Integrated researches of topical problems of fast reactors thermal physics

A P Sorokin, Yu A Kuzina, V V Alekseev, V A Grabezhnaya, Yu I Zagorulko, A A Kamaev and Yu I Orlov

State Scientific Centre of the Russian Federation – Leypunsky Institute for Physics and Power Engineering, Joint-Stock Company (IPPE JSC), 249033, Obninsk, Kaluga region, Bondarenko sq., 1

Sorokin@ippe.ru

Abstract. The results of actual studies are presented to justify the thermophysical characteristics and safety of fast reactors with sodium and lead coolants. The influence of a stratified coolant flow on the formation of velocity and temperature fields in a fast reactor vessel with a sodium coolant is shown. The development of the degradation of the fuel assembly during the development of a severe accident with the loss of sodium flow rate in a fast reactor, the blockage of the cross sections of the fuel assembly and the release of marker materials beyond it are demonstrated. The expediency of the combined sodium purification system built into the reactor vessel is shown, in which “cold traps” are an indispensable element, and “hot traps” provide accelerated oxygen purification during operation of the NPP in nominal regime. The results of studies of a large-modular sodium-water steam are presented. As applied to a reactor with a lead coolant, the influence of spacer grids on heat transfer in a fuel assembly of the core is shown. On the model of the steam generator of the reactor installation with lead coolant, it was found that the values of the steam temperature at the outlet of both collectors coincide. The temperature of lead at the exit from the lowering section and in the main path of lead is coincides also. The state and prospects of the development of technology for heavy liquid metal coolants are analyzed. The principle possibility of providing the required parameters of a high-temperature fast reactor with a sodium coolant for the production of hydrogen is shown. The problems of further thermophysical research are analyzed.

1. Introduction
In Russia, there are all the prerequisites (technological and organizational) for the implementation of resource-independent energy, namely, a two-component nuclear energy system with a nuclear fuel cycle closure (NFC) [1–3], which includes nuclear power plants with thermal and fast reactors with sodium coolant that produce energy and reproduce fuel, and a centralized closed NFC. The development of a two-component nuclear energy system with a NFC closure creates the basis for nuclear-hydrogen energy, when nuclear plants will produce electricity, nuclear fuel and hydrogen [4]. In order to increase the safety, economy, environmental friendliness and justification of the reliability of new generation fast reactors with sodium and lead coolants, implemented in various projects [5], extensive studies have been conducted, the main results of which are summarized in this article.
2. Thermophysical studies of fast reactors with sodium coolant

In the design of a promising fast reactor with sodium coolant, a number of principle new technical solutions were used, which provided a significant improvement in the technical characteristics of the power unit, including [5]:

- improved design of the reactor and steam generator (reduction of material consumption);
- decay heat removal system with autonomous heat exchangers built into the reactor vessel (increased reliability);
- internal reactor sodium purification system of the primary circuit (exclusion of pipelines with radioactive sodium and systems serving them);
- internal sodium reactor control system.

As a result there was a need to solve a complex of tasks of computational and experimental substantiation of the above new technical solutions.

2.1. Studies of temperature and coolant velocity fields on the integral model of a fast reactor under various conditions

The circulation loop of the coolant of a fast reactor is a complex combination of series and parallel connected elements with different orientations in the field of gravity, the geometric characteristics of the passage sections of which change dramatically in the direction of the coolant. The coolant in nuclear power plants (NPPs) is always non-isothermal due to the non-uniformity release of energy, the temperature difference between the units of the circulation circuit, the features of heat removal during transient and accident operation.

The results of experimental studies of the IPPE, NRU MEI and JIHT RAS on the integral water model of a fast reactor showed that thermogravitational forces lead to temperature stratification with the occurrence of stagnant and recirculation formations, the restructuring of the flow regime and temperature field. Modeling of thermohydraulic processes in a sodium-cooled reactor vessel using small-scale water models was considered in [6].

