The problem of decay heat removal under accidental conditions in the liquid metal cooled fast reactor

Iu.E. Shvetsov[1], E.O. Adamov[1], V.I. Rachkov[1], Iu.S. Khomyakov[1]

[1]ITC "PRORYV" Project, Moscow, Russia

E-mail contact of main author: Iu.E. Shvetsov - shuye@proryv2020.ru

Abstract. The problem of decay heat removal in case of reactor shutdown remains urgent to date, and it has been confirmed by the Fukushima accident. Besides, the problem becomes more severe for the large size reactors.

Based on the results of calculations on the reactor decay heat removal transients made by 3D thermohydraulic codes, comparative analysis was made of heat removal effectiveness of the large size reactors cooled by lead and sodium.

Two types of decay heat removal systems (DHRS), with different locations of decay heat exchanger (DHX) are compared. In the first option DHX is situated in the reactor upper plenum, and so heat is removed from the core by the coolant flowing in the circuit "DHX – upper plenum – inter-subassembly space (ISS) of the core – upper plenum". In the second case, DHX is located in the gap downcomer of the reactor, and the check valve is provided in the circuit, which does not prevent hot coolant flow from the reactor upper plenum to the DHX inlet in the decay heat removal mode but stops this coolant flow under normal reactor operation.

It has been demonstrated that both DHRS options are more effective as applied to the sodium cooled reactor. In case of the large size reactor cooled by lead DHRS with check valve is also capable of removing decay heat without exceeding permissible temperature values. However, in the option with DHX located in the upper plenum, fuel element temperature exceeds safe operation limit for a short time.

Key Words: fast reactor, liquid metal coolant, accidents, decay heat removal.

1. Introduction

Fast neutron reactors have been chosen as the basic component of the new Russian integrated technological platform aimed at solving not only the problem of safe electricity production in the integrated nuclear power body, but also the problems related to reprocessing and refabrication of nuclear fuel within closed fuel cycle. Solving these problems is the main goal of ROSATOM's activity within the framework of "PRORYV" innovative project. By now reactors cooled by sodium and lead have been chosen as the basic options in this project. As applied to the reactor designs considered in "PRORYV", as well as the other power reactors, the problem of decay heat removal from the reactor core in case of accident resulting in failure of normal heat removal systems remains urgent. What is more, importance and complexity of this problem increase with the reactor size, and possible design approaches, as a rule, mean the attempt to reach a compromise between reliability and technical efficiency of some system, on the one hand, and its cost, on the other hand.

Studies on the optimization of decay heat removal system (DHRS) design as applied to fast reactors with sodium coolant have been carried out for many years both in Russia and abroad.

While Intermediate Reactor Auxiliary Cooling System (IRACS) with special decay heat removal loop "hung" on the secondary system was commonly used in the early designs of sodium cooled fast reactors (SFR), now reactor designers in Russia and in other countries prefer to use Direct Reactor Auxiliary Cooling System (DRACS) with decay heat exchangers (DHX) submerged in the primary coolant. The main advantages of this system are as follows: i) self-sustainability and ii) use of passive means (natural coolant flow). Its drawbacks include required additional expenses for development, justification and operation. DRACS has been chosen for some foreign reactor designs, namely: European Fast Reactor (EFR), China Experimental Fast Reactor (CEFR), Indian demonstration reactor PFBR, Korean demonstration reactor KALIMER, Japanese demonstration reactor DFBR and Russian reactor BN1200.

There are various designs of DRACS type DHRS. In most reactor designs, decay heat exchangers of DHRS are located directly in the upper reactor plenum. In this case, as a rule, decay heat is removed from the core by two parallel coolant flows. Some part of "cold" sodium entering reactor upper plenum from the DHX outlet, flows down in the peripheral subassemblies of the core and radial shielding, enters core diagrid and then returns to the upper plenum flowing inside the subassemblies. The other part of sodium flows into the inter-subassembly space (ISS). Thus the heat is removed from the fuel in two ways: by sodium flowing inside fuel subassemblies and by sodium flowing in the inter-subassembly space (with heat transfer through the core subassembly duct wall) [1]. The basic drawback of this option is lower heat removal effectiveness caused by additional thermal resistance provided by the subassembly duct wall (separating cold sodium flowing down in the inter-subassembly space and sodium upflow inside the subassembly) and by the decrease of coolant flow rate in the circuit because of rather high ISS hydraulic resistance.

