ISSN 0040-6015, Thermal Engineering, 2016, Vol. 63, No. 2, pp. 130–139. © Pleiades Publishing, Inc., 2016.
Original Russian Text © V.M. Alipchenkov, A.M. Anfimov, D.A. Afremov, V.S. Gorbunov, Yu.A. Zeigarnik, A.V. Kudryavtsev, S.L. Osipov, N.A. Mosunova, V.F. Strizhov, E.V. Usov, 2016, published in Teploenergetika.

Fundamentals, Current State of the Development of, and Prospects for Further Improvement of the New-Generation Thermal-Hydraulic Computational HYDRA-IBRAE/LM Code for Simulation of Fast Reactor Systems

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Abstract—The conceptual fundamentals of the development of the new-generation system thermal-hydraulic computational HYDRA-IBRAE/LM code are presented. The code is intended to simulate the thermal-hydraulic processes that take place in the loops and the heat-exchange equipment of liquid–metal cooled fast reactor systems under normal operation and anticipated operational occurrences and during accidents. The paper provides a brief overview of Russian and foreign system thermal-hydraulic codes for modeling liquid–metal coolants and gives grounds for the necessity of development of a new-generation HYDRA-IBRAE/LM code. Considering the specific engineering features of the nuclear power plants (NPPs) equipped with the BN-1200 and the BREST-OD-300 reactors, the processes and the phenomena are singled out that require a detailed analysis and development of the models to be correctly described by the system thermal-hydraulic code in question. Information on the functionality of the computational code is provided, viz., the thermal-hydraulic two-phase model, the properties of the sodium and the lead coolants, the closing equations for simulation of the heat–mass exchange processes, the models to describe the processes that take place during the steam-generator tube rupture, etc. The article gives a brief overview of the usability of the computational code, including a description of the support documentation and the supply package, as well as possibilities of taking advantages of the modern computer technologies, such as parallel computations. The paper shows the current state of verification and validation of the computational code; it also presents information on the principles of constructing of and populating the verification matrices for the BREST-OD-300 and the BN-1200 reactor systems. The prospects are outlined for further development of the HYDRA-IBRAE/LM code, introduction of new models into it, and enhancement of its usability. It is shown that the program of development and practical application of the code will allow carrying out in the nearest future the computations to analyze the safety of potential NPP projects at a qualitatively higher level.

Keywords: system thermal-hydraulic code, liquid–metal coolant, fast reactor system, lead, sodium, heat–mass exchange, closing equations, verification, validation

DOI: 10.1134/S0040601516020014

INTRODUCTION

In Russian nuclear power engineering, nuclear power plants widely use VVER water-moderated power reactors that, during the entire period of their operation, have proven to be highly safe. Despite the fact that the world reserves of natural uranium are sufficient to provide it for the existing nuclear power plants for the next one hundred years, in the nuclear power industry of Russia and other countries, designs of fast-neutron breeders are being developed that will allow for natural uranium to be used on a considerably greater scale and for the problem of accumulation of spent fuel to be solved by introducing closed nuclear fuel cycle technology and providing the nuclear power supply system with a fuel produced from ^{238}U [1, 2]. This is the development of new technologies based on practical exploitation of the fast reactors' advantages that will significantly facilitate the solution of the power engineering problems of the third millennium.

Russia is the only country in the world where a sodium-cooled BN-600 fast reactor has been successfully operated at Beloyarsk NPP for more than 30 years, and the first launch of a power-generating unit with a BN-800 reactor has reached it first criticality at the end of 2013. The problem of closing the fuel cycle with extended—and, at the same time, meeting the nonproliferation criterion—breeding of fissile isotopes in complete compliance with accepted safety requirements that has become more stringent in recent years in connection with the Fukushima-1 nuclear accident has to be solved by a power-generating unit with a BREST-OD-300 inherent-safety fast reactor system with lead coolant; the design of this reactor, alongside with the design of the BN-1200 reactor, is being developed within the framework of the *Proryv* project of the *New-Generation Nuclear Energy Technologies for 2010–2015 and until 2020* Federal Target Program.

