**Fundamental AND APPLIED investigatons OF the   
LIQUID-METAL COOLed FAST REACTOR THERMAL HYDRAULICS   
(ACHIEVED RESULTS AND FURTHER investigaton issues)**

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**Abstract**

The results of experimental and computational-theoretical investigations in the area of hydrodynamics and heat transfer of fast reactors and accelerator-driven system with liquid metal coolants are presented, and the problems and tasks for further investigations are formulated. Physical phenomena, effects, laws and process characteristics occurring in reactor power plants, including a flow path and a fast reactor core: the velocity and temperature fields, the structure and characteristics of turbulent transfer of momentum and energy, the hydraulic resistance of channels and fuel rod assemblies, the collector system hydrodynamics, the vibroacoustics, the heat transfer in channels and fuel rod assemblies with liquid metal cooling and inter-channel exchange are considered and analyzed. The data of experimental investigations are presented on a single-tube model of a large-module steam generator and on a fragmentary thermohydraulic model of a steam generator of a reactor power plant with twisted steam-generating tubes operating at subcritical and supercritical water pressure. The results of investigations of temperature and velocity fields on a small-scale water model of a fast reactor vessel with an integral layout in nominal, transient and emergency operating regimes are demonstrated. The investigation results on model of a fast reactor vessel with an integral layout are demonstrated that the effect of thermogravitational forces leads to temperature stratification with stagnant and recirculating formations, internal waves appear at the stratified boundaries, temperature pulsations, thermal fatigue of structural materials and a decrease in the service life of the equipment. It is shown that the liquid metal boiling process in fuel rod assemblies is formed under the influence of various factors, has a complex structure, is characterized by both stable (nucleate, annular-dispersed) and pulsation (slug) regimes with significant fluctuations of technological parameters, which can cause a crisis of heat transfer. Heat transfer was studied, a cartogram two-phase flow was obtained for liquid metal boiling in fuel rod assemblies, the effect of the surface roughness of fuel rods on liquid metal heat transfer and boiling regimes in fuel rod assemblies was found. The possibility of long-term stable cooling of the core during sodium boiling was shown by using “sodium cavity” over the reactor core. The kinetic and mechanical characteristics of the degradation process of the simulated fuel assembly in fast reactor core during the thermal interaction of the uranium-containing simulators of fuel with static sodium and their dependence on the parameters and design system were determined.

Information about the key problems of thermophysical investigations of a high-temperature fast reactor with a sodium coolant for the production of hydrogen and reactors cooled by water at supercritical pressure is given.

1. INTRODUCTION

Over the course of more than sixty years of experience in the development of liquid metal coolants at the IPPE JSC, the scientific foundations of their application in nuclear power have been created [1–3]. Much attention was paid to methods of physical modeling of experimental investigations of hydrodynamics and heat transfer in a nuclear power plant with liquid metal coolants [4]. The possibility of modeling the hydrodynamics of incompressible media, including liquid metals in experiments with air, heat transfer in liquid metals Na, Na-K, Li, Hg, Pb, Pb-Bi, etc., using modeling media has been experimentally proved. At all stages of investigations, much attention was paid to the methods and techniques of measurements, including the development of unique sensors for velocity, flow rate, pressure, level, temperature, etc.

The complex of hydrodynamic and liquid-metal thermohydraulic facilities created at IPPE JSC [5] made it possible, together with institutes and design organizations that develop nuclear power plants, to scientifically substantiate the thermohydraulic parameters of nuclear power plants, develop and practically implement highly efficient devices and systems that ensure the successful operation of operating nuclear power plants, as well as organize work on new generation NPP projects [6].

In the early 2000s, the main investigations in the field of thermal hydraulics was aimed at improving the safety of operating nuclear power plants (BN-600), extending their service life, justifying innovative projects (BN-800), developing computational codes, and conducting fundamental investigations. Wide international scientific and technical cooperation was carried out with organizations and firms from Germany, Japan, Italy, France, USA, Sweden, Republic of Korea, China and international organizations (IAEA) etc.

The commissioning in 2010 of the Federal Target Program “Nuclear Energy Technologies of a New Generation for the Period of 2010–2015 and for the Perspective until 2020” initiated the development of projects for fast reactors of a new generation. The tasks of introducing fundamentally new technical solutions into projects were solved, providing a significant improvement in the design and technical characteristics of power units, including a reactor and a steam generator (reducing material consumption), an emergency heat removal system with autonomous heat exchangers built into the reactor vessel (increasing reliability), in-reactor purification systems and control of sodium in the primary circuit.

At present, the IPPE JSC has a complex of modern extra-reactor thermophysical facilities and installations for investigations to substantiate design solutions and the safety of liquid metal cooled fast reactor [5].

1. INVESTIGATION OF THE STATISTICAL CHARACTERISTICS OF VELOCITY AND TEMPERATURE FIELDS IN A TURBULENT FLOW

Extensive experimental material on turbulent hydrodynamic characteristics in fuel rod assemblies was obtained under the joint investigations program of INR and IPPE [7]. The results of these studies are still relevant today.

As a result of comprehensive and systematic investigations of the statistical characteristics of velocity and temperature fields in a turbulent flow of water and liquid metals, the characteristics of velocity and temperature pulsations, intensity this pulsations, correlation functions, spectral density and probability distribution have been studied [8]. The obtained results made it possible to gain a deeper understanding of the heat transfer processes mechanisms and outline the directions for creation of a physically substantiated theory of turbulent heat transfer.

As before, the topical problems of researching the processes and characteristics of hydrodynamics and heat transfer in fuel rod assemblies with liquid metal cooling are the study of the structure and characteristics of turbulent transfer of momentum and energy in complex shape channels, their modeling taking into account the transfer anisotropy and the effect of spacers, determination of the turbulent Prandtl number; determination of the influence of various similarity criteria, identification of self-similarity areas, creation of reliable methods for experimental modeling of heat transfer processes.

1. HYDRODYNAMICS IN CHANNELS AND STRUCTURal elements OF NUCLEAR POWER PLANTS

***Hydrodynamics in channels and structural elements of nuclear power plants***. Despite the fact that there are a lot of reference books on hydraulics, many designs in nuclear power engineering cannot be calculated, so model experiments were required.

Hydraulic resistances, velocity distributions for single-phase flows are calculated most simply (± 10%), but here, too, problems arise. There are known cases in the hydraulics of circuits where the real flow rates differ from the calculated ones. In the practice of calculations, the mutual influence of hydraulic resistances is ignored when two local resistances are nearby (for example, two turns or an equalizing grid after input of the heat exchanger apparatus) [9].

These studies made it possible to create physically based methods and codes for calculating the hydrodynamic characteristics of the channels in the core of liquid metal cooled reactors [10].

