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Fundamental and applied investigations of the liquid-metal cooled fast reactor thermal hydraulics (achieved results and further investigation issues)

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Experimental studies are the basis for solving thermophysical problems in fast reactors

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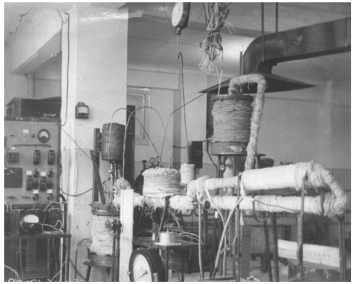
One of the most important problems in design and development of new type fast reactors always has been the thermophysics problems: the choice and mastery of coolants, the heat exchange and hydrodynamics study



First Pb-Bi facility (1951)

Aerodynamic facility «SGDI»

«TT-1M» facility with Pb and Pb-Bi for technological studies

«6B» facility with Na and Na-K for thermohydraulic studies

The needs to develop methods for numerical simulation of hydrodynamics and heat transfer require experimental studies:

- ✓ **to obtain data on the physical laws of thermophysical processes, characteristics of hydrodynamics and heat transfer in reactor facilities**
- ✓ **the relations for closing the equations of numerical models**
- ✓ **verification of computational codes**

To solve these problems, a complex of experimental liquid metal test facilities was constructed, the equipment, modeling methods, experimental techniques and measurement procedures, as well as sensors and automation tools for the acquisition and processing of experimental information were developed.

The problems of the modeling theory for thermophysical processes

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When thermophysical experiments are prepared and conducted, one of the most important issues is fulfillment of the conditions of mechanical, thermal and thermodynamic similarity, which establishes the dependence of the physical properties of a moving heat-conducting medium on the parameters of the state and the most general dependences for describing heat transfer in various liquids under a wide variety of conditions of hydrodynamics and heat transfer in the objects under study.

- ✓ **The similarity analysis and the following criteria and asymptotic solutions should be applied in full. In the event that a sufficiently complete analytical solution or numerical study of a mathematical model is possible, the purpose of the experiments is to test its basic premises and also to refine the calculated coefficients.**
- ✓ **Over the past decades, many monographs and textbooks have been written on this topic. At the same time, S.S. Kutateladze notes that the external simplicity of the basic analysis and the ever-increasing multiparametric nature of the problems of physical and mathematical modeling lead to a lot of misunderstanding and direct errors.**

The aim of this work is to analyze the application of the principles of modeling the processes of hydrodynamics and heat transfer in liquid metal coolants and the theory of the similarity of thermophysical processes when experimental studies are conducted on the models with the use of other media to substantiate hydrodynamics and heat transfer in reactors with liquid metal coolants and to transfer data to the reactor conditions.

Features of liquid (molten) metals

Liquid (molten) metals (Li, Na, K, Cs, Pb, Bi, Hg, Ga, In, and their alloys (Na–K, Pb–Bi, Pb–Li in terms of heat transfer form a special class of coolants with significant volumetric heat capacity and high thermal conductivity. Their coefficient of kinematic viscosity is much lower than the coefficient of thermal diffusivity or, which is the same, the Prandtl number is much lower than one (Pr << 1).

In liquid metal thermal perturbations associated with molecular thermal conductivity propagate deep into the flow at a much greater distance than perturbations of the velocity fields caused by the action of molecular friction. The dynamic boundary layer "drowns" in the thermal boundary layer, and the notion of a thermal boundary layer that extends to the channel center loses its meaning.

Due to a good thermal conductivity induced by electronic conductivity, they are characterized by a high heat transfer coefficient, thus ensuring temperature conditions acceptable for operation of heat transfer surfaces at a high heat flux density.

Liquid metals have a high boiling point and do not require a lot of pressure to prevent boiling.

For most liquid metals, the disadvantage is their high reactivity when they interact with atmospheric oxygen, water, and structural materials. Under certain conditions, this fact will impair heat transfer.

A feature that should be taken into account during heat transfer in liquid metals in the region of the heat-release surface is the state of the impurity composition of the liquid-metal coolant during its circulation close to the heat transfer surface under nominal conditions and in case of deviations from the normal operation conditions.

Thermohydraulic studies of fast reactors with liquid metal coolants

Over the course of more than sixty years of experience in studying liquid metal coolants at the SSC RF - IPPE, the scientific foundations for their application in nuclear energy have been created.