The plans to develop the nuclear power industry in Russia [2] that embrace development and putting into operation of fundamentally novel reactor systems that do not have any analogues either in Russia or abroad necessitate in-depth research and development, including those carried out using computational codes [3]. Against the background of the political situation in the world that has arisen since the beginning of 2014, for these plans to be successfully implemented, the dependence of the Russian NPPs on foreign software should be eliminated, since the sanctions imposed by a number of European countries and the United States already partly apply to such software.

For this purpose, within the framework of the *New-Generation Nuclear Energy Technologies for 2010–2015 and until 2020* Federal Target Program, a system of the domestic new-generation software is being developed [4], a component of which is the system HYDRA-IBRAE/LM code designed to calculate the thermal-hydraulic characteristics in the loops and the heat-exchange equipment of the NPPs with liquid coolants, such as sodium, lead, and lead–bismuth. The codes of this class are the basic tool for analysis of transients and emergencies in the operation of NPPs, including accidents, since they ensure a quick solution of dynamic problems of heat–mass exchange in two-phase flows. They are used:

(1) to determine the safe operation margins and for NPP safety justification;

(2) to provide computational support for the experimental programs, including carrying out of R&D;

(3) to model individual processes that occur in nuclear power plant units;

(4) for check calculations; and

(5) personnel teaching.

The aim of this article is to present the fundamentals that underlie the system computational HYDRA-IBRAE/LM code and information on the current state of its development and the prospects for further improvement.

RUSSIAN AND FOREIGN SYSTEM THERMAL-HYDRAULIC CODES FOR SIMULATION OF LIQUID–METAL COOLANTS

The thermal-hydraulic codes referred to as system codes are based on the solution of an equation system that expresses the fundamental laws of conservation of mass, energy, and momentum and is closed by the thermodynamic state relations of the coolant and the relations that describe the phase interactions and the interactions of the phases with the channel walls called the closing equations. The phase interactions and the interactions with the channel walls depend on the interface area to a considerable extent, the characteristic dimensions of the regular structures, viz., bubbles, droplets, films, etc., and the interfaces between the phases and the wall. The latter are found in accordance with the so-called flow regime maps.

Since water-cooled reactors have become the most widespread nuclear power plant units in the world, the system thermal-hydraulic codes designed to compute two-phase water coolant flows, including those with an admixture of noncondensable gases, are currently the most developed codes, i.e., those that have a high computation speed and stability, are validated on a wide range of experimental data, include special models to consider the design features of modern NPPs, and have an advanced graphical user interfaces to set input data and visualize the computational results. The most widely used foreign system thermal-hydraulic computational codes are:

(1) RELAP5-3D (Idaho National Laboratory, United States) [5], which is the latest version of the RELAP5 code series intended for analysis of transients and accidents at NPPs equipped with both the most widespread water-cooled and fast reactors;

(2) CATHARE (EDF, AREVA-NP, IRSN, and CEA, France) [6], a system code to analyze and substantiate the safety of NPPs with pressurized-water reactors (PWRs), to develop the manuals on management of accidents, to license, to determine the safe operation margins, and to carry out R&D; and

(3) TRAC code family (Los Alamos National Laboratory and Idaho National Laboratory) [7–9] that are designed to simulate the entire range of accidents in the boiling-water reactors (BWRs) and the PWRs and associated facilities.

The above foreign system thermal-hydraulic codes are applied by numerous machine-building and development companies, power engineering companies, reactor equipment manufacturers, and governmental supervisory bodies. In recent years, adaptation of the above computational codes to simulation of liquid–metal coolants have been started and already partly carried out. It should be noted, however, that while RELAP5-3D, CATHARE, and TRAC codes have a long history of development and application to the water coolant, which ensures a high-quality and reliable software product, the situation about the liquid metals is different: the codes are neither verified nor validated on the required amount of experimental data; they have not been tested for a long time with regard to applied problems; the closing equations used in the codes do not describe all characteristic features of the fast reactors; when developing the models, the main attention is paid to the sodium coolant; and all the above foreign codes lack models for the lead coolant. In addition, differences between design features of the foreign and domestic NPPs and the fact that these codes have not been validated on the basis of the domestic experimental data make their application to Russian fast reactor projects difficult. Consequently, while acknowledging the unquestionable usefulness of using the foreign system codes to ground the design solutions for potential liquid–metal cooled reactors in Russia, we should point out some limitations as to their application that can become decisive factors in analysis of the Russian NPP projects, which once more stresses a necessity of developing a domestic specialized system thermal-hydraulic code.

