Simulations of Liquid Metal Flows by DNS CONV-3D Code

Vladimir V. Chudanov, Anna E. Aksenova, Valerii A. Pervichko

Abstract—To simulate the thermal hydraulics processes in fast reactors with a liquid metal coolant DNS CFD code CONV-3D has been developed. The paper presents the results of the application of CONV-3D code for simulation of sodium natural convection in the upper plenum of the MONJU (Japan) reactor vessel, the calculation results of the experiment conducted on the Phenix facility (France) with sodium coolant, mixing of sodium flows with different temperatures in the T-tube (Russia). The results of simulation of heavy-liquid metal (LBE) flow and heat transfer along a hexagonal 19-rod bundle with wire spacers (KALLA, Germany) are presented also. A satisfactory agreement of the numerical predictions with experiments is demonstrated, in particular for the temperature distribution vs the coordinates. The results obtained allow to conclude that using of CONV-3D code with high predictive power can be recommended for reactor applications.

***Keywords*—**CFD, CONV-3D, sodium, LBE, liquid metal coolant.

I. Introduction

ONE of the practical examples of the use of CFD in the nuclear industry may be modeling the thermal stratification of the coolant. The operating experience of the BN-600, experimental study and numerical simulation of BN-800 showed that in the upper reactor chamber and at the inlet of the heat exchanger there is a temperature stratification of the heat carrier (stratification) [1]. This issue is relevant for reactors of the fourth generation. Thermal stratification of the coolant leads to the formation of stagnant and recirculation zones with large gradients and temperature surges at the boundaries of isothermal zones, which cause additional thermal Cycling loads on the equipment and can have a significant influence on the service life of load-bearing structures.

The investigation of thermal hydraulic characteristics of stratified flow in the elements of the circuit of a fast reactor in different modes of operation, development of recommendations for reducing the temperature non-uniformity, as well as confirmation of the adopted design decisions on the effectiveness of a passive emergency heat removal with submersible heat exchanger in the upper chamber of the reactor are basic objectives of the numerical simulation. These tasks require a significant increase in computational costs compared to the traditional use of CFD codes and applications of supercomputers.

For the simulation of the thermalhydraulic processes in fast reactors with liquid metal coolant DNS CFD code CONV-3D has been developed [2—4]. This code has ideal scalability and is very effective for calculations on high performance cluster computers. The code has been validated on the set of analytical tests and experiments in a wide range of Rayleigh and Reynolds numbers, in particular, at extremely small Prandtl numbers [5—7]. The paper presents the results of the application of CONV-3D code for simulation of sodium natural convection in the upper plenum of the MONJU (Japan) reactor vessel. A satisfactory agreement of the numerical predictions with experiments is demonstrated. The calculation results of the experiment conducted on the Phenix facility (France) with sodium coolant are demonstrated. The experiment focuses on the mixing of two fluxes at different temperatures in the secondary circuit of reactor facility with liquid metal coolant in the presence of a bending tube. A small pipe is connected via T-connection to the main pipe and unloads of sodium in the main pipe at a temperature which is higher than in the main pipe. A satisfactory agreement of the numerical predictions with experiments is demonstrated, in particular for the temperature distribution vs the coordinates. A satisfactory agreement of the numerical predictions with experiments is demonstrated also for mixing of sodium flows with different temperatures in the T-tube (Russia). The results of simulation of heavy-liquid metal (LBE) flow and heat transfer along a hexagonal 19-rod bundle with wire spacers (KALLA, Germany) are presented.

The results obtained allow concluding that using of CONV-3D code with high predictive power can be recommended for reactor applications.

II. Modeling of reactor MONJU

MONJU — reactor facility of loop type with fast reactor with sodium coolant. The geometry of the upper chamber of the reactor MONJU has a complicated structure. System of tubes located above the exits from the active zone, creating a very complex flow in the region under the column. The size of

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these tubes is significantly less than the size of the camera. However, their influence on the course should be taken into account for the successful modeling of physical processes in the upper reactor chamber.

