# Experimental AND CALCULATION INVESTIGATIONS OF HYDRODYNAMICS AND HEAT EXCHANGE IN LIQUID METAL TURBULENT FLOWS IN FAST REACTOR FUEL ASSEMBLIES

Yu.A. KUZINA. A.P. SOROKIN, N.A. DENISOVA

IPPE JSC

Obninsk, Russian Federation

Email: ukyzina@ippe.ru

**Abstract**

The formation of velocity and temperature fields in the fuel assemblies of fast reactor core occurs under the influence various factors. It is shown, that the most important factors include a complex multiply connected geometry, subject to deformation during the campaign under the influence of temperature non-uniformity and radiation effects. The results of studies of the velocity fields and shear stress, turbulence microstructure are presented. An intensification of turbulent momentum transfer in channels in azimuthal and radial directions in the area of gaps between the rods is demonstrated. The performed analysis is indicate that there are a significant difference between the experimental dependences for turbulent momentum transfer coefficients in the radial and azimuthal directions and calculated within the framework of semi-empirical models of turbulent momentum transfer as well as the anisotropy coefficients of the turbulent momentum transfer in rod bundles. The results of benchmark on the fuel assembly thermohydraulics showed that the common commercial codes describe experimental data only approximately. It is shown, that the intensification of turbulent momentum transfer in the channels of rod bundles is due to the appearance of large-scale turbulent momentum transfer (secondary flows). The contribution of large-scale turbulent momentum transfer to the turbulent momentum transfer coefficients in the channels of rod bundles is calculated. The dependence for coefficient of inter-channel turbulent exchange of momentum is obtained and the intensification of inter-channel turbulent exchange in close-packed rod lattices is explained. A dependence for the dissimilarity coefficients of forced inter-channel convective exchange of momentum and mass, as well as energy and mass in rod bundles spaced by wire winding is obtained. The calculation methods and the numerical modeling results for temperature regime of fuel assemblies with randomly distributed initial parameters by the Monte Carlo method are presented as well as the thermomechanical analysis of the temperature field in the fuel assemblies during the campaign. An idea about the equilibrium configuration of a rod bundle in a hexagonal jacket during irradiation, the stress-strain state of an individual fuel rod and a fuel assembly jacket is obtained. The tasks of further investigations discussed.

## INTRODUCTION

The development of fast reactors (FR) with liquid metal coolants (LMC), which have been carried out at the IPPE since 1950, required a large set of experimental and theoretical thermohydraulic studies of the reactor core [1–6]. For this purpose, a complex of experimental facilities with liquid metal cooling was created, methods for modeling thermohydraulic processes, sensors and measurement techniques were developed as well as appropriate methods for thermohydraulic calculation of the fast reactor core.

Ensuring highly efficient operation and prolongation of the service life of the 3rd block of the BNPP with the BN-600 reactor, justification of the safety of the 4th block of the BNPP with the BN-800 reactor, the development of new schematic-layout and technical solutions in the design of a new generation nuclear reactor with sodium cooling (BN-1200), with lead cooling (BREST-OD-300) and lead-bismuth alloy cooling (SVBR-75/100) set new tasks [7–10]. Research, as a rule, have a complex experimental and computational nature and is ongoing.

In the presented work, problem-oriented, prospecting and applied thermohydraulic studies of the fast reactor core with liquid metal coolants are discussed, and individual problems and tasks for further research are formulated.

## DESIGN, TECHNOLOGICAL AND PHYSICAL FEATURES OF THE FAST REACTOR CORE

***Design features.*** The main components of the reactor are the core and the breeding zone, recruited from hexagonal fuel assemblies (FA) with variable flow rates, in which cylindrical fuel elements (fuel rods) are located and in the near-wall area there are displacers to align the heating over the fuel assembly cross-section.

The modularity of the core determines the difference in the velocity and temperature fields in the fuel assembly, the need to organize an optimization procedure for calculating of the fuel assembly characteristics (flow rate, coolant heating). A significant number of fuel rods in assemblies determines the large assembly dimension. The technological gap in the fuel assembly, the statistical nature of the geometric parameter distributions in the fuel assembly determine the difference between the geometry of the periphery and the central areas of the fuel assembly.

Wire spacing of fuel rods of the “rib-by-cladding” type (with a tape on the wall fuel rods or three-head ribs in the breeding zone) initiates an intense transverse convective exchange between channels in a fuel assembly, which has a complex structure and affects the distribution of velocity and temperature fields within the channel groups and in the volume of the fuel assembly as a whole. The resulting coolant leakage in the gaps between the fuel assemblies causes intense heat transfer through the fuel assembly jackets.

