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Concept of Small Power Reactor Installation without Refueling during Lifetime (SVBR-75/100)

Principles of Providing Passive Safety and Passive Safety Characteristics of SVBR Type RIs for Small Power Nuclear Plants

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CONTENTS

Nomenclature

AEP	– additional emergency protection
APE	– annual permissible exhaust
CPS	– control and protection system
CR	– compensating rod
CRP	– coordinated research project
EP	– emergency protection
FR	– fast reactor
FSA	– fuel sub-assemblies
GFP	– gas fission products
GS	– gas system
LBC	– lead-bismuth coolant
LOCA	– loss of coolant accident
LOHS	– loss of heat sink
MCC	– main circulating circuit
MCP	– main circulation pump
MSP	– main steam pipeline
NPP	– nuclear power plants
OCR	– operating compensating rod
OSL	– operating safety limits
PHRS	– passive heat removal system
RC	– reactor compartment
RI	– reactor installation
RMB	– reactor monoblock
RR	– regulating rod
SAC	– system of autonomous launching and cooling-down
SG	– steam-generator
SPNP	– small power nuclear plant
SRWOR	– small reactors without on-site refueling
SVBR	– lead-bismuth fast reactor
TGI	– turbo-generator installation

1. Introduction

In many developing countries and in the certain regions in Russia there are serious deficiencies in electricity, heat or fresh water. It is possible to make up this deficiency by use of nuclear energy sources. The specific requirements, which are not peculiar to the large nuclear power plants (NPP), must be produced to such energy sources:

- as for the capital costs and the cost of production, the small power nuclear plant (SPNP) must be competitive with alternative variants of energy sources that operate by using fossil fuel;
- the electric power of the SPNP must be small and lie within the range 10…100 MWe;
- to construct the different power SPNPs, it is expedient to use a modular principle of designing the NPP power-unit that includes several identical reactor installations operating for one or several turbine installations;
- the reactor installation (RI) must possess the inherent self-protection and passive safety properties which deterministically eliminate a possibility of severe accidents requiring population evacuation beyond the NPP fence in the events of simultaneous multiple equipment failures, superposition of personnel's errors or malevolent actions. At this, the RI safety characteristics must allow location of such NPPs near the population centers;
- the uranium enriched in uranium-235 not more than 20 % must be used as fuel. This complies with IAEA recommendations concerning non-proliferation;

- the RI must be completely factory-fabricated in the Supplier-Country. An opportunity of safe transportation of ready RIs to the User-Country and back must be assured. At this, the RI is a property of the Supplier-Country and is leased to the User-Country for the period determined by the core lifetime;
- the core lifetime of the RI must be not less than 7-10 years.

All the stated requirements are met by a standardizated reactor installation for multi-purpose usage – SVBR-75/100 (**L**ead-**B**ismuth **F**ast **R**eactor of equivalent electric power being within 75 – 100 MWe, depending on the steam parameters [1]) that is being designed by SSC RF IPPE named after A.I. Leypunsky and FSUE EDO “Gidropress” on the basis of unique experience of constructing and operating the reactors with lead-bismuth coolant (LBC) [2]. Currently, RI SVBR-10 with equivalent electric power of 10 MWe is under development.

Today the conditions for implementation of this technology in the civilian nuclear power have been formed. This is conditioned by the fact that the natural properties of lead-bismuth coolant and physical features of fast reactors realized with due account of experience gained for RIs make it possible to construct the RI in which the principle of inherent self-protection against the certain severest accidents has been realized to the most extent, i.e. the causes of arising such accidents have been eliminated.

First of all, this is conditioned by the fact that to obtain the high temperature of liquid-metal coolant does not require to increase pressure in the reactor vessel. Such reactor installations with passive safety properties do not need a great number of safety systems and therefore the construction of the nuclear power complex will be cheapened and its operating will be simplified.

The SVBR-75/100 reactors belong to a class of small reactors without on-site refueling (SRWOR). For this, under the supervision of IAEA the certain countries carry out a coordinated research project (CRP) within the frameworks of which has been approved a neutron-physical benchmark, which computation results will be presented in 2007.

This report presents a concept of providing safety and the results of computations of some severe accidents, which verify an extremely high potential of safety of such type reactors.

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2. Safety providing concept

The concept of providing safety of the RI is based on the following provisions:

- choosing the type of the reactor, primary circuit coolant and design solutions allowing use of inherent self-protection and passive safety properties to the maximal possible extent and deterministically eliminate the severe accidents (such as LOCA, LOHS etc.);
- heightening the reliability and safety of the RI due to considerable reduction a number of safety systems and their simplification and assigning the safety functions to the normal operating systems.

The developed RIs with LBC (SVBR-75/100 and SVBR-10) use fast neutron reactors (FR). Such features of the FR as lack of poisoning effects, low value of negative temperature reactivity coefficient, compensation for fuel burn-up processes by plutonium build up along with a special algorithm of controlling the control and protection system (CPS) rods provide the operative reactivity margin in the operating reactor to be less than the delayed neutron fraction. Therefore, a possibility of prompt neutron reactor runaway has been eliminated.

A special algorithm of controlling the CPS rods presumes that in the beginning of the lifetime all compensating rods (CR) are immersed into the core and are surely switched off from the automatic control system. Operating at power is performed during the micro-lifetimes which duration is about five-seven months each. Maintenance of reactor criticality and power control during the micro-lifetime are realized with the help of regulating rods (RR) and operating compensating rods (OCR), which reactivity margin is essentially less than 1 $. In the end of the micro-lifetime the value of the current reactivity margin equals to zero and its restitution is realized by re-compensation.