In 1995, tests were conducted on the crash of the MONJU reactor during its operation at a power level of 40 %. The transition of the reactor emergency cooling were simulated by applying a signal AZ "the Failure of the main condenser" and shutdown of the turbine. The stationary flow of the coolant in the MONJU reactor chamber in the cooling mode was investigated experimentally. The starting point corresponds to the mode of operation of the reactor at 40 % power.

To study the convection of sodium in the upper chamber of the reactor MONJU as the estimated area of the selected sector of the reactor 60º [9]. The outlets of fuel assembly presented hexagonal. Because of the complexity of the design all of the elements in above zonal region are not explicitly taken into account.

As input boundary conditions at the ends of the fuel assemblies are specified flow rate and temperature. The output from model is given by the condition of zero perturbations on the static pressure. On the planes clipping of sector as well as at the upper boundary of the computational domain, simulating the "sodium-argon" interface, the slip condition at the wall was defined. On all the walls the condition of no permeability set with a turbulent boundary layer, characterized by a logarithmic law of variation of the tangential component of the velocity.

The numerical simulations were performed for two modes of operation of the reactor MONJU: stationary regime at a power of 40% and in the cooling mode. The results of calculations were obtained by code CONV-3D using the quasi-DNS approach and the computational grid with size 321x257x513 (number of elements ~ 42 million), which was built using three-dimensional Geometry Editor [10]. Figure 1 shows the temperature profiles in the altitude chamber of the reactor MONJU in the location of the probe for 1) stationary and 2) mode of cooling at the time of time of 120 seconds, obtained in the experiment and by means of the CFD code CONV-3D.

In both cases, a satisfactory coincidence of results of numerical predictions with experiment is observed. Mode cooling results are also satisfactory, with the standard deviation in time is 120 seconds received code-CONV-3D is 7.1 °C (~3,4 %).

Figure 2 shows graphs of changes with time of the temperature of sodium at the points of location of the thermocouples of the probe, namely for thermocouple TE3 [9], obtained for cooling mode in the experiment and calculation codes CONV-3D. The temperature change in both cases is similar.

Thus, for both modes were obtained satisfactory results. The root mean square deviation of the calculated average values of temperature in comparison with the readings of the thermocouples in the experiment for the top of the mixing chamber does not exceed 9%.

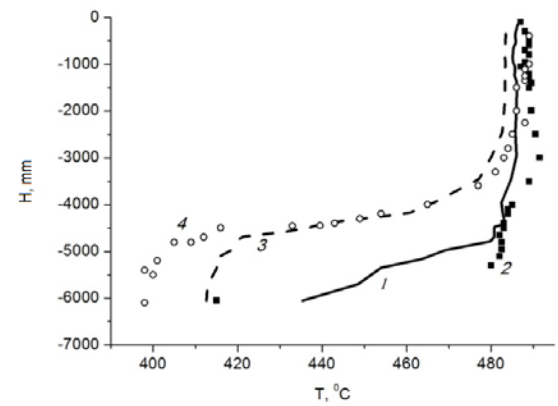


Fig. 1 profile of temperature along the height of reactor chamber: stationary (1) and experiment (2); in the mode cooling at the 120 s (3) and experiment (4)

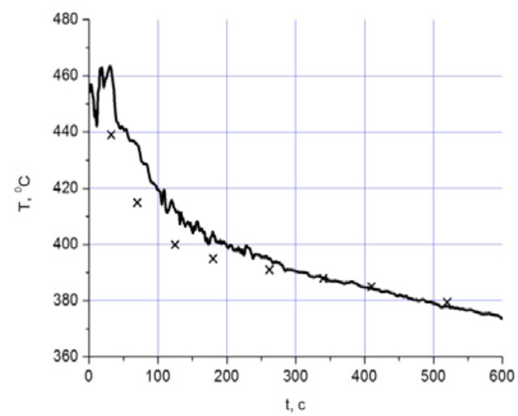


Fig. 2 profile of temperature change in the position of the sensor TE3 (—) and experimental (×)

For stable mode there is observed a decrease of the temperature distribution on the interval from -5000 to -6000 mm that can be caused by the simulating in 1/6 part of the total geometry. But even in this case, the divergence in values does not exceed 10 degrees, which is about 10% and is within the error.

