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Research Article

Actual problems of modeling thermal-hydraulic processes in fast neutron reactors*

Yulia A. Kuzina[1], Alexander P. Sorokin[1]

1 *IPPE JSC, 1 Bondarenko Sq., 249033 Obninsk, Kaluga Reg., Russia*

Corresponding author: Alexander P. Sorokin (apsorokin@ippe.ru)

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Abstract

The results of studies on hydrodynamics and heat transfer processes in fast neutron reactors are presented. Data on turbulent momentum transfer in rod bundles are analyzed. It is shown that the intensification of turbulent momentum transfer in the rod bundle channels is due to large-scale turbulent momentum transfer (secondary currents). The intensification of interchannel turbulent exchange in close-packed lattices of rods is explained. A dependence is obtained for the dissimilarity coefficients of interchannel convective exchange forced by wire wrapping in bundles of rods. The methods and results of numerical modeling of thermal hydraulics using the Monte Carlo method, thermomechanical analysis of the temperature field in fuel rod assemblies in the lifetime process are presented. The results of modeling based on a water model of temperature fields and the structure of coolant movement in the primary circuit of the reactor in various regimes are presented. A stable temperature stratification of the coolant was revealed in the peripheral zone of the upper chamber of the reactor above the side screens. It is shown that the process of boiling liquid metals in fuel assemblies has a complex structure, characterized by stable and pulsating regimes and a heat transfer crisis. The agreement between the results of experimental and numerical modeling is shown. A cartogram of the flow regimes of a two-phase flow of liquid metals in fuel rod assemblies has been plotted. The influence of the surface roughness of fuel elements on the boiling process and heat transfer during boiling of liquid metals is analyzed. Long-term cooling of a fuel assembly with a "sodium cavity" above the reactor core in accident regimes with boiling of liquid metals is shown. The objectives of further research are formulated.

Keywords

fast reactor, reactor tank, fuel rod assemblies, hydrodynamics, heat transfer, experiment, computational modeling, turbulent transfer, interchannel exchange, coolant temperature stratification, boiling regimes, roughness, two-fluid model

Introduction

The processes of hydrodynamics, mass and heat exchange and the properties of coolants have a major impact on the reactor neutronic performance, corrosion processes, and the reliability and safety of the reactor facility. Lack of proper attention to thermal-physical problems affects the solution of these topical issues.

Thermophysical justification for the developed NPPs required the establishment of the respective experimental framework, and the development of special equipment and novel techniques for experimental and computational

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simulation of current processes (Efanov et al. 2015; Thermophysical Bench Base of the Nuclear Power Industry in Russia and Kazakhstan 2016; Sorokin and Kuzina 2019).

The advancements achieved as a result of developing liquid metal coolants have made it possible to propose a liquid metal technology for a variety of engineering applications: sodium for NPPs with fast neutron reactors; sodium and sodium-potassium for metallurgical and chemical industries; sodium-potassium, cesium and lithium for outer space applications, and lithium for fusion and thermonuclear reactors, etc. (Rachkov et al. 2014).

Further evolution of nuclear power in Russia, implementation of the strategy of two-component nuclear power with a closed fuel cycle using sodium cooled fast neutron reactors (Ponomarev-Stepnoy 2016), ensuring its competitiveness, and preserving Russia's leadership in the field of NPPs with fast neutron reactors requires undertaking more integrated problem-oriented studies, primarily on innovative designs of fast neutron power reactors, such as BN-GT and BN-VT.

An analysis into the experience of building and operating nuclear power installations with liquid metal coolants shows that improving their performance and safety requires one to understand thoroughly the fundamental laws of hydrodynamics and heat exchange processes that define the physicochemical and mass exchange processes in the NPP circuits. Essential in this regard is to investigate the processes of turbulent mass and heat exchange in the reactor channels, structural components and tank for nominal, transient and emergency modes, including emergency cooldown and the occurrence and development of sodium boiling in the reactor core.

The paper discusses the results of problem-oriented studies into the thermal hydraulics of liquid metal cooled fast reactors the authors believe to be topically important, and formulates the objectives for further studies.

Hydrodynamics and heat exchange in the fast reactor core channels and fuel rod assemblies

A similarity analysis and the criteria and asymptotic solutions that follow from same were used to plan the experimental studies and generalize their results (Sorokin and Kuzina 2019).

All investigation stages relied primarily on measurement methods and techniques, including the development of unique velocity, flow rate, pressure, level, temperature and other detectors. Different flow meter designs were developed to measure the flow rates of liquid metals. Later, methodologies and techniques were developed to measure electromagnetically the vectors of the liquid metal local flow rates (velocities) in channels and rod bundles (Subbotin et al. 1975). Miniature thermocouples were developed to measure temperature in protective shrouds with an outer diameter of 0.3 to 0.8 mm designed to operate in a temperature range of 300 to 1800 °C.

Hydrodynamics of fuel rod assemblies with smooth heat exchange surfaces

High thermal conductivity and minor dependence of the thermophysical performance of liquid metals on temperature lead to a minor dependence of the temperature profile in a flow of liquid on the heat flux value. As a result, the heat flux has minor effect on the hydraulic resistance of the liquid metal current. The data for rough pipes did not reveal as well any peculiarities of the liquid metal current. Therefore, the analysis results make it possible to simulate the hydrodynamics of liquid metals using water and gases (Sorokin and Kuzina 2019). The decisive criterion in simulating hydrodynamic processes in channels is the Re criterion.