Elimination of possibility of prompt neutron reactor runaway at launching is achieved due to limiting the velocity of movement of the absorbing rods [3], which are extracted in turn, and use of the CPS rods which effectiveness is much less than 1 $. At this, an unauthorizedly inserted positive reactivity has time to be compensated by feedbacks without dangerous increase of the core temperature.

Option for LBC in the primary circuit of the considered RIs is conditioned by natural properties of this coolant, namely:

- the high boiling point (~1670°C) and therefore impossibility of LBC boiling heightens the reliability of heat transfer from the core and safety due to lack of a heat removal crisis phenomenon. Moreover, there is no need to maintain high pressure in the primary circuit. These all result in simplification of the RI design, heightening of its reliability, practical elimination of the possibility of over-pressurization of the primary circuit and thermal explosion of the reactor in an event of accidental overheating of coolant. In addition, low pressure in the primary circuit reduces the risk of its tightness failure and enables to lessen the thickness of reactor vessel walls and reduce the limitations for the rate of changing the temperature by the terms of thermo-cyclic strength;
- LBC reacts very slightly with water and air. Progress of accidental processes caused by tightness failure in the primary circuit and inter-circuit leaks in the steam-generator (SG) occurs without release of hydrogen and any exothermic reactions. Moreover, within the core and RI there are no materials releasing hydrogen as a result of thermal and radiation effects and chemical reactions with coolant. Therefore, the likelihood of chemical explosions and fires is virtually eliminated.

The basic RI systems which provide safety functions are as follows [1]: a system of reactor shutdown and maintaining the reactor in a sub-critical state; a system of autonomous launching (without connection with turbo-generator installation (TGI)) and cooling (SAC) that provides a function of emergency cooling of the reactor; a passive heat removal system (PHRS) that removes heat decay from the reactor and performs reactor cooling and localizing system's functions; primary circuit's gas system that performs a function of protection and localizing system in an event of inter-circuit leak in SG modules.

2.1 Shutdown reactor system

A system of reactor shutdown and its maintenance in a sub-critical state consists of three sub-systems of converting the reactor into a sub-critical state, namely:

- an emergency protection (EP) sub-system;
- an additional emergency protection (AEP) sub-system;
- a sub-system of OCRs group.

An emergency protection sub-system is a protective safety system and is provided for safe and reliable conversion of the reactor core into a sub-critical state in all modes of RI operating, namely, automatically – by emergency signals and automatedly – by a command of the operator. The system includes 6 EP rods in the dry channels [1]. The rods are equipped with springs and electromagnetic clutches. The rods are inserted into the core at the signals of the control system or in an event of de-energizing.

In order to heighten the RI safety, the RI design provides the AEP sub-system that assures passive actuation of the 12 AEP rods due to actuation of fusible locks when the coolant temperature reaches the dangerous values at the reactor outlet [1].

In an event of de-energizing the RI, the emergency protection function is also performed by a system consisting of 13 OCRs, which executive mechanisms have springs providing drop of the rods when electromagnetic clutches are de-energized [1].

2.2 The system of autonomous launching and cooling-down (SAC)

The RI structure includes two cooling channels, each is capable to remove ~3 % of nominal power of the reactor. Each channel consists of a cooling condenser connected with a separator and cooled by technical water and a pipeline of discharging a condensate with a valve-regulator of direct action that is opening when the pressure in the separator increases over the nominal one.

In normal operating conditions the autonomous cooling circuit is used in the modes of launching and cooling-down.

In the waiting mode the condensers are flooded with water and there are virtually no heat losses. When due to some reason the pressure increases up to a certain value, the valves are opening and the condensate is discharged into the separator. In this way a heat-exchanging surface is opened and steam begins to condense unless the steam pressure is lowered up to a specified level.

2.3 Passive heat removal system

As the PHRS tank with water is a component of the RI, there is an opportunity to remove heat decay from the core via a wall of the reactor monoblock (RMB) vessel. In an event of failure of all RI systems, the PHRS tank assures passive removal of heat decay from the RMB vessel, which is realized due to evaporation of water from the tank when water boils and steam removal via the air vents to the atmosphere. There is enough water in the tank for cooling the reactor during 4 days without damage of the core.

The PHRS and safeguard casing also eliminate the impermissible LBC losses by the core cooling conditions in an event of a postulated accident being beyond the design basis with failure of the monoblock vessel's tightness. In this case the LBC leak will be localized within the safeguard casing and further the leaking LBC will freeze while contacting with the safeguard casing's walls cooled by water.

The volume of the space between the basic monoblock vessel and safeguard casing determines a maximal possible loss of LBC in an event of failure of the basic vessel's tightness. In an event of such LBC loss, the LBC level in the monoblock does not drop lower than the limiting value that assures maintenance of the conditions for natural circulation of LBC in the monoblock.

So, the PHRS performs both protection and localizing safety functions.

2.4 Primary circuit gas system

The primary circuit gas system performs a function of protection and localizing safety system by removing steam from the primary circuit's cover gas to the gas system's condenser and then to the reservoir for condensate intake in an event of inter-circuit leak in the SG modules. The system of removing steam from the gas system is designed for guillotine rupture of a single tube of the evaporator module. At the same time the gas system is also a technical tool for overcoming an accident being beyond the design basis, which is caused by a postulated guillotine rupture of several tubes of the evaporator module.

