Safety assurance of the new generation of the Russian fast liquid metal reactors

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Abstract. The paper presents approaches to the safety ensuring and justification of the new generation of fast liquid metal reactors with sodium and lead coolants being currently developed in Russia. The optimal design solutions, minimization of potential nuclear hazard, application of passive elements affecting reactivity and passive systems of residual heat removal, increased reliability of main equipment and safety systems ensure the enhanced safety of these reactors. The main directions of improving the software and regulatory framework outlined in the Russian Federation are presented, and the high potential role of international cooperation in this field is noted.

Key Words: Safety, fast neutron reactor installation, BN-1200, BREST-OD-300.

1. Introduction

During the development of the innovative new generation fast neutron reactors within the framework of the Federal Target Program "Nuclear Energy Technologies of the New Generation for the Period of 2010-2015 and for the Prospect up to 2020" [1] the basic approaches to ensuring their safety have been formed, including:

- enhancing the intrinsic properties of safety (self-protection) by minimizing the maximum reactivity excess, the void effect of reactivity, by strengthening negative feedback and the use of passive elements affecting reactivity on the hydraulic and temperature principle;
- application of a reactor installation (RI) integral configuration, excluding a loss-of-coolant accident;
- development of systems for removing residual heat from the primary circuit due to natural circulation with passive principles of initiating their operation and ensuring a long-term accumulation of residual energy release without reaching a dangerous temperature level;
- enhancing the reliability of main equipment and safety systems.

As a basic requirement for the new generation RI, the task is set to eliminate severe accidents, leading to the potential need for protective measures such as evacuation and resettlement of civilian population. Also, there are the economic requirements to the safety systems - quantity and quality of the systems providing the enhanced safety should not lead to essential increase of expenses for RI construction and operation.

The Russian experience in the development and operation of the BN-600 and development BN-800 has convincingly demonstrated the high level of nuclear and radiation safety achieved. Nevertheless, there is reason to believe that the potential for improving the safety of the fast neutron RI is far from being worked out. The paper demonstrates the approach to safety of new Russian power units with the BN-1200 reactor with sodium coolant and the BREST reactor with lead coolant.

In order to make optimal common decisions, the new designs of power units with fast reactors are covered by one design direction named "PRORYV" ("breakthrough") that also includes R&D to support justification of closed fuel cycle facilities for these reactors.

2. Common features and approaches ensuring the enhanced safety of fast reactors with sodium and lead coolants

The BN and BREST-type reactors being developed are characterized by a number of major safety-related features that are typical for reactors with liquid metal coolants and integrated configuration of the first circuit. These properties are summarized in Table 1.

TABLE I: Common features of intrinsic self-protection typical for BN and BREST-type reactors

Features	Achieved effect
Full integration of the 1st circuit (absence of pipelines and equipment out of the reactor vessel)	Elimination of the possibility of a radioactive coolant leak from the 1st circuit based on a high reliability of the reactor vessel
Large heat capacity of the integral type reactor (main equipment of the first circuit is in a reactor tank)	Low rate of the coolant average temperature increase in the first circuit in case of the termination of reactor heat removal after emergency protection system triggering that provides safe activation of the system of emergency heat removal (SEHR), including its activation by passive means, and considerable time to perform emergency management actions.
Low overpressure in a reactor gas plenum due to the high boiling temperature of sodium	Helps to achieve a small leak of radioactive gases from reactor at normal operation and in case of reactor gas plenum leak.
Absence of phase transitions of the coolant in case of circuit depressurizing	Helps to keep a reliable reactor cooling in case of emergency due to the forced or natural coolant circulation
Steady negative factor of reactivity against reactor capacity and temperature of reactor core materials	Restriction of reactor capacity escalation at unauthorized input of positive reactivity, decrease of capacity at emergency reactor heating
High thermal conductivity	Efficient heat removal at normal and emergency operation

While developing the fast reactors designs, the initial events have been revealed and the scenarios that can potentially lead to nuclear and radiation accidents have been defined:

- Leak of coolant from the 1st circuit;
- Blocking of a fuel assembly clear opening;

- Loss of heat removal from reactor core (failure, including SEHR);
- Termination of forced circulation of coolant in the first circuit (main circulation pump (MCP) halt) without activation of emergency protection (ULOF);
- Input of positive reactivity (self-feed of control and protection system (CPS) rods) without activation of emergency protection (UTOP).

