Codes of new generation - current state of development and prospects for further development

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Abstract. The work on codes of new generation development was initiated in 2010. Codes of new generation are intended for the safety assessment of NPP with FR (BREST-OD-300 or BN-1200) with closed nuclear fuel cycle. Under “codes of new generation” is understood commercializable software product with the following characteristics: based on the state-of-the-art level of the theoretical models and experimental data; founded on the efficient numerical algorithms; written according to the up to date requirements of the programming languages standards and adjusted to the up to date computing technics; provided with the graphical user interface; documented: users’ guide, reference manual, programmers guide; automatically connected with the CAD systems. There is one more requirement to the developed codes of new generation, that is obligate according to the Russian federal rules and regulations to use them for NPP safety assessment: codes should be certified by Rostechnadzor. The first priority was to develop the codes that did not have the Russian analogues but that are required for safety assessment. Starting from 2010 the high-level team was formed including the specialists from scientific institutes, organizations of the State Atomic Energy Corporation ROSATOM, institutions of high education, combining specialist from different areas. This allows achieving the following key results to the mid of 2018: 26 codes have been developed with the different level of detailization and complexity, including integral multi-physical codes (neutronics MCU-FR, ODETTA, CORNER, NDP-ACE, BPSD, COMPLEX; thermohydraulic HYDRA-IBRAE/LM, LOGOS, CONV-3D, KUPOLE-BR; fuel rod behavior code BERKUT; codes for analysis of fission products transport in the environment, including safety validation of the disposal of all types of prepared radioactive wastes, ROM, ROUZ, Sibylla, GeRa; codes for modeling processes of the closed nuclear fuel cycle KOD TP and VIZART; code for probabilistic safety analysis CRIS 5.3; integral codes for NPP safety justification SOCRAT-BN and EUCLID), 6 of them are certified in Rostechnadzor; the codes of new generation are distributed by the User License Agreements; the First Codes of New Generation User School was conducted for the young specialists of atomic and other industries; the codes of new generation are used for NPP with BREST-OD-300 and BN-1200 reactors safety assessment. At the next stage it is assumed to develop the codes of new generation that does not have the foreign analogues (in particular, codes for closed nuclear cycle objects safety assessment), to improve the certain models, to create the positive public image of the developed software products by taking part in the international benchmarks and conferences, to expand the codes field of application by validation of the codes on the unique novel experiments.
1. Introduction
In 2010, Russia commissioned works on creating national system of new generation codes (NGC) to substantiate design solutions and safety of nuclear power plants with liquid metal cooled fast reactors (BREST-OD-300, BN-1200) and closed nuclear fuel cycle. “New generation codes” refer to commercialized software with the following characteristics:

- based on the state-of-the-art theoretical knowledge and experimental data on physical processes and phenomena;
- employs efficient numerical algorithms;
- developed in accordance with up-to-date requirements of programming language standards and adapted to modern computing technology;
- has a user-friendly interface;
- supplied with a full package of documentation (users’ guide, reference manual, programmers guide);
- employs automated work with CAD models.

In addition, to ensure the possibility of using the developed codes in materials justifying the safety of nuclear facilities, another mandatory requirement has been provided for them - certification at Rostechnadzor. Brief information about the “New Generation Codes” project and the state of codes development as of the end of 2016 and 2017 is given in papers [1, 2]. This paper provides information on the development status of NGC in mid-2018. Note that since the average time of single development cycle of a code is about 5 years (a single development cycle refers to the process from the approval of terms of reference to the submission of a code for certification), since the beginning of work on the project, most of the new generation codes have passed in 2017–2018 to the life cycle stage of "industrial use" and began to be used to solve relevant applied problems. This paper focuses on the models and development status of NGC, but not on their application.