Mean absolute temperature deviation Δ_T °C is of 5.6 °C, mean relative deviation in temperature, δ_T % – 2,7 %.

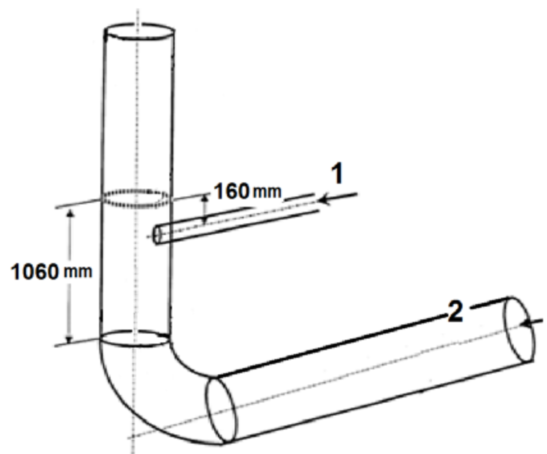
The following formula is used for variance calculation:

$$\Delta_T = /T_{exp} - T_{CFD}/ \qquad \text{and} \qquad \delta_T = \frac{/T_{exp} - T_{CFD}/}{T_{max} - T_{min}} \cdot 100\%$$

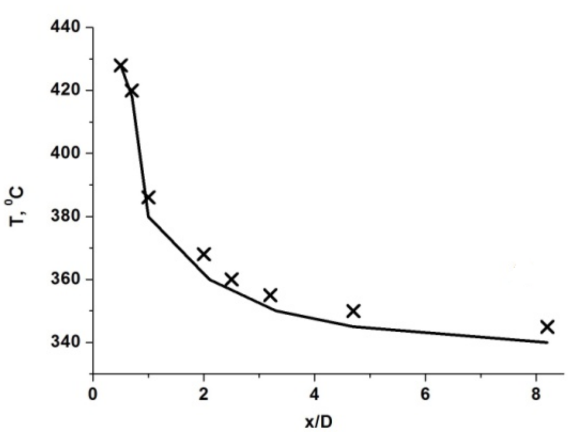
where T_{exp}, T_{CFD} – the experimental and calculated temperature at the selected by height point, correspondingly; T_{max}, T_{min} – maximum and minimum temperature in the experiment, correspondingly.

III. Simulation experiment Phenix with sodium cooled

This experiment is designed to study the mixing of two flows at different temperatures in the secondary circuit. During normal operation, sodium at a low temperature flows into the main pipe of the second circuit. A small pipe connected via T-connection with the main pipe, carries the sodium in the main pipe at a temperature higher than in the main pipe [11]. At figure 3a geometry of the experimental setup Phenix is shown. There are the following notations used: 1 — small pipe, the temperature of the coolant 430 °C with a consumption of 7 kg/s, d=68 mm; 2 — big pipe, the temperature of the coolant 340 °C with a flow rate of 800 kg/s, D=494 mm. The agreement between calculated and experimental temperature is satisfactory. A difference of numerical predictions by means of CONV-3D code and experimental results does not exceed 5 % (see fig. 3 b).



a)



b)

Fig. 3 geometry of the experimental setup Phenix and numerical predictions with comparison experiment

IV. Heavy-liquid metal (LBE) flow along a 19-rod bundle with wire spacers

Description of the experiment is given in [12]. For a better understanding of heat transfer and pressure drop in rod bundles with liquid metal cooling in typical conditions of the reactor, was running an experimental campaign in the Laboratory liquid metals Karlsruhe (KALLA) of the Karlsruhe Institute of Technology (KIT).