Russia has a certain advantage over the European countries and the United States in the development of physical models of thermal-hydraulic processes that occur in liquid metals and in the amount of available experimental data that creates the necessary prerequisites for development of a Russian new-generation system thermal-hydraulic code designed to simulate the NPPs with liquid–metal cooled reactors based on the latest achievements of science and the experimental data bank gathered over several decades and supported by a service necessary for comfortable use.

To fulfill the tasks assigned under the *New-Generation Codes* project within the framework of the *Proryv* program, the system thermal-hydraulic HYDRA-IBRAE/LM code is being developed. The fundamentals that underlie this code are set forth below. The development of the code was based on the experience gained by development organizations when creating and applying the domestic programs TRIANA (by AO OKB Gidropress), DINROS (by AO GNTs RF–FEI), and BURAN and DIN800 (by AO OKBM Africantov) to simulate the thermal-hydraulic processes in liquid–metal coolants.

SPECIFIC FEATURES OF SIMULATION OF THE LIQUID–METAL COOLED REACTORS

The main application field of the HYDRA-IBRAE/LM computational code is simulation of steady-state conditions, transients, and accidents of the BREST-OD-300 and BN-1200 reactor systems, although the models and the algorithms used in the code enable this software tool to be applied to reactors of other types, viz., BN-350, BN-600, BN-800, and SVBR-100 reactors as well as to research reactors, e.g., MBIR and BOR-60. When developing the computational code, special attention was paid to its capability of describing the specific design features of the engineering solutions assigned when developing nuclear power plants.

With respect to simulation of reactors of the BREST-OD-300 type, the computational code is to ensure correct simulation of the following equipment and systems [10]: lead-cooled fast reactor core, upper an lower plenums, hot and cold legs, blanketing gas, pumps, valves, residual heat removal system and emergency core cooling system, steam generators, and passive core coolant flow rate feedback system. In the secondary loop, the steam generators, the feedpumps, the steam condenser, the pipelines, the fittings, and the turbine are to be simulated.

Similarly, for the sodium-cooled BN-1200 fast reactor, the following is to be simulated: the primary (I) and the secondary (II) (intermediate) sodium loops, the third (III) water–steam loop (within the boundaries of the steam generator), the intermediate sodium loop of the emergency core cooling system (ECCS) (EHRS), and the ECCS air loop. Loop I of a BN-1200 reactor comprises a core and a reflector, an upper plenum, a pressure header, emergency heat exchangers, intermediate heat exchangers, circulating pumps, valves, entrainment filters, a reactor vessel cooling system, and a loop I pressurizer. Loop II of a BN-1200 reactor comprises intermediate heat exchangers, steam generators, a loop II pressurizer, pipelines, a circuit II main circulating pump, and a safety injection. Loop III consists of steam generators, a steam pipeline, a deaerator, water heaters, a condenser, a turbine, and water reservoirs.

Analysis of the engineering processes and the specific design features of the BREST-OD-300 and the BN-1200 reactors allows identifying the processes and phenomena that require an additional detailed study and development of the models for their correct description by the system thermal-hydraulic code in question in addition to the models developed for the water-cooled VVER reactors, PWRs, BWRs, and RBMK high-power channel reactors as follows:

(1) the heat–mass exchange in the loops with the following combinations of the coolant and the working fluid: lead–water and sodium–water;

(2) two-phase flows with the boiling sodium coolant;

(3) flows with gas-phase inclusions, including those that occur under emergency conditions, for example, in the case of steam generator tube rupture and severe accidents accompanied by fuel rod failure;

(4) a complex hydrodynamics of the liquid–metal coolant flow in the core, the steam generator, the upper and lower plenums;

(5) processes in the blanketing gas filled with an inert gas, e.g., argon, considering the latter's dissolution and transport by the liquid–metal coolant;

(6) the coupled heat–mass exchange under ingress of oxygen and corrosion products of the structural material components into the coolant;

(7) heat–mass exchange processes in the normal and the emergency core cooling systems;

(8) pressure drop due to individual specific local constraints; and

(9) variations in the geometry characteristics of the flow region in the course of computation.