2. The list and development status of new generation codes, including automated systems, in mid-2018
Table 1 summarizes the list of new generation codes and their development status in the middle of 2018. As compared to the data in [1, 2], the list of neutron-physical codes was expanded. In particular, a decision was made on the feasibility of developing a new generation diffusional neutron-physical code. The peculiarity of this code lies in the actual rejection of medium homogenization, which makes it possible to solve the main problem of a lower accuracy of the solution compared to codes that take into account the angular dependence of the neutron flux (Sn method) or solve the integral equation (Monte Carlo method).

In addition, in a short time, a code was developed for calculating the burnout and a code to substantiate the radiation safety of a fast reactor and fuel cycle facilities. More information about these codes will be given below.

The scope of application of a number of codes (for example, the BERKUT-U fuel rod code and EUCLID/V2 integral code) has been significantly expanded to include new modules and models. Certification of 6 new generation codes is completed, 8 more codes are at the final stage of certification.

Table 1. Development status of new generation codes in mid-2018.

| Code name   | Description                             | Development status                                  |
|-------------|-----------------------------------------|----------------------------------------------------|
| CRISS 5.3   | Code for probabilistic safety assessment | Validated, certified                                |
| BERKUT      | Fuel rod code, engineering version       | Validated, recommended for certification by section No. 4 of the Expert Council on Software Certification at Rostechnadzor |
| Code name        | Description                                                                 | Development status                                  |
|------------------|-----------------------------------------------------------------------------|-----------------------------------------------------|
| BERKUT-U         | Fuel rod code, improved version                                              | Partially validated                                 |
| **Neutronics Codes** |                                                                                  |                                                     |
| MCU-FR           | Code based on the Monte Carlo method                                          | Validated, submitted for certification              |
|                  | Code for calculating reactor shield based on the S_n method and the finite  | Validated, submitted for certification              |
|                  | element method                                                              |                                                     |
| ODETTA           | Code based on the S_n method and the finite difference method                | Partially validated                                 |
| CORNER           | Code based on the S_n method and the finite difference method                |                                                     |
|                  | Code based on diffusion approximation                                        | Under development                                   |
| DOLCE VITA       | Nuclide kinetics, calculation of activity and decay heat                     | Partially validated                                 |
| BPSD             | Nuclear data processing program and Code for justification of radiation     | Under development                                   |
| NDP-ACE          | Code for safety of a fast reactor and nuclear fuel cycle facilities          | Under development                                   |
| **Thermohydraulic Codes** |                                                                                  |                                                     |
| HYDRA-IBRAE/LM/V1| System thermohydraulic code                                                  | Validated, certified                                |
| LOGOS            | RANS CFD code                                                                | Validated, submitted for certification              |
| CONV-3D          | DNS CFD code                                                                 | Validated, submitted for certification              |
| CONV-3D/TwoPhase | Module of CONV-3D code for calculating heat and hydrodynamics for two-phase flows | Under development                                   |
|                  | Code for modeling the thermohydrodynamics processes occurring in lead-cooled fast neutron reactor, taking into account the physico-chemical processes | Partially validated                                 |
| KUPOL-BR         | Code for modeling mass transfer processes in Reactor Containment Building    | Validated, submitted for certification              |
| **Codes for analyzing the transport of fission products in the environment** |                                                                                  |                                                     |
| ROM              | Code for assessment of radiation situation out of NPP site                  | Validated, certified                                |
| ROUZ             | Code for calculation of on-site atmospheric transport                        | Validated, submitted for certification              |
| Sibylla          | Calculation of exposure through water ways                                   | Validated, certified                                |
| GeRa/V1          | Code for assessment of safety of RW disposal                                 | Validated, certified                                |
| **Integral Codes** |                                                                                  |                                                     |
| SOCRAT-BN/V1     | Comprehensive analysis of normal operating conditions, normal operation failure, design basis accidents and beyond design basis accidents, | Validated, certified                                |
| SOCRAT-BN/V2     |                                                                                  | Validated, submitted for certification              |
The codes for simulation of processes of closed nuclear fuel cycle