There is an alternative DHRS design option [2], in which DHX outlet is connected not to the upper reactor plenum but to the core diagrid by special pipeline with check valve (for instance, hydraulically suspended flapper), which opens in a passive manner when the primary coolant flow rate decreases down to some preset value. In this option natural coolant flow is set up through the circuit: "DHX – check valve – piping – core diagrid – reactor core". The drawback of this design is the presence of additional equipment (check valve), its reliability practically determining that of the whole system.

As regards lead cooled fast reactors (LFR), there is much less experience gained in solving the problem of decay heat removal. Nevertheless, it is initially clear that the relatively low thermal conductivity of lead would aggravate the problem. Therefore the possibility to apply in the lead cooled reactors the approaches used for decay heat removal in sodium cooled reactors requires justification.

In [3] presented are the results of comparative analysis of different decay heat removal system designs as applied to the large size (N = 4,200 MW) sodium cooled reactor. It has been demonstrated that the most effective decay heat removal from the reactor is assured by IRACS design with the intermediate circuit of decay heat removal system connected to the secondary circuit of reactor plant, thus assuring sodium flow in the intermediate heat exchanger (IHX). The effectiveness of heat removal of DRACS option with the check valve is a little bit lower. In case of use of classical DRACS design with submerged DHX the highest core temperatures are observed in the mode of decay heat removal.

The possibility of decay heat removal using DRACS type DHRS in the large size reactors (N = 2,800 MW) cooled either by sodium or by lead are considered below. Decay heat removal systems with DHX submerged in the upper reactor plenum and those with the check valve are compared in terms of their effectiveness for the two types of the reactors. In addition, possible modifications of the lead cooled reactor core design are considered in order to increase the effectiveness of heat removal from the fuel elements. The calculations have been made using Russian 3D thermohydraulic codes: GRIF [4] and SVIR.

GRIF is a single-phase universal thermohydraulic code designed for calculations of dynamics of thermohydraulic parameters of sodium cooled reactor in both steady state and transient operation modes. The important feature of this code is the possibility to use the models with various geometrical dimensions for different reactor structural elements and simulate thermohydraulics of not only the main reactor circuit but also inter-subassembly flow path. From the standpoint of analysis of decay heat removal modes it is most important that this code is capable of modelling thermohydaulics of the reactor taking into account heat and mass transfer with the inter-subassembly space in the reactor core.

The code includes the following modules:

- 3D thermohydraulic model (based on "porous body" model) for analysis of sodium velocity, pressure and temperature patterns in the primary circuit of the reactor;
- 3D model for analysis of sodium velocity, pressure and temperature patterns in the core inter-subassembly space;
- package of 1D, 2D, and 3D models for analysis of temperature patterns in the "impermeable" elements (fuel pins, fuel subassembly ducts, etc.);
- thermohydraulic model of the intermediate and decay heat exchangers;
- primary pump model;
- 1D model of the secondary circuit;
- 1D model of DHRS intermediate circuit;
- neutron kinetics point model.

Thermohydraulic module designed for calculations of sodium velocity, pressure and temperature 3D patterns in the primary circuit in cylindrical r-φ-z geometry is the main component of the code. Heat and mass transfer equations set includes mass, momentum and energy balance equations presented within the framework of "porous body" model. The liquid is considered incompressible, and stratification effects are taken into account using Boussinesq approximation.

The similar equation set is solved for the sodium flowing in the inter-subassembly space. Solutions of both equation sets are sewed together on the outer circuit of the subdomain simulating reactor core, since two sodium flows (the main flow in the subassemblies and that in the inter-subassembly space) only join on the boundaries. Heat transfer across the subassembly duct wall between sodium flows throughout the core is taken into account.

SVIR code designed using the same principles has the same structure, and it is used for the analysis of decay heat removal processes in lead cooled reactors.

2. Sodium cooled large size reactor

Sodium cooled reactor of 2,800 MW power with core composed of ducted fuel subassemblies is considered. Reactor parameters for rated power mode are presented in column two of Table I.

