Fuel element diameter is 8.6 mm. [3].

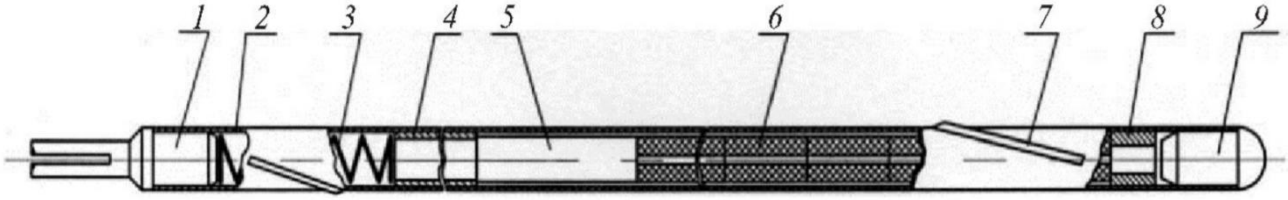


Figure 2 – The IBR-2M reactor fuel element:
1 – upper tip; 2 – cladding; 3 – spring; 4 – insert; 5 – reflector; 6 – fuel pellets;
7 – wire; 8 – sleeve; 9 – plug

2 Calculation methods

To perform these tasks, a calculation thermal-hydraulic model of the IBR-2M reactor fuel assembly was developed using the modern ANSYS SP.

ANSYS SP is a new interdisciplinary platform that allows engineers to create virtual prototypes of their designs directly from solid models almost instantly. The developer of this software package is Ansys Corporation. ANSYS SP was released in September 2017.

The ANSYS SP allows you to numerically solve stationary and non-stationary thermal, strength, aero-and hydrodynamic problems in an interactive mode using the finite-difference method. In this case, for the calculation of the expert it is enough to import a geometric model of the structure under study, to perform incomplete model parametrization (if the analysis of the different design options behavior is necessary), using the advantages of direct simulation, as well as to set the boundary conditions. The calculation is performed in real time, and the user receives instant feedback in the form of a response, including an interactive change in the geometry of the model [4].

Software features allow you to give fine control over the calculation model, set the properties of materials and related parameters, and allows you to customize the visualization, change the detail and display quality of the model. In this paper there were used the highest possible level of model detail, which can lead to a significant increase in the calculation time.

An evaluative calculation of the fuel assembly’s thermal-hydraulic parameters, which are additionally carried out, is used for comparison with the results of numerical simulation and allows to estimate the probable errors of the calculation model, as well as to make assertions about its further application.

3 Thermophysical calculation of the IBR-2M reactor fuel assembly

3.1 Estimative calculation of thermal-hydraulic parameters

The estimative thermal-hydraulic calculation was carried out for the IBR-2M reactor fuel assembly, based on the data, presented in tables 1 and 2.

Table 1 – Thermophysical parameters of liquid sodium at $t_{in} = 290°C$ [5]

Name	Notation, units	Value
The density of liquid sodium	ρ_{Na}, kg/m^3	883.8
Heat capacity of liquid sodium	C_p, J/(kg$\cdot$ $°C$)	1308
Kinematic viscosity of liquid sodium	ν, m^2/s	$0.396 \cdot 10^{-6}$

Table 2 – Thermal-hydraulic parameters of the IBR-2M reactor fuel assembly [2, 8]

Name	Notation, units	Value
The height of the fuel assembly	H, m	1.25
The coolant temperature at the inlet to the fuel assembly	T_{in}, $°C$	290
Average coolant heat drop in the fuel assembly	Δt, $°C$	65
The turn-key size of the fuel assembly	B, mm	26.2
The power of the fuel assembly	N, MW	2
The total number of the fuel assemblies	$n_{\text{твс}}$	69
Number of fuel elements	$n_{\text{fuel elements}}$, штук	7
Number of fuel displacers	$n_{\text{fuel displacers}}$, штук	6
The diameter of the fuel element	d, mm	8.6
Fuel rod pitch	x	1.062
Fuel rod wire pitch	s, m	0.1