At the stratified interfaces between the coolant regions with different temperatures, internal waves arise that cause temperature pulsations on the walls of the reactor equipment [7] (Figure 1). This leads to thermal fatigue of structural materials and reduce the life of the reactor equipment. The established regime of natural circulation is characterized by much smaller temperature gradients in the vertical direction above the lateral blankets.

![Figure 1](image_url)

Figure 1. Distributions of the averaged coolant temperature (a) and intensity of temperature pulsations (b) along the height of the upper chamber, obtained by moving movable thermal probes along the height of the upper chamber, in the nominal operating regime of the installation

The data can be used to verify design codes DINROS, GRIF and codes of the new generation LOGOS, HYDRA, SOCRATES--BN.
2.2. Thermohydraulic studies of a single-tube model of a steam generator in starting, transient regimes of operation and regimes at incomplete power

As a result of studies carried out at the IPPE, data on thermal hydraulics were obtained on a single-tube model at the SPRUT facility (Figure 2) to justify the design parameters of a large-modular steam generator for new construction of large power fast reactor, in which the processes of vaporization and superheating are combined in one casing.

Particular attention was paid to the heat transfer crisis and the associated temperature pulsations of the heat transfer wall. It should be noted that currently there are no experimental data on the critical vapor contents and the corresponding heat fluxes obtained at a pressure of 17–18 MPa and a mass velocity of 1100–1400 kg/(m²·s).

It is fundamentally incorrect to transfer the data on the temperature fluctuations of the heat transfer wall obtained on electrically heated pipes to pipes with liquid metal heating.

The obtained experimental data on the critical heat flux at a pressure below 15 MPa are in satisfactory agreement with the data of skeletal tables for calculating the critical heat flux in the pipe. With increasing pressure, an increase in the density of the heat flux and a decrease in the critical (boundary) vapor content are noted. In all regimes (starting, normal, and transitional), an unsteady heat transfer crisis was observed, characterized by a shift in the crisis zone either to the exit from the steam-generating channel or to the entrance to it. Using the values of wall temperature pulsations, the maximum range of temperature pulsations was determined depending on the flow rate of feed water (Figure 3).

2.3. Experimental studies of the degradation of model fuel assemblies in accidents with uncontrolled loss of sodium flow rate

In IPPE [8], a 19-rod model fuel assembly (FA) was studied at the Pluton facility under conditions simulating an accident with an uncontrolled loss of sodium flow rate. The energy release in the experiment was provided by the reaction of the thermite mixture Al + Fe₂O₃ with stoichiometric composition (working heat of combustion of the mixture \( Q_p = 1.6 \) MJ/kg).

The main objectives of the experiment are to identify the main mechanisms of degradation of the shells of fuel element simulators, and to assess the distribution of marker materials (Cu, Mo, Mg) the height of the assembly in its final state, the study of the phenomena of blockage the cross sections of the model assembly, the release of marker materials beyond the assembly volume.

When conducting experiments under conditions simulating an uncontrolled loss of sodium flow rate, the region of global degradation of the claddings of fuel element simulators was about 65 % of its height and was mainly localized in the part of the rod bundle with an increased termite charge density. According to the results of experimental studies, three main causes of shell degradation were
identified: temperature stresses in the shell material; shell melting; dynamic effects due to the rapid conversion of the thermal energy of the corium simulator melt into mechanical work during the thermal interaction of the melt with sodium. The calculated value of the conversion coefficient (transfer of thermal energy into mechanical work) was 0.115% at an energy output of 4.85 kW.

As a result of the analysis of the distribution of marker materials over the assembly height, fragments of cladding of fuel element simulators, domains of hardened steel melts of iron and iron, conglomerates of thermite reaction products, powdery products of the thermal interaction of the melt of the corium simulator with sodium, and solidified steel flows were found (Figure 4).