FUNCTIONALITY OF THE HYDRA-IBRAE/LM CODE

The computations by the HYDRA-IBRAE/LM code yielded the fields of the coolant and the working fluid's velocities, temperature, and pressure, as well as the void fractions in the cells into which the simulated domain—fast reactor NPP loops, heat-exchange equipment, etc.—is divided. The flow of a two-phase medium (steam and liquid) with an admixture of noncondensable gases is described in the HYDRA-IBRAE/LM code by the thermal-hydraulic two-fluid model. In the computational code, the behavior of the fluid film and that of fluid droplets suspended in the gas core are described separately in the annular-dispersed flow regime. For this purpose, the momentum conservation equations are written separately for the gas–droplet core, which includes the dispersed part of the fluid, and for the fluid film.

The following assumptions are also made:

(1) the gas phase consists of steam and/or noncondensable gases;

(2) the pressures of the fluid and the gaseous phases are equal;

(3) the noncondensable gases and the coolant's steam phase are in thermal equilibrium;

(4) the noncondensable gases satisfy the ideal gas equation; and

(5) the gas phase is described by the Dalton law.

The balance mass, energy, and momentum conservation laws are written for the computational cells into which the simulated domain is divided (the channel approximation). As compared with the standard notation used for water coolants, e.g., [11], the second member of the energy conservation equation for the fluid phase comprises the heat flux resulting from the longitudinal heat conduction; this heat flux is used only for liquid–metal coolants, since their thermal conductivity is by approximately two orders of magnitude higher than that of a water coolant.

The conservation equations are closed by the relations that define:

(1) the phase interactions, viz., the interface parameters, the interphase heat–mass exchange, and the interphase friction;

(2) the interactions between the phases and the channel walls, viz., friction on the channel walls, local constraints pressure drops, and the heat exchange with the walls; and

(3) the state equations and the thermophysical properties of the phases of the coolant and the noncondensible gases.

The state equations for the sodium coolant are realized in the form of the following quantities:

(1) the phase enthalpies as functions of the pressure and the temperature;

(2) the phase densities as functions of the pressure and the temperature; and

(3) the thermal conductivities and viscosities of the phases as functions of the temperature.

To describe the thermophysical properties of the liquid and the gaseous sodium coolant, well-known formulae presented in [12, 13] are used in the HYDRA-IBRAE/LM code.

The thermophysical properties of lead are realized using the relations derived by analyzing the data available in the literature and the experimental data on the heat capacity and the thermal conductivity of lead obtained during the investigations under the *Proryv* project carried out in 2013 at the Institute of Thermophysics (Siberian Branch, Russian Academy of Sciences) and the National Nuclear Research University Moscow Engineering and Physics Institute. The state equations for the lead coolant are realized in a form similar to that for the sodium coolant.

Experimental investigations carried out in the Soviet Union [14, 15], Russia [16], and abroad [17, 18] have shown that the basic two-phase flow regimes of alkaline metals are the same as those of the water coolant. In the HYDRA-IBRAE/LM code, the phase interactions and the interactions with the walls are defined for the sodium coolant by the flow regime map that comprises the following flow regimes: the single-fluid and the steam flows, the bubble flow, the annular-dispersed flow, and the dispersed flow. In the sodium-cooled reactors, the operation and emergency processes are realized at low pressures, i.e., under the conditions that correspond to the most part of the available experimental data on the flow regimes of two-phase media obtained in the experiments with water and other fluids; therefore, use of conventional flow regime maps for the sodium coolant is even more justified than for the water coolant. For the same reason, the relations of the droplet entrainment rate by the flow core resulting from experiments with ordinary fluids are applied to two-phase liquid–metal coolant flows, since the available experimental data on the characteristics of the annular-dispersed liquid–metal flow patterns are not sufficient to drive these relations.

The closing equations used in the code for the sodium coolant are based on experimental data the most part of which was obtained at the Physical Power

Engineering Institute (GNTs RF-FEI) [19]. In particular, the closing equations to simulate the hydraulic pressure losses and the heat-exchange coefficients in the channels and the fuel assemblies (FAs) are presented in [20].

