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Research Article

Thermohydraulic studies of alkali liquid metal coolants for justification of nuclear power facilities*

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Abstract

The paper presents and discusses the results of experimental and computational studies obtained by the authors on hydrodynamics and heat exchange in fuel assemblies of the alkali liquid metal cooled fast reactor cores, and experimental data on hydrodynamics of flow paths in the heat exchanger and reactor header systems. Investigation results are presented on in-tank coolant circulation obtained using a well-developed theory of approximation simulation of the nonisothermic coolant velocity and temperature fields in the fast neutron reactor primary circuit and demonstrating stable stratification and thermal fluctuations in the coolant. Results are presented from experimental and computational simulation of the alkali liquid metal boiling process based on fuel assembly models during an emergency situation caused by an operational occurrence involving simultaneous loss of power for all reactor coolant pumps and the reactor scram rod failure. Objectives are formulated for further studies, achieving which is essential for the evolution of the liquid metal technology, as dictated by the need for the improved safety, environmental friendliness, reliability and longer service life of nuclear power facilities currently in operation and in the process of development.

Keywords

Alkali liquid metals, fast reactors, hydrodynamics, heat exchange, core, reactor tank, distributing header system, steam generator, boiling

Introduction

In the autumn of 1950, as part of discussing the proposals by A.I. Leypunsky, the Section of the Chief Directorate's Scientific and Technical Council recommended that Laboratory "V" (currently, the Institute of Physics and Power Engineering or the IPPE) focus its activities on development of liquid metal cooled reactors. Requirements were formulated with respect to coolants taking into account their influence on physical, process, corrosive and thermohydraulic characteristics of reactors, toxicity and cost. The list of liquid metals and alloys used or considered as candidates for application in nuclear power includes lithium, sodium, eutectic sodium and potassium alloy, potassium, cesium, lead, eutectic lead and bismuth alloy, and gallium.

On 24 June 1954, a heat engineering department was established at the IPPE transformed further into the thermophysical sector which was led by V.I. Subbotin, and later by P.L Kirillov and A.D. Yefanov. The key research

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fields were thermal hydraulics, mechanisms of turbulent heat exchange, processes of liquid metal boiling and condensation, systematization, analysis and generalization of thermophysical data, establishment of an experimental thermophysical data base, heat tubes, thermal physics of thermionic generators, high-temperature nuclear power systems for outer space applications, and thermonuclear plants (Efanov et al. 2015).

The need for providing a scientific thermophysical rationale for nuclear power plants and nuclear power facilities of a new type currently in the process of development required new methodologies, dedicated equipment and an experimental framework to match the challenges posed by the BR-10, BOR-60, BN-350, BN-600, BN-800 and BN-1200 fast reactor projects. An integrated system of hydrodynamic, liquid-metal thermohydraulic and process test benches built at the IPPE has made it possible to implement these projects and prepare for the experimental justification of innovative solutions for nuclear power facility (fast reactor) designs of a new generation (Thermophysical Bench Framework 2016).

A sixty-year experience of adopting alkali liquid metals (sodium, eutectic sodium-potassium alloys, lithium, cesium), jointly with the industry's institutes, academies of sciences and experimental design bureaus engaged in development of nuclear power and propulsion systems, has led to the scientific basis set up for their application in nuclear power, and thermohydraulic parameters and processes justified, which have provided for the successful operation of fundamentally new nuclear power facilities. The combined experience of operating sodium cooled fast neutron nuclear power facilities (BR-10, BOR-60, BN-350, BN-600, BN-800) exceeds 200 years and is over 6 years and a half in the event of those with sodium-potassium coolant for spacecraft applications (BUK, TOPOL, TOPAZ) with rated parameters. Long operating times (of several decades) with the use of a sodium-potassium alloy have been recorded for the BR-5, DFR and RAPSODIE operation (Rachkov et al. 2014).

This has been contributed by many years of international cooperation in using sodium in nuclear power facilities with foreign countries (Great Britain, Germany, the Republic of Korea, the USA, France, the Czech Republic, Japan, and others).

The progress achieved as a result of adopting alkali liquid metal technologies have made it possible to propose the liquid-metal technology for various engineering applications: NPPs with fast neutron reactors (sodium), metallurgy and chemical industry (sodium and sodium-potassium), spacecraft propulsion systems (sodium-potassium, cesium, lithium), fusion or thermonuclear reactors (lithium), etc.

Implementing the strategy of a two-component nuclear power with a closed fuel cycle using sodium cooled fast neutron reactors (Ponomarev-Stepnoi 2016), achieving a competitive edge, and maintaining the priority enjoyed by Russia in the field of NPPs with sodium cooled fast neutron reactors, including designs of a fast neutron reactor with a gas turbine unit (GTU) and a high-temperature reactor for nuclear hydrogen power (BN-VT), require further problem-oriented thermohydraulic studies.