To reduce or even to eliminate the negative consequences of these accidents, various measures are foreseen for BN and BREST-type reactors, which will be described further in the text.

In the given section, it is necessary to discuss the general possibility, that is typical for these reactors, to minimize consequences of UTOP accident in relation to the most severe case, such as self-feed of all compensating rods, i.e. full liberation of reactivity reserved for compensation of reactivity loss because of fuel burning. Simultaneous extraction of all CPS rods from reactor core is practically excluded due to the used approach to their management and circuit design of control system. However, such scenario needs to be analyzed for perspective RI considering its high potential danger. For the purpose of reducing the danger of this scenario, the nitride fuel was selected in the Project “PRORYV” as the basic fuel. Using of nitride fuel allows implementing a reactor core without zones of reproduction with breeding ratio of the core BRC $\geq$1. In turn, such solution allows reaching two effects:

- Minimizing of a reactivity margin against a fuel burn-up;
- Excluding a possibility of weapon-grade plutonium generation and extraction of plutonium as it is that provides solution of nonproliferation problem.

Calculations performed for reactor cores of BN-1200 and BREST-OD-300 reactors have demonstrated a possibility to reduce the maximum reactivity margin of the reactor by the value of less than β_{eff} ~0/37% ΔK/K for compensation of calculated and technological errors by various technical measures. Due to the practical impossibility of instant removal of all CPS rods from a reactor core, the value β_{eff} is considered as a reference point. Consequences of a self-feed of all CPS rods (UTOP accident) should be analyzed taking into account all conditions: speeds of rod removal, reactivity feedback, etc.

3. Safety of BN-1200 design

Safety of RI BN-1200 is based upon the experience accumulated during the operation of reactor BN-600 (since 1980), the results of safety analysis and technical safety solutions adopted for the BN-800 reactor that was put into commercial operation in 2016 [2,3]. We shall note the following for the accidents listed in the Section 2.

The accidents with radioactive sodium leakage from the 1st circuit are almost excluded in the BN-1200 design. Unlike reactors BN-600 and BN-800, the 1st circuit is completely integrated in the design BN-1200 – there are no sodium pipelines of auxiliary systems (system of sodium purification and the quality and sodium activity monitoring systems of the 1st circuit). In the BN-1200 design, auxiliary systems are placed into the reactor vessel. As for a sodium leak from the reactor vessel, it is eliminated in all BN-type reactors due to a safety case. The analysis shows that the probability of leakages of both cases is less than 10^{-7} 1/year. Permanent monitoring of the safety case state allows one to reveal depressurization of any of cases and to take the corresponding measures in due time.

The accident with instant blocking of the entire fuel assembly clear opening is a postulated one, snice there is no technical possibility of such an accident. Consequences of this accident for all BN reactors, including BN-1200 reactor, are limited to triggering of emergency protection system against radioactivity increase in the cladding failure detection and location system. At a failure of emergency protection system in this accident, the scale of reactor core damage is limited to melting of emergency fuel assembly. At this, the release of activity from the reactor occurs only due to release of radioactive gas from the reactor that is allowed in the design. Doses of radiation exposure for the population do not exceed the natural background, and, according to regulatory documents, no protection measures for the population reside out of the NPP site are required.

To reduce the probability and scale of consequences of other accidents specified in section 2 for BN-1200 design, a number of new technical solutions were made. They are presented in table 2.

TABLE II: New technical solutions related to enhanced safety of RI BN-1200

Technical safety solutions	BN-600	BN-800	BN-1200
Emergency protection:			
▪ Active	+	+	+
▪ Passive, based upon hydraulic suspended absorber rods	-	+	+
▪ Passive, based upon temperature principle	-	-	+
System of emergency heat removal: ▪ Air exchangers are connected with 2nd circuit	In one loop of second circuit with sodium circulation	In all loops of second circuit with sodium and air circulation	-
▪ Air exchangers are connected with 1st circuit	-	-	+ 4 loops with sodium and air circulation
▪ EHRS activation:			
▪ active	+	+	+
▪ passive	-	-	+

In the BN-1200 design, as well as in the design BN-800, a passive system based upon hydraulic suspended absorber rods is used. In the working conditions, the system supports the rods in a position above the reactor core in the uncoupled state. In case of flow decrease below 50% of the nominal value, the rods fall into the reactor core under their own weight, providing reactor transfer into a subcritical state. Performance of these functions of the system is checked at every reactor shutdown. This system prevents the most dangerous consequences of temperature increase in a reactor core in case of failure of main circulation pump and emergency protection system. Finally, because of MCP switching-off against the set protective signals, this system will be activated upon increase of the sodium temperature in the reactor.