![Figure 4](image_url)

**Figure 4.** The beginning of the zone of global degradation (**a**), signs of brittle destruction shells of fuel element simulators (**b**), melting material of shells of fuel element simulators (**c**)

The presence of noticeable concentrations of marker materials (Cu and Mo) in the samples along the entire assembly height indicates intense mixing of the melt inside the fuel element simulators until their cladding is destroyed. The total number of thermite reaction products ejected beyond the assembly volume was 75–80% of the initial mass of the thermite mixture. An almost complete blockage of the passage section of the model assembly was found in its lower part.

In [9], models were developed that were implemented in the BRUT calculation code, and it was shown for the first time that the time to shell penetration was 10 s. The results obtained allow verification of the settlement codes to justify accident scenarios of the ULOF type (Unprotected Loss of Flow – termination of the coolant flow through the reactor).

2.4. **Research of physicochemical processes and technology of sodium coolant**

Tasks of sodium technology: purification of sodium from impurities and control over their content, safe operation of the reactor installation in operating conditions and during repair work were successfully solved when creating domestic plants BR-5, BOR-60, BN-350, BN-600, BN-800 [10, 11]. It was decided to place all systems with radioactive sodium in the reactor vessel [5], the dimensions of the I loop purification system, and therefore, their performance and impurity capacity, are limited.

The required quality of the sodium coolant is supported by special purification agents using various physical methods: sedimentation, distillation, filtration, and purification with cold and hot traps (CT and HT). Based on the results of the studies, the last two methods were selected [10, 11].

The research results determined the domestic approach to designing a cold trap, which has three consecutive zones: a cooled settling tank, a zone of final cooling, and an isothermal filter. CT tests showed that in it sodium is purified from oxygen and hydrogen efficiently (when the sodium stays in the trap for more than 15 minutes, the impurity retention coefficient is close to one). Purification of sodium from corrosion products, especially carbon, is less effective.

To increase the capacity of CT by impurities, it is necessary that the distribution of deposits in the trap be uniform.

Placing CT in the reactor vessel leads to dangers associated with the accumulation of hydrogen in it, using argon for cooling CT at a pressure of 1.5 MPa, and the possibility of heating it with sodium in the reactor vessel. To eliminate these defects, specialists from IPPE proposed at the output of I and II
circuits to provide temperatures of 150 and 120 °C, respectively, which would eliminate the accumulation of hydrogen in the cold trap of the primary circuit. The calculations confirmed the possibility of implementing such regimes [10].

According to the studies performed on the models, there is a fundamental possibility of creating CT with an admixture capacity several times higher than that laid down in previous projects. For this, the design thermohydraulic and mass transfer characteristics of the CT are optimized using codes TURBOFLOW и MASKA–LM [11].

The use of a cold trap built into the reactor vessel with a sodium cooling system allows increasing the CT admixture capacity, the purification rate, and improving safety compared to the gas cooling option. The temperature level of the coolant in modern installations allows the use of getters to purify sodium from impurities [12].

These results indicate the feasibility of developing a variant of a combined purification system, in which CT is an essential element of the purification system built into the reactor vessel, and accelerated purification from oxygen can be carried out in the HT during operation of the NPP in nominal regime.

Among operational methods for controlling the amount of impurities in sodium, the main focus is on the use of cork indicators (PI), sensors with diffusion membranes, and electrochemical methods [13]. An electrochemical hydrogen control sensor with a nickel membrane based on stabilized zirconia is currently being introduced at nuclear power plants. Using models of homogeneous and heterogeneous mass transfer of impurities in sodium circuits, computer codes have been developed for calculating the integral mass transfer of hydrogen and tritium, as well as corrosion products of structural materials [14].