Hydrodynamics and heat exchange in channels and fuel rod bundles

Methodology of investigations

All stages of investigations have given a great deal of attention to measurement methodologies and techniques, including development of unique velocity, flow rate, pressure, level, temperature and other sensors. Microthermocouples were developed to measure temperature in safety cans with an outer diameter of 0.3 to 0.8 mm operating in a temperature interval of 300 to 1800 °C. Flow meters of different designs were developed to measure flow rates of liquid metals. Methodologies and a technique were developed later for measuring electromagnetically the liquid metal local flow rate (velocity) vectors in channels and fuel rod bundles, and measuring the coolant mixing characteristics in experiments in air with a small fraction of gaseous tracers added in the form of Freon or propane (Rachkov et al. 2018, Sorokin et al. 2021b).

Much attention is given to methods of simulating physically experimental studies on hydrodynamics and heat exchange in liquid metal cooled nuclear power facilities. It has been experimentally shown that it is possible to simulate hydrodynamics of incompressible fluids, including liquid metals in experiments with air, and heat exchange in liquid metals, such as Na, Na-K, Li, Hg, Pb, Pb-Bi, etc., using simulating fluids (Sorokin and Kuzina 2019). The said methodologies have made it possible to undertake a broad range of fundamental and applied experiments.

Channels and fuel rod bundles

Extensive studies have been undertaken into hydrodynamics of irregularly shaped channels, including rod bundles, and flow paths of reactor facilities; maximum attention was given to measurement of velocity fields, distribution of tangential stresses, and turbulent characteristics.

The results of experimental studies into hydrodynamic turbulent characteristics in fuel assemblies in air using a Pitot tube have shown (Fig. 1a, b) that there is a local maximum observed in the distribution of tangential stresses along the wetted perimeter of the regular fuel lattice cell at the point with the greatest channel expansion, which can be explained by the secondary vortex impact (Sorokin et al. 2021b).

In a deformed lattice (Fig. 1c, d), the distribution of tangential stresses along the wetted surfaces is practically symmetrical with respect to the geometrical symmetry axes of the flow section, though anomalies are observed in some of the FA portions, which can be explained by the impact of certain secondary vortexes not only inside the channels, but also at the boundary. The velocity distribution along the normal to the wetted perimeter is described by a universal law if the local tangential stress value is used to calculate the dynamic velocity. There is a substantial intensification of turbulent

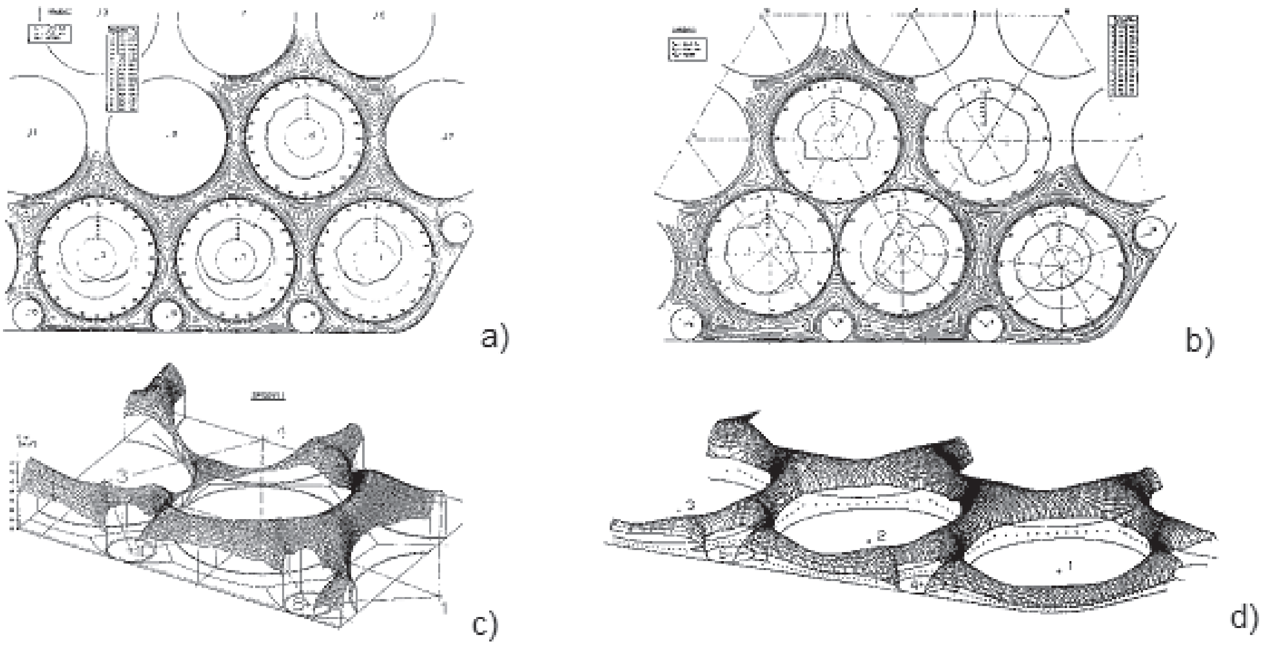

Figure 1. Distributions of tangential stresses on the fuel rod surface and velocity in the channel cross-section in the peripheral region of a model FA with displacers in the peripheral channels with a nominal geometry (**a, b**) and during a fuel lattice strain in the FA peripheral region (**c, d**).

velocity fluctuations observed in the bundle's peripheral region as compared with an infinite lattice (Sorokin et al. 2021b).