Much attention was paid to the methods of physical modeling of experimental studies of hydrodynamics and heat transfer in nuclear power plants with liquid metal coolants. 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 proven.

At all stages of research, much attention was paid to measurement methods and techniques, including the development of unique sensors for speed, flow, pressure, level, temperature, etc., and means for automating the collection and processing of experimental information.

The complex of hydrodynamic and liquid-metal thermal-hydraulic stands created at the IPPE made it possible to scientifically substantiate the thermal-hydraulic parameters of nuclear power plants, to develop and practically implement highly efficient devices and systems that ensured successful operation and safety, extending the life of existing installations (BOR-60, BN-600), and also to organize work on projects new generation fast reactors with sodium coolant (BN-1200) and heavy coolants – lead-bismuth (SVBR) and lead (BREST-OD-300).

The tasks of introducing fundamentally new technical solutions into projects that improve the design and technical characteristics of power units were solved.

Fulfillment of conditions for physical modeling of hydrodynamic and heat transfer processes

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Direct physical modeling consists in the fact that the process of the same physical nature as in a full-scale specimen is reproduced, but the characteristics of the same name are changed (reduced or increased) by some constant factors. In other words, any two physically similar phenomena can form a pair: a full-scale specimen – a model.

Physical modeling with fairly well-defined confidence boundaries of the results obtained with its help is possible only on the basis of some pre-selected mathematical model.

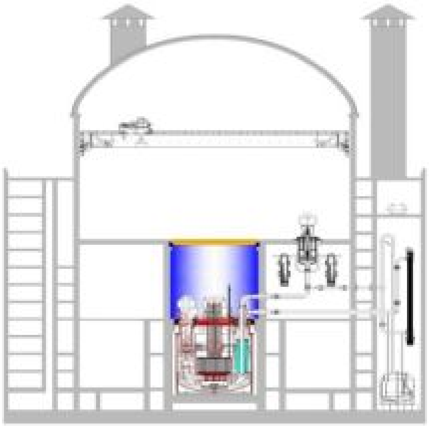
In practice, direct modeling can be unlimitedly applied only to the processes, whose determined similarity numbers are functions of only the geometric simplices of the system and one determining criterion. The presence of two defining criteria, such as the Re and Pr numbers during heat transfer, significantly complicates the simulation. With three defining criteria, direct modeling is usually not feasible. In such cases, it is necessary to set up systematic multivariate experiments.

The purpose of such modeling experiments is to actually reveal the effects permitted by a very general mathematical model, but not reproducible at the modern level of mathematical technologies, neither analytically or in numerical studies. For example, elucidation of complex vortex structures in viscous fluid flows, correlation functions of specific turbulent flows, etc.

In this paper, we present the results of an analysis of the features of the application of the similarity theory of thermophysical processes to modeling:

- ✓ Hydrodynamics and heat transfer in liquid metals are presented for channels of a complex shape.
- ✓ Thermal hydraulics in rod systems with liquid-metal coolants (core and heat exchangers of fast reactors).
- ✓ Temperature and velocity fields in hot chamber of the reactor in various operating modes.

Features of the design of a reactor plant with a fast neutron reactor of high power



Characteristics

Rated heat output 2900 MW

Number of outlet loops 4 electric power, gross 1220 MW

Coolant temperature in the primary circuit, at the inlet/outlet of PTO 550/410°C

Coolant temperature in the secondary circuit, at the inlet/outlet of the SG 527/355°C

Third circuit parameters:

- ✓ flow temperature 510 °C
- ✓ flow pressure 14 MPa
- ✓ feed water temperature 240 °C

Features of the reactor plant

Project full integration of sodium systems and primary circuit equipment in the reactor tank (primary circuit cleaning system in the reactor vessel)

Use of safety systems based on the passive principle of operation, including emergency heat removal systems with autonomous heat exchangers

Built into the reactor pressure vessel use of vessel steam generators

New design solutions are carefully worked out by carrying out computational and experimental studies