In BN-1200 design, a new passive system of absorber rods based upon the temperature principle is used. The rods fall into the reactor core after temperature increase at the exit of the reactor core, providing reactor transfer into a subcritical state. However, the ability of such system, possessing a universal principle of operation, to perform their functions during the reactor operation cannot be checked. The characteristics of this system can be obtained only in bench conditions, thus it is considered as a tool to manage accidents.

The system of emergency heat removal (SEHR) in the reactor BN-1200 foresees a heat removal to atmosphere through air heat exchangers (AHE) directly from the first circuit using independent heat exchangers (IHE), installed in the reactor tank. Air heat exchangers are connected with IHE by intermediate sodium circuit. Unlike the BN-800 design, in which AHE are connected to the second circuit, the scheme accepted for BN-1200 enhances the reliability of SEHR functioning as it excludes the influence of the second circuit on its work.

SEHR includes four independent channels. In all three circuits of each channel - the first (sodium), intermediate (sodium) and air - the heat removal is carried out by natural circulation of coolants. Each AHE incorporates dampers that have a passive principle of opening, while IHE have spherical valves that open a path of sodium circulation directly through the reactor core under their own weight, if a forced circulation of sodium through the reactor is terminated.

For the beyond design-basis accidents with coolant overheat, from the safety point of view, it is important to use the reactor core design with fuel elements that have an upper sodium plenum. This is also used for BN-800 design. Filling of this plenum with steam at the exit of coolant from reactor core, in the conditions of accident with the reactor core overheat leads to increase in neutrons leak from the top part of the reactor core and, correspondingly, to decrease of reactivity and reactor power (in the situations, when boiling does not extend over the entire height of the reactor core).

The analysis made for beyond design-basis accidents related to heat-removal and reactivity shows that in case of improbable initial events of failures of all active safety systems and individual failures of passive elements of safety systems, depressurization of cladding of a part of fuel elements occurs, there is no fuel melting, the reactor vessel remains intact. The activity released into the environment at operation of a hydro-shutter in the conditions of given accidents, is basically caused by accumulation of inert radioactive gases in the reactor during normal operation. In this case, the doses of radiation exposure for the population do not exceed the natural background radiation exposure that, according to the regulatory documents, does not require any protection measures for the population out of the NPP site. Change of parameters in case of a heat-removal accident is given in fig. 1; more details and basic results of analysis for the given accidents are presented in [4].