| Code name          | Description                                                                 | Development status          |
|--------------------|----------------------------------------------------------------------------|------------------------------|
| EUCLID/V1          | Comprehensive analysis of normal operating conditions, normal operation failure, design basis accidents and beyond design basis accidents, including severe accidents, for NPPs with sodium-cooled fast reactors with oxide fuel | Validated, submitted for certification |
| EUCLID/V2          | Including severe accidents, for NPPs with sodium-cooled, lead-cooled and lead-bismuth-cooled reactors and fuel rods with oxide or nitride fuel | Partially validated |
| VIZART             | The codes for simulation of processes of closed nuclear fuel cycle          | Partially validated          |
| KOD TP             | Code to simulate the operation of process workflows Automated Systems     | Partially validated          |
| SO                 | Management system for calculated safety justifications for nuclear fuel cycle facilities (“System shell”) | Put into trial operation |
| SURRK-ALM, SURRK-Portal | Management system for processes of calculation codes life cycle | Put into industrial operation |

Section 2 provides information on codes that were not considered in publications [1, 2] or the application scope of which was significantly extended.

3. Brief information about the new generation codes developed in 2017–2018 or the application scope of which has been extended

3.1. Code BPSD for nuclide kinetics, activity and decay heat calculations

Code for nuclide kinetics, activity and decay heat calculations BPSD is designed to calculate the change in the nuclide composition of materials. The code is used to simulate generation/transmutation of actinides and fission fragments to determine their errors by given initial concentration errors [3], half-lives, neutron flux values, cross-sections, and to calculate nuclide transmutation with an error estimate in non-fuel material compositions present in the core - structural materials, coolant (sodium, lead), absorber (boron carbide, dysprosium hafnate).

The code is validated based on the results of post-irradiation studies of oxide fuel samples irradiated in the BN-350 reactor and nitride fuel samples irradiated in the BN-600 reactor. Of particular note is the validation of the BPSD code based on the unique radiochemical studies of a sample of mixed nitride uranium-plutonium (MNUP) fuel from a central fuel element irradiated as part of a combined experimental fuel assembly (KETVS-1) in the core of the BN-600 reactor during 65 to 68 cycles between refueling. The isotopic composition of the fuel and the mass of nuclides are in good agreement with the experimental data.

3.2. Diffusion neutronics code DOLCE VITA

DOLCE VITA code was created to solve the transfer equation in the multi-group diffusion approximation, taking into account the heterogeneity of the medium. The following may be solved using the code:

- homogeneous neutron transport equation (direct and conjugate);
• nonhomogeneous neutron transport equation with a different type of fixed source, including
  subcritical breeding;
• non-steady-state neutron transport equation, both the exact solution of the direct equation, and
  its approximate solutions, as well as the inverse kinetics equation;
• solution of the neutron transport equation in the cavities using the solution of the integral
  neutron transport equation.

The code uses the neutronics constants prepared by CONSYST/BNAB-RF code developed by JSC
"SSC RF-IPPE"

To the main advantages of the calculation code in comparison with the codes used in practice for
calculation justification of reactor installations, one can attribute:
• non-homogenization of cells;
• the number of channels (types of heterogeneous FA description) is 14: from homogeneous
  cells with one, seven or thirteen points per channel to heterogeneous cells using 7 to 271 fuel
  rods, pellets or pipes and separating boundary cells with can and wall layer of coolant;
• the core can be modeled either in full volume or partially using sectoral symmetry, where it
  can be represented by a sector of 180°, 120° or 60°.

As an example, table 2 shows the calculation results of the reactor with cavities (simulation of
passive feedback system (PFBS) of the BREST-OD-300 reactor). It can be seen that the code provides
high accuracy comparable to the accuracy of precision calculation codes, while requiring significantly
less time resources.