Most attention has been given to thermohydraulic studies for the most heated and essential component of a reactor plant (the reactor core) affected, in the course of life, by a number of factors, including design, mode, process, radiation and operating factors (Rachkov et al. 2018). It was shown that there was no thermal (contact) resistance at the coolant – heat-exchange surface interface when the concentration of impurities in the coolant did not exceed their solubility at the circulating metal temperatures. In these conditions, heat transfer to liquid metals, such as Na, Na-K, Li, Hg, and Pb-Bi, in tubes is described by a single criterial relationship close to the Lion formula. When the coolant is saturated with impurities, the heat transfer coefficients are expected to decrease by a factor of one and a half to two, which corresponds to experiments by the Energy Institute (ENIN), the Central Boiler and Turbine Institute, and the IPPE (Rachkov et al. 2018).

Experimental and computational studies have shown the need for solving the 'conjugate' problem of heat removal from fuel rods with regard for their thermophysical properties. P.A. Ushakov developed a theory of approximate thermal similarity of fuel rods in regular lattices (Sorokin and Kuzina 2019) that has made it possible to simulate fuel rods by multilayer tubes electrically heated from the inside and is used in all investigations for fast reactor cores. Detailed experimental data have been obtained for thermal hydraulics of full-scale core models provided there are fuel rod deflections, asymmetrical displacements and deformations of components, closure of different core parts, and counter flows.

As a result of experimental studies and a computational and theoretical analysis of the mass, pulse and energy exchange among the channels in bundles of smooth and wire wrap finned fuel rods, physically justified methods and programs (TEMP, MIF) have been developed for thermohydraulic calculations of reshaped fast neutron reactor core fuel assemblies (Zhukov et al. 1991).

The influence of the fuel rod geometry and materials, and the radiation-induced swelling and creep effects on the FA temperature mode has been investigated, and peculiarities of the core temperature mode formation in the course of operation (life) have been identified for fast neutron reactors. The efficiency of using differently directed wire wraps leading to oppositely directed coolant flows in transverse directions has been shown.

Hydrodynamics of the heat exchanger and reactor header system flow paths

Extensive and prolonged experimental studies were conducted in a wind tunnel bench and on a water table for hydrodynamics of flow paths in different types of axisymmetrical flat-plate and cylindrical distributing header systems (DHS) with different conditions of liquid supply and removal (Gabrianovich and Delnov 2016).

DHSs with central supply and side removal of liquid

A water flow pattern has been obtained for the flow path of a flat-plate DHS with central supply and side removal of water. It has been found that the liquid flow pattern in the header is defined by the DHS dimensional ratio and design.

Liquid flow models for cylindrical-type DHSs

The liquid flow in the cylindrical-type DHS flow path is of a complex nature and is defined predominantly by the DHS dimensional ratio and design, the liquid flow pattern, and the hydraulic resistance coefficient for the outlet component flow path. Representative models of liquid flows in the flow path of the above DHS type (Fig. 2) have been obtained with regard for results of various experimental studies (Gabrianovich and Delnov 2016).

DHSs with side supply and central removal of liquid

A water flow pattern was obtained on a water table for the flow path of a flat-plate DHS with side supply and central removal of water. It has been found that the liquid flow pattern in the header is defined by the DHS dimensional ratio and design.

The liquid flow in the flow path of a cylindrical-type DHS is of a complex type and is defined predominantly by the DHS dimensional ratio and design, the liquid flow pattern, and the hydraulic resistance of the lattice. Representative models of the liquid flow in the flow path of the considered DHS type have been obtained with regard for the results of experimental studies (Gabrianovich and Delnov 2016). The most graphic peculiarities of the liquid flow manifest themselves in the inlet, main and outlet portions of the flow path in the DHS under consideration (Fig. 3).

Scientific discoveries

As a result of the studies, the earlier unknown regularity and phenomenon, dealing with nuclear, space, metallurgical and chemical fields of science and technology, have been identified and registered as scientific discoveries.

Regularity

It has been found that there is an earlier unknown regularity of the liquid distribution at the outlets of the flow paths in distributing header systems that consists in that axisymmetrical regions form as the liquid exits the header, the characteristics of which are defined by the design and process peculiarities of the header system (liquid supply point, motion path, jet parameters, hydraulic resistance, etc.) (Delnov et al. 2019).

Phenomenon

It has been found that there is an earlier unknown phenomenon of the hydrodynamic identity occurrence in distributing header systems that consists in the similarity of the hydrodynamic characteristics of the flow paths in axisymmetrical distributing header systems, e.g., nuclear power facilities and heat exchangers with different conditions of supplying and removing the liquid flowing in the system (Delnov 2021).