The problem of the initial superheating of sodium upon the coolant boiling up is extremely important. Superheating of zero to hundreds of degrees has been recorded experimentally [17, 21, 22]. High-temperature superheating is fraught with destructive consequences upon the coolant boiling up, since the entire energy stored in the metastable superheated fluid is transformed into the kinetic energy of the moving coolant. As the magnitude of superheating depends on numerous time-variable operating conditions and engineering factors, viz., the velocity of the coolant, the condition and the material of the heating wall, and the presence of dissolved gases, a correct choice of the initial superheating becomes a matter of expertise or may be considered to be a random quantity. For this reason, the HYDRA-IBRAE/LM code provides a possibility of setting the superheating value by the user.

When designing the reactor systems, it is advisable to use a superheating value that does not exceed several tens of degrees. Since the superheating depends on a great number of extraneous factors with a high degree of uncertainty, it is practicable to carry out multiple-choice computations when analyzing accidents accompanied by boiling of sodium.

The two-fluid model with equal-pressure phases that is widely applied to the water coolant is not able to adequately describe the dynamics of a growing steam bubble under high-temperature superheating, since the difference between the pressures in the steam and the fluid phases becomes significant. For such conditions, the HYDRA-IBRAE/LM code uses a model of a missile-like bubble that is similar to the model used, for example, in the SAS4 code [23] and presented in [24]. For a detailed description of the applied closing relations refer to the manual to the HYDRA-IBRAE/LM code models.

Blocking of the FA flow areas that may result in changes in the thermal-hydraulic characteristics of the flow, flow reversals in individual FAs, and formation of stagnation zones necessitates an individual 3D core model. For this purpose, in the HYDRA-IBRAE/LM code, a module based on a porous-body model [25] is used that currently allows computation of the processes in a single-phase coolant. To simulate the boiling processes, it is planned to improve the module to enable description of two-phase flows.

To simulate the transients in the secondary sodium loop under interloop water–sodium leakiness of the steam generators in the BN-type reactor systems, two models are used.

(1) A single "small" leakiness model. If a "small" leakiness is formed, a high-temperature flame occurs in the fracture area [26] that is caused by a reaction of sodium with water; the reaction has a high rate and results in formation of hydrogen and intensive corrosion–erosion destruction of the tube bundle material. The interaction model considers (currently) only the basic reaction of water with sodium that results in formation of the most part of hydrogen and the generated heat. Under the leakages in question, the rate of the chemical reaction can be thought to be infinitely high; therefore, the rate of interaction is determined by the inleak of the steam [27]. A significant role in this phenomenon is plaid by transport of gaseous hydrogen and other reaction products along the sodium circuit. To simulate the transport of the reaction products by the HYDRA-IBRAE/LM code, a two-fluid model is used with a common flow regime map for the interphase friction and the heat exchange between the reaction products and liquid sodium.

(2) A "large" leakiness model to compute the maximum accident. In the case of "large" leakiness, high rates—2 kg/s and above—of the water outflow into sodium are assumed; such rates lead to occurrence of an intensive local source of hydrogen in the secondary loop; therefore, to calculate the growth of the hydrogen bubble and the motion of liquid sodium in this case, a model of a projectile gas volume growth is used that is simpler as compared with the two-fluid approximation, which allows a reduction in the calculation time without loss of accuracy. When calculating "large" leakiness, one should determine the maximum short-time loads on the equipment and the pipelines of the secondary loop, which were experimentally observed and studied in [28, 29], and establish whether the margin of safety in terms of the pressure will be exceeded or not. The model of a projectile gas volume growth is based on the model of sodium's explosive boiling upon superheating which, for the sake of simplicity, assumes, in particular, that sodium does not evaporate at the phase interface.

Within the entire range of pressures and temperatures characteristic of fast reactors developed within the framework of the *Proryv* project, the lead coolant can be considered to be a single-phase fluid. This assumption underlies the HYDRA-IBRAE/LM code model for the lead coolant. The assumption that underlies the development of the model is that the designers of the BREST-OD-300 reactor system will ensure the maintenance of the oxygen potential in a narrow permissible range, 10^{-2}–10^{-5}. Under such conditions, the relations for calculation of frictional and heat-exchange pressure losses can be used that were experimentally obtained at GNTs RF–FEI, the Central Boiler-Turbine Institute (NPO TsKTI) [29–32], and the State Technical University in Nizhny Novgorod [10, 33–35] using heavy liquid–metal coolants in the same oxygen potential range. The relation used in the HYDRA-IBRAE/LM code to calculate the friction on the walls and the heat exchange with the wall are presented in [36].