Extensive studies were undertaken to investigate the hydrodynamics of intricately shaped channels, including rod bundles and flow areas of reactor facilities. As much attention as possible was paid to measuring velocity, distribution of tangent stresses, and turbulent characteristics. This made it possible to understand more thoroughly the mechanism of turbulent heat exchange and outlined the ways for building a physically sound theory of turbulent heat exchange (Ibragimov et al. 1978).

The extensive experimental data obtained on the turbulent hydrodynamic characteristics in fuel rod assemblies show (Sorokin et al. 2021) that there is a local maximum observed at the channel largest expansion point when one considers the distribution of tangential stresses along the cell's wetted perimeter in a regular fuel lattice, this being potentially attributed to the secondary vortex effect.

In a deformed lattice, the distribution of tangential stresses along wetted surfaces is nearly symmetrical with respect to the geometric axes of the flow path symmetry, although some FA regions have anomalies that can be explained by the effect of individual secondary vortices not only inside of channels but also on the boundary. The distribution of velocity along the normal to the wetted perimeter is described by the universal law if one uses the local value of the tangent stress to calculate the dynamic velocity.

There is a major intensification of the turbulent velocity fluctuations observed in the bundle's peripheral region as compared with an infinite lattice (Sorokin et al. 2021).

The analysis undertaken by the authors has shown that data obtained by different authors on the distribution of the radial turbulent diffusion and azimuthal turbulent diffusion coefficients along the normal to the wall as part of simulating the turbulent movement of uncompressible liquid in the approximation of Reynolds equations in triangular rod lattices differ greatly (Figs 1, 2). The data shown in Figs 1, 2 correspond to the following references to literature sources: 1 – Nijsing and Eifler 1971; 2 – Zhukov et al. 1986b; 3 – Mühlbauer et al. 1989; 4 – Yang and Jiang 1988; 5 – Neelen 1986; 6 – Shimizu et al. 1990; 7 – Ibragimov et al. 1978; 8 – Sorokin et al. 2021; 9 – Reichardt 1951; 10 – Kjellstrom 1974; 11 – Ramm and Johannsen 1975. The values of the momentum transfer anisotropy coefficients in the azimuthal and radial directions, as calculated based on dependencies from different authors, also differ by an order of magnitude or more (Fig. 3).

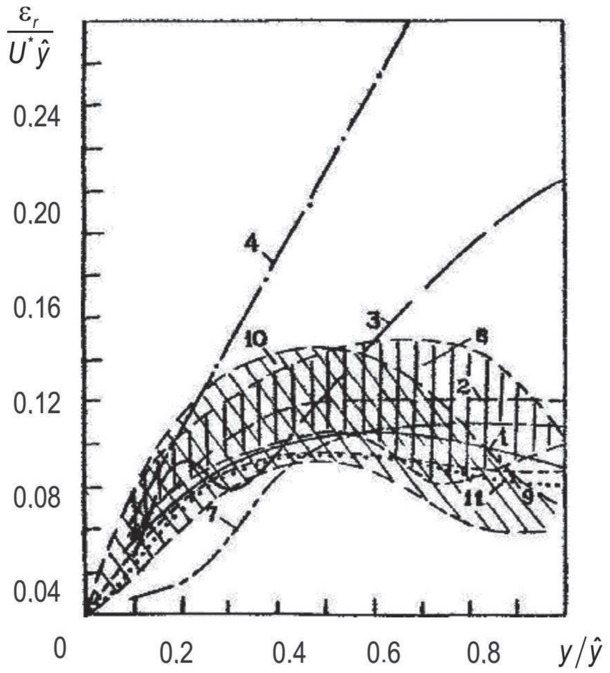

Figure 1. Distribution of the radial turbulent diffusion coefficient along the normal to the wall according to data from different authors.

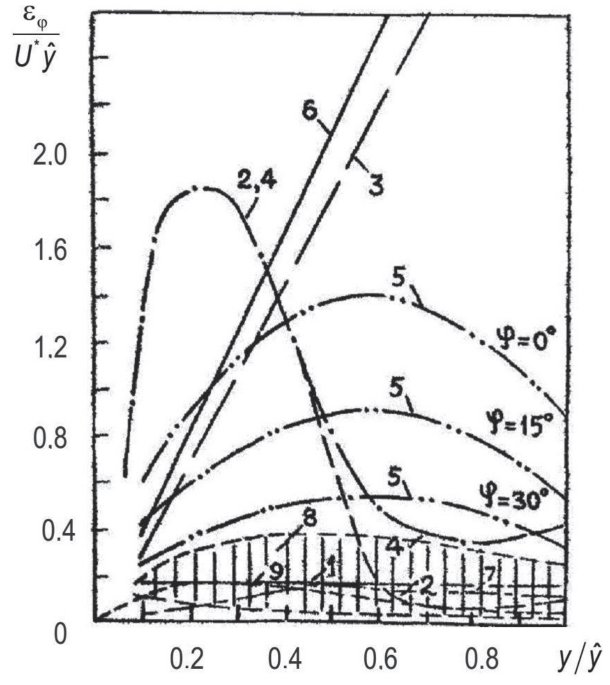

Figure 2. Distribution of the azimuthal turbulent diffusion coefficients along the normal to the wall according to data from different authors.