According to [6], the evaluation of the pressure drop in the fuel assembly includes consideration of friction resistance (Δp_{fr}), local resistance (Δp_m), acceleration pressure losses (Δp_{acc}), levelling pressure losses (Δp_{lev}) and is determined by the following formula:

$$\Delta p = \sum \Delta p_{fr} + \sum \Delta p_m \pm \Delta p_{acc} \pm \Delta p_{lev}, \tag{1}$$

After performing simple mathematical transformations of the formulas given in [6], the expression for estimating the pressure drop in the fuel assembly will take the following form:

$$\Delta p = \rho_{Na} \cdot [w^2 \cdot (\frac{\xi \cdot l}{2 \cdot d_h} + \frac{\zeta_m}{2} + \frac{\rho_{in} - \rho_{out}}{\rho_{out} \cdot \rho_{in}}) + g \cdot H], \tag{2}$$

where w – the coolant speed through the fuel assembly of the core, m/s;

l – the length of the fuel assembly, in this case it is equal to the height of the fuel assembly, m;

d_h – equivalent or hydraulic diameter, m;

ζ_m – local loss coefficient, in this case it is equal to 1;

ρ_{in} – coolant density at the inlet of the fuel assembly, kg/m^3;

ρ_{out} – coolant density at the outlet of the fuel assembly, which is equal to the density of the liquid sodium at $t_{out} = 355°C$, i. e., $\rho_{out} = 869.7$, kg/m^3 [5];

ξ – hydraulic resistance coefficient.

The coefficient of resistance ξ is determined with an error ±15 % for the rods beam of fuel elements located in the fuel assembly on a triangular mesh under the condition of touching the "edge on the rod" type according to the following formula [6].

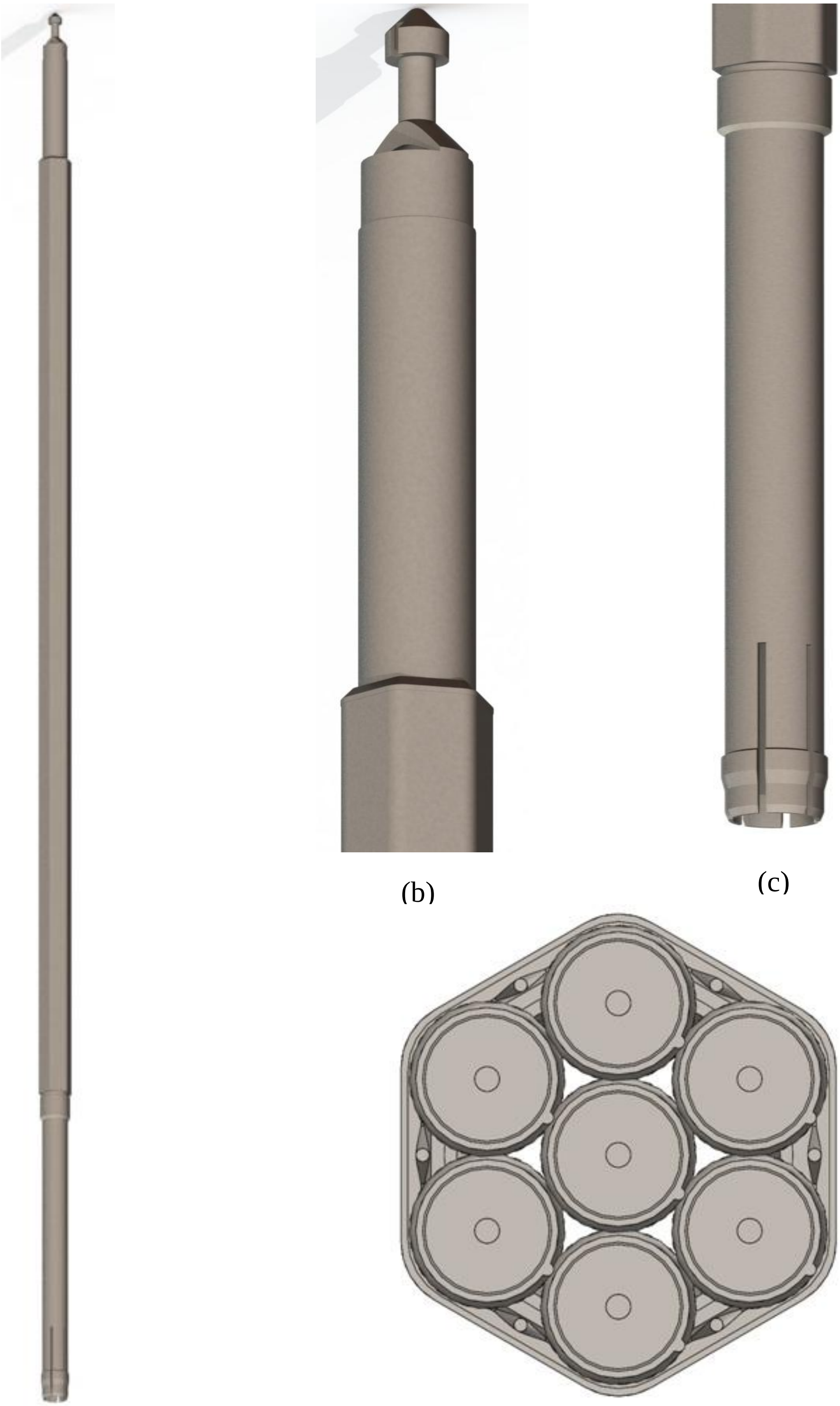
$$\xi = \frac{0.21}{\mathrm{Re}^{0.25}} \cdot \left\{ 1 + \frac{1.24}{(s/d)^{0.25}} \cdot [1.78 + 1.485 \cdot (x-1)] \cdot (x-1)^{0.32} \right\}. \qquad (3)$$