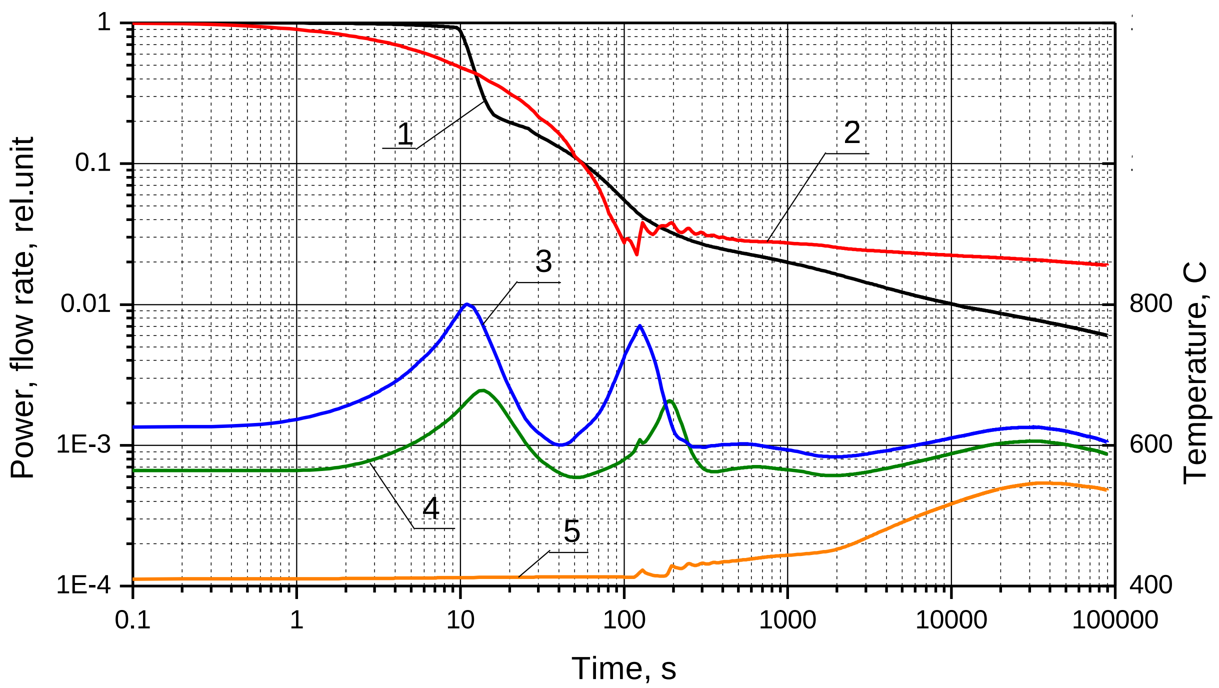

Fig. 1. The basic reactor parameters in case of blackout and failures of all active safety systems and individual failures of passive elements of safety systems

1 – Reactor power; 2 – coolant flow through reactor core; 3 - temperature of cladding of mostly stressed fuel elements; 4 - average temperature of the coolant at exit from fuel assembly; 5 - coolant temperature at entry into reactor core

(Left Y-scale – power, flow in relative units, right Y-scale – temperature in °C, X-scale – time in s)

A preliminary analysis of ULOF and UTOP accidents with failures of all active and passive systems of reactor shutdown has also been done. The probability of these accidents is estimated as less than 10^{-8} 1/year. A part of fuel assemblies in a reactor core is melted in these accidents. Analysis of consequences of these accidents, performed using the SOCRAT-BN and COREMELT codes, shows that, due to the system of emergency release containment foreseen in the design, there is no necessity to evacuate or resettle the population of the nearest settlements in the conditions of these accidents. More details of the severe accident analysis are given in section 5 and the paper [5].

4. Safety of BREST-OD-300 reactor installation

The purpose of development of BREST-OD-300 reactor is to demonstrate the ability to achieve a high level of safety and competitiveness by using a new coolant and dense nitride fuel, as well as to gain experience of closed fuel cycle operation of lead coolant fast reactor within the frameworks of prototype power system [10].

Taking into account a number of general properties typical for reactors with liquid-metal coolant, the basis for BREST-OD-300 development is more than 40 years of practical experience in design and operation of fast sodium reactors and experience of heavy coolant application in nuclear submarine reactors and benches. The concept of BREST technology with its closed fuel cycle was developed in the late 80s - early 90s as a result of search for compromise between two contradictory requirements for the future large-scale nuclear power

(NP), namely, for its safety which excludes radiation accidents exceeding the limit values of radioactive releases for population and environment, and for its effectiveness which ensures the competitiveness of nuclear power with alternative energy sources.

The selected reactor power level N = 300 MW(e) (700 MW(t)) is close to the minimum at which the characteristics of the prototype meet the conditions of complete plutonium breeding in core (BRC ~ 1), operation in an "equilibrium regime" with a small reactivity excess ($\Delta\rho<\beta_{ef}$). The specific characteristics of future commercial BREST reactor are modeled to a considerable extent; and the continuity of many design and technical solutions that will be demonstrated during trial operation of the prototype is ensured.

As noted above, a significant part of the properties of high internal self-protection of lead-cooled reactors is common with the properties of sodium reactors. But there are also the features and differing technical solutions for safety ensuring.

With regard to exclusion of leakage of the radioactive coolant of 1 circuit, the following can be noted. The reactor installation was designed with an integral configuration in a metal-concrete multilayered vessel, i.e. all the equipment of the first circulation circuit is located within the vessel and coolant does not leave the vessel. In case of postulated leak of a boundary of the first circuit, the coolant is localized by concrete filler and, due to the high freezing temperature, do not go out of vessel limits. Such a solution makes negligible the possibility of loss of core cooling due to coolant loss. The calculated justification showed that the selected vessel design ensures a possibility of leak with partial loss of coolant not more than 9.7×10^{-10} 1/year [11]. At the same time, the vessel performs the function of radiation protection providing a natural radiation background on surface.