***Technological features.*** The low sodium specific heat leads to high heating in the reactor, in fuel assemblies and individual cells, which significantly complicates the solution of the problems of thermohydraulics and thermomechanics in fuel assemblies, and is especially important for analyzing the shape change of fuel assembly. High thermal conductivity leads to high values of the heat transfer coefficient from the fuel rod to sodium, so that the maximum values of the temperature difference between the cladding surface and the coolant do not exceed ~ 20 °C. Thus, in determining of the fuel rod temperature in assembly, the decisive role is played by the exact determination of the coolant temperature distribution in the fuel assembly volume.

Since the maximum sodium temperature under nominal operating conditions in a typical power reactor does not exceed 650–670 °C, and the boiling point is about 900 °C, there is a very high margin of boiling point under normal operating conditions at any point in the core (~ 400 °C).

***Physical features.*** An important factor for determining the temperature field is the distribution of the heat generation along the core radius and the change in this distribution during the campaign. The distribution of the specific heat generation over the core volume at a given time depends on the core volume, the type and design of the surrounding shields, the number of enrichment zones, the presence of reproducing inserts inside the core, the presence and type of reactivity control rods, the state of the reactor in relation to the refueling time and etc.

Non-uniformity heat generation in fuel assemblies causes additional unevenness of the temperature field in the fuel assembly cross-section, additional non-uniformity in distribution of the specific heat flux along the fuel rod perimeters, as well as inter channel heat exchange due to the thermal conductivity of fuel rods.

Due to various mechanisms in fuel assemblies there is an exchange of mass, momentum and heat between cells, which is an important factor in the formation of velocity and temperature fields in the fast reactor core (as well as other reactors with cylindrical fuel rods).

Being exposed to a high dose of radiation exposure (fast neutron fluence ~ 1023 sm-2), being in inhomogeneous temperature and neutron fields, hexagonal fuel assembly jackets and fuel rod bundles can have significant changes in shape caused by both radiation effects (swelling, creep) and thermal expansion. This is reflected in the curvature of fuel assemblies, an increase in their turnkey size, in the curvature of fuel rods and a change in their diameter. The character of the shape change depends on the fuel assembly geometry, the material of the fuel assembly jacket and the fuel rod cladding.

The current level of requirements for calculating the operability of core elements, modern knowledge and describing the processes of shape change, mass transfer and corrosion, strength characteristics, which largely determine the core operability, requires from the thermohydraulic calculation to find not only the maximum (or average) fuel rod temperature, but also significantly higher detailing: the temperature distribution along the fuel rod perimeter, the maximum azimuthal non-uniformity of the fuel rod temperature, as well as the temperature distribution along the perimeter of the fuel assembly jacket, taking into account their influence [5].

## Pacing FACTORS of THE OPERATING TEMPERATURE For FAST REACTOR CORE

The operating temperature of the fuel rods in the fast reactor core is determined by a large number factors with a regular and statistical character [3, 4].

At the stage of the reactor design, the nominal value of the heat generation field in the core and in the lateral breeding zone, found taking into account burnout, rearrangements of fuel assemblies, and the influence of control and safety rods, is set. Hydraulic profiling of the core is carried out with this in mind.

Inside the fuel assembly, the formation of the temperature field, in addition to the “global” factors – the fuel assembly power and the coolant flow rate through it – is influenced by the neutron flux gradient over the cross-section, as well as the complex processes of the coolant flow, characterized by the presence of intense transverse mass and heat transfer between channels.

Radiation deformation of the hexagonal jacket and fuel rod cladding during the campaign leads to a change in the width of the gaps between the fuel rods, between the fuel rods and the fuel assembly jacket, to a change in the flow area of the cells, to longitudinal deformation of the fuel rods and the jacket, thereby changing the initial temperature regime of the fuel rods and the stressed state of the core elements.

The data for the “cold” of fuel assemblies of the BN-600 reactor show that the technological gap is distributed in such a way that the distributions of the flow cross-sections of the central and peripheral cells are described by the Weibull law. Peripheral fuel rods with a spacer tape are separated with a certain gap from the fuel assembly jacket, the rest gap is distributed in the inner area of the fuel assembly. For individual fuel rods, a shift in the wire winding orientation from the nominal value is observed.

A model of the statistical distribution of parameters in fuel assemblies during the campaign, taking into account the swelling of fuel rods, was formulated in [6]. The statistical dispersion of swelling is reflected in the dispersion of the distances between the fuel rods and the flow cross-sections of the cells and, accordingly, on the temperature of the fuel rod cladding.