Using the data given in tables 1 and 2, it is possible to estimate the pressure drop in fuel assembly by the formula (2):

$$\Delta p = 71747.37 \pm 9306.28 \text{ Pa} \approx 71.75 \pm 9.31 \text{ kPa}.$$

3.2 Description of the IBR-2M reactor fuel assembly calculation model

With the use of the ANSYS PS [4], a calculation model of the IBR-2M reactor fuel assembly was developed, which includes fuel elements, a displacer, fuel assembly cask, a support collet, a backlate, a bottom, upper and bottom fuel assembly nozzles. In turn, the fuel elements include a nuclear fuel rod, a fuel elements cladding, a support wire, an upper tip and a lower plug. Also, the fuel assembly model of the IBR-2M reactor includes a gas gap with thermophysical parameters of helium. Thus, the calculated thermal-hydraulic model of the IBR-2M reactor fuel assembly is composed of many smaller parts using the "Сборка" mode and does not take into account the upper cylindrical bush-reflector, as well as the spring of the nuclear fuel rod. The calculation model of the IBR-2M reactor fuel assembly is shown on figure 3.



(a) (b) (c) (d)

Figure 3 – The calculation model of the IBR-2M reactor fuel assembly:
a – general view; b – upper nozzle; c – bottom nozzle; d – cross-section at the core center

In addition, the space between the fuel assembly cans, as well as the thermal and physical characteristics of the coolant and structural materials, are taken into account in the development of the calculation thermal-hydraulic model of the fuel assembly. According to the description, presented in section 1.1, the IBR-2M reactor is a pulsed fast research reactor, and also has a special unit for generating neutron flashes – a reactivity modulator. However, these features of the reactor installation will not be taken into account in this work due to the limited field of ANSYS PS use and the high complexity of the initial data. In other words, the numerical simulation of fuel assembly is carried out using the average power approximation of the reactor core.

Numerical simulation of the IBR-2M reactor normal operation was carried out using the boundary conditions presented in table 3.

Таблица 3 – Boundary conditions for numerical simulation

Name, units	Value
Coolant flow through the fuel assembly, m^3/h	100
Pressure at the outlet of the fuel assembly, kPa	136
Coolant temperature at the inlet of the fuel assembly, $°C$	290
Structural material	Stainless steel

In accordance with the requirements of paragraph 2.2.1.2 NP-048-03 [7], the design of fuel assembly and fuel elements of the IBR-2M reactor during normal and abnormal operation conditions, including design accidents, must ensure that the relevant limits, such as the maximum permissible temperatures of the fuel elements cladding and fuel, taking into account shock and vibration effects, thermal-cycling loading, fatigue and other factors that worsen the mechanical characteristics of the core materials and the integrity of the fuel elements cladding, are not exceeded.