When localization of the SG leak is performed, one of the basic elements is a LBC circulation circuit with coolant streams up-going to the free coolant's levels. This provides the reliable separation of steam-water mixture and prevents the drag of steam to the core by a descending primary circuit's coolant stream.

In events of small SG leaks (up to full damage of a single SG tube), two gas system's condensers cooled by technical water are used. At this, their efficiency enables to maintain pressure in the gas system of the RI within 0.5 MPa.

To withstand the large SG leaks (postulated damage of several SG tubes – an accident that is beyond the design basis), the gas system is connected with the PHRS tank. This line is overlapped by a membrane-protection device designed for the rupture at the gas system pressure of 1 MPa that is not dangerous for the RMB vessel [1]. At this, the volatile radionuclides of the cover gas will remain in the tank water but the radioactive non-condensing gases will escape via a filtrate system to the atmosphere. Radioactivity release will not exceed the permissible level.

3. The results of safety analysis

The results of the analysis of the most dangerous pre-accidental situations and beyond the design basis accidents, which have been obtained in the process of proving safety of RI SVBR-75/100 are presented below [4]:

- unauthorized extraction of the OCR or RR from the reactor core while operating at a power level and at launching (including failure of the emergency protection and shutdown of the main circulation pump (MCP));
- inter-circuit leak in the SG evaporator modules (guillotine rupture of one, two or more tubes);
- core cooling in an event of rupture of a steam pipeline;
- tightness failure in the primary circuit gas system;
- partial blocking of the flow area at the reactor core inlet;
- coolant leak in the reactor monoblock primary circuit;
- blacking out of the RI;
- damage of the power-unit building, blacking out and failure of tightness in the primary circuit gas system (GS) with direct contact of the alloy surface with an air atmosphere within a single monoblock.

3.1. The analysis of the pre-accidental situations

3.1.1 Unauthorized extraction of the OCR or RR from the reactor core while operating at a power level

To analyze the accident that is within the design basis, one should consider as an initial event the unauthorized extraction of the heaviest rod of the CPS (RR) in compliance with an algorithm of safe control. While operating at power levels, this results in increase of power and EP actuation by a signal corresponding to exceeding the specified power by the real one or by a signal corresponding to exceeding the specified accidental setpoint by the coolant temperature at the core outlet.

In the computation the initial event of the accidental situation is accepted as follows. At 5 s of the calculated time one begins the unauthorized extraction of the RR at a rate of 100 mm/s. The setpoint for EP actuation by a signal corresponding to exceeding the specified power by the real one has been accepted as follows: $N_{real} \geq 1.13\ N_{specified} + 0.07$.

In ~ 5 s after beginning of extracting the RR, the neutron power achieves the accidental setpoint value and the reactor EP actuates. A maximal value of neutron power in the transient process is 1.267 N_n, a maximal value of coolant temperature at the core outlet does not exceed 487 °C. Change of the RI parameters in the accidental situation is within the design basis limits.

3.1.2 Inter-circuit leak in the SG modules

An initial event with failure of a single tube of the SG evaporator module was studied from the standpoint of probable reactivity disturbance in an event of ingress of steam-water mixture into the core. To eliminate over-pressurization of the reactor vessel in events of small SG leaks (up to guillotine rupture of a single SG tube), two gas system's condensers cooled by technical water are used.

Ingress of steam-water mixture into the core has been eliminated by a circulation scheme adopted for the main circulating circuit (MCC). In fact, in an event of arising leak, the water contacts with primary circuit coolant and evaporates (the minimal temperature of the primary circuit coolant $t \geq 200$ °C while the saturation temperature for maximal pressure of ~0,5 MPa in the inter-tube SG gap is $t_{saturation}$~150 °C).

Simultaneously with pressure increase in the gas system a EP signal is produced and the RI will be shutdown. After the MCP has been shutdown, capture of steam in LBC is eliminated as the LBC velocity at natural circulation is much less than 0,5 m/s.

The conclusion of the mentioned above is as follows: coolant capture of steam-water mixture on the MCC from the free level of LBC in the buffer chamber to the core is virtually eliminated.

Along with this, there have been performed the computations on reactivity change in an event of postulated ingress of saturated steam into the core under pressure of ~ 1 MPa. It was presumed that steam with a corresponding density (~ 0,005 g/cm^3) had been homogeneously stirred in LBC. Change of reactivity for the coolant's steam volumetric fraction of ~ 20 % was calculated.

It has been revealed that in an event of ingress of such steam quantity into the core, the positive reactivity is ~ 0,09 $ that is managed easily by the control system or EP system.

3.1.3 Core over-cooling in an event of steam pipeline rupture

In an event of rupture of any of steam pipelines, the EP actuates and the RI is changed over into a cooling-down mode. Effectiveness of the EP rods assures sub-criticality of the reactor at any temperature.

The main steam pipeline (MSP) is installed outside the reactor box. In an event of its rupture, the gate valves, which cut off the RI from the MSP, are actuating and the RI is changed over into cooling-down mode with the help of the SAC. The number of the gate valves mounted in turn from a separator to the MSP is 2 and in events of the accident situations being within the design basis, localization of leaks is realized by shutting down the gate valve. The RI has two steam-generators with separators, which are completely autonomous for the secondary circuit. "Freezing" of LBC in the SG modules has been eliminated by a relatively low temperature of melting the eutectic Pb-Bi alloy (~125 °C) and by option of feeding water parameters.

The rupture of the pipeline on the section from a separator to a first gate valve results in steam leak into the RI box. The accidental SG is cut off from the MSP and the second SG. In such accidental situation being within the design basis the RI is cooled-down via the second SG.