Table 2. Efficiency of emptying the PFBD to a height of 70 cm from the core top.

| Parameter | MCU-FR | DOLCE VITA |
|-----------|--------|------------|
| Δk/k, %   | 0.22   | 0.22       |

3.3. Code for justification of radiation safety of a fast neutron reactor and nuclear fuel cycle facilities

Code for nuclide kinetics, activity and decay heat calculations BPSD is designed to calculate the
change in the nuclide composition of materials. The code is used to simulate generation/transmutation
of actinides and fission fragments to determine their errors by given initial concentration errors [3],
half-lives, neutron flux values, cross-sections, and to calculate nuclide transmutation with an error
estimate in non-fuel material compositions present in the core - structural materials, coolant (sodium,
lead), absorber (boron carbide, dysprosium hafnate).

Code for justification of radiation safety of a fast reactor and nuclear fuel cycle facilities is
designed to determine the power of radiation sources and develop measures to reduce the flux density
of these radiations to justify compliance of the degree of personnel and public protection with the
standards and rules for designing and operating nuclear power plants with closed nuclear fuel cycle,
including calculations of:
• radiation safety of the reactor and technological circuits;
• radiation characteristics of fresh fuel, spent nuclear fuel and radioactive waste to determine the
  requirements for technologies of nuclear materials management during their processing,
  refabrication, storage and final isolation;
• radiation characteristics of FA and SFA (including structural materials) to determine the
  requirements for transport containers for fresh and spent fuel assemblies;
• determining the possibility of reuse of structures, etc.
• The radiation safety justification code is an integral code based on the developments of the
  “New Generation Codes” project and includes the following basic modules:
• reactor module based on neutronics codes for calculating the core and material irradiation
  conditions by the Monte Carlo method (MCU-FR), based on diffusion approximation
  (DOLCE VITA) and kinetic code based on Sn approximation (CORNER);
• nuclide kinetics module (BPSD);
• radiation source calculation module;
• neutronics module for calculation of radiation protection based on Sn approximation (CORNER, ODETTA) and the Monte Carlo method (MCU-FR);
• constant support system (CONSYST/BNAB-RF);
• module for calculating the dose rate from neutron and photon radiation (integrals of radiation sources).

The code determines the intensity of radioactive sources and radiation characteristics of nuclear fuel, coolant and various materials after irradiation, as well as radiation doses.

Currently, the calculation code is under development.

3.4. Improved fuel rod code BERKUT-U

The improved fuel rod code BERKUT-U is designed to calculate the thermomechanical behavior and substantiate the performance of a single container-type fuel rod with pellets of oxide (dioxide and MOX) and nitride (mononitride and MNUP) fuel, with a gas and liquid metal filled fuel-cladding gap during normal operating conditions, normal operation failure of prospective cores of fast reactors with liquid metal coolant. By mid-2018, BERKUT-U code was validated based on the results of post-irradiation studies of the BORA-BORA and KETVS-1, 2, 3, 6 fuel elements irradiated in the BN-600 reactor. The obtained results showed that the code makes it possible to accurately describe an array of experimental data on release of fission products and mechanical state of fuel rods.

At the end of 2017, models that allow simulating the behavior of fuel elements with a liquid metal filled fuel-cladding gap were included in the BERKUT-U code. To do this, the code thermodynamic model for describing the interaction of nitride fuel and the steel cladding of a fuel rod with liquid-metallic (sodium, lead) filled gap was extended. An analysis of the thermodynamic data available in literature on interaction of liquid metal filled gap with irradiated MNUP fuel made it possible to formulate the problem of determining the composition of substances in a gap filled by coolant as a calculation of thermochemical equilibrium in a system with elemental isotopic composition specified by fission products (FP) emitted from a fuel pellet with addition of Pb or Na element, and a phase composition, in which a phase of liquid lead or sodium is added to the list of condensed phases already present in the model. In addition, about 30 chemical compounds of lead and sodium are added to the model and it is considered that the initial quantitative composition of the system is formed from:

• condensed components of the solid solution "(U, Pu) N - FP", which form a layer of fuel adjacent to pellet border;
• gas phase consisting of gaseous chemical compounds releasing from pellet with the addition of lead or sodium vapors and their gaseous compounds;
• phase of liquid lead (sodium).