The campaign focuses on two goals: to justify existing empirical correlation for the operating conditions of fuel assemblies for the reactor MYRRHA [13] and obtain reliable experimental data for validation of modern CFD simulations and analysis tools of subchannels. Only stationary, adiabatic and forced-convection conditions are considered in [12]. After the experiment, 2014, an experiment was conducted in 2016: for eutectic bismuth+lead in the 19-rod bundle with wire spacers.

A test domain, installed vertically with upward flow, consists of a bundle of nineteen electrically heated rods embedded in a hexagonal channel. For practical research, the concept of double-casing was chosen. The inner casing forms a hexagonal channel for the flow, while the exterior it acts as a case of high external pressure. The staging domain is filled with a fluid – static eutectic lead-bismuth (LBE). Figure 4 shows a schematic view of fuel assembly (top view).

In the experiment [12] evaluated the coefficient of friction. In the calculations used properties from [12]. The calculation of the coefficient of friction for the task with winding is carried out according to the formula: $f_1 = \frac{\Delta p}{\rho} \frac{d_r}{\Delta z u^2 / 2}$. The dimensions of the computational domain are 66x60x1230 mm.

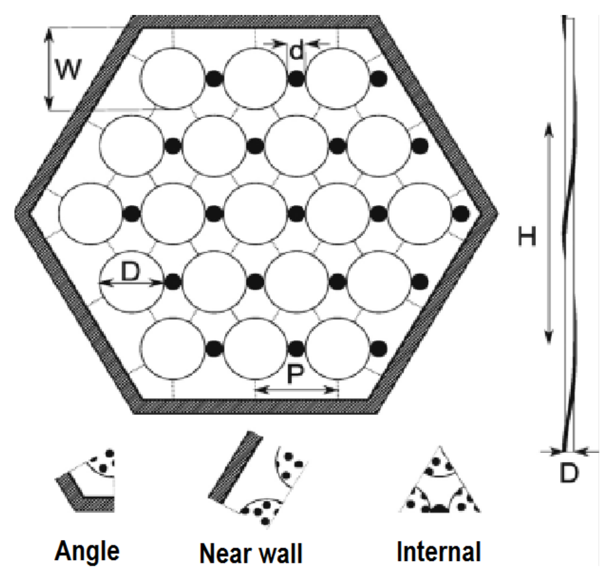


Fig. 4 a schematic view of fuel assembly (top view)

Formula Novenstern (the accuracy of the equation – 30%)

$$f_2 = \frac{0{,}3164}{Re^{0{,}25}}\left[\frac{1{,}034}{(P/d)^{0{,}124}} + \frac{29{,}7(P/d)^{6{,}94}Re^{0{,}086}}{(H/d)^{2{,}239}}\right]^{0{,}885},$$

where

$$1{,}06 \le s/d \le 1{,}42;$$
$$2{,}6 * 10^3 \le Re \le 2 * 10^5; 8{,}0 \le h/d \le 96.$$

The formula is (accuracy of equation – 15%)

$$f_3 = \frac{0{,}210}{Re^{0{,}25}}\left\{1 + \frac{124 * Re^{0{,}06}}{(H/d)^{1{,}65}} A\big((S/d) - 1\big)\right\},$$

where

$$A = \left[1{,}78 + 1{,}485\big((S/d) - 1\big)\right],$$
$$1{,}0 \le s/d \le 1{,}5;\ 10^4 \le Re \le 2 * 10^5; 8{,}0 \le h/d \le 50.$$

Figure 5 shows a grid spacer (bottom view). On the one hand, this design allows you to put thermocouples into it at selected temperature measurement locations. On the other hand, it leads to relatively large roughness; its value was estimated at 30 microns.