The project adopted a canless design of fuel assemblies with smooth fuel elements and spacer grids. The use of canless fuel assemblies ensures a higher level of safety in comparison with covered fuel assemblies. Calculations have shown that the postulated shuttering of coolant flow at inlet of 7 canless fuel assemblies in the central part of core does not lead to an increase in temperature of fuel element surface exceeding the acceptance criterion of 800°C. For the canless fuel assemblies, heat is removed by cross-flow of coolant from adjacent fuel assemblies if the coolant flow is blocked at the inlet.

There are no shutoff valves in the first circuit; and it contributes to heat removal in the event of loss of forced circulation and excludes hydraulic shocks possible with false operation or failure of valves. Coolant circulation scheme is organized with difference of free levels; this solution ensures the continuation of circulation under de-energizing. A large volume of lead coolant in the circuit (~ 1000 m^3) helps to reduce the rate of temperature rise in case of violations of normal operation.

A positive property of lead coolant is its chemical inertness (incombustibility) with respect to water and air. This property allows the possibility to take away the heat directly from the lead circuit by placing air heat exchangers in it with natural air circulation, which increases the reliability of the system as its branching decreases. Due to the chemical inertness of lead, a two-circuit system of heat transfer to turbine with the placement of steam generators inside the vessel is applied, which allows reducing the costs for the main equipment. To ensure safety when the pipes leak, a localization system was developed which prevents overpressure in the vessel even when several pipes start a leak simultaneously.

A part of side reflector blocks adjoining the active zone is made in the form of vertical channels with lead columns closed on top and open below (like a gas bell). The level of lead columns monitors pressure head (flow rate) of the coolant and affects neutron leakage. With the help of these devices - the channels of passive feedback system (PFS), reactivity (and power) of reactor passively communicates with pressure head (flow rate) of the coolant through reactor core, which is an important factor in reactor safety and control. The passive feedback system is designed to compensate the reactivity effects by passive method when the lead coolant flow rate in the primary circuit changes in energy range of RI power variation (30-100) % of rated power in order to maintain the reactivity excess compensated by CPS rods <$ß_{эф}$. PFS includes 20 independent channels.

Safety analysis of BREST-OD-300 reactor [12] showed that, when the initial events of violations of normal operation with superposition of postulated multiple failures of systems and equipment or personnel errors take place, there is no violation of the limits of safe operation of power unit. Two classes of the most conservative initial events leading to violation of heat removal from core and initial events leading to unauthorized insertion of design reactivity excess are considered. It is shown that partial damage of fuel elements which have a large burnup depth is possible even in case of the worst scenarios. However, melting of the claddings of fuel elements and fuel does not occur; and the integrity of circulation circuit is ensured. Due to the high boiling point of lead (T_b>1700°C), void effect of reactivity related to boiling of the coolant does not take place in core. Release of fission products from the RI for the first day is not more than 6.14 × 10^{10} Bq (does not exceed the control level of releases per day during normal operation). The probability of implementing such a scenario is 2.8 × 10^{-9}. Fig. 2 shows the change of parameters in a heat removal accident.