Modeling of thermal-hydraulic processes in the tank of fast neutron reactors

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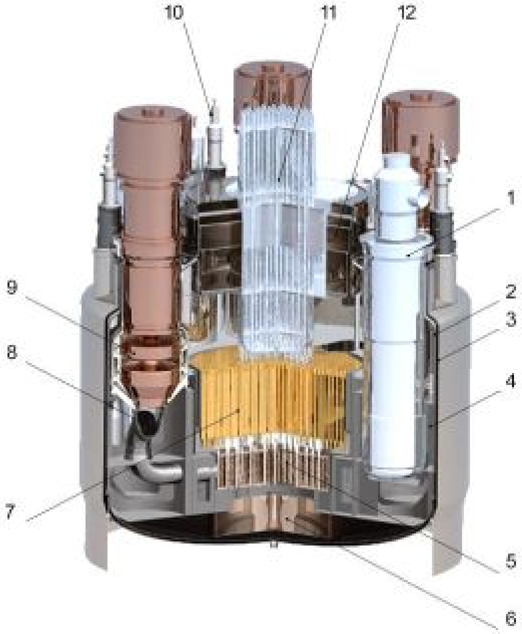
The first circulation loop of a fast reactor with a tank design is a complex combination of elements connected in series and in parallel with different orientations in the gravity field, the shape and geometric characteristics of the flow sections of which change sharply in the direction of motion.



1 - intermediate heat exchanger (IHX);

2, 3 - main and safety cases, respectively;

4 - support belt;

5 pressure chamber;

6 - corium trap (pallet);

7 - core;

8 - pressure pipeline;

9 - MCP-1;

10 - autonomous heat exchanger (AHX);

11 - overload mechanism;

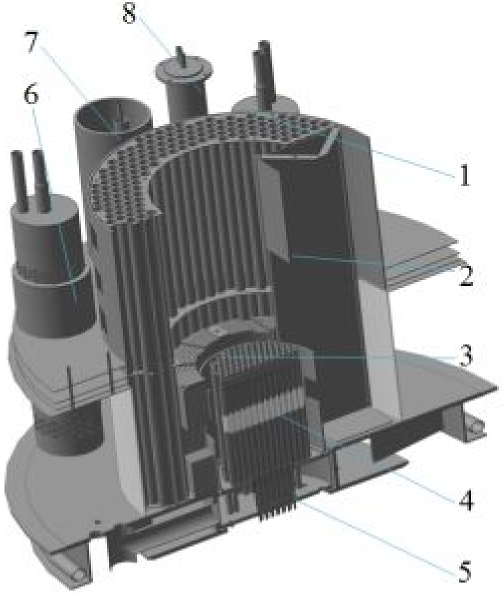
12 - rotary plugs

The use of large-scale models with full-scale coolant leads to a high cost of experimental setups and research.

Errors in modeling thermal hydraulics on fragmentary sector models with isothermal flow are associated with the neglect of spatial 3D effects and temperature inhomogenelity of the flow.

Model of a fast reactor with an integrated layout on a water facility

Studies of the thermal hydraulics of the primary circuit of a fast reactor with a liquid metal coolant with an integral layout of equipment were carried out on a water model on a scale of ~ 1:10.



Experimental water model of the primary circuit of the reactor:
1, 6 – intermediate heat exchanger; 2 – elevator enclosure;
3 – elements of intra-tank protection; 4 – core (fuel assembly simulators); 5 – pressure chamber; 7 – simulator MCP-1;
8 – autonomous heat exchanger

General view
of the experimental model

In vessel equipment experimental model

Modeling of thermal-hydraulic processes in a tank of fast reactors

Accurate modeling of hydrodynamics and heat transfer in the reactor tank on small-scale models with full-scale coolant (liquid metal) is impossible due to the impossibility of simultaneously observing the most important similarity criteria:

Reynolds numbers (Re = wl/ν), Peclet numbers (Pe = wl/a), Froude (Fr = $w^2/g\beta \Delta Tl$).

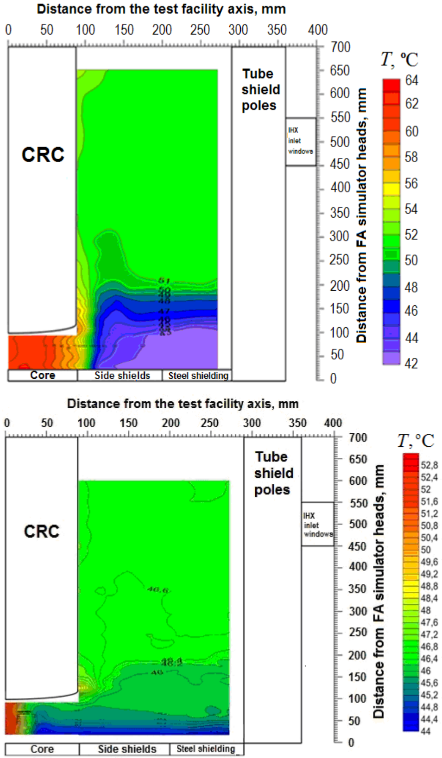
The conducted studies have shown that when the number Re > 10^4, the sizes of stagnant and recirculation formations do not change at equal values of the Froude number for the model and the reactor ($Fr_m = Fr_r$).