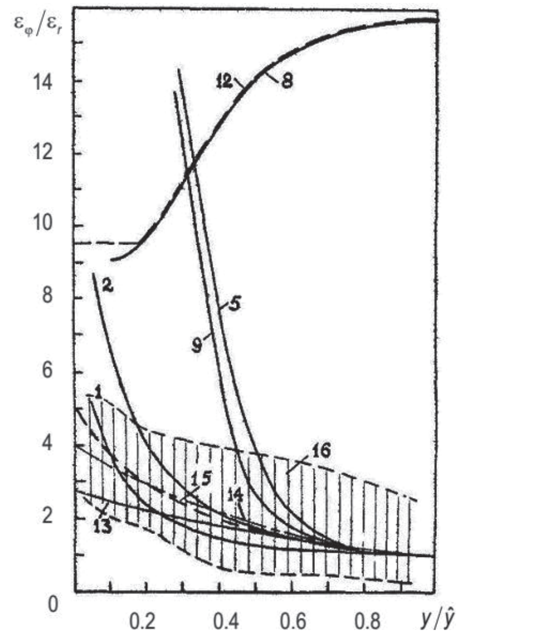

Figure 3. Comparison of dependences from different authors for the coefficient of the momentum turbulent transfer anisotropy in triangular fuel lattices: 1, 2, 5 – Saller (Shimizu et al. 1990); 3 – Slagger (Launder and Spalding 1974); 9 – Yang and Jiang 1988; 12 – Mühlbauer (Mühlbauer et al.(1989); 13 – Mantlik (Zhukov et al. 1986b); 14 – Ibragimov (Ibragimov et al. 1978); 15 –Reichardt 1951; 16 –Sorokin et al. 2021.

Along with small-scale turbulent diffusion, a peculiarity of the turbulence structure in intricately shaped channels is the presence of large-scale turbulent transport (large-scale turbulent vortices or so-called secondary currents).

Secondary currents occur, firstly, due to the unsteady flow in the initial length of channels, and, secondly, are generated due to the heterogeneity of turbulence in the channel wall area. These occur in channels with an asymmetrical cross-section, such as the rod assembly channels. For a fully developed flow, they have a closed vortex shape.

The analysis results show that it is necessary to improve the simulation of turbulent momentum transfer in rod bundles, which requires accurate and reliable experimental data on the structure and characteristics of turbulent momentum transport.

Large-scale momentum transport takes place by moles moving from high velocity regions to low velocity region (and vice versa). When large-scale moles move, they break down and form small-scale moles which increase the kinetic energy of turbulent fluctuations. Dissipation of the large-scale vortex energy serves in full to increase the kinetic energy of small-scale turbulence.

We use the relation developed from the empirical approximation by Nijsing et al. 1971 to describe the intensity of secondary currents, and the Buleev formula (Buleev and Zinina 1981) to estimate the dissipation of the large-scale vortex kinetic energy (Buleev and Zinina 1981) obtained as a result of solving the equation of the mole motion in the process of its interaction with the environment. Following the required transformations, we find that the increase in the kinetic energy of turbulent fluctuations along the normal from the rod surface is described by the formula

$$\Delta K^{+}=\frac{\Delta K}{(u^{*})^{2}}=\frac{1.2}{\sqrt{\lambda/8}}\frac{\frac{d}{d\varphi}\sqrt{\tau_{w}^{+}}}{\left(\frac{d}{d\varphi}\sqrt{\tau_{w}^{+}}\right)^{\max}}\left[\left(\tau_{w}^{+}\right)^{\max}-\left(\tau_{w}^{+}\right)^{\min}\right]\times \tag{1}$$

$$\times\left\{\left|\frac{d}{d\varphi}\sqrt{\tau_{w}^{+}}f(r\varphi)\right|+\left|\left[\left(\frac{d}{d\varphi}\sqrt{\tau_{w}^{+}}\right)^{\max}-\left(\frac{d}{d\varphi}\sqrt{\tau_{w}^{+}}\right)^{\min}\right]f(r\varphi)\right|\right\}$$

$$f(r\varphi)=\frac{1-e^{-\alpha r\varphi}}{\alpha r\varphi} \tag{2}$$

$$\alpha r\varphi\sim\frac{2}{\mathrm{Re}}\frac{\left(d_{h}/d\right)\left(r\varphi/\alpha\right)}{r_{m}^{2}} \tag{3}$$

where K is the kinetic energy of the velocity turbulent fluctuations, u is the coolant velocity, λ is the hydraulic resistance coefficient of the channel, τ_w is the tangential stress on the rod surface, φ is the azimuth coordinate, Re is the Reynolds number, d is the rod diameter, d_h is the hydraulic diameter of the channel, and r_m is the mole radius.

Comparing the calculations based on formulas (1) through (3) with experimental data for the lateral channel without displacers shows (Sorokin et al. 2021) that the calculated value of the turbulence kinetic energy near the wall is higher than in the experiment, and that in the flow core is lower than in the experiment (Fig. 4a). Similar data have been obtained for the case of the bundle deformation (Fig. 4b).

The value of the coefficient of azimuthal turbulent momentum diffusion due to large-scale pulse transfer is according to the following formula:

$$\varepsilon_{\varphi}^{+}=\frac{\varepsilon_{\varphi}}{u^{*}\hat{y}}=-\frac{\overline{w'u'}}{\frac{1}{r}\frac{\partial u}{\partial\varphi}\frac{u^{*}\hat{y}}{y+d/2}}=\frac{1{,}2\sqrt{\lambda/8}\left[(\tau^{*})^{\max}-(\tau^{*})^{\min}\right]\sqrt{\frac{2}{3}\Delta K^{+}}}{\frac{\hat{y}}{r}\left(\frac{\partial}{\partial\varphi}\sqrt{\tau^{+}}\right)^{\max}} \tag{4}$$

The value $(\varepsilon_{\varphi}^{+})^{\max}$ for the lateral channel is ~ 0.32, and that for the deformed channel is ~ 1.