The problems of modeling the mixing of flows with different temperatures are encountered in almost all power plants (water, sodium, lead. Pb-Bi, gas, etc.). Reliable methods have been developed for modeling the hydraulics of various media in air using the supply of a well-controlled impurity (halide method). At the boundary of two flows with different temperatures, temperature pulsations arise, leading to fatigue temperature stresses in the materials of the circuit [11].

***Hydrodynamics of collector systems.*** Studies on hydrodynamic models have shown that the coolant flows at the output from each circulation loop occupy their own specific region in the core and are weakly mixed. As a result of a large complex of investigations of the hydrodynamics of collectors and flow paths of reactors, heat exchangers and steam generators, a previously unknown regularity of liquid distribution at the outlet from flow paths of distributing collector systems was established, it registered as a discovery by Delnov V.N., Gabrianovich B.N. and Yuryev Yu.S. [11–12], which consists in the fact that at the outlet of the liquid from the flow parts of the distributing collector systems axisymmetric zones are formed, the characteristics of which are determined by the design and technological features of the collector system (locations of fluid supply, flow trajectory, jet parameters, hydraulic resistance, etc.).

***Vibroacoustic investigations.*** A number of new theoretical and experimental results in the direction of vibroacoustics of inhomogeneous media were obtained at the IPPE JSC [13]. The theory of effective properties of inhomogeneous media developed at IPPE makes it possible to take into account the effect of interphase interaction on the acoustic characteristics of such media and on the vibration characteristics of equipment operating in them. Based on the developed theory:

* A new equation of fluid filtration in inhomogeneous anisotropic porous media is obtained, taking into account the effect of inertial interaction of the fluid and the skeleton of the porous medium.
* Theoretical dependences and experimental data on added masses and damping coefficients for rod bundles in one- and two-phase flows were obtained.
* Experimental data on vibration characteristics of rod elements of power equipment in one- and two-phase flows were obtained.
* Experimentally established the phenomenon of self-oscillations of the flutter type of rods with artificial roughness (ribbing) at anomalously low velocities of the longitudinal fluid flow.
* Mathematical models of sound propagation in dispersed media, dispersion-reinforced composites, saturated porous media and rod systems have been developed.

As a result of theoretical and experimental research, new fundamental phenomena were discovered:

* Theoretically obtained a resonance dependence of the effective shear viscosity of emulsions on the frequency of impacts.
* Theoretically predicted and experimentally confirmed the existence of a resonant frequency dependence of the effective density and translational viscosity of gas-liquid media, associated with the resonance of dipole-quadrupole oscillations of bubbles in a liquid.
* Theoretically predicted and experimentally confirmed the existence of a resonant dispersion of the velocity of sound at a frequency equal to half the natural frequency of the quadrupole oscillations of bubbles. In contrast to the previously known dispersion of sound due to the resonance of compressibility of bubbles, the discussed resonance lies in the frequency range characteristic of oscillations of the heat exchange equipment of a nuclear power plant.

The results of fundamental investigations in the area of vibroacoustics of inhomogeneous media, carried out at the IPPE, can significantly increase the information content of vibroacoustic diagnostics of the technical state of NPP structural elements.

1. heat TRANSFER IN CHANNELS AND FUEL rod ASSEMBLIES OF THE CORE AND STEAM GENERATORS WITH LIQUID METALS COOLING

The greatest attention was paid to studies of heat transfer in the most highly stressed and critical unit of the reactor plant – the reactor core, which is exposed to structural, operational, technological, radiation and operational factors during the campaign [2].

As a result, if we consider the problems of heat transfer on a large scale, then over the past decades, for liquid metal cooled reactors, the key problem of a fundamental-applied nature in the area of heat transfer in tubes and rod bundles has been solved. And although, in general, we understand the physics of these processes, but the heat transfer coefficients for new structures still have to be obtained from experiments for real or model conditions.

***Modeling heat transfer in the liquid metal flows in tubes***. Intensive studies of heat transfer in liquid metals began in the 1950s in connection with the development of nuclear power engineering and the development of nuclear reactors based on Pb-Bi alloy and continues up to the present time [14, 15]. When studying heat transfer for a steady flow of liquid metals, the hypothesis was used that Reynolds number can be approximately eliminated in the criterion equation Nu = *f*(Re, Ре) [4]. Works indicating additional errors in its use are unknown.

During the first experiments on heat transfer, the purity of metals was not controlled and no measures were taken to clean them from impurities. It is noted that in “clean” conditions, higher values of the heat transfer coefficient are observed. It was shown that thermal (contact) resistance at the interface between the coolant and the heat exchange surface is absent when the impurity concentration in the coolant does not exceed their solubility at the temperatures of the circulating metal. Under these conditions, heat transfer to liquid metals Na, Na-K, Li, Hg, Pb-Bi in tubes is described by a single criterion dependence close to the Lyon formula obtained using the integral calculation method and the condition of equality to the unit of the turbulent Prandtl number.

When the coolant is saturated with impurities, the heat transfer coefficients can decrease by 1.5–2 times, which corresponds to the experiments of ENIN, CKTI and IPPE.

Already in the 1960s, when the methods of purification of the Pb-Bi alloy were still developing, measurements of heat transfer in mercury and the Pb-Bi alloy gave similar values.

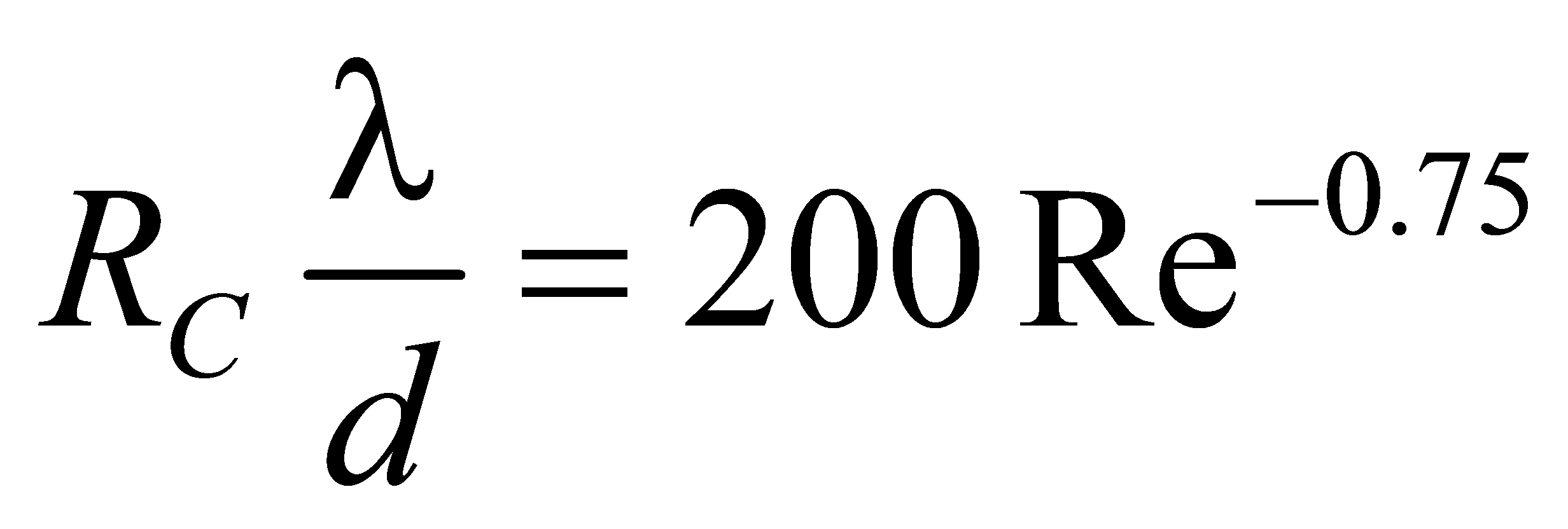
The data obtained as a result of processing measurements of the wall temperature along the length of all working sections made of different materials and temperature distributions in the sodium flow (Pe = 40–1150) turned out to be the same and are described with an accuracy of ± 10% by the following formula:

Nu = 5 + 0.025Pe0.8. (1)

***Contact thermal resistance.*** The performed set of computational and theoretical works and generalization of the experimental material made it possible to reveal the mechanism of heat transfer on the heat transfer surface in liquid metals.

As a result of systematic experimental investigations carried out at the IPPE [2], it was shown that the main cause of contact thermal resistance is the deposition of solid particles suspended in a liquid metal flow on the heat exchange surface. To calculate the maximum value of the contact thermal resistance, P.L. Kirillov in [16] proposed the formula:

. (2)

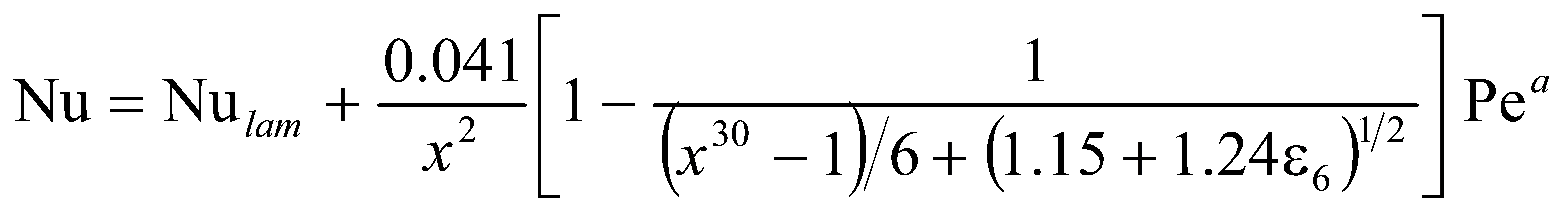


Studies of the dependences of the contact thermal resistance during heat exchange in liquid metals in channels of complex shapes on various factors, primarily on the oxygen content in the coolant, should be continued.

***Heat transfer and azimuthal temperature irregularities in fuel bundles.*** Experimental and computational studies have shown the need to consider “conjugate” problems of heat transfer with fuel rods for channels of complex shape. Developed by P.A. Ushakov, the theory of approximate thermal similarity of fuel rods located in regular lattices [14], made it possible to simulate fuel rods with multilayer or single-layer tubes with electric heating from the inside. Generalizations, recommendations and formulas obtained on the basis of experiments and calculations for determining the heat transfer coefficients and temperature irregularities during heat removal by liquid metals in regular fuel rod lattices take into account the thermophysical properties of fuel rods by criterion of their thermal similarity 17].

To calculate heat transfer and temperature fields in triangular lattices of fuel rods A.V. Zhukov and etc. the formula is recommended [17]:

, (3)



where *a* = 0.56 + 0.19*x* – 0.l/*x*80.

Range for use of the formula (3) are 1 < *x* = *s*/*d* < 2, 1 < Pe < 4000 and 0.01 < ε6 < ∞, where ε6 is criterion for the approximate thermal similarity of fuel rods in a triangular lattice, the characteristic dimension is the equivalent hydraulic diameter. The error in formula (3) is 12–15% for *x* = 1.12–1.14, the calculation can be carried out using the formula for round tube.

Calculations of the fuel rod temperature irregularities along the length are performed according to the formula [31].

Since heat exchange is carried out by convective heat transfer, high thermal conductivity does not guarantee the absence of large temperature irregularities in structures cooled by liquid metals. The temperature field is determined by the degree of heating of the coolant, and not by the heat transfer coefficients, which significantly increases the requirements for the organization of the hydrodynamics of reactors.

It has been established that during the flow of liquid metal coolants in fuel assemblies of reactors, and even more so in core of a cassette type, there is no thermal stabilization along the length. We can talk about quasi-stabilization for the central region of the fuel assembly and consider the thermohydraulic processes in it as in an infinite lattice of fuel elements. Temperature irregularities along the perimeter of fuel rods surrounded by non-standard cells (corner, peripheral fuel rods), as a rule, change over the entire height of the core.

As a result of numerous experimental and computational studies, extensive data have been obtained on heat transfer and temperature fields of fuel rods for various non-nominal conditions and operating regimes of the core in the presence of fuel rod deflections, asymmetric displacements and deformations of fuel rods, overlapping of various parts of the core, the presence of counter flows, bursts of heat generation, statistical distribution of parameters (overheating factors), etc. [15, 19].

***Heat transfer in wide rod lattices with various spacers.*** The experiments were carried out at the “6B” facility on 37 rod model assemblies of fuel rod simulators with *s*/*d* = 1.28 (Fig. 1) [20] applied to a reactor with a heavy liquid metal coolant using a eutectic sodium-potassium alloy (22% Na + 78% K), which has a value of the Prandtl number close to the Prandtl number for lead (Ре = 202–1192).

Experimental data processing was carried out separately for the area between the spacer grids (smooth sections) and in the area of the spacer grids. In the studied range of Peclet numbers, the effect of the spacer grids on the fuel rods is small. Spacer grids lead to a local heat transfer burst, which turns out to be higher than for smooth regions of fuel rod simulators (between the grids). Heat transfer increases as the coolant moves in the grid. Inside the spacer grid, periodic temperature irregularities appear along the perimeter of the measuring simulator, caused by touching the grid elements.

The results of experiments for fuel rod assembly with *s*/*d* = 1.33 have shown that, all other things being equal, in wide rod lattices, the spacing of fuel elements by transverse grids looks more preferable from the point of view of the characteristics of temperature fields and heat transfer than spacing with two-head ribs according to the “rib-by-rib” principle.

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| *a*) | *b*) | *c*) |

*Fig. 1. Cross-section (a) and general view of the model (b), dependence of Nusselt numbers on Peclet for a central simulated fuel rod in assembly with s/d = 1.28 (c): ―●― – for smooth region of fuel rod simulators spaced by transverse grids,   
 – – – for lattice of smooth fuel rods*

***Inter-channel exchange in fast reactor fuel assemblies.*** An essential feature of the thermal hydraulics of fuel rod assemblies is the effect on the distribution of the parameters of inter-channel mass and heat transfer. between medium flows in adjacent fuel assembly channels.