2.5. The main tasks of the further development of thermohydraulic, physicochemical and technological research as applied to fast reactors

Thermohydraulic studies:

- development of modeling methods for physicochemical, thermohydraulic and technological processes in all sections of the hydrodynamic path (CORE, IHE, AHE, DHR), taking into account their non-stationarity;
- obtaining basic constants for heat transfer and temperature fields of fuel elements for all operating conditions and regimes (changing geometry, bursts of energy release, statistical distribution of parameters, overheating factors, etc.);
- improving methods for calculating local thermohydraulic turbulent characteristics for single-phase and two-phase flows of liquid metal in channels and large volumes, taking into account large-scale vortex flows, stratification of coolants.

Research in the field of physical chemistry, mass transfer and coolant technology:

- obtaining data on the solubility, diffusion characteristics and dispersion of complex liquid metal heterogeneous systems, the behavior of such systems, taking into account the spontaneous formation of crystalline phase nuclei from supersaturated solutions;
- study of the mechanism and kinetics of the formation and decomposition of complex oxides and carbon compounds in a non-isothermal circuit;
- determination of minimum concentrations of oxygen and other impurities in sodium, at which the performance of structural materials is maintained;
- obtaining fundamental data on physicochemical processes for ternary (for example, sodium – iron – oxygen, sodium – chromium – oxygen) and more complex systems in sodium, which are necessary to justify models embedded in codes, and conducting experiments to verify codes;
- creation of a combined system for the purification of sodium from impurities, including radionuclides and suspensions, built into the reactor vessel;
- improvement of sodium impurity control devices.
Safety studies:

- using a complex of codes, analysis of the features of homogeneous (hydrogen, tritium) and heterogeneous (impurities with low solubility: corrosion products, carbon, fuel, fission products) mass transfer of impurities and development of measures to minimize their accumulation in stagnant zones, to eliminate the occurrence of abnormal situations: task maximum to exclude the possibility of the formation of a “depot” of impurities in liquid metal systems;
- substantiation of nominal conditions excluding the formation of vortices in the collector of the core and on the surface of sodium (gas capture), passive circulation zones;
- verification of a new technical solution for core cooling in regimes with sodium boiling (sodium cavity above the reactor core): it is necessary to determine the boundaries of unstable operation, to study the dynamics of the distribution of the boiling region in a real fuel assembly;
- to increase the safety of a large module steam generator, the development of materials and structures that ensure the slowdown of self-development and leakage processes and operational repair of SG after a water leak into sodium.

3. Thermohydraulic studies to substantiate a reactor with a lead coolant

3.1. Experimental studies of the thermal hydraulics of a fuel assembly and a steam generator

Studies of the thermophysical characteristics of the reactor core with a heavy coolant on a model fuel assembly with spacer grids of the peripheral zone of the core showed that, at Peclet numbers \(200 < \text{Pe} < 1200\), the influence of these grids on heat transfer is insignificant.

In the temperature distribution for the central measuring fuel element simulator inside the spacer grid, there are three local maximums that occur at the points of contact between the rod and the spacer grid. For use in the calculations of data on temperature rises, they are dimensionless and presented in the form of formulas. In [15], it was shown that the values of periodic temperature non-uniformities are within acceptable limits for the reactor.

Work in support of the steam generator of the BREST reactor began in 2011 at the SPRUT facility on a three-pipe model of the steam generator (Figure 5).

Earlier, experiments were carried out on the same lead circuit to determine the heat transfer coefficients on the lead side in the transverse flow of the coolant, which partially covered the range of coolant velocities typical for starting conditions (Figure 6).

![Figure 5. An experimental model of a steam generator of a reactor installation with twisted pipes on a loop of the SPRUT facility with a lead coolant](image)

![Figure 6. The dependence of the dimensionless heat transfer coefficients on the lead side in the transverse flow of the coolant in the model of the steam generator on the Peclet number: 1 – calculation according to the CKTI formula; 2 – calculation by the IPPE formula; 3 – when heated by lead; 4 – when heated by an alloy of lead with bismuth](image)
3.2. State and prospects of development of technology for heavy liquid metal coolants

Given the positive experience in operating transport-type reactor facilities (nuclear submarines, etc.), today we can distinguish three main tasks that can be solved using coolant technology (Pb-Bi or Pb) in civilian reactor plants [16]:

- ensuring the purity of the coolant and the surfaces of the circulation circuit to maintain the design of thermohydraulic characteristics with large operating resources;
- prevention of corrosion and erosion of structural materials during long-term operation;
- ensuring modern safety requirements at various stages of the operation of the reactor installation (preparation of the coolant, start-up, current operation, repairs and overloads, depressurization, regimes of deviation from the conditions of normal operation).