3.1.4 Tightness failure in the primary circuit gas system

When performing the computation assessments of radiation consequences of the accident in which the initial event was vital failure of tightness in the primary circuit GS, the following conditions of accident's progressing were considered: before the accident the RI had been long operating at power near the nominal one; as for the number of fuel elements and their damage, the state of the RI fuel elements was near the operating safety limits (OSL), i.e. gas leaks in 63 fuel elements and 7 leaky fuel elements with contact between coolant and fuel; it was presumed that in an event of arising large leak in the GS the reactor was shutdown immediately; after the accident had happened, the conditions of air ventilation in the reactor compartment (RC) and the regime of its access were the same (as in the normal operating conditions); the repetition factor of air ventilation in the RC was 3 h^{-1}; after the accident the relative rate of leaking the cover gas was accepted to be 2 h^{-1}; the relative rate of releasing gas fission products (GFP) from coolant to cover gas was accepted to be $1 \cdot 10^{-5}$ s^{-1}.

The results of the computation assessments of radiation consequences of the GS accident at the fuel elements' state being in accordance with OSL are as follows:

- during the first hour of the accident in the reactor compartment a volumetric activity of GFP in the air can achieve 0.55 Ci/m^3. During the first hour of staying in the RC the individual irradiation dose of radioactive gases released in the RC air as a result of the accident can achieve 4.5 mSv;
- during the 24 hours of the accident the GFP activity in the exhaust to the atmosphere is assessed as 655 Ci that is much less than $1.86 \cdot 10^{+4}$ Ci – the annual permissible exhaust (APE) [5];

- during the 24 hours from the moment of GS damage the exhaust of iodine and long-lived cesium radionuclides into the atmosphere (with ventilation of the RC, with filtration with 99.9 % of purification efficiency) will not exceed $1.36\cdot10^{+7}$ Bq or ≈ 0.37 mCi. This value is nearly by two orders less than a control level of the month exhaust of VVER-1000 ($1.5\cdot10^{+9}$ Bq) [5]. At this, the cumulative exhaust of iodine radionuclides in the accident is ≤ $1.8\cdot10^{+5}$ Bq that is 10^5 times less than the annual permissible exhaust of iodine radionuclides (APE $\Sigma I = 1.8\cdot10^{+10}$ Bq). Activity of Cs^{134} in the accidental exhaust is ≤ $7\cdot10^{+6}$ Bq that is 100 times less than the APR of this radionuclide ($9\cdot10^{+8}$ Bq), and activity of Cs^{137} is ≤ $6.4\cdot10^{+6}$ Bq that is more than 100 times less than APE ($2\cdot10^{+9}$ Bq). During the 24 hours from the moment of the accident the activity of Po-210 in the ventilating exhaust can achieve $1.1\cdot10^{+5}$ Bq;
- an individual dose of GFP irradiating the personnel in the RC air and that caused by inhalation of radioactive aerosols during the first hour of accident progressing (respiratory organs' protection means are not used) will not exceed 30 mSv (3 rem). Moreover, in the RC the contribution of the external irradiation from the GFP to this dose will be 4,5 mSv. The contribution of the polonium-210 aerosols to the total dose of accidental irradiation can be ~ 20 mSv. During the first hour of accident progressing the inhalation of air with iodine and cesium radionuclide aerosols in the RC is equivalent to ~ 5 mSv of irradiation;
- the volumetric activities of iodine, cesium and polonium radionuclides in the layer of the air near the ground with due account of exhaust filtration in the ventilation system will be much less than the values of annual volumetric activities of these radionuclides that is permissible for the population. I.e. the accidental situation does not practically deteriorate the radio-ecological situation beyond the NPP fence.

3.2. Analysis of the accidents being beyond the design basis

3.2.1 Unauthorized release of the operative reactivity margin

The computation analysis of the installation's parameters has been performed in an event of unauthorized extraction of the whole RR under operating the installation at nominal power without EP actuation and MCP shutdown.

The used algorithm of safe control makes it possible to perform minimization of the operative reactivity margin while operating at a nominal power level. In order to analyze the accidental situation, it has been presumed that a regulating rod releases positive reactivity of ~0.25 $ in case of full lifting to the upper end. An initial event for an accidental situation is lift of the regulating rod with a velocity of 100 mm/s (maximal velocity of lifting). The results of computations are presented in Fig. 1.

It has been presumed in computations that an event of releasing the reactivity margin of 0.25 $ is a cause of postulated failure in the system of automatic control of neutron power of the reactor but does not result in actuation of the EP of the RI. In the process of progressing the accidental situation, power is raised to the level of 1.51 N_{nom} and is stabilized at that level in the mode of power self-regulation due to influence of negative temperature reactivity feedbacks.