Thus, modeling by Reynolds number is not required.

In this case, in a viscous fluid, approximate modeling is possible in the forced circulation regimes using the Froude and Peclet numbers without preserving the Reynolds number.

Approximate modeling for regimes with natural circulation is provided by conservative modeling in terms of the Euler number Eu = $\Delta P/ \rho w^2$.

Thermal-hydraulic processes in the reactor vessel in the nominal mode and in the steady state mode of natural convection



The average temperature of the coolant in the upper chamber when moving movable thermal probes height in nominal and steady state cooling by natural convection

The research results showed that the effect of thermogravitational forces leads to temperature stratification with the appearance of stagnant and recirculation formations, the restructuring of the nature of the flow and temperature regime.

The most commonly used characteristics of a stratified current are the Weissal-Brent frequency and the buoyancy scale.

$$N^2 = -\frac{g}{\rho}\left(\frac{\partial \rho}{\partial z}\right)^{-1}; \qquad l_{пл} = \rho\left(\frac{\partial \rho}{\partial z}\right)^{-1}$$

In a stably stratified turbulent flow, the maximum size of the eddies 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. Internal waves create temperature fluctuations in the equipment wall material with a frequency $f \leq N$.

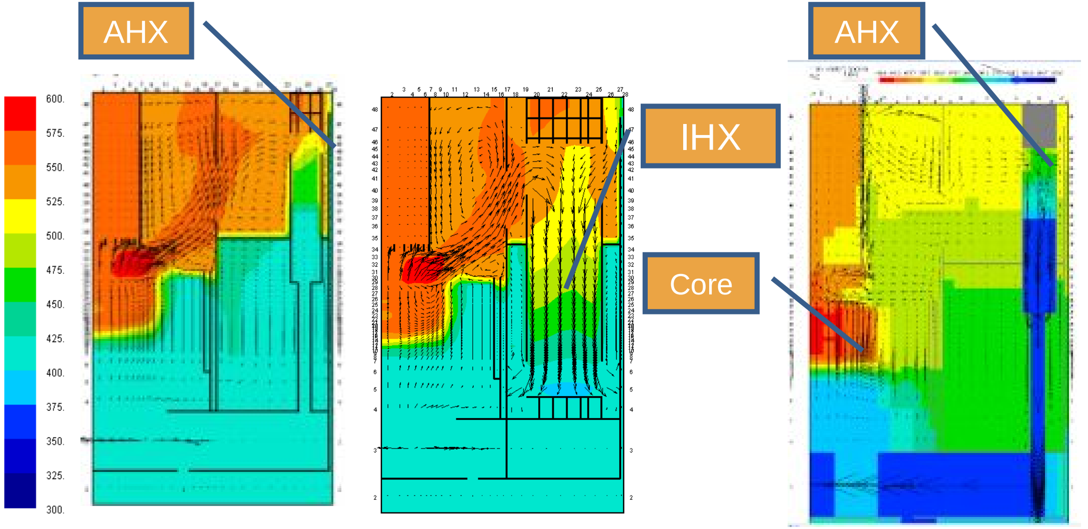
This leads to thermal fatigue of structural materials and a decrease in the service life of reactor equipment.

The steady state mode of natural circulation (cooling down the reactor by natural convection) is characterized by much smaller temperature gradients in the vertical direction above the side screens.

Computational studies according to the GRIF code

GRIF is a complex thermal-hydraulic code designed to calculate the dynamics of thermal-hydraulic parameters in a liquid-metal nuclear reactor, both in stationary and transient modes

The coolant velocity and temperature fields in the reactor vessel



Cross section according to AHX 1 (nominal mode)

Cross section according to IHX 1 (nominal mode)

Cross section according to AHX 1 (cooling mode – τ = 60000 s)

Calculations using the GRIF code showed that the maximum temperature of the fuel cladding in the core is reached in a short-term maximum a few seconds after the initial failure and is 650 °C, which is below the safe operation limit for fuel claddings, and practically does not depend on the number of connected ESAOT loops

Investigations of hydrodynamic characteristics in channels and structures of nuclear power plants