To solve these problems, methods and means of coolant technology are being developed, which include:

- hydrogen purification of the coolant and the circuit from slag-forming impurities;
- regulation of dissolved oxygen in the coolant for corrosion protection of steel against corrosion;
- coolant and protective gas filtration;
- coolant control both in reactor and in non-reactor conditions.

4. Innovative sodium coolant fast reactor for hydrogen production and other innovative applications

As a result of neutron-physical and thermophysical studies of a reactor with a sodium coolant (BN–VT) with a thermal power of 600 MW, it was shown that it is possible in principle to provide the required parameters of a high-temperature fast reactor for the production of a large amount of hydrogen, for example, based on one of the thermochemical cycles or high-temperature electrolysis with a high coefficient of thermal use of electricity [4]. Safety requirements will be respected. Relatively small dimensions, the type of coolant, the choice of fissile material and structural materials make it possible to create a reactor with increased nuclear and radiation safety.

The system for purification high-temperature sodium from hydrogen is based on a fundamentally new method – evacuation through special membranes, while the permeability coefficient of the system for purification the second loop from tritium should exceed 140 kg/s.

Further study is required by the processes occurring in high-temperature heat resistant materials upon irradiation [4].

The results presented in this article do not exhaust the range of scientific and technical problems that need to be solved when creating new generation fast reactors with liquid metals.

5. Franco-Russian cooperation in the field of fast reactors thermal physics

Franco-Russian scientific and technical cooperation in the field of thermophysics of fast reactors with sodium coolant has been carried out since the beginning of the 1990s. The purpose of the cooperation is to increase the level of scientific and technical substantiation of design solutions and characteristics of fast reactors with sodium coolant, increase the efficiency and safety of their work [17].

6. Conclusion

As a result of the studies of hydrodynamics, heat and mass transfer in liquid metal coolants in relation to the thermophysical substantiation of fast reactors, the mechanism of heat transfer in liquid metals has been clarified, the laws of hydrodynamics and heat transfer in channels of complex shape and equipment elements, the fundamental physico-chemical laws of the coolant – impurity – protective gas – materials system have been investigated. Thermohydraulic parameters and highly efficient technological processes are scientifically substantiated, devices and systems have been developed and practically implemented that ensure the successful operation of fast reactors with original technical solutions that have no analogues in world practice.
References

[1] Ponomarev-Stepnay N.N. A two-component nuclear energy system with a closed nuclear fuel cycle based on BN and VVER // Atomic energy. 2016. V. 120. Issue 4. P. 183–191

[2] Klinov D.A., Gulevich A.V., Kagramanyan V.S., Dekussar V.M., Usanov V.I. Challenges and incentives for the development of sodium fast reactors in modern conditions // Atomic energy. 2018. V. 125. Issue 3. P. 131–136

[3] The use of liquid metals in nuclear, thermonuclear energy and other innovative technologies / V.I. Rachkov, M.N. Arnoldov, A.D. Efano, S.G. Kalyakin, F.A. Kozlov, N.I. Loginov, Yu.I. Orlov, A.P. Sorokin // Thermal engineering. 2014. No. 5. P. 20–30

[4] Neutron-physical and thermophysical studies in support of high-temperature nuclear energy technology with a fast reactor with a sodium coolant for hydrogen production / S.G. Kalyakin, F.A. Kozlov, A.P. Sorokin, G.P. Bogoslovskaya, A.P. Ivanov, M.A. Konovalov, A.V. Morozov, V.Yu. Stogov // Izv. universities. Nuclear energy. 2016. No. 3. P. 104–115