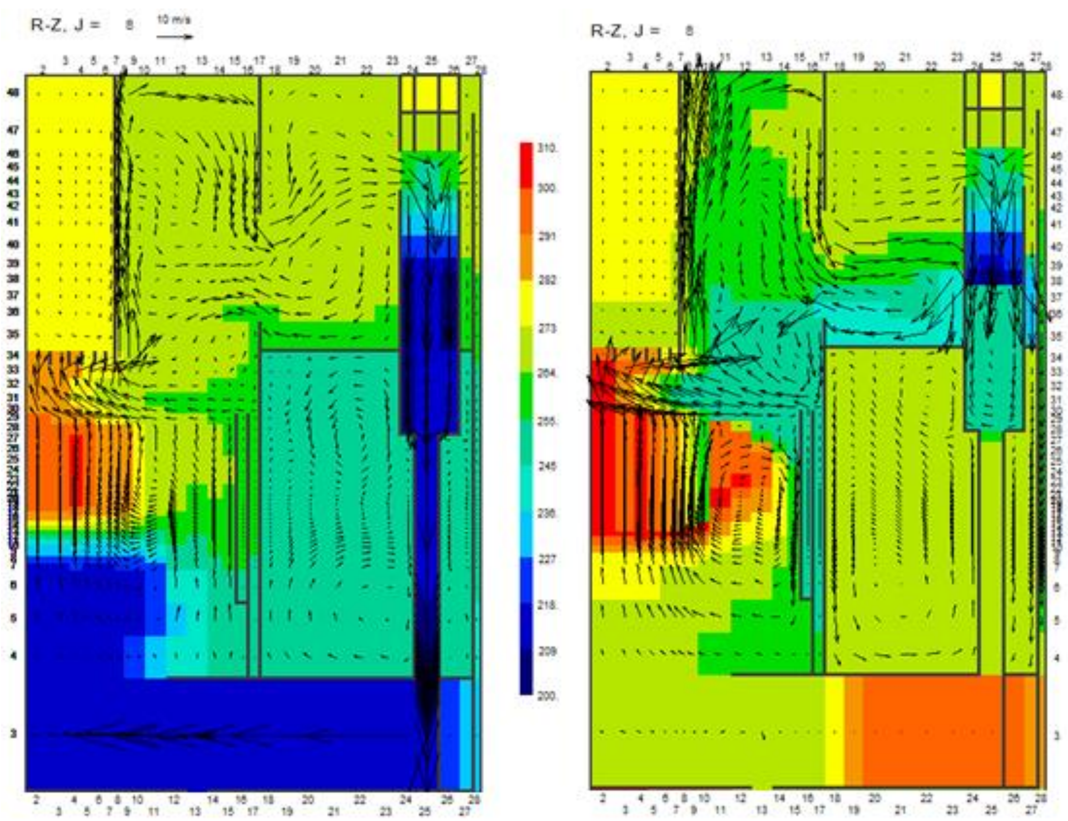
Table I. Reactor parameters for rated power mode

Parameter	Value	
Coolant	Sodium	Lead
Rated Power, Mwt	2800	2800
Core flow rate, kg/s	16350	158400
Coolant temperature at core inlet, ºC	410	420
Coolant temperature at core outlet, ºC	560	520
Max. pin cladding temperature, ºC	593	640

Postulated scenario of LOF accident for SFR:

- Initial failure – – loss of power supply at the moment τ=0s;
- SCRAM in two seconds (at τ=2s);
- Primary and secondary pumps run down during 100 seconds;
- Gate valves of the air heat exchanger are opened in 25 seconds;
- DHX check valves are opened at τ=50s.

Decay heat removal from the core is carried out by three circuits (sodium-sodium-air) using straight-tube DHX located in the upper reactor plenum. Two types of DHRS are compared: one with submerged DHX and one with check valve. It should be noted that all DHX parameters (flow cross section, pressure drop, and heat transfer surface) are equal in both options. Failure of primary and secondary sodium pumps is assumed as the accident initiating event. Safety rods are dropped into the core and gate valves of the air heat exchanger are opened to set up natural coolant flow in the DHRS intermediate circuit and in the reactor enhancing heat removal from the core. Fig.1 shows sodium velocity and temperature patterns in the reactor elevation on DHX cross section, calculated for two options in decay heat removal mode (τ = 60,000 s). It should be noted that sodium velocity and temperature patterns shown in the vicinity of subassembly ducts (where two flows exist, namely: one – inside subassembly and the other – in the inter-subassembly space) correspond to the ISS flow. It can be seen that in case of submerged DHX the circuit is closed through the inter-subassembly space, where vortex region is formed. So, in the option with check valve, cold coolant is supplied to the core diagrid (Fig. 1a), and in case of submerged DHX (Fig. 1b) the coolant returns to the upper plenum and its considerable portion enters inter-subassembly space in the core periphery.



a) DHRS with check valve *b) DHRS with submerged DHX*

FIG. 1. Sodium velocity and temperature patterns in the mode of decay heat removal in the reactors with different DHRS designs (calculation was made using GRIF code).