3.3 Results of numerical simulation

Numerical simulation of the coolant behavior was carried out using the developed calculation model of the IBR-2M reactor fuel assembly. For these purposes, a thermal-hydraulic problem, solved by the finite-difference method, was formulated and set. During the numerical simulation the maximum possible density of the computation mesh was used. As a result of the simulation, the pressure distribution, temperature and velocity fields, appearing in the fuel assembly and characterizing the behavior of the coolant, are estimated. Figures 4, 5 and 6, 7 show the evaluations of the coolant velocity field and the pressure distribution in the fuel assembly, respectively.

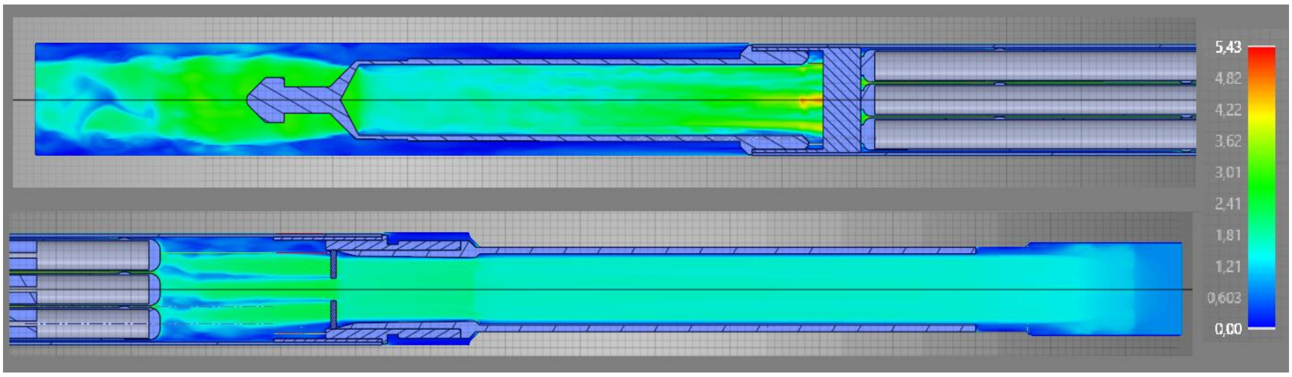


Figure 4 – Coolant velocity field in the longitudinal cross-section of the bottom (from the bottom) and upper (from the top) nozzles

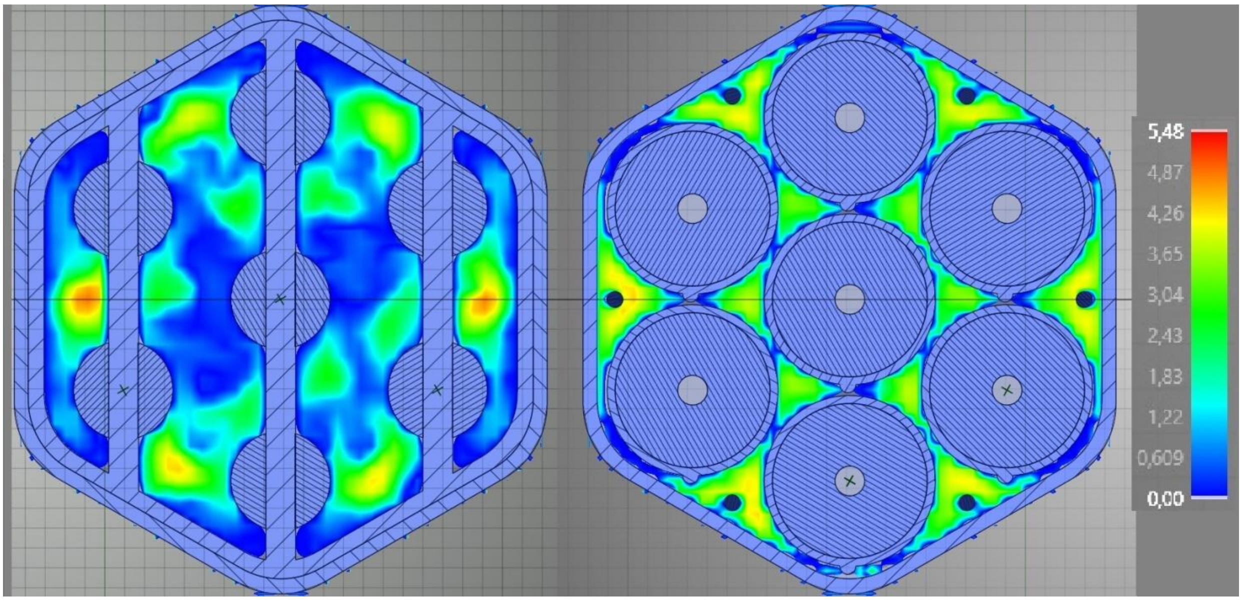


Figure 5 – Coolant velocity field in the cross-section on the top (on the left) and on the bottom (on the right) views