[5] The concept of a promising power unit with a fast reactor BN-1200 / V.I. Rachkov, V.M. Poplavsky, A.M. Tsibulya, Yu.E. Bagdasarov, B.A. Vasiliev, Yu.L. Kamanin, S.L. Osipov, N.G. Kuzavkov, B.N. Ershov, N.R. Amirmetov // Atomic energy. 2010. V. 108. Issue 4. P. 201–205

[6] Ushakov P.A., Sorokin A.P. Modeling problems of emergency natural convection heat removal in the upper plenum of LMR using water // Proc. of 8th Intern. Conf. on Nuclear Engineering (ICONE-8). Baltimore, USA. April 2–6, 2000. ICONE-8078

[7] Experimental studies of temperature and velocity fields on integrated water model of fast reactor in various operating modes / A.N. Opanasenko, A.P. Sorokin, A.A. Trufanov, N.A. Denisova, E.V. Sviridov, N.G. Razuvanov, V.G. Zagorsky, I.A. Belyaev // Atomic energy. 2017. Vol. 123. Issue 1. P. 28–35

[8] Zagorulko Yu.I., Kashcheev M.V., Ganichev N.S. The mechanisms of initial degradation of fuel assemblies of fast reactors // Atomic energy. 2015. V. 119. Issue 2. C. 75–79

[9] Kashcheev M.V., Sorokin A.P. Design study of severe accidents in fast reactors with sodium coolant // VANT Ser.: Nuclear and reactor constants. 2017. Issue 4. P. 24–36

[10] Kozlov F.A., Sorokin A.P., Konovalov M.A. Purification systems for sodium as a coolant of nuclear power plants with fast reactors (retrospective perspective view) // Izv. universities. Nuclear energy. 2015. No. 3. P. 5–19

[11] Sodium coolant purification systems for nuclear power plants with a BN-1200 reactor / V.V. Alekseev, Yu.P. Kovalev, S.G. Kalyakin, F.A. Kozlov, V.Ya. Kumaev, A.S. Kondratiev, V.V. Matyuhkin, E.K. Pirgov, G.P. Sergeev, A.P. Sorokin // Thermal engineering. 2013. No. 5. P. 9–20

[12] Kozlov F.A., Konovalov M.A., Sorokin A.P. Purification by getters of liquid metal systems with sodium coolant from oxygen // Thermal engineering. 2016. No. 5. P. 63–69

[13] Kozlov F.A., Sorokin A.P., Konovalov M.A. Sodium purification systems for nuclear power plants with fast neutron reactors // Izv. universities. Nuclear energy. 2015. No. 3. P. 5–15

[14] Calculation of impurities mass transfer in cold traps with sodium cooling / V.V. Alekseev, F.A. Kozlov, A.P. Sorokin, E.V. Varseev, V.Ya. Kumaev, A.S. Kondratiev // Atomic energy. 2015. V. 118. Issue 5. P. 257–261

[15] Studies of temperature fields and heat transfer in model fuel assemblies of a reactor with a heavy coolant (homogeneous geometry) / Yu.A. Kuzina, V.V. Privzentesve, A.P. Sorokin, K.S. Rymkevich // VANT. Ser.: Nuclear reactor constants. 2017. Issue 4. P. 15–23

[16] Martynov P.N., Orlov Yu.I. Modern approaches to the technology of heavy coolants // New industrial technologies. 2011. No 1. P. 3–6

[17] Sorokin A.P., Poplavsky V.M., Trufanov A.A., Kozlov F.A., Kuzina Yu.A., Alekseev V.V., Latger C. Cooperation with French experts in the field of fast reactors thermal physics // Questions of atomic science and technology. Series: Nuclear and Reactor Constants, 2017. Issue 2. Electronic Materials http://vant.ippe.ru/archiv/year2017.html