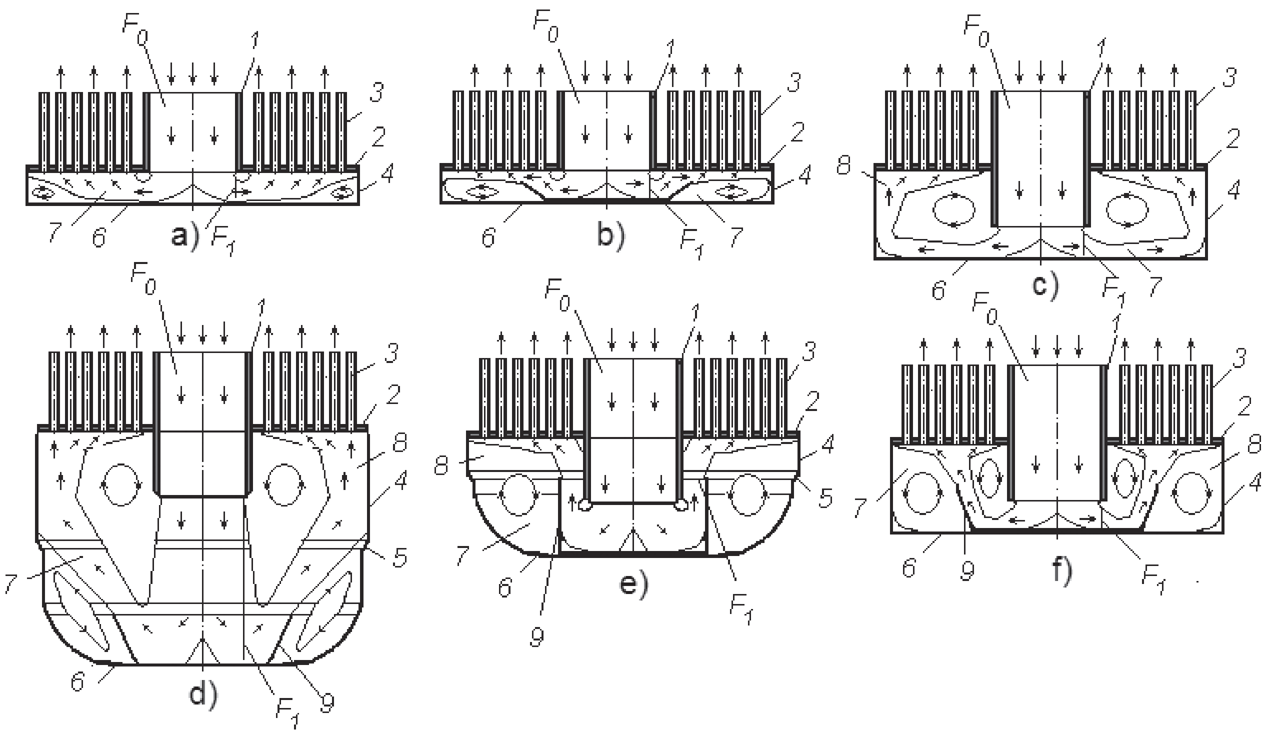

Figure 2. Typical designs and liquid flow models in axisymmetrical DHSs of a cylindrical type with central supply of water to the header: **a, b.** Superconstricted DHSs without the central tube extension with and with no distributor respectively; **c.** constricted DHSs with the central tube extension and a header with a constricted inlet section; **d.** Constricted DHS with a distributor, the central tube extension and a header with a free inlet section; **e, f.** Constricted DHSs with a distributor, the central tube extension, and a header with constricted and free inlet sections; 1 – central tube; 2 – tube sheet; 3 – bundle tube; 4 – housing; 5 – stage; 6 – bottom; 7 – header; 8 – side annulus, 9 – distributor.