The purely statistical factors influencing the formation of the temperature field of fuel rods and fuel assemblies, as a rule, include factors that are stable in time. These factors, which parameters are set at the campaign beginning, remain unchanged for any moment of the reactor operation. These include most of the factors that are determined by industrial errors, such as the dispersion of the nuclear fuel content in the fuel rods, the dispersion of the geometric dimensions of the fuel rods and the hexagonal jacket, and the errors of hydraulic profiling. This also includes errors in the thermophysical properties of structural and fissile materials, metrological errors arising at the stage of monitoring the manufacturing processes of individual elements and units of the core, and other factors.

As a result of taking into account the influence of random factors on the temperature regime using various statistical methods (dispersion method, Monte Carlo method, method of external methods) or their combinations, temperature distribution laws are obtained with the ability to correctly calculate the probability of exceeding specified permissible limits.

The displacers are deformed in fuel assemblies. The bending of the displacers leads to an increase in the peripheral gap and to the convergence of the fuel rods in the central area of the fuel assembly. The spreading of the peripheral fuel rods by the displacers significantly increases the flow areas of these cells and, accordingly, leads to the squeezing of adjacent areas of the fuel assemblies. The shape change of the jacket and fuel rods leads to deformation of the temperature field in the fuel assembly, which in turn affects the shape change of the fuel assembly. Deformation of the rod bundle is observed with the formation of a hollow inside the bundle.

## EXPERIMENTAL STUDIES OF HyDRODYNAMICS IN CHANNELS AND ROD bundles

### Hydrodynamics of the central area in smooth rod bundles

***Velocity distributions.*** Among the experimental achievements, in addition to specific data on the velocity profiles and hydraulic resistances, the most interesting are three principal results [1].

First, the friction resistance coefficient in channels with cross-sections having narrow parts (a triangle with a small angle, close-packed fuel rods, a completely eccentric annular channel, etc.) is noticeably lower than in simpler channels, for example, in round tubes, if the equivalent hydraulic diameter is taken as the characteristic dimension.

Second, it was shown that even in very complex channels, for example, in close-packed fuel rods, the velocity distribution on the normal to the cell surface is dimensionless with good practice accuracy is approximated by a universal law for the boundary layer, for example, by the Prandtl-Karman formula. This position is used in many semi-empirical calculation methods.

Third, it was established that for a small radius of the perimeter curvature, the velocity profile differs from the universal velocity profile for the boundary layer. This should be taken into account, for example, when calculating annular channels, rod bundles with large pith-to-diameter ratio.

***Turbulent flow microstructure.*** Experimental studies of the turbulent flow microstructure in the rod lattices were carried out on an experimental model of the rod bundles with pith-to-diameter ratio *s*/*d* = 1.17 by the heated filament method using the DISA apparatus [7].

Turbulent characteristics in central area of the model (constituting of the turbulence intensity *U*', *V*’, *W*') with an increase in the distance from the wall in the measured area, they monotonically fall down to the line of maximum velocities. In addition to the expected minimum at the widest point, there is another minimum here – in the gap between the rods – with practically the same level. The minima in the narrowest places are most likely caused by the same momentum transfer mechanism as in *U*, that is, the secondary flow.

An intensification of turbulent transfer in the azimuthal direction in the area of the gap between the rods is observed. The anisotropy coefficient can reach values of 30–40.

### Hydrodynamics of the peripheral area in smooth rod bundles

***Velocity and shear stress distributions in the peripheral area of smooth rod bundles.*** Significant non-uniformities were found in the velocity distribution along the perimeter of the lateral and angular rods in the fuel assembly models, reaching 60% of the level of the average velocity along the rod perimeter.

The measurement results show (Fig. 1 *a*, *b*) that in the shear stress distribution along the wetted perimeter of the cell in a regular rod lattice, a local maximum is observed at the place of the greatest expansion of the channel, that can be explained by the effect of a secondary vortex.

In the deformed lattice (Fig. 1 *c*, *d*) the shear stress distribution along the wetted surfaces is almost symmetric with respect to the geometric symmetry axes of the flow section, although in some areas of the fuel assembly anomalies are observed that can be explained by the action of individual secondary vortices not only inside the channels, but also at the boundary. The velocity distribution of along the normal to the wetted perimeter is described by a universal law, if the local value of the shear stress is used to calculate the dynamic velocity. The influence of the Reynolds number in the range 4·104–2·105 on the shear stress distribution in a regular lattice is negligible.

|  |  |
| --- | --- |
| *a*) | *b*) |
| *c*) | *d*) |

*Fig. 1. Distributions of shear stresses on the fuel rod surface and velocities in the channel cross-section in the peripheral area of the fuel assembly model with displacers in the peripheral channels at the nominal geometry (a, b) and deformation of the fuel rod lattice in the peripheral area of the assembly (c, d)*

Based on the data obtained from measurements of the signal distributions along the tube perimeter by the electromagnetic sensors, the velocity distributions over the model cross-sections were calculated [25].