Processing the hydrodynamic investigation data for the fuel assembly's peripheral area with the peripheral fuel rod displacement along the shroud perimeter or with the group displacement of fuel elements into the bundle interior has shown that the intensity of the turbulent momentum exchange among the channels in close-packed lattices is higher than in expanded ones (Sorokin et al. 2021).

The empirical relationship for the inter-channel turbulent pulse exchange coefficient is (Bogoslovkaya et al. 1999):

$$\mu^{*}=\frac{0.0293-0.051(s/d-1)}{\left[(2\sqrt{3}/\pi)(s/d)^{2}-1\right]\mathrm{Re}^{0.1}d},1/m \tag{5}$$

$$1.035\le s/d\le 1.25;\ 6.5\cdot 10^{4}\le \mathrm{Re}\le 18.1\cdot 10^{4}$$

where s is the fuel lattice spacing.

An analysis of experimental data on the turbulent exchange among the channels in lattice-spaced fuel assemblies (Rowe and Chapman 1973) shows that dependencies for smooth rod assemblies can be used approximately.

The results of numerically simulating the velocity and temperature field in the model FA using commercial three-dimensional CFD codes show (Zhukov et al. 2005) that three-dimensional codes used by experts in different countries, including Russia (D.A. Afremov, V.P. Smirnov, D.A. Yashnikov, BRS-TVS.R code), Japan (H. Ohshima, SPIRAL and AQUA codes), Spain (A. Pena, G.A. Esteban, FLUENT code), the Netherlands (J. Karlsson, H. Weider, STAR-CD code), and the Republic of Korea (K. Sah, MATRA and CFX codes), describe rather approximately the presented experimental data. Unfortunately,

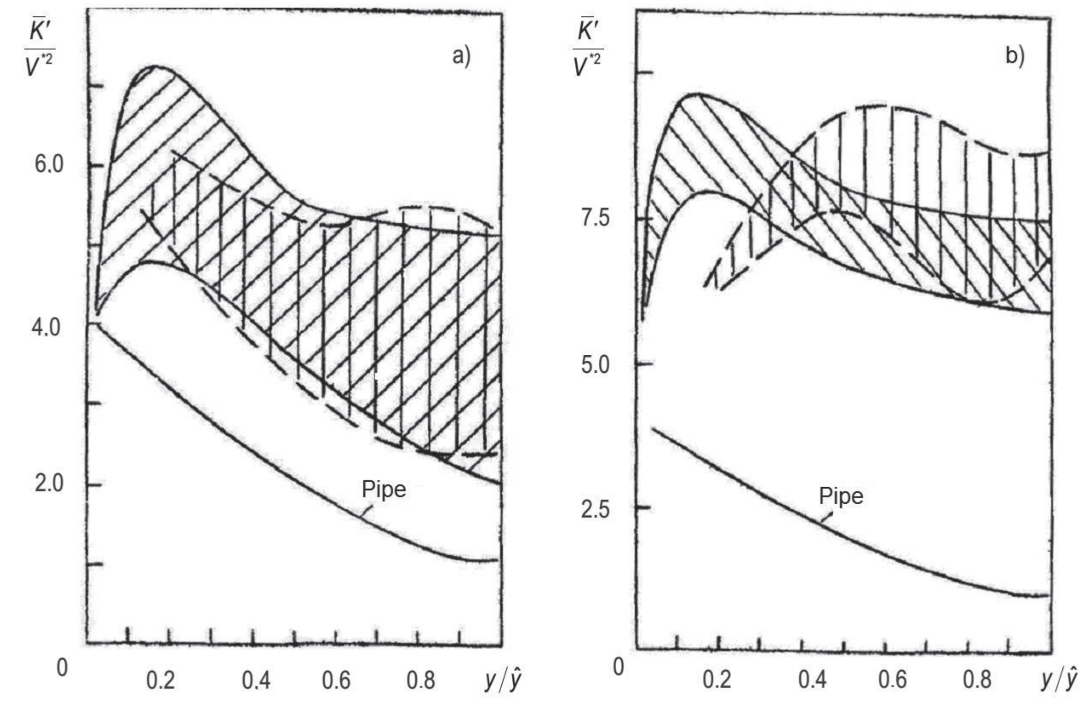

Figure 4. Comparison of calculated and experimental values for the kinetic energy of turbulent fluctuations in the lateral channel region adjoining the corner channel (**a**) and in the lateral channel region adjoining the deformed channel (**b**): IIIIII – experiment; *IIIIII* – calculation.

the turbulent transport models used in codes do not take into account the sufficient extent the anisotropy of turbulent transport, as well as large-scale turbulent transport.

Hydrodynamics of wire-wrap finned fuel rod assemblies

The results of experimental studies in a bundle with wire-wrap spacing of the "rib-on-cladding" type show that the velocity field in the rod bundle channels is affected to a great extent by the wrap spacing coolant flow swirling.

The results of experimental studies have shown that there is nonequivalence between the convective transport of a substance (momentum and heat) among the channels and mass. The value of the coefficient of nonequivalence between the transport of a substance and mass is a function of the assembly geometry, mode parameters, portable properties of coolant, and the nature of the substance change lengthwise the assembly channels. The value of the averaged coefficient of nonequivalence between heat and mass transport varies between 0.6 and 1, increases as the parameter h/d grows, and decreases as Re, Pr and s/d decrease (Sorokin et al. 2020).