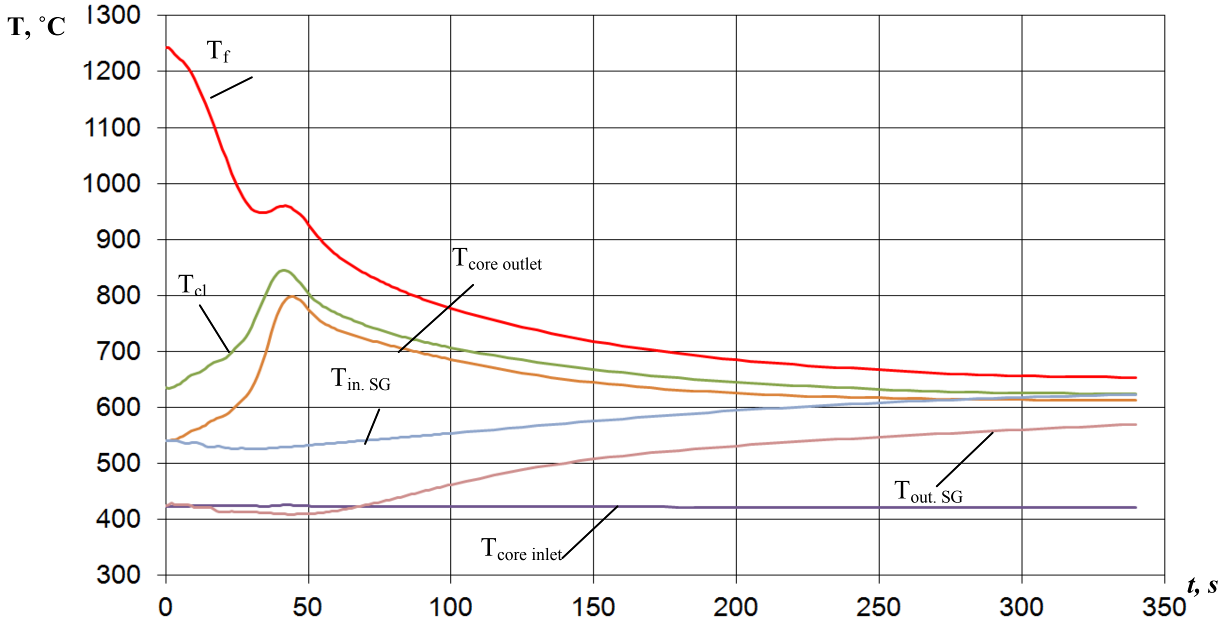

Fig. 2. Diagram of transient process under blackout with failure of all shutdown systems. Temperature of fuel, fuel element cladding, lead coolant at the inlet and outlet of core and steam generator are shown.

Additional information on severe accidents is given in the Section 5.

5. Calculation codes of new generation for safety justification

Computational safety justification of the projects with sodium and lead coolants is carried out using calculation codes that allow modeling the main processes and phenomena in RI and power unit. For a sodium coolant, the calculation codes and experimental data for their validation are available because of sufficiently high level of technology development. In addition, these codes are successfully used in justification of BN-600 and BN-800 projects, which makes it possible to speak with more confidence about the applicability of these codes to the RI with BN-1200 [6]. Among the codes developed within the framework of the "New Generation Codes" project, special attention is paid to SOCRAT-BN [5] code for analysis of emergency processes, including severe accidents with core melting.

At the same time, the availability of calculation codes and experimental data for lead coolant is much lower. The lack of reactor installations-prototypes also does not allow full-scale validation of calculation codes for heavy metal coolant. In these conditions, development of a new generation code system is of particular importance. For these codes, special attention is paid to multi-physical modeling using modern software tools based on adequate physical models and modern methods (CFD, Monte Carlo, etc.), including high-performance computing systems [8], This allows increasing the predictive ability of codes when verifying them on less large-scale experiments. A special emphasis in the development is made on universality of calculation tools with respect to heat carrier, which significantly expands the verification base by experiments on model liquids (for example, using the Rose's alloy). In addition, development and use of precision codes can reduce the deficit of experimental data needed for projects justification.

Such universal software used to analyze the safety of LMC installations is the following calculation codes: HYDRA-IBRAE/LM, BERKUT, hydrodynamic calculation codes and codes describing the migration of fission products in the atmosphere, water and geological rocks. HYDRA-IBRAE/LM is the system thermal-hydraulic calculation code for liquid-metal coolants; BERKUT is the code for mechanistic analysis of fuel behavior, both oxide and nitride in steel claddings of fuel elements. It should also be noted the integral calculation code of EUCLID intended for system analysis of normal and emergency modes of operation of RI with LMC.

General information on the code system being developed is given in the paper [9], and detailed description of the calculation codes is given in the specific messages for each code.