***Turbulent flow microstructure in peripheral area of the fuel assemblies without displacers.*** A significant intensification of turbulent velocity fluctuations is observed in peripheral area of the fuel assembly in comparison with an infinite lattice.

***Turbulent flow microstructure in peripheral area of the fuel assemblies with displacers.*** Comparison of the data shows that the presence of displacers leads to a fundamental change in the flow nature in the channel as compared to the geometry without displacers. In the angular and lateral areas, a sharp decrease in the longitudinal turbulence intensity is seen to a level comparable to a smooth tube. Displacers reduce the intensity of the transverse transfer of momentum between the angular and adjacent channels.

## MODELS OF TURBULENT MOMENTUM TRANSFER AND DEPENDENCES FOR THE TURBULENT VISCOSITY COEFFICIENT IN THE rod bundles

Turbulent flow of an incompressible liquid (gas) can be described by the Reynolds equations. The system of equations is closed on the basis of the hypotheses and models of I. Boussinesq, L. Prandtl, T. Karman, K. Taylor, A.M. Obukhov, N.I. Buleev, M.D. Millionshchikov, V.L. Ievlev, V.I. Launder, D.V. Spalding, V.I. Subbotin, M.Kh. Ibragimov, V.P. Bobkov and others. [8].

A relations obtained by various authors for the turbulent viscosity coefficient in triangular rod lattices differ significantly (Fig. 2, 3). Accordingly, the values of the anisotropy coefficient of momentum transfer in the azimuthal and radial directions calculated from the dependences of various authors, also differ by an order or more (Fig. 4).

|  |  |  |
| --- | --- | --- |
|  |  |  |
| *Fig. 2. Distribution of the radial turbulent diffusion coefficient along  the normal to the wall according  to various authors: 1 – Niyasing;  2 – Mantlik; 3 – Milbauer; 4 – Yang and Jiang; 5 – Neelen; 6 – Shimizu;  7 – Ibragimov; 8 – experiment [7];  9 – tube according to Reichard;  10 – Kjelström; 11 – Ramm* | *Fig. 3. Distribution of the azimuthal turbulent diffusion coefficient along the normal to the wall according  to various authors: 1 – Niyasing;  2 – Mantlik; 3 – Milbauer; 4 – Yang and Jiang; 5 – Neelen; 6 – Shimizu;  7 – Ibragimov; 8 – experiment [7];  9 – tubee according to Reichard;  10 – Kjelström; 11 – Ramm* | *Fig. 4. Comparison of dependences for the anisotropy coefficient  of turbulent transfer of momentum  in triangular rod lattices:  1, 2, 5 – Seale; 3 – Slugter; 9 – Yang and Jiang; 12 – Milbauer;  13 – Mantlik; 14 – Ibragimov;  15 – tube according to Reichard;  16 – experiment [7]* |

M.Kh. Ibragimov et al. proposed an approach to calculating momentum transfer coefficients based on the analysis of the interaction processes between the averaged and pulsating motions in the flow and the use of dependencies for statistical characteristics.

N.I. Buleev [9] proposed a phenomenological theoretical model of turbulent exchange, which allows one to approximate all six components of the symmetric tensor of turbulent stresses in the entire area of the liquid flow, right up to the walls. In this case, the turbulent liquid flow is considered as a result of the imposition of disordered unsteady vortices on the main flow.

A feature of the turbulence structure in complex shape channels is the presence of large-scale turbulent transfer (large-scale turbulent vortices or so-called secondary flows) as well as small-scale turbulent diffusion. Secondary flows arise, firstly, due to unsteady flow in the initial section of the channels, and secondly, they are generated due to inhomogeneity of turbulence in the near-wall area of the channels. They occur in channels with an asymmetrical cross-section, such as channels in a rod bundle. For a fully developed flow, they have closed vortices form.

The turbulent flow of a substance is presented by I.O. Khintse in [10] as a superposition of small-scale turbulent diffusion and large-scale turbulent transfer

, (1)

where *u*1 and *u*2 are pulsating velocities of small and large vortices; *c* is substance.