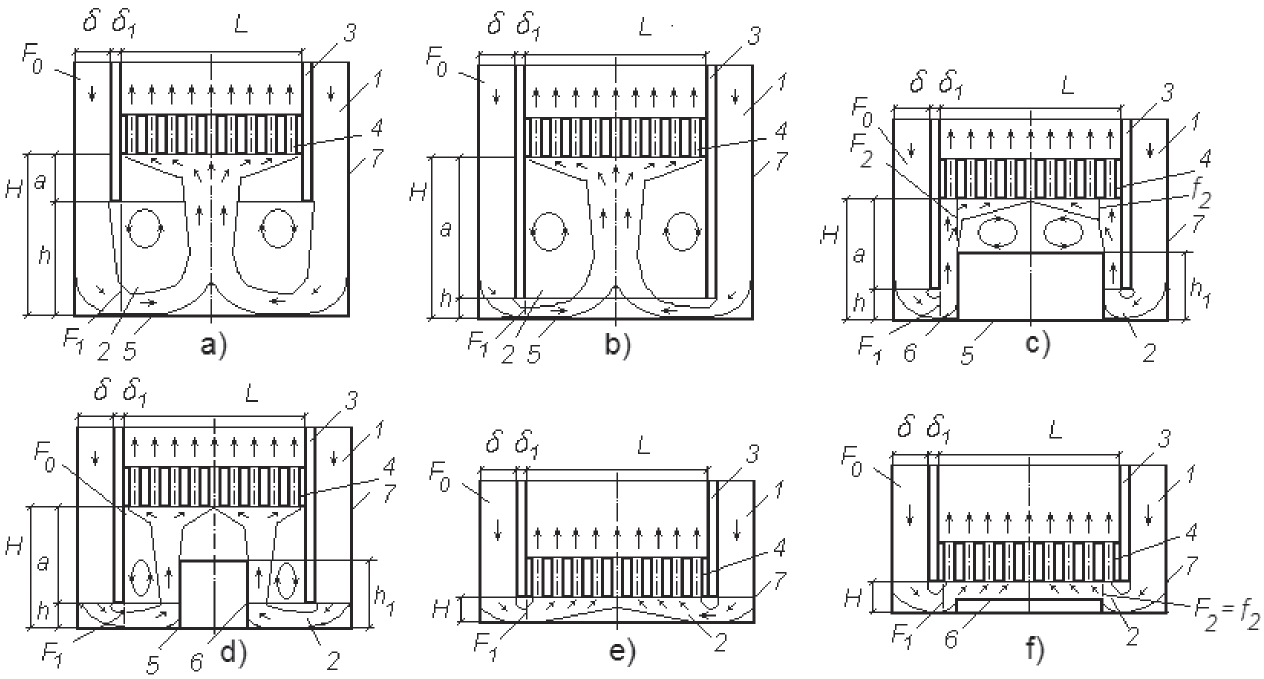

Figure 3. Typical designs and liquid flow models in the flow paths of axisymmetrical DHSs of a cylindrical type with side supply of liquid to the header: **a, b.** constricted DHSs with the displaced tube sheet in a shell and with free and constricted inlet sections respectively; **c, d.** constricted DHSs with a constricted inlet section, the displaced tube sheet in a shell and inserts of relatively large and small diameters respectively; **e, f.** superconstricted DHSs with a header constricted by the inlet section, with no lattice displacement in a shell with no and with inserts respectively; 1 – annulus; 2 – header; 3 – shell; 4 – lattice; 5 – bottom; 6 – insert; 7 – housing.

Differences in designs and hydrodynamics of flow paths in DHSs of different types

DHS design differences: a tube sheet and a system of plates are used as the outlet component respectively in cylindrical and flat-plate DHSs:

- axisymmetrical round, cylindrical, conical and annular jets occur in DHSs of a cylindrical type, and jets with a rectangular (square) cross-section are typical of flat-plate DHSs;
- in cylindrical DHSs, one type of jets is transformed upstream into another type, and flat-plate DHSs have the common jet divided into individual parts or individual parts of the jet converge into one jet;
- DHSs with different liquid supply and removal points differ in the sequence for transformation of certain types of jets into others.

General characteristic of scientific discoveries

Scientific discoveries

- have changed the existing scientific concepts in the field of hydrodynamics of axisymmetrical DHSs;
- have explained the scientific facts and experimental data not rationalized in scientific terms earlier;
- have shown an extremely strong effect of minor changes in the DHS design and dimensional ratio on the DHS liquid flow;
- have made it possible to obtain empirical relationships for determining the hydrodynamic irregularities at the outlet of different DHS designs and types;
- have allowed predicting, prior to experiments and calculations, the DHS designs with the required liquid flow profile at the header outlet;
- contain data on the models of liquid flows in different DHS types;
- have identified the existence of constricted and free DHSs the liquid flow in which differs greatly from that in thoroughly studied classic superconstricted and free DHSs.

Scientific discoveries have been used to justify the flow paths of DHSs in reactors and heat exchangers of nuclear power facilities and to develop and verify the DHS flow fluid dynamics codes.

In-vessel circulation, equipment components

In-vessel circulation

Experimental studies based on an integrated water model of reactors using a developed theory of approximation simulation, temperature fields and the structure of the nonisothermic coolant movement in the primary circuit components of a fast neutron reactor for forced circulation modes, and changeover to the cooldown mode and emergency cooldown by natural coolant circulation (Opanasenko et al. 2017), demonstrate a substantial and stable temperature stratification of the coolant in the peripheral region of the reactor's upper (hot) chamber above the side shields, in the cold and discharge chambers, in the elevator baffle, in the reactor vessel cooling system, and at the intermediate and auxiliary heat exchanger outlets in different modes of the heat exchanger operation (Figs 4, 5).

High gradients and fluctuations of temperature have been recorded at the interfaces of stratified and recirculating formations. The obtained results can be used for verification of codes and coarse estimation of the reactor plant parameters during recalculation based on similarity criteria.

Heat exchangers and steam generators

The Protva and Ugra codes, set up based on a well-developed theory of an anisotropic porous body for the calculation of complex flows in reactors, heat exchangers and steam generators, were used to prove the possibility of using in the BN-800 design the heat exchangers with the same heat-transfer surface as in the BN-600 design. Characteristics of heat exchange, critical heat fluxes and circulation stability have been studied for steam generators of reactor plants with the BN-350, BN-600 and BN-800 reactors and a fundamentally new large-module steam generator for an advanced fast reactor (Grabezhnaya and Mikheev 2015).