As follows from Fig. 2, comparison made in terms of temperature shows that the effectiveness of DHRS with submerged DHX is somewhat lower, however this system is also capable of removing decay heat from the core without excess heating of the fuel elements. The decrease of the effectiveness is caused by the additional thermal resistance of the subassembly duct wall, through which heat is transferred to the sodium flowing in the inter-subassembly channels. As a result, the general temperature level in the reactor with the system using submerged DHX turns out to be higher by 70 – 80 ºC, although not exceeding design limit of 650 ºC adopted for the stainless steel cladding of the fuel elements used in the Russian fast reactors.

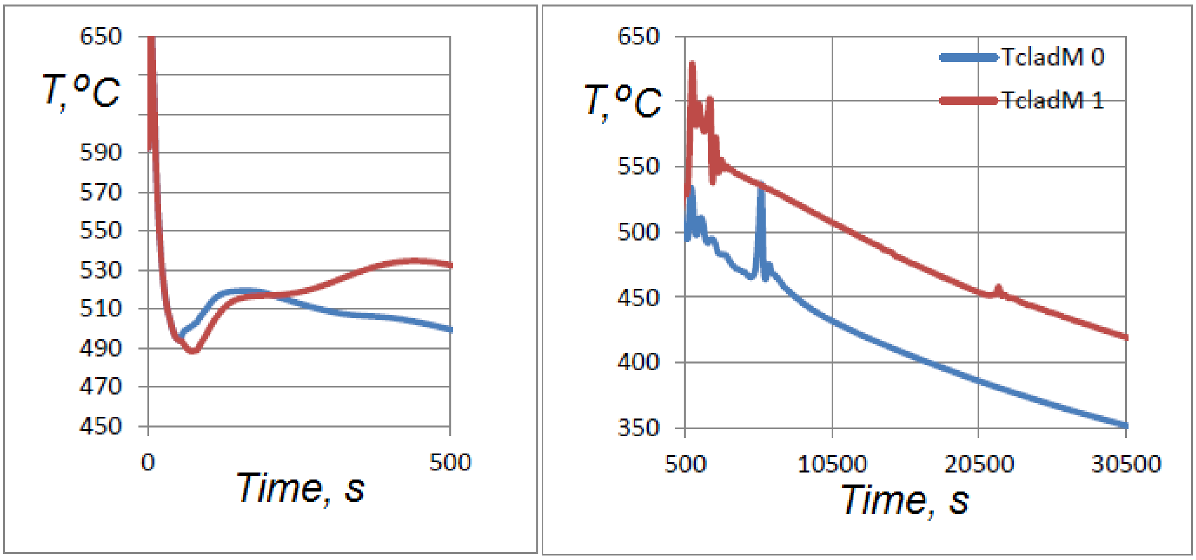


FIG. 2. Maximum fuel element cladding temperature in sodium cooled fast reactor with different DHRS designs: (0) – with check valve, (1) – with submerged DHX.

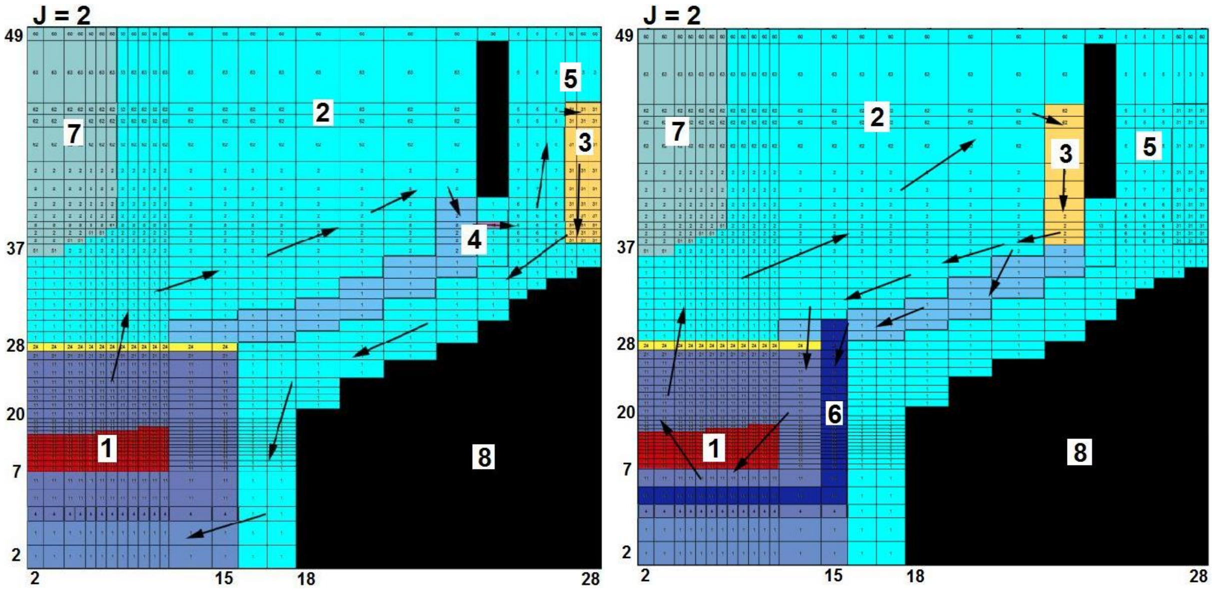
3. Large size fast reactor with lead coolant