The Reynolds equation for the averaged liquid flow in the presence of secondary flows in the channels can be written in the form

, (2)

where term *II* denotes convective transfer due to secondary flows, and term *III* denotes the Reynolds stress due to turbulent pulsations.

The results of the performed analysis indicate the need to improve the modeling of turbulent momentum transfer in rod bundles, which requires accurate, reliable experimental data on the structure and characteristics of turbulent momentum transfer.

### 5.1. Calculation of large-scale turbulent momentum transfer

Large-scale momentum transfer is carried out by moles moving from high-velocity area to low-velocity area (and vice versa). When large-scale moles move, they are destroyed and small-scale moles are formed, which increase the kinetic energy of turbulent pulsations. The energy dissipation of the large-scale vortices is completely spent on increasing the kinetic energy of small-scale turbulence, measured in experiments.

To describe the intensity of secondary flows, we use the relation developed from the empirical Niyasing approximation, and to estimate the kinetic energy dissipation of large-scale vortices the N.I. Buleev’s formula [9], obtained as a result of solving the equation of the mole motion when it interacts with the environment. After performing the necessary transformations, we find that the increment in the kinetic energy of turbulent pulsations along the normal from the rod surface is described by the formula



, (3)

where λ is coefficient of hydraulic resistance of the channel;

, (4)

, (5)

where *rm* is mole radius.

Comparison of calculations by the formula with experimental data for a lateral channel without displacers shows that the calculated value of the kinetic energy of turbulence near the wall is higher than in the experiment, in the flow core it is lower than in the experiment (Fig. 5 *a*). Assuming that the energy dissipation of turbulent moles is isotropic, we find that the calculated intensity of pulsations in the axial and azimuthal directions is lower than in the experiment. Similar data have been obtained for the case of bundle deformations (Fig. 5 *b*).

### 5.2. Contribution of large-scale turbulent exchange to inter-channel momentum exchange

The intensity of momentum exchange can be calculated from the relations that can be easily obtained from the momentum balance for interacting channels

**, (6)

where ω is channel flow area; π is wetted perimeter; Δ*Рij* is impulse flow in the gaps between channels *i* and *j*.

|  |  |
| --- | --- |
| *a)* | *b)* |

*Fig. 5. Comparison of the calculated and experimental values of the kinetic energy of turbulent pulsations in the area   
of the lateral channel adjacent to the angular channel (a) and in the area of the lateral channel adjacent to the deformed channel (b):* IIIIII *– experiment; IIIIII – calculation*

The processing of hydrodynamic research data, performed for the peripheral area of the fuel assembly when the peripheral fuel rod is displaced along the jacket perimeter or when the fuel rods are grouped into the inner area of the rod bundle [8], has shown that the intensity of inter-channel turbulent exchange of momentum in close-packed lattices is higher than in extended ones. The empirical dependence for the coefficient of inter-channel turbulent momentum exchange has the form:

, (7)

1.035 ≤ *s*/*d* ≤ 1.25; 6.5·104 ≤ Re ≤ 18.1·104.

Dependence (7) indicates the effect of intensification of inter-channel turbulent exchange in close-packed lattices, which is not revealed when analyzing only small-scale turbulent diffusion.

To take into account the contribution of large-scale turbulent transfer to the inter-channel turbulent exchange, we use the turbulent substance exchange model. Assuming that the secondary flows transfer the average substance for each of the channels from the central part of cells *i* and *e*-area of the gap between them and the medium mixed substance for channels *i* and *e* from the area of the gap to the central part of the channels, we obtain

, (8)

where  is average velocity of the large-scale vortices movement; Δ*Sie* is width of transversal one-way flow of secondary flows.

We have received the formula for the coefficient of inter-channel turbulent momentum exchange [11]:

, (9)

1.0 ≤ *s*/*d* ≤ 1.6; 104 ≤ Re ≤ 2·105.

Comparison of the obtained dependence (9) with experimental data shows that dependence (9) with an accuracy of ± 15% agrees with the empirical dependence of the authors (7), but ~ 30% lower than the generalizing dependences obtained by L. Ingesson and J.T. Rogers.

Analysis of experimental data by D.S. Rowe et al on inter-channel turbulent exchange in fuel rod bundles spaced by grids shows that it is approximately possible to use dependences for smooth rod bundles.