Boiling of alkali liquid metals in fuel rod bundles

Investigations into the liquid metal boiling based on fuel assembly (FA) models have shown three patterns for a two-phase liquid metal flow in fuel rod bundles (bubble, slug and annular dispersed flow patterns) which is limiting with respect to the assembly cooling. It has been shown that long-term cooling of the core is conceptually possible in emergency modes involving boiling of liquid metals. Heat transfer during boiling of liquid metals in fuel rod bundles has been studied, the effects of the fuel rod surface roughness on the development of the boiling process has been investigated, and a diagram of the two-phase liquid metal flow patterns in fuel rod bundles has been plotted [Sorokin et al. 2018, 2019].

The results of computational studies for a system of parallel FAs (Fig. 6), undertaken based on an upgraded version of the SABENA code (Sorokin et al. 2021a) implementing a two-liquid model of a two-phase liquid metal flow in an approximation of equal pressures in a steam or liquid phase, reproduce the variation of temperature, the development of single-phase (bubble, slug) flow patterns, and the liquid metal flow rate fluctuations obtained in experimental studies, and demonstrate antiphase coolant flow rate fluctuations in parallel FAs, the interchannel stability characterized by a major growth in the coolant flow rate fluctuation amplitude in parallel FAs as compared with single FAs, a periodic drop in the FA coolant flow rate practically to zero, and potential FA exposure (departure from nucleate boiling).

To exclude the development of an emergency caused by an operational occurrence with simultaneous loss of power for the reactor coolant pumps and a failure of the reactor scram rods (an ULOF accident), a design solution has been proposed with a 'sodium plenum' arranged above the reactor core. Comparing the calculation and experiment results has shown the possibility of heat removal by the boiling coolant in a model FA with a 'sodium plenum' during thermal loads of 10% to 15% and the sodium flow rate level of about 5% of the nominal values (Sorokin et al. 2018).

The activities to systematize thermophysical data in the field of alkali cooled fast reactors were undertaken under the auspices of the IPPE's Thermophysical Data Center in many fields of fast reactor thermal physics: velocity and temperature fields in the core, in the hot

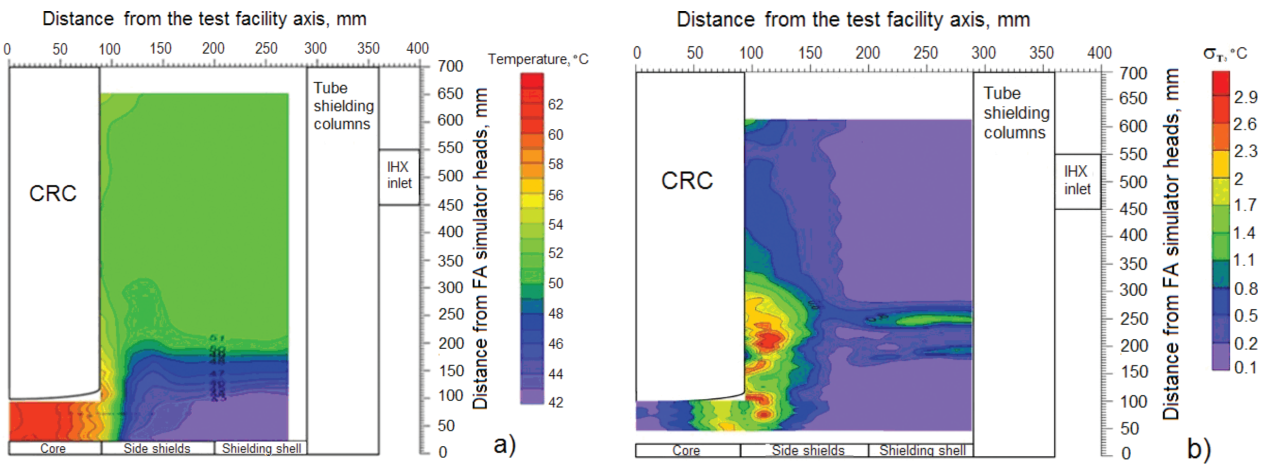

Figure 4. Axial upper chamber distributions of the averaged coolant temperature (**a**) and the intensity of temperature fluctuations (**b**) obtained during the movement of mobile temperature probes along the upper chamber height in the nominal mode of the plant operation.

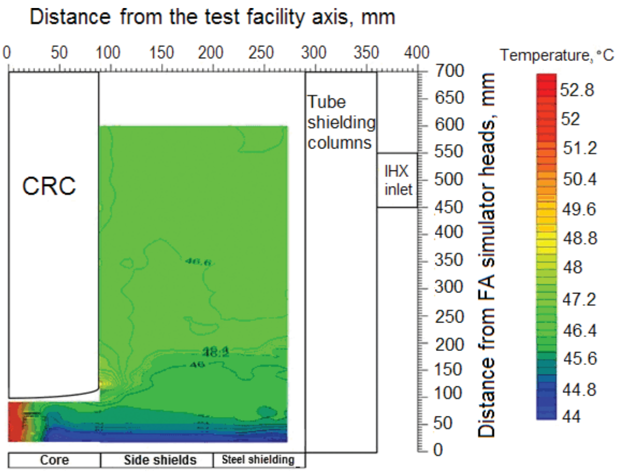

Figure 5. Averaged temperature field along the upper chamber height in the steady-state natural convection cooldown mode.