Analytical studies on decay heat removal mode have been carried out for lead cooled reactor which has much in common with sodium cooled reactor design previously considered, namely:

- the same rated power (2,800 MW);
- reactor core composed of ducted fuel subassemblies;
- three circuits decay heat removal system (lead – lead – air).

Reactor parameters for rated power mode were presented in column three of Table I. Postulated scenario of LOF accident for LFR is almost the same as that for SFR. The difference is that DHX check valves are opened earlier - at τ=5s and primary and secondary flow rates decrease during 25 seconds instead of 100 seconds.

As in case of sodium cooled reactor, two DHRS designs are compared: one with check valve (Fig. 3a) and one with DHX submerged into the reactor upper plenum (Fig. 3b). However, as it is shown in Fig. 3a, DHRS with check valve in lead cooled reactor is somewhat different from that used in sodium cooled reactor. In reactor with lead coolant, there are special channels with check valves provided to connect the upper plenum with the downcomer, where decay heat exchangers are located. Under normal operating conditions of the reactor, heat removed from the core by coolant flow provided by the pumps is transferred to the steam generators located outside the reactor, and the check valve is closed. In case of accident caused, for instance, by the loss of power, pumps run down and safety rods are dropped into the core. The main flow circuit is broken because the steam generators are located at the level higher than that of lead in the reactor, and natural coolant flow in the reactor and decay heat exchangers is possible because the check valve is opened. Heat from the intermediate lead circuit is transferred to the air of the third circuit through special heat exchanger.



1 – reactor core; 2 – upper plenum; 3 – decay heat exchanger; 4 – check valve; 5 – downcomer; 6 – channels enhancing coolant flow in ISS; 7 – control rods column; 8 – impermeable elements

a) DHRS with check valves *b) DHRS with submerged DHS*

FIG. 3. SVIR code calculation domain simulating lead cooled large size reactor

In Fig. 4 presented are curves showing dynamics of maximum fuel element cladding temperature in decay heat removal transient of lead cooled reactor under study using different DHRS designs.