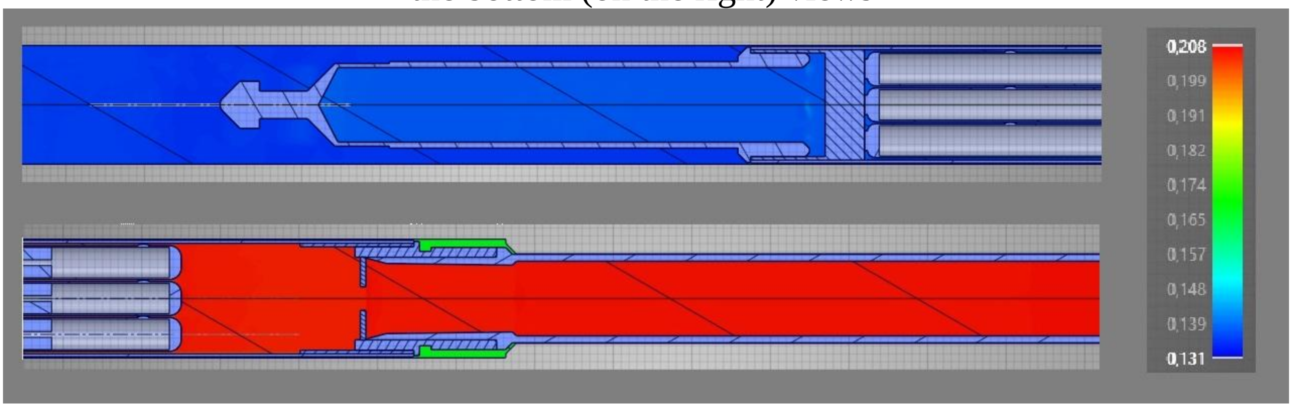


Figure 6 – Pressure distribution in the longitudinal cross-section of the bottom (from the bottom) and upper (from the top) nozzles