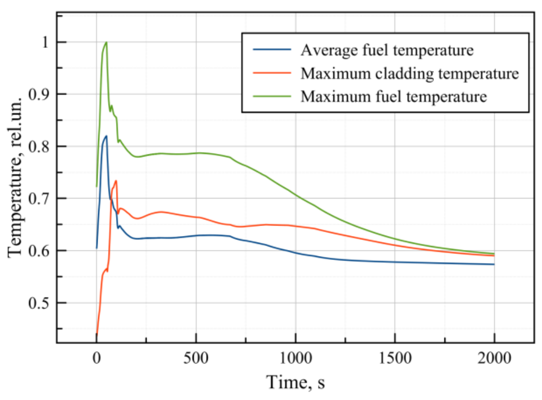

Fig. 3. Insertion of total excess of reactivity at RI BREST-OD-300

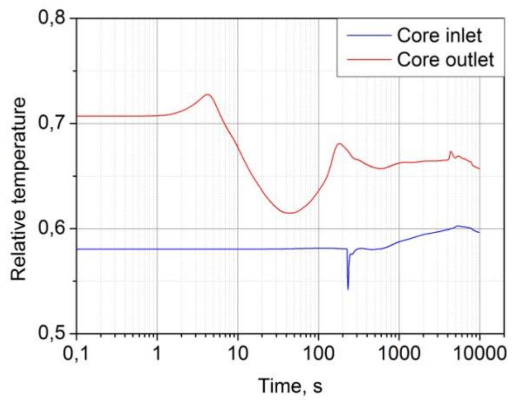

Fig. 4. Scenario for loss of system power supply of RI BN-1200

A number of operation modes and design basis accidents for reactor installations with sodium and lead coolants have been calculated using EUCLID calculation code. As an example, Fig. 3 shows the results of calculating an accident accompanied by insertion of total excess of reactivity for RI BREST-OD-300 without triggering of emergency protection. It is shown that the core materials during the accident do not melt.

Fig. 4 shows the results of calculations of emergency mode of RU BN-1200 for the scenario of loss of system power supply. It follows from the results of calculation that exceeding of the cladding temperature limits of safe operation does not occur.

An analysis of possible radiation consequences of a number of highly unlikely hypothetical severe ULOF, UTOP and TIB accidents for RU BN-1200 was carried out using the SOCRAT-BN calculation code.

Thus, it is assumed in the ULOF accident that the system and reliable power supply of RI will be completely lost with simultaneous failure of all passive and active protection systems (CPS). For the UTOP accident, uncontrolled withdrawal of shim rods (SR) rods and control rods (CR) with additional failures of passive and active elements of emergency protection is postulated. For the TIB accident, the possible consequences of complete blockage of cross-section of one assembly with failures of cladding failure detection system and active elements of emergency protection are investigated. It should be noted again the extremely low probability of the above-mentioned accidents because it takes a large number of independent simultaneous failures of the elements of reactor safety systems (more than 15 elements according to preliminary estimates).

Fig. 5, 6 show the preliminary results of analysis of the UTOP accident using SOCRAT-BN integrated code. For this scenario, the calculations forecast increase of pressure in gas plenum of reactor resulting in short actuation of hydroseal and release of a part of volatile and gaseous fission products through a special ventilation system into the environment.

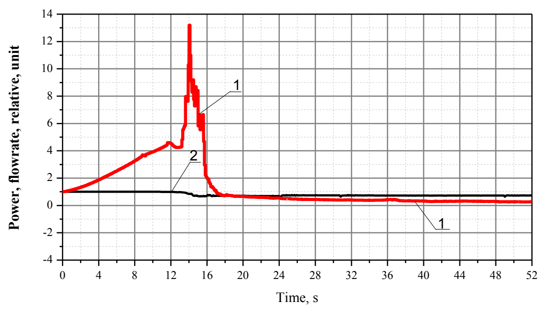

Fig. 5. Relative power of the reactor (1) and flow rate of coolant through the core. (2)

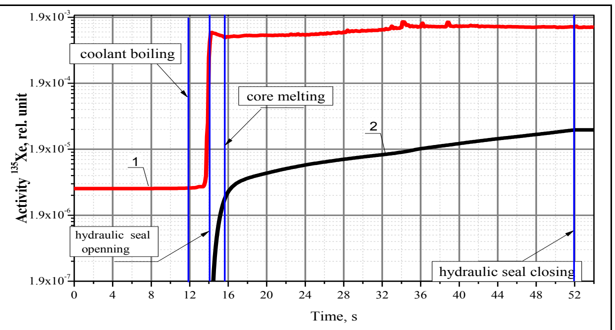

Fig. 6. Relative activity of Xe in gas plenum of the reactor (1) and integral release into the environment (2)

Figures 5 and 6 show the relative power of energy release, the activity of Xe135 isotope in the gas plenum of the reactor and integrated release of Xe135 isotope into the environment calculated by codes of time dependence. It was confirmed by calculation that despite the predicted melting of a part of RI core, only a small fraction of accumulated activity is released into the environment. An analysis of radiation situation showed that no measures will be taken to protect the population outside the site in this case.