To measure the coefficients of inter-channel interaction, thermal and electromagnetic methods, the freon tracer method, as well as sensors for measuring the temperature and flow rate of the coolant along the assembly height were developed. Convective inter-channel transfer in fuel rod bundles is provided by spacer spiral ribs or wire winding. Computational models for all mechanisms of inter-channel exchange were developed, generalizing dependencies for the characteristics of inter-channel exchange were obtained, which made it possible to close the system of equations for the channel-by-channel thermohydraulic analysis of fuel assemblies [21, 22].

As a result of experimental studies and computational and theoretical analysis of inter-channel exchange of mass, momentum, and energy in bundles of smooth and ribbed by spiral spacing wire winding of fuel rods, physically substantiated methods and programs for thermohydraulic calculation of shaped fuel assemblies in fast reactor core were created (TEMP, MIF).

The influence of the geometry and materials of fuel rods, the effects of radiation swelling and creep on the temperature regime of fuel assemblies were investigated, the temperature regime features in core during operation (campaign) for fast reactors were revealed 15. The efficiency of using multidirectional wire winding creating oppositely directed coolant flows in transverse directions is shown.

***Thermal hydraulic studies of a large-scale steam generator.*** Studies were carried out for the parameters of a large-scale steam generator on a single-pipe model at the ”SPRUT” facility (Fig. 2*a*), in one housing of which the processes of evaporation and superheating of steam were combined in regimes 12.5% and 75% of the nominal sodium flow rate at full-scale parameters in the sodium loop and high pressure water loop.

The obtained experimental data on the critical heat flux are in satisfactory agreement with the data of skeletal tables for calculating the critical heat flux in pipes. A strong influence of water pressure on both the critical (boundary) quality and the value of the heat flux density is noted. With an increase in pressure, an increase in the heat flux density and a decrease in the value of the critical (boundary) flow content are noted. In all modes, a non-stationary process took place, characterized by a shift in the crisis zone. Based on the records of wall temperature pulsations, the maximum amplitudes of temperature pulsations were determined depending on the feed water flow rate.

***Experimental studies on a fragmentary thermohydraulic model of a steam generator of a reactor installation with a lead coolant.*** As a result of thermohydraulic studies carried out at the “SPRUT” facility for the first time on a fragmentary model of a steam generator, consisting of 18 twisted steam generating pipes (Fig. 2 *b*), it was found that when operating at subcritical water pressure (about 18 MPa), the temperature of the steam at the model outlet was equal (503–509) С in the entire range of changes in water flow rates. At a water flow rate of (80–120)% of the nominal value, no pulsations of the water flow rate at the model inlet were found, although according to the calculations at the inlet there should be fluctuations in the water flow with a magnitude of 50 to 150%. At supercritical water pressure (about 25 MPa), the tests were carried out under the same operating parameters as at 18 MPa. No noticeable differences in the steam temperature at the model outlet were found, the discrepancy did not exceed (2–3) С. The absence of fluctuations in the feed water flow rate, pressure in the circuits indicates stable modes when operating at partial parameters [11].

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| *a*) | *b*) |

*Fig. 2. Single-pipe model of a steam generator (a) and general view of the model of a steam generator with twisted pipes on the lead loop (b) at the "SPRUT" facility*

1. INVESTIGATIONS OF TEMPERATURE AND VELOCITY FIELDS IN A FAST REACTOR vessel WITH AN INTEGRAl layout OF EQUIPMENT IN VARIOUS OPERATING conditions

A complex combination of series and parallel connected elements with different orientations in the gravity field, the geometric characteristics of the flow sections of which change sharply along the movement, is the coolant circulation loop in a fast reactor.

Studies have shown that modeling of thermal hydraulics in a fast reactor vessel can be performed on small-scale water models [23]. The experimental facility, located at the “V-200” facility at the IPPE, included an integral three-circuit model of a fast reactor on a scale 1:10 (Fig. 3 *a*). The second circuit like first circuit in model is a closed circulation system with a pump filled with distillate and designed to provide heat removal from model intermediate heat exchangers (IHX) and autonomous heat exchangers (AHX).

The model contains all the main elements of the primary circuit: the reactor vessel (Fig. 3 *a*): the core, the IHX and the AHX, the central rotary column (CRC), support belts, heat shields, steel shells and pillars of in-vessel protection, reflector, pressure chamber with simulators of core assemblies, simulators of main circulation pumps (MCP-1). Additionally, simulators of filter traps, simulators of IR control of the energy level and subcritical state, a device for closing (opening, closing) the reflector holes above the upper plate of the pressure chamber are installed in the upper chamber of the model.

To avoid the difficulties of rigorous criterial modeling, the heat transfer processes in the core and heat exchangers were simulated by uniform volumetric heat generation while maintaining the values of the resistance coefficients for the model and the reactor, which made it possible to exclude the similarity criteria associated with the effect of heat transfer and heat capacity of the system in the transient regimes.

In the forced convection mode, the simulation was carried out in terms of the Fr and Pe numbers using water as the modeling liquid. The coolant in the reactor tank is always non-isothermal due to non-uniformity of power release, temperature difference between the nodes of the circulation loop, peculiarities of heat removal in transient and emergency operating modes. The action of thermogravitational forces leads to a radical change in the structure of the movement of the coolant in comparison with the isothermal flow. Temperature stratification of the flow occurs, especially in circuit elements with large volumes, during transient and emergency operating modes, which is characterized by the occurrence of stagnant and recirculating formations with significant temperature gradients at stratified interfaces.

The criteria determining the similarity of flows in stably stratified zones of the coolant are the numbers Fr, Pe and the local gradient Richardson number Ri =*g*β(∂*t*/∂*z*)/(∂*w*/∂*z*)2. The characteristics of a stably stratified coolant flow are the Vaisal-Brent frequency *N*2= (*g*/ρ)(ρ∂/∂*z*), where ρ – density of the medium, and the scale of buoyancy *lb* = ρ(∂ρ/∂*z*)-1. In a stably stratified turbulent flow, the maximum eddy size cannot exceed the buoyancy scale. Therefore, large-scale eddies larger than the buoyancy scale are suppressed and spread along the stratified interface in the form of internal waves, which create temperature pulsations in the material of the equipment walls with a frequency *f* ≤ *N*. In an extreme case, for example, when a quick emergency protection is triggered, when *lb* < *lК* [*lК* = (ν3/ε)0.25 – Kolmogorov scale, where ε – the rate of dissipation of the kinetic energy of turbulence, there is a complete suppression of convective transfer through the interface between the cold and hot regions of the flow.