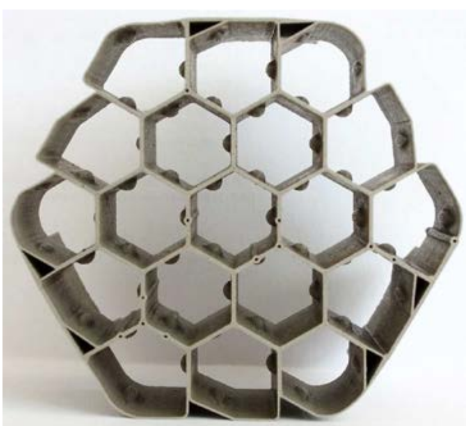


Fig. 5 a grid spacer (bottom view)

Evaluation of friction coefficient on the basis of numerical experiment for the geometry "as is" in the experiment, with turbulence at the inlet and outlet are summarized in table I.

Tab. I friction coefficient for the geometry “as is”

№	Exp.	f_1 (pre-dictions)	f_2 (Novenstern)	f_3 (normative document)
1	0,028	0,022	0,029 $f_2^{-30\%} = 0{,}020$	0,023 $f_3^{-15\%} = 0{,}020$
2	0,025	0,018	0,022 $f_2^{-30\%} = 0{,}016$	0,018 $f_3^{-15\%} = 0{,}015$

Test 1 performed on the grid 129x129x513 with skew symmetrical scheme with upwind differences.

The following characteristics for calculations are used: average velocity $\bar{w} = 549\ mm/s$; a hydraulic diameter $d_г^{с\ нав.} = 7{,}5mm$; this correspond to the Reynolds number $Re = 1{,}8 * 10^4$.

Test 2 performed on the grid 129x129x513. The following characteristics for calculations are used: average velocity $\bar{w} = 1637\ мм/с$; a hydraulic diameter $d_г^{с\ нав.} = 7{,}5мм$; this correspond to the Reynolds number $Re = 5{,}3 * 10^4$.

V. Mixing of sodium flows with different temperatures in the T-tube

To study processes of mixing of flows with different temperatures the sodium test facility was developed, manufactured and put into operation in Institute of Continuous Media Mechanics of the Ural Branch of Russian Academy of Science (ICMM UB RAS) [14].

The test facility consists of “cold” and “hot” branches and a test section. Each branch includes a sodium-air heat exchanger or a heater and an electromagnetic pump and measuring instrumentation for coolant flow and temperature. Basic parameters of sodium test facility are given in Table II.

Tab. II parameters of sodium test facility

Parameter	Value
Electric heater power, not less than, kW	20
Heat exchanger power, not less than, kW	20
Maximum flow rate of “hot” sodium through the experimental model, l/s	1.0
Maximum flow rate of “cold” sodium through the experimental model, l/s	0.5
Sodium temperature in the test section, not more than, °C	250

Experimental studies on mixing of sodium flows with different temperatures were made at test facility on a T-tube model (see Fig. 6). There are the following notations used: 1 – mixing portion; 2 – “cold” sodium inlet portion; 3 – thermal couples at the “cold” branch inlet; 4 – thermal couples at the “hot” branch inlet; 5 - “hot” sodium inlet portion; 6 – thermal couples in flow.

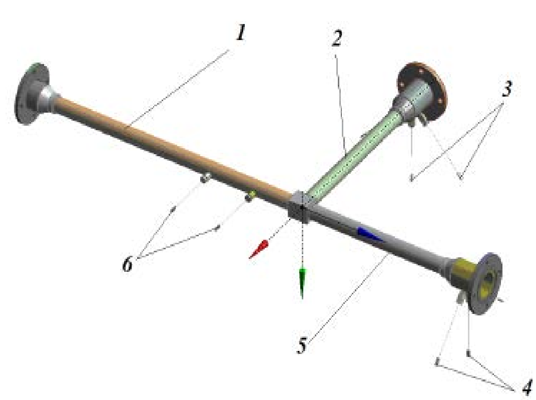


Fig. 6 the T-tube model

This is a T-shaped connection of straight stainless steel tubes of 40 mm inner diameter; tube wall thickness is 1.5 mm. In the T-tube the "hot" sodium circulates in the through mode, and "cold" sodium circulates via the side supply. "Hot" and "cold" sodium enters the T-tube via honeycombs. The selected thickness of the T-tube model wall ensures low heat retention, which allows measuring temperature pulsation at the external surface of the circuit using an infra-red imager. Low-inertia micro thermal couples were used to measure temperature in sodium flow.