6. Conclusion

The paper shows that power units with reactor installations BN-1200 and BREST-OD-300 designed within the framework of the FTP NETNG (Federal Target Program “Nuclear Energy Technologies of the New Generation”) fully comply with the safety requirements for the Generation IV installations. Based on the results of the analysis performed so far, the requirement in the "PRORYV" project to exclude the need for evacuation of the population in the nearest settlements under any possible accidents at the power units is ensured.

For the well-formed detailed description of the processes occurring in the RI, including in case of accidents, the development of new generation codes is under way.

References

1 Adamov E.O., Aleksakhin R.M., Bolshov L.A., Dedul A.V., Orlov V.V., Pershukov V.A., Rachkov V.I., Tolstoukhov D.A., Troyanov V.M. The “PRORYV” Project is a technological foundation for large-scale nuclear power engineering, Izvestiya RAN. Energetika, No. 1, p. 5-12, 2015.

2 I.A. Kuznetsov, V.M. Poplavsky. Safety of NPP with fast neutron reactors. Under the general editorship of Corresponding Member of the Russian Academy of Sciences V.I. Rachkov. Moscow, Publishing House, 2012.

3 Realization of the natural safety principle in the BN-1200 project. B.A. Vasiliev, V.I. Evseev, N.G. Kuzavkov, S.L. Osipov, S.F. Shepelev. Safety of Nuclear Technology and the Environment No. 1 (25), p. 62-65, 2012.

4 Anfimov, A., Evseev, V., Osipov, S., Rogozhkin, S., Shepelev, S. “Safety Assurance for BN-1200 Power Unit During Accidents” // Paper for this conference

5 Rtishchev, N.; Chalyy, R. “SOCRAT-BN integral code for safety analysis of NPP with sodium cooled fast reactors: development and plant applications” (Int. Conf. on Fast Reactors and Related Fuel Cycles (FR17)), Paper CN- 281, Ekaterinburg, 2017.

6 Shepelev S.F., Osipov S.L., Marova E.V., Poplavsky V.M., Ashurko Yu.M., Strizhov V.F., Mosunova N.A., Calculation platform for the project RI BN-1200 justification. IV International Scientific and Technical Conference. IV ISTC NIKIET. Collection of papers, V.2, P.47-53, Moscow, Publishing house JSC "NIKIET", 2016.

7 Y.M. Ashurko, A.V. Volkov, K.F. Raskach. COREMELT2D Code for Analysis of Severe Accidents in Sodium Fast Reactor. In: Intern. Conf. on Fast Reactor and Fuel Cycle: Safe Technologies and Sustainable Scenarios (FR13), Paris, France, 2013 – IAEA, 2013, T1-CN-199/453, track 03.

8 Bolshov, L. A., Mosunova, N. A., Strizhov, V. F.; Shmidt, O. V. Next Generation Design Codes for a New Technological Platform for Nuclear Power. Atomic Energy, 2016, V. 120, № 6, P: 369-379.

9 L.A. Bolshov, V.F. Strizhov, N.A. Mosunova. Codes of New Generation Developed for the “Proryv” Project. (Int. Conf. on Fast Reactors and Related Fuel Cycles (FR17)), Paper CN-561, Ekaterinburg, 2017.

10 Dragunov Yu.G., Lemehov V.V., Moiseev A.V., Smirnov V.S., Yarmolenko O.A., Vasyukhno V.P., Cherepnin Yu.S. Technical project RI BREST-OD-300: stages of development and justification. IV ISTC NIKIET. Collection of papers, V.1, P.21-30, Moscow, Publishing house JSC "NIKIET", 2016.

11 Bazhanov A.A., Zhuravleva I.V., Ivochkin M.Yu., Malysheva E.Yu., Shiversky E.A. Probabilistic analysis of the NPP safety with RI BREST-OD-300. Problems of Nuclear Science and Engineering. Series: NPP safety assurance. Issue. 36, p. 13-18., M., 2016.

12 Lememov V.V., Moiseev A.V., Kogut V.A., Muratov A.G. and others. Deterministic analysis of the RI BREST-OD-300 safety. Problems of Nuclear Science and Engineering. Series: NPP safety assurance. Issue. 36, p. 5-12., M., 2016.