The results of analyzing and generalizing data on the hydraulic resistance in assemblies with a triangular lattice of smooth rods and wire-wrap finned rods of the "rib-on-cladding" type (see Zhukov et al. 1986a) are described by the formula:

$$\lambda_p = \frac{0.210}{\mathrm{Re}^{0.25}}\left\{1 + \frac{124}{(h/d)^{1.65}}\left[1.78 + 1.485(s/d - 1)\right](s/d - 1)\right\}$$

$$1.0 \le s/d \le 1.5;\ 10^4 \le \mathrm{Re} \le 2 \times 10^5;\ 8.0 \le h/d \le 50 \qquad (6)$$

where h is the fuel rod wire wrap pitch.

Formula (6) has a simple structure, offers a threshold transition to the formula for smooth rods in the case of the close-packed lattice (s/d = 1.0), and agrees with experimental data with an accuracy of ± 15%.

Heat exchange in channels and fuel assemblies when cooled by liquid metals

It has been demonstrated as a result of experimentally investigating the temperature fields in liquid metal coolant flows, the distribution of temperature and heat fluxes along the perimeter of intricately shaped channels, and the effect of the impurity content on thermal and hydraulic performance that the theory-predicted intensive heat removal by liquid metal coolants is possible.

The generalization of the experimental material revealed the mechanism of heat exchange on the heat-transfer surface in liquid metals: if the concentration of impurities in the coolant does not exceed their solubility at the temperature of the circulating liquid metal, there is no contact thermal resistance at the interface between the coolant and the heat exchange surface.

The most reliable data on heat exchange in liquid metal in pipes, which excludes the effect of contact thermal resistance, were obtained for the temperature distribution in the liquid metal flow (Kirillov 2017; Sorokin and Kuzina 2019). It has been shown therefore that, under these conditions, heat transfer to liquid metals, such as Pb, Pb-Bi, Hg, Na, Na-K, Li and others, is described by a single criterion dependence which is close to that obtained using the Lyon formula (Sorokin and Kuzina 2019).

The generalizations, recommendations and formulas obtained based on experiments and calculations for determining the coefficients of heat transfer and temperature irregularities in the process of heat removal by liquid metals in regular fuel lattices take into account the thermophysical properties of fuel elements through the criterion of their thermal similarity (Sorokin and Kuzina 2019). As a result of multiple experimental and computational studies, extensive data were obtained on heat transfer and fuel element temperature fields for different off-nominal conditions and operating modes of the fast reactor core (geometry change, power excursions, statistical distribution of parameters, overheating factors, etc.) (Bogoslovkaya et al. 1999).

Computational models have been developed based on the investigation results for all mechanisms of exchange among channels (convective, molecular, turbulent, caused by the thermal conductivity of the fuel rod). Generalizing dependencies were obtained for the characteristics of exchange among the channels, which has made it possible to close the system of equations for the thermal-hydraulic channel-by-channel analysis of fuel assemblies (Sorokin et al. 2020).

Data from many-year experimental studies were employed to develop the MIF code for thermal-hydraulic channel-by-channel calculations of FAs used for design and in-service calculations of fast reactor cores (Zhukov et al. 1991). For each of the FAs, it is possible to undertake a standard calculation (based on the average parameters of the bundle) or a statistical calculation (based on the distributed parameters) of thermal and hydraulic characteristics. Statistical calculation of FAs involves a sequential multiple calculation of the temperature field for different options with channel flow areas and fuel element power density arbitrarily distributed using the Monte Carlo method, as well as statistical processing of the results (finding the mathematical expectation and dispersion of the maximum fuel cladding temperature, heating of coolant in the channels, and non-uniformity of temperature along the fuel rod perimeter (Gordeev et al. 2016).

Calculations performed based on the SDT-MIF thermomechanical code for FAs in the BN-600 reactor high-enrichment fuel region with a burnup of 7% of heavy atoms, which corresponds to a dose of about 60 dpa, have shown that the cell area change along the FA height is of a complex nature for a dose of more than 42 dpa (Kuzina et al.

2023). The difference between the maximum temperature of the fuel cladding at the beginning of refueling interval 1 and at the end of refueling interval 12 is in a range of 10 to 15 °C, and the maximum azimuthal non-uniformity of the fuel rod temperature varies in the same limits.

The calculation results for the stress-strain state of the fuel cladding for the fuel assembly geometry of the BN-600 reactor have shown that the maximum stress values occur in the cross-section with the maximum form change (Sorokin et al. 2016). As of the BOL time, the maximum axial compressive stresses of 18 kg/mm^2 fit the peak temperature, and the greatest tensile stresses are 14 kg/mm^2. As of the EOL time, due to the effect from the non-uniform swelling of the cladding material, the compressive stresses on the inner cladding fiber turn to tensile stresses and have the same nature of the temperature distribution along the cladding perimeter.

Thermal hydraulics in the tank of a fast neutron reactor with an integral layout of equipment in different operating modes

The primary coolant circulation circuit (tank design) of a sodium cooled fast neutron reactor represents a complex combination of components connected in series and in parallel with different orientation in the gravity field, the geometric characteristics of the flow areas in which change rapidly in the direction of motion. The errors involved in simulation of thermal hydraulics in fragmented sector models with an isothermal flow are largely caused by the spatial 3D effects and the temperature heterogeneity of the flow not taken into account. The coolant is always non-isothermal due to power peaking, temperature difference between the circulation circuit components, and the peculiarities of heat removal during transients and emergency operating modes.