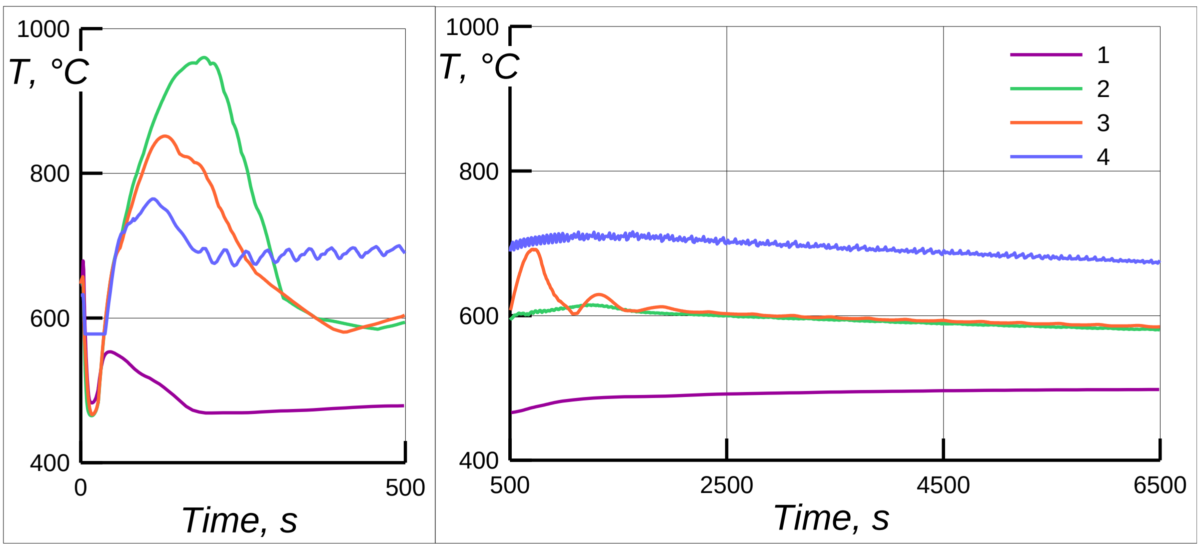


FIG. 4. Maximum fuel element cladding temperature behavior in lead cooled reactor with different DHRS designs: (1) check valves, (2) submerged DHX, (3) submerged DHX + ISS channels, (4) submerged DHX + ductless core fuel subassemblies.

Comparison of different options shows that if DHX is only relocated from the downcomer area to the upper reactor plenum, then maximum cladding temperature reaches 960 ºC value, which exceeds safe operation limit (T_{clad} < 800 ºC). In order to enhance heat removal, it was proposed to provide special “channels” on the core periphery and in the lower section of ISS enabling “cold” coolant supply to this area (Fig. 3a). These “channels” would allow to decrease maximum cladding temperature down to 700 – 850 ºC depending on the time required for lead discharge from the steam generators. After all, it also follows from Fig. 4 that the use of ductless fuel subassemblies in the whole core could be a comprehensive solution of the problem of decay heat removal. In this case maximum cladding temperature is as low as 760 ºC.

4. CONCLUSION

The results of studies are summarized in the Table II below.

TABLE II: Maximum temperature of the fuel element cladding in decay heat removal transient (in the second stage of the process, when mass transfer in the circuit is only caused by natural flow).

Reactor type	**Sodium cooled**		**Lead cooled**			
DHRS design	With check valve	With submerged DHX	With check valve	With submerged DHX		
Core design	Ducted	Ducted	Ducted	Ducted	Ducted, with ISS channels	Ductless
$T_{max\ clad}$, °C	530	630	553	940	850	764

The main conclusions are as follows:

(1) DHRS design with check valve is most effective from the standpoint of decay heat removal for both sodium cooled and lead cooled reactors. The difference between lead and sodium coolants in terms of their thermal properties results in slightly higher maximum cladding temperature in lead cooled reactor.

(2) Use of DHRS design with submerged heat exchangers in sodium cooled reactor does not lead to unacceptable increase of the fuel element temperature, although maximum temperature in this case is higher by 100 ºC.

(3) “Traditional” design of decay heat removal system with submerged decay heat exchangers as applied to the reactor with lead coolant is incapable of effectively removing decay heat from the core unless some major core design changes are introduced. Exceeding safe operation temperature limit of the fuel elements can be avoided by modification of the core design aimed at the enhancement of coolant flow in the inter-subassembly space. The use of ductless core design is even more effective. In this case maximum cladding temperature exceeds safe operation limit for only short time.

REFERENCES

[1] S.F. SHEPELEV, IU.E. SHVETSOV, et al., Analytical and experimental justification of decay heat removal system of BN-1200 design, Atomnaya Energiya, v. 116, No. 4, April 2014, pp. 222–228.

[2] I.A. KUZNETSOV and IU.E.SHVETSOV, et al., “Calculation of thermal-hydraulic parameters of fast neutron reactor with account of inter-fuel-assembly space influence”, Book of extended synopses. (International meeting FR09, 2009), p.483, IAEA, CN-176, Kyoto (2009).

[3] IU. SHVETSOV, IU. AHURKO, et al., “Comparative Analysis of Effectiveness of Various Emergency Core Cooling System Design Options for Sodium Fast Reactors of High Power”, Book of extended synopses. (International meeting FR13, 2013), IAEA, T1-CN-199/472, Paris (2013).

[4] IU. SHVETSOV and A.V. VOLKOV, “GRIF and HYDRON – 3D Codes for Analysis of Thermal and Hydraulics Parameters or Reactors with 1-Phase Incompressible coolant”, Report (10-th International Meeting of IAHR Working Group on Advanced Nuclear Reactors Thermal Hydraulics), PFBR, Obninsk (2001) 119–137.