### Thermal hydraulic benchmark for fuel assemblies with spacer grids

The results of numerical simulation of the velocity and temperature fields in a fuel assembly model using commercial three-dimensional CFD codes indicate [12] that three-dimensional codes used by specialists from different countries (Russia (D.A. Afremov, V.P. Smirnov, D..A. Yashnikov, code BRS-TVS.R), Japan (H. Ohshima, codes SPIRAL, AQUA), Spain (A. Pena, G.A. Esteban, code FLUENT), Netherlands (J. Karlsson, H. Weider, code STAR-CD), Republic of Korea (K. Sakh, codes MATRA, CFX) very approximately describe the presented experimental data. Typically used in codes *k-*ε the models of turbulent transfer, unfortunately, do not sufficiently take into account the anisotropy of turbulent transfer in the area of channel gaps in fuel assemblies, as well as large-scale turbulent transfer.

## HYDRODYNAMICS of WIRE-Wrapped ROD bundles with spacing of the “RIB-to-CLADDING” TYPE

***The distribution velocity in wire-wrapped rod bundles with spacing of the “rib-to-cladding” type.***The results of experimental studies in a bundle with spacing by wire wrapping “rib-by-cladding”type indicate a significant effect on the velocity field in the channels of a rod bundle by swirling the coolant flow by spacing wrapping (Fig. 6).

|  |  |
| --- | --- |
| *a)* | *b)* |

*Fig. 6. Coolant velocity distribution over the channel cross-section in a fuel assembly model with wire wrapped fuel rods by “rib-to-cladding” type (a), a schematic representation of the wire wrapping effect on the coolant flow and the character of the transverse velocity component distribution along the rod azimuth, depending on its orientation (b)*

***Convective inter-channel exchange in wire wrapped bundles with spacing of the “rib-to-cladding” type.*** The possibility of approximating the local change in the coolant temperature along the cell length in the framework of the integral model of inter-channel exchange is confirmed by the results of computational and experimental studies carried out when heating the inner rod in the fuel assembly [11].

The results of the numerical calculation of the temperature field, performed by solving the system of equations for the macro-energy transfer within the framework of the local and integral exchange models for the case of heating a single rod, show that at *z*/*h* ≳ 2

, (10)

so that

, (11)

where  is average coefficient of nonequivalence of heat and mass transfer.

Based on the description generality of the exchange process of various substances, we find that the convective coefficients of the substance exchange have the following form:

. (12)

Calculations based on the channel-by-channel model of exchange show [12] that the distribution of the average coolant heating in the channels adjacent to a single heated rod in assembly, found using the “molar” model of convective exchange, coincides with the calculated distribution obtained using the local spatial exchange model if

. (13)

Thus, the average nonequivalence coefficient between substance and mass transfer is expressed by the following ratio

. (14)

Let us assume that the characteristic mole diameter is equal to the width of the gap between the fuel rods, then

. (15)

Thus, the value of the nonequivalence coefficient between substance and mass the transfer is a function of the assembly geometry, the operating parameters, the coolant properties and the character of the substance change along the length of the assembly channels. The value of the averaged nonequivalence coefficient between heat and mass transfer varies from 0.6 to 1, increases with an increase in *h*/*d* and decreases with a decrease in Re, Pr, *s*/*d*.

***Hydraulic resistance in smooth and wire-wrapped rod bundles*** ***with spacing of the “rib-to-rib” type.*** The results of analysis and generalization of data on hydraulic resistance in assemblies with a triangular lattice of smooth and wire-wrapped rods with spacing of the “rib-by-rib” type, presented in [3], are described by the formula:

 (16)

1.0 ≤ *s/d* ≤ 1.5; 104 ≤ Re ≤ 2⋅105; 8.0 ≤ *h/d*≤ 50.

Formula (16) is simple in structure and agrees with the experimental data ± 15%.

## HEAT TRANSFER IN CHANNELS AND fuel assemblies WITH LIQUID METALS COOLING

As a result of numerous experimental and computational studies, extensive data on heat transfer and temperature fields of fuel rods have been obtained for various non-nominal conditions and operating modes of the fast reactor core (changes in geometry, bursts of heat generation, statistical distribution of parameters, overheating factors, etc.) [1–3]. It has been established that during the flow of liquid metal coolants in reactor fuel assemblies, and even more so in reactor core of a cassette-free type, there is no thermal stabilization along the length. Only for the central area of the fuel assembly can one speak of quasi-stabilization and consider the thermohydraulic processes in it as in an infinite lattice of fuel rods.

An important characteristic of the thermohydraulic analysis of fuel rod assemblies is the transverse exchange of mass, momentum, and energy between the channels. The thermal and electromagnetic methods, the freon tracer method and sensors for measuring the temperature and flow rate of the coolant along the assembly height have been developed and are used for an experimental study of the inter-channel exchange coefficients. As a result of the studies carried out the computational models have been developed for all mechanisms of inter-channel exchange (convective, molecular, turbulent and due to fuel rod thermal conductivity). Generalizing dependences for the inter-channel exchange characteristics have been obtained, which made it possible to close the system of equations for the channel-by-channel thermohydraulic analysis of fuel rodassemblies [11–13].