In a viscous fluid, simulations with Froude and Richardson numbers cannot be performed keeping the Reynolds number constant. Research with numbers Re > 104 showed that when the values of the Froude number are equal, calculated from the parameters of the experimental model and the reactor (Fr*m* = Fr*r*), the sizes of stagnant and recirculating formations do not change, therefore, accurate modeling by Re numbers is not required.

An approximate simulation of the natural convection regime was provided by the Euler number Eu = Δ*р*/(ρ*w*2), where Δ*р* – pressure difference at two characteristic points of the flow.

The results of thermohydraulic studies of the primary loop of a liquid metal cooled fast reactor and on a water model for forced circulation regimes, transition to cool down mode and emergency cool down by natural convection have been shown that the temperature stratification with the generation of stagnant and recirculation formations, restructuring of the flow pattern and temperature regime occurs in peripheral region of upper (hold) plenum over side shield, in cold and pressure chambers, elevator baffle, reactor vessel cooling system, at the output from intermediate and autonomous heat exchangers in various regimes of their operation, (Fig. 3 b) under influence of themogravitation force.

In this case, waves generated at the interfaces between stratified and recirculation formations cause temperature pulsations on the walls of the reactor equipment and, ultimately, lead to thermal fatigue of structural materials and a decrease in the service life of the reactor equipment. The steady-state natural convection regime is characterized by significantly lower temperature gradients in the vertical direction over the side shields (Fig. 3 *c*).

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| *a*) | *b*) | *c*) |

*Fig. 3. Experimental water model of the primary circuit of a fast reactor with an integral layout of equipment (a), the averaged coolant temperature in the upper plenum when moving the movable thermal probes along the height in the nominal regime (b) and the steady-state cooling regime by natural convection (c): 1, 6 – intermediate heat exchanger;   
2 – elevator baffle; 3 – elements of in-vessel protection; 4 – core (fuel assembly simulators); 5 – pressure chamber;   
7 – simulator MCP-1; 8 – autonomous heat exchanger*

1. STUDIES OF BOILING AND CONDENSATION OF LIQUID METALS

***Studies of the liquid metal boiling.*** Studies of the metal boiling in installations equipped with electric heating, movable microthermocouples and X-ray devices for transmission, made it possible to identify the specific features of the process, to obtain recommendations for calculating heat transfer and critical thermal loads. This made it possible to better understand the physics of the phase transition process [25].

Filming and measurements of the temperature of a single vapor bubble showed that the vapor bubble is surrounded by a liquid that has left the heating surface with it and is overheated relative to the saturation temperature. The study of the dynamics of vapor bubbles during the boiling of wetting liquids on wires has shown that the separation of bubbles from the boiling surface is carried out by forces of non-gravitational origin. The experimental results are confirmed by hydromechanical calculations. Data were also obtained on the transition from the natural convection regime to film boiling (bypassing the nucleate boiling regime) on small-diameter wires. The results obtained have opened the way to change and refine the concepts of the surface boiling mechanism.

The principal possibility of two regimes of boiling – stable and unstable – has been revealed. Experiments carried out on metals and water have shown the possibility of a significant delay in the onset of boiling due to intense convective and conductive heat removal.

State-of-the-art modeling of dynamic alkali liquid metal boiling is of prime importance for an integrated analysis of the neutronic and thermohydraulic characteristics of a fast reactor core in accident conditions (UTOP, ULOF) in the course of their safety justification [26]. In accident conditions, the sodium boiling regime can lead to an increase in reactivity due to the positive sodium void effect of reactivity, overheating of fuel elements, depressurization and melting of their cladding and fuel, etc. Experimental studies of the liquid metal boiling were carried out on simulated fast reactor fuel assemblies with a different number of fuel rod simulators for conditions as close as possible to full-scale conditions.

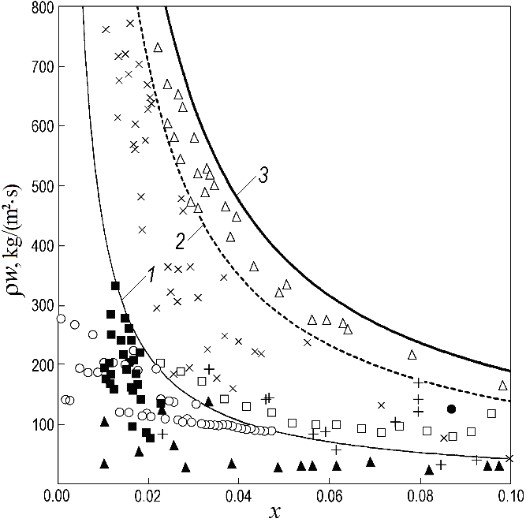
The results of experimental studies carried out in 1995–2015 at IPPE on models of fast reactor fuel assemblies (Fig. 4 *a*) on a high-temperature liquid metal “AR-1” stand showed (Fig. 4 *b*) that the boiling process of liquid metals in channels and fuel rod assemblies is formed under the influence of various factors, has a complex structure, is characterized by both stable (nucleate, annular-dispersed) and pulsation (slug) modes with significant fluctuations of technological parameters (flow rate, pressure, temperature), which can last for tens of seconds and lead to a heat transfer crisis [26, 27]. The effect of the surface roughness of fuel elements on heat transfer and boiling modes of liquid metal in fuel rod assemblies has been revealed [28].

Heat transfer during boiling of liquid metals in bundles of fuel elements has been studied experimentally using models of fuel assemblies [28]. It should be noted that with a forced flow of a vapor-liquid mixture of metals in a tube (at a pressure of about 0.1 MPa) already at a mass quality (1–5)% a annular-dispersed regime occurs, characterized by the fact that (95–99) % liquid is in the form of drops in the central area of the flow. The heat transfer coefficient under these conditions has approximately the same value as for pool boiling. Generalization of experimental data on heat transfer at the liquid metal boiling in a criterion form indicated the possibility of transferring data obtained with sodium-potassium coolant to sodium [27].

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| *a)* | *b)* |

*Fig. 4. Experimental facility with a 7-element fuel assembly model for studying sodium-potassium coolant boiling (a), heat flux density on fuel rod simulators, central simulator wall temperature, coolant temperature, coolant mass flow at the fuel assembly model inlet (b)*

The boundaries of the flow regimes of a two-phase liquid metal flow in fuel rod assemblies are determined. A cartogram of flow regimes at the liquid metal boiling in the fuel rod assemblies was constructed (Fig. 5) [27]. The principal possibility of long-term stable cooling of fuel assemblies in the core at the sodium boiling is shown for a new technical solution – “sodium cavity” over reactor core [28].