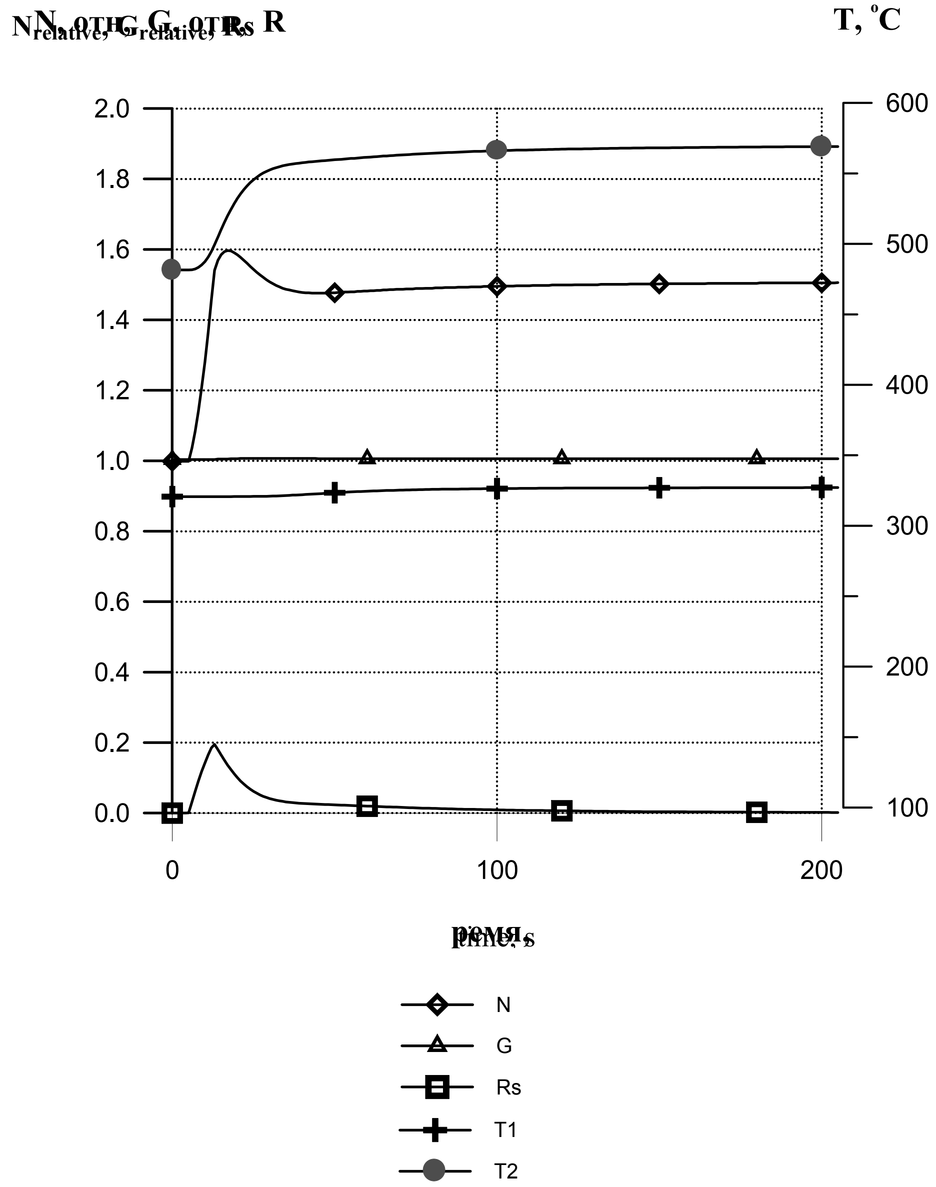
The performed computations has revealed that in an event of releasing the reactivity margin of 0.25 $ at nominal power without EP actuation, the coolant temperature in the primary circuit increases to 569 °C, and the maximal temperature of the cladding increases to 590 °C (without accounting the factors of over-heating) and that is much less than short-time allowed temperatures for the primary circuit's equipment materials.

3.2.2 Partial blocking of the flow area at the reactor core inlet

At the existing proved technologies and methods of purifying the LBC from additive agents and foreign bodies, the initiation of blockages in the fuel sub-assemblies (FSA) or in the part of the core is a very unlikely event. Nevertheless, their influence on thermo-technical reliability of the core should be estimated.

The purpose of the computations was investigation into the stationary hydro-dynamic and thermal processes in the core which follow partial blocking of the core flow area and estimation of influence of these processes on the core thermo-hydraulics.

The computations of the velocity and temperature differences distributions were performed for a nominal mode and for three simulated modes, namely: a mode with blocking the central FSA at the inlet, a mode with blocking a half of the flow area of the core at the inlet, a mode with blocking the central FSA at the level of fuel section's beginning. The obtained distributions of LBC velocities and temperature differences for the mode with blocking a half of the core flow area at the inlet are shown in Fig. 2.



N – reactor power, relative units;
G – flow rate of LBC via the core, relative units;
Rs – reactivity, in fractions of $;
T_1 – LBC temperature at the core inlet, °C;
T_2 – LBC temperature at the core outlet, °C.

FIG. 1. Computation change of RI parameters
when extracting the RR at nominal power without EP rods actuation