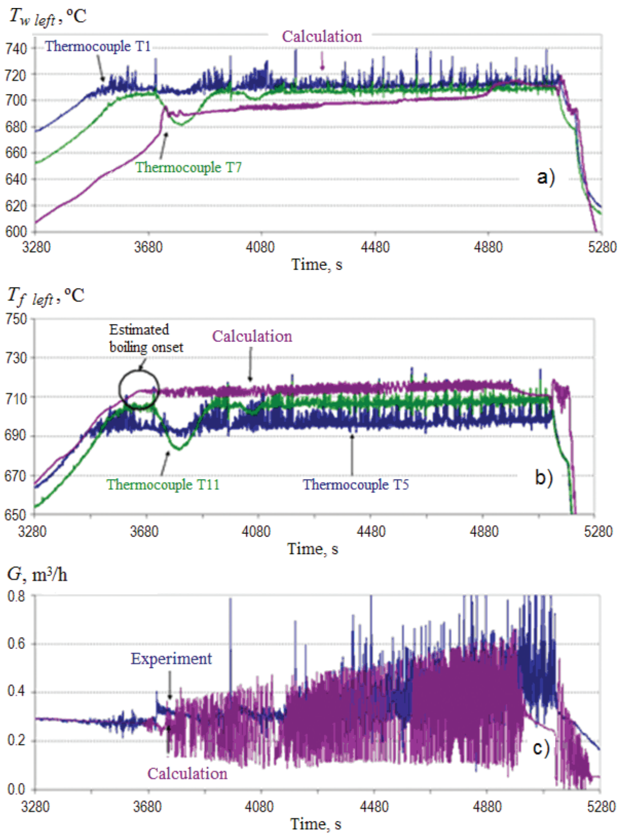

Figure 6. Comparison of calculated and experimental time-dependent distributions of the fuel simulator surface temperature (**a**), coolant temperature (**b**) and coolant flow rate (**c**) in the left-hand FA during parallel operation of FAs.

chamber, and in heat exchangers and steam generators, hydraulic resistance and heat transfer in channels and fuel rod bundles for a single-phase flow, heat transfer and the diagram of two-phase flow patterns, departure from nucleate boiling in assemblies, and thermophysical properties of coolants and reactor materials. Technical guidelines and reference books have been developed (Kirillov et al. 2010).

Objectives of further thermohydraulic studies in nuclear power f1acilities for alkali liquid metal coolants:

- refining methods to calculate local turbulent characteristics of pulse and energy transport for single- and two-phase liquid metal flows in channels and in large volumes taking into account large-scale eddy currents, and the coolant flow stratification effects;
- development and justification of a system of verification tests;
- development of a verified system of codes taking into account the interconnection of nuclear physical, thermohydraulic, physicochemical, thermomechanical, mass-exchange and engineering processes taking place in a nuclear power facility, for justifying its service life with regard for the entire combination of its operating processes and modes;
- evaluation of temperature fluctuations directly in the coolant flow and on the channel walls, and investigation of the effects these fluctuations have on structural strength;
- justification of the core's temperature mode taking into account random deviations of parameters (geometrical, operating and process parameters, uncertainties of thermophysical properties, calculated constants, etc.);
- analyzing the consequences of potential nonstandard modes (interlocks, emergency heat removal, boiling) and development of measures to prevent these from progressing into a severe accident;
- investigating the dynamics of the sodium boiling region propagation in a real FA with a 'sodium plenum' above the core;
- justification of nominal modes that exclude the formation of eddies in the core header and on the sodium surface (gas capture), and passive circulation zones (stratification phenomena, temperature fluctuations);
- estimation of temperature fluctuations directly in the coolant flow and on the channel walls, and investigation of the effects these fluctuations have on structural strength;
- development of a system for detection of anomalies in an individual FA or in a number of FAs prior to the fuel cladding failure;
- justifying, for large-unit SGs, thermohydraulic modes and hydrodynamic stability, and integrated tests of the automatic steam generator protection systems (all scram systems shall be supplied and assembled to support the tests, and the scram system bench shall be put into operation);
- development of the material and design for improving the safety of the large-unit steam generator to provide for the slowdown of the leakage self-development and development processes, and online repair of SGs after the water escape into sodium.

Conclusions

The existing experience in adoption of liquid metal coolants makes it possible to believe that these have gained a rightful place in nuclear power on an equal basis with water coolants. However, despite this fact, one cannot think that all problems have been resolved and it only remains to replicate the accumulated experience when building new reactor facilities. Therefore, objectives are formulated in the paper achieving which is essential for the development of the liquid metal technology. An important methodological conclusion follows from previous works: the greatest efficiency of studies is achieved by combining fundamental experimental and computational and theoretical studies, building pilot prototypes on their basis, and investigating their characteristics with the subsequent transition to commercial products.