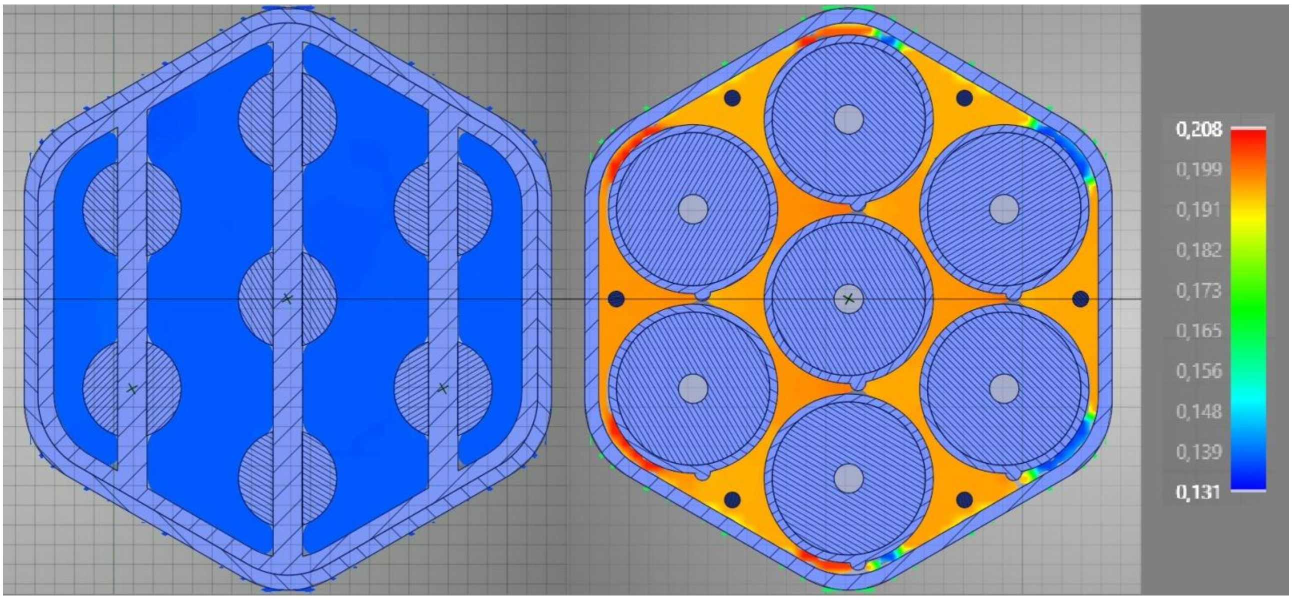


Figure 7 – Pressure distribution in the cross-section on the top (on the left) and on the bottom (on the right) views

As it can be seen on figure 4, the area of the bottom nozzle is characterized by slightly varying properties of the coolant, due to the constant flow cross-section. The velocity profile in this section has a classical form, and the average velocity value is set at 1.6 m/s. In the area under the fuel elements (figure 4), when the coolant passes through the bottom of the fuel assembly cask, the coolant flow and velocity profile are redistributed to seven clear flows due to the changed flow cross-section. In this section of fuel assembly, the average velocity value increased to 2.1 m/s. The velocity reaches its maximum value of 5.48 m/s when the coolant passes through the core of the fuel assembly in the area of displacers (figures 4 and 5).

As it can be seen on figure 6, the maximum pressure value, equals to 0.208 MPa, is achieved in the bottom nozzle area and specified by indicating the weight flow as the boundary condition at the entrance to the fuel assembly model, while the minimum value of the pressure is determined by the boundary condition at the outlet of fuel assembly model. The resulting pressure drop is described by various losses, additionally estimated in section 3.1.

Also, the temperature distribution evaluations in the IBR-2M reactor fuel assembly were obtained, which are presented on figures 8 and 9.

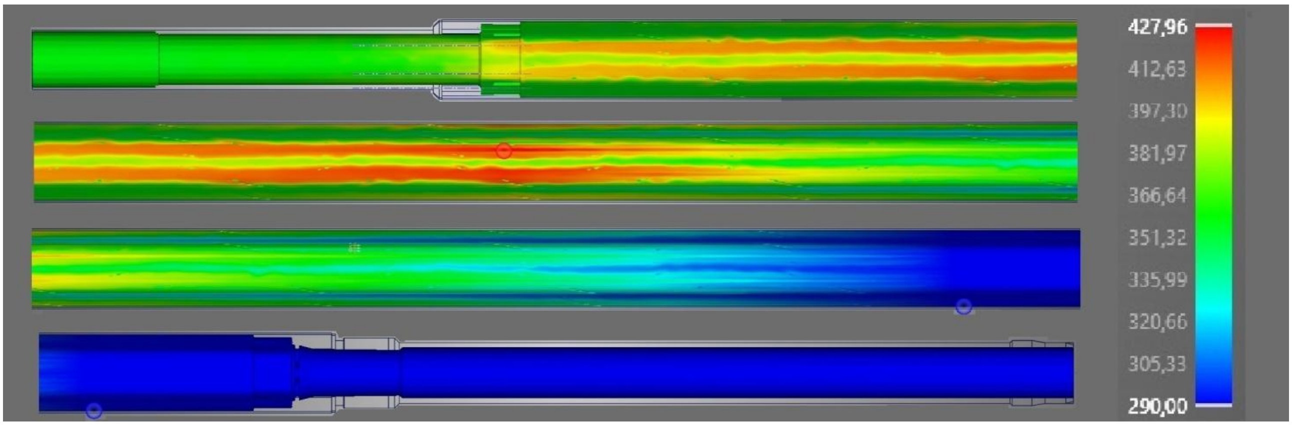


Figure 8 – Temperature distribution in the longitudinal cross-section of the fuel assembly