*Fig. 5. Cartogram regimes of two-phase flow liquid metal coolants:*

*1 – boundary of nucleate and slug boiling regimes; 2 – boundary of slug and annular-dispersed boiling regimes;  
3 – boundary of transition to post-CHF heat transfer;* ***○****,* ***□****– nucleate and slug regimes, IPPE JSC data on sodium boiling;  
▲,* ***+****, ● – the first stable regime, pulsation and second stable regime, respectively, according to Yamaguchi [8];  
■, ,* ***△****– IPPE JSC data on sodium-potassium alloy boiling: nucleate, slug and annular-dispersed regimes, respectively*

Additional studies of the boundary of unstable sodium boiling regimes in bundles on dependence of various factors and the study of the dynamics of the boiling region propagation in a real large-scale fuel assembly with a “sodium cavity” are required.

***Studies of the liquid metal condensation.*** The mechanism of the metal drop condensation process is investigated. If for water and other liquids the flux of heat and mass in some cases occurs through the surface areas free from drops (micro-film-droplet condensation), then for metals with high surface tension all heat and mass transfer occurs through the droplet surface (only droplet condensation). Theoretical calculations and experiments made it possible to obtain the data and to explain the mechanisms of droplet and micro-film-droplet condensation and the regions of their existence for various liquids as well as to explain the contradictions existing in the literature on this issue [29].

1. EXPERIMENTAL STUDIES OF MODEL FUEL ASSEMBLy DEGRADATION IN ACCIDENTS WITH UNCONTROLLED LOSS OF SODIUM FLOW RATE

The study of the final state of a 19-rod model assembly was carried out at the “Pluton” facility under conditions simulating an accident with an uncontrolled loss of sodium flow rate [30]. The energy release in the experiment was provided by the thermite reaction of the mixture Al + Fe2O3 with stoichiometric composition (*Qp* = 1.6 MJ/kg). According to experimental estimates of the propagation kinetics of the thermite reaction front in channels of similar geometry, the transition time of the entire mass of the initial thermite mixture into the melt Fe + Al2O3 (*Т* = 3100 К) is less than 1 s.

The main objectives of the experiment were to identify the main degradation mechanisms of the cladding of fuel element simulators, assess the distribution of materials over the height of the assembly in its final state, assess the phenomena of blockage of flow sections of the model assembly, and assess the release of materials outside the assembly volume.

The introduction of markers (Cu, Mo, Mg) into the composition of the thermite mixture of materials, in the initial state localized in a strictly defined position along the height of individual fuel rod simulators, made it possible, on the basis of data on their final distribution, to estimate the nature of the melt movement within the model assembly.

Under conditions simulating an uncontrolled loss of sodium flow rate, the region of global degradation of fuel rod simulator cladding was ~ 65% along its height and was predominantly localized in the part of the rod bundle with an increased density of thermite charge.

Based on the experimental study results, three main mechanisms of cladding degradation were identified: temperature stresses in the cladding material, melting of cladding materials, dynamic effects caused by the rapid conversion of thermal energy of the corium simulator melt into mechanical work during thermal interaction of the melt with sodium. The calculated conversion rate was 1.1510–1% at power release 4.85 kW.

As an analysis result of the material distribution along the assembly height, materials were found in the following states: fragments of the fuel rod simulator cladding, domains of solidified melts of steel and iron, conglomerates of thermite reaction products, powdery materials of products of thermal interaction of the molten corium simulator with sodium, solidified beads of steel.

The detection of noticeable concentrations of marker materials (Cu and Mo) in material samples along the entire height of the assembly indicates intense mixing of the melt inside the fuel element simulators until their cladding is destroyed. Powdery materials are distributed uniformly along the height of the assembly with some excess of the concentration of molybdenum in the lower part and copper – in the upper part. The total number of thermite reaction products ejected outside the assembly volume was 75–80% from the initial mass of the thermite mixture. Almost complete blockage of the flow section of the model assembly was found in its lower part.

As a result of the conducted studies of localization of displaced materials along the height of fuel assemblies and blocking of flow sections under conditions of loss of sodium flow rate and behavior of corium at the interface with reactor structures, possible mechanisms of destruction of the cladding of fuel elements stress, thermal interaction of corium with sodium), computational models have been developed. For the first time it has been shown that the time until the cladding melts is 10 s. The obtained results make it possible to verify the computation codes to justify the scenarios of the ULOF type accidents.

Calculations have shown that the time to penetration of the cladding is 10 s. The results obtained make it possible to verify the design codes to justify the scenarios of accidents of the ULOF type.

***Calculation methods and codes.*** Widely recognized three-dimensional model of turbulence by N.I. Buleev [31] give impetus to the development of a complex theoretical and computational studies.

Calculation models have been proposed and substantiated for the development of calculation codes describing the hydrodynamics, heat and mass transfer and physicochemical characteristics of processes in all sections of the hydrodynamic path of nuclear power plants (core, IHX, DHRS et al.) for homogeneous and heterogeneous systems (taking into account the finely dispersed phase in coolants) with liquid.

A model of channel-by-channel thermohydraulic calculation in fast reactor fuel assemblies has been developed for nominal and non-nominal operating modes, taking into account the influence of various factors, including the shape change of the structural elements in reactor core during the campaign [21]. The theory of an anisotropic porous body is proposed and developed as applied to the calculation of complex flows in reactors, heat exchangers and steam generators [14].

To simulate LMC, calculation codes have been created: MIF (core), UGRA and PROTVA (intermediate heat exchanger), MASKA-LM and TURBO-FLOW-LMC (mass transfer of impurities) et al [32].

A problematic issue is the improvement of methods for calculating local turbulent characteristics for single-phase and two-phase flows of liquid metal in channels and large volumes, taking into account large-scale vortex flows, the coolant stratification effect. The problems of validation of methods of thermohydraulic calculation and verification of calculation codes are topical [33].To carry them out, it is necessary to develop verification matrices, systematize the results of experimental studies, create databases on the thermohydraulics of the cores of liquid-metal-cooled reactors, develop a system of verification tests and carry out standard tasks, implement a program of experimental work to obtain additional precision data on the velocity and temperature fields and thermohydraulic characteristics of fuel assemblies with liquid metal cooling.

1. THERMAL HYDRAULIC STUDIES OF THE TARGET MODEL OF THE ACCELERATED-DRIVEN SYSTEM

The target model of the accelerated-driven system is an axisymmetric construction (Fig. 6) installed on the liquid metal “AR-1” facility. The coolant is fed from bottom to top into the horizontal outer annular channel of the model, turns, hitting on the membrane surface, and leaves into the inner channel (tube) through the profiled inlet grid, returning to the circulation pump inlet through the branch pipe located in the lower part.