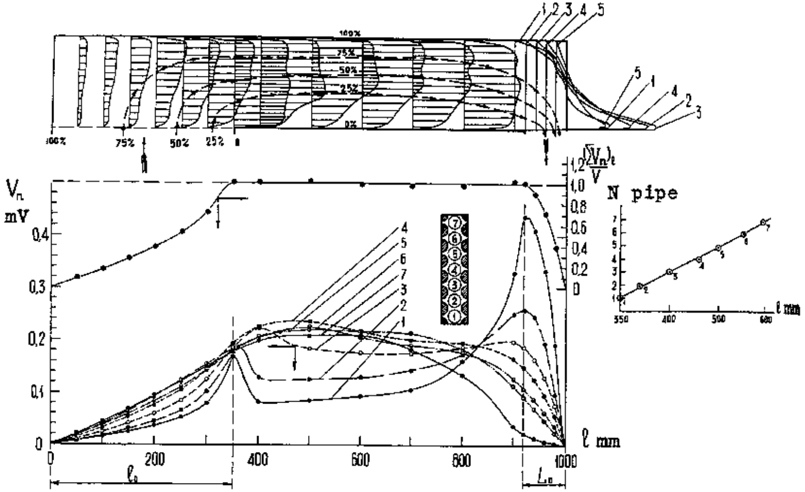
As a result of extensive research, it has been established that hydraulic resistances, velocity distributions for single-phase flows in channels and bundles of rods are calculated simply (± 10%).

But cases are known in the hydraulics of circuits when the real costs 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 a leveling grid after entering the heat exchanger housing).

Despite the fact that there are a lot of reference books on hydraulics, many designs cannot be calculated. Model experiments were required.

Experimental data obtained on the liquid metal by the electromagnetic method of measurements on the velocity distribution in the volume of the intermediate heat exchanger of the fast reactor made it possible to optimize the geometry and increase its efficiency.



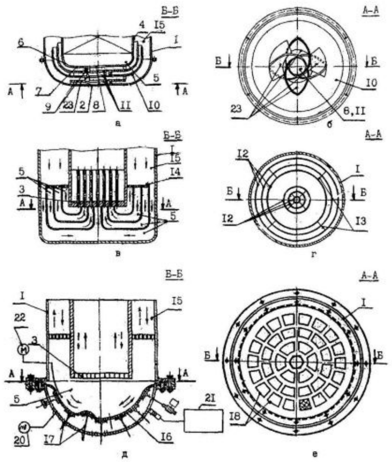
As a result, physically based methods and codes have been created for calculating the hydrodynamic characteristics of the channels of the active zones of reactors, heat exchangers and steam generators with liquid metal coolants.

Features of hydrodynamics in collector systems of reactors, heat exchangers and steam generators

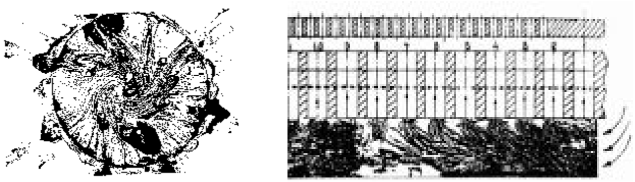
Studies on hydrodynamic models have shown that the coolant flows at the outlet of each circulation loop occupy their own specific area in the core and are weakly mixed.

As a result of a large complex of studies of the hydrodynamics of collectors and the flow part of reactors, heat exchangers and steam generators, a previously unknown pattern of liquid distribution at the outlet of the flow parts of distributing collector systems was established, which was registered as a discovery by V. N. Delnov, B. N. Gabrianovich. and Yuriev Yu.S.

It was found that axisymmetric zones are formed at the liquid outlet from the flow parts of the distributing collector systems, the characteristics of which are determined by the design and technological features of the collector system (fluid supply points, movement trajectory, jet parameters, hydraulic resistance, etc.).