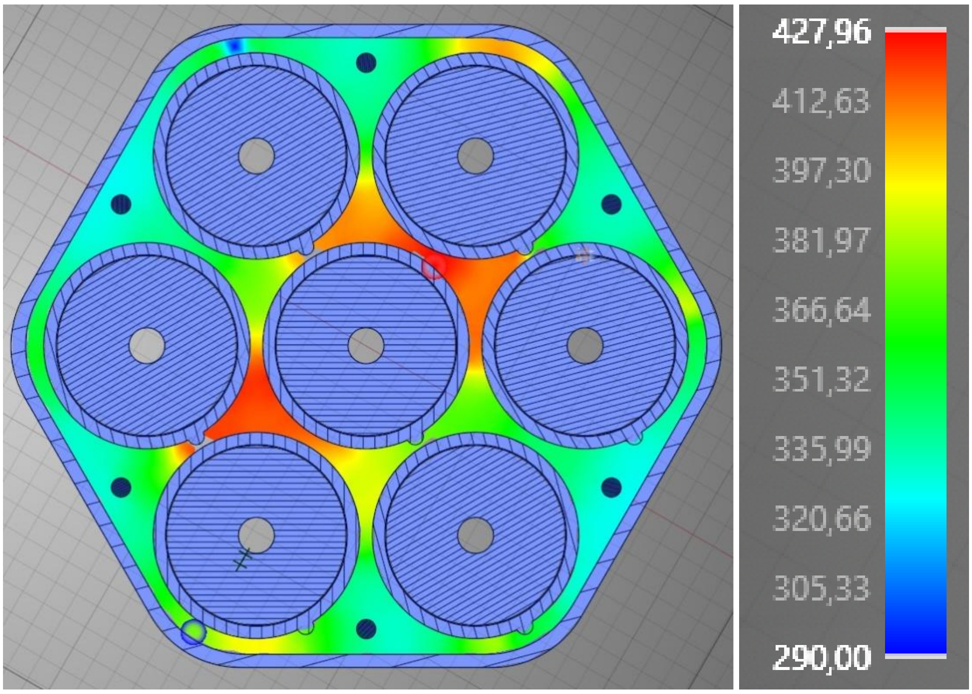


Figure 9 – Temperature distribution in the cross-section of the fuel assembly

Figures 8 and 9 show that the average temperature of the coolant at the outlet of the fuel assembly is 353 $°C$, with heat drop through the reactor core, which is equal to 63 $°C$. The maximum temperature, equals to 427.96 $°C$, is achieved in a limited area of the upper part of the central fuel assembly, and the average temperature in the fuel assembly vessel is set at 323 $°C$.

The results presented on figures 6, 7 and 9, among other things, demonstrate the influence of the support wire on the coolant flow regime through the fuel assembly, instigating the turbulence of the flow, the effects of which are not considered in this paper.

With the help of standard methods of mathematical statistics, the root mean square deviation (RMSD) of the required value, caused by the deviation of each term from its mean value, is determined. The mean value can be found by the following formula [9]:

$$\overline{\Delta p} = \frac{\Delta p_{eval} + \Delta p_{calc}}{2} = 71.87 \pm 0.02 \text{ kPa.} \quad (4)$$

The values of the pressure drop in fuel assembly, obtained as a result of evaluation and numerical simulation, have a good agreement with the passport values, which allows us to make assertions about the correctness of the used assumptions and the applicability of the developed calculation model for further research.

Conclusion

The object of study in this paper is considered fuel assembly of the IBR-2M research reactor of the JINR Frank Laboratory of neutron physics, as well as a general description of the reactor installation, including the thermal-hydraulic characteristics of fuel assembly.

As a result of this work, the ANSYS PS was mastered, with the help of which, a calculation thermal-hydraulic model of the IBR-2M reactor fuel assembly was developed. In addition, the evaluative calculations of the thermal-hydraulic characteristics of the fuel assembly were carried out, the results of which were used for comparison with the calculations of numerical simulation.

The paper presents the evaluations of pressure distribution, temperature distribution and velocity field in fuel assembly, obtained by numerical simulation of the developed calculation model using the finite-difference method.

It is shown that the scheme with the use of three-dimensional numerical simulation for the evaluation of thermal-hydraulic characteristics, proposed in this paper, allows to obtain a better and informative understanding of the coolant flow process, which is not achievable using standard methods and approaches. As a result, it is recommended to use this model for further research.

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