*Fig. 6. Longitudinal view of the thermal model of the target:*

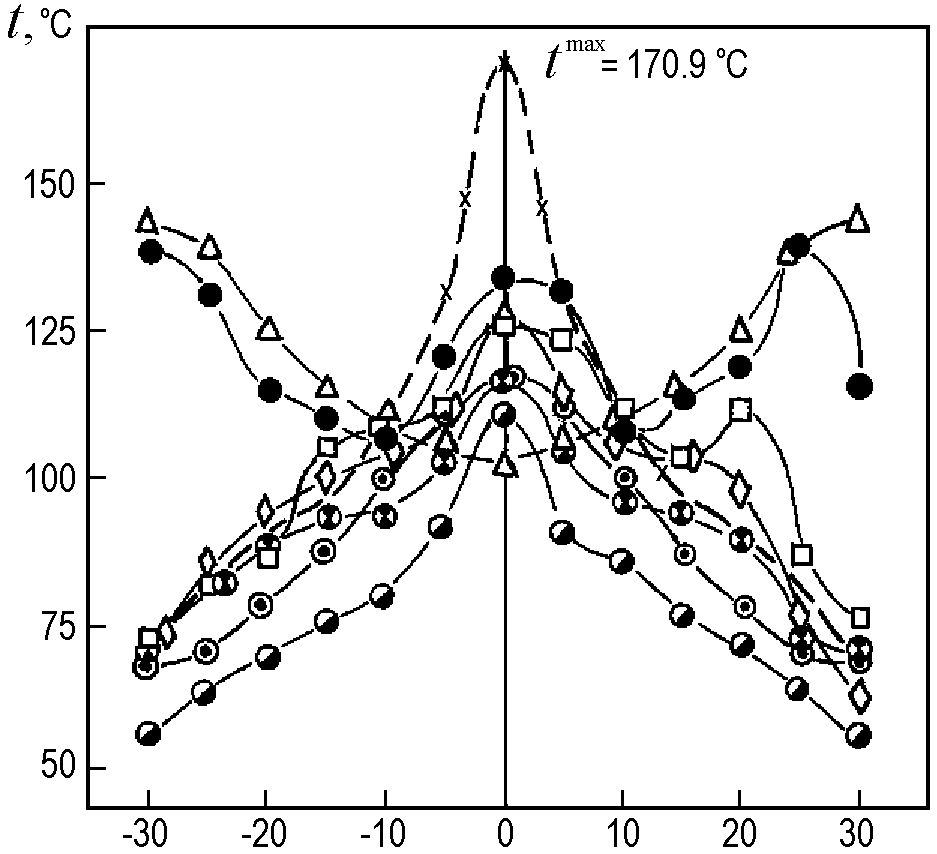
*1 – copper piece with the nichrom heaters inserted; 2 – target membrane;3 – distribution grid; 4 – mobile thermocouple probe*

The membrane is a shaped thin-walled plate made of stainless steel 1.5 mm thick. The heat flux on the membrane surface is created using a copper rod 65 mm in diameter, them end is silver-soldered to the membrane surface. The rod is heated by the “thermal wedge” method. The heating block is a copper bar with a diameter of 130, a length of 200 mm, in which 19 holes with a diameter of 19 mm and a depth of 165 mm are drilled. Ceramic bushings with a diameter of 19 mm made of magnesium oxide with four holes for a spiral with a diameter of 4 mm are inserted into the holes of the block. Study is carried out on a eutectic sodium-potassium alloy as a coolant, which has a close Prandtl number with a lead-bismuth eutectic alloy.

The temperature distribution along the diameter of the central tube was measured in two directions: vertical and horizontal using a movable thermocouple probe. The thermocouple probe consists of two thermocouple combs located opposite the corresponding holes in the profiling grid (Fig. 7). A thermocouple probe was used to measure the instantaneous temperature of the coolant in the cross sections of the inner channel of the target located at 15, 20, 40, 60, 80, 100, 150, 200, 250, 300 mm along the symmetry axis of the target from the outer surface of the membrane, as well as in the space between the membrane and a profiling grid. A narrow “torch” of a high-temperature coolant, moving from the membrane along the axis of the inner channel was observed, the significant fluctuations in the coolant temperature were noted.

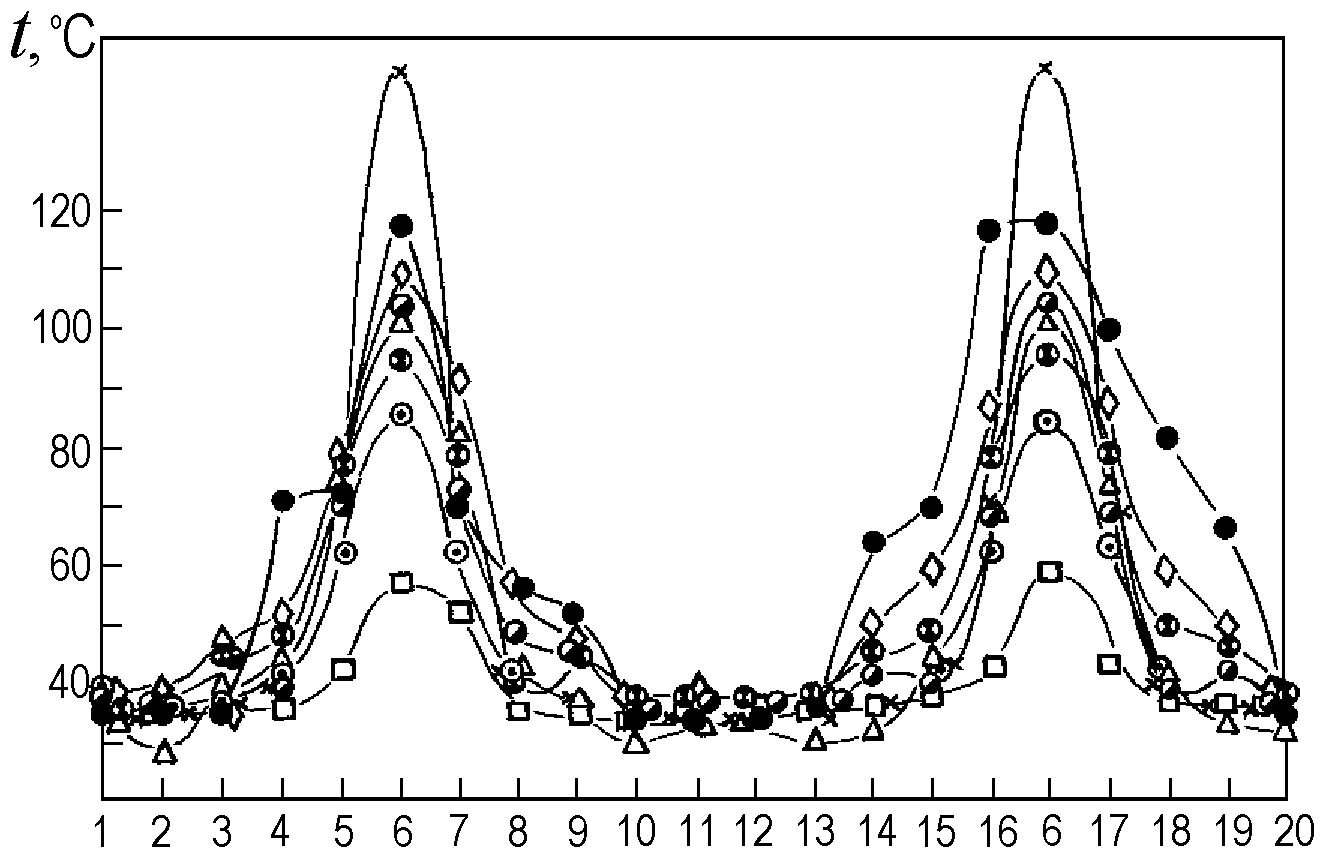
The radially variable temperature near the membrane surface on the flow side was measured with a movable thermocouple located in a capillary 0.6 mm in diameter. The thermocouple moved along the diameter of the heated membrane area in a special channel made on the membrane surface from the coolant side. The data were obtained for the instantaneous temperature of the membrane surface, characterized by a high level of temperature pulsations.