Distribution manifold systems with lateral inlet and central outlet of coolant



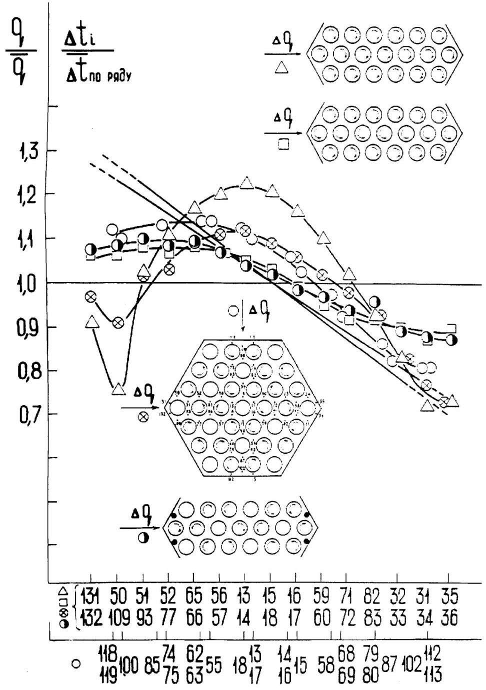
Influence of a Horizontal Planar Vortex on the Distribution of the Coolant Flow in the Reactor Pressure Chamber

Studies of heat transfer in pipes and bundles of fuel elements in a turbulent flow in the reactor core

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, regime, technological, radiation and operational factors during the campaign.

Considering the problems of heat transfer on a large scale, over the past decades, for reactors cooled by liquid metals, a key problem of a fundamental and applied nature in the field of heat transfer in pipes and fuel bundles has been solved.

As a result of numerous experimental and computational studies, extensive data on heat transfer and temperature fields of fuel elements have been obtained for nominal and various non-nominal conditions and operating modes of the core in the presence of deflections of fuel elements, asymmetric shifts and deformations of elements, overlapping of various parts of the core, the presence of counter flows, bursts energy release, statistical distribution of parameters (overheating factors), etc.



Temperature distribution in the cross section at the outlet of fuel assemblies for various types of fuel rod spacing by wire winding

Research of statistical characteristics of velocity and temperature fields in turbulent flows

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Statistical characteristics of velocity and temperature fields in turbulent flows of liquid metals, pulsation characteristics of velocities and temperatures, pulsation intensity, correlation functions, spectral density and probability distribution have been studied.

It has been established that the intensity of pulsations is proportional to the heat flux and essentially depends on the liquid velocity. Spatial correlation coefficients and integral scales of temperature perturbations in the wall differ significantly in individual directions.

As a result of systematic experimental studies, it has been shown that the main cause of contact thermal resistance is the deposition of solid particles suspended in the flow of liquid metals on the heat exchange surface. A strong influence of the contact thermal resistance on the characteristics of temperature fluctuations in the wall is found. Thus, under conditions of unstable contact resistance at moderate thermal loads, temperature fluctuations in the channel wall reached tens of degrees.

The results obtained made it possible to better understand the mechanisms of hydrodynamic and heat transfer processes and outline the ways to construct a physically based theory of turbulent momentum and energy transfer. And although, in general, we understand the physics of these processes, but the heat transfer coefficients for new designs still have to be obtained from experiments for real or model conditions.

The study of the structure and characteristics of the turbulent transfer of momentum and energy in channels of complex shape, their modeling taking into account the anisotropy of the transfer and the influence of spacers are still topical problems in studying the processes and characteristics of hydrodynamics and heat transfer in fuel rod bundles with liquid metal cooling.

Thermal-hydraulic studies of a large-module steam generator of a high-power reactor plant

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For the first time, studies were carried out on a single-tube model at the SPRUT stand to substantiate the design parameters of a new design of a large-module RP steam generator, in which the processes of evaporation and superheating of steam are combined in one housing, in modes of 12.5% and 75% of the nominal sodium flow rate at full-scale parameters in sodium circuit and high pressure water circuit.

The obtained experimental data on the critical heat flux agree satisfactorily with the data of the skeletal tables on the calculation of the critical heat flux in pipes.

There is a strong influence of water pressure both on the critical (boundary) vapor quality and on the value of the heat flux density. With an increase in pressure, an increase in the heat flux density and a decrease in the value of the critical (boundary) vapor content are noted.

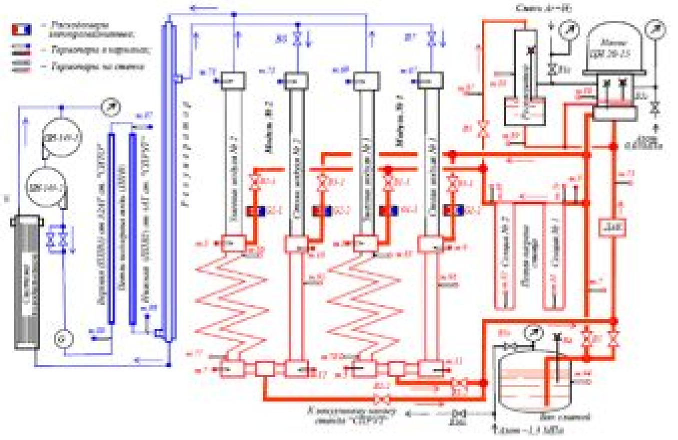
Thermal-hydraulic studies at the "SPRUT" facility to justify the project of a lead-cooled steam generator