To describe the interaction of liquid metal filling the gap with a steel clad, the process of oxidation of steel with impurity oxygen is neglected in the model, but the processes of dissolution of steel components in liquid metal are taken into account. To describe the transport of impurities dissolved in the liquid metal filled gap, the model uses rapid dissolution and fast radial diffusion approximations, and the problem of fission products transport in liquid metal filled gap is reduced to calculating the nonlinear diffusion equation for multicomponent concentration vector.

Calculations of test problem were performed, in which the behavior of CEFA-1 assembly was modeled assuming that the fuel-clad gap is filled with liquid metal (sodium or lead). It is shown that the calculation results obtained for release of gaseous fission products from fuel, dissolution of steel clad in liquid metal, and distribution of dissolved steel components in liquid metal filled gap are in good agreement with theoretical estimates based on the solubility of steel components in liquid metal.

3.5. Two-phase CFD module CONV-3D/TwoPhas

The two-phase 3D CFD module is designed for direct numerical simulation of thermo- and hydrodynamics in a two-phase medium, taking into account interphase heat exchange and mass
transfer using a stiffened state equation of condensed gas without flow regime map for personal computer. The module is based on the technique for modeling heat and mass transfer in a two-phase medium using HLLC (Harten-Lax-van Leer-Contact) solver and two-step predictor-corrector MUSCL algorithm (Monotonic Upwind Scheme for Conservation Laws is a monotonous scheme with directed differences for solving conservation laws) [4]. The developed code for direct numerical simulation of two-phase gas-dynamic flows is adapted for massive parallel supercomputers and is scalable in a wide range of computational grids.

In addition, a model consisting of 11 differential equations has been developed for direct simulation of compressed two-phase flows, taking into account capillary effects. The surface tension forces in non-equilibrium model with respect to pressure are taken into consideration using the Hamilton principle of least action. As a result, in the equations for momentum and total energy of the two-phase mixture, the additional tensor components containing the second spatial derivatives of mass fraction of liquid phase are appeared. In the equilibrium evaporation model, interphase heat transfer and phase transitions bring the two-phase system to a state with the same values of pressures, temperatures, and Gibbs energies. This allows determining the values of the thermodynamic parameters after mass transfer using the procedures of instantaneous relaxation (equalization): pressures, temperatures and Gibbs energy. Each of the relaxation procedures is iterative.

By the middle of 2018, the first version of two-phase module was developed on the basis of a multidimensional technique using the stiffened state equation to solve liquid-gas interaction problems with separation of the interface. The method has been tested for solving the following problems: disintegration of rupture and cavitation for water, dodecane, sodium. Adaptation of the two-phase module for modeling flows with sodium coolant was carried out taking into account mass exchange for the stiffened or Noble-Abel state equation. For module validation, an experiment was simulated to study the process of heat and mass transfer in vertical channel during sodium boiling [5]. A satisfactory agreement of numerical predictions with experiment was demonstrated.

3.6. EUCLID/V2 integral code for the safety assessment of NPPs with fast reactors

The EUCLID/V2 code is designed to analyze and justify the safety of NPPs with fast reactors with liquid metal coolants in normal operating conditions, normal operation failure, design basis accidents and beyond design basis accidents, including severe accidents. It is based on and includes all models of the first version, EUCLID/V1, detailed information about which can be found in [6, 7].