Undertaking thermal-hydraulic studies using large-scale models with real coolant leads to a high cost of experimental facilities. The studies conducted by the authors have shown that no accurate simulation of fluid dynamics and heat exchange based on small-scale models with real coolant (liquid metal) is possible due to the failure to comply simultaneously with a number of the most important similarity criteria: the Reynolds number ($Re = wl/\nu$), the Peclet number ($Pe = wl/a$), and the Froude number ($Fr = w^2/g\beta\Delta Tl$) (Nijsing et al. 1971). With the Re number > 10^4, the dimensions of stagnant and recirculation formations, the Froude number values being equal for the model and the reactor ($Fr_m = Fr_r$), do not change. Therefore, no simulation based on the Reynolds number is required. In this case, forced circulation modes were simulated using the Froude and Peclet numbers. The approximating simulation of natural circulation modes was based on the Euler number, $Eu = \Delta P/\rho W^2$.

Based on this, experimental studies were undertaken at the V-200 test bench at IPPE JSC based on a water phantom of the SARKh fast reactor with an integral layout of equipment (scale: ~ 1:10 (Fig. 5)) (Opanasenko et al. 2017).

Spatial distributions with an azimuthal temperature non-uniformity in the upper chamber were investigated by three movable temperature probes (MPs) installed at an angle of ~ 150 ° with 15 horizontally arranged thermocouples in each (Fig. 6).

The results of the experiments show that the impact of heat-gravitational forces leads to a temperature-induced stratification that causes the occurrence of stagnant and recirculation formations, and the flow pattern and temperature mode restructuring (Fig. 7a, b). Internal waves occur at the stratified interfaces which cause temperature fluctuations on the reactor equipment walls. This leads to the thermal fatigue of structural materials and to shorter lifetimes of the reactor components. Steady-state natural circulation is characterized by much smaller vertical temperature gradients above the side shields (Fig. 7b).

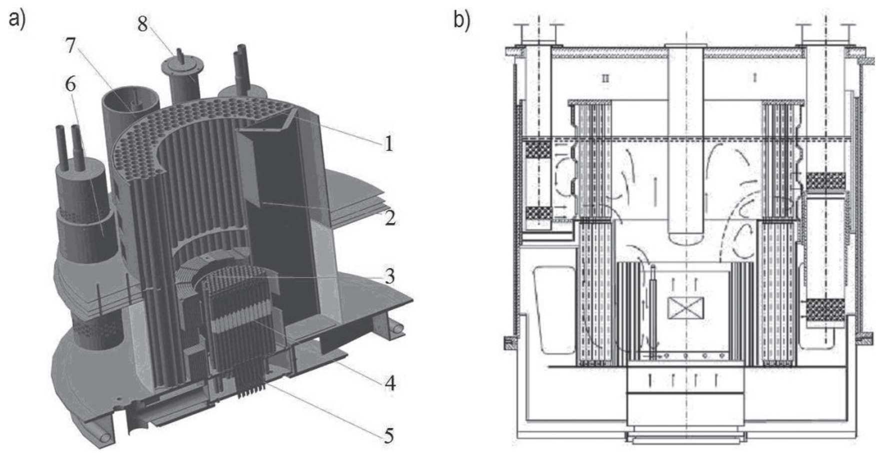

Figure 5. Key components of the water model primary circuit at the V-200 **a.** Test bench, diagram of coolant circulation in the EHRS experimental model (I – nominal mode; II – NC mode); **b.** 1, 6 – intermediate heat exchangers (IHX); 2 – elevator compartment; 3 – in-tank shielding components, 4 – reactor core (dummy FAs), 5 – pressure chamber, 7 – dummy RCP 1; 8 – emergency heat exchanger (EHX).

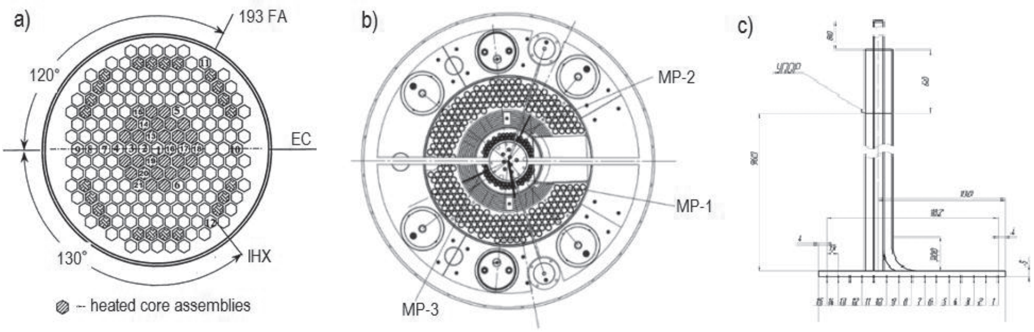

Figure 6. Map of the core assembly dummies (**a**); arrangement of movable temperature probes (MP) (**b**); arrangement of thermocouples on MPs (**c**).