Lead temperature at the model input – 540°C

Water temperature at the model input – 340°C

Water flow rate – 100%, 80% и 120% from the nominal value

Steam generator model



Scheme of a lead circuit with a model of a lead-cooled steam generator

When operating at subcritical pressure (about 18 MPa), the steam temperature at the outlet of the model is in the range of 503–509 °C over the entire range of water flow rates.

At a water flow rate of 80-120% of the nominal value, no fluctuations in the water flow rate at the entrance to the model were detected, although according to NIKIET calculations for the SPRUT stand, there should be fluctuations in the water flow rate at the entrance with a magnitude of 50 to 150%.

At supercritical pressure (about 25 MPa), the tests were carried out under the same regime parameters as at 18 MPa. No noticeable differences in the steam temperature at the outlet of the model were found, the discrepancy did not exceed 2–3°C. Water flow pulsations were not noticed.

Violation of normal operation conditions in case of leakage and burning of sodium in the secondary circuit

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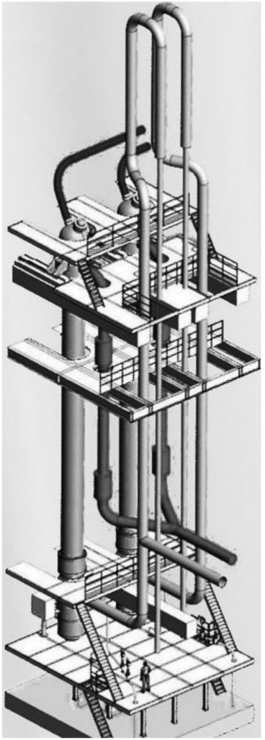

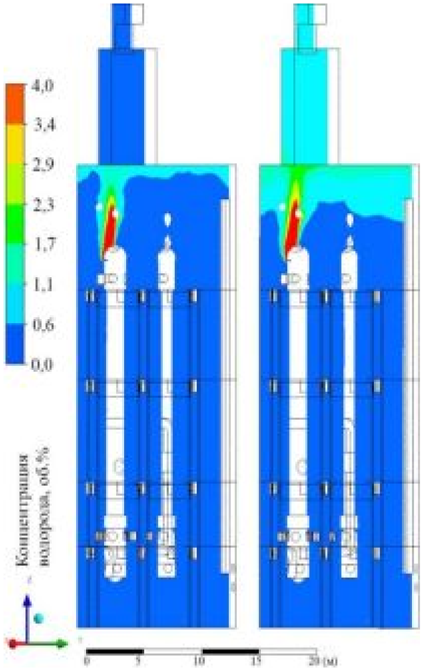

Ventilation from 100% to 33%,
hydrogen consumption 5.1 g / s

In the steam generator box of a reactor plant with a sodium coolant, there is a possibility of the formation of hydrogen-containing mixtures, for example, if the steam generator case is damaged due to water leakage into sodium followed by a circuit burn-through or sodium spills into the room.

Calculations based on the developed methodology showed that during normal operation of ventilation, only a small local cloud of a hydrogen-containing mixture is formed, which is removed from building structures, which ensures that they are not affected by a shock wave, regardless of the combustion mode.

The higher the source of hydrogen is located, the lower its power and total mass, all other things being equal, the conditions for the appearance of hydrogen.

Investigation of sodium boiling in a model of fuel assembly in a fast reactor core