A significant discrepancy in the calculated temperature distribution on the membrane surface in Fig. 7 can be explained by the difference in the approaches used in the calculations for modeling the profiling grid and heat generation.



*Fig. 7. Temperature distribution on the membrane surface, obtained experimentally and as a result of calculations using various codes (y is the distance from the membrane axis):  
● – experiment;△ – FLUENT RNG;□ – STAR-CD; ◊ – AQUA-TM;*  *– AQUA-ASM;* *– FLUENT RSM;◉ – Phoenics;   
× – ANSIS version 5.7.1*

The participants in the benchmark investigated a larger number of turbulence models used in the calculation codes [34]. Not all of the studied turbulence models generated turbulization of the flow due to the interaction of the jets behind the profiling grid (Fig. 8). So only two models of turbulence RNG и “Zero Equation Turbulence Model” reproduced the generation of turbulent viscosity due to the interaction of jets behind the grid. The other four turbulence models available in the ANSYS code gave zero values of turbulent viscosity in almost the entire volume of the inner tube, except for a thin near-wall layer.

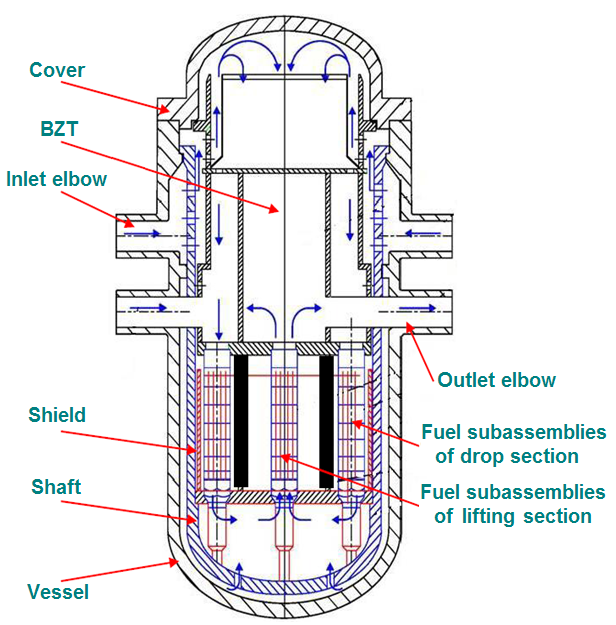


*Fig. 8. The coolant temperature distribution, measured by thermocouples located in the thermocouple probe, and calculated using various codes (distance from the central thermocouple head to the membrane is 1 mm):**● – experiment;△ – FLUENT RNG;□ – STAR-CD; ◊ – AQUA-TM;*  *– AQUA-ASM;* *– FLUENT RSM;◉ – Phoenics;   
× – ANSIS version 5.7.1*

***High-temperature nuclear power plant for hydrogen production and other innovative applications.*** Currently, the main method of hydrogen production is steam reforming of methane. In the long term, large-scale production of hydrogen, this method is not viable due to the consumption of non-renewable resources and the release of greenhouse gases. Alternative methods of producing hydrogen by splitting water using thermochemical or electrolysis processes require a high temperature heat source. The carried out neutron-physical and thermophysical studies have shown that there is a principal possibility of providing the required parameters for a high-temperature reactor plant (900–950 °C) with a fast reactor of 600 MW (th) with a sodium coolant for hydrogen production, based on one of the thermochemical cycles or high-temperature electrolysis with a high coefficient of thermal power utilization [35].The developed methodology and calculation program make it possible to take into account, in the fuel assemblies shaped during the campaign, the effect on the temperature distribution of the cladding of fuel elements and temperature irregularities along the perimeter of fuel elements in fuel assemblies of interchannel exchange, the distribution of flow cross-sections of channels and energy release of fuel elements according to a random law using the Monte Carlo method, as well as others.

1. THERMAL HYDRAULIC STUDIES FOR A SUPERCRITICAL WATER Fast REACTOR PROJECT

The work on the project of a fast reactor plant with supercritical water (SCW) is due to the vast experience in the development and operation of plants with SCW in conventional power engineering. A set of experimental and computational studies was carried out [36] to substantiate the design of an installation with an electrical power of 1700 MW and an experimental reactor with a thermal power of 30 MW.



*Fig. 9. Schematic representation of the arrangement of equipment for the variant with a two-pass flow diagram of the coolant in the VVER-SKD reactor core*

As a result of processing the obtained experimental data, an empirical dependence for calculating the length of a zone with a local deterioration of heat transfer on the mass velocity and heat flux density is proposed. In close bundles, the heat transfer deterioration occurred at low mass velocities and high heat fluxes; in wide bundles, no heat transfer deterioration was observed. The heat transfer increasing in the bundles is facilitated by spacer and mixing grids, which destroy the wall blocking layer, them prevents heat transfer from the wall to the flow center at supercritical pressure. It is necessary to accumulate experimental data and additional analysis of investigation results.

1. Conclusion

Further development of the nuclear power industry in Russia, the implementation of the strategy of two-component nuclear power with the closure of the fuel cycle using fast reactors with sodium coolant, requires the continuation of the implementation of a set of problem-oriented investigations on fast reactor projects.

It should be noted that the constants obtained as a result of studies for the implementation of thermohydraulic computational codes for fast reactors can differ significantly. Their analysis allows us to determine the main components that need to be taken into account, and those that can be neglected in each specific case. These data, along with the experimental determination of the closing relations and verification tests, taking into account the materials used in promising fast reactors, operating parameters and design solutions, will make it possible to carry out a justified verification of the codes.

It should also be borne in mind other thermophysical tasks related to the extension of the service life of fast reactors, the development and creation of promising nuclear power plants with fast reactors with increased coolant parameters, the improvement of methods and devices for cleaning coolants from impurities, devices for controlling the content of impurities, the creation of databases on thermal hydraulics, physical chemistry and technology of fast reactors.

The formation of a technological platform for resource-independent energy stimulates the expansion of the nuclear energy use in the economy. This is the next technological order – atomic-hydrogen energy, according to which nuclear power plants will produce electricity, nuclear fuel and hydrogen.

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