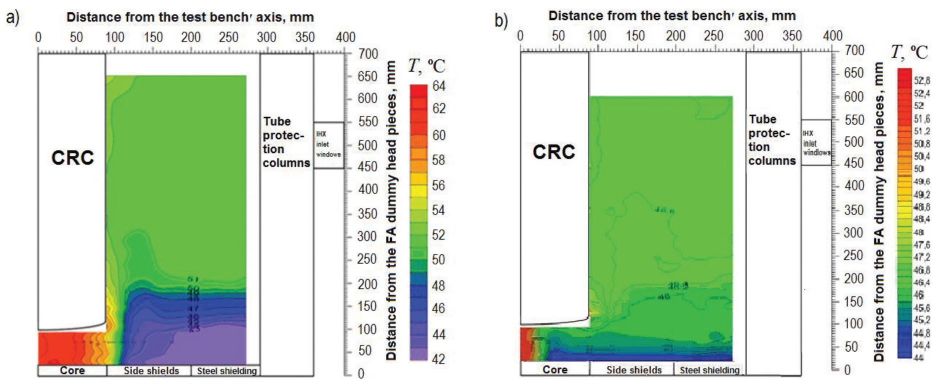

Figure 7. Averaged temperature fields along the upper chamber height obtained by the movement of movable temperature probes, MP-1 and MP-2, in nominal mode (**a**), and chambers in the steady-state mode of cooldown by natural convection (**b**).

Heat exchange during liquid metal boiling in fuel assemblies

Boiling of liquid metals in constricted FA channels is a complex and dynamic high-temperature process (the sodium saturation temperature at an atmospheric pressure of 883 °C). The dynamics of the vapor phase formation can be explosive, specifically given the potential overheating of liquid metal against the boiling saturation temperature.

Studies into the liquid metal boiling undertaken at IPPE JSC based on fuel assembly models have shown three flow patterns for a two-phase liquid metal flow in fuel rod bundles: bubble flow, slug flow, and annular-dispersed flow, which is the limiting one relative to the assembly cooling. The steady-state bubble boiling mode in model fuel assemblies is observed only in the limited area of heat fluxes, its transition to the unstable pulsating slug boiling mode being defined by a variety of factors (Sorokin et al. 2019).

In assemblies with smooth heat exchange surfaces, the development of an unstable (slug) pattern with abrupt fluctuations in the coolant flow rate and the dummy wall overheating is expected to lead at once to a departure from nucleate boiling; there is no practically critical heat flux. For assemblies with an engineering surface roughness, there is a transition from an unstable slug pattern to a steady-state annular-dispersed pattern due to the formation of a liquid film on the heat exchange surface.

The boundaries of transition from the bubble flow pattern to the slug, annular-dispersed and dispersed patterns of a two-phase liquid metal flow in fuel rod bundles are approximated by simple dependencies. It should be noted that the map of the liquid metal two-phase flow patterns (see Fig. 8) differs greatly from the respective map for water.

The experimental data on heat transfer in the process of sodium and potassium boiling in large-size volumes and channels, as well as of sodium-potassium alloy boiling in a seven-rod bundle in a criterion form (Fig. 9) are described by dependence (7) of the Nusselt number (Nu) on the Peclet number (Pe) and the complex (K_p):

$$\mathrm{Nu} = 8.7\cdot10^4\,\mathrm{Pe}^{0.7}\,K_p^{\,0.7} \qquad (7)$$

where $\mathrm{Nu}=\frac{\alpha}{\lambda'}\sqrt{\frac{\sigma}{g(\rho'-\rho'')}}$ is the Nusselt number; $\mathrm{Pe}=\frac{qc\rho'}{r\rho''\lambda'}\sqrt{\frac{\sigma}{g(\rho'-\rho'')}}$ is the Peclet number; and $K_p=\frac{p}{\sqrt{\sigma g(\rho'-\rho'')}}$.

Therefore, the possibility has been shown for transferring data on heat exchange in the process of boiling, obtained for the sodium-potassium coolant, to sodium.

The results of computational studies for a system of single and parallel FAs undertaken based on the upgraded version of the SABENA by-channel code (Sorokin et al. 2006), using a two-fluid model of a two-phase liquid metal flow assuming that pressures are equal in the vapor and liquid phases, reproduce the temperature course, the development of two-phase flow patterns, and the flow rate fluctuations, obtained in experiments,

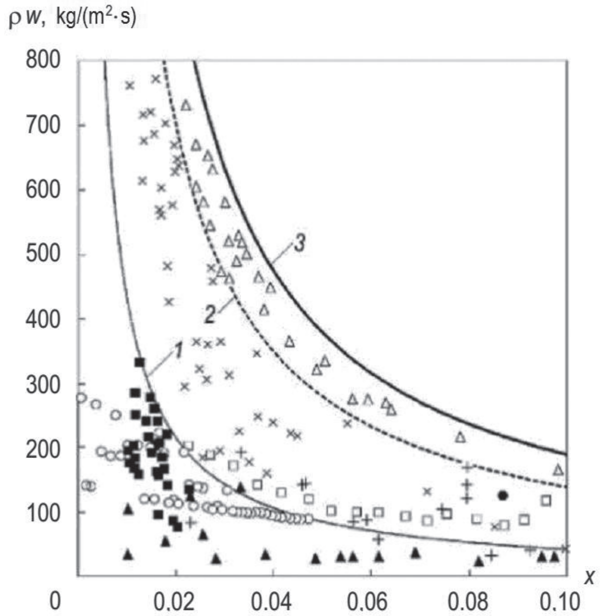

Figure 8. Map of a two-phase liquid metal coolant flow patterns during boiling: 1 – boundary of bubble and slug flow patterns; 2 – boundary of slug and annular-dispersed flow patterns; 3 – boundary of transition to post-CHF transfer; black up-pointing triangle, Multiplication sign, black circle – sodium boiling steady-state mode 1, pulsating mode and steady-state mode 2 according to data from Yamaguchi 1987; black square, plus sign, white up-pointing triangle – data from IPPE JSC on sodium-potassium alloy boiling: bubble, slug and annular-dispersed flow patterns respectively; IPPE JSC's data on sodium boiling: white circle, white square – bubble and slug flow patterns respectively.