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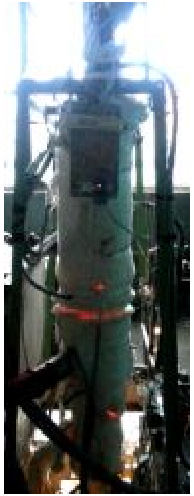
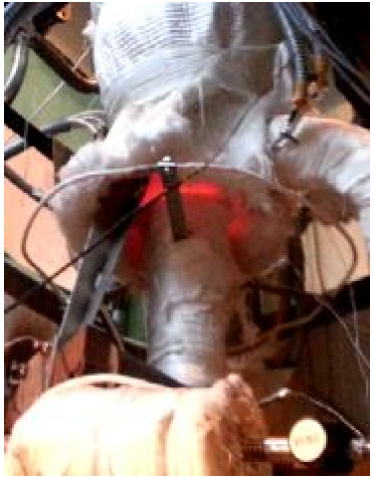
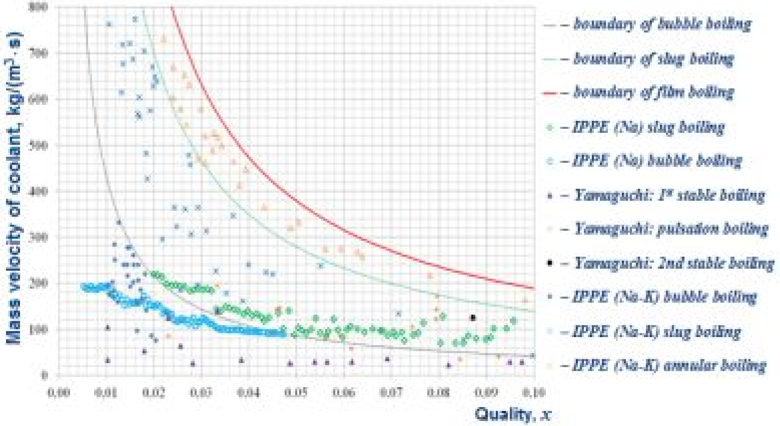




Experimental model in the process of experiment



Cartogram of a two-phase flow of liquid metal in fuel assemblies

The purpose of experimental research is to find design solutions and modes that can provide crisis-free boiling of sodium during heat removal, excluding depressurization of fuel rods in emergency regimes.

For the first time on a fuel assembly model in the presence of a "sodium cavity" above the energy release section, stable sodium boiling for 10 minutes was recorded in forced and natural convection modes

As a result of the data analysis, a cartogram of a two-phase flow of liquid metal coolants in the fuel assemblies of reactors was developed. The obtained data was used to verify the COREMELT code.

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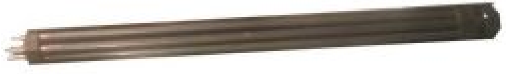
Simulation of an accident in fast reactors with uncontrolled loss of coolant flow

In the zone of global degradation, there were fragments of shells with tear-shaped inclusions of steel and iron

When opening the reaction chamber, removing the 19-rod assembly and dismantling it, it was found that sodium was present on the walls of the reaction chamber in the form of plaque mixed with the products of its thermal interaction with the corium simulator.
The cover of the assembly had burns, on one of the faces the lower collector was partially melted.

Rod bundle



The original configuration of the rod bundle was preserved in the outer layer in its upper part over a length of ~130 to 140 mm. The inner layer of simulators retained its original configuration over a length of 60 to 80 mm. The tip of one of the simulators of the second layer of the beam has shifted to the zone of global degradation.

The lower half of the MSS is a zone of global degradation of shells with multiple fractures, pronounced signs of melting of the shell material, the presence of longitudinal and transverse cracks in the preserved shell fragments.

The end outlets of the assembly are completely blocked by the melted steel, the drainage windows on the edges of the cover are partially blocked.

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Experimental study of the behavior of the corium at the interface between the corium and reactor structures

The phase separation of the melt into metallic (Na) and ceramic (fuel) phases is experimentally shown. The chemical composition of the near-wall layer in the zones of phase localization corresponds to the main component of the phase.

The effect of thermal interaction of corium with nitrogen, which simulates the vapor phase of sodium, is considered. Experimental estimates of melt ejections from the interaction zone due to expansion of the gas phase are made wave, regardless of the combustion mode.

Wear of materials and deformations of samples are registered only for categories subjected to impact action of the melt.

The obtained results make it possible to improve the high-temperature part of the COREMELT code (core melting).

Stratification of the corium simulant melt into (a) metal and (b) ceramic phases

Conclusion

Thermal-hydraulic processes in liquid metal-cooled fast reactors are complex and are formed under the influence of many factors.

The obtained results of extensive experimental studies, the created calculation codes, and the carried out calculation studies made it possible to perform a thermal-hydraulic justification of nuclear reactors with sodium cooling and heavy coolants.

Progress in the field of thermal hydraulics of the core of reactors cooled by liquid metal coolants is determined by the expansion of the range of tasks to be solved, the accumulation of experimental data and their generalization, the development of calculation methods and codes for numerical simulation of thermal hydraulic processes.

Thank you for your attention

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