The following modules function together as part of the EUCLID/V2 code, providing multi-physical and consistent simulation of various processes and phenomena:

- thermohydraulic (HYDRA-IBRAE/LM), which includes the module of transport and behavior of fission products, corrosion and activation in the primary circuit and the gas system of the reactor (AEROSOL-LM), the module of transport of solid-phase impurities in the primary circuit of the reactor with a heavy liquid metal coolant (OXID), the module of tritium migration in the coolant of the primary, secondary and tertiary (if any) circuits (TRITIUM);
- sub-channel module (CELSIST) for 2D simulation of coolant flow in fuel assemblies;
- diffusion and Sn neutronics modules (DN3D and CORNER, respectively), modules for calculation of burnout (BPSD) and decay heat (OSTB);
- fuel rod module (BERKUT and BERKUT-U) with models for calculating sources of fission products in the reactor core in the event of cladding leakage, designed for numerical simulation of the behavior and calculation justification of operability of fuel elements with nitride and oxide fuel;
- module for calculating the destruction of fuel elements and the core of fast reactor (SAFR);
- module for mass transfer and distribution of fission products in reactor containment building (KUPOL-BR or HYDRA-IBRAE/LM);
- module for calculating the radiation conditions at the mesoscale (outside the site of nuclear facility) ROM.
• SMART/LM integration shell provides a consistent calculation of the modules. The structure of the EVCLID/V2 integral code is shown in figure 1.

![Figure 1. The structure of the EVCLID/V2 integral code.](image)

In 2018, the CELSIST two-dimensional sub-channel module developed by NIKIET JSC was included into the EUCLID/V2 code. The CELSIST module is intended for the thermo-hydraulic calculation of the FA of reactors with liquid metal coolant. Fuel rods in the fuel assembly may be spaced using spacer grids or wire wrapping on the fuel rods. A fuel assembly is considered as a system of thermohydraulically interconnected cells-channels formed by fuel elements, shell, CPS channels and supporting rods. Each cell can contact the external environment with the specified parameters (temperature, heat transfer coefficient, wall conductivity). The CELSIST module solves a system of non-stationary finite-difference equations of conservation of mass, momentum and energy written for the average values in the cells into which the FA flow cross section is divided. The equations of momentum transfer in transverse direction are also solved. Together with the equations of convective heat transfer in cells, the heat conduction equations in fuel elements are solved. The system of finite-difference equations is solved by the SIMPLER method, widely used in three-dimensional CFD codes.

A standalone version of the CELSIST module was validated based on the following experiments.

- 37-rod assembly of the MONJU reactor with sodium coolant (Japan): heat transfer during flow around the blockage and formation of recirculation vortex in the area downstream of the blockade;
- 19-rod assembly in the THEADES lead-bismuth circuit of KALLA installation (Germany): heat transfer with uniform heating of the rods;
- 19-rod FFM Bundle 2A assembly with sodium coolant (USA): heat transfer under different heating conditions;
- 37-rod assembly of 6B bench of SRC RF-IPPE, JSC, with Na-K coolant (Russia): heat transfer under different conditions of uniform and non-uniform heating of the rods over the cross section.

Figure 2 shows a comparison of the experimental (measured at the exit of the assembly) and calculated temperatures for the zone heating experiment (7 central simulated fuel elements of 37 were heated) conducted on the 6B bench at SRC RF-IPPE, JSC. For this centrally symmetric problem, the groups of coolant cells were numbered depending on the distance to the central axis of the assembly.
In figure 2 these cell group numbers are given on the X axis. It can be seen that the calculated values lie within the experimental uncertainty.

![Figure 2. Distribution of coolant heating in cells depending on the cell radius for mode No. 3 (“zone heating”) in an experiment conducted on 6B bench at SRC RF-IPPE, JSC.](image)

3.7. **Code to simulate the balance of materials and nuclide flows in the CNFC - VIZART**

The VIZART code is intended for calculating the material flows of process flows and separate production sites in the stationary and dynamic modes, taking into account the evolution of isotopic composition. It is used for the calculation support of the technological process development of CNFC and the justification of technical solutions, including the number of process support personnel. In 2017–2018, the library of computational models of individual process flows of the VIZART code has been significantly expanded; the possibility of performing related calculations using the VIZART code and neutron physical codes to justify the nuclear and radiation safety of process flows and devices has been implemented.