The processes occur during steam generator tube rupture are simulated in the HYDRA-IBRAE/LM code by the thermal-hydraulic two-fluid model considering two media, lead and steam or water–steam mixture, with neglect of the mass exchange between lead and steam/water–steam mixture. For the lead coolant, the steam or the water–steam mixture is an equivalent inert gas. The choice of the relation to calculate a two-phase flow is based on the accepted flow regime map. The flow regimes are classified by the value of the void fraction (steam or the water–steam mixture). The following regimes are distinguished: a bubble flow, a transient flow, and an annular flow. Single-phase flow regimes are the extreme cases of the bubble and the annular flows. In the transient flow regime, the relations to calculate the friction, the heat exchange, and the parameters of the interface surface are determined by interpolation of the coefficients in the bubble and the annular flow regimes. In the HYDRA-IBRAE/LM code, a common flow regime map is used to calculate the interface area that falls at a volume unit and the coefficients of the interphase friction and heat exchange.

The HYDRA-IBRAE/LM code also enables simulation of the processes in the water loops of the reactor systems. The flow regimes and the heat exchange maps that allow description of the processes that occur in the water loops of the NPPs with liquid–metal cooled fast reactors are based on the maps used in the KORSAR code [37], since the latter enable description of all regimes of practical interest. The used closing equations are presented, for example, in [38].

The finite difference approximation of the governing equation system is based on the following fundamental principles:

(1) the spatial discretization of the equations is underlain by the control-volume method, the difference equations of the phases' mass and energy conservation being derived by the balance method in the conservative form;

(2) a staggered grid is used in the spatial coordinate when the scalar variables of the flow, viz., the pressure, the enthalpy, and the phases' volume fractions, are determined in the computational cell centers, while the vector variables, the phase velocities, are computed at the computational cell boundaries;

(3) approximation of the convective components of the phases' mass and energy transport is carried out according to the upwind scheme; and

(4) semi-implicit numerical time-approximation is selected.

To describe the heat transfer processes, a heat-conduction model in the two-dimensional formulation and a radiative heat-exchange model are implemented in the HYDRA-IBRAE/LM code. In addition, to correctly calculate the fast reactor systems, models of their equipment have been developed, viz., of valves and pumps, including a model of the main circulating pump for the BN reactor, a simplified model of the turbine, and the library of the local resistances. The HYDRA-IBRAE/LM code can be applied not only to substantiation of the safe operation of the prospective reactor systems, but also to planning and analysis of experimental results. For this purpose, the code provides for a possibility of using the closing equations for channels of various geometries, such as round tubes, annular channels, and triangular- and square-arrangement rod bundles.

USABILITY AND THE COMPUTATION PROCEDURE

The first version of the code, HYDRA-IBRAE/LM/V1, is a cross-platform development with regard to the Windows and the Linux OSs and is supplied to the users with a documentation package that includes:

(1) a users' guide that contains information on installation and program start, general information on in- and output files, a brief description of the used models, and a detailed description of the input file format with examples;

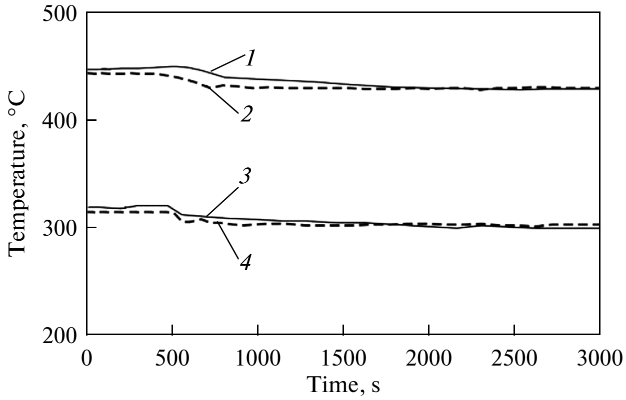
(2) a reference manual that describes the basic system of the thermal-hydraulic equations solved in the code, the flow regime and the heat-exchange regime maps for the sodium, the lead (lead–bismuth) and the water coolants, a description of the closing equations and the state equations, models used to describe the heat transfer in the solid structural components and the gas gaps, a model to compute the radiative heat exchange, and the approaches to simulating the pumps, the valves, an the other equipment components and their numerical implementation; and

(3) a report on the results of the verification and the validation calculations.