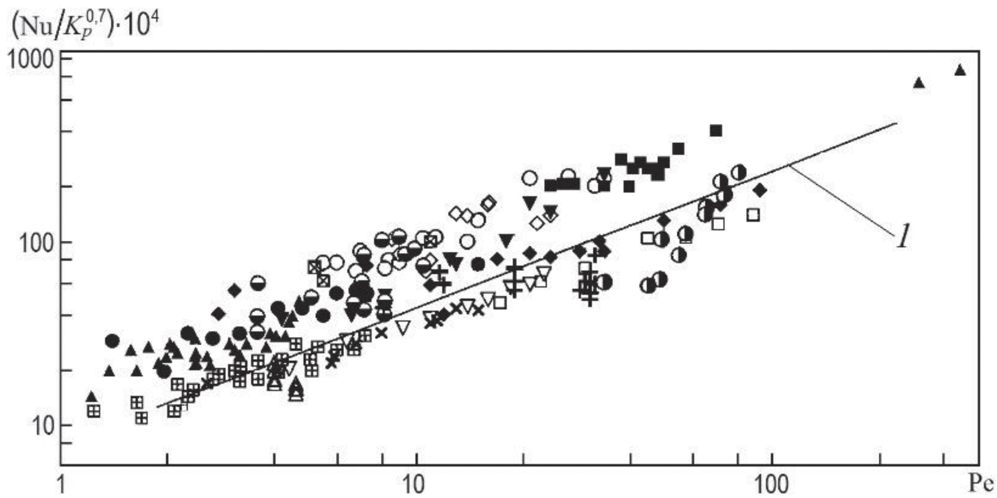

Figure 9. Experimental data on heat transfer in the process of liquid metal boiling in a large volume, in crevices, pipes and a seven-rod bundle: squared plus, black circle, plus sign, white down-pointing triangle, white square – potassium boiling in a large volume with p = 0.12; 0.11; 0.07; 0.04 and 0.004 MPa respectively; black up-pointing triangle, circle with upper half black – sodium boiling in a large volume with p = 0.1 and 0.0472 MPa; white diamond, black down-pointing triangle – sodium boiling in a large volume with p = 0.026 and 0.01 MPa; black diamond – sodium boiling in a large volume in normal conditions; black square – sodium boiling, crevice of d = 2 mm; white circle – sodium boiling, crevice of d = 4 мм; squared times – sodium boiling, crevice of d = 6 mm; white up-pointing triangle, multiplication sign – potassium boiling, pipe of d = 22 and 8.3 mm; data: circle with right half black – sodium-potassium alloy boiling in a seven-rod bundle; 1 – dependence calculated based on formula (7).

and also demonstrate anti-phase fluctuations of the coolant flow in parallel FAs, the inter-channel instability characterized by a major increase in the amplitude of the coolant flow fluctuations in parallel FAs as compared with single FAs, the periodic drop in the FA coolant flow rate to practically zero, and the potential FA drying (departure from nucleate boiling).

Comparing the results of calculations and experiments has shown that it is possible to remove heat by boiling coolant in a model fuel assembly with a "sodium cavity" above the reactor core to prevent the development of an emergency with an operational occurrence (ULOF accident) (Sorokin et al. 2019), during thermal loads of 10 to 15%, and the sodium consumption level of about 5% of rated values.

Conclusions

The experience accumulated in the development of fast neutron reactors with liquid metal coolants allows one to believe that they have rightfully their own niche in nuclear power. However, it cannot be thought that all issues have been resolved, and it remains just to replicate the experience gained in building new reactor facilities.

The objectives for further thermal-hydraulic studies include:

- development of techniques for physical simulation of thermal-hydraulic processes in all portions of the reactor's hydrodynamic path (core, IHX, EHRS) taking into account the instability, e.g., the development of natural circulation in the event of sodium boiling in FAs;
- refinement of techniques to calculate local turbulent characteristics of momentum and energy transport for single-phase and two-phase flows of liquid metal in channels and large volumes taking into account large-scale vortex currents, and the effect of the coolant flow stratification;
- experimental determination of the base constants and closing ratios for heat exchange and temperature fields for all operating conditions and modes (geometry deformation, power excursions, statistical distribution of parameters, etc.) taking into account the overheating factors used in numerical implementation of the code suite;
- analysis of consequences from potential off-nominal modes (interlocks, EHRS, boiling) and development of measures to exclude their progression into a severe accident;
- estimation of temperature fluctuations directly in the coolant flow and on the channel walls, investigation of the temperature fluctuation effect on the strength of structures;
- justification of nominal modes excluding the formation of vortices in the core header and on the sodium surface (gas entrainment), passive circulation zones (stratification phenomena, temperature fluctuations);
- investigation of the two-phase sodium flow modes, dynamics of the sodium boiling area extension in emergency modes in a real FA with a "sodium cavity" above the fast neutron reactor core, and determination of the stable core cooling boundary;
- establishment and justification of a system of verification tests;
- development of a verified set of codes taking into account the interconnection of neutronic, thermal-hydraulic, physicochemical, thermomechanical, mass-exchange and technological processes occurring in a nuclear power plant to justify its service life with regard for the entire combination of processes and operating modes.

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