The total values of the inter-channel exchange coefficients increase with an increase in the pith-to-diameter ratio of lattice

, . (17)

The degree of the convective exchange effect can be estimated by the relation [11]:

, (18)

The intensity of molecular-turbulent exchange is commensurate with the intensity of convective exchange in close-packed lattices (*s*/*d* < 1.05), with an increase in the wire wrapping pitch to 500 mm and for heat exchange with a decrease in Pe to ~ 50. The total value of the heat transfer coefficient decreases with growth of Pe.

## THERMOHYDRAULIC ANALYSIS

A code MIF for channel-by-channel thermohydraulic calculation of fuel assemblies in the fast reactor core was created to simulate the nominal and non-nominal operating modes, taking into account the influence of various factors on the temperature regime of fuel rods and fuel assembly jackets, including the shape change of the structural elements of the reactor core during the campaign, non-uniformity heat generation, statistical deviations of parameters and other factors, using the data of many years of experimental research [13]. The results of calculations by MIF code for a fuel assembly model with non-uniformity heat generation in the cross-section are in agreement with the experimental data R.A. Markley, F.C. Engel [14]. At the same time, comparison of the experimental results R.A. Markley, F.C. Engel with calculations by COTEC, SUPERENERGY and COBRA-IV codes indicates a significant difference.

## SIMULATION OF THERMOMECHANICS AND STRESS-STRAIN STATE OF FUEL ASSEMBLIES

The research results indicate that one of the most significant factors determining the formation of the temperature field of fuel rods and fuel assembly jackets during the campaign is the initial distribution of geometric parameters of fuel assemblies and their redistribution as a result of mechanical interaction of elements and radiation deformation of structural materials of fuel assemblies.

Calculations performed using the SDT-MIF thermomechanical code for fuel assemblies in the area of high fuel enrichment for BN-600 reactor with 7% burnup of heavy atoms, that corresponds to a dose of about 60 displacements per atom (dpa), have shown [15] that for a dose over 42 dpa the change in the cells area along the height of the fuel assembly is complex. The difference in the maximum temperature of the fuel rods cladding at the beginning of the first and at the end of the twelfth micro-campaign lies in the range of 10–15 °C, the maximum azimuthal non-unifotmity of the fuel rod temperature varies within the same limits.

The results of calculating the stress-strain state of the fuel rod cladding for the geometry of the fuel assemblies of the BN-600 reactor showed that the maximum stress values occur in the section with the maximum shape change [16]. At the end of the campaign, due to the influence of uneven swelling of the cladding material on the inner fiber of the cladding, compressive stresses transform into tensile stresses and repeat the nature of the temperature distribution along the cladding perimeter. The maximum axial tensile stresses at the end of the campaign are 34 kg/mm2, the highest axial stresses at the exit from the core are 6.9 kg/mm2, the tangential stresses are 5 kg/mm2.

## Conclusion

Progress in the thermal hydraulics of the fast reactor core with liquid metal coolant is determined by the expansion of the solved problems, the accumulation of experimental data, analysis and generalize them, the development of numerical modeling methods and a qualitatively new level of the used computer technology.

It is shown that the intensification of turbulent momentum transfer in smooth rod bundles due to the appearance of large-scale transfer (secondary flows). The contribution of large-scale turbulent transfer to the kinetic energy of turbulent pulsations, azimuthal turbulent shear stresses, and turbulent momentum transfer coefficients is calculated. The intensification of turbulent inter-channel exchange of momentum in close-packed and deformed rod lattices is explained. The dissimilarity of the inter-channel exchange of momentum and mass, as well as energy and mass in ribbed rod assemblies, is explained, and a dependence for the dissimilarity coefficients is obtained.

The results of the analysis indicate that the refinement of the dependences for the coefficients of turbulent transfer of momentum in rod bundles within the framework of semi-empirical theories of turbulent transfer requires the use of accurate, reliable experimental data on the structure and characteristics of turbulent transfer.

Recommendations were developed for calculating the coefficients of inter-channel exchange. An effective engineering calculation method for shaped during the campaign of fast reactor fuel assembly was implemented based on the channel-by-channel method of thermohydraulic calculation.

Further development of complex codes for neutron-physical, thermohydraulic and thermomechanical calculation of the fast reactor core characteristics during the campaign is urgent. The validating tasks of the thermohydraulic calculation methods and verification of computational codes are topical [17].