In addition to the executable files of the computational code and the above documentation, the supply package also contains a graphical user interface that enables visualization and storage of the computational results in a form convenient for the user as well as the test problems, which can be used as tutorials to develop own input files.

To enhance the efficiency of the computations, algorithms for parallel computations are introduced into the code that allow for the computations to be carried out on both several PC cores and cluster computing systems.

During the operation, the user presets the nodalization, i.e., partitioning into the computing elements, of the simulated domain, the initial and the boundary conditions, and the parameters that govern the computation in the special-format input file. Then, the executable file of the computational code is started that processes the preset information and performs the computation. The results are generated in the text- and the binary-format files, which can be displayed



Comparison of the predicted temperature values (*2*, *4*) of the lead–bismuth eutectic at the core inlet and the core outlet with experimental data (*1*, *3*). (*1*, *2*) at the core outlet and (*3*, *4*) at the core inlet.

with the help of the graphical user interface or browsed manually.

VERIFICATION AND VALIDATION OF THE HYDRA-IBRAE/LM COMPUTATIONAL CODE

The correctness of the chosen physical models and numerical method, their correct implementation in the form of operators in the chosen programming language (C++), and errors of computation of the basic parameters are determined by validation and verification computations.[1] The preliminary stage of validation and verification is the development of a verification and validation matrix. The matrix is usually represented in the form of a table in which the selected phenomena representative for the reactor system in question correspond to the lines, and the columns represent the operating conditions of the reactor to be simulated by the computational code and experimental setups on which the corresponding phenomena are studied, as well as numerical and analytical tests. At the intersection of the lines and the columns, the numbers of the experimental data sources are placed and it is also shown whether the phenomenon in question is significant for simulation under the given conditions and whether the performed experiment allows assessment of the errors of certain predicted parameters. A fragment of the verification and validation matrix of the HYDRA-IBRAE/LM code for the BREST-OD-300 reactor is presented in the table.

The HYDRA-IBRAE/LM verification and validation matrices for liquid–metal coolants are founded on a unique knowledge base gained in Russia in the course of operating fast reactors and on modern data both available in the literature and obtained within the framework of the *Proryv* project.

Recently, the computational code has been verified and validated by solving the problems that have an analytical solution, as well as on the basis of the available experimental data, and the errors of prediction of certain parameters have been assessed. In the future, given the lack of experimental data on particular phenomena/processes, the results of computations by the precision CFD-codes are supposed to be used as reference estimates of the corresponding parameters and characteristics.

As an example, the figure shows the results of simulation by the HYDRA-IBRAE/LM code of the operating conditions of the integral two-circuit TALL reactor system (Sweden) with excessive heat removal in the heat exchanger owing to the increased secondary loop coolant rate [39]. In loop I of the system, which consists of a core simulator, pressurizer, and a parallel-tube heat exchanger in which the heat is removed to loop II with glycerin as the working fluid, the lead–bismuth eutectic solution flows. The circulation of the coolant is effected by an electromagnetic pump. Starting from the 400th second, the glycerin rate increases by 2.4 times. For comparison, the figure presents the absolute temperature values of the lead–bismuth coolant at the core in- and outlet; the figures were computed by the HYDRA-IBRAE/LM code and obtained experimentally. The differences in the initial section can be explained by the different laws according to which the characteristics of the pumps in the primary and the secondary circuits varied in the experiments and the computations by the code since such information is not available in the description of the experiment. The mean relative errors of the computation of these conditions in terms of the temperature of the lead–bismuth eutectic solution at the core in- and outlet were less than 1% compared to the experiment.

PROSPECTS FOR THE FURTHER DEVELOPMENT OF THE COMPUTATIONAL CODE

At present, the first version of the computational code, HYDRA-IBRAE/LM/V1, described in this article is installed at the State Nuclear Power Engineering Corporation *Rosatom* (AO NIKIET, AO OKBM Africantov, and the Private Enterprise ITTsP *Proryv*).

The second version of the computational code, HYDRA-IBRAE/LM/V2, will incorporate the following additional modules:

(1) a three-fluid model to describe the annular-dispersed flow regime; the model is presented in the form of the laws of mass, momentum, and energy conservation for each of the three components, viz., film, gas,

[1] Validation is a process of determining the adequacy of a mathematical model to a real system (nature) that is simulated by comparing the predictions based on the model in question with the observation data obtained on a real system or with experimental data. Verification is analysis of the software implementation for its compliance with the description given in the documentation.