3.8. **Code for simulating the operation of technological schemes KOD TP**

The KOD TP is designed to simulate the operation of technological schemes (in real time) in order to study the operability, controllability and optimization of technological lines, including monitoring and control systems. In 2017, the flowsheet of fuel fabrication and re-fabrication module (FRM) was assembled in the KOD TP and functionality was implemented to simulate emergency modes and equipment failures. In addition, electronic procedures for management of the carbothermic synthesis section of FRM were developed (automated sequences of operator actions during the execution of process operations based on available manuals/operating instructions), which allowed identifying the inconsistencies in operating instructions and production line regulations prior to installation and commissioning.

3.9. **Commercial code STAR-CCM+ with a block for modelling the transport of oxygen, gas impurities, corrosion and activation products along the coolant circulation circuits**

The code is intended for computational modelling of thermohydrodynamic processes occurring in lead-cooled fast reactor, taking into account the following physico-chemical processes:

- growth of multilayer oxide film on the surfaces of the fuel rod claddings and other surfaces of the primary circuit which the coolant flows around;
- release and behavior of corrosion products in lead coolant;
- transport and change in activity of dissolved oxygen;
interaction of polydisperse particles with each other and with the surface;
accumulation of solid-phase deposits on the surface in a lead coolant;
destruction of the oxide film due to erosion in the coolant flow.

The calculation takes into account the degradation of heat transfer due to the formation of an oxide film.

The code was validated based on the results of experiments on transport and absorption of oxygen in the flow of a heavy liquid metal coolant in the models of fuel assemblies made by SRC RF-IPPE, JSC (table 3).

Table 3. Comparison of the calculation results with experimental data on the thickness of the oxide film on the surface of the fuel assembly tube.

| Element type         | Experiment, microns | Calculation, microns |
|----------------------|---------------------|----------------------|
| Fuel elements simulators | 10                  | 10                   |
| Spacer grids         | 10                  | 10                   |

4. Experiments to Validate New Generation Codes

We will review two scenarios of the two-component NPS development. The first one implies development of Russian nuclear power industry by 2100 up to 85 GW (without units export), the second one - up to 170GW (with export). At figure 2 scenarios are marked with a) and b) letters, correspondingly.

Simultaneously with the development of new generation codes, experimental studies are being actively carried out to determine the physical processes in order to create their physical and mathematical models and validate the NGC.

A program has been launched for experimental studies of thermohydraulic and physico-chemical processes proceeding in a lead coolant which are performed at NIKIET JSC, SRC RF-IPPE, JSC, Institute of Thermal Physics of Siberian Branch of the Russian Academy of Sciences (IT SO RAN) and other entities.

In particular, in 2017 IBRAE RAN and IT SO RAN jointly performed well-instrumented experiments in order to validate the codes, including CFD, and substantiate design solutions for components of reactor facilities with heavy liquid-metal coolant. Comprehensive experimental studies were performed of the flow of a model heavy liquid metal coolant in a 7-rod model of a fuel assembly as close as possible to the fuel assembly geometry of the BREST-OD-300 reactor, including spacer grids. Data on the non-uniformity of the axial and axial temperature distribution over the wall of the fuel rods simulators and data on the influence of the spacer grid on the temperature field distribution on the surface of the fuel rods simulator have been obtained. A series of experimental studies on the interphase interaction of gas - model heavy liquid metal coolant has been carried out, in which the pulsed interaction of gas with coolant is most pronounced, leading to the occurrence of dynamic processes associated with fluctuations in pressure and coolant level. A comprehensive experimental study of the process of outflow of an inert gas into a heavy liquid metal coolant with the determination of the parameters of resulting gas bubbles has been carried out. A study was performed of the patterns of heat transfer between an inert gas and a heavy liquid metal coolant under different flow conditions (bubble, slug flow) of a gas into a lead melt or Rose's alloy located in a vertical adiabatic channel. One of the main results of the work is the identification of regularities in the flow of a heavy liquid metal coolant, in which the three-dimensional structure of the flow is most clearly expressed (figure 3). A database of experimental measurements was obtained, including detailed values of local averaged and oscillatory temperature distributions, data on local and integral heat transfer. The obtained results are
already used as benchmarks for validating CFD codes used in hydrodynamic calculations of reactor plants in non-isothermal modes of coolant flow.