Experimental studies were performed for three modes which differ in ratios of flow rates and temperatures between "hot" and "cold" coolant, coming to the inlet portions of the T-tube model (see Table III below).

Tab. III basic characteristics of experimental models

Mode's number	Qh, ml/s (hot heat flux)	Qc, ml/s (cold heat flux)	Th, °C (temperature of hot heat flux)	Tc, °C (temperature of cold heat flux)
1	1040	490	207	153,4
2	556	231	222,1	126,3
3	915	830	184,9	140,8

The model for numerical simulation of the process of mixing of sodium flows with different temperatures in the T-tube includes honeycombs and diffusers on inlet portions in addition to the T-tube. The assigned boundary conditions correspond completely to the conditions under which the experimental studies were performed [14].

The coolant temperature distribution is shown in figure 7.

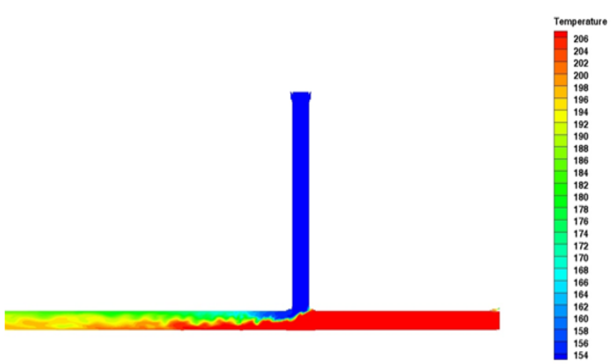


Fig. 7 coolant temperature distribution

The average temperature dependence ($\Delta_T = /T_{exp} - T_{CFD}/$) in monitoring points T1-T10 is shown in figure 8. Good agreement between experimental and calculation data was obtained.

For the problem of mixing of non – isothermal flows of sodium coolant in a tee the average temperatures at the control points T1 – T10 have relative errors not exceeding 5-10% for the three experimental regimes.

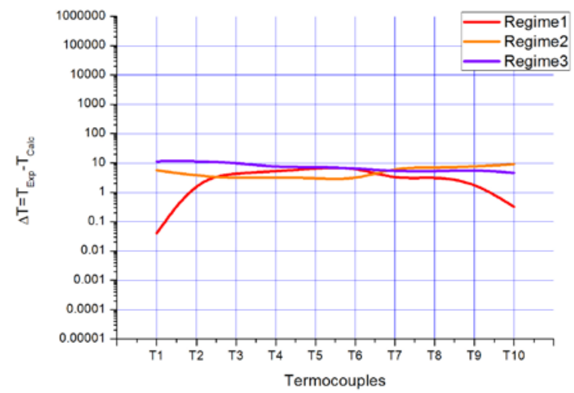


Fig. 8 average temperature dependence ($\Delta_T = /T_{exp} - T_{CFD}/$) in monitoring points

Moreover, the average temperatures at the control points Na1 and Na2 located in the sodium stream have also relative errors not exceeding 5-10% for the three experimental regimes (see table IV).

Tab. IV the average temperatures at the control points Na1 and Na2

Monitor. Points	Regime1	Regime2	Regime3
Na1	3,02	4,48	4,48
Na2	0,3	0,8	2,99

VI. Conclusion

Developed DNS CFD code CONV-3D for the simulation of thermal hydraulic processes in fast reactors with liquid metal coolant allows to simulate currents in the structural components of nuclear power plants in a wide range of parameters Ra < 10^{16} and Re=10^3—10^5. This is evidenced by the results of the qualitative and quantitative coincidence with the experiment.

According to the results of numerical simulation of MONJU reactor root mean square deviation of the calculated average values of temperature in comparison with the readings of the thermocouples in the experiment for the top of the mixing chamber does not exceed 9%.