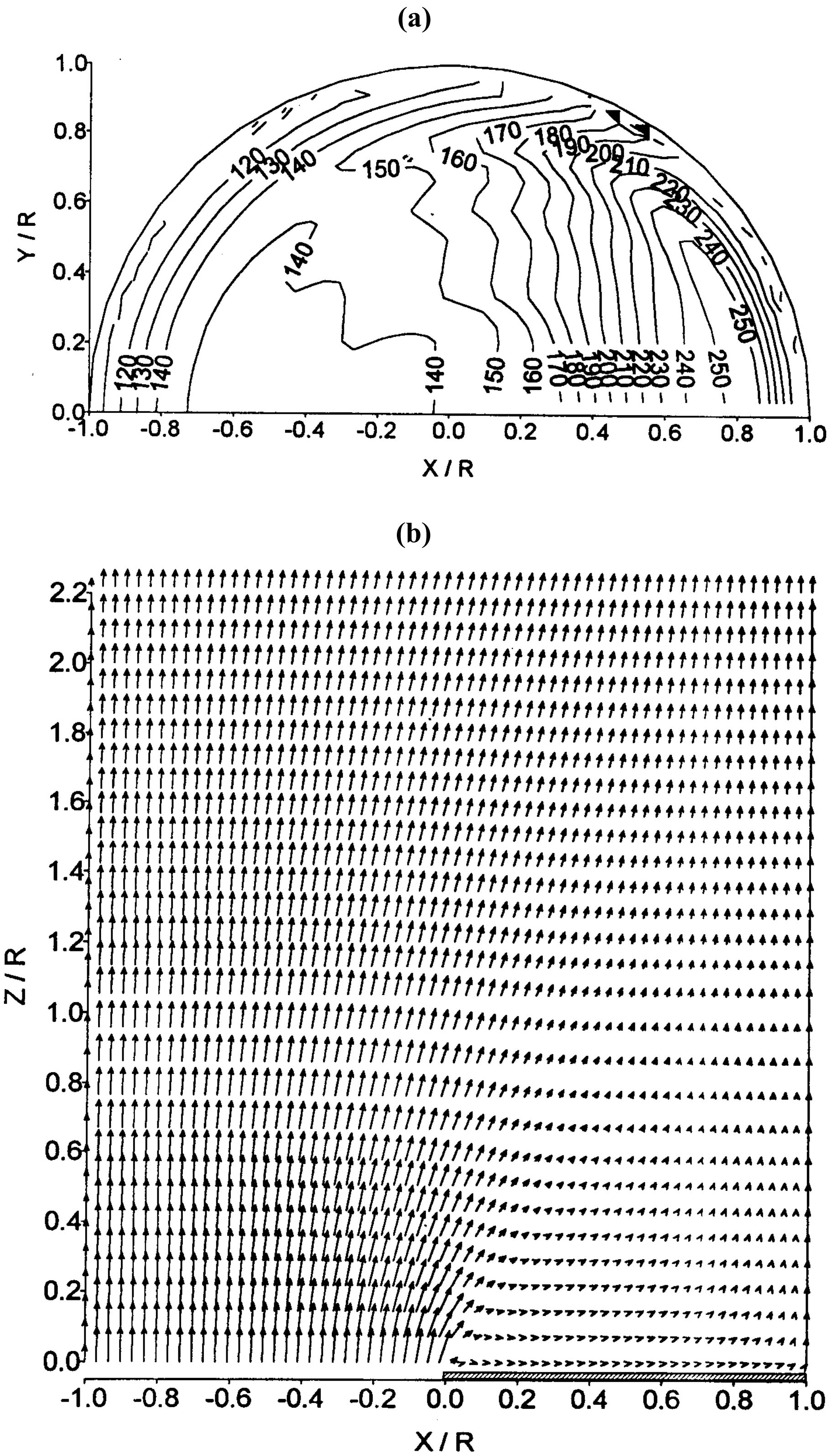


FIG. 2. LBC temperature differences, °C, in the outlet section of the core (a) and velocity distribution in the vertical section (b) (R – the radius of the core)

In all considered modes the absolute maximum of LBC temperature is achieved near the outlet section of the core. The absolute maximum of the fuel elements claddings' temperature is achieved there too. The characteristic values of temperatures at the core outlet are presented in Table 1.

Table 1. The characteristic values of coolant heating and temperatures at the core outlet at an inlet temperature of LBC being 320 ^oC

Mode	$\Delta T_{f,cp}$	$\Delta T_{f,min}$	$\Delta T_{f,max}$	$T_{f,max}$	$T_{w,max}$
Nominal	164	135	186	506	520
FSA blocking at the inlet	164	135	186	503	520
Blocking a half of the core at the inlet	164	107	258	578	592
FSA blocking at the height of ~H/2	164	134	263	583	596

($\Delta T_{f,av}$, $\Delta T_{f,min}$ and $\Delta T_{f,max}$ – average, minimal and maximal values of coolant heating in the cross section of the core, $T_{f,max}$ – maximal value of the coolant temperature in the core, $T_{w,max}$ - maximal value of the temperature of the fuel element's cladding in the core; all values are in ^oC).

The obtained results show very high thermo-technical reliability of the core that is a result of without casing structure of FSA, high transverse thermo-mass-exchange and "shift" of the active part of the fuel element to the upper part of the core.

3.2.3 Blacking out of the RI

In the computations of the situation with blacking out it was presumed that the heat removed from the RMB vessel with the help of the water of the PHRS tank. At this, the water (110 tons) is heated with further boiling and generation of steam. The steel structures of 12 heat-exchangers with total mass of 16 tons installed in the PHRS tank and the PHRS tank of 62 tons are heated as well. At this, it was taken into account that the PHRS heat-exchangers perform additional removal of heat from water that equals to 100 kW under the temperature of 99.6 ^oC due to natural convection of the inter-circuit water contained in the heat-exchangers tubes.

The heat from the upper part of the monoblock vessel that is located in the air (or steam) is taken by the wall of the PHRS tank. From the PHRS tank wall via the air gap of 5 mm in width the heat is transferred by air convection and irradiation to the monoblock well concrete which width is ~1 m in the side part.

In the computations of the blacking out situation it was presumed that in the separators existed a protective tearing membrane via which the generated steam of the secondary circuit was dumped into the atmosphere if pressure exceeded 11 MPa. At this, a flow rate of dumping steam was determined by critical outflow proportional to the steam pressure. Moreover, the following was accounted in the computations:

- emergency cooling of the reactor with using SAC was not actuated;
- supply of feeding water to the steam-generators was terminated;
- LBC circulation in the primary circuit is realized by natural convection;
- supply of the cooling water to the PHRS heat-exchangers was terminated.

In the computations of the blacking out situation it was presumed that at the moment of de-energizing ($\tau = 0$) the simultaneous insertion of 5 EP rods (out of 6 EP rods due to failure of the single EP rod) and 12 OCR rods was beginning. Complete insertion of the EP rods into the core is realized during 1 second, complete insertion of the OCR rods into the core is realized during 3 seconds. The results of computation of the final stage of the accidental situation are presented in Fig. 3.

Evaporation of water in the PHRS tank is completed at the 4-th day from the moment of happening the accidental situation.

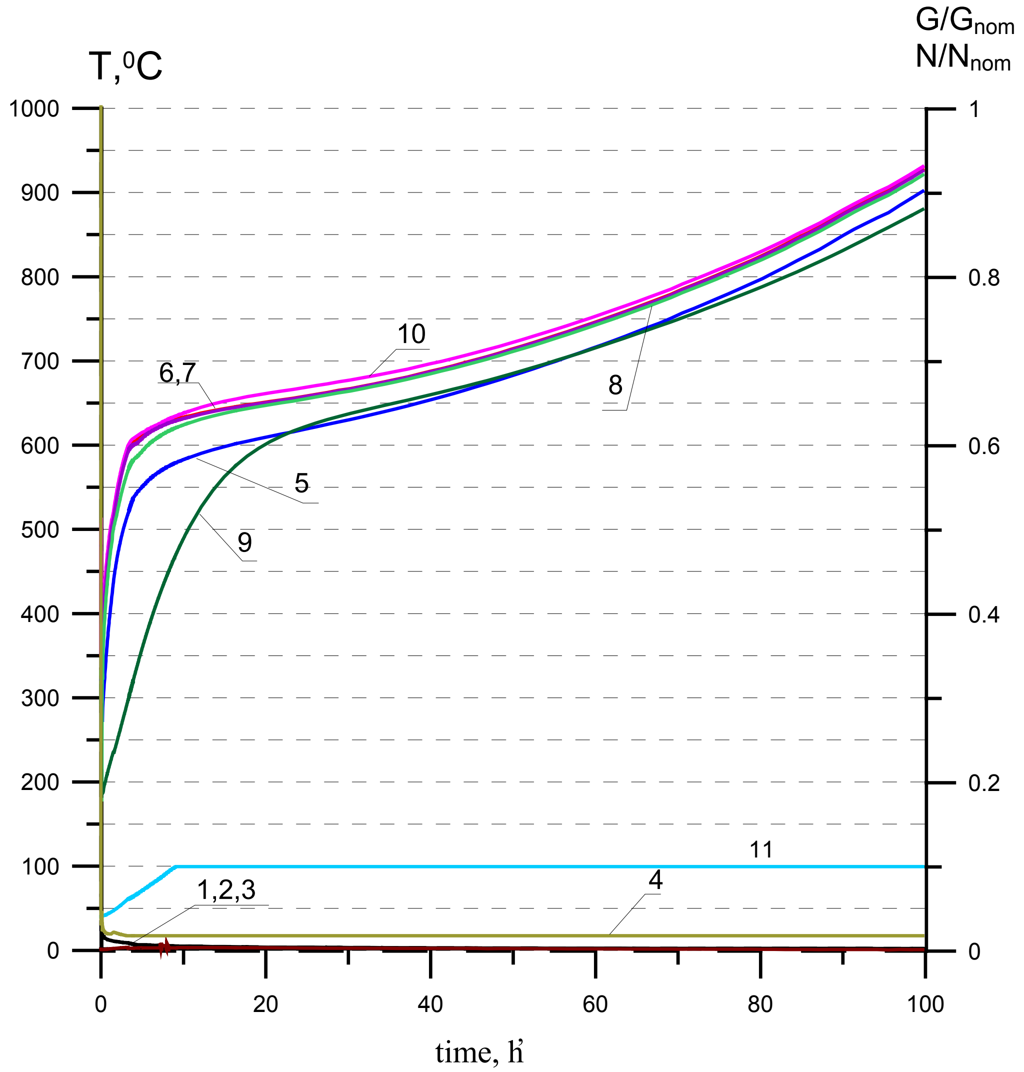


FIG. 3. The dynamics of the parameters at the final stage of the accidental situation with blacking out

(N_{res}/N_{nom}, N_n/N_{nom}, $N(\tau)/N_{nom}$ – the ratio of the residual power release, neutron power and total power to the nominal power correspondingly;

G/G_{nom}- relative flow rate of the primary circuit coolant;

T^{core}_{inlet}, T^{core}_{outlet} - coolant temperatures at the core inlet and core outlet;

T^{SG}_{inlet}, $T^{SG}_{\text{outlet top}}$ and $T^{SG}_{\text{outlet buttom}}$ - coolant temperatures at the inlet, outlet from the top and at the outlet from bottom of the SG;

$T^{core}_{\max}$ - maximal coolant temperature in the core;

T_{PHRS} - water temperature in the PHRS tank):

1-N_{res}/N_{nom}; 2- N_n/N_{nom}; 3- $N(\tau)/N_{nom}$; 4-G/G_{nom};

5 -T^{core}_{inlet}; 6- T^{core}_{outlet}; 7- T^{SG}_{inlet}; 8- $T^{SG}_{\text{outlet top}}$; 9-$T^{SG}_{\text{outlet buttom}}$; 10-$T^{core}_{\max}$; 11-$T_{PHRS}$

By the end of the first day the maximal temperature of the fuel element's cladding made of steel EP-823 will be 667 ^{o}C. At that moment the average temperature of fuel will not be higher than the values obtained under the nominal mode. Therefore, there will be no direct thermo-mechanical interaction of the slightly over-heated cladding and fuel. Thus, a conclusion is made that the fuel element's claddings is remaining tight and up to the end of the first day of the pre-accidental situation the operability of the core is assured.

The basic vessel and the safeguard casing of the monoblock will keep their integrity as during the 100 hours of accident progressing the allowable temperature for the monoblock vessels made of austenitic steel is not exceeded.

Nevertheless, in order to maintain the operability of the RI and to eliminate development of the pre-accidental situation into the accident, there should be taken the measures on recharging the water in the PHRS tanks with the help of off-design sources and providing with electricity the pumps of the inter-circuit and technical water.

3.2.4 Damage of the power-unit building, blacking out and failure of tightness of the primary circuit gas system with direct contact of the LBC surface with an air atmosphere

The predictive assessments are based on a following scenario of accident progressing:

- it is presumed that the powerful external impact has damaged the NPP unit containment and damaged (broken up) a lid of the reinforced concrete box in which reactor module SVBR-75/100 is installed;
- the equipment of the SVBR-75/100 primary circuit's gas system that is installed beyond the monoblock vessel is much damaged; there is a direct contact of the radioactive LBC surface with an atmosphere;
- malfunction in electricity supply is a cause of shutting down the circulators of all installation's circuits and failure of the standard system of heat removal; the reactor EP actuates at the first seconds of the accident;
- the coolant is heated up to ~600 °C on the surface of the alloy in the circuit;
- the accidental situation is lasting for 24 hours, then as a result of the actions on accident elimination the emergency cooling system will be switched on, the coolant temperature will be decreased to 300-400 °C, the surface of the coolant's contact with an atmosphere will be essentially diminished due to construction of the temporary sheath over the damaged reactor box and organization of ventilation of the box according to the timer scheme with holding up the aerosols on the filters;
- as a result of accidental over-heating the fuel elements' claddings, the gas-untight fuel elements become the ones «damaged and contacting», in which the coolant comparatively easily penetrates via the fuel elements' cladding to the fuel pellets and is a cause of the event in which the gas and volatile fission products accumulated earlier in their cavities are exhausted from the fuel elements into the coolant;
- the state of the fuel elements of the reactor is near the operating OSL for fuel elements.

In conditions of the accident the radioactivities of gas and volatile fission products will dominate in the accidental radioactive exhaust into the atmosphere.

During the period of the accident's peak, the most dangerous for the population is inhalation of the exhaust's radioactive aerosols. The dose of that factor of irradiation can be up to 91 % of the total doze in the beginning of the accident. During one hour of man's staying in the accidental zone without respiratory protective means the irradiation dose caused by inhalation of polonium aerosols will not exceed 27 μSv or will be less than 6 % of the total dose during the period of accident's peak at the distance of 1 km from the source of the exhaust (~0.5 mSv).

To live for a long time on the territory that has been radioactive-contaminated due to the accidental exhaust is dangerous because of three irradiation factors:

- external irradiation from radionuclides deposited on the surface of the ground, growth and buildings;
- internal irradiation caused by inhalation of radioactive dust blown by wind (it is only vital during the first year on happening the accident);
- internal irradiation caused by intake of food and water contaminated by exhaust radionuclides.

As for the first two factors, the assessments of the population's irradiating doses have revealed that at the distances over 1 km from the source of exhaust, the individual annual doses will not exceed 20 mSv. Therefore, taking into account that irradiation by food and water can be prevented with the help of prohibiting the intake of the foodstuff obtained on the contaminated territory and the water from the unprotected sources, it is proposed to implement such prohibition as a major measure of population's protection. In case of implementation of this protection measure, the dose loads for the population will be within the frameworks specified by the Russian standards for radiation safety requiring population evacuation (over 50 mSv [6]).

4. Conclusion

The realized preliminary analysis of safety have revealed that for the SVBR type RI the inherent self-protection properties result in lack of the accidents being within the design basis as at this design stage there are no scenarios of progressing the accidental situations which cause exhaust of radioactivity and/or ionizing radiation in quantities exceeding the established safety limits beyond the provided normal operation design limits.

The inherent self-protection properties coupled with an algorithm of controlling the CPS rods, design of the RMB and safety systems of passive actuation deterministically eliminate the certain accidental situations (such as prompt neutron reactor runaway, ingress of steam-water mixture into the reactor core and over-pressurization of the RMB vessel in an event of guillotine rupture of several tubes of the SG evaporator module, primary circuit coolant loss) and the initial events with superposition of multiple failures, which cause the accidents being beyond the design basis (such as release of the total operative reactivity margin without EP actuation, blacking out of the RI, destruction of the RI building with large scale failure of tightness in the gas system, direct contact of the alloy surface with an air atmosphere and coolant's overheating) do not result in catastrophic consequences and do not require population evacuation beyond the NPP sanitary-protection zone.

The properties and safety parameters of the SVBR type RI (such as inherent self-protection, deterministical elimination of severe accidents, lack of catastrophical consequences and necessity of population evacuation in the events of severe accidents being beyond the design basis) which are virtually obtained «free of charge» due to only use of a fast neutron reactor, heavy liquid-metal coolant and an integral design of the primary circuit enable economical-effective use of those RIs in the different purpose SPNPs located nearby the population centers.

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