References

- Delnov VN (2021) Application for Discovery: No. A-678, dated 25.02.2021. Discovery Priority: 2018. Discovery Diploma No. 524 from MAANOiI. The Phenomenon of the Emergence of Hydrodynamic Identity in Distributing Header Systems. [In Russian]
- Delnov VN, Gabrianovich BN, Yuriev YuS (2019) Application for Discovery: No. 70, dated 17.04.2019. Discovery Priority: 26.11.2012. Discovery Diploma No. 518 from MAANOiI. Regularity of the Liquid Distribution at the Distributing Header System Flow Path Outlets. [In Russian]
- Efanov AD, Kozlov FA, Rachkov VI, Sorokin AP, Chernonog VL (2015) Scientific School of the SSC RF – IPPE "Heat and Mass Transfer, Physical Chemistry and Technology of Coolants in Power Systems". In: Sorokin AP, Trufanova AA, Vereshchagina TN (Eds) Scientific and Technical Collection "Results of Scientific and Technical Activities of the Institute of Nuclear Reactors and Thermal Physics in 2014". Obninsk. SSC RF – International Production & Processing Expo (IPPE) publication, 24–51. [In Russian]
- Gabrianovich BN, Delnov VN (2016) Peculiarities of flow path hydrodynamics in header systems of the nuclear power plant heat exchangers and reactors. Sarov. RFYaTs – VNIIEF publication, 216 pp. [In Russian]
- Grabezhnaya VA, Mikheev AS (2015) Experience in experimental validation of the steam generators used in NPP with fast reactors. Atomnaya energiya 119(2): 106–116. https://doi.org/10.1007/s10512-015-0036-0
- Kirillov PL, Bobkov VP, Zhukov AV, Yuriev YuS (2010) Handbook on thermohydraulic calculations in nuclear power engineering. V. 1. Moscow. IzdAT publication, 770 pp. [in Russian]
- Opanasenko AN, Sorokin AP, Trufanov AA, Denisova NA, Sviridov YeV, Razuvanov NG, Zagorsky VG, Belyaev IA (2017) Experimental studies of the coolant temperature and velocity on an integrated water model of a fast reactor in different operating modes. Atomnaya energiya 123(1): 21–27. https://doi.org/10.1007/s10512-017-0295-z
- Ponomarev-Stepnoi NN (2016) Two-component nuclear power system with a closed nuclear fuel cycle based on BN and VVER reactors. Atomnaya energiya 120(4): 233–239. https://doi.org/10.1007/s10512-016-0123-x
- Rachkov VI, Arnoldov MN, Yefanov AD, Kalyakin SG, Kozlov FA, Loginov NI, Orlov YuI, Sorokin AP (2014) Use of liquid metals in nuclear and thermonuclear power engineering and other innovative technologies. Teploenergetika 61(5): 337–347. https://doi.org/10.1134/S0040601514050085 [In Russian]
- Rachkov VI, Sorokin AP, Zhukov AV (2018) Thermohydraulic studies of liquid metal coolants in nuclear power plants. Teplofizika vysokikh temperatur 56(1): 121–136. https://doi.org/10.1134/S0018151X18010145 [In Russian]
- Sorokin AP, Kuzina YuA, Ivanov YeF (2018) Heat transfer during boiling of liquid metal coolants in fuel assemblies of fast reactors in emergency modes. VANT. Ser. Yaderno-reaktornye konstany 3: 176–194. https://vant.ippe.ru/year2018/3/thermal-physics-hydrodynamics/1548-17.html [In Russian] [or] https://vant.ippe.ru/en/year2018/3/thermal-physics-hydrodynamics/1527-17.html [English version] [accessed Dec. 05, 2021]
- Sorokin AP, Kuzina YuA (2019) Physical simulation of hydrodynamic and heat transfer processes in NPPs with liquid metal cooling. Teploenergetika 8: 1–9. https://doi.org/10.1134/S0040363619080095 [in Russian]
- Sorokin AP, Kuzina YuA, Ivanov YeF (2019) Peculiarities of heat exchange during liquid metal boiling in fuel assemblies of fast reactors in emergency modes. Atomnaya energiya 126(2): 73–82. https://doi.org/10.1007/s10512-019-00518-0
- Sorokin AP, Ivanov YeF, Kuzina YuA, Denisova NA, Nizovtsev AA, Privezentsev VV, Sorokin GA (2021a) Experimental and computational studies into heat exchange and stability of circulation during liquid metal boiling in fuel assemblies of fast neutron reactors in emergency modes. Teploenergetika 68(10): 752–762. https://doi.org/10.1134/S0040601521090093 [In Russian]
- Sorokin AP, Kuzina YuA, Denisova NA (2021b) Hydrodynamics of turbulent flows in fast reactor fuel assemblies (velocity field and microstructure of turbulence). VANT. Ser. Yaderno-Reaktornye Konstany 2: 139–166. https://doi.org/10.55176/2414-1038-2021-2-139-166 [In Russian]
- Thermophysical Bench Framework of the Nuclear Power Industry in Russia and Kazakhstan (2016) In: Pershukov VA, Arkhangelsky AV, Kononov OYe, Sorokin AP (Eds) Sarov. RFYaTs – VNIIEF publication, 160 pp. [In Russian]
- Zhukov AV, Sorokin AP, Matyukhin NM (1991) Interchannel exchange in fuel assemblies of fast reactors (codes and practical application). Energoatomizdat Publisher, Moscow, 224 pp. [In Russian]