A satisfactory agreement of numerical predictions with experiment on the setting of the Phenix (France) with sodium coolant was demonstrated. A difference numerical predictions and experimental results does not exceed 5%.

Good agreement between experimental and calculation data was obtained the problem of mixing of non – isothermal flows of sodium coolant in a tee. The average temperatures at the control points Na1 and Na2 located in the sodium stream, and at the control points T1 – T10 have relative errors not exceeding 5-10% for the three experimental regimes.

According to the results of modeling the flow of the LBE in the 19th rods assemblies, rods, twisted in the range of the Reynolds number $\leq 10^4$ obtained with the help of CONV-3D code results have a relative error not exceeding 5 – 10 % coefficient of hydraulic resistance and coefficient of friction.

References

[1.] S. L. Osipov, et al. “Experience and problems of verification of CFD codes in the projects in the fast reactor”, in *Book of abstracts the scientific-technical seminar Problems of verification and application of CFD codes in nuclear engineering,* N. Novgorod, 2012. 62 p.

[2.] V.V. Chudanov, et al., “A numerical study on natural convection of a heat-generating fluid in rectangular enclosures”, *Int. J. Heat Mass Transfer*, 37, 1994, p. 2969.

[3.] V.V. Chudanov, et al., “Methods of direct numerical simulation of turbulence with use DNS and LES approaches in the tasks of thermal hydraulic fuel Assembly”, *Izvestia of Russian Academy of Sciences, Energetics*, 6, 2007, p.47.

[4.] V.V. Chudanov, et al., “New method for solving of CFD problems at clustered computers petaflops performance”. Software systems: theory and applications, 1, 2014, p.3.

[5.] J. Mahaffy, “Synthesis of Results for the *T*-junction benchmark”, in *Proc. CFD4NRS-3 Conf. on Computational Fluid Dynamics for Nuclear Reactor Safety Applications*, USA, 2010, vol. 1.

[6.] P. Betts, et al., “Experiments on turbulent natural convection in an enclosed tall cavity”, *Int. J. Heat Fluid Flow*, 21, 2000, p.675.

[7.] B.L. Smith, et al., “Report of the OECD/NEA KAERI Rod Bundle CFD Benchmark Exercise”, *Report NEA/OECD*, №NEA/CSNI/R(2013)5, 2013. 124 p.

[8.] P.L. Kirillov, et al., “Thermophysical properties of liquid metal coolants”, Review, IPPE-0291 CNII ATOMINFORM, 2000, 42 p.

[9.] INTERNATIONAL ATOMIC ENERGY AGENCY, Benchmark Analyses of Sodium Natural Convection in the Upper Plenum of the Monju Reactor Vessel. *Final Report of a Coordinated Research Project 2008–2012*. IAEA-TECDOC-1754, Vienna, 2015.

[10.] Methods of computational fluid dynamics to analyze the security of FEC objects. Under the General editorship of L. A. Bolshov. *Proceedings of IBRAE RAS.* Vol. 3. Moscow: Nauka, 2008. 210 p.

[11.] INTERNATIONAL ATOMIC ENERGY AGENCY, Validation of fast reactor thermomechanical and thermohydraulic codes. *Final report of a co-ordinated research project 1996–1999*. IAEA-1318, Vienna, 2002.

[12.] J. Pacio, et al. “Experimental study of heavy-liquid metal (LBE) flow and heat transfer along a hexagonal 19-rod bundle with wire spacers, *Nucl. Engin. and Design*. 301, 2016, p.111.

[13.] H.A. Abderrahim, et al., “MYRRHA – a multi-purpose fast spectrum research reactor”, in *Proc. 10th International Conference on Sustainable Energy Technologies SET-2011*, 2012.

[14.] A. Vasiliev, I. Kolesnichenko, A. Mamykin, P. Frick, R. Khalilov, S. Rogozhkin, V. Pakholkov, “Turbulent convective heat transfer in an inclined tube filled with sodium”, *Technical Physics*, 60, №9, 2015, p.1305.