References

1. Subbotin, V., Ibragimov, M., Ushakov, P., Bobkov, V., Zhukov, A., Yuriev, Yu., Hydrodynamics and Heat Transfer in Nuclear Power Plants (Calculation basics), Atomizdat, Moscow (1975).
2. Zhukov, A., Kirillov, P., Matyukhin, N., Sorokin, A., Tikhomirov, B., Ushakov, P., Yuriev, Yu., Mantlik, F. et al, Calculation of fuel assemblies for fast reactors with liquid metal cooling, Energoatomizdat, Moscow (1985).
3. Bogoslovkaia, G., Cevolani, S., Ninokata, H., Rinejski, A., Sorokin, A., Zhukov, A., LMFR Core and heat exchanger thermohydraulic design: Former USSR and present Russia approaches, IAEA-TECDOC-1060, Vienna (1999).
4. Sorokin, A., Kuzina, Yu., Trufanov, A., Kamaev, A., Orlov, Yu., Alekseev, V., Grabezhnaia, V., Zagorulko, Yu., Actual problems of thermal physics of fast reactors, Thermal Engineering, **65** 10 (2018) 725–731.
5. Erbacher, P., Cladding tube deformation and core emergency cooling in a loss of coolant accident of a pressurized water reactor, Nuclear Engineering and Design, **103** 1 (1987) 55–64.
6. Tikhomirov, B., Savitskaya, L., Poplavsky, V., Sorokin, A., Models of statistical accounting of the radiation shape change of structural materials in the calculation of the temperature regime of fuel assemblies of fast reactors, Report at the French-Soviet seminar “Problems of the fast reactor core of thermal hydraulics”, Cadarache, France (1986).
7. Zhukov, A., Sorokin, A., Ushakov, P., Matyukhin, N., Tikhomirov, B., Titov, P., Mikhin, V., Mantlik, F., Geina, Ya., Vosaglo, L., Chervenka, Ya., Hydrodynamic characteristics in fuel assemblies of fast reactors, Preprint IPPE-1816, Obninsk (1986).
8. Sorokin, A., Zhukov, A., Bogoslovskaia, G., Titov, P., Ushakov, P., Analysis of the features of the turbulent microstructure of the flow in the rod bundles of nuclear reactors, Preprint IPPE-2274, IPPE, Obninsk (1992).
9. Buleev, N., Spatial model of turbulent exchange, Science. Ch. ed. physical-mat. lit., Moscow (1989).
10. Khintse, I., Turbulence, its mechanism and theory, Fizmatgiz, Moscow (1963).
11. Zhukov, A., Sorokin, A., Matyukhin, N., Interchannel exchange in fuel assemblies of fast reactors (theoretical foundations and physics of the process), Series: Physics and technology of nuclear reactors, **38**, Energoatomizdat, Moscow (1989).
12. Zhukov, A., Kuzina, Yu., Sorokin, A., Analysis of a benchmark experiment on hydraulics and heat transfer in the assembly of fuel element simulators with liquid metal cooling, Atomic Energy, **99** 5 (2005) 336–348
13. Zhukov, A., Sorokin, A., Matyukhin, N., Interchannel exchange in fuel assemblies of fast reactors (calculation programs and practical application), Series: Physics and technology of nuclear reactors, **43**, Energoatomizdat, Moscow (1991).
14. Markley, R., Engel, F., LMFBR blanket assembly heat transfer and hydraulic test data evaluation, Specialist's Meeting “Thermodynamics of FBR fuel subassemblies under nominal and non-nominal operating conditions”, IAEA, Karlsruhe, February 5–7 (1978), IWGFR/29 229–253.
15. Sorokin, A., Efanov, A., Zhukov, A., Problems of fast reactors thermohydraulics with liquid metal cooling, ICONE-11, Tokyo, Japan, April 20–23 (2003) ICONE-11-36131.
16. Sorokin, A., Bogoslovskaia, G., Trufanov, A., Denisova, N., Investigation of the influence of radiation-induced deformation of fuel assemblies on the temperature regime and stress-strain state of the fuel element cladding, Atomic Energy, **120** 6 (2016) 341–346.
17. Sorokin, A., Yuriev, Yu., Bogoslovskaia, G., Kartashov, K., Abdulkadyrov, V., Planning of experiments on thermophysical stands for filling in the verification matrices of integral codes in terms of thermohydraulic codes, Scientific and technical collection edited by S. Kalyakin, O. Kukharchuk, A. Sorokin, SSC RF – IPPE, Obninsk (2013) 99–116.