**Figure 3.** An example of 3D temperature distribution on the surface of a T-junction when mixing two different-temperature flows of a heavy liquid metal coolant: the ratio of hot and cold flow of coolant $Q_h/Q_c \sim 0.63$ (left); $Q_h/Q_c \sim 0.07$ (right).

For 2018-2021, IRM JSC plans to obtain reactor experimental data and dependencies of the kinetics of the release of gaseous fission products from MNUP fuel required to adjust the parameters of models and validate the BERKUT-U fuel rod code.

The irradiation of experimental fuel assemblies with fuel rods with MNUP fuel in the BN-600 reactor continues. The data obtained is used, inter alia, to validate the new generation integral codes and the fuel rod code.

5. **Prospects for Further Development of New Generation Codes**

The further development of new generation codes is aimed at expanding their field of applicability through the completion of models based on the results of new experimental studies and validation of calculation codes. In addition, it is planned to develop new generation codes that do not have world analogues, in particular, precision codes for the safety assessment of closed nuclear fuel cycle facilities.

Among other trends in the development of new generation codes, it is worth noting the final drift from foreign components and platforms, thus providing a solution to the problem of import substitution, as well as support of advanced techniques and approaches to the design of complex products using new generation codes. Such techniques and approaches include multi-criteria optimization, robust design, as well as “forward” and “generative” modeling.

The traditional design process involves creating a product prototype by designer, then the prototype is calculated by engineer using the codes, which leads to several iterations of the design. Advanced techniques and approaches will allow moving (where applicable) to the process when, based on the developed requirements for product (in terms of dimensions, mass, consumer properties, etc.), and as a result of calculating by codes, a design optimized by the specified criteria is created. Then designer modifies the resulting model by CAD (computer-aided design system). At the same time, to reduce the design time, the designer is able to calculate quickly the model obtained in the course of work using appropriate calculation code built into CAD.

In connection with the beginning of active use of calculation codes in practice, in 2017, the first School-Workshop on new generation codes training was conducted, which confirmed the interest of
experts and specialists of various organizations in the use of created software products. In the future, such School-Workshops are planned to be held annually.

To use the new generation codes effectively and to ensure a gradual transition to the domestic software in the practice of calculated justification, it is necessary to propagate the created codes in higher education institutions. As a result, the university graduates, when coming to work at R&D organizations, will easily perform the calculated justifications using NGC. The corresponding work is also put on the priority list.

6. Conclusion

Thus, the system of new generation codes has been developed. The system, after completion of certification of the whole system, will allow making well-grounded design decisions and carrying out the calculations to justify NPP safety (including the power complex) with the BREST-OD-300 or BN-1200 reactor installations in closed nuclear fuel cycle at the advanced level of knowledge about the processes and phenomena that are taking place, the physical and mathematical models and the available array of experimental data. Due to the use of precision models, the NGC allow making predictive calculations in order to achieve optimal technical and economic indicators of the nuclear technologies being developed. Already now, the NGC are used in R&D organizations, being delivered through the conclusion of licensing agreements, to solve the urgent problems of mathematical modeling, including those beyond the “Proryv” project.

A program of further development of new generation codes has been outlined, which from 2018 will take on a fundamentally new nature: until 2018, the main objective was to provide the “Proryv” Project with the codes necessary to justify design solutions and safety, and create a domestic substitution for the currently used foreign software. From 2018, the tasks of creating precision computational models that have no world analogues, as well as implementation of the developed tools into practice of computational justification and the programs of higher education institutions, including annual user school-workshops. At the same time, there is still a priority to preserve the formed highly qualified team, which includes the experts from research institutes, enterprises of the "Rosatom" State Corporation, and universities, uniting the leading experts of various profiles.

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