Fragment of the HYDRA-IBRAE/LM verification and validation matrix for the BREST-OD-300 reactor

Processes and phenomena	Experimental reactor facilities							
	TALL	LIFUS 5	THEADES	EUST	SPRUT	TsKTI test bench	VNIITF test bench	NIKIET test bench
Hydraulic resistance of								
heavy liquid metal in heated and nonheated channels with a simple shape (tubes and annular channels), including those with an oxide layer	+	–	–	–	–	–	–	–
heavy liquid metal in heated and nonheated bundles (triangular arrangement), including those with a spacer grid with the coil bundle flown-around	–	–	–	–	–	+	–	+
with the coil bundle flown-around	–	–	–	–	+	–	–	–
Outleakage of the water–stream mixture into the heavy liquid-metal coolant flow	–	+	–	–	–	–	+	–
Motion of vapor formations in the heavy liquid-metal coolant flow in the core, the seam generator, and other sections of the coolant circuit	–	+	–	–	–	–	+	–
Heat exchange:								
of heavy liquid metal in channels with a simple shape (tubes and annular channels) considering oxides	+	–	–	–	–	–	–	–
with the coils flown-around by the heavy liquid metal	–	–	–	–	+	–	–	–
of heavy liquid metal in smooth fuel element bundles, including those with spacer grids	–	–	+	–	–	+	–	–
to the water–stream mixture in different zones of the once-through steam generator	–	–	–	–	+	–	–	–
to the air in the emergency core cooling system	–	–	–	+	–	–	–	–

The plus/minus signs indicate that the published information is sufficient/insufficient to evaluate the correctness of simulation of processes/phenomena by the code.

and liquid droplets; the model will allow a reduction in the level of empiricism in the closing equations and a more detailed simulation of the heat-exchange processes associated with the dryout of the liquid film;

(2) a model to describe the dynamics of the changing sizes of dispersed particles, such as droplets and bubbles. For this purpose, the equation of the interface area density is supposed to be used, which will consider all factors that determine the particle sizes, viz., steam generation, disintegration of the particles under the impact of different mechanisms, coagulation, etc. Use of the equation of the interface area density will allow for the flow regime map to be simplified, since the equation uses the correlations that are not related to a particular flow regime;

(3) a module of a porous medium designed to simulate the core, the intermediate heat exchanger, and the steam generator in a 3D approximation, since the processes that occur in these system components are essentially non-one-dimensional; and

(4) an interface to the CFD codes that will ensure modeling of particular components of the reactor equipment and systems with the required accuracy.

For convenient use of the code, improvements will be made to the graphical user interface that allows setting the initial data.

In accordance with the federal requirements and rules of computational support for the development of reactor systems—including those of new types—in nuclear power engineering, verified and certified computational codes are to be used. In connection with this, on completion of verification and validation and presentation of the first version of the computational code for certification, the verification and the validation of the second version will be commenced.

For further improvement of the usability of the HYDRA-IBRAE/LM code and enhancement of the computation stability, the user community is supposed to be extended by supplying the code to relevant higher educational establishments, using the code for training and developing the training programs.

Consequently, the key features of the HYDRA-IBRAE/LM computational code are:

(1) use of tested computation techniques and physical models based on a wide range of experimental data, including the unique data obtained in recent years as a result of implementing the *Proryv* project;

(2) consideration of the specific structural features of the fast reactors developed within the framework of the *Proryv* project and creation of special-purpose models to ensure the required quality of simulation;

(3) exploitation of potentialities of modern computing systems that enhance the efficiency of computations;

(4) availability of the computational code for the users independent of the developers (alienability) and orientation toward a circle of users as wide as possible; and

(5) validation using a wide experimental data base.

The program of further development of the computational code provides for further improvements to the models and the algorithms aimed at enhancement of the quality of simulating the liquid-metal-cooled reactor systems. Introduction of the HYDRA-IBRAE/LM code into design institutions will allow for the computations to be carried out in the nearest future to substantiate the safe operation of potential NPP projects at a qualitatively higher level.